Nucleopedia™

Essentials in Nuclear Engineering
2009 Edition

AREVA

edia

Foreword

The Nucleopedia™ 2009 Edition is an update of the 1991 English version of Framemo, first published in 1981. The guide has been given a major revamp, with color, more pages, clearer layout, full technical data and details of notions specific to the nuclear field. All of which makes it an essential tool for any engineer.

Also new for this edition are details of the latest AREVA products, such as the 1600+ MWe EPR™ reactor, the first Generation III+ reactor under construction, the KERENA™* boiling water reactor, and ATMEA 1™ reactor, the world's most advanced 1100 MWe pressurized water reactor.

One product of AREVA's creation in 2001 has been the integration of developments in the front and back ends into the nuclear fuel cycle. Details on licensing in Germany and the USA are also included, together with full and up-to-date energy data. The guide contains standard French and German fluid system abbreviations, the most commonly-used acronyms, and the most useful website addresses. Another new feature in this edition is a glossary.

Whatever your interest in nuclear energy, you'll find all the essentials in nuclear engineering in AREVA's Nucleopedia™.

** SWR 1000 now branded as KERENA™*

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AREVA, THE REACTOR VENDOR

1.1. The nuclear program

AREVA is the world leader in the design and construction of nuclear power plants and research reactors, modernization, maintenance and repair services, component manufacture and nuclear fuel supply.

With headquarters in Paris, France, the company has two main subsidiaries in Germany and the United States. With offices in Western and Central Europe, the United States, Asia and South Africa, AREVA can meet the local requirements of all its customers. AREVA offers experience, R&D, and unrivalled innovation and expertise in engineering, fuel supply, and equipment and services for all types of reactors, primarily PWRs and BWRs.

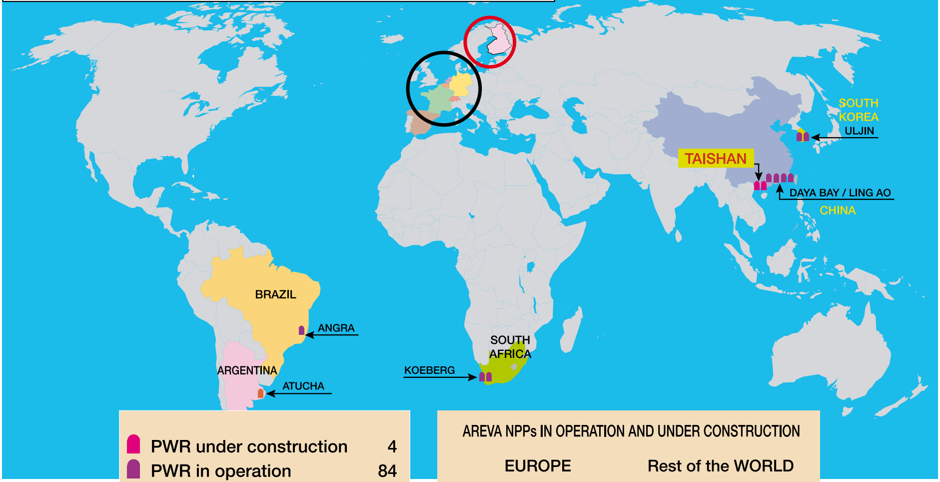
The business sector related to plants has the necessary expertise in the full range of nuclear engineering and technical disciplines to handle new plant construction, and modernization and backfitting projects for both pressurized and boiling water reactors. To date, AREVA has built or is building 102 nuclear power plants in 11 countries, representing almost 30% of existing nuclear power installations in the world. Its latest major projects include the construction of new EPR™ units: one unit at the Olkiluoto site in Finland, one unit at the Flamanville site in France and two units at the Taishan site in China.

The business sector in charge of Services currently provides a full range of inspection, repair and maintenance services for all types of pressurized water reactors including VVER, boiling water reactors and heavy water reactors (including CANDU plants).

The fuel business sector is world leader in the nuclear fuel sector thanks to its extensive research and development activities, which have yielded optimized materials, high-burnup fuel and improved control assemblies. To date, AREVA has supplied more than 170,000 fuel assemblies to 137 PWRs and 57 BWRs in the U.S., Europe, Asia, South America and South Africa. Fuel assembly fabrication involves a fabrication network with manufacturing locations in all three regions.

AREVA provides the industry‘s broadest spectrum of manufacturing capability for nuclear power plant equipment and components. The company has designed and built primary system components for more than 90 reactors worldwide. It has manufactured 76 reactor pressure vessels, 301 steam generators, 70 pressurizers and 220 reactor coolant pumps. It has taken part in the replacement of more than 100 steam generators, including 50% of those replaced in the United States over the last eight years.

Fig.1.1-1 Schematic representation of 102 AREVA NPPs around the World



PWR under construction	4
PWR in operation	84
BWR in operation	6
PHWR in operation	1
Shutdown Reactors	7
Total plants	102

AREVA NPPs IN OPERATION AND UNDER CONSTRUCTION

EUROPE		Rest of the WORLD	
Belgium	3	Argentina	1
Finland	1	Brazil	1
France	59	China	6
Germany	17	South Africa	2
Netherlands	1	South Korea	2
Spain	1		
Switzerland	1		

4 EPR™ reactors under construction:
Flamanville 3, France
Olliluoto 3, Finland
Taishan 1&2, China

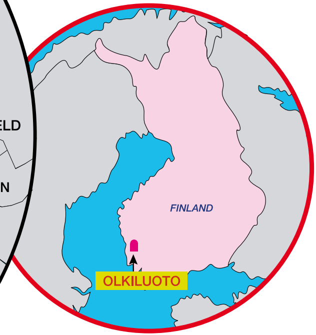
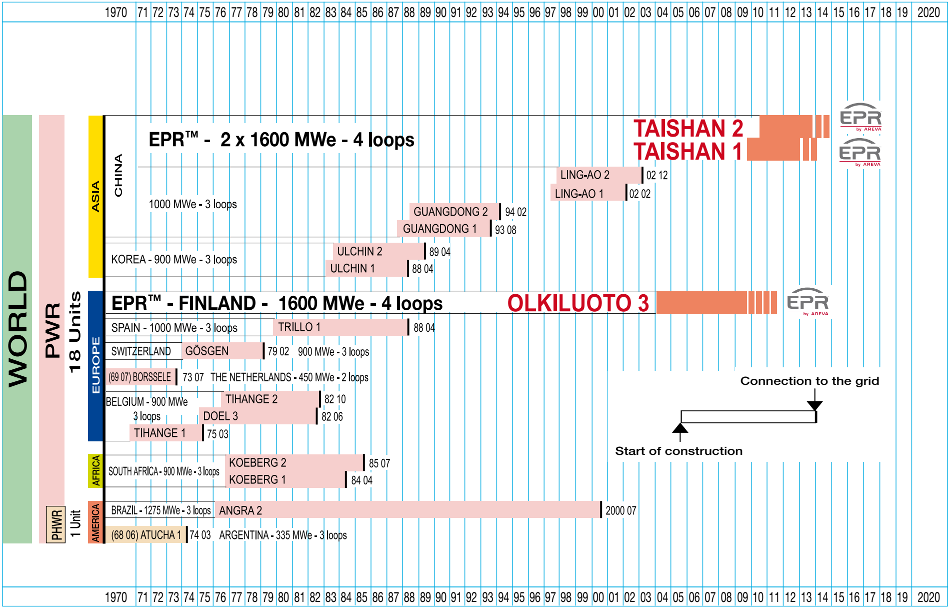
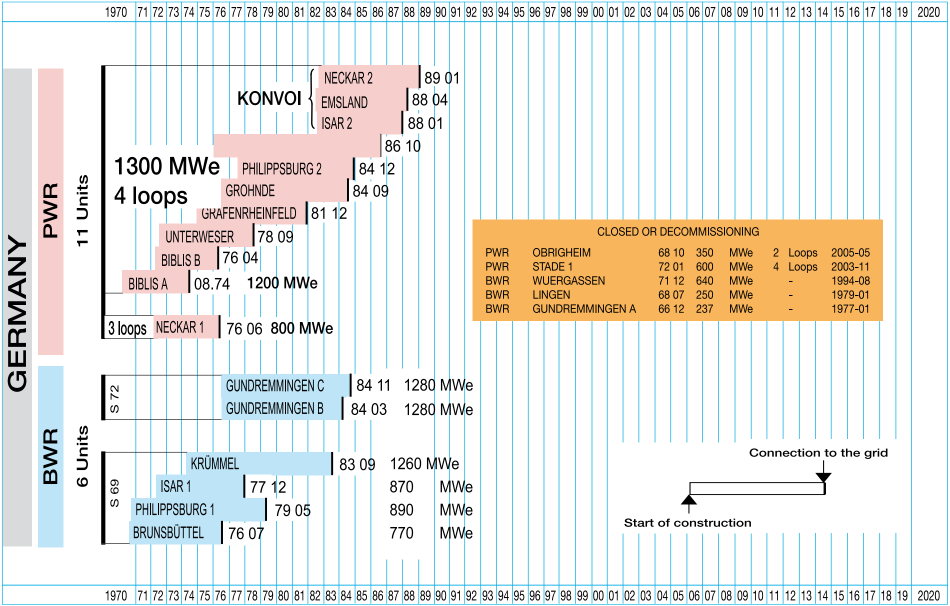


Fig. 1.1-2 Bar chart representing AREVA reactors construction program

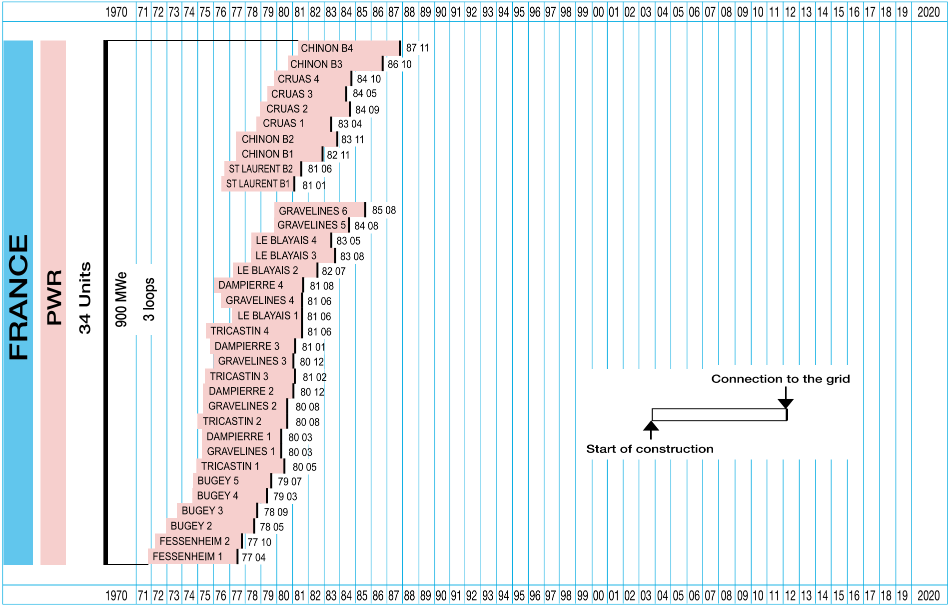


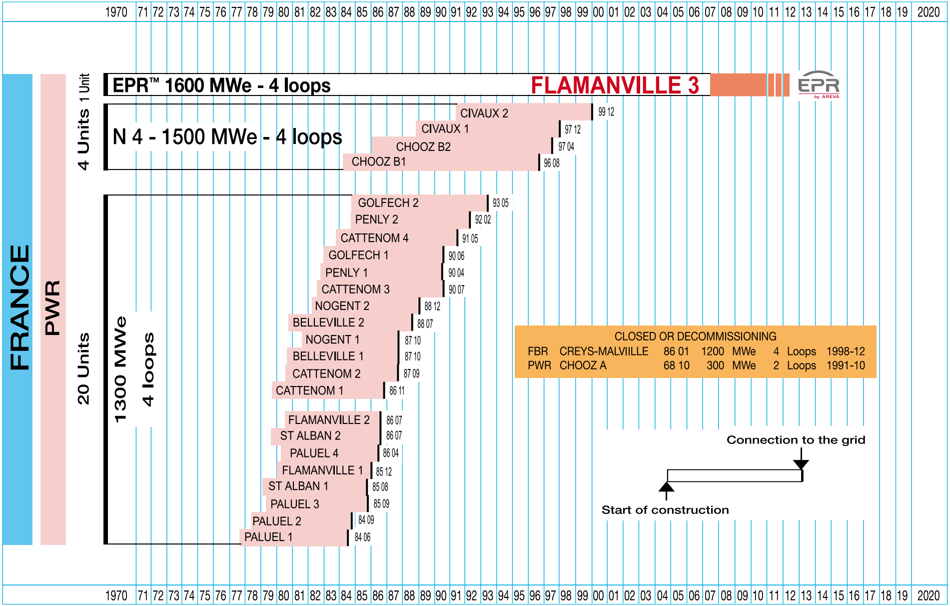


CLOSED OR DECOMMISSIONING

PWR	OBRIGHEIM	68 10	350	MWe	2	Loops	2005-05
PWR	STADE 1	72 01	600	MWe	4	Loops	2003-11
BWR	WUERGASSEN	71 12	640	MWe	-		1994-08
BWR	LINGEN	68 07	250	MWe	-		1979-01
BWR	GUNDREMMINGEN A	66 12	237	MWe	-		1977-01

Fig. 1.1-2 Bar chart representing AREVA reactors construction program





1970 71 72 73 74 75 76 77 78 79 80 81 82 83 84 85 86 87 88 89 90 91 92 93 94 95 96 97 98 99 00 01 02 03 04 05 06 07 08 09 10 11 12 13 14 15 16 17 18 19 2020
FRANCE
PWR
1 Unit
EPR™ 1600 MWe - 4 loops
FLAMANVILLE 3
EPR by AREVA
4 Units
N 4 - 1500 MWe - 4 loops
CIVAUX 2 99 12
CIVAUX 1 97 12
CHOOZ B2 97 04
CHOOZ B1 96 08
20 Units
1300 MWe
4 loops
GOLFECH 2 93 05
PENLY 2 92 02
CATTENOM 4 91 05
GOLFECH 1 90 06
PENLY 1 90 04
CATTENOM 3 90 07
NOGENT 2 88 12
BELLEVILLE 2 88 07
NOGENT 1 87 10
BELLEVILLE 1 87 10
CATTENOM 2 87 09
CATTENOM 1 86 11
FLAMANVILLE 2 86 07
ST ALBAN 2 86 07
PALUEL 4 86 04
FLAMANVILLE 1 85 12
ST ALBAN 1 85 08
PALUEL 3 85 09
PALUEL 2 84 09
PALUEL 1 84 06
CLOSED OR DECOMMISSIONING
FBR CREYS-MALVIILLE 86 01 1200 MWe 4 Loops 1998-12
PWR CHOOZ A 68 10 300 MWe 2 Loops 1991-10
Connection to the grid
Start of construction
1970 71 72 73 74 75 76 77 78 79 80 81 82 83 84 85 86 87 88 89 90 91 92 93 94 95 96 97 98 99 00 01 02 03 04 05 06 07 08 09 10 11 12 13 14 15 16 17 18 19 2020

1.2. Characteristics of AREVA reactors

AREVA's line of reactors includes the EPR™ and ATMEA 1 ™ reactors which are PWR's and a BWR, the KERENA™ reactor. All are Generation III+ reactors which bring major advances in terms of competitiveness and safety.
All AREVA reactors are based on existing, proven technologies incorporating innovative systems. These models have a very high level of safety thanks to significant technology advances that help prevent and reduce the risk of an accident and provide greater protection for the neighbouring population. They are also designed to withstand the crash of a commercial airplane.

1.2.1. EPR™

The EPR™ reactor is the most powerful reactor marketed by AREVA with a net electrical output in the range of 1600+MWe. The safety level is further improved thanks to a combination of fourfold redundancy and diversity for the safeguard systems.

Fig.1.2-1 EPR™ reactor cutaway

1.2.2. ATMEA 1 ™

The ATMEA joint venture, officially formed in November 2007 by Mitsubishi Heavy Industries, LTD (MHI) and AREVA NP in equal shares, is working on the design of the ATMEA 1 ™ reactor, which will have approximately 1,100+ MWe of power. It features advanced safety systems, high thermal yields, and a flexible 12 to 24 month operating cycle. The ATMEA 1 ™ reactor will be ready for the market in 2010/2011.

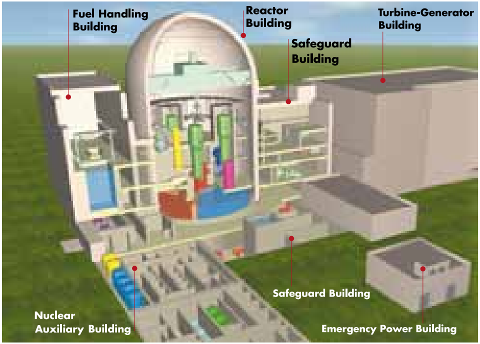


Fig.1.2-2 ATMEA 1 ™ reactor cutaway

1.2.3. KERENA ™

AREVA is developing its latest boiling water reactor. Positioned in the medium-capacity market, its electrical output is 1,250+MWe. This reactor incorporates primarily passive safety systems while keeping a certain number of active systems, ensuring a high level of safety and substantial operating flexibility.

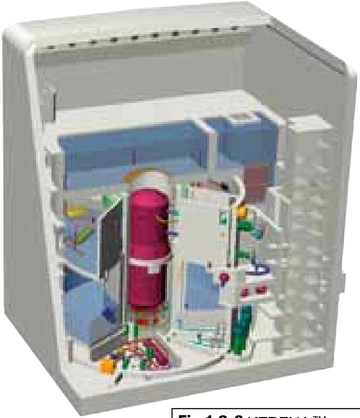


Fig.1.2-3 KERENA ™ reactor cutaway

1.2.4. Main characteristics of AREVA reactors

The figures given in the following table correspond to typical values, representative of the different types of reactors already built or under construction. From one unit to another, slight variations can be observed essentially resulting from the characteristics of the plant site itself (heat sink temperature, ambient air conditions, etc.)

Note: in the following table, **"n.a."** means not applicable and **"-"** means not available.

		CPY 900 MWe	Ling Ao 1000 MWe	P4 or P'4 1300 MWe	N4 1500 MWe	Konvoi	EPR ™ 1660 MWe	BWR 72[0] 1300 MWe	KERENA ™
GENERAL CHARACTERISTICS									
Single (S) or Twin (T) units		T	T	S	S	S	S	T	S
Number of loops		3	3	4	4	4	4	n.a.	n.a.
NSSS thermal power (MWth)		2785	2905	3817	4270	3850	4590	3840	3370
Net electrical power to the grid (MWe)[1]		920	1000	1320	1470	1269	1660	1284	1250
CORE									
Active height (m)		3.66	3.66	4.27	4.27	3.90	4.20	3.71	3.0
Pitch between fuel assemblies (cm)		21.5	21.5	21.5	21.5	23.0	21.5	-	-
Control Rod Pitch [mm]		n.a.	n.a.	n.a.	n.a.	n.a.	n.a.	305	363
No. of fuel assemblies		157	157	193	205	193	241	784	664
Quantity of UO_2 (ton)		82	82	118	125	103	145	136	138.5
Enrichment (mass %):									
1st core	Region 1	1.8	1.8	1.5	1.8	1.9	2.1[2]	1.16	1.5[4]
	Region 2	2.4	2.4	2.4	2.4	2.5	3.2[2-3]	1.85	2.17[4]
	Region 3	3.1	3.1	2.95	3.1	3.2	4.2[2-3]	2.12/2.59	2.99[4]

(0) Gundremmingen B and C; Data per unit
(1) Typical value -(2) Depending on the Fuel Management strategy
(3) Burnable poison pins made of Gadolinia are used as soon as the 1st fuel cycle
(4) Value at average discharge burnup at equilibrium of 45,000 MWd/ton

	CPY 900 MWe	Ling Ao 1000 MWe	P4 or P'4 1300 MWe	N4 1500 MWe	Konvoi	EPR ™ 1660 MWe	BWR 72[0] 1300 MWe	KERENA ™
At equilibrium [5]	3.25	3.7[6]	3.1[6]	3.4[7]	2.0 [8]	4.95[2]	2.8 up to 5.47 with MOX	4.68 / 3.54[4]
Average discharge burn-up at equilibrium (MWd/ton)	47,000	39,000	55,500	48,000	46,000	Up to 65000[2]	Up to 60,000	65,000
Core thermal power (MWth)	2775	2895	3800	4250	3850	4590	3840	3370
Average power density (kW/core liter)	104	107	100	105	95.3	96.6	56.8	51.36
Average linear power density (W/cm)	178	186	170.2	179.2	166.6	166.7	204	126.88
Maximum linear power density at 118% full power (W/cm)	590	590	590	590	-	590	n.a.	n.a.
Core Heat exchange surface (m^2)	4524	4525	6480	6480	6775	8005	7856	8050.4
Overall fuel assembly length (m)	4.10	4.10	4.85	4.85	4.82	4.80	4.47	3.76
Fuel assemblies cross section (mm)	214 x 214	213.4 x 213.4	213.4 x 213.4	213.4 x 213.4	229.6 x 229.6	213.4 x 213.4	131 x 131	159.9 x 159.9
Fuel assembly mass (kg)	664	670	773	773	830	782	255	325
No. of fuel rods per assembly	264	264	264	264	300	265	80/91	128
Number of grids	8	8	10	10	9	10	7	7
Fuel rod outer diameter (mm)	9.5	9.5	9.5	9.5	9.5	9.5	10.05	10.05
Fuel rod pitch (mm)	12.6	12.6	12.6	12.6	12.7	12.6	12.96	12.95
Array	17 x 17	17 x 17	17 x 17	17 x 17	18 x 18	17 x 17	10 x 10	12 x 12
Fuel pellet diameter (mm)	8.2	8.2	8.2	8.2	8.05	8.2	8.67	8.67

(0) Gundremmingen B and C; Data per unit
(1) Typical value - (2) Depending upon the Fuel Management strategy
(3) Burnable poison pins made of Gadolinia are used as early as the 1st fuel cycle
(4) Value at average discharge burnup at equilibrium of 45,000 MWd/ton

(5) Operation with annual refueling and a load factor of 80%
(6) One-third of fuel assemblies replaced by next assemblies each year
(7) One-fourth of fuel assemblies replaced by next assemblies each year
(8) Reload Enrichment

	CPY 900 MWe	Ling Ao 1000 MWe	P4 or P'4 1300 MWe	N4 1500 MWe	Konvoi	EPR ™ 1660 MWe	BWR 72[0] 1300 MWe	KERENA ™
REACTOR COOLANT SYSTEM								
Operating pressure (bar abs)	155	155	155	155	158	155	70.6	75
Design pressure (bar abs)	172.3	172.3	172.3	172.3	175	176	86.3	89
Pressurizer temperature (°C)	343	343	343	343	350	343	286	290.5
Coolant temperature at the vessel inlet (°C)	292.4	292.4	292.8	292.2	291.3	295.2	-	-
Volume of coolant (m^3)	274	283	408	380	400	460	n.a.	n.a.
Boric acid concentration (ppm)	0 to 2500	0 to 2500	0 to 2500	0 to 2150	0 to 2200	0 to 2600[1]	n.a.	n.a.
STEAM-WATER SYSTEM								
Rated feedwater temperature (°C)	219	226	229.5	229.5	-	230	215	220
Full-load steam pressure (bar)	56	67.5	71	72	65	78[2]	70.6	75
Maximum moisture content at the SG outlet (%)	0.25	0.25	0.25	0.25	0.25	0.25	< 0.2	< 0.2
Max. steam flow rate per SG / steam line (kg/s)	606.6	537	537	601	524	651	2077[3] (520 per line)	1849[4] (617 per line)
Steam line inside diameter (mm)	753	748	698	692	670 NI / 680 CI	DN 750	600 (NW)	600
Feedwater line inside diameter (mm)	377	356	364	410	DN 550	DN 500	450 (NW)	546
MAIN AUXILIARY SYSTEMS								
Reactor coolant system letdown flow rate (kg/s)	3.8 to 7.5	3.8 to 7.5	5 to 10	5 to 10	10 to 31 normal 19.5	10 to 20	n.a.	n.a.
Purification at cold shutdown (kg/s)	40	n.a.	14	25	10 to 31	18	n.a.	n.a.

(1) EPR ™ reactor uses ^{10}B enriched boron
(2) at tube bundle outlet
(3) 4 lines
(4) 3 lines

		CPY 900 MWe	Ling Ao 1000 MWe	P4 or P'4 1300 MWe	N4 1500 MWe	Konvoi	EPR ™ 1660 MWe	BWR 72[0] 1300 MWe	KERENA ™
Spent fuel pit storage capacity (number of fuel assemblies)		216	690	612	612	772	954[1]	3219	1650
Refueling water storage tank volume (m^3)		1600	1600	3000	3400	-	1900 to 2000	-	-
Water Volume in Flooding pool and wetwell (m^3)		n.a.	n.a.	n.a.	n.a.	4 x 470	n.a.	3150	6100
REACTOR VESSEL									
Inside diameter (m)		4.0	3.99	4.39	4.48	500	4.885	6.62	7.12
Thickness at the core level (mm)		200	200	220	225	250 + 6	250	163 + 8	183
Overall height with closure head (m)		13.2	13.2	13.6	13.6	12.4	13.1	22.35	23.8
No. of adapters for control rod clusters		61	61	73	73	-	89	-	-
No. of closure studs		58	58	54	54	52	52	72	76
Mass (empty, with closure head and studs) (ton)		330	330	435	453	486	552	783	980
REACTOR INTERNALS									
Lower internals	mass (ton)	84	84	116	115	106	186	100[2]	121
	height (m)	9.92	9.92	10.15	10.25	8.00	9.22	8.1	7.2
	diameter (m)	3.91	3.91	4.37	4.48	4.67	4.82	5.4	5.7[3]
Upper internals	mass (ton)	43.7	43.7	46	69	51	84	164[4]	290
	height (m)	4.2	4.2	4.46	4.54	3.92	4.44	-	13
	diameter (m)	3.91	3.91	4.37	4.48	4.67	4.11	-	6.7
No. of guide tubes		53	61	65 + 8	73	61	89	193	157

(1) For Finland OL3 Unit only
(2) incl. core
(3) core shroud support
(4) cap, water separator, steam dryer

	CPY 900 MWe	Ling Ao 1000 MWe	P4 or P'4 1300 MWe	N4 1500 MWe	Konvoi	EPR ™ 1660 MWe	BWR 72(0) 1300 MWe	KERENA ™
STEAM GENERATORS								
Model	51 M	55/19	68/19	73/19E	54 SK	79/19 TE	n.a.	n.a.
No. of SGs	3	3	4	4	4	4		
Unit thermal power (MW)	928	968	954	1067.5	966.8	1154		
Heat exchange surface per SG (m^2)	4745	5430	6935	7330	5412	7960		
Overall height (m)	20.60	20.85	22.30	21.90	21.3	24.6		
Outside diameter, max. (m)	4.46	4.46	5.04	4.76	4.81	5.17		
Tube sheet thickness (mm)	534	555	605	590	700	620		
No. of U-tubes	3361	4474	5342	5599	4118	5980		
Outside diameter of U-tubes (mm)	22.2	19.05	19.05	19.05	22	19.05		
U-tube thickness (mm)	1.27	1.09	1.09	1.09	1.2	1.09		
Total mass, empty (ton)	300	310	440	400	420	550		
REACTOR COOLANT PUMPS / RECIRCULATION PUMPS								
Model	93D	100	100	N24	-	EPR ™-50Hz(1)	Internal Pump	Internal Wet Motor Pump
No. of pumps	3	3	4	4	4	4	8	8
Rated flow (m^3/h)	21,240	23,790	22,890	24,500	22,700	28,315	8,900	8,106
Pressure head in normal operation (m)	90.7	97.2	99	106	90	100.2	31	37
Rotating speed (rpm)	1500	1500	1500	1500	1500	1500	600-2000	700-2000
Power consumed in normal operation, per unit (kW)	5400	6680	6500	7200	5500	8680	1030	~800
Power consumed when cold, per unit (kW)	7200	9050	8700	9600	7400	10850	-	-
Height of the motor-driven pump unit (m)	8.2	8.2	8.2	8.5	9.8	9.4	7	-

(1) EPR ™–60Hz for US EPR ™ reactor

	CPY 900 MWe	Ling Ao 1000 MWe	P4 or P'4 1300 MWe	N4 1500 MWe	Konvoi	EPR ™ 1660 MWe	BWR 72[0] 1300 MWe	KERENA ™
Mass of the pump and motor together (ton)	86.5	104	104	102	102	112	16 (unit B) 11 (unit C)	-
PRESSURIZER								
Internal volume (m^3)	39.75	39.75	59.5	59.5	65	75	n.a.	n.a.
Max. heating power (MW)	1.44	1.44	2.16	2.16	2.1	2.6		
No. of heaters	60	60	90	90	102	108		
Unit heater power (kW)	24	24	24	24	21	24		
Design opening pressure (bar abs)	172.3	172.3	172.3	172.3	170/176	176		
Mass, empty (ton)	79	79	117	117	135	150		
PRESSURIZER RELIEF TANK								
Volume (m^3)	37	37	41	60	38	40	n.a.	n.a.
Mass, empty (ton)	9	9	10	14.1	20	31		
CONTROL ROD DRIVE MECHANISMS								
Number	-	57 + 4[1]	65	73	61	89	193	157
Step length (mm)	15.9	15.9	15.9	15.9	10.0	10.0	-	-
Lifting and lowering speed (m/mn)	1.2	1.14	1.14	1.14	0.375 or 0.75	0.375 or 0.75	30mm/s	30mm/s
Overall height without drive rod (m)	5.7	5.7	6.3	6.1	4.7	5.5	3.66	-
Total mass (kg)	680	680	690	690	115	404	-	~700
REACTOR COOLANT SYSTEM PIPING								
Hot leg inside diameter (mm)	737	736	736	736	750	780	n.a.	n.a.
Cold leg inside diameter (mm)	698	698	698	698	750	780		
Crossover leg inside diameter (mm)	787	787	787	787	750	780		
Thickness (mm)	60 to 85	64 to 71	62 to 71	63 to 84	52 to 90	76 to 97		

(1) Four extra CRDMs for greater flexibility in the refueling strategy

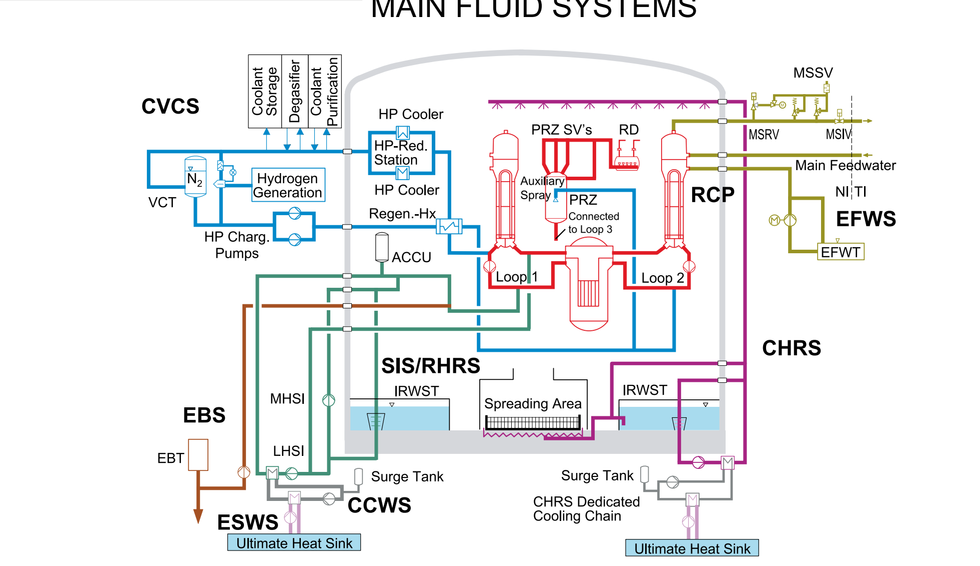
	CPY 900 MWe	Ling Ao 1000 MWe	P4 or P'4 1300 MWe	N4 1500 MWe	Konvoi	EPR™ 1660 MWe	BWR 72[0] 1300 MWe	KERENA™
IN-CORE INSTRUMENTATION								
Mobile in-core								
Type of system	RIC	RIC	RIC	RIC	Aeroball / SPND sensors	Aeroball / LVD sensors	Low / Inter-mediate range power detection	Low / Wide range detection
Vessel Penetration	Bottom	Bottom	Bottom	Bottom	Top	Top	Bottom	Bottom
No. of miniature detectors	5	5	6	6	n.a.	n.a.	n.a.	n.a.
Speed of flux detector movement (m/mn)	1.5 and 9	3 and 18	3 and 18	3 and 18	n.a.	n.a.	n.a.	n.a.
No. of flux measurement channels	50	50	58	60	28	40	3/5/44	39
No. of selectors	5	5	6	6	n.a.	n.a.	-	-
Total required duration of activation (mn)	≈ 60	≈ 60	≈ 60	≈ 60	3	≈ 5	-	-
Fix in-core								
No. of thermocouples	51	40	50	52	24	36	-	-
No. of lance fingers	n.a.	n.a.	n.a.	n.a.	8	12	-	-
No. of self-powered neutron detectors (SPND) per lance finger	n.a.	n.a.	n.a.	n.a.	6	6	-	-

		CPY 900 MWe	Ling Ao 1000 MWe	P4 or P'4 1300 MWe	N4 1500 MWe	Konvoi	EPR™ 1660 MWe	BWR 72[0] 1300 MWe	KERENA™
EX-CORE INSTRUMENTATION									
Detectors									
Source level	number	2	2	4	4	2	3	n.a.	n.a.
	length (m)	0.755	0.755	0.755	0.760	0.56	0.727		
	range (n/cm^2.s)	$1x10^{-1}$ to $1x10^{5}$	$1x10^{-1}$ to $1x10^{5}$	$1x10^{-1}$ to $1x10^{5}$	$1x10^{-1}$ to $1x10^{5}$	$1x10^{-2}$ to $1x10^{5}$	$5x10^{-2}$ to $5x10^{4}$		
Intermediate level	number	2	2	4	4	4	4		
	length (m)	0.55	0.55	0.55	0.76	0.67	0.655		
	range (n/cm^2.s)	$2.5x10^{2}$ to $2.5x10^{10}$	$2x10^{2}$ to $5x10^{10}$	$2.5x10^{2}$ to $2.5x10^{10}$	$2.5x10^{2}$ to $2.5x10^{10}$	$1x10^{2}$ to $1x10^{8}$	$1x10^{2}$ to $1x10^{10}$		
Power level	number	4	4	4	4	4	4		
	length (m)	3.17	3.17	3.96	3.85	0.67	0.655		
	range (n/cm^2.s)	$1x10^{7}$ to $5x10^{10}$	$5x10^{2}$ to $5x10^{10}$	$2x10^{7}$ to $5x10^{10}$	$2x10^{7}$ to $5x10^{10}$	$1x10^{5}$ to $1x10^{8}$	$1x10^{5}$ to $1x10^{10}$		
REACTOR CONTAINMENT									
Type		Single, with liner	Single, with liner	Double, no liner	Double, no liner	Double with spherical steel liner	Double, with liner	Prestressed concrete with liner	Double with liner
Inner containment	inside diameter (m)	37	37	43.8	43.8	globe d = 56.0	46.8	29	33
	inside height (m)	58	59.4	58.0	58.0		65.4	40.1	34.7
	equipment hatch diameter (m)	7.4	7.4	8	8		8.3	1.95 x 1.7	3.1
design pressure (bar)		5	-	4.8 - 5.2	5.3	6.3	5.5	3.3 (10) [1]	3.5 (8.3) [1]

(1) Design pressure for design base accidents (DBA); containment also stable at severe accidents (100% zircon oxidation) and containment pressure in brackets.

1.2.5. General sketches of an EPR ™ reactor

Fig. 1.2-4 EPR ™ reactor Main Fluid systems



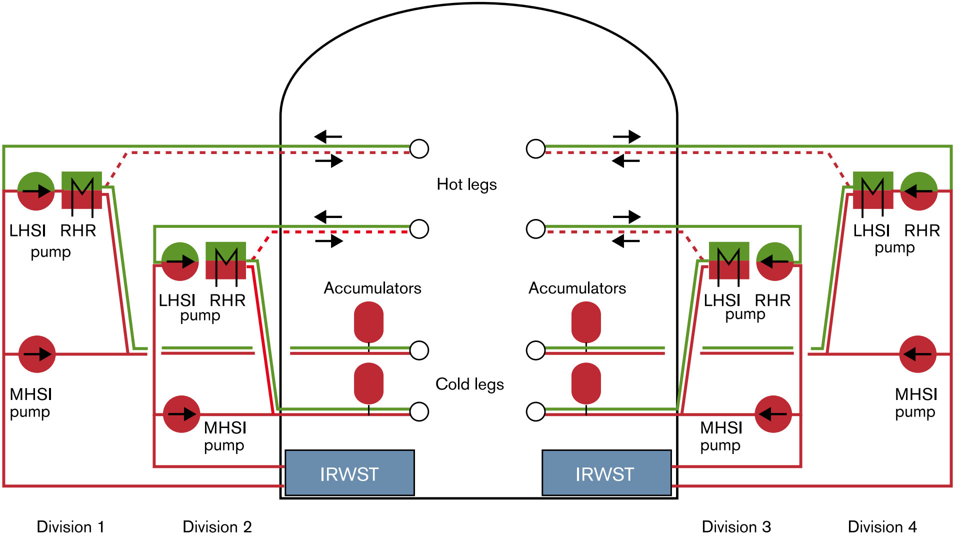


Fig. 1.2-5 Safety Injection/Residual Heat Removal System

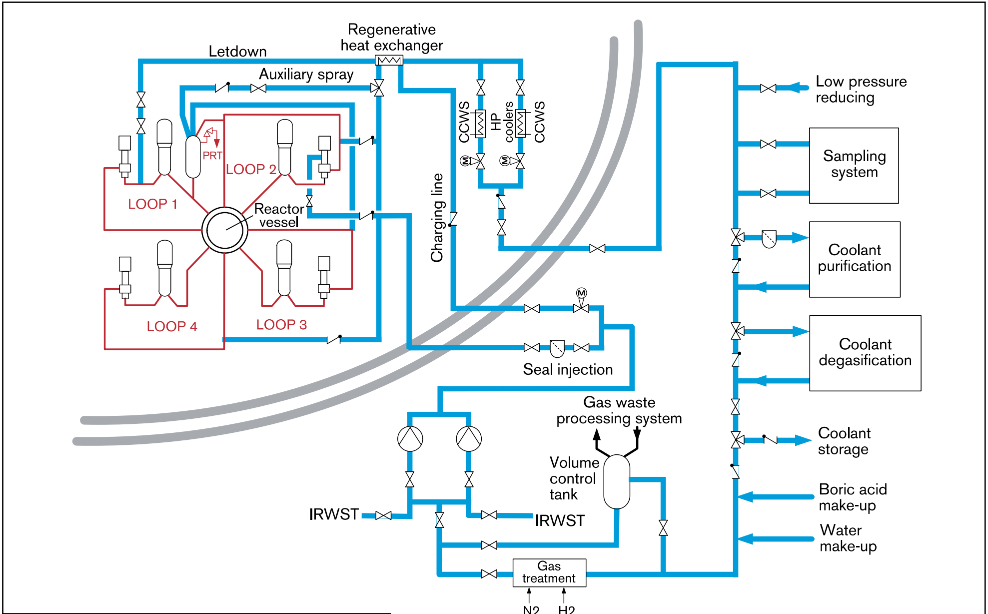


Fig. 1.2-6 Chemical and Volume Control System

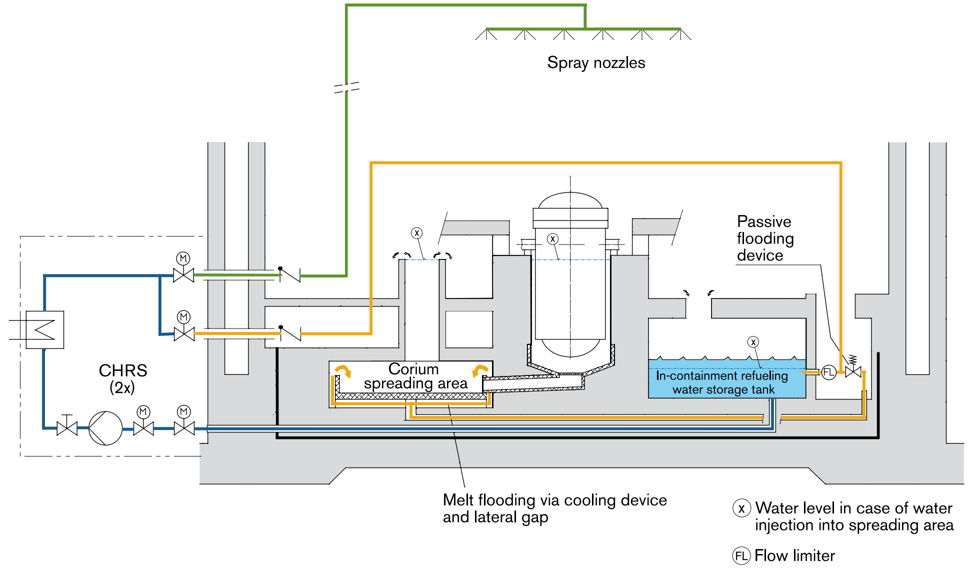


Fig. 1.2-7 Containment Heat Removal System

- A set of quadruple redundant safeguard systems with independent and geographically separated trains.

Fig. 1.2-8 Four Train Concept and corium spreading area

- The outer shell (in blue) protects against external hazards the Reactor building, Spent fuel builing and two of the four Safeguard buildings including the control room.

- A double concrete shell: an inner prestressed concrete housing internally covered with a metallic liner and an outer reinforced concrete shield.

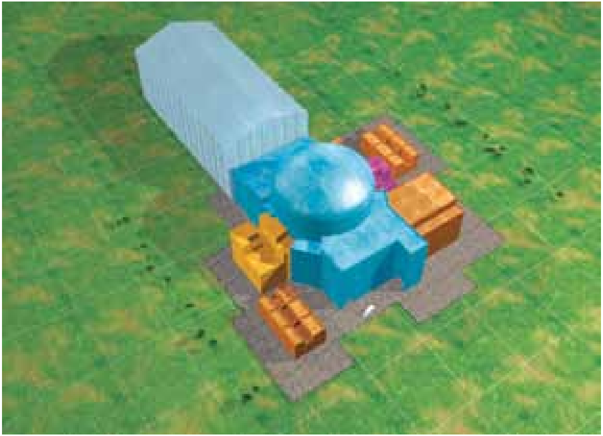


Fig. 1.2-9 Protection against airplane crash

Fig. 1.2-10 Double-walled Containment

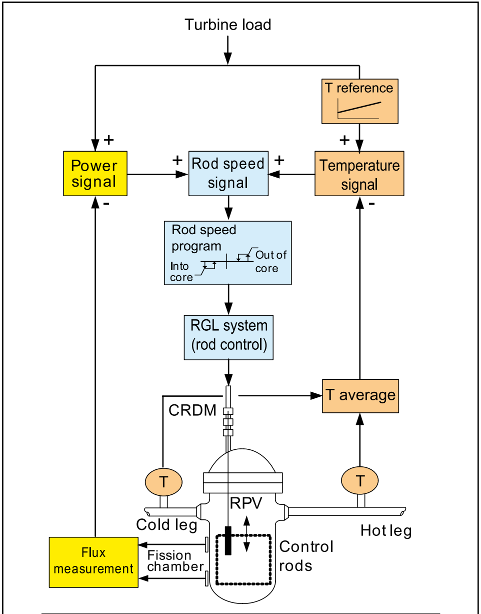


Fig. 1.2-11 Reactor Coolant System temperature

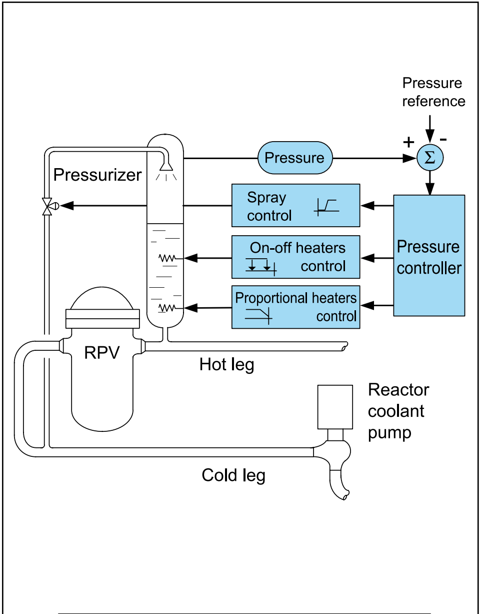


Fig. 1.2-12 Reactor Coolant System pressure

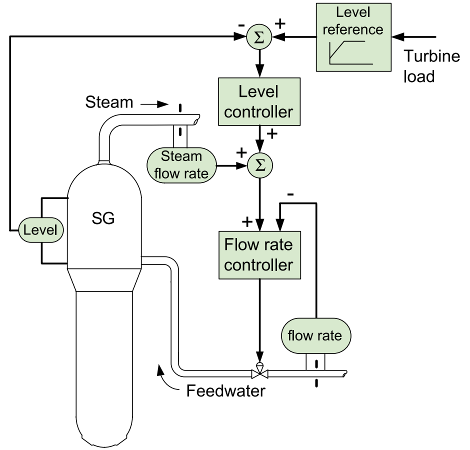


Fig. 1.2-13 Steam Generator Water level

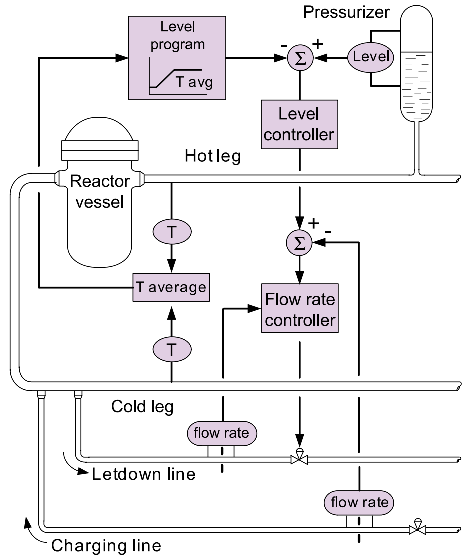


Fig. 1.2-14 Pressurizer Water level

1

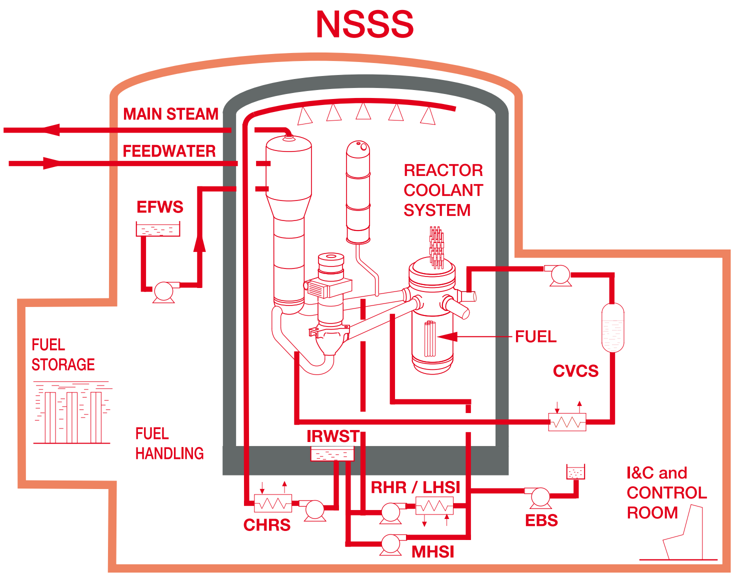


Fig. 1.2-15 Nuclear Steam Supply System

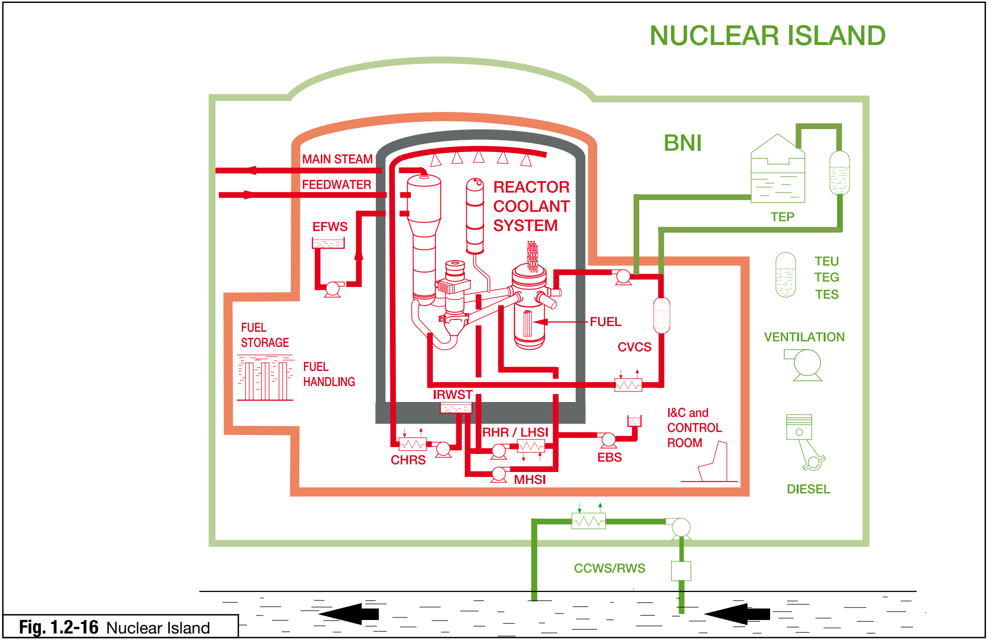


Fig. 1.2-16 Nuclear Island

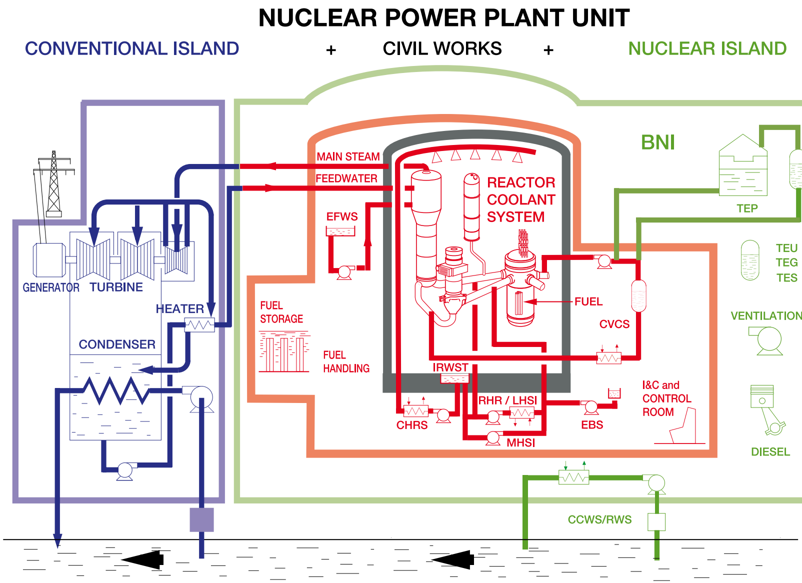


Fig. 1.2-17 Nuclear Power Plant Unit

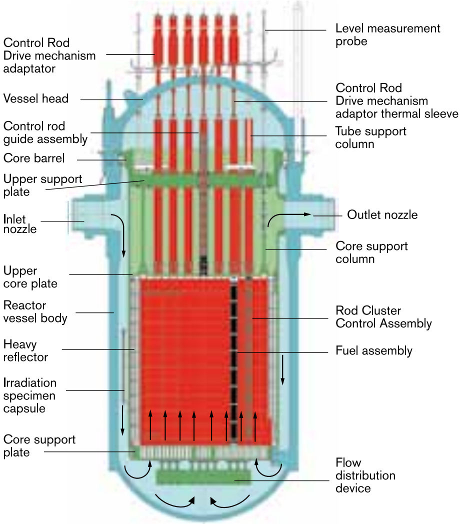


Fig. 1.2-18 EPR™ technology Reactor Pressure Vessel

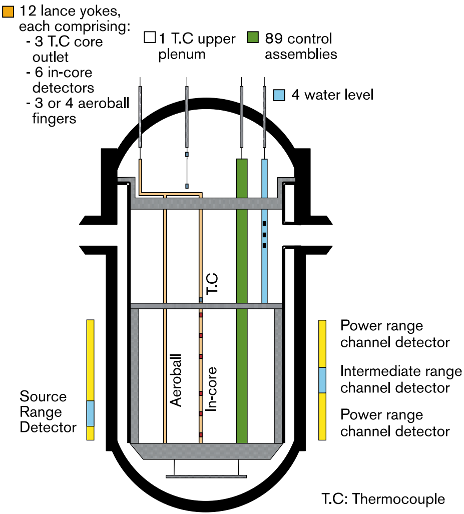


Fig. 1.2-19 Typical initial core loading

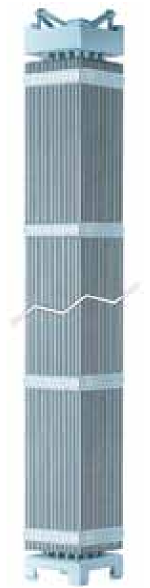


Fig. 1.2-20 Fuel assembly

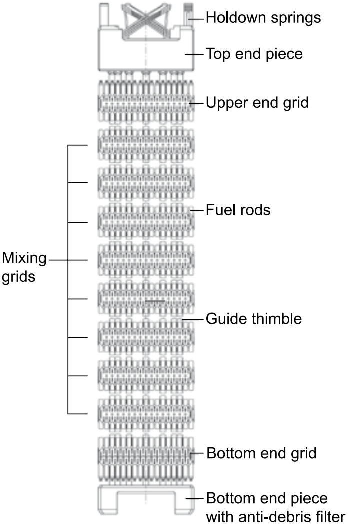


Fig. 1.2-21 PWR Fuel assembly

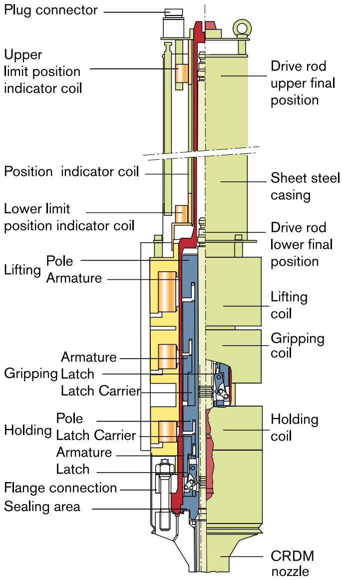


Fig. 1.2-22 Control Rod Drive Mechanisms

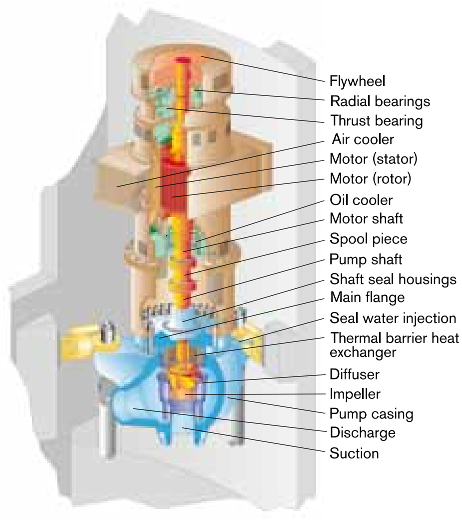


Fig. 1.2-23 Reactor Coolant Pump

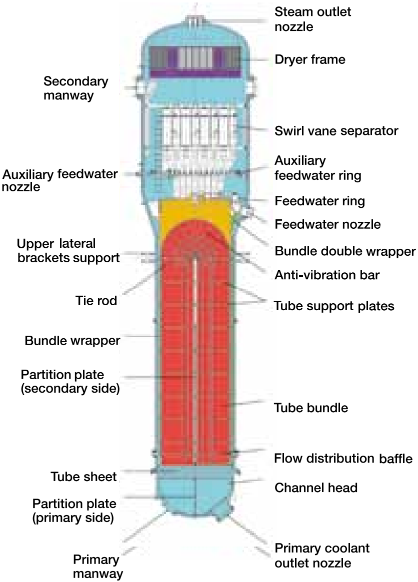


Fig. 1.2-24 Steam Generator Cutaway

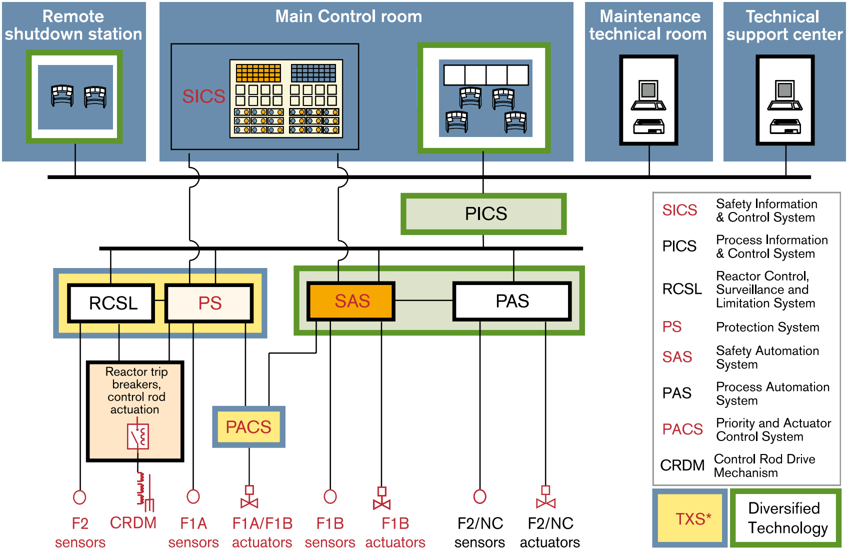
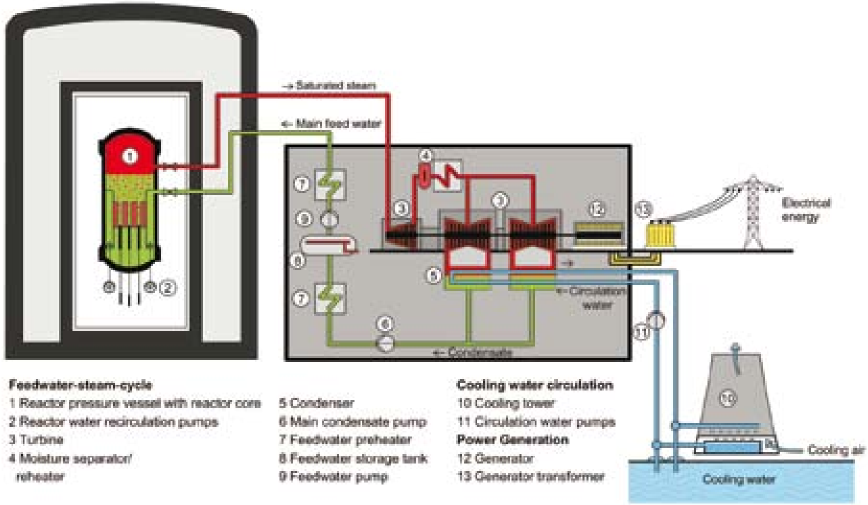


Fig. 1.2-25 I&C Architecture

1.2.6. General Sketches of BWR



Feedwater-steam-cycle
1 Reactor pressure vessel with reactor core
2 Reactor water recirculation pumps
3 Turbine
4 Moisture separator/ reheater
5 Condenser
6 Main condensate pump
7 Feedwater preheater
8 Feedwater storage tank
9 Feedwater pump

Cooling water circulation
10 Cooling tower
11 Circulation water pumps

Power Generation
12 Generator
13 Generator transformer

Fig. 1.2-26 General sketch of BWR

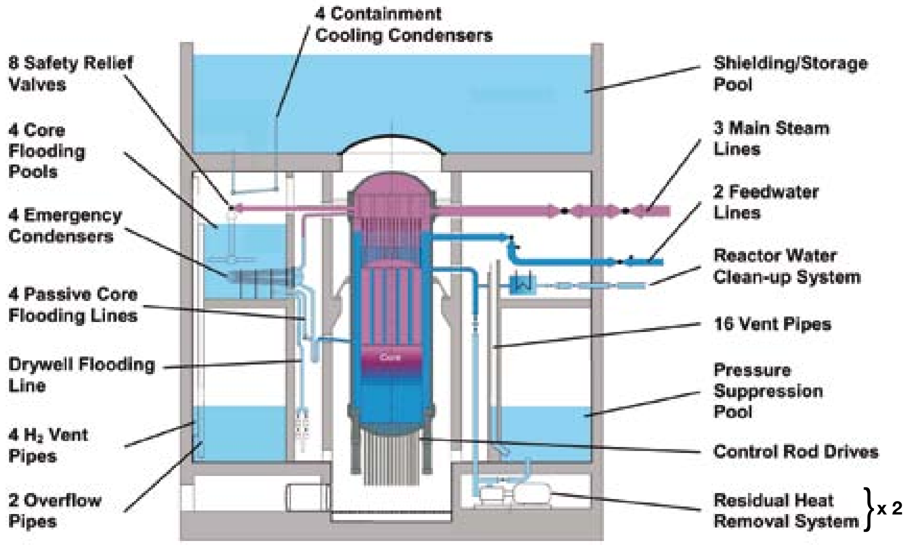
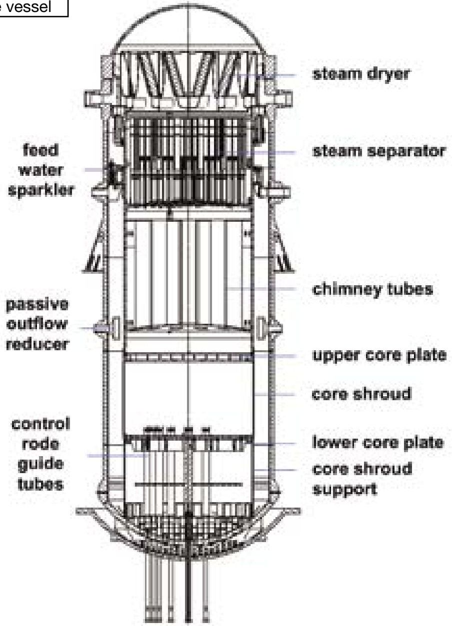


Fig. 1.2-27 KERENA™ reactor main fluid systems

Fig. 1.2-28 KERENA™ reactor pressure vessel



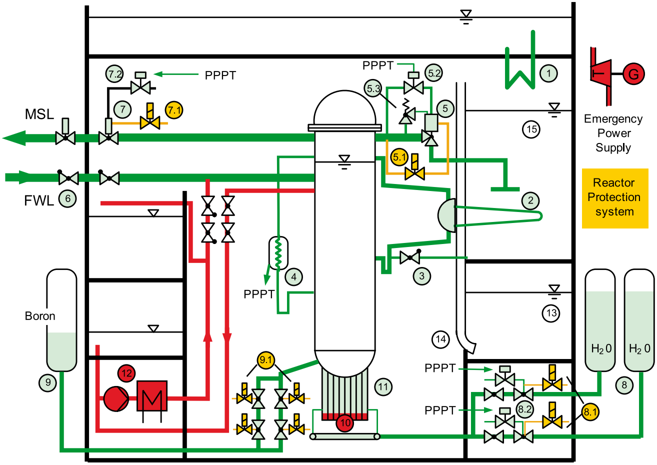


Fig. 1.2-29 KERENA™ passive and active safety systems

Passive Systems

1	Containment cooling condensers
2	Emergency condensers
3	Passive flooding lines
4	Passive pressure pulse transmitters (PPPT)
5	Safety-relief valves (SVR)
5.2	Diaphragm pilot valves for SRV
5.3	Spring loaded pilot valves for SRV
6	Feedwater line isolation valves
7	Main steam line isolation valves (MSIV)
7.2	Diaphragm pilot valves for MSIV
8	Scram system
8.2	Diaphragm pilot valves for scram system
9	Boron shutdown system
11	Hydraulic control rod drives

External Signal for Passive Systems

5.1	Solenoid pilot valves for SRV
7.7	Solenoid pilot valves for MSIV
8.1	Solenoid pilot valves for scram system
9.1	Solenoid pilot valves for boron shutdown system

Active Systems

10	Fine motion control rod drives
12	RHR and LPCI system

Containment

13	Wetwell
14	Vent pipes
15	Flooding pool

Fig. 1.2-30 KERENA™ passive safety systems (Containement Cooling condenser)

KERENA™ after a severe accident (core meltdown event). The drywell part of the containment is flooded by the drywell flooding line. The molten core as well as the molten internal RPV structures remain in the RPV bottom. Arising decay heat is transferred into the surrounding drywell water and the developing steam ascends to the ceiling. The steam condenses at the containment cooling condencers thus the decay heat is removed out of the containment.

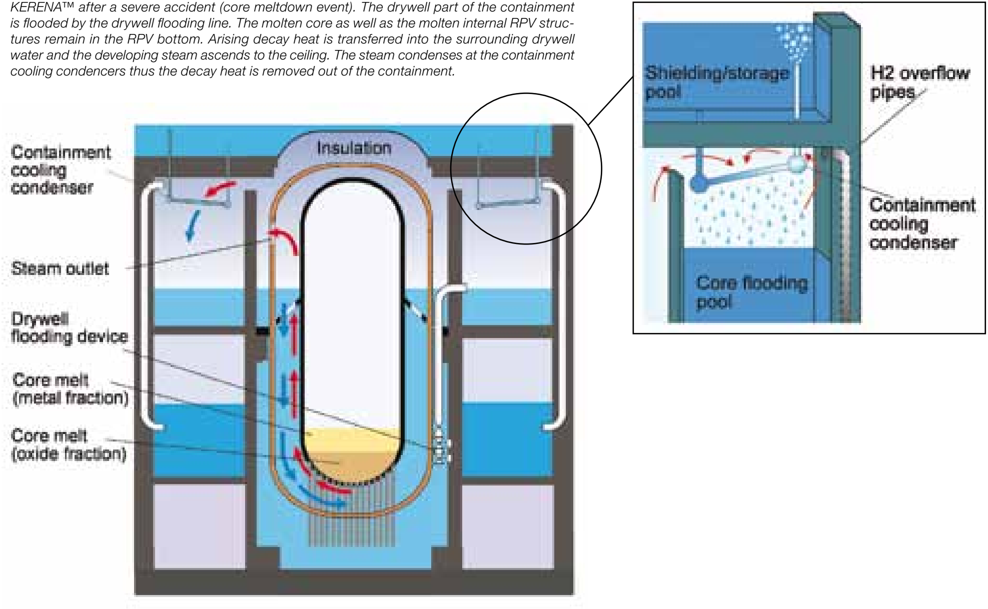


Fig. 1.2-31 KERENA™ passive safety systems (others systems)

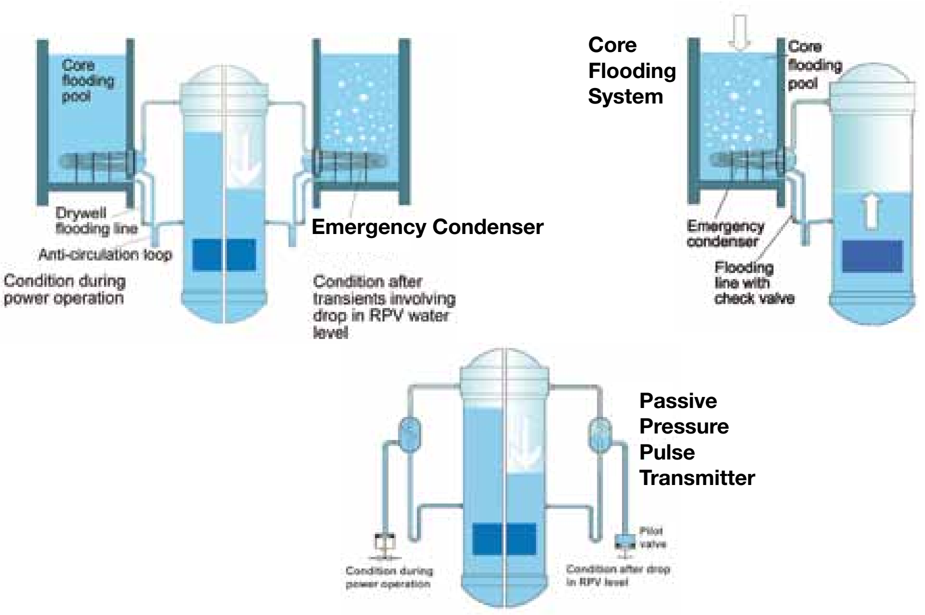
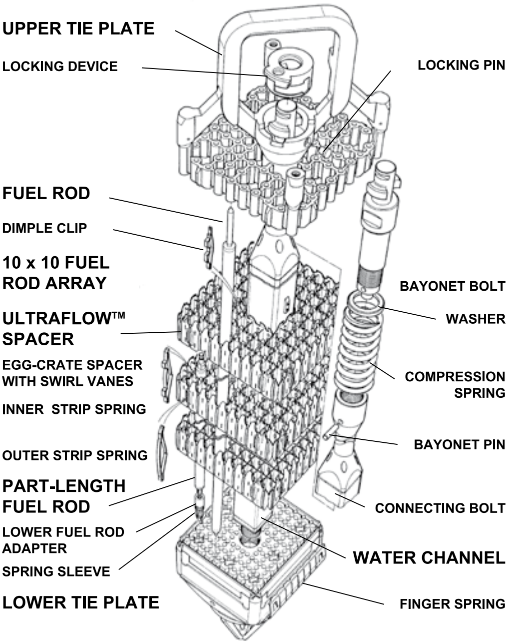


Fig. 1.2-32 ATRIUM 10 fuel assembly



1.3. History

1958 • Framatome,now AREVA NP, founded as a PWR nuclear engineering firm under Westinghouse license.

1961 • First Framatome order obtained for the 300 MWe Chooz A unit.

1964 • 300 MWe Obrigheim (PWR) plant ordered from Siemens and commissioned in October 1968. (In 1955, a branch for developing nuclear reactors - Reaktor Entwicklung - was founded by Siemens).

1967 • Chooz A commissioned.
- 662 MWe four-loop Stade plant ordered from Siemens by NWK (later PreussenElektra) and handed over in 1972.

1968 • Würgassen, Germany's first commercial BWR (670 MWe), ordered in the wake of two 250 MWe demonstration BWRs at Gundremmingen and Lingen (ordered in 1962 and 1964 respectively).
- Atucha 1 PHWR plant (335 MWe) ordered by Argentina and order booked for Atucha 2 (700 MWe).

1969 • Framatome-lead Franco-Belgian consortium receives a Franco-Belgian order for Tihange 1 (900 MWe PWR, commissioned in 1975). Framatome coordinates the overall NSSS design and supplies a number of reactor coolant system components.
- KWU created as an AEG and Siemens joint venture, merging the AEG Boiling Water Reactor (BWR) and SIEMENS Pressurized Water Reactor (PWR) technology.
- Between 1969 and 1976, eight Siemens orders received for PWRs in Germany, including the Biblis A (1200 MWe) contract in 1969. This 1200 MWe plant (the world's largest at the time) was handed over to RWE in 1975.
- 452 MWe Borssele plant ordered by the Netherlands handed over in 1973.

1970 • Order received from EDF for Fessenheim 1 and 2 (2 x 900 MWe PWRs).
- Between 1970 and 1974, orders received by AEG for six BWRs in Germany (700 to 1300 MWe).
- KWU receives order for Zwetendorf BWR plant in Austria which was decommissioned immediately before start up due to a change in domestic policy.

1971
- 900 MWe NSSSs ordered by EDF for Bugey 2 and 3.
- Order for 1000 MWe Biblis A plant in Germany. Further six orders for 1300 MWe plants (based on Biblis A) in the next five years, as well as a number of provisional orders.

1972
- Framatome becomes an NSSS manufacturer, acquiring heavy component factories at Le Creusot and Chalon-sur-Saône, France. Production begins three years later.

1973
- EDF orders a further two 900 MW PWRs (Bugey 4 and 5).
- Creation of KWU from nuclear divisions of AEG and Siemens.
- Two KWU orders for 1000 MW PWRs in Switzerland and Spain.

1974
- First "multi-unit" contract signed with EDF for sixteen 900 MW PWRs.
- Orders for two 900 MW plants, Tihange 2 and Doel 3 with Framatome as leader; plants commisioned in 1982.
- Overseas, among others, the KWU was engaged in the construction of the nuclear power plant (NPP) in Gösgen, Switzerland. From 1974 on, KWU was involved in the construction of the NPP at Bushehr (Iran), cancelled for political reasons, construction of the heavy water reactors Atucha I and Atucha II in Argentina (the latter is still uncompleted today). In addition to that they also built the Austrian commercial NPP Zwetendorf, which was decommisioned immediately before start up due to a change in domestic policy.

1975
- Multi-unit contract signed between Framatome and EDF for the first eight 1300 MWe plants.
- KWU receives oder for two 1300 MWe plants in Brazil (Angra 2) and Iran (Bushehr) (Bushehr uncompleted due to political reasons).

1976
- EDF places a second multi-unit contract with Framatome for 10 additional 900 MW PWRs.
- Turnkey contract awarded by Eskom (South Africa) to a French consortium for two 900 MW plants at Koeberg, commissioned in 1984 and 1985.
- KWU receives order for Gundremmingen B and C. Two BWR blocks, each 1300 MW, start operating in 1984.
- R&D agreement signed by Framatome, EDF, the CEA and Westinghouse and renewed in 1982.

1977
- Iran's AEOI orders two 900 MWe class units (Karun 1 & 2) from the same French consortium. This contract was cancelled in 1979 due to political upheaval in Iran.

1978 • 1400 MWe PWR Unterweser (at this time the world's largest) was handed over to the customer Nordwestdeutschen Kraftwerke AG (later Preussen Electra) by KWU.

1980 • Framatome receives EDF order for eight more 1300 MWe NSSS and the final two 900 MWe NSSS.
• Korean order for two Framatome 950 MWe units (Ulchin 1 and 2).

1981 • Framatome and Cogema pool their reactor fuel activities and set up two joint companies, Fragema (design, marketing, and sales) and CFC (manufacture of fuel assemblies).
• Framatome's licencing agreement with Westinghouse is replaced by a high-level technical cooperation agreement.

1982 • Framatome signs a contract with Britain's Central Electricity Generating Board to design the reactor pressure vessel for the Sizewell B PWR NPP.
• KWU receives orders for three PWR Konvoi units: Isar 2, Emsland and Neckarwestheim 2, built between 1982 and 1989. First German example of standardization (Konvoi).

1984 • Paluel 1 & 2, the first two 1300 MWe class units completed are connected to the grid.
• EDF 1500 MWe Chooz B 1-2 units (first in N4 series).

1985 • Takeover of the mechanical and heavy boilermaking activities of Creusot Loire.

1986 • Framatome signs contract for two 1000 MWe nuclear islands at Guangdong in China.
• CETIC, the maintenance and training center, founded with EDF in Chalon.
• KWU acquires Exxon Nuclear Company (BWR and PWR fuel) in Richland, USA.

1987 • Final unit in the EDF 900 MWe series connected to the grid (Chinon B4).
• First MOX (mixed oxide UO2 – PuO2) fuel loading at Saint Laurent B1.
• Creation of B&W Fuel Cy in the US (20% Framatome owned).
• KWU buys EXXON Nuclear Fuels in USA.
• KWU incorporated into Siemens.

1988
- PWR Konvoi unit Emsland starts commercial operation.

1989
- Agreements with Siemens and founding of Nuclear Power International for the development of new reactors and the export of PWR NSSSs and NIs.
- B&W Nuclear Services set up in the USA as an equal partnership between Framatome and Babcock and Wilcox.
- Framatome – Novatome merger (fast breeder reactor activities including Superphenix which reached full power in 1986).

1990
- Creation with Cogema of Melox. MOX fuel fabrication begins in 1995.
- First steam generator replacement for Dampierre 1.

1991
- Order from EDF for Civaux 1-2 (1500 MWe, N4 class).

1992
- Takeover of FBFC, Cerca and Cezus.

1994
- Start of steam generator replacement in ten 900 MWe units (completed in 2004).

1995
- Order received for the supply of two 1000 MWe units at Ling Ao, China, commissioned in May 2002 and January 2003.

1998
- Major digital I&C and electrical systems implemented (Teleperm XP).

1999
- Comprehensive safety upgrade of Biblis A and B begun.

2000
- Civaux 2, the last 1500 MWe N4 plant to be built in France, is connected to the grid.
- Angra 2 (Brazil) is connected to the grid.

2001
- On January 2001, Framatome and Siemens merge to form Framatome ANP, now AREVA NP. The company has its Head Office in Paris, with regional subsidiaries in Germany and the USA. Framatome has a 66% stake and Siemens 34%.
- On September 2001, CEA-Industry, COGEMA and FRAMATOME are combined to form AREVA. The group is the world leader in nuclear power and the only company to cover all industrial activities in this field.

2003
- AREVA and SIEMENS sign a contract with Finnish utility TVO for the supply of the world's first Gen III+ reactor: a 1600 Mwe turnkey EPR™ unit at Olkiluoto.

2004
- Construction of a first-of-a-kind EPR™ reactor launched at Olkiluoto.

2005
- Uddcomb, an engineering company in Sweden, is bought by AREVA.
- AREVA and Constellation Energy create a new company UniStar Nuclear, LLC, to market the EPR™ reactor in the United States.

2006
- AREVA NP and France Essor sign an agreement finalizing the purchase of SFARSTEEL - Creusot Forge.
- The nuclear entity of Jeumont SA, an AREVA NP subsidiary, becomes JSPM. It is specialized in solutions for pumps and control rod drive mechanisms used in nuclear power plants.

2007
- 100^{th} reactor order received by AREVA for the supply of an EPR™ NSSS in Flamanville, France.
- AREVA and MHI form a joint venture: ATMEA. It will develop, market, license and sell the ATMEA 1™ reactor, a new 3^{rd}-generation 1000 MWe PWR.
- Record contract with CGNPC of China for the supply of two EPR™ Nuclear Islands, on the Taishan site, and a number of fuel reloads.

2

OPERATION OF PWR NUCLEAR STEAM SUPPLY SYSTEMS

2.1. Energy release

2.1.1. Nuclear fission energy

The energy supplied by a single fission is close to 200 MeV depending on the fissile nucleus type. It is delivered through the kinetic energy of the fission products which are formed during fissile nucleus fission and are slowed down inside the fuel matter in roughly 1 x 10^{-2} mm. The corresponding mass default is very small since it is equal to one micro-gram per day.

The slowing down of the fission products generates a large amount of heat which is extracted from the core by the reactor coolant and transformed into steam inside the steam generators.

2.1.2. Energy equivalents of fission

1 joule	3.121×10^{10} fissions
1 fission ~ ~	3.204×10^{-11} joules (or watt.s) 8.906×10^{-18} kWh thermal 3.711×10^{-22} MWd
1 kWh (thermal) ~ ~	1.124×10^{17} fissions 4.384×10^{-5} g of fissioned ^{235}U 5.23×10^{-5} g of ^{235}U destroyed
1 MWd (megawatt-day) ~ ~	2.697×10^{21} fissions 1.052 g of fissioned ^{235}U 1.25 g of ^{235}U destroyed

3.1×10^{10} fissions/s ~ 1 watt = 624×10^{10} MeV/s

Complete fission of 1 g of ^{235}U ~ 23×10^{3} kWh
$\cong$ 1 MWd

2

PRINCIPAL FISSION PRODUCTS

Mass	Atomic No.	Element	Fission Yield %	Half-life	Radiant Energy (MeV)	
					β	γ
85	36	Krypton	0.3	10.3 years	0.7	-
89	38	Strontium	4.7	53 days	1.5	-
90		Strontium	5.8	28 years	0.6	-
90	39	Yttrium(1)	-	64 hours	2.2	-
91		Yttrium	5.8	60 days	1.5	-
95	40	Zirconium	6.3	63 days	0.4	0.7
95	41	Niobium(1)	-	35 days	0.16	0.75
99	42	Molybdenum	6.1	67 hours	1.2	0.04-0.08
99	43	Technetium	-	200,000 years	0.3	-
103	44	Ruthenium	2.9	40 days	0.22	0.5
106		Ruthenium	0.4	1 year	0.04	-
106	45	Rhodium	-	30 seconds	3.5	Up to 0.5

(1) Isotopes followed by (1) are the direct descendants, by β emission, of the isotope just above them in the table. The "mother" and the "daughter" isotopes thus have the same mass.

(2) ^{120}Te and ^{137}Ba are not β radioactive. They are transformed by internal rearrangement of the nucleus, with γ emission.

Mass	Atomic No.	Element	Fission Yield %	Half-life	Radiant Energy (MeV)	
					β	γ
129	52	Tellurium [(2)]	0.3	33 days	-	0.11
131	53	Iodine	3.0	8 days	0.6	0.36
133	54	Xenon	6.5	5.3 days	0.34	0.08
137	55	Cesium	6.2	30 years	0.5	-
137	56	Barium[(1)(2)]	-	2.6 minutes	-	0.66
140		Barium	6.3	12.8 days	1.0	0.16-0.5
140	57	Lanthanum[(1)]	-	40 hours	0.8-2.2	0.1-2.5
141	58	Cerium	5.7	33 days	0.4-0.6	0.14
144		Cerium	6.0	284 days	0.17-0.3	0.03-0.13
144	59	Praseodymium[(1)]	-	17.5 minutes	3.0	0.7-2.2
143		Praseodymium	6.2	13.7 days	0.9	-
147	60	Neodymium	2.6	11.3 days	0.4-0.8	0.1-0.5
147	61	Promethium	-	2.6 years	0.22	-

Fig 2.1-1 Relations between transuranic elements

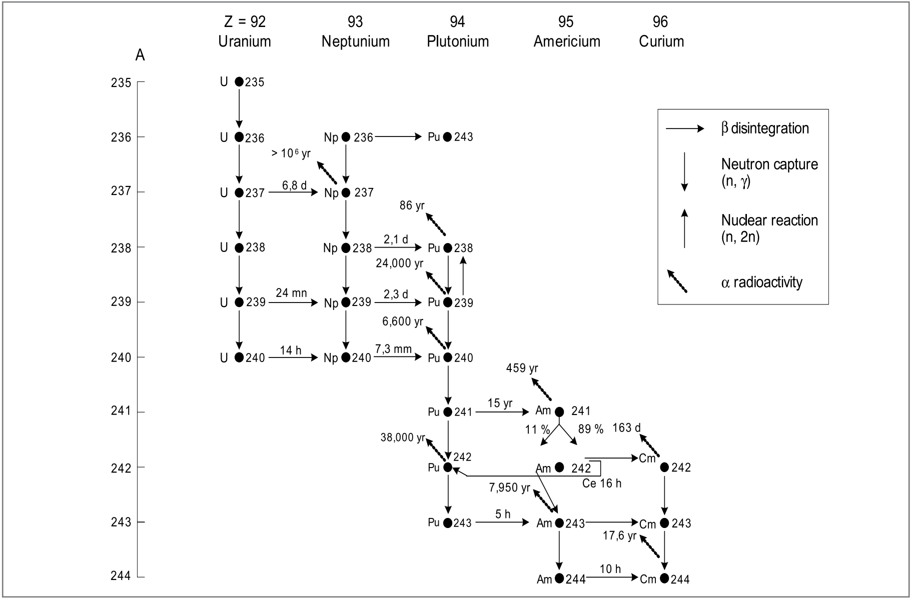
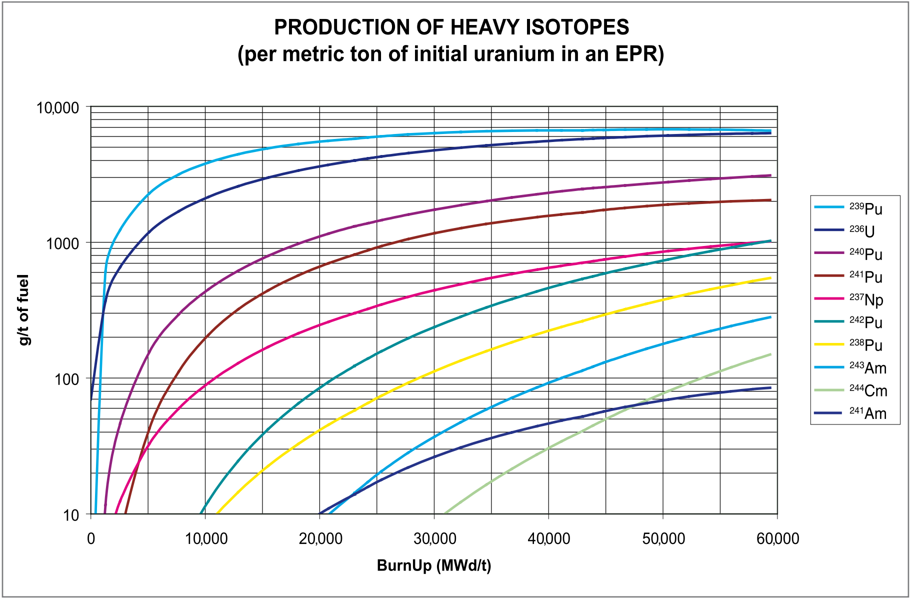


Fig 2.1-2 Production of Heavy Isotopes



For each metric ton of uranium initially in the nuclear fuel, during reactor operation the 41 kg of ^{235}U consumed (enrichment reduced from 5% to 0.9%) as well as the 42 kg of ^{238}U consumed are converted into 61 kg of various fission products, to which is added a mixture of heavy isotopes.

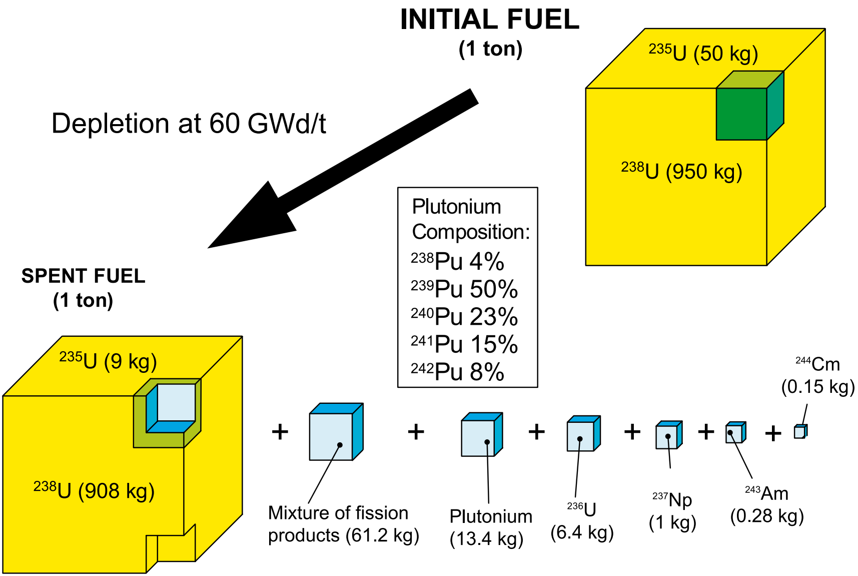
CONTENT AND SPECIFIC ACTIVITY OF THE HEAVY ISOTOPES FOR A DISCHARGE BURN-UP EQUAL TO 60 GWd/T

Element	Isotope	Content (g/ton)	Half-life (years)		Specific activity (Bq/g)
			α	β	
Np	^{237}Np	1015	2.14×10^6		2.61×10^7
Pu	^{238}Pu	546	87.7		6.34×10^{11}
	^{239}Pu	6635	2.41×10^4		2.29×10^9
	^{240}Pu	3107	6.56×10^3		8.51×10^9
	^{241}Pu	2044	6.00×10^5	14.4	3.83×10^{12}
	^{242}Pu	1023	3.74×10^5		1.47×10^8
Am	^{241}Am	~85	433		1.27×10^{11}
	^{243}Am	281	7.37×10^3		7.40×10^9
Cm	^{244}Cm	149	18.1		2.99×10^{12}

If the irradiated fuel is cooled for one year outside the reactor before reprocessing, about 40 g of ^{241}Am per metric ton of initial fuel is created from the ^{241}Pu.

NOTE: The residual ^{235}U content of the spent uranium is still higher than in natural uranium. Recovered at the outlet of the reprocessing plant and reinjected into the enrichment installation, this uranium contributes to reducing the consumption of natural uranium by about 20%.

Fig 2.1-3 Consumption of PWR Fuel



2.1.3. Residual power after shutdown

The residual power after reactor shutdown is the sum of three contributions:

1) Residual fissions (due to delayed neutrons), which depend on the negative reactivity introduced. The influence of this factor quickly becomes negligible.

2) Heavy nuclei: especially ^{239}U and ^{239}Np in a UO_2 core at the decay times below (their highest contribution is 20%).

3) Fission products: after several tens of seconds, these are responsible for the preponderant part of the residual power.

Indicative values in a UO_2 core							
Time t after shutdown		10 sec	1 min	1 hour	1 day	1 month	1 year
Contribution	Fissions	7.87	2.95	0.00	0.00	0.00	0.00
	Heavy nuclei	0.33	0.33	0.19	0.12	0.01	0.00
	Fission products	4.95	3.57	1.24	0.48	0.15	0.03
Total $P/Po \times 10^{-3}$		13.15	6.85	1.43	0.60	0.16	0.03

P=The total power emitted at time t after shutdown.

Po=The total power, reactor in operation.

All the values indicated in this table include an uncertainty factor equal to 1.645 σ

Reminder ▶ with the reactor in operation, $P\gamma_{\text{fission}} \cong 40 \times 10^{10}$ MeV/s per Watt.
$P\gamma_{\text{total}} \cong 65 \times 10^{10}$ MeV/s per Watt.

2.2. Thermal-hydraulics: production of energy in the core

The energy produced in a nuclear steam supply system (NSSS) results from the release of heat due to fission reactions in the assembly fuel rods. The reactor coolant that flows around these rods evacuates this heat energy and transfers it to the steam generator(s).

The heat extracted from the core by the reactor coolant is characterized by a certain number of magnitudes. For a 1600 MWe class NSSS, these values are:

Rated Core power	Q = 4590 MWth
Mass flow rate	m = 83,380 ton/h
Increase in enthalpy	H = 210 kJ/kg
Average power: - of a fuel assembly - of one fuel rod	 = 19 MW = 72 kW
Average power density (related to fuel rods) - linear, q' - surface, q" - volume, q"'	 ≅ 166 W/cm ≅ 57 W/cm^2 ≅ 241 W/cm^3

2.2.1. Neutronics

The distribution of the power released in the core is not uniform and large disparities exist between the different zones, resulting from the non-uniform distribution of neutron flux in the core due to nuclear physics.

To recap, neutron flux is expressed in n/cm^2.s and is related at each point in the fuel to the thermal power released, by the equation:

$$\boxed{q''' = \Sigma_f \, \phi_{th} \times 3.2 \times 10^{-11}}$$

where Σ_f is the effective fission cross-section in cm^{-1} and ϕ_{th} is the thermal flux in n/cm^2.s.

2

SHAPE OF THE NEUTRON FLUX

In considering the form of the neutron flux, one distinguishes between 1) the radial flux distribution and 2) the axial flux distribution.

1) Radial Flux Distribution

For a homogeneous reactor (a theoretical case), the radial flux distribution is a Bessel function J0 (shape 1). Because of the presence of a neutron reflector (the water surrounding the core or a metallic structure installed on purpose in the case of the EPR ™ reactor), and because of different fuel-enrichment zones loaded according to a certain fuel loading pattern, the flux distribution as a function of the radius is in reality more complex than the Bessel function as shown in the accompanying figure. According to the fuel management strategies used in the 80s and the fuel loading patterns which followed on from these, the most enriched fuel assemblies were loaded at the periphery of the core and the lowest at the centre in order to get a flattened power distribution (shape 2). Modern fuel management strategies currently used require that the highest fuel-enrichment zone be loaded at the centre of the core and the lowest at the periphery in order to decrease the amplitude of neutron leakage and then increase the initial core reactivity and consequently the fuel cycle length (shape 3). Obviously, the immediate consequence is an increase in the core peaking factors. The safety analysis must be reviewed to demonstrate that the safety margins are still acceptable.

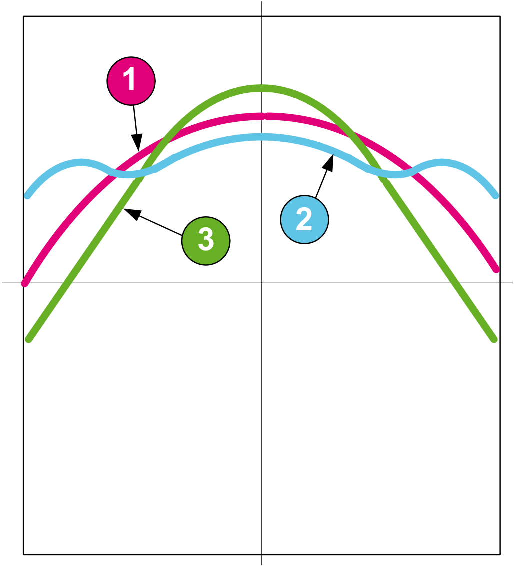


Fig 2.2-1 Axial flux distribution

2) Axial Flux Distribution

When the core is homogenous and at the beginning of life (BOL) and zero power (shape 1), the curve of flux distribution has the shape of a cosine function.

The reactor power level increasing to full power, the coolant temperature being higher towards the top of the core while being roughly constant towards the bottom (according to the core average temperature variation versus the power level), neutron moderation is more effective towards the bottom of the core where the water density is the highest (less effective towards the top where the water density is the lowest). This leads to an axial gradient of reactivity that induces a slight bulge in the axial flux distribution towards the base of the core (shape 2).

After a few months of operation at full power, the fuel is burned up faster in the region where the neutron flux is higher i.e. a little bit lower than the mid-axis. On the other hand, the fuel is burned up more slowly at the axial core extremities. The consequence is a so-called "double-humped" axial curve (shape 3). This is typically the case at the end of the fuel cycle (EOL).

The shapes of the flux curves shown in the figure are based on the assumption that all of the control rods have been withdrawn.

The combination of the radial and axial flux shapes creates a "hot point" in the reactor where the neutron flux (and thus the thermal flux) is at a maximum.

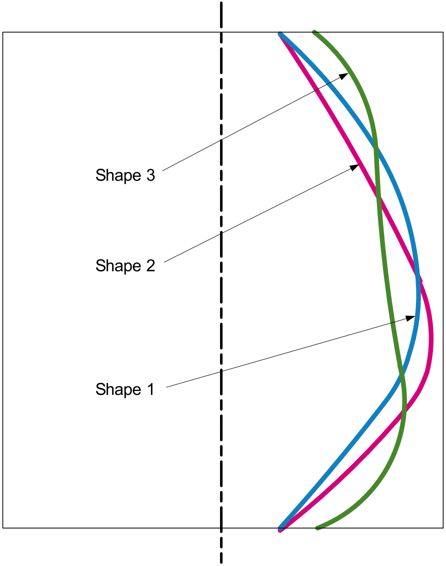


Fig 2.2-2 Axial flux distribution

As indicated by the figure on the facing page, insertion of the control rods modifies the shape of the axial flux distribution curve, by moving the position of maximum flux clearly into the lower half of the core. This movement is accompanied by an increase in the value of the maximum neutron flux in the same region.
It is therefore necessary, as we will see later on, to verify that the thermal flux at this point does not exceed certain limiting values.

This situation also makes it necessary to operate at normal rated reactor power with minimum insertion of the control rods in the core, which allows more homogenous fuel burn-up and ensures a greater negative reactivity reserve in case of reactor trip. Nevertheless, the fast reactivity variations, related to rapid changes in the power level must be compensated for by the control rod clusters. That means a compromise must be looked for and a “certain” insertion is acceptable.

To be able to keep the control rod clusters at the position indicated in the figure, which is in the so-called “reference” zone, there must be another means of reactor control available: soluble boron. This element is present in the form of boric acid, diluted in the reactor coolant. Thus, slow reactivity variations, in particular those related to progressive burn-up of the fuel, shift over from cold shutdown to hot shutdown, and scheduled slow reactor power changes are compensated for by causing the boron concentration to vary correspondingly. This adjustment is quite slow (some 10 minutes is necessary to change the boron concentration) but homogenous i.e. does not influence the power distribution. The drawback relates to the quantity of liquid effluents produced at each change in the boron concentration of the reactor coolant.
Immediately after the first power escalation after reactor refueling, at full power, the boron concentration is approximately 1000 ppm according to the fuel management strategy and the fuel loading pattern. Normal fuel burn-up reduces the concentration by 3 to 4 ppm per day.

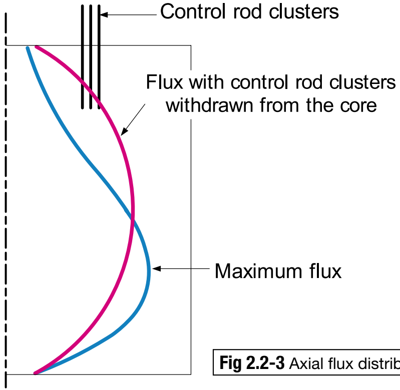


Fig 2.2-3 Axial flux distribution (2)

THE XENON EFFECT

Some of the fission products formed strongly capture free neutrons and are known as “poisons”. This is particularly the case of xenon 135, which can cause considerable variations in the core reactivity within the space of a few hours, thus leading to difficulties in controlling the reactor power (overall effect) and in controlling the power distribution (local effect).

The *xenon effect* is responsible for the following phenomena:

- **Startup:** If one starts the reactor from a situation where no xenon is present (after a shutdown lasting for several days), the progressive formation of ^{135}Xe makes it necessary to withdraw the control rod clusters or to reduce the concentration of soluble boron in order to maintain the desired neutron flux or power level, until the xenon has attained its equilibrium concentration.
- **Power increase:** Starting from a situation where the xenon is in a state of equilibrium, it is necessary to temporarily increase the boron concentration or to insert the control rod clusters, whereas power increase is obtained by the opposite movement.
- **Power decrease:** Starting from a situation in which the xenon is in equilibrium, one obtains an opposite effect from that mentioned just above, but one that is much more pronounced. This is called the xenon peak. After a reactor trip, it can limit the return to criticality and then to full reactor power for periods of several hours.
- **Stable power distribution:** The production of xenon depends upon the disappearance of iodine 135, the local quantity of xenon depends on the evolution of the flux at the point under consideration. The distribution of xenon in the core influences the power distribution and can cause instabilities in the latter. The figure below shows the evolution of the negative reactivity due to xenon over time, as a function of the power level variations (PN stands for the nominal reactor power level).

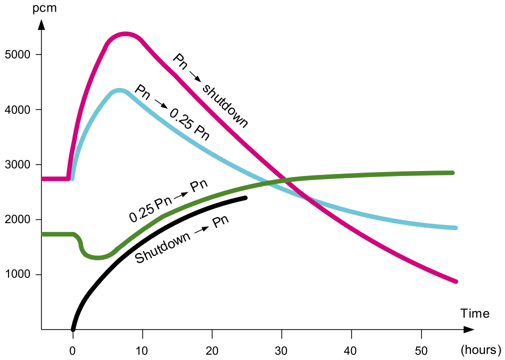


Fig 2.2-4 Evolution of xenon negative reactivity

2

AXIAL OFFSET

The theoretical distribution of neutron flux (or of the resulting thermal power) is determined by core physics computations for the various reactor operating modes (normal or accident), taking into account the fuel burn-up and subsequent refueling. As a first approximation, it can be assumed that the flux distribution along the core axis (z) is the same at every (x,y) position. This average axial flux distribution (or axial flux shape) depends on a certain number of factors, such as the power level, the control rod positions and the fuel burn-up.

The axial flux shapes can be symmetrical (cosine form, for example), offset towards the bottom (control rods inserted), or offset towards the top (end of cycle, after control rod cluster withdrawal), as shown in the figure.

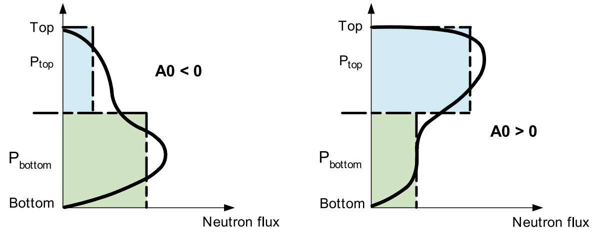


Fig 2.2-5 Axial offset

The flux shape is characterized by its relative axial power unbalance or "axial offset" (AO), the value of which plays an important role in the core design:

$$\mathbf{AO} = \frac{P_{Top} - P_{Bottom}}{(P_{Top} + P_{Bottom})} \times 100\ (\%)$$

where Ptop and Pbottom represent the average power of the upper and lower halves of the core. The axial offset value is expressed in % i.e. without dimension.

The average power of the upper and lower halves of the core is measured by the currents supplied by the flux-measurement ionization "half chambers" (1) located outside the pressure vessel.

(1) The "Axial Flux Difference" or AF Dis also used with:

$$AFD = \frac{P_{Top} - P_{Bottom}}{(P_{Top} + P_{Bottom})_{nominal}}$$ expressed in % of Nominal power.

Thus AFD = AO x Pr where Pr is the relative power expressed in % of nominal power.

2.2.2. Transfer of energy from the fuel to the coolant

TEMPERATURE DISTRIBUTION IN A FUEL ROD

Generated in the uranium oxide, the heat must pass through four regions to reach the coolant and be removed (see the accompanying figure):

I) The central region, consisting of the UO_2 fuel pellet with its low thermal conductivity (~ 2W/m/°C), strongly resists passage of the heat. The result is a large temperature difference between the center of the pellet and the periphery, which is given by the equation:

$$\boxed{\Delta T\ UO_2 = \frac{q''' \times a^2}{4 \times \lambda}}$$

Where:

q''' = the heat flux per unit of fuel volume (W/cm^3)
λ = the thermal conductivity, and
a = the fuel pellet radius

II) The second region through which the heat must pass is the helium film in the gap between the fuel pellet and the inner surface of the fuel cladding.

III) The third region consists of the cladding, which has high heat conductivity.

IV) The fourth region is the thin contact zone (or film) between the fuel cladding and the coolant. This zone is characterized by a heat exchange coefficient that depends largely on the flow regime, the fluid enthalpy, and the calorie flux. This very thin film has physical properties that differ from those of the average fluid in the hydraulic channel.

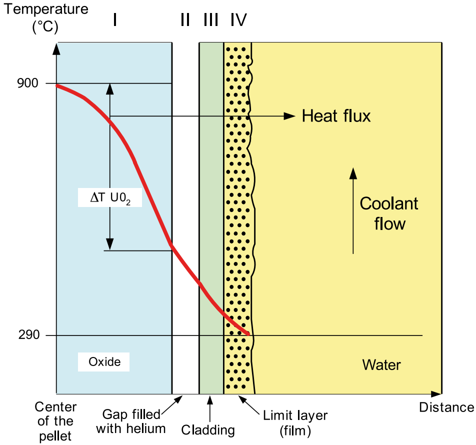


Fig 2.2-6 Temperature distribution in a fuel rod

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The temperature difference ΔT across the film is related to the heat exchange coefficient h and the thermal flux q" by the equation:

$$\Delta T \text{ film} = \frac{q''}{h}$$

$$\text{or } T \text{ cladding} = T \text{ fluid} + \frac{q''}{h}$$

In single-phase forced convection, this gives:
$q'' \cong 600{,}000\ W/m^2$, $h \cong 40{,}000\ W/m^2/K$,
or: ΔT film $\cong$ 15°C.

The thermal flux and the temperature of the fluid being fixed, it can be seen that the temperature of the cladding depends on the heat exchange coefficient h, as the figure on the previous page shows.

PRINCIPLE OF HEAT EXCHANGE ALONG AN ISOLATED AND HEATED CHANNEL

The heat exchange coefficient depends on the local properties of coolant flow, which evolve all along the hot channel. This coefficient is characterized by its $F\Delta h$ and the coolant by:

- Ti = the inlet temperature
- g = the mass flow rate
- T_{sat} = the saturation temperature
- X = the quality = steam mass/mixture mass, and
- α = the void fraction = steam volume/mixture volume.

A first assumption is made that the channel is isolated and exchanges neither mass nor energy with the neighboring channels. This hypothesis, in fact highly penalizing, is not verified in a real PWR.

As the coolant rises along the channel (see figure 2.2-7), its physical properties are modified, because its temperature increases, along with the temperature of the channel wall. The height of the channel can be divided into a certain number of zones with different properties:

(1) A lower zone, in which the wall temperature and the coolant temperature are below the saturation temperature. In this zone, the flow is single phase and the heat exchange regime is one of forced convection. The heat exchange between the cladding and the coolant is good and the temperature difference ΔT remains small, not exceeding several tens of degrees.

(2) Starting from a certain length of tube, the wall temperature exceeds the coolant's saturation temperature, T_{sat}, whereas the coolant remains at a temperature less than T_{sat}. Bubbles then begin to appear along the cladding wall, while the coolant remains strongly under saturation. These bubbles improve the thermal exchange, because they do not remain stuck to the wall, but are carried along by the coolant flow. Consequently, they transmit calories from the wall to the coolant.

(3) Since the coolant continues to heat up, the density and the size of the bubbles increase. Suddenly, there is a coalescence of the bubbles and the creation of a stable vapor film along the cladding wall. From this moment on, the heat exchange degenerates (h decreases, thus $T_{cladding}$ increases).

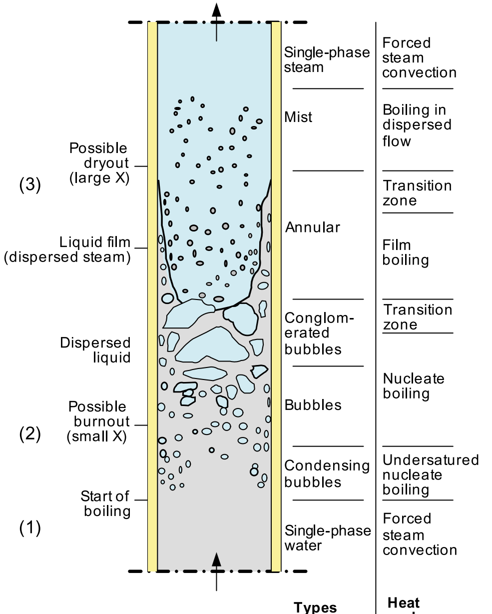


Fig 2.2-7 Boiling phenomenon

This degraded heat exchange is explained, among other reasons, by the fact that steam has a lower thermal conductivity than water. It occurs when a certain value of thermal flux has been reached, and leads to "Departure from Nucleate Boiling" (DNB) and burnout.

BURNOUT

- Zone 1-2 corresponds to the **forced convection** regime.
- Zone 2-3 corresponds to the **nucleate boiling** regime, which is interesting because the steep slope (large heat-exchange coefficient) shows that a considerable amount of heat flux is extracted by means of a small increase in the wall temperature.
- Starting at point 3 on the curve, which is the **critical temperature**, the exchange of heat becomes less efficient. Although heat continues to be removed, the temperature rises quickly, which risks damaging the cladding material (as well as the oxide fuel pellets). Furthermore, this zone is unstable, with oscillations of the flow rate and the temperature. The regime in this zone is called **transition boiling** (region 3).
- Beyond point 4, stable film boiling is achieved, with forced convection in a single-phase vapor flow. The value q"c corresponding to point 3 is thus a threshold value, beyond which there is a risk of dangerously high wall temperatures. This is therefore called **the critical flux**. Theoretically, this is a limiting value, never to be exceeded under normal operating conditions. As we will see later, however, in PWRs a rule of always remaining well below this critical value is imposed.

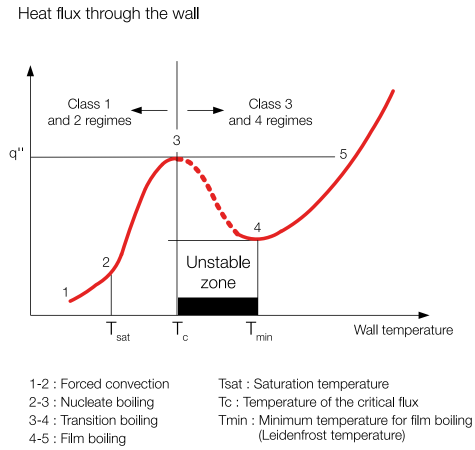


Fig 2.2-8 Burnout

2.2.3. Reactor control

The cores of PWR units are:

- stable with respect to the xenon effect, from the radial viewpoint,
- generally unstable with respect to the xenon effect, from the axial viewpoint.

Reactor control consists in using the means available to affect the reactivity (boron concentration, control rod cluster insertion or withdrawal), to adjust the reactor power level in accordance with the required grid characteristics, while respecting the criteria that define reactor safety. Because of the reactor's stability, there is no need for radial control. Simply, the control rod positions have been determined so that at high power levels they disturb the radial flux distribution as little as possible. So as to avoid excessive axial flux peaks, which would be penalizing for the reactor from the point of view of safety, the control rod movement on the axial plane is adapted to control not only the power but also the axial power distribution required by the axial offset control.

Several types of reactor control are used in AREVA NP reactors:

- **Mode A:** In this mode, the core reactivity is controlled by controlling the reactor coolant temperature. Four standard control banks called D, C, B, A, of which the reactivity value is close to - 1000pcm are slaved to the reactor coolant temperature. Constant overlaps exist between D and C, C and B and B and A. But the necessity to keep the AFD value inside a + or - 5% band requires that the position of the D control bank be adjusted by varying the boron concentration in the reactor coolant. This boron concentration variation induces a reactor coolant temperature that is detected by the temperature control channel which provokes the right control bank motion.

The xenon evolutions are compensated for by adjusting the boron concentration; simultaneously, the D bank must be kept at the suitable insertion for controlling the AFD value in the +/- 5% band (actual boron concentration variation is the sum of both effects). The six French CP0 900 MW class units are operated with the A mode.

The D, C, B, A bank insertion must meet the Insertion limit threshold which is a function of the power level. That guarantees the right reactor trip reactivity value known as the “Shutdown Margin”.

This control mode is most suitable for base-load unit operation and slow load changes. Fast load changes are limited by the boron dilution rate, which is slow, especially near the end of the fuel cycle.

• **Mode G:** This mode uses the equipment known as the RAMP (Reactor Advanced Maneuverability Package). The reactivity effects related to the changes in power level are totally compensated for by control banks ranked in four banks of progressively greater reactivity worth, which are inserted into the core in accordance with a predetermined program (calibration), as a function of the power required by the turbine. These banks are called "power-compensation banks". The corresponding control channel is of the open loop type.

This makes possible fast load changes and provides a fast return to full power without notice at any time during the low power period of load-following transients.

The power compensating banks are composed of four banks G1, G2, N1, N2 of which the reactivity value is respectively close to 250, 400, 1000, 1000 pcm. The first two banks are made of stainless steel absorbing pins which are less absorbent than the standard pin made of an alloy of Silver-Indium-Cadmium. This is the reason why they are called "grey" banks as opposed to the standard control banks which are called "black" banks. The last two banks N1 and N2 are standard. When reducing the power level from the nominal value, G1 is the first bank to be inserted followed by G2, N1 and N2 according to optimized constant overlaps in such a way that a small disturbance of the axial power distribution (axial-offset) is introduced while the power level is still high. When increasing the power level from zero, N2 is the first bank to be withdrawn.

A different bank which is called "R" is slaved to the average reactor coolant temperature in a closed-loop control channel; it is used to finely adjust the reactor coolant temperature and then the core reactivity when residual reactivity effects may occur. The R bank is standard i.e. a "black bank" of which the reactivity value is close to 1000 pcm. The R bank is also used to control the AFD: as with the A mode, R bank motions are induced by using adequate boron concentration variations. A boron concentration decrease (dilution) will induce an increase in the average coolant temperature and then R bank insertion. Conversely, a boron concentration increase (boration) will lead to R bank withdrawal.

The R bank insertion must meet the insertion limit threshold which guarantees the right reactor trip reactivity value known as the "Shutdown Margin".

The xenon evolutions are compensated for by adjusting the boron concentration in the reactor coolant. The French CP1 and CP2 900 MWe class units, as well as the 1300 MWe class units, are equipped with the RAMP as well as the four units at Guangdong and Ling Ao in the People's Republic of China.

• **Mode X:** Mode X uses an even more advanced equipment package, called the DMAX for providing an automatic control of both average reactor coolant temperature and axial-offset at the same time. For this purpose the mode X permanently adjusts the insertion of five different banks X1, X2, X3, X4, X5 and their respective overlaps which means the risk of generating xenon axial oscillations does not exist while the global reactivity of the core is controlled whatever the grid demand characteristics. The reactivity value of the five Xi banks is ranked following the same principles as mode G. Unlike the other core control modes, mode X favors rather deeper bank insertion (mainly X1) at high power levels. It is planned to permute the X1 bank with one of the two sub-banks which compose the X3 bank every two weeks in order to manage the burn-up shadowing effect induced by insertion of the XI bank (interchange sequence procedure).

The mode X provides Utilities with a high operating flexibility since Utilities can choose either a return to full power without notice strategy or a liquid effluent saving strategy according to their own daily needs. The control room operator is still responsible for the boration/dilution operations by following the information provided by a specific computer called the "Shutdown Margin Computer". Since the overlaps between the Xi control banks are not constant, it is not possible to predict their respective insertion which is a consequence of the control channel response to the power and xenon variations which cannot be anticipated.

As a consequence the Insertion Limit concept will be much too conservative and cannot be used. This is the reason why a specific computer was developed and installed. The SDM algorithm is based on the "perturbation theory". It is able to calculate the reactivity value corresponding to the current position of the Xi banks and to infer the SDM value.

The DMAX system is available on the four N4 units (4 loops, 1450 MWe units).

2

• **Mode T [R] :** The core control mode developed for the EPR ™ reactor is a fully automatic control mode. It will provide automatic control of three variables at the same time: the coolant temperature, the axial-offset and the return to power capacity. In other words, the boration/dilution actions are automatically performed by a third control channel fitted into both temperature and axial-offset control channels. In case of a grid demand, the maximum power level to be reached at any time during the low power period of a load-follow transient is now chosen by the operator by setting the corresponding set-point value in the control room. The EPR ™ reactor control mode uses five banks called P1 to P5. All of them are made of standard absorbing pins. The Pi bank reactivity value is respectively close to 500, 500, 1000, 1000, 1000 pcm. It has to be noted that the P1 and P2 banks are composed of four RCCAs instead of eight.

To illustrate the bank motion logic we can take the following example. At full power the P1 bank is inserted at 60 cm for example; P2, P3, P4, and P5 are inserted at the same altitude close to 20 cm and constitute a solid bank which is considered as a "heavy" bank called "H" (H stands for heavy). The overlap between the P1 and the H banks is variable in order to control the axial-offset parameter. When decreasing the power, the P1, P2 and possibly the P3 banks are inserted in the core. The H bank is then composed of P4 and P5 and still participates in axial-offset control according to the coolant temperature decrease. The P1/P2 and P2/P3 overlaps are constant. The overlap between P3 and H is variable. If the operator selects the return to full power strategy a boron dilution is initiated for xenon build up in order to keep the P1, P2, P3 inserted at the altitude which compensates for the power defect. If the operator selects the liquid waste optimization strategy the xenon build up is compensated for by withdrawing the P1, P2, P3 banks while the axial-offset is simultaneously controlled by the H bank.

When increasing the power level up to the nominal value, the P3 and P2 banks are withdrawn up to the P4 and P5 altitude to constitute the H bank again. P1 remains slightly inserted in the core. Nevertheless, an interchange sequence procedure between P1 and another Pi sub-bank is planned to manage the burn-up shadowing effect due to P1 bank insertion.

2.2.4. Core safety limits

The fundamental objective of the core design is to establish the conditions under which the maximum heat can be extracted, while remaining within the limits that are set by the mechanical properties of the materials used. These limits are defined by a certain number of safety rules drawn up by the designer and the competent safety authorities.

The conditions to be respected (or "criteria") vary depending on the class of the operating regime (see Chapter 4.B). The main one is: no damage to the fuel cladding in the hottest channel for any Class 1 or 2 operating conditions. A certain limited deterioration is admitted (for example, a fraction of the fuel pins experiencing DNB) for Class 3 or 4, but the maintenance of the heat-exchange surface and of the coolant passage cross section, as well as the free insertion of the control rod clusters, must, for the most part, be assured.

NO MELTING AT THE CENTER OF THE FUEL PELLETS

This first operating limit is intended to prevent any cladding damage. The temperature at the center of the UO_2 fuel pellets is limited to 2260°C, which corresponds to a linear power density at the hot spot of 590 W/cm. It thus corresponds to a limit on the maximum power level of the NSSS, taking into account the distribution of power in the reactor core. A dedicated protection channel ensures that this limit is not exceeded.

SATURATION LIMIT

Another operating limit is imposed by the enthalpy at the hot channel outlet. It is necessary to prevent boiling at this outlet, because the measurement of coolant heating during its passage through the core equal to the difference between the outlet temperature and the inlet temperature or ΔT, would then no longer be representative of the power. Furthermore, the correlation used to forecast the critical flux (see below) would no longer be valid.

The temperature at the hot channel outlet is given by the equation:

$$\boxed{T_o = T_{in} + \frac{P}{G\,ACp}}$$

Where:

P = the hot channel power
G = the mass flow rate
A = the channel flow cross section, and
Cp = the specific heat of the coolant

During operation at rated power, the evolutions of the different temperatures in the average channel and the hot channel have the shapes indicated in the figure below.

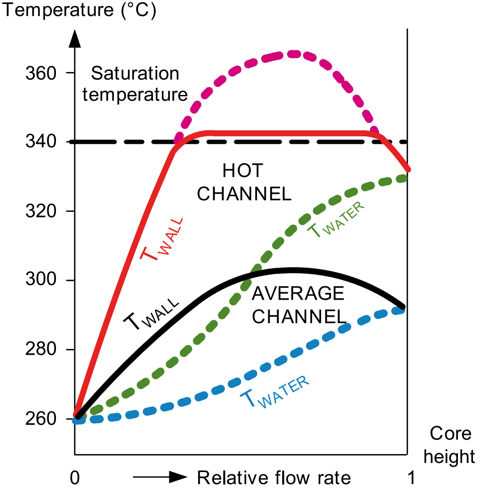


Fig 2.2-9 Saturation limit

NON-APPEARANCE OF BURNOUT (DNB LIMIT)

The aim is to avoid any damage to the fuel cladding due to an excessive increase in temperature. We have seen that the heat flux q" must not exceed a certain value q"c (critical-flux). For this, a parameter that plays a fundamental role is used, the **DNBR (Departure from Nucleate Boiling Ratio)**, which is equal to the ratio of the critical flux to the real flux at any time:

$$\boxed{\mathbf{DNBR} = \frac{q''c}{q}}$$

The critical flux is experimentally determined. Correlations have been established that allow, for a given channel, the critical flux q"c to be determined as a function of the flow characteristics (pressure, flow rate, inlet and outlet enthalpies, quality, etc.) and the geometrical characteristics of the channel (hydraulic diameter, grid spacing and type). These correlations make it possible to predict the critical flux, under the given conditions, within an uncertainty margin. For the correlation used by AREVA NP, this uncertainty margin leads to never dropping below a given DNBR value (typically 1.30, see the figure), which corresponds to a critical flux of about 185 W/cm^2. The result for the reactor protection system is a limitation on the power level, the pressure, the temperature, and the coolant flow rate in the core.

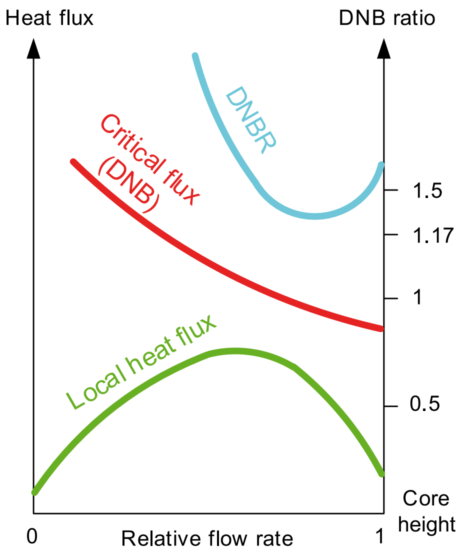


Fig 2.2-10 Non-appearance of burnout (DNB limit)

OTHER CRITERION

Let us also note the limits on the cladding stress, during class 2 power transients likely to cause the cladding to crack (consequence of the fuel pellet-cladding mechanical interaction).

Operating feedback showed this phenomenon may occur when the power quickly increases after a long period of operation at low or zero power. The French Safety Authorities impose a check if the “overpower ΔT” or the “High Linear Heat Rate” protection channels (see below) are able to avoid the risk of pellet clad interaction during class 2 accidents. To do so, complex calculations mixing nuclear and thermo-mechanical aspects were performed and have shown that it will be the case if the Extended Low Power Operation periods are limited. Acceptable durations and sequences are indicated in the **Operating Technical Specifications** (OTS).

Regarding the Loss Of Coolant Accident (LOCA) which corresponds to breaks of different areas affecting the primary circuit pipes, the main associated criteria are the fuel cladding temperature which must remain below 1200°C and the fuel cladding oxidation rate which must be lower than 17%. The corresponding calculations show that these criteria are met provided that the maximum Linear Heat Rate before the accident expressed in W/cm is lower than a certain value close to 400 W/cm (typical value). It is the operator’s role to meet this value during operation by controlling the axial power distribution in the core through the Axial Flux Difference control and by keeping it inside the “Class 1 Operating Domain”.

THE CORRESPONDING CORE PROTECTION CHANNELS

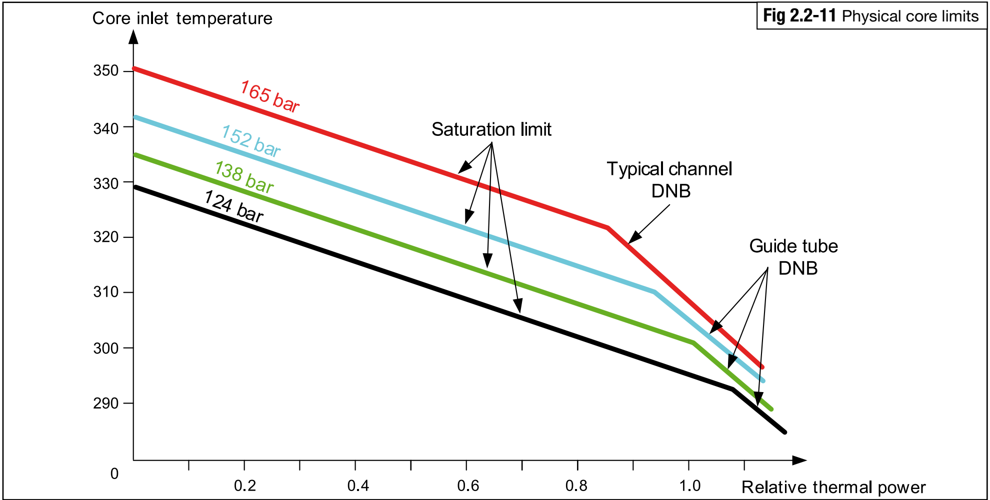
The above limits are never reached during normal operation but might be during accidents. In order to avoid fuel cladding damage due to the above physical phenomena during accidents, two protection channels were designed and installed to trip the reactor before the safety criteria are met.

On the 900 MW 3-loop reactors, these protection channels are called “Overpower and Overtemperature ΔT channels” dealing respectively with the “center fuel melting” and “DNB” risks. By means of analog circuits, these channels elaborate the maximum acceptable power level or ΔT_A as a function of different parameters such as the inlet temperature, the primary flow rate, the primary pressure and the axial Flux Difference. If the ΔT_A becomes higher than the measured ΔT_M a reactor trip signal is emitted, causing all the control banks to drop.

The ΔT channel set-points are determined after complex calculations. One of the calculation phases is illustrated on the accompanying figure.
The limits corresponding to the saturation and DNB phenomena (mentioned above) are calculated and drawn up as a function of the primary pressure in a (P, T_{IN}) plan in order to determine the forbidden zones (see figure 2.2-11).

On the 1300 MW and 1450 MW 4-loop reactors and taking advantage of the digital technology available at the time of the design phase, these ΔT channels were replaced by the "High Linear Heat Rate" and the "Low DNBR" protection channels able to calculate the linear power at the core hot spot factor FQ and the Departure from Nucleate Boiling Ratio values in real time conditions. When the current values corrected with the adequate uncertainty overstep the respective 590 W/cm threshold value (by increasing values) and the 1.30 (typical value) threshold (by decreasing values), a reactor trip is actuated.

Fig 2.2-11 Physical core limits



2.2.5. Symbols used in computing core power distribution

F^{N}_{XY}	Radial peaking hot spot factor	F^{N}_{U}	Nuclear uncertainty factor
F^{N}_{Z}	Axial peaking hot spot factor	$F^{N}_{\Delta H}$	Nuclear enthalpy rise hot channel factor
$F_{XY}(Z)$	Radial peaking factor at elevation Z	$F^{E}_{\Delta H}$	Enthalpy rise engineering hot channel factor
F_{Z}	Axial peaking factor	$F_{\Delta H}$ $F^{T}_{\Delta H}$	Total enthalpy rise factor
F^{N}_{Q}	Nuclear heat flux hot spot factor[1]	$P(z)$	Axial power shape
F^{E}_{Q}	Engineering heat flux hot spot factor[1]	F_{B}	Penalty factor for rod bow
F_{Q} F^{T}_{Q}	Total peaking factor or heat flux hot spot factor*	$S(z)$	Densification penalty

[1] **Some people incorrectly use the word "channel" in place of the correct word "spot"**

2

2.3. Limiting conditions of operation

Whatever the technology used for the cited protection channels, their response time is not compatible with fast accidents because one of the signals used to measure the core power level, i.e. the hot leg temperature, is impacted by some lag due to the measurement of the hot leg temperature in the hot leg by-pass.

This is the reason why it is necessary to design and install several "specific" protection channels with fast response times. Each of them relates to one fast accident. In order to get a fast response time, the specific protection channels are very simple: one parameter and only one is measured and compared with a predetermined threshold. In the case of overstepping, a reactor trip signal is emitted.

Theoretical calculations are necessary to demonstrate that the core melting and DNBR criterion values are not overstepped when the trip signal emitted by a specific protection channel occur is because all the parameters needed to characterize the DNB ratio or the core hot spot factor F_Q are not measured. For example, the primary pump speed signal which represents the core primary flow rate, is not a correct picture of the DNB ratio since this ratio depends upon several other parameters such as the inlet temperature, the primary pressure, and the axial power distribution. Assumptions dealing with the parameters which are not measured and the corresponding values before the accidents must be used and defined to perform the calculations. These assumptions become a part of the safety case and must not be violated at any time during normal operating conditions (class 1). In case of violation, alarms are actuated in the control room and operators must restore an acceptable situation as soon as possible.

Regarding the Linear Heat Rate aspect, the most limiting initial condition to be met is governed by the Loss of Coolant Accident which is a class 4 accident (LOCA).

For 900 MW 3-loop reactors, LHR monitoring during class 1 operation is performed through the Axial Flux Difference AFD measured by the 2-section excore detectors.

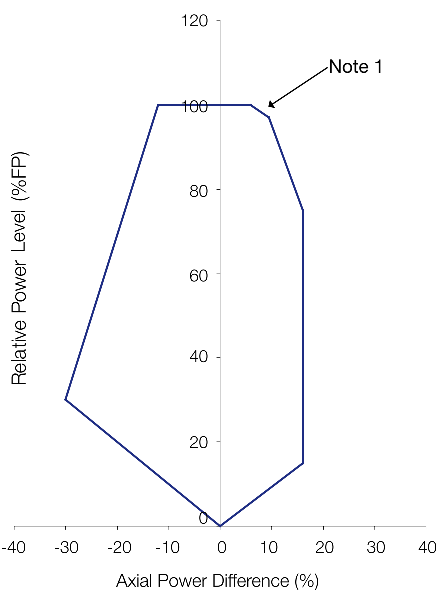
The measured AFD value must be kept inside boundaries which look like a trapezium (see figure 2.3-1) and constitute the class 1 operating domain (already mentioned above). But it is recommended to avoid strong variations in axial power distribution in order to avoid the generation of axial xenon oscillations later and to keep the (P, AFD) operating point close to its reference value.

For 1300 MW and 1450 MW 4-loop reactors, and taking profit from digital technology and from an improvement in the Excore detectors able to measure the real axial power distribution in the core and not only the AFD, Linear Heat Rate monitoring during class 1 operation is directly performed by calculating the current value of the maximum LHR F_Q and by comparing this value with the LOCA limit (400 W/cm) corrected with the adequate uncertainty (see figure 2.3-2). The new kind of excore detector is characterized by the presence of six sections instead of two. Thanks to a mathematical expansion process, it is possible to fit a curve through the six measuring points and to reconstruct the current axial power distribution. It has to be noted that the LOCA limit determined during the LOCA analysis is not constant versus the core elevation z. The 400 W/cm typical value corresponds to the lower core half.

For the EPR ™ reactor, the same principle as for the French 4-loop reactors is kept but excore detectors are replaced by a new kind of neutron detectors which are permanently located at fixed positions inside the core. These new detectors are made of ^{59}Co and are called **"Self Powered Neutron Detectors"** (SPND).

Regarding the Departure from Nucleate Boiling aspect, the most limiting initial condition to be met is governed by class 2 accidents, depending on the kind of reactors.

Fig 2.3-1 Operating Domain



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For 900 MW 3-loop reactors, DNBR monitoring during class 1 operation is performed through AFD monitoring. The aforementioned class 1 operating domain, which is determined according to LOCA considerations, is generally consistent with the limiting accident related to the DNBR.

If not, the operating domain must be corrected and most of the time narrowed, slightly reducing operating flexibility (see Note 1 on figure 2.3-1).

For 1300 MW, 1450 MW and the EPR™ reactors, DNBR monitoring during class 1 operation is directly performed by calculating the current DNBR value and by comparing this value with a "low-DNBR" threshold, the value of which depends on the limiting accident and the DNBR monitoring channel uncertainty (see figure 2.3-3). Cobalt SPNDs are used instead of six-section excore detectors for the EPR™ reactor.

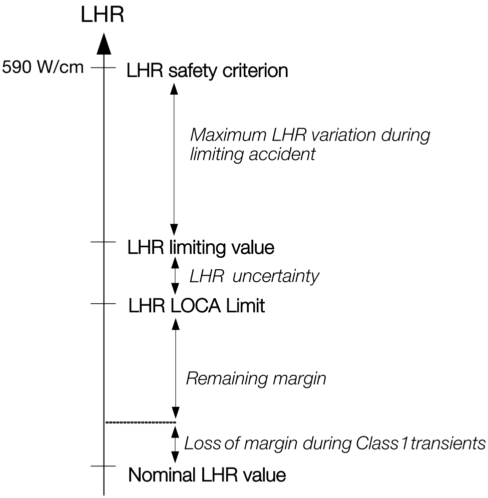


Fig 2.3-2 Principle of Linear Heat Rate Monitoring in class 1 condition

Considering the electronic equipment design aspect, it has to be noted that the **Limiting Conditions of Operation** monitoring function is ensured by the core protection system for 900 MW and 1300 MW plants. This LCO monitoring function is performed in dedicated equipment called "LCO Monitoring Units" which are separated from the core protection system for the N4 and the EPR™ units. This design choice was made to improve the core operating margins (LOCA and DNBR margins). The compromise between response time and accuracy was oriented towards the accuracy improvement induced by a core mesh improvement to the detriment of the response time due to the calculation time increase. This accuracy improvement allows us to gain margins and to improve the core operating flexibility.

The deterioration of the response time has no consequence when monitoring the core power distribution during normal operation, which is characterized by base load operation or slow power transients. Such an evolution was obviously not allowable for the protection system for which a quick response time is mandatory.

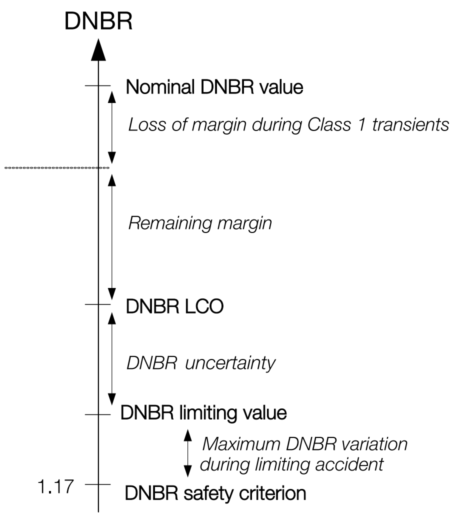


Fig 2.3-3 Principle of the Departure from Nucleate Boiling Ratio monitoring in class 1 condition

2.4. Instrumentation & Control Systems

The digital Instrumentation & Control (I&C) system for the EPR ™ and the KERENA ™ reactors consists in the following basic subsystems (Fig 2.4-1):

- Operational I&C system,
- Safety I&C system,
- Screen-based man-machine interface.

These subsystems are implemented using two different digital I&C platforms:

- SPPA-T2000 for operational functions,
- TELEPERM XS for safety functions.

Both systems have already been implemented in power plants in new installations and in backfitting projects. They offer excellent plant performance and proven operational behavior.

The operational I&C encompasses all systems and components required for plant process control in normal operating conditions (power operation, shutdown, refueling, etc.), such as:

- The Process Automation System including sensors, automatic controls and component protection functions in **Function Units** (FU),
- The Process Information and Control System including the man-machine interface systems in the **main control room** (MCR) and the **remote shutdown station** (RSS) both driven by **Communication Processors** (CP),
- Plant Bus and Terminal Bus systems,
- Diagnostic and Engineering System.

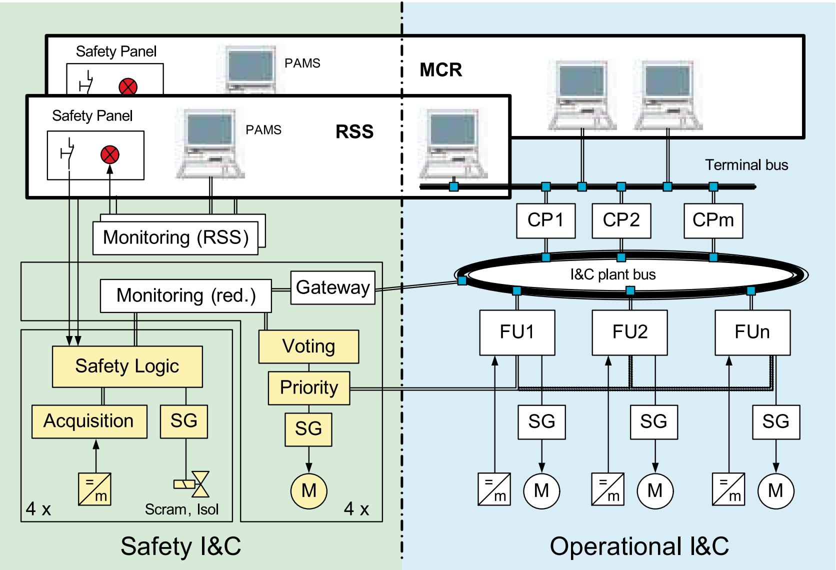


Fig 2.4-1 Instrumentation & Control System Concept

The screen-based Process Information and Control System is the overall information source with a plant overview panel on large screens and multiple screen-based workplaces. Intelligent information processing and compression enables this system to display all process conditions and process sequences with high level assistance to operators for both safety and operational tasks. Comprehensive archiving functions for off-line analysis of plant events and interface capability for intelligent diagnostic and maintenance systems is also provided.

The Process Information and Control System is supplemented by an independent Safety Information and Control System ("Safety Panel" and "PAMS") that provides manual control capability and screen-based accident monitoring. This system is driven by the safety I&C system and is different from the operational man-machine interface system.

The safety I&C system is subdivided into a number of redundant subsystems, the most important one being the Reactor Protection System (RPS). The task of the RPS is to process and monitor key process variables important for reactor safety and environmental protection. This system detects accident conditions and automatically initiates measures (e.g. shutdown, isolation, heat removal, pressure release, etc.), maintaining the reactor conditions within safe limits.

The safety I&C does not generate any safety action during normal undisturbed operation, but takes priority over all operational I&C system actions when required.

The process variables are measured, acquired and processed by several redundancy channels. The digital protection system processes the measured signals, compares them with limit values (some of which may be the result of calculations), generates logic trip demands (mainly reactor trip and emergency safety feature actuation demands), processes those demands through logic gates (majority voting) and transmits the resulting reactor trip and emergency safety feature actuation signals to the reactor trip breakers and safeguard actuators.

The reactor protection system is a digital system which offers excellent accuracy and performance.

The RPS is designed to withstand postulated failure situations according to the "single failure criterion". Complementary measures are taken to cope with common cause failures: some actions of the highest importance are generated by functions implemented in an automation system other than the RPS, i.e. the control system.

Furthermore, for the KERENA ™ reactor, even if the RPS totally fails in an accident situation, the passive safety equipment is able to bring the reactor back to a safe state as an independent and diverse back-up control system.

The various I&C subsystems are interconnected via redundant serial bus systems, implemented using fiber-optic cables. This reduces the electromagnetic sensitivity to a minimum and provides excellent performance capabilities.

Both the operational and the safety I&C system provide all the benefits of digital I&C:

- Fewer hardware module types and thus fewer spare parts,
- Engineering systems enabling plant staff to make application software modifications, if necessary. The top-down and straightforward design approach of the engineering systems ensures consistency of the actual system functions and their documentation,
- High reliability and failure detection capability due to on-line self-monitoring and self-test features,
- Low maintenance due to immediate failure detection and diagnostics; low failure rates and short repair time, proven by operating experience in many applications,
- Low effort for periodic tests, especially for the safety I&C. Since comprehensive and immediate failure detection is provided, the time interval between successive periodic tests can be extended so that periodic tests are carried out only during plant refueling outages, while retaining excellent reliability from the point of view of safety.

Most of the instrumentation and control (I&C) systems are supplied with power at a constant voltage of + 24 V by the DC systems via DC/DC converters.

Where necessary, the power is supplied through a pair of diodes by two different channels: rectification of AC supplied by an uninterruptible power source (rectifier, battery and inverter) or directly from the battery (through DC/DC converter) both of which are backed up by the main and station blackout diesel generators.

2.5. Materials under irradiation

2.5.1. Terminology

FRACTURE TOUGHNESS

This is a general term used to characterize the resistance of a material to withstand sudden crack propagation. This property can be assessed from several quantities (notch impact toughness, stress intensity factor, etc.)

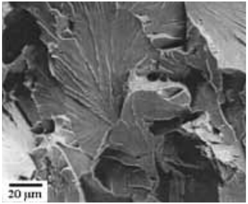


Fig 2.5-1 Brittle fracture

CHARPY V - NOTCH IMPACT TOUGHNESS

This is the energy absorbed during the impact testing of a V-notched bending specimen (Charpy test). Depending on whether the energy is normalized by the cross-sectional area under the notch, the notch impact toughness is expressed in Joules or in daJ/cm^2, using the notations KV or KCV respectively. Specimens with a U-shaped notch are sometimes used, in which case the notations become KU and KCU.

For materials exhibiting a ductile-brittle transition, fracture at low temperature gives a low value of impact toughness and high degree of crystallinity of the fracture surfaces (see figure 2.5-1). Above the transition temperature, the fracture surface becomes fibrous, which increases with temperature (see figure 2.5-2).

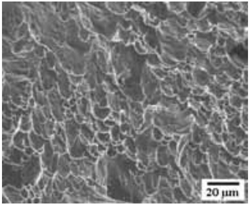
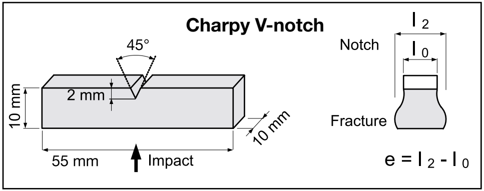


Fig 2.5-2 Ductile fracture

A transition temperature characterizing the fracture surface appearance (FATT or T_{K50}) is defined as the temperature at which the fibrous and brittle areas on the fracture surface are equal.

After the impact test, the lateral expansion of the specimen (see figure 2.5-3) is also measured, which enables fracture ductility to be assessed.

Fig 2.5-3 Impact Test specimen



WELD COEFFICIENT

For a pressure vessel butt weld, a weld coefficient (less than or equal to one) is applied to the design stress in the weld.

STRESS INTENSITY FACTOR (K_I)

Linear elastic fracture mechanics can be used to describe the strain and stress fields in the vicinity of a crack tip using the stress intensity factor K_I. This factor is a function of the crack size and shape, the stress, and the geometry of the structure:

$$\boxed{K_I = \alpha.\sigma\sqrt{\pi a}}$$

expressed in the SI system in $MPa\sqrt{m}$, where:

α = is a numerical coefficient that characterizes the geometries of the structure and the crack,

σ = is the stress applied to the part of structure remote from the crack, and

a = is the depth of the crack.

Fracture occurs when K_I attains a critical value K_{IC}, characteristic of the material, which depends on the temperature and the loading rate.

NIL-DUCTILITY TRANSITION TEMPERATURE, NDTT

This is the maximum temperature at which a standard drop-weight specimen (Pellini test), unable to withstand plastic deformation, is considered broken in accordance with a specific procedure. The deformation of the notched specimen is limited by a deflection stop so that it always remains elastic.

REFERENCE TEMPERATURE, RTNDT

This is a temperature which covers both the NDTT and a minimum notch impact toughness in the transverse direction at NDTT + 33°C. Above the RT_{NDT} , the risk of brittle fracture is low. This reference temperature is used to assess steel fracture toughness characteristics in mechanical analyses.

K_{IC} DESIGN CURVE

This is the lower bound of the critical stress intensity factor values versus temperature measured during low loading rate tests, with the reference temperature (RT_{NDT}) as the origin.

K_{IR} DESIGN CURVE

This is the lower bound of the critical stress intensity factor values measured during static and dynamic loading tests, with the reference temperature as the origin.

FATIGUE

When a structure is subjected to cyclic loading, a crack can be initiated and propagated until catastrophic failure of the structure occurs. The fatigue resistance of a material is characterized by the number of cycles, for a given load range, that it can withstand before failure (see figure 2.5-4). Two fatigue regimes are distinguished: low or high cycle fatigue. In design, this kind of damage is prevented by adopting transposition factors with respect to the average fatigue life curves determined for each material using small polished specimens.

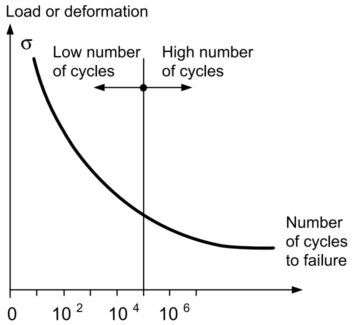


Fig 2.5-4 Fatigue resistance of a material

USAGE FACTOR

The usage factor is the ratio between the number of stress cycles to which a structure has actually been subjected and the number that would lead to the initiation of a crack. The end of life of a structure corresponds to a usage factor of 1. When a spectrum of stresses is present, the total usage factor is calculated as the sum of the usage factors corresponding to each mode.

In design, the curves used to compute the usage factor take into account the transposition factors defined either for the fatigue stress range or the cyclic life.

FATIGUE CRACK GROWTH RATE

For each material, the fatigue crack growth rate is defined as the crack extension per loading cycle, da/dN, and can be related to the range of the stress intensity factor during the cycle, ΔK_I. This crack growth rate is a function of the temperature and, for surface-breaking cracks, of the surrounding chemical environment, to varying degrees depending on the steel.

2.5.2. Effects of irradiation on materials

Neutron irradiation can initiate nuclear reactions and create unstable radioactive isotopes within the reactor's structural materials. It also modifies the physical properties of these materials. Four specific areas, particularly important in the technology of nuclear power plants, are considered below.

EFFECTS ON METALS

Irradiation of all materials by high energy particles such as neutrons, α particles and other heavy ions and electrons causes atom displacements from their equilibrium cristallographic locations thereby creating atomic scale point defects, i.e vacancies and interstitials. Neutrons generate larger cascades as the energy transfer to the displaced atoms is greater so that the displaced atoms in turn continue and cause knock-on collisions and even more atom displacements. Subsequent diffusion of point defects to various sinks leads to significant changes in microstructure and mechanical properties in metallic materials. In addition, neutron capture reactions induce transmutation reactions and hence changes in chemical composition.

EFFECTS ON REACTOR VESSELS AND STAINLESS STEEL STRUCTURES NEAR THE CORE

$$\left.\begin{array}{l}\text{Yield strength, E}\\ \text{Ultimate tensile strength}\\ \text{Hardness}\\ \text{Brittleness}\end{array}\right\}\uparrow$$

$$\left.\begin{array}{l}\text{Ductility}\\ \text{Impact toughness}\end{array}\right\}\downarrow$$

IRRADIATION MEASUREMENT UNITS

Integrated Flux or Fluence

This is the integrated flux, nvt, received per cm^2 during the entire irradiation. The minimum energy of the fast neutrons taken into account must be specified. Neutrons with an energy higher than 0.1 or 1 MeV are usually considered as harmful to metals.

Displacement Per Atom (DPA)

The DPA is the number of displacements made, on average, by an atom during irradiation. This quantity is not directly measurable, and instead calculated. It is used for characterizing neutron damage in fast breeder reactor materials and for light water reactor internals.

REACTOR PRESSURE VESSEL BEHAVIOR

The fast fracture toughness of ferritic steels varies according to temperature:

- At low temperature (below 0°C, depending on material specifications), the fracture toughness is low and a relatively small stress may be sufficient to initiate and propagate a fracture from an internal or surface defect. This is the brittle rupture zone.
- At higher temperature, much greater stress is necessary to propagate the same pre-existing defect. This is the ductile rupture zone.
- In the zone in which the fracture toughness increases rapidly with the temperature, a particular transition point is defined: the reference temperature RT_{NDT} (see above).

During hydraulic testing and under operating conditions, stress limits are imposed so as to stay safely below the material K_{IC} toughness curve. This leads to operating diagrams (see figure 2.5-5) defining a zone of authorized operation, i.e. the possible operation pressure versus temperature. When thermal stresses are generated by temperature gradients in addition to pressure loads, more severe curves are established for several rates of temperature variation.

Operating diagrams are based on material toughness curves drawn as a function of the temperature, and linked to the reference temperature RT_{NDT} of the material.

Under the effect of irradiation by neutrons escaping from the core, during the operating life of the reactor vessel, steel embrittlement occurs and the transition temperature rises. This embrittlement is calculated using formulae given in the codes, or measured by the amplitude of the shift in the transition temperature for a given level of impact energy (see figure 2.5-6) in the frame of the surveillance program.

This gives the variation in the reference temperature ΔRT_{NDT}. As a result, the material toughness curve shifts with RT_{NDT} and the operating diagram shifts with time accordingly.

The extent of irradiation embrittlement is a function of the neutron fluence received and the chemical composition of the reactor vessel steel. Sensitivity to embrittlement depends mainly on the copper and phosphorus contents, which are kept low in the irradiated zone of the vessel.

In the context of the surveillance program, capsules containing mechanical test specimens, made of the same material as the reactor vessel, are placed inside the vessel and irradiated at the same time as the vessel wall, although at a higher rate, making it possible to anticipate vessel wall ageing. Samples are removed at regular intervals to verify that the embrittlement of the metal conforms to the initial forecast. If any variation is observed, the curves of the operating diagrams may be revised.

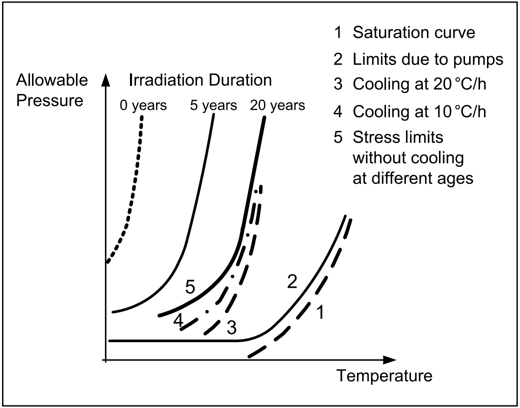


Fig 2.5-5 Operating diagram (pressure vs temperature)

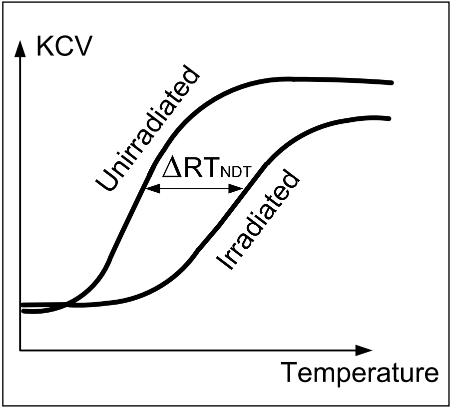


Fig 2.5-6 Reactor pressure vessel material behavior

2.6. Activity of the reactor coolant

Different sources of radionuclides are considered for the design and operation of a nuclear power plant. They are listed in the Safety Analysis Report to support the safety case, radiation protection of workers, and effluent treatment before discharge to the environment.

2.6.1. Activation of water

Several radioactive isotopes are produced by neutron capture reactions at each passage through the core. After a sufficient number of passages, the radioactivity at saturation is:

$$\boxed{A_\infty = N \sigma \phi \frac{1-e^{-\lambda\theta}}{1-e^{-\lambda T}}}$$

where N is the number of target atoms, σ is the macroscopic activation cross-section (per kg of water), λ is the radioactive decay rate constant (1.44 x half-life)$^{-1}$, θ is the irradiation time and T the time of one cycle in the primary system.

2.6.2. Activation reaction of water

Initial isotope	Neutron	Isotope formed	Half-life	Particle emitted	Remarks
^{2}H	Thermal	^{3}H	12.35 yrs	γ	Negligible (low σ)
^{16}O	Fast	^{16}N	7.11 s	β, γ	Preponderant reaction (high-energy γ rays)
^{17}O	Fast	^{17}N	4.16 s	β, η	Activity $\cong 1x10^{-2}$ of ^{16}N
^{18}O	Thermal	^{19}O	29 s	β, γ	Activity < $1x10^{-1}$ of ^{16}N

- During normal operation, the ^{16}N activity is as high as 3 to 5 $x10^{12}$ Bq/t depending on the reactor power. This radioactivity determines the thickness of the biological shielding around the reactor coolant system in the reactor building. It is also often measured in the steam leaving the generator as a tracer for primary to secondary leaks.
- At shutdown: negligible.

2.6.3. Chemical species dissolved in the water

Reactivity is controlled by means of a boric acid solution, whose pH is increased by adding lithium hydroxide for corrosion control. This chemistry regime (boron/lithium) leads to the formation of tritium, principally through the $^{10}B\ (n,2\alpha)\ ^{3}H$ and $^{6}Li\ (n,\alpha)\ ^{3}H$ reactions. This explains why lithium hydroxide is used as ^{7}Li enriched to 99.99% in the primary coolant to limit tritium production by this last reaction. The other sources come from ternary fission (~1 x10^{-4} atom of ^{3}H per fission), which diffuses through the fuel cladding and from neutron source rods made of Sb-Be. Depending on the power of the reactor, the total quantity of tritium released per year in liquid effluents may vary between 10 TBq and 50 TBq; this figure also depends on the type of fuel management.

The water contains very low concentrations of dissolved ions (Na, Cl, Al, Ca, etc.) and gases (argon), whose total contribution to the radioactivity level is negligible.

2.6.4. Activation of corrosion products

Depending on the type and design of the reactor, the surface of the reactor coolant system in contact with the water (essentially that of the steam generator tubes) is of the order of 15,000 to 30,000 m^2. Due to general corrosion, metallic ions and particles are released into the water, mainly at the start of power plant operation. These corrosion products are transported through the reactor coolant system, and are therefore susceptible to activation by passing through and staying in the core and are finally released from the core and deposited on the out-of-core surfaces. They are predominantly responsible for the irradiation of workers during plant shutdown.

The most troublesome isotopes, in decreasing order, are ^{60}Co and ^{58}Co, followed by ^{54}Mn, ^{59}Fe, and ^{51}Cr and ^{110m}Ag in the event of abnormal contamination. Since their half-lives are rather long, they determine the precautions to be taken during reactor maintenance in the vicinity of primary and auxiliary systems.

Radionuclide	Activation reaction	Half-life	Sources
^{58}Co	Ni (n,p)	70.7 days	Ni base alloys
^{60}Co	Co (n,γ)	5.27 years	Stellites, material impurities
^{59}Fe	^{58}Fe (n,γ)	45.1 days	All metallic materials
^{54}Mn	^{54}Fe (n,γ)	312.5 days	All metallic materials
^{51}Cr	Cr (n,γ)	27.7 days	All metallic materials
^{110m}Ag	Ag (n,γ)	249.9 days	Control rods and seals

2.6.5. Fission products

The fuel rod cladding may present minor cracks. Nevertheless it is possible to operate with leaking fuel rods provided that the total reactor coolant inventory in the fission products is limited and remains under technical specification values. This inventory depends on the utilities' practices and local regulations. In France, a water fission product inventory equivalent to 20 MBq/kg of equivalent ^{131}I may be accepted in normal operation (equivalent ^{131}I is calculated as $^{131}I + ^{132}I/30 + ^{133}I/4 + ^{134}I/50 + ^{135}I/10$). The rate of diffusion of the fission products in the fuel rods depends on their physical-chemical properties. The fission products that escape are essentially rare gases (^{85}Kr, ^{85m}Kr, ^{87}Kr, ^{88}Kr and ^{133}Xe, ^{133m}Xe, ^{135}Xe, ^{138}Xe) and iodine and cesium (^{131}I, ^{132}I, ^{133}I, ^{134}I, ^{135}I and ^{134}Cs, ^{136}Cs, ^{137}Cs, ^{138}Cs). The corresponding activity can reach 20 MBq/kg of equivalent ^{131}I, and 500 MBq/kg of rare gases without shutdown. In practice, experience shows that the actual activity is on average about 10 times less.

2.7. Fuel consumption

REFERENCE DATA

- One gram of fissioned uranium produces around 1.2 MWd.
- In a PWR core, the average content of ^{235}U is reduced, over a period of fuel assembly irradiation of 3 × 18 months, from 4.2% to between 0.8 and 0.6%.
- A 1% decrease in ^{235}U content is equivalent to 15,300 MWd/ton of uranium (usually abbreviated as MWd/tU).
- If captures without fission (not producing energy but consuming uranium) and plutonium fissions are included (Pu being produced from ^{238}U), the following correspondence is established for 3 × 18 months of irradiation in a PWR (52,000 MWd/tU): 1 MWd ≅ 0.66 g of ^{235}U + 0.72 g of ^{238}U
- 1 megawatt-day (MWd) $= 8.64 \times 10^{10}$ W.s $= 8.64 \times 10^{10}$ joule
  $= 2.4 \times 10^4$ kWh (thermal)
  $= 2.06 \times 10^{10}$ calorie
  $= 8.81 \times 10^9$ kgm
  $= 5.19 \times 10^{12}$ BTU
  $= 9.6 \times 10^{-7}$ kg-mass equivalent.

2

SPECIFIC BURNUP

The consumption rate characterizes the level of use of the nuclear fuel. It can be expressed as the product of the flux and the time during which it is produced, i.e. the integrated flux or fluence (expressed in neutrons/cm^2) used by physicists.

Engineers characterize this irradiation by a more technological term, specific burnup, expressed in megawatt-days per ton of uranium, i.e. the thermal energy expressed in megawatt-days (= 24,000 kWh = 0.864 x 10^{11} J) extracted from one ton of uranium (^{235}U + ^{238}U) contained in the initial fuel load.

FUEL BURNUP

Fuel burnup is the energy produced in the nuclear fuel per mass unit of fissile and fertile uranium:

Natural U with graphite or heavy water moderator	6,000 to 10,000 MWd/tU
Light water moderator	52,000 to 56,000 MWd/tU with refueling by 3 × 18 months
Fast breeder reactor	Up to 100,000 MWd/(t[U + Pu])

PRODUCTION OF PLUTONIUM (IN A PWR)

On average, at the end of a 3 × 18 months cycles, for each kg of enriched uranium in the initial load of fuel there are:

- 12.2 g of various Pu isotopes,
- 5.4 g of ^{236}U,
- 0.8 g of ^{237}Np,
- 0.3 g of ^{241}Am and ^{243}Am,
- 0.1 g of ^{244}Cm,
- 55 g of fission products.

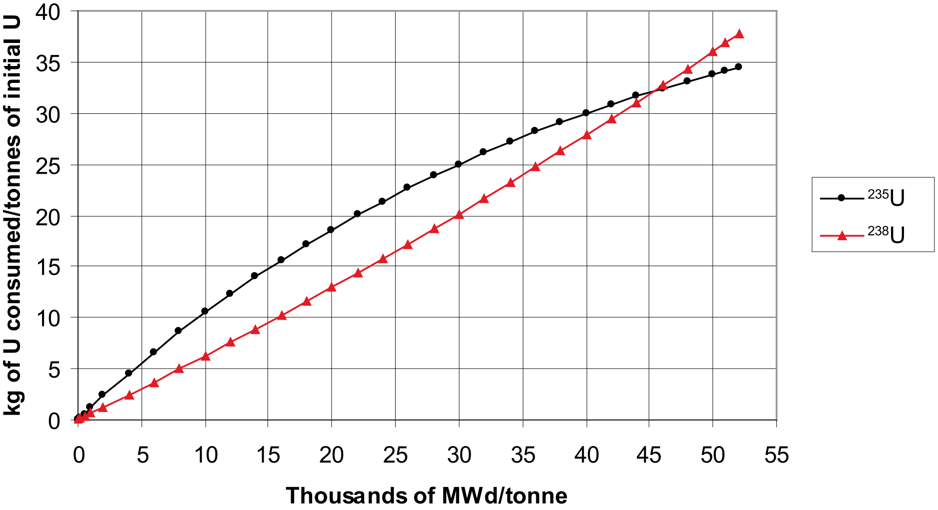


Fig 2.7-1 Consumption of ^{235}U and ^{238}U with respect to the initial mass of contained uranium

CONVERSION

Production, starting from a fertile substance, of a fissile substance different from the fissile substance being burned up.

CONVERSION FACTOR

$$\textbf{The conversion factor, } R = \frac{\text{The number of fissile nuclei produced}}{\text{The number of fissible nuclei destroyed (by fission or capture)}}$$

applies to an interval of time or a given instant.

- For a PWR, $R \cong 0.55$
- For a **fast breeder reactor** (FBR), this term is sometimes called **the regeneration factor**. Its average overall value, with blankets, varies from 1.15 to 1.20, but the evolution of core reactivity is essentially the product of the internal regeneration factor, which is about 0.80 to 0.85.

CONVERSION (OR REGENERATION) GAIN

Excess quantity of fissile atoms created by the fission of one fissile:

$$\boxed{\textbf{atom: } G = R-1}$$

In a FBR:

$G_{core} \cong -0.15$ to -0.20,

$G_{blankets} \cong +0.35$,

and $G_{overall} \cong +0.15$ to $+0.20$.

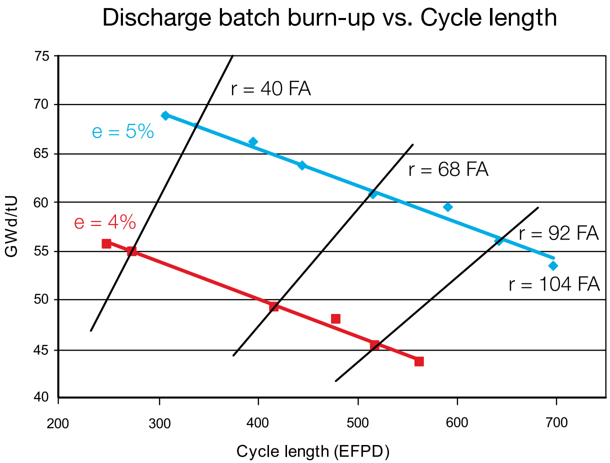
ORIGIN OF THE ENERGY

Because of conversion during the fuel cycle of 3 × 18 months in a PWR, the fissions producing energy occur in:

^{235}U	^{239}Pu	^{241}Pu	^{238}U
56-60%	30-31%	4-5%	5-7% (fast fission)

Note: Near the end of the cycle, only 30% of the fissions occur in ^{235}U, while 54% occur in ^{239}Pu, 10% in ^{241}Pu, and 6% in ^{238}U. One can thus see that even without plutonium recycling, plutonium makes a considerable contribution as a nuclear fuel. Pu is now being recycled thanks to the introduction of mixed-oxide (UO_2 and PuO_2) fuel, called "MOX", for use in PWRs.

Fig 2.7-2 EPR ™ reactor sketch of fuel management



e = enrichment
FA = Fuel Assembly
r = reload
EFPD = Equivalent Full Power Day

3

OPERATION OF NUCLEAR POWER PLANTS

THE FUEL CYCLE

B. THE NUCLEAR FUEL CYCLE **134**

3.1. Thermodynamics of an LWR unit, example of Tricastin 900 MWe PWR

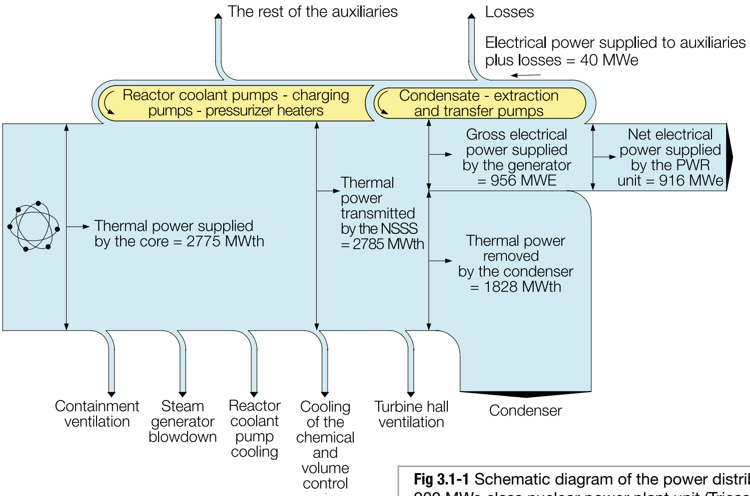


Fig 3.1-1 Schematic diagram of the power distribution in a 900 MWe class nuclear power plant unit (Tricastin 1)

3.2. Definitions

3.2.1. Powers

RATED CORE POWER (P): the thermal power engendered in the reactor core. This power is limited by safety considerations, and it is to this power that safety authorities refer in their definition of a maximum authorized value.

RATED NSSS POWER: the thermal power exchanged in the steam generators. This is equal to the rated core power increased by the heating of the reactor coolant due to pumping and decreased by the thermal losses through the walls of the reactor coolant loops.

The rated NSSS power can also be defined at the water-steam sides, at the limit of the NSSS scope of supply. In this case, the thermal losses through the walls of the secondary part of the NSSS scope of supply must be deducted. The value can also include the losses due to chemical control extractions (primary side) and to SG blowdowns (secondary side).

EXPECTED RATED POWER: for a given NSSS and fuel, this is the rated thermal power that is expected to be achieved, after unit startup, without modifying the limiting values, in particular for the fuel, once the unit's real behavior is known (real power distribution in the core, real performance of the SGs, etc.).

The auxiliary reactor systems and the secondary system are generally dimensioned as a function of the expected power increase (of the order of 5%), which is also known as a stretch or uprating. This power is then attained by varying certain quantities (for example, the temperature of the reactor coolant or the steam pressure), without any change in the limiting conditions for the hot channel.

NET ELECTRICAL OUTPUT POWER (P_{en}): the electrical power supplied by the generator, decreased by the power consumption of the unit auxiliaries and the losses in the main transformer.

NET CONTINUOUS POWER (P_{cn})[1] : the maximum net electrical output power that the installation can supply in continuous operation, the totality of the installation assumed to be in good operating condition. This value can vary during the year, along with the generator output power, depending on the temperature of the condenser cooling water. When determining the load factor (see section 3.3.), the reference power is the P_{cn} corresponding to the most unfavorable seasonal conditions (most often in the summertime).

NUCLEAR POWER: the total power released inside the reactor as a direct consequence of the fissions occurring therein. It is directly proportional to the number

(1) Or Net Maximum Continuous Power (EDF).

3

of fissions per second. The nuclear power includes the energy released by the slowing down of neutrons inside the moderator.

THERMAL OUTPUT: the thermal power produced by nuclear reactions in the reactor core. It also includes the energy released by the reactor coolant pumps. This is the power available for producing steam in the SGs.

THE POWER LEVEL is measured by different measurement channels: the Excore detectors (Power Range Detector), enthalpy differences between the SG inlet and outlet (secondary side enthalpy balance) or between the core inlet and outlet (primary side enthalpy balance). According to the measuring channels, the power level signal is affected by different uncertainties and time responses. Attention must be paid when using the power level signal in the protection system or the LCO monitoring system.

GROSS ELECTRICAL OUTPUT POWER (P_{EG}): this value, used quite frequently in statistics, is the electrical power at the generator output.

AVAILABLE POWER (P_a): the maximum electrical power, gross or net, that can be supplied by the installation at a given moment of its operation, under its real operating conditions at the time.

UNAVAILABLE POWER (P_u): this is equal to P_{cn} - P_a.

3.2.2. Normal operation

BASE LOAD OPERATION: at the origin of the nuclear industry, nuclear power plants are operated at full power (100% of the rated power) all the time. After three days following the first power escalation after refuelling, xenon equilibrium conditions are reached and the core is stable. Long term reactivity variations occur due to uranium depletion and plutonium build-up. Such operating conditions correspond to an economic optimum.

When the number of nuclear power plants delivering power on the grid increases, it is obvious that the nuclear plants must contribute to the right power production/consumption balance. So the power level may vary over the day.

LOAD FOLLOWING: electrical consumption on an interconnected grid varies during the day. Power plants must be able to adapt their operation to follow these load variations. Certain installations (such as run-of-the-river hydropower plants) operate continuously to supply the base load. For the others, on the basis of the forecast demand, the central dispatching office defines, each day, a load profile to be followed by each plant unit. An analysis of the behavior of NSSSs during load following is made on the basis of different typical profiles defined beforehand, for example with a 17-hour plateau at P_{cn}, a six-hour plateau at 50% of P_{cn} and two half-hour plateaus for power changes. It has to be noted that power level variations induce rather high xenon level variations of which the consequences are seen either in terms of global reactivity (temperature variations) or axial power perturbations (AFD oscillations). Both perturbations must be monitored and countered by the operator.

AUTOMATIC (OR LOCAL) FREQUENCY CONTROL: the method described above allows only a rough adjustment to the grid load variations. Slaving of the plant unit to the grid ensures instantaneous adjustment at the turbine-generator, by means of the latter's speed/load adjustment. The power variation requirements transmitted from the turbine-generator to the NSSS, as a function of the frequency differences of the grid voltage with respect to the reference frequency of 50 Hz, are defined by the turbine-generator's offset adjustment. In France, this is generally set at 4%, i.e. a frequency difference of 20 millihertz leads to a load demand change of 1%. The amplitude of the corresponding power variations belongs to a -3% / +3% of NP band.

REMOTE FREQUENCY CONTROL: starting from an energy balance (production vs consumption) and measurement of the general grid frequency, the central dispatching office elaborates a signal that is sent to all the plant units participating in remote frequency control. Each of them contributes to making the necessary adjustment, up to a certain fraction of its rated power (up to +10 and -10% of rated power for conventional thermal units and up to +5 to -5% for nuclear power units).

TECHNICAL MINIMUM: the minimum output power at which the unit can operate during an unlimited time under good conditions of stability, in automatic control.

3.2.3. Transients caused by grid faults

GRID FAULT RESISTANCE: this is the aptitude of the installation to withstand a grid fault without tripping out; it is time-limited.

Beyond the time limit, the installation must either be tripped out or shifted to house load.

HOUSE LOAD: in the event of general power grid trouble (an excessive frequency drop to 47 Hz, for example) due to too great a variation between production and consumption, the plant units shift to house load. This means that they disconnect themselves from the grid and rapidly reduce their power level (load release) to that necessary for powering the auxiliaries alone i.e. the technical minimum. This prevents their complete shutdown due to overload and makes them available for a fast power buildup and return to load, when the load fault is finally eliminated.

3.2.4. Shutdown conditions

REFUELING SHUTDOWN: cold temperature and atmospheric pressure conditions with the reactor vessel closure head removed.

The spent and the fresh fuel assemblies are located in the fuel building and more accurately in the Spent Fuel Pit filled with borated water. The sub-criticality level required by the Technical Specifications is reached thanks to a high boron concentration (typically 2500 ppm). The residual heat, i.e. the heat mainly released by the beta radioactivity of the fission products contained in the spent fuel assemblies, is removed by the Reactor Cavity and Spent Fuel Pit Cooling and Treatment System. According to the current Plant Outage phase, the reactor pit or cavity may be filled fully or partially with borated water of the same concentration as above.

COLD SHUTDOWN: cold temperature and low pressure conditions with the reactor vessel head shut. The sub-criticality level required by the Technical Specifications is reached thanks to a combination of control bank insertion and boron concentration. The required boron concentration is a function of the mean core burn up. The residual heat is removed by the Residual Heat Removal System.

HOT SHUTDOWN: the temperature and the pressure values are the nominal values at zero power defined by the part load diagram.

The sub-criticality level required by the Technical Specifications is reached thanks to a combination of control bank insertion and boron concentration. The required boron concentration is a function of the mean core burn-up. The residual heat and the heat released by the primary pumps are removed by the Steam Generators. The primary water temperature is controlled thanks to the secondary pressure control channel. The steam is evacuated by the condenser steam bypass device.

HOT STANDBY: the temperature and the pressure values are the nominal values at zero power defined by the part load diagram. The corresponding duration period is limited.
The reactor is critical at a power level close to 2%. Control bank insertion and boron concentration enable the power to be escalated to the rated value as quickly as possible. The primary water temperature is controlled thanks to the secondary pressure control channel. The steam is evacuated by the condenser steam bypass device.

CYCLE: in the restricted sense, this is the period of time between two refueling shutdowns. In the broad sense, the fuel cycle encompasses the complete history of the fuel from its fabrication, starting from the mineral state up to its reprocessing after irradiation in the reactor.

CAMPAIGN: the period of time including the length of the cycle as defined above (restricted sense), to which is added the outage time necessary for maintenance and refueling of the reactor.

3.3. Performance Indicators

3.3.1. Introduction

A great number of terms exist to characterize the operation of nuclear power plants. Their definitions vary according to the authors, the organizations, or even the departments of these organizations, both in France and abroad.

This leads not only to a lack of homogeneity, but even ambiguity or imprecision in the values published in the professional literature. These defects are aggravated by other difficulties:

- Sometimes the coefficients are established on the basis of gross power, sometimes on the basis of net power.
- The reference power to be taken into account is ill-defined. It may be the NSSS power evaluated during the design phase, that authorized by the safety authorities, engraved on the turbine-generator nameplate, or the maximum power possible for the installation during the summer (which can lead to a load factor in excess of 100% in winter, when the temperature of the condenser cooling water is more favorable).
- Some utilities, to ensure high performance coefficients, characterize the reliability of the installation, or even a part of it, by the percentage of time during which it operates independently of its power level. This explains why the NSSS can be said to be available, even if the rest of the installation is not in operating condition, or even when the NSSS power does not exceed several percent of its rated power. Even when the turbine generator is out of order, the NSSS can still be considered to have an “availability” of 100%. This availability is sometimes even defined as the complement of 1 of the unavailability (which is easier to characterize).

3.3.2. Main indicators

ENERGY-BASED INDICATORS

The most representative and most frequently used terms refer to factors, expressed as a percentage, that characterize the production of energy (see the table on the following pages).

Load Factor, LF or Ke: for a given period of time the Load Factor of a Unit or Station is the net energy that it produces (MWh) compared to the output at generator terminals, multiplied by the calendar hours.

Energy Availability Factor, EAF or Kd: for a given reference period T (month, year, etc.) the Energy Availability Factor is the ratio of the energy available at the unit output to the maximum energy that this unit would produce if operating full time at its net continuous power.

3

Unit Capability Factor, UCF: the Unit Capability Factor, defined by UNIPEDE[(1)] in conjunction with INPO[(2)], is the ratio of the available energy over a given time period to the maximum amount of energy which could be produced at maximum capacity if operated continuously over the same period. Available energy is the energy that could have been produced considering only limitations within the control of the plant management.

TIME-BASED INDICATORS

The availability of a unit is also characterized by factors that characterize the unit operating hours or days.

Availability Factor: the Availability Factor is the ratio of the time a Unit or item of equipment is capable of operating to the total time in a given time period, usually a year.

The Availability Factor is also used by the EPRI[(3)]-URD[(4)]. It is better adapted to new design specifications than the Unit Capability Factor defined by UNIPEDE. The UNIPEDE definition currently implemented by WANO[(5)], is more adapted to the operation phase to represent Unit performances.
If the plant is capable of rated power production when in operation, then the Availability Factor is roughly equal to the Unit Capability Factor.

Forced Unavailability Factor: the Forced Unavailability Factor is the ratio of the time a Unit is unable to produce power due to equipment or system failure to the total time in a given time period, usually a year.
For the same reason as for the availability objective, the unavailability in time i.e. Forced Unavailability Factor is preferred to Unplanned Capability Loss Factor defined

(1) Union Internationale des Producteurs et Distributeurs d'Énergie Électrique founded in 1925, is a professional organization which comprises the companies responsible for producing, transporting and distributing electricity.

(2) Institute of Nuclear Power Operations created in 1979 by the nuclear electric utility industry. INPO's mission is to promote the highest levels of safety and reliability in the operation of nuclear plants.

(3) The Electric Power Research Institute was established in 1973 as an independent, nonprofit center for public interest energy and environmental research.

(4) Utility Requirement Document is a set of recommendations or requirements established by a group of nuclear electric utilities for Advanced Light Water Reactors to be built beyond the 80's.

(5) World Association of Nuclear Operators is an organization created to improve safety at every nuclear power plant in the world. WANO was formed in May 1989 by nuclear operators worldwide uniting to exchange operating experience in a culture of openness, so members can work together to achieve the highest possible standards of nuclear safety.

by UNIPEDE/WANO as the ratio of unplanned (i.e. less than 4 weeks notice) unavailable energy over a period of time, to the maximum amount of energy which could be produced over the same period.

NOTE: the ratio of net energy supplied by the plant unit (during a given period) to the energy available during the same period is called **Output Factor Ku**. It characterizes the demand structure, as well as the economic constraints and the nature (or age) of the equipment, which leads to operating certain power plants units more than others (for example, those with lower production costs).

Structure of unit/unavailable capacity according to IAEA:

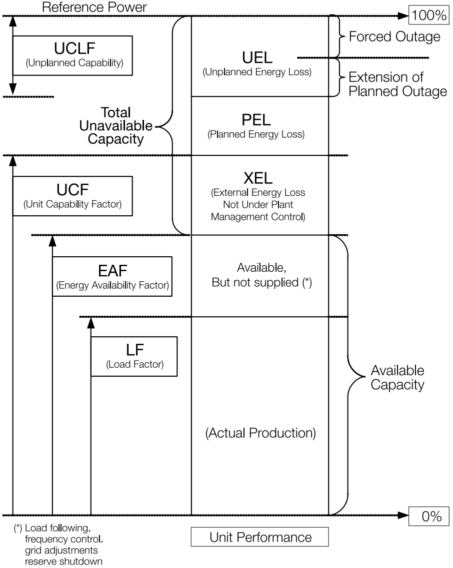


Fig 3.3-1 Structure of unit total available/unavailable capacity according to the IAEA

Terminology	Definition	French equivalent
Load factor[1] or Capacity factor (EE)	$LF = K_e = \frac{\text{Net energy produced}}{P \times T}$ P: output at generator terminals[2] T: number of hours in the reference period (calendar hours)	Facteur de charge Ke
Energy Availability Factor	$EAF = K_d = \frac{\text{Producible energy}^{(3)}}{P_{cn} \times T}$	Coefficient de disponibilité (en énergie du matériel) Kd
Output factor	$K_u = \frac{\text{Energy produced}}{\text{Energy producible}}$	Coefficient d'utilisation (de l'énergie pendant la disponibilité) Ku
Availability factor	$AF = \frac{\text{Time unit is available}^{(4)}}{T}$	Coefficient de disponibilité en temps (du matériel)

(1) Often expressed in equivalent hours (or days) at full power.
(2) Output at generator terminals is also named Gross Power. However, the IAEA also calculates in addition to the "Gross" Load Factor a "Net" Load Factor
(3) The installation being connected to the grid or in reserve (whatever the level of power).
(4) During which the installation can supply electricity (whatever the level of power).
IAEA: International Atomic Energy Agency - EEI: Edison Electric Institute

3.4. Economic analysis

3.4.1. Principles

CONSTANT-MONEY LEVELIZED GENERATION COST METHOD

In general, power plants take two to six years to build, depending on their type. While the lead time of power plants is rather long, their lifetime is much longer still: 25 to 40 years and up to 60 years in the case of new nuclear power plants.
Building a new power plant is therefore a long-term investment which, as such, requires to be backed by sound economic analysis, with a view to comparing the respective advantages of each technology available and, ultimately, making the best decision possible on the basis of the available data.
The most commonly selected criterion is the kWh cost discounted over the lifetime of the plant, generally referred to as the levelized generation cost.
Analysts are faced with two problems when using this approach:

- Expenses vary over time.
  Expense forecasts are altered by inflation.
  Moreover, an overall national inflation rate does not faithfully reflect the monetary changes encountered by each specific item. On top of this, the constituent parts of the cash flow are liable to drift above or below the assumed values.

- Investment and production do not occur at the same time. From an economic point of view, it is not without incidence that an expense occurs at a specific point in time rather than at another, even when the amount remains unchanged in constant money figures.
  Therefore a generation cost cannot be calculated by simply dividing expenses by production figures when they occur at different times.

The first problem is dealt with in two stages:

- Analysis in constant money terms
  All costs are expressed in constant monetary units of a specified reference year, thus avoiding the need to forecast future inflation trends.

- Handling of cash flow drift
  Assumptions on drifts (such as those relating to fossil fuels: oil, natural gas, etc.) are dealt with through scenarios.

The second problem – which has more to do with methodology – is dealt with by giving a specific time value to money (for example, this is what a bank does - through interest rates - when lending money to its customers). It is important to note that the interest rate not only compensates for monetary losses due to inflation, but also allows the lender to make some net profit.

In the economic analysis, this objective is met through the use of a "discount rate" acting exactly like a net interest rate, i.e. inflation free. It should be emphasized that this time-valuation applies both to expenses and to electricity generation, the latter being the source of income.

- In countries where electricity generating plants are state-owned, the government decides the value of the discount rate and in this way keeps control over the investment choices relating to power plants. Nonetheless, the overall economic context, which is fairly well reflected in the interest rates, is taken into account when determining the discount rate value. In France, the 2006 discount rate was 8%.

- Whenever electricity generating plants are privately owned, the owners are at liberty to choose the discount rate they want to use, the main driver being the financial markets. In some cases this leads to low discount rates (such as in Japan where it stands at 3%, or in Finland where the figure is 5%). In other cases, the discount rates in use are much higher (10% or even more), reflecting the fear investors have of long-term investments and the difficulty in securing funds in the context of liberalized electricity markets.

Once the discount rate has been selected, the levelized generation cost per kWh can be calculated.
In order to do this, there must be an economic time matching between expenses and electricity generation so that the former can validly be divided by the latter.
This is particularly true for investment expenses, as opposed to fuel and O&M expenses which occur roughly at the same time as electricity generation.
To achieve the said time matching, a reference date (t_0) is selected to which both expenses and generation are discounted; most often t_0 is the plant commissioning date.

Taking for instance an expense E_t occurring in year "t" - its economic equivalent E_{t0} calculated at time t_0 is given by the formula:

$$\mathbf{E_{to} = E_t \times (1+r)^{(to-t)}}, \text{ r being the discount rate}$$

- Expenses incurred before the reference date are increased in this process. This applies mainly to the initial capital investment, but also to the front-end expenses incurred for the first core. Conversely, expenses incurred after the reference discounting date are reduced in the calculation process. These are mainly the fuel and Operating & Maintenance expenses, but also revamping investment and the plant dismantling expenses.
- The increase to which the initial investment is subjected is known as Interests During Construction (IDC). The reference equation is:

Total Capital Investment = Overnight cost + IDC

- Electricity generation (kWh) that takes place after the reference date is also reduced in the calculation.

The levelized generation cost per kWh (c) is obtained by dividing the total discounted expenses figure by the discounted amount of electricity generated. In order to simplify the calculation, cost and electricity generation figures are assumed to occur at mid-year. It is important to note that the levelized generation cost stays constant, irrespective of the selected discount date.

$$\mathbf{C = \frac{\text{Sum of discounted expenses}}{\text{Sum of discounted electricity}}}$$

The higher the discount rate, the greater the effect of the expenses/production time gap. The levelized generation cost of electricity is then very sensitive to the investment cost and the construction duration. This is typical of nuclear power plants – high investment cost and long construction time – the levelized generation cost of which is very sensitive to the selected discount rate. On the contrary, a gas-fired power plant, characterized by low capital investment cost and short construction time, benefits from a levelized generation cost that is totally insensitive to

the selected discount rate (but is subject to potentially high fluctuations in gas prices).
The "Reference costs for electricity generation" (coûts de référence de la production électrique) published every five years or so by the French Ministry for the Economy, Finance and Industry (DIDEME) are calculated in this way. The last study was published in 2003 and the one before that in 1997. In both studies, a discount rate of 8% was used.
Some international organizations (OECD, IAEA) also use this methodology which is well-suited to international comparisons. The last study performed by the OECD (Projected costs of Generating electricity) was published in 2005. The calculations were performed twice, with discount rates of 5% and 10%, thereby covering the range of discount rates used in most countries.

OTHER WAYS OF EVALUATING THE AVERAGE ELECTRICITY GENERATION COST

- Making allowance for the way the capital investment is financed.

In a strictly economic approach, the investment cost is taken as a single figure, the financing of which is assumed to be a non-issue. Here, on the contrary, the investment is split between "equity" and "debt". In the calculation process, the debt portion is handled through a reimbursement schedule. In other words, the expenditure that occurred during construction in the economic approach now occurs when the debt is being repaid, i.e. when the plant is operating. This leads, of course, to different figures because the installments are subject to interest. as for the equity portion, the calculation is identical to that used in the economic approach. altogether, the overall capital investment expenses schedule is completely changed since it now includes expenses occurring before and after plant start-up. Moreover the interest rate applying to the debt plays a key role in the debt repayment figures.

Therefore, the analysis performed is no longer a strictly economic analysis in which the discount rate is the sole parameter measuring the time value of money. Here the investor, be it private or state-owned, will likely take the opportunity to include an external loan, often of foreign origin, in his economic evaluation of the power plant.
But, once this is done, the subsequent steps of the calculations are performed in a way similar to that used in the conventional economic approach in order to obtain a levelized generation cost of the kWh.

It is worth noting that in this case, the discount rate, which is the rate of return on 100% of the investment in the conventional economic evaluation, expresses the rate of return on equity.

Taking into account the available loan reduces the kWh cost should the loan net interest rate (i.e. inflation rate removed) be lower than the selected discount rate.

Since the loan interest rates are always expressed as a nominal value (inflation trends included), this type of analysis is generally performed in current money. Nonetheless, by using a nominal discount rate (inflation trends included) to discount expenses, it is possible to obtain a levelized generation cost expressed in constant money at the reference date, as with the conventional economic approach.

This methodology, which was formalized by the IAEA, was applied in the economic evaluation of several international bids for nuclear power plants in the nineties.

- Taking into account the electricity wholesale price and the return on equity.

One of the features of the conventional economic approach is that when the wholesale price of the generated electricity is equal to the levelized generation cost, the return on investment is equal to the discount rate.

The principle of this new approach is to determine the wholesale electricity price that will ensure the return required by the equity investors. It can also be used the other way round, starting from a wholesale electricity price, such as the one that the market can afford, and then deducing the maximum possible return on equity. This makes it possible to see whether the investment is viable or not.

In practice, this type of analysis is performed in conjunction with an equity/debt sharing as described above in the same paragraph. It should be noted that the conventional economic method is an upper bound case of this approach where the investment consists of 100% equity.

Other parameters, such as income tax, are often included in this type of calculation.

3.4.2. Results (constant-money levelized generation cost method)

These results are taken from the aforementioned OECD report, Projected costs of Generating electricity – 2005 update.

This study is the sixth in a series (the first was published in 1983) carried out by the OECD. Nineteen OECD countries and three non-OECD countries provided information and cost data for some 130 power plants, including 27 coal-fired plants, 23 gas-fired plants and 13 nuclear power plants, all of which are either under construction or liable to be commissioned between 2010 and 2015.

When comparing these energy sources (coal, gas, nuclear) in the context of base-load generation, two main findings emerge.

GENERATION COSTS RANGE BY PLANT TYPE

Generation costs cover the data provided by all participating countries, bearing in mind that, in order to have more meaningful results, the 5% highest and lowest values were excluded.

5% and 10% refer to the discount rate used in the calculation.

In the OECD report, generation costs were expressed in US dollars of mid 2003 per kWh, but these absolute figures are now obsolete due to the combined effects of fossil-fuel prices large (2 to 3-fold) increase and investment costs increases.

For instance, in the meantime, the price of raw materials has soared e.g. from 2005 to 2007 :

- Stainless steel : + 88%
- Copper : + 130%
- Nickel : + 238%

Therefore it appeared more meaningful to convert these absolute figures to relative figures on a 0 – 100 scale, as shown in the figure below.

At a 5% discount rate, in terms of the levelized generation cost of electricity, nuclear power plants are always competitive with gas-fired power plants and are generally very well placed when compared to coal-fired power plants.

At a 10% discount rate, nuclear power plants are generally competitive with gas-fired power plants and often well placed when compared to coal-fired power plants.

The above results were obtained without allocating any cost to CO_2 emissions from fossil-fired power plants. Including such costs would have further improved the position of nuclear power generation in all cases.

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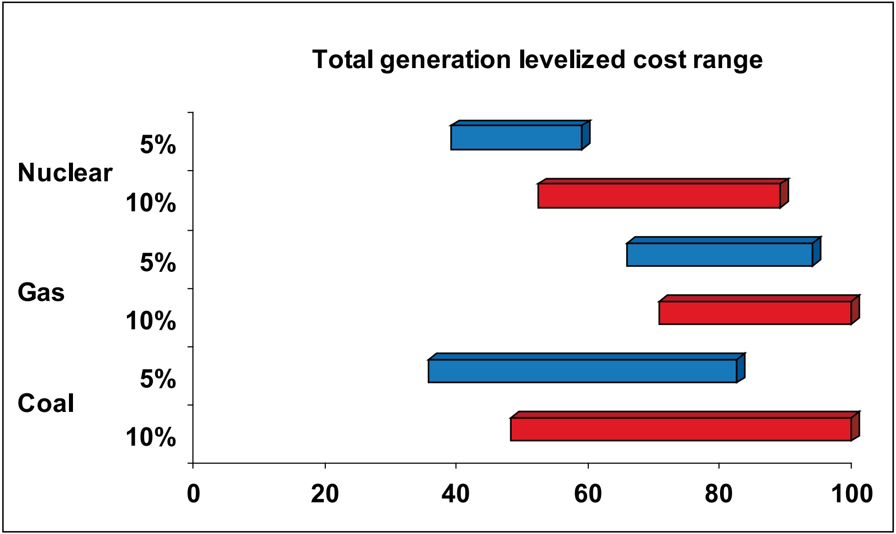


Fig 3.4-3 Total generation levelized costs range

AVERAGE COST STRUCTURE BY PLANT TYPE

In the case of nuclear power, the investment cost is the major component of the levelized generation cost, ranging from 50% at a 5% discount rate to 67% at a 10% discount rate. The fuel cost (front end plus back end) is the lowest component (13% at a 10% discount rate, 21% at a 5% discount rate).

The case of gas is radically different: the fuel cost is by far the largest contributor to the levelized generation cost, ranging from 72% at a 10% discount rate to 77% at a 5% discount rate. The investment cost is a minor contributor (17% at a 5% discount rate, 23% at a 10% discount rate).

In the case of coal, the contributions of the investment cost and the fuel cost are much more balanced. at a 5% discount rate the fuel cost is still the major factor (52% versus 30%), but at a 10% discount rate the investment cost is slightly higher (44% versus 41%).

Naturally these results are average values only and percentages may vary case by case. Nonetheless the average values reported here are a fairly faithful reflection of the general situation.

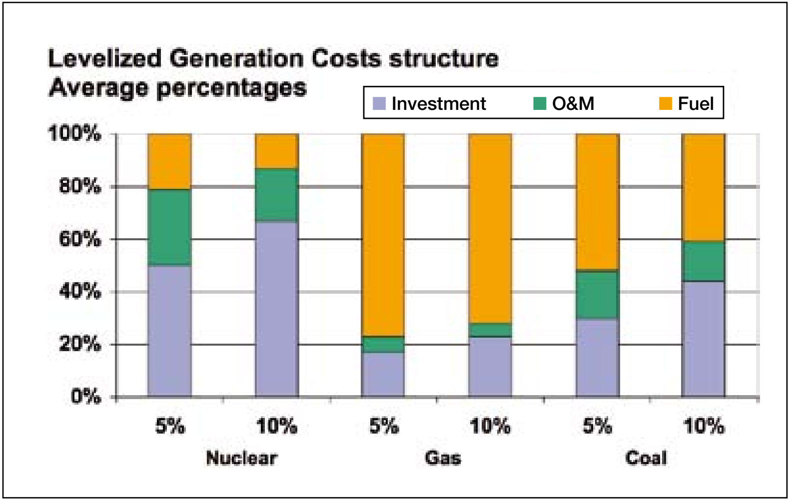


Fig 3.4-4 Levelized generation costs structure average percentage

3.4.3. CO_2 emissions and global warming

GENERALITIES

Concern over global warming explains the decision of some industrialized countries to internalise the impact of CO_2 emissions by assigning them a value to be included in the generation cost of fossil-fired power plants.

Such move adds a new element to the marginal cost of fossil-fired electricity, thereby increasing its price. The nuclear/fossil-fired power plants economic comparison needs a new assessment.

INCIDENCE ON THE GAS/COAL/NUCLEAR COMPETITIVENESS

Carbon emissions valuation:

In industrialized countries, the average amounts of CO_2 emissions by technology can be taken as follows:

- Coal-fired power plant : 850 kg/MWh
- Gas-fired power plant : 400 kg/MWh

The graph below gives the additional generation cost borne by Coal- and gas-fired power plants when the CO_2 emission valuation varies from 0 to €100/ton.

For instance a carbon emission valuation of 20 €/ton of CO_2 induces an additional generation cost of :

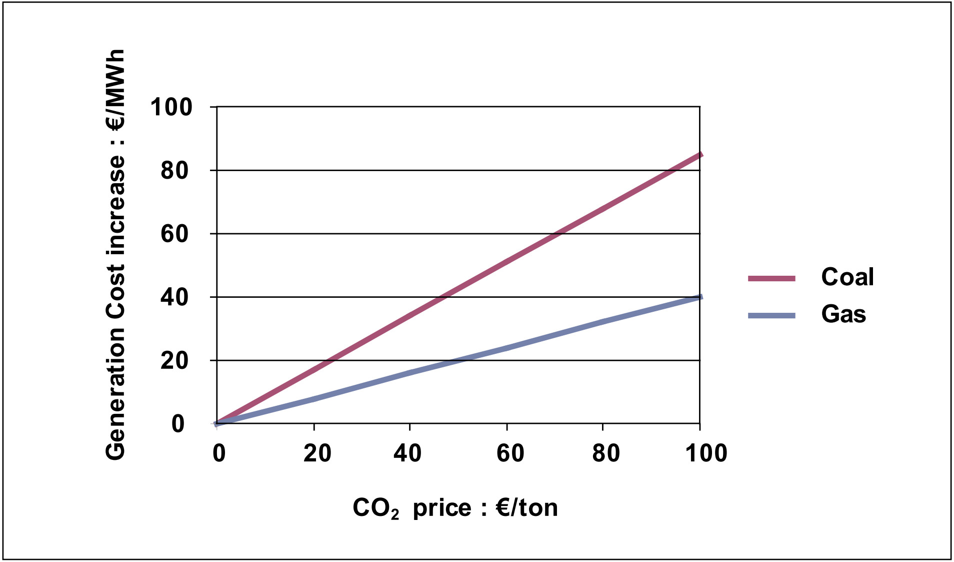
- 17€/MWh for coal
- 8€/MWh for gas

This graph may also be used to estimate the economic advantage of nuclear power plants when compared to fossil-fired plants subject to a CO_2 emissions cost.

In the specific case of a nuclear generation cost being higher than a fossil-fired generation cost prior to any CO_2 emissions valuation, the graph allows to determine the level of CO_2 emissions price necessary to restore the competitiveness of the former.

Example :
Let us assume that in a given country, prior to any CO_2 emissions valuation, an economic comparison puts nuclear, coal and gas on a par.

A CO_2 emissions valuation of €30/ton will make nuclear more competitive than coal and gas by €25.5/MWh and €12/MWh respectively.



3.4-5 Sensitivity of coal/gas generation cost to CO_2 emission valuing

3.4.4. Impact of the financing scheme on the generation cost estimation

Requirements coming both from private investors and lending institutions leads to generation cost calculations which need other approaches than the OECD constant money analysis (see section 3.4.1).

The prevailing financing environment is taken into consideration with the investment cost being split between "Equity" and "Debt".

Each of these two components is handled separately in the calculation: the debt portion through its interest rate and payback period and the equity portion through the return on equity required by investors over a specified period of time.

Equity investors require a high return specially if they must wait a long time to get it. Typically rates of 15%/year (nominal rate) are not uncommon.

Lending institutions offer lower rates (i.e. 8%) but with a rather short payback period (15 years). Moreover such loans will only be granted if a significant share of the investment (some 25%) comes from equity investors.

Example:

	Equity	Debt
Share	25%	75%
Rate of return	15%	8%
Inflation rate	3%	3%
Rate return (real)	11.7%	4.9%

A parameter often used in investments comparison is the Weighted Average Cost of Capital (WACC):
WACC (nominal) = 0.15x0.25 + 0.08x0.75 = 9.8%
WACC (real) = 0.117x0.25 + 0.049x0.75 = 6.6%

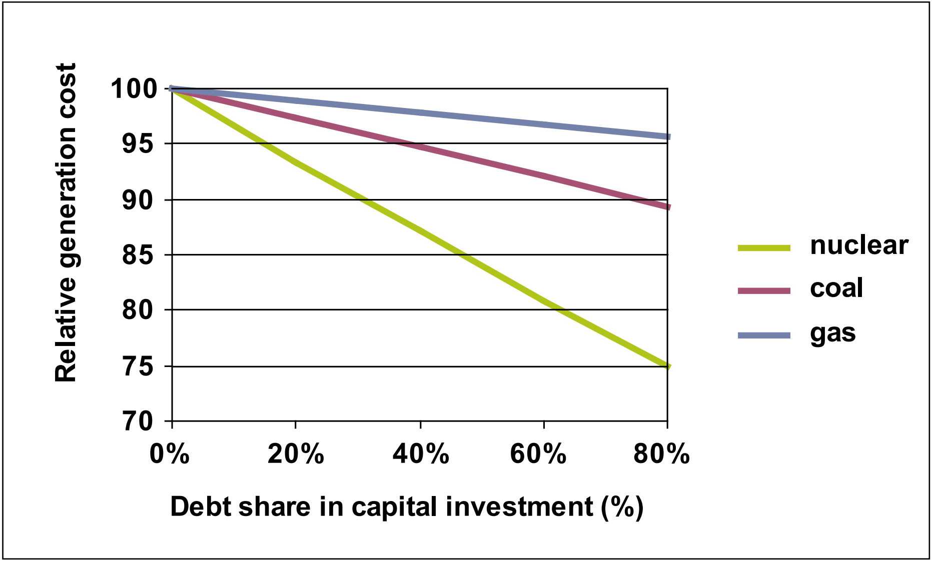
(The above formulas disregard the fact that return on equity is generally taxed while debt repayment is not. In order to maintain the same level of return on investor equity after tax, one must increase the first term of the above sums and hence, the WACC.)

Loan interest rates are lower than equity rates of return. Therefore increasing the debt portion is economically beneficial to the plant economy and lowers its electricity generation cost. And since the investment component is more important in nuclear power plants than in coal-fired or gas-fired power plants, this effect will be important for nuclear, moderate for coal and low for gas.

This is typically shown on the graph below where, starting with 0 debt (hence 100% equity), generation costs are 100 (relative values).

Assuming for instance that with 100% equity the generation cost of nuclear is 5% below the one of coal, reducing this equity portion to 40% (Debt = 60%) will increase the economic competitiveness of nuclear to 17% (0.95x80.7=76.7 for nuclear vs. 92 for coal).

Further decreases of the equity portion would give an even bigger economic advantage to nuclear.



3.4-6 Impact of debt share on the generation cost

3.4.5. Fuel cycle costs

THE NUCLEAR FUEL CYCLE

Once uranium has been mined, it has to go through the milling, conversion, enrichment and fuel fabrication stages before it can be used in a nuclear reactor. These steps make up the "front end" of the nuclear fuel cycle.

After uranium has been used in a reactor to produce electricity it is known as "used fuel" and may undergo a further series of steps including temporary storage, reprocessing, and recycling before final disposal as waste. Collectively these steps are known as the "back end" of the fuel cycle.

The various activities associated with the production of electricity from nuclear reactors are referred to collectively as the nuclear fuel cycle. The nuclear fuel cycle starts with uranium mining and ends with the disposal of nuclear waste. With the reprocessing of used fuel as an option for nuclear energy, the stages form a true cycle.

NUCLEAR FUEL COSTS

From the outset, the basic attraction of nuclear energy has been its low fuel costs compared with coal, oil and gas-fired plants, even if uranium has to be processed, enriched and fabricated into fuel elements. It is worth noting that this nuclear fuel cost also includes the management of used radioactive fuel and the ultimate disposal of this used fuel or the waste separated from it. But even with these included, the total fuel costs of a nuclear power plant in the OECD are typically about one third of those for a coal-fired plant and between a quarter and a fifth of those for a gas combined-cycle plant.

Structure of the Cycle Cost
(approximate values, at January 2007 US$ rates)

• Purchase of natural uranium	50%
• Conversion into UF_6	5%
• ^{235}U enrichment	25%
• Fabrication of fuel assemblies	10%
• Back end	10%
Total	**100%**

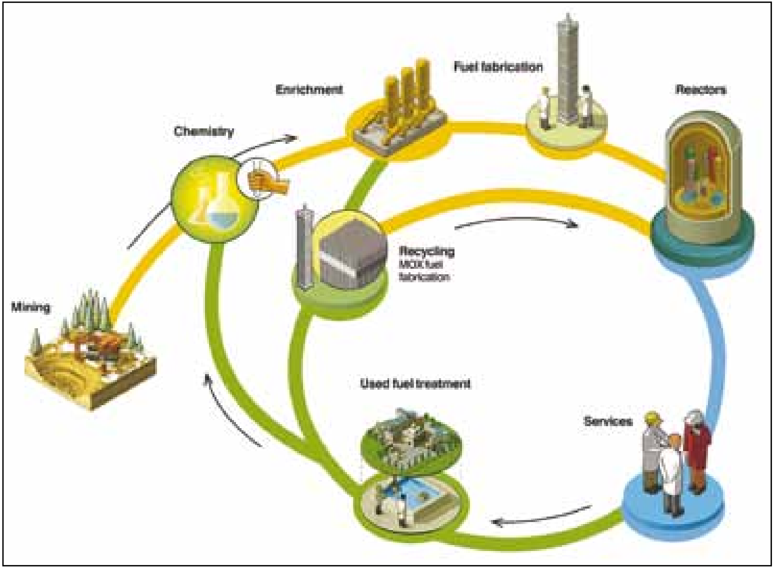


Fig 3.4-7 The Nuclear fuel cycle

3.5. Natural uranium

3.5.1. World uranium production and demand

World uranium production in 2006 amounted to 40,039 tU.

World uranium requirements for 2006 amounted to 65,937 tU.

(Source: NUKEM Data Feature – May 2007).

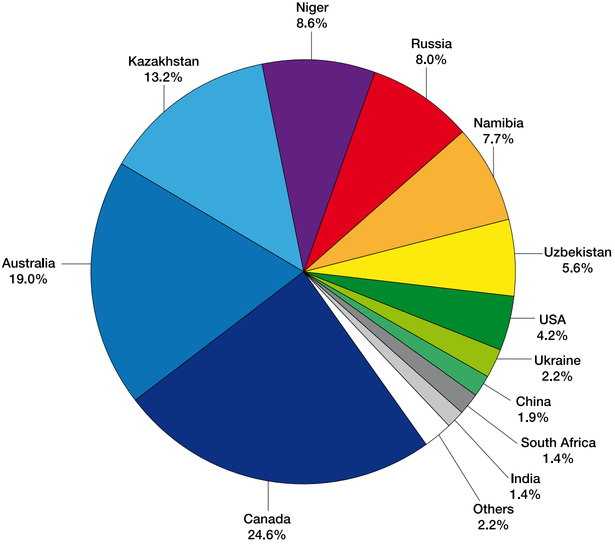


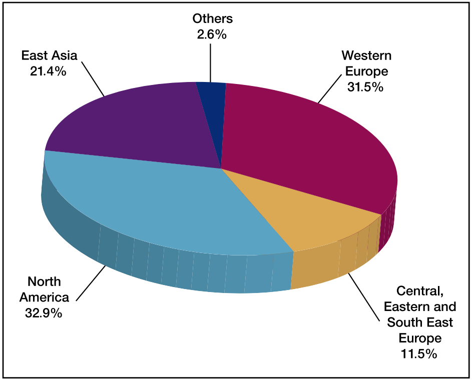
Fig 3.5-1 World Uranium Production in 2006 by country

World natural uranium production by country in 2006 [1]
(Source: NUKEM Data Feature – May 2007)

COUNTRY	Tons U	Share of Total (%)
Canada	9,862	24.6
Australia	7,593	19.0
Kazakhstan	5,281	13.2
Niger	3,434	8.6
Russia	3,200	8.0
Namibia	3,077	7.7
Uzbekistan	2,242	5.6
United States	1,699	4.2
Ukraine	900	2.2
China	750	1.9
South Africa	559	1.4
India	550	1.4
Subtotal	**39,147**	**97.8**
Others [2]	**892**	**2.2**
TOTAL	**40,039**	**100.0**

(1) Countries sorted according to production in 2006
(2) Others includes notably: Brazil, Czech Republic, Pakistan, and Romania.

Fig 3.5-2 Uranium Requirements in 2006 by zone in tU



Source: WNA Market Report 2007 - September 2007

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NOTE: in 2006, uranium production represented 60% of uranium requirements. The gap is covered by so-called "secondary sources" such as dilution of highly enriched uranium, recycling of off-spec material, use of MOX fuel and reprocessed uranium, and drawdown from inventories.

The concentrate extracted from mines is called yellowcake and comes in the chemical form of uranates or oxide (U_3O_8).

ASSAY OF NATURAL URANIUM (BY WEIGHT)

^{238}U: 99.284 %
^{235}U: 0.711 % (= 1/140)
^{234}U: 0.0053 %

NOTE: the other isotopes ^{232}U, ^{233}U, ^{236}U, ^{237}U are not present in natural uranium.

3.5.2. Sustainability of uranium resources

INTRODUCTORY REMARKS AND DEFINITIONS

The mineral resources and reserves in the earth's crust extracted by man contain a variety of materials (chemical elements or compound minerals, etc.) and can be distinguished as follow:

Resources: A mineral resource is defined as a mineral-bearing concentration or indicator of a natural, solid inorganic or fossilized organic material in or on the Earth's crust, and which is present in such form, quantity, concentration or quality to indicate that there are reasonable prospects for economic extraction. The location, quantity, and quality of a resource are estimated from geological evidence and knowledge.

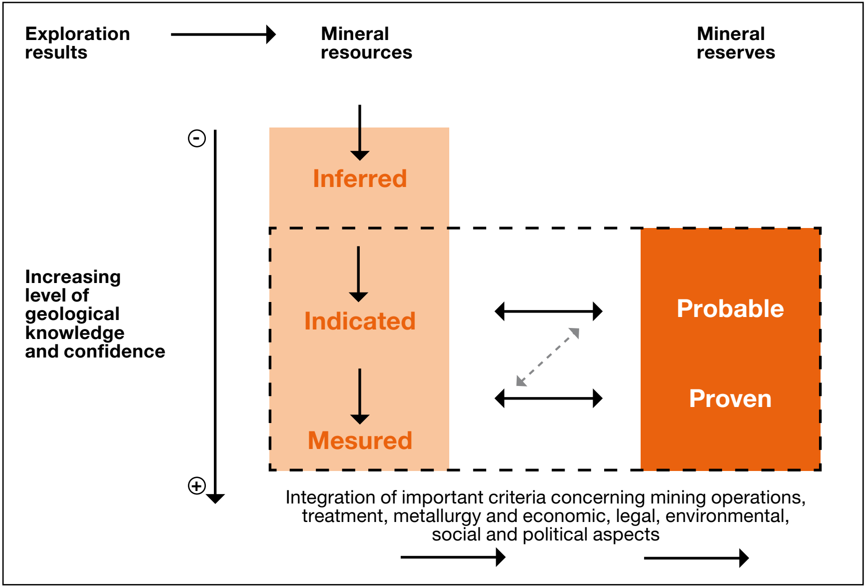
Resources are subdivided into categories, depending on a level of confidence. In particular, the IAEA/OECD "Red Book", which addresses worldwide uranium resources, uses the following breakdown for Identified Resources, by level of increasing confidence:

- Inferred Resources,
- RAR (Reasonably Assured Resources).

NOTE: depending on the reporting organization, definitions and breakdown may vary.

Reserves: A mineral reserve is the economically mineable (or recoverable) part of a resource, demonstrated at least by a preliminary feasibility study.
The diagram below shows the organization and hierarchy between resources and reserves:

Fig 3.5-3 Resources and reserves organization and hierarchy



WORLDWIDE URANIUM RESOURCES

a) Reasonably Assured Resources (RAR) by country and cost range

Source: OECD/IAEA "Red Book 2005" / Data as of 1 January 2005, in tons U

Country	< US$130/kgU	< US$80/kgU	< US$40/kgU
Australia	747,000	714,000	701,000
Kazakhstan	513,897	378,290	278,840
Canada	345,200	345,200	287,200
United States	342,000	102,000	NA
South Africa	255,593	177,147	88,548
Namibia	182,556	151,321	62,186
Niger	180,466	180,466	172,866
Brazil	157,700	157,700	139,900
Russian Federation	131,750	131,750	57,530
Others	1,187,527	305,469	159,313
Total	**3,296,689**	**2,643,343**	**1,947,383**

Note 1: the figures given in each column are the RAR available at an extraction cost below the threshold indicated in the first line. Figures are cumulative, e.g. Namibia totals 182,556 tons of RAR, of which 151,321 tons are recoverable at a cost of less than US$80, and of which 62,186 at a cost of less than US$40.

Note 2: for each category of resources, quantities are given according to an extraction cost range: less than US$40, US$80, and US$130. However, this is not directly linked to economic considerations, and it cannot be translated into reserves.

b) Inferred Resources

Besides the RAR, the world Inferred Resources recoverable at a cost < US$130/kgU are estimated at 1,446,164 tons U, leading to a total Identified Resources of 4,743 million tons U (< US$130/kgU). This represents 85 years of global nuclear electricity generation at 2004 levels, considering the current fuel cycle. Changes in nuclear technology, especially recycling and the fast reactor fuel cycle, would considerably extend this period.

c) Uranium resources by category (in million tons U)

Source: Nuclear Energy Agency "Uranium 2005: Resources, Production and Demand"

Cost of recovery $/kgU	Conventional				Unconventional
	Identified (deposits)		Undiscovered		
	Reasonably assured resources	Inferred resources (1)	Prognosticated resources (2)	Speculative Resources (3)	
< 40	1.95	0.80	1.7	4.6	
40 to 80	0.70	0.36			
80 to 130	0.65	0.29	0.82		
> 130	-	-	-	2.9	
Subtotal	3.30	1.45	2.52	7.5	
General total	**4.75**		**10.0**		**15 to 25**

- General total of conventional resources = 14,750,000 t
- World demand in 2006 < 70,000 t
- Resources > 200 times 2006 demand

(1) Based on direct geological evidence
(2) Based on indirect geological evidence
(3) Extrapolated values

3.5.3. Conversion

Prior to enrichment, the ore concentrates from uranium mines (yellowcake) need to be purified and converted into UF_6. The purification and conversion process can typically be presented as follows:

Yellowcake (U_3O_8) $\xrightarrow{\text{purification using TBP}}$ $UO_2(NO_3)_2$

$\xrightarrow{\text{Denitration, calcination}}$ UO_3 $\xrightarrow{H_2 \text{ reduction}}$ UO_2

$\xrightarrow{\text{Hydrofluorination } F_2}$ UF_4 $\xrightarrow{\text{Fluorination } F_2}$ UF_6

> **Note:** this may vary depending on the mining technique, the chemical nature of ore concentrate and the industrial process.

UF_6 can be a gas at relatively low temperature and atmospheric pressure, and therefore usable in enrichment facilities. Fluorine has only one isotope ^{19}F and makes mass differentiation between ^{235}U and ^{238}U easier.

The world conversion capacities are as follows:

Country	Conversion capacity (in tons of U/UF_6)
Russia (Rosatom)	20,000
France (AREVA/Comurhex)	14,000
United States (Converdyn)	12,500
Canada (Cameco)	12,500
United Kingdom (Westinghouse/ Cameco)	6,000

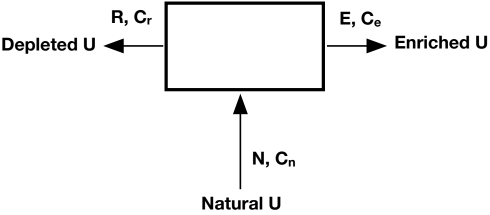
China, Japan and Brazil also have some small conversion capacities.

3.5.4. Enriched uranium

The enrichment of uranium involves increasing the assay in the fissile isotope ^{235}U, from C_n=0.711% to an assay C_e usable in power reactors, typically C_e=3 to 5%. Two processes are currently used on an industrial scale: gaseous diffusion and centrifuge technology. They both process UF_6 in a gaseous form.

CONSUMPTION OF NATURAL URANIUM

The enrichment operation can be presented as follows:



Uranium	Quantity of U (kg)	^{235}U Concentration (enrichment)
natural	N	C_n
enriched	E	C_e
depleted	R	C_r

From the conversion ratios of the total quantity of U and ^{235}U, the following equation is drawn:

$$\boxed{K = \frac{N}{E} = \frac{C_e - C_r}{C_n - C_r}}$$

K is a dimensionless ratio that gives the quantity of natural uranium (in kg) necessary to produce one kg of enriched uranium at a concentration C_e, when the concentration of the depleted U is C_r.

SWU: SEPARATIVE WORK UNIT (1)

Separative work is a unit used to identify the work needed:

- to enrich a given quantity N of natural U (assay C_n of ^{235}U) to an assay C_e of ^{235}U,
- while obtaining a ^{235}U tails assay equal to C_r for depleted U.

(1) This is the only unit in the entire document that does not have a simple and natural definition and signification.

To every assay c of ^{235}U one can associate a dimensionless function called the "value", as follows:

$$\boxed{V(c) = (2c - 1) \log \frac{c}{1 - c}}$$

The "mass value" of the uranium is therefore:

- before enrichment: $U_1 = N.V(C_n)$,
- after enrichment: $U_2 = E.V(C_e) + R.V(C_r)$.

As c and V(c) are dimensionless, U_1 and U_2 are homogeneous to a mass (kg).
This enables us to calculate the separative work thus:

$$\Delta U = U_2 - U_1$$

for a quantity homogenous to a mass, measured in kg. Taking the above ratios, one can determine the amount of separative work (expressed in kg SWU) needed to produce 1 kg of U, having an assay C_e from natural U (assay C_n) with a tails assay of C_r is:

$$\Delta U = V(e) + (K-1)\, V(C_r) - K.V(C_n)$$

The following two tables give a number of values illustrating this equation.

3

Note: the above functions give the separative work needed, regardless of the enrichment technology used. The actual number depends on the efficiency of the enrichment plant, and may vary with plant design and technology (centrifuge or gaseous diffusion).

Examples

- The separative work for 4.00% enrichment with a tails assay of 0.20% requires 6.544 kg SWU and a consumption K of 7.436 kg of natural uranium.
- For e = 0.711 (natural uranium), $\Delta U = 0$ and K = 1.

Note: separative work is homogeneous to a mass, not energy. However, the different processes consume more or less electricity to produce their SWUs. The total electricity consumption, including the installation's auxiliaries, of a gaseous diffusion factory like Eurodif is nearly 2,800 kWh/SWU, whereas centrifuge technology typically consumes 50 times less electricity.

SEPARATIVE WORK UNITS (SWU) AND MASS OF NATURAL URANIUM (KG) NEEDED TO PRODUCE 1 KG OF ENRICHED UF_6 WITH A TAILS ASSAYS OF 0.2 AND 0.3%, FOR DIFFERENT ENRICHMENTS

The error margin of the following values is less than 5%.

Enrichment (% of ^{235}U by weight)	Tails assay = 0.2%		Tails assay = 0.3%	
	SWU	Natural U (kg)	SWU	Natural U (kg)
2.00	2,194	3.523	1,697	4.136
2.20	2,602	3.914	2,028	4.623
2.40	3,018	4.305	2,367	5.109
2.60	3,441	4.697	2,714	5.596
2.80	3,871	5.088	3,066	6.083
3.00	4,306	5.479	3,425	6.569
3.20	4,746	5.871	3,787	7.056

Enrichment (% of ^{235}U by weight)	Tails assay = 0.2%		Tails assay = 0.3%	
	SWU	Natural U (kg)	SWU	Natural U (kg)
3.40	5,191	6.262	4,154	7.543
3.60	5,638	6.654	4,525	8.029
3.80	6,090	7.045	4,899	8.516
4.00	6,544	7.436	5,276	9.002
4.20	7,001	7.828	5,656	9.489
4.40	7,460	8.219	6,039	9.976
4.60	7,922	8.611	6,423	10.462
4.80	8,385	9.002	6,810	10.949
4.90	8,618	9.198	7,004	11.192
4.95	8,734	9.295	7,101	11.314
5.00	8,851	9.393	7,198	11.436

CONVERSION OF THE MAIN COMPOUNDS OF URANIUM

Multiply the mass of the compound Ai by the factor (ij) to obtain the mass of the compound Bj containing the same quantity of uranium.

Bj \ Ai	U	UO_2	UO_3	U_3O_8	UF_4	UF_6
U	1.0	0.8815	0.8322	0.8480	0.7580	0.6762
UO_2	1.1344	1.0	0.9441	0.9620	0.8599	0.7671
UO_3	1.2017	1.0593	1.0	1.0190	0.9109	0.8125
U_3O_8	1.1792	1.0395	0.9814	1.0	0.8939	0.7974
UF_4	1.3192	1.1629	1.0979	1.1187	1.0	0.8921
UF_6	1.4789	1.3036	1.2307	1.2541	1.1210	1.0

1 kg U = 2.600 lb U_3O_8 = 3.261 lb UF_6

1 lb U_3O_8 = 0.3846 kg U

1 lb UF_6 = 0.3067 kg U

URANIUM AND SWU CONSUMPTION

The above two tables give the mass of the different compounds required when fuelling reactors.

Examples:

- SWU required for a reload of 72 assemblies containing 530 kg of uranium each, enriched to 4.95% in mass:

  with a tails assay of 0.2% → 72 x 530 x 8,734 = 333,289 SWU
  with a tails assay of 0.3% → 72 x 530 x 7,101 = 270,974 SWU

- Mass of natural uranium required for the same reload:

with a tails assay of 0.2% ➔ 72 x 530 x 9.295 = 354.7 tons
with a tails assay of 0.3% ➔ 72 x 530 x 11.314 = 431.7 tons

- Corresponding mass of U_3O_8:

with a tails assay of 0.2% ➔ 354,700 / 2.600 = 136,400 lbs
with a tails assay of 0.3% ➔ 431,700 /2.600 = 166,000 lbs

Note: the tails assay is provided by the customer to the enricher. From this calculation, it is clear that SWU is a substitute for natural uranium, the tails assay being the tuning factor.

URANIUM ENRICHMENT CAPACITIES IN 2006 (10^6 SWU)

USEC (United States) - production	5.0	Gaseous Diffusion
USEC – Russian HEU	5.5	Dilution of Highly Enriched Uranium
AREVA / Eurodif (France)	10.8	Gaseous Diffusion
URENCO Enrichment Company (United Kingdom, Netherlands, Germany)	8.1	Centrifuge technology
Rosatom (Russia)	12	Centrifuge technology
CNNC (China)	1.1	Centrifuge technology
Others (Japan, Brazil)	0.3	Centrifuge technology
World Total (million SWUs)	**42.8**	

Source: AREVA

REQUIREMENTS FOR URANIUM ENRICHMENT (10^6 SWU)

Country	1995	2000	2005
FRANCE	5.8	6	7.1
TOTAL FOR EUROPE	12.2	12	23.9
JAPAN	5.6	5.4	5.0
UNITED STATES	10.0	10.5	13.4
WORLD TOTAL	**29.6**	**30**	**47.7**

Source: AREVA / WNA

3.6. Fuel design and manufacturing

Designing fuel assemblies with high burnup capability and high performance level requires control over key design fields: thermal-mechanical performance of the pellets, pellet-cladding and cladding-water interfaces, which ensure the efficiency of heat generation, control over the metallurgy of zirconium and of advanced alloys, such as M5® for PWR products, which confers integrity and high performance properties on the assemblies.

The grids, key components in terms of heat removal, optimize mixing for good heat exchange between the claddings and the reactor coolant. The design and manufacturing of the skeletons meet the most demanding rigidity and robustness criteria. The continuity of rod-grid contact and the integrity of the fuel depend on this. The dimensional and geometrical stability of the fuel assemblies is essential for ensuring good handling ability during reloading operations.

The fineness of the fuel assembly and reactor core neutronic modeling and the deployment of gadolinium allow a heightened performance level for optimized core management.

After design, a fuel assembly is broadly the product of three steps: the manufacturing of the uranium pellets, the manufacturing of the fuel claddings and structural components, the insertion of the pellets into the claddings and the loading of the rods into the structures.

- Once extracted from the mine, natural uranium is enriched and transported in containers to the Fuel Manufacturing plant. The manufacturing of the uranium pellets starts by converting gaseous UF_6 into uranium dioxide powder. The powder is then pressed to form a cylinder, then sintered at 1780°C to make it into a compact and solid pellet.

- The claddings and structural components are made from zirconium. After the zirconium sponge is melted and forged, the grids, water channels and cans are manufactured from the flat products. The cylindrical pre-products are made into claddings and guide thimbles.

- The grids are assembled and the skeletons are put together. The claddings receive the pellets, and then the rods are filled with helium and sealed with plugs. Once the rods are put together, they join the skeletons to form a fuel assembly.

The following tables present the characteristics of AREVA fuel for PWR and BWR reactors. Mark-B™, Mark-BW™, AFA 3G™, HTP™, DUPLEX™ are AREVA NP trademarks and AGORA® and M5® are registered trademarks of AREVA NP. Figures of AREVA NP fuel assemblies for PWR and BWR are represented in chapter 1.

PWR design data

	CE-HTP ™	Mk-B-HTP ™	Mk-BW ™ 17	AFA 3G ™	AFA 3G ™	AGORA ® 5A(*)
Assembly geometry	14x14	15x15	17x17-12’	17x17-12’	17x17-14’	15x15
No. of fuel rods per assembly	176	208	264	264	264	205
Overall assembly length (mm)	3990	4214	4059	4060	4795	4061
Overall assembly width (mm)	206	217	214	214	214	214
Rod length (mm)	3715	3937	3865	3863.4	4497.3	3876
Rod outside diameter (mm)	11.18	10.92	9.5	9.5	9.5	10.77
Pellet length (mm)	11.0	11.9	10.2	13.46	13.46	11 / 13.5
Pellet outside diameter (mm)	9.66	9.49	8.19	8.19	8.19	9.33
Pellet density (g/cm^3 or TD)	10.52	10.52	10.52	10.4	10.4	10.52
Clad material	Zy4 / M5®	M5®	M5®	Zy4/M5®	Zy4/M5®	M5®
Clad thickness (mm)	0.67	0.635	0.57	0.57	0.57	0.635
Grid material	Zy4 / M5® / Alloy718	M5® / Alloy718	Zy4 / M5® / Alloy718	Zy4+Alloy718	Zy4+Alloy718	M5® + Alloy718
Max burnup (MWd/kgU)	up to 60	up to 60	up to 65	up to 60	up to 60	up to 65

(*) AGORA ® A features AFA 3G™ mixing grids
AGORA ® H features HTP™ mixing grids

PWR design data (continued)

	AGORA® 7A(*) / AGORA® 7H(*)	AGORA® 7HE(*) / AFA 3G™ LE(*)	AGORA® 4H(*)	AGORA® 6H(*)	AGORA® 8H(*)	HTP™
Assembly geometry	17x17-12'	17x17-13.8' (EPR™)	14x14-(16+1)	16x16-(20)	18x18-(24)	17x17-(24+1)
No. of fuel rods per assembly	264	265	179	236	300	264
Overall assembly length (mm)	4062	4805	2900	4827	4827	4057
Overall assembly width (mm)	214	214	197.2	229.6	229.6	214
Rod length (mm)	3863.4	4550	2635	4405	4405	3853
Rod outside diameter (mm)	9.5	9.5	10.77	10.75	9.5	9.55
Pellet length (mm)	13.46	13.5	11.0	11.0	9.8	9.37
Pellet outside diameter (mm)	8.19	8.19	9.11	9.11	8.05	8.17
Pellet density (g/cm^3 or TD)	10.52	10.52	10.45	10.45	10.45	10.45
Clad material	M5®	M5®	M5®	Duplex ™ / M5®	Duplex ™/ M5®	Optimised Zry4 / Modified Zry4 / Duplex ™ / M5®
Clad thickness (mm)	0.57	0.57	0.725	0.725	0.64	0.61
Grid material	M5®/ Alloy718	M5®/ Alloy718	M5® / Alloy718	M5® / HPA-4™ / Alloy718	M5® / HPA-4™ / Alloy718	Modified Zry4 / M5® / HPA-4™ / Alloy718
Max burnup (MWd/kgU)	up to 65	up to 65	up to 65	up to 65	up to 65	up to 65

BWR design data

	ATRIUM ™ 10A or B	ATRIUM ™ 10XP	ATRIUM ™ 10XM
Assembly geometry	10x10	10x10	10x10
Total number of rods per assembly	91	91	91
– Full length + part length fuel rods	83 + 8	81 + 10	79 + 12
– Water structure	Water channel replaces 3x3 fuel rod positions		
Overall assembly length (mm)	4470	4470	4470
Overall assembly width (mm)	134	134	134
Rod length (mm)	4081.4	4081.4	4081.4
Rod outside diameter (mm)	10.05	10.28	10.28
Pellet length (mm)	10.5	10.5	10.5
Pellet outside diameter (mm)	8.67	8.87	8.87
Pellet density (g/cm^3)	10.55 (liner) 10.45 (no liner)	10.6	10.6
Clad material	Zy2, LTP2, Fe enhanced Zr liner	LTP2 / Fe enhanced Zr liner	LTP2 / Fe enhanced Zr liner
Clad thickness (mm)	0.605	0.620	0.620
Grid material	Zy strips / Alloy718 springs	Alloy718	Alloy718
Average discharge burnup (MWd/kgU or HM)	65	66	66
Maximum assembly burnup (MWd/kgU or HM)	71	70	70

3.7. Nuclear Fuel Treatment (Reprocessing)

Used nuclear fuel treatment (reprocessing) operations are aimed at recovering materials (uranium and plutonium) that can be recycled, and packaging ultimate waste as effectively as possible.

3.7.1. Reprocessing

The following are the main stages of reprocessing as carried out at the La Hague plant:

- unloading and underwater interim storage of spent fuel assemblies in a pond for a minimum of three years to cool them and reduce their radioactivity,
- these assemblies are then sheared to break the fuel rods and grids into pieces and to separate the end pieces from the assembly,
- the pellets from the rod pieces are dissolved in boiling nitric acid in a rotary dissolver,
- the uranium and plutonium are extracted using an organic solvent (TBP) which allows them to be separated from the aqueous solution resulting from the dissolution process (liquid/liquid extraction),
- the rest of the solution containing the fission products and minor actinides (Np, Am, Cm) is then sent to a special workshop for vitrification and production of glass canisters,
- the pieces of cladding which do not dissolve (hulls) and the debris from the fuel assembly structural parts (in particular the end pieces) are rinsed and sent to the compacting workshop where they are pressed into compacted metal waste packages, forming a canister geometrically identical to the glass canister,
- the retrieved plutonium is transformed into plutonium oxide while the uranium is stored in the form of uranyl nitrate which will later be transformed into uranium oxide.

3.7.2. Characteristics of irradiated light water fuel (1300 MWe PWR class unit)

- Average burnup: 43,500 MWd/tU
- Average load factor: 85%
- Cooldown in the reactor pool: at least 180 days
- Used nuclear fuel characteristics 3 years after discharge:
  - Residual radioactivity:

- fission products:	1000 Ci/kgHM
- actinides:	17 Ci/kgHM

  - Residual power: 5 W/kgHM
  - Quantity of used nuclear fuel to reprocess: 30 tHMi/yr

- Average composition, per ton (for a burnup of 46,800 MWd/tU and initial enrichment of 4%):

- Uranium: 940 kg
- Plutonium 11 kg
- Minor actinides (Np, Am, Cm) 1 kg
- Fission products: 48 kg

3.7.3. Used nuclear fuel treatment (Reprocessing) plants

The commercial reprocessing of light water reactor fuel is commonly conducted on a large scale in France and the United Kingdom, and more recently in Japan.

Country / Operator	Area / Plant	Capacity (THM/yr)
France / AREVA	LA HAGUE / UP2 AND UP3	1700
Japan / JNFL	ROKKASHO-MURA / RRP	800
UK / NDA-BNGS	SELLAFIELD / THORP	900
Russia / Rosa-tom	CHELYABINSK / RT1	400

3.8. Recycling

Recycling valuable fissile materials recovered from used nuclear fuel reprocessing, namely plutonium and uranium, ensures sustainable resource management by reducing the uranium requirement of a nuclear power plant by up to 30%.

3.8.1. MOX fuel

MOX (**M**ixed **OX**ide) fuel is a mixture of plutonium and uranium oxides. Depleted uranium is generally used since plutonium is intended as a substitute for ^{235}U.

Today, plutonium is recycled as MOX fuel in light water reactors (PWRs and BWRs), making it possible to save enriched uranium by replacing it with plutonium, thereby preventing plutonium from ending up in ultimate waste.

Viewed from the outside, MOX fuel for PWRs or BWRs is identical to the enriched uranium fuel it replaces – same assembly structure, same spacing, same rods, claddings, grids, and springs. The pellets enclosed in the claddings are the same size – the only difference is their composition, and therefore, their manufacturing process.

NB: the characteristics given below are typical values and concern the materials recovered from PWR fuel assembles of the 17 x 17 array type (1300 MWe class units), initially enriched to 3.7% and having a burnup of 45,000 MWd/tUi. Similarly, the equivalent contents refer to typical conditions.

CHARACTERISTICS OF MIXED-OXIDE (MOX) FUEL

- Plutonium quality
  - The isotopic composition and quality of the plutonium is determined by the reactor type in which it is formed, the initial enrichment of uranium, the discharge burnup and the intermediate storage time of the reprocessed fuel.
  - Plutonium quality is expressed as ($^{239}Pu + ^{241}Pu$) 100/Pu_{tot}. The higher the plutonium quality, the higher the beginning-of-life reactivity, and the larger the reactivity slope versus burnup curve. This affects the average initial plutonium concentration required for achieving "burnup equivalence".
  - Below is an example of plutonium isotopic and quality figures. Initial enrichment is 3.7% ^{235}U, with burnup at 45,000 MWd/tiU, after a period of 3 years from when the fuel is unloaded from the reactor core.

^{238}Pu:	2.9%	^{241}Pu:	12.9%
^{239}Pu:	52.1%	^{242}Pu:	7.8%
^{240}Pu:	24.3%		

Plutonium quality = 65.0
Other average plutonium quality figures:

Magnox Plutonium ~75
Second generation Pu < 60
Weapons grade Plutonium ~95

- Equivalent Pu content (for the same energy produced):

$$T = \frac{\text{Mass of Pu} + {}^{241}\text{Am}}{\text{Mass of U} + \text{Pu} + {}^{241}\text{Am}} = 7.1\%\ \text{Pu, equivalent to } 3.7\%\ {}^{235}\text{U}$$

- Carrier material for plutonium

Three options for carrier material have been investigated for commercial plutonium recycling strategies: natural uranium (U_{nat}), tails uranium from the enrichment process (U_{tails}) and uranium recycled through used fuel reprocessing (U_{rep}).

Current MOX fuel is fabricated mainly with tails uranium as the carrier, for economic reasons. Typical ^{235}U enrichments in tails assays are 0.2-0.3%.

- Zoning in the assembly: a typical 17x17 PWR MOX fuel assembly with an average plutonium concentration of 7.2 % Pu includes:

Angle zones	4 x 3 fuel rods at 3.7% Pu
Peripheral zone	68 fuel rods at 5.2% Pu
Central zone	184 fuel rods at 8.2% Pu

The remaining locations within the assembly are dedicated to guide tubes (24) and a centrally located instrumentation tube.

- MOX licensing: in-core fuel management with MOX fuel assemblies is common practice in a large number of nuclear power plants in several countries in Europe and in Japan. Commercial reactors using around 30% MOX fuel assemblies in the core (up to around 50%) are in operation. Higher MOX loadings (up to 100%) are being investigated and their feasibility has already been demonstrated in numerous studies (EPR™ technology).

- A single recycling of plutonium increases the energy derived from initial uranium by about 12% and saves up to this percentage of natural uranium requirement and the energy associated with its conversion and enrichment processes as well.

MOX FUEL PLANTS

Country / Operator	Area / Plant	Capacity (THM/yr) (1)
In operation		
France / AREVA	Marcoule / MELOX	195
UK / NDA-BNGS	Sellafield / SMP	120
Under construction		
USA/SHAW-AREVA -MOX Services (2)	Savannah river	70
Planned		
Japan / JNFL	Rokkasho-Mura/ J-MOX	100
Russia (2)	Tomsk	70

(1) THM: Tons of Heavy Metal
(2) The objective of this MOX fuel program is to recycle US and Russian weapons-grade plutonium

3.8.2. U_{rep} fuel

U_{rep} (**Rep**rocessed **U**ranium) can be recycled either by blending with HEU (Highly Enriched Uranium) or by re-enrichment:

It is possible to recycle the uranium recovered from used Light Water Reactor fuel, either in LWRs or in other reactor types.

- By blending, the concentration of ^{235}U is increased to the level required in fresh fuel by diluting the HEU prior to fuel fabrication.
- Re-enrichment involves converting reprocessed uranium into UF_6 then feeding it to an enrichment plant prior to fuel fabrication.

French, German, Japanese, Belgian and Swiss LWRs are already using U_{rep} fuel.

CHARACTERISTICS OF REPROCESSED URANIUM FUEL (UREP)

During irradiation, the initial ^{235}U inventory is depleted by fission and neutron captures, while other minor uranium isotopes ^{232}U, ^{234}U and ^{236}U are built up, along with the very short-lived ^{237}U (6.7 days). The minor uranium isotopes are each important for various reasons:

- ^{232}U has a decay period of only 68.9 years, but its decay chain includes ^{208}Ti which represents a very important source of occupational exposure in fuel fabrication operations,

3

^{234}U has a small but significant role as an alpha emitter which contributes to internal dose uptake in fuel fabrication operations, and has a small neutron absorbing effect in the reactor,

- ^{236}U is important as a neutron absorber; its neutron capture effect needs to be compensated by over-enriching the ^{235}U,

- Typical isotopic composition of U_{rep} based on LWR fuel with 3.7% initial enrichment, 45 GWd/tU burn-up and 10 years after discharge :

  ^{232}U: 3.24 ppb
  ^{234}U: 0.02%
  ^{235}U: 0.72%
  ^{236}U: 0.52%
  ^{238}U: 98.74%

- Equivalent enrichment in ^{235}U (for the same energy produced) 4.1% Urep equivalent to 3.7% U_{nat},

- A single recycling of Urep increases the energy derived from initial uranium by some 10% and saves up to this percentage in natural uranium requirements.

4

RADIOLOGICAL PROTECTION
NUCLEAR SAFETY

4.1. Attenuation of radiation

NB: no allowance is made here for the case of neutrons, which requires the use of relatively complex techniques, and which is rarely encountered in most reactors (except during the reactor design and, in the case of the EPR™ reactor, during access to the Reactor Building for power operation). The various types of basic data discussed below give only orders of magnitude and are only valid for gamma rays with energy values ranging from 0.5 to several MeV.

4.1.1. Effects of distance

Like all flux[1], the γ radiation is proportional to the power of the source S (MeV/s) and inversely proportional to the square of the distance D between the source and the receiver:

$$\boxed{\Phi = \frac{S}{4\pi D^2}\ e}$$

- This equation is essentially valid for dimensions of S not exceeding D/20.
- For larger dimensions, point-by-point integration is necessary; nomographs exist in the specialized literature[2].
- The dose received can be deduced from this relation: a flux of 1×10^6 MeV/cm^2.s $\simeq$ 20 mSv/h (see Chapter 4.2.).

4.1.2. Absorption by matter

Attenuation of the flux Φ varies exponentially with the thickness through which it passes:

$$\boxed{\Phi = \Phi_o e^{-K(\rho X)} B}$$

where:

Φ_o is the flux without protection,

Φ is the flux with radiation protection,

ρ is the density of the material, and

x is the thickness of the material.

(1) Number of "particles" that pass, per unit of time, through a sphere of unitary cross-section. The flux is equal to the product of the number of particles per cm^3 and their average velocity. The AFNOR term is "flux density" or "fluence density". The flux received during a given time is the "fluence", or, more commonly, the "integrated flux".
(2) Reactor Handbook, Vol. III Part 8: "Shielding", or INSTN Génie Atomique, Vol. 1 Part C.

The exponent is equal to the product of:

- A coefficient K, **the total macroscopic cross-section of mass absorption**, normally written as Σ/ρ (cm^2/g), which is found in tables. Slightly larger for elements with a high atomic number, it decreases with the γ ray energy (see the ta urface area.

The entire term is multiplied by a correction factor B (buildup factor), which takes into account the γ rays that are only subjected to diffusions and are not exponentially attenuated, but which contribute to increasing the flux Φ. Tables give the values for B[3].

TOTAL ABSORPTION CROSS-SECTION (K)

(including Compton diffusion)
for several materials, in cm^2/g

Energy (MeV)	Water $\rho = 1$	Concrete $\rho = 2.3$	Iron $\rho = 7.8$	Lead $\rho = 11.3$
0.3	0.119	0.107	0.110	0.378
0.5	0.0967	0.0870	0.0840	0.152
0.6	0.0894	0.0804	0.0769	0.119
0.8	0.0786	0.0706	0.0668	0.0866
1.0	0.0706	0.0635	0.0598	0.0703
1.2	0.0642	0.0578	0.0537	0.0610
1.5	0.0576	0.0517	0.0484	0.0523
1.7	0.0538	0.0482	0.0452	0.0485
2.0	0.0493	0.0445	0.0422	0.0413

[3] *Reactor Handbook, Vol. III, Part B, or INSTN Génie Atomique Vol. 1, Part C. As a first approximation, this factor can be taken to be equal to Σx (so-called "linear" buildup).*

NECESSARY THICKNESSES TO OBTAIN A GIVEN ATTENUATION (in cm)

Attenuation	Water		Concrete		Iron		Lead	
	A	B	A	B	A	B	A	B
10	70 (35)	120 (83)	25 (17)	50 (36)	10 (6)	14 (9)	9 (3)	6 (5)
100	115	220	45	90	16	25	9	11
1.000	160	300	65	130	22	35	12	17
100.000	240	500	100	200	34	56	20	28
10.000.000	320	700	140	280	46	77	27	39

A = γ rays with an energy of 1.2 MeV (^{60}Co)
B = γ rays with an energy of 6 MeV (^{16}N, emitted with the reactor in operation)

NB: the figure in parentheses in the first line is that given by exponential attenuation without application of the buildup factor (channeled geometry, for which the diffused γ rays are considered to have been absorbed).

Example: determine the dose rate 50 cm from a 3.7 TBq ^{60}Co source and the influence of a protective lead screen 20 cm thick.

In the case of ^{60}Co, two γ rays with an energy of 1.2 MeV (in reality, one with 1.17 MeV and one with 1.33 MeV) are emitted at each disintegration. Since the source equals 3.7×10^{12} disintegrations per second, the flux received at a distance of 50 cm is:

$$\Phi = \frac{3.7 \times 10^{12} \times 2 \times 1.2}{4\pi \times 50^2} = 2.8 \times 10^8 \text{ MeV / cm}^2\text{.s}$$

Using the value of the dose factor for γ rays, η (see Chapter 4.2.), the dose rate is equal to $2.8 \times 10^8 \times 2 \times 10^{-8} = 5.60$ Sv/h. The attenuation due to the lead shielding, ignoring the effect of the buildup factor is:

$$e^{-\Sigma\rho x} = e^{-0.061 \times 11.3 \times 20} = e^{-13.9} = 9 \times 10^{-7}$$

With a linear buildup factor, the result becomes $9 \times 10^{-7} \times 13.9 = 1.3 \times 10^{-5}$ (whereas the above table gives a value of 1×10^{-5}).

4.2. Dosimetry

4.2.1. Physical units

Unit	Type of radiation measured	Medium concerned	Basis of definition	Remarks
C/kg	X-rays γ rays (< 3 MeV)	Air	No. of ionization/kg of air	Exposure dose
gray, Gy	Any	Any	1 joule/kg	Absorbed dose Common unit for all categories of radiation
sievert, Sv	Any	Any organic tissue (including bones)	No. of gray times the radiological weighting factor	Dose equivalent

Activity

This term characterizes the radioactivity of a source by the number of disintegrations occurring per unit of time:

- **1 becquerel (Bq)** = 1 disintegration/s,
- **1 curie (Ci)** = 3.7 x 10^{10} disintegrations/s (no longer used in SI system, still used in USA).

3 x 10^6 g of ^{238}U, 16 g of ^{239}Pu, 1 g of ^{226}Ra, and 0.00088 g of ^{60}Co all have the same activity: one curie or 37 GBq.

Exposure Dose

This is the quantity of X or γ radiation received at a point, determined on the basis of its property of producing ions in the air. The exposure dose is measured in C/Kg.

Roentgen

The roentgen is an old unit, whose use is now discouraged, for measuring **exposure**. It is based on the energy released by ionization in the air, and in practice is not much different from the rad, when applied to muscles, bones, and fatty tissue. In the SI system, the roentgen is replaced by coulomb/kg and the rad is replaced by Gray.

1 roentgen = 2.58 x 10^{-4} C/kg.
1 rad = 10^{-2} Gy

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4.2.2. Biological units

TERMINOLOGY

Irradiation
It is generally admitted that the impact of different types of radiation on the human body are, ultimately, directly related to their ionizing powers, i.e. their interactions with the peripheral electrons of the atoms belonging to the different biological tissues exposed to this radiation. The harmful effect is proportional to the energy lost by the radiation in the human body.

Absorbed Dose
The absorbed dose is the quantity of energy communicated by the radiation to a material, independent of its condition or composition:

$$\boxed{D = \frac{dW_D}{dm} = \frac{dW_D}{\rho . dV}}$$

where dW_D is the energy absorbed by an element of volume dV and $dm = \rho dV$ is the mass of the material of density ρ in the elementary volume.
The absorbed dose measurement unit since 1975, in the SI system, is the **gray** (Gy). One gray equals 1 joule/kg or 100 rad.

$$\boxed{1 \text{ Gy} = 1 \text{ J/kg} = 6.24 \times 10^9 \text{ MeV/g}}$$

Dose Rate
The dose rate is the dose delivered per unit of time: $\Delta = dD/dt$. The measurement unit is:

$$\boxed{1 \text{ Gy/s} = 1 \times 10^{-3} \text{ W/g} = 100 \text{ rad/s} = 6.24 \times 10^9 \text{ MeV/g.s}}$$

External exposure occurs whenever a person is on the path of radiation emitted by a radioactive source located outside the human body.

External contamination occurs each time that radioactive substances are brought into contact with the skin.

Internal contamination occurs each time that radioactive substances penetrate inside the human body (by inhalation, swallowing, through open wounds, or by passing through the skin).

DOSE EQUIVALENT (H)

The dose equivalent (in general incorrectly called the dose) is the quantity that makes it possible to assess the degree to which irradiation affects living organisms. It takes account of the fact that to produce the same biological effect, the absorbed dose varies depending on the type of radiation, the nuclide, the irradiation, etc. The dose equivalent makes it possible to compare the effects of different types of radiation and, in particular, to take into account (by summing) the combined effects of external and internal exposures.

The unit of measurement for the dose equivalent is the **Sievert** (Sv), one Sievert being defined as the radiation dose that produces the same biological effect as the absorption of one gray of 250 kV X-rays. In practice, for γ and X rays there is an equivalence between the Sv and the Gray.

H cannot be directly measured. Instead, it is computed on the basis of the absorbed dose (in Gy), to which two dimensionless coefficients are applied: the radiological weighting factor (W_R) and the N factor:

$$\boxed{H = D \times W_R \times N}$$

Type of radiation	W_R
Photons	1
Electrons, muons	1
Neutrons < 10 KeV	5
10 keV < Neutrons < 100 keV	10
100 keV < Neutrons < 2 MeV	20
2 MeV < neutrons < 20 MeV	10
Neutrons > 20 MeV	5
Protons > 2 MeV	5
Alphas, fission fragments, heavy nuclides	20

W_R: This factor takes into account the biological effect for different types of radiations and energies (for a given absorbed dose, α particles cause much more damage than X-rays). N is the product of all other modifying factors. Such factors might take into account, for example, the absorbed dose rate and the fractionation. At present, the value 1 is assigned to N.

COLLECTIVE DOSE

The collective dose is the sum of the doses received by a group of people, expressed in "**man.sieverts**". This quantity serves to evaluate long-term effects. Thus, 10 **man.sieverts** correspond, in terms of overall damage, to 10 people having each received one Sv, or 100 people having each received 0.1 Sv. This unit only applies to a group of individuals and has no meaning for any particular individual in the group.

EXPOSURE VALUES (in mSv/yr)

- Terrestrial materials: 0.50 to 5, depending on the regions (India and Nepal: 2.8)
- Cosmic rays: at sea level, 0.30; at 1500 m altitude, 0.80
- Natural radiation of the human body: 0.24
- Irradiation due to human activities:
  - Radiological examination: 0.05 to 10 (average of 0.50)
  - Color television: 0.10
  - Transatlantic flight: 0.03 to 0.05 per flight
  - Luminous watch dials: 0.02
  - Fallout from nuclear bomb tests: 0.03
  - Nuclear power plants: < 0.01

(For the maximum allowable doses, see Chapter 4.3.)

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RADIATION FLUX EQUIVALENCES

For external irradiation, the dose factor η is used to convert ambient radiation flux into the received dose rate.

Radiation type	Approximate dose factor (Sv)
γ (1.2 MeV)	*1 MeV/cm².s →2 x 10^{-8}
Fast neutrons (≥ 1 MeV)	1 n/cm².s →1.3 x 10^{-6}
Thermal neutrons	1 n/cm².s → 4 x 10^{-8}

** Depending on the energy of the γ rays*

4.3. Regulations and Standards

4.3.1. Regulations

The International Commission on Radiological Protection (ICRP) formulates and publishes recommendations at international level. These recommendations have been adopted in the form of standards in the different national regulations. This section only makes reference to French regulations.

In France, the main decrees and application ordinances concerning people working in environments subject to irradiation are as follows:

- Article 37 of the Euratom treaty: obligation of each member country to inform the Commission of **any plan to release radioactive effluents** that could affect the territory of any other member country,
- Ordinance no. 2001-270 dated March 28, 2001, regarding transposition of European requirements for protection against radiation,
- Decree no. 2002-460 dated April 4, 2002, regarding general protection of people against radiation,
- Decree no. 2003-296 dated March 31, 2003, regarding the protection of workers against radiation hazards,
- In general, surveillance within nuclear sites is carried out by and under the responsibility of the site operator. Offsite inspections are carried out by the ASN (Autorité Sureté Nucléaire) and IRSN (Institut Radioprotection et Sureté Nucléaire), which can, under their responsibility, request assistance from the facility operator.

4.3.2. Definition of the people to whom the standards apply

The regulations distinguish three distinct categories of people to whom the standards apply:

- **Category A workers:** these are people whose regular working conditions are liable to involve exceeding three-tenths of the prescribed annual exposure limits,
- **Category B workers:** these are people whose regular working conditions should not normally involve exceeding three-tenths of the prescribed annual exposure limits,
- **The general public**, which forms the rest of the population.

4.3.3. Standards

- Dose limits (in mSv) for 12 months in succession.

Exposure	Category A	Category B
Whole body	20	6
Hands, forearms and ankles	500	150
Skin	500	150
Crystalline lens of the eye	150	50
Public	1	

4.3.4. Marking of controlled areas[1]

Green Area (French practice)
In terms of the external and internal exposure of the whole body, the **supervised area** is taken as meaning the total dose likely to be received in 1 hour where this is less than 0.0075 mSv. Above this value, and up to 0.025 mSv, the term "controlled **green area**" is used.

Yellow Area (2.5 x 10^{-2} to 2 mSv/h)
The amount of time spent in such areas is subject to regulations:
– precautions to avoid contamination,
– controlled time spent in the area.

Orange Area (2 mSv/h to 100 mSv/h)
The amount of time spent in such areas is limited:
– access subject to special authorization,
– special radiation protection measures.

Red Area (over 100 mSv/h)
Access to such areas is normally forbidden and physically prevented.

(1) The zoning principle exists for all nuclear power plants, but the color areas and corresponding dose rates mentioned here are applicable only to French PWR units.

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4.4. General principles - Definitions

NUCLEAR SAFETY

Nuclear Safety is the set of provisions made at all stages in the design, construction, operation and decommissioning of nuclear facilities to protect man and his environment against the dispersal of radioactive substances under all circumstances, or in other words to:

- ensure the facilities are operating normally,
- prevent incidents and accidents,
- mitigate the consequences of any incidents or accidents that may occur.

To confine radioactive products and prevent their dispersal in the event of incidents or accidents, Nuclear Safety is based on the following two main principles:

- application, at all stages of plant life, of the **defense-in-depth principle** which consists in systematically considering possible failures and taking appropriate measures to prevent their consequence through successive lines of defense,
- interposition of **leaktight barriers** between radioactive products and the environment.

DEFENSE-IN-DEPTH PRINCIPLE

Application of the concept of defense-in-depth in the design of a plant provides a series of levels of defense (inherent features, equipment and procedures) aimed at preventing accidents and ensuring appropriate protection in the event that prevention fails.

It consists of five levels:

- The first level, **"Prevention"**, is a combination of conservative design, quality assurance and surveillance activities in order to prevent departure from normal operation,
- The second level, **"Monitoring and Detection"**, is the detection and identification of protection devices to cope with deviations from normal operation,
- The third level **"Accident management"** is afforded by those features and systems that are provided to mitigate design basis accidents and consequently to prevent their evolution into severe accidents,
- The fourth level **"Severe accident mitigation"** ensures that radioactive releases are kept as low as practicable when the first three levels have failed. The most important objective of this level is the protection of the containment function,
- The fifth level **"Mitigation of the radiological consequences of potential releases of radioactive materials"** requires the provision of an adequately equipped emergency control centre and plans for off-site emergency response.

RADIOLOGICAL BARRIERS

When a nuclear reactor is in operation, there is a build-up of radioactive products in the fuel. Therefore, there are three strong, leaktight and independent **"barriers"** to prevent their release into the environment:

- First barrier: fuel cladding,
- Second barrier: main primary system,
- Third barrier: reactor building containment.

FUNDAMENTAL SAFETY FUNCTIONS

To prevent barrier failure or to limit the consequences of their failure it is necessary to monitor and maintain the three fundamental safety functions:

- reactivity control,
- heat removal,
- radioactive material confinement.

SINGLE FAILURE CRITERION

The single failure criterion is implemented in order to ensure that more than the minimum number of components is provided to carry out essential functions. This redundancy requirement helps to ensure the high reliability of safety systems designed to maintain the plant within its deterministic design basis.
The systems which perform safety functions during incidents (for instance the protection systems) or accidents (for instance the safeguard systems) and their supporting systems (electrical power supply, cooling and control air) must be capable of performing their functions in the event of failure of any one of their components.

The failures taken into consideration are:

- For electrical systems, it is assumed that any component has failed when the system is required to operate.
- For mechanical systems carrying fluids: a distinction is drawn between **active components** which require the movement of some of their components to perform the required function (pumps, motor-operated valves, etc.) and **passive components** (pipes, heat exchangers, tanks and check valves).

Any active failure may be the refusal of an active component to operate. A passive failure may be either a break in the retaining wall or a mechanical failure affecting the fluid flow within the system.
A system is designed according to the single failure criterion if it is able to fulfill its functions in spite of a random failure independent of the event, the control of which necessitates system operation. The postulated single failure can be active in the short and long term or passive over the long term (> 24 hours).

Note: *for the EPR™ reactor as well as for the KERENA™ reactor and all German NPP's in operation, in addition to the single failure criterion (considered as a N+1 criterion) and to improve plant availability, it is assumed that a train of the safeguard systems is undergoing maintenance (N+2 criterion). As a consequence, two safeguard trains are assumed to be unavailable in the analyses of the design basis conditions.*

SAFE STATE

The safety analysis is performed up to a safe shutdown state.

The safe shutdown state is defined as a state in which:

- the core is subcritical,
- radioactivity release is maintained within the prescribed limits,
- decay heat is removed durably.

When possible, safe shutdown is reached when the residual heat removal system (RHRS) is connected. In that state the core heat removal is durable. Another safe state can be Primary System Feed and Bleed when consistent with the prescribed limits of radioactivity release.

SAFETY CLASSIFICATION

The objectives of safety classification are as follows:

- Contribute through design, manufacturing and operating requirements to the quality of the systems, components and construction work that contribute to plant safety,
- Obtain a sufficient level of quality of the safety systems and components under all their expected operating conditions. The most stringent requirements correspond to the most important safety functions.

Safety categorization and classification is carried out on the following basis:

- definition of safety functions,
- identification of equipment and structures involved in each function,
- assignment of each item of equipment or each structure to a safety category, generally according to the highest safety level of the function it has to perform.

The associated requirements may concern:

- systems: application of the single failure criterion, physical separation, emergency power supply, periodic tests,
- components: qualification, use of design and construction codes,
- or both: earthquake design, quality assurance.

DETERMINISTIC APPROACH

In the deterministic approach, buildings, systems and components are engineered taking into account the following events:

- Events of internal origin:
  - incidents and accidents (e.g. failure of plant systems),
  - other events of internal origin that can also affect plant safety: fire, high energy piping rupture, high energy missile, flooding, etc.

- Events of external origin (man-made or natural) such as aircraft crash, explosion, earthquake, flooding, extreme meteorological conditions.

Any event occurring in a nuclear power plant must be analyzed with two aspects in mind:

- the consequences for the core and the radioactive releases into the environment (radiological consequences),
- the consequences on pressure vessels and on safety classified buildings.

CORE AND RADIOLOGICAL CONSEQUENCES

The plant operating conditions are the different configurations that can be encountered, either as the result of a voluntary action by the operators or due to an abnormal "event" [e.g. the event consisting of a break in a reactor coolant system pipe places the NSSS in the "loss of coolant accident" (LOCA) condition].

Not all possible events have the same probability of occurrence or the same consequences. The operating conditions are classified according to their frequency of occurrence and the gravity of their consequences. The more probable an event, the less severe the consequences should be.

The events taken into consideration in the deterministic approach are divided into categories. To simplify this approach similar events are gathered under some representative initiating events. The considered transients guarantee the acceptability of transients that are not studied.

The Design Basis Conditions (DBC) are divided into four categories as follows:

- Condition I: Normal reactor operation and normal operating transients,
- Condition II: Incidents of moderate frequency,
- Condition III: Infrequent incidents,
- Condition IV: Limiting faults.

Accidents are analyzed on the basis of a set of conservative assumptions about the integrity of the three barriers.

The analysis of condition II to IV occurrences lays down the reactor protection system requirements and determines the setpoints of this system and associated systems.

The analysis of condition III and IV occurrences ensures that the engineered safeguard systems have been correctly designed.

Furthermore, special measures are implemented to cover the total loss of redundant systems used under condition I and II operating conditions. Specific assumptions are used for the analysis of these complementary conditions and for the design of the resulting specific means implemented to mitigate them.

In addition, operating conditions not considered to be plausible are taken into account to ensure the protection of the public in the event of a severe and hypothetical accident, with an extremely low probability of occurrence. These analyses are carried out with realistic assumptions.

The objective for operating plants in France is that the following dose equivalents measured at the site boundary shall not be exceeded, based on a two-hour residence time and on calculations carried out with realistic assumptions:

- Condition III: 5 mSv for the whole body and 15 mSv for the thyroid equivalent dose,
- Condition IV: 150 mSv for the whole body and 450 mSv for the thyroid equivalent dose.

For the ™, the incidents and accidents considered at the design stage are classified as **Design Basis Conditions** (DBC-2 to DBC-4) and **Design Extension Conditions** (DEC-A and DEC-B).

The DEC-A events are essentially related to the prevention of core melt. They are event combinations including multiple failure events such as, for example, initiating events combined with a **Common Cause Failure** (CCF) of a required safety system. These event combinations are called "complex sequences".

The DEC-B events, also called "severe accidents", assume that a low-pressure core melt has occurred and relate to the prevention of large radioactive releases. The mitigating means implemented during severe accidents must be consistent with the emergency plan.

The objectives associated with conditions DBC-3 and DBC-4 are as follows:

- Effective dose 10 mSv
- Thyroid dose 100 mSv

CONSEQUENCES FOR PRESSURE VESSELS

The order of November 10, 1999 ("Arrêté Exploitation") constitutes the regulatory reference for operation of the Main Primary and Main Secondary systems. It identifies three categories of equipment operating conditions according to a probabilistic approach.

The three categories with the corresponding main requirements are :

- Category 2: Normal operating transients – the maximum pressure remains inferior to design pressure,
- Category 3: Exceptional situations – Overpressure protection devices shall limit pressure to 110% of the design pressure if all safety valves are available and to a maximum of 120% of design pressure in the event of unavailability of some safety valves,
- Category 4: Highly improbable situations – Loss of integrity of equipment shall be avoided.

INTERNAL HAZARDS

Internal hazards are events that originate on the plant site and have the potential to cause adverse conditions or even damage inside or on safety classified buildings. These effects can potentially lead to a common cause failure within the systems used to reach and maintain the plant in a safe state.

The aim is to sufficiently protect the equipment needed for the three main safety functions from the unacceptable effects of internal hazards.

The internal hazards taken into account for the design basis and the layout of safety-related equipment are the following:

- failure of pipes,
- failure of vessels, tanks, pumps and valves,
- missiles,
- load drop,
- internal explosion,
- fire,
- internal flooding.

Some hazards are avoided by administrative procedures or by design. The consequences of others are taken into account in the design of the plant through separation rules, adapted layout and equipment design to withstand loads.

EXTERNAL HAZARDS

External hazards, of natural origin or arising from human activity, potentially affect the safety of the plant. These hazards (nature and level) depend on the site under consideration.

The objective of the design provisions is to ensure that the safety functions of the systems and components required to return the plant to the safe shutdown state will not be unacceptably affected. The combination of an externally generated hazard with an independent DBC is not considered.

At the design stage, special attention is paid to three hazards: earthquake, aircraft crash, and external explosion. They are taken into account on the basis of standardized and/or specific load combinations for the design of some civil works structures and/or equipment.

Other external hazards liable to condition the nuclear island design are taken into account at the design stage according to site specific characteristics:

- snow, wind and rain conditions,
- air temperature and extreme relative humidity conditions,
- extreme temperature conditions of the heat sink,
- lightning and electromagnetic interference.

PROBABILISTIC APPROACH

The deterministic approach is supplemented by **Probabilistic Safety Analyses** (PSA). They identify, through systematic analysis, any possible sequence of events, including those not taken into account in the deterministic approach, which could lead to undesirable events such as core damage or large release. They make it possible to reveal weak points in the design and to check that the probability of a severe accident is acceptably low.

For new generation plants, PSA is performed for licensing to check that the overall Core Damage Frequency and Large Release Frequency goals are met.

RISK INFORMED APPROACH

A Risk-Informed approach to regulatory decision-making is a process whereby risk insights, in particular the results and findings resulting from PSA, are considered together with deterministic factors to establish optimized safety requirements. When appropriate, the Risk-Informed approach can identify areas of unnecessary conservatism in the deterministic approaches. However, the Risk-Informed approach can also identify areas with insufficient conservatism and the need for additional safety requirements. PSA is used to identify risk significant items.

The main Risk-Informed applications relate to:

- Operating Technical Specification.
- In-Service Inspection.
- In-Service Testing.
- Reliability Centered Maintenance.

4.5. Technical Regulations, codes and standards

4.5.1. French general technical regulations

The general technical regulations include all texts of a general nature laying down the technical rules in the field of nuclear safety; they include statutory laws (administrative orders) such as the law of June 13, 2006 (on transparency and safety in the nuclear field, commonly referred to as the TSN law) and para-regulatory documents (circulars, basic safety rules (RFS), guides).

MINISTERIAL AND INTERMINISTERIAL ORDERS

- **Pressure vessels**

Basic Nuclear Installations (BNI) house two types of pressure vessels: those which are specifically nuclear, in other words those which contain radioactive products, and those which are more conventional and are not specific to nuclear facilities.

The applicable regulations are detailed in the following table:

<table>
<tr><th></th><th colspan="2">Nuclear</th><th></th><th>Conventional</th></tr>
<tr><th></th><th>Main primary system of pressurized water reactors</th><th>Main secondary system of pressurized water reactors</th><th>Other equipment</th><th></th></tr>
<tr><td rowspan="2">Construction</td><td>Decree of April 2, 1926
Order of February 26, 1974</td><td>Decree of April 2, 1926
RFS II-3.8 (June 8, 1990)</td><td>Decree of April 2, 1926
Decree of January 18, 1943
Or Decree no. 99-1046 of December 13, 1999</td><td>Decree no. 99-1046 of December 13, 1999</td></tr>
<tr><td colspan="3">or Order of December 12, 2005 (known as the “ESPN Order”)</td><td></td></tr>
<tr><td>Operation</td><td colspan="2">Order of November 10, 1999</td><td>Decree of April 2, 1926
Decree of January 18, 1943 (1)</td><td>Decree no. 99-1046 of December 13, 1999
Order of March 15, 2000</td></tr>
</table>

(1) Starting from 2011, the ESPN order will apply to the operation of nuclear pressure vessels, with the exception of main primary and secondary systems.

• Quality Assurance

Concerning quality, the ministerial order and circular of August 10, 1984 stipulate the general rules for quality assurance and the organisation to be followed by operators at the BNI design, construction and operating stages.

• Prevention of the nuisances and external risks resulting from operation of a Basic Nuclear Installation

The operation of a nuclear facility can lead to nuisances and risks for the overall environment, notably for the surrounding installations and those working in them, but also for the public and the environment. It aims at preventing and limiting the risks for the installations by making sure that the following are applied:

- the order of December 31, 1999 laying down the general technical rules intended to prevent and limit the nuisances and external risks resulting from operation of the basic nuclear installations, modified by the order of January 31, 2006,
- the legislation applying to Installations Classified for Environmental Protection (ICPE) for those installations within the perimeter of the BNI.

TEXTS ISSUED BY THE FRENCH NUCLEAR SAFETY AUTHORITY (ASN)

• Statutory decisions of a technical nature

In application of the law of June 13, 2006, the ASN can make decisions to supplement the terms and conditions of application of decrees and orders governing nuclear safety and radiation protection, with the exception of those concerning occupational medicine.

Those relating to nuclear safety are ratified by the Ministers in charge of nuclear safety and those relating to radiation protection by the Ministers in charge of radiation protection.

• Basic Safety Rules and ASN guides

The ASN has issued **Basic Safety Rules** (RFS: “Règles Fondamentales de Sûreté”) on various technical subjects concerning both PWRs and other BNIs. These rules constitute recommendations identifying the safety goals to be reached and describing accepted practice which ASN deems compatible with these aims.

They are not, strictly speaking, regulatory documents. A plant operator may decide not to adopt the provisions laid down in a Basic Safety Rule, providing it can demonstrate that the safety goals underlying the rule can be reached by alternative means, which it has to propose.

There are currently about forty Basic Safety Rules, which, together with the other technical rules issued by the ASN, can be consulted on the ASN website (www.asn.fr).

Due to the changes currently being made to the general technical regulations, the Basic Safety Rules will gradually be replaced by safety guides.

- **Technical Guidelines for the design and construction of the next generation of PWRs**

The Technical Guidelines for the design and construction of the next generation of Nuclear Power Plants with Pressurized Water Reactors (adopted during the GPR/German experts plenary meetings held on October 19 and 26, 2000) present the opinion of the **French Standing Group for Nuclear Reactors** (GPR: “Groupe Permanent Réacteur”) on the safety philosophy and approach, as well as the general safety requirements to be applied for the design and construction of the next generation of nuclear power plants of the PWR (pressurized water reactor) type.

The EPR ™ reactor shall comply with these Technical Guidelines which were issued on September 28, 2004 through a letter issued by the Head of the nuclear safety and the radiation protection.

FRENCH NUCLEAR INDUSTRY CODES AND STANDARDS

French regulatory practice relative to nuclear safety requires that plant operators have a written presentation of the rules, codes and standards corresponding to the industrial practice which is implemented in the design, manufacturing and commissioning of safety-related equipment.

The French nuclear industry (AFCEN with EDF and designers) has issued a complete set of design and construction rules known as **“RCCs”** (“Règles de Conception et de Construction” – Design and Construction Rules), governing all aspects of the design, manufacturing and construction of PWR nuclear islands.

The collection of RCCs covering PWRs is in five volumes: RCC-I (Fire Protection), RCC-G (Civil Work), RCC-M (Mechanical Equipment), RCC-E (Electrical Equipment) and RCC-C (Nuclear Fuel).

In addition, a code of “mechanical equipment in-service surveillance rules” **(RSE-M)** was drafted to deal with operation of this equipment. This document was modified recently, notably to bring it into line with the order of November 10, 1999 related to the surveillance of the main primary and secondary systems of pressurized water reactors during operation.

Drafting these documents is the responsibility of industry, not the ASN. The ASN nonetheless examines them to ensure that they comply with the general technical regulations, which in most cases leads to drafting of an RFS or a decision which thus recognises overall acceptability on the date of the version in question.

In the field of pressure vessels, the **European Pressure Equipment Directive** (PED) dated May 29, 1997 gives essential requirements which remain the minimum basis for nuclear safety related equipment, safety rules taking precedence in case of conflict.

Nevertheless, being issued in a context where nuclear safety aspects remain under the responsibility of National states, nuclear pressure equipment was considered outside the scope of the PED, allowing every national body to issue specific nuclear safety regulations.

In France, the same approach was followed under the form of a specific Order dated December 12, 2005, which refers to PED provisions, supplementing them depending on safety aspects and potential radioactive releases.

In order to satisfy these requirements, specific industrial codes and standards can be applied, subject to a justification demonstrating compliance with the essential requirements in the regulation, or a reference to harmonized standards may be chosen, completed where necessary in the nuclear field to fulfill specific rules in the nuclear regulations.

4.5.2. German Legal and Regulatory Rules

Following a brief introduction to the main laws, the regulations, guidelines and technical rules which are relevant for nuclear licensing and supervision in Germany are given.

LEGISLATIVE BACKGROUND

With the amendment of the **Atomic Energy Act (Atomgesetz, AtG)** – which entered into force on 22 April 2002 - essential parts of German Nuclear Energy Law have been redesigned.

The new goal of the Act is to phase out the use of nuclear energy for the commercial generation of electricity in an orderly manner. Still, the existing nuclear power plants will have to be operated at a high level of safety for their residual operating lives.

For the implementation of this aim, the AtG stipulates e. g. that no further licenses shall be granted for the construction and operation of new nuclear power plants and facilities for the reprocessing of irradiated nuclear fuels; operating licences for existing commercial reactors will become extinct as soon as certain electricity amounts have been exhausted which have been determined separately for each nuclear power plant. Nevertheless, the current licensing procedure continues to be applicable as both essential modifications of existing installations and the construction of non-industrial new facilities (e. g. research reactors such as FRM II in Munich) require licensing.

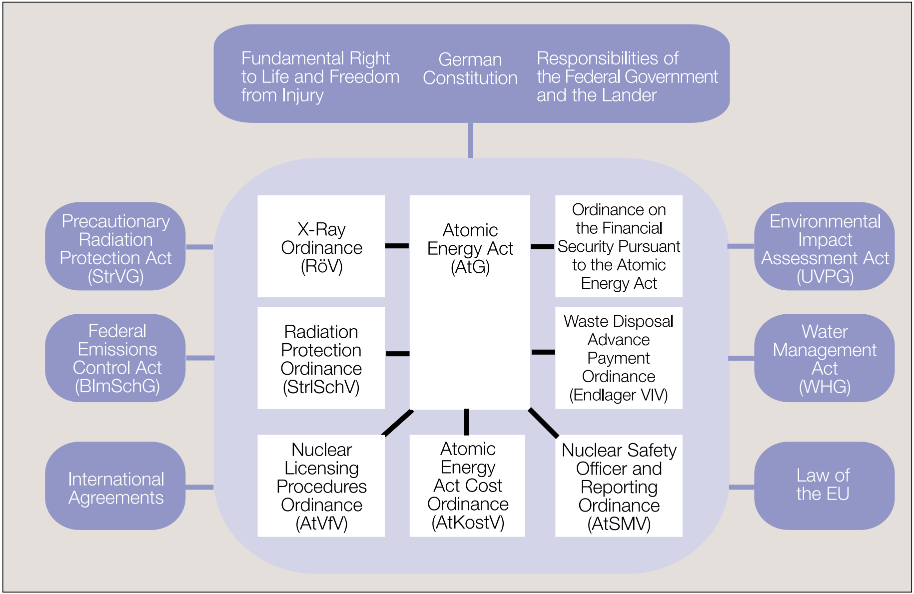

Fig 4.5-1 Legal Basis of peaceful use of nuclear power

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GERMAN CONSTITUTION

According to the German Constitution, the Länder are responsible for implementation of the Atomic Energy Act (AtG) on behalf of the Federal Government (federal executive administration). Each Land is under federal supervision to execute this Act, in compliance with the other Länder. The Federal Government is allowed to issue instructions concerning the expedience and legality of the execution.

THE ATOMIC ENERGY ACT

The Atomic Energy Act (AtG) sets forth the legal prerequisites for the peaceful use of nuclear power. The Act came into force in 1960 and has been amended several times since then, due to technical/scientific developments and political changes.

The goal of this Act is the protection of the population against the danger of nuclear power and it guarantees the correct phase-out of the use of nuclear energy.

The Atomic Energy Act regulates the following:

- **Section 3 AtG**: import and export of nuclear fuel,
- **Sections 4, 4a, 4b AtG**: transport of nuclear materials,
- **Section 5 AtG**: authorized ownership and safe custody of nuclear fuel,
- **Section 6 AtG**: storage of nuclear fuel outside state custody,
- **Section 7 AtG**: erection, operation, holding and decommissioning of stationary nuclear facilities for the production, treatment, fission or reconditioning of irradiated nuclear fuel,
- **Section 9 AtG**: treatment, processing or other utilization of nuclear fuel outside installations in need of a license,
- **Section 9a AtG**: utilization of residual radioactive waste. Section 9a, para. 3 stipulates the organization of state collection facilities in each Land to temporarily store radioactive waste in their territory. The role of the Federal Government is it to set up facilities for the safe keeping and the final storage of the radioactive waste. Section 9b AtG states that a plan approval procedure including an environmental impact assessment has to be executed for these federal institutions, concerning erection, operation and any major alteration,
- **Section 19 AtG**: the a.m. Government supervision is organized in such a way that the supervisory authority has to control compliance with the Atomic Energy Act and that it may order to remove hazardous states and may issue directions upon violation. In addition to that, this authority shall have access to nuclear facilities at any time and shall be able to examine these facilities.

The AtG also includes a series of regulations which refer (mainly) to the following areas:

- **Section 11 and 12 AtG**: enabling provisions for the issue of ordinances by the Federal Government,
- **Section 17**: regulations on restrictions, conditions and revocation of licenses,
- **Sections 22 to 24 AtG**: distribution of responsibilities,
- **Sections 25, 26, 31 AtG**: liability regulations,
- **Sections 18, 28, 29, 30 AtG**: compensation regulations,
- **Sections 46, 49 AtG**: criminal and administrative fines regulations.

LEGAL REGULATIONS AND TECHNICAL RULES

Further Laws supplement or specify the Atomic Energy Act:

- Wasserhaushaltsgesetz **(WHG): Water Management Act,**
- Bundesmissionsschutzgesetz **(BImSchG): Federal Emission Control Act,**
- Gesetz über die Umweltverträglichkeitsprüfung **(UVPG): Environmental Impact Assessment Act.**

Essential legal regulations are:

- Strahlenschutzverordnung **(StrlSchV): Radiation Protection Regulation,**
- Atomrechtliche Verfahrensverordnung **(AtVfV): Nuclear Licensing Procedure Regulation,**
- Atomrechtliche Deckungsvorsorge-Verordnung **(AtDeckV): Regulation on the Financial Security Pursuant to the Atomic Energy Act,**
- Atomrechtliche Kostenverordnung **(AtKosV): Atomic Energy Act Cost Regulation,**
- Atomrechtliche Sicherheitsbeauftragten- und Meldeverordnung **(AtSMV): Nuclear Safety Officer and Reporting Regulation,**
- Endlagervorausleistungsverordnung **(EndlagerVIV): Waste Disposal Advance Payments Regulation**.

The safety requirements have not been specified in great detail. Thus, different technical solutions are possible, which nonetheless have to meet the same protection objectives. The licensing and supervisory authority then decides on a case-to-case basis if the objectives are met or not.

A variety of safety regulations which have to be considered are subordinate. They serve the purpose of demonstrating adequate provisions against damages. There are a lot of guidelines, provisions and technical rules which have to be considered depending on the specific task. The main requirements which have to be taken into account are:

- BMI-Safety Criteria for NPPs,
- BMI- / BMU-Guidelines,
- RSK Guidelines,
- Resolutions of the Länder Committee for Nuclear Energy,
- Hazardous Incident Ordinance,

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- KTA rules,
- ISO / DIN-standards,
- IEC-rules and standards,
- Pressure Vessel Code,
- German Accident Prevention Regulations,
- RSK and SSK recommendations,
- Relevant international agreements, foreign rules and regulations, e.g. for the transport of radioactive substances across national borders.

To get more information on implementing guidelines, administrative regulations, recommendations, technical rules and guidelines, one should have a look at the "Handbuch Reaktorsicherheit und Strahlenschutz" (Reactor Safety and Radiation Protection Manual), which is edited by the Federal Ministry for the Environment, Nature Conservation and Nuclear Safety (Bundesministerium für Umwelt, Naturschutz und Reaktorsicherheit **(BMU)** and published by the Federal Office for Radiation Protection (Bundesamt für Strahlenschutz **(BfS)**.

4.5.3. United States Legal and Regulatory Rules

LEGISLATIVE BACKGROUND AND REGULATORY AUTHORITY

The basic legislation that provides the legal basis for nuclear regulation in the U.S. is the **Atomic Energy Act** of 1954 (AEA). Originally signed into law on August 30, 1954 and amended numerous times since that date, the AEA governs essentially all activities associated with both the military and civilian uses of nuclear energy in the U.S. The discussion in this section focuses on the peaceful uses of nuclear energy, and in particular, the production and utilization of "special nuclear material," which is defined in the AEA as plutonium, the fissile isotopes of uranium, and materials that are artificially enriched in those substances.

The AEA establishes the regulatory structure for civilian uses of nuclear energy and provides broad guidelines for the regulatory agency to follow in its activities; in particular, requirements are established for civilian uses of nuclear energy and radioactive materials to be licensed by the regulatory agency. The law also empowers the regulatory agency to develop its own rules and regulations to implement the legal guidelines, and to enforce those rules and regulations. It is important to note that the principal criterion established in the AEA for determining whether a license should be granted for any activity is that the activity can be conducted with "reasonable assurance of adequate protection of public health and safety and the environment." The specific methods for determining what comprises "reasonable assurance" and "adequate protection" are left to the regulatory agency to develop.

The major chapters of the AEA (as amended) that are of primary importance in the civilian production of nuclear energy include:

- **Chapter 5:** Production of Special Nuclear Material,
- **Chapter 6:** Special Nuclear Material,
- **Chapter 10:** Atomic Energy Licenses,
- **Chapter 14:** General Authority,
- **Chapter 16:** Judicial Review and Administrative Procedure,
- **Chapter 18:** Enforcement.

There are other U.S. laws that are relevant to the use and regulation of nuclear power, and related activities. In general, these laws provide additional guidance on either (1) the structure and operation of the regulatory authority, or (2) requirements pertaining to the regulation of specific activities. These include:

- Energy Reorganization Act of 1974,
- Reorganization Plan No. 3 of 1970,
- Reorganization Plan No. 1 of 1980,
- Nuclear Waste Policy Act of 1982 (as amended),
- Low-Level Waste Policy Amendments Act of 1985,
- Uranium Mill Tailings Radiation Control Act of 1978,
- Nuclear Non-Proliferation Act of 1978,
- Administrative Procedures Act,
- National Environmental Policy Act of 1969,
- Hazardous Materials Transportation Uniform Safety Act of 1990 (as amended).

The legislation listed above, with relevant amendments, and several other items that are of interest primarily as historical documents, are published periodically by the U.S. Nuclear Regulatory Commission (NRC) in a report entitled, "Nuclear Regulatory Legislation," NUREG-0980, which is available on the NRC's public website.

NUCLEAR REGULATION

Pursuant to the authority provided in the AEA, regulations governing the use of civilian nuclear power in the U.S. are developed by the NRC and are compiled in the **Code of Federal Regulations** (CFR), the compendium of administrative and regulatory law of Executive Branch departments and agencies of the U.S. Federal government. The CFR is divided into 50 "titles," each of which covers a broad area in which the Federal government exercises regulatory oversight. NRC regulations appear in Title 10, "Energy," Chapter 1, "Nuclear Regulatory Commission," which comprises Parts 1-199. Citations of regulations in the CFR are indicated by the title number, the part number, and (if applicable) the section and/or subsection indicator, which includes the part number and section number (e.g.., 10 CFR Part 20; 10 CFR 50.46).

The bases for the NRC's regulations can include laws passed by the U.S. Congress and signed by the President;

Executive Orders issued by the President; regulations or standards issued by other Executive Branch departments or agencies; or technical information gathered or developed by or for the NRC to address nuclear safety issues. Regulations associated with legislation or Executive Orders are, in most cases, related to regulatory procedures, rather than safety issues. For example, the AEA specifies that the duration of licenses for activities associated with the use of nuclear energy or nuclear materials can be no more than 40 years, and that such licenses can be renewed. The 40-year limitation and renewal capability are reflected, for nuclear power reactors, in 10 CFR 50.51. Congress also sets the NRC's annual budget and establishes the fraction of those funds that must be recovered through fees and other charges imposed on the NRC's licensees.

Regulations that are based on standards of other agencies reflect activities that cross administrative boundaries in the Executive Branch, such as environmental protection and transportation. They can arise as a result of legislation that gives those agencies the responsibility for establishing the standards, and requires the NRC to issue regulations implementing them. For example, the U.S. Environmental Protection Agency (EPA) is required to establish radiation protection standards for a high-level waste repository; the NRC must then reflect those standards in its regulations.

However, the vast majority of the NRC's regulations that pertain to the safety of civilian nuclear power and related activities are developed by the agency based on technical information that it compiles from internal or external sources, or develops from its own research programs. Most of the technical safety requirements that apply to the design, construction, and operation of nuclear power plants are contained in 10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities," while siting requirements are in 10 CFR Part 100, "Reactor Site Criteria." Key elements of the regulations include:

- **10 CFR Part 50, Appendix A**, "General Design Criteria for Nuclear Power Plants",
- **10 CFR Part 50, Appendix B**, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants",
- **10 CFR 50.46**, "Acceptance Criteria for Emergency Core Cooling Systems for Light-Water Nuclear Power Reactors" along with 10 CFR Part 50, Appendix K, "ECCS Evaluation Models",
- **10 CFR Part 100**, Appendix A, "Seismic and Geologic Criteria for Nuclear Power Plants".

To facilitate the implementation of regulatory requirements, the NRC also issues guidance documents. The methods and criteria reflected in these guidance documents are not requirements (except insofar as they reflect specific regulatory requirements or acceptance criteria); rather, they reflect methods that the NRC has determined to be acceptable. As such, they cannot be imposed on NRC licensees, and alternative methods can be proposed.

Guidance to NRC licensees is provided in the form of **Regulatory Guides** (RGs), which generally include substantial detail and interpretation of regulatory requirements. The NRC also publishes guidance for use by NRC technical reviewers in evaluating applications and supporting material submitted by licensees or applicants for licenses. For nuclear power plant licensing, the most important such document is the "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants," NUREG-0800, generally referred to as the SRP. While this information is intended for use by NRC reviewers, it is also used by applicants and licensees to help ensure that license-related submissions are consistent with NRC reviewer expectations.

In addition to technical requirements reflected directly in nuclear regulations, the NRC makes extensive use of **consensus codes and standards** developed by standards-development organizations (SDOs). In some cases, these standards are incorporated by reference into the NRC's regulations, thus becoming regulatory requirements; in others, the standards are endorsed by NRC RGs or the SRP, reflecting an acceptable method for meeting NRC requirements, but allowing alternative approaches. Examples of standards in both of these categories include:

- ASME Boiler and Pressure Vessel Code (incorporated by reference in 10 CFR 50.55a),
- ASME Code for Operation and Maintenance of Nuclear Power Plants (incorporated by reference in 10 CFR 50.55a),
- IEEE Standard 603-1991, "Criteria for Safety Systems for Nuclear Power Generating Stations" (incorporated by reference in 10 CFR 50.55a),
- ASME-NQA-1-1994, "Quality Assurance Program for Nuclear Facilities" (endorsed in the SRP, Section 17.5),
- ANSI/ANS-3.5-1998, "Nuclear Power Plant Simulators for Use in Operator Training and Examination" (endorsed in RG 1.149, "Nuclear Power Plant Simulation Facilities for Use in Operator Training and License Examinations").

The NRC may also impose new regulatory requirements by **issuance of Orders**, which are usually used if a significant issue is determined to require immediate or near-term action to ensure reasonable assurance of adequate protection of public health and safety and the environment. Orders may be issued to individual licensees for plant-specific issues, or to all holders of licenses for specified types of facilities (e.g., PWRs, BWRs, or all nuclear power plants).

The NRC also issues generic communications (GCs) to licensees to alert them to emergent issues related to regulations and/or safety. In order of increasing urgency of the issue, these documents include Information Notices, Regulatory Information Summaries, Generic

Letters, and Bulletins. In some cases, licensees are required to provide written responses to these communications describing how they are addressing the relevant issues. Unlike Orders, GCs cannot require licensees to take action, but licensees who do not take appropriate action voluntarily in response to GCs may subsequently be ordered to do so.

The NRC occasionally publishes official Policy Statements, which are published in the Federal Register, the daily compendium of documents issued by Executive Branch departments and agencies. Policy Statements are not regulations; rather, they express broad aims and objectives of the Commission with respect to important issues, and may serve as the basis for future rules and regulatory requirements. Perhaps the best-known of these documents is the Safety Goals Policy Statement, issued in August 1986, which provided the Commission's views on the question of "how safe is safe enough" with regard to the operation of nuclear power plants, and established quantitative health objectives to guide the evaluation of reactor safety.

4.6. Nuclear safety organization

4.6.1. Nuclear safety organization in France

In France, Nuclear Safety and Radiation Protection are controlled essentially by three different Authorities: Parliament, Government and the ASN (Nuclear Safety Authority) (see fig 4.6-1).

Parliament intervenes in the control of Nuclear Power by voting laws.
Two major laws were voted in 2006 in the field of Nuclear Safety and Radiation Protection:

- Law no. 2006-686 dated June 13, 2006, on Nuclear Transparency and Safety, stipulating in particular that the ASN report to Parliament, notably by presenting its annual report to the OPECST (Parliamentary Office of Assessment of Scientific and Technological Choices),
- Law no. 2006-739 dated June 28, 2006, relating to the Sustainable Management of Radioactive Materials and Radioactive Waste.

The Government promulgates the General Technical Rules relating to Nuclear Safety and Radiation Protection. The law dated June 13, 2006, also obliges the Government to take major decisions on Basic Nuclear Installations. The Government is advised by the **CIINB** (Interministerial Commission for Basic Nuclear Installations), the **HCTISN** (High Council for Transparency and Information concerning Nuclear Safety) and the **HCSP**

(High Council for Public Health) on matters relating to Nuclear Safety and Radiation Protection.

Finally, the law dated June 13, 2006, creates an independent authority, the ASN (Nuclear Safety Authority), which is responsible for controlling Nuclear Safety and Radiation Protection.

The ASN prepares draft texts for the Government and clarifies rules and regulations by way of technical decisions. Nuclear safety and radiation protection inspectors, working for this organization, supervise and control nuclear activities. Finally, the ASN contributes to informing the public. The ASN receives technical support in the form of expertise provided by the IRSN (Institute for Radiation Protection and Nuclear Safety) and of groups of experts (Standing Group (GP) - Standing Nuclear Section (SPN)).

Just like other independent administrative authorities in France or its counterparts abroad, the ASN is steered by a college of administrators and lays down the general policy of the ASN in Nuclear Safety and Radiation Protection.

The ASN College of Administrators comprises five members, appointed by decree:

- three are appointed by the President of the Republic,
- one is appointed by the President of the Senate,
- one is appointed by the President of Parliament.

The ASN consists of Central Services, including the Corporate Administration Department, the Legal and Organizational Department, six other departments, as well as eleven Regional Territorial Delegations (see fig 4.6-2).

The **IRSN** was created in February 2002 by article 5 of law no. 2001 – 398 dated May 9, 2001, and by implementation of the order dated February 22, 2002.

The IRSN's field of expertise covers all the risks related to ionizing radiation used within industry or medicine, or even natural radiation. Its sphere of activity comprises numerous activities including the safety of nuclear plants and the transportation of radioactive and fissile materials, the protection of the public and the environment against ionizing radiation, the protection and control of nuclear materials and products that could potentially be used to manufacture weapons, the organization and training of the crisis management team, and the protection of nuclear plants and transportation operations against any malicious acts. The IRSN provides the public with information concerning nuclear related items.

The Standing Groups (GP) of industry experts are consulted by the ASN on safety or technical issues. They are in charge of the problems relating to nuclear reactors' long-term storage facilities for radioactive waste, and other Basic Nuclear Installations (INB). The GP for nuclear reactors studies the technical problems regarding safety that may be caused by the creation, operation and decommissioning of a nuclear plant and its auxiliary buildings. The GP examines the preliminary, provisional and final safety analysis reports of each INB. The GP has at its disposal reports containing the results of the analyses carried out by the IRSN. They give their opinion together with their recommendations.

Created by law no. 83-609 dated July 8, 1983, the **OPECST** is a parliamentary delegation whose responsibility is to inform Parliament of the potential consequences of scientific or technological choices, to help clarify the decision-making process. The office thus fulfils its legislative function while working to develop and apply the law.
The **CSSIN** created by decree 73-278 dated March 13, 1973, and modified by decree 87-137 dated March 2, 1987, is composed of persons chosen for their scientific, technical, or economic competence, or for their position in labor unions, trade associations, or other associations. It provides the Ministry of Industry or the Ministry of the Environment with any recommendations that it considers useful to enhance the overall effectiveness of the actions taken in France in the field of nuclear safety.

The **CIINB** created by the decree dated December 11, 1963, and modified by decree 73-405 dated March, 27, 1973 concerning Basic Nuclear Installations, is consulted by the Minister of Industry and the Minister of the Environment on the following subjects: requests for licensing of new basic nuclear facilities, or modification of existing ones and drafting of the regulations concerning these facilities.

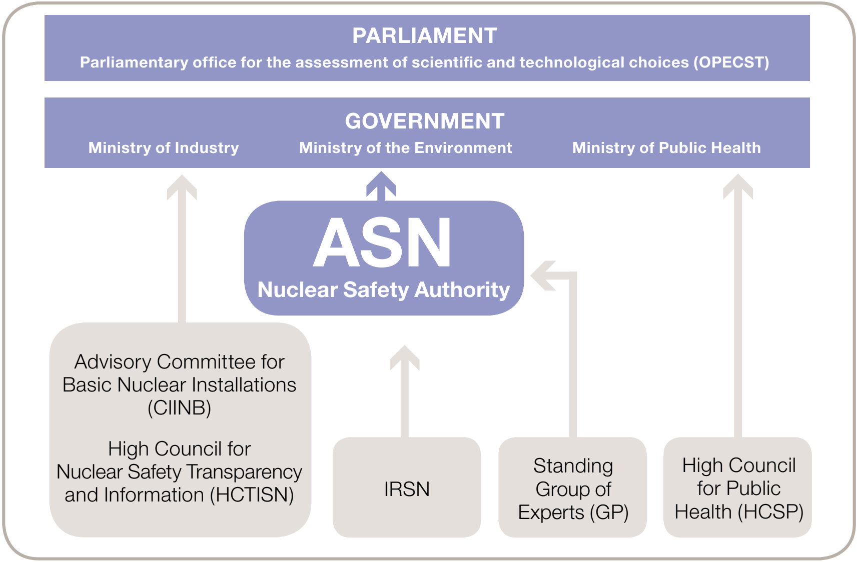

Fig 4.6-1 French nuclear safety organization

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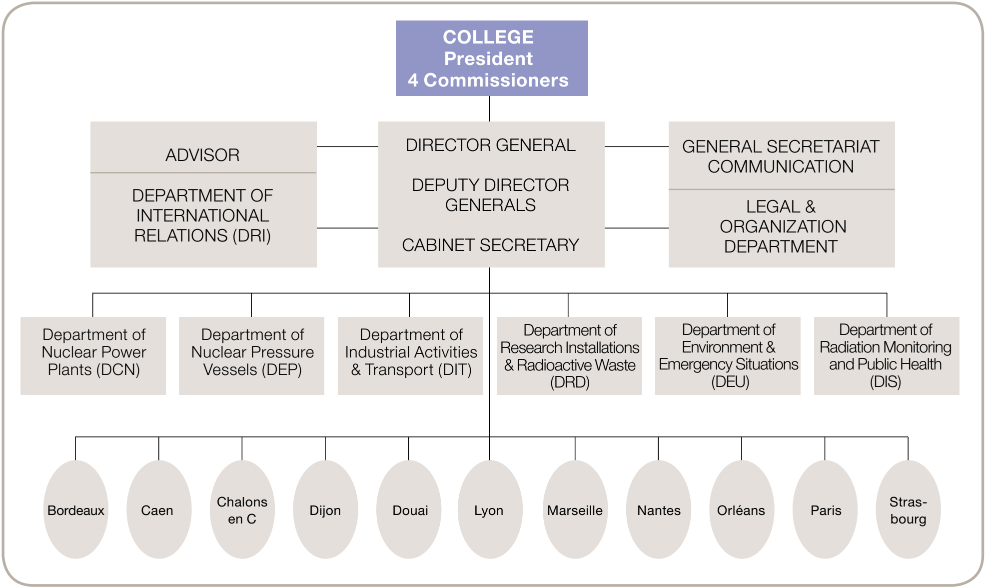
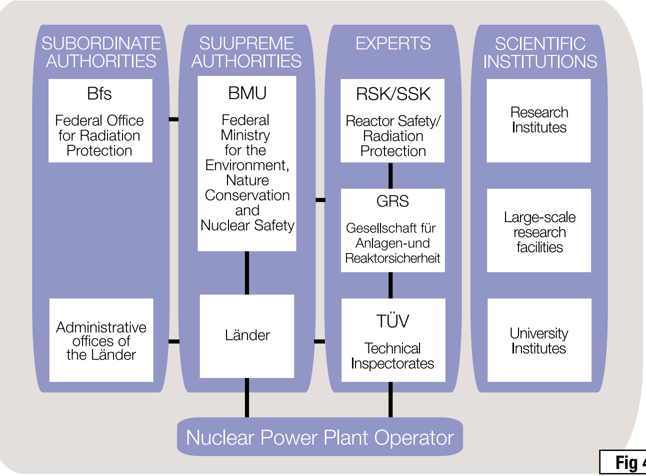


Fig 4.6-2 ASN organization

4.6.2. Nuclear safety organization in Germany

This section gives a brief introduction into the organisation of the German authorities and expert organisations which are responsible for nuclear licensing and supervision.
As stated in section 4.5.2, the Länder are responsible for the implementation of the Atomic Energy Act on behalf of the Federal Government. The Länder have ministries that are responsible for licensing of construction, operation, essential modification and decommissioning of nuclear power plants. To ensure a uniform implementation of the AtG, the Länder are subject to federal supervision. The federal government has the right to issue directives concerning the legality and expediency of the implementation.



The competent federal authority for federal supervision is the **Federal Ministry for Environment, Nature Conservation and Nuclear Safety** (BMU). It is assisted by the **Federal Office for Radiation Protection** (BfS), the **Reactor Safety Commission** (RSK) and the **Radiation Protection Commission** (SSK) as well as by **Gesellschaft für Anlagen und Reaktorsicherheit** (GRS), the central nuclear safety expert organization for the federal government.

Licences for the construction and operation of nuclear facilities pursuant to Section 7 AtG are granted by the competent Länder authorities, see Figure 4.6-3.

Fig 4.6-3 Nuclear safety and radiation institutions

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These are also responsible for the supervision of the nuclear facilities in operation.

On the basis of the submitted documents from the applicants, the licensing authority of the Land examines whether the prerequisites for granting a licence have been fulfilled. In doing so it is advised scientifically by experts, generally by one or more Technical Inspectorates **(Technischer Überwachungs-Verein, TÜV)**. In parallel the Federal Ministry for Environment as well as all central, regional and local authorities affected are involved in the process, and the general public is informed.
Within its framework of authorities, the Land is only granted the competence to implement duties to execute instructions. This legal capacity includes activities relating towards third parties, as it covers, for example, the conclusion of public law contracts.

The competence of excercising factual assessment and substantive decisions is allocated to the Länder as well. Due to the fact that this competence is apposed to the competence to execute duties, it may optionally be assumed by the Federal Government. If this is the case, the Federal Government is able to inspect and to comment on the applicant's documents submitted to the responsible Land authority. If Land and Federal Government asses these documents differently, the Federal Government may order instructions, which are binding for the Länder authorities.

Federal supervision is exercised by the Federal Ministry for Environment, Nature Conservation and Nuclear Safety (BMU). The BMU involves other Federal Ministries that are concerned. In performing its role as federal supervisor, the BMU is advised in particular by its expert committees, the RSK and the SSK. These expert committees are composed of independent experts of different scientific disciplines.

After thorough examination of the planned project, the competent authority (the Land) drafts a license which is submitted to BMU for approval. RSK and SSK give a recommendation on the draft to the BMU. The BMU analyses this recommendation and then provides its comment to the respective competent licensing authority. This comment has to be considered in the decision-making process of the Land authority in charge. In addition GRS provides expert opinions to the BMU.
The Federal Office for Radiation Protection (BfS) provides the BMU with scientific and administrative experts, ensuring federal supervision by commenting and suggesting improvements in nuclear facility safety. In addition it is in charge of the safe custody of nuclear fuels by the state, for the construction and operation of disposal facilities as well as for licensing transport and storage of nuclear fuels.

The exchange of experience between the Federal Government and the Länder and co-ordination to ensure an equal procedure of all Länder in the field of nuclear safety and radiation is organized by the Länder Committee for Nuclear Energy and its technical committees (chaired by the BMU).

Reference: Report "Nuclear Licensing and Supervision in Germany" from GRS (Gesellschaft für Anlagen- und Reaktorsicherheit (GRS) mbH), 4th edition, published in 2002.

4.6.3. Nuclear safety organization in the United States

The Atomic Energy Act of 1954 (see section 4.5.3) established the Atomic Energy Commission (AEC) as the civilian agency in charge of both the development and regulation of civilian nuclear power and related activities in the U.S. For the next 20 years, the AEC pursued the development of nuclear power technology, primarily by means of research at U.S. national laboratories, and the oversight of that technology in a separate regulatory organization. However, the combination of these two functions in a single agency was increasingly seen as creating a conflict of interest that could impair the ability of the regulatory arm of the AEC to act as an independent safety authority. This resulted in enactment of the Energy Reorganization Act of 1974 (ERA), which established the **Nuclear Regulatory Commission** (NRC) as an independent agency, and the Energy Research and Development Administration (ERDA) as the developmental organization. ERDA was subsequently folded into the Department of Energy (DOE). The term "Nuclear Regulatory Commission," as used in the ERA, actually refers to the Commissioners who serve as the governing and policy-making body of the NRC. However, as used today, "NRC" and "Commission" generally refer to the agency as a whole, while "Commissioners" refers to the governing group.

As an independent Federal agency, the NRC is treated, in most respects, as part of the Executive Branch, but it has responsibilities and prerogatives that go beyond those of Executive Branch departments and agencies. The NRC combines the three functions of government—executive (enforcement), legislative (rulemaking), and adjudicatory (hearings)—in a single agency. The NRC can be sued in Federal court, and the agency is subject to Congressional oversight, but Commission decisions cannot, in general, be challenged other than by due judicial process.

The ERA specifies that there be five Commissioners; each Commissioner is appointed by the President and must be confirmed by a majority vote of the U.S. Senate. No more than three of the five Commissioners can be members of one political party. The nominal term of each Commission is five years, beginning on July 1 of the year of appointment, with one Commissioner's term expiring on June 30 of each year. However, it is important to recognize that a Commissioner's term is determined by the "seat" to which he or she is appointed. That is, if a Commissioner's term expires and he or she is not reappointed, that seat remains vacant until a new Commissioner is appointed to fill the position. The new Commissioner's term expires on June 30 five years after the previous Commissioner departed, not five years after the new Commissioner joins the NRC.

The Chairman is designated by the President from among the serving Commissioners. Senate confirmation of the appointment is not required, and a new Chairman can be designated at any time, at the pleasure of the President. If a Chairman is replaced by Presidential action prior to the expiration of his or her term, the individual remains as a Commissioner. In general, Commissioners cannot be removed from office by the President except for malfeasance in office. The bipartisan nature of the Commission and the designation of the NRC as an independent agency (see below) tend to insulate the agency, to some extent, from direct political pressure.

The ERA also discusses the structure of the agency under the Commission. Three offices are required by statute: the Office of Nuclear Reactor Regulation, which is responsible for nuclear reactor safety oversight; the Office of Nuclear Material Safety and Safeguards, which is responsible for oversight of non-reactor (e.g. medical and industrial) uses of radioactive material (except for medical x-rays); the Office of Nuclear Regulatory Research, which is responsible for developing the technical bases for NRC regulations.

The Three Mile Island accident in 1979 exposed some of the weaknesses in the structure of the five-member Commission, particular with regard to the exercise of executive responsibility. Reorganization Plan No. 1 of 1980 was thus issued as an Executive Order, and later enacted into law by Congress. The Plan designated the NRC Chairman as the principal executive officer of the NRC and further defined the responsibilities of the Chairman and the other Commissioners in the conduct of NRC business and oversight of the activities of the NRC staff. The structure described in Reorganization Plan No. 1 of 1980 continues to be the basis of present-day NRC operation.

While the Chairman is the NRC's chief executive, each Commissioner has equal power with respect to the establishment of NRC policies and rules, and in the adjudicatory functions of the Commission, e.g. in licensing decisions. Actions are approved by majority vote of the sitting Commissioners.

The regulation of civilian nuclear power facilities and research reactors (except those reactors operated by and for the Department of Energy) is the sole responsibility of the NRC. However, regulation and oversight of some non-reactor uses of radioactive materials, including medical and industrial applications and low-level waste disposal, can be undertaken by State governments under the NRC's "Agreement State" program. States participating in this program have signed agreements with the NRC to impose regulatory requirements that, as a minimum, are consistent with the NRC's rules and regulations.

In carrying out its oversight responsibilities, the NRC (Commissioners and staff) are assisted by several advisory groups. The **Advisory Committee on Reactor Safeguards** (ACRS) is mandated by the AEA, and consists of senior experts drawn from industry (retirees), universities, and national laboratories, in various technical areas associated with nuclear reactor technology and safety. Members are appointed by the Commissioners and serve for four-year terms. There is a nominal limit of three terms, but the limit can be waived by the Commission. A small full-time staff, headed by an Executive Director, supports the ACRS. The **Advisory Committee on Nuclear Waste and Materials** (ACNW&M) was created by the Commission; it is not mandated by statute. It was originally created as an offshoot of the ACRS, from members with expertise in waste and materials issues. It is similar to, but smaller than, the ACRS, and employs the same staff. The **Advisory Committee on the Medical Uses of Isotopes** (ACMUI) is composed of experts from the health and medical fields, and provides advice on the use of radioactive materials in medical applications. Members are appointed by the Commissioners for a term of three years, and can serve up to two consecutive terms. Advisory committee members are considered to be "special government employees," a designation that imposes restrictions on the number of days each year that they can work on committee business.

Advisory committees meet on a regular schedule, and may appoint subcommittees to handle specific tasks and issues. Reports are issued by letter to the Commission and the NRC's Executive Director for Operations (EDO). In general, while the advisory committees are required to provide advice to the Commission, the Commissioners are not bound to implement advisory committee recommendations.

The NRC's adjudicatory responsibilities are carried out in part by the **Atomic Safety and Licensing Board Panel** (ASLBP). Members of the ASLBP may be either part-time or full-time employees, and are designated as legal or technical experts. The ASLBP is responsible for conducting hearings, as directed by the Commission, covering a wide range of technical and administrative issues. Depending on the topic and the complexity of the issue, hearings may be conducted by a single judge, or by a three-person panel composed of two technical judges and one legal judge drawn from the (current) ASLBP membership of about 26 judges. The ASLBP is assisted by a small technical and legal staff.

ASLBP decisions can be appealed to the Commissioners, and the Commission may also review ASLBP decisions that are not appealed. At its discretion, the Commission may choose to act as the hearing panel in lieu of the ASLBP.

The technical work associated with the conduct of the NRC's regulatory responsibilities is carried out by the NRC staff. While major licensing issues and decisions are reviewed by the Commission, routine issues and decisions are delegated by the Commission to the appropriate NRC staff manager. When new issues related to NRC policy are identified in the course of the staff's technical and administrative responsibilities, the NRC staff will generally make recommendations to the Commission on the resolution of the issues, which then votes on whether to approve in part, approve in full, or disapprove the staff's recommendations.

The upsurge in interest in new nuclear power plants in the U.S., along with pending efforts by the Department of Energy to license a high-level waste repository, have resulted in recent reorganizations of the NRC staff to more efficiently deal with the increased workload. The major NRC staff offices and their primary responsibilities now include:

- Office of Nuclear Reactor Regulation – oversight of currently operating reactors,
- Office of New Reactors – review of applications related to new reactor design, licensing, and siting,
- Office of Nuclear Material Safety and Safeguards – oversight of activities related to the nuclear fuel cycle,
- Office of Federal and State Materials and Environmental Management Programs – oversight of industrial and medical uses of radioactive materials, and interface with Agreement States,
- Office of Nuclear Regulatory Research – development of technical bases for NRC regulations,
- Office of Nuclear Security and Incident Response – oversight of security programs and operation of the NRC's Incident Response Center.

The NRC maintains four Regional Offices around the U.S., each headed by a Regional Administrator. The staff in these offices is composed primarily of inspectors who serve as first points of contact with NRC licensees in their designated region. The NRC also assigns Resident Inspectors (RIs) to each operating nuclear power plant and other selected facilities. Depending on the number of reactors at a power plant, two or more RIs are assigned to the site, with one designated as the Senior Resident Inspector. RIs have offices at the plant site and provide a direct NRC interface with plant management and operations personnel.

4.7. Licensing procedures

4.7.1. Generic licensing process

National regulations are enacted to ensure the safety of nuclear power plants under construction and in operation, in order to protect site employees, the public and the environment from potential adverse effects.

These regulations result in precisely organized licensing processes imposed on plant operators, whose responsibility is to apply for and obtain from the safety authority, in a step by step manner, all authorizations to construct, test and operate new plants. Throughout this licensing process, the plant operator is technically supported by its plant supplier.

In many countries where nuclear power generation is in use, the main steps of the licensing process of a new power unit are quite similar.

Typically, licensing is organized in sequential phases. Before construction of a new plant, a construction permit has to be obtained from the safety authority. This authorization is conditioned by satisfactory results of the examination of a Preliminary Safety Analysis Report (PSAR) and related documentation (such as environmental impact and risk analysis reports) prepared under the responsibility of the operator. The PSAR is analyzed, at the request of the safety authority, by safety technical experts. In some countries, at this stage, regulations also provide for a local public inquiry. In its decision-making process to issue a construction authorization, the safety authority is generally advised by a standing committee of senior safety experts (Permanent Group in France, Advisory Committee for Reactor Safety –ACRS– in the USA).

During the construction phase, compliance with regulations, quality standards and the PSAR commitments is closely supervised by the authorities (a particularly illustrative example is the specific supervision enforced on the manufacture and testing of the primary coolant system and its components).

Before plant operation, a commissioning program has to be set up under the responsibility of the operator. It is organized in a step-by-step approach, from the individual testing of equipment and systems to the zero and the power operation of the plant. It is added to the PSAR to constitute the Final Safety Analysis Report (FSAR) which has to be approved by the safety authority before the initial fuel loading and any radioactive effluent release. The commissioning program is carried out, under safety authority supervision, to verify the compliance of the plant with the statements of the FSAR. During the program implementation, the step-by-step power increase of the plant is submitted for successive approvals by the safety authority.

After completion of the commissioning phase, the FSAR is revised (RFSAR) to reflect the actual status under which the plant will be operated and the safety authority is in a position to issue the plant commercial operation authorization.

Near the generic licensing process, a few specificities appear for the countries depending on how the national institutions are organized. Examples of these specificities for France the United States and Germany are summarized below.

4.7.2. French specificities

The main French specificities are the following:

The Declaration of Public Utility Procedure **(DUP)** which includes a public inquiry and an environmental impact study enabling acquisition of land to commence. The Ministry of Industry and the Environment engages this procedure, on the basis of an information package prepared by the plant operator.

Completion of the procedure is certified by a decree signed by the Prime Minister and the Minister of Industry and the Environment.

The three principal steps for authorizing the construction and operation of a nuclear power plant unit are:

- First, the plant operator sends a **construction license application** to the Ministry of Industry and Environment, including the Preliminary Safety Analysis Report (PSAR). In particular, six months before the beginning of the startup tests, the plant operator submits a revised PSAR accompanied by an application for prior approval for fuelling and for operation at low power,
- Secondly, six months before the first core loading, the plant operator submits the Final Safety Analysis Report (called "Rapport Provisoire de Sûreté") and requests a **fuel loading and low-power operating license,**
- Finally, when construction is complete and the startup tests finished, the plant operator sends the safety authority the Revised Final Safety Analysis Report (called "Rapport Définitif de Sûreté") and requests the **commercial operating license** (ministerial approval).

The French licensing process is presented in figure 4.7-1.

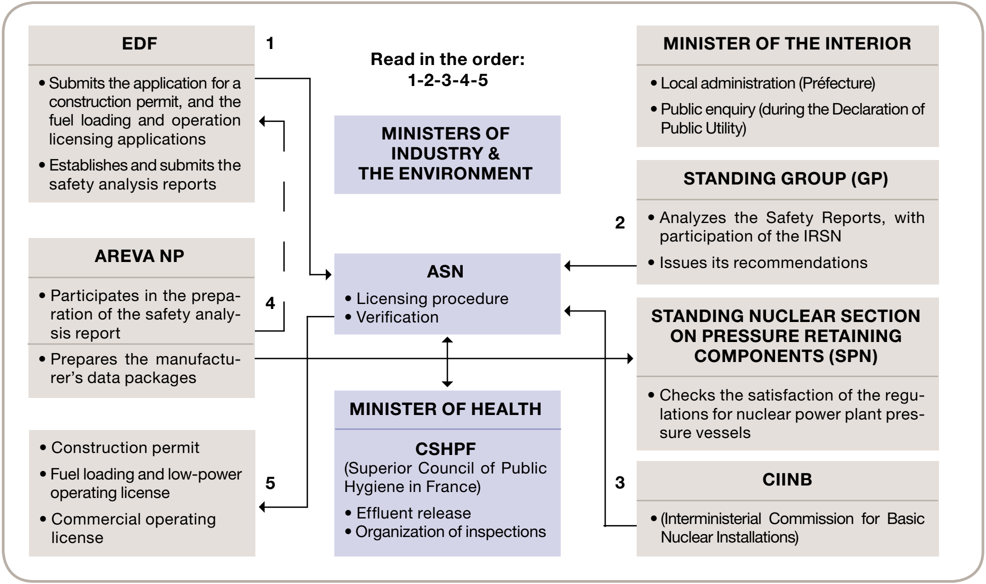


Fig 4.7-1 Licensing procedure in France

4.7.3. United States specificities

The entire licensing process, notably including public involvement, is overseen by the U.S. Nuclear Regulatory Commission (NRC).

Before 1989, nuclear power plants were licensed under a two-step licensing process described in **10 CFR Part 50**, requiring both a **construction permit** (preliminary safety analyses, environmental review, financial and antitrust statements) and an **Operating license**, which requires the review of a Final Safety Analysis Report. As described previously, the technical requirements governing the review of nuclear power plant license applications are located primarily in 10 CFR Part 50 and 10 CFR Part 100. The NRC staff conducts these reviews based on the guidance provided in the NRC's Standard Review Plan, NUREG-0800.

In 1989, the NRC established **10 CFR Part 52**, which provides for the issuance of a single combined construction permit and operating license (COL), usually referred to simply as a combined license. As the name suggests, a COL replaces the previous "two-step" licensing process with a single licensing proceeding; upon issuance of a COL, a licensee can construct a nuclear power plant and proceed directly to plant operation without the need to obtain a separate operating license. Part 52 also describes other processes that support the COL process. An application can be submitted for an **early site permit**, which approves a site for future siting of a nuclear power plant. A standardized nuclear power plant design can also be reviewed under an application for **a standard plant design certification**. An early site permit is considered a license under NRC regulations, and involves the submission of a site environmental report and a safety analysis, which are subject to NRC staff review and a subsequent hearing. A design certification is not a license; rather, a generic safety analysis, at the approximate level of a Final Safety Analysis Report (along with supporting documentation) is reviewed by the NRC staff. Upon staff approval of the design, through issuance of a Final Safety Evaluation Report (FSER), the design is certified by means of a rulemaking, with the key design parameters incorporated into a design certification rule that is published as an Appendix to 10 CFR Part 52. It is important to note that the governing technical requirements in 10 CFR Part 50 and 10 CFR Part 100 apply to reviews conducted under 10 CFR Part 52. Part 52 is concerned principally with the various licensing (and supporting) processes discussed above, not with technical requirements. Under either the Part 50 or Part 52 process, before an applicant can build and operate a nuclear power plant, it must obtain approval from the NRC.

An application for a COL may incorporate by reference a standard design certification, an early site permit, both, or neither. This approach allows early resolution of safety and environmental issues. The issues resolved by the design certification rulemaking and during the early site permit hearing are not considered during the combined license review. These processes are presented in figure 4.7-2.

An important element of the COL process, as indicated in Fig. 4.7-2, is the "inspections, tests, analysis, and acceptance criteria", or ITAAC, process. ITAAC is the bridge between the plant design, as proposed in the COL application, and the plant as constructed. As part of the COL, important plant design and operational parameters are identified for inclusion as ITAAC. Each ITAAC entry describes a particular design attribute, the inspections, tests and/or analyses that will be used to demonstrate that the constructed plant fulfills that design attribute, and the acceptance criteria that must be satisfied as part of that demonstration.

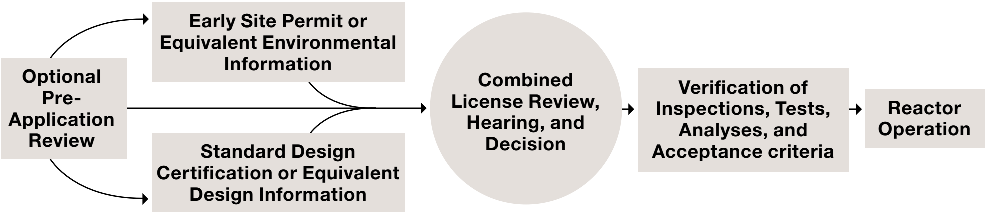
By the end of plant construction, the licensee must certify to the NRC that all ITAAC requirements have been satisfied, and the NRC must verify that is the case, before the licensee is given permission to load fuel into the plant and commence operation. If a member of the public can provide substantive evidence of failure to meet one or more ITAAC requjrements, a request for a hearing can be submitted to the NRC prior to plant operation. The Commission may, at its discretion, grant or deny that request. It is also within the Commission's discretion to grant a request for a hearing, but to allow the licensee to load fuel and commence operation in the meantime.

It should be noted that all currently-operating nuclear power plants in the U.S. were licensed under the "two-step" process in 10 CFR Part 50. As of early 2008, the NRC has issued three early site permits and four standard design certifications. The first-ever COL applications were submitted near the end of 2007, and NRC review of these applications has begun.

The NRC's regulations also provide separate processes for the licensing of non-reactor nuclear facilities. For example, 10 CFR Part 70 provides the licensing requirements and process for nuclear fuel fabrication plants. Non-reactor licensing is not discussed further in this section.

All documents and correspondence related to a application for a license, early site permit, or design certification are placed in the NRC's Public Document Room (PDR) in Rockville, Maryland, which can be accessed through the Electronic Reading Room on the NRC's public website. Members of the public may access the Electronic Reading libraries from computers with internet access. NRC documents may also be obtained from public libraries throughout the country, in locations near NRC-licensed facilities.

Fig 4.7-2 Relationships between combined licenses, early site permits, and standard design certification



4.7.4. Germany specificities

For construction, operation or any other holding of a stationary installation for the production, treatment, processing or fission of nuclear fuel or modifying such installations or its operation a license is required.

Since the private power utility companies are the operators of nuclear power plants, they also act as applicants of the nuclear licensing procedures. Written applications are submitted to the competent licensing authority of the Land, in which they intend to construct the plant. This application must be accompanied by several documents. One of the application documents is the safety report, which includes the site plans and the survey diagrams. It shows the plant and its operation and the associated effects. In addition to this, it explains the preventive measures.

The licensing authority examines whether the license prerequisites have been met. Because of the great scope of examination, usually experts (and expert organizations) are engaged to support the licensing authority. According to the federal executive administration, the licensing authority also involves the BMU, which consults the Reactor Safety Commission for Radiological Protection and sometimes also the GRS for support. The general public is also involved by the licensing authority, concerning the public announcement of the project and the holding of a public hearing.

Due to Germany's federal states, the licensing of nuclear installations is the responsibility of the individual Länder. They act in case of the use of nuclear energy under the supervision of the Federation (BMU) to execute in a uniform manner. The licensing authority of each Land examines, supported by scientific experts (usually by the German TÜV), if the required prerequisites to receive a license have been fulfilled. Simultaneously the Federal Environment Ministry and the federal, regional and local authorities responsible are involved and the general public is informed.

The involved authorities can be seen in the following diagram.

One of the important tasks of the Licensing Authority is to involve the general public to protect citizens who might be affected by the planned modification. The following steps are set out in Section 5 of the Nuclear Licensing Procedures Regulations (AtVfV) and are obligatory:

- Public announcement of the project in the Federal Bulletin and public press,
- Carry out hearings where objections can be discussed between the licensing authority, applicant and objectors,
- Minutes describing the subject of the licensing procedure, the course and the results of the hearing,
- Public display of application documents at the licensing authority and at one or more locations near the site for a period of two months,
- Serving of the licensing authority's decision concerning the objections.

Another important constituent of the nuclear licensing procedure is the environmental impact assessment (UVP: Umweltverträglichkeitsprüfung), which has to be carried out as stated in Section 1a AtVfV. The environmental impact assessment is performed on the basis of the requirements of the nuclear and radiation protection regulations and is finally the basis for the decision about acceptance of the project. The exact UVP procedure can be found in the Environmental Impact Assessment Act (UVPG), which is an addition to the Nuclear Licensing Procedures Regulation.

Comments

1 **Examination of application by the RSK, SSK.**

2 **RSK, SSK give a recommendation to the BMU.**

3 **BMU analyses recommendation & provides comments to responsible Licensing Authority.**

4 **The GRS (Gesellschaft für Reaktorsicherheit) provides expert opinions to the BMU.**

5 **Experts support the licensing authority. They do not have the authority to make decisions.**

6 **To involve the general public is an important task of the licensing authority.**

7 **BMU comments have to be considered in the decision-making process.**

8 **Licensing authority examines and rejects the application or grants the license.**

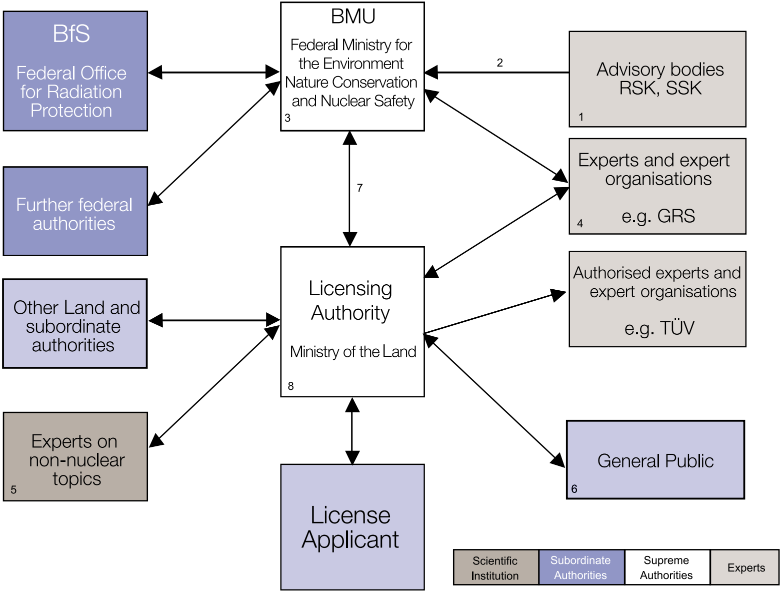


Fig 4.7-3 Nuclear Licensing Procedure in Germany

4

The following figure illustrates in brief how this assessment has to be executed.

As stated above, the licensing authority examines whether the prerequisites pursuant to Section 7 para. 2 AtG are met in the application. The following points are examined:

- Opinions and evaluation reports of the authorized experts,
- BMU statement,
- Statements of other authorities involved,
- Objections by the general public,
- Results of the UVP.

The decision process of the licensing authority is based an all the points above as well as on the application documents. All procedural requirements must be fulfilled for a legal decision. The authority is finally responsible for (based on the AtVfV):

- Rejecting the application if the licensing prerequisites have not been met (Section 15, paragraph 2),
- Granting the license (Section 16, 18).

The decision made has to include (based on Section 16 AtVfV):

- The name and place of residence of the applicant,
- Information that the license will be granted with an indication of the legal basis,
- Definition of the subject of the license, including facility location,
- Supplementary provisions,
- Reasons for decision and treatment of the objections.

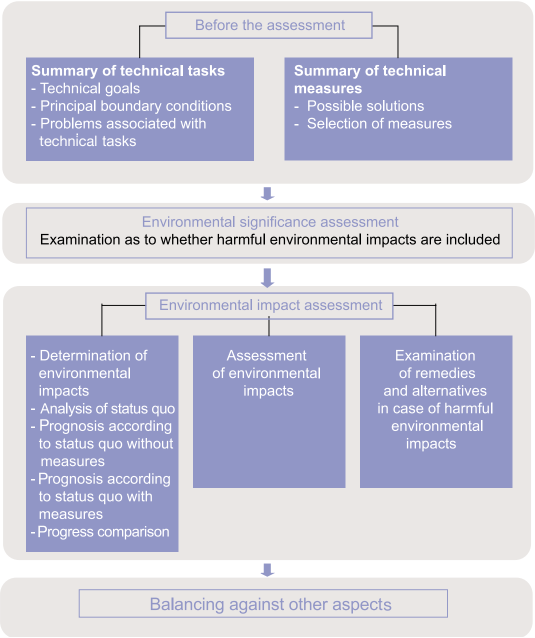


Fig 4.7-4 Environmental Impact Assessment

C. CRISIS ORGANIZATION

The French organization in the event of an incident or accident implying radiological risks is based on a structure including:

- **Decision makers:** "Postes de Commandement Direction" or PCDs (Management Command Posts),
- **Evaluation teams:** Emergency Response Teams.

This organization is set up in parallel by the French government authorities and by the utility (EDF), with the **technical support of AREVA NP**.

When such an event occurs, EDF implements its on-site Emergency Plan (PUI, or Plan d'Urgence Interne), including:

- On site, setting up of Command Posts (CPs):
  - Management Command Post, solely responsible for the decisions to be taken to ensure the safety of facilities, the protection of personnel and the safeguarding of equipment, also handles official communications with the central authorities,
  - Local Command Post, in charge of ensuring the control and safeguard of the affected nuclear unit,
  - Monitoring Command Post, in charge of centralizing the radiological measurements and synthesizing them,
  - Logistics Command Post, responsible for internal logistics,
  - Local Emergency Response Team, responsible for analyzing and evaluating the situation, and for keeping the central authorities' emergency response teams informed of the situation.
- At national level:
  - A Utility Command Post, led by the manager of EDF's Nuclear Production Division, handles the relations with the public authorities at national level (Poste de Commandement Direction or PCD),
  - A National Emergency Response Team responsible for providing additional information and recommendations to the PCD.

Under such emergency situations, AREVA NP provides the utility, upon request, with fast and specific technical assistance adapted to the severity of the situation, by triggering its own **Emergency Response Plan**, based on its own **Emergency Response Organization.**

The AREVA NP Emergency Response Plan includes three phases:

- Activation of the plan,
- Setting up of the AREVA NP **Emergency Response Team (ERT)** in the AREVA NP **Emergency Response Center** (located in Paris La Défense),

- Mobilization of the **Technical Emergency Assistance Team (TEAT)** and, if required, a **Local Emergency Assistance Team (LEAT).**

The AREVA NP Emergency Response Organization includes:

- A dedicated and permanent management structure,
- The **ERT** composed of 5 teams of 13 experts on call in turn,
- The **TEAT** composed of about 200 experts whose private contact details are listed, continuously updated and kept in a safe in the Emergency Assistance Center,
- The **LEAT**, made available to the utility on site.

In parallel, the government authorities set up their own emergency response organization:

- Locally and at county (department) level, the Prefect initiates a specific action plan coordinated through a Command Post that includes various department authorities, the representative of EDF and the representatives of the different Ministries concerned. An **Operational Command Post** is set up as near as possible to the site.
- At the level of the central authorities:

  Within the Ministry of the Interior, the Directorate of Civil Emergency Preparedness coordinates the activities of the different agencies responsible for prevention and assistance measures.

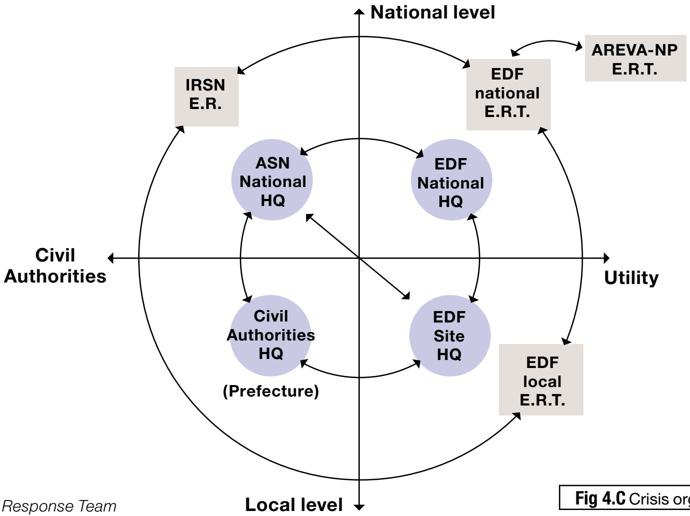
  The health physics branch of **IRSN** (Institut de Radioprotection et de Sûreté Nucléaire), supported by the National Committee of Medical Experts, provides technical advice to the public authorities concerning radiological protection measures. The Ministry of Industry provides national coordination of public and media information activities.

The General Secretary of the Inter-ministerial Committee for Nuclear Safety keeps the President and the Prime Minister continuously informed of the situation.

- The **ASN** (Autorité de Sûreté Nucléaire – Safety Authoritiy) sets up an organization including:
  - A Management Command Post located in their offices in Paris,
  - An Emergency Response Team located in Fontenay-aux-Roses, near Paris, under the authority of the **IRSN**,
  - A local mission split between the site (local emergency response team) and the Prefecture concerned, which is responsible for keeping the ASN emergency response team informed.

A dozen drills are conducted annually with EDF, about half of them involving the various government authorities.

Moreover, AREVA NP provides the same kind of remote technical assistance to foreign customers. Within an hour of a call, our Emergency Response Team would be operational to address any difficulties encountered by the Guang Dong or Ling Ao plants in China, or by the Koeberg plant in **South Africa**. An annual drill is conducted with each utility.



HQ: Headquarters
E.R.T.: Emergency Response Team

Fig 4.C Crisis organization in France

5

ENERGY

UNITS OF MEASUREMENT

5.1. Statistical units and equivalences

5.1.1. Introduction

To compare several different forms of energy, it is convenient to convert the corresponding quantities into a single unit of measurement. Generally, the unit employed is the **Tonne of Oil Equivalent (TOE)** which is the thermal energy that can be released by the complete combustion of one tonne of oil. All the quantities of energy (produced or potential resources) are then expressed in TOE. Another equivalent unit that was historically used is the **Tonne of Coal Equivalent (TCE)** which has a similar definition but concerns the combustion of coal. Today the TCE is no longer used.

Depending on the country, certain differences exist in the equivalence coefficients used.

5.1.2. Units and physical equivalence coefficients

These values are derived from physical units, thus used worldwide:

1 kWh	**= 3.6 x 10^6 J**	**= 3412 btu**
1 thermie	**= 1 x 10^6 cal**	**= 4.1858 MJ**
1 GWd	**= 24 x 10^3 kWh**	
1 btu	**= 1055.056 J**	

Other units are used to assess energy reserves:

1 quad	**= 1 x 10^{15} btu**	**= 1.055 EJ (exajoule)**

The comparison between the different sources of energy is presented in TOE. The value of the TOE is based on the definition of standardized oil which represents the average oil found on the earth.

The international community uses the following value for the TOE:

1 TOE	**= 10,000 thermies**	**= 41.86 GJ**

This means that the complete combustion of one tonne of oil will produce 41.86 GJ of thermal energy (without consideration of the final use of this energy).

The heating power of coal was evaluated at about 7 th/kg **(1 TCE = 0.7 TOE).** This value is no longer used.

In addition, see fig. 5.7-1 Energy units conversion, “page 235”.

5.1.3. Electrical energy production

AMERICAN STATISTICS

Official energy statistics from the U.S. Government are provided by the DOE (Department Of Energy) which is part of the EIA (Energy Information Administration).

This administration defines the following values:

Content (in btu) of common energy units.

Energy units	Content in btu	Equivalence in TOE
1 barrel of crude oil	5.8×10^6 btu	1.044 TOE/metric ton
1 Gallon of gasoline	1.24×10^5 btu	0.937 TOE/metric ton
1 Gallon of heating oil or diesel fuel	1.39×10^5 btu	1.051 TOE/metric ton
1 cubic foot of natural gas	1.026 btu	0.914×10^{-6} TOE/m^3
1 short ton of coal	20.68×10^6 btu	0.58 TOE/metric ton

Note: these values represent only a thermal energy equivalent.

Example: the complete combustion of one barrel of crude oil will produce 5.8×10^6 btu.

The equivalence coefficients for final electricity consumption depend on the efficiency of the production plants.

Plant	Quantity	btu	TOE	Efficiency
Geothermal plants	1 MWh	21.02×10^6	0.53	16.2%
Nuclear plants	1 MWh	10.42×10^6	0.260	33.0%
Fossil fueled plants	1 MWh	10.24×10^6	0.258	33.3%
Other plants (hydro, solar, wind)	1 MWh	3.412×10^6	0.086	ignored

Important comments: for nuclear electricity, the value of 0.260 TOE is the quantity of oil required to produce 1 MWh in a plant whose net efficiency is 33%. In the "Other plants" the value are based on the "method of the energetic contents"; this implies that the efficiency of these plants is ignored.

EUROPEAN AND FRENCH STATISTICS

Since the beginning of 2002, the French "Observatoire de l'Energie" has adopted the method proposed by the international organizations concerned: International Energy Agency, EUROSTAT, etc.

The proposed coefficients are given in the tables below.

ENERGY	Physical Units	GJ (LHP) [(1)]	TOE (LHP) [(1)]
Coal	1 tonne	26	0.619
Coke	1 tonne	28	0.667

ENERGY	Physical Units	GJ (LHP) [(1)]	TOE (LHP) [(1)]
Lignite agglomerates and briquettes	1 tonne	32	0.762
Lignite and residual products	1 tonne	17	0.405
Crude oil, heating fuel and diesel fuel	1 tonne	42	1
LPG	1 tonne	46	1.095
Car and jet fuel	1 tonne	44	1.048
Heavy fuel oil	1 tonne	40	0.952
Oil coke	1 tonne	32	0.762

The equivalence coefficients for final electricity consumption depend on the efficiency of the production plants.

Plant	Quantity	GJ	TOE	Efficiency
Geothermal plants	1 MWh	3.6	0.86	10.0%
Nuclear plants	1 MWh	3.6	0.260	33.0%
Other plants (fossil fueled, hydro, solar, wind)	1 MWh	3.6	0.086	ignored

The same comment applies here. One major difference between the American and European positions concerns fossil fueled plants, for which the IEA and French OE retain a value based only on energy content, ignoring the efficiency of the plant.

(1) LHP: Lower Heating Power.

VOLUME-WEIGHT EQUIVALENCES OF PETROLEUM PRODUCTS

Simplified Conversions and Equivalences

1 metric ton of crude oil	≃ 7.1 US barrels
1 US barrel of crude oil	≃ 0.14 metric ton
1 barrel	≃ 159 liters
1 m^3 of liquefied natural gas	≃ 600 m^3 of gas

ENERGY EQUIVALENTS OF NATURAL AND ENRICHED URANIUM

There are no official equivalents; the following coefficients depend on several factors:

- The thermal energy produced by the complete fission of the fissile uranium isotope
  1 gram of ^{235}U = 0.95 MWd = 1.96 TOE

- Different parameters intervene in the case of nuclear power (tails assay for the enrichment operation, energy consumed during enrichment, discharge burnup, re-fueling strategy, etc.).

Discharge burnup has a direct influence on thermal energy production, expressed in TOE per kg of uranium present in the nuclear fuel (natural or enriched, depending on the type of reactor):

Reactor type	Discharge burnup MWd/(ton of U)	Energy equivalent TOE/(kg of U)
PWR without Pu recycling	30,000	61.8
PWR without Pu recycling	40,000	82.4
PWR with Pu recycling	30,000	68.0
Natural uranium graphite and gas cooled reactor	2,900	6.5
Natural uranium and heavy water	5,800	13.0
Fast breeder reactor (case of a total combustion of the uranium)		500 TOE per kg of natural uranium

The previous table gives the energy produced from the uranium present in the nuclear fuel (enriched uranium in PWRs). The value for the breeder reactors is obtained after intensive recycling of the uranium and the plutonium produced.

To express this production from natural uranium, it is necessary to know the quantity of natural uranium required to obtain one kilogram of enriched uranium. This depends on two main factors: enrichment and the tails assay reached during the enrichment operation.

Tails assay	Natural uranium required to obtain 1 kg of enriched uranium		
	3.4% ^{235}U	3.7% ^{235}U	4% ^{235}U
0.2%	6.262 kg	6.849 kg	7.436 kg
0.25%	6.833 kg	7.484 kg	8.134 kg
0.30%	7.543 kg	8.272 kg	9.002 kg

Example:

Consider a PWR whose equilibrium cycle involves:
- a fuel containing 3.4% enriched uranium with a 0.2% tails assay,
- a discharge burnup of 40,000 MWd/t,
- no Pu recycling.

The equivalent thermal energy required is 82.4/6.262 = 13.15 TOE per kg of natural uranium.

5.1.4. Nuclear energy and fossil fuel savings

The oil required to generate the same amount of electric energy depends on the net efficiency of the nuclear plant.

The typical net efficiency of a nuclear plant is 33%. This means only a third of the thermal energy produced in the reactor is converted into electricity available for consumption. This means that the electricity produced by a kilogram of natural uranium, in the preceding example, would require three times the TOE used to obtain the corresponding thermal energy. This amounts to some 40 TOE saved by this plant.
For comparison, the total hydro-electric annual production in France, which was 56 TWh in 2005, represents an annual saving of 14.5 x 10^6 TOE.

This means that the same amount of electricity, produced in a fossil fuel plant, would have burned 14.5 x 10^6 TOE.

5.2. Energy and electricity

5.2.1. World energy and electricity

WORLD ENERGY CONSUMPTION IN 2005 (BY SOURCE, IN MTOE)

Country or Region	Final consumption of solid fuels	Final consumption of oil	Final consumption of gas	Final consumption of electricity
AMERICA (North)	48.7	954.3	364.5	365.7
United States	43.4	863.8	316.3	322.0
Canada	5.3	90.5	48.2	43.7
AMERICA (Latin)	14.3	0.3	68.4	80.5
AMERICA (South)	n.a.	n.a.	n.a.	n.a.
AMERICA (Central)	n.a.	n.a.	n.a.	n.a.
Argentina	0.6	20.9	19.7	7.8
Brazil	9.5	79.9	9.6	31.1
AFRICA	0.017	107.3	22.9	36.9
ASIA	0.6	903.4	134.1	385.5
China	434.3	268.4	39.0	168.1

Country or Region	Final consumption of solid fuels	Final consumption of oil	Final consumption of gas	Final consumption of electricity
Japan	46.4	21.5	26.9	81.4
MIDDLE EAST	1.2	194.4	88.2	43.3
SOVIET UNION (ex)	56.4	138.9	174.7	81.5
CEI	56.0	134.5	172.6	79.9
EUROPE (1)	84.8	666.2	311.3	267.3
EUROPEAN UNION (25) (2)	63.9	593.7	284.9	230.9
Denmark	0.2	7.6	1.7	2.9
France	5.8	87.6	34.6	36.0
Germany	13.4	112.9	62.4	44.3
Italy	4.9	67.8	42.7	25.7
Netherlands	2.3	29.0	22.7	9.0
Spain	2.0	62.3	17.7	20.7
Sweden	1.6	13.6	0.5	11.3
United Kingdom	4.7	78.0	51.0	29.7
WORLD	**819.8**	**3427.8**	**1178.7**	**1270.4**

(1) EUROPE means EUROPE of 27 countries without Romania and Bulgaria and with Switzerland, Norway
(2) EUROPEAN UNION (25) means EUROPE of 27 countries without Romania and Bulgaria

WORLD INSTALLED ELECTRICITY CAPACITY BY TYPE IN 2005 IN GWe

Country or Region	Installed electricity capacity	Installed thermal electricity capacity	Installed hydro-electricity capacity	Installed nuclear electricity capacity	Installed geothermal electricity capacity	Installed solar electricity capacity	Installed wind electricity capacity
AMERICA (North)	1205.6	916.3	163.6	112.5	3.0	0.5	9.7
United States	1082.7	878.2	91.9	100.0	3.0	0.5	9.1
Canada	122.9	381.0	71.7	12.5	n.a.	0.01	0.5
AMERICA (Latin)	272.7	124.9	142.8	4.4	1.4	n.a.	0.2
AMERICA (South)	198.6	67.9	127.5	3.0	n.a.	0.007	0.06
AMERICA (Central)	8.3	4.7	4.1	n.a.	0.5	n.a.	0.09
Argentina	27.1	16.3	9.8	1.0	n.a.	n.a.	0.03
Brazil	99.4	24.3	73.1	2.0	n.a.	0.007	0.03
AFRICA	117.9	91.9	23.7	1.8	0.2	0.009	0.3
ASIA	1240.4	919.0	233.6	79.6	3.5	1.4	4.7
China	533.0	410.6	115.0	6.5	n.a.	0.02	0.8
Japan	278.2	182.5	46.7	47.7	0.5	1.4	0.8

Country or Region	Installed electricity capacity	Installed thermal electricity capacity	Installed hydro-electricity capacity	Installed nuclear electricity capacity	Installed geothermal electricity capacity	Installed solar electricity capacity	Installed wind electricity capacity
MIDDLE EAST	143.1	132.8	10.2	n.a.	n.a.	0.001	0.1
SOVIET UNION (ex)	359.3	253.5	68.3	37.5	0.01	0.003	0.10
CEI	349.0	247.0	65.8	36.2	0.01	n.a.	0.04
EUROPE (1)	868.8	477.1	210.4	137.6	0.9	n.a.	41.4
EUROPEAN UNION (25) (2)	729.2	421.3	134.4	131.0	0.7	n.a.	40.3
Denmark	13.9	10.8	0.0	0.0	n.a.	0.0	3.1
France	117.8	28.2	25.6	63.4	n.a.	0.03	0.6
Germany	130.4	80.2	9.9	20.4	n.a.	1.5	18.4
Italy	85.5	62.2	21.0	0.0	0.7	0.04	1.6
Netherlands	22.5	20.8	0.037	0.4	n.a.	0.05	1.2
Spain	77.5	41.0	18.6	7.9	n.a.	0.06	10.0
Sweden	33.2	7.4	16.6	8.6	n.a.	0.004	0.5
United Kingdom	80.8	63.1	4.3	12.1	n.a.	0.01	1.4
WORLD	**4256.6**	**2953.6**	**863.5**	**372.2**	**9.4**	**n.a.**	**57.1**

(1) EUROPE means EUROPE of 27 countries without Romania and Bulgaria and with Switzerland and Norway
(2) EUROPEAN UNION (25) means EUROPE of 27 countries without Romania and Bulgaria

WORLD CONSUMPTION OF ELECTRICITY IN TWh

Country or region	1990	1995	2000	2005
AMERICA (North)	3049.2	3485.6	3977.1	4252.5
United States	2633.6	3042.0	3499.3	3744.0
Canada	415.6	443.6	477.8	508.5
AMERICA (Latin)	499.2	618.6	784.6	936.3
AMERICA (South)	364.1	457.8	567.6	n.a.
AMERICA (Central)	12.2	16.2	22.2	n.a.
Argentina	40.4	55.4	75.5	90.9
Brazil	210.8	256.5	319.4	361.7
AFRICA	246.5	285.6	340.7	429.1
ASIA	1830.5	2538.9	3189.8	4477.9
China	481.7	766.6	1044.3	1954.4
Japan	752.1	860.6	947.8	946.5
MIDDLE EAST	194.4	273.0	367.7	504.0

Country or region	1990	1995	2000	2005
SOVIET UNION (ex)	1247.7	925.3	872.0	947.8
CEI	1247.7	910.1	856.5	929.4
EUROPE (1)	2371.2	2515.9	2842.1	3108.5
EUROPEAN UNION (25) (2)	2022.7	2183.3	2465.4	2685.3
Denmark	28.4	30.9	32.4	33.2
France	301.9	342.6	385.4	419.1
Germany	455.1	451.2	490.2	514.9
Italy	214.2	237.8	272.5	298.7
Netherlands	73.5	83.1	97.9	104.6
Spain	125.8	140.9	188.5	240.8
Sweden	120.3	124.6	128.7	131.3
United Kingdom	274.4	294.7	329.4	344.8
WORLD	**9597.8**	**10,805.0**	**12,568.0**	**14,763.6**

(1) EUROPE means EUROPE of 27 countries without Romania and Bulgaria and with Switzerland and Norway
(2) EUROPEAN UNION (25) means EUROPE of 27 countries without Romania and Bulgaria

WORLD PRODUCTION OF ELECTRICITY IN TWh

Country or region	1990	1995	2000	2005
AMERICA (North)	3700.7	4142.1	4658.1	4910.5
United States	3218.6	3582.1	4052.5	4282.1
Canada	482.1	560.0	605.6	628.4
AMERICA (Latin)	611.8	770.4	979.4	1182.2
AMERICA (South)	446.2	565.7	706.5	n.a.
AMERICA (Central)	14.5	19.8	27.8	n.a.
Argentina	51.0	67.2	89.0	105.3
Brazil	222.8	275.6	349.2	402.9
AFRICA	316.9	364.3	438.8	555.5
ASIA	2215.2	3104.4	3947.7	5537.0
China	621.2	1007.7	1355.6	2475.0
Japan	844.7	971.4	1062.7	1057.5

Country or region	1990	1995	2000	2005
MIDDLE EAST	240.0	338.6	463.6	632.8
SOVIET UNION (ex)	1727.0	1293.4	1272.3	1399.5
CEI	1727.0	1266.8	1248.2	1370.4
EUROPE (1)	2845.8	3068.4	3427.7	3733.2
EUROPEAN UNION (25) (2)	2413.4	2631.6	2927.9	3186.7
Denmark	26.0	36.7	36.0	36.8
France	420.7	494.1	541.1	575.4
Italy	216.6	241.5	276.6	302.4
Germany	550.0	537.3	571.4	619.8
Netherlands	71.9	81.0	89.7	100.2
Spain	151.9	167.1	224.5	294.9
Sweden	146.5	148.4	145.3	159.0
United Kingdom	319.7	334.0	377.3	399.5
WORLD	**11,847.0**	**13,266.0**	**15,413.7**	**18,138.5**

(1) EUROPE means EUROPE of 27 countries without Romania and Bulgaria and with Switzerland and Norway
(2) EUROPEAN UNION (25) means EUROPE of 27 countries without Romania and Bulgaria

5.2.2. French energy and electricity

FRENCH PRIMARY ENERGY PRODUCTION BY ENERGY SOURCE IN MTOE

	1973	1980	1990	2000	2005	2006
COAL	17	13	8	2	0.3	0.2
OIL	2	2	4	2	1.3	1.3
NATURAL GAS	6	6	3	2	0.9	1
PRIMARY ELECTRICITY (1)	8	22	87	114	123	123
OF WHICH NUCLEAR	4	16	82	108	118	117
HYDROELECTRIC, wind, photovoltaic)	4	6	5	6	5	5.5
COMBUSTIBLE RENEWABLES (wood, etc.)	10	9	11	12	13	13
TOTAL	**44**	**52**	**112**	**132**	**138**	**138**
Energy Independance ratio %	24	27	50	50	50	50.5

(1) Nuclear, hydroelectric, wind and photovoltaic included

FRENCH PRIMARY ENERGY CONSUMPTION BY ENERGY SOURCE IN MTOE

	1973	1980	1990	2000	2005	2006
COAL	28	31	19	14	13	12
OIL	121	107	88	95	92	92
NATURAL GAS	13	21	26	38	41	40
PRIMARY ELECTRICITY (1)	8	22	83	109	117	118
COMBUSTIBLE RENEWABLES (wood, etc.)	10	8	12	13	13	13
TOTAL	**180**	**190**	**229**	**269**	**276**	**275**

(1) Nuclear, hydroelectric, wind and photovoltaic included

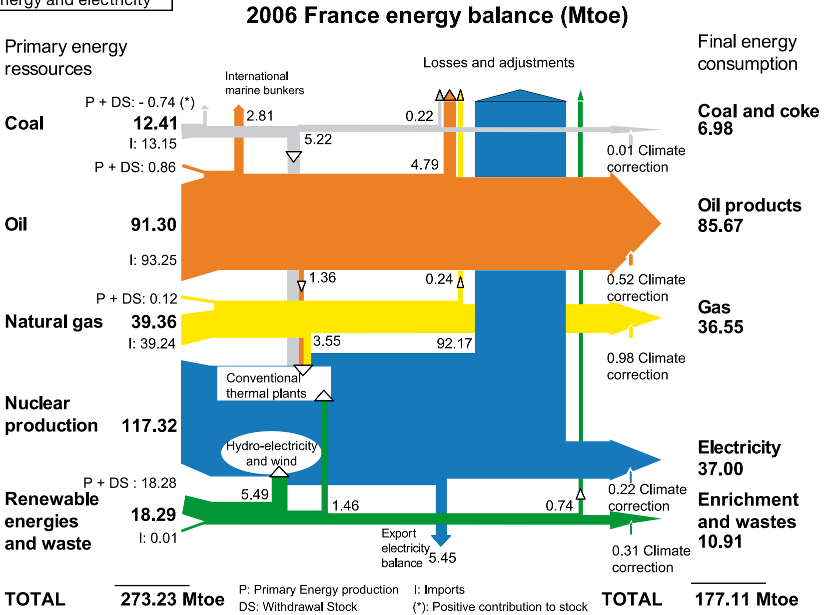
FRENCH ELECTRICITY PRODUCTION IN TWH

	1973	1980	1990	2000	2005	2006
CONVENTIONAL THERMAL	119	126	48	53	67	60.5
NUCLEAR	15	61	314	415	451	450
HYDROELECTRIC, WIND, PHOTOVOLTAIC	48	71	58	73	58	64
TOTAL	**182**	**258**	**420**	**541**	**576**	**575**

FRENCH ELECTRICITY PRODUCTION AND CONSUMPTION BALANCE IN TWH

	1973	1980	1990	2000	2005	2006
IMPORTS	5	n.a.	7	4	n.a.	8.5
EXPORTS	-8	n.a.	-52	-73	n.a.	72
EXTERNAL BALANCE	-3	n.a.	-46	-70	n.a.	63
PUMPING	0	n.a.	-5	-7	n.a.	n.a.
Consumption of ancillaries	-8	n.a.	-20	-24	n.a.	n.a.
France CONSUMPTION	171	n.a.	350	441	n.a.	480

Fig 5.2-1 French energy and electricity



5.3. The international system of units (SI)

Application of the International System of Units is obligatory in France, in the European Community (ISO 1000), and in numerous other countries with the exception of the USA, where it is only recommended by the NIST (National Institute of Standards and Technology).

5.3.1. Base units

Quantity	Unit	Symbol	Definition
Length	meter	m	The meter is the length of the path traveled by light in a vacuum during a time interval of 1/299, 792, 458 of a second (1).
Mass	kilogram	kg	The kilogram is the unit of mass; it is equal to the mass of the International prototype of the kilogram in the BIPM (France).
Time	second	s	The second is the duration of 9, 192, 631, 770 periods of the radiation corresponding to the transition between the two hyperfine levels of the ground state of the cesium 133 atom.

(1) This implies that the speed of light in vacuum is 299, 792, 458 m/s exactly.

Quantity	Unit	Symbol	Definition
Electric current	ampere	A	The ampere is that constant current which, if maintained in two straight parallel conductors of infinite length, of negligible circular cross-section, and placed 1 meter apart in a vacuum, would produce between these conductors a force equal to 2×10^{-7} Newtons per meter of length.
Thermodynamic temperature	Kelvin	K	The Kelvin, unit of thermodynamic temperature, is the fraction of 1/273.16 of the thermodynamic temperature of the triple point of water.
Amount of substance	mole	mol	The mole is the amount of substance of a system which contains as many elements as there are atoms in 0.012 kilogram of carbon 12. When the mole is used, the elementary entities must be specified and may be atoms, molecules, ions, electrons, other particles or specified groups of such particles.
Luminous intensity	candela	cd	The candela is the luminous intensity, in a given direction, of a source that emits monochromatic radiation at a frequency of 540×10^{12} hertz and that has a radiant intensity in that direction of 1/683 watt per steradian.

Important note: the unit of mass is the kilogram whose symbol is kg. It is an exception that the symbol of a base unit is a multiple of another symbol g; this symbol can be used as any other symbol. So it is legal to use the symbol mg (1 mg = 1×10^{-6} kg) to measure of very small masses.

5.3.2. Derived units

Derived units are usually expressed as a function of base units. Some have special names and symbols.

Quantity	Name of the derived SI unit	Symbol	Expressed in other SI units	Expressed in SI base units
Plane angle	radian	rad	1	$m.\ m^{-1}$
Solid angle	steradian	sr	1	$m^2.\ m^{-2}$
Frequency	hertz	Hz		s^{-1}
Force	Newton	N		$m.kg.s^{-2}$
Pressure, stress	pascal	Pa	N/m^2	$m^{-1}.kg.s^{-2}$
Work, energy, amount of heat	joule	J	N.m	$m^2.kg.s^{-2}$
Power	watt	W	J/s	$m^2.kg.s^{-3}$
Electric charge, amount of electricity	coulomb	C		s.A
Electric potential, potential difference, electromotive force	volt	V	W/A	$m^2.kg.s^{-3}.A^{-1}$
Capacitance	farad	F	C/V	$m^{-2}.kg^{-1}.s^4.A^2$
Electric resistance	ohm	Ω	V/A	$m^2.kg.s^{-3}A^{-2}$

Quantity	Name of the derived SI unit	Symbol	Expressed in other SI units	Expressed in SI base units
Electric conductance	siemens	S	A/V	$m^{-2}.kg^{-1}.s^3.A^2$
Magnetic flux	weber	Wb	V.s	$m^2.kg.s^{-2}.A^{-1}$
Magnetic flux density	tesla	T	Wb/m^2	$kg.s^{-2}.A^{-1}$
Inductance	henry	H	Wb/A	$m^2.kg.s^{-2}.A^{-2}$
Celsius temperature (see note below)	degree Celsius	°C		K
Luminous flux	lumen	lm	cd.sr	cd
Illuminance	lux	lx	lm/m^2	$m^{-2}.cd$
Activity (of a radionuclide)	becquerel	Bq		s^{-1}
Absorbed dose	gray	Gy	J/kg	$m^2.s^{-2}$
Dose equivalent	sievert	Sv	J/kg	$m^2.s^{-2}$
Catalytic activity	katal	kat		$mol.\ s^{-1}$

Note: the degree Celsius is not a measurable quantity, but it is observable. It is defined by the difference $t = T - T_0$ between two thermodynamic temperatures, T and T_0, with $T_0 = 273.15$ K.

5.3.3. Other authorized units - Units for specific applications

Quantity	Unit		
	Name	Symbol	Value
Volume	liter	l or L (1)	1×10^{-3} m^3
Length	angström	Å	1×10^{-10} m
Mass	Metric ton	t	1×10^{3} kg
Pressure	bar	bar	1×10^{5} Pa
Surface area of agrarian fields and estates	hectare	ha	1 ha = 1×10^{4} m^2
Plane angles	degree	°	$1° = \frac{\pi}{180}$ rad
	minute of angle	’	$1' = \frac{\pi}{10800}$ rad
	second of angle	”	$1'' = \frac{\pi}{648000}$ rad
Time	minute	mn	1 mn = 60 s
	hour	h	1 h = 3600 s
	day	d	1 d = 86 400 s
Area	barn	b	1 b = 1×10^{-28} m^2

(1) The symbol of the liter can be a lowercase or an uppercase character. It has to be chosen to avoid any confusion with the number 1.

5.3.4. Units defined independently of the seven base units (their value, expressed in SI units, is not precisely known)

Quantity	Unit		
	Name	Symbol	Definition and Experimental Value in 1973
Mass	atomic mass	amu	The atomic mass unit is equal to 1/12 of the mass of an atom of the C12 nuclide. 1 amu = 1.66053×10^{-27} kg
Energy	electronvolt	eV	The electronvolt is the kinetic energy acquired by an electron passing through a potential difference of 1 V in a vacuum. 1 eV = 1.60218×10^{-19} J.
Length	astronomical unit	AU	The astronomical unit is the average distance between the earth and the sun (center to center distance). 1 AU = 1.49600×10^{11} m (System of astronomical constants of the International Astronomical Union).
	parsec	pc	The parsec is the distance at which one AU subtends angle of one second. 1 pc = 206,265 AU = 3.0857×10^{16} m

5.3.5. Multiples of the SI units

Factor	Prefix	Symbol	Factor	Prefix	Symbol
10	deca	da	10	deci	d
10^{2}	hecto	h	10^{-2}	centi	c
10^{3}	kilo	k	10^{-3}	milli	m
10^{6}	mega	M	10^{-6}	micro	μ
10^{9}	giga	G	10^{-9}	nano	n
10^{12}	tera	T	10^{-12}	pico	p
10^{15}	peta	P	10^{-15}	femto	f
10^{18}	exa	E	10^{-18}	atto	a
10^{21}	*zetta*	*Z*	*10^{-21}*	*zepto*	*z*
10^{24}	*yotta*	*Y*	*10^{-24}*	*yocto*	*y*

Note: the prefixes zetta, yotta, zepto and yocto are listed here to conform to the SI reference text. But, even in this SI text, it is noted that their usage is not recommended.

5.4. Non SI units and Anglo-Saxon units

Due to its definition, the calorie (symbol cal) can be given several values in SI Unit:

- 1 cal th = 4.184 J
- 1 cal IT = 4.1868 J
- 1 cal (15°C) = 4.1858 J

An average value of 4.1855 is often used, but the value retained in this document is the value of the calorie (15°C) given in the following table.

Quantity	Unit		
	Name	Symbol	Value in SI Unit
Energy	calorie	cal	= 4.1858 J
	British thermal unit	btu	= 1.05506 x 10^3 J
	quad	Q	= 1 x 10^{15} btu = 1.05506 x 10^{18} J
Length	inch	in	= 0.0254 m
	foot	ft	= 12 in = 0.3048 m
Volume	gallon	gal	= 3.7854 x 10^{-3} m^3
Mass	ounce	oz	= 0.0283495 kg
	pound	lb	= 16 oz = 0.45359237 kg
Temperature	degree Fahrenheit	°F	T/°F = 1.8 T/°C + 32
Pressure	pound per square inch	psi	= 6.8948 x 10^3 Pa

5.5. Rules and style conventions for expressing values of quantities

A statement assigns a value to a quantity.
This value is expressed as the product of a number and a unit specifier.

Example: P = 1.5 x 10^3 W

The value and the unit must be separated by a space. A number is formed of an integral part, a decimal marker (optional), a decimal part (optional), an exponent between braces (optional and meaning a power of ten).

The decimal marker can be a dot (English usage) or a comma (French usage).

When the integral or decimal parts contain more than 3 digits, it is recommended to write these digits in groups of 3 digits, separated by a space. This practice of grouping digits does not apply in engineering drawings or in financial statements.

The exponent between braces is an engineering usage rather than a rule expressed in BIPM or NIST reference texts.

A unit specifier is formed of a prefix (optional) and a combination of one or several units.

Examples: cm, km/h, MWd/t

Note that symbols for units are treated as mathematical entities.

Thus we can write: **T/°F = 1.8 T/°C + 32**

The two terms of this statement are dimensionless. Such expressions are commonly used in labels of tables and in labels of the axes of a graph.

Symbols and names of the prefix and the unit must not be mixed:

kW or kilowatt	**but not**	kwatt or kiloW
m/s	**but not**	meter/s or m/second

Both unit symbols and unit names are invariable in the plural.

L = 6.5 m	**but not**	L = 6.5 ms or L = 6.5 meters

5.6. Common symbols and prefixes

A	ampere	**F**	farad
Å	angstrom	**°F**	degree Fahrenheit
atm	atmosphere	**ft**	foot
bar	bar	**G**	giga
bbl	barrel	**g**	gram
Bq	Becquerel	**gal**	gallon
btu	British thermal unit	**GWd/t**	gigawatt-day per metric ton
C	coulomb	**Gy**	gray
°C	degree Celsius	**H**	henry
Ci	curie	**h**	hecto
c	centi	**h**	hour
cal	calorie	**Hz**	hertz
cm	centimeter	**in**	inch
d	deci	**J**	joule
d	day	**K**	Kelvin
da	deca	**kg**	kilogram
deg	degree	**kWh**	kilowatt-hour
dm	decimeter	**l**	liter
eV	electronVolt	**lb**	pound

lm	lumen	**p**	pico
lx	lux	**rad**	radian
M	mega	**s**	second (of time)
MWe	megawatt-electric	**Sv**	sievert
MWth	megawatt-thermal	**sq.mile**	Square mile
MWd/t	megawatt-day per metric ton	**T**	tera
m	meter	**T**	tesla
m	milli	**t**	metric ton
mn	minute	**th**	thermie
mo	month	**V**	volt
mol	mole	**VA**	volt-ampere
μ	micro	**W**	watt
N	newton	**Wb**	weber
n	nano	**Wh**	watt-hour
Ω	ohm	**yd**	yard
oz	ounce	**yr**	year
Pa	pascal	**°**	degree (unit of angle)
pcm	millinile	**’**	minute (unit of angle)
Pl	poiseuille	**”**	second (unit of angle)
psi	pound per square inch		

5.7. Energy units and equivalents

The values in the following table are written using only five significant digits.

One \ Equals	joule	kilowatt-hour	calorie	electronvolt	British thermal unit
joule	1	2.7778×10^{-7}	0.23890	6.2415×10^{18}	9.4782×10^{-4}
kilowatt-hour	3.6000×10^{6}	1	8.6005×10^{5}	2.2469×10^{25}	3.4122×10^{3}
calorie (15°C)	4.1858	1.1627×10^{-6}	1	2.6126×10^{19}	3.9674×10^{-3}
electronvolt	1.6022×10^{-19}	4.4506×10^{-26}	3.8276×10^{-20}	1	1.5186×10^{-22}
British thermal unit	1.05506×10^{3}	2.9307×10^{-4}	252.08	6.5857×10^{21}	1

ENERGY UNITS GRAPH

The following chart shows the large range of values of the different energy units used.

To complete the above table and the following chart some other practical equivalents are given below:

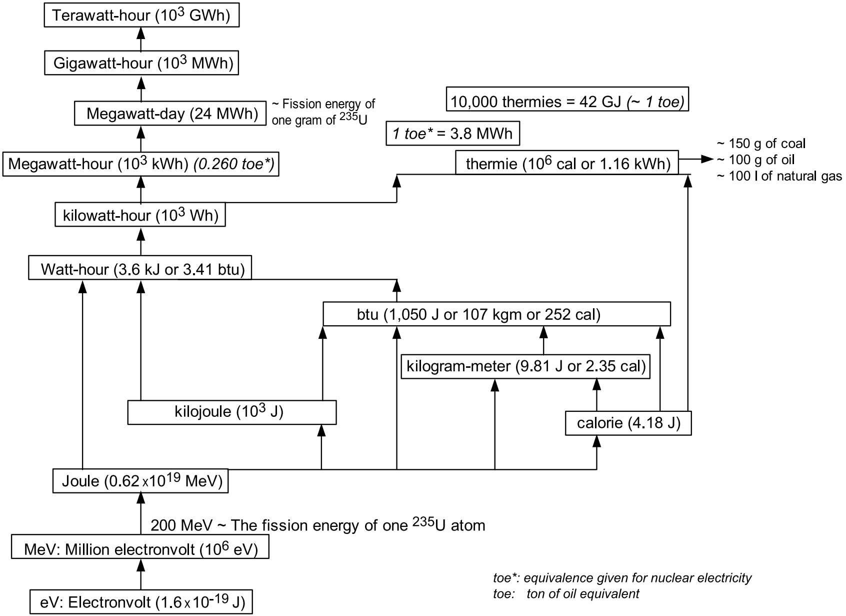
200 MeV $\cong$ the energy produced by the fission of one atom of ^{235}U

1 Thermie = 1 Mcal by definition
$\cong$ 1.16 kWh
$\cong$ the energy produced by the combustion of 150 g of coal, or 100 g of oil or 100 l of gas.

1 MWd $\cong$ the energy produced by the fission of one gram of ^{235}U

THE LARGE RANGE OF VALUES OF DIFFERENT ENERGY UNITS

Fig 5.7-1 Energy unit conversion



toe: equivalence given for nuclear electricity*
toe: ton of oil equivalent

5.8. Conversion tables

Example: = 1 angström = 10^{-10} meter

Multiply	by	to obtain
A		
ampere/cm^2	6.4516	A/in^2
ampere/in^2	0.1550	A/cm^2
angström	3.9370×10^{-6}	inch
angström	10^{-10}	meter
angström	10^{-4}	micrometer
are	100	m^2
atmosphere (1)	1.01325	bar
atmosphere	1.03323	kg/cm^2
atmosphere	1013.25	millibar
atmosphere	1.01325×10^5	pascal
atmosphere	14.696	psi
B		
bar	0.98692	atmosphere
bar	1.0197	kg/cm^2
bar	10^5	pascal
bar	14.504	psi

Multiply	by	to obtain
barn	10^{-24}	cm^2
barn	1.0765×10^{-27}	ft^2
barrel (maritime)	100	ft^3
barrel (maritime)	2.8317	m^3
barrel (oil)	42	US gallon
barrel (oil)	158.987	liter
barrel (oil)	0.158987	m^3
becquerel	1	disintegrations/s
becquerel	2.7027×10^{-11}	curie
btu	252.08	calorie
btu	777.95	ft-lb
btu	1055.06	joule
btu	107.59	kilogram-meter
btu	2.9297×10^4	kilowatt-hour
btu	2.5208×10^{-4}	thermie
btu	6.584×10^{15}	MeV
btu	1.2208×10^{-8}	MWd
btu	3.29×10^{13}	fissions of 200 MeV each
btu	1.28×10^{-8}	g of fissioned ^{235}U

Multiply	by	to obtain
btu/ft^2.h	2.7133	kcal/m^2.h
btu/ft^2.h	3.1546	W/m^2
btu/ft^2.h.°F	4.8839	kcal/m^2.h.K
btu/ft^2.h.°F	5.67826	W/m^2.K
btu/h	0.070022	cal/s
btu/h	0.29307	watt
btu/h.ft	9.6106 x 10^3	watt/cm
btu/h.ft.°F	1.4481	kcal/h.m.K
btu/mn	12.996	ft.lb/s
btu/mn	1.7580 x 10^{-2}	kilowatt
btu/lb	0.55555	cal/g
btu/lb	2326.0	J/kg
btu/lb.°F	1	cal/g.K
btu/lb.°F	4185.8	J/kg.K
btu/s	1.05506	kW
C		
calorie (internat.)	4.1868	J
calorie (thermoch.)	4.1840	J
calorie (at 15°C)	4.1858	J
calorie (average)	4.1855	J

Multiply	by	to obtain
calorie	3.9671 x 10^{-3}	btu
calorie	2.6126 x 10^{19}	electronvolt
calorie	3.0871	ft.lb
calorie	0.42680	kilogram-meter
calorie	1.1626 x 10^{-6}	kWh
calorie	1.305 x 10^{11}	fiss. of 200 MeV each
calorie	5.09 x 10^{-11}	g of fissioned ^{235}U
cal/cm^2.s	3.6 x 10^4	kcal/m^2.h
cal/cm^2.s	4.1855	W/cm^2
cal/g	1.7994	btu/lb
cal/g	4185.5	J/kg
Celsius (degree)	(9/5)°C + 32	Fahrenheit (degree)
centimeter	10^8	angström
centimeter	10^4	micrometer
centimeter	0.032808	foot
centimeter	0.39370	inch
centimeter	0.010936	yard
cm^2	10^{24}	barn
cm^2	10^{-4}	m^2
cm^2	1.0764 x 10^{-3}	ft^2

Multiply	by	to obtain
cm^2	0.15500	in^2
cm^2	1.1960×10^{-4}	yd^2
cm^3	3.5315×10^{-6}	ft^3
cm^3	6.1024×10^{-2}	in^3
cm^3	1.3080×10^{-6}	yd^3
cm^3	2.6417×10^{-4}	gallon
cm/s	1.9685	ft/mn
cm/s	0.39370	inch per second
cm/s	0.0360	km/h
cm/s	0.6	m/mn
cm/s	2.2369×10^{-2}	mi/h
cm/s^2	0.032808	ft/s^2
coulomb	2.7778×10^{-4}	Ah
coulomb/s	1	A
curie	3.7000×10^{10}	becquerel
D		
day	1440	minute
day	86 400	second
degree (angle)	0.0174533	radian
degree/s	0.0174533	radian/s

Multiply	by	to obtain
degree/s	2.7778×10^{-3}	rpm
degree (Celsius)	(9/5)°C + 32	degree (Fahrenheit)
degree (Fahrenheit)	5/9(°F - 32)	degree (Celsius)
°F/in	0.21872	K/cm
E		
electronvolt	0.38276×10^{-19}	calorie
electronvolt	1.6022×10^{-19}	joule
electronvolt	0.163×10^{-19}	kilogram-meter
eV	(see electronvolt)	
exajoule	10^{18}	joule
exajoule	$\cong 1$	quad
exajoule	$\cong 23 \times 10^6$	TOE
exajoule	$\cong 35 \times 10^6$	TCE
F		
Fahrenheit (degree)	5/9(°F - 32)	Celsius (degree)
fission	3.2041×10^{-11}	joule
fission	8.9058×10^{-18}	kWh
fission	200	MeV
fission	3.7108×10^{-23}	MWd

Multiply	by	to obtain
foot	30.480	cm
foot	12	inch
foot	0.30480	meter
foot	0.33333	yard
ft^2	9.2903×10^{-4}	are
ft^2	929.03	cm^2
ft^2	0.092903	m^2
ft^2	144	in^2
ft^2	0.11111	yd^2
ft^3	2.8317×10^4	cm^3
ft^3	1728	in^3
ft^3	0.037037	yd^3
ft^3	7.4805	gallon
ft^3	28.317	liter
ft^3	0.028317	m^3
ft^3/mn	471.95	cm^3/s
ft^3/mn	0.12467	gal/s
ft^3/mn	0.47195	l/s
ft^3/mn	1.6990	m^3/h
ft^3/s	0.64632	millions of gal/d

Multiply	by	to obtain
ft/mn	0.018288	km/h
ft/mn	5.080×10^{-3}	m/s
ft/s	30.480	cm/s
ft/s	1.0973	km/h
ft/s	0.3048	m/s
ft/s	0.68182	mi/h
ft/s^2	30.480	cm/s^2
ft/s^2	0.30480	m/s^2
ft/s^2	0.68182	mi/h.s
ft.lb	1.2851×10^{-3}	btu
ft.lb	0.32393	calorie
ft.lb	1.3558	joule
ft.lb	3.2393×10^{-4}	kcal
ft.lb	0.13825	kilogram-meter
ft.lb	3.7662×10^{-7}	kWh
ft.lb/mn	1.2851×10^{-3}	btu/mn
ft.lb/s	4.6264	btu/h
ft.lb/s	3.2393×10^{-4}	kcal/s
ft.lb/s	0.13825	kg.m/s
ft.lb/s	1.3558	watt

Multiply	by	to obtain
G		
g (acceleration due to gravity ISO)	980.665	cm/s^2
g	32.1740	ft/s^2
g	9.80665	m/s^2
gallon, US	0.13368	ft^3
gallon, US	231	in^3
gallon, US	4.9511×10^{-3}	yd^3
gallon, US	3.7854×10^{-3}	m^3
gallon, US	3.78541	liter
gal/mn, US	8.0208	ft^3/h
gal/mn, US	2.2280×10^{-3}	ft^3/s
gal/mn, US	0.063090	l/s
gal/mn, US	0.227125	m^3/h
gram	2.20462×10^{-3}	pound
gram (force)	9.80665×10^{-3}	newton
gram (force)	2.20462×10^{-3}	pound (force)
g/cm	5.5997×10^{-3}	lb/in
g/cm^2	9.80665×10^{-4}	bar
g/cm^2	98.0665	pascal

Multiply	by	to obtain
g/cm^2	2.0482	lb/ft^2
g/cm^2	1.4223×10^{-2}	psi
g/cm^3	1000	kg/m^3
g/cm^3	62.428	lb/ft^3
g/cm^3	3.6127×10^{-2}	lb/in^3
g/liter	6.2428×10^{-2}	lb/ft^3
g.cm	9.2947×10^{-8}	btu
g.cm	2.3430×10^{-5}	calorie
g.cm	7.2330×10^{-5}	ft.lb
g.cm	980.665×10^{-7}	joule
g.cm	10^{-5}	kg.m
gray	1	joule/kg
gray	100	rad
H		
hectare	10^4	m^2
hectare	1.07639×10^5	ft^2
hectobar	1.0197	kg/mm^2
hour	1.1408×10^4	year (Tropical)
hour	4.1667×10^{-2}	day
hour	60	minute

Multiply	by	to obtain
hour	3600	second
I		
inch	2.5400	centimeter
inch	0.083333	foot
inch	0.02540	meter
inch	1.5783×10^{-5}	mile
inch	0.027778	yard
in^2	6.4516	cm^2
in^2	6.9444×10^{-3}	ft^2
in^2	7.7160×10^{-4}	yd^2
in^3	16.387	cm^3
in^3	5.7870×10^{-4}	ft^3
in^3	4.3290×10^{-3}	gallon
in^3	16.387×10^{-3}	liter
J		
joule	9.4782×10^{-4}	btu
joule	0.23892	calorie
joule	0.73756	ft.lb
joule	2.3892×10^{-4}	kcal
joule	0.10197	kilogram-meter

Multiply	by	to obtain
joule	1	watt-second
joule	2.7778×10^{-7}	kWh
joule/kg	4.2992×10^{-4}	btu/lb
joule/kg	2.3892×10^{-4}	cal/g
K		
K/cm	4.5720	°F/in
kilocalorie	3.96707	btu
kilocalorie	3087.1	ft.lb
kilocalorie	4185.5	joule
kilocalorie	426.80	kilogram-meter
kilocalorie	1.1626×10^{-3}	kWh
kcal/h	1.1024×10^{-3}	btu/s
kcal/h	1.1626	watt
kcal/kg	1.8	btu/lb
kcal/kg.K	1	btu/lb.°F
$kcal/m^3$	0.11233	btu/ft^3
$kcal/m^2.h$	0.36855	$btu/ft^2.h$
$kcal/m^2.h$	1.1626	W/m^2
$kcal/m^2.h.K$	0.20476	$btu/ft^2.h.°F$
$kcal/m^2.h.K$	1.1626	$W/m^2.K$

Multiply	by	to obtain
kcal.m/m^2.h.K	8.0611	btu.in/ft^2.h.°F
kcal.m/m^2.h.K	1.1626	W.m/m^2.K
kcal.m/m^2.h.K	1.1626 x 10^{-2}	W.cm/cm^2.K
kilogram	2.2046226	pound
kilogram (force)	9.80655	newton
kilogram (force)	2.2046	pound (force)
kgf/cm^2	0.96784	atmosphere
kgf/cm^2	0.980665	bar
kgf/cm^2	9.80665 x 10^4	pascal
kg/cm^{2}kgf/cm^2	14.223	psi
kg/dm^3	62.428	lb/ft^3
kgf/m^2	9.80665 x 10^{-6}	bar
kg/m^2	10^{-4}	kg/cm^2
kgf/m^2	9.80665	pascal
kgf/m^2	14.223 x 10^{-4}	psi
kg/m^3	10^{-6}	kg/cm^3
kg/m^3	10^{-3}	kg/dm^3
kg/m^3	3.6127 x 10^{-5}	lb/in^3
kg/m^3	0.062428	lb/ft^3
kg/mm^2	100	kg/cm^2

Multiply	by	to obtain
kgf/mm^2	9.80665 x 10^6	pascal
kgf/mm^2	1422.3	psi
kilogram-meter	9.2949 x 10^{-3}	btu
kilogram-meter	2.3430	calorie
kilogram-meter	7.2330	ft.lb
kilogram-meter	9.80665	joule
kilogram-meter	2.3430 x 10^{-3}	kcal
kilogram-meter	2.7241 x 10^{-6}	kWh
kg.m/s	9.80665 x 10^{-3}	kilowatt
kilometer	3280.8	foot
kilometer	0.62137	mile
kilometer	1093.6	yard
km^2	100	hectare
km^2	1.0764 x 10^7	ft^2
km^2	0.38610	mi^2
km/h	0.91133	ft/s
km/h	0.27778	m/s
km/h	0.62137	mi/h
km/h.s	0.91133	ft/s^2
km/h.s	0.27778	m/s^2

Multiply	by	to obtain
km/h.s	0.62137	mi/h.s
kilowatt	3412.1	btu/h
kilowatt	238.92	cal/s
kilowatt	737.56	ft.lb/s
kilowatt	860.11	kcal/h
kilowatt	0.23892	kcal/s
kilowatt	101.97	kg m/s
kWh	3412.1	btu
kWh	8.6011×10^{5}	calorie
kWh	2.6552×10^{6}	ft.lb
kWh	3.6×10^{6}	joule
kWh	860.11	kcal
kWh	41.667×10^{-6}	MWd
kWh	2.2469×10^{19}	MeV
kWh	3600	kilojoule
kWh	3.6709×10^{5}	kilogram-meter
kWh	1.1235×10^{17}	fiss. of 200 MeV each
kWh	4.3840×10^{-5}	g of fissioned ^{235}U
kWh(e)	0.333×10^{-3}	TCE
kWh(e)	0.222×10^{-3}	TOE

Multiply	by	to obtain
kW/l	9.6644×10^{4}	btu/ft^{3}.h
kW/l	0.23892	cal/cm^{3}s
L		
liter	1000	cm^{3}
liter	61.024	in^{3}
liter	3.5315×10^{-2}	ft^{3}
liter	0.26417	gallon, US
liter	0.001	m^{3}
l/s	2.1189	ft^{3}/mn
l/s	15.850	gal/mn
l/s	3.6	m^{3}/h
lbf.s/ft^{2}	478.81	poise
lbf.s/ft^{2}	47.881	poiseuille
lb.ft	0.13826	m.kg
lb.ft	1.3558	m.N
lb.ft^{3}	1.6018×10^{-2}	g.cm^{3}
lb.ft^{3}	5.7870×10^{-4}	lb.in^{3}
lb.in^{3}	27.680	g.cm^{3}
lb.in^{3}	1728	lb.ft^{3}
lb/ft	1.4882	kg/m

Multiply	by	to obtain
lb/ft.s	14.882	poise
lb/ft.s	1.4882	poiseuille
lb/in	17.858	kg/m
lumen/m^2	1	lux
lumen/$foot^2$	1	foot candle
lumen/$foot^2$	10.764	lux
M		
megajoule	238.92	kcal
megawatt	860.11	th/h
MWd	8.189×10^7	btu
MWd	24×10^3	kWh
MWd	5.3916×10^{23}	MeV
MWd	2.696×10^{21}	fiss. of 200 MeV each
MWd	1.052	g of fissioned ^{235}U
meter	10^{10}	angström
meter	3.2808	foot
meter	39.370	inch
meter	10^6	micrometer
meter	6.2137×10^{-4}	mile
meter	1.0936	yard

Multiply	by	to obtain
m^2	10^{-4}	hectare
m^2	10.764	ft^2
m^2	1550.0	in^2
m^2	1.1960	yd^2
m^3	6.1024×10^4	in^3
m^3	1.3080	yd^3
m^3	264.17	gallon
m^3	1000	liter
m^3/h	0.58858	ft^3/mn
m^3/h	4.4028	gal/mn
m^3/h	16.667	l/mn
m^3/h	0.27778	l/s
m^3/mn	16.667	l/s
m/s	196.85	ft/mn
m/s	3.6	km/h
m/s	2.2369	mi/h
m/s^2	3.2808	ft/s^2
m/s^2	0.10197	g (acceleration of gravity)
m.kg	10^5	cm.g

Multiply	by	to obtain
m.kg	9.80665	joule
m.kg	7.2330	ft.lb
m.N	0.10197	m.kg
m.N	0.73756	ft.lb
MeV	1.602×10^{-13}	joule
µm	3.9370×10^{-2}	mil
µm	10^{-3}	millimeter
µm	10^{4}	angström
mil	0.001	inch
mil^2	6.4516×10^{-4}	mm^2
mile	5280	foot
mile	1.609344	kilometer
mile	1760	yard
mi^2	2.58999×10^{2}	hectare
mi^2	2.58999	km^2
mi^2	27.878×10^{6}	ft^2
mi^2	3.0976×10^{6}	yd^2
mi/h	1.4667	ft/s
mi/h	1.6093	km/h
mi/h	0.44704	m/s

Multiply	by	to obtain
millibar	100	pascal
millimeter	0.039370	inch
millimeter	39.370	mil
millimeter	1000	micrometer
mm^3	1	microliter
minute (angle)	0.016667	degree
minute (angle)	2.9089×10^{-4}	radian
minute	6.944×10^{-4}	day
minute	9.9206×10^{-5}	week
N		
newton	101.97	gram (force)
newton	0.22481	pound (force)
N/m	1	J/m^2
N/m	0.10197	kg/m
$N.s/m^2$	10	poise
$N.s/m^2$	1	poiseuille
O		
ounce	28.3495	gram
ounce	0.062500	pound
oz/in^2	0.062500	psi

Multiply	by	to obtain
P		
parsec	3.0857×10^{16}	meter
pascal	9.8692×10^{-6}	atmosphere
pascal	10^{-5}	bar
pascal	1.0197×10^{-5}	kgf/cm^2
pascal	0.10197	kgf/m^2
pascal	1.0197×10^{-7}	kg/mm^2
pascal	0.01	millibar
pascal	1	N/m^2
pascal	1.4504×10^{-4}	psi
picometer	10	angström
picometer	10^{-6}	millimeter
poise	0.0020885	$lbf.s/ft^2$
poise	0.1	poiseuille
poiseuille	1	$N.s/m^2$
poiseuille	10	poise
poiseuille	0.020885	$lbf.s/ft^2$
pound	453.59237	gram
pound	16	ounce
psi	0.068046	atmosphere

Multiply	by	to obtain
psi	0.068948	bar
psi	0.070307	kgf/cm^2
psi	7.0307×10^{-4}	kgf/mm^2
psi	6.8948×10^{3}	pascal
psi	144	lb/ft^2
Q		
quad	10^{15}	btu
quad	1.055×10^{18}	joule
quad	$\cong 3.65 \times 10^{7}$	TCE
quad	$\cong 2.44 \times 10^{7}$	TOE
quad	1.055	exajoule
R		
rad	0.01	gray
radian	57° 17’ 45 ”	degree (angle)
radian	3437.75	minute (angle)
radian	206265	second (angle)
radian	0.15915	revolution
rem	BEF	rad
S		
second (angle)	2.7778×10^{-4}	degree (angle)

Multiply	by	to obtain
second (angle)	1.6667×10^{-2}	minute (angle)
second (angle)	4.8481×10^{-6}	radian
second (time)	2.7778×10^{-4}	hour
second (time)	1.6667×10^{-2}	minute
sievert	100	rem
T		
TCE	3.000	kWh(e)
TCE	0.6667	TOE
therm (USA)	10^{5}	btu
thermie	3967.4	btu
thermie	10^{6}	calorie
thermie	4.1855×10^{6}	joule
thermie	1000	kcal
TOE	4 500	kWh(e)
TOE	1.5	TCE
ton (long)	1.01605	metric ton
ton (short)	0.907185	metric ton
tonne (metric ton)	1000	kilogram
tonne	2204.62	pound
tonne	0.98421	ton (long)

Multiply	by	to obtain
tonne	1.1023	ton (short)
U		
V		
W		
watt	3.4121	btu/h
watt	5.6869×10^{-2}	btu/mn
watt	0.23892	cal/s
watt	0.73756	ft.lb/s
watt	1	J/s
watt	0.86011	kcal/h
watt	2.3892×10^{-4}	kcal/s
watt	0.10197	kgm/s
Wh	3.4121	btu
Wh	860.11	calorie
Wh	2.6552×10^{3}	ft.lb
Wh	3600	joule
Wh	0.86011	kcal
Wh	367.10	kgm
W.s (=J)	9.4782×10^{-4}	btu
W.s	6.2419×10^{12}	MeV

Multiply	by	to obtain
W.s	3.1211×10^{10}	fiss. of 200 MeV each
W.s	1.217×10^{-11}	g of fissioned ^{235}U
W/cm	104.05	btu/h.ft
W/cm.K	57.803	btu/h.ft.°F
W/cm^2	3171.2	btu/h.ft^2
W/cm^2 K	1761.8	btu/h.ft^2.°F
W/in^2	1.5500	kW/m^2
weber/m^2	1	tesla
weber/m^2	6.4516×10^{-4}	weber/in^2
weber/in^2	1550	tesla
X		
Y		
yard	91.440	centimeter
yard	36	inch
yard	3	foot
yard	0.91440	meter
yd^2	8361.3	cm^2
yd^2	1296	in^2
yd^2	0.83613	m^2

Multiply	by	to obtain
yd^2	9	ft^2
yd^3	7.6455×10^5	cm^3
yd^3	27	ft^3
yd^3	46 656	in^3
yd^3	201.97	gallon
yd^3	764.55	liter
yd^3	0.76455	m^3
yd^3/mn	0.45	ft^3/s
yd^3/mn	3.3662	gal/s
yd^3/mn	12.743	l/s
yd^3/mn	45.874	m^3 /h
year (Tropical)	8765.8	hour
year (Tropical)	3.1557×10^7	second
Z		

ABBREVIATIONS, ACRONYMS AND WEBSITES

PART ONE - Abbreviations used for systems

PART TWO

PART THREE

PART ONE - Abbreviations used for systems:

French Three-Letter Codes (ECS)

	BUILDING ACRONYMS	KKS
APEC	Fuel removal workshop	
BAA	General and administration building	
BAC	Auxiliary building for conditioning (preparation of concrete casks for storage of radioactive wastes)	
BAG	General auxiliary building	
BAM	Mechanical auxiliary building	
BAN	Nuclear auxiliary building	UKA
BANG	General nuclear auxiliary building	
BAS	Safeguard building	UJH
BD	Diesel Buildings	
BDS	Control and security room	
BEX	Operation building (2-unit N4 site)	
BK	Fuel building	UFA

	BUILDING ACRONYMS	KKS
BL	Electrical building	
BR	Reactor building	UJA/UJB
BTE	Effluent treatment building	UKS
BW	Peripheral building	
BZ	Gas and chemical products storage building	
CPN	Nuclear production centre	
SDC	Control room	
SDM	Turbine Hall	UKE

ECS		KKS
A	**Feedwater Supply**	
AAD	Startup and Shutdown Feedwater System	
ABP	Low Pressure Feedwater Heater System	
ACO	Feedwater Heaters Drain Recovery System	
ADG	Feedwater Tank and Deaerator	
AET	Feedwater Pump Turbine Gland System (CP1)	
AFR	Turbine-driven Feedwater Pump Control Fluid System	

ECS		KKS
AGR	Feedwater Pump Turbine Lubrication and Control Fluid System	
AHP	High Pressure Feedwater Heater System	LAD
ANG	Main Feedwater System (CP0)	
APA	Motor-driven Feedwater Pump System	
APG	Steam Generator Blowdown System	LCQ
APP	Turbine-driven Feedwater Pump System	
APU	Feedwater Pump Turbine Drain System (CPY-CP1)	
APV	Feedwater Pump Turbine Drain System (CPY-CP2)	
ARE	Feedwater Flow Control System	
ARG	Feedwater Flow Control System (FSH)	
ASG	Auxiliary Feedwater System	LAR
ATD	Water treatment at startup	
ATH	Lubrication Polishing System	
C	**Condenser**	
CAP	Condenser Makeup and Discharge System	
CAR	Turbine Exhaust Water Spray System	
CAX	Feedwater pump turbine condenser (N4)	

ECS		KKS
CET	Turbine Gland System	
CEX	Condensate Extraction System	
CFI	Circulating Water Filtration System	
CGR	Circulating Water Pump Lubrication System	
CPA	Cathodic Protection System – Seaside Units (1300 – N4)	
CRC	Circulating Water Recycle System (CP2)	
CRF	Circulating Water System	
CSI	Circulating Water Isolation System (CP1)	
CTA	Condenser Tube Cleaning System	
CTE	Circulating Water Treatment System	
CTF	Cooling tower acid or raw water vaccination system	
CVF	Cooling towers – forced ventilation	
CVI	Condenser Vacuum System	
D		
DA (x)	Nuclear Island Elevators Systems (BR – BAN – BK) (CPY)	SN
DAK	Fuel building lifts and freight elevators	

ECS		KKS
DAL	Electrical building lifts and freight elevators	
DAN	Nuclear auxiliary building lifts and freight elevators	
DAR	Reactor building lifts and freight elevators	
DCC	Control Room Air Conditioning System (CP0)	
DCL	Main Control Room Air Conditioning System	SAB
DCM	Control Room – Diesel building – pump room – heating ventilation system (CP0)	
DEG	Nuclear Island Chilled Water System	
DEL	Electrical Building Chilled Water System	
DEQ	Waste Treatment Building Chilled Water (1300 – N4)	
DER	Operational Chilled Water System	QNA
DFL	Electrical Building Smoke Exhaust System	
DM (x)	Handling equipment	
DMD	Diesel building handling equipment	
DMK	Fuel building handling equipment	
DML	Safeguard auxiliary / electrical building handling equipment	
DMM	Turbine hall handling equipment	

ECS		KKS
DMN	Nuclear auxiliaries building handling equipment	
DMP	Circulating water pumping station handling equipment	
DMQ	Waste treatment building handling equipment	
DMR	Reactor building handling equipment	
DMS	Site handling equipment	
DN	Normal Lighting System	
DNX	6.6 kV Electric Power	
DS (x)	Emergency Lighting	
DSA	Emergency and Security Lighting (CPY)	
DSI	Site security system (Detection and Intrusion Surveillance system)	
DTE	Pneumatic sample transport	
DTF	Computer network system (office applications)	
DTL	Closed-Circuit Television System	CYP01
DTV	Site Communication System (transmission - telephone - paging - time distribution)	CYP01
DVA	Hot Workshop and Warehouse Ventilation System	
DVB	Administration Building Ventilation System	

ECS		KKS
DVC	Control Room Air Conditioning System	
DVD	Diesel Building Ventilation System	SAD
DVE	Cable Floor Ventilation System	
DVF	Electrical Building Smoke Exhaust System	
DVG	Auxiliary Feedwater Pump Room Ventilation System	
DVH	Charging Pump Room Emergency Ventilation System	
DVI	Component Cooling Room Ventilation System (CPY)	
DVK	Fuel Building Ventilation System	
DVL	Electrical Building Main Ventilation System	
DVM	Turbine Hall Ventilation System	
DVN	Nuclear Auxiliary Building Ventilation System	
DVP	Circulating Water Pumping Station Ventilation System	
DVQ	Waste Auxiliary Building Ventilation System (P'4 – N4)	
DVR	Electronic Equipment Room Air Conditioning System (1300 – N4)	
DVS	Safety Injection and Containment Spray Pump Motor Room Ventilation System	
DVT	Miscellaneous Room Air Conditioning System (P4 – N4)	

ECS		KKS
DVU	BDS and Main Access Control Ventilation and Air Conditioning System	
DVW	BW Rooms Ventilation System (1300 – N4)	
DVX	Electrical Rooms Access Ventilation (1300 – N4)	
DVY	Demineralization plant ventilation and air-conditioning system	
DVZ	Electrical Rooms Access Emergency Supplied Ventilation (P4 – N4)	
DWA	Hot Workshop and Warehouse Ventilation System	
DWB	Restaurant Ventilation System	
DWK	Fuel Building Ventilation System	KLL
DWL	Safeguard Building Controlled-area Ventilation System	KLC
DWN	Nuclear Auxiliary Building Ventilation System	KLE
DWQ	Waste Treatment Auxiliary Building Ventilation System (N4)	KLF
DWW	Access Building Ventilation System	
DWX	Miscellaneous Rooms Ventilation System (P'4)	
E	**Containment**	
EAS	Containment Spray System	
EAU	Containment Instrumentation System	JMY

ECS		KKS
EBA	Containment Sweeping Ventilation System	KLA
ECF	Containment Leakage Monitoring System (CP0)	
EDE	Secondary Containment Isolation (1300 – N4)	
EIE	Containment Isolation System (CP0 – 1300)	
EPP	Containment Leakage Monitoring System	
ETY	Containment Atmosphere Monitoring System	
EVC	Reactor Pit Ventilation System (CPY)	
EVF	Containment Cleanup System	
EVR	Containment Continuous Ventilation System	
EVU	Containment Heat Removal System	JMQ
G	**Turbine Generator**	
GCA	Turbine shutdown maintained-action	MAN
GCT	Turbine Bypass System	MAN
GEA	Standby offsite system transformer	BCT
GEV	Power Transmission System	
GEX	Generator Excitation and Voltage Regulation System	

ECS		KKS
GFR	Turbine Control Fluid System	
GGR	Turbine Lubrication, Jacking, and Turning System	
GHE	Generator Seal Oil System	
GME	Turbine measure elements (N4)	
GPA	Generator and Power Transmission Protection System	
GPV	Turbine Steam and Drain System	
GRE	Turbine Control System	
GRH	Generator Hydrogen Cooling System	
GRV	Generator Hydrogen Supply System	
GSE	Turbine Protection System	MYA
GSS	Moisture Separator Reheater System	
GST	Stator Cooling Water System	
GSY	Grid Synchronization and Connecting System	
GTH	Turbine Lube Oil Treatment System	
GTR	Turbine Generator Remote Control System	

ECS		KKS
H	**Buildings**	
HA	Site buildings and installation	
HB	Operational Control Center (POE)	
HB	Medecine, restaurant, training (MRF)	
HC	Releases, coolants	
HD	Diesel generator buildings (BD1, BD2, BD3, BD4)	
HD	Auxiliary diesel generator buidlings (BDSBO1, BDSBO4)	
HE	Switchyard	
HF	Non-classified electrical building (BLNC)	
HG	Galleries	
HH	Old SG temporary storage buildings	
HI	Central heating (hot water production)	
HJ	Auxiliary transformer platform (TA)	
HK	Fuel building (BK)	
HL	Electrical buildings (BL1, BL2 , BL3 , BL4)	
HL	Auxiliary safeguard buildings (BAS1, BAS2, BAS3, BAS4)	

ECS		KKS
HM	Turbine hall (SDM)	
HN	Nuclear auxiliary building (BAN)	
HO	Water storage building	
HP	Pumping station (including release) (SDP)	
HQ	Waste treatment building (BTE)	
HR	Reactor building (BR)	
HS	Environment, site	
HT	Main transformer platform	
HU	Site protection and main access	
HW	Access building to electrical building (ECS: peripheral buildings of reactor building and service building)	
HY	Demineralization building	
HZ	Gas and chemical product storage building	
J	**Fire**	
JAC	Classified Fire Water Production System	
JAN	Non-classified fire water production	

ECS		KKS
JDT	Fire Detection System	CYE
JPD	Fire Fighting Water Distribution	
JPH	Turbine Oil Tank Fire Protection System	
JPI	Nuclear Island Fire Protection System	SGB
JPL	Electrical Building Fire Protection System (CPY – P’4)	
JPP	Fire Fighting Water Production System	
JPS	Mobile and Portable Fire Fighting Equipment (CPY – N4)	
JPT	Transformer Fire Protection System	
JPU	Site Fire Fighting Water Distribution System (CP1)	
JPV	Diesel Generator Fire Protection System (CPY – N4)	
K	**Control**	
KAA	Analog Data Acquisition System (N4)	
KAC	Recording Aid System	
KBS	Cold Junction Boxes	
KCC	Data Transmission to national crisis center	
KCG	BAG Control System	

ECS		KKS
KCH	Demineralization I&C equipment (P'4)	
KCM	Turbine hall instrumentation and control equipment monitoring system	
KCO	Common Data Logic Relaying System (1300 – N4)	
KCP	Common Data Logic Relaying System – KDO Extension (1300)	
KCQ	Common Data Logic Relaying System – KDO Extension (1300)	
KCS	Common Data Logic Relaying System BDS – KDO Extension (1300)	
KCT	Common Data Logic Relaying System BTE – KDO Extension (P'4)	
KCU	Pumping station instrumentation and control equipment monitoring system	
KCY	Demineralisation plant instrumentation and control equipment	
KDO	Test Data Acquisition System	
KDS	Site Computer	
KER	Nuclear Island Liquid Radwaste Monitoring and Discharge System	
KGA	Digital Logic Automation System (CONTROBLOC)	
KGD	Site Data Monitoring System (N4)	
KGE	Waste management system	
KHY	BAN Hydrogen Detection System (1300)	

ECS		KKS
KIC	Operating Computer System	
KIR	Loose Parts and Vibration Monitoring System	JYF (loose parts) JYG (vibrations)
KIT	Centralized Data Processing System	
KKK	Site and Building Access Control System	
KKO	Energy Metering and Perturbography System	
KKV	Site video surveillance	
KLE	Local company network	
KLI	Local industrial network	
KME	Test Instrumentation System	
KOS	Trouble recorder	
KPE	Fast trouble recorder	
KPP	HV station perimeter surveillance system	
KPR	Remote Shutdown Panel	CXA
KPS	Safety Panel	
KRA	Nuclear Island Nitrogen Detection System (1300)	

ECS		KKS
KRC	Contamination and Dosimetry Monitoring System	
KRG	General Control Analog Cabinets	
KRH	Nuclear island hydrogen detection system	
KRS	Site Radiation and Meteorological Monitoring System	
KRT	Plant Radiation Monitoring System	CPR
KSA	Alarm Processing System (CPY)	
KSB	Demineralization Alarm Processing System (CP1)	
KSC	Main Control Room	
KSD	Monitoring and diagnostic aid system	
KSF	NSSS fatigue monitoring system	
KSN	Nuclear Auxiliary Building – Local Control Panels and Boards (CPY)	
KSD	Monitoring and diagnostic aid system	
KSF	NSSS fatigue monitoring system	
KSQ	Waste treatment building control room	
KSR	Remote shutdown panel	
KSU	Safeguard and alert panel	

ECS		KKS
KTG	Foundation block and cooling tower testing	
KZC	Controlled Area Access Monitoring System	
KZR	Unit Interface Computer CASOAR	
L	**Electricity**	
LA (x)	Power Supply and Distribution of DC Control and Display Voltages (230 V)	BT
LAK/L/M/N	Control rod drive mechanism power supply	
LAU/V	Turbine generator auxiliaires and control rod drive mechanism	
LB (x)	125 V DC Power Supply Systems	
LC (x)	48 V DC Power Supply Systems	
LD (x)	30 V and 48 V DC Power Supply	
LE	28 V DC Power System	BRW
LEC	Lighting (CP0)	
LEI	Switch / Circuit Breaker (CP0)	
LEX	Emergency Lighting (CP0)	
LG (x)	Preferred MV Network	BB
LGA/B/C/D	Conventional island 10 kV normal distribution system	

ECS		KKS
LGE	10 kV normal distribution Operational Control Station	
LGF/G/H/I	Nuclear island 10 kV normal distribution system	
LGK/N	Conventional island 10 kV normal distribution system	
LGR	Auxiliary transformer platform power supply	
LH (x)	Emergency Supply and Distribution MV Network	BD
LHA/B/ C/D	Emergency supplied 10kV distribution system (nuclear island + Essential service water system)	
LHP/Q/R/S	10 kV diesel generator set Division 1 - 2 - 3 - 4	
LI (x)	Normal 380 V Power Distribution (Extension) (CPY)	BF
LIA/B/C/D	Conventional island 690 V Normal Distribution system	
LIF/LII	Nuclear island 690 V Normal Distribution system	
LIL/M/N	Conventional island 690 V Normal Distribution - pre-treatment demineralization water	
LJ (x)	690 V Power Supply and Distribution Network	BM
LJA/B/C/D	690 V AC Emergency supplied distribution use on nuclear island	
LJF/LJI	690 V AC Emergency supplied distribution use on conventional island	
LJK/L	Emergency supplied 690 V AC use on conventional island	

ECS		KKS
LJR/S	690 V Emergency diesel generators Divisions 1 - 4	
LJU/X	Emergency supplied 690 V AC use on nuclear island, diesel buildins	
LJZ	Emergency supplied 690 V AC use on nuclear island, third fuel pool cooling system pump	
LK (x)	Normal 380 V Power Distribution (including Lighting)	BH
LKA/B/C/D	400 V normal distribution C.I.	
LKF/G/H/I	400 V normal distribution C.I. pre-treatment demineralisation water	
LKK/L	400 V normal distribution N.I.	
LKP/Q/R/S	400 V normal distribution divisions 1 - 2 - 3 - 4	
LL (x)	Emergency Supplied 380 V Power Distribution (including Lighting)	BM
LLA/B/C/D	400 V emergency-supplied distribution N.I.	
LLF/G/H/I	400 V emergency-supplied distribution N.I.	
LLK/L	400 V emergency-supplied distribution C.I.	
LLP/Q/R/S	400 V emergency-supplied distribution N.I.	
LM	Non-regulated 220 V Power Distribution	
LN (x)	220 V AC Power Supplies (Vital and Uninterrupted)	

ECS		KKS
LO (x)	400 V Power Supply and Distribution Network	
LOA/B/ C/D	400 V regulated distribution system, N.I.	
LS (x)	Test Loops	
LSI	Site lighting system	
LTA	Auxiliary Transformer (CP0)	
LTP	Main Transformer (CP0)	
LTR	Grounding Circuit	BAW
LV (x)	400 V uninterruptible Power Supply and Distribution Network	BR
LVA/B/ C/D	400 V uninterrupted power supply production and distribution N.I.	
LVF/G/H/I	400 V uninterrupted power supply production and distribution N.I	
LVK/L	400 V uninterrupted power supply production and distribution C.I	
LYS	Batteries Test Loops (CPY – 1300)	
P	**Spent Fuel Pool**	
PMA	Fuel Handling and Storage System	

ECS		KKS
PMB	Fuel containerizing and handling system	
PMC	Fuel Handling and Storage System	FAA
PME	New and spent fuel examination facility	
PMG	Stud Tensioning Machine System	
PML	Immersed lighting	
PMO	Handling Tools	
PMT	Fuel handling: Transfer and elevator	
PTR	Reactor Cavity and Spent Fuel Pit Cooling and Treatment System	FAE/ FAF/ FAK/ FAL
R	**Reactor**	
RAM	CRDM Power Supply	
RAZ	Nuclear Island Nitrogen Distribution System	
RBA	Automatic Boration (1300)	
RBS	Extra Borating System	JDH
RCP	Reactor Coolant System	JE

ECS		KKS
RCV	Chemical and Volume Control System	KBD (chemical) KBA (volume)
RDE	Steam Generator Decontamination System (CP0 – CPY)	
REA	Reactor Boron and Water Makeup System	KBC
REN	Nuclear Sampling System	KU
RES	Secondary Side of Steam Generator Sampling System	
RGL	Full-length Rod Control System	
RGN	Fuel Cladding Monitoring System (BGY)	
RHY	Hydrogen Distribution and Storage System (Nuclear Uses)	
RIC	In-Core Instrumentation System	JKS
RIS	Safety Injection System	JND/JNG
RPE	Nuclear Island Vent and Drain System	KT
RPN	Nuclear Instrumentation System	
RPR	Reactor Protection System	JR
RRA	Residual Heat Removal System	JNA
RRB	Boron Heating System	

ECS		KKS
RRC	Reactor Control System	JS
RRI	Component Cooling System	KAA/KAB
RRM	CRDM Ventilation System	
S	**General Services**	
SAA	*Production d'air respirable*	
SAD	Data Acquisition System (CPY)	
SAP	Compressed Air Production System	SCA/SCB
SAR	Instrument Compressed Air Distribution System	
SAT	Service Compressed Air Distribution System	
SBE	Hot laundry and decontamination system	
SCA	Auxiliary Steam Production System (CP0)	
SCO	CO_2 Storage (1300)	
SDA	Demineralized Water Production System	
SDB	Demineralization sludge treatment	
SDD	Nuclear island (reactor) demineralized water distribution (storage included)	
SDE	Waste Monitoring System (CPY)	

ECS		KKS
SDP	Demineralized Water Pretreatment System	
SDR	Conventional island Ph9 demineralized water distribution (storage included)	
SDS	*Production d'eau Déminéralisée par Dessalement*	
SDX	Reagent Storage and Demineralization Effluents Neutralization System	
SEA	Demineralized Water (Water Pre-treatment)	
SEB	Raw Water	
SEC	Essential Service Water System	PE
SED	Nuclear Island Demineralized Water Distribution	
SEF	Water Intake, Screening and Filtering System	
SEH	Waste Oil and Inactive Water Drain System	
SEI	Industrial Water System (1300 – N4)	
SEK	Conventional Island Liquid Waste Discharge System	
SEL	Electrically produced Hot Water Supply System	
SEN	Auxiliary Cooling Water System	
SEO	Plant Drainage System	GM
SEP	Drinking Water Distribution system	GKB

ECS		KKS
SER	Conventional Island Demineralized Water Distribution System	
SES	Hot Water Production and Distribution System (Nuclear Island)	
SEU	Rainwater drainage system	
SEZ	Ground water control system	
SFI	Raw Water Filtering System	
SGC	Carbon Dioxide Supply System	
SGH	Hydrogen Gas Supply System	QJC
SGL	Oil distribution system	
SGN	Nitrogen Gas Supply System	QJB
SGO	Oxygen Gas Supply System	QJA
SGZ	General Gas Storage and Distribution System	
SHM	Primary Motor-Pumps Oil Filling-Draining System (P4 – N4)	
SHY	Hydrogen Production and Distribution System (1300)	
SIR	Chemical Reagents Injection System	QC
SIS	Primary System Acoustic Monitoring	
SIT	Feedwater Chemical Sampling System	

ECS		KKS
SKH	Oil and Grease Storage System	
SKL	Oil storage	
SKR	Primary pump motor oil filling and draining	
SKZ	Gas Storage System	
SNL	Steam Generator Cleaning System	
SNV	Turbine Hall Cleaning (CP2)	
SRE	Sewage Recovery System (NI – Workshops – Site Laboratory) (BGY – CPY)	
SRI	Conventional Island Closed Cooling Water System	
SRU	Ultimate Cooling	
STB	Demineralization Sludge Treatment	
STE	Electrical Tracing System (Except boron heating)	
STF	Electrical tracing system (nuclear systems)	
STR	Steam Transformer System	
SVA	Auxiliary Steam Distribution System	LBG/LCN
SVE	Preoperational Test Steam Production and Distribution System	
SXS	Conventional Island Drain System (CP0)	

ECS		KKS
T	**Waste Treatment**	
TEG	Gaseous Waste Treatment System	KPL
TEK	Nuclear island radwaste monitoring and discharge system	
TEN	Waste Auxiliary Sampling System (P'4 – N4)	
TEP	Boron Recycle System	
TER	Liquid Waste Discharge System	
TES	Solid Waste Treatment System	KPA
TEU	Liquid Waste Treatment System	KPF
TPE	Waste Treatment Building Drain and Vent System (P'4)	
TRI	Waste Auxiliary Component Cooling System (P'4 – N4)	
V	**Main Steam**	
VCD	Turbine Bypass, Including to Atmosphere (CP0)	
VDA	Turbine Bypass to Atmosphere (EPR™ technology)	
VPU	Steam Line Drain System	
VTN	Main Turbine Driven Pump (CP0)	
VVP	Main Steam System	LB

ECS		KKS
X	**Auxiliary Steam**	
XAA	Auxiliary Boiler Feedwater Supply and Deaeration System (CPY – P4)	
XCA	Auxiliary Boiler	QH
Y	**Temporary Installation**	
YGV	Steam generator insrumentation system	

German Three-Letter Codes (KKS)

The KKS-System is used for uniform identification of plant buildings, systems and components. Development of the KKS-System started in 1970 in Germany. It is used in many NPPs around the world and is the basis for the international rules in IEC and ISO. It can be used for PWR as well as for BWR reactor types.

ORGANISATION OF THE KKS

The identifier is divided into breakdown levels and from left to right denotes progressively smaller units.
It employs the following basic format:

Breakdown Level	Total plant	Function					Equipment unit				Component		
Data unit	G	F_0	F_1	F_2	F_3	F_N	A_1	A_2	A_N	A_3	B_1	B_2	B_N
type (*)	A or N	N	A	A	A	NN	A	A	NNN	A	A	A	NN

(*) A for letter; N for digit

Function key

The following extract from the key contains the most important classifying parts of the function keys.

German Three Letter Codes (KKS)

KKS		ECS
A	**Grid and distribution systems**	
AC	380 (420) kV systems	
ACA	400 kV switchyard (on site)	
ACB	400 kV switchyard (off site)	
AE	110 - (150) kV systems	
AEA		
AR	Protection equipment	
ARB	Protection equipment	
AU	Open-loop control, checkback and auxiliary equipment	
AUA	400 kV switchyard auxiliary relay cubicle	
AV	Marshalling racks	
AVB	Optical cable panel	
B	**Power transmission and auxiliary power supply**	
BA	Power transmission	
BAA	Generator leads	

KKS		ECS
BAB	Generator bus duct auxiliary cabinets	
BAT	Main unit transformer	GEV
BAW	Earthing and lightning protection systems	LTR
BB	Medium-voltage distribution boards and transformers, normal power supply system	LG
BBA-BBD	10 kV unit switchgear	LGA - LGD
BBE	10 kV medium-voltage distribution board normal power supply system, NI division 1	
BBF	10 kV medium-voltage distribution board normal power supply system, NI division 2	
BBG	10 kV medium-voltage distribution board normal power supply system, NI division 3	
BBH	10 kV medium-voltage distribution board normal power supply system, NI division 4	
BBK	10 kV switchgear for electrical aux. boiler	
BBL	10 kV switchgear for electrical aux. boiler	
BBR	10 kV switchgear (NI maintenance power supply)	
BBS	10 kV switchgear (NI maintenance power supply)	
BBT	High-voltage auxiliary power transformer	GEV
BBY	Medium-voltage supply to TVO	
BC	Medium-voltage distribution boards and transformers, general purpose	GE

KKS		ECS
BCT	Standby offsite system transformer	GEA
BD	Medium-voltage distribution boards, (diesel) emergency power supply system	LH
BDA-BDD	10 kV medium-voltage distribution board emergency power supply system, NI division 1 - division 4	LHA - LHD
BF	Low-voltage distribution boards and transformers, normal power supply system	LI / LK
BFA-BFD	690 V unit low-voltage switchgear	LIA - LID
BFE-BFM	400 V low-voltage distribution board normal power supply system, NI division 1 - division 4	
BFN	690 V low-voltage distribution board normal power supply system, NI division 1	
BFR	690 V low-voltage distribution board normal power supply system, NI division 4	
BFT	690 V / 400 V low-voltage Unit Transformer	LIA
BH	Low-voltage main distribution boards and transformers for general purpose, normal power supply system	LI / LK
BHA-BHF	690 V unit low-voltage switchgear, station service	
BHK	400 V low-voltage subdistribution board normal power supply system, NI division 1	
BHL	400 V low-voltage subdistribution board normal power supply system, NI division 2	
BHM	400 V low-voltage subdistribution board normal power supply system, NI division 3	

KKS		ECS
BHN	400 V low-voltage subdistribution board normal power supply system, NI division 4	
BHT	Low-voltage unit transformer, station service	
BJ	Low-voltage subdistribution boards and transformers, normal system	
BJA	400 V subdistribution board (TI)	
BJB	400 V subdistribution board (TI)	
BJC	400 V subdistribution board (TI)	
BJD	400 V subdistribution board (TI)	
BJT	400 V regulating transformers (TI)	
BK	Low-voltage distribution boards and transformers for 30UKA and 30UKS, normal power supply system	
BKA	690 V low-voltage distribution board normal power supply system for 30UKA, NI division 1	
BKD	690 V low-voltage distribution board normal power supply system for 30UKS, NI division 4	
BKH	400 V low-voltage distribution board normal power supply system for 30UKS, NI division 4	
BKT	Low-voltage auxiliary power transformer	

KKS		ECS
BL	Low-voltage subdistribution boards and transformers for general purpose, (diesel) emergency power supply system	
BLA	400 V subdistribution for emergency diesel generator, NI division 1	
BLB	400 V subdistribution for emergency diesel generator, NI division 2	
BLC	400 V subdistribution for emergency diesel generator, NI division 3	
BLD	400 V subdistribution for emergency diesel generator, NI division 4	
BLE	400 V subdistribution for station blackout diesel generator, NI division 1	
BLH	400 V subdistribution for station blackout diesel generator, NI division 4	
BLT	Low-voltage auxiliary power transformer	
BM	690 V low-voltage distribution boards and transformers, (diesel) emergency power supply system	LJ / LL
BMA-BMD	690 V low-voltage switchgear emergency power supply system, NI division 1 - division 4	LJA - LJD
BME-BMM	690 V subdistribution emergency power supply system, NI division 1 - division 4	
BMN	690 V subdistribution emergency power supply system, NI division 1	
BMP	690 V subdistribution emergency power supply system, NI division 4	
BMR	690 V emergency power supply switchgear (TI)	

KKS		ECS
BMS	400 V emergency power supply switchgear (TI)	
BMT	Low-voltage auxiliary power transformer	
BMU	400 V transformer, emergency power supply (TI)	
BMW	690 V low-voltage switchgear emergency power supply system, NI division 1	
BMZ	690 V low-voltage switchgear emergency power supply system, NI division 4	
BN	400 V low-voltage distribution boards and transformers, (diesel) emergency power supply system	LL / LO
BNA-BNH	400 V low-voltage switchgear emergency power supply system, NI division 1 - division 8	
BNT	Low voltage auxiliary power transformer	
BNJ	400 V low-voltage switchgear emergency power supply system, NI division 1	
BNK	400 V low-voltage switchgear emergency power supply system, NI division 2	
BNL	400 V low-voltage switchgear emergency power supply system, NI division 3	
BNM	400 V low-voltage switchgear emergency power supply system, NI division 4	
BNP	400 V AC distribution 12h uninterruptible power supply, NI division 1	
BPA	Sub-distribution for 30UYA	
BPB	Sub-distribution for 30UYA	

KKS		ECS
BPC	Sub-distribution for 30UYA	
BPD	Sub-distribution (back-up diesel) for 30UYA	
BPE	Sub-distribution (pump station) for 30UYA	
BPF	Sub-distribution (30UYA heating) for 30UYA	
BPG	Sub-distribution for 30UYA	
BPH	Sub-distribution for 30UYA	
BPJ	Sub-distribution (mechanical workshop) for 30UYA	
BPK	Sub-distribution (equipment lift) for 30UYA	
BPL	Sub-distribution for 30UYA	
BPM	Sub-distribution for 30UYA	
BPN	Sub-distribution for 30UYA	
BPP	Sub-distribution for 30UYA	
BPQ	Sub-distribution for 30UYA	
BPR	Sub-distribution for 30UYA	
BPS	Sub-distribution for 30UYA	
BPT	Sub-distribution (ventilation and AC room) for 30UYA	

KKS		ECS
BPU	Sub-distribution for 30UYA	
BPV	Sub-distribution for 30UYA	
BPW	Sub-distribution for 30UYA	
BPX	Sub-distribution for 30UYA	
BNQ	400 V AC distribution 12h uninterruptible power supply, NI division 2	
BNR	400 V AC distribution 12h uninterruptible power supply, NI division 3	
BNS	400 V AC distribution 12h uninterruptible power supply, NI division 4	
BP	Distribution boards for 30UYA	
BR	Low-voltage distribution boards, inverters and AC/DC-converters, uninterruptible power supply	LV / LL
BRA-BRK	400 V AC uninterruptible power supply switchgear for 2 h uninterruptible power supply, NI division	LVA - LVI
BRM	400 V AC uninterruptible power supply distribution board (TI)	
BRN	400 V AC uninterruptible power supply distribution board (TI)	
BRT	Converter (rotary)	
BRU	400 V AC Inverters for uninterrupted power supply	

KKS		ECS
BRV	AC/DC-converters for control voltage 220 V DC of switchgear	
BRW	28 V DC Power System	LE
BRX	AC/DC-converters for power supply I&C 24 V DC, non-safety	
BT	Battery systems	LA / LF
BTA	220 V battery	LF
BTD	220 V battery	LYS
BTJ	Battery	
BTL	Rectifier, battery charger	
BTP	220 V DC battery chargers	
BTQ	Rectifier, battery charger	
BTR	Rectifier, battery charger	
BTU	Rectifier, battery charger	
BTV	Battery feeder cubicles (TI)	
BTW	Common battery equipment	
BTX	Discharge converter	
BTY	Common equipment	

KKS		ECS
BU	DC distribution boards, normal system	LA
BUA-BUD	220 V DC control rod power supply switchgear, rod control, NI division 1 - division 4	LAK - LAN
BUE	220 V DC distribution from 2 h battery, NI division 1	
BUF	220 V DC distribution from 2 h battery, NI division 2	
BUG	220 V DC distribution from 2 h battery, NI division 3	
BUH	220 V DC distribution from 2 h battery, NI division 4	
BUK	220/24 V DC converters	
BUM	220 V DC switchgear (TI)	
BUN	220 V DC switchgear (TI)	
BUW	DC/DC-converters for power supply I&C 24 V DC, safety	
BUX	DC/DC-converters for power supply I&C 24 V DC, non-safety	
BV	Direct voltage distribution board, emergency power system 1	
BVA-BVS	Direct voltage emergency distribution board	
BW	Direct voltage distribution boards, emergency power system 2	
BWE-BWS	Direct voltage emergency distribution board	
BX	Fluid supply system for control and protection equipment	

KKS		ECS
BZ	Cables and lighting small power	
BZB	Cable trays	
BZK	Cables	
BZL	Lighting and small power system	
C	**Instrumentation and control equipment**	
CA	Protective interlocks	
CB	Functional group control, subloop control	
CBP	Synchronisation 400 kV unit circuit breakers	
CBQ	10 kV High speed bus bar transfer system	
CC	Binary signal conditioning	
CD	Drive control interface	
CE	Annunciation	
CF	Measurement, recording	
CFQ	Interface cabinets for fault recording system	
CG	Closed-loop control (excl. power section)	
CH	Protection (excl. reactor protection)	

KKS		ECS
CHA	Electrical generator and HV transformer protection, synchronization, transformer voltage regulation	
CHB	Auxiliary standby transformer protection, measuring, metering, transformer voltage regulation	
CHY	Metering and measuring, Interface cabinet for 400 kV grid connection	
CJ	Unit coordination level	
CJJ	Instrumentation and control cabinets for steam turbine set	
CJN	Instrumentation and control cabinets for steam turbine set, generator voltage controller	
CK	Process computer system	
CKG	Online supervisory and diagnostic computer	
CL	Safety I&C systems	
CLE-CLH	Safety I&C systems (Schranke)	
CLJ-CLZ	Reactor protection	
CM	Reactor power instrumentation and control	
CMA-CMD	Cabinets for reactor control, surveillance and limitation system, division 1 - division 4	
CME-CMH	Instrumentation and control equipment	

KKS		ECS
CMN	Instrumentation and control equipment	
CMV-CMY	Instrumentation and control equipment	
CN	Core monitoring, incl. loose parts and vibration monitoring of secondary system	
CNC	Cabinets for expert systems	
CNM	POWERTRAX/E online core monitoring system	
CNN	Cabinets for aeroball system	
CP-CQ	Instrumentation and control equipment	KRA/ KRT
CPE	Seismic monitoring system	
CPF	Boron concentration measurement system	
CPR	Plant Radiation Monitoring System	KRT
CPS	Environmental monitoring system	
CR	PAS - process automation system	
CRA	PAS - process automation system	
CRB	PAS - process automation system	
CRD	PAS - process automation system	
CRT	PAS - process automation system, remote I&C cabinets for FINGRID	

KKS		ECS
CRU	PICS - process information and control system	
CRV	Configuring system	
CRX	PAS - isolation cabinet	
CRY	PAS - process automation system	
CRZ	PAS - process automation system	
CS	Environmental Monitoring System	
CSE	Cabinets for severe accident I&C, division 1	
CSF	Cabinets for severe accident I&C, division 2	
CSG	Cabinets for severe accident I&C, division 3	
CSH	Cabinets for severe accident I&C, division 4	
CT	Local control panels	
CTD	Local control panel 30KBD	
CTK	Local control panels 30KBE	
CU	Closed-loop control (power section)	
CUL	Cabinets for interposing relays for solenoid valves	
CV	Marshalling racks	

KKS		ECS
CW	Main control room	KSC
CWA	Main control desk (reactor and turbine operator)	
CWB	Main control desk (shift supervisor)	
CWD	Main control desk (auxiliary systems)	
CWE	Main control console	
CWJ-CWU	Protection system panel; part of SICS (30CWY)	
CWV	Additional equipment in control rooms	
CWW	Control room	
CWX	Control room	
CWY	SICS - safety information and control system	
CX	Local control stations (e. g. for solid fuel supply system, ash removal system, cooling water system, diesel engine plant, monitoring of generator cooling, emergency control station)	
CXA	Remote shutdown station	KPR
CXB-CXZ	Emergency diesel instrumentation and control cabinets	
CY	Communication equipment	DTV / DTL

KKS		ECS
CYA	Telephone system	
CYB	Intercommunication system (control console telephone system)	
CYC	Public address and alarm systems	
CYE	Fire Detection System	JDT
CYP	Optical monitoring system (plant TV system)	
CYP01	Closed-Circuit Television System	DTL
CYQ	Detection system for toxic gases	
CYT	Communication system (TVO)	
CYV	VIRVE radio system	
CZ	Physical protection systems	
CZD	Security alarm system	
CZE	Physical protection closed cuircuit television system	
CZG	Physical protection intercommunication system	
CZK	Access control system	
CZP	Control stations and desks	
CZQ	Physical protection power supply	

KKS		ECS
D	**Instrumentation and control equipment**	
DA-DQ	- blocked -	
DMA	Plant overview displays	
DMB	Function unit overview displays	
DMC	Function complex overview displays	
DMD	Process information area overview displays	
DR	SAS - safety automation system	
DRA	SAS - safety automation system	
DRB	SAS - safety automation system	
DRD	SAS - safety automation system	
DRX	SAS - isolation cabinet	
DRY	SAS - safety automation system	
DS	- blocked -	
E	**Conventional fuel supply and residues disposal**	

KKS		ECS
F	**Handling of nuclear equipment**	
FA	Storage of fuel assemblies (also includes breeder and reflector assemblies) and other radioactive components	
FAA	New fuel storage and new fuel handling tool	PMC
FAB	Irradiated fuel assemblies store (fuel pool)	PMC
FAE	Reactor Cavity and Spent Fuel Pit Cooling and Treatment System	PTR
FAF	Reactor Cavity and Spent Fuel Pit Cooling and Treatment System	PTR
FAK	Fuel pool cooling system	PTR2
FAL	Fuel pool purification system	PTR3
FB	Handling of fuel assemblies (also includes breeder and reflector assemblies) and other reactor core internals	
FBA	Fuel assemblies and other reactor core internals testing equipment	PMC
FBB	Fuel assemblies and other reactor core internals repair equipment	
FBC	Fuel assemblies and other reactor core internals cleaning equipment	
FC	Refueling and transport equipment for fuel assemblies (also includes breeder and reflector assemblies) and other reactor core internals	
FCB	Refueling equipment at reactor	

KKS		ECS
	Refueling machine	
FCD	Refueling equipment in store for fuel assemblies and other reactor core internals	
FCJ	Transport equipment	PMC
FJ	Erection and in-service inspection equipment	
FJA	Reactor pressure vessel - closure head equipment	
FJB	Tools and erection equipment for reactor pressure vessels internals	
FJC	Non destructive examination of the reactor pressure vessel	
FJE	Tools and erection equipment for reactor coolant system components	
FJF	Non destructive examination of the reactor coolant system	
FK	Decontamination equipment (excluding cleaning equipment for fuel assemblies store *FAM* and fuel assemblies *FBC*)	
FKE	Decontamination system for process equipment and vessels	
FKK	Decontamination equipment for small machine components (in 30UKS building)	
FX	Fluid supply system for control and protection equipment	
FY	Control and protection equipment	

KKS		ECS
G	**Water supply and disposal**	
GA	Raw water supply	
GB	Treatment system (carbonate hardness removal) incl. cooling tower make-up water treatment system	
GC	Treatment System (demineralization)	
GD	Steam Generator Blowdown Demineralizing System	APG2
GDA	Steam generator blowdown demineralizing main system	
GDN	Chemicals supply system	
GDP	Regeneration, flushing equipment	
GH	Distribution Systems (not drinking water)	
GHC	Demineralized Water Distribution System	SDD/ SDR
GHD	Distribution system after treatment	
GHW	Seal Water Supply System	SDD2
GK	Potable and Sanitary Water Systems	
GKB	Drinking Water Distribution System	SEP
GM	Plant Drainage System	SEO

KKS		ECS
GMA	Process drains collection and disposal system for transformer structures	
GMB	Process drains collection and disposal system for 31/32/33/34UBP	
GMC	Process drains collection and disposal system for 30UQA	
GMD	Process drains collection and distribution system for 30UBA	
GME	Process drains collection and distribution system for 30UBZ	
GMJ	Process drains collection and disposal system for 31/32/33/34UJH	
GMK	Process drains collection and disposal system for 30UKE	
GMM	Central disposal system	
GMQ	Process drains collection and disposal system for 31/32/33/34UQB	
GMT	Process drains collection and disposal system for 30UTH	
GMZ	Process drains collection and disposal system for 31UMZ	
GN	Process Drains Treatment System	
GQ	Domestic Waste Water Collection and Drainage System incl. sewage	SEO
GQA	Domestic waste water collection and drainage system incl. sewage for 31/32/33/34UJH, 31/32/33/34UJK, 30UKE	
GU	Rainwater collection and drainage system	

KKS		ECS
GUA	Rainwater collection and drain system for NI	
J	**Nuclear heat generation**	
JA	Reactor System	RCP
JAA	Reactor pressure vessel incl. support	RCP8
JAB	Reactor pressure vessel accessories	RCP6
JAC	Reactor pressure vessel internals	RCP7
JAH	Reactor pressure vessel thermal insulation	
JD	Reactor control and shutdown equipment	
JDA	Control rod drive mechanism	RGL
JDE	Solid Neutron Absorber Shutdown System	
JDH	Liquid Neutron Absorber Shutdown System (extra borating system)	RBS
JDJ	2nd Backup Liquid Neutron Absorber Shutdown System (boron shutdown system)	
JE	Reactor Coolant System	RCP
JEA	Steam generator	RCP1
JEB	Reactor coolant pump	RCP2
JEC	Reactor coolant line incl. surge line	RCP3

KKS		ECS
JEF	Reactor Coolant Pressurizing System (incl. pressurizer, valves, piping, etc.)	RCP4
JEG	Pressurizer Relief Discharge System (incl. pressurizer relief tank)	RCP5
JEW	Reactor Coolant Pump Seal Injection and Leak-off System	RCV2
JK	Reactor core with appurtenances	
JKA	Fuel assemblies	
JKB	Rod control cluster assemblies, neutron sources, flow restrictors (thimble plug assemblies)	
JKQ	Aeroball System	RIC2
JKS	In-Core Instrumentation System	RIC
JKT	Excore instrumentation	RPN
JM	Containment	EPP
JMA	Containment liner	ETY
JMB	Core melt stabilization system	
JME	Equipment hatch	EPP2
JMF	Personnel airlock	EPP3
JMG	Emergency airlock	EPP4

KKS		ECS
JMH	Construction opening	
JMJ	Steam generator pressure equalization ceiling	
JMK	Piping penetrations	
JML	Cable penetrations	
JMM	Leakage Exhaust and Monitoring System	EPP1
JMQ	Containment Heat Removal System	EVU
JMT	Combustible gas control	
JMU	Hydrogen Monitoring System	ETY2
JMY	Containment Instrumentation System	EAU
JN	Safety Injection and Residual Heat Removal System	RIS/ RRA
JNA	Residual Heat Removal System	RRA
JNB	Emergency Residual Heat Removal System	RIS
JND	Medium Head Safety Injection System	RIS
JNG	Low Head Safety Injection System	RIS
JNK	In-containment refueling water storage tank	RWST
JNP	Function Testing System	

KKS		ECS
JNY	Control and protection equipment	
JQ	Hardwired back-up system	
JQE	Hardwired back-up system, division 1	
JQF	Hardwired back-up system, division 2	
JQG	Hardwired back-up system, division 3	
JQH	Hardwired back-up system, division 4	
JR	PS - Protection System	RPR
JRE	PS - protection system, division 1	
JRF	PS - protection system, division 2	
JRG	PS - protection system, division 3	
JRH	PS - protection system, division 4	
JRY	Information displays	
JS	Reactor Control Surveillance and Limitation System	RGL/ RRC
JSA	Reactor control surveillance and limitation system, division 1	
JSB	Reactor control surveillance and limitation system, division 2	
JSC	Reactor control surveillance and limitation system, division 3	

KKS		ECS
JSD	Reactor control surveillance and limitation system, division 4	
JSN	Reactor control surveillance and limitation system, belong more divisions	
JT	Reactor Operational, Protective and Status Limitation System	
JY	Control and protection equipment (other than *JR*, *JS*, *JT*)	
JYC	Contamination monitor	
JYE	DIROM - diagnostic of rotating machinery	
JYF	Loose Parts Monitoring System	KIR1
JYG	Vibration Monitoring System incl. main coolant pump	KIR2
JYH	Leakage Monitoring System	EPP
JYK	Radioactivity Monitoring System	KRT
JYL	FAMOS - fatigue monitoring system	
JYM	Vibration monitoring system for main steam line and main feedwater line	
JYV	ADAM - valve monitoring system	
JZ	Severe accident I&C system	
JZE	Severe accident I&C system, division 1	
JZF	Severe accident I&C system, division 2	

KKS		ECS
JZG	Severe accident I&C system, division 3	
JZH	Severe accident I&C system, division 4	
K	**Reactor auxiliary systems**	
KA	Component Cooling Systems	
KAA	Component Cooling Water System safety-related	RRI1
KAB	Component Cooling Water System process-related	RRI2
KB	Coolant treatment	
KBA	Volume Control System	RCV
KBB	Coolant Supply and Storage System	TEP1
KBC	Reactor Boron and Water Makeup System	REA
KBD	Chemical Control System	RCV3
KBE	Coolant Purification System	TEP
KBF	Coolant Treatment System	TEP3
KBG	Coolant Degasification System	TEP4
KJ	Nuclear Refrigerant System	
KJM	Chilled water system for gaseous radioactive waste processing	

KKS		ECS
KL	Heating, ventilation, air-conditioning (HVAC) system in controlled areas and exclusion areas	
KLA	Containment Ventilation System	EBA/ EVF
KLB	Annulus Ventilation System	EDE
KLC	Safeguard Building Controlled-area Ventilation System	DWL
KLD	Access building and office and staff amenities building ventilation system	
KLE	Heating, ventilation and air-conditioning system (HVAC) in controlled areas and exclusion areas	DWN
KLF	Waste Treatment Auxiliary Building Ventilation System (N4)	DWQ
KLK	Sampling Activity Monitoring Systems	KRT
KLL	Fuel Building Ventilation System	DWK
KLX	Fluid supply system for control and protection equipment	SAR/ SAT
KP	Radioactive waste processing	
KPA	Solid Waste Processing System	TES1
KPB	Solid Radioactive Waste Processing System	
KPC	Radioactive Concentrates Processing System	
KPD	Filter changing equipment	TES3

KKS		ECS
KPE	Solid Waste Storage System	TES
KPF	Liquid Waste Processing System	TEU
KPK	Liquid Waste Storage System	TEK
KPL	Gaseous Waste Processing System	TEG
KQ	- blocked -	
KT	Nuclear Island Drain and Vent Systems	RPE
KTA	Nuclear island vent/drain systems - primary effluents	
KTB	Nuclear island vent/drain systems - process drains	
KTC	Nuclear island vent/drain systems - floor drains 1	
KTD	Nuclear island vent/drain systems - floor drains 2	
KTE	Nuclear island vent/drain systems - floor drains 3	
KTL	Leakage detection system on nuclear systems outside 30UJA	
KU	Nuclear Sampling System	REN
KUA	Nuclear sampling system - active liquid samples	
KUB	Nuclear sampling system - slightly active liquid samples	
KUF	Nuclear sampling system - gasous samples	

KKS		ECS
KUL	Post accident atmosphere sampling system	
L	**Steam, water, gas cycles**	
LA	Feedwater System	
LAA	Deaerator, feedwater tank	ADG
LAB	Feedwater Piping System	ARE
LAC	Feedwater Pump System	APA1
LAD	HP Feedwater Heating System	AHP
LAH	Startup and Shutdown Piping System	APD
LAJ	Startup and Shutdown Pump System	APD
LAR	Emergency Feedwater System	ASG
LAS	Emergency Feedwater Pump System	ASG
LAY	Feedwater control and protection equipment	
LB	Main Steam System	VVP
LBA	Main Steam Piping System	VVP/ VDA
LBB	Hot Reheat Piping System	
LBC	Cold Reheat Piping System	

KKS		ECS
LBG	Auxiliary Steam Piping System	SVA1
LBJ	Moisture separator/reheater	GSS
LBQ	Steam piping system for HP feedwater heating	
LBS	Steam piping system for LP feedwater heating	
LBX	Fluid supply system for control and protection equipment	
LC	Condensate System	
LCA	Main Condensate Piping System	CEX
LCB	Main Condensate Pump System	CEX
LCC	Main Condensate Heating System	
LCE	Condensate Desuperheating Spray System	
LCH	HP Heater Drains System	
LCJ	LP Heater Drains System	
LCM	Clean Drains System	
LCN	Auxiliary Steam Condensate System	SVA2
LCQ	Steam Generator Blowdown System	APG1
LCR	Standby Condensate Distribution System	

KKS		ECS
LCS	Reheater Drains System (MSR)	
LCT	Moisture Separator Drains System (MSR)	
LCW	Sealing and Cooling Drains System	
LD	Condensate polishing plant	
LDB	Condensate Mechanical Filtering System	
LDD	Electromagnetic Polishing System	
M	**Main machine sets**	
MA	Steam turbine plant	
MAA	HP steam turbine	
MAC	LP steam turbine	
MAD	Turbine bearing	
MAG	Condensing System	
MAJ	Air Removal System (steam turbine plant)	GCA
MAK	Shaft turning system	
MAL	Turbine Drains System	
MAM	Leak-off Steam System	

KKS		ECS
MAN	Turbine bypass station	GCT
MAQ	Vent system for air removal system	
MAV	Lubricant Supply System (steam turbine plant)	GGR
MAW	Gland Steam System	
MAX	Control Fluid System	
MAY	Electrical control and protection equipment	
MB	Gas turbine plant	
MBA	Turbine, compressor rotor with common casing	
MBB	Turbine casing and rotor	
MK	Generator (general)	
MKA	Generator Cooling System	GRH
MKC	Exciter set	
MKD	Generator bearing	
MKF	Stator Winding Cooling System	
MKG	Gas Cooling System	
MKQ	Exhaust gas system	

KKS		ECS
MKV	Lubricant Supply System	
MKW	Seal Oil System	
MKX	Generator waste fluid system	
MKY	Generator protection, synchronizing equipment	
MP	Common installations for main machine sets	
MPS	Drying and layup system	
MV	Lubricant Supply System	
MX	Fluid supply system for control and protection equipment	
MY	Control and protection equipment	GSE
MYA	Turbine Protection System	GSE
N	**Process energy/fluid supply for external users**	
P	**Cooling water systems**	
PA	Circulating Water System	
PAA	Circulating water screening plant	
PAB	Circulating Water system	CVF
PAC	Circulating Water Pump system	CRF

KKS		ECS
PAD	Recirculation Cooling System, Outfall Cooling System	
PAE	Cooling Tower Pump System	
PAH	Condenser Cleaning System	CTA
PAL-PAQ	- blocked -	
PAR	Make-up Water Piping System	
PAS	Make-up Water Pump System	
PAV	Lubricant Supply System	
PAY	Control and protection equipment	
PB	Circulating (main cooling) Water Treatment System	
PBB	Filtering, Mechanical Cleaning System	
PBC	Aeration, Gas Injection System	
PBN	Chemicals Supply System	
PBQ	Injection system for main fluid	
PBR	Flushing water and residues removal system, incl. neutralization	
PC	Service Water System (conventional)	
PCA	Extraction, mechanical cleaning	

KKS		ECS
PCB	Auxiliary cooling water system	
PCC	Auxiliary cooling water pump system	
PCH	Heat exchanger cleaning system	
PD	Service (secondary cooling) Water Treatment System, conventional area	
PE	Essential Service Water System	SEC
PEA	Essential service water system extraction for direct cooling	
PEB	Essential service water piping system	
PEH	Essential service water cooler cleaning system	
PF	Service (secondary cooling) Water Treatment System for secured area	
PG	Closed Cooling Water System	
PGB	Closed cooling water piping system	
PGC	Closed cooling water pump system	
PGD	Closed cooling water heat exchanger	
PH	Closed Cooling Water Treatment System for conventional area	
PJ	Closed Cooling Water System for secured area	
PK	Closed Cooling Water Treatment System for secured area	

KKS		ECS
PM	Closed Cooling Water System for transformers	
PS	Cooling Tower Blowdown System	
PU	Common equipment for cooling water systems	
PUE	Evacuation system for cooling water system	
PUR	Cathodic protection	
Q	**Auxiliary systems**	
QC	Central chemicals supply	SIR
QCA	Hydrazine storage, supply and distribution system	
QCB	Mobile hydrazine injection system for nuclear island	
QF	General control air supply	
QFB	Control air system	
QH	Auxiliary Steam Generating System	XCA
QHA	Auxiliary steam boiler	
QHB	Auxiliary boiler support structure, enclosure	
QHE	Blowdown system, flash drain system	
QHG	Boiler water circulation system	

KKS		ECS
QHX	Fluid supply system for control and protection equipment	
QHY	Control and protection equipment	
QJ	Gas distribution and storage	SG(x)
QJA	Qxygen (O2) gas distribution system	
QJB	Hydrogen Gas Supply System	SGN
QJC	Nitrogen Gas Supply System	SGH
QJD	Argon-Methane (ArCH4) gas distribution system	
QJE	Argon (Ar) gas distribution system	
QJH	Oxygen Gas Supply System	SGO
QK	Chilled water systems for conventional area	SGO/ SGN
QKA	Safety chilled water system	
QKB	Distribution lines to the HVAC system 30SAB - trains in the main control room	
QKC	Distribution lines to the HVAC system 30SAC/30KLC - cooling coils in the electrical and mechanical part of the safeguard buildings	
QKF	Safety chilled water system in fuel building	
QL	Feedwater, steam, condensate cycle of auxiliary steam generating and distribution system	

KKS		ECS
QLA	Feedwater System	
QLB	Steam System	
QLF	Common equipment for auxiliary steam generation and distribution systems	
QM	Air Humidification System	
QMA	Air humidification system QM central in 30UMA building	
QME	Air humidification system in 30UKE building	
QMK	Air humidification system in 30UKA building	
QMS	Air humidification system in 30UKS building	
QN	Operational chilled water systems	
QNA	Operational Chilled Water System	DER
QNB	Operational chilled water system for gaseous waste processing system	
QNC	Operational chilled water system in 31/34UJK building	
QND	Supply to KPL/ KBG components	
QNE	Operational chilled water system in 30UKE building	
QNJ	Supply to the HVAC system KLA - cooling coils in the reactor building (30UJA)	
QNK	Operational chilled water system in 30UKA building	

KKS		ECS
QNM	Operational chilled water system PG central in 30UMA building	
QNS	Operational chilled water system in 30UKS building	
QNX	Operational chilled water system, distribution lines to 30SAM, cooling coils in 31/32/33/34UJE building	
QNY	Operational chilled water system in 30UYA building	
QU	Sampling System water/steam cycle	REN
QUA	Sampling system for feedwater systems	
QUB	Sampling system for steam systems (*LB*)	
QUC	Sampling system for condensate systems	
QUD	Sampling systems for auxiliary steam generation systems	
QUG	Sampling systems for demineralized water systems	
QUP	Sampling systems for cooling water systems	
S	**Ancillary systems**	**DV(x)/ DW(x)**
SA	Ventilation and Air Condition Systems	
SAB	Main Control Room Air Conditioning System	DCL
SAC	Electrical division of safeguard building ventilation system	

KKS		ECS
SAD	Diesel Building Ventilation System	DVD
SAG	Smoke confinement system of nuclear island	
SAH	Ventilation system auxiliary boiler building 30UTH	
SAJ	Ventilation system anti-icing pump room 30UQX building	
SAL	Station black-out diesel building ventilation system	
SAM	Ventilation systems for turbine building	
SAQ	Ventilation system for essential service water pump building 31/32/33/34UQB	
SAU	Ventilation and heating in building 30UYA	
SAZ	HVAC systems for pipe and cable ducts	
SB	Space Heating System	SES
SBA	Space heating system SB central in 30UMA building	
SBC	Space heating system in 30UBA building	
SBE	Space heating system in 30UKE building	
SBH	Space heating system in 30UTH building	
SBK	Space heating system in 30UKA building	

KKS		ECS
SBM	Space heating system in 30UMA building	
SBQ	Space heating system in 30UQA building	
SBS	Space heating system in 30UKS building	
SBY	Space heating system in 30UYA building	
SBZ	Space heating system pipe & cable ducts	
SC	Stationary compressed air supplies	
SCA	Compressed Air Generation System	SAP
SCB	Compressed Air Distribution System	SAR
SD	Cleaning systems	
SE	Stationary Welding Gas System	
SF	Heating and Fuel Gas System	
SG	Stationary Fire Protection Systems	JP(x)/ JA(x)
SGA	Fire Water System, conventional area	JP(x)
SGB	Fire Water Distribution System inside nuclear island	JPI
SGC	Spray Deluge Systems	JPI
SGD	Spray Deluge System, nuclear area	

KKS		ECS
SGE	Sprinkler Systems	JA(x)
SGJ	Gaseous fire extinguishing systems	
SGK	Halon Fire-fighting System	
SGM	Fire protection equipment	
SGY	Fire detection and control system	
SM	Cranes, stationary hoists, lifts	DM(x)
SMA	Cranes, stationary hoists and conveying appliances	
SMB	Cranes, stationary hoists and conveying appliances	
SME	Cranes in access building 30UKE	
SMF	Cranes in fuel building	
SMJ	Cranes in reactor building	
SMK	Cranes in nuclear auxiliary building and radioactive waste processing building	
SMM	Cranes in turbine building	
SMQ	Cranes in cooling water pump building	
SMZ	Cranes in ducts	
SN	Elevators	DA(x)

KKS		ECS
SNF	Elevator in fuel building 30UFA	
SNJ	Elevator in reactor building and safeguard building	
SNK	Elevator in nuclear auxiliary building	
SNM	Elevator in turbine building	
SP	Rallway installations	
SQ	Road installations	
SR	Workshop, stores, laboratory equipment and staff amenities inside controlled area	
SRA	Hot workshop equipment	
SRC	Maintenance area in controlled area	
SRG	Hot laboratory equipment	
SRH	Health physics laboratory equipment	
SRP	Hot laundry facilities	
ST	Workshop, stores, laboratory equipment and staff amenities outside controlled area	
STG	Laboratory equipment	
U	**Structures**	
UA	Structures for grid and distribution systems	

KKS		ECS
UAA	Switchyard	
UAB	Grid system switchgear building	
UB	Structures for power transmission and auxiliary power supply	
UBA	Switchgear building	HLT
UBC-UBG	Structure for transformers	
UBH	Structure for oil collecting pits	
UBJ	Structure for transformer tracks	
UBN	Non-safety emergency power generator (TI)	
UBP	Emergency power generator	HDA - HDD
UBS	Cable shaft	
UBY	Bridge structure	
UBZ	Cable duct from 30UBA to 31UJH and 31UJK	HMG/ HLP
UC	Structures for instrumentation and control	
UE	Structures for conventional fuel supply and residues disposal	
UEJ	Structure for storage of liquid fuels	
UF	Structures for the handling of nuclear equipment	

KKS		ECS
UFA	Fuel building	BK
UG	Structures for water supply and disposal	
UGA	Structure for raw water supply	
UGC	Structure for demineralized water storage	
UGD	Structure for demineralization system	
UGG	Structure for drinking water supply	
UGJ	Structure for water supply and disposal	
UGK	Flocculant mixing chamber	
UGL	Flocculator structure, flocculator	
UGN	Treated water basin	
UGP	Sludge thickener	
UGU	Structure for effluent disposal	
UGV	Structure for sewerage plant	
UGX	Waste water buffer tank	
UGZ	Piping corridor	
UH	Structures for conventional heat generation	

KKS		ECS
UJ	Structures for nuclear heat generation	
UJA	Reactor building	HRA
UJB	Reactor building annulus	HRB
UJE	Main steam valve room	
UJF	Equipment air lock enclosure	
UJG	Gantry (reactor building)	
UJK	Safeguard building electrical	
UJH	Safeguard Building	HLF/ HLG/HLH
UJP	Reactor pit	
UK	Structures for reactor auxiliary systems	HLA
UKA	Nuclear auxiliary building	BAN
UKE	Access Building	HM
UKH	Vent stack	
UKS	Waste treatment building	BTE
UKZ	Personnel connection to existing Plant	
UL	Structures for steam-, water-, gas-cycles	

KKS		ECS
ULB	Emergency feed building	
ULD	Structure for condensate polishing plant	
UM	Structures for main machine sets	
UMA	Turbine building	HM(x)
UMY	Pipe bridge	
UN	Structure for process energy supply	
UP	Structures for circulating (cooling) water systems (e. g. circulating water intake)	
UPA	Circulating water intake rock tunnel	HP(x)
UPB	Service (secondary cooling) water intake culvert	
UPC	Circulating water intake structure	HP(x)
UPD	Service (secondary cooling) water intake structure	
UPN	Circulation (cooling) water inlet culvert	
UPP	Service (secondary cooling) water inlet culvert	
UPR	Structure for circulating (cooling) water system	
UPS	Access pit for anti-icing pipes	
UPT	Screen wash water cleaning structure	

KKS		ECS
UQ	Structures for circulating (cooling) water systems (e. g. circulating water pumps and outfall)	
UQA	Circulating water pump building	
UQB	Essential service water pump	
UQH	Debris handling building	
UQJ	Circulating water seal pit	
UQM	Service water collecting	
UQN	Circulating water outfall rock tunnel	
UQQ	Circulating water outlet structure	
UQX	Anti icing pump structure	
UQZ	Concrete channel between 30UQJ and 30UQA	
UT	Structures for auxiliary systems	
UTG	Central gas supply building	
UTH	Auxiliary boiler building	
UU	Shaft structures	
UUA	Pump shafts outside buildings	

KKS		ECS
UUB	Inspection shafts outside buildings	
UUC	Collecting shafts	
UY	General service structures	
UYA	Office and staff amenities building	
UYH	Simulator building (training facilities)	
UZ	Structures for transport, traffic, fencing, gardens and other purposes	
UZE	Track system (rails if necessary)	
UZJ	Fencing and gates	
UZR	Harbour	
UZT	Outdoor area	
X	**Heavy machinery (not main machine sets)**	
X	Heavy machinery (not main machine sets)	
XJ	Diesel engine	
XJA	Engine	
XJG	Emergency diesel low temperature water system	
XJN	Emergency diesel fuel oil system	

KKS		ECS
XJQ	Emergency diesel air intake system	
XJR	Emergency diesel exhaust gas system	
XJV	Emergency diesel lube oil system	
XJX	Emergency diesel compressed air system	
XK	Diesel generator	
XKA	Generator frame	
XKY	Electrical generator protection and synchronization	
XY	Control and protection equipment	
XYA	Emergency diesel safety I&C	
Z	**Workshop and office equipment**	
Z	Workshop and office equipment	
ZJ	Simulator	
ZJA	Full scope simulator	

PART TWO:

Acronyms

A	
ABB	Asea – Brown Boveri
ABWR	Advanced Boiling Water Reactor, a GE, Toshiba & Hitachi BWR reactor
AC	Alternating Current
ACR	Advanced Candu Reactor
ACRS	Advisory Committee on Reactor Safeguards (USA)
-	Accumulator - *Accu Accumulateur*
AEA	Atomic Energy Authority (UK)
AECL	Atomic Energy of Canada Ltd.
AFA	AREVA NP PWR Advanced Fuel Assembly
AGR	Advanced Gas-cooled Reactor (UK)
AIC	Argent Indium Cadmium (Composition of control rods)
AIF	Atomic Industrial Forum (USA)
ALARA	As Low As Reasonably Achievable
ALWR	Advanced Light Water Reactor
ANDRA	National Radioactive Waste Management Agency (France)

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AIF	Atomic Industrial Forum (USA)
ALARA	As Low As Reasonably Achievable
ALWR	Advanced Light Water Reactor
ANDRA	National Radioactive Waste Management Agency (France)
ANS	American Nuclear Society

ANSI	American National Standards Institute
AO	Axial Offset
AP	Activation Product
AP1000	The W PWR advanced 1117 to 1154 MWe nuclear power plant
ASA	American Standards Association
ASN	Nuclear Safety Authority (France)
ASME	American Society of Mechanical Engineers
ASTM	American Society for Testing ANS Materials
ATMEA	Joint venture between AREVA and MHI
ATOMPROM	Russian nuclear giant company
ATP	Authorization to proceed
ATWS	Anticipated transient without scram
ATWT	Anticipated transient without trip
AVB	Antivibration bar
AVT	All-volatile treatment

B	
BCI	Balance of Conventional Island
BDBA	Beyond Design Basis Accident
BINE	Beijing Institute of Nuclear Engineering (PRC)
BNES	British Nuclear Energy Society (UK)
BNFL	British Nuclear Fuels (UK)
BNI	Balance of Nuclear Island
BOC	Beginning of cycle - *DDC*
BOL	Beginning of life - *DDV*
BOP	Balance of plant
-	Boron Recycle System - *TEP Traitement des effluents primaires*
BOT	Build, operate, and transfer
BTU	British thermal unit
BWR	Boiling water reactor
C	
CANDU	CANadian Deuterium – Uranium reactor
CCF	Common Cause Failure - *DCC*

CCS	Carbon Capture and Sequestration
CCVS	Containment Cooling and Ventilation - *EVR*
CCWS	Component Cooling Water System - *RRI Circuit de réfrigération Intermédiaire*
CDF	Core Damage Frequencies
CE	Combustion Engineering Inc. (USA)
CFR	Code of Federal Regulations (USA)
CGNPC	China Guangdong Nuclear Power Corporation
CHP	Combined Heat and Power
CHRS	Containment Heat Removal System - *Système de refroidissement de l'enceinte*
CI	Conventional Island - *ICC Ilot conventionel*
CMF	Common Mode Failure
CNNC	China National Nuclear Corporation
CO_2	Carbon dioxide
CRDM	Control Rod Drive Mechanism - *MCDG Mécanismes de commande de grappes - RGL*
CSS	Containment Spray System - *EAS Système d'aspersion de l'enceinte*
CSVS	Purging of the Reactor Building - *EBA*
CTL	Coal to Liquids

C	
CVCS	Chemical and Volumetric Control System - *RCV Circuit de contrôle volumétrique et chimique*
CWFS	Sea Water Filtering - *CFI*
D	
DBA	Design basis accident
DBE	Design basis earthquake - *SSD*
DC	Direct Current
DIN	Deutsche Institut für Normen (Germany)
DNB	Departure from nucleate boiling
DNBR	Departure from nucleate boiling ratio
DOE	U.S. Department of Energy
E	
EBRD	European Bank for Reconstruction and Development
EBS	Emergency Boration System or Extra Borating System - *RBS Système de borication du réacteur*
EC	Eddy current
ECCS	Emergency core cooling system
EDG	Emergency Diesel Generator

EEC	Commission of the European Economic Community
EFPD	Equivalent full power days - JEPP
EFPH	Equivalent full power hours
EFQM	European Foundation for Quality Management
EFWS	Emergency Feedwater System - *ASG Système d'alimentation de secours des générateurs de vapeur*
EIA	Energy Information Administration (US DOE)
EnBW	A German electricity company
ENDESA	Empresa Nacional De Electricidad SA (Spain)
ENEL	Ente Nazionale per l'Energia ELettrica (Italy)
ENS	European Nuclear Society
EOL	End of life
EPA	US Environmental Protection Agency
EPRI	Electric Power Research Institute (USA)
ESBWR	Economic, Simplified BWR, the new GE BWR reactor
ESF	Engineered Safety Features
ESKOM	A South African electricity company
ESWS	Essential Service Water System - *SEC Circuit d'eau brute secourue*

E	
ETC	European Technical Codes
EUR	European Utilities Requirements
F	
FB	Fuel Building
FBR	Fast neutron Breeder Reactor
FDS	Fire Detection - *JTD*
FOB	Free on board
FORATOM	European Atomic Forum, *an EU body*
FP	Fission Product
FROG	AREVA Owner's Group
FSAR	Final safety analysis report
-	Fuel handling and storage system *PMC Poste de manutention du combustible ou Manutention et stockage du combustible*
FW	Feedwater
FWL	Feedwater Lines - *ARE Eau alimentaire*
FWLB	Feed Water Line Break

G	
GA	General Atomics Corp. (USA)
-	Gaseous Waste Treatment System - *TEG Traitement des effluents gazeux*
GCR	Gas-cooled reactor
GDC	General Design Criteria
GE	General Electric (USA)
GENERATION IV or GEN IV	The development of next-generation nuclear energy systems
GFR	Gas-cooled Fast Reactor
GIF	Generation IV International Forum
GNPJVC	Guangdong Nuclear Power Joint Venture Company (CRP)
GNPS	Guangdong Nuclear Power Station
GRS	Gesellschaft für Reaktor Sicherheit
GT-MHR	Gas Turbine Modular High temperature Reactor
H	
HEU	High Enriched Uranium
HHSI	High-Head Safety Injection

H	
HFE	Human Factors Engineering
HMI	Human-Machine Interface
HP	High Pressure
HPSI	High Pressure Safety Injection
HTP	AREVA NP BWR Fuel Assembly
HTR	High Temperature Reactor
HVAC	Heating, Ventilation, Air Conditioning
HWR	Heavy Water Reactor
HX	Heat exchanger
I	
I & C	Instrumentation and Control - *CC Contrôle-commande*
IAEA	International Atomic Energy Agency, *a UN body*
ICRP	International Commission on Radiological Protection - *CIPR*
ID	Inside diameter
IEA	International Energy Agency
IEEE	Institute of Electrical and Electronics Engineers

INES	International Nuclear Event Scale, *IAEA sponsored*
INPO	Institute of Nuclear Power Operators, *a USA professional association*
INSAG	International Nuclear Safety Advisory Group
IRIS	International Reactor Innovative and Secure, *a novel type of water-cooled reactor*
IRPA	International Radiation Protection Association
IRSN	Institute for Nuclear Safety and Radiation Protection (France)
IRWST	In-containment Refuelling Water Storage Tank, PTR storage tank - *Piscine RIS*
ISI	In-service inspection
ISO	International Organization for Standardization
ITER	International Thermonuclear Experimental Reactor
K	
KEPCO	Korea Electric Power Corporation (Korea)
KWU	Kraftwerke Union, Germany, *now in AREVA NP*
L	
LANPC	Ling Ao Nuclear Power Company (CRP)
LBB	Leak Before Break - *FAR*
LCO	Limiting Condition of Operation

L	
LHSI	Low-Head Safety Injection
-	Liquid Waste Treatment System - *TEU Traitement des effluents liquides*
LMFBR	Liquid Metal Fast Breeder Reactor
LNG	Liquefied Natural Gas
LNPS	Ling Ao Nuclear Power Station
LOCA	Loss-of-coolant accident
LOI	Letter of Intent
LOOP	Lost Of Offsite Power
LPSI	Low Pressure Safety Injection
LV	Low voltage
LWPS	Non-Recycled Liquid Waste Treatment - *TEU*
LWR	Light Water Reactor
M	
MFWS	Steam Generator Main Feedwater System - *ARE*
MHI	Mitsubishi Heavy Industry
MHSI	Medium-Head Safety Injection

MMI	Man-Machine Interface
MOU	Memorandum of Understanding
MOV	Motor Operated Valve
MOX	Mixed-oxide fuel assembly
MSB	Main Steam Bypass - *GCT*
MSIV	Main Steam Isolation Valve - *VIV*
MSL	Main Steam Line - *VVP Circuit vapeur*
MSLB	Main Stream Line Break
MSR	Molten Salt Reactor
MSS	Main Steam System - *VVP*
MSSV	Main Steam Safety Relief Valve
MTBF	Mean Time Between Failures
N	
NA	Not available, not attached, not applicable
NAB	Nuclear Auxiliary Building
NDE	Nondestructive examination
NDT	Nondestructive testing

N	
NDTT	Nil-ductility transition temperature
NEA	Nuclear Energy Agency (OECD)
NEI	Nuclear Energy Institute (USA)
NEPA	National Environmental Policy Act
NGNP	Next Generation Nuclear Plant, *a very high temperature gas-cooled nuclear reactor prototype*
NI	Nuclear Island - *IN Ilot nucléaire*
NNSA	National Nuclear Safety Authority (UK)
NOK	A Swiss electricity company
NPIC	Nuclear Power Institute of China
NPP	Nuclear Power Plant
NPQJVC	Nuclear Power of Qinshan JVC (CRP)
NPSH	Net Positive Suction Head
NRC	Nuclear Regulatory Commission (USA)
NSS	Nuclear Sampling System - *REN Echantillonnage Nucléaire*
NSSS	Nuclear Steam Supply System (N triple S) - *Chaudière*
NVDS	Nuclear Vents and Drains - *RPE*

O	
OBE	Operating basis earthquake
OD	Outside diameter
OPECST	Parliamentary Office for the Assessment of Scientific and Technological Choices (France)
OSART	Operational Safety Analysis Review Team, *an IAEA entity*
P	
P & ID	Piping and instrumentation diagram
PAMS	Post-Accident Monitoring System
PAS	Process Automation System
PBMR	Pebble Bed Modular Reactor, *a type of high-temperature reactor developed in South Africa*
PCI	Pellet-Clad mechanical Interaction - *IPG interaction pastille gaine*
PCRD	Program Common Research and Development, *European Commission*
PHWR	Pressurized Heavy Water Reactor
PIE	Postulated Initiating Event
PORV	Power Operated Relief Valve
PP	Primary Pump - *GMPP Groupe Moto-Pompe Primaire*
PRA	Probabilistic Safety Assessment

P	
-	Primary Coolant System - *RCP Circuit primaire de refroidissement*
PRT	Pressurizer Relief Tank - *RDP Réservoir de décharge du pressuriseur*
PRV	Pressure Relief Valve
PZR	Pressurizer - *Pressuriseur*
PSA	Probabilistic Safety Analysis, *a technique employed in nuclear safety studies - EPS*
PSAR	Preliminary safety analysis report
PV	Photovoltaic
PWR	Pressurized Water Reactor - *REP*
Q	
QA	Quality assurance
QC	Quality control
QM	Quality manual
R	
R&D	Research and Development
RAM	Random Access Memory
RAM	Reliability, Availability, Maintainability

RBMK	Reactor Bolshoi Moschmosti Kanalynyl, *a Russian reactor model* Light water graphite-moderated reactor
RCCA	Rod Cluster Control Assembly
RCCM	Rod Cluster Control Mechanism
RCS	Reactor Primary Coolant - *RCP*
RCP	Reactor coolant pump
RCPB	Reactor Coolant Pressure Boundary - *CPP*
RBWMS	Reactor Boron and Water Make-up System - *REA Circuit d'appoint en eau et bore*
REX	Experience Feedback
RHRS	Residual Heat Removal System - *RRA Système d'évacuation de la chaleur résiduelle ou Refroidissement du réacteur à l'arrêt*
RHX	Regenerative heat exchanger
RPS	Reactor Protection System - *RPR*
RPV	Reactor Pressure Vessel
RPVI	Reactor Pressure Vessel Internals
RSG	Replacement steam generator
RSK	Reaktor Sicherheit Kommission (Germany)

R	
RTE	French National Grid
RTNDT	Reference nil-ductility transition temperature
RV	Reactor Vessel
RVH	Reactor Vessel Head
RVHR	Reactor Vessel Head Replacement
RWST	Refueling Water Storage Tank
S	
SAR	Safety Analysis Report
SBO	Station Blackout
SCC	Stress Corrosion Cracking
SCWR	Super Critical Water-cooled Reactor
SFC	Single Failure Criteria - *CDU Critère de défaillance unique*
SFR	Sodium-cooled Fast Reactor
SG	Steam Generator - *GV Générateur de vapeur*
SGBS	Steam Generator Blowdown System - *APG*
SGR	Steam Generator Replacement

SGTR	Steam Generator Tube Rupture
SI	Safety injection
SIS	Safety Injection System - *RIS Système d'injection de sécurité*
SKI	The Swedish Nuclear Safety Authority
SLB	Steam Line Break
SMIRT	Structural Mechanics in Reactor Technology
-	Solid Waste Treatment System - *TES Traitement des effluents solides*
SS	Stainless steel
SSC	Systems, structures and components
SSE	Safe shutdown earthquake - *SMS Séisme Majoré de Sécurité*
SSPB	Secondary System Pressure Boundary
STUK	Sätellyturvakeskus, *a Finnish centre for radiation and nuclear safety*
SWTS	Solid Waste Treatment - *TES*
SWU	Separative Work Unit
SYSTEM 80	A Combustion Engineering, now W, PWR reactor
SYSTEM 80+	A CE/ABB/W NRC certified design, *basis for the Korean APR 1400*

T	
T/G	Turbine generator
TACIS	Technical Assistance to the Community of Independent States
T_{AVG}	Average temperature
TI	Turbine Island - *IT Ilot turbine*
TIG	Tungsten inert gas (welding)
TMI	Three Mile Island nuclear power plant (USA)
TOE	Tons of Oil Equivalent
TPL	Turn – Push – Light
TVA	Tennessee Valley Authority (USA)
TVO	Teollisuuden Voima Oy, *a Finnish nuclear operator*
U	
UCWS	Ultimate Cooling Water System
UNSCEAR	United Nation Scientific Committee on the Effects of Atomic Radiation
URD	Advanced light water Reactor Utilities Requirements Document (EPRI)
USEC	United States Enrichment Corporation (USA)
USNRC	US Nuclear Regulatory Commission

UT	Ultrasonic testing
V	
VD	Ten-yearly in-service inspection, *a process used in EDF management*
-	Vent and Drain System - *RPE Purges et évents*
VHR	Vessel Head Replacement
VHTR	Very High Temperature Reactor
VVER	Vodo Volceni Energetikesthy Reactor (Russian)
W	
W	Westinghouse
WANO	World Association of Nuclear Operators, *a professional association promoting peer review*
WEC	World Energy Council
WENRA	West Europe Nuclear Regulatory Association, *an EU body*
WNA	World Nuclear Association
WTB	Waste Treatment Building

PART THREE:
Websites

AECL	Atomic Energy of Canada Limited	http://www.aecl.ca
AEI	American Energy Independence	http://www.americanenergyindependence.com
ANS	American Nuclear Society	http://www.ans.org *The official Web site for the American Nuclear Society. It provides Society information including meeting schedules, conference information, periodical information and current events related to nuclear technologies*
AREVA	French energy group	http://www.areva.com
AREVA NC	AREVA Nuclear Cycle	http://www.areva-nc.com
AREVA NP	AREVA Nuclear Power	http://www.areva-np.com
AREVA TC	AREVA Training Center	http://www.areva-np-training.de
ASN	French safety authority	http://www.asn.gouv.fr
ATOMIC	Atomic Insights	http://www.atomicinsights.com
BERKELEY UNIVERSITY	Nuclear Engineering Department	http://www.nuc.berkeley.edu
BNFL	British Nuclear Fuel Limited	http://www.bnfl.com
CALVERT CLIFFS	Calvert Cliffs nuclear power plant	www.calvertcliffs.com

CANDU	CANDU Org	http://www.candu.org/nuclear_links.html
CASE	Clean And Safe Energy	http://www.cleansafeenergy.org
CEA	French Nuclear Energy Research Organization	http://www.cea.fr/english_portal
CNA	Canadian Nuclear Association	http://www.cna.ca
CNS/SNC	Canadian Nuclear Society	http://www.cns-snc.ca
CONSTELLATION	Constellation Energy	http://www.constellation.com/portal/site/constellation
CRYPTOME		http://eyeball-series.org/npp2/npp2-eyeball.htm *Nice US picture plants. See also Eyeballing 63 Nuclear Power Plants for overhead photos.*
DAtF	Deutsches Atomforum e.V.	http://www.atomforum.de
DOE	US Department of Energy	http://www.doe.gov
EC	European Comission	http://www.ec.europa.eu/energy
EDF	French electricity utility	http://www.edf.fr
EFN/AEPN	Environmentalists For Nuclear Energy	http://www.ecolo.org
ENERGIE-FAKTEN	Energy-Fakten (Germany)	http://www.energie-fakten.de

ENS	European Nuclear Society	http://www.euronuclear.org
EPRI	Electric Power Research Institute	http://www.epri.com
EPR-UK	EPR ™ Reactor	http://www.epr-reactor.co.uk
EURATOM	European Atomic Energy Community	http://www.euratom.org
EURONUCLEAR	Euronuclear	http://www.euronuclear.org/info/links.htm
FORATOM	European Atomic Forum	http://www.foratom.org
FORT FREEDOM	Fort Freedom	http://www.fortfreedom.org
FREEDOM FOR FISSION	Freedom for Fission	http://www.freedomforfission.org.uk
FRONTLINE NUCLEAR REACTION	Frontline Nuclear Reaction	http://www.pbs.org/wgbh/pages/frontline/shows/reaction
GA	General Atomics	http://www.ga.com
GEN IV	**The Generation IV International Forum**	**http://www.gen-4.org**
GRS	German Nuclear Safety Authority	http://www.grs.de/en/index.html
HSW	How Stuff Works - Nuclear Radiation	http://www.howstuffworks.com/nuclear.htm

IAEA	International Atomic Energy Agency	http://www.iaea.org *Official Site of the International Atomic Energy Administration. As might be expected, this well supported site is full of information and the latest in Web design features*
ICRP	International Commission on Radio-logical Protection	http://www.icrp.org
IEA	International Energy Agency	http://www.iea.org *an agency of OECD*
INF	International Nuclear Forum	http://www.climatechange.org *Nuclear Energy and Climate Change*
INSC	International Nuclear Safety Center US DOE	http://www.insc.anl.gov
IPCC	Intergovernmental Panel on Climate Change	http://www.ipcc.ch
IRPA	International Radiation Protection Association	http://irpa.net/index.html
IRSN	Nuclear Safety and Radiation Proctection Institute	www.irsn.org
ITER	International Torus Experimental Reactor	http://www.iter.org
JAEA	Japan Atomic Energy Agency	http://www.jaea.go.jp

KERNENERGIE	German Atomforum	http://www.kernenergie.de
LA RECHERCHE		www.larecherche.fr
NE	Nuclear Energy	http://www.cna.ca
NEA	OECD Nuclear Energy Agency	http://www.nea.fr
NEI	The Nuclear Energy Institute	http://www.nei.org *A US policy organization for nuclear energy. Interesting Power-Point presentations.*
NIA	Nuclear Industry Association	http://www.niauk.org
Nick's World list	A US policy organization for nuclear energy. Interesting Power-Point presentations.	http://www.calytrix.biz/radlinks
NIRS	Nuclear Information and Resource Services	http://www.nirs.org
NOVA	University of Missouri- Rolla	http://nova.nuc.umr.edu/~ans *American Nuclear Society Student Chapter. This site, one of several excellent productions by nuclear engineering students, provides a great list of frequently asked questions related to nuclear power, radiation health, and public interest*
NRC	Nuclear Regulatory Commission	http://www.nrc.gov

NUCLEAR COM		http://www.nuclear.com
NUCLEAR FAQ		http://www.nuclearfaq.ca *A privately maintained page with hard-to-find information about Canada's CANDU heavy water reactor plants*
NUCLEAR OIL	Nuclear Oil	http://www.nuclearoil.com
NUCLEAR POWER NOW	Nuclear Power Now	http://www.nuclearnow.org
NUCLEAR REACTION	Nuclear Reaction	http://www.pbs.org/wgbh/pages/frontline/shows/reaction/maps
NUCLEARINFO	Nuclearinfo.net	http://nuclearinfo.net
NUKE WORKER	NukeWorker	http://www.nukeworker.com *an indispensable resource for people in the nuclear industry*
NUPEC	Nuclear Power Engineering Corporation (Japan)	http://www.nupec.or.jp
NUSTART	A consortium for new nuclear energy development in the US	http://www.nustartenergy.com
ONP	One Nuclear Place	http://www.1nuclearplace.com
PBMR	Pebble Bed Modular Reactor	http://www.pbmr.com *The South African Pebble Bed Modular Reactor company*

RADWASTE	Radwaste - listing of government & regulatory agencies	http://www.radwaste.org/gov.htm
RSH	Radiation, Science, and Health, Inc,	http://www.radscihealth.org/rsh *The home of a non-profit organization made up of scientists and engineers dedicated to unveiling the truth about the health effects of radiation*
SCKCEN	Belgian Nuclear Research Centre for peaceful, medical and industrial applications of nuclear energy	www3.sckcen.be/sckcen_fr
SEAL	Sustainable Energy for All	www.sealnet.org
SFEN	French Nuclear Energy Society	http://www.sfen.org
SNETP	Sustainable Nuclear Energy Technology Platform	http://www.snetp.eu
SONE	Supporters of Nuclear Energy	http://www.sone.org.uk
STANDFORD UNIVERSITY	Nuclear Energy FAQs	http://www-formal.stanford.edu/jmc/progress/nuclear-faq.html
STC	Save The Climate	http://www.savetheclimate.org
TECHNO-SCIENCE		www.techno-science.net
THIS WEEK IN NUCLEAR	This Week in Nuclear	http://www.thisweekinnuclear.com

TPN	The Podcast Network	http://atomic.thepodcastnetwork.com
TVO	**Finnish Safety Authority**	**http://www.tvo.fi**
UIC	Uranium Information Center -(Australia)	http://www.uic.com.au *An interesting perspective on the nuclear energy industry. The UIC newsletter, which is archived at this site, is a particularly valuable resource*
UNISTAR	Partnership to develop the EPR™ reactor in USA	http://www.unistarnuclear.com
UNSCEAR	United Nations Scientific Committee on the Effects of Atomic Radiation	http://www.unscear.org/unscear/index.html
URENCO	Enrichment services by ultracentrifugation	http://www.urenco.com
VNT	The Virtual Nuclear Tourist	http://www.virtualnucleartourist.com *Valuable information and some pictures that help readers understand how nuclear reactors work and how nuclear energy compares to other available energy alternatives. Well worth a visit!*

W	Westinghouse	http://www.westinghousenuclear.com *a Toshiba group company*
WANO	World Association of Nuclear Operators	http://www.wano.org.uk
WEC	World Energy Council	http://www.worldenergy.org
WIN	Women in Nuclear	http://www.win-global.org
WNA	World Nuclear Association	http://www.world-nuclear.org *This organization bills itself as "the only independent, non-governmental, global organization to offer a forum for research and debate on economic, technical and political issues affecting the peaceful use of nuclear energy." The site is well organized and worth regular visits*
WNU	World Nuclear University	http://www.world-nuclear-university.org
WONUC	World Council of Nuclear Workers	http://www.wonuc.org

GLOSSARY OF NUCLEAR TERMS

A

Actinide
Element with atomic number of 90 through 103; includes uranium and plutonium.

Alpha particle
Positively charged particle emitted by various radioactive materials during decay. It consists of two neutrons and two protons and a helium atom nucleus. Alpha radiation can be stopped by a sheet of paper.

Atom
Basic component of matter. An atom is the smallest part of an element having all the chemical properties of that element. An atom consists of a nucleus (that contains protons and neutrons) and surrounding electrons.

B

Background radiation
Radiation arising from natural sources always present in the environment, including solar and cosmic radiation from outer space and naturally radioactive elements in the atmosphere, the ground, building materials, and the human body. The background radiation level is about 2.4 mSv per year.

Becquerel (Bq)
Measure of the rate of decay of a radioactive substance. One Bq is 1 disintegration per second. The human body has thousands of disintegrations due to the presence of potassium-40.

Beta particle
Negatively charged particle (an electron) emitted in radioactive decay of unstable atoms. A beta moves faster than an alpha and can be stopped by a thin piece of aluminum or a short span of air.

B

Boiling water reactor (BWR)
Nuclear reactor in which water, used as both coolant and moderator, boils in the reactor core. The steam from the boiling water is used to turn the turbine-generator.

Breeder reactor
Nuclear fission reactor that makes more usable new fuel (plutonium^{239}) than it consumes.

British thermal unit (Btu)
Quantity of heat required to raise the temperature of one pound of water by one degree Fahrenheit.

C

Canister
A corrosion-resistant container used to enclose high-level nuclear waste during long-term storage.

Ceramic
Material from which pottery, earthenware, or porcelain is made. Uranium fuel is made into a ceramic at a fuel fabrication plant.

Chain reaction
Continuing series of nuclear fission events that take place within the fuel of a nuclear reactor. Neutrons produced by a split nucleus collide with other nuclei and/or split them causing a chain of fission events.

Chemical energy
Energy released when the chemical makeup of materials changes. The energy in coal is released when the coal is burned leaving sulfur dioxide, carbon dioxide, ash, etc.

Chemical reaction
Reaction that occurs between the electrons of atoms; the reaction does not change the element itself.

Cladding
Metal tube that surrounds the nuclear reactor fuel.

Condenser
Equipment that cools steam and turns it back into water.

Containment building
Structure made of steel-reinforced concrete that houses a nuclear reactor. The containment is designed to prevent the escape of radioactive materials into the environment.

Containment
A heavy structure completely surrounding a nuclear reactor to prevent radioactivity from getting into the atmosphere in the event of a major accident.

Contamination
Unwanted pollution of working surface, devices, rooms, etc. by radioactive substances.

Control rods
Devices that can be raised and lowered in the reactor core to absorb neutrons and regulate the chain reaction. The speed of the chain reaction is controlled by control rods.

Control room
The area in a nuclear power plant where the plant operators work. The equipment in the control room tells operators what is happening in the reactor and other parts of the plant.

Conversion plant
Plant where mined uranium is converted into a gas and purified.

Coolant
Gas or liquid used in a nuclear reactor to remove the heat generated by the fission process.

Coolant/moderator
Substance used to cool the reactor and to slow neutrons. In most nuclear power plants, water is used for cooling to keep the reactor from getting too hot and to slow down neutrons so they are more likely to cause ^{235}U to fission.

Cooling tower
Structure in a nuclear power plant used to remove heat from cooling water from the condenser. The cooling tower prevents the temperature of the water in lakes and rivers from rising.

Core
Part of the nuclear reactor where the fission chain reaction takes place.

Core catcher
Core meltdown retention device.

Core meltdown
An event or sequence of events that result in the melting of part of the fuel in the reactor core.

Cosmic radiation
Radiation originating directly or indirectly from extraterrestrial sources.

C

Criticality
The condition of a nuclear reactor where a reaction initiates its own repetition.

Curie
Name for the former unit of activity.

D

Decay chain
Process that certain elements pass through to become stable.

Decommissioning
Closing a nuclear power plant after it has operated.

Deuterium
Hydrogen isotope with a nucleus containing a neutron and a proton.

Dose
Amount of radiation or energy absorbed (often measured in mSv).

E

Electron
Elementary particle with a negative electrical elementary charge.

E

Electron volt
Unit equal to the energy of one electron moving through a potential difference of one volt.

Element, chemical
Atom that has a unique number of protons in its nucleus.

Energy conversion
Process of changing one energy form into another.

Energy
Ability to do work. Energy is found in forms such as mechanical, chemical, electrical, nuclear, heat, and light.

Enrichment
Process by which the share of a certain isotope in an element is increased.

F

Fast reactor
Reactor in which the fission chain reaction is maintained mainly by fast neutrons.

Final energy
Form of energy available to the user following the conversion from primary energy carriers - crude oil, natural gas, nuclear energy, coal, regenerative energies. Final forms of energy include, among others, heating oil, fuels, gas, current, domestic heat.

Fertile
Material that becomes fissile upon absorbing a neutron.

Film badge
Piece of film worn by workers in order to see if they have been exposed to radiation.

Fissile
Material that will fission, i.e. split into two or more lighter materials, upon absorbing a neutron.

Fission
Process of splitting the nucleus of a heavy atom into two or more lighter atoms when the heavy atom absorbs a neutron. Fission also releases a large amount of energy and two or more neutrons.

Fission products
The atoms that remain when uranium is split in a nuclear reactor. Fission products are usually radioactive. E.g.: ^{85}Km, ^{90}Sr, ^{131}Cs.

Fuel assembly
Structure that often has 264 fuel rods (265 for the EPR™ reactor) containing uranium dioxide pellets. Fuel for a nuclear power plant is loaded in the reactor core in fuel assemblies.

Fuel
Material that can be converted into useful energy.

Fuel pellet
Cylindrical shape into which nuclear fuel is formed for use in a reactor.

Fuel rods
Geometrical form in which fuel pellets surrounded by cladding material are inserted into a reactor.

Fusion
Combining the nuclei of two light atoms into one heavier nucleus (a process that releases an enormous amount of energy; more energy than from fission). This requires a very high temperature.

G

Gamma radiation
High energy, high speed, and short wavelength electromagnetic radiation emitted by the radioactive decay of an unstable atom. Gamma radiation is highly penetrating and can be stopped by high density materials such as lead. X-rays are a form of gamma rays.

Gaseous diffusion plant
A facility where uranium hexafluoride gas is processed to increase its percentage of uranium 235 from about 0.7 percent to 3-4 percent.

Geiger-Müller counter
Radiation detection and measuring device. It consists of a gas-filled tube in which an electric discharge takes place when it is penetrated by ionizing radiation. The discharges are counted and signify a measurement of the radiation intensity.

Geothermal energy
Energy from heat inside the earth.

Graphite
A pure form of carbon used as a moderator in some nuclear reactors.

Gray
Basic unit of absorbed dose of ionizing radiation. 1 Gray = 1 J/kg. Symbol Gy.

H

Half-life
Time for a radioactive substance to lose half of its activity due to radioactive decay. At the end of one half-life, 50% of the original radioactive material has decayed.

H

Heavy water

Water in which the hydrogen atoms contain one neutron in their nucleus in addition to the characteristic proton.

High level waste (HLW)

Highly radioactive solid material that results from chemical reprocessing of used fuel from a nuclear fission reactor. HLW consists mainly of fission products, but also trace amounts of uranium and plutonium, plus other transuranic elements.

High-temperature gas-cooled reactor (HTGR)

Nuclear reactor cooled with helium.

I

International Nuclear Event Scale (INES)

A scale with seven levels to evaluate the events occurring in nuclear installations according to uniform international criteria.

Ion

Atom, or group of atoms, that carries a positive or negative electrical charge as a result of having gained or lost one or more electrons.

Ionizing radiation

Radiation that has enough energy to remove an electron from a struck atom, thus leaving positively charged particles (ions) behind. High enough doses of ionizing radiation can cause cellular damage.

I

Isotope
Atom of the same element but with a different mass number. Isotopes contain the same number of protons in their nucleus (hence, the same chemical properties), but different numbers of neutrons, e.g. isotopes of carbon are carbon 12, carbon 13, and carbon 14.

K

Kilowatt-hour (kWh)
Energy unit defined as 1,000 watts of electricity for one hour (equivalent to 3,413 Btu).

L

Linear hypotheses
Assumption that any radiation causes biological damage according to a straight-line graph of adverse health effects versus dose.

Liquid metal fast breeder reactor (LMFBR)
Nuclear reactor that uses a liquid metal, such as sodium, to transfer heat from the reactor to a steam generator. A breeder reactor makes more fuel than it uses by converting uranium 238 to plutonium 239.

Liquid Metal-cooled Reactor (LMR)
A nuclear reactor in which the heat is removed by a liquid metal coolant (usually sodium). Since the highly energetic neutrons created during the fission process are not slowed down very much by this relatively heavy coolant, the neutrons remain at a fairly high speed. Hence, the LMR is often called a fast reactor.

L

Low level waste (LLW)
Waste from nuclear processes containing very low amounts of radioactivity, requiring essentially no shielding or heat removal.

M

Modular High Temperature Gas Reactor (MHTGR)
A nuclear reactor in which the heat is removed by a gas (usually helium). This reactor operates at higher coolant temperatures than other systems because the coolant is already in a gaseous form (negating concerns of coolant boiling).

millisievert
Unit of radiation dosage equal to one thousandth of a sievert. The average person receives about 3.6 mSv per year from all sources.

Mill tailings
Crushed rock left after uranium has been removed from uranium ore.

Moderator
Substance that slows down neutrons so they are more likely to cause fission.

Molecule
Smallest unit into which a substance can be divided and still keep all its characteristics.

MOX
Mixed plutonium and uranium oxide fuel.

M

Monitored Retrievable Storage (MRS)
Facility in which spent nuclear fuel can be stored and monitored on a temporary basis.

N

Natural uranium
Uranium as mined (containing 0.7% ^{235}U and 99.3% ^{238}U).

Neutron
One of three basic particles in all atoms except hydrogen. Neutrons are located in the atom nucleus, are electrically neutral, and each has a mass about equal to a proton.

Nuclear chain reaction
A sequence of nuclear fissions, in the course of which the neutrons released cause further fissions, which in turn yield further neutrons causing further fissions.

Nuclear energy
Energy, usually in the form of heat or electricity, produced by the process of nuclear fission within a nuclear reactor. The coolant that removes the heat from the nuclear reactor is normally used to boil water, and the resultant steam drives steam turbines that rotate electrical generators.

Nuclear fission
cf. Fission.

Nuclear fuel
Fissionable material that can be “burned” (fissioned) in a nuclear reactor (e.g. ^{235}U).

N

Nuclear waste
Waste produced within the nuclear enterprise. This includes nuclear power plants, hospitals and medical laboratories, and numerous industrial users of nuclear products.

Nucleus
The central part of an atom that contains protons and neutrons. The number of protons uniquely defines the chemical element.

Nuclides
General term used to describe the full range of elements and their family of isotopes.

O

Once-through
Fuel cycle in which used fuel is not reprocessed.

Operating cost
Expense to keep a power plant running after it has been built, e.g. workers' salaries, repair, plant upkeep.

Operating license
Permission given by law to operate something, e.g. a nuclear power plant.

Periodic table
A chart containing all the nuclides, i.e. all elements and their family of isotopes.

Pitchblende
Ore from which uranium and radium are obtained.

Plasma
A gas so hot that all electrons are stripped away from the atoms. As such, the gas has a positive charge and can be confined in a magnetic field. High-temperature plasma is used in controlled fusion experiments.

Plutonium
Radioactive element used to produce nuclear energy.

Pressurized water reactor (PWR)
Nuclear reactor in which water is kept under pressure in a vessel to prevent boiling. Steam is made in a second vessel.

Primary energy source nuclear, hydro, solar and tidal
Energy raw materials in their natural form prior to any technical conversion, e.g. coal, lignite, mineral oil, natural gas, uranium, water, solar radiation.

Primary system
The enclosed coolant system within a nuclear reactor in which heat is directly removed from the fuel elements.

Proton
One of three basic particles in an atom. Protons are located in the atom nucleus, have a positive electrical charge, and each has a mass about equal to a neutron.

Q

Quad
A measure of energy equivalent to a quadrillion (a million times a million times one thousand, or 10^{15}) British thermal units (Btu).

R

Radiation
Particles and electromagnetic rays (waves) emitted from the center of an atom during radioactive disintegration.

Radiation dose
Radiation received during a time interval.

Radioactive
Giving off energy in the form of particles and rays by the disintegration of atomic nuclei.

Radioactive decay
Spontaneous change of an atom into a different atom or a different state of the same atom.

Radioactive isotope
Element that emits ionizing radiation when it decays. Radioactive isotopes are commonly used in science, industry, and medicine.

Radioactivity
Spontaneous emission of radiation from the unstable nucleus of an atom.

Radiography
Use of ionizing radiation to produce shadow images on a photographic film. Some of the gamma or X-rays pass through an item being evaluated while others are partially or completely absorbed by more opaque parts of the item and cast a shadow on the photographic film.

Radionuclide
Any species of an atom that is radioactive. A generic word used to replace radioisotope, which is limited to one element.

Radiotoxicity
The toxicity to human cells caused by absorption of high doses of radioactive substances. Many chemicals are toxic or poisonous at high doses.

Radium
Radioactive metallic element. Discovered by Pierre and Marie Curie in 1898, with an atomic number of 88 and atomic weight of 226. Its symbol is Ra.

Radon
Heavy, natural, radioactive gas formed by the radioactive decay of radium, a decay product of uranium. Its atomic number is 86 and its atomic weight is 222. Its symbol is Rn.

Reactor
Part of a nuclear power plant where neutrons produce a fission chain reaction.

Recycling
A term made popular in the environmental movement to reuse materials that otherwise would be discarded as waste. Within the nuclear industry, it is a synonym for reprocessing used fuel.

R

Reprocessing
The mechanical and chemical processing of spent nuclear fuel to separate useable products (i.e. uranium and plutonium) from waste materials (i.e. fission products).

Risk assessment
Science of studying the amount of risk associated with doing something.

S

Safety system
Procedure and equipment designed to keep accidents from happening or to provide corrective action.

Scintillation counter
Detector that measures the amount of ionizing radiation in materials. Used in medical and nuclear research and in looking for a radioactive material.

Secondary system
The enclosed coolant system within a nuclear power plant in which heat is removed from the primary system, through a steam generator and passed to the turbine to generate electricity.

Shielding
Material used to protect people or living things from ionizing radiation. Lead can shield from gamma rays.

Shipping cask
A container that provides appropriate shielding and structural rigidity for the transportation of radioactive material.

S

Sievert
Unit that measures the effect of radiation on the body. A calculated number based on dose and the body organ (e.g. a dose to the eye would give a different number from the same dose to the liver).

Spent (used) fuel
Nuclear fuel elements removed from a nuclear reactor after they have been used to produce power. Spent fuel has great potential for use as a fuel after reprocessing; thus “used fuel” is a more accurate term.

T

Technical Specifications (Tech Specs)
Document that records the mandated specific limits on equipment conditions and on the state of systems that a nuclear plant must maintain to keep operating safely and in accordance with NRC requirements. The Tech Specs define the limiting conditions for plant operation; these conditions ensure that the plant does not get into a situation where it poses an undue risk.

Thermal energy
Energy in the form of heat.

Thermal reactors
Nuclear reactors that employ slow (“thermal”) neutrons to sustain the fission process. Water is commonly used in such reactors as a coolant since the hydrogen contained in water is very effective in slowing down the highly energetic neutrons generated during fission.

Time, distance, shielding
The three most important factors for limiting exposure to radiation.

T

Toxicity
A measure of significant biological damage done by the absorption of a foreign substance. Many toxic materials, including radiation, are not toxic at low levels.

Transmutation
The process of changing one isotope into another using nuclear reactions.

Transuranics (TRU)
Nuclides with an atomic number greater than uranium (i.e. greater than 92). The principal transuranics are neptunium (No. 93), plutonium (No. 94), and americium (No. 95).

Tritium
Radioactive isotope of hydrogen with two neutrons and one proton in the nucleus. Tritium is manmade and heavier than deuterium. Tritium is used in industrial thickness gauges and as a label in chemical and biological experiments.

Turbine
Wheel with many blades that is spun when steam pushes the blades. A turbine converts heat energy into mechanical energy.

U

Uranium carbide
Fuel used in high-temperature gas-cooled reactors.

U

Uranium dioxide
Chemical form of uranium when it is made into fuel pellets.

Uranium enrichment
Process that increases the percentage of ^{235}U isotopes in uranium fuel from 0.7 percent to 3-4 percent.

Uranium hexafluoride
Gaseous form of uranium made from yellowcake and fluoride. The gas is made and purified at a conversion plant and shipped to a gaseous diffusion plant for enrichment.

Uranium
The heaviest element normally found in nature. The fissile isotope ^{235}U is the principal nuclear fuel material used in today's nuclear power reactors. Uranium is a hard, shiny, metallic radioactive element. Its atomic number is 92, its atomic weight is 238, and its symbol is U.

Used fuel
cf. spent fuel.

V

Vitrification
The process of placing nuclear high level waste (HLW) into a glass form for long-term disposal.

X

X-rays
Electromagnetic radiation with more energy than visual light, usually produced by an X-ray machine. Physically, X-rays and gamma rays are similar.

Y

Yellowcake
Yellow powder that is mostly uranium. Yellowcake is produced by pouring crushed uranium ore into an acid that dissolves uranium. The acid is drained from the crushed ore and dried, leaving a yellow powder called yellowcake.

Z

Zircaloy
Zirconium alloy based on zircon and tin used as a material for fuel rod cladding.

The terms in this glossary are taken from a variety of sources (ANS, DOE, ENS, IAEA, NRC, etc.).

Notes

Notes

For any question, comment, remark, send an e-mail to the following address: nucleopedia@areva.com

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