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SSI report

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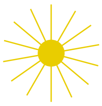
Inquiry into the radiological consequences of power uprates at light-water rectors worldwide

Tea Bilic Zabric, Bojan Tomic, Klas Lundgren and Mats Sjöberg

Statens strålskyddsinstitut
Swedish Radiation Protection Authority

SSI's Activity Symbols

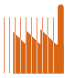


Ultraviolet, solar and optical radiation
Ultraviolet radiation from the sun and solariums can result in both long-term and short-term effects. Other types of optical radiation, primarily from lasers, can also be hazardous. SSI provides guidance and information.

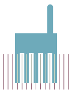
Solariums
The risk of tanning in a solarium are probably the same as tanning in natural sunlight. Therefore SSI's regulations also provide advice for people tanning in solariums.

Radon
The largest contribution to the total radiation dose to the Swedish population comes from indoor air. SSI works with risk assessments, measurement techniques and advises other authorities.

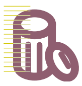
Health care
The second largest contribution to the total radiation dose to the Swedish population comes from health care. SSI is working to reduce the radiation dose to employees and patients through its regulations and its inspection activities.



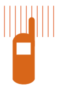
Radiation in industry and research
According to the Radiation Protection Act, a licence is required to conduct activities involving ionising radiation. SSI promulgates regulations and checks compliance with these regulations, conducts inspections and investigations and can stop hazardous activities.



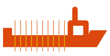
Nuclear power
SSI requires that nuclear power plants should have adequate radiation protection for the generalpublic, employees and the environment. SSI also checks compliance with these requirements on a continuous basis.



Waste
SSI works to ensure that all radioactive waste is managed in a manner that is safe from the standpoint of radiation protection.



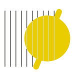
Mobile telephony
Mobile telephones and base stations emit electromagnetic fields. SSI is monitoring developments and research in mobile telephony and associated health risks.



Transport
SSI is involved in work in Sweden and abroad to ensure the safe transportation of radioactive substances used in the health care sector, industrial radiation sources and spent nuclear fuel.

Environment
"A safe radiation environment" is one of the 15 environmental quality objectives that the Swedish parliament has decided must be met in order to achieve an ecologically sustainable development in society. SSI is responsible for ensuring that this objective is reached.

Biofuel
Biofuel from trees, which contains, for example from the Chernobyl accident, is an issue where SSI is currently conducting research and formulating regulations.



Cosmic radiation
Airline flight crews can be exposed to high levels of cosmic radiation. SSI participates in joint international projects to identify the occupational exposure within this job category.

Electromagnetic fields
SSI is working on the risks associated with electromagnetic fields and adopts countermeasures when risks are identified.

Emergency preparedness
SSI maintains a round-the-clock emergency response organisation to protect people and the environment from the consequences of nuclear accidents and other radiation-related accidents.

SSI Education
is charged with providing a wide range of education in the field of radiation protection. Its courses are financed by students' fees.

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TITLE / TITEL: Inquiry into the radiological consequences of power uprates at light-water rectors worldwide. / Utredning av radiologiska konsekvenser i samband med effekthöjningar i lättvattenreaktorer världen över.

DEPARTMENT / AVDELNING: Department of Occupational and Medical Exposures / Avdelning för för personal- och patientstrålskydd.

SUMMARY: In Sweden, most of the nuclear power plants are planning power uprates within the next few years. The Department of Occupational and Medical Exposures at the Swedish Radiation Protection Agency, SSI, has initiated a research project to investigate the radiological implications of power uprates on light-water reactors throughout the world.

The project was divided into three tasks:

1. A compilation of power uprates of light-water reactors worldwide. The compi-lation contains a technical description in brief of how the power uprates were carried out.
2. An analysis of the radiological consequences at four selected Nuclear Power Plants, which was the main objective of the inquiry. Affects on the radiological and chemical situation due to the changed situation were discussed.
3. Review of technical and organisational factors to be considered in uprate projects to keep exposures ALARA.

The project was carried out, starting with the collecting of information on the imple-mented and planned uprates on reactors internationally. The information was catalogued in accordance with criteria focusing on radiological impact. A detailed analysis followed of four plants selected for uprates chosen according to established criteria, in line with the project requirements. The selected plants were Olkiluoto 1 and 2, Cofrentes, Asco and Tihange. The plants were selected with design and operation conditions close to the Swedish plants. All information was compiled to identify good and bad practices that are impacting on the occupational exposure. Important factors were discussed concerning BWRs and PWRs which affect radiation levels and occupational exposures in general, and especially at power uprates.

Conclusions related to each task are in detail presented in a particular chapter of the report. Taking into account the whole project and its main objective the following conclu-sions are considered to be emphasized

Optimisation of the work processes to limit the duration of the time spent in the controlled areas is especially important. Leadership, composition and organization of the large demanding tasks are critical for successful implementation of power uprate and keeping received doses at a minimum. Good planning and preparation, which reflects experience from similar projects elsewhere, adherence to procedures and supervision from plant personnel as well as consequential application of ALARA principles and good practices are important factors.

It has not been found a direct relationship between the uprates and the occupational expo-sures. The occupational doses on some plants seem to be higher after the uprate, while on others seem to be lower. However the general trend in light-water reactors worldwide is gradually reduced occupational exposures.

There is no obvious correlation of the power uprate and fuel failures. However, perform-ance of fuel for PWRs and BWRs went in opposing directions, improving for PWRs and deteriorating for BWRs.

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For BWRs investment in the condensate cleanup efficiency results in favourable water chemistry conditions that can be maintained, or even improved, after the power uprate. The higher steam velocity after a power uprate can increase the radiation levels around main steam lines and other turbine components due to a considerable increase in steam moisture content. This problem can be overcome with a recent design and installation of new steam dryers in the reactor pressure vessel to reduce steam moisture.

Issues of relevance for PWRs include: Increase in the rate of production of H-3 due to higher boron concentration and power level, especially for longer fuel cycles; Control of pH and Lithium as an essential means of controlling the corrosion level and thus radiation levels. Fuel related corrosion problems are shown to be less visible with good pH control and shorter fuel cycles.

SAMMANFATTNING: I Sverige pågår eller planeras effekthöjningsprojekt i de flesta av kärnkraftverken. Statens strålskyddsinstituts avdelning för personal- och patientstrålskydd har initierat ett forskningsprojekt för att samla in information om effekthöjningar i lättvattenreaktorer världen över med fokus på de radiologiska konsekvenserna som effekthöjningen innebär.

Projektet delades in i tre delprojekt:

1. Insamling av fakta om reaktorer som höjt eller står i begrepp att höja effekten med en kortfattad teknisk beskrivning om effekthöjningen.
2. Analys av de radiologiska konsekvenserna på fyra utvalda kärnkraftverk, vilket var det egentliga huvudsyftet med projektet. Påverkan på den radiologiska och kemiska situationen pga. den ändrade situationen diskuterades.
3. Genomgång av tekniska och organisatoriska faktorer som bör beaktas vid en ef-fekthöjning för att hålla stråldoser så låga som rimligen är möjligt (ALARA).

Projektet genomfördes med att initialt samla in internationell information om genomförda och planerade effekthöjningar. Informationen katalogiserades i enlighet med kriterier som fokuserade på radiologisk påverkan. En detaljerad analys genomfördes av fyra utvalda kärnkraftverk valda enligt fastställda kriterier som uppfyllde projektets anspråk. De valda kraftverken var Olkiluoto 1 and 2, Cofrentes, Asco and Tihange. Dessa kraftverk valdes med hänsyn till att konstruktion och driftbetingelser ligger nära de svenska kraftverken. Information samlades in för att identifiera “good and bad practices” som påverkar perso-nalens stråldoser. Viktiga faktorer som diskuterades berör strålnivåer och personalens stråldoser generellt och vid effekthöjningar i synnerhet.

Varje delprojekts slutsatser är presenterade i delprojektets kapitlet. Vi vill dock med hän-syn till projektet och dess mål framhålla följande slutsatser:

Optimering av arbetsprocesser för att begränsa tiden på kontrollerat område är av stor vikt. Ledarskap, sammansättning och organisation av stora utmanande projekt är kritiska faktorer för att genomföra en effekthöjning med lyckat resultat. God planering och förbe-redelse som tar hänsyn till liknande projekt vid andra anläggningar, användning av in-struktioner och att kraftverkets egen personal utför en noggrann övervakning av arbeten samt införande av ALARA principer och användandet av goda förebilder är viktiga fakto-rer.

Någon direkt relation mellan effekthöjning och personalens stråldoser har inte kunnat påvisas. Personalens stråldoser på några kraftverk verkar öka medan på andra sjunker doserna. En generell trend i lättvattenvattenreaktorer världen över är gradvis minskade doser till personalen.

Det har inte kunnat påvisas någon tydlig korrelation mellan effekthöjning och bränsleska-dor. Emellertid har bränsleskadeutvecklingen för PWR och BWR gått motsatta vägar. För PWR har den förbättrats men för BWR har den försämrats.

För BWR har det visat sig att investeringar i att effektivisera kondensatreningssystemet har resulterat i lika bra eller t.o.m. bättre kemiförhållande. Den högre ånghastigheten efter en effekthöjning kan resultera i högre strålnivåer runt ångledningarna och turbinkompo-nenter pga. ökad fukthalt i ångan. Detta problem kan åtgärdas genom installation av en modern konstruktion av ångseparator i reaktortanken.

I tryckvattenreaktorer ökar H-3 produktionen genom högre effektnivå och ökad borkon-centration, speciellt med långa driftcykler. Kontroll av pH- och litiumnivå är väsentliga hjälpmedel för kontroll av korrosion och därigenom strålnivån. Problem med bränslet kan undvikas genom god kontroll av pH och begränsning av bränslecykellängden.

esKONSULT ALARA Engineering ENCO

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List of abbreviations

AB	Auxiliary building
ABB	ASEA merged with BBC
ACEC	Alstom ACEC Energie
ACLF	Grouping of ACEC/C-L//Framatome/WNE
AEE	Atomenergoexport
AFW	Auxiliary Feedwater
ALARA	As Low As Reasonably Achievable
AO	Axial Offset
AOA	Axial Offset Anomaly
ASEA	Allmänna Svenska Elektriska Aktiebolaget (General Swedish Electrical Limited Company)
BBC	Brown Boverie et Cie
BOC	Beginning Of Cycle
BRAC	BWR Radiation Assessment and Control
BWR	Boiling Water Reactor
°C	Degrees Centigrade
CCU	Condensate Clean-Up
CE	Combustion Engineering
CILC	Crud Induced Local Corrosion
C-L	Creusot Loire
CMI	Cockerill Mechanical Industry
CNC	Confrentes nuclear power plant
CRD	Control Rod Drives
CVCS	Chemical Volume Control System
DB	Deep Bed
DG	Diesel Generator
DH	Dissolved Hydrogen
DO	Dissolved Oxygen
dP	Delta Pressure
DR	Dose Rate
E	Extended power
EPU	Extended Power Uprate
DZO	Depleted Zinc Oxide
EBA	Enriched Boron Acid
EC	European Commission
ECP	Electrochemical (or corrosion) Potential
EFPH	Effective Full Power Hour
EOC	End Of Cycle
EPRI	Electric Power Research Institute (www.epri.com)
°F	Degrees Fahrenheit
F + DB	Filter + Deep Bed
FD	Filter Demineralizer
FPHD	Forward Pumped Heater Drains

FRAM	Framatome
FRAMACECO	Framatome-ACEC-CO
FW	Feedwater
GBq	Giga Becquerel
GE	General Electric
Gpm	Gallons per minute
HP	High Pressure
HWC	Hydrogen Water Chemistry in BWRs with injection of hydrogen in order to reduce the risk of environmental assisted cracking
HWC-M	Moderate HWC
IAEA	International Atomic Energy Agency
ICRP	International Commission on Radiological Protection
IGSCC	Intergranular Stress Corrosion Cracking
ISOE	Information System on Occupational Exposure
KWU	Kraftwerk Union
LEFM	Ultrasonic feedwater flow measuring system
LP	Low Pressure
MP	Measuring Position for dose rate
MSL	Main Steam Line
MSLR	Main Steam Line Radiation
MU	Measurument Uncertainty
MUR	Measurument Uncertainty Recapture
MWt	MW thermal power
NM	Noble Metal
NMCA	Noble Metal Chemical Addition
NPP	Nuclear Power Plant
NRC	Nuclear Regulatory Commission
NSSS	Nuclear steam supply system
NWC	Normal Water Chemistry in BWRs without injection of hydrogen
O&M	Operation and Maintenace
OECD	Organisation for Economic Cooperation and Development
OL1	Olkiluoto-1
OL2	Olkiluoto-2
OLNC	On Line NobleChem
PCI	Pellet Cladding Interaction
PCMI	Pellet Cladding Mechanical Interaction
PCS	Power Conversion System
pH300	pH at 300 □C
PI	Performance Indicators
PLR	Primary Loop Recirculation (i.e. recirculation lines)
PS	Pressure Suppression
PWR	Pressurized Water Reactor
R2	Ringhals 2
R3	Ringhals 3
R4	Ringhals 4
RB	Reactor Building

RCS	Reactor coolant system
RFO	Refuelling Outage
RHR	Residual Heat Removal
Rpm	Rounds per minute
RPV	Reactor Pressure Vessel
RTD	Reactor Temperature
RW	Reactor Water
RWCU	Reactor Water Clean-Up
Rx	Reactor
S	Stretch power
SG	Steam Generator
SGR	Steam Generator Replacement
SHE	Standard Hydrogen Electrode
SPF	Spent Fuel Pit
SPU	Stretch Power Uprate
SS	Stainless Steel
SSI	Swedish Radiation Protection Authority
STP	Standard Temperature and Pressure
T Ave	Average temperature
$T_{½}$	Half-life for radionuclide
TMI	Three Mile Island
TVO	Teollisuuden Voima Oy
WANO	World Association of Nuclear Operators (www.wano.org.uk)
WEC	Westinghouse
WNE	Westinghouse Nuclear
XO	Extra Outage

1. Introduction

Most nuclear power plants in Sweden are planning power uprates within the next few years. Permission to increase power is given by the Swedish Government. The Swedish Radiation Protection Authority, SSI, is one of the bodies to which the application for power uprates is referred for consideration. The Department of Occupational and Medical Exposures at SSI has initiated an inquiry to consider the radiological implications of thermal power uprates on light-water reactors throughout the world. The information gained from the research will be firstly used as a reference and background information source and then toreview the the applications for power uprates and in the assessment of the after-effects of these uprates.

Available information shows that a relatively high percentage of all operating NPPs in the world have implemented, or are considering, some form of power increase (power uprate). Such uprates vary significantly. Small uprates of a few percent of the plant's power may be achieved by modification of the power conversion system and/or adjustments to control systems. Conversley large uprates, sometimes in excess of 30% nominal power, may be undertaken which require substantial changes on the reactor side, including fuel, operating regime and limits, etc.

The majority of power uprates are in the middle range (between 5 and 10 % of nominal power) and typically involve changes to both reactor and power conversion system (PCS). However, all power uprates require either major or minor modification to operating practices and conditions. The radiological doses to personnel are related to these upgrades, both during normal operation and during outages, whilst also being sensitive to differing materials and operating regimes. Integral doses could often be found from WANO indicators and other sources of information. However, these had not been systematically analysed to determin which specific features of the uprates were influencing radiological doses.

2. Project description

2.1 Project overview

The aim of the inquiry was to investigate what specific conditions and practices affect the operational doses received when reactor power is uprated. Identification of these factors on a worldwide basis should then allow power uprates to be planned in way that provides better exposure optimisation.

The inquiry was divided into three tasks:

1. A compilation of power uprates of light-water reactors worldwide. The compilation contains a technical description in brief of how the power uprates were carried out.
2. The main emphasis of the inquiry was an analysis of the radiological consequences at four selected Nuclear Power Plants. Affects on the radiological situation due to the changed situation was discussed by checking areas of special interest, such as
   - degradation of material resulting in more repair work,
   - verification of safety and security resulting in more testing and
   - work performed in controlled areas in relation to the uprate.
3. Experience from the reconstruction period with bearing on the radiation protection of workers

This report is a compilation of all three tasks. Each task has its own chapter and for task 2 the analysis of the selected plants, are shown in three different subchapters. Task 3 is divided into two subchapters where the technical factors to control radiation fields are discussed in one and the organisational issues in the second.

2.2 Working method

This inquiry was implemented with three specific elements (tasks), starting with the collecting of information on the implemented and planned uprates on PWR and BWR reactors internationally. The information was catalogued in accordance with criteria focusing on radiological impact. A detailed analysis of plants selected for uprates, were chosen according to established criteria, in line with the project requirements. For BWR were two plants selected, one with 12% power increase and another with 25%. For the PWRs two uprates in the range of 10% power increase were selected. The plants were selected with design and operation conditions close to the Swedish plants. The aim was a detailed analysis of causal relations between uprates and the radiological content, thus the project was organized into three tasks that were implemented one after another.

Data collection for the detailed analysis was carried out through personal contacts. Data was specified which was specific for the particular type of reactor and sent to the contact persons. There was a good response for the data sought for the BWR's. For the PWRs detailed radiological data was received but less technical and chemistry. The PWR Task 2 report is therefore, not the detailed analysis we hoped to make. Though a detailed analysis on radiological data has been performed.

2.3 Participants

Three experienced companies carried out the inquiry. The experts involved in the project were:

Tea Bilic Zabric and Bojan Tomic from Enconet Consulting, Vienna

Klas Lundgren from ALARA Enginnering, Skultuna

Mats Sjöberg ES-konsult, Solna

3. Compilation of power uprates

3.1 Introduction

Quite a few NPPs have implemented, or are considering increasing the power level on which they operate (power uprate). Those power uprates vary significantly, from small ones of a few percentages to large uprates above 30% of the nominal power. The alterations to the plants particularly those with larger power uprates, require changes and modifications to the plant, in both hardware and operating arrangements (procedures, operating regime, and operating window). Apart from different requirements (on power conversion system) fuel remains an important (and also a limiting) issue during an uprate.

In addition to the safety impact of uprates that is normally verified in depth before a licensee for an uprate is issued the radiological aspects, the consequences of an uprate are also of interest. Changes in operating regimes, but also changes and operation with new hardware might have an impact on the operational doses and their distribution. Increased power level may also have certain impact on the effluents, especially on tritium.

The first task was to compile a database using worldwide sources. This database will be accompanied by a discussion of the sources used and the initial conclusions that could be drawn from the data.

3.2 Data Collection and Sources

The data collected within this task relates to the worldwide uprates performed (or planned) and which shows the radiological consequences and compares the data before and after the uprates. The data was collected from the literature sources; including a variety of databases, regulatory filings, analyses and other available information.

The following sources were used:

WANO Performance indicators

WANO maintains five programmes for information exchange, promoting mutual communications and benchmarking. Two of them: ‘Exchange of Operating Experience’ and ‘Performance Indicators’ – a series of standardised parameters for the comparison of power plants, were reviewed for data collection within this project. The data, readily available from the WANO performance indicators database is more general and cannot be used to determine doses in e.g. outages. The data is available from 1992. In some samples for multi unit sites the doses in WANO indicators database are just a fraction (1/3 or 1/2) of the plant’s total value. The WANO database was used for the initial review of occupational doses and to support some generic conclusions.

OECD ISOE data base

The ISOE Programme is the world’s largest occupational exposure database, established by a network of radiation protection experts from operators and regulators. The ISOE data serves as a point for the exchange of information and experience but it is also used to support analysis. ISOE structure supports the collection of doses during outages with specific doses for particular activities. The ISOE database distinguishes plant personnel and subcontractors. ISOE database was established in 1992, but the data on collective doses start in 1977.

The ISOE database was used to extract the information on occupational doses during outages and during normal operation (annual doses).

Nuclear Engineering

The World Nuclear Industry Handbook is a reference guide to the nuclear power industry. It is updated each year. Among other information the Handbook contains information on power reactors, a country-by-country summary of reactors showing type, status, location, main contractors and key dates; main data on each unit including technical detail on core, vessel containment, fuel, coolant, moderator, control, fuelling, operating strategy, turbine and more.

IAEA

The IAEA is a leading publisher in the nuclear field. It's scientific and technical publications cover fifteen subject areas. They include the proceedings of major international conferences, as well as international guides, codes, standards, reports, documents and conventions. IAEA PRIS data base and publication was used for reviewing information and proceedings from conferences.

NRC

The NRC's REIRS system provides the latest available information on radiation exposure to the workforce at certain NRC licensed facilities. REIRS contains several data bases that record the radiation exposure information. We used 'Effluent Database for Nuclear Power Plants', which was developed to track annual aqueous and atmospheric effluent release data and offsite doses calculated for each nuclear power plant in the United States. The data is available from year 1998. The OECD developed the document 'Thermal Power Uprating in Europe' which reflects the cooperation of many experts in Europe.. In addition to the uprating data, also included was some 'Plant data' and general information available from relevant countries, mostly based on the "IAEA PRIS" data.

EC

The Commission periodically publishes reports on releases to the environment of radioactive substances in airborne and liquid effluents from Nuclear Power Stations and Nuclear Fuel Reprocessing Sites in the European Union. These reports cover discharges from Nuclear Power Stations of capacity greater than 50 MWe as well as from (former) Nuclear Fuel Reprocessing Sites.

Comments on the comparison of data across the sources of data

There is no systematic collection of data covering uprates and related activities, nor there is any specific collection of radiological exposure information. Therefore, the information of relevance for radiological impact of uprates was collected from a combination of sources. While some of the sources were traditional ones, in others the data collection started only recently. Because of that, a meaningful comparison in relation with earlier uprates is not possible. The main source of data that was used for radiological releases cover the years after 1995. Therefore, it was not possible to assess the effects of earlier uprates. For the occupational exposures, the main source of data was the ISOE data base.

To assure the correctness and to be able to corroborate the ISOE data, a comparison with the WANO data base entries for analysed plants was undertaken.

The table below compares ISOE and WANO data for the average recorded Occupational doses during outages (for a period before and after an uprate) with the recorded Occupational dose for the outage during which an uprate was implemented. The assessment was made for several plants that are comparable in their characteristics.

As can be seen from the data in the table, significant differences are visible in some cases between ISOE and WANO data. Even after evaluating the reporting requirements, the explanation for those differences could not be found.

To assure the consistency of any analysis within this project, a decision was taken to exclusively use the ISOE data as a figure of merit for the occupations exposures during normal operation and outages. The ISOE Programme is the world's largest collection of information on occupational exposure. The ISOE data collection is structured in a way to relate doses during outages with specific activities undertaken. Moreover the ISOE data separate plant's personnel and subcontractors, thus allowing for a comparison of plants that use external support differently.

Examples	Occupational dose in operating year				Dose during uprate (annual doses in year uprate was implemented)	
	Before the uprate (manSv)		After the uprate (manSv)		manSv	
	ISOE data	WANO data	ISOE data	WANO data	ISOE data	WANO data
Plant1	1.202	1.14	0.879	0.83	3.30	1.49
Unit1/Plant2	1.282	0.99	0.677	0.78	3.22	2.03
Unit2/Plant2	1.432	0.99	0.986	0.78	1.41	2.03
Plant3	0.488	0.5	0.752	0.83	1.54	0.90

3.3 Content of the data base

Keeping in mind the overall objective of the project, the criteria for cataloguing the information to be collected was established. These reflected the knowledge of elements that are impacting on the radiological doses, and that could be related to the uprate activities. The printout of the Data base containing all information collected is provided in the Appendix 1.

The table below describes the fields that are included in the data base and discusses the contents of each of those.

Field #	Title	Description/comments	Main Reference
0	Country	Country where plant is located	
1	Plant name	Plant name with unit indication (if more uprates were performed, year of the uprate follow the unit designator, i.e. Thiange 2/2001)	WANO PI
2.1	Vendor of NSSS	Vendor (GE,WEC, FRAM, etc)	WANO PI
2.2	Commercial data of operation	Month, Day, Year,	WANO PI
3.1	Reactor type	(PWR, BWR)	OECD, Nuclear Engineering
3.2	Initial power	Original thermal power in MWt	OECD, Nuclear Engineering
4.1	Thermal power	Uprated thermal power in MWt	OECD, IAEA, NRC
4.2	Year implemented	Year when the uprate was implemented (in some cases approved, data sometimes inconsistent)	OECD, NRC
4.3	Uprate type & total power increase in the percentage for particular year	-MU -Measurement uncertainty (uprates are less than 2 %) -S -Stretch power (uprates typically up to 7) -E - Extended power (uprates greater than the stretch) Example: 1985 S 4.1% 2001 E 19.4% The first uprate was in 1985 (4.1%) and the second in 2001 (15.4). The % always indicate the total power increase compared with original design (thus 19.4% in second)	OECD, NRC
4.4	Technical solution	Technical solution for an uprate. Example: Whether the uprate was implemented by increase of Rx T Avg (with the same FW mass flow) OR T Avg remains same (but FW/MSL mass flow and pressures were increased) Data entered where available	OECD, NRC
4.5	Equipment	Where available, list of main equipment modified/affected	OECD, NRC

Field #	Title	Description/comments	Main Reference
5.1	Fuel cycle	Length of the fuel cycle - in the months	Nuclear Engineering
5.2	Average linear fuel rating (before uprate)	Fuel rating - in kW/m	Nuclear Engineering
5.3	Average linear fuel rating (after uprate)	Change in fuel rating – kW/m	Nuclear Engineering
5.4	Fuel type	Type of fuel used. Sometimes different types used, depends on core design. Equilibrium cycle fuel entered (when available).	Nuclear Engineering
6.1	Annual liquid effluents (before uprate)	Total Liquid release in GBq. (Data from EC used, available from 1995. For US, data available from 1998)	EC
6.2	Annual liquid effluents (after uprate)	As above	EC
7.1	Annual gaseous effluents (before uprate)	Total Airborne releases in GBq	EC
7.2	Annual gaseous effluents (after uprate)	As above	EC
8.1	Annual occupational dose (before uprate)	Average value of the total collective dose over the three-year period before the uprate in manSv (when three years not available, it is noted in table)	ISOE, NRC
8.2	Annual occupational dose (after uprate)	Average value of the total collective dose over the three-year period after the uprate in manSv (when three years not available, it is noted in table)	ISOE, NRC
9.1	Occupational dose during outage (before uprate)	Average value of the collective outage dose over the three-year period before the uprate - in manSv (when three years not available, it is noted in table)	ISOE
9.2	Occupational dose during outage (after uprate)	Average value of the collective outage dose over the three-year period before the uprate - in manSv (when three years not available, it is noted in table)	ISOE
10	Occupational dose during uprate	Doses received during the uprate (if equipment was changed) – in manSv. Data not systematically available. In some cases data cover annual occupational dose for the year when the uprate was implemented. Of no relevance for MU. Of limited relevance for S, except when major plant modifications implemented. Sometimes involve SGR, if performed in parallel	ISOE

Classification of uprates

There are considerable economic benefits to uprates because they allow more value (energy) to be generated by the existing plant. Whilst the fuel costs may marginally rise the remaining costs do not increase. This makes uprates highly attractive to the utilities. Nevertheless, the complexity and significance of the safety and operational issues associated with uprates make additional gains anything but easy. Comprehensive safety analysis and, depending on the country, re-licensing by the regulator are important elements of every uprate project. Changes in operating practices and methods of maintenance organisation following uprates lead to radiological consequences both in normal operation and during outages.

The aim of every uprate is to increase the electrical power output available from the main generator. This can be achieved by modifying the power conversion system (e.g. turbine, generator, associate equipment) and/or by increasing the reactor energy output.

The development of technology of, in particular, turbines in the last decade is such that many plants increased the power by installing new turbines or parts of them, and achieved the power increase of up to 3%. As this project is focused on radiological issues, the increase of the generated energy through modifications on the PCS does not introduce any effects of interest.

The second way to increase generation is to increase the power of the reactor. Typically there are three distinctive categories of power increase (although in the first category, MU, the reactor power is, physically, not increased), as follows:

Measurement Uncertainty (MU) Recapture Power Uprate: uprates of 1 to 2 percent power, typically achieved using more precise techniques for measuring Feedwater flow and/or performing analysis to reduce unnecessary conservatism

Stretch Power (S) Uprate: uprates of 5 to 7 percent power, typically achieved by changing instrumentation set points, re-analysis (to recover excessive margins) together with a small number of major plant modifications

Extended Power (E) Uprate: uprates of up to 20 percent power, achieved by major changes of core design and significant modifications to major plant equipment

The majority of power uprates implemented or planned are in the middle range of between 5 and 10 % nominal power. These typically involve changes to both the reactor and the power conversion system. Some plants have performed different types of uprate on two or more occasions.

Improved measurement and analysis techniques have allowed utilities to increase the licensed power limits of existing plants as a cost-effective method of increasing power. Currently, 76 PWR units (19 Europe and 57 USA) and 45 BWR units (11 Europe, 32 USA and 2 Mexico) have uprated thermal power. Of these, 24% were small uprates of up to 2% increase, 49% were stretch and 27% were large power uprates.

A number of European and Asian utilities are planning to implement power uprates within the next few years:

Dukovany 1-4 (PWR) is planing to increase power for 2.3% in the years 2005-2008 with installation of new turbine blades. It plans a 10% uprate at a later date.

Brokdorf (PWR) is planning 3.9% uprate (date not known)

Emsland (PWR) is planning 4.9% uprate (date not known)

Grafenrheinfeld (PWR) is planning 4.9% uprate (date not known)

Grohnde (PWR) is planning 4.5% uprate (date not known)

Gundremmingen B and C (BWR) are planning 6.8% uprate (date not known)

Isar-1 (BWR) is planning 7% uprate (date not known)

PAKS 1, 2, 4 (PWR) are planning 9.1% uprate in 2006

PAKS 3 (PWR) is planning 9.1% uprate in 2007

Kori 3 & 4 (PWR) are planning 5% uprate in 2006

YGN 1 & 2 (PWR) are planning 5% uprate in 2006

Higasidory (ABWR) is planning uprate (% and date not known)

Shika (BWR) is planning uprate (% and date not known)

Forsmark-1 (BWR) is planning 19.9% uprate in 2010

Forsmark-2 (BWR) is planning 19.9% uprate in 2009

Forsmark-3 (BWR) is planning 25.0% uprate in 2011

Oskarshamn-3 (BWR) is planning 29.8% uprate in 2008

Ringhals-1 (BWR) is planning 11.9% power uprate in not known

Ringhals-3 (PWR) is planning 7.8% uprate in year 2007 and 13.5% uprate in 2007

Ringhals -4 (PWR) is planning 13.5% uprate in 2011

The following table describing intended future power uprates in the USA is based on information obtained from a survey of all licensees conducted in March 2006.

Fiscal Year	Power Uprates Expected	MUR	SPU	EPU	MWt
2006	4	1	0	3	1470
2007	6	5	1	0	431
2008	0	0	0	0	0
2009	10	2	3	5	1792
2010	2	2	0	0	76
2011	1	1	0	0	26

3.4 Conclusions

Some conclusions are raised from the review of the data collected. Issues of interest are discussed in section below.

Relevance of yearly occupational doses:

The data table provided the annual occupational doses for (usually) 3 years average before and after the uprate has been implemented. While this provides (some) insights related with occupation doses, the annual occupational doses are often driven by processes and activities that have nothing to do with uprate, rather with specific repairs and interventions during outages and/or some specific operational issues (i.e. unusual leaks, change of chemistry, material used, etc). In some case, unusual events might add to the collective annual doses. While the averaging over a longer period (i.e. 3 years) remove some of the impact of unusual events or specific repairs, it does not remove it completely. Therefore, it is difficult to make any global conclusions and relate the uprate with any of the annual occupation doses as documented in the data sources used.

Cycle length

In the last decade, many plants decided to extend their fuel cycle (new design of fuel allowed for higher burnup) to increase the plant's availability. In many cases the extension of the fuel cycle coincides with the plant modernisation/modification (which required extensive safety analysis and which were then used to justify the extension of the cycle), which in many cases coincided with the uprate. When a fuel cycle is extended beyond one year, the fact that there was an outage in a given year dominates the annual occupational dose. The three-year averaging tends to remove some obvious peaks (and valleys) , which are present in the data, but not completely, and the exact time of the outage varies and the lengths of about 15 month present additional challenges. Moreover, if the cycle duration was changed simultaneously with the uprate, then the comparison of the annual occupational dose before and after the uprate is (almost) meaningless.

Recognising of the impact of uprates

During the process of data collection, some analysis of the data were undertaken to both focus the data collection (and presentation) and make initial conclusions. Some interesting patterns, as supported by the graphical presentation below, emerge:

Throughout the world, occupational doses at NPPs have been steadily decreasing over the past decade, mainly through better application of ALARA principles, the use of better shielding material, but also increased attention paid to occupational dose issues. The ICRP suggested reduction in annual limits for radiation workers also impacted the overall doses

Occupational exposure in the BWR plants is typically about 50% higher than in PWR plants, due to specific of the design.

No direct relationship between the uprates and the occupational doses could be established. The occupational doses on some plants seem to be higher after the uprate, while on others seem to be lower. Without a detailed analysis (on a plant specific level) no general conclusion could be raised.

As it can be seen when comparing units at multiple units sites (that are presumably operated in a same fashion) even with 3 years averaging does not remove the variations caused by specific events

The three years averaging is helping in “smoothing” some of the obvious variations in the annual occupational doses. Nevertheless (as discussed above) even the 3 years average does not always allow for removing of external effects

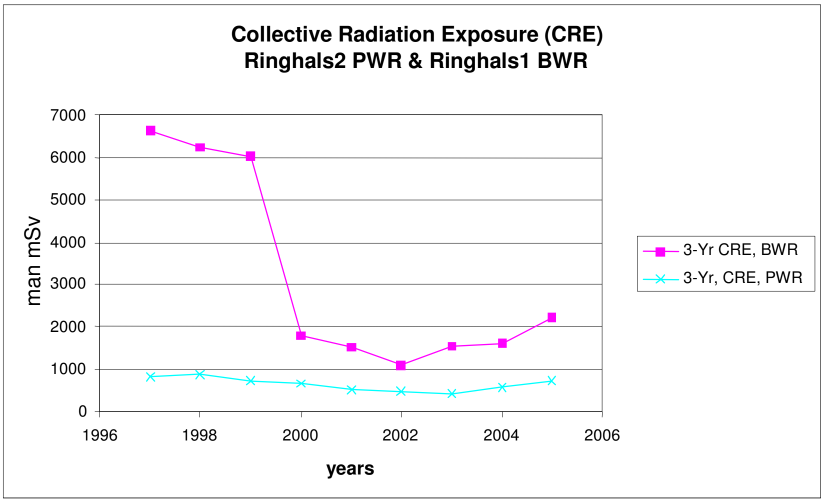


Figure 3.1: PWR to BWR variation

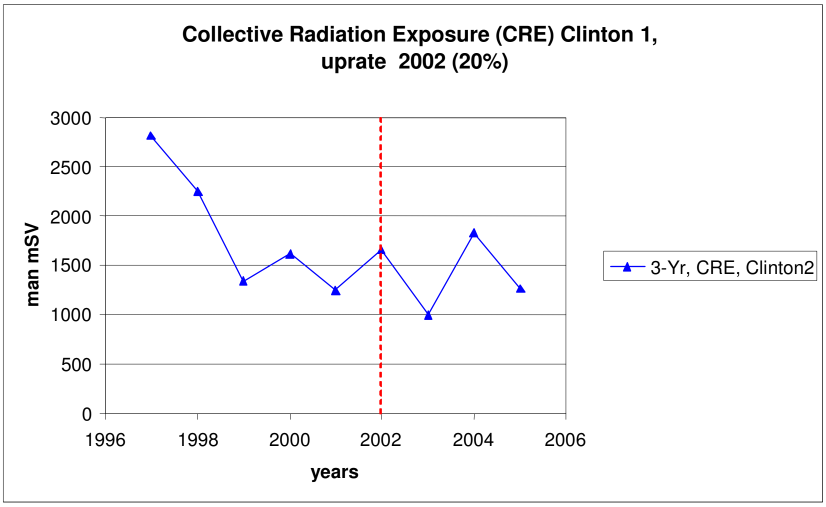


Figure 3.2: Overall trend in occupational doses and impact of an uprate

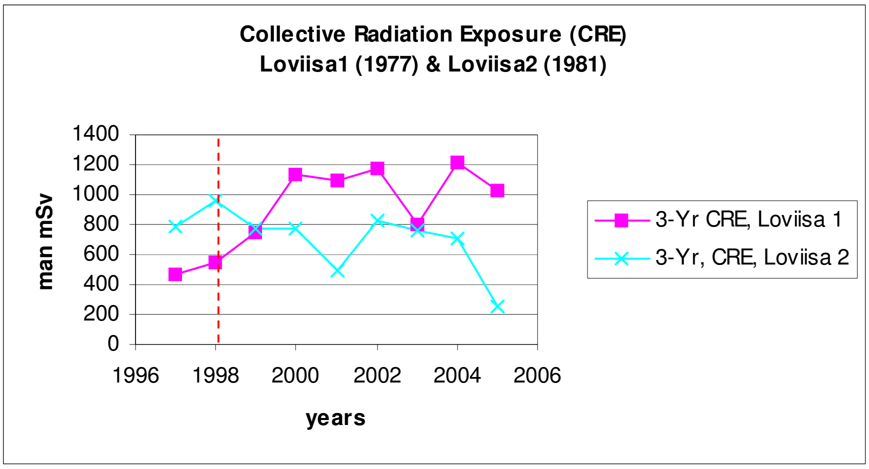


Figure 3.3: Same site variation among units

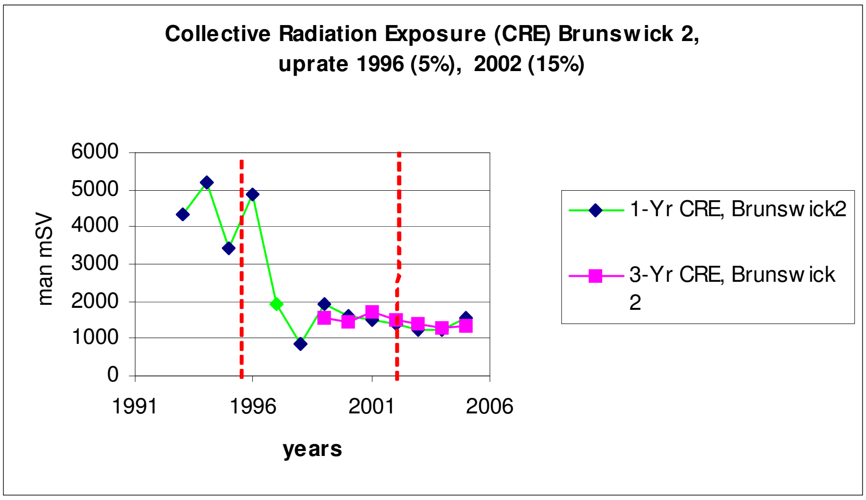


Figure 3.4: Effects of the averaging on presentation

Effects of the uprate on fuel failures

The status of and the effects on fuel are one of the most important elements of the uprates, in particular for extended uprates, where originally established fuel margins are exceeded (requiring new design of fuel). This is the reason that the Uprate Database contains information on the fuel used in each NPP evaluated. Moreover, fuel failures have a direct negative impact on occupational doses in normal operation and accident releases. Therefore, it is of interest to review the effect of the uprates on fuel by evaluating fuel failures in relation to the time of uprate.

The total number of reported fuel failures since January 2000 has decreased in the US (the trend is the same for scrams and general operational events). However, the number of units experiencing fuel failures increased in the same period (about 80% of all units reported fuel failures). The performance of fuel in PWRs and BWRs went in opposite directions, improving for PWR and deteriorating for BWR, although the failure rate (failed assemblies per 1,000 installed) for PWRs is still higher than for BWRs. It appears that whilst all fuel vendors have experienced fuel failures these failures are clustered on specific fuel models.

In PWRs, the dominant fuel failure modes are grid-to-rod fretting followed by debris-related failures (this being greatly reduced by fuel filter and better control during outages with open reactor). On BWRs, the dominant failures are debris fretting (five times higher than for PWRs) and pellet-clad interaction/stress corrosion cracking (PCI-SCC). BWR fuel designs are moving toward more closely packed fuel arrays (10 x 10), increasing the potential for debris-induced failures. With smaller channel dimensions, the possibility of debris-induced failures is greater. On PWRs, most failures are occurring on 11 units (16%). Among BWRs, there are 8 units (25% of total) where most failures occur.

The debris-related failures are hard to relate to an uprate, as it depends mainly on the operation and maintenance processes itself. The rod fretting and PCI are dependant on the fuel loading, but higher fuel loading might be a consequence of an uprate, but also of an extended fuel cycle. However, uprates increased thermal duty in both PWRs and BWRs. Therefore, from a mechanistic point of view, power uprates would be likely to result in reduced margins for fuel.

Contrary to expectation, the evaluation on the distribution of fuel failures and correlating it with the uprates offer a highly inconclusive picture. Only 17 US units did not perform any power uprate at all. Of those, 14 experienced at least one fuel failure. On the other hand, 14 units performed large power uprates (12 BWR and 2 PWRs). Of those only 9 experienced a fuel failure. This suggests that there is no obvious correlation of the power uprate with fuel failures.

Review of recent operational events related to uprates

During the process of data collection the project team initiated some limited-collection and review of information on operational events that occurred as a consequence or are otherwise related to uprates. The aim of this activity was to help identify any specific aspect that could be of interest to consider either during the data collection within or on in depth analysis on selected plants. While no specific issues were identified, some insights of interest were noted, as below.

A review identified more than 40 events that have occurred over the past five years as a result of inadequate design or implementation of uprates. The events involved equipment issues, unanticipated responses to conditions, or challenges for operating staff. The number and types of events indicate that more significant consequences could occur if uprates are not conducted in a controlled manner.

None of the events below had direct consequences on doses to the personnel or releases. However, all of them might have contributed and/or raised the probability of incidents/accidents that could have increased occupational dose or releases.

Significant aspects of events include:

- Loose parts as a result of a flow-induced, high-cycle fatigue failure on a steam dryer cover plate

- Operational transients and equipment damage due to lack of training of plant staff on changes to PCS operating characteristics
- Unanticipated challenges and degraded performance from reductions in margins
- Operation beyond licensed power levels for extended periods due to errors in thermal power calculations following uprates

Steam Dryer Damage at a BWR

After an extended power uprate (18 %), increased steam flow rates led to a high-cycle fatigue failure of a steam dryer cover. The plate broke into several pieces, resulting in a 10-day forced outage to retrieve the loose parts. This condition was not anticipated because the effects of the increased steam flow conditions in combination with existing steam dome forces on the steam dryer were not well understood.

Extended Operation in an Overpower Condition of a BWR

A BWR with stretch uprate was operated at power level greater than 100 % because changes to the process-computer calibration constants for feedwater flow were not identified when the feedwater transmitters were replaced.

Unexpected Feedwater Heater Problems at a BWR

Existing feedwater heater material condition was recognized in the preparation for a stretch uprate, but not implemented due to budget limitations. The problem was identified BEFORE an event occurred. 50 % of the nozzles on the feedwater heaters required repair to mitigate the condition.

Turbine Control System Changes Result in Unanticipated Operational Challenges at a PWR

After a stretch uprate on a PWR, operators experienced difficulty controlling turbine speed and generator load. The need for new operating strategies was not recognized before implementation of the uprate.

Power Reduction at a PWR

Stretch uprate resulted in a reduced-stator cooling water differential-temperature operating margin. A power reduction was required to cope with the situation.

Reactor Instability in a Core after Subsequent Trip of Both Recirculation Pumps in BWR

In parallel with the extended uprate, new fuel elements of GE11 type (9x9 fuel with part length rods) were introduced in a small BWR4 core, thus having a mixed core of GE11 and GE8 (8x8 fuel). During the performance of stability measurements, as part of an uprate, power oscillation was observed. Before this event the plant had not experienced any core power oscillations.

Flow-Induced Vibration Issues (FIV issues) and steam dryer cracking

The commercial nuclear industry has experienced several incidents of steam dryer cracking and FIV issues at nuclear power plants operating at extended power uprate conditions.

After installation of new steam dryers in two BWR units in the middle of the year, which had an improved design to increase their structural capability, the licensee discovered significant degradation of the electrometric relief valves (ERVs) at the end of the year. The licensee shut down the units to repair the ERVs and restarted the units with operation up to pre-uprate power levels.

BWR plants had operated for several years at the extended power uprate level with the modified steam dryers without significant damage. Cracking was found later in two units. The licensee repaired the cracks and installed additional modifications to the steam dryers. The licensee plans to replace the dryers.

During outage inspection activities cracking was identified on a lower guide rod follower bracket at the base of the steam dryer in the BWR plant, but only after several years of operation at 5 percent power uprate conditions.

Abnormalities in Ultrasonic Flow Meter Instrumentation

Use of ultrasonic flow meter of the type used for MUR power uprates has led to unexpected but small differences in power level indications at some plants.

No single event listed above has any casual relation with radiological impact at affected plants, but it does not mean that the above events could not be precursors to these events having radiological impact. Moreover, it could be argued that some of the events (e.g. steam dryer) contributed to occupational doses due to need for repair (in the area with increased radiation level).

It should also be noted that most of the events are occurring at units with a power increase of 5% or more, possibly indicating that the system interactions and PCS issues are not always well understood or addressed during the planning or implementation of an uprate.

4. Analysis of the selected plants

Based on the data collected in the task #1, two PWR and two BWR plants were selected for additional analyses.

4.1 Olkiluoto 1 and 2

4.1.1 Introduction

One of the BWR plants selected for detailed analyses was Olkiluoto nuclear power station consisting of two twin BWR units, Olkiluoto 1 and 2 (OL1 and OL2, both reactors were included in the review). Main reasons for that selection were:

- A considerable power uprate of 25% compared to the initial thermal power level.
- Reactor design and operation conditions very close to most of the Swedish BWRs.
- A good availability of reactor data.

The following sections present the result of the indepth review performed for the OL1/2 plants. Data for the review had been obtained from the TVO utility owning and operating the plants, and great acknowledgment is given to them for supplying the large amount of information.

4.1.2 OL1/2 power uprate

4.1.2.1 Power uprate characteristics

On the west coast of Finland, in Eurajoki, Teollisuuden Voima Oy (TVO) operates two 840 MWe boiling water reactors, Olkiluoto 1 and 2 (OL1/2). Construction work began at Olkiluoto early in 1974. The first reactor unit OL1 was supplied on a turnkey basis by the Swedish company ASEA-ATOM AB (now Westinghouse Electric Sweden AB). In September 1975 construction work began on the second identical plant unit. The OL2 unit was supplied on the same principle with the exception that TVO was responsible for the civil construction work. The major subcontractors for the units were STAL-LAVAL Turbin AB (turbine plant), ASEA AB (electrical equipment, generator), Uddcomb Sweden AB (reactor pressure vessel), Finnatom (reactor internal parts, mechanical components), Oy Strömberg Ab (electrical equipment) and Atomirakennus (OL1 civil construction). The OL2 civil construction was carried out by a Finnish-Swedish consortium, Jukola. The OL1 unit was first connected to the national grid in September 1978 and the OL2 unit in February 1980.

The units have been uprated twice since the commissioning. The thermal power of each reactor was increased from 2000 MW to 2160 MW in 1984 and to 2500 MW in 1998. The corresponding nominal values of the net electrical output were 660 MWe, 710 MWe and 840 MWe, respectively. The present study focuses on the second uprate, resulting in a thermal power level of 125% compared to the initial power level. The net electrical output from the plants during the period 1990-2006 is shown in **Figure 4.1.1**.

The latter uprate was a part of an extensive modernization program implemented in 1994–2006. After the modernization, the plant units fulfilled most of the safety and technical requirements for new nuclear power plants. The modernization program was in line with TVO´s policy to keep the plant units continually up-to-date technically.

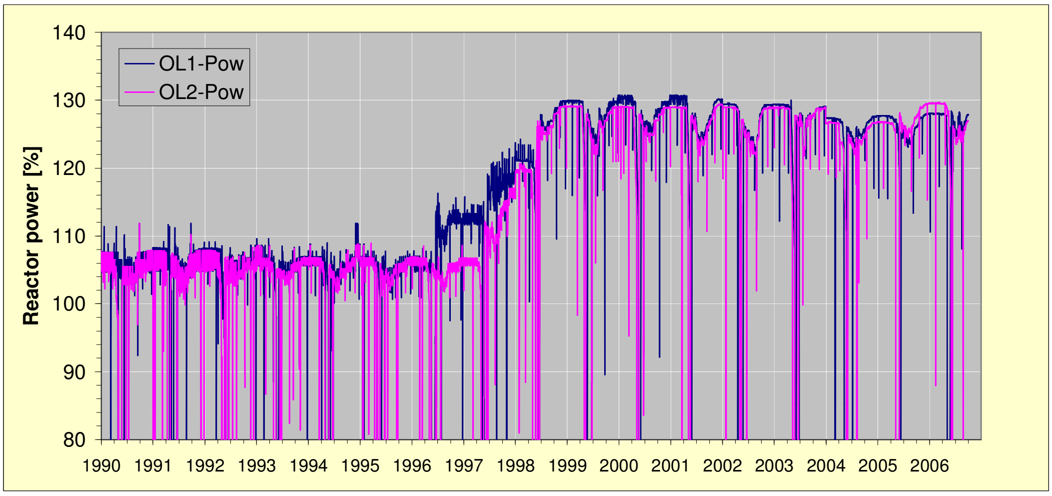


Figure 4.1.1: OL1/2 – Relative reactor power based on net el. output(100% = 660 Mwe)

The reactor building (**Figure 4.1.2**) is the dominant and highest building of the plant. It encloses the primary containment of the reactor and serves as a secondary containment. The reactor service room, at the top of the building, contains the reactor and fuel pools with storage racks for fuel and internals, the reactor service bridge for refuelling operations, and the overhead crane for handling the containment dome, the reactor vessel lid and other heavy components. The bottom part of the reactor building contains separate compartments for important safety-related systems, such as the emergency core cooling systems. The reactor containment is a pre-stressed concrete vessel.

The containment is based on the principle of pressure-suppression. This allows for a compact containment design, with the minimum of equipment installed inside the containment. The use of internal main circulation pumps has allowed further reduction of the containment volume. All components requiring regular service during normal operation of the reactor are located outside the containment. The tightness of the containment is ensured by a steel liner embedded in concrete. The steel liner is protected by the concrete against corrosion, thermal transients and hot water and steam jets or missiles that may occur in the event of a pipe rupture. The containment is inerted, i.e. filled with nitrogen gas during operation. The containment is divided into compartments by internal structures, the upper and lower drywell, and the wetwell. Access to the containment is gained through air locks at the bottom of the lower drywell, and at the floor of the upper drywell. The cylindrical part of the containment vessel extends to the top of the reactor vessel. The condensation pool is enclosed in the annular space between the containment vessel wall and an inner cylindrical wall, which also carries the biological shield.

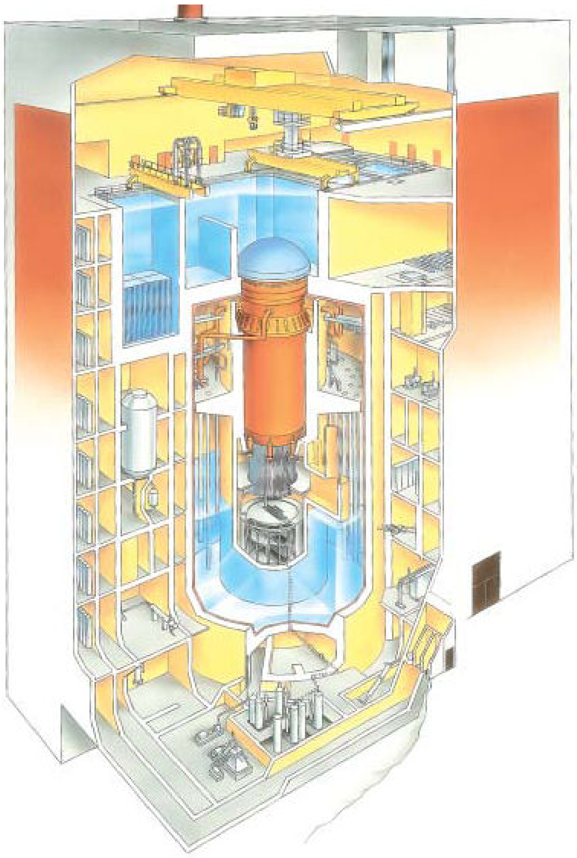


Figure 4.1.2: OL1/2 - Section through the reactor building and the reactor containment

1 – Reactor, 2 - Main steam lines, 3 - Fuel storage pool, 4 - Reactor service bridge,

5 - Reactor service room crane, 6 - Main circulation pumps, 7 - Control rod drives,

8 - Reactor containment vessel, 9 - Control rod service platform, 10 - Blow-down pipes,

11 - Embedded steel liner. 12 - Condensation pool, 13 - Scram system tanks, 14 – Venturi scrubber

The reactor pressure vessel (**Figure 4.1.3**) is made of low-alloy steel, with a lining of stainless steel. All major pipe nozzles are located above the top of the core, to ensure that the core is kept flooded in the event of a pipe rupture in the primary systems. The reactor vessel hangs on top of the biological shield by means of a welded-on support skirt. The vessel support skirt is located near the primary system pipe connections, an arrangement which minimizes the pipe stresses resulting from the thermal expansion of the vessel. This location also allows for more maintenance space around the recirculation pumps.

The reactor internals are designed to allow for fast and safe handling during refuelling operations. Apart from the moderator tank support skirt and the pump deck, which are welded to the reactor vessel, all internals are removable. The internals are held in position in the reactor vessel by means of resilient beams in the reactor vessel cover. When the cover has been removed, the internals can be lifted out of the reactor without breaking any bolted joints. Another related feature is the thermal insulation of the reactor cover, which is fastened to the inside of the containment dome, they are removed together when the reactor is to be opened. The procedures for removing the reactor vessel cover have been considerably simplified by eliminating all the external pipe connections to the reactor cover and making the connection inside the reactor.

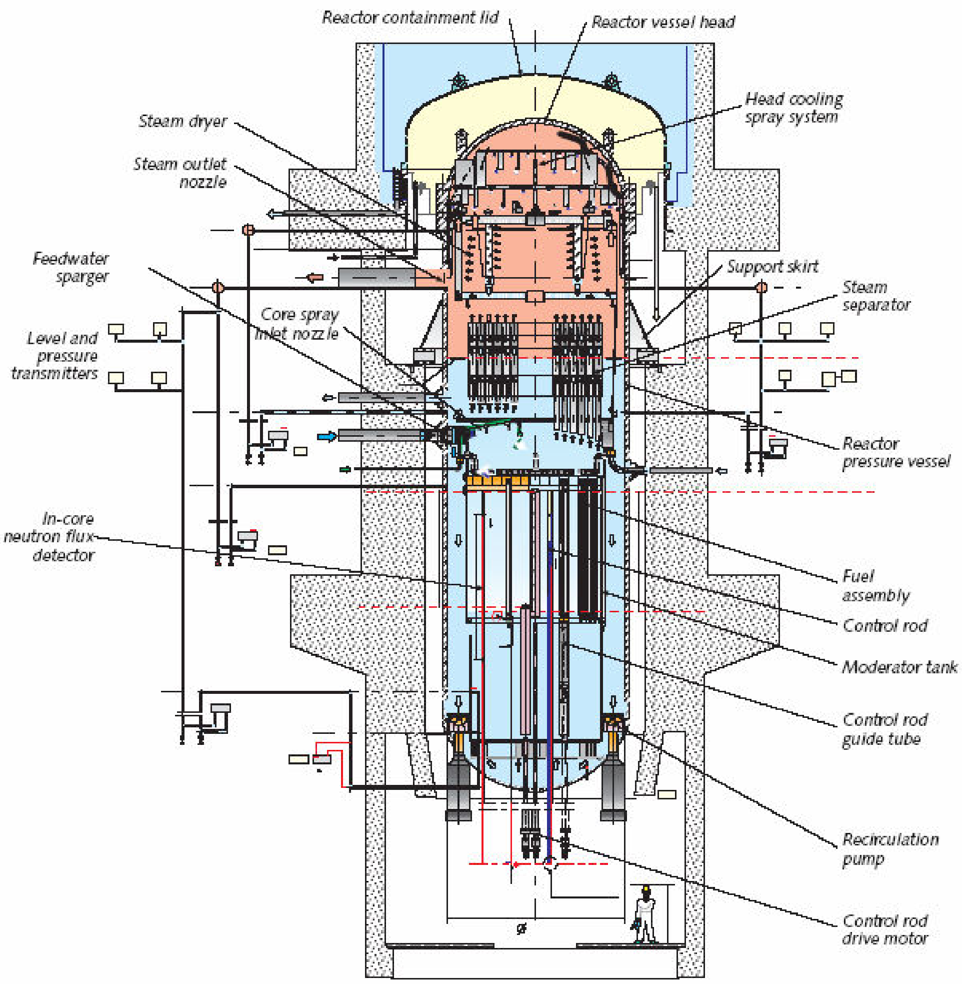


Figure 4.1.3: OL1/2 - Sectional view of the reactor pressure vessel

The coolant flow through the core is maintained by means of six internal circulation pumps. The internal circulation pump design is based on the use of wet motors, thus eliminating shaft seals. The motor housing forms an integral part of the reactor vessel. Internal circulation pumps offer a number of advantages over external pumps:

- no risk of major pipe rupture below the top of the core
- compact containment design
- low circulation pressure drop improves natural circulation and decreases auxiliary power demand
- lowered drywell background radiation level contributes to very low occupational exposure during pump motor maintenance and inspection
- significant reduction of primary system weld length.

A split shaft design allows for convenient assembly and disassembly. The pump shaft extends into the hollow motor shaft and power is transmitted from the motor shaft through a coupling that can be disassembled from the bottom of the motor housing. A pump motor or impeller can thus be removed or replaced without draining the water from the reactor vessel.

The turbine plant comprises a single turbine-generator unit. It is a 3000 rpm tandem-compound, single-shaft machine with one high pressure (HP) cylinder and four low pressure (LP) cylinders. The turbine is equipped with a single-pass condenser, mounted across the longitudinal axis of the turbine. The condenser is sea water cooled, and equipped with titanium tubing. The heating of the condensate and feedwater up to a temperature of 185°C is carried out in five stages. Both the LP heaters and the HP feed heaters are split up into two half-capacity, parallel circuits, each equipped with a bypass system.

The purpose of the offgas system (**Figure 4.1.4**) is to limit the emission of radioactive gases from the plant. The system employs charcoal absorption, and consists basically of two decay vessels, two dryers, two fans and three charcoal columns. The gas from the turbine ejectors flows through the recombiner system, the first decay vessel, one of the dryers, one of the charcoal columns, one of the fans and finally the second decay vessel.

The xenon content in the offgas flow will be absorbed in the charcoal. When the absorption capacity of the column has been used, the flow is automatically routed through another column. The “used” column is then connected to the turbine condenser, and the xenon in its charcoal is desorbed by a small flow of “cleaned” air to the condenser.

The main difference between this offgas system and other charcoal-type systems is that it uses a relatively small quantity of charcoal and that the radioactive gases decay in sand beds instead of charcoal columns. Thus, only a small fraction of the radioactive gases, essentially low energy radiation emitters, will reach the charcoal columns.

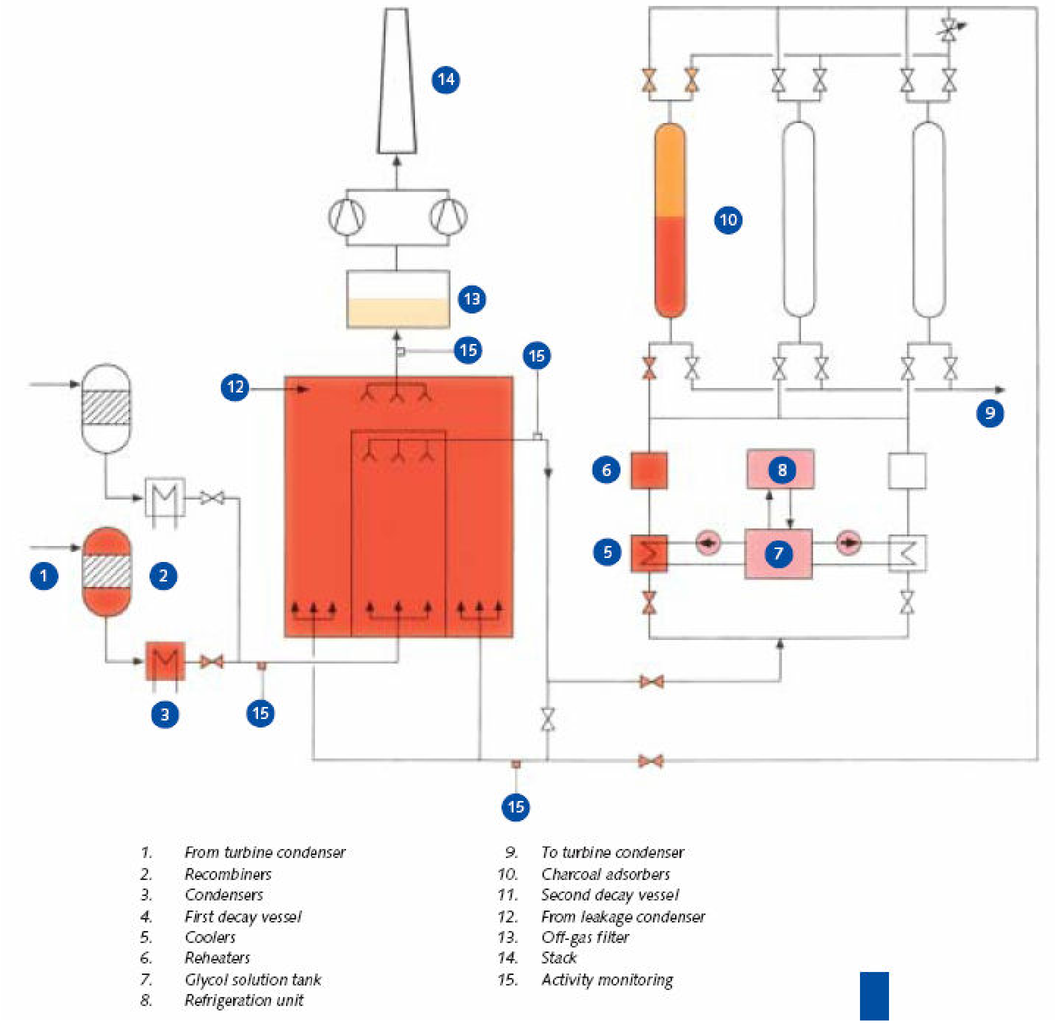


Figure 4.1.4: OL1/2 – Offgas system

The primary circuit of the reactor operates without chemical additives to the coolant. That means that neither hydrogen gas[1] nor zinc[2] is injected as in many other BWRs. Feedwater chemistry corresponds to that of "neutral water", i.e. water with very low electrical conductivity. High reactor water purity contributes substantially to reliable operation of the reactor, prevents crud deposits on the fuel rods and reduces the radioactive contamination of the primary systems, thus ensuring better accessibility and lower occupational radiation exposure.

The primary circuit water is treated by two independent, coordinated cleanup systems, the reactor water cleanup and the condensate cleanup systems (**Figure 4.1.5**). The reactor water cleanup system comprises two ion exchanger units of radial flow, bed filter type. Each unit is capable of producing the normal cleanup flow, corresponding to 2% of the maximum feedwater flow, so that the flow may be doubled when necessary. The flow

[1] Hydrogen Water Chemistry (HWC) in order to reduce Stress Corrosion Cracking. The operation utilized in OL1/2 without hydrogen injection is normally classified as Normal Water Chemistry (NWC).

[2] Injection of Depleted Zinc Oxide (DZO) to the reactor water is applied in many BWRs in order to reduce radiation fields.

rate in the reactor water cleanup system was increased after the power uprate to maintain the 2% capacity per filter unit. The cleanup flow is generated by one pump in the shutdown cooling system, through two regenerative heat exchangers and one cooler, to one of the ion exchangers, and returns to the reactor via the regenerative heat exchangers. A portion of it flows through the scram system, purging and cooling the control rod drives.

The condensate cleanup (CCU) system, connected between the first and second condensate preheaters, comprises seven parallel coupled trains with deep-action, rod-type precoat filters. Note that all turbine drainage is cascaded back and cleaned in the CCU system (**Figure 4.1.5**).

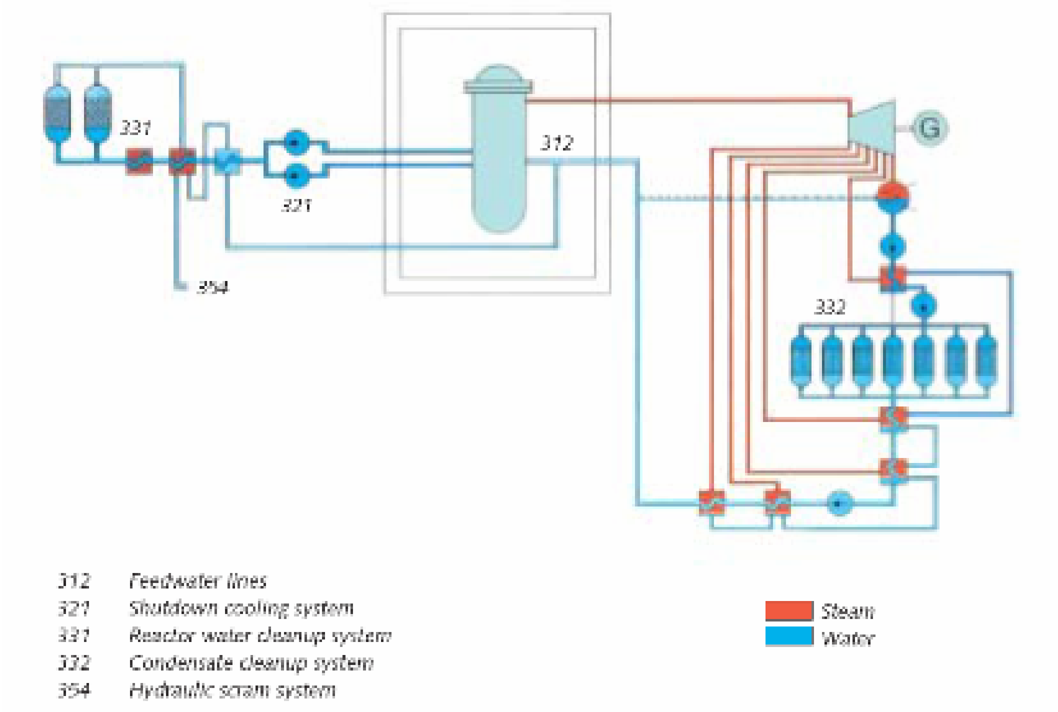


Figure 4.1.5: OL1/2 – Reactor water and condensate cleanup system
(Reactor water: deep bed filters, condensate: pre-coat filter demineralizers)

The reactor cores of OL1 and OL2 contain 500 fuel assemblies each, which means 5 MW_{th} per fuel assembly at present maximum power level, compared to 4 MW_{th} per assembly at initial design power. The fuel assemblies for the first cores and subsequent reloads of OL1 and OL2 were of 8x8 design. Since then, new fuel designs have gradually been adopted. At present 10x10 fuel is used in both reactors. This type of fuel has new features that have made reactor power uprating and more efficient fuel utilizations possible. The linear heat load has in principle been maintained compared to the initial core.

Main data for the OL1 and OL2 reactors are provided in the **Table 4.1.1**.

Table 4.1.1: OL1/2 – Main data

General		
Reactor thermal power	MWth	2500
Electrical output, net	MWe	840
Electric output, gross	MWe	870
Reactor steam flow	kg/s	1260
Reactor operating pressure	bar	70
Feedwater temperature	°C	185
Core		
Number of fuel assemblies		500
Total fuel weight	ton U	86-90
Core diameter (equiv.)	mm	3880
Core height	mm	3680
Fuel		
Fuel rods per assembly		91-100
Fuel rod outer diameter	mm	9.62-10.30
Cladding material	Zr-2	
Fuel density	kgUO$_2$/1	10.5
Weight of fuel assembly (incl. channel)	kg	292-331
Control rods		
Number of control rods		121
Absorber length	mm	3650
Total length	mm	6380
Absorber material		B$_4$C and Hf
Pressure vessel		
Inner diameter	mm	5540
Inner height	mm	20593
Wall thickness, carbon steel (ASME A533B, A508Gr2)	mm	134
Thickness of stainless steel lining	mm	5
Design pressure	bar	85
Design temperature	°C	300
Weight of vessel	ton	524
Weight of cover	ton	107
Turbine plant		
Turbine		
Nominal rating	MW	870
Live steam pressure	bar	67
Live steam temperature	°C	283
Live steam flow	kg/s	1260
Rated speed	rpm	3000
HP cylinder design	Axial, 2-flow	

General		
Reactor thermal power	MWth	2500
Electrical output, net	MWe	840
Electric output, gross	MWe	870
Reactor steam flow	kg/s	1260
Reactor operating pressure	bar	70
Feedwater temperature	°C	185
Core		
Number of fuel assemblies		500
Total fuel weight	ton U	86-90
Core diameter (equiv.)	mm	3880
Core height	mm	3680
Fuel		
Fuel rods per assembly		91-100
Fuel rod outer diameter	mm	9.62-10.30
Cladding material	Zr-2	
Fuel density	kgUO$_2$/1	10.5
Weight of fuel assembly (incl. channel)	kg	292-331
Control rods		
Number of control rods		121
Absorber length	mm	3650
Total length	mm	6380
Absorber material		B$_4$C and Hf
Pressure vessel		
Inner diameter	mm	5540
Inner height	mm	20593
Wall thickness, carbon steel (ASME A533B, A508Gr2)	mm	134
Thickness of stainless steel lining	mm	5
Design pressure	bar	85
Design temperature	°C	300
Weight of vessel	ton	524
Weight of cover	ton	107
Turbine plant		
Turbine		
Nominal rating	MW	870
Live steam pressure	bar	67
Live steam temperature	°C	283
Live steam flow	kg/s	1260
Rated speed	rpm	3000
HP cylinder design	Axial, 2-flow	

A summary of all modifications that have been introduced in the OL1/2 plants during the period 1994 – 2006 is presented in **Table 4.1.2** and **Table 4.1.3**. Note that all modifications are not necessarily due to the power uprates, but in many cases were part of the ongoing plant modernization programs carried out. Those two aspects are not always easy to separate; introduced modifications are in many cases addressing both aspects. The major bulk of work in association to the power uprates was performed during the 1996, 1997 and 1998 outages in OL1, and during the 1997 and 1998 outages in OL2. Note also that major modernization work was carried out during 2005 in OL2, and during 2006 in OL1. The variation in outage lengths 1990 – 2006 is presented in **Figure 4.1.6**. The normal strategy is that a larger outage is performed at one of the plants during one year, but only a shorter outage for refuelling at the other plant, and vice versa the following year. The years 1997, 1998 and 2000, however, display somewhat prolonged outages in both plants. Generally speaking, both plants are characterized by short outages and high plant availability, and the modernization and power uprate programs have been carried out with a rather small impact on average outage length (impact on occupational exposures will be addressed later in the report).

Table 4.1.2: OL1/2 – Summary of modernization work performed 1994 - 1999

YEAR	OL1	OL2
1994	• Replacement of two vessels in Condensate system to SS-steel. • Replacement of two heat exchangers in Condensate system.	• Replacement of core grid. • Replacement of RHR syst. piping app. 185 m outside of containment. • Replacement of Main generator. • Replacement of two vessels in Condensate system to SS-steel.
1995	• Modifications of operating equipment of personnel air lock doors of PS-containment	• Modifications of operating equipment of personnel air lock doors of PS-containment. • Replacement of four heat exchangers in Condensate system
1996	• Installation of two new safety relief valves with blow down piping. • Replacement of LP3- and LP4-turbines. • Replacement of Main generator. • Modifications of HP-turbine. • Modernization of turbine automation. • Increase of capacity of Secondary cooling systems. • Replacement of condensate lines to SS-steel (app. 50 m) • Modernization of Reactor service bridge. • **Increase of power to 113% (2268 MW).**	• Preparation works for the 1997 modernization.
1997	• Installation of new Moderator tank head / Steam separator unit. • Replacement of 2 valves and some piping in RHR syst. • Replacement of some piping in the Core spray syst. • Replacement of LP1- and LP2-turbines. • Modernization of Condensate pumps P1-P6 and modifications of Condensate pipelines. • Change of Generator switch.	• Installation of new Moderator tank head / Steam separator unit. • Installation of two new safety relief valves with blow down piping. • Replacement of LP3- and LP4-turbines. • Modifications of HP-turbine. • Replacement of some piping in the RWCU syst. • Increase of capacity of Secondary cooling systems.

YEAR	OL1	OL2
1998	• Modifications of HP-turbine and replacement of shaft sealings. • Replacement of six vessels in the Reheater and moisture separator syst. to larger. • Installation of new moisture separators in steam lines to the Reheater and moisture separator syst. • Related to the modifications of turbine plant, replacement of plenty of pipelines in Reheater and moisture separator and Steam extraction syst. • Modernization of the reheaters in the Reheater and moisture separator syst. • Replacement of HP-regulator valves to larger and changes for main steam lines. • Lifting of level of two condensate tanks (1 m) • Modernization of Boron syst. • **MODE-project complete. Increase of power to 2500 MWth (840 MWe net)**	• Replacement of 2 valves and some piping in RHR syst. • Replacement of LP1- and LP2-turbines. • Modifications of HP-turbine and replacement of shaft sealings. • Replacement of six vessels in the Reheater and moisture separator syst. to larger. • Installation of new moisture separators in steam lines to the Reheater and moisture separator syst. • Related to the modifications of turbine plant, replacement of plenty of pipelines in Reheater and moisture separator and Steam extraction syst. • Modernization of the reheaters in the Reheater and moisture separator syst. • Replacement of HP-regulator valves to larger and changes for main steam lines. • Lifting of level of two condensate tanks (1 m) • Modernization of condensate pumps P1-P6 and modifications of condensate pipelines. • Change of Generator switch. • Modernization of Boron syst. • **MODE-project complete. Increase of power to 2500 MWth (840 MWe net)**
1999	• Modifications of Core spray pipelines inside of RPV. • Assuring the opening of two safety relief valves during severe reactor accident conditions. • Commutation of halon to halotron in Fire extinguisher system.	• Modifications of Core spray pipelines inside of RPV. • Assuring the opening of two safety relief valves during severe reactor accident conditions. • Commutation of halon to halotron in Fire extinguisher system.

Table 4.1.3: OL1/2 – Summary of modernization work performed 2000-2006

YEAR	OL1	OL2
2000	• Replacement of two valves in the RHR syst. and parts of the piping after these valves. • Replacement of steam extraction pipelines from LP1- and LP2-turbines to one LP feedwater heater. • Replacement of the main part of the piping of the Flange cooling syst. in- and outside of the containment. • Changes for the suction lines of pumps P1 and P2 and modification of the tank T1 of Boron syst. • Replacement of the pumps P1-P3 in the Turbine oil syst.	• Replacement of steam extraction pipelines from LP1- and LP2-turbines to one LP feedwater heater. • Changes for the suction lines of pumps P1 and P2 and modification of the tank T1 of Boron syst.

2001	• Modifications introduced for lower personnel air lock of PS-containment by considering possible steam explosion.	• Replacement of one valve in the RHR syst. and part of the piping after the valve. • Replacement of the main part of the piping of the Flange cooling syst. in- and outside of the containment. • Install of a new service valve in Flange cooling system. • Modifications for the fluid couplings and changes of the rotor wheels of the feedwater pumps P301. P302 and P304.
2002	• Replacement of the water mixing location of Feedwater and Auxiliary feedwater system. • Install of a new service valve in Flange cooling system. • Modifications for the fluid couplings of the feedwater pumps P301 and P302.	• Modifications introduced for lower personnel air lock of PS-containment by considering possible steam explosion.
2003	• Modifications for the fluid couplings of the feedwater pumps P303 and P304.	• Replacement of the feedwater spargers in the RPV. • Condensate cleanup project, including process modifications, replacement of heat exchanger E1 and increasing the pressure of condensate pumps P5 and P6 with 3.5 bar.
2004	• Replacement of the feedwater spargers in the RPV. • Main steam line supports replacements. • Condensate cleanup project, including process modifications, replacement of heat exchanger E1 and increasing the pressure of condensate pumps P5 and P6 with 3.5 bar. • Preparative works for the replacement of Moisture separator / reheaters and retrofit of HP turbines at 2006.	• Main steam line supports replacements. • The "old" feedwater spargers were placed back. • The preparative works for the replacement of moisture separator reheaters and retrofit of HP turbines at 2005.
2005	• Preparative works for the replacement of Moisture separator / reheaters and retrofit of HP turbines at 2006. • Main steam line supports replacements.	• The replacement of Steam dryer in RPV. • The replacement of repaired feedwater spargers. • The replacement of HP-turbine (rotor including blades, inner casing including stationary blades, steam inlet piping). • The replacement of steam Reheater / moisture separators and drainage piping and tanks. • Main steam line supports replacements.
2006	• The replacement of Steam dryer in RPV. • The replacement of HP-turbine (rotor including blades, inner casing including stationary blades, steam inlet piping). • The replacement of steam reheater / moisture separators and drainage piping and tanks. • Main steam line supports replacements. • Replacement of one inner isolation valve in RHR syst. and one valve in the Flange cooling syst. in the containment. • Inspections and repair of the LP4 rotor because of the stress corrosion cracks.	• Main steam line supports replacements. • The "old" Steam Dryer were reinstalled because of repair of the "new" Steam Dryer.

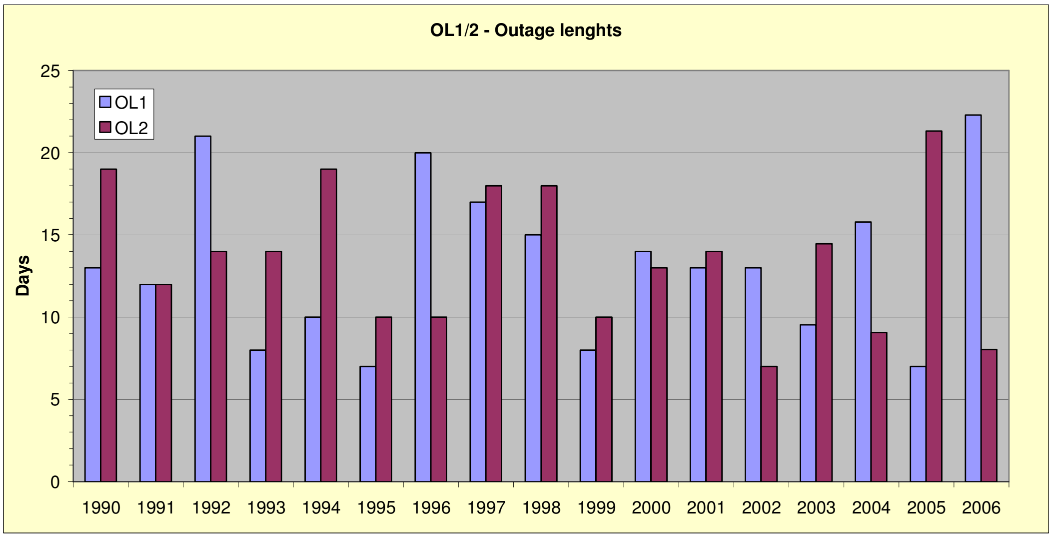


Figure 4.1.6: OL1/2 – Outage lengths 1900 - 2006

4.1.2.2 Water chemistry and radiochemistry

The water chemistry and radiochemistry conditions in the plants before and after the power uprates are presented a series of diagrams, **Figure 4.1.7** - **Figure 4.1.15**. The reactor power of each plant is included in the diagrams for identification of the power uprate history.

Figure 4.1.7 shows measured concentrations of iron in the final feedwater. The efficiency of the condensate cleanup plant (CCU) is of a large importance for the feedwater chemistry, and a power uprate implies a certain strain on that system due to higher flow rates. However, an improvement of this system had already been introduced at the beginning of the 90:ies, and good iron removal efficiency was also maintained after the power uprate. The increase of iron in OL2 during the end of the period is deliberate because of recommendations from the fuel vendor to that plant to maintain a certain excess of iron in the fuel crud.

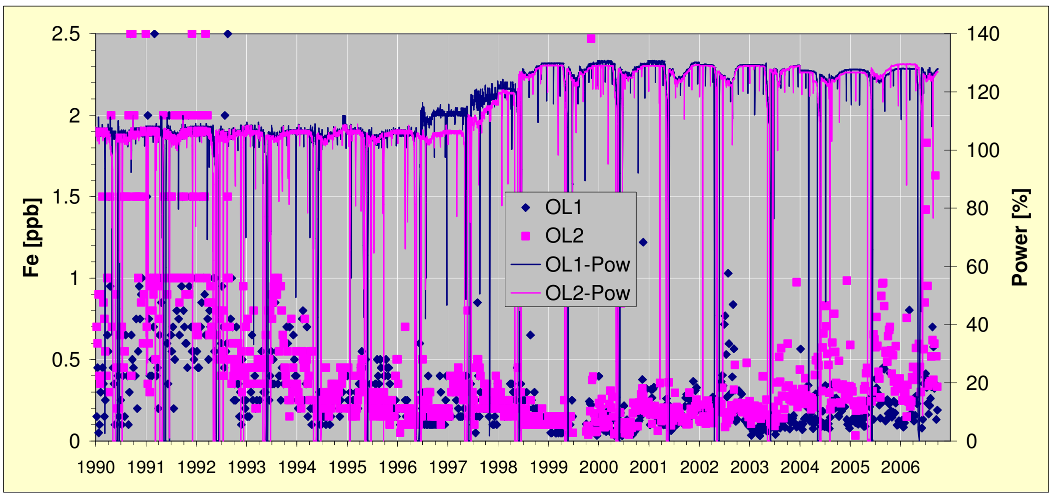


Figure 4.1.7: OL1/2 – Feedwater Fe before and after power uprate

The CCU system is, as stated, important for maintaining good water chemistry. However it is also the major source of sulphate, one of the most detrimental impurities in the reactor water from a materials integrity point of view. (**Figure 4.1.8**). The source of the sulphate is mainly from degradation of ion exchange resin, normally from the CCU system. There are three important factors that affect the rate of degradation of CCU resin:

1. Temperature - a power uprate normally results in an increased temperature, which is detrimental. The increase in OL1/2 was from 60°C to 70°C.

2. The quantity of hydrogen peroxide (H_2O_2) in the condensate. Both carry over with the steam and the decomposition of H_2O_2 which are affected by a power uprate. Normally an increase of H_2O_2 is experienced.

3. The total loading of iron on the resin. The higher flow rates associated with uprates normally imply increased loading of iron on the resin. The higher loading of iron may demand more frequent backflushing of the CCU filters, which, on the other hand, results in an increased production of radwaste.

Much effort has been spent in the OL1/2 plants to reduce the degradation of CCU resin. Important modifications were introduced in 2003 in OL2 and in 2004 in OL1 (**Table 4.1.3**), which resulted in a decrease of operation temperature for the CCU filters from about 70°C to about 50°C. This measure has significantly improved the situation and both plants are today basically meeting the recommended limit (WANO limit: 2 ppb).

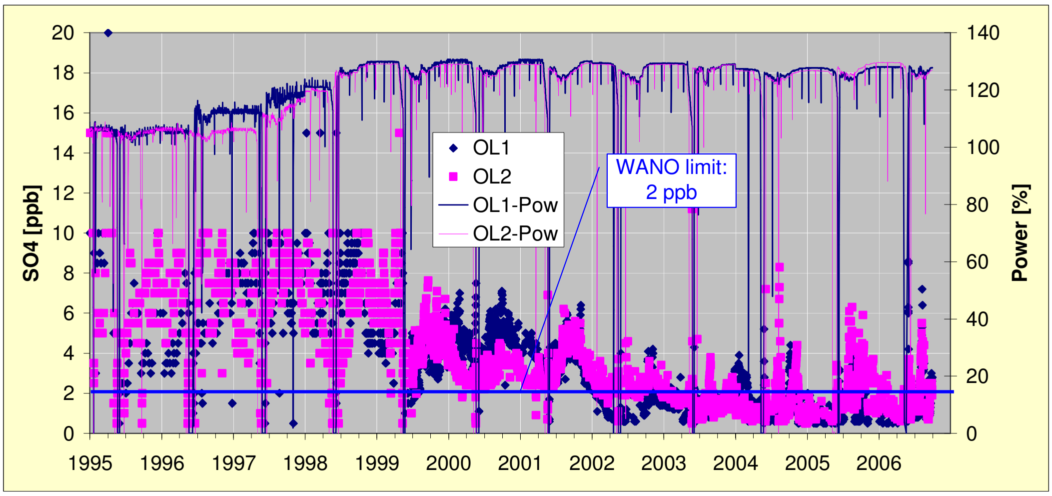


Figure 4.1.8: OL1/2 – Reactor water sulphate before and after power uprate

Some important corrosion products, Ni, Co and Cr, in the reactor water are shown in **Figure 4.1.9**, **Figure 4.1.10** and **Figure 4.1.11**, respectively. No dramatic influence of the power uprates in 1998 is seen (the reducing trend for Ni is probably due to improvements in sampling procedures). Note, however, an increase of Cr after the 2005 outage in OL2. This increase is most likely due to the installation of a new moisture separator in the RPV (**Table 4.1.3**) with a large unfilmed stainless steel surface. Similar increases have been seen in other plants when installing new components with large stainless steel surfaces.

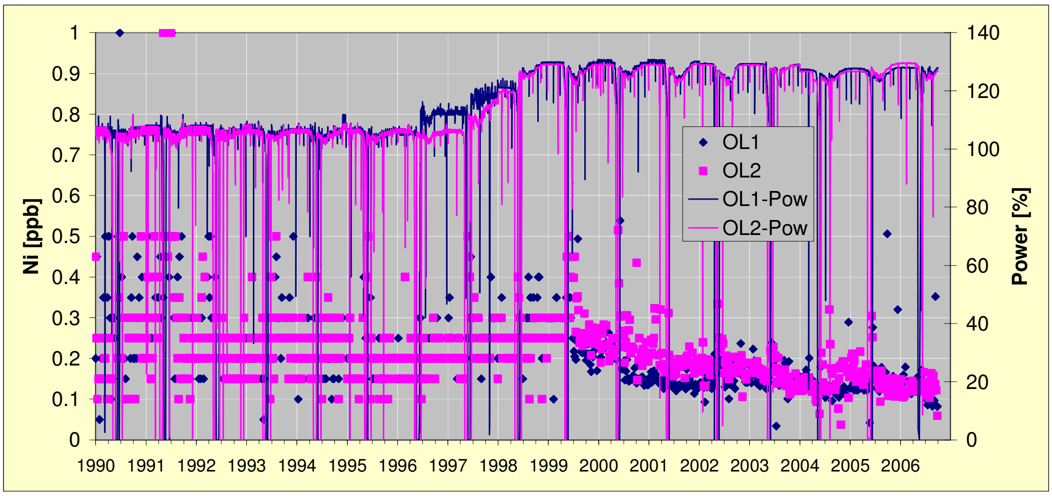


Figure 4.1.9 OL1/2 – Reactor water Ni before and after power uprate

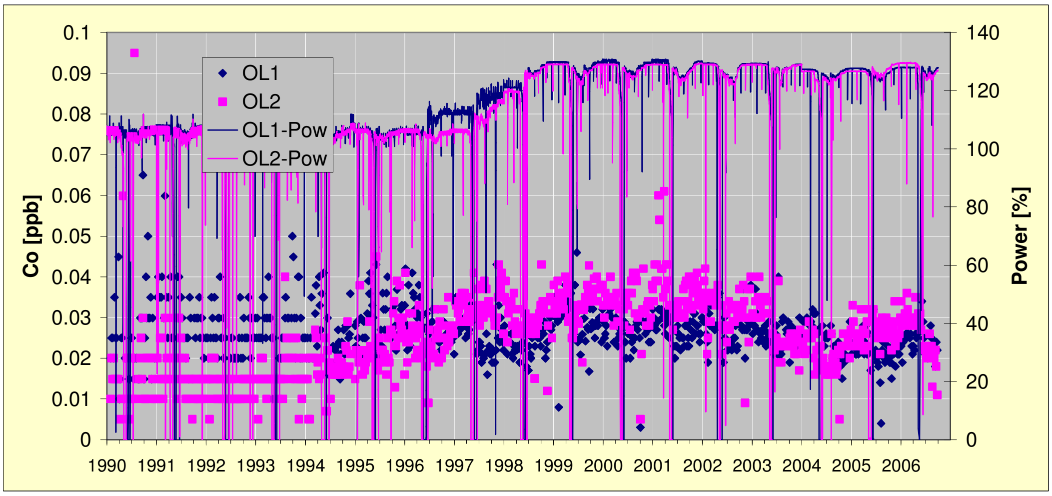


Figure 4.1.10: OL1/2 – Reactor water Co before and after power uprate

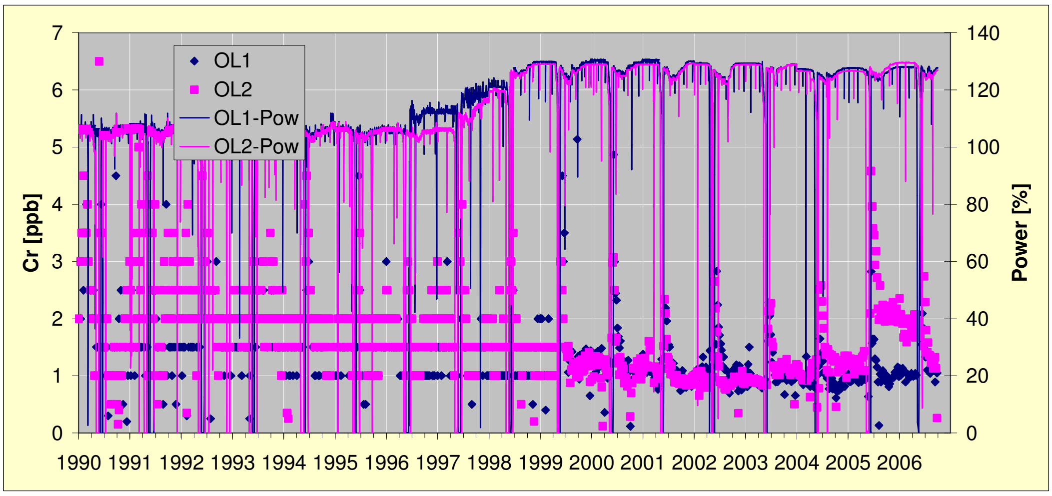


Figure 4.1.11: OL1/2 – Reactor water Cr before and after power uprate

Radiation levels in a BWR plant are very much determined by the activated corrosion products, e.g. Co-60, and measured reactor water concentrations of some of these are presented in **Figure 4.1.12** - **Figure 4.1.15**, and are commented on below:

– Co-60: The most important activated corrosion product, being responsible for the dominant fraction of the occupational exposures in the BWR plants. The Co-60 levels in OL1/2 show some variations, but these variations are not easily correlated to the power uprates. The variation in reactor water activity is larger than the variation in Co-60 activity on primary piping, which is described in a later section. The Co-60 reactor water concentration is affected by other factors than the inflow of cobalt, e.g. the corrosion product composition of the fuel crud.

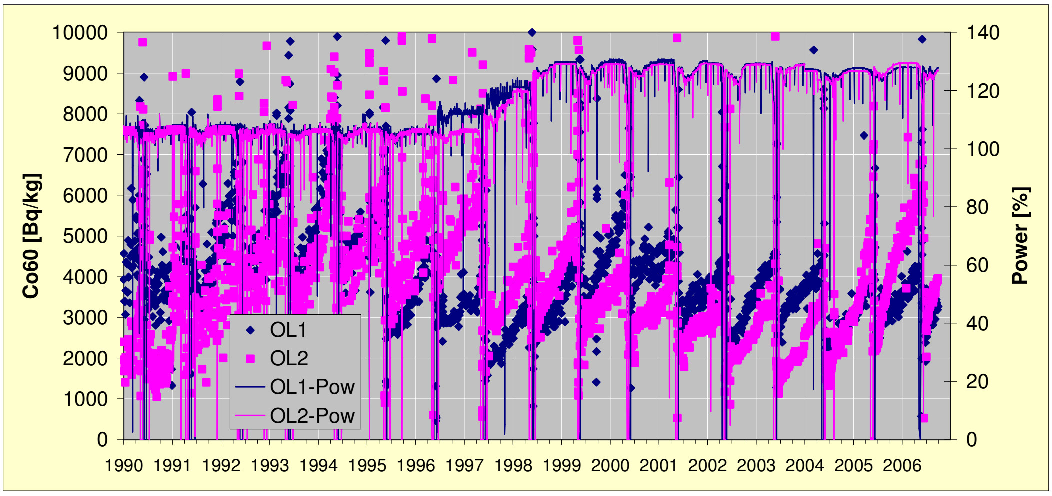


Figure 4.1.12: OL1/2 – Reactor water Co-60 before and after power uprate

– Co-58 is due to neutron activation of Ni, and the reactor water concentration is very much determined by the surface area of in-core Inconel in the reactor, i.e. Inconel in fuel spacers. This is evident by studying the two sister reactors OL1 and OL2:
 - OL1: Have during the period been loaded with fuel with a small fuel spacer Inconel area.
 - OL2: -1999: about 0.5 m^2 per assembly, 1999 – 2003, about 0.1 m^2 per assembly, 2003-: about 0.9 m^2 per assembly.
The OL1/2 data show an influence of fuel design but not of power uprate.

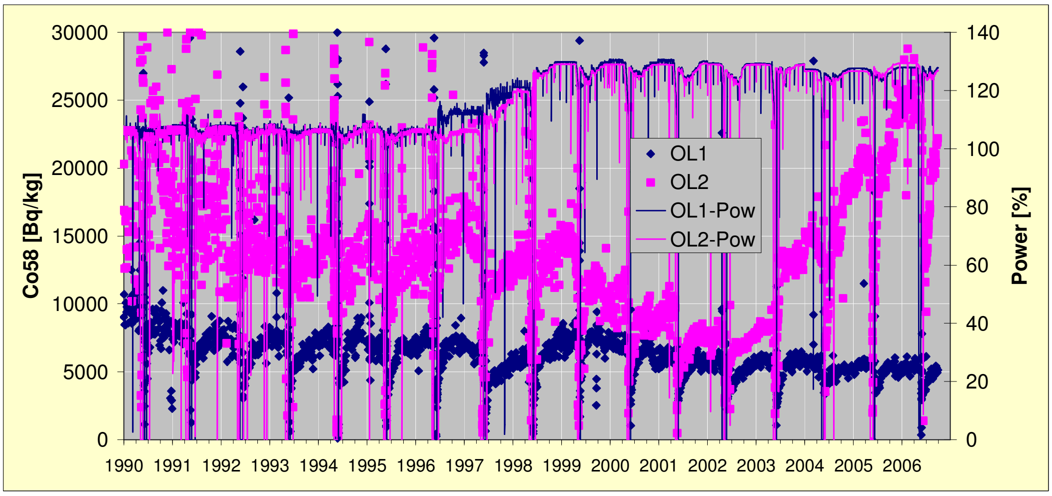


Figure 4.1.13: OL1/2 – Reactor water Co-58 before and after power uprate

– Mn-54 is produced by neutron activation of iron. The effect of reduced feedwater iron inflow due to improved CCU performance is clearly visible. Recent slight increases are deliberate, i.e. a small increase of feedwater iron is arranged in order to follow fuel vendor recommendations.

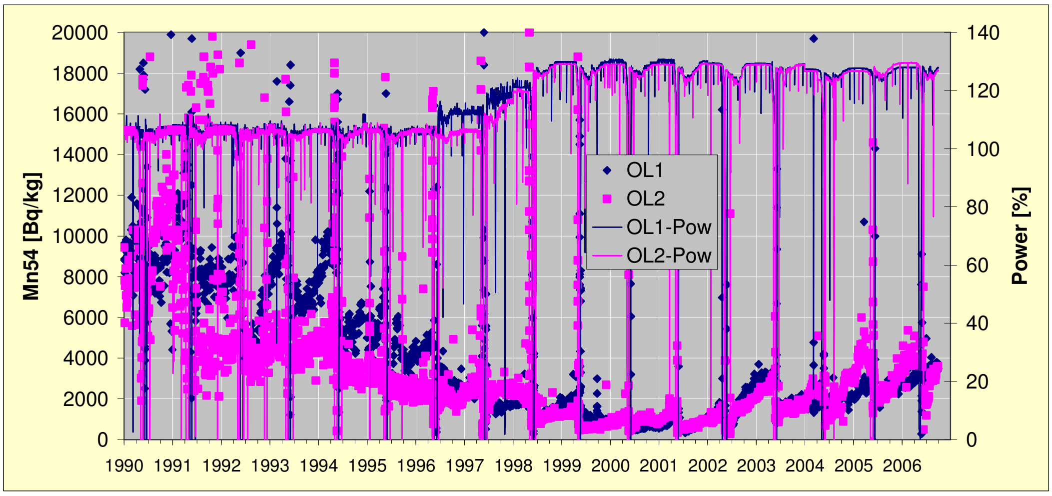


Figure 4.1.14: OL1/2 – Reactor water Mn-54 before and after power uprate

– Sb-124 is produced through neutron activation of antimony, and only small quantities of antimony are needed to produce a significant amount of Sb-124. The resulting amount of Sb-124 is primarily determined by the inflow of antimony, but also by the corrosion product balance in the reactor water[3]. A suspected main source of antimony is antimony-containing graphite sealings in the feedwater pumps, a material that has been shown to be not easily be replaced by the antimony-free substitutes. The Sb-124 activity does not seem to be directly associated with the power uprate.

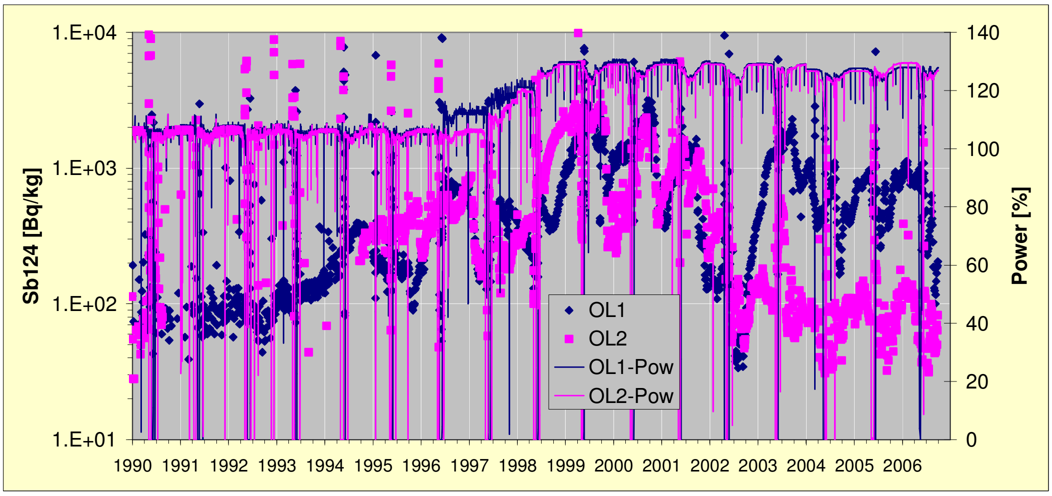


Figure 4.1.15: OL1/2 – Reactor water Sb-124 before and after power uprate

The amount of moisture in the steam leaving the reactor pressure vessel is supervised by comparing measured Na-24 in condensate and reactor water. The results of such measurements in OL1/2 are presented in **Figure 4.1.16**. A significant variation is seen through the years:

– 1996: Some problems with enhanced steam moisture were already experience during the period before the power uprate in 1998. The average moisture content was, however, controlled at a level of about 0.1% through optimized fuel loading.

– 1996-1997: Some further enhancement was seen, especially in OL1 when the power was gradually increased.

– 1997-1998: New moderator tank lids including steam separators were installed in both plants during the 1997 outage. Some reduction in moisture was seen during the following cycle.

– 1998-2005: The power uprate after the 1998 outage resulted in a significant enhancement of moisture, with typical levels of 0.3 – 0.4%. This increase resulted

[3] Excess of Fe compared to Ni (+Zn) means low solubility and low reactor water concentrations, and vice versa.

in a considerable buildup of radiation fields around main steam lines and other turbine components (discussed in more detail in a later section).

- 2005-2007: New improved steam dryers were installed in the reactor pressure vessels, 2005 in OL2 and 2006 in OL1. The new dryers turned out to have a dramatic effect on the steam moisture, with a reduction down to about 0.01%. A crack in a weld in the new dryer in OL2 was observed during the 2006 outage, which resulted in the temporary reinstallation of the old dryer and a return to the enhanced steam moisture level during the present cycle (The crack is currently being repaired, and the new dryer is planned to be reinstalled during the 2007 outage).

The influence on the radiation fields of the variation in steam moisture will be discussed in a later section.

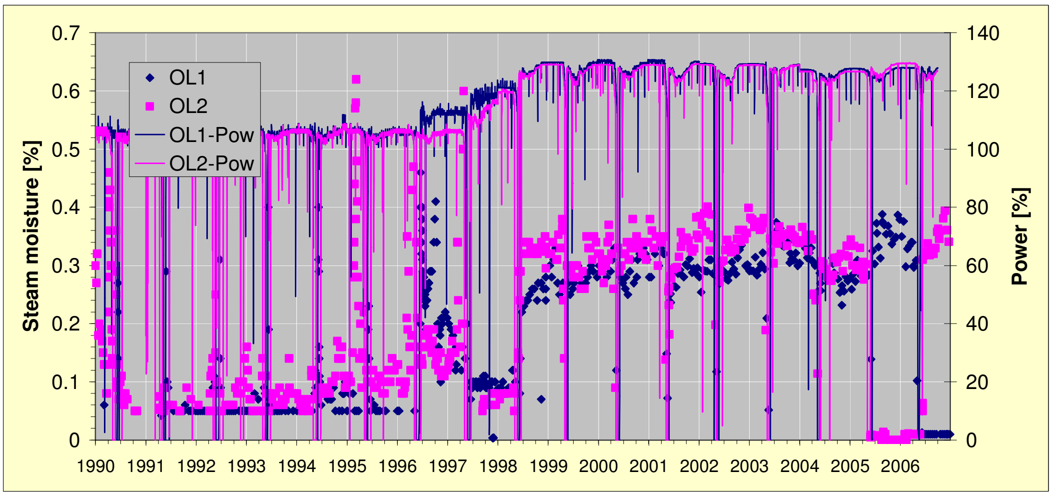


Figure 4.1.16: OL1/2 – Steam moisture content before and after power uprate
New steam dryers: OL1: 2006 - , OL2: 2005-2006

4.1.2.3 Radiation levels

4.1.2.3.1 During operation

The radiation levels during operation is mainly determined by short-lived nuclides such as N-16 ($T_{½}$ = 7.12 s). The production occurs through neutron activation of the ^{16}O isotope in the coolant, which means that the production rate is basically proportional to the thermal reactor power. The N-16 radiation source term in different systems is affected by the production rate, but also by the decay time from the reactor circuit due to the N-16 nuclide's short half-life. Power uprates affect the flow rates, and hence the decay times. The distribution of N-16 between the steam and the reactor water is dramatically affected by injection of hydrogen gas, so called HWC, with an increase of steam line activity to about a factor of five. Operation without hydrogen gas injection, so called NWC, has, however, been maintained in both OL1 and OL2, and any introduction of HWC is not considered.

One way to assess the overall affect of power uprates on radiation levels during operation is to study the occupational exposures obtained during reactor operation. Data from OL1/2 are summarized in **Figure 4.1.17**. It must be noted, that the annual exposure during operation is normally only a small fraction, 10-15%, of the total occupational exposure, and is not always well distributed between the different reactors. The data in **Figure 4.1.17** show, therefore, some variation, but no clear correlation to the 1998 power uprate is seen.

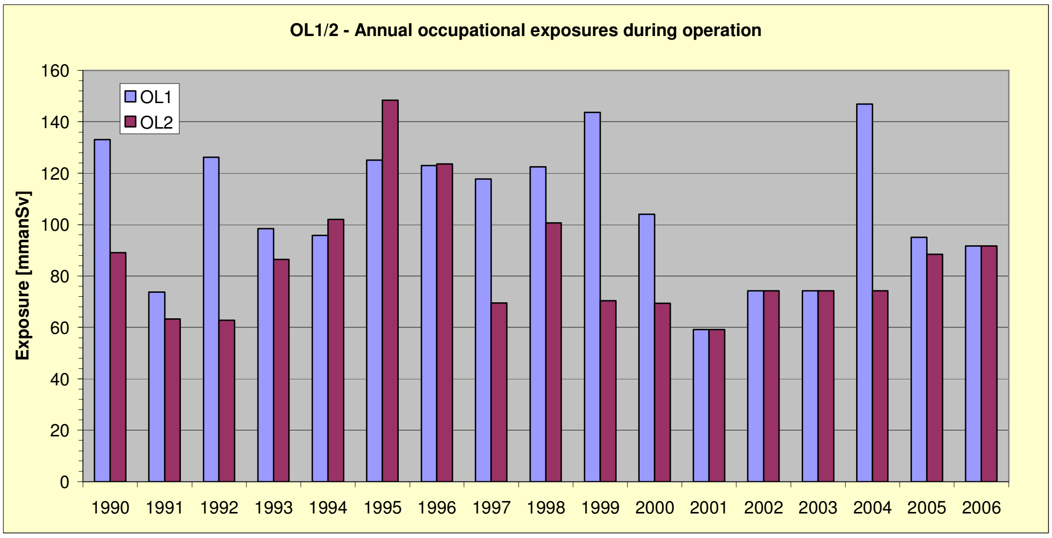


Figure 4.1.17: OL1/2 – Annual occupational exposures during reactor operation before and after power uprate

4.1.2.3.2 During outages

The radiation fields during outage conditions in OL1/2, as well as in other Nordic BWRs, have been regularly followed-up. These measurement campaigns have included both dose rate surveys as well as gamma scanning measurements revealing the contribution from different radionuclides to the radiation fields. The variation of average contribution from different nuclides is shown in **Figure 4.1.18** and **Figure 4.1.19 F** for OL1 and OL2, respectively. Co-60 is by far the most important nuclide, with a contribution of 65-80%. Co-58 is the second most important with a 5-15% contribution. Note especially the difference between OL1 and OL2, which correlates well with the difference in Co-58 reactor water activity (**Figure 4.1.13**). The Mn-54 contribution has been gradually reduced in parallel to a reduced inflow of feedwater iron (**Figure 4.1.7**). The Sb-124 contribution varies, but during some years it reaches about 10%. The contribution from other nuclides (Fe-59, Cr-51, Zr-95/Nb-95, etc.) is normally <10%.

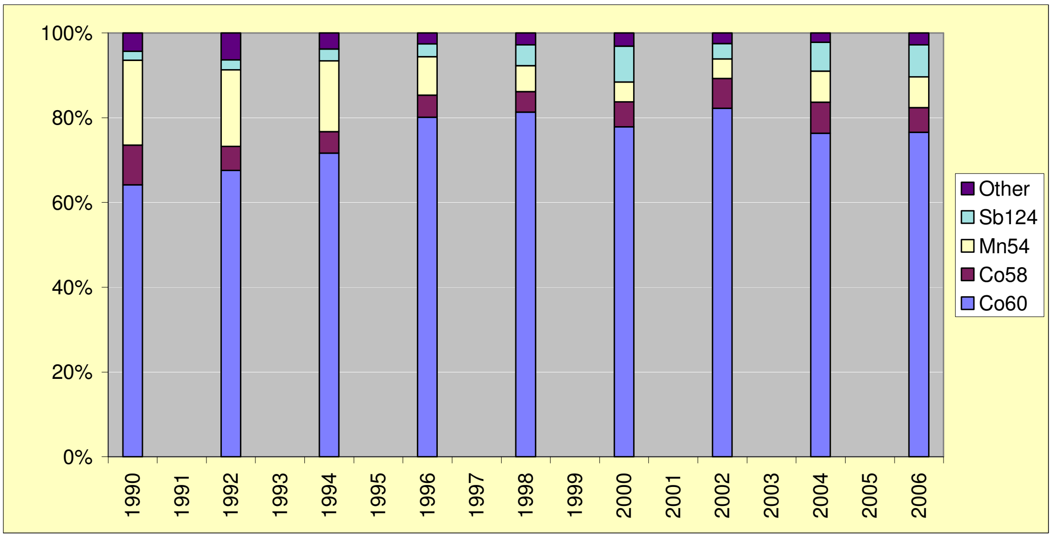


Figure 4.1.18: OL1 – Contribution from different radionuclides to outage radiation levels
Average value of 12 different measurement locations

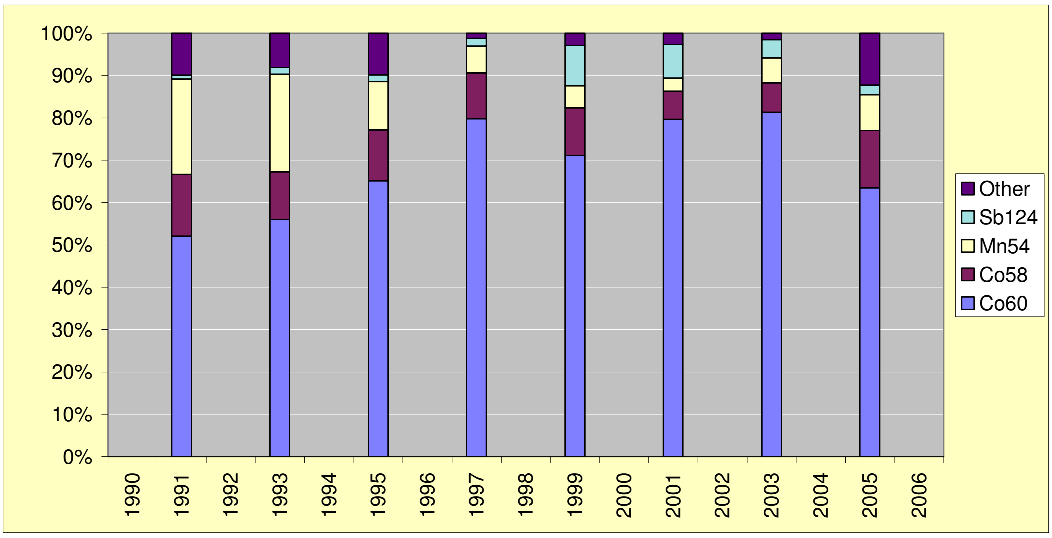


Figure 4.1.19: OL2 – Contribution from different radionuclides to outage radiation levels
Average value of 12 different measurement locations

The OL1/2 gamma scanning data are compared to corresponding data from other Nordic BWRs in **Figure 4.1.20**. The general picture is quite similar in all plants.

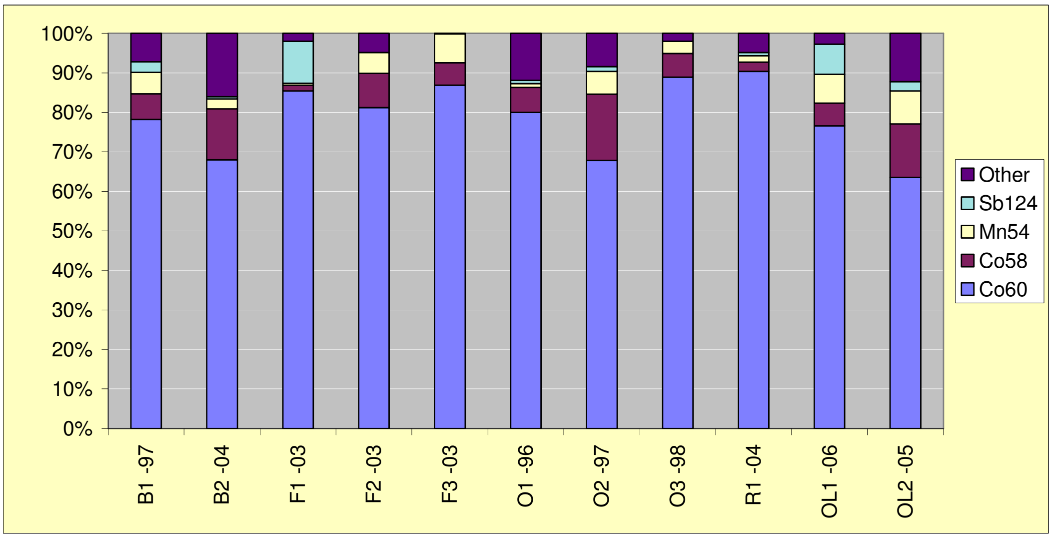


Figure 4.1.20: Contribution from different radionuclides to outage radiation levels
Comparison of 11 Nordic BWRs

Measured dose rates in different locations at outage conditions in OL1/2 during the period 1990 – 2006 are presented below in a series of diagrams. The OL1/2 dose rates are also compared with the database covering all eleven Nordic BWRs (denoted "BWRs", the general database is not updated in the same degree as the OL1/2 data).

Of large importance for occupational exposures are the radiation fields around primary piping containing hot reactor water during operation. Such a standard location in the Nordic BWRs is on a vertical pipe just outside the containment in the RHR system, and measured dose rates are shown in **Figure 4.1.21**. The corresponding gamma scanning results in OL1/2 is shown in **Figure 4.1.22**. The radiation levels display some variation, but OL1/2 has basically remained on the same level as before the 1998 power uprate. The OL1/2 levels are on the lower end of the database. The gamma scanning data do not show any significant change in contribution from different radionuclides.

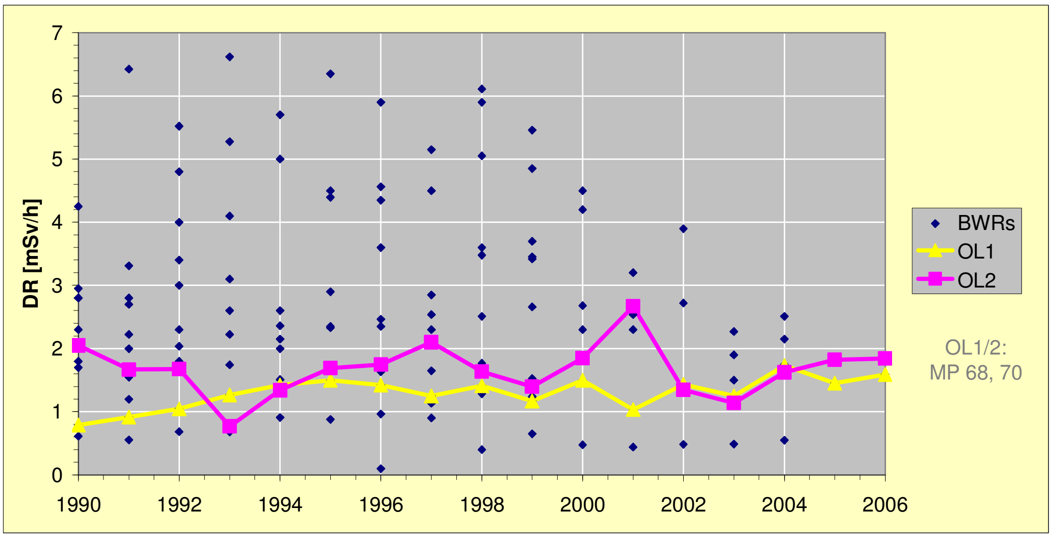


Figure 4.1.21: OL1/2 – Radiation levels on vertical RHR lines compared to database for Nordic BWRs (power uprates in OL1/2 during 1998)

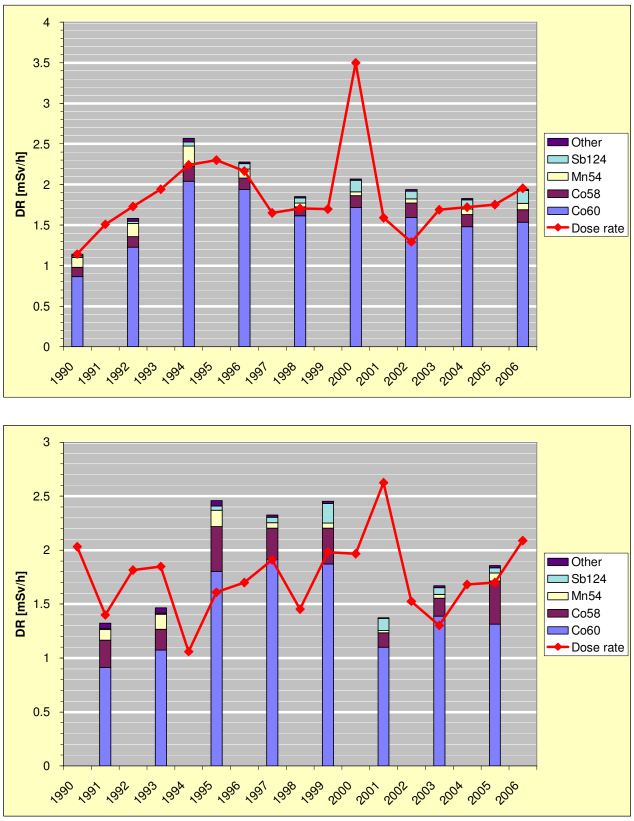


Figure 4.1.22: OL1/2 – Gamma scanning data and radiation levels RHR/RWCU pipes, average for 3 locations (power uprates in OL1/2 during 1998)

The contact dose rates on primary piping are mainly determined by the buildup of an active oxide layer on the inside of the pipe. However, general radiation levels and dose rates on certain components are also affected by the contribution from sedimented crud ("hot spots") in locations especially prone to accumulate particles (stagnant flow, low-points, etc.).

The general radiation levels in rooms for RHR pumps are presented in **Figure 4.1.23**. The corresponding general radiation levels in rooms with piping for the RWCU system are shown in **Figure 4.1.24**. Dose rates on regenerative heat exchangers in the RWCU system are presented in **Figure 4.1.25** and the dose rates on coolers in the RWCU system in **Figure 4.1.26**. The radiation levels on water tanks in the scram system are shown in **Figure 4.1.27**, and the general radiation levels in the area below the motors for the control rod drives in **Figure 4.1.28**. After the 1998 power uprate quite stable radiation levels were experienced in all these locations. Some variations between the plants can be explained by other factors. E.g. the difference in radiation levels below the control rod drives (**Figure 4.1.28**) between OL1 and OL2 during the 90's is explained by the combination of fuel failures with fuel dissolution and some flaking fuel channels in the OL2 plant in the beginning of the 90's (gamma scanning measurements revealed a rather high fraction of Zr-95/Nb-95 activity). Both factors contributed to the release of insoluble fuel crud, and the control rod drives and core instrumentation flanges below the reactor pressure vessel are especially prone to accumulate such activity. This difference between OL1 and OL2 is also seen in some other locations.

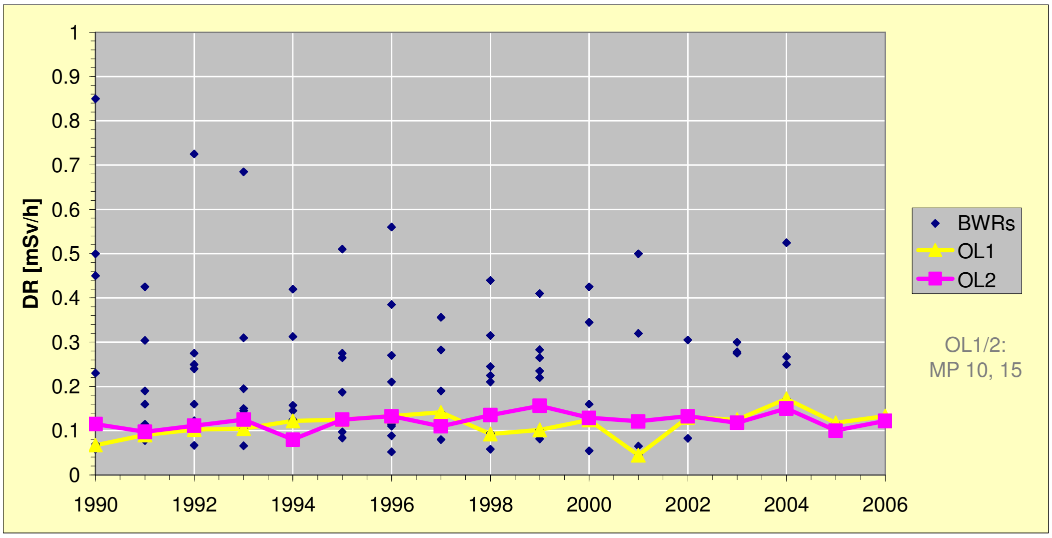


Figure 4.1.23: OL1/2 – General radiation levels in RHR pump rooms compared to databasefor Nordic BWRs (power uprates in OL1/2 during 1998)

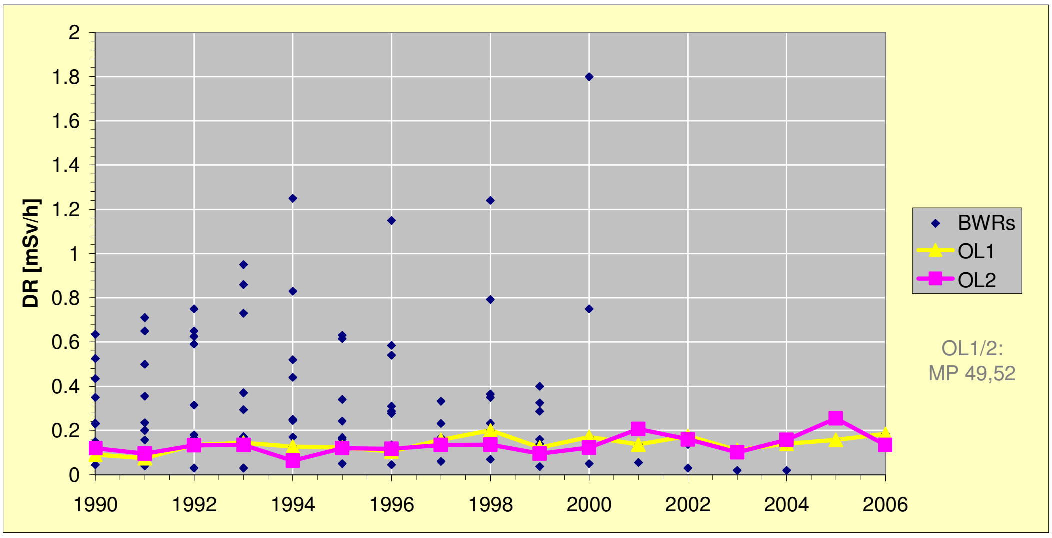


Figure 4.1.24: OL1/2 – General radiation levels in rooms with RWCU piping compared to database for Nordic BWRs (power uprates in OL1/2 during 1998)

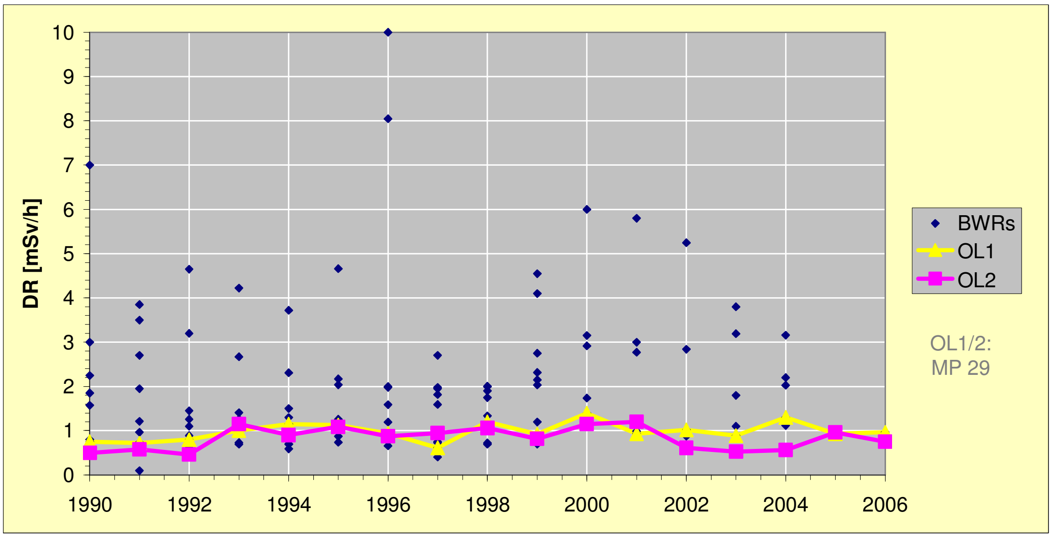


Figure 4.1.25: OL1/2 – Radiation levels on RWCU regenerative heat exchangers compared to database for Nordic BWRs (power uprates in OL1/2 during 1998)

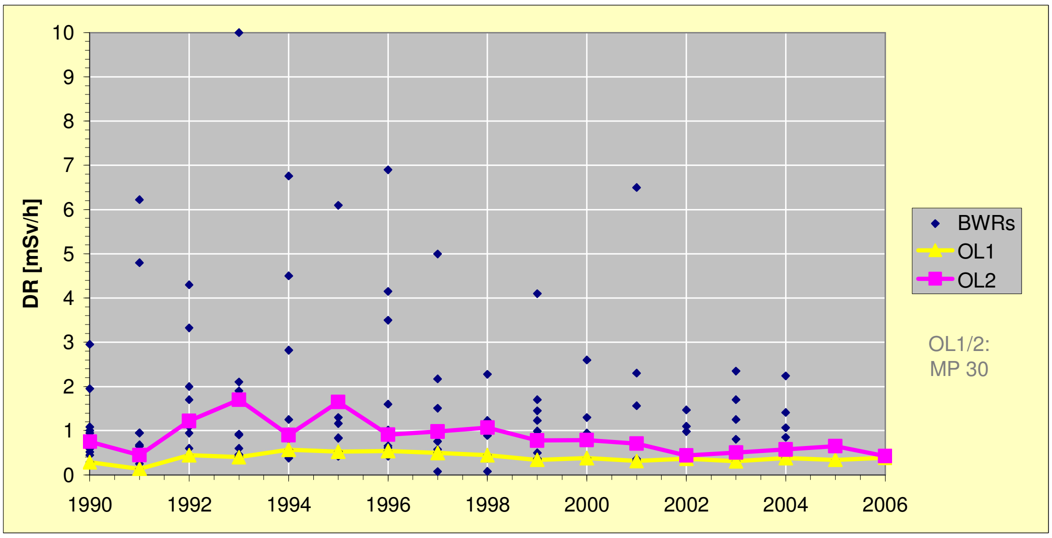


Figure 4.1.26: OL1/2 – Radiation levels on RWCU coolers compared to database for Nordic BWRs (power uprates in OL1/2 during 1998)

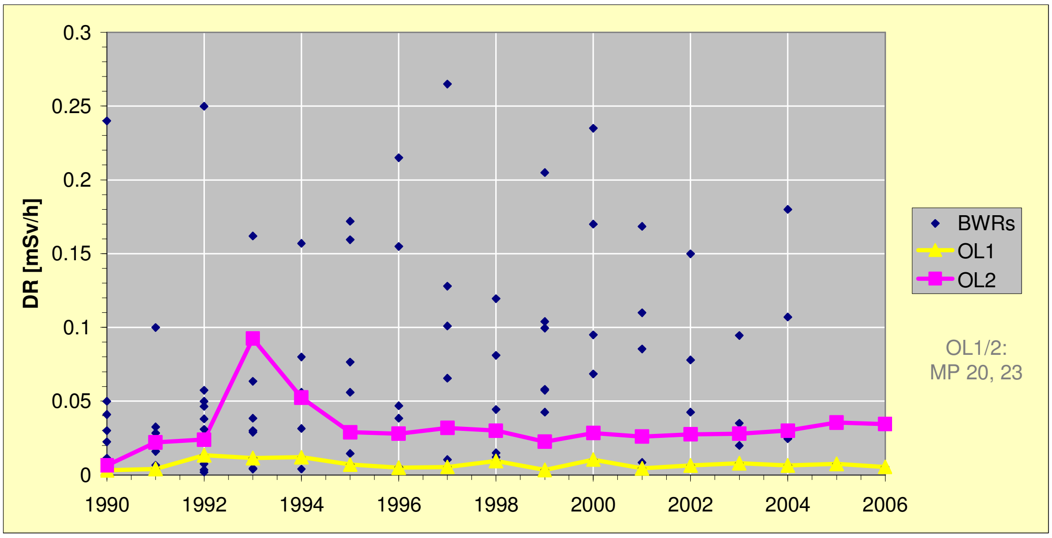


Figure 4.1.27: OL1/2 – Radiation levels on water tanks in scram system compared to database for Nordic BWRs (power uprates in OL1/2 during 1998)

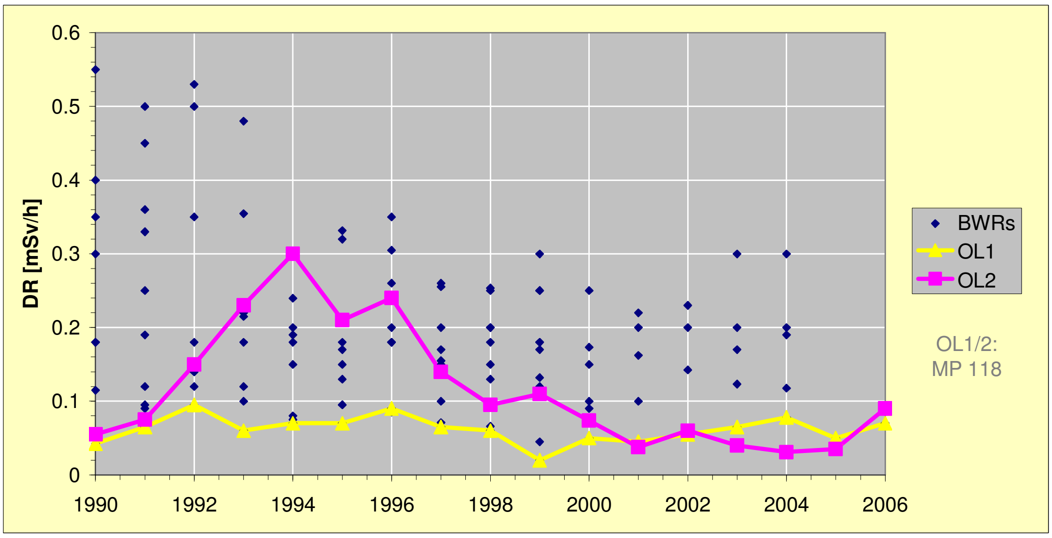


Figure 4.1.28: OL1/2 – General radiation levels below control rod drive motors compared to database for Nordic BWRs (power uprates in OL1/2 during 1998)

A different picture is seen for the radiation levels on main steam lines (**Figure 4.1.29**, **Figure 4.1.30**) and other turbine system components, e.g. the moisture separator / steam reheater (**Figure 4.1.31**). The radiation levels especially on the main steam lines are greatly affected by the steam moisture content (**Figure 4.1.16**), with a considerable increase after the 1998 power uprate when the moisture was enhanced. On the other hand, the reduced moisture content in OL2 during the cycle 2005-06 resulted in a rather quick decrease of radiation fields during only one cycle down to about the pre-1998 levels, and further operation with very low moisture content is expected to result in further reductions. A high contamination level of the turbine plant has a large impact on the occupational exposures, and a reduction of the turbine plant radiation fields is consequently an important ALARA measure.

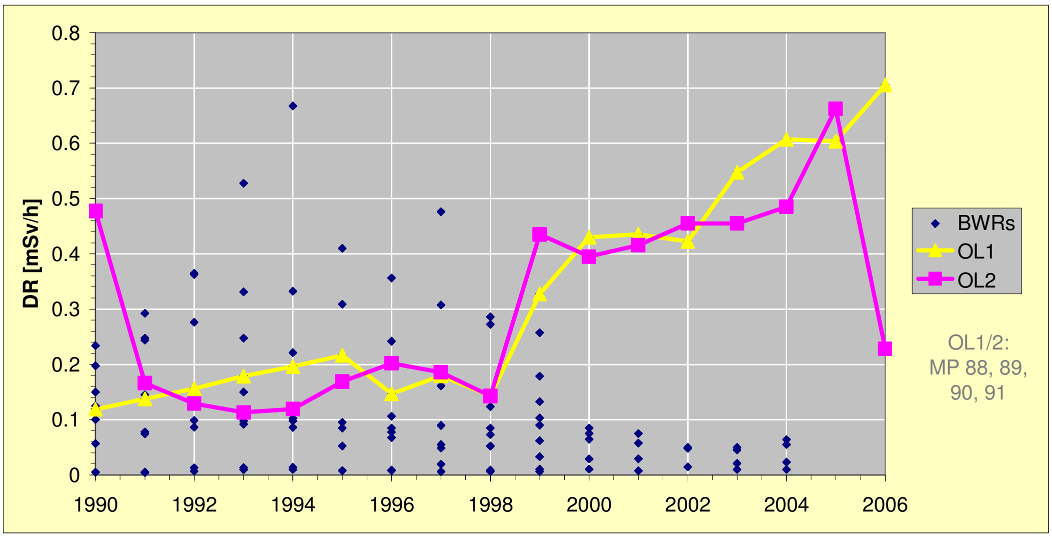
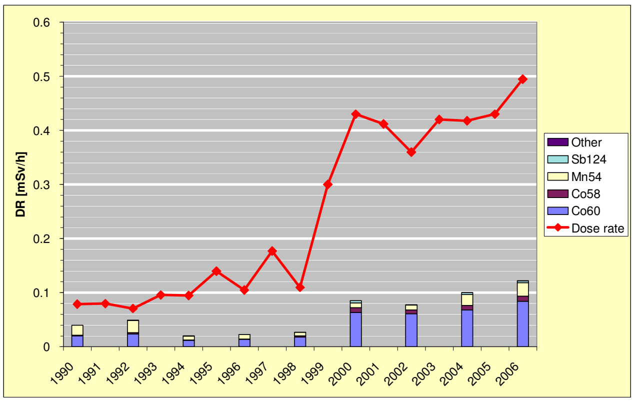


Figure 4.1.29: OL1/2 – Radiation levels on main steam lines outside containment compared to database for Nordic BWRs (power uprates in OL1/2 during 1998)

OL1



OL2

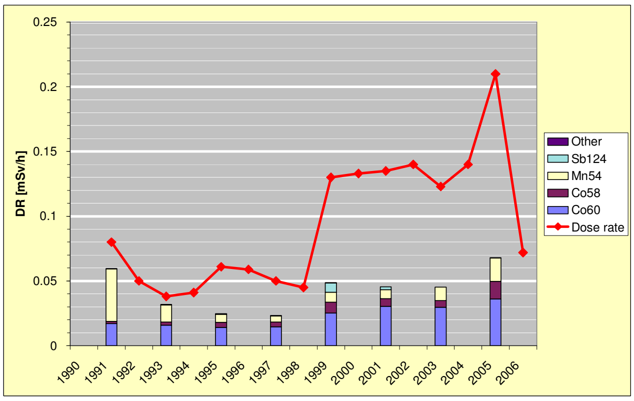


Figure 4.1.30: OL1/2 – Gamma scanning data and radiation level on main steam line (power uprates in OL1/2 during 1998)

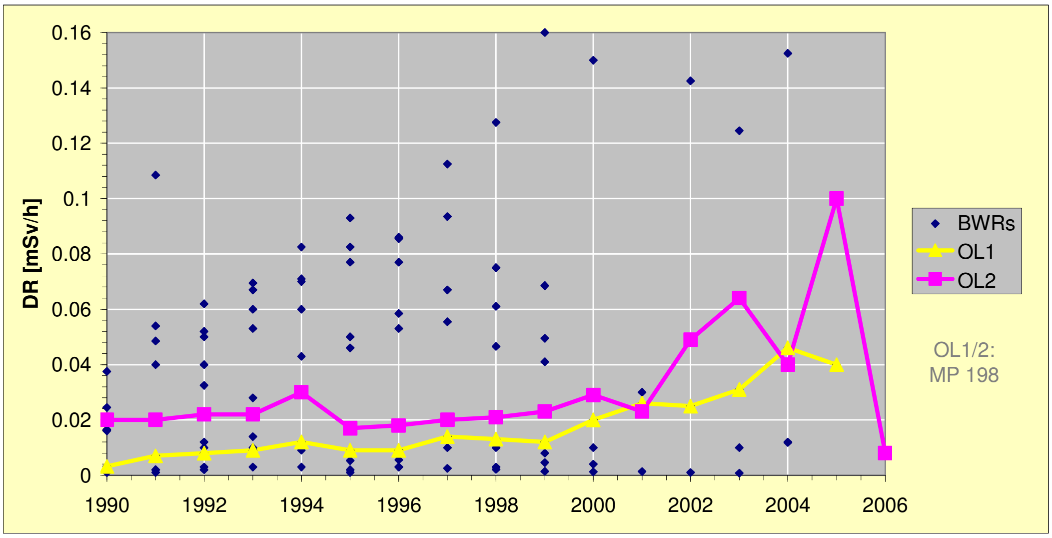


Figure 4.1.31: OL1/2 – Radiation levels on moisture separators / steam reheaters compared to database for Nordic BWRs (power uprates in OL1/2 during 1998)

At the above locations radiation levels are mainly determined by activated corrosion products, especially Co-60, with only a small contribution from fission products. However the radiation levels around the offgas system (**Figure 4.1.32**) are strongly dependent on fission products, especially noble gas daughters such as Ba-140 and La-140. The amount of noble gas daughters are to a large extent determined by the amount of tramp uranium on the core, This is illustrated by comparison between OL1 and OL2 in **Figure 4.1.32.** As discussed, the OL2 plant suffered from fuel failures in the early 90's resulting in significant tramp uranium and a consequent increase in radiation fields around the offgas system. The memory effect of tramp uranium on the core is considerable, and it takes 10 years or more until the radiation levels are restored. During the 90's OL1 had a rather low tramp uranium contamination, and the radiation levels around the offgas system were consequently low. Fuel failures with some tramp uranium contamination occurred around 2000, resulting in an increase of radiation levels. Learning from this, the utilities have introduced operational strategies to restrict the amount of tramp uranium. It is, therefore, unlikely that the OL2 conditions occurring at the beginning of the 90's will be repeated today.

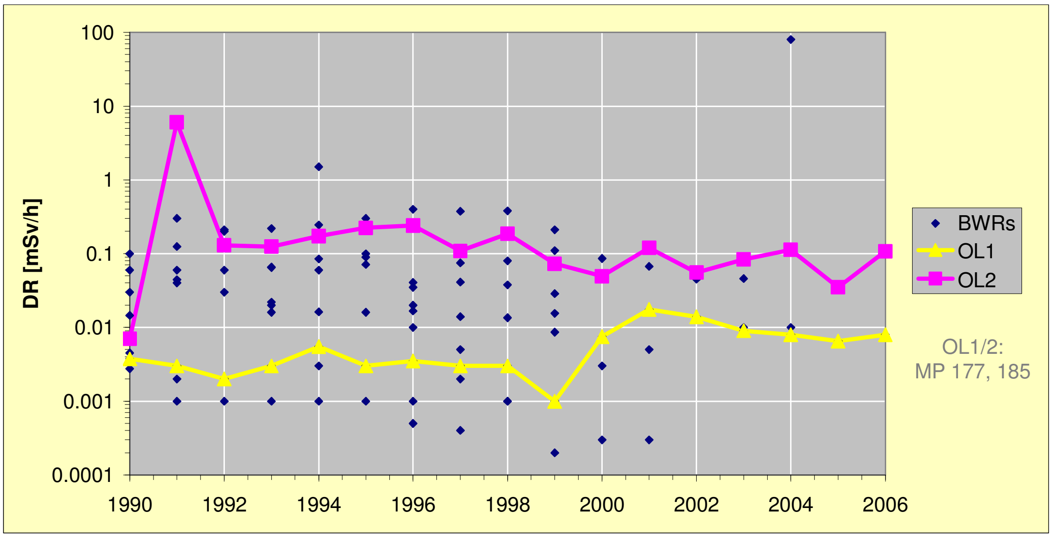
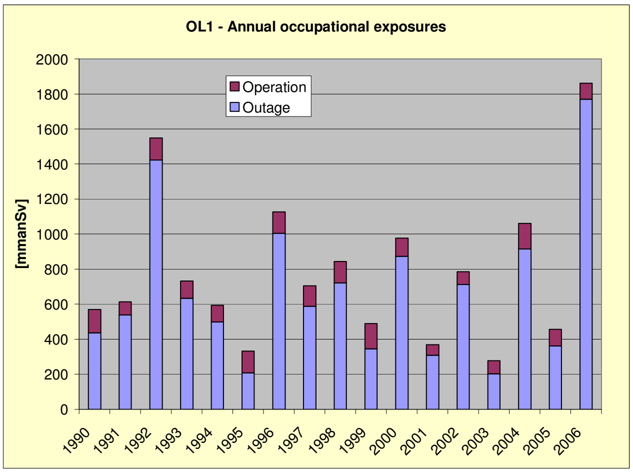


Figure 4.1.32: OL1/2 – Radiation levels on inlet pipe to offgas system compared to database for Nordic BWRs (power uprates in OL1/2 during 1998)

4.1.2.4 Occupational exposures

The annual occupational exposures in OL1 and OL2 during the period 1990 – 2006 are presented in **Figure 4.1.33**. The exposures are divided into contribution during outage and operation conditions. The exposures during operation are only a small fraction of the total exposures, and have been quite constant during the period (**Figure 4.1.17**). A clear correlation with the outage lengths (**Figure 4.1.6**) is seen, i.e. outages involving large efforts for the power uprate and plant modernization projects have consequently implied increased exposure (compare with **Table 4.1.2** and **Table 4.1.3**). Data on exposure per outage day are compiled in **Figure 4.1.34**. The exposure per outage day shows some variation, but no trend of increasing exposure after the 1998 power uprate is seen.



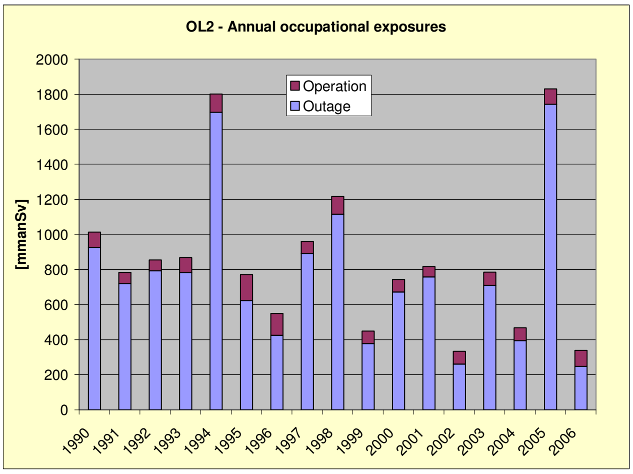


Figure 4.1.33: OL1/2 – Annual occupational exposures showing proportion due to outage and operation conditions

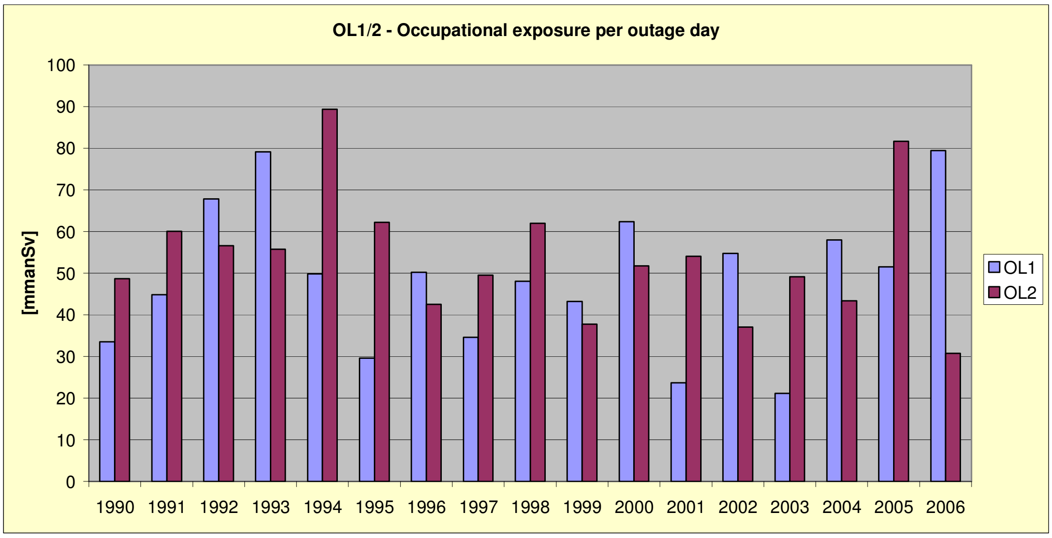


Figure 4.1.34: OL1/2 – Occupational exposures per outage day

The OL1/2 annual exposures are compared to some international BWR data in **Figure 4.1.35** The OL1/2 exposures are rather low compared to the average international BWR. Only years with considerable efforts in modernization projects, e.g. 2005 in OL2 and 2006 in OL1, result in exposures comparable with the average US and Japanese plants.

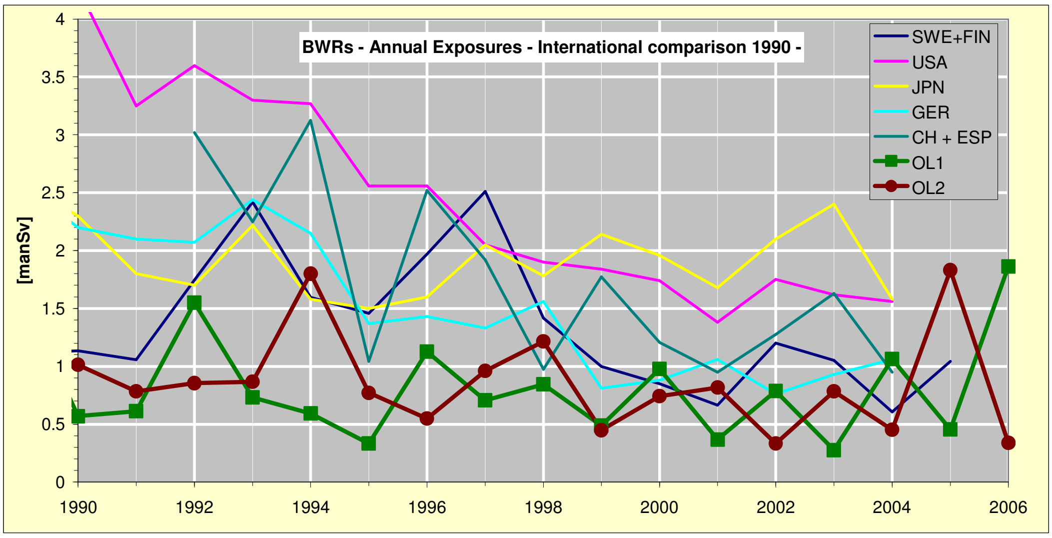


Figure 4.1.35: OL1/2 – Annual occupational exposures compared to international data

4.1.2.5 Summary and conclusions

The data and experience from the OL1/2 plants before and after the 1998 power uprate has been been reviewed and resulted in the following conclusions:

- The plants have been uprated twice since the commissioning. The thermal power of each reactor was increased from 2000 MW to 2160 MW in 1984 and to 2500

MW in 1998. The present study has focused on the second uprate, resulting in a thermal power level of 125% compared to the initial power level.

- The 1998 uprate was part of an extensive modernization program implemented in 1994–2006. After the modernization, the plants fulfil most of the safety and technical requirements for new nuclear power plants.
- It was possible to implement the modernisation program, with a reasonable impact on outage lengths, due to good planning (maximum annual outage length 22 days compared to typically 7-14 days).
- The water chemistry is characterized by “as clean as possible”. Standard normal water chemistry (NWC) without hydrogen injection is maintained and no zinc is injected. The condensate polishing plant is operated to meet the power uprate conditions e.g. a reduction of operation temperature. The flow rate in the RWCU system has been increased to continue to maintain the 2% of the maximum feedwater flow capacity after the power uprate. The overall result is that favourable water chemistry conditions have been maintained, or improved, after the power uprate.
- Some water chemistry transients have, however, been related to major replacement of reactor components, e.g. the steam dryers. The most dominant specie in the transients is chromium. The transients are restricted to the first cycle after the replacement.
- Co source reduction was considered in the modernization, e.g. replacement of Stellite in valves had been considered in cases of valve modification. An initial Stellite area of about 94 dm^2 in reactor system valves per plant has been reduced by about 25% in OL1 and about 35% in OL2. Although the turbine plant contains approximately 190 dm^2 Stellite per unit downstream of the condensate cleanup system the reduction of these Stellite areas has so far been limited. A moderate overall reduction of Co inflow to the primary circuit seems to be indicated.
- The exposures during operation have been maintained at a constant and low level since the uprate, even if an unavoidable increase of the source term of short-lived N-16 activity has occurred. One important factor is that the plants have been maintained on NWC, i.e. the increase of N-16 main steam line radiation connected to HWC operation has been avoided.
- Radiation levels during outage of reactor systems have been maintained at a low and constant level since the power uprate. Conversly radiation levels around main steam lines and other turbine components have increased considerably after the uprate in spite of the installation of new steam separators in both plants in 1997. The reason for the increase is a considerable rise in steam moisture content after the uprate. The problem has been addressed by the installation of new, recently designed steam dryers in the reactor pressure vessel. As a result, steam moisture and radiation levels have been considerably reduced.
- The large manhour efforts expended during some outages for power uprate and plant modernization program have resulted in increased occupational exposures. However, the exposure per outage day has been maintained at a fairly constant level. In spite of the large efforts for power uprate and plant modernization, the

average annual exposures in the OL1/2 plants have been kept on a rather low level compared to international BWR data.

4.2 Cofrentes

4.2.1 Introduction

The second of the BWR plants selected for detailed anlyses was Cofrentes (CNC). The main reasons for this selection were:

- A considerable power uprate of 12% compared to the initial thermal power level.
- Water chemistry conditions including hydrogen injection (HWC) and the use of depleted zinc injection (DZO). Both HWC and DZO have great impact on the radiological situation, and are applied in some of the Swedish BWRs.
- A good availability of reactor data.

The following sections present the results of the indepth review performed for the CNC plant. Data for the review has been supplied by the Iberdrola utility owner and operator of the plants. The utility is greatly acknowledged for supplying the large amount of information.

4.2.2 CNC power uprate

4.2.2.1 Power uprate characteristics

Cofrentes nuclear station (CNC) is a single unit BWR, located in Spain, on the province of Valencia, by the Jucar River. It is 100% owned by Iberdrola, which is the main private utility in Spain. The plant design is General Electric BWR-6, with Mark III containment, and belongs to the group of post-TMI plants, and had already incorporated the lessons learned from TMI into the original design.

The CNC plant has 624 fuel assemblies in the core, had an original rated power of 2894 MWt, and is currently operating at 3237 MWt (111.85%). Commercial operation started in 1984 with 12-month cycles at rated power. Both cycle length and thermal power have been increased. Power has been uprated in several steps:

1. 2952 MWt (102%) in cycle 4 (Apr-88)
2. 3015 MWt (104.2%) in cycle 11 (Mar-98)
3. 3184 MWt (110%) in cycle in cycle 14 (June-02)
4. 3237 MWt (111.85%) in cycle 15 (Oct-03), by taking credit of lower feedwater flow uncertainties.

There are some tentative plans to bring the plant up to about 120%.

Fuel cycle length increased to 18 month in cycle 7, cycle 15 length was 20 months, cycle 16 is planned to 22 months and cycle 17 will be a 24 months cycle. Fuel assembly design has evolved from the original 8x8 array to 9x9 in cycle 8 and 10x10 in cycle 12, which has advanced designs suitable for more demanding operation conditions. This evolution has resulted in mixed cores with bundles from the same supplier up to cycle 11, and from different suppliers starting in cycle 12. The present power level means an average of 5.19 MWt per fuel assembly.

The variation of relative thermal power level 1996 – 2006 is presented in **Figure 4.2.1**.

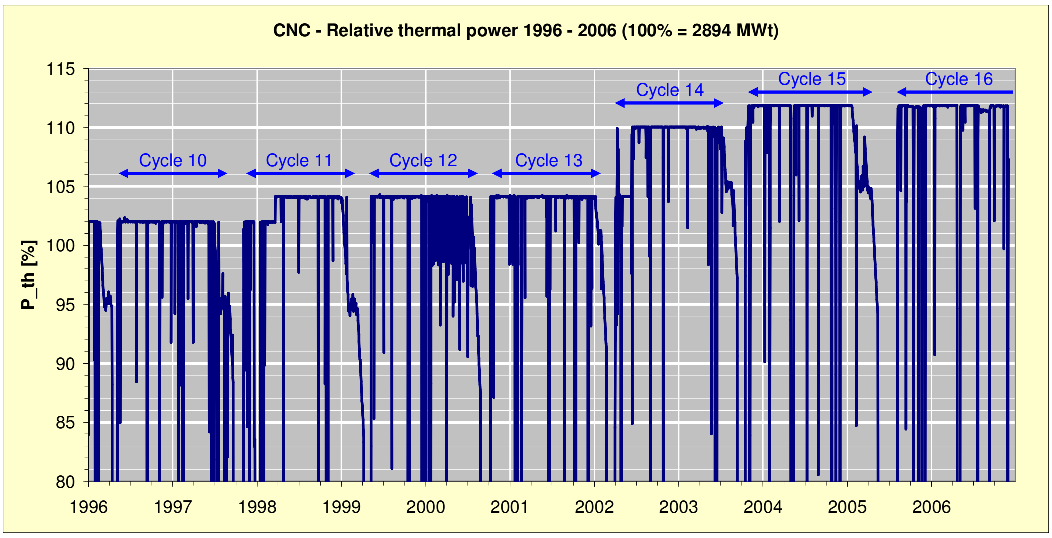


Figure 4.2.1: CNC – Relative reactor power with thermal output (100% = 2894 MWt)

The design of the recirculation system is shown in **Figure 4.2.2**. The plant has two external recirculation loops located at the bottom region of the reactor building, inside the containment. Each recirculation loop consists of a centrifugal pump, a flow control valve, two shutoff valves and jetpumps. The jetpumps are located in the downcomer inside the reactor pressure vessel (RPV), and are used to drive the circulating core coolant flow. The use of jet pumps in the recirculation system has the advantage that only about 1/3 of the core flow has to be withdrawn outside the reactor vessel. The suction lines of the recirculation pumps are attached to the lower part of the RPV and take water from the downward flow in the bottom of the downcomer. The water is then pumped and distributed through a manifold to which a number of riser pipes are connected and returned to the reactor through the jetpumps. The recirculation loop pumps are of vertical mechanical seal type. Cooling of the pump seal and the pump motor is provided by a closed cooling water system.

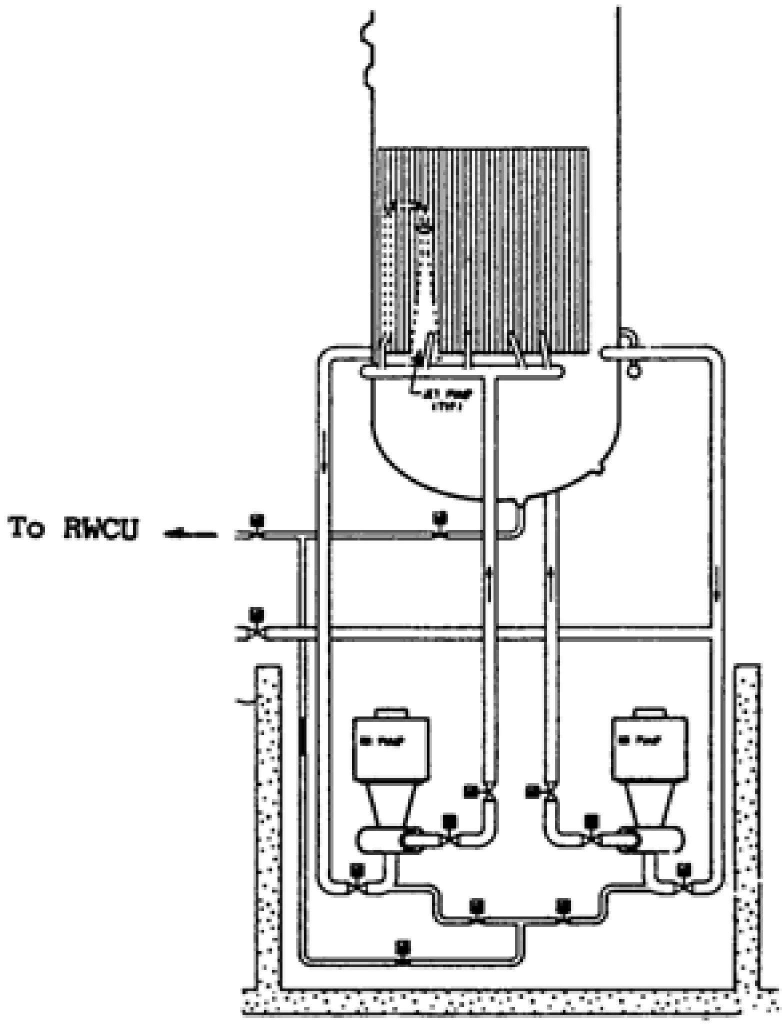


Figure 4.2.2: CNC – Reactor recirculation system with two pumps (B33)

The reactor water cleanup (RWCU) system at CNC is operated during startup, shutdown, and refuelling operation as well as during normal plant operation. A flow sheet of the RWCU system is shown in **Figure 4.2.3**. The main components in the system are:

– Two parallel pumps receiving hot reactor water from the recirculation loops and the bottom of the RPV. The pumps have mechanical seals.

– One cooling line, consisting of three regenerative heat exchangers and two coolers.

– Two parallel filter demineralizers. Original design flow was 2x125 gpm (2x8.0 kg/s), later increased to 2x150 gpm (2x9.5 kg/s).

The RWCU system takes out a water flow from both of the recirculation suction lines, and from the bottom of the RPV. The RWCU lines are made of carbon steel, as opposed to the recirculation lines of stainless steel. The water is cooled through the regenerative heat exchangers and coolers before passing the filter demineralizer units. The cleaned water is returning to the reactor circuit via the feedwater lines. The maximum cleanup flow corresponds to about 1% of the maximum feedwater flow, and the filters are operated at a temperature of about 120°F (50°C). The filter demineralizers are of the precoat type, and the backwashing frequency is typically once per two weeks. Resin traps are installed both upstream and downstream of the demineralizers.

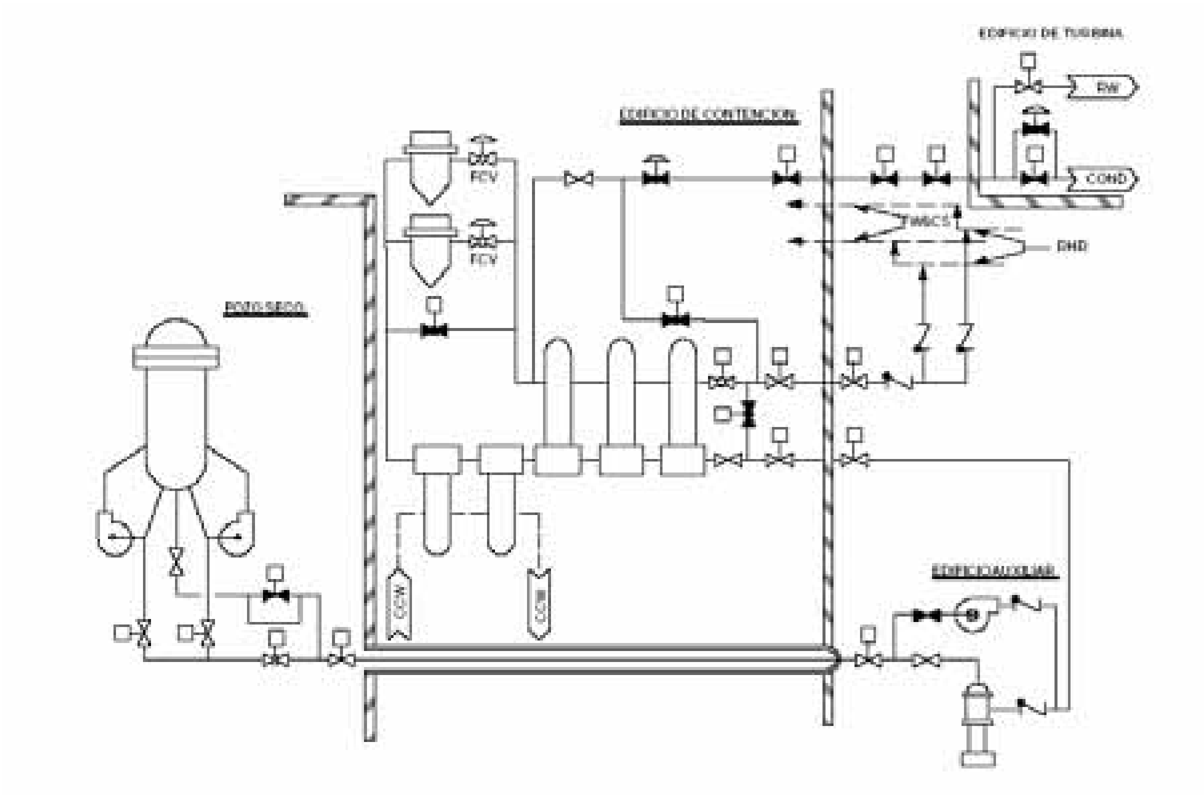


Figure 4.2.3: CNC – Simplified RWCU scheme

The concentration of corrosion products in the final feedwater is of great importance for the activity buildup in BWR plants. This concentration is greatly determined by the design and materials selection of the turbine plant.

A flow chart showing the main turbine design at CNC is shown in **Figure 4.2.4**. The design is of the type “forward pumped heater drains” (FPHD), which means that the high pressure turbine drains are not passing the condensate cleanup system. The high pressure drains at CNC amount to about 30% of the total feedwater flow. Two parallel SALA High Gradient Magnetic Filters are installed in order to remove some iron from the forward pumped drains.

The condensate cleanup (CCU) system in CNC consists of five precoat filter demineralizers. The operation temperature of the CCU system is in the range 50-55°C.

The condenser tubes at CNC were originally of Admirality brass, but in RFO-12 (September 2000) were replaced with titanium tubes. The steam extraction lines were originally of carbon steel, which accounted for a rather high erosion corrosion rate and, thus, elevated levels of iron in the forward pumped drains (somewhat reduced through the SALA filters). An extensive replacement of the steam extraction pipes to low-alloy steel has been carried out, which has dramatically reduced the forward pumped heater drains contribution of feedwater iron. As in most other BWRs, Stellite is present in several valve applications both in the reactor and turbine system. The control rods in CNC were originally of the type with "pins and rollers" of Stellite. A gradual replacement of old control rods has been performed, and the new rods are without Stellite.

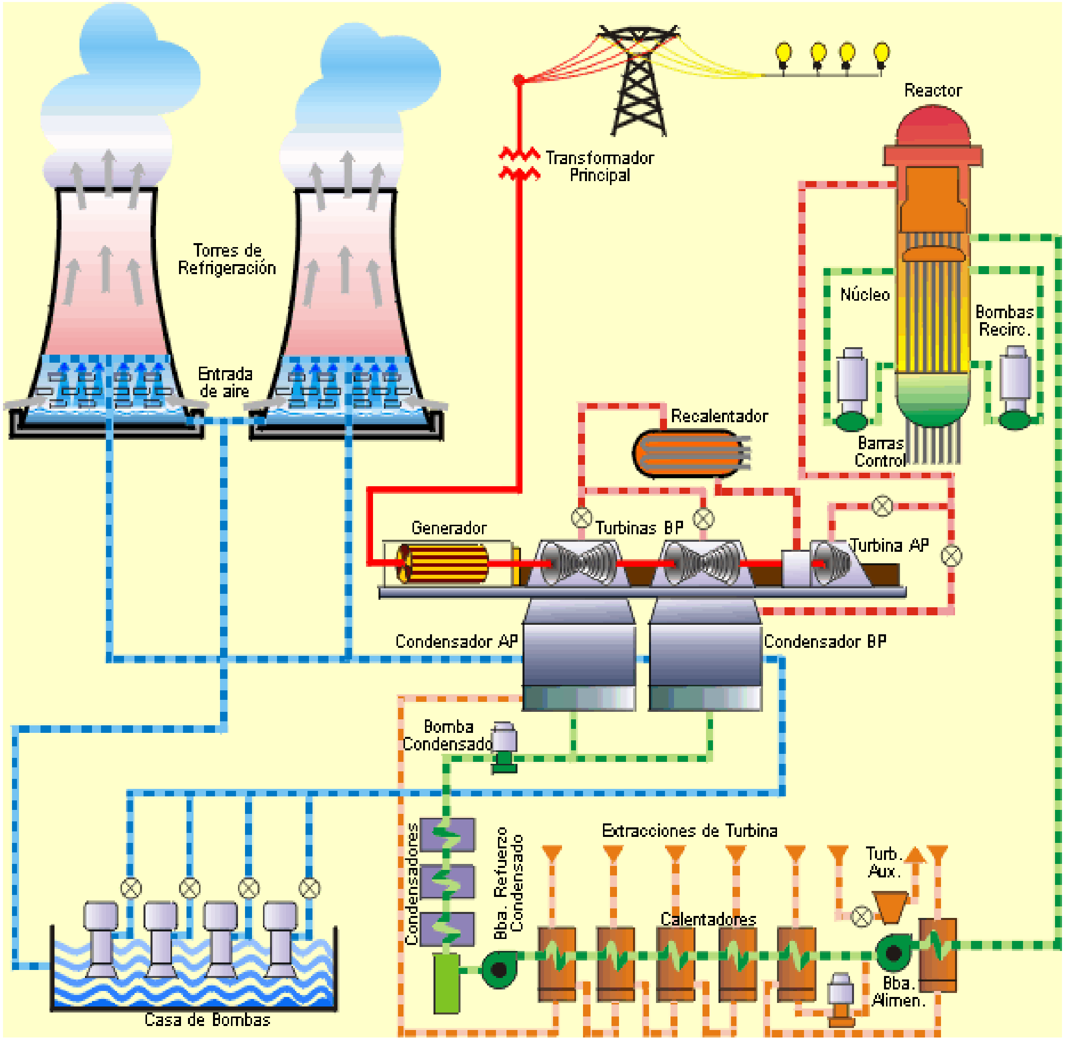


Figure 4.2.4: CNC – Main balance of plant flow diagram

The main power uprate of the plant was in 2002, but the necessary plant system and operation modifications have been carried out over approximately 10 years:

- July 1996: Start injection of zinc (depleted zinc oxide – DZO). The feedwater already contained earlier significant amounts of zinc due to the brass condenser tubes (further discussed below).
- March 1997: Start operation of hydrogen injection to obtain Hydrogen Water Chemistry (HWC) conditions. However, the initial effect of the hydrogen injection on corrosion potential (ECP) was limited, due to considerable amount of reactor water copper.
- September 1997 (RFO-10): Turbine moisture separator and reheater tube sheets were replaced.

– September 2000 (RFO-12): Condenser revamping to titanium. The reason for the modification was twofold; (1) reduction of the reactor water copper and (2) an increase of the condenser capacity to allow the future power uprates.

– March 2002 (RFO-13): Low pressure turbine steam paths modified for power uprate. Heater drain pumps replaced for power uprate. Feedwater pump turbines modified for power uprate. Feedwater flow measuring system type LEFM installed. Modification in one of the condensate filter vessels to install pleated filter elements. Partial decontamination of the RWCU system and the lower sections of the recirculation loops (the original planned closed loop decontamination could not be performed due to problems with the recirculation suction nozzle plugs).

– September 2003 (RFO-14): High pressure turbine steam path modifications for power uprate. Heater drains flow control valves modified for power uprate.

– May 2005 (RFO-15): Successful decontamination of recirculation loops, RWCU and RHR system (for dose control, not directly for power uprate). Replacement of RWCU heat exchangers and RWCU pump internals (not directly for power uprate). Steam dryer inspection resulted in two cracks being found in banks cover. Weld repair had to be done in the reactor pool by divers. The defects had not affected the moisture content, but the defect could be related to flow induced vibrations and may, therefore, be related to the earlier power uprates.

The power uprates in CNC has included a gradual increase of reactor pressure (**Figure 4.2.5**) and reactor water temperature (**Figure 4.2.6**). The increase of reactor pressure has implied that the steam velocity in the top of the RPV and in the main steam lines has not been affected very much by the power uprate. This is a difference when compared to the Scandinavian BWRs, where power uprates have been carried out with maintained reactor pressure level, i.e. the steam velocity has been increased in the same proportion as the power uprate.

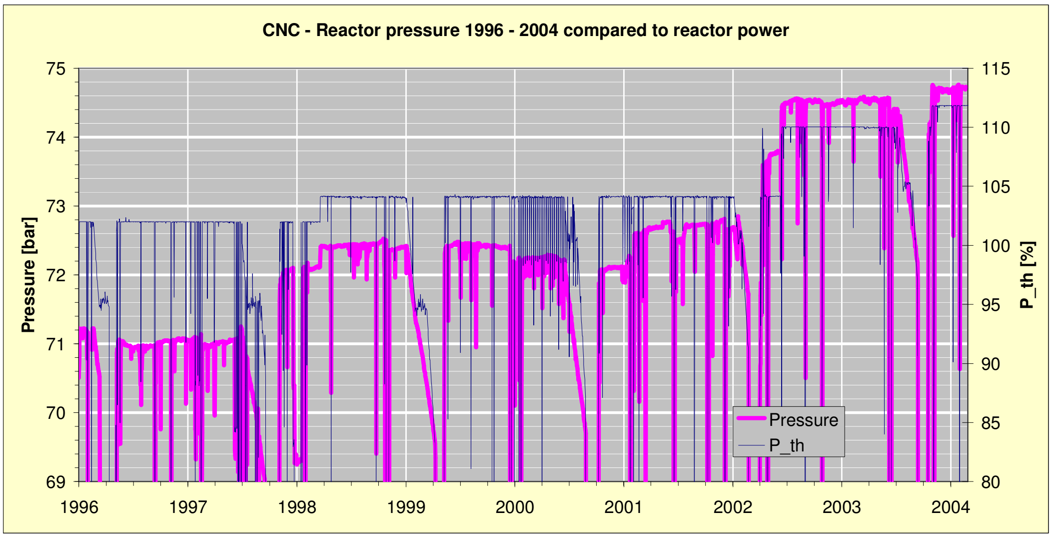


Figure 4.2.5: CNC – Reactor pressure 1996 - 2004

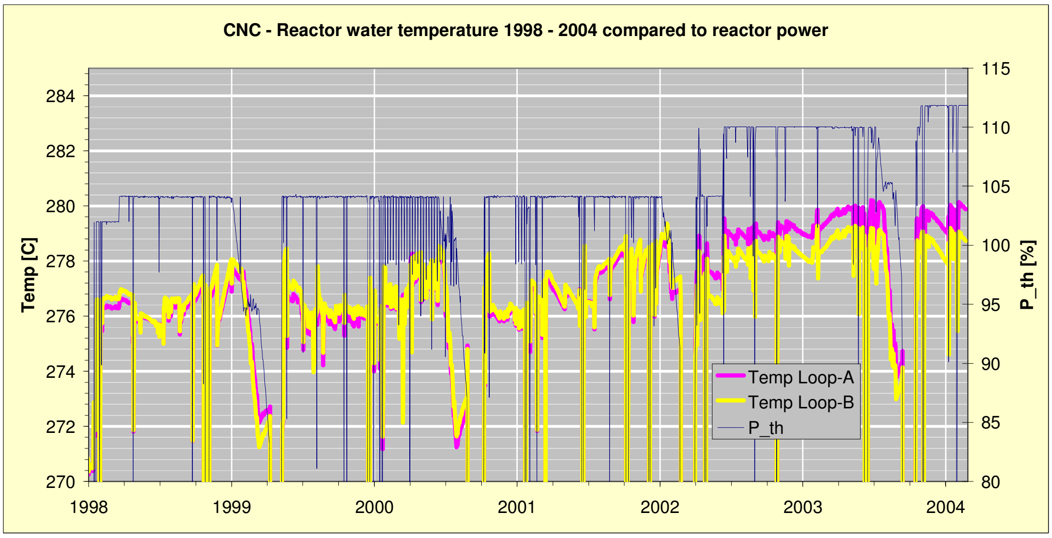


Figure 4.2.6: CNC – Temperature in recirculation loops (A and B) 1998 - 2004

The evolution of refuelling outage lengths is shown in **Figure 4.2.7**. When compared to earlier outages, the 2002 RFO-13, where most modifications for the power uprate were introduced, does not actually differ in length. Note that the extra long outage in 2005 was affected by an external side corrosion problem that occured in the CRD piping in the vessel pedestal penetrations that caused the extension of the RFO-15 for more than 30 days and gave some extra exposure.

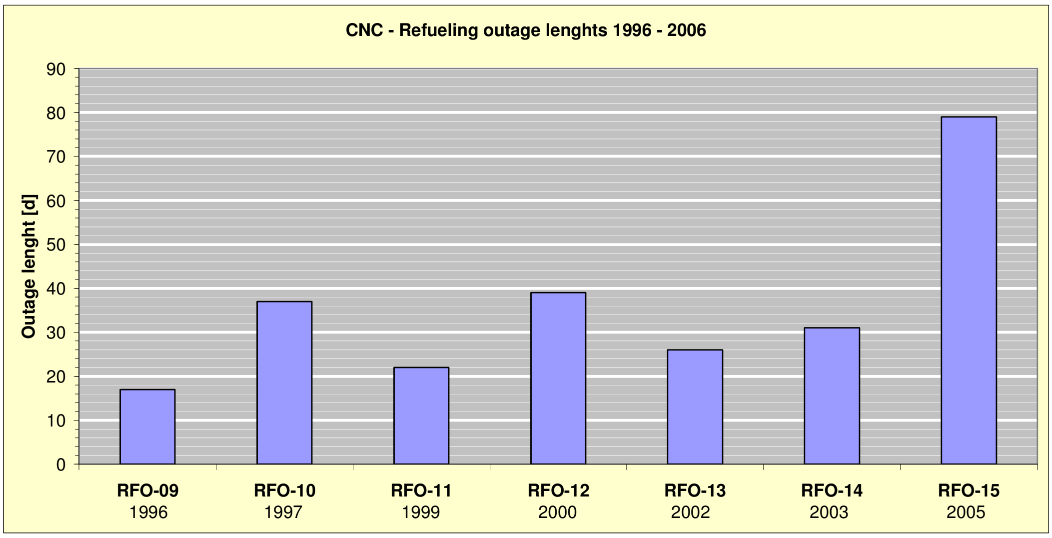


Figure 4.2.7: CNC – Outage lengths 1996 - 2006

4.2.2.2 Water chemistry and radiochemistry

The CNC feedwater chemistry is characterised by periodically high levels of iron (**Figure 4.2.8**), copper (**Figure 4.2.9**) and zinc (**Figure 4.2.10**). The amount of iron shows some variation, but the average level has not been significantly changed during the last 10 years when power uprates have occurred. The earlier quite high contribution from the forward pumped heater drains has been dramatically reduced by the replacement of carbon steel steam extraction lines with low-alloy steel pipes. However, this decrease has been compensated for by an increase of iron inflow to the condensate polishing plant (10-15 ppb Fe in the mid-90's, presently increased to 30-35 ppb Fe), and today's feedwater iron is mainly determined by the leakage through the filter demineralizers. Efforts have been made to increase the cleanup efficiency of the condensate polishing plant, but the optimization of the polishing plant is a balance between cleanup efficiency and long run-lengths to obtain low waste production. The present level of feedwater iron is somewhat high compared to recent water chemistry guidelines[4].

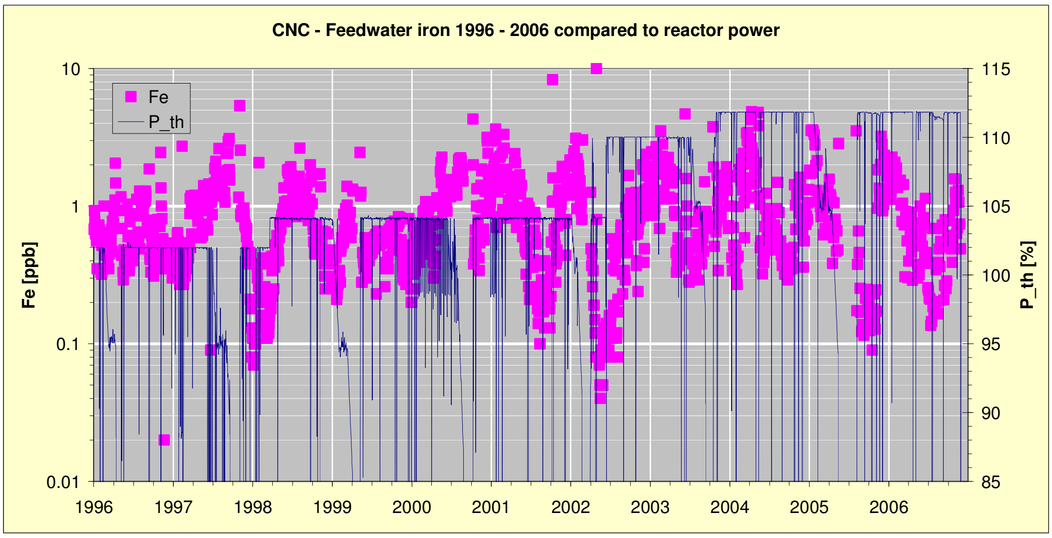


Figure 4.2.8: CNC – Feedwater iron before and after power uprate

The feedwater copper historically has been rather high, but a considerable decrease is seen after the September 2000 condenser tube replacement of brass tubes to titanium. The reduction of feedwater zinc due to the change of condenser tubes is met by an increase of injected zinc, i.e. the feedwater zinc level has basically been maintained or slightly has been increased, but the natural zinc from the brass condenser tubes has been replaced with DZO with low level of ^{64}Zn[5]. The injection of feedwater zinc is controlled so that a target level of reactor water zinc is maintained.

4 EPRI water chemistry guidelines, 2004 revision, specifies 0.1-1.0 ppb feedwater iron for plants on HWC.

5 DZO contains about 1% of ^{64}Zn, compared to about 50% in natural zinc. Neutron irradiation of ^{64}Zn results in production of the radioactive nuclide ^{65}Zn.

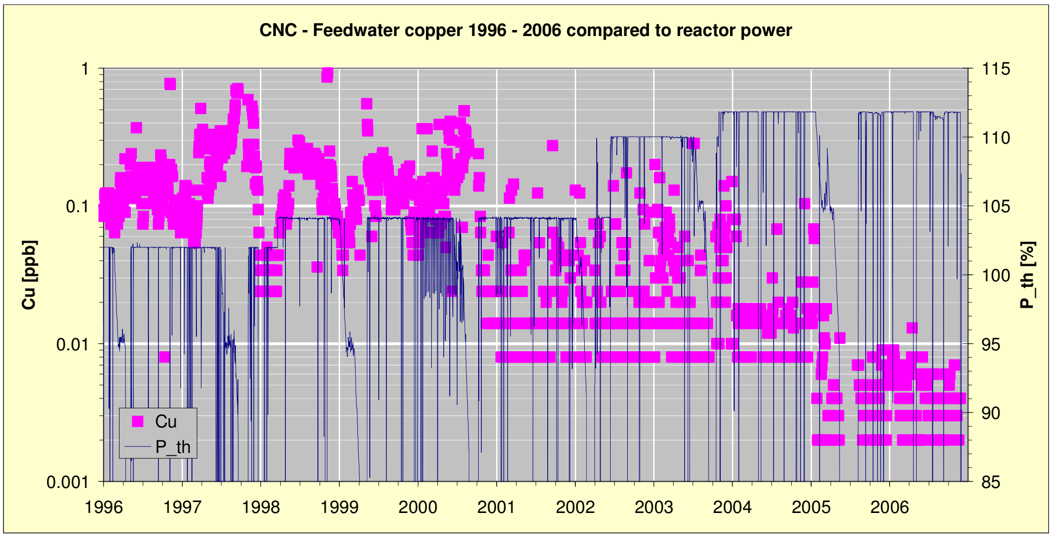


Figure 4.2.9: CNC – Feedwater copper before and after power uprate

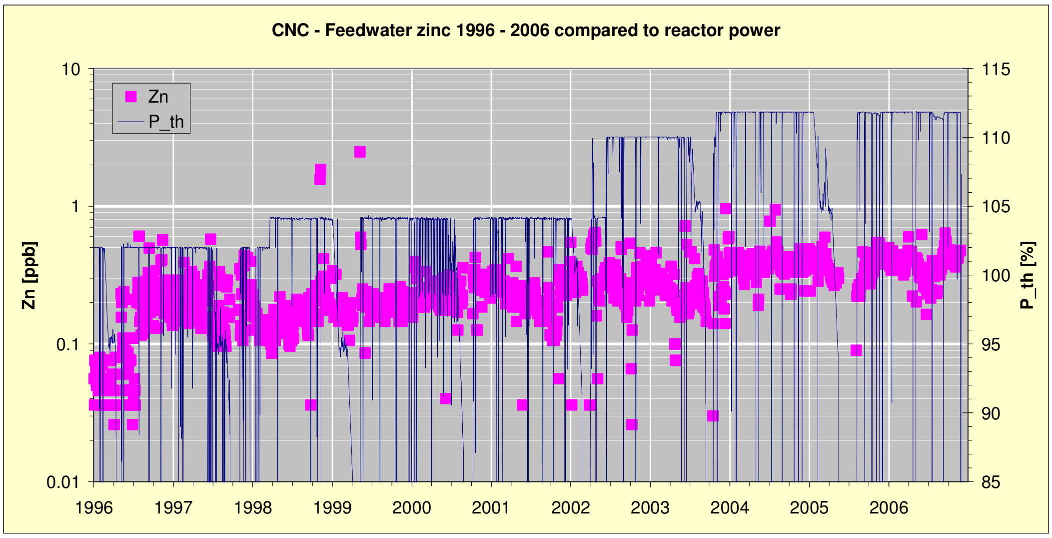


Figure 4.2.10: CNC – Feedwater zinc before and after power uprate

The feedwater dissolved oxygen (**Figure 4.2.11**) is control within the range 20-200 ppb, where the erosion-corrosion of carbon steel is maintained at a reasonable level, and the risk of environmental cracking is low. Hydrogen injection started in March 1997, but, in the present study, feedwater dissolved hydrogen (DH) data has only been available from the end of 1998 (**Figure 4.2.12**). During most of the period with hydrogen injection, the feedwater DH has been kept at 1 ppm, but during 2000 and in a recent fuel cycle it was increased to about 1.67 ppm. The combination 1 ppm of feedwater DH and significant remaining amounts of copper in the reactor water has probably not resulted in a considerable decrease of ECP in the recirculation loops, and especially not for the reactor internals. The gradual decrease of reactor water copper, particularly when the hydrogen injection was increased during the recent cycle meant that ECP levels were sufficiently low for protection of the reactor materials and has probably been obtained both in the recirculation lines and the bottom of the RPV.

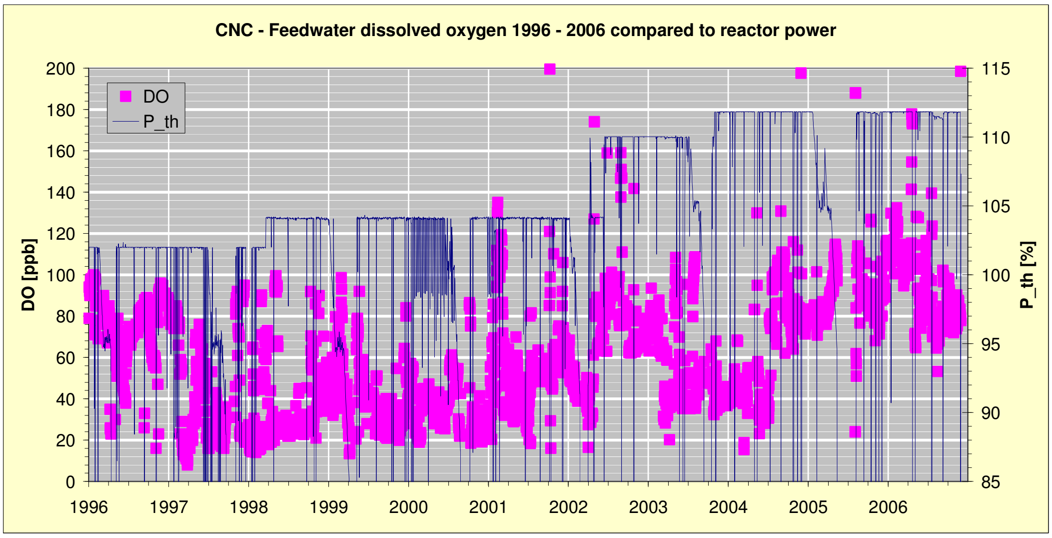


Figure 4.2.11: CNC – Feedwater dissolved oxygen before and after power uprate

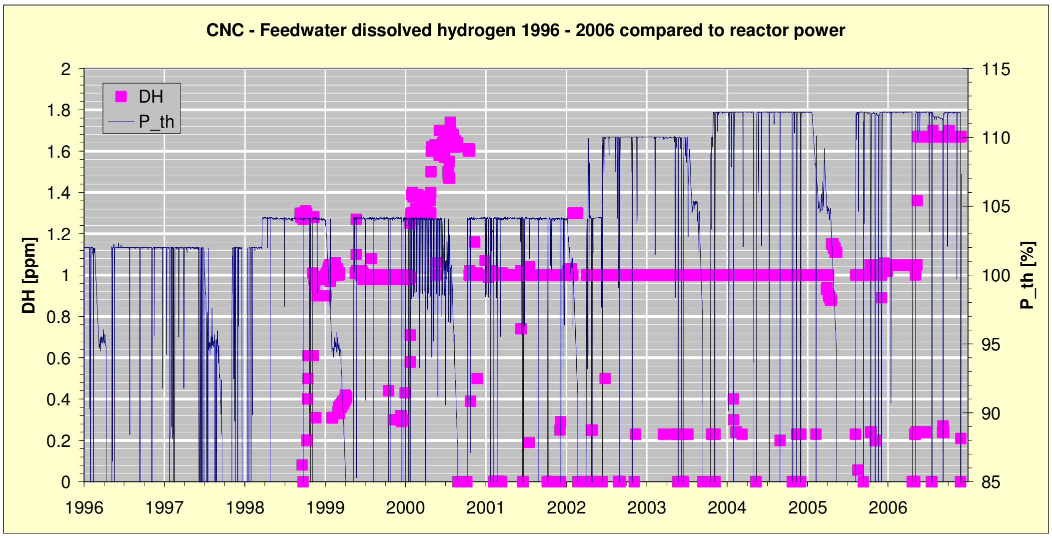


Figure 4.2.12: CNC – Feedwater dissolved hydrogen (injected) before and after power uprate (Note: HWC operation started already March 1997, see **Figure 4.2.14**)

As mentioned, the CCU system is of importance to maintain good water chemistry, but ironically, is also the major source of one of the most detrimental impurities in the reactor water from materials integrity point of view, sulphate. The source of the sulphate is mainly degradation of ion exchange resin, probably from the CCU system. There are three important factors that affect the rate of degradation of CCU resin:

1. The temperature, a power uprate normally results in an increased temperature, which is detrimental.

2. The content of hydrogen peroxide (H_2O_2) in the condensate. Carry-over with the steam and decomposition of the H_2O_2 are affected by a power uprate, normally an increase of H_2O_2 is experienced.

3. The total loading of iron in the resin. The higher flow rates in connection with uprates normally imply increased loading of iron on the resin. The higher loading of iron may demand more frequent backflushing of the CCU filters, which on the other hand results in an increased production of radwaste.

To a large extent this type of resin degradation has been avoided in the CNC plant, mainly through the relatively low temperature in the CCU plant. Low levels of anionic impurities in the reactor water are exhibited by low reactor water conductivity (**Figure 4.2.13**).

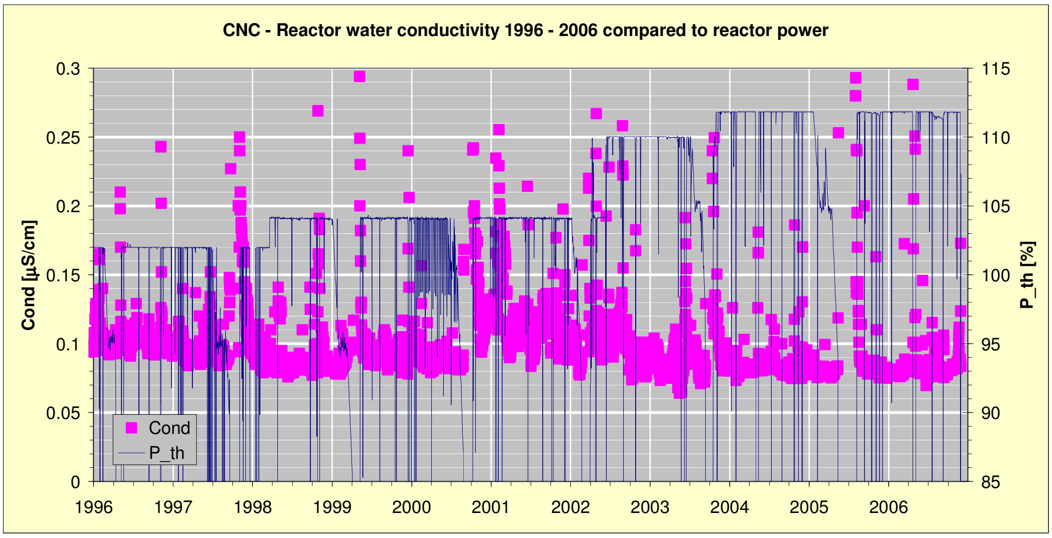


Figure 4.2.13: CNC – Reactor water conductivity before and after power uprate

Measured reactor water dissolved oxygen (DO) is presented in **Figure 4.2.14**. Low ECP levels demand DO levels ≤1 ppb, and it is only during the recent cycle that sufficiently low oxygen levels have been achieved. However, it must, be noted that the measurement of very low oxygen levels may be disturbed by small intrusion of air, e.g. if the measuring equipment is connected with plastic tubing.

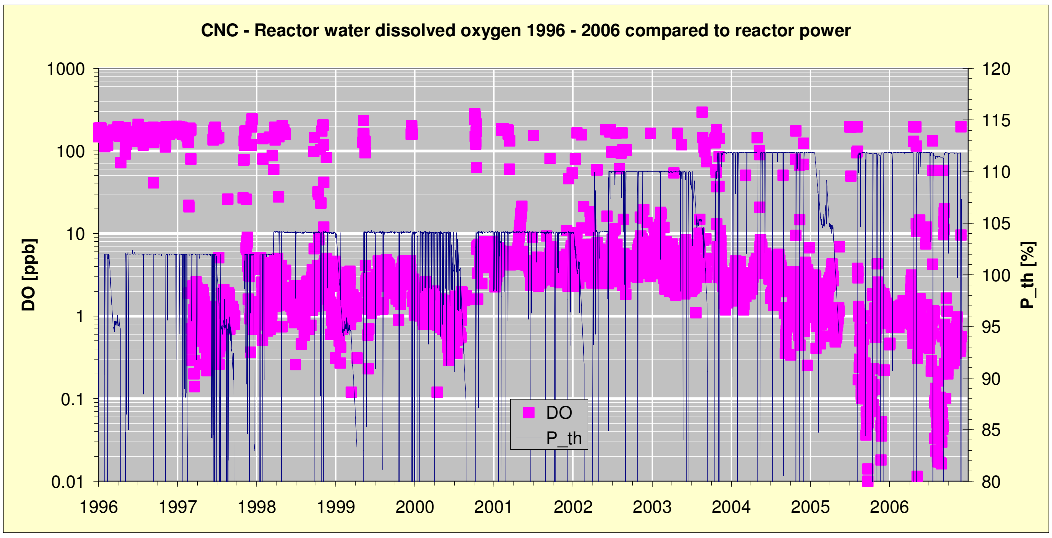


Figure 4.2.14: CNC – Reactor water dissolved oxygen before and after power uprate

Some important corrosion products, copper, zinc and cobalt, in the reactor water are shown in **Figure 4.2.15**, **Figure 4.2.16** and **Figure 4.2.17**, respectively. The reactor water copper is gradually reduced after the condenser retubing, but it takes some years until sub-ppb levels are reached. A further delayed effect of the previous copper chemistry is that system oxides contain some remaining copper. The successful decontamination campaign in 2005 eliminated this effect for the recirculation loops.

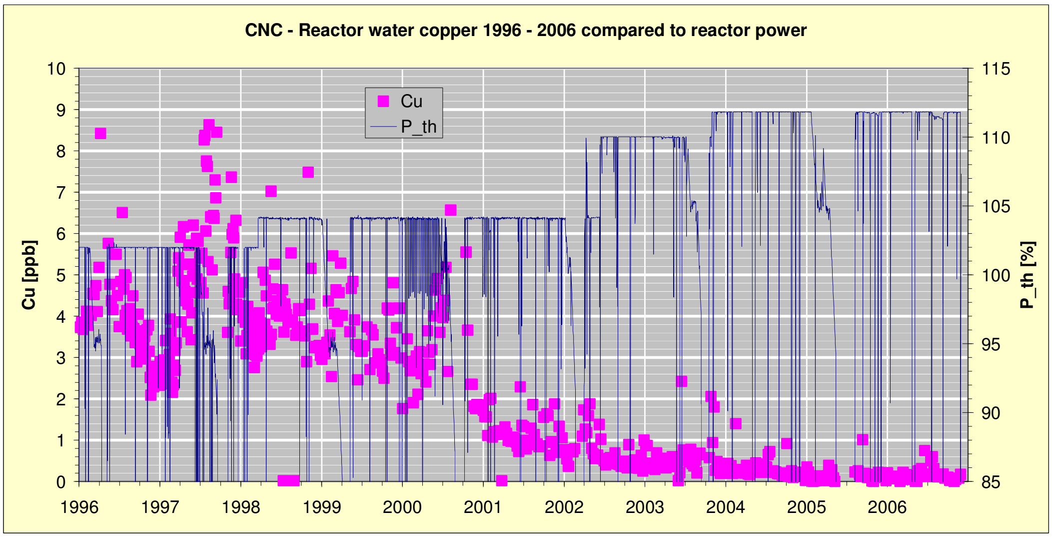


Figure 4.2.15: CNC – Reactor water copper before and after power uprate
(cfr. Feedwater Cu, see **Figure 4.2.9**)

The reactor water zinc was maintained at typically ≤5 ppb before the 2003 RFO, but increased to ≥5 ppb after this outage. The reason for this change was the increased radiation level experienced during the 2003 RFO, and on recommendation issued by EPRI.

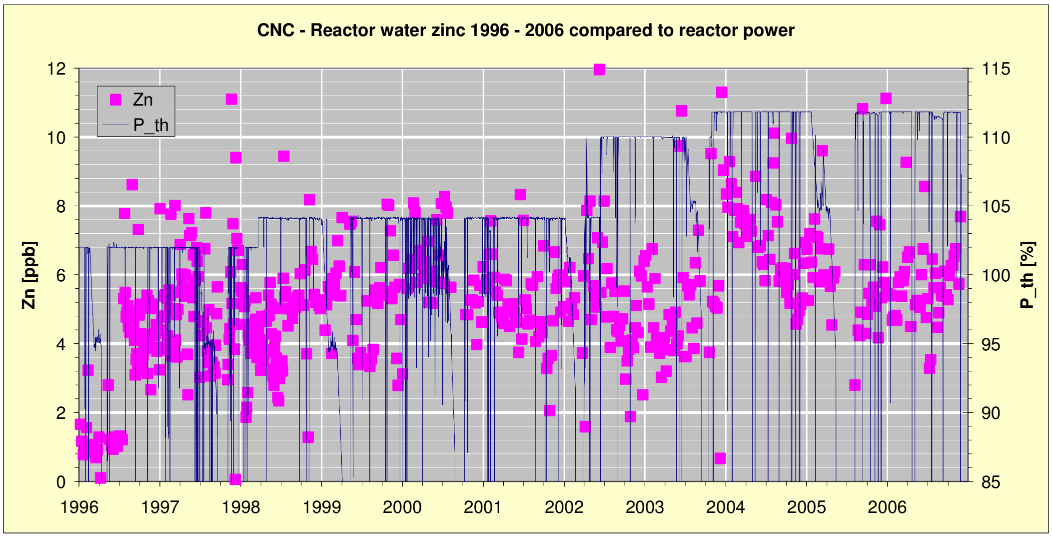


Figure 4.2.16: CNC – Reactor water zinc before and after power uprate
(cfr. Feedwater Zn, see **Figure 4.2.10**)

The measured reactor water cobalt has been biased by cobalt containing Stellite in the sampling system valves. Improved cobalt analysing procedures were introduced in 2003, and the sampling valves were replaced with Stellite-free in 2005, and more realistic reactor water cobalt concentrations, about 0.02 ppb are recorded.

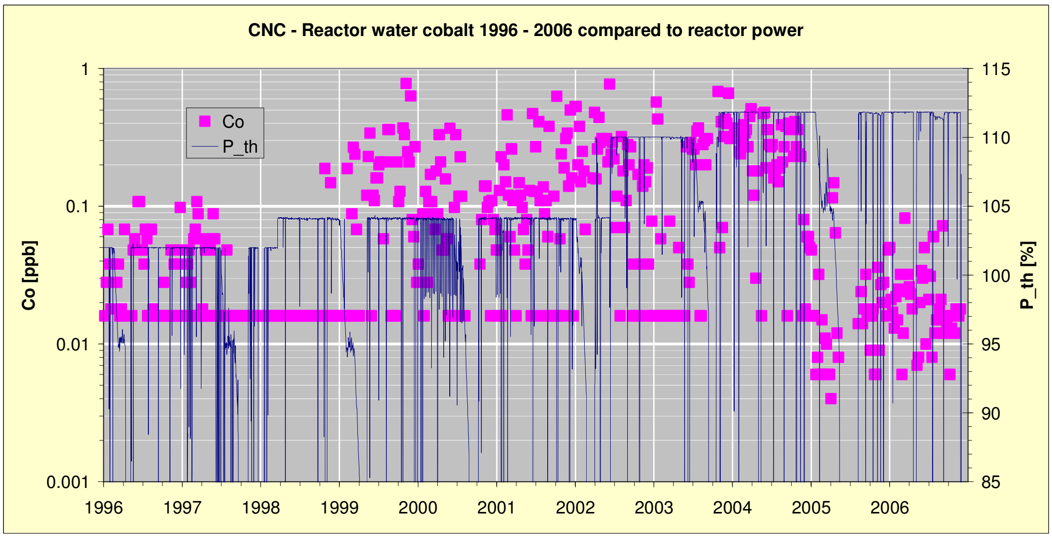


Figure 4.2.17: CNC – Reactor water cobalt before and after power uprate

Radiation levels in a BWR plant is very much determined by the activated corrosion products, e.g. Co-60, and measured reactor water concentrations of some of these are presented in **Figure 4.2.18**- **Figure 4.2.21**, and are commented below:

– Co-60: The most important activated corrosion product, being responsible for the dominant fraction of the occupational exposures in the BWR plants. The Co-60 levels in CNC show some variations, but these variations are not easily correlated to the power uprates. The general trend in CNC is gradually decreasing reactor water Co-60 concentration.

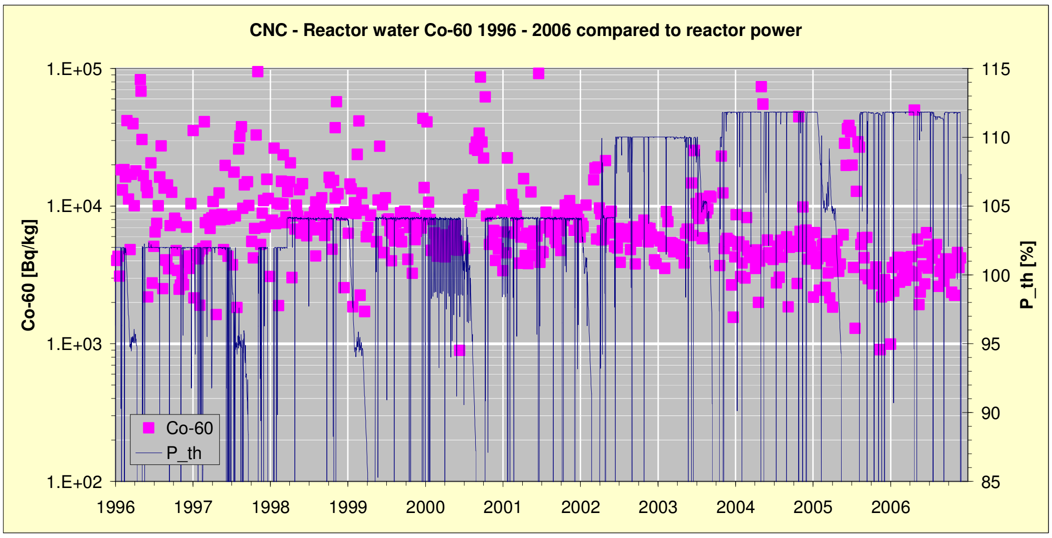


Figure 4.2.18: CNC – Reactor water Co-60 before and after power uprate

– Co-58 is due to neutron activation of Ni, and the reactor water concentration is very much determined by the surface area of in-core Inconel in the reactor, i.e. Inconel in fuel spacers. The CNC Co-58 levels have been maintained at a fairly constant level during the last 10 years.

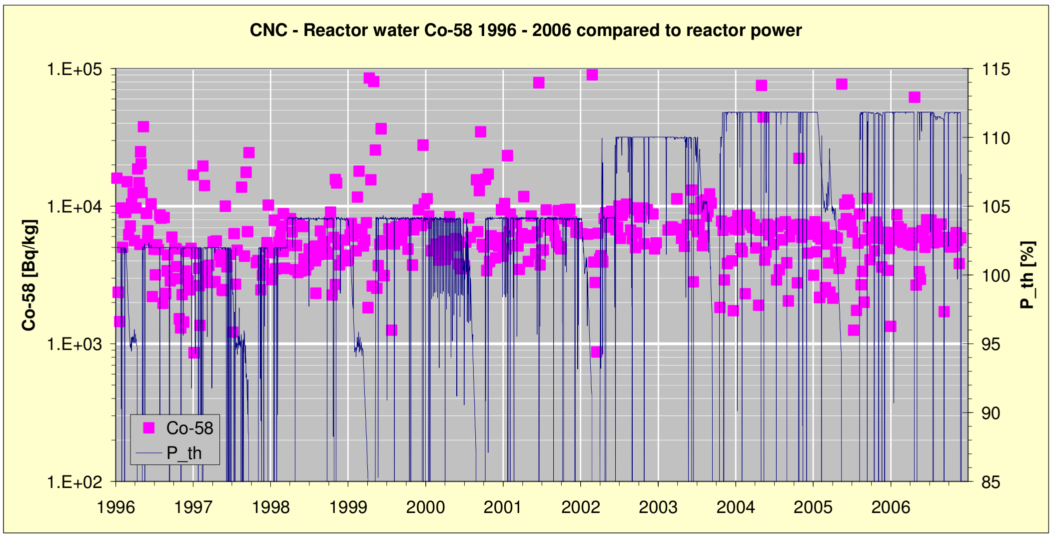


Figure 4.2.19: CNC – Reactor water Co-58 before and after power uprate

– Mn-54 is produced by neutron activation of iron. The CNC reactor water Mn-54 activity shows a rather large scattering, enhanced by a relatively large fraction of insoluble activity. A clear trend is not evident; the average level has been kept at about the same level during the last 10 years.

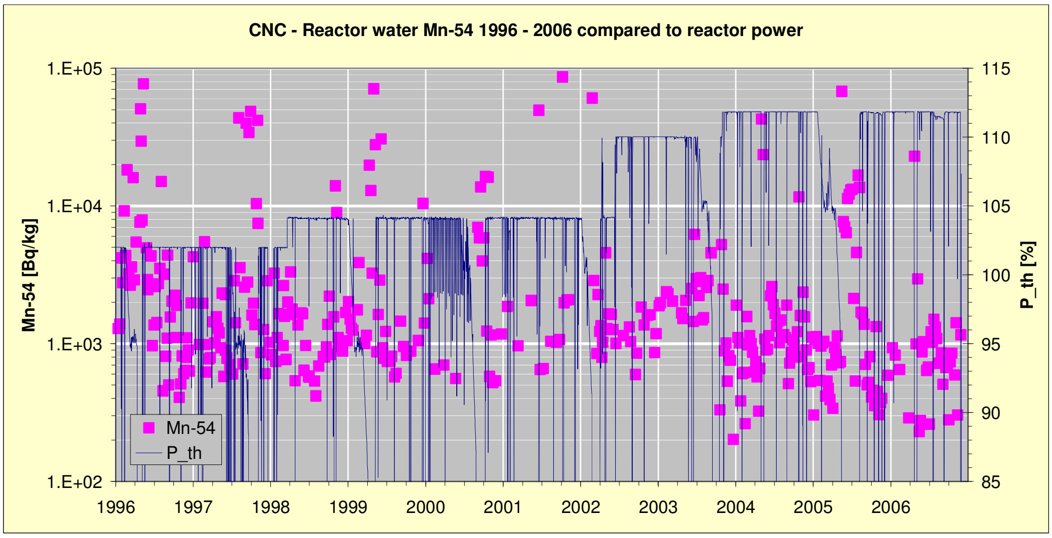


Figure 4.2.20: CNC – Reactor water Mn-54 before and after power uprate

– Reactor water Zn-65 is shows a decreasing trend after the retubing of the turbine condenser, i.e. when most of the feedwater zinc has been replaced with DZO with low content of ^{64}Zn.

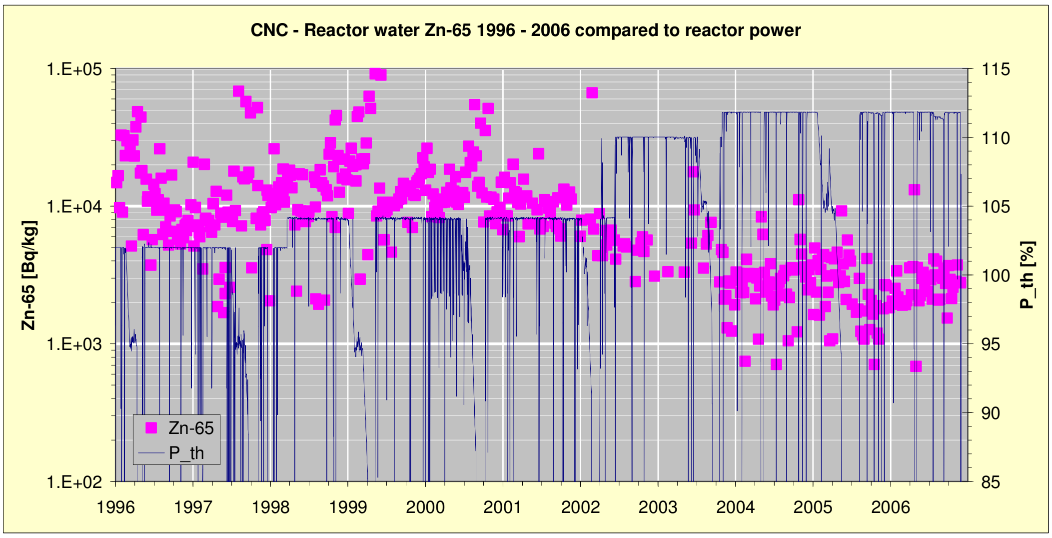


Figure 4.2.21: CNC – Reactor water Zn-65 before and after power uprate

Occasional fuel failures have been experienced in CNC, e.g. during 2004 and 2006 (I-131 activity in the reactor water are presented in **Figure 4.2.22**). The fuel leakages have normally been caused by debris fretting, and are not judged to be related to the power uprates.

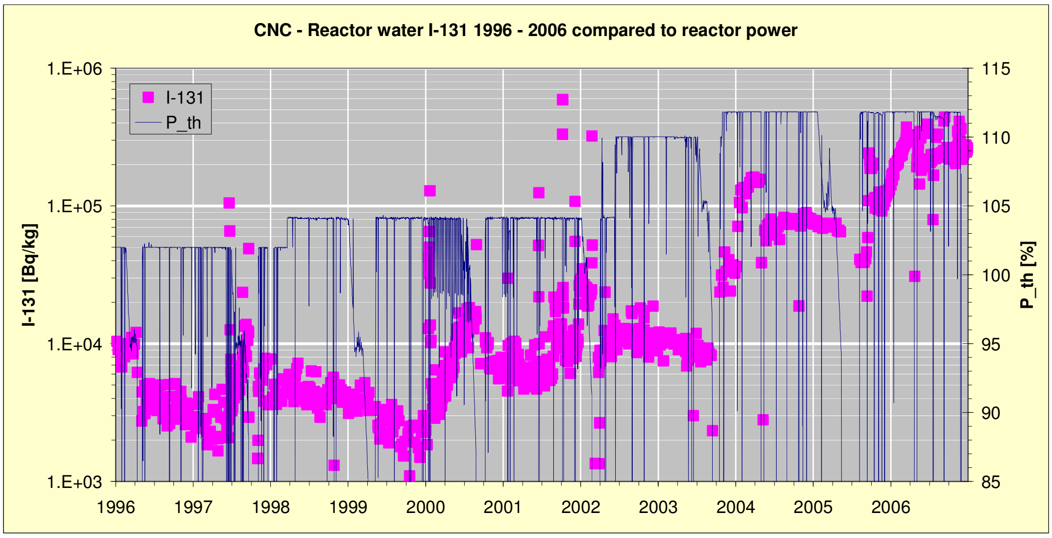


Figure 4.2.22: CNC – Reactor water I-131 before and after power uprate

The moisture content in the steam in the CNC plant, both before and after the power uprates, has been maintained on a low level, with very low contamination levels of the turbine plant. Recent level of steam moisture content is typically 0.02%.

4.2.2.3 Radiation levels

4.2.2.3.1 During operation

The radiation levels during operation are mainly determined by short-lived nuclides such as N-16 ($T_{½}$ = 7.12 s). The production occurs through neutron activation of the ^{16}O isotope in the coolant, which means that the production rate is basically proportional to the thermal reactor power. The N-16 radiation source term in different systems is affected by the production rate, and also by the decay time from the reactor circuit due to the N-16 nuclide's short half-life. Power uprates affect the flow rates, and hence the decay times. The distribution of N-16 between the steam and the reactor water is dramatically affected by injection of hydrogen, so called HWC, with an increase of steam line to about a factor of five. CNC has been operating with HWC since March 1997.

Measuring campaigns of radiation levels at a great number of locations were carried out at four different power levels, 92%, 104.2%, 107% and 110%, in April 2002. The corresponding dose rates were measured at 111.8% power level in November 2003, i.e. at the present maximum reactor power level. All measurements were performed at HWC conditions with 1 ppm of feedwater DH.

The radiation levels outside main steam lines at the April 2002 measuring campaigns are summarized in **Table 4.2.1**. The radiation levels are increased with average 23.8% between 92% and 110%, i.e. for a 19.6% power increase. The increase of radiation levels seems to be very close to the power increase.

Table 4.2.1: CNC – Main Steam Line Radiation (MSLR) monitoring April 2002 at four different power levels and standard HWC conditions

mSv/h	92% Pow. 1 ppm H_2	104,2% Pow. 1 ppm H_2	107% Pow. 1 ppm H_2	110% Pow. 1 ppm H_2	Δ 110%/ 92% (ΔPow. 19.6%)
Steam line A	17.60	20.00	21.40	22.00	25%
Steam line B	17.10	19.40	20.40	21.10	23%
Steam line C	18.00	20.30	21.30	22.30	24%
Steam line D	19.00	21.80	22.80	23.40	23%
Av. line A-D	17.93	20.38	21.48	22.20	+23,8%

Dose rate measurements, inside and outside the plant buildings, for five power levels between 92% and 111.8% have been assessed and are presented in **Figure 4.2.23** and **Figure 4.2.24**. The different locations show some variation. The highest power level, 111.8%, was measured on a later occasion when the operating conditions differed. It shows lower radiation levels than the four previous power levels. Because of this, these readings were excluded and the evaluation was based on the four remaining power levels measured close to each other. The average relative increase between 92% and 110% is close to 50%, i.e. somewhat higher than experienced for the main steam lines. Whilst the measured radiation levels vary by almost 6 orders of magnitude, the average relative increase is similar for different radiation levels, see **Figure 4.2.24**.

The annual occupational exposures 1990 – 2006 obtained during reactor operation are presented in **Figure 4.2.25**. The exposure level has been quite stable over the last ten years, and seems not to have be significantly affected by either the change to HWC in 1997 or the recent power uprates. The overall conclusion must be that the N-16 radiation source term is not the dominant contributor to the occupational exposure during operation.

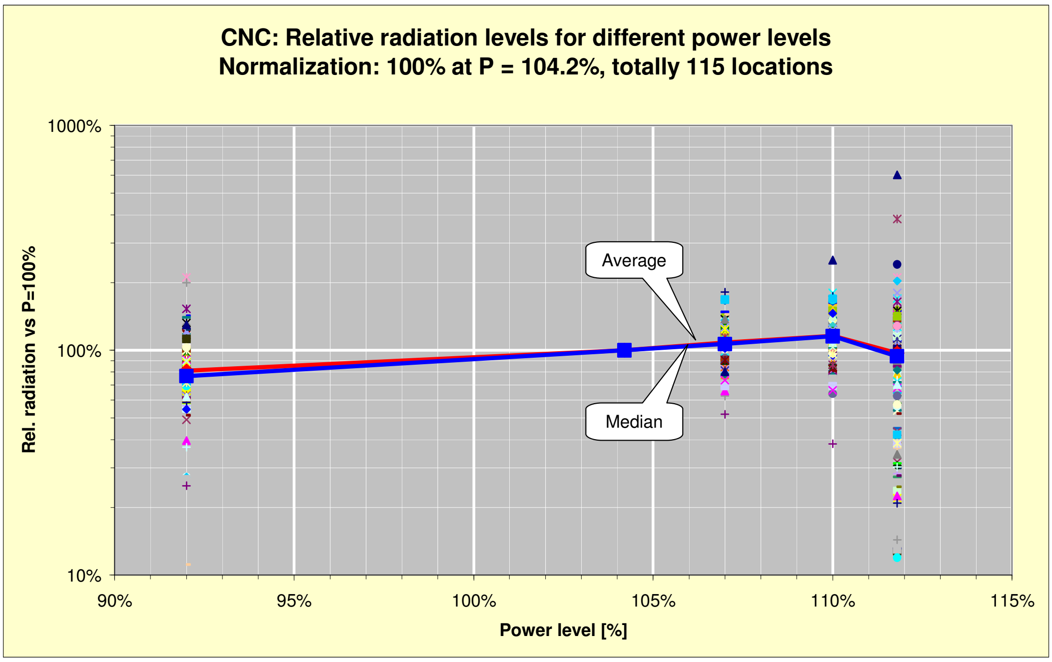


Figure 4.2.23: CNC – Measured relative radiation levels at five different power levels Normalization: 100% at P=104.2%, totally 115 measuring locations, P=92%, 104:2%, 107% and 110% measured at April 2002, P=111.8% at November 2003

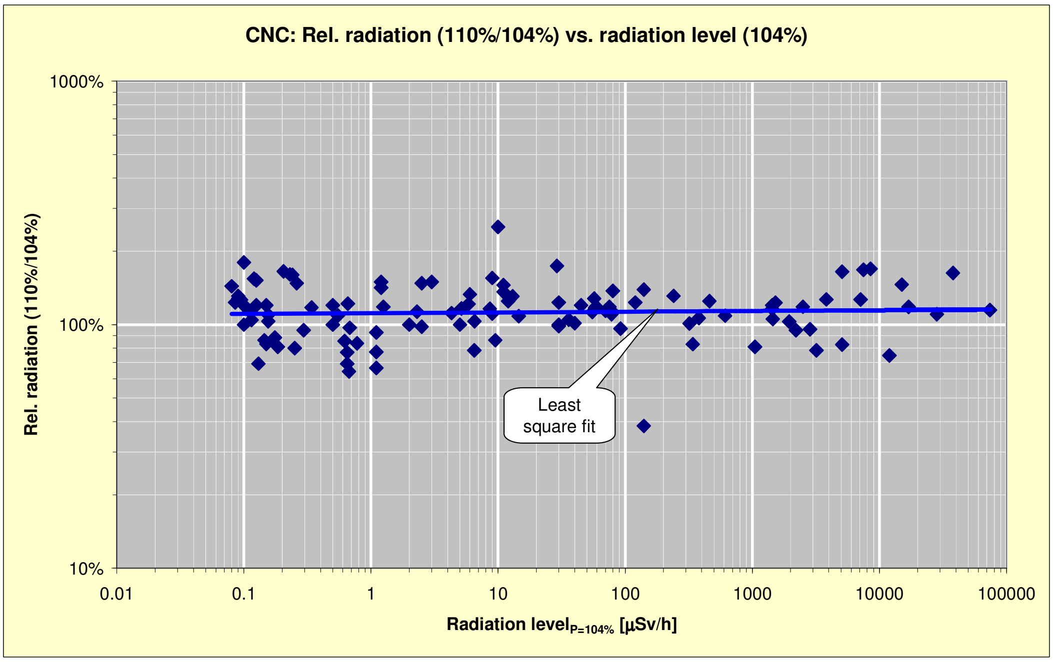


Figure 4.2.24: CNC – Relative increase of measured radiation levels between P=104.2% and P=110% versus radiation level at P=104.2%(Totally 115 measuring locations both inside and outside plant buildings)

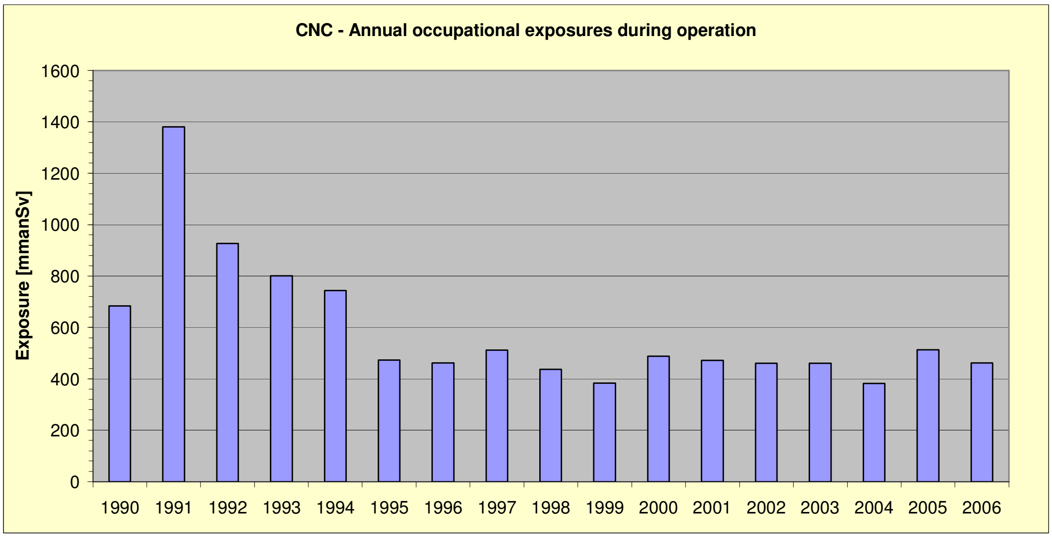


Figure 4.2.25: CNC – Annual occupational exposure during reactor operation

4.2.2.3.2 During outages

About 70% of the occupational exposure in CNC is received during refuelling outages, and about 70% of this exposure comes from the drywell area. This area contains mainly the recirculation loops with pipes, pumps and valves. The area is accessible only during outages.

The radiation levels on the recirculation loops are surveyed in great detail during outages due to the loops large impact on the occupational exposures. The different measuring locations are shown in **Figure 4.2.26**, and some of the locations especially addressed in the present study are indicated in the figure. Note that the decontamination performed at the 2002 RFO only included the lower parts of the loops due to problems with some plugs at the RPV nozzles.

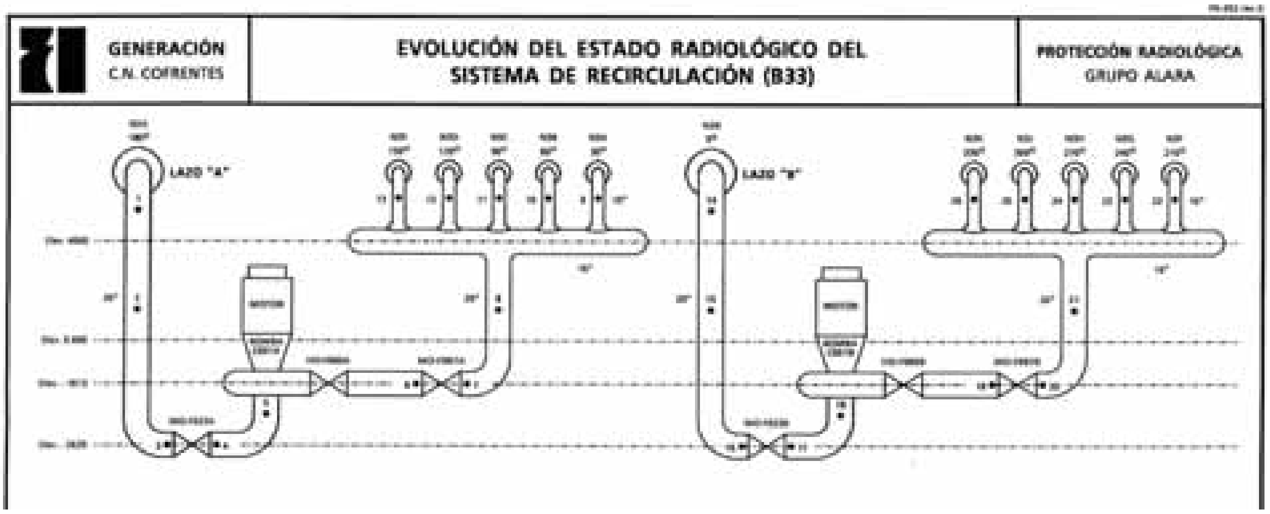


Figure 4.2.26: CNC: BRAC measuring locations on recirculation loop A and B Average of points 2/8 and 15/21 for loop A and B, respectively, used as reference in **Figure 4.2.27** Point 4 (decontaminated) compared to point 15 (not decontaminated) in **Figure 4.2.28**

The vertical sections on the recirculation loops are used as reference locations for the radiation surveys, and the historical dose rate data on these locations are shown in **Figure 4.2.27**. These locations were not included in the 2002 RFO decontamination campaign. Dose rates on RWCU piping are also included in the figure. Note that the RWCU piping is of carbon steel, while the recirculation loops are of stainless steel. The radiation levels were quite stable up to the 2002 RFO, but a considerable increase was experienced especially on the recirculation lines at the following outages. Notably was that this considerable increase was not associated with any dramatic change in measured water chemistry conditions, e.g. reactor water Co-60 (see **Figure 4.2.18**). Also notable was that the increase was most pronounced in locations on the recirculation lines that were not affected by the 2002 RFO decontamination campaign, and that decontaminated pipes in both the recirculation loops (**Figure 4.2.28**) and the RWCU system showed a more reasonable recontamination rate.

Large efforts have been spent to understand the post-RFO-13 behaviour in CNC. The analysis has identified, that a key factor has been the evolution of HWC conditions due to the gradual decrease of reactor water copper (**Figure 4.2.15**). The sub-ppb levels of reactor water copper reached during cycle 14 meant that at least decontaminated pipes were reaching low ECP levels, while probably non-decontaminated pipes with remaining copper in the oxide were staying on a somewhat higher ECP level promoting precipitation and incorporation of soluble corrosion products in the oxide layer. This radiation problem has been solved in the following way:

- A successful decontamination was performed in the 2005 RFO, including all parts of the recirculation loops.
- The hydrogen injection has been increased to assure more stable reducing conditions in the recirculation loops and in the bottom of the RPV (**Figure 4.2.12**).

– The zinc injection has been increased after the 2003 RFO to obtain somewhat higher reactor water zinc levels (**Figure 4.2.16**).

The implemented actions seem to have been quite successful, and the recontamination of the recirculation lines is quite slow (**Figure 4.2.27)**, measurements were performed at an extra outage during 2006). The RWCU pipes of carbon steel, on the other hand, show a more rapid recontamination. The activity buildup in the CNC plant seems to be primarily affected by the water chemistry conditions, and in a very small degree by the power uprating.

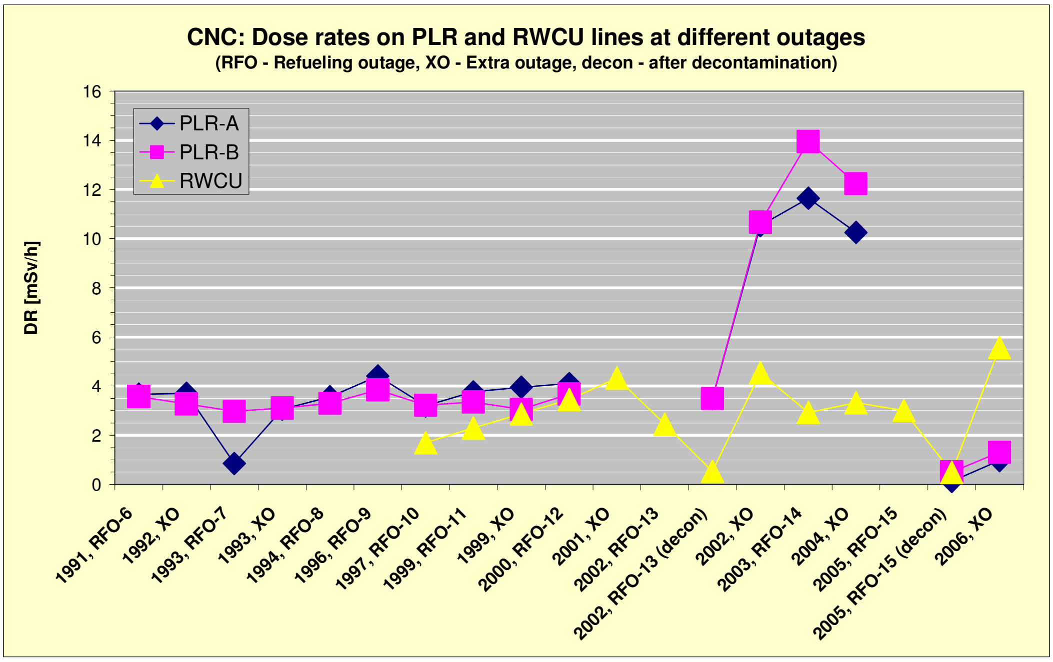


Figure 4.2.27: CNC: Measured dose rates on recirculation lines (PLR) and RWCU lines at different outages 1991 – 2006 (PLR-A and -B: average 2 locations (**Figure 4.2.26**), RWCU: average several locations)

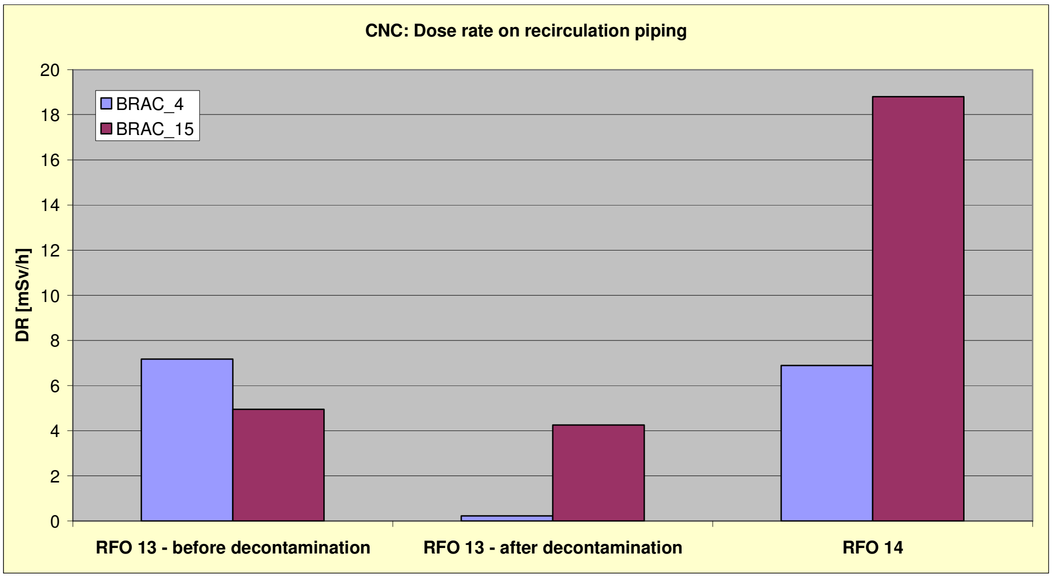


Figure 4.2.28: CNC: Dose rate on two different locations on recirculation lines during RFO-13 (2002, before and after decontamination) and RFO-14 (2003) BRAC_4: Affected by decontamination, BRAC_15: Not affected (**Figure 4.2.26**)

4.2.2.4 Occupational exposures

The annual occupational exposures in CNC during the period 1990 – 2006, have been divided into contributions during reactor operation and outage conditions, are presented in **Figure 4.2.29**. Some years are without RFO in CNC, which means a low annual occupational exposure. Therefore, the rolling 3-year average has been included in the diagram to obtain information about the long-term exposure trend. The exposure levels have been quite stable during the 10 last years, but with a slight increasing trend. The last three RFOs have been associated with somewhat higher exposure levels due to the combined effect of large modification and maintenance efforts (2002, 2006), and the above discussed increased radiation levels after the 2002 RFO. The introduced measures to reduce radiation fields are likely to imply somewhat lower future exposure levels, at least if maintenance efforts are kept on a reasonable level.

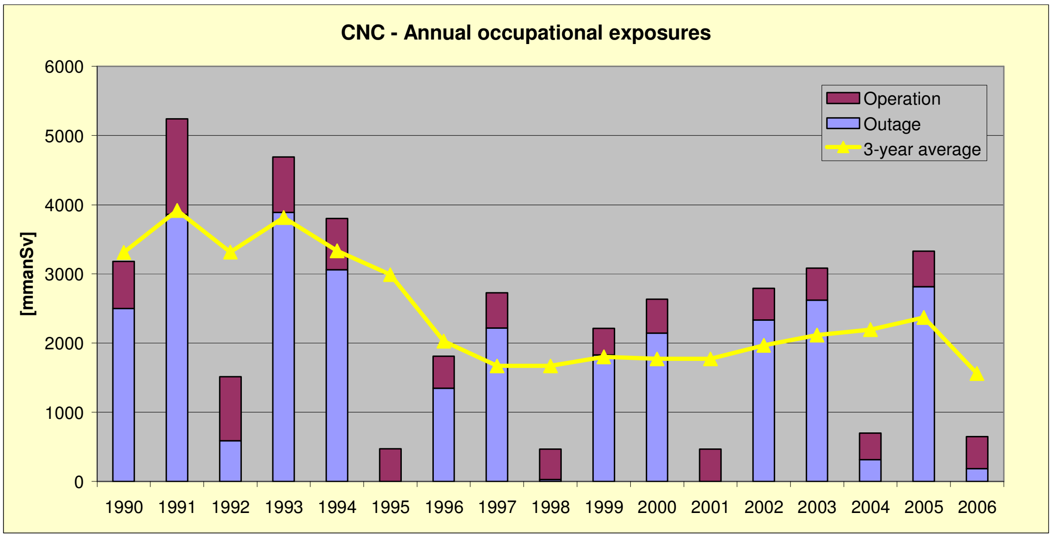


Figure 4.2.29: CNC – Annual occupational exposures split-up on outage and operation conditions plus rolling 3 year-average based on total annual exposure

The CNC annual occupational exposures, presented as rolling 3-years average, are compared with some international exposure data in **Figure 4.2.30**. The CNC data are quite similar to the US BWR experience, which is understandable due to the similar reactor designs. However, as discussed above, the slightly increasing trend seen in CNC during last 10 years is contrary to the US experience, where a slightly decreasing trend is observed. A similar increasing trend to CNC is also seen in Japanese plants. The main reason for the Scandinavian and German plants being about half of the CNC exposure level is probably due to the fact, that most of these plants are of a different design with internal recirculation pumps, i.e. without the recirculation loops responsible for a large fraction of the CNC exposures.

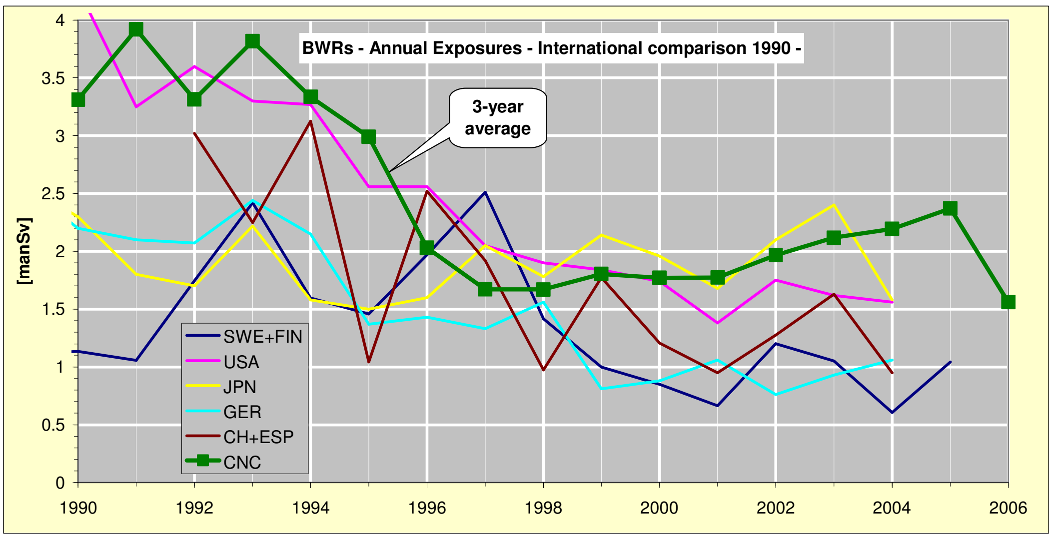


Figure 4.2.30: CNC – Annual occupational exposures compared to international data
Note: the CNC data are rolling 3-year average

4.2.2.5 Summary and conclusions

A review of data and experience from the CNC plant before and after the power uprates 1998-2003 has been performed. The review has resulted in the following conclusions:

- The present power level corresponds to 111.8% of the initial thermal power level, which means on average 5.19 MWt per fuel assembly. The main power increase was introduced in 2002, when the power level was increased from 104.2% to 110%.
- The fuel cycle lengths have been increased in parallel to the power uprates, initially from 12 months then up to 22 months for the present fuel cycle 16. Fuel assembly design has evolved from the original 8x8 array to 9x9 in cycle 8 and 10x10 in cycle 12, which has necessitated more advanced designs for the more demanding recent operation conditions.
- The modifications introduced have mainly affected the turbine plant:
  - Turbine moisture separator and reheater tube sheets
  - Condenser revamping, also including a change of tube material from brass to titanium
  - Low pressure turbine steam paths
  - Turbine heater drain pumps
  - Feedwater pump turbines
  - Feedwater flow measuring system
  - High pressure turbine steam path
  - Heater drains flow control valves.

  Due to low contamination level of the turbine plant it was possible to carry out these modifications with a low exposure.
- The main change on the reactor side is that the reactor pressure and reactor temperature have been slightly increased. This increase means that the steam velocity in the reactor and in the main steam lines has been only marginally affected by the power uprates.
- The reactor water chemistry is strongly determined by the design and materials selection of the turbine plant. The turbine design with forward pumped heater drains and initially carbon steel in most of the piping, resulted in a rather high inflow of iron. Most of the steam extraction pipes have currently been replaced with low-alloy steel pipes which has considerably reduced the forward pumped heater drain contribution to the feedwater iron, however, a significant contribution passing the condensate polishing plant still remains. The earlier brass material in the condenser tubes resulted in a rather high inflow of copper and zinc, but this inflow has been eliminated by changing to titanium tubes. Though a high inflow of zinc has been maintained by the introduction of zinc injection in order to control radiation fields.
- Hydrogen water chemistry (HWC) was introduced in 1997. However, the reduction of corrosion potential was small due to high levels of reactor water copper. The gradual reduction of reactor water copper, and an increase in the amount of injected hydrogen have most likely resulted in low corrosion potentials meeting the HWC requirements during recent cycles.

- The radiation levels during operation in different locations both inside and outside of plant buildings are affected by the production and distribution of the short-lived nuclide N-16. Surveys at power levels in the range of 92% - 111.8% indicate an average effect of the power increase in the order of +15% - +30%. Such increase is reasonable considering the combined effects of higher N-16 production rate and shorter transport times due to the increased flow rates. Of larger importance for the general radiation levels during operation was probably the introduction of HWC in 1997, that resulted in an increase of approximately a five times in carryover of N-16 activity with the steam to the turbine plant. However, the annual operational exposures have been maintained at a relatively stable level during the last 10 years and do not seem to be significantly affected by either the change to HWC in 1997 or the recent power uprates. The overall conclusion must be that the N-16 radiation source term is not the dominant contributor to occupational exposure during operation.

- Radiation levels during outage conditions are of importance for occupational exposures and are very much determined by the recirculation loops and the RWCU piping. A considerable increase, particularly in the radiation fields around recirculation loops, was experienced at the 2003 refuelling outage. The increase does not seem to be due to the power uprate, but rather to the combined water chemistry effect of gradually decreasing reactor water copper and HWC operation, resulting in the restructuring of the oxide layers inside the recirculation loops. Several measures were introduced to mitigate the increase (decontamination campaign in 2005, increased injection of hydrogen and zinc), and at present, the recirculation loop radiation fields seem to be low and well controlled.

- The annual occupational exposures at CNC have displayed a slightly increasing trend during the last 10 years. This trend is explained by the combined effect of increased radiation fields and considerable efforts spent in modifications and maintenance during recent outages. The CNC annual exposures show large similarities with the exposures in US BWRs of a similar design, but the US plants generally show a slightly decreasing trend during recent years. A future decreasing trend is also expected in the CNC plant due to the improved control of radiation fields around the recirculation loops mentioned above.

- The general feeling is that the CNC power uprates have had a negligible impact on occupational exposures, or at least are shadowed by more important factors such as:
  - water chemistry (HWC, copper, zinc, feedwater iron, etc.)
  - fuel failures, recent failures have been caused by debris fretting, and do not seem to be attributed to the power uprates
  - some additional maintenance work, e.g. earlier postponed maintenance on the RWCU system and experienced corrosion problems on the CRD piping in the vessel pedestal penetrations.

  Most of the works related to power uprates have concerned the turbine plant with very low exposures. The only high dose rate work that could be attributed to the power uprate (with some reservation) is the underwater repair of the steam dryer during the 2005 refuelling outage. However, that work contributed rather little to the total exposure during the outage.

4.3 Asco and Tihange

4.3.1 Introduction

Based on the data collected in the task #1, two PWR plants were selected for additional analyses. Main criteria for selection were:

- Reactor design of the Westinghouse or Framatom type.
- Significant power increase, up to 10%.
- Steam Generator (SG) Replacement (New SG with tubes of Inconel 690 (i.e. a high fraction of Ni in corrosion products).
- ≤18 months fuel cycles, boron – lithium chemistry.
- Plants where two years operational experience exists with the uprated plant, to be able to assess the effects on the occupational doses.

Several PWR plants were considered but finally two PWR plants were selected for detailed analyses: Asco and Tihange. NPP Asco and NPP Tihange are, by design, very similar plants. The fundamental operating basics at this type of Pressurised Water Reactor (PWR) are set below.

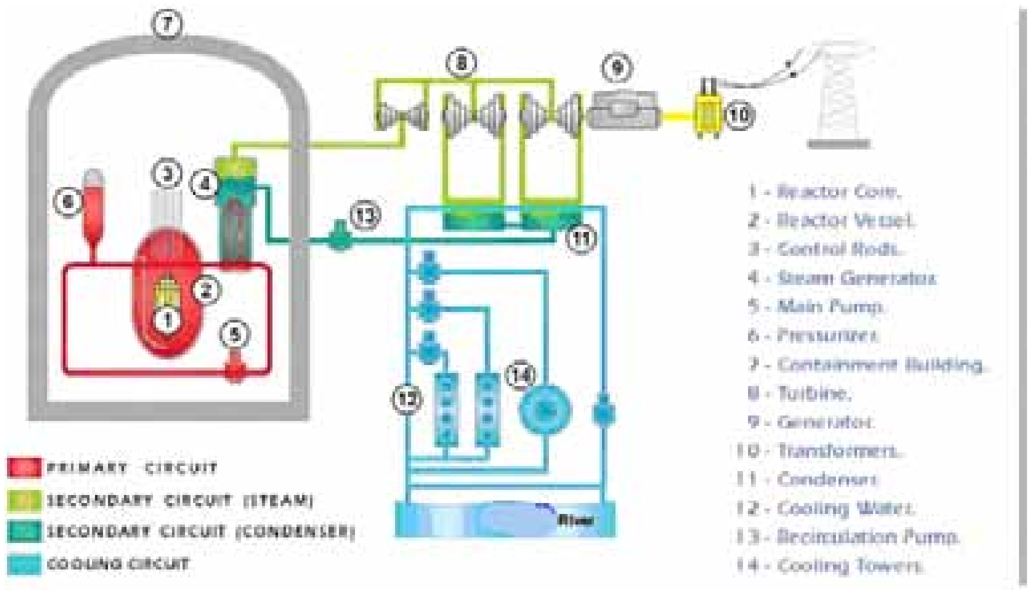


Figure 4.3.1: A PWR plant – Operating diagram

The containment building (dry, ambient pressure, **Figure 4.3.1**) is the dominant and highest building of the plant designed to accommodate normal operating loads, functional loads resulting from a loss-of-coolant accident, and the most severe loading predicted for seismic activity. It provides biological shielding for both normal and accident conditions and collection and holdup for leakage from the containment vessel. Inside the containment structure, the reactor and other NSSS components are shielded with concrete. The primary circuit (**Figure 4.3.2**) consists of three independent loops connected to the reactor vessel. Each loop includes a steam generator and a main pump. All these elements are connected by a main pipe, forming a closed and completely watertight assembly to ensure that all radioactive fluids are confined to the system. To ensure the continuous flow characteristics and to prevent evaporation of the coolant water in the reactor, the entire system is adequately pressurized relative to saturation pressure.

The power supply system to the reactor coolant pumps is designed so that adequate coolant flow is maintained to cool the reactor core under all conceivable circumstances. All pump parts in contact with the coolant are made of austenitic steel or stainless steel covered.

The steam generators, one per loop, are vertical U-tube units that contain Inconel 690 tubes.

The reactor coolant piping and all of the pressure containing and heat transfer surfaces in contact with reactor water are stainless steel except the steam generator tubes and fuel tubes, which are Inconel 690 and zircaloy or ZIRLO, respectively. Reactor core internals, including control rod drive shafts, are primarily stainless steel.

An electrically heated pressurizer connected to one reactor coolant loop maintains RCS pressure during normal operation, limits pressure variations during plant load transients, and keeps system pressure within design limits during abnormal conditions. The pressurizer is located in one of the loops.

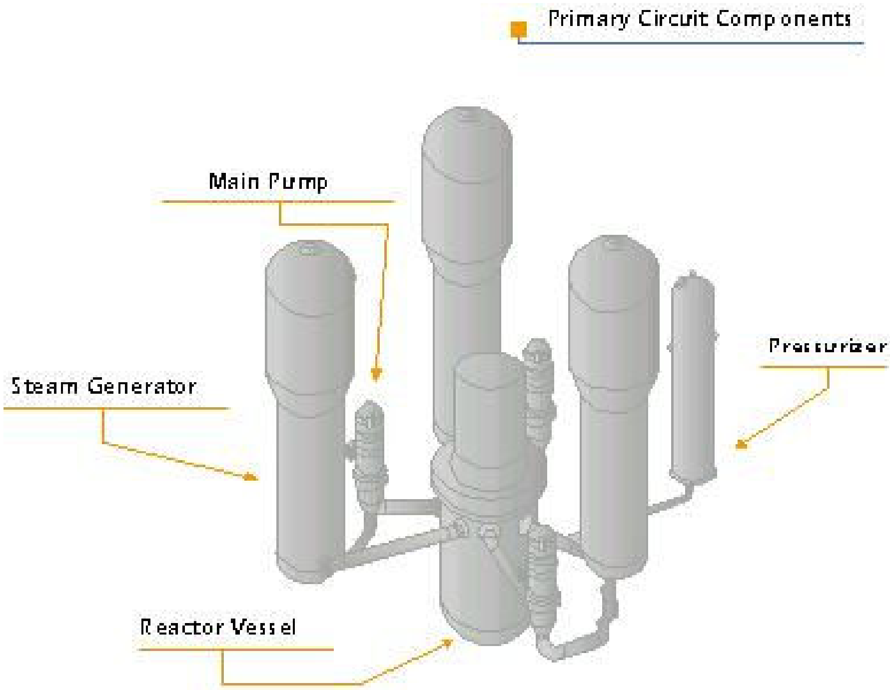


Figure 4.3.2: A PWR plant – Primary circuit

The turbine plant comprises a single turbine-generator unit. Saturated steam is supplied from the steam generators to the turbine. The turbine is a 1500-rpm, tandem-compound unit with one high pressure and two low-pressure cylinders. The steam flows through the high-pressure turbine where it expands to low-pressure condition and then through moisture separator reheater units where it is dried and reheated to superheated condition and directed to the low-pressure turbines where it expands to condenser pressure condition. The steam cycle is closed by condensing the steam from LP turbines, heating the condensate in the feedwater system and returning the water to the steam generators through the feedwater pumps.

The reactor pressure vessel (**Figure 4.3.3**) is cylindrical with a welded hemispherical bottom head and a removable, bolted flanged and gasketed, hemispherical upper head. The vessel contains the core, core support structures, control rods, and other parts directly associated with the core. Outlet nozzles are arranged on the vessel to facilitate optimum layout of the Reactor Coolant System equipment. The inlet nozzles are tapered from the coolant loop vessel interfaces to the vessel wall to reduce loop pressure drop. The bottom head of the vessel contains penetration nozzles for connection and entry of the nuclear in-core instrumentation. Internal surfaces of the vessel that are in contact with primary coolant, are weld overlay with stainless steel or Inconel.

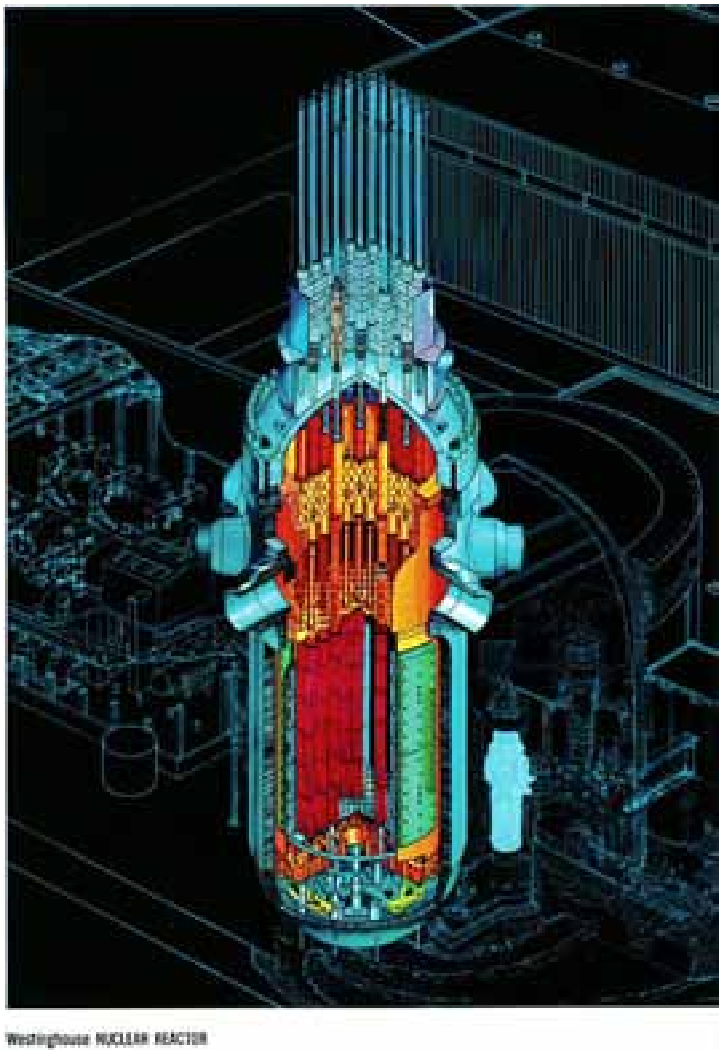


Figure 4.3.3:A PWR plant – Reactor Vessel

The reactor core is composed of 157 fuel assemblies. The fuel assemblies are of square cross section and are arranged in the reactor vessel in a way to optimally use the available cylindrical space. Square spacer grid assemblies and the upper and lower end fitting assemblies support the fuel rods in fuel assemblies. Each fuel assembly is composed of 17 x 17 = 289 rods; of these, most of the places are used by fuel rods; the remaining places, which are evenly and symmetrically distributed across the cross section of the assembly, are provided with thimble tubes which may be reserved for control rods, and control instrumentation tube for incore thimble.

Control rod assemblies are used for reactor control and consist of clusters of stainless steel clad neutron absorber rods inserted into guide tubes, one per tube. The absorber rods move within the guide tubes. Above the core, each cluster of absorber rods is attached to a spider connector and drive shaft, which is raised and lowered by a drive mechanism mounted on the reactor vessel head. Downward stroke of the control rod cluster is by gravity.

Water Chemistry

The RCS water chemistry is selected to minimize corrosion. A periodic analysis of the coolant chemical composition is performed to verify that the reactor coolant quality meets the required specifications. The Chemical and Volume Control System provide a means for adding chemicals to the RCS which control the pH of the coolant during initial startup and subsequent operation, scavenge oxygen from the coolant during startup and control the oxygen level of the coolant due to radiolysis during all power operations subsequent to startup. The pH control chemical employed is Lithium 7 hydroxide. This chemical is chosen for its compatibility with the materials and water chemistry of borated water/stainless steel/zirconium/inconel systems. In addition, lithium is produced in solution from the neutron irradiation of the dissolved boron in the coolant. The concentration of Lithium 7 hydroxide in the RCS is maintained in the range specified for pH control. If the concentration exceeds this range, the cation bed demineralizer is employed in the letdown line in series operation with the mixed bed demineralizer.

Dissolved hydrogen is employed to control and scavenge oxygen produced due to radiolysis of water in the core region.

Components with stainless steel sensitized in the manner expected during component fabrication and installation will operate satisfactorily under normal plant chemistry conditions in PWR systems because chlorides, fluorides and oxygen are controlled to very low levels.

4.3.2 Asco NPP power uprate

4.3.2.1 Power uprate characteristics

Ascó NPP is located on the right bank of the Ebre River in the Ribera d'Ebre region. It is located between the villages of Flix and Ascó (Tarragona), some 65-Km from the city of Lleida and some 110-Km from the Ebre River's mouth. The plant takes its name from the village of Ascó, which is located 2-Km south of the plant. The village of Flix is located north of Ascó NPP, holding the oldest chemical facilities in Spain.

Asco NPP consists of two 1028 MWe units, Westinghouse design 3 loop plant with 2912 Mwt pressurised water reactors. Supplier of the nuclear scope was Westinhouse and the supplier of the non-nuclear design was Initec/Inypsa. Endesa have total ownership of Asco 1 whereas at Asco 2, Endesa own 85% and Iberdorola 15%. Asco 1 had first criticality June 17, 1983 and stared commercial operation December 10, 1984. Asco 2 had first criticality September 11, 1985 and stared commercial operation March 31, 1986.

Both units have been uprated since commissioning. The thermal power of Asco 1 reactor was increased from 2696 MW to 2900 MW in 2000 and to 2951 MW in 2003. The corresponding nominal values of the net electrical output were 930 MWe, 1028 MWe and 1033 MWe, respectively. The present study focuses on the Asco 1 first uprate, resulting in a thermal power level of 8% compared to the initial power level. The net electrical output from the plant during the period 1983 – 2005 is shown in **Figure 4.3.4**.

The latter uprate was an uprate of 1.5 %, achieved using more precise techniques for measuring feedwater flow.

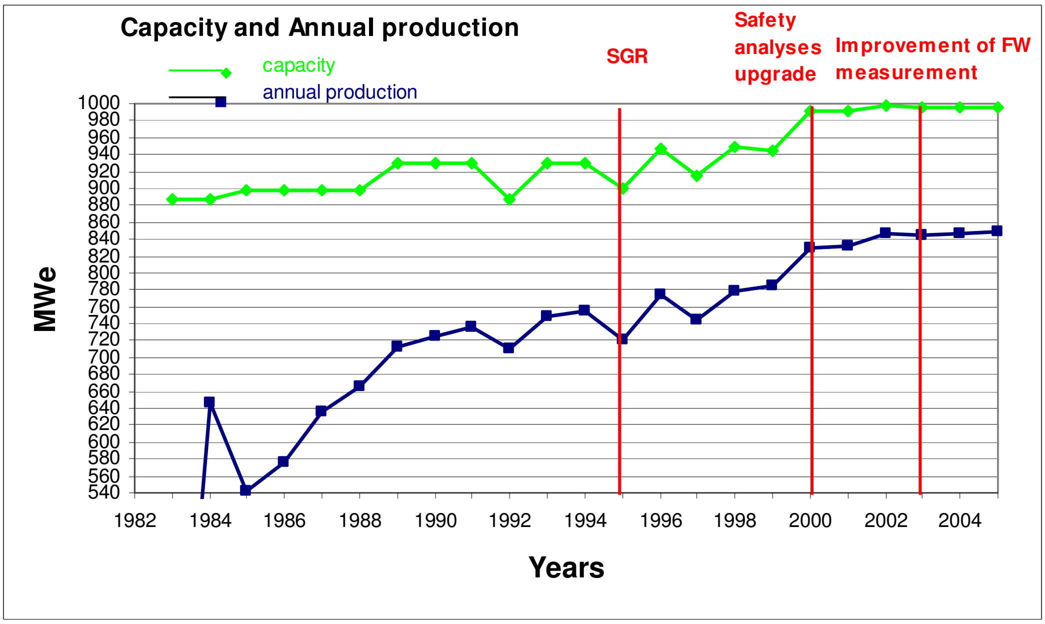


Figure 4.3.4: Asco 1 – Net electrical output

The reactor core of Asco 1 contains of 157 fuel assemblies, which means 18.80 MWth per fuel assembly at present maximum power level, compared to 17.20 MWth per assembly at initial design power. The fuel assemblies are 17x17 design, for the first cores STANDARD design, with average enrichment 3.5 %. Since then new fuel designs have been adopted. At present it is MAEF-IFM+RP (Enusa) with average enrichment 4.7%. Each 17x17 MAEF+IFM+RP is composed of a bundle of 264 fuel rods. These rods are arranged in a square lattice of 17 x 17 positions and are supported by twelve grids, two of which are end grids, six intermediate, three mixer and one protective. The grids, together with 24 thimbles tubes, an instrumentation tube and two nozzles at the ends, form the structural skeleton of the fuel tube. Main specification of fuel is given below.

FUEL ASSEMBLY	
Lattice	17x17
No. fuel rods	264
No. thimble tubes	24
No. instrumentation tubes	1
No. grids (without IFM)	9
No. IFM grids	3

FUEL ROD	
Composition of pellets	UO_2
Composition of neutronic poison pellets	$UO_2 - Gd_2O_3$
Cladding material	ZIRLO™
Spring material	stainless steel
Plug material	ZIRLO™

MATERIALS	
End grids (strap)	INCONEL-718
Intermediate grids	ZIRLO™
IFM grids	ZIRLO™
Protective grid (strap & dimples)	INCONEL-718
Upper nozzle	stainless steel
Lower nozzle	stainless steel
Spring	INCONEL-718
Thimble tubes	ZIRLO™
Instrumentation tube	ZIRLO™

A summary of the modifications that have been introduced in the Asco 1 plant during the period 1995 – 2003 is presented in **Table 4.3.1**. Not all modifications are due to the power uprates, but in many cases form part of the ongoing plant modernization programs that have been carried out. Those two aspects are not always easy to separate; introduced modifications are in many cases addressing both aspects. The major modernization work was performed in 1995-1997 when was replaced SG and both HP and LP turbines were modified. Electrical power was increased after turbine modifications (1995, 1997), thermal power was increased in year 2000 when upgrade of Safety Analyses was performed.

Table 4.3.1 – Asco 1 Summary of modernization work

Year	Work performed
1995	SG Replacement Modification of High Pressure Turbine
1997	Modification of Low Pressure Turbine
2000	Upgrade of Safety and System Analysis Turbine modification for increased flow Upgrades of main electric equipment
2003	Improvement of the feed water flow measurement

The variation in outage lengths in period 1990 – 2004 is presented in **Figure 4.3.5.** On the beginning of operation outages duration were mostly about 40 days. Planned outages show trend of shortening during last decade. In the year 1995 outage was prolonged due to SG replacement and RTD Bypass replacement. Impact on occupational exposures is addressed later in the report.

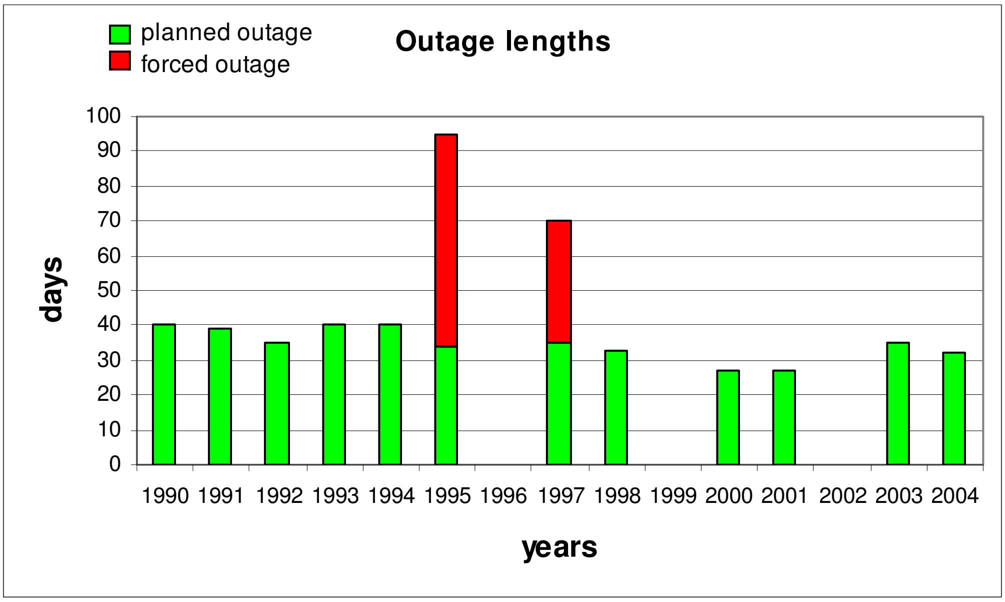


Figure 4.3.5: Asco 1 – Outage lengths for years 1990-2004

4.3.2.2 Water chemistry and radiochemistry

Detailed analyses of chemistry and radiochemistry were not possible to perform due to non-availability of that data. However, a short Reactor Coolant System (RCS) chemistry specifications were available:

Parameter	Value
Solution pH	Determined by the concentration of boric acid and alkali present. Expected values range between 7.04 (high boric acid concentration) to 7.1 (low boric acid concentration)
pH Control Agent (Li^7OH)	3.5 ppm
Boric Acid, ppm B	Variable from 0 to approximately 3000

4.3.2.3 Radiation levels

4.3.2.3.1 During operation

A way to assess the overall affect of power uprates on radiation levels during operation is to study the occupational exposures obtained during reactor operation. Data from Asco 1 are summarized in **Figure 4.3.6**. It must be noted, that the annual exposure during operation is normally a small fraction, 2-20%, but mostly less than 10%, of the total occupational exposure. Therefore, the data in Figure 4.3.6 shows some variation, but no clear correlation to the power uprate is seen.

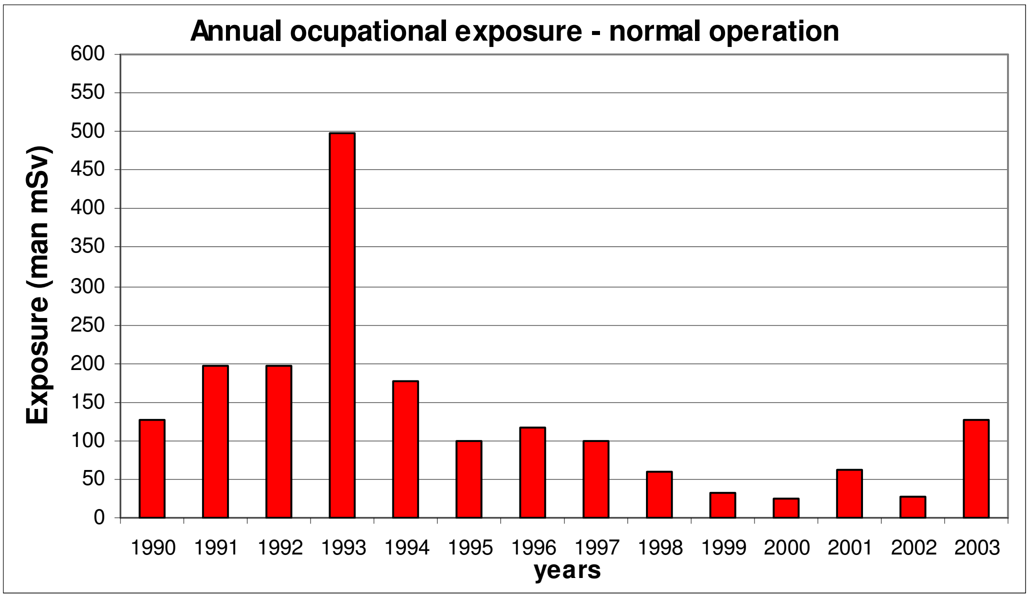


Figure 4.3.6: Asco 1 – Annual occupational exposure

4.3.2.3.2 During outages

The dose rates during outage conditions have been regularly followed-up. Measured dose rates on survey points around primary piping (primary circuit and channel head hot and cold legs) at outage conditions are shown on **Figure 4.3.7**. The dose around the channel head is the biggest fraction, approximately 95% of the total measured dose on survey points. From the figure it is obvious that radiation level decreases, starting with SG Replacement, year 1995.

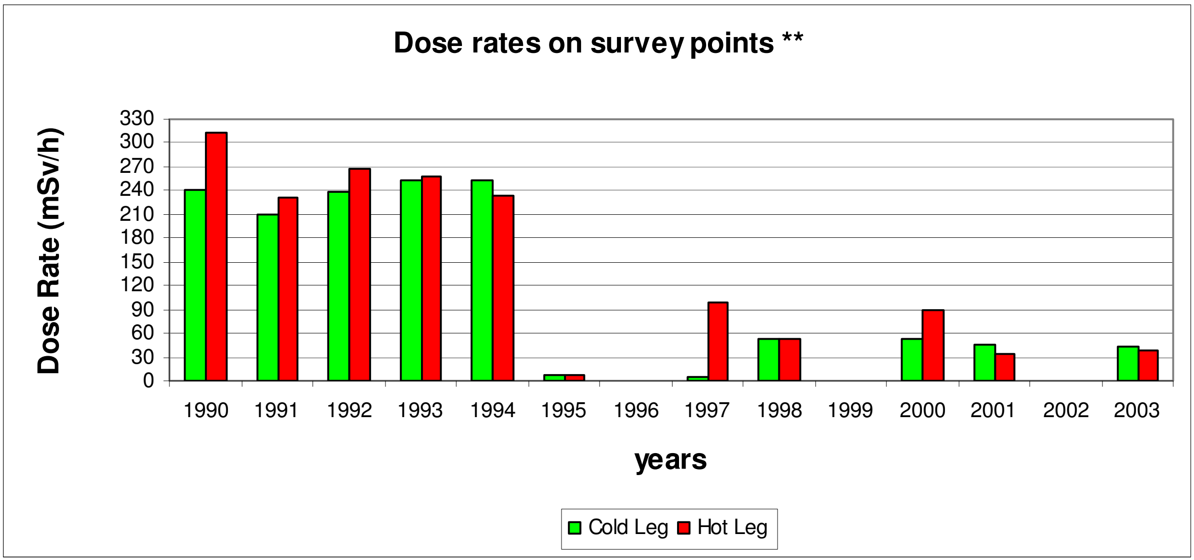


Figure 4.3.7: Asco 1– Dose rates on survey points**

**survey points: Channel head /Primary circuit

4.3.2.4 Occupational exposures

The annual occupational exposures during the period 1990 – 2003 are presented in **Figure 4.3.8.** The exposures are divided into contribution during outage and operation conditions. The exposures during operation are only a small fraction of the total exposures, and have trend shown on the **Figure 4.3.6**. A clear correlation with the outage lengths (**Figure 4.3.5**) is seen, i.e. outages involving large efforts for the power uprate and plant modernization projects have consequently implied increased exposure).

Data on exposure per outage day are compiled in **Figure 4.3.9**. The exposure per outage day shows some variation, but no trend of increasing exposure after SG Replacement and the power uprate is seen.

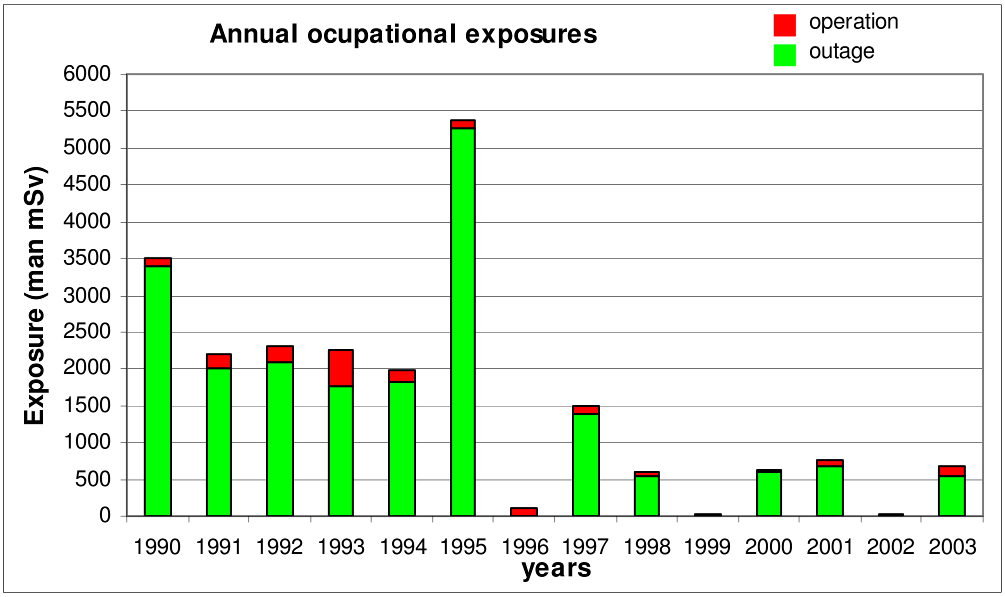


Figure 4.3.8: Asco 1 Annual occupational exposures split-up on outage and operation conditions

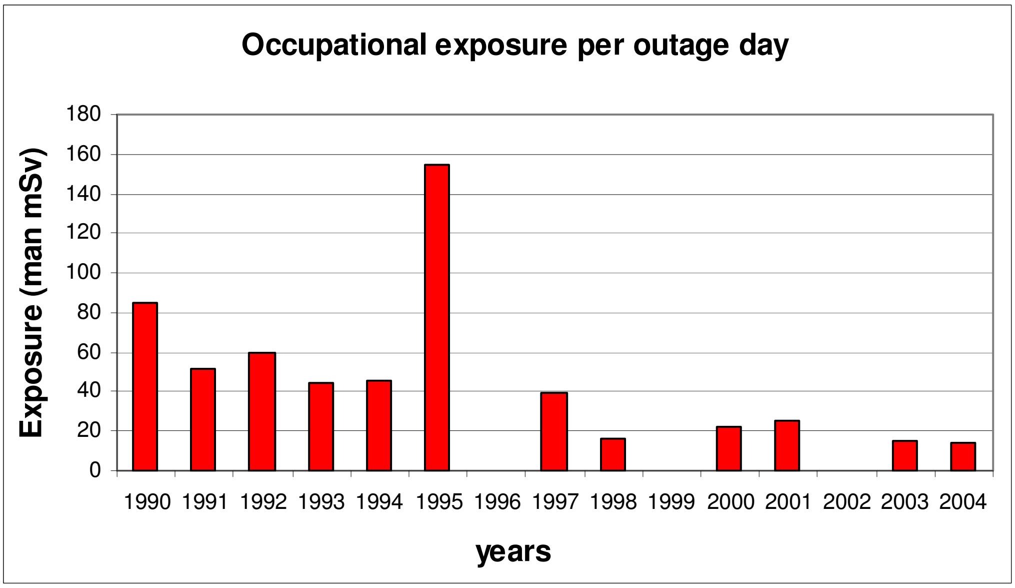


Figure 4.3.9: Occupational exposures per outage day

The Asco 1 annual exposures are compared to some international PWR data in **Figure 4.3.10**. The Asco 1 exposures are in the last decades rather low compared to the average international PWR. Only years with considerable efforts in modernization projects, e.g. 1995-1997, imply higher exposures comparable with the average European plants.

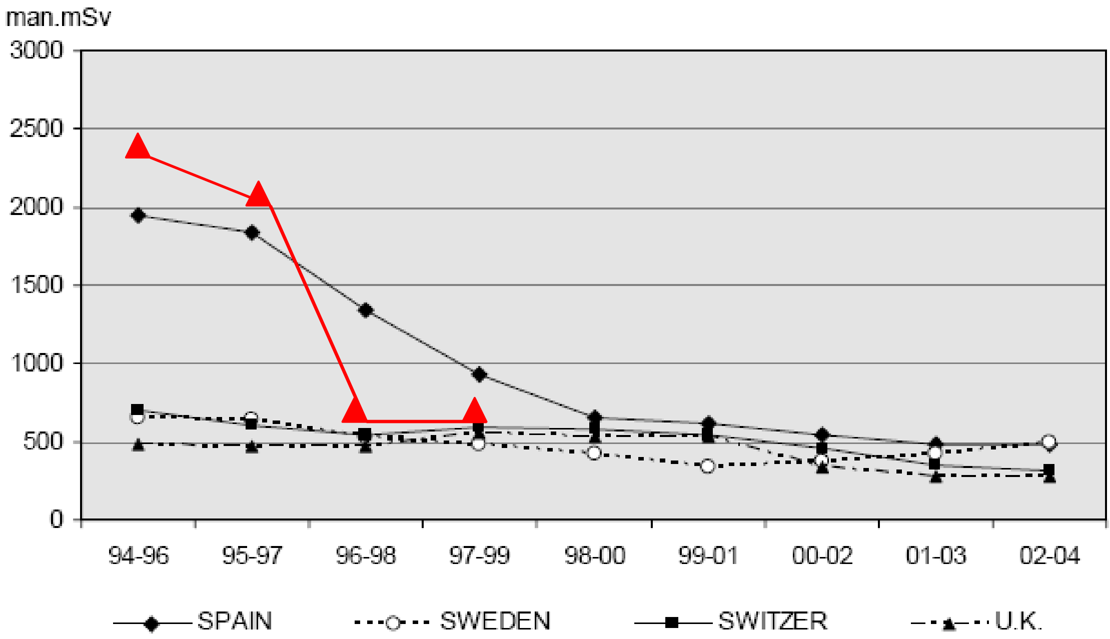


Figure 4.3.10: Average outage collective dose per reactor type and per country

4.3.2.5 Summary and conclusions

A review of data and experience from the Asco 1 plant has been performed. The review has resulted in the following conclusions:

- The plant has been uprated twice since the commissioning. The thermal power of reactor was increased from 2696 MW to 2900 MW in 2000 and to 2951 MW in 2003. The SG Replacement was performed in 1995 but all safety analyses necessary for power uprate were performed in 2000. The present study has focused on the first uprate, resulting in a thermal power level of 8% compared to the initial power level. The latter uprate was an uprate of 1.5 %, achieved using more precise techniques for measuring feedwater flow.
- In year 1995 when a SG Replacement was performed the planned outage length was 34 days. The reality was it took 61 forced outage days. Typical outage lengths are between 30 and 40 days.
- Standard PWR water chemistry was maintained, no zinc is injected.
- Operational exposure has been maintained on a constant and rather low level since the uprate.
- Dose rates, during outage, on reactor systems have been maintained on a rather low and constant level after SG Replacement.
- Considerable manhour efforts spent during some outages for the plant modernization program have, of course, resulted in some increase of occupational exposures. The exposures per outage day have decreased in the last decade. The average annual exposures in the Asco 1 have been kept on a level comparable to international values for PWR plants.

4.3.3 Tihange NPP power uprate

4.3.3.1 Power uprate characteristics

Tihange NPP is located on the bank of the Meuse River in the Liege region. It is located in Huy, some 30-Km from the city of Liege, the municipality includes the old communesof Ben-Ahin and Tihange.

Tihange NPP consists of three units, Framatome/Westinghouse design 3 loop plant with pressurised water reactors. Westinghouse and Framatome, associated with Alstom ACEC Energie and Cockerill Mechanical Industry (CMI) supplied the nuclear scope, the supplier of the non nuclear design was Alsthom ACEC. Tihange 1 is 50% owned by Electrabel 50% and 50% by EDF. Whereas Electrabel own 96% of Tihange 2 & Tihange 3 and the Belgian utility SPE own the remaining 4%. Tihange 1 had first criticality February 21, 1975 and started commercial operation on October 1, 1975. Tihange 2 had first criticality October 13, 1982 and started commercial operation on Jun 6, 1983. Tihange 3 had first criticality Jun 5, 1985 and started commercial operation on September 1, 1985.

Tihange 1 and Tihange 2 have been uprated since the commissioning. The thermal power of Tihange 1 reactor was increased from 2665 MW to 2875 MW in 1995. The corresponding nominal values of the net electrical output were 914 MWe, 1009 MWe. The thermal power of Tihange 2 reactor was increased from 2785 MW to 2905 MW in 1995 and to 3064 MW in 2001. The corresponding nominal values of the net electrical output were 945 MWe, 1007 MWe and 1055 MWe. The present study focuses on the Tihange 2 second uprate, resulting in a thermal power level of 10% compared to the initial power level. The net electrical output from the plant during the period 1983 – 2005 is shown in **Figure 4.3.11**.

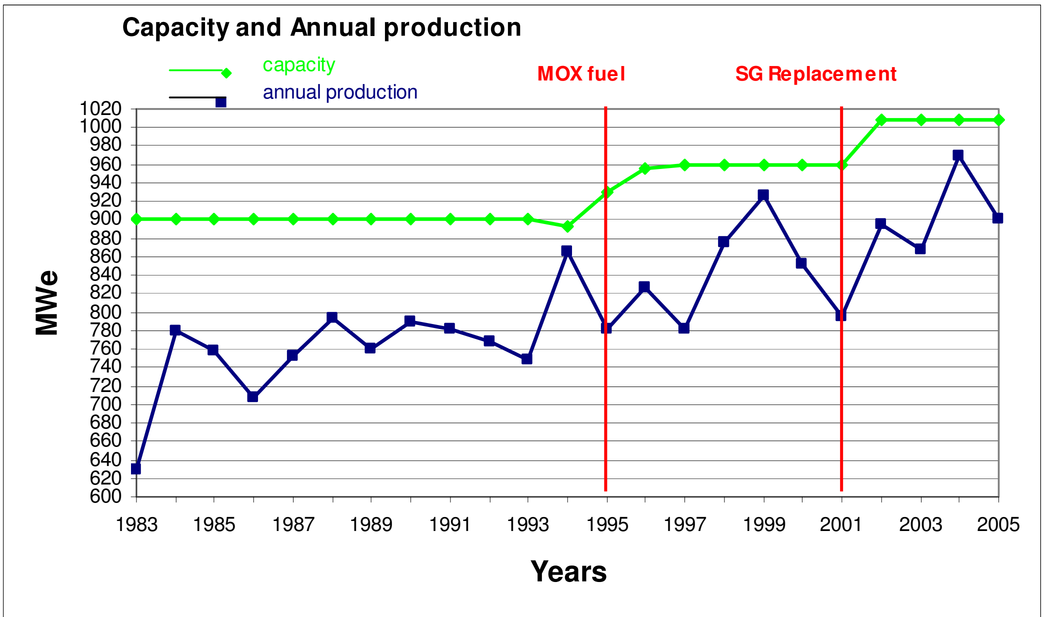


Figure 4.3.11: Tihange 2 – Net annual electrical output and Capacity

The reactor core of Tihange 2 contains 157 fuel assemblies, which means 19.52 MWth per fuel assembly at present maximum power level, compared to 17.74 MWth per assembly at initial design power. The fuel assemblies are 17x17 design, for the first cores STANDARD design, with enrichments 1.6%/2.2%/3.5 %. Since then, new fuel designs have been adopted, MOX (Belgonucleaire) with average enrichment 3.8%. Each fuel assembly is composed of a bundle of 264 fuel rods. A MOX element is simply a fissile element that weighs approximately 450 kg and is composed of a mixture of 415 kg of uranium oxide and 35 kg of plutonium oxide. By replacing a standard fuel element composed of enriched uranium by a MOX element, 9 kg of plutonium is consumed instead of 5 kg that is produced.

The recovering of plutonium in MOX elements reduces the quantity of plutonium produced in nuclear power plants and economizes uranium ore needs.

A summary of modifications that have been introduced in the Tihange 2 plant during the period 1995 – 2001 is presented in Table 2. Modifications are not mainly due to the power uprates, but in many cases are part of ongoing plant modernization programs. Those two aspects are not always easy to separate; introduced modifications are in many cases addressing both aspects. Major modernization work was performed in 1995 when a new fuel design was introduced, and in 2001 when SG were replaced and a revision of the Safety Analyses was performed. In both cases thermal power was increased.

Table 4.3.2 – Tihange 2 Summary of modernization work

Year	Work performed
1995	First loading of MOX fuel in Tihange 2 (March)
2001	During the outage of Tihange 2, which started on 9 June and ended on 11 August, all three Steam Generators were successfully replaced. The steam generator replacement was executed in the new record time of 17.5 days. Revision of safety analyses

The variation in outage lengths 1990 – 2004 is presented in **Figure 4.3.12**. Normal plant outages duration is about 30 days. In 1997 the outage was prolonged due to RCP inspections and repairs. In 2001 the outage was prolonged due to SG replacement. Plant availability is mostly about 86%, and the modernization and power uprate programs have not impacted on the average outage length (impact on occupational exposures is addressed later in the report).

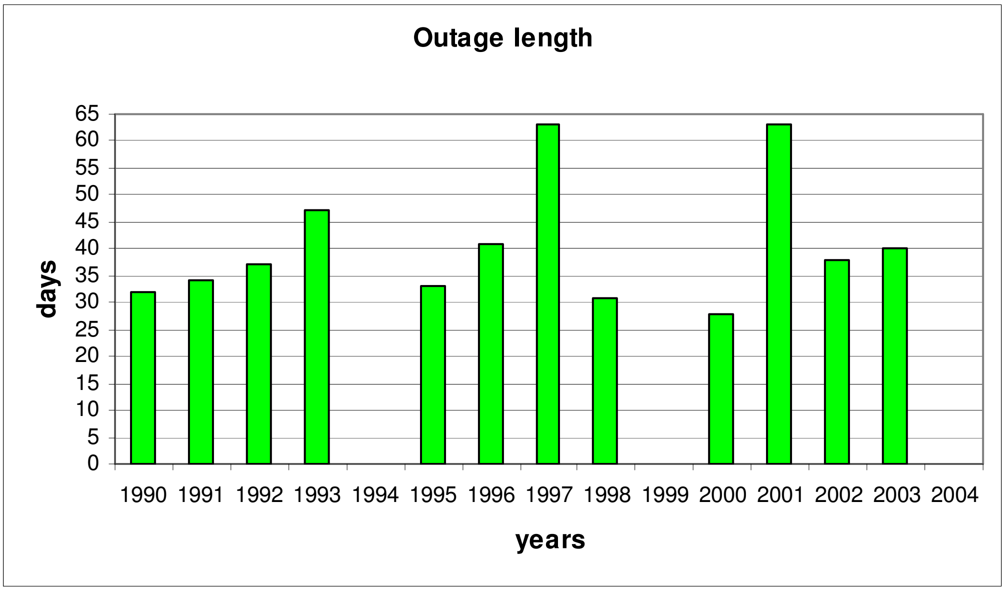


Figure 4.3.12: Tihange 2 – Outage lengths for years 1990-2004

4.3.3.2 Water chemistry and radiochemistry

Detailed analyses of chemistry and radiochemistry were not possible to perform due to non-availability of that data. However, a short Reactor Coolant System (RCS) chemistry specifications were available:

Parameter	Value
Solution pH	Determined by the concentration of boric acid and alkali present. Expected values approximately 7
pH Control Agent (Li^7OH)	3.5 ppm
Boric Acid, ppm B	Variable from 0 to approximately 3000

4.3.3.3 Radiation levels

4.3.3.3.1 During operation

A way to assess the overall affect of power uprates on radiation levels during operation is to study the occupational exposures obtained during reactor operation. Data from Tihange 2 is summarized in **Figure 4.3.13**. It must be noted, that the annual exposure during operation is normally only a small fraction, 5-20% of the total occupational exposure. Therefore, the data in **Figure 4.3.13** shows some variation, but no clear correlation to the power uprate is seen. The occupational exposure has been decreasing in the last decade.

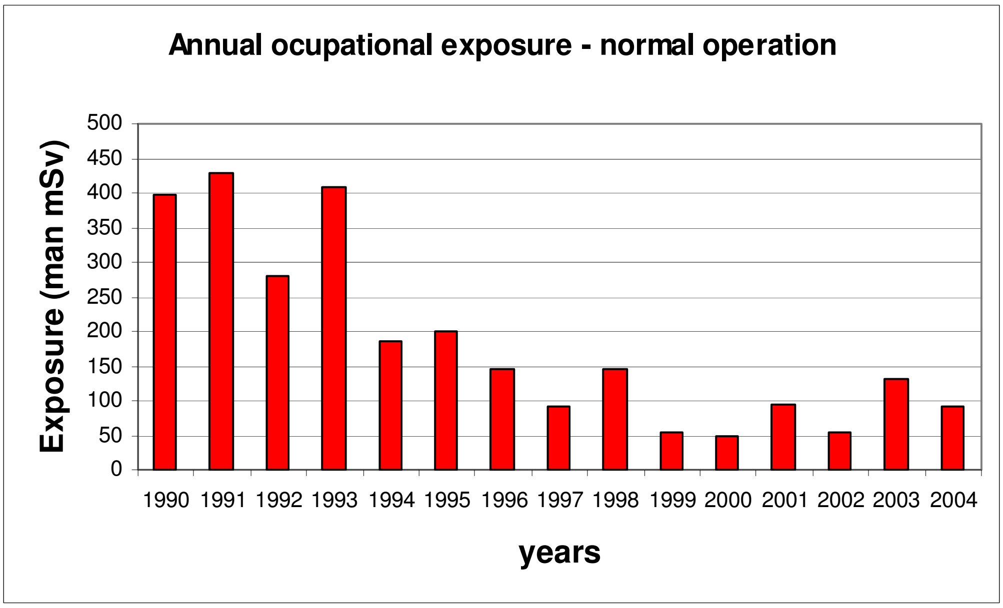


Figure 4.3.13: Tihange 2 – Annual occupational exposure

4.3.3.3.2 During outage

Dose rate during outage conditions have been regularly followed-up. Measured dose rates on survey points around primary piping (primary circuit and channel head hot and cold legs) at outage conditions are shown on **Figure 4.3.14**. The dose around the channel head is the biggest fraction, approximately 95%, of the total measured dose on survey points. It can be seen that radiation level decreases after SG replacement, year 2001.

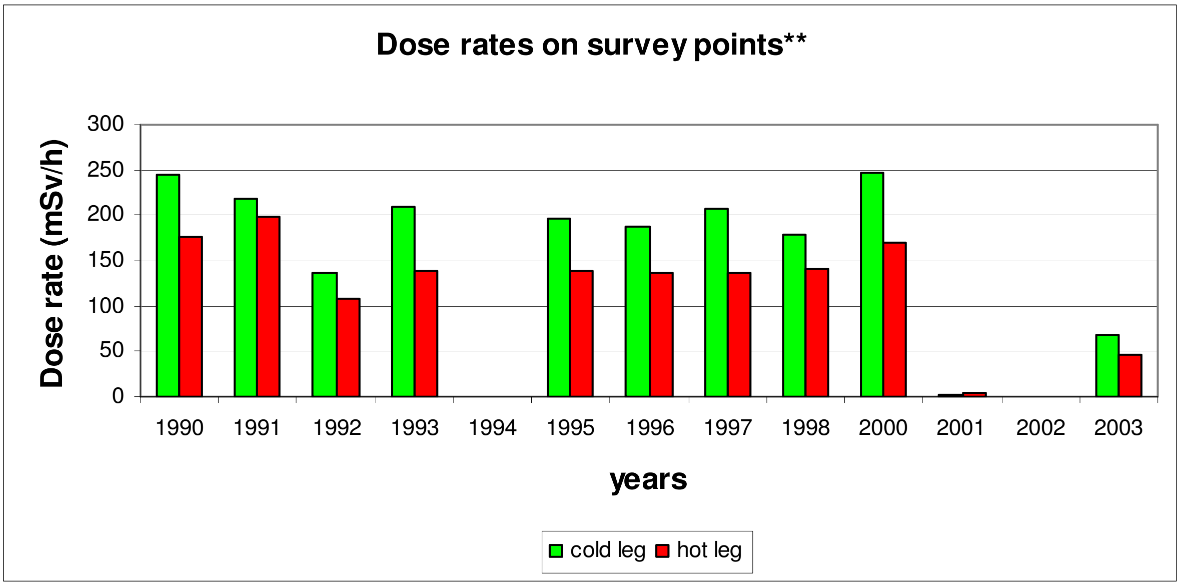


Figure 4.3.14: Tihange 2 – Dose rates on survey points**

* There was an outage in the year 2002 but dose rate data is missing for year 2002

**survey points: Channel head / Primary circuit

4.3.3.4 Occupational exposures

The annual occupational exposures during the period 1990 – 2003 are presented in the **Figure 4.3.15**. The exposures are divided into contribution during outage and operation conditions. The exposures during operation are only a small fraction of the total exposures; the trend is shown on the **Figure 4.3.13**. Data on exposure per outage day is compiled in **Figure 4.3.16**. The exposure per outage day shows some variation, but what is important, it that it a shows decreasing trend over the last decade. The outages are responsible for the major part of the collective doses. In the year 2001, the steam generator replacement was responsible for half of the collective dose in Tihange 2.

There is a correlation with the outage lengths (**Figure 4.3.12**), i.e. outages involving large efforts for the power uprate and plant modernization projects have consequently implied increased exposure. It can also be seen from exposure data for 1997, that the outage length was longer than most of the others but the exposure was not very high; no bigger modernization activities were performed.

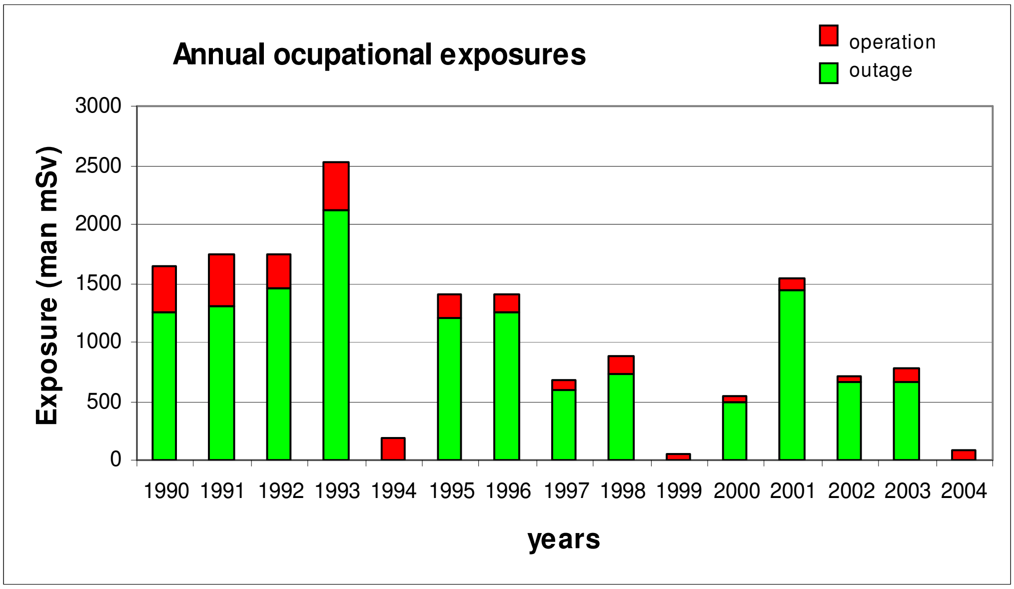


Figure 4.3.15: Tihange 2 Annual occupational exposures split-up on outage and operation conditions

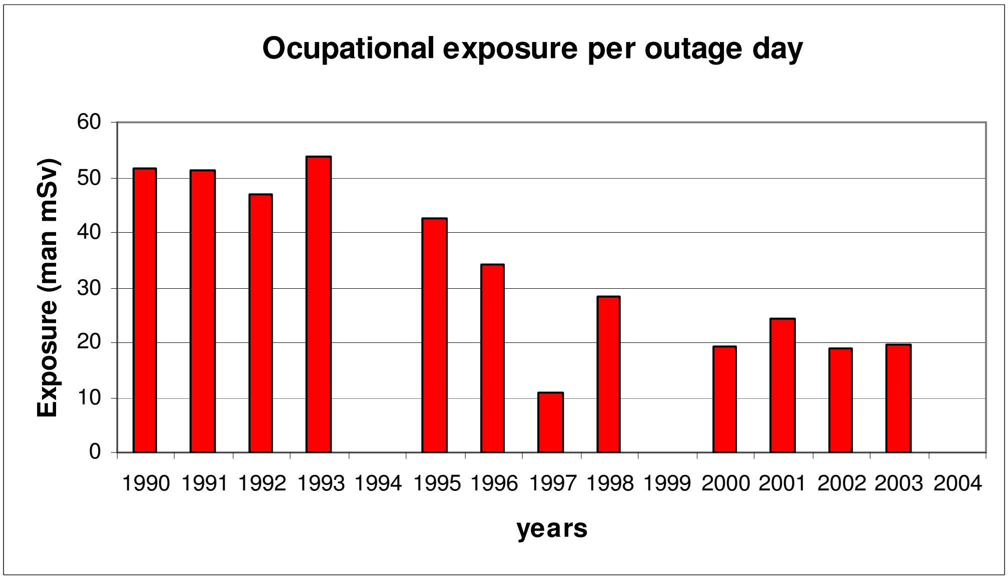


Figure 4.3.16: Tihange –Occupational exposures per outage day

The Tihange 2 annual exposures are compared to Belgium and some international data in **Figure 4.3.17**. Collective doses in Tihange are increasing compared to 2000. This is due to the outage and steam generator replacement. In 2003 and 2002 there was outage in Tihange 2. In 2003, there was a supplementary stop at Tihange 2 for pressuriser's leg welding inspection. There was insufficient data after SG replacement to evaluate the real exposure trend but it seems to be decreasing.

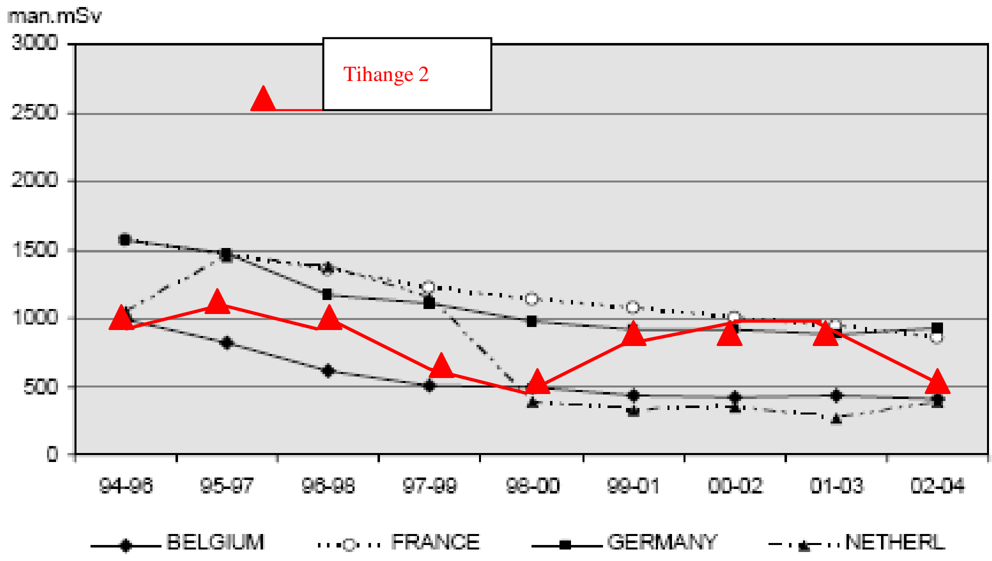


Figure 4.3.17: Average outage collective dose per reactor type and per country

4.3.3.5 Summary and conclusions

A review of data and experience from the Tihange 2 plant has been performed. The review has resulted in the following conclusions:

- The plant has been uprated twice since the commissioning. The thermal power of the reactor was increased from 2785 MW to 2905 MW in 1995 and to 3064 MW in 2001. The present study focuses on the Tihange 2 second uprate, resulting in a thermal power level of 10% compared to the initial power level.
- In the 2001 three SG were replaced in 63 days. Typical outage lengths are between 30 and 40 days. The total annual exposure in 2001 was 1539 man mSv. Outage exposure was 1446 man mSv, and half of it was due to SG replacements, which indicates good organization.
- Standard PWR water chemistry is maintained, no zinc is injected.
- Since the uprate exposures during operation have been maintained on a constant and rather low level.
- Radiation levels during outage on reactor systems have been maintained on a rather low and constant level after SG Replacement.
- Considerable manhour efforts spent during some outages for the plant modernization program have of course resulted in some increase of occupational exposures. The exposures per outage day have shown a decreasing trend in the last decade. The average annual exposures in the Tihange 2 have been kept on a level comparable to international values for PWR plants.

4.3.4 Comparison of two PWR uprates

For two selected PWR plants, Asco 1 and Tihange 2, the SG replacement and the modernisation activites were analysed for tehri contribution to occupational doses.

Typical values for:

- Outages length
- Number of personnel (plant and outside)
- Collective dose by job and type of personnel
- Collective dose by task and type of personnel
- Collective dose by occupational category and type of personnel

were reviewed and are presented in the **Table 4.3.3** and **Figure 4.3.18-Figure 4.3.25**.

Table 4.3.3: Specific elements of the PWR plants for the year of uprate

	ASCO 1	TIHANGE 2
Annual collective dose by type of personnel (man mSv) • Plant • Outside	 129 5239	 152 1387
Outage length (SGR + other outage activities)	95 days 61 days per SGR	63 days 52.5 days per SGR
Number of outage personnel • Plant • Outside	 335 2057	 324 1235
Collective outage dose (man mSv) • Plant • Outside	Planned out. Forced out.* * SGR 113 1.35 1329 3827	Planned out. Forced out. 90 - 1356 -
Collective dose by task (man mSv) SG Replacement	 2443 man mSv	 648 man mSv

If normal tasks are compared with tasks performed in the outage per particular year:

- Refuelling
- Reactor Vessel activities (Inspection, Maintenance)
- Activities on SG Primary side
- Activities on SG Secondary side
- HRS&SIS systems
- RC pumps
- Valve work
- Routine Inspection
- Scaffolding
- Insulation
- Large tasks (as SG Replacement)

There is no big difference in the extent of work per outage, there are some deviations but not significant. Most of the activities at Tihange plant are performed with smaller occupational exposure. **Table 4.3.3**, shows that a large task like SG Replacement is performed in a more optimised manner at Tihange plant.

If a comparison of exposure per occupational category in both plants is made, high exposure is received by mechanical personnel, inspection personnel and decontamination personnel

The following tables show comparisons for Tihange and Asco for the selected task of SG replacement. The comparison period was two years prior to the replacement and two years after. It has to be noted that Asco 1 performed SG replacement in 1995 and Tihange 2 in 2001.

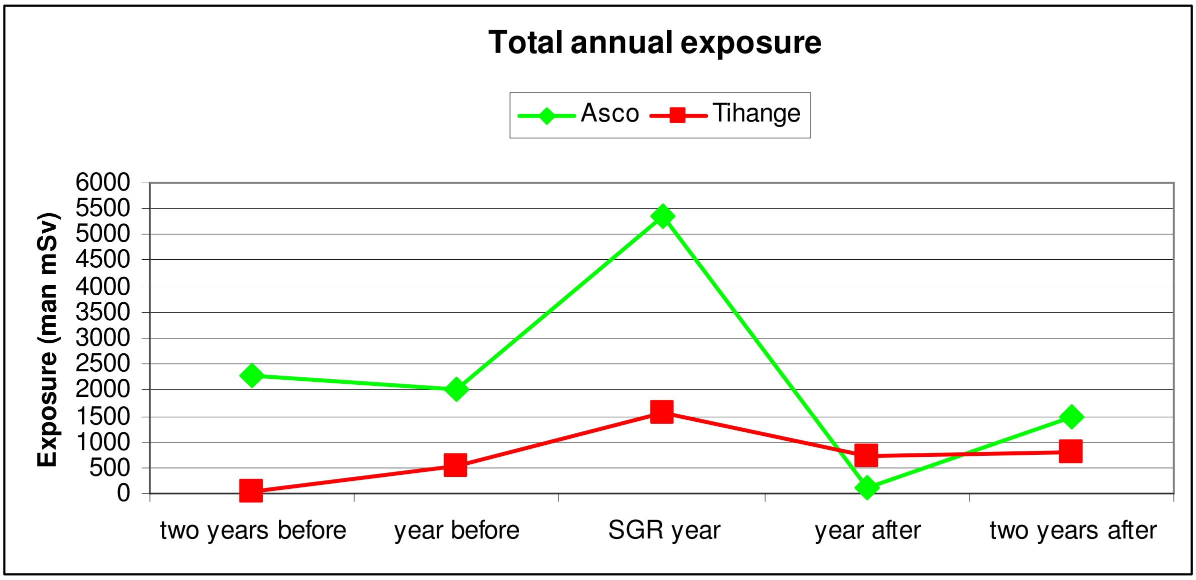


Figure 4.3.18: Total annual exposure for five years period

** Asco 1 did not have outage in the year after SG Replacement

Tihange 2 did not have outage two years before SG Replacement

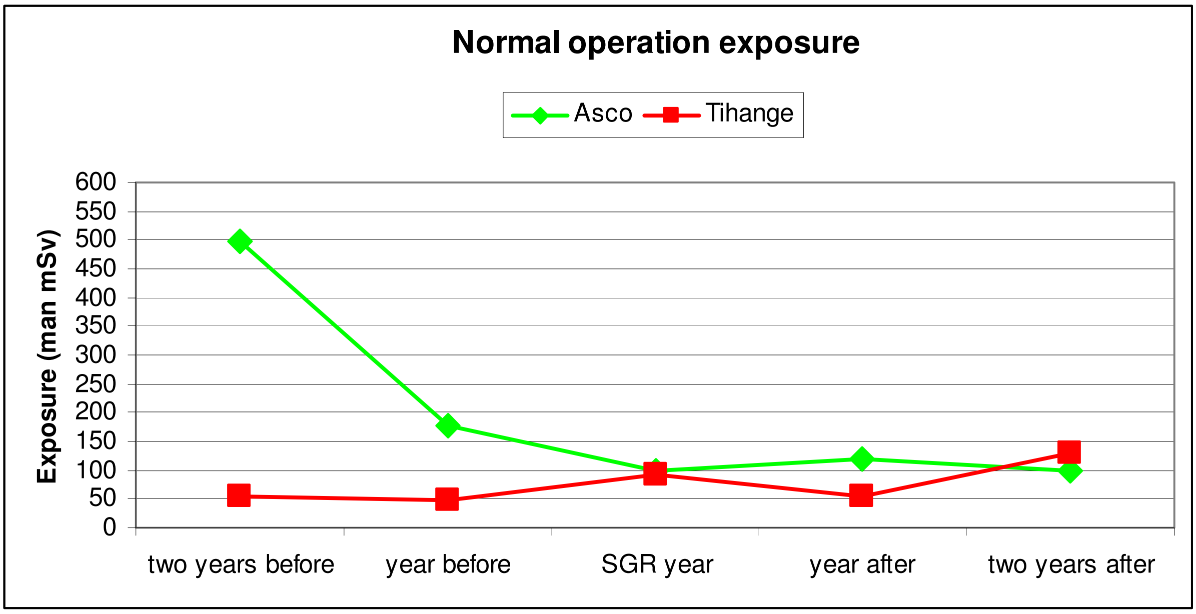


Figure 4.3.19: Normal operation exposure for five years period

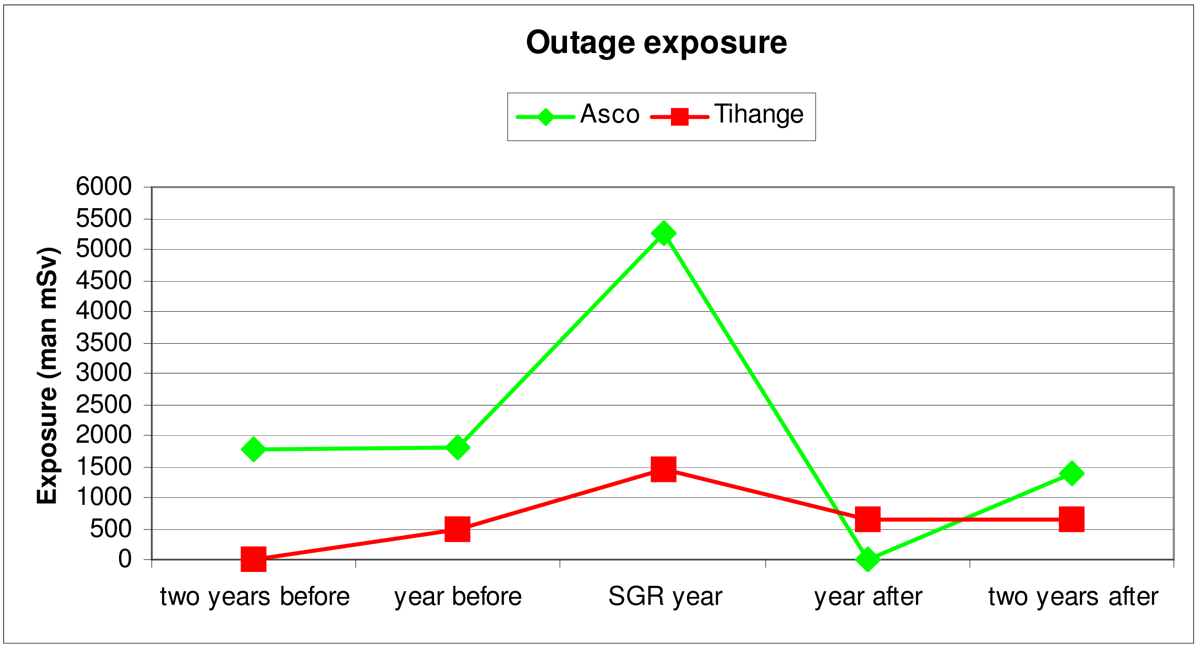


Figure 4.3.20: Outage exposure for five years period

** Asco 1 did not have outage in the year after SG Replacement

Tihange 2 did not have outage two years before SG Replacement

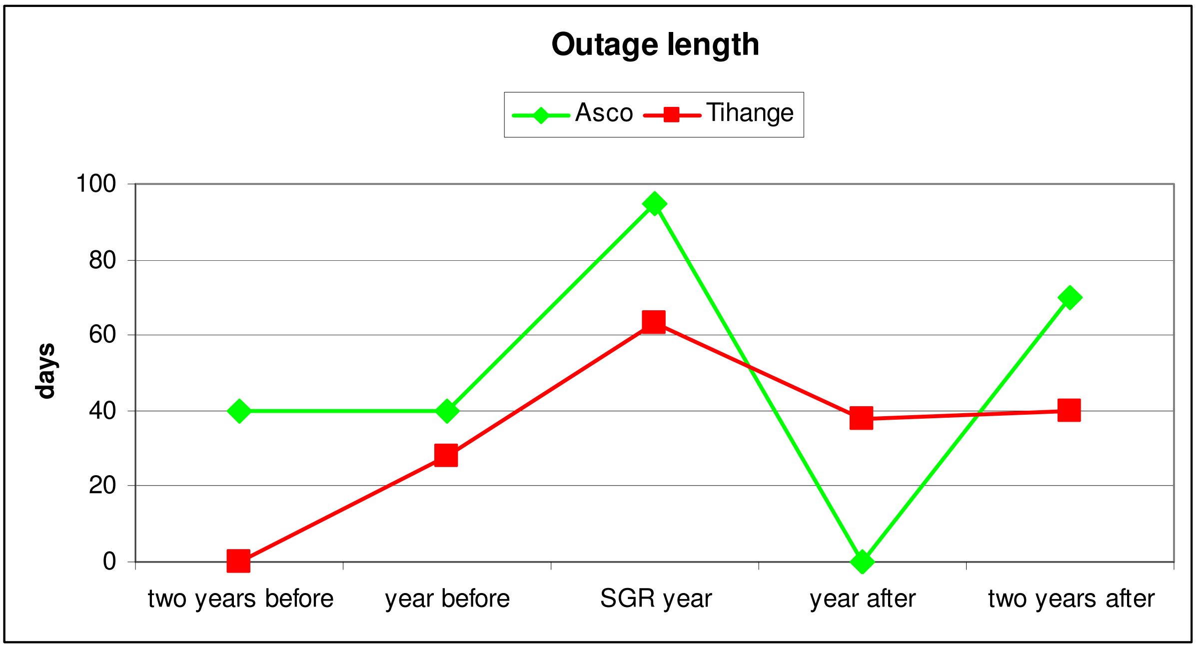


Figure 4.3.21: Outage length for five years period

** Asco 1 did not have outage in the year after SG Replacement

Tihange 2 did not have outage two years before SG Replacement

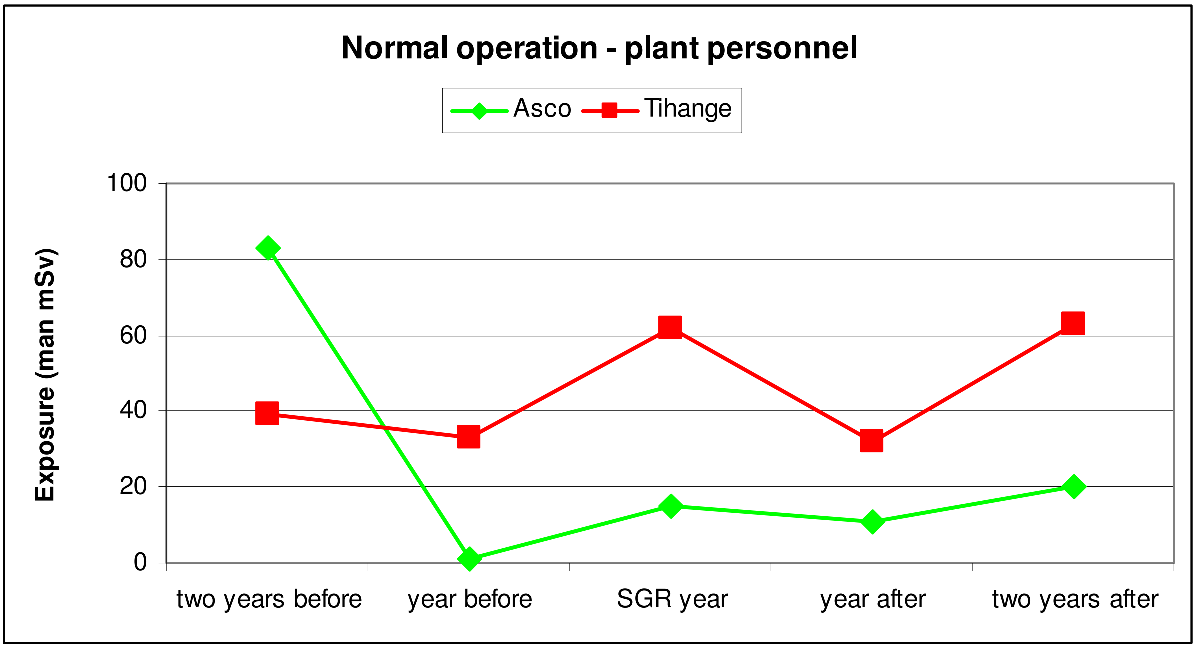


Figure 4.3.22: Normal operation exposure - plant personnel

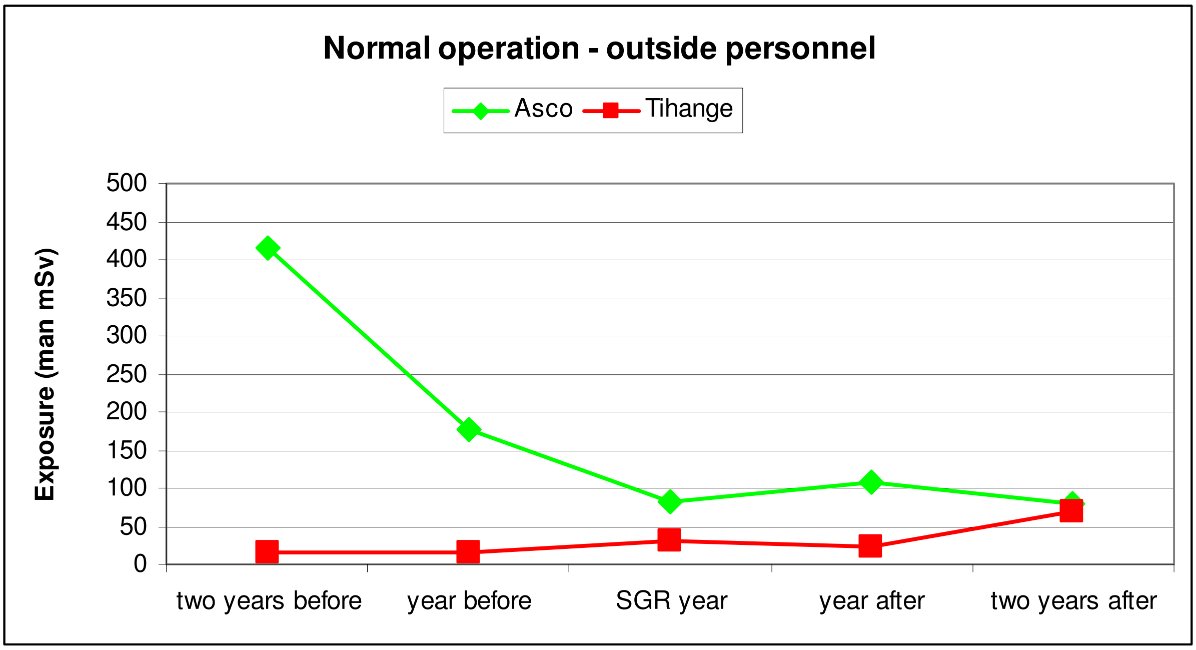


Figure 4.3.23: Normal operation exposure-outside personnel

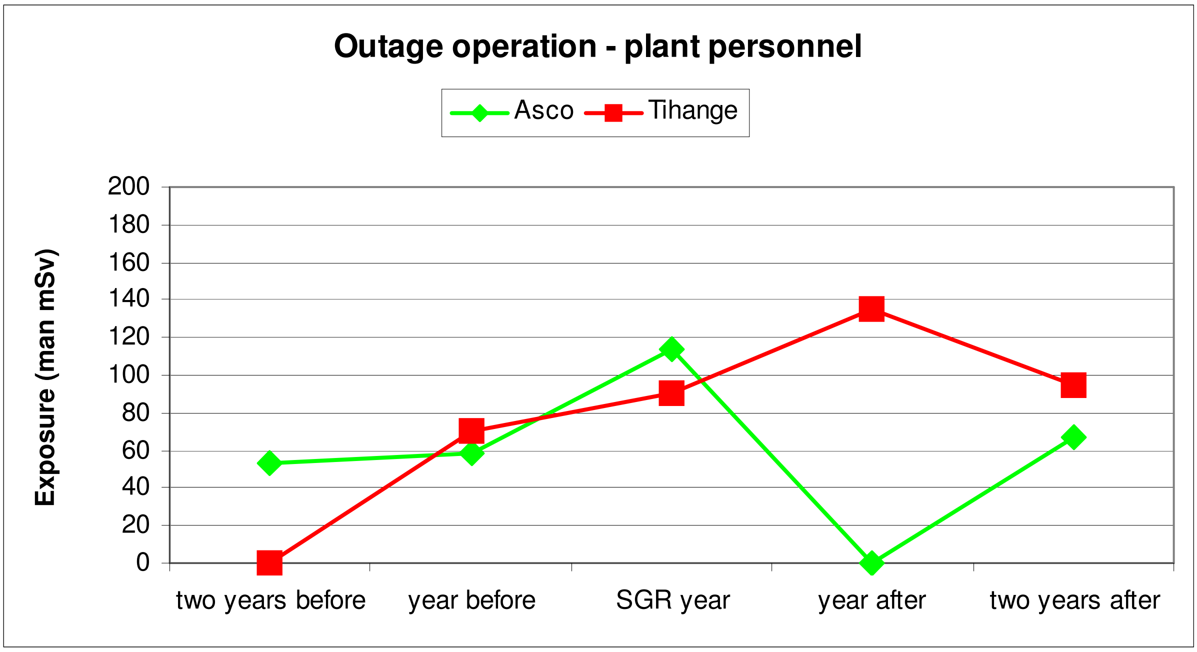


Figure 4.3.24: Outage operation exposure-plant personnel

** Asco 1 did not have outage in the year after SG Replacement

Tihange 2 did not have outage two years before SG Replacement

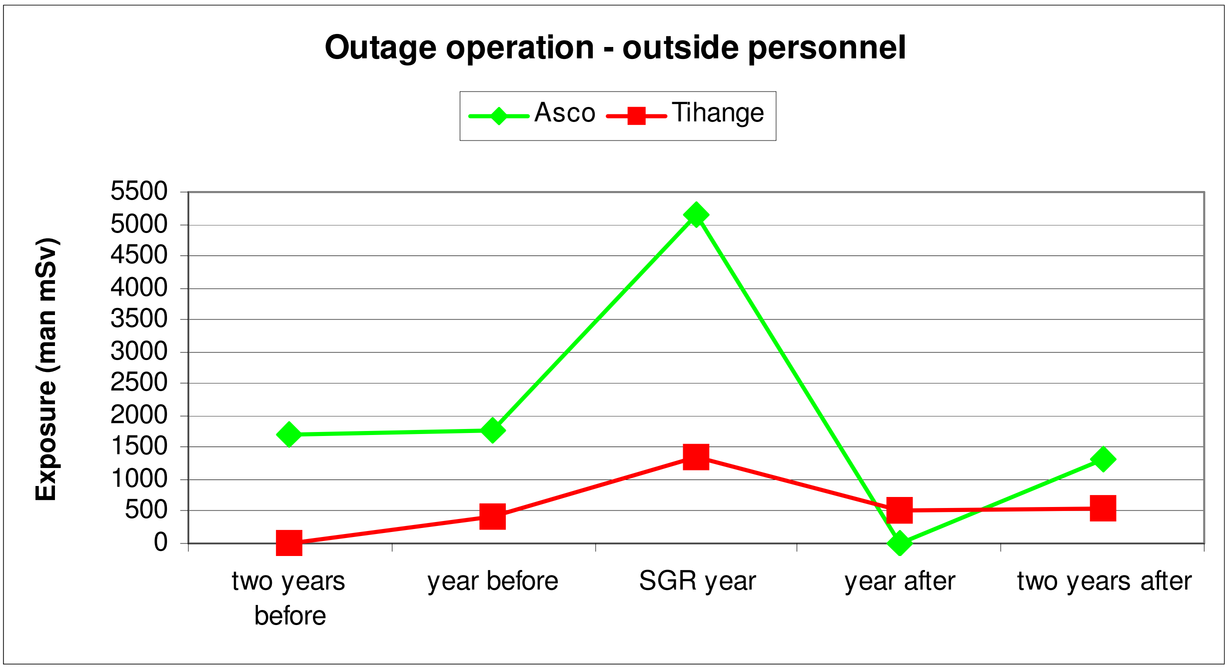


Figure 4.3.25: Outage operation exposure-outside personnel

** Asco 1 did not have outage in the year after SG Replacement

Tihange 2 did not have outage two years before SG Replacement

From the figures **4.3.18-4.3.25** and **Table 4.3.3** following conclusions emerge:

- For the five year period (two years before and after SGR), total exposure, including outage and normal operation exposures are greater for Asco plant. The only exception is the year when there was no outage in the Asco plant. Asco plant has longer outages than Tihange plant and contractors are more loaded. Even during normal operation most of the activity in the Asco plant is performed by outside personnel.
- There are some visible differences in exposures data for Asco 1 and Tihange 2 unit. Some facts have to be underlined:
  - Leadership, composition and organisation of the power uprate, including radiological activities, especially large demanding tasks are critical for successful implementation of power uprate and received doses. When exposures are, compared for SG Replacement it can be noted that occupational exposure in Asco 1 was almost four times higher than in the Tihange 2. However, it must be noted that SG replacement in Asco was performed in year 1995, and SG Replacement in Tihange was performed in year 2001.
  - A prerequisite for applying the principle of optimisation to occupational radiation protection is the appropriate and timely exchange of data, techniques and experience on doses and dose reduction methods. The Information System on Occupational Exposure (ISOE) was launched in year 1992 after a two-year pilot programme, with aim of helping utilities in sharing information and experience worldwide. Thus, Tihange plant had more available data,..
  - Most of the activity in normal operation and outage in the Asco plant are performed by outside personnel. Event reports from worldwide Operating Experience demonstrate that it is not recommended to allocate the responsibility for important activities to outside personnel without strong supervision by plant personnel. Outside personnel are not as well-trained or as acquainted with plant design details as plant personnel. If outside personnel perform activities they have to be trained and retrained to stress the potential for abnormally high or rapidly changing radiological conditions, including the the actions required when these conditions occur. Training should also emphasize the importance of a high level of awareness and sense of individual responsibility with regard to personnel radiation protection.
- Although the exposures are higher for Asco plant, what should be emphasised is the decreasing trend of exposures during last decade. The most important impact in the reduction of collective dose was the removal of the old RTD System, the SG Replacement, the substitution of the satellite and the ALARA Program.

5. Reconstruction experience

The aim of task#3 was to identify good (and bad) practices that are impacting on the doses to personnel and could be related to an uprate. From the perspective of occupational exposure it should give an answer to what kind of design, implementation and operational arrangement are the best, Optimisation of the work processes to limit the duration of the time spent in the controlled areas is specially highlighted. Leadership, composition and organization of the large demanding tasks are critical for successful implementation of power uprate and received doses

5.1 Technical factors

5.1.1 Introduction

In the following sections, analysis of the outputs of task 2 and task 3 of the project have consisted of a review of the important factors in BWRs and PWRs which affect radiation levels and occupational exposures in general, and especially at power uprates.

The following sections review technical factors important for radiation fields in BWRs and PWRs, and how these are affected by power uprates.

5.1.2 BWR uprates

5.1.2.1 Water chemistry issues

5.1.2.1.1 Corrosion product balance

The water chemistry control in BWRs to combat radiation buildup is largely based around the optimisation of the corrosion product balance in the primary circuit. Six different general types of corrosion product balance are schematically illustrated in the **Figure 5.1.1**. It is concluded that the fuel crud composition should be well balanced, i.e. the relation between iron (Fe) and nickel (Ni) plus zinc (Zn) should be maintained close to the spinel relation, with only a small excess of Fe. A significant inflow of Fe may results in fuel corrosion problems, especially if the Fe inflow is occurring together with a considerable inflow of Zn (and copper[6] (Cu)). High inflow of Fe normally results in a fuel crud that is prone to release particles resulting in hot spot radiation sources in the plant. However, a too low inflow of Fe may lead to formation of less stable monoxides, resulting in increased reactor water activity concentrations, and, particularly in connection to somewhat increased steam moisture content and increased turbine plant radiation levels. Fuel crud with a too high Ni/Fe ratio also seem to be involved in increased local fuel cladding corrosion, especially for fuel with spacers of Inconel.

[6] Cu shall be avoided for several reasons. Cu is known in several cases to cause fuel cladding corrosion (CILC). Cu also makes HWC operation less effective. There are, however, some indications that a moderate amount of Cu in the fuel crud may form oxide forms that have a high affinity for cobalt (Co), resulting in reduced Co-60 reactor water activity.

The above general recommendations indicate different actions whether the plant is operating with NWC, HWC or has applied NMCA.

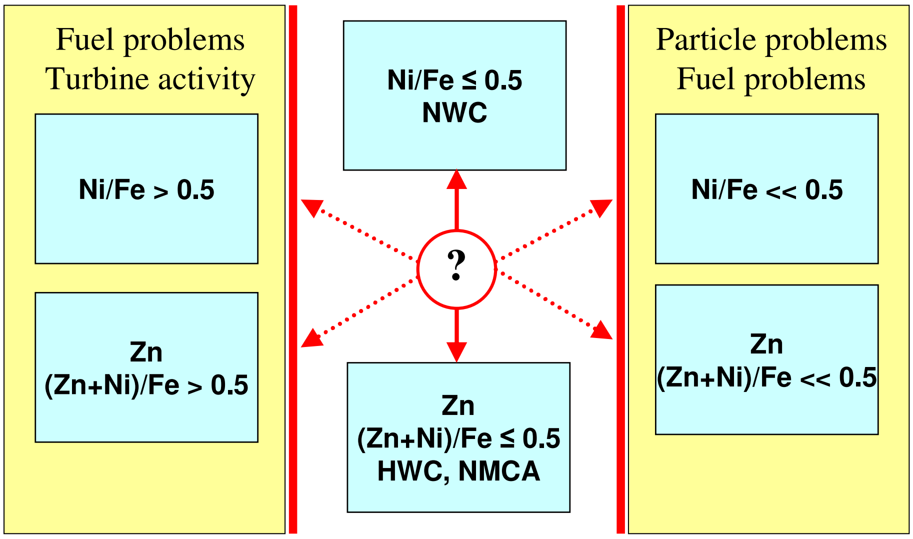


Figure 5.1.1: *BWR* Corrosion product balance Fe/Ni/Zn - Six different cases: Where to go?

The Fe inflow in plants operating with NWC is very much dominated by the inflow by the feedwater. That means that the key to controlling the fuel crud composition is to control the feedwater chemistry, and especially the feedwater Fe. The most recent EPRI water chemistry guideline proposes that the feedwater Fe concentration during NWC conditions should be maintained in the interval 0.5 – 1.5 ppb. The proposed amount of Fe is probably rather conservative, especially in the case with low feedwater Zn. Experience from Scandinavian BWRs has shown, that a well balanced fuel crud with respect to Fe and Ni can be maintained with a feedwater Fe concentration as low as about 0.2 ppb, if the Zn level is low. This level is supported by typical fuel crud Ni amounts measured. Zn in the feedwater, natural or injected, increases the amount of feedwater Fe needed to maintain a fuel crud of the spinel type:

Eq. 1 $$^{FW}C_{Fe} \geq 0.2 + 2.3 \cdot {}^{FW}C_{Zn}$$

where:
$^{FW}C_{Fe}$ *– Feedwater Fe concentration [ppb]*
$^{FW}C_{Zn}$ *– Feedwater Zn concentration [ppb]*

A control of the feedwater Fe according to ***Eq. 1*** will result in a fuel crud close to the ideal spinel type, i.e. further injection of Zn is not needed and will only result in an increased demand of feedwater Fe. On the other hand, if the minimum feedwater Fe is not easily obtained, Zn injection can help to improve the characteristic of the fuel crud. The 1.5 ppb EPRI 2004 upper bound of feedwater Fe corresponds to a feedwater Zn level of maximum 0.6 ppb, which corresponds to the recommended maximum EPRI level. Higher Zn (and Fe) levels may result in increased fuel cladding corrosion.

The prerequisites for Fe control change considerably when HWC operation is applied, and even more when NMCA application is performed. The feedwater is no longer the only source of Fe, considerable contribution is also expected from sources in the reactor circuit obtaining low corrosion potential. These internal sources are not easily monitored, and significant variation may exist between different plants. Other than reactor design features, the influencing factors are the degree of H_2 injection in the case of HWC, the degree of NM coverage in the case of NMCA, and the pre-history with respect to feedwater Fe and Zn inflow. The recent EPRI recommendations consider this effect, and low feedwater Fe, acceptable. Recommended interval for feedwater Fe is 0.1 – 1 ppb. The lower limit represents the limit of today's US BWR experience, and a practical interpretation is that actually no lower limit exists in the case of HWC and NMCA plants. In practice, large efforts are made in US plants to lower the feedwater Fe input.

In the case of HWC and NMCA plants, Zn injection seems to be especially effective in forming a more stable fuel crud composition. However, the amount Zn injection needed is not so easily determined due to the above mentioned, non-monitored internal sources of Fe. The recent EPRI guidelines propose that a reactor water Zn level of >5 ppb shall be maintained in HWC plants. In the case of NMCA plants a relation between soluble ^{60}Co and Zn is proposed instead ($<2.0\cdot10^{-5}$ μCi/g per ppb, or <720 Bq/kg per ppb), which in reality, normally means a somewhat lower reactor water Zn level than 5 ppb. However, the reactor water specifications have to consider the proposed feedwater Zn limits, <0.6 ppb in HWC plants and <0.4 ppb in NMCA plants, which may override the reactor water limits. The feedwater Zn limits are due to fuel concerns.

As mentioned above, the Zn injection may be complicated to control due to the non-monitored sources of Fe in HWC and NMCA plants. Therefore, one Scandinavian HWC plant with low feedwater Fe has used an alternate way of controlling the feedwater Zn injection based on relation between reactor and feedwater Zn:

Eq. 2 $$^{FW}C_{Zn} \geq 2 \cdot \frac{f_{RWCU}}{f_{FW}} \cdot {}^{RW}C_{Zn}$$

Eq. 3 $$^{RW}C_{Zn} \leq {}^{Max}C_{Zn}$$

where:
$^{FW}C_{Zn}$ – Feedwater Zn concentration [ppb]
$^{RW}C_{Zn}$ – Reactor water Zn concentration [ppb]
$^{Max}C_{Zn}$ – Maximum allowed reactor water Zn concentration [ppb]
f_{RWCU} – Reactor water cleanup flow [kg/s]
f_{FW} – Feedwater flow [kg/s]

The above relation ***Eq. 2*** means that at least about half of feedwater Zn shall be consumed in restructuring of fuel crud and system surface oxides, and maximum about 50% of the feedwater Zn is allowed to be cleaned-up by the RWCU. The feedwater Zn is primarily adjusted to reach the reactor water Zn target level, $^{Max}C_{Zn}$ (***Eq. 3***). If the Zn target level cannot be reached together with the relation ***Eq. 2***, the Zn injection is decreased to a point where ***Eq. 2*** is fulfilled and a somewhat lower reactor water Zn level than the target is accepted. This operation strategy is to ensure that a certain iron surplus in the fuel crud is maintained.

A power uprate in a BWR plant certainly has the potential to affect the corrosion product balance. The demands on the condensate polishing plant are normally increased due to higher flow rates, increased operation temperature and higher inflow of Fe from erosion corrosion. In many uprate projects the turbine design is also changed from a full-flow condensate cleanup to forward pumped heater drains with less cleanup of the condensate. In other words, there is a risk that the outcome of a power uprate is increased feedwater inflow of corrosion products, especially Fe, which has to be addressed in the project. On the other hand, the turbine design modifications included in a power uprate project imply a potential to introduce improvements. Replacements of e.g. steam extraction pipes from previous carbon steel to low-alloy steel will significantly reduce the erosion corrosion Fe, something that is especially important in forward pumped designs (see **Section 4.2**). Another modification sometimes introduced is the revamping of the turbine condenser from previous brass to titanium with a reduction of the Cu inflow to the primary circuit (see **Section 4.2**). Furthermore, large efforts are spent in many BWRs to improve the performance of the condensate polishing plants. An example of such an improvement are the modifications performed in the OL1/2 plants (see **Section 4.1**). A review among the 39 BWRs (36 North American, 3 European) that participated in the EPRI BWR chemistry monitoring database in 2005 has been made and was reported [2] at the 2006 water chemistry conference in Korea. There are three configurations of condensate polishing systems employed: deep bed demineralizers only, filter + deep bed, and precoated filter demineralizers. Feedwater iron, feedwater copper, and reactor water sulphate are three key parameters that are of particular interest for assurance of fuel reliability, mitigation of intergranular stress corrosion cracking (IGSCC), and achievement of dose reduction goals.

Deep bed (DB) demineralizers have a high capacity to remove ionic impurities but limited capability to remove particulate impurities (such as iron and copper oxides).

Filter + Deep Bed (F+DB) systems have deep bed demineralizers downstream of high efficiency backwashable filters. These filters are commonly called prefilters and are used without precoat materials. In North America, all prefilter systems currently employ filter septa with pleated filter media. The Filter + Deep Bed design provides both efficient removal of insoluble crud particles and high capacity to remove ionic impurities.

Filter Demineralizers (FD) employ vertical cylindrical filter septa (either pleated or wound) that are precoated with filter aid materials made up totally or partially of powdered cation and anion exchange resins. This design provides efficient removal of insoluble crud particles but has only limited capacity to remove ionic impurities.

Measures taken by the industry to improve iron control have been successful. This is evident in the downward trend from 1997-2005 in average feedwater iron results for 39 BWRs shown in **Figure 5.1.2**. A 63.8% reduction in average feedwater iron occurred between 1997 and 2005, with a 19.3% reduction between 2004 and 2005, as shown in **Figure 5.1.3**, this trend is expected to continue as more Deep Bed Only plants retrofit prefilters and as more Filter Demineralizer plants implement the use of lower particle retention rating septa.

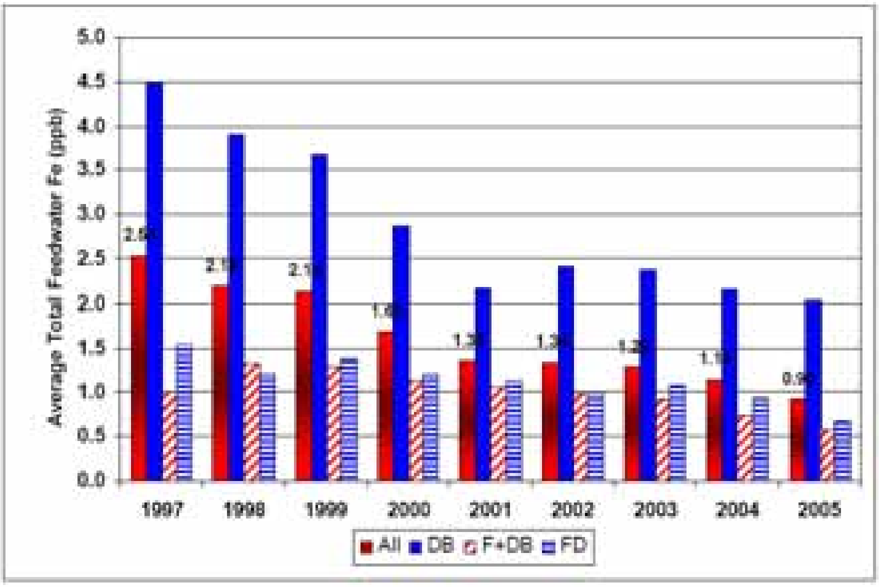


Figure 5.1.2: BWR Feedwater Iron Trend from 1997-2005
for 39 BWRs by Condensate Polishing System [2]

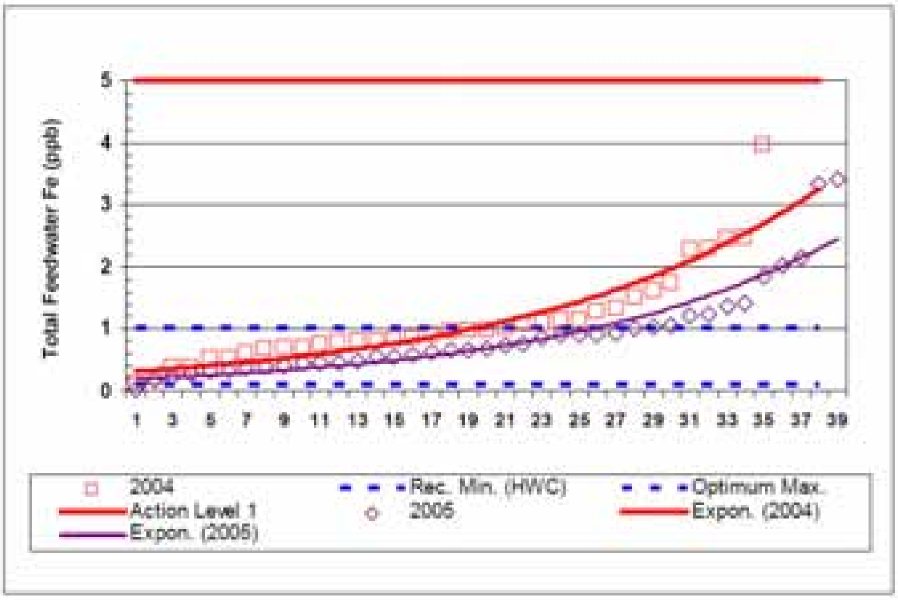


Figure 5.1.3: Individual BWR Feedwater Iron Trend for 2004 and 2005 [2]

When planning a power uprate, the effects of increased flow and increased inlet iron must be addressed. The effects of increased flow and increased inlet iron are multiplicative; i.e., a 10% increase in flow combined with a 10% increase in inlet iron results in a 21% increase in filter iron loading.

Increased condensate flow results according to [2] in:

- Higher area flow rates which affect filtration and ion exchange;

– Higher filter dP resulting in shorter runs and shorter septa life and higher radwaste volumes; and

– Lower system dP margin which may further limit useful septa life.

Increased inlet iron results in [2]:

– Higher dP resulting in shorter runs and higher radwaste volumes; and

– More solids to be processed in radwaste than would be expected by flow increase alone.

Stations have addressed these issues in a variety of ways, including:

– Installation of an additional service vessel;

– Increasing septa length and/or diameter;

– Planned bypass flow during times when a vessel is out of service for backwashing.

The overall BWR trend is improved feedwater chemistry in spite of power uprate projects in many BWRs.

5.1.2.1.2 Cobalt sources

Co-60 is the dominant radiation source in BWR plants, and it is, therefore, of utmost importance to assess, and possibly reduce, the sources of cobalt (Co) in the plant. The measurement of the very low concentrations of Co in feed- and reactor water is not an easy task, and the industry has spent lots of efforts to map the actual Co sources. It has become evident, that the material that contributes most to the Co source is Stellite, which contains about 60% of Co. This material is especially used in valve seats and in pumps, in both reactor and turbine systems. Some low-pressure turbines also have Stellite on the blade tips. Of special interest is the use of Stellite as "pins and rollers" on control rods. The Stellite on control rods is neutron activated, and the corrosion means an inflow of Co-60 to the primary circuit in parallel to the Co inflow. Many plants have been successful in Co reduction programs, but many applications with Stellite are not easily replaced with Co-free substitutes.

Available Stellite replacement materials are further discussed in the PWR section.

Forsberg et al., 2004 [3], report laboratory measurements of Co release from Stellite in different BWR water chemistries, see **Figure 5.1.4**. Release rates are lower in a reducing environment than in a oxidising one and dosage with Zn causes a further reduction. A change from NWC conditions without Zn to HWC conditions with Zn causes a reduction in Co release by almost an order of magnitude. The effect is one of several explanations for the beneficial effect of Zn injection, especially at HWC conditions.

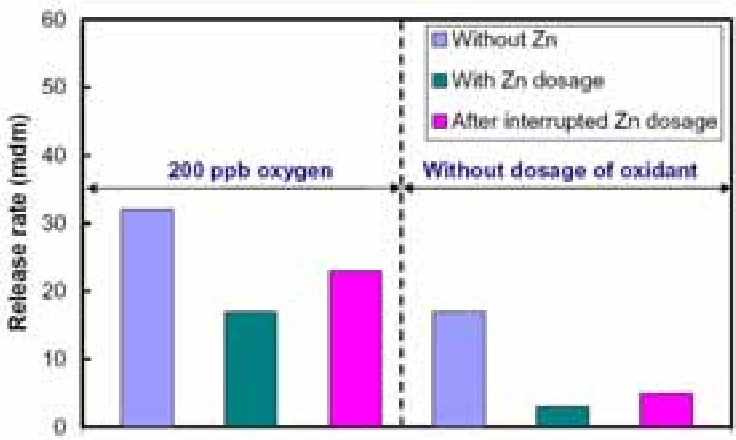


Figure 5.1.4: Laboratory testing of Co release rate from Stellite –
Influence of HWC and Zn, Forsberg et al., 2004 [3]

Stellwag & Staudt, 2004, report data from a German plant indicating that the replacement of control rods containing Stellite with Co-free substitutes has resulted in a significant reduction of reactor water Co-60, see . 116 out of 145 control rods were replaced, and an about 50% reduction of the reactor water Co-60 was observed, and a corresponding long-term reduction of radiation fields are expected. Similar assessments have been made for other BWRs, and somewhat lower contribution to the Co-60 activity has been obtained.

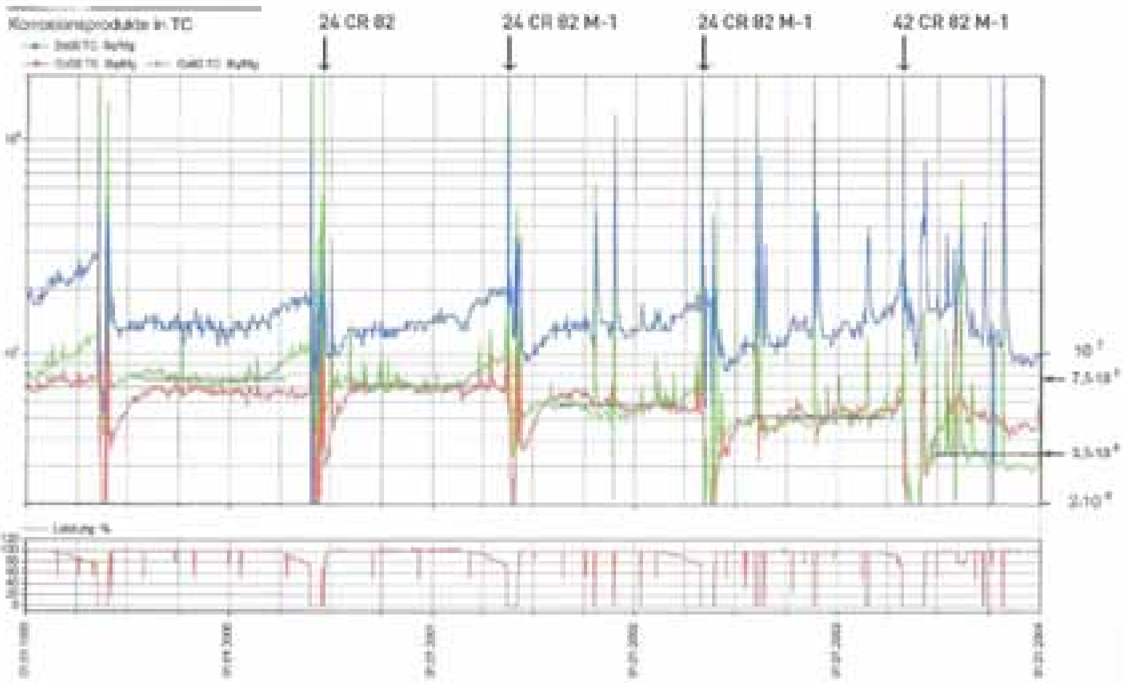


Figure 5.1.5 Reduction of Co-60 *RW* activity in a German *BWR* due to control rod replacement without Co containing buttons and rollers (116 out of 145 CRs replaced) (green line: Co-60, red line: Co-58, blue line: Zn-65 [Bq/Mg]), Stellwag & Staudt, 2004

Plants with full-flow condensate cleanup normally have only a minor contribution to the total Co inflow from the turbine plant, i.e. the major Co contribution is from Stellite in reactor systems. Plants with turbine design employing forward pumped heater drains (FPHD) normally display feedwater Co input that is comparable or even higher than the reactor system sources, i.e. FPHD plants normally have a somewhat higher total input of Co to the primary circuit than full-flow cleaned-up plants.

Most plants have Co reduction programs, where a gradual replacement of Stellite in valves and other applications are planned. However, a problem with such programs is the lack of knowledge of the exact Co contribution from different sources[7]. The outcome of performed modifications is not always easy to measure due to the extremely low concentrations of Co and sometimes there are problems with high background levels due to sources of Co in the sampling system (e.g. Stellite in sampling valves). A good way to monitor the long term trend of Co inflow to the reactor circuit is to regularly perform fuel scraping campaigns to access the accumulation of Co and other corrosion products in the fuel crud.

One additional way to reduce Co inflow is to introduce improved methods to avoid intrusion of Co due to grinding of valve seats during outages. Such methods are strongly emphasized, being an important part of the overall Co reduction program.

The BWR power uprates studied in more detail, OL1/2 (**Section 4.1**) and CNC (**Section 4.2)** have not included major replacement of Stellite in valves. The experience of such replacements is rather sparse. The Ringhals 1 plant has conducted a considerable reduction of Stellite in the recirculation loops and in the LP turbines during performed modernization projects, and a significant gradual reduction of Co-60 radiation has been experienced. Some reduction of feedwater Co has also been seen in the Swedish forward pumped heater drain plants Forsmark 3 and Oskarshamn 3 after the replacement of Stellite in the turbine plants.

5.1.3.1.3 NWC – HWC – NMCA – OLNC

The method employed to establish reducing conditions in parts of the BWR primary circuit is to inject hydrogen into the feedwater. This technique was first demonstrated in 1979 in the Swedish Oskarshamn 2 plant, and was introduced in several plants during the 80's. More than 50% of all BWRs are today on HWC in various degrees. The most recent EPRI water chemistry guidelines from 2004 specifies that BWR components <u>must</u> be exposed to reducing conditions (i.e. <-230 mV (SHE)) to mitigate stress corrosion cracking initiation and growth. According to EPRI this goal can be achieved in two different ways:

1. By injecting enough hydrogen to the feedwater to maintain the electrochemical corrosion potential (ECP) of reactor internals below -230 mV (SHE). This method is defined as moderate hydrogen water chemistry (HWC-M).
2. By treating the reactor internals with noble metals (Pt and Rh) and injecting small amounts of hydrogen into the feedwater to maintain a hydrogen to oxygen molar ratio of ≥3 (which assures an ECP value <-230 mV (SHE) for reactor internals). This method is called noble metal chemical addition (NMCA).

[7] There are examples of small regulation valves where erosion of valve seats means a considerable loss of Co material.

Most European plants, especially Scandinavian and German BWRs with internal recirculation pumps, maintain operation on NWC. In most of the European plants, the reason for this difference between plants is due to more resistant materials from environmental cracking point of view The HWC technique relies on suppressing the radiolytic production of O_2 and H_2O_2, the oxidants responsible for promoting high ECP values on BWR components, from values of several hundred ppb to ppb or sub-ppb levels. Because of the complexity of the interaction between radiolysis production, removal of radiolysis products by boiling and decomposition, the concentration of oxidizing species (O_2 and H_2O_2) is not uniform around the circuit. The net result is that different components reach the -230 mV (SHE) specification at different hydrogen injection rates. In addition, the hydrogen demand to maintain a given location below -230 mV (SHE) can change considerably between different plants and during the cycle.

During the 1990's a new concept capable of coping with many of the observed problems with HWC was developed. This concept was a technique to transform all system surfaces to respond readily and quickly to HWC and the reducing conditions as a noble metal redox electrode. The increased response to hydrogen only demands a stoichiometric excess of hydrogen, i.e. at least twice a much dissolved hydrogen as oxygen on the molecular basis dissolved in the water, as opposed to the sometimes several hundred times higher hydrogen over-stoichiometry required in ordinary HWC. This new concept, called NMCA, Noble Metal Chemical Addition, involves an addition of noble metal, NM, compounds, i.e. directly to the primary circuit. The NMCA will lead to NM deposits by an electro less method in situ, resulting in the requested behaviour, as shown in **Figure 5.1.6**.

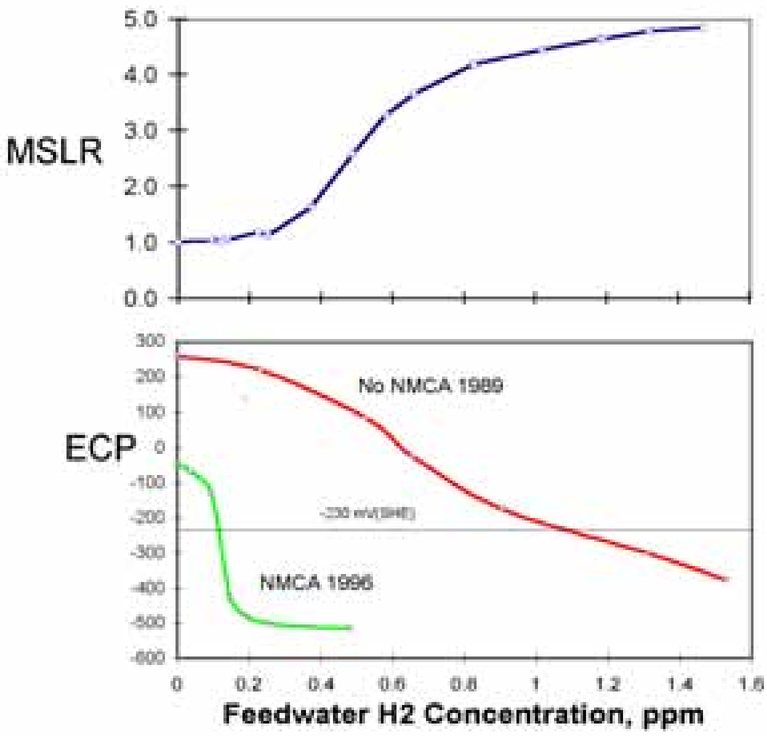


Figure 5.1.6: Schematic behaviour of stainless steel and platinum in BWR water as a function of feedwater hydrogen addition, Hettiarachchi, S, 2000 [5]

The number of plants having applied NMCA or NobleChem™ has grown drastically during the last few years. Recently, 24 U.S. BWRs (70% of the U.S. BWR fleet), 1 European, and 3 Japanese BWRs have applied NMCA. Hence 28 of 88 BWRs worldwide apply NMCA. Fifteen plants have operated for at least two cycles with NMCA, but only one plant has finished more than three cycles, and that is Duane Arnold. A couple of plants have applied NMCA but do not also perform hydrogen addition, for various reasons. If the rate of introduction of NMCA is compared with other important water chemistry adjustments in U.S. BWRs in the last two decades, i.e. HWC and zinc injection, see **Figure 5.1.7**, it is obvious that much less experience was been gained from NMCA compared to the other regimes.

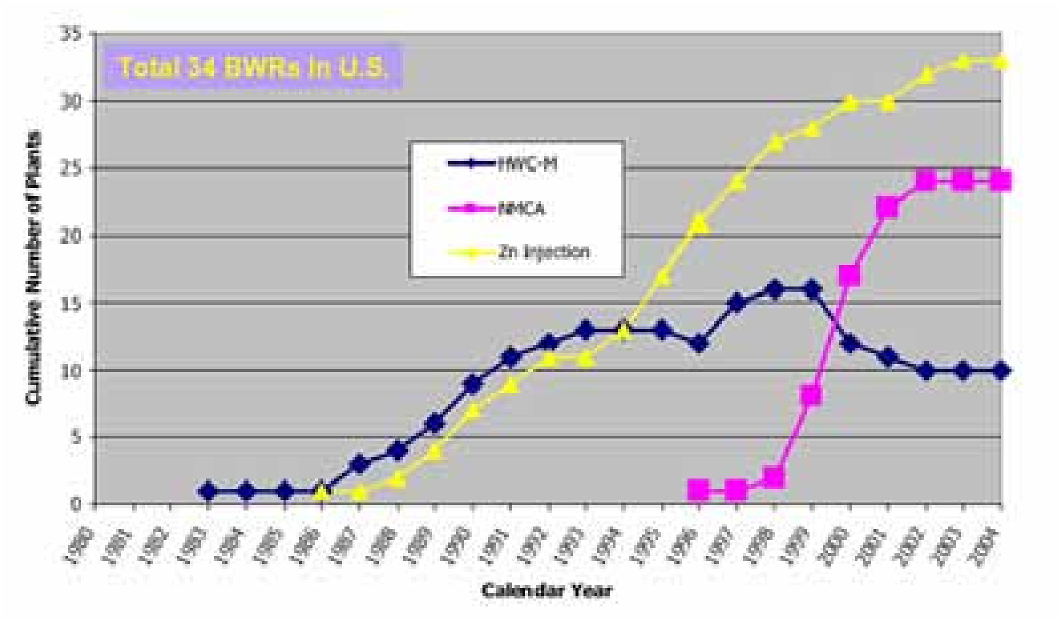


Figure 5.1.7: Water chemistry changes in US BWRs, Jones, 2004

The water chemistry will be rather drastically affected by the NMCA, as has been recorded to at least to some degree in all cases where NMCA has been employed. One effect is an initial increase in the MSLR levels. This increase has been found to be up to a 100% increase in the MSLR even at NWC, and occasionally equally high transients are obtained once the initial increase at NWC has declined and HWC is started. The transient can last from a few weeks to several months as described more in detail in a later section.

Another effect is the increase in conductivity following NMCA. This increase in conductivity can be rather significant. It was e.g. reported in Peach Bottom-3 that the maximum soluble iron level was 44 ppb during the transient. If it is assumed that the iron was in the form of (Fe^{2+} + 2 OH^-), a conductivity of 0.407 would be expected, which, at least in this case, fully explains the conductivity transient. The largest conductivity transient reported was of peaking at 1.23 µS/cm. This corresponds to 140 ppb of fully dissociated $Fe(OH)_2$. If, as seem to be the general case, the increase in conductivity is from soluble iron it will probably not have any effect on cracking, as opposed to conductivity enhancements caused by sulphate or chloride transients. Nevertheless, an elevated conductivity in the range of 0.3 to 1.2 $\mu S\cdot cm^{-1}$ could conceal significant additional conductivity transients caused by the more detrimental ions. It is recommended to operate the plant with both zinc injection and HWC before NMCA in order to minimize the iron restructuring.

Another chemistry effect that seems coupled to the iron restructuring, is the release of tramp uranium from system surfaces depositing on the fuel cladding surfaces. Other chemistry effects appear to be more directly related to the establishment of a reducing environment, as the enhanced iodine volatility and the MSLR increase, described above. There are also a number of other water chemistry effects that are less straightforward to understand. A summary of different water chemistry effects, and their grouping in duration, is presented in **Table 5.1.1**.

Table 5.1.1: Tentative grouping of post-*NMCA* chemistry effect duration.

Duration	Effect
Months	MSL radiation increase
	Short-lived nuclides increase
	Elevated conductivity
	Elevated soluble iron
Years	Elevated Co-60
	Lowered sulphate (and large hide-out return transients)
Long term	Tramp uranium vagabonding
	Elevated offgas
	Lowered RW iodine

A summary of the radiation behaviour of seventeen plants that started hydrogen injection immediately after NMCA is shown in **Figure 5.1.8** where the BRAC standard dose rate locations measured after the first and second cycles are plotted versus the cycle median reactor water Co-60(s)/Zn(s) ratio. Seven of the plants were below the EPRI recommended ratio of 0.74 Bq/ml/ppb for both cycles and their BRAC dose rates remained in the less than 1 mSv/h range. Several of the plants experienced higher dose rates after operating well above the 0.74 Bq/ml/ppb ratio value during the first post NMCA cycle followed by a decreased ratio during the second cycle. These results clearly indicate that those plants that operate their first post NMCA cycle at values below the 0.74 ratio will have low dose rates as long as they stay at or below that value. For those plants that operate their first post NMCA cycle at values above the 0.74 ratio will "lock in" higher dose rates and will probably have to conduct a piping decontamination and reapplication and then operate at or below the 0.74 ratio to reach BRAC levels of 1 mSv/h or lower. Three plants have utilized this strategy of conducting a piping decontamination followed by a low temperature NMCA application.

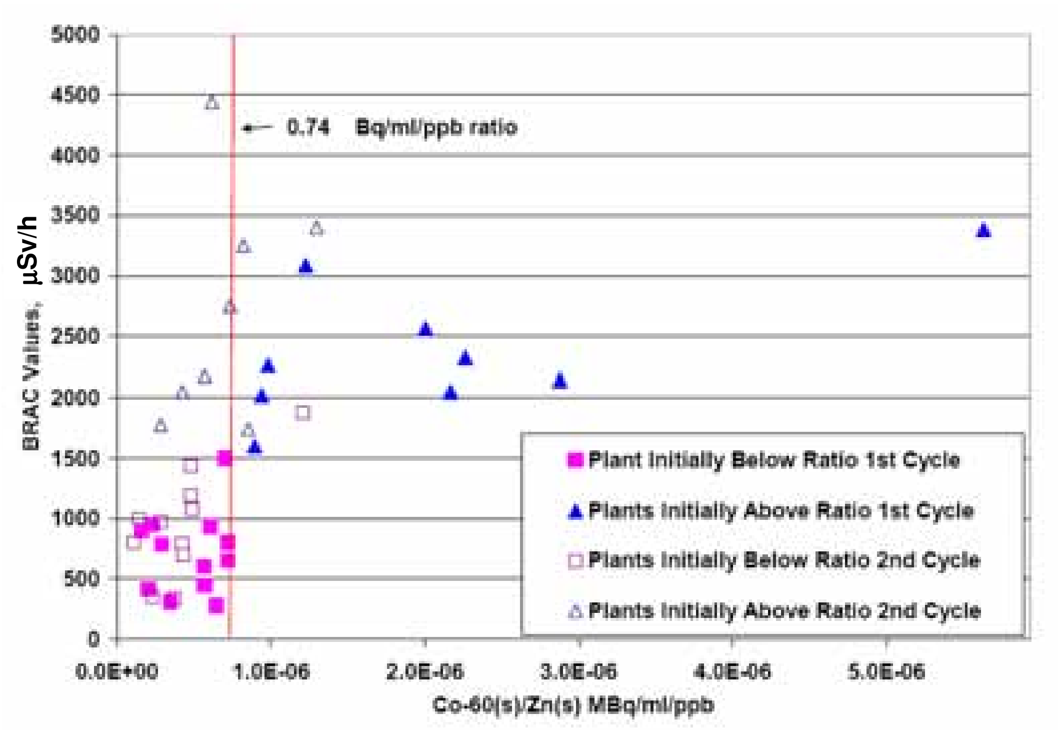


Figure 5.1.8: Post NMCA BRAC dose rate measured at the end of the first and second fuel cycles with highest dose rate plant omitted (Data available as of Jan. 26, 2006 - Post NMCA Zinc Demand) [7]

Analysis of BRAC shutdown dose rates of plants that use moderate HWC chemistry and zinc injection has revealed an apparent grouping based on the recirculation piping electrochemical corrosion potential (**Figure 5.1.9**). All plants shown (with the exception of two), inject zinc according to the recommended reactor water soluble Co-60 to soluble zinc ratio of 0.74 Bq/ml/ppb. The data reveals two clear groupings between the plants. Group 1 plants enjoy average BWR Radiation Assessment and Control (BRAC) dose rates similar to those experienced by 1st post NMCA cycle plants that follow the recommended cobalt-to-zinc ratio, while Group 2 plants have considerably higher dose rates.

Cowan and Hussey, 2006 [7] conclude, that the apparent difference between Group 1 and Group 2 plants is the electrochemical corrosion potential (ECP) in the recirculation piping. Although in both groups the recirculation piping ECP is below -230 mV SHE (Standard Hydrogen Electrode) and has sufficient mitigation protection against intergranular stress corrosion cracking (IGSCC), the recirculation piping from Group 1 plants has an ECP in the range of the theoretical limit from hydrogen injection. Recall that NMCA plants effectively have theoretical ECP at the surface in the presence of small amounts of hydrogen. Therefore, the trends are consistent; if the recirculation piping surfaces maintain the theoretical hydrogen ECP, then the plant should observe similar radiological benefits. This is further supported by the decreased BRAC shutdown dose rates for Grand Gulf. The figure shows the transition from higher dose rates to lower dose rates after the hydrogen injection rate was increased.

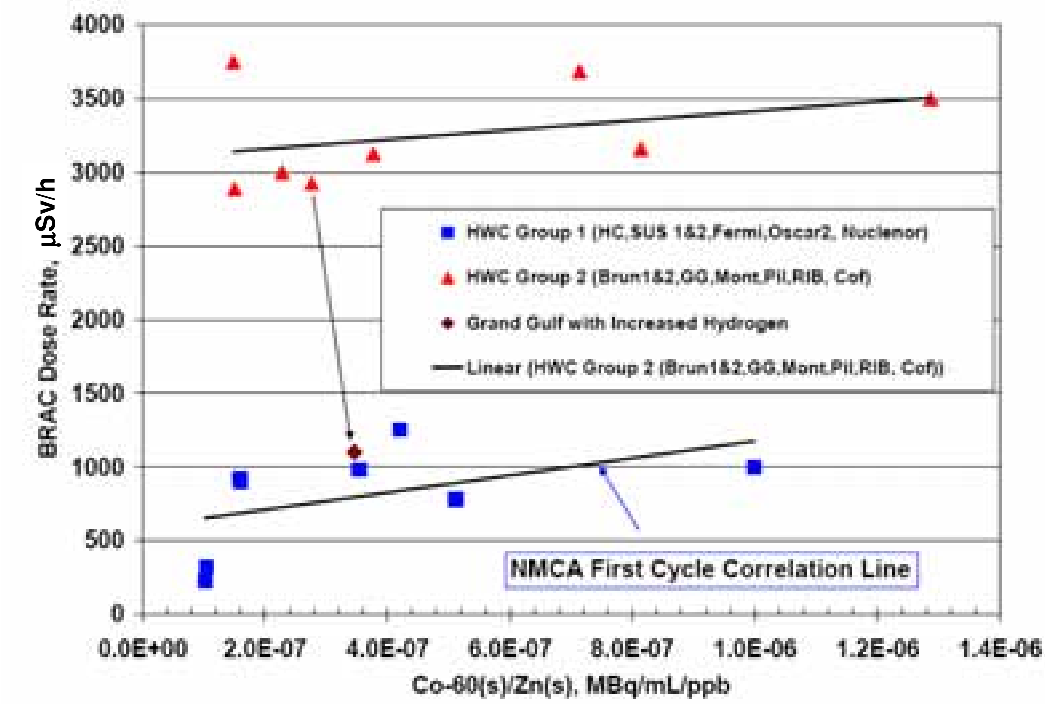


Figure 5.1.9: Grouping of HWC plants that inject zinc based on recirculation piping ECP [7]

Cowan and Hussey, 2006 [7] also discuss the influence of feedwater iron on shutdown dose rates for plants that inject zinc. Iron would appear to have a direct impact on the shutdown dose rates, but is actually secondary to its impact on zinc. Iron inhibits the performance of zinc injection in three ways; (1) it acts as a competing metal for incorporation into the oxide film, (2) it may combine with zinc in the reactor coolant and prevent zinc incorporation into the corrosion film, and (3) fuel performance concerns limit the total metal oxide concentration in the reactor coolant, hence the more iron in the feedwater, the less zinc can be used. Iron reduction strategies were discussed in an earlier chapter. The overall message is that reduction of feedwater iron is a very efficient way to reduce radiation fields in plants on HWC/NMCA together with DZO.

The overall conclusion is that low radiation fields can be maintained in HWC and NMCA plant if the above recommendations are followed. Whilst these recommendations are not directly affected by a power uprate they may be easier to realise if considered within a plant modernisation project.

5.1.2.1.4 Zinc injection

The recommendation to inject zinc in especially plants on HWC or NMCA has previously been addressed. A recent example of a successful application of Zn injection is the Oskarshamn 2 plant. Of special interest is the experience of the combined effect of low feedwater iron, HWC operation and zinc injection in the Oskarshamn 2 plant, reported by Lejon, 2006 [8]. The HWC operation in the Oskarshamn 2 plant resulted in a considerable increase of radiation levels, with an increase of RHR dose rate up to about 7 mSv/h at the 2003 refuelling outage (**Figure 5.1.11**) The increase in the dose rate was in spite of very low feedwater iron and rather low reactor water Co-60 level (**Figure 5.1.11**), and a main reason for the increase was believed to be due to a significant contribution of soluble iron from certain reactor systems reaching very low corrosion potential because of the HWC operation. A decision was made to perform a system decontamination during the 2003 outage, and thereafter start DZO injection during the following cycle. The DZO injection resulted in an increase of reactor water Co-58 and Co-60 (**Figure 5.1.10**), but the RHR dose rate has remained on a low level (**Figure 5.1.11**).

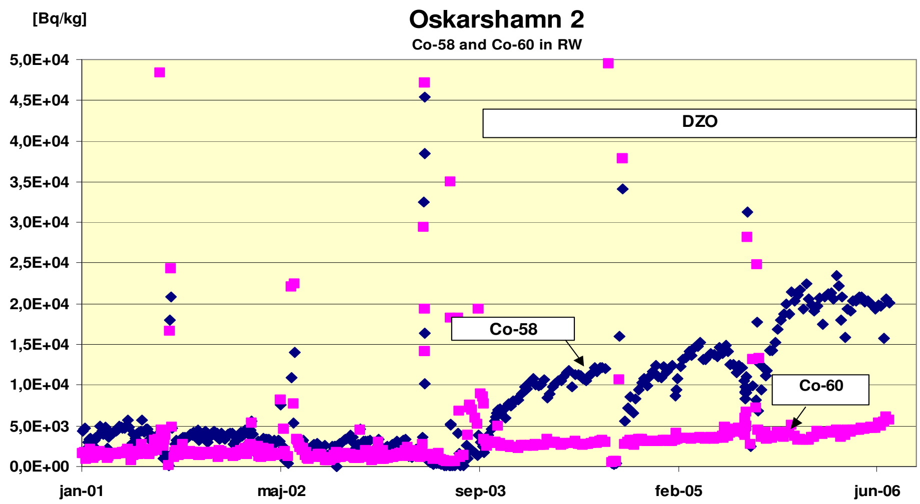


Figure 5.1.10: Oskarshamn 2 – Reactor water Co-58 and Co-60 before and after DZO injection

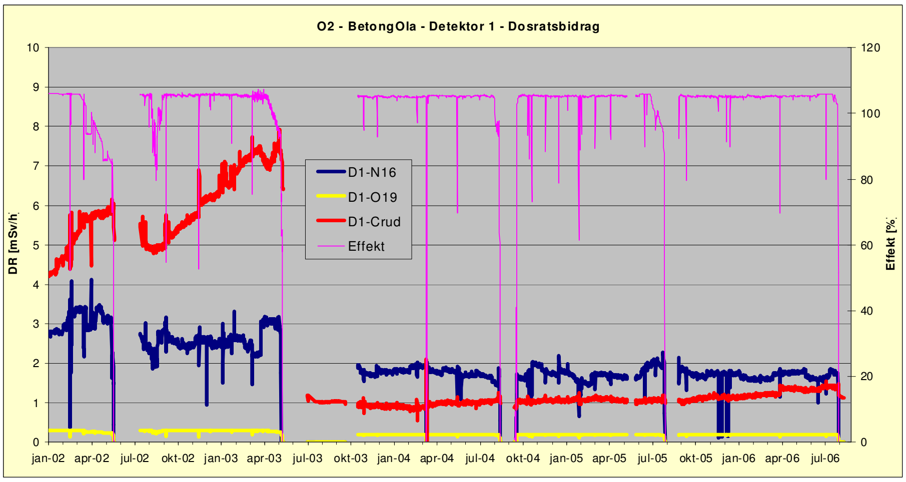


Figure 5.1.11: Oskarshamn 2 – On-line follow-up of contact dose rate on RHR pipe with separation of different radiation sources (DZO injection started Oct. 2003)

The water chemistry conditions in Oskarshamn 2 during DZO injection can be summarized as follows [8]:

- Feedwater Zn: about 0.2 ppb
  Reactor water zinc: 4-5 ppb
  - i.e. about 50% of injected Zn cleaned up in RWCU (2% RWCU capacity)
- Feedwater Fe: typically 0.02 ppb

- HWC operation:
  - PLR piping: about -300 mV (SHE)
  - RHR piping: about -500 mV (SHE)

- Fuel crud composition in fuel crud that has only seen DZO conditions:
  - (Fe/(Ni+Zn)) ~ 2.2, see **Table 5.1.2.**

The outcome in Oskarshamn 2 coincides quite well with the proposed optimized corrosion product balance described previously. The overall corrosion product balance must be addressed in any power uprate project.

Table 5.1.2: Oskarshamn 2: Fuel deposit sampling data one cycle after DZO introduction [8]

		Fuel assembly	
Burn-up	**[MWd/kgU]**	**10.7**	**41.3**
	[EFPH]	**7942**	**37337**
		DZO only	**DZO 1 of 5**
Fe	**[g/m^2]**	**1.0**	**5.9**
Ni	**[g/m^2]**	**0.22**	**0.97**
Zn	**[g/m^2]**	**0.31**	**0.19**
Cr	**[g/m^2]**	**0.14**	**0.14**
Mn	**[g/m^2]**	**0.05**	**0.15**
Co	**[g/m^2]**	**0.005**	**0.013**
Cu	**[g/m^2]**	**<0.01**	**<0.01**
Σ	**[g/m^2]**	**1.7**	**7.3**
[Fe/(Ni+Zn)]		**2.2**	**5.4**

5.1.2.2 Radiation fields during operation

The radiation fields during operation are dominated by the contribution from the short-lived radionuclide N-16 ($T_{½}$ = 7.13 s). Most of the produced N-16 is decaying in the reactor vessel and recirculation lines, this activity is well shielded and does not contribute to the occupational exposure. However, the carry-over of N-16 to the turbine plant is of great importance, and particularly the resulting radiation levels outside the turbine plant due to air-scatter radiation through the roof of the turbine building (s.c. N-16-skyshine). The radiation source term of N-16 activity in the turbine plant is affected by a power uprate in two different ways:

1. The power uprate in itself increases the production of N-16 in the reactor coolant in proportion to the power increase.

2 The increased steam flow rate in association with the power uprate means a reduced transport time from the reactor core to the turbine plant. A typical transport time is about six seconds, and e.g. a 30% power increase with a corresponding increase of steam flow rate means a reduction in transport time of about 1.4 s. The overall effect of such a power increase would be about a 50% increase of the N-16 inventory in the turbine plant.

On the other hand, such an increase of N-16 activity in the turbine plant is much less dramatic than the increase experienced when HWC is introduced, when the N-16 carry-over to the steam is typically increased by a factor of five. Therefore, the biggest concern from the perspective of N-16 skyshine, is for plants combining HWC operation with a significant power uprate. It has also to be noted, that different plants have different well shielded turbine plants. In practice the difference in N16 skyshine levels outside a well shielded, compared to less effectively shielded, plant under the same operational conditions can be almost an order of magnitude. Almost all BWRs were originally designed for NWC operation, i.e. HWC operation is not considered in the shielding design.

Nevertheless, the power uprates studied in both the OL1/2 (**Section 4.1**) and CNC (**Section 4.2**) plants show only marginal impact on the occupational exposures during reactor operation, and these exposures remain ..only a small fraction of the total exposures. The OL1/2 plants are on NWC, while the CNC plant is on HWC operation.

5.1.2.3 Radiation fields during outages

As discussed in the previous sections, the radiation fields during outage conditions are mainly controlled by the overall corrosion product balance. This corrosion product balance is affected by a power uprate, but correctly addressed it can be maintained and even improved after the uprate. This is verified by the experience of the plants in the present study **Section 4.1** and **Section 4.2**.

The normal BWR case is low steam moisture content and low shutdown radiation levels in the turbine plant. However, there is experience of situations with increased steam moisture, resulting in significant radiation levels and occupational exposures in the turbine plant (see e.g. the OL1/2 plants **Section 4.1**). Such situations of increased steam moisture content are in many cases associated with plant power uprates. Critical reactor components determining the steam moisture content are the steam separators above the core plus the steam dryer in the top of the reactor pressure vessel. These components are sensitive to the core and steam flow, and the increase of flow rate in association with power uprates may result in a break-through of moisture. The lesson learnt is that the design of steam separators and dryers must be carefully reviewed when planning an uprate, and that significant redesign of these components may be necessary.

5.1.2.4 Fuel failure issues

During the past three years the number of US BWRs with cladding defects has doubled, so that today about one-third of the US plants are operating with at least one leaking fuel rod. Understanding and reversing this trend is a top industry priority because fuel cladding failures lead to increased O&M costs, as well as to operational restrictions and outage duration increase.

Determining the root causes of the recent failures is complicated by the fact that fuel designs and duties, cladding materials and primary water chemistries have all changed significantly during the past decade. The root cause evaluations performed indicate that a considerable fraction of recent failures is associated with water chemistry and crud deposition, see **Figure 5.1.12**. However, it is necessary to remember that a large fraction of the crud related failures seen in the figure is from one single plant with a large number of failures. Research programs have been initiated to gain better understanding of the mechanisms, and possible identify what water chemistry changes shall be introduced to minimize the formation of deleterious fuel crud. It should be observed that the increase of fuel crud related failures coincides with the extensive introduction of changed chemistry conditions in the form of zinc injection and NMCA, see e.g. Keys, 2004 [9]. The considerable effort in US to reduce feedwater iron is, to a large extent, driven by the aim of reducing crud induced fuel failure problems.

A similar trend of increased fuel crud associated fuel failures are not seen in Scandinavian and German plants, further indicating that the main root cause of the US experience is due to the significant change of water chemistry conditions introduced during last years.

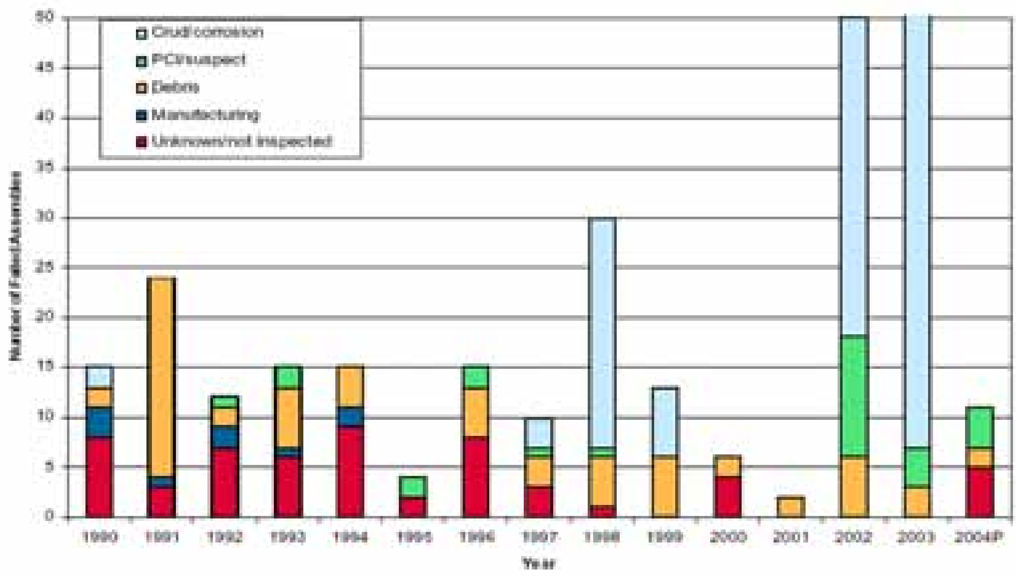


Figure 5.1.12: BWR fuel failure root cause trend in US BWRs, Turnage, 2004 [1]

Fuel failures, including dissolution of fuel material, result in increased release of fuel crud particles to the coolant, which in turn results in the formation of radiation sources (hot spots) in the plant. The responsible mechanism is knock-out reactions due to fissions in the formed tramp uranium. A significant tramp uranium contamination results in long-term effects, which take about 10 years to return to the background level. Plant experience has shown that a tramp uranium contamination of the order 50 – 100 g may increase the annual exposure with typically 20%. There are also several other negative impacts of fuel dissolution resulting in tramp uranium on the core:

– High background activity levels in reactor water and off-gases making it more difficult to detect additional fuel failures.
– Increase of release to the environment.
– Increased waste production, especially of actinides.
– Increased radiation levels around certain turbine components due to accumulation of noble gas daughters (e.g. Ba-140 and La-140).

The way to avoid this problem is to introduce a proper fuel failure management at the plant. Methods should be introduced to efficiently monitor situation with fuel dissolution. The earlier method using measurements of reactor water Np-239 has been shown not to work in HWC and NMCA conditions. Therefore, many plants have introduced a method where Sr-92 reactor water activity is used as a measure of the amount of tramp uranium. This method is efficient both in oxidizing and reducing conditions. The employed method must be combined with a determined level of maximum amount of accepted tramp uranium. Most Scandinavian BWRs have introduced such a level, typically in the order of 10 g total of tramp uranium. In situations where there is a risk that this level will be exceeded the reactor would be shut-down, the leaking fuel identified and removed.

Both the OL1/2 (**Section 4.1**) and CNC (**Section 4.2**) plants have recently experienced failures of the foreign debris type. Several plants take actions to reduce the risk of fuel failures of the debris type by introducing fuel with debris filters and by installing cyclotron filtration in the main feedwater lines. The crud induced failures recently seen in some US plants are the results of bad feedwater chemistry (high levels of Fe, Zn and Cu), and actions are underway to improve the situation, see e.g. **Figure 5.1.3**.

5.1.2.5 BWR conclusions

- BWR plants worldwide continue to show low annual occupational exposure, typically 1 – 2 manSv per plant and year. The lowest exposures are experienced in BWR designs with internal recirculation pumps. No significant difference between plants being power uprated or non uprated is seen.
- A key factor in achieving low radiation levels seems to be to maintain a well balanced corrosion product inflow to the primary circuit, forming low amounts of fuel crud of the spinel type. Efforts have to be spent to improve condensate cleanup, especially in connection to power uprates, in order to achieve that. Zn injection seems to be an effective method to reduce radiation fields in plants on HWC or NMCA.
- NMCA plants that have already maintained the recommended reactor water zinc level during the first NMCA cycle show low piping dose rates still after two cycles, around 1 mSv/h. For those plants that operate their first post NMCA cycle at lower Zn levels will “lock in” higher dose rates and will probably have to conduct a piping decontamination and NMCA reapplication to reach dose rates of 1 mSv/h or lower.
- HWC plants that are operating with the piping ECP levels close to the theoretical for Pt (<-450 mV (SHE) show lower radiation fields than plants operating closer to the -230 mV (SHE) limit. The “Low-ECP” HWC plants show radiation fields at about 1 mSv/h, i.e. similar to the NMCA plants.
- European BWRs show similar radiation levels to US BWRs in spite of some differences in reactor design and water chemistry conditions. One Scandinavian BWR demonstrates that low radiation fields can be achieved with the combination HWC, Zn injection and ultra-low feedwater Fe (<0.1 ppb).
- In many cases increased steam moisture content is associated with plant power uprates. Critical reactor components determining the steam moisture content are the steam separators above the core plus the steam dryer in the top of the reactor pressure vessel. The lesson learnt is that the design of steam separators and dryer must be carefully reviewed when planning an uprate, and that significant redesign of these components may be necessary.

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5.1.3 PWR uprates

5.1.3.1 Water chemistry issues

The primary side reactor water chemistry in a PWR has a large impact on the activity buildup, and hence on the radiation fields and occupational exposures. The chemistry is to a large extent determined by the power regulation with boron, and a power uprate will certainly affect this boron control. The influence of a power uprate on the water chemistry conditions, and the resulting impact on radiation levels, are discussed in this section.

5.1.3.1.1 Boron

The reactor power, and the corresponding variation of boron in the reactor water during the period 1995 – 2005 in the Swedish PWR Ringhals 3 (R3), is shown in **Figure 5.1.13**. A boron concentration of at least 2000 ppm is maintained during outage conditions to assure under-criticality. The boron content is lower at power operation, but in the beginning of a fuel cycle (BOC) considerable boron content is needed to compensate for the over-reactivity of the newly loaded core. The boron content at BOC varies somewhat due to how long the fuel cycle and how long the coast-down operation are planned for. The boron cycle is almost linearly decreased during the cycle down to 3-4 ppm, when coast-down operation is started.

A typical recent fuel cycle in R3 is shown in **Figure 5.1.14**.

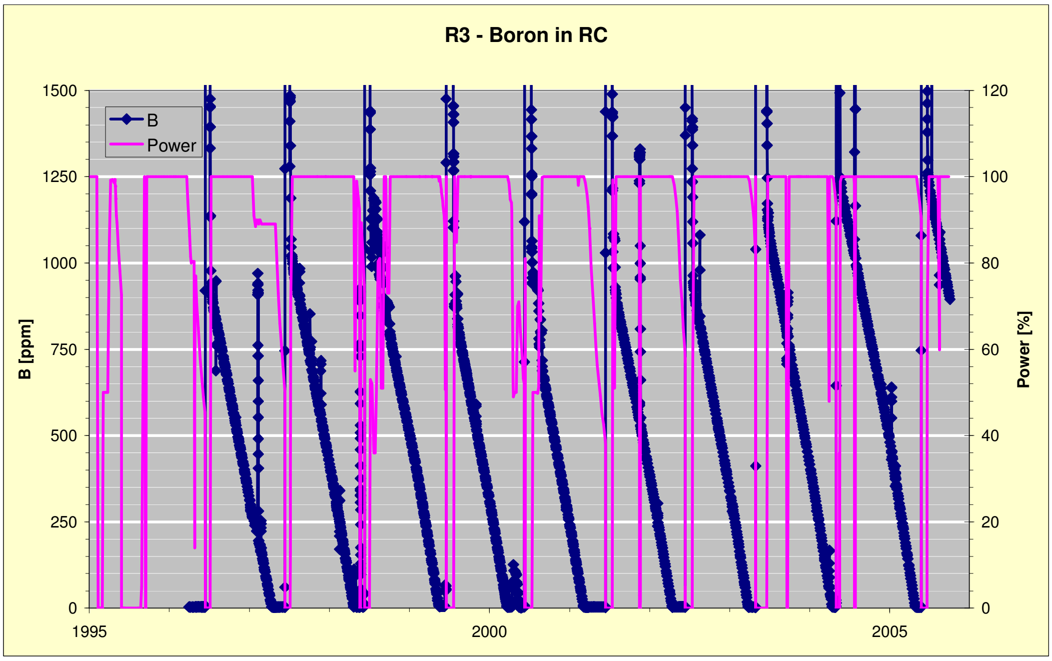

Figure 5.1.13: R3 – Boron reactor water concentration and reactor power 1995 - 2005

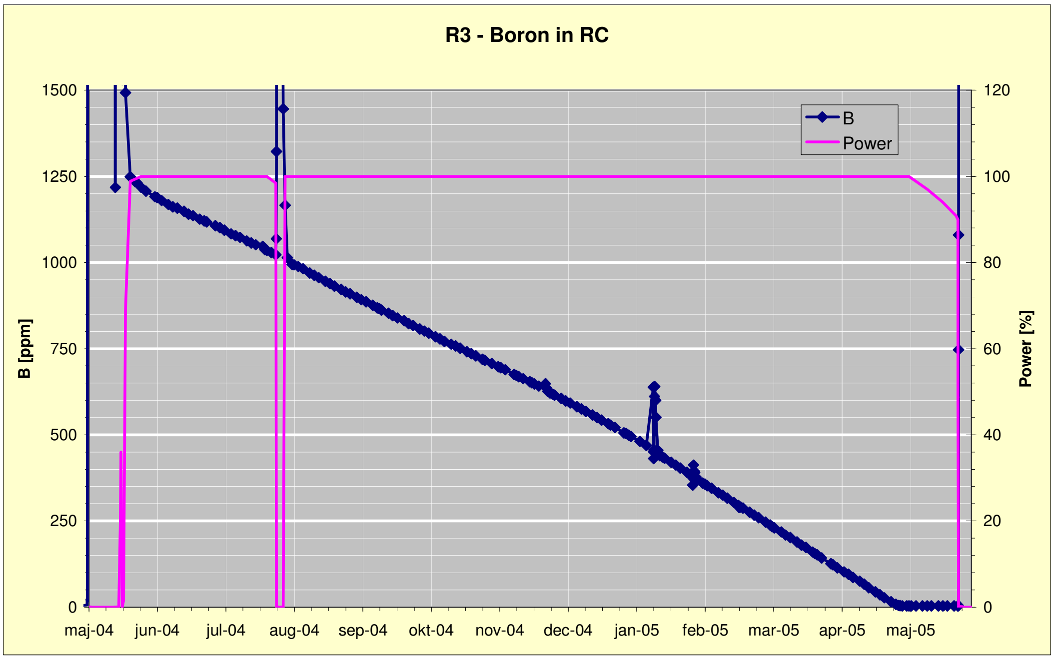

Figure 5.1.14: R3 – Boron reactor water concentration and reactor power during the fuel cycle 2004-05

An increase of power level means that the BOC boron content must be increased. This is exemplified in **Figure 5.1.15**, where an increase from 2873 to 3160 MWth is assumed in the R3 plant. The BOC reactor water boron is increased from 1250 to 1460 ppm, and the average boron conten from 634 to 741 ppm. Note that longer fuel cycles mean a further increase of average boron content, as illustrated by data from a PWR operating with 18 months cycles (**Figure 5.1.16**).

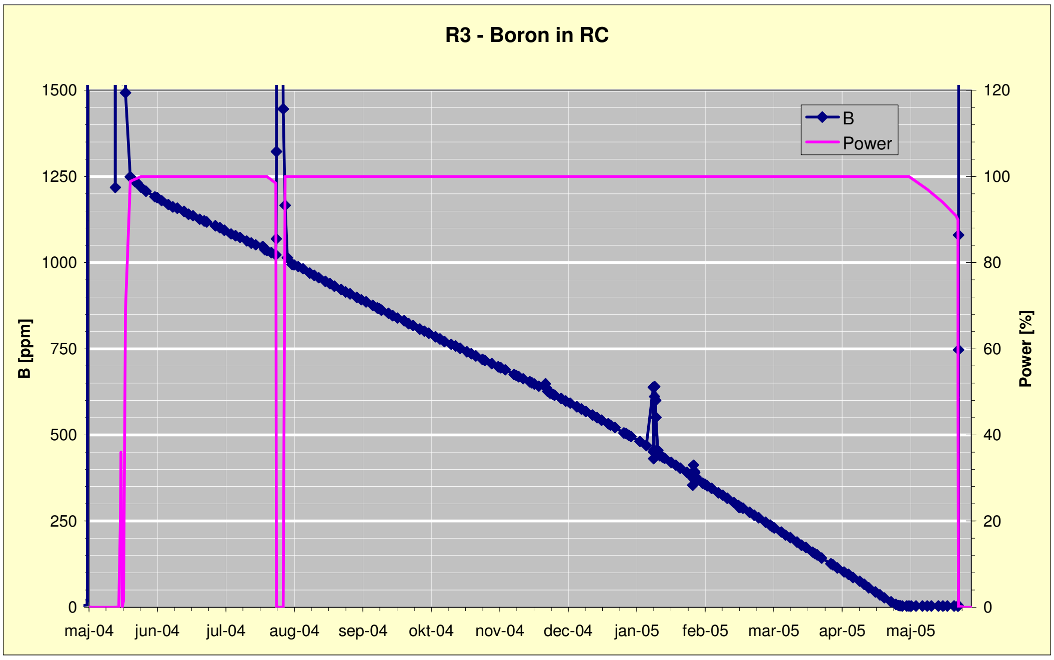

Figure 5.1.15: R3 – Typical variation of reactor water boron at 2873 to 3160 MW_{th}

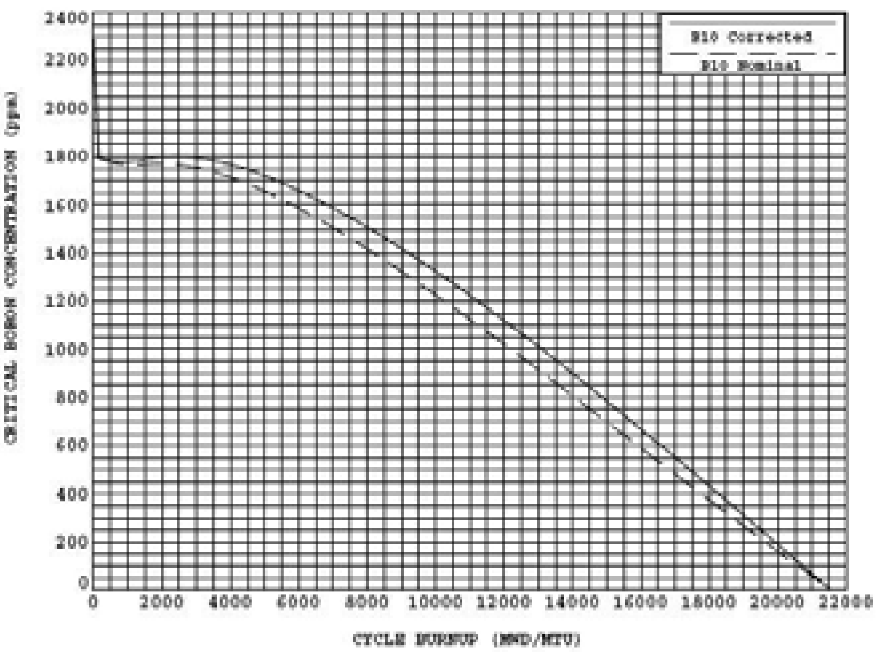

Figure 5.1.16: Variation of reactor water boron in a plant operating with 18 months cycles

The main impact of the boron content is on pH control and on the risk of being affected by Axial Offset Anomaly (AOA). Both these issues are discussed further in later sections.

The content of boron also affects the production of tritium (H-3, $T_{½}$ = 12.33 y) in the coolant. Whilst the production occurs through several reactions it is dominated by the following:

- B-10 (n,2α) H-3

The combination of increased neutron flux and increased average boron content in the reactor water means that the production of H-3 will increase alsont four time with the reactor power increase. The increased H-3 production will only marginally affect the radiological in-plant conditions, but has a significant impact on the activity release from the plant.

The corresponding quadratic increase with reactor power will be experienced for the radionuclide carbon-11 (C-11, $T_{½}$ = 20.385 m):

- B-11 (n,p) C-11

C-11 has a rather small effect on radiation levels during operation and activity release from the plant.

So called EBA (Enriched Boron Acid), where the fraction of the neutron absorbing nuclide B-10 has been increased, is used in some plants. EBA with typically 30% B-10 compared to 19.7% in natural boron is used, which means that the reactor water boron content can be reduced with about 30%. The use of EBA has advantages in pH control and to avoid AOA, see below. Furthermore, the production of C-11 is reduced. However, the production of H-3 in the coolant is not affected due to a maintained content of B-10.

5.1.3.1.2. pH control

Adding boron acid to the reactor coolant means acid conditions, which is counteracted by adding 7-litiumhydroxide ($^{7}LiOH$) to the coolant. The selection of ^{7}Li as counter-ion to the hydroxide is made from materials and reactor physics considerations[8]. Early operation in US PWRs was made with constant concentration of ^{7}Li in the coolant, which resulted in a gradual increase of alkaline conditions during the fuel cycle. This operation resulted in high radiation fields around steam generators (SGs) and primary piping, the main reason associated with corrosion and solubility of corrosion products. The corrosion rate of SG material was high at the BOC conditions with less alkaline conditions, and the corrosion products were predominantly deposited on the hot fuel surfaces. The more alkaline conditions at the end of cycle (EOC) conditions resulted in a dissolution and redistribution of the neutron activated fuel crud to the colder SG tubes and primary piping.

The solution to that problem was the introduction of so called "coordinated" chemistry, where the content of ^{7}Li in the reactor water is varied in order to maintain a constant high temperature pH (pH_{300}) during the cycle. However, the question was what pH_{300} to select? Initially the solubility of magnetite was selected as a base for the decision, resulting in a selection of pH_{300} = 6.9. Later it was realized that the PWR chemistry was more affected by the solubility of nickel oxides, especially in the case of SG tubes of Inconel 600

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[8] ^{7}Li is also produced in neutron absorptions in ^{10}B. ^{7}Li has a low neutron cross section as compared to ^{6}Li, which means a low production of H-3. Added ^{7}Li is specified with a low level of ^{6}Li (<0.04%).

or 690. Therefore, an effort has been made to introduce operation at increased pH_{300}, up to 7.4 during most of the cycle (”Elevated” pH chemistry), where the nickel oxides seem to have minimum solubility. The operation experience with such elevated pH chemistry has generally been favourable, with reduced radiation levels during outages around SGs and primary piping.

The relation between content of boron and Li for some high temperature pH is shown in **Figure 5.1.17**. The figure shows that a considerable Li concentration is needed to maintain a pH_{300}=7.4 at BOC. A sensitive question in the PWR society is the maximum level of Li in the coolant that is acceptable for the fuel cladding. A maximum level of 2.2 ppm has been adopted for a long time [11]. This limit, however, means that the high temperature pH is low during a large part of the beginning of the cycle, and the optimum pH_{300}=7.4 is only obtained at the end of the cycle. The fuel vendors have through materials development allowed higher maximum Li concentrations, and the limit used during several years at the Ringhals PWRs is 3.5 ppm. Even at Li_{max}=3.5 ppm a large fraction of the fuel cycle is not operated at a pH_{300}<7.4, especially in the case of uprated power with increased boron concentration. Presently projects are underway to increase the maximum Li content in US and European BWRs, and the Ringhals PWRs have recently achieved approval from the fuel vendor to increase the maximum Li limit to 6 ppm. It is used in R2 and R3 PWRs to be able to start at BOC at a pH_{300}=7.2, and that Li content is maintained up to pH_{300}=7.4, thereafter pH_{300}=7.4 is maintained by adjusting the Li content.

The conclusion is, that with an allowed increased maximum Li level, the pH control may be maintained or even improved after a power uprate. The threat against such an operation strategy is the risk that future fuel experience would reveal that the maximum Li content should be decreased, however such a development is not likely and not supported by recent fuel inspections. One possible solution to handling such a situation is to introduce EBA with reduced boron content and thereby lead to a reduced need of Li in the coolant. Another possibility is to introduce Zn injection in the PWR primary circuit. Zn injection in PWRs has been introduced in some US and German PWRs with favourable impact on the radiation buildup.

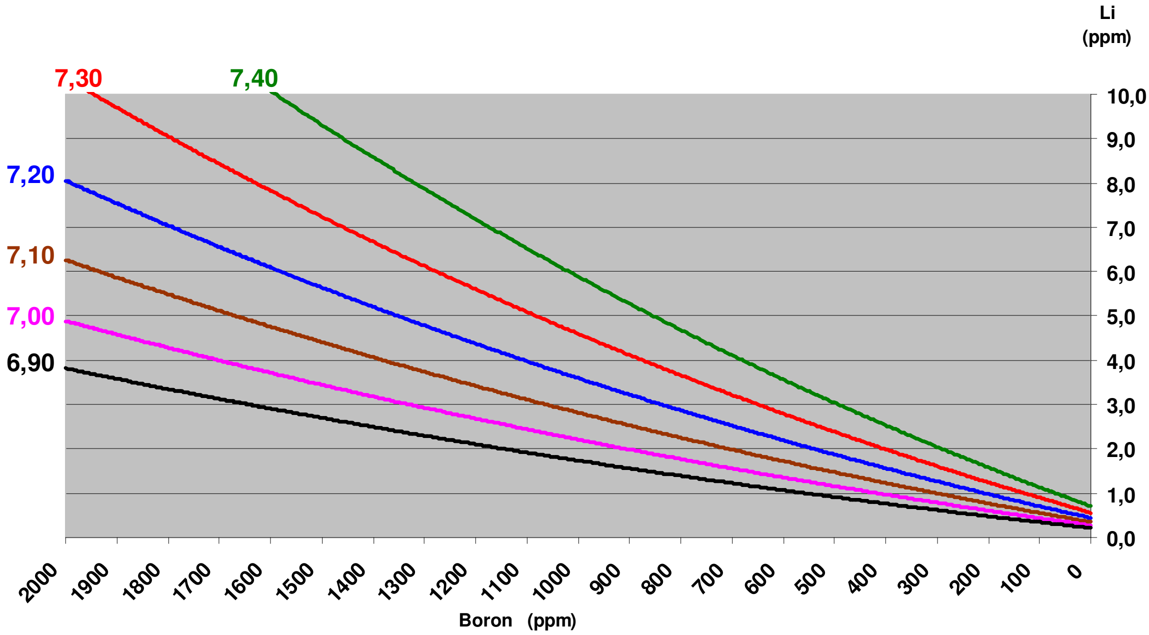


Figure 5.1.17: Relation between boron and Li for pH_{300} = 6.9, 7.0, 7.1, 7.2, 7.3 and 7.4 [10]

5.1.3.1.3 Corrosion products

The corrosion products in PWR primary circuit is to a high degree determined by the selection of material in the SG tubes, in combination with the pH control applied. The selection of Inconel 600 or 690 as tube material will mean the chemistry is very much determined by nickel and its oxides. The corrosion product chemistry will mainly be determined by these three factors working together:

1. The materials selection of the SG tubes. The present widespread use of Inconel 690 shows excellent corrosion performance, especially on the secondary side. German PWRs instead use Inconel 800, which seems to have some advantages on the primary side due to its higher content of Fe compared to the Inconel 690 (and Inconel 600).

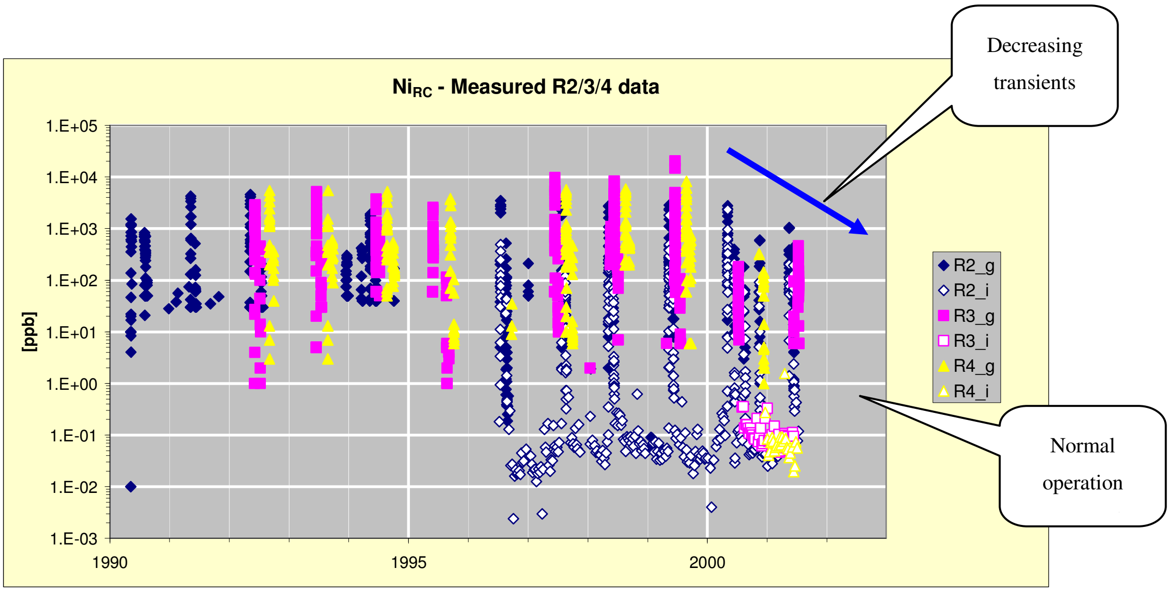
2. The pH control adopted. A pH_{300}=7.4 seems to be close to optimal in the case of Inconel 690 and 600 due to low solubility for nickel oxides.

3. The amount of dissolved hydrogen (DH). A typical value of about 35 ml (STP)/kg used in most PWRs seems to be quite enough to ensure reducing conditions in the whole primary circuit. However, the optimum DH concentration is subject to discussions, and both higher and lower values than present are proposed. The present guidelines may be changed in the future.

Measured reactor water Ni concentrations during normal operation and shutdown transients in the Ringhals PWRs are shown in **Figure 5.1.18**. The shutdown transients have been followed during several years, while the normal operation concentrations have only been followed during recent years through the introduction of integrated sampling techniques. A couple of interesting observations can be made:

- The normal operation levels are low, about 0.1 ppb. Some US PWRs that have experienced fuel AOA problems have experienced considerable higher levels, up to 10 ppb [13].

- The shutdown transients show a considerable reduction in the R2 and R3 plants after the increase of high temperature pH to 7.4 that was introduced around 2000. This is further shown in **Figure 5.1.19**, where the accumulated annual amount of Ni in transients in the R3 plant is shown. The increased pH introduced after the 1999 outage has resulted in a dramatic decrease in Ni transient, and a similar development is seen in the R2 plant with the same pH control, but not in the R4 plant that has maintained the old pH control with a somewhat lower pH.



Figuer 5.1.18: R2/3/4 – Measured reactor water Ni during normal operation and shutdown transients (g – grab samples, i – integrated sampling)

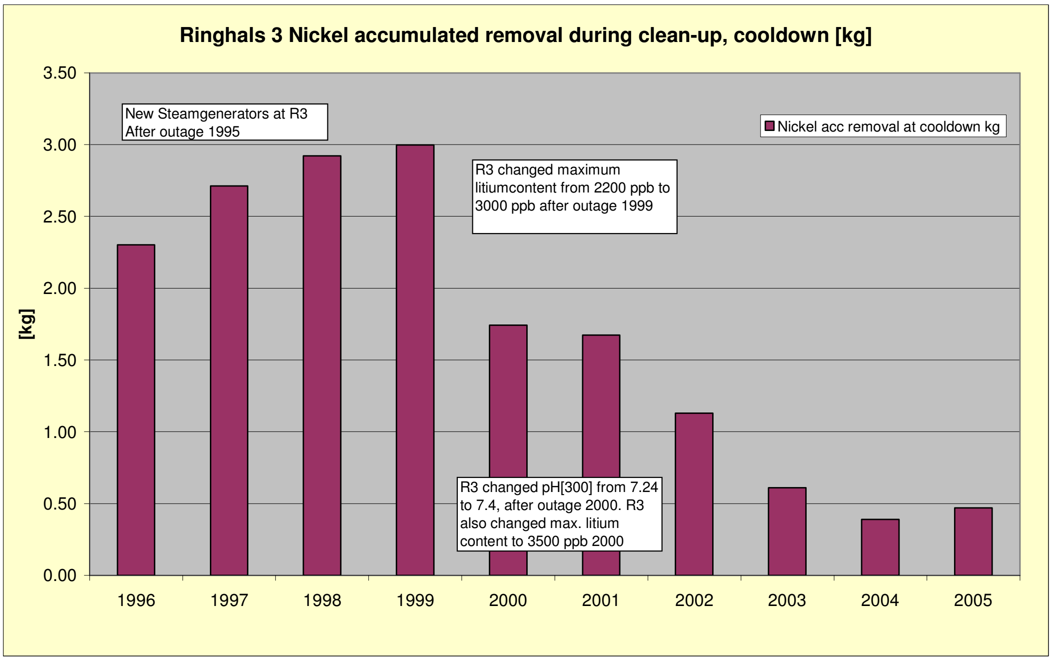


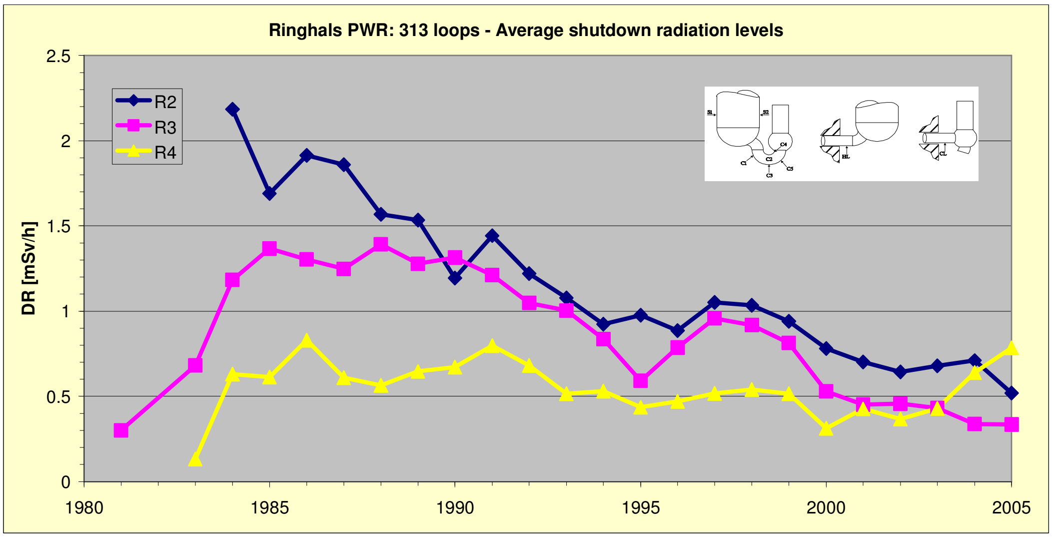
Figure 5.1.19: R3 – Evaluation of release of Ni during shutdown transients 1996 – 2005 [12]

5.1.3.2 Radiation fields during operation

The radiation levels during operation are very much determined by prompt radiation from the core fission process, and the production of short-lived radionuclides such as N-16. The increase is approximately proportional to power increase. Experience from operating plants shows a rather small contribution to occupational exposure from these sources.

5.1.3.3 Radiation fields during outages

Radiation levels during outage conditions in PWRs are dominated by the nuclides Co-60 and Co-58. The radiation levels in plants that have adopted improved water chemistry control during recent years, e.g. R2 and R3, normally show reducing radiation levels, see **Figure 5.1.20**. On the other hand, plants that have maintained previous water chemistry with somewhat lower pH, e.g. the R4 plant, show a less favourable development.



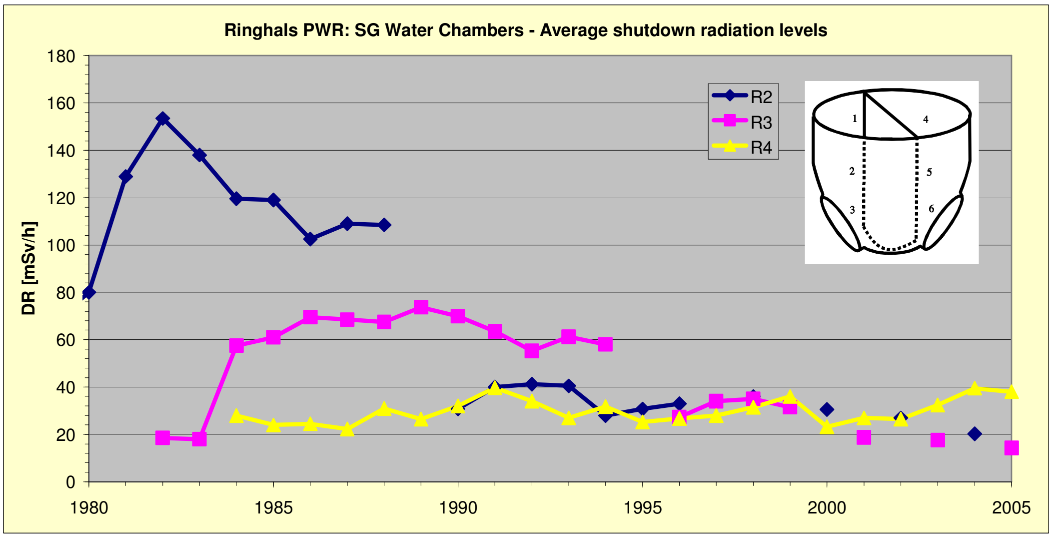


Figure 5.1.20: Ringhals PWR – Dose rate trends on primary piping and in SG water chambers

5.1.3.4 Fuel failure issues

5.1.3.4.1 Axial Anomaly Offset (AOA)

Axial Offset (AO) is defined as the relative power difference between the upper and lower part of the reactor core:

Eq. 4
$$AO = \frac{P_t - P_b}{P_t + P_b} \cdot 100$$

where:
P_t – Power in the upper part of the core
P_b – Power in the lower part of the core

The AO is calculated and measured during each cycle, and must be kept within specified limits. If there is a difference between measured and calculated AO of at least 3%, an Axial Offset Anomaly (AOA) exists. An example of AOA in a Westinghouse PWR is shown in **Figure 5.1.21**. The beginning of the cycle shows a reasonable agreement between measured and calculated AO <3%. In the middle of the cycle the measured AO is low compared to the calculated value, changing to a high measured AO compared to calculation at the end of the cycle.

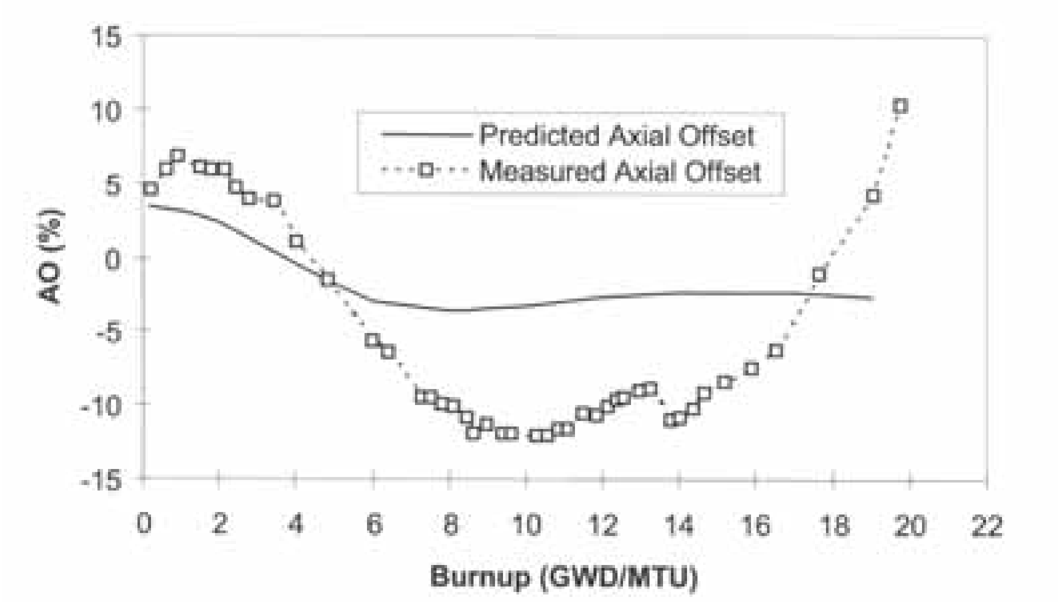


Figure 5.1.21: Example of AOA in a Westinghouse PWR [11]

All details for the AOA phenomenon is not well know, but the mechanism can qualitatively be explained in the following way (see also **Figure 5.1.22**):

– Ni-rich fuel crud is accumulating in the beginning of the cycle in high-power fuel bundles. This is promoted by the coexistence of several factors:
  - High contents of reactor water Ni, e.g. due to initial corrosion of a new SG (with Inconel 690 or 600, AOA is never observed in German BWRs with Inconel 800 SG tubes) .
  - Less good pH control due to high initial boron and too low Li. This problem is especially pronounced in plants operating on long cycles, 18 or 24 months.
  - High-power fuel assemblies with considerable sub-cooled boiling in the top. The degree of boiling in the top of the core is normally substantially increased in a reactor which has undergone power uprate.

– Boron is enriched in the formed fuel crud layer through boiling. The enrichment of boron results in a not-predicted local power decrease.

– Dissolution of the fuel crud occurs at the end of the cycle, e.g. at power reductions. The dissolution of the fuel crud results in local increase of the power level due to lower burnup than predicted due to the earlier boron accumulation.

– A large fraction of the accumulated fuel crud will be dissolved during the shutdown transient. The dissolution of the fuel crud contributes to the activity buildup and the shutdown radiation fields.

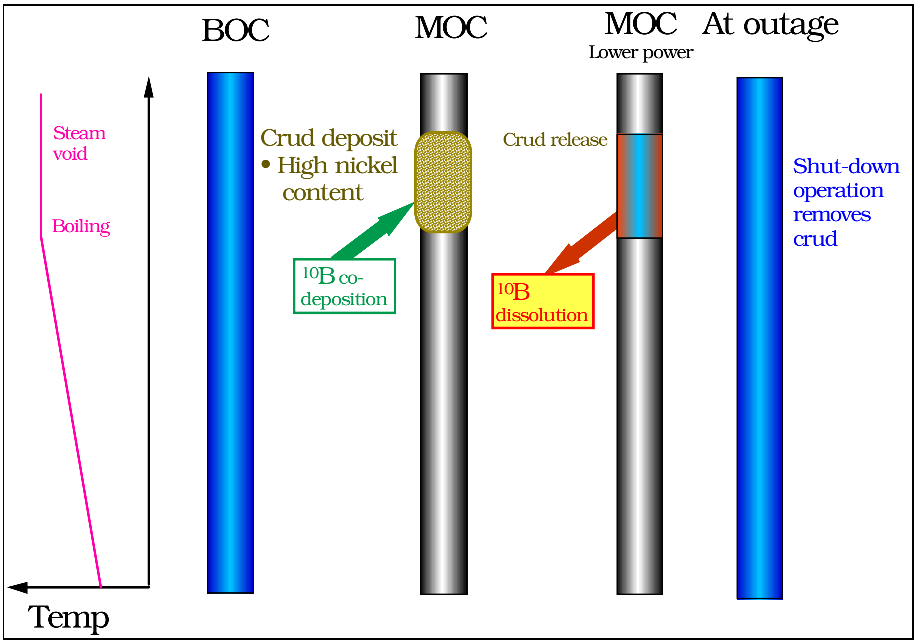


Figure 5.1.22: Proposed mechanism for AOA

Significant problems with AOA have so far, only been experienced in US PWRs (long cycles, maximum Li 2.2 ppm, Inconel 600 (690?), and high-duty cores). It is very obvious, that power uprates significantly increase the risk of AOA to be developed, something that has to be addressed in an uprate project. Twelve months cycles and improved pH control with pH_{300}=7.4 are factors that reduce the risk. It may be a good decision to avoid a power uprate immediately after a SG replacement when the release of Ni is expected to be high due to the initial corrosion.

5.1.3.4.2 Other factors

A number of different types of fuel failures in PWRs are discussed in [14]. Some examples:

– Wear between spacers and cladding. One special case is so called “baffle jetting” due to leakage through the baffle plates. These problems are avoided through improved design.

– So called “debris fretting”. This failure mechanism is avoided by a high cleanliness during outage, and by use of fuel with debris filters.

– Crud-related fuel failures have occurred in some plants. Is avoided by a good pH control.

– Bending of guide tubes. This problem seems to be solved by improved fuel design and improved material selection.

- Pellet cladding interaction (PCI). More frequent in BWRs than in PWRs. Are normally avoided in PWRs by restrictions on the rate of power changes.
- Pellet cladding mechanical interaction (PCMI). This failure mechanism is coupled to the burnup of the fuel, and not primarily to the power level.

The performed review indicates, that the different failures have a very little connection to the power level, and other factors, e.g. design and fuel material, are of larger importance.

5.1.3.5 PWR conclusions

- A power uprate in a PWR means increased content of boron in the coolant, especially in the beginning of cycle. One consequence is a quadratic increase of H-3 production. This effect is further enhanced if the plant is operated with long fuel cycles (18 or 24 months).
- Operation with increased Li, with a high temperature pH around 7.4, has shown to have a favourable impact on radiation levels. Earlier concerns of high-Li operation with respect to fuel material have been reduced due to improved fuel materials. Improved pH control in most PWRs is the main reason for the present rather low radiation fields and exposures in PWRs. Power uprates with increased boron in the reactor water call for further increased reactor water Li in order to maintain the favourable conditions. One alternative, however costly, is to introduce EBA enriched in B-10 in order to reduce the boron content in the coolant.
- An additional way to achieve exposure reduction is to apply Zinc injection with depleted zinc (DZO) also in PWRs. The effect of zinc on radiation fields seems to be especially large in connection to SG replacements.
- One great concern for PWRs is the risk of having fuel problems due to so called Axial Offset Anomaly (AOA). The development of AOA is promoted by the combination of (i) high levels of Ni, e.g. after a SG replacement, (ii) poor pH control with too low Li levels in BOC, and (iii) high power fuel assemblies with a substantial sub-cooled boiling rate. The combination of power uprate and a SG replacement means an increased risk, and it would be wise to avoid a power uprate the first cycle after a SG replacement. Factors that reduce the risk is maintaining a good pH control, and operating at shorter (twelve) months cycles.
- Other fuel failure mechanisms in PWRs seem to be mostly associated with fuel design and materials factors, and only marginally with the power level.

5.2 Organisational factors

5.2.1 Reducing the exposures

Every NPP has stringent administrative processes and rules for performing the work that has radiological impact on plant staff. Typically, these processes and rules apply to all NPP staff. However, the rules are not always strictly followed, which might, in particular, be a problem with external (e.g. non-permanent) staff and contractors. Because of that, the "administrative barriers" are not always effective as they are meant to be.

Moreover, the economic pressures resulted in refuelling and maintenance operations being streamlined and concentrated as much as possible, which may sometimes have a negative impact on the occupational exposure.

The occupational exposures are best managed through appropriate job preparation, planning and implementation, and post-performance review. This is to ensure that exposures are kept "as low as reasonably achievable" (ALARA). The use of operating experience and exchange of best practices is an important prerequisite for applying the principle of optimisation to occupational radiation protection.

NPPs worldwide have made significant progress over the past 10 years in reducing the radiation dose to workers. The most important organizational elements in decreasing exposures are:

5.2.1.1 Leadership and Supervisory Monitoring

– Recognition of the importance of radiation safety to all nuclear power plant workers is an important issue for the nuclear industry and has to be communicated clearly. Leaders need to emphasize the importance of the individual having a high awarness of their personal responsibility in regard to radiation protection. This should include the need to pay specific attention to complying with rules and following administrative procedures and processes.

– The radiation protection management team has to be fully integrated into:

  - Work planning activities
  - Outage planning and scheduling activities
  - Plant modification reviews
  - Plant strategic decision-making processes.

– For those jobs where large doses could be received in a short period of time, direct involvement by both radiation protection staff and work supervisors is required in:

  - ALARA reviews
  - Planning

- Preparation
- Performance of the job

Prejob briefings, review of prior operating experience, and adequate radiation exposure reduction reviews are important measures required by ALARA planning. Radiation protection staff and work supervisors should ensure that necessary surveys and other radiation protection measures required by ALARA planning are implemented at the work site.

5.2.1.2 High quality procedures

The activities with the potential for high radiation exposure have to be controlled by written procedures and/or specific radiation work permits.

- Procedures or radiation work permits should contain specific radiation protection instructions to prevent unplanned exposures.
- Procedures or radiation work permits should contain specific radiation protection instructions such as permissible cumulative doses allowed for a task and, the requirments relating to radiation monitoring (continuous or at what frequency, physically or through remote monitoring methods).
- Procedures or radiation work permits should contain specific instructions concerning the frequency and type of radiation surveys and at what dose rates or accumulated dose to an individual worker the area has to be evacuated.
- Electronic dosimetry, with alarms for dose rate and cumulative dose appropriately set for the task, should be required by procedures or radiation work permits and when specified should be used.by personnel
- Entry into very high radiation areas should have the written approval from the radiation protection manager.

5.2.1.3 Training

Important elements in reducing occupational exposures:

- Training of each ALARA activity
- Retraining of plant staff
- Training of contract employees in radiological protection expectations and practices at NPP.

The training and retraining programs should stress the potential for abnormally high or rapidly changing radiological conditions, including the actions required when these conditions occur. Training should also emphasize the importance of a high level of awareness and sense of individual responsibility with regard to personnel radiation protection.

5.2.1.4 Operation Experience

It was found that identifying and trending minor radiological protection violations and problems that occur in an NPP helps to identify weaknesses, so that the defences and barriers can be strengthened before leading to a more significant events.

5.2.1.5 Assessment of Radiation Conditions

- Periodic reviews and assessment of radiation protection deficiencies, including those involving human performance errors assures that radiological controls are adequately addressed. These reviews should ensure that work control systems, including radiation work permits, procedures, and work orders, adequately address radiological controls, including the potential for changing radiological conditions.
- Additionally, periodic self-assessments of radiation protection department performance should be performed to identify and correct radiation protection program and process weaknesses.
- Review of plant areas should be performed to ensure that all areas with existing or potential high radiation exposure rates are identified and properly posted and controlled.

One important prerequisite for applying the principle of optimisation to occupational radiation protection is the appropriate and timely exchange of data, techniques and experience on doses and dose reduction methods.

5.2.2 Implementation of power uprate

Many of the undesirable outcomes can be avoided if the uprate project organisation is well resourced, staffed with personnel with extensive plant experience and focused on identifying and resolving potential operational and other impacts. The elements, which point out good organizational aspects, are:

5.2.2.1 Power Uprate Project Team Organisation

- The uprate project team includes a full-time project manager, and the project manager and team members have no additional duties.
- Involvement of the operations, radiation safety and training departments early in the project.
- Sufficient lead-time is assured to allow plant personnel adequate data analysis, identification of procedure changes, and reviews during all stages of the project.
- Sufficient time is assured for training plant personnel and for incorporating changes in the control room simulator.

5.2.2.2 Feasibility Study Phase

- Feasibility studies considered existing equipment problems or limitations. Feasibility studies that used original design parameters often underestimated the scope of work required.

- Some project schedules can be challenged when detailed analyses identified major additional equipment modifications.
- Following the feasibility study contributes to more effective reviews of power uprate related plant changes, modifications, training, and procedure revisions

5.2.2.3 Detailed Analysis/Design Phase

- Plant engineering, radiation safety experts and operations personnel perform detailed reviews of power uprate project analyses. Interdisciplinary occupation is very important
- Ensuring that present plant conditions and equipment performance problems are addressed during system and components design reviews.
- Insight into the critical topical areas - fuel
- Perform Loading Pattern Risk Assessment to address possibility of:
  - Crud Induced Power Shift
  - Steady state Hot-Leg Streaming
  - Axial Xenon Stability
  - Fuel Performance – Fuel Rod Design Criteria
  - Axial Offset Anomalies

5.2.2.4 Implementation Phase

- For extended power uprates it is recommended to use a two-stage implementation strategy to implement the changes; for example, new fuel design and power uprate (good solution to avoid Axial Offset Anomalies after SG Replacement and power increase)
- In the power ascension tests is included an approach in which power is increased in increments, with hold points at predetermined intervals. These hold points can last several days to gain experience or resolve technical issues at the new power level
- During the testing, actual plant parameters are compared to expected values
- A contingency plan is available to identify actions needed to address unexpected plant situations.

5.2.2.5 Ongoing Post-Power Uprate Operation

- Staff are aware that operating margins are reduced and various plant systems and components are changed
- Contingency plans are available in providing guidance to the plant staff when expected or unexpected conditions are encountered (vibration, flow accelerated corrosion, fuel condition)

- Following power uprates, some plants operate with equipment that was previously used as spares. Removal of these components from service for maintenance following power up-rates will require a power reduction from rated power. This has to be correctly addressed because it has prompted changes to on-line maintenance strategies.
- Power uprate projects sometimes require operating strategy changes. The traditional operating philosophy of operating at 100 percent reactor thermal power may not be applicable following many power uprates because a plant system or component may be the power limiting factor.
- Ongoing equipment problems may occur due to flow-accelerated corrosion and vibration. Because of greater feed-water and steam flows associated with power uprates, there is an increased potential for flow-accelerated corrosion that could lead to failure. Additionally, increased vibration of components in systems experiencing increased flow rates has caused fatigue-induced failures. These conditions may not be readily identified during the analysis phase of the uprate project. Thus, the scope and frequency of flow-accelerated corrosion and vibration monitoring programmes may need to be reconsidered following power uprates.

Economic evaluations for adding electrical generation under present industry conditions are likely to show justification for the continued power uprating of existing nuclear facilities.

Industry experience has shown that power uprates can be implemented safely and successfully. However, industry experience illustrates the need for additional focus on the importance of using a thorough, deliberate approach when planning and executing power uprates to avoid undesirable consequences.

5.2.3 Lesson learned

Leadership, composition and organisation of the power uprate especially large demanding tasks are critical for successful implementation of power uprate and received doses. Lessons learned comprise different aspects:

- Human factor and work planning

  Work in difficult conditions, like SG replacement, should be carefully planned and supported with sufficient number of workers with the necessary skills. Otherwise, underestimation could lead to: unnecessary injuries of the overloaded workers, job extension with influence on the outage critical path and poor quality of the work performed.

- Scheduling, documentation

  Final documentation necessary for all modifications which are going to be performed have to be presented at the site early enough, otherwise it could result in insufficient time for the procurement of necessary parts and components and prolong the time of outage.

- Project

  Each large demanding task has to start sufficiently early in order to prevent the acceptance of inappropriate requirements due to time pressure. This can lead to acceptation of some deviations, which are normally not been accepted. More emphasis should be put on early planning and involvement of interdisciplinary teams.

- Subcontractors

  Most of the large activities require the involvement of subcontractors. They have to be included in the project early enough for quality work preparation. Non-quality performed work invariably led to additional work hours, which can result in increased occupational exposures for both plant and outside personnel.

  It has to be noted that it is not recommended to give responsibility for important tasks to outside personnel unless there is strong supervision by plant personnel. Outside personnel are not necessarily as well-trained or acquainted with plant design details as plant personnel.

6. Conclusions

This report presents the result of the three tasks of the Inquiry into the radiological consequences of power uprates at light-water reactors worldwide. The review has resulted in the following conclusions:

Compilation of power uprates

Worldwide collection of information on the uprates for PWR and BWR reactors that were implemented or are planned to be implement are summarised in the database. Through a process of data collection and its review the following initial conclusions were obtained:

- Throughout the world, occupational doses at NPPs have steadily decreased over the past decade, mainly through better application of ALARA principles, better use of shielding material, but also increased attention to occupational dose issues.
- Occupational exposure in the BWR plants are typically about 50% higher than in PWR plants, due to differences in the design
- No direct relationship between the uprates and the occupational doses could be established. The occupational doses on some plants seem to be higher after the uprate, while on others seem to be lower.
- There is no obvious correlation of the power uprate and fuel failures. However, performance of fuel for PWRs and BWRs went in opposing directions, improving for PWRs and deteriorating for BWRs.
- Through the data collection process events were identified that have occurred as a result of inadequate design or implementation of uprates. These events involved equipment issues, unanticipated responses to conditions, or challenges for operating staff , for example:
  - Loose parts as a result of a flow-induced, high-cycle fatigue failure on a steam dryer cover plate (BWR plants)
  - Operational transients and equipment damage due to lack of training of plant staff on changes to PCS operating characteristics
  - Unanticipated challenges and degraded performance from reductions in margins
  - Operation beyond licensed power levels for extended periods due to errors in thermal power calculations following uprates
- None of the above events had direct consequences on doses to the personnel or releases. However, some of them might have had an indirect influence on occupational exposure or releases (replace or repair of damage equipment).

Analysis of selected BWR plants:

Olkiluoto 1 and 2

The plants have been uprated twice since commissioning. The thermal power of each reactor was increased from 2000 MW to 2160 MW in 1984 and to 2500 MW in 1998.

The 1998 uprate was part of an extensive modernization program implemented in 1994–2006.

- Good planning of modernization program has reasonable impact on outage lengths (maximum annual outage length 22 days compared to typically 7-14 days).
- Investment in the cleanup capacity (still maintained after the power uprate) results in favourable water chemistry conditions that can be maintained, or even improved, after the power uprate.
- Reduction or replacement of materials (Stellite) results in Co source reduction.
- Exposures during operation are maintained on a constant and rather low level after the uprate. One important factor is that the plant is maintained on chemistry without hydrogen injection.
- Radiation levels during outage on reactor systems are maintained on a rather low and constant level after the power uprate.
- The installation of new steam separators can increase the radiation levels around main steam lines and other turbine components due to a considerable increase in steam moisture content.. This problem can be overcome with a recent design and installation of new steam dryers in the reactor pressure vessel to reduce steam moisture.
- The exposure per outage day is maintained on a fairly constant level even though the considerable man-hour efforts during some outages for the power uprate and plant modernization program have resulted in increase of occupational exposures.
- The average annual exposures in the Olkiluoto plants is kept on a rather low level compared to international BWR data in spite of the large efforts for power uprate and plant modernization.

Cofrentes

The present power level corresponds to 111.8% of the initial thermal power level, which means on average 5.19 MWt per fuel assembly. The main power increase was introduced in 2002, when the power level was increased from 104.2% to 110%.

- Extension of the fuel cycle often goes in parallel with the power uprates. Due to the margin in the core fuel assembly, design is changed from the original 8x8 array to 9x9 and finally to 10x10 necessary for the more demanding recent operation conditions.
- Modifications, which mainly affect the turbine plant, result in low exposure due to low contamination level of the turbine plant.
- Increase in the reactor pressure and temperature due to uprate only moderately affects the steam velocity in the reactor and the main steam lines.

- The reactor water chemistry is significantly influenced by the design and materials selection of the turbine plant. Low-alloy steel pipes, instead of carbon steel, considerably reduce contribution to the feedwater iron.
- Radiation fields are well controlled by the introduction of zinc injection
- Long term introduction of hydrogen water chemistry (HWC) results in reduction of corrosion materials.
- The annual exposures during operation, affected by the production and distribution of the short-lived nuclide N-16, are maintained on a rather stable level and do not seem to be significantly affected by the change to HWC or the power uprates. Radiation levels during operation indicate an average effect due to the power uprates in the order +15% - +30%. This is probably due to the introduction of HWC with increased carry-over of N-16. Overall it can be concluded that the N-16 radiation source term is not the dominant contributor to the occupational exposure during operation.
- The recirculation loops and the RWCU piping significantly influence radiation levels during outage conditions are of importance to occupational exposures
- An increase of the radiation fields around recirculation loops is experienced due to the specific water chemistry situation (gradually decreasing reactor water copper and HWC operation resulting in restructuring of the oxide layers inside recirculation loops). Several measures are introduced to mitigate the increase, and the recirculation loop radiation fields seem at present to be low and well controlled.
- The annual occupational exposures at CNC display a slightly increasing trend during the last 10 years. This trend is explained by the combined effect of increased radiation fields and the considerable modifications and maintenance that have taken place during recent outages. A future decreasing trend is expected due to the above-mentioned improved control of radiation fields around the recirculation loops.
- The CNC power uprates have had a negligible impact on occupational exposures, or at least are shadowed by more important factors such as water chemistry.
- *Analysis of selected PWR plants:*

Asco

The plant has been uprated twice since the commissioning. The thermal power of the reactor was increased from 2696 MW to 2900 MW in 2000 and to 2951 MW in 2003. The SG Replacement was performed in 1995 but all safety analyses necessary for power uprate were performed in 2000. The present study has focused on the first uprate, resulting in a thermal power level of 8% compared to the initial power level. The latter uprate was an uprate of 1.5 %, achieved by using more precise techniques for measuring feedwater flow.

- SG Replacement significantly affected outage length and doses. The activity required 95 outage days. Typical outage lengths are between 30 and 40 days.
- Standard PWR water chemistry is maintained, no zinc is injected.
- The exposures during operation are maintained on a constant and rather low level after the uprate.

- Dose rates, during outage, on reactor systems are maintained on a rather low and constant level after SG Replacement.
- The exposures per outage day have decreased in the last decade even though the considerable manhours incurred during some of the plant modernisation outages resulted in increased occupational exposure.
- The average annual exposures in the Asco 1 have been kept on a level comparable to international values for PWR plants.

Tihange

The plant has been uprated twice since the commissioning. The thermal power of the reactor was increased from 2785 MW to 2905 MW in 1995 and to 3064 MW in 2001. The present study focuses on the Tihange 2 second uprate, resulting in a thermal power level of 110% compared to the initial power level.

- SG replacements have affected outage length but have not significantly affected doses due to good project organization. Three SG were replaced in 63 days. Typical outage lengths are between 30 and 40 days.
- Standard PWR water chemistry is maintained, no zinc is injected.
- The exposures during operation are maintained on a constant and rather low level after the uprate.
- Radiation levels during outage on reactor systems are maintained on a rather low and constant level after SG Replacement.
- The exposures per outage day have decreased in the last decade even though the considerable manhours incurred during some of the plant modernisation outages resulted in increased occupational exposure.
- The average annual exposures in the Tihange 2 have been kept on a level comparable to international values for PWR plants.

Reconstruction experience

Technical factors controlling radiation fields BWR

- BWR plants worldwide continue to show low annual occupational exposure, typically 1 – 2 manSv per plant and year. The lowest exposures experienced are in BWR designs with internal recirculation pumps. No significant difference is seen between uprated and non-uprated plants.
- A key factor in achieving low radiation levels seems to be maintaining a well-balanced corrosion product inflow to the primary circuit. Efforts have to be spent to improve condensate cleanup, especially in connection to power uprates. Zn injection seems to be an effective method to reduce radiation fields in plants on HWC or NMCA.
- European BWRs show similar radiation levels as US BWRs in spite of some differences in reactor design and water chemistry conditions. One Scandinavian BWR dem-

onstrates that low radiation fields can be achieved with the combination HWC, Zn injection and ultra-low feedwater Fe (<0.1 ppb).

- In many cases increased steam moisture content is associated with plant power uprates. Critical reactor components determining the steam moisture content are the steam separators above the core plus the steam dryer in the top of the reactor pressure vessel. The lesson learnt is that the design of steam separators and dryer must be carefully reviewed when planning an uprate, and that significant redesign of these components may be necessary.
- Several plants take actions to reduce the risk of fuel failures of the debris type by introducing fuel with debris filters and by installing cyclotron filtration in the main feedwater lines. The crud induced failures recently seen in some US plants are the result of bad feedwater chemistry (high levels of Fe, Zn and Cu), and actions are underway to improve the situation.

Technical factors controlling radiation fields PWR

- A power uprate in a PWR means increased content of boron in the coolant, especially at the beginning of the cycle. One consequence is a quadratic increase of H-3 production. This effect is further enhanced if the plant is operated with long fuel cycles (18 or 24 months).
- Operation with increased Li, a high temperature pH of around 7.4, has shown to have a favourable impact on radiation levels. Earlier concerns of high-Li operation with respect to fuel material have been reduced due to improved fuel materials. Improved pH control in most PWRs is the main reason for present rather low radiation fields and exposures in PWRs. Power uprates with increased boron in the reactor water necessitate further increased reactor water Li in order to maintain the favourable conditions. One alternative, however costly, is to introduce EBA enriched in B-10 in order to reduce the boron content in the coolant.
- An additional way to achieve exposure reduction is to apply Zinc injection with depleted zinc (DZO) in PWRs. The effect of zinc on radiation fields seems to be especially large in connection to SG replacements.
- One great concern for PWRs is the risk of having fuel problems due to so-called Axial Offset Anomaly (AOA). The development of AOA is promoted by the combination of (i) high levels of Ni, e.g. after a SG replacement, (ii) poor pH control with too low Li levels in BOC, and (iii) high power fuel assemblies with a substantial sub-cooled boiling rate. The combination power uprate and a SG replacement means an increased risk, and it would be sensible to avoid a power uprate for the first cycle after a SG replacement. Factors that reduce the risk is maintaining a good pH control, and operating in shorter (twelve months) cycles.
- Fuel failure mechanisms in PWRs mostly seem to be associated with fuel design and materials factors, and only marginally with the power level.

Organizational factors on uprates

- Leadership, composition and organisation of the power uprate, including radiological activities, especially for large demanding tasks are critical for the successful implementation of power uprate and received doses.
- One important prerequisite for applying the principle of optimisation to occupational radiation protection is the appropriate and timely exchange of data, techniques and experience on doses and dose reduction methods.
- It is not recommendable to give the responsibility for important activities to contractors without strong supervision by plant personnel. Outside personnel are not always well-trained and acquainted with plant design details as the plant personnel.
- NPPs worldwide have made significant progress over the past 10 years in reducing the radiation dose to workers. The most important organizational elements in decreasing exposures are:
    - Strong leadership and supervisory monitoring by plant staff
    - Recognition of radiation safety as an important responsibility in NPP industry with its fully integration into the whole project process
    - Strict implementation of ALARA principles
    - High quality procedures, which have to control all activities which have the potential for high radiation exposure
    - Strict control of entrance into very high radiation areas
    - Training and retraining of plant staff and contractors in ALARA activity and radiation protection
    - Operation Experience
    - Assessment of Radiation Conditions

- During and after uprates many of the undesirable outcomes can be avoided due to quality and professional power uprate project organisation, which is of great importance:
    - Project manager and team members should work full time
    - Involvement of the operations, radiation safety and training departments early in the project
    - Detailed review process and sufficient time for adequate safety/data analysis, identification of procedure changes, and reviews during all stages of the project, which guarantee that all problems and critical topical area (e.g. fuel) are properly addressed.
    - Sufficient time for training personnel and for incorporating changes in the control room simulator
    - Consideration of existing equipment problems or limitations.
    - For extended power uprates it is recommended to use a two-stage implementation strategy; 1) new fuel design and 2) power uprate

- For the Post-Power Uprate Operation phase it should be ensured that the staff is aware that operating margins are reduced and various plant systems and components are changed
- Contingency plans should be available, both for the test phase and Post-Power Uprate Operation, providing guidance to the plant staff when expected or unexpected conditions are encountered (vibration, flow accelerated corrosion, fuel condition, etc)

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Appendix 1: Data base with NPP uprates

Country	1. Station name	2. Key data		3. Plant type		4. Uprate data				
Country		2.1 Vendor NSSS	2.2 . Comm. Operation	3.1 Reactor type	3.2. Initial power	4.1 Uprate power	4.2 Year implemented	4.3 Uprate type & power increase	4.4 Technical solution	4.5 Equipment
					MWt	MWt		(E, S, MU)* %		
Belgium	Doel 2	WEC	12.01.75	PWR	1192	1311	2004	E 10%	Revision of safety nalyses	SGR
	Doel 3	FRAM	10.11.82	PWR	2785	3064	1993	E 10%	Revision of safety nalyses	SGR
	Tihange 1	ACLF	09.01.75	PWR	2665	2875	1995	E 8%	Revision of safety nalyses	SGR
	Tihange 2/1995	ACLF	06.01.83	PWR	2785	2905	1995	S 4,3%	Use of improved core layout	Use of MOX fuel
	Tihange 2/2001	ACLF	06.01.83	PWR	2785	3064**	2001	E 10%	Revision of safety nalyses	SGR
Finland	Loviisa 1	AEE	05.09.77	PWR	1375	1500	1998	E 9%	Complete revision of all safety analyses	Numerous "tunings" of existing equipment; largest changes in the high-pressure turbine
	Loviisa 2	ASEA	01.05.81	PWR	1375	1500	1998	E 9%	Complete revision of all safety analyses	Numerous "tunings" of existing equipment; largest changes in the high-pressure turbine
	Olkiluoto 1/1984	ASEA	10.01.79	BWR	2000	2160	1984	E 10,8%	Revision of safety analyses & substantial	Large modifications in reactor recirculation pumps, turbine,balance of plant equipment (pumps, steam turbine, balance of plant equipment (pumps, steam dryer) Modifications in I&C
	Olkiluoto 1/1998	ASEA	10.01.79	BWR	2000	2500**	1998**	E 25%**		
	Olkiluoto 2/1984	ASEA	07.01.82	BWR	2000	2160	1984	E 10,8%	Revision of safety analyses & substantial	Large modifications in reactor recirculation pumps turbine, balance of plant equipment (pumps, steam dryer)modifications in I&C
	Olkiluoto 2/1998	ASEA	07.01.82	BWR	2000	2500**	1998**	E 25%**		
Germany	Brokdorf	KWU	01.11.86	PWR	3765	3900	?	S 3,9%		
	Emsland	KWU	06.20.88	PWR	3765	3850	1990	S 2,3%	Increase of average coolant temparature	
		KWU	06.20.88	PWR	3765	3900	?	S 3,9%		
	Grafenrheinfeld	KWU	06.17.82	PWR	3765	3950	?	S 4,9%		
	Grohnde	KWU	02.01.85	PWR	3765	3850	1990	S 2,3%	Increase of average coolant temparature	
		KWU	02.01.85	PWR	3765	3900	?	S 4,5%		
	Gundremmingen B	KWU	07.01.84	BWR	3840	4100	?	S 6,8%		
	Gundremmingen C	KWU	01.01.85	BWR	3840	4100	?	S 6,8%		
	Isar-1	KWU	03.21.79	BWR	2575	2755	?	S 7,0%		
	Isar-2/1991	KWU	04.09.88	PWR	3765	3850	1991	S 2,3%	Increase of average coolant temparature	
	Isar-2/1998	KWU	04.09.88	PWR	3765	3950**	1998**	S 4.9%**		
	Neckar-2/1991	KWU	04.15.89	PWR	3765	3850	1991	S 2,3%		
	Neckar-2/2005	KWU	04.15.89	PWR		3965**	2005**	S 5.3 %**		
	Philippsburg-2.1991	KWU	12.17.84	PWR	3765	3803	1991	MU 1%		
	Philippsburg-2/1992	KWU	12.17.84	PWR	3765	3850	1992	S 2.3%		
	Philippsburg-2/2000	KWU	12.17.84	PWR	3765	3950***	2000***	S 4.9%***	Increase of thermal reactor power	
	Unterwester	KWU	10.01.79		3733	3900	2000	S 4.5%	Increase of thermal reactor power	
Hungary	PAKS1	AEE	11.03.83	PWR	1375	1500	2006	E 9,1%		
	PAKS2	AEE	09.21.84	PWR	1375	1500	2006	E 9,1%		
	PAKS3	AEE	11.03.86	PWR	1375	1500	2007	E 9,1%		
	PAKS4	AEE	11.19.87	PWR	1375	1500	2006	E 9,1%		
Japan	Higasidory		05.03.05	BWR	?	?	?			
	Shhika	HIT	07.30.93	BWR	1593	?	?			
Korea	Kori 3	WEC	10.01.86	PWR	2775	2914	2006	S 5%		
	Kori-4	WEC	04.29.86	PWR	2775	2914	2006	S 5%		
	YGN 1	WEC	08.25.86	PWR	2775	2914	2006	S 5%		
	YGN 2	WEC	06.10.87	PWR	2775	2914	2006	S 5%		

Country	1. Station name	2. Key data		3. Plant type		4. Uprate data					
Country		2.1 Vendor NSSS	2.2 . Comm. Operation	3.1 Reactor type	3.2. Initial power	4.1 Uprate power	4.2 Year implemented	4.3 Uprate type & power increase		4.4 Technical solution	4.5 Equipment
					MWt	MWt		(E, S, MU)*	%		
Mexco	LV 1	GE	07.29.90	BWR	1931	2027	1999	S	5%		
	LV 2	GE	04.10.95	BWR	1931	2027	1999	S	5%		
Spain	Almaraz-1	WEC	09.01.83	PWR	2696	2739	2003	MU	1.6%	Improvement of the feed water flow measurement	Turbine and SG change
	Almaraz-2	WEC	07.01.84	PWR	2696	2739	2003	MU	1.6%	Improvement of the feed water flow measurement	Turbine and SG change
	Asco-1/2000	WEC	12.01.84	PWR	2696	2900	2000	E	8,0%	Safety and system Analysis Turbine modification for increased flow	High pressure turbine change Low pressure turbine change Upgrades on main electric equipment
	Asco-1/2003	WEC	12.01.84	PWR	2696	2951**	2003	MU	9.5 %**	Improvement of the feed water flow measurement	
	Asco-2/1999	WEC	06.01.85	PWR	2696	2900	1999	E	8,0%	Safety and system Analysis Turbine modification for increased flow	Turbine change Upgrades on main electric equipment
	Asco-2/2004	WEC	06.01.85	PWR	2696	2952	2004	MU	9.5 %**	Improvement of the feed water flow measurement	
	Cofrentes	GE	03.11.85	BWR	2894	2952	1988	S	2%	Optimised core layout	
	Cofrentes	GE	03.11.85	BWR	2894	3015	1998	S	4,2%	Optimised core layout	
	Cofrentes	GE	03.11.85	BWR	2894	3184	2002	E	10%	Fuel loading, turbine optimisation	
	Cofrentes	GE	03.11.85	BWR	2894	3237	2003	E	12%		Upgrading of I&C functions
	Vandellos-2/1999	WEC	03.08.88	PWR	2775	2900	1999	S	4,5%	Safety and system Analysis/No changes on systems except electric	Electric system
	Vandellos-2/2002	WEC	03.08.88	PWR	2775	2941	2002	E	10,6%	Improvement of the feed water flow measurement	
Sweden	Forsmark-1	ASEA	12.01.80	BWR	2711	2928	1986	E	8,4%	Use of large existing margins	Improved fuel
		ASEA	12.01.80	BWR	2711	3253	2008	E	19,1%		
	Forsmark-2	ASEA	07.01.81	BWR	2711	2928	1986	E	8,4%	Use of large existing margins	Improved fuel
		ASEA	07.01.81	BWR	2711	3250	2009	E	19,1%		
	Forsmark-3	ASEA	08.01.85	BWR	3020	3300	1989	E	9,3%	Use of large existing margins	Improved fuel
		ASEA	08.01.85	BWR	3020	3775	2010	E	25,0%		
	Oskarshamn-2	ASEA	01.01.75	BWR	1700	1800	1982	S	5,9%	Use of large existing margins	Improved fuel
	Oskarshamn-3	ASEA	08.15.85	BWR	3020	3300	1989	E	9,3%	Use of large existing margins	Improved fuel
		ASEA	08.15.85	BWR	3020	3900	2008	E	29,8%		
	Ringhals-1	ASEA	01.01.76	BWR	2270	2500	1989	E	10,1%	Use of large existing margins	Improved fuel
		ASEA	01.01.76	BWR	2270	2540	2006	MU	11,9%		
	Ringhals-2	WEC	05.01.75	PWR	2440	2660	1989	E	9%		SGR
	Ringhals-3	WEC	09.01.81	PWR	2783	3000	2006	E	7,8%		
		WEC	09.01.81	PWR	2783	3160	2007	S	13,5%		
	Ringhals-4	WEC	11.21.83	PWR	2783	3160	2011	E	13,5%		

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					MWt	MWt		(E, S, MU)*	%		
Switzerland	Leibstadt/1985	GE	12.15.84	BWR	3012	3138	1985	S	4,2%		Modification of Turbine System
	Leibstadt1998	GE	12.15.84	BWR	3012	3327	1998	S	6%		
	Leibstadt.1999	GE	12.15.84	BWR	3012	3420	1999	E	9%		
	Leibstadt/2000	GE	12.15.84	BWR	3012	3515	2000	E	12%		
	Leibstadt/2003	GE	12.15.84	BWR	3012	3600	2002	E	14,7%		
	Mühleberg/1976	BBC	11.06.72	BWR	947	997	1976	S	5,3%		
	Mühleberg/1993	BBC	11.06.72	BWR	947	1047	1993	S	5%		
	Mühleberg/1994	BBC	11.06.72	BWR	947	1097	1994	E	10%		
	Gösgen/1985	KWU	11.01.79	PWR	2808	2900	1985	S	3,3%		
	Gösgen/1992	KWU	11.01.79	PWR	2808	3002	1992	S	6,9%		
USA	ANO-2	CE	03.26.80	PWR	2815	3026	2002	E	7,5%	Revision of Safety analyses (SLB increase design containment pressure), revision of transition reload safety analyses	SGR, modifications to the BOP changes to the Main Unit Turbine/Generator, the Main Unit Condenser, and accessories and associated supporting systems.
	Beaver Valley 1/2001	WEC	10.01.76	PWR	2652	2689	2001	MU	1,4%		
	Beaver Valley 1/2006	WEC	10.01.76	PWR	2652	2900	2006	E	9,4%	Revision of Safety analyses	Upgrading of existing eqquipment, component i
	Beaver Valley 2/2001	WEC	11.17.87	PWR	2652	2689	2001	MU	1,4%		
	Beaver Valley 2/2006	WEC	11.17.87	PWR	2652	2900	2006	E	9,4%	Revision of Safety analyses	Upgrading of existing eqquipment, component i
	Braidwood 1	WEC	07.29.88	PWR	3411	3587	2001	S	5,2%		
	Braidwood 2	WEC	10.17.88	PWR	3411	3587	2001	S	5,2%		
	Browns Ferry 2/1998	GE	03.01.75	BWR	3294	3458	1998	S	5,0%		
	Browns Ferry 2/2007	GE	03.01.75	BWR	3294	3952	2007	E	20%	Revision of Safety analyses, updated analysis methods to predict the load's experience by its steam dryers during power uprate operation.	Upgrading of existing eqquipment, component instrumentation of a steam dryer to collect actual pressure load data during extended power uprate operation,
	Browns Ferry 3	GE	03.01.75	BWR	3294	3458	1998	S	5,0%		
	Browns Ferry 3/2007	GE	03.01.75	BWR	3294	3952	2007	E	20%	Revision of Safety analyses, updated analysis methods to predict the load's experience by its steam dryers during power uprate operation.	Upgrading of existing eqquipment, component instrumentation of a steam dryer to collect actual pressure load data during extended power uprate operation,
	Browns Ferry 1	GE	01.08.74	BWR	3293	3952	2007	E	20%	Revision of Safety analyses, updated analysis methods to predict the load's experience by its steam dryers during power uprate operation.	Upgrading of existing eqquipment, component instrumentation of a steam dryer to collect actual pressure load data during extended power uprate operation,
	Brunswick 1/1996	GE	03.18.77	BWR	2436	2558	1996	S	5,0%		
	Brunswick 1/2002	GE	03.18.77	BWR	2436	2923	2002	E	20,0%		
	Brunswick 2/1996	GE	11.03.75	BWR	2436	2558	1996	S	5,0%		

Country	1. Station name	2. Key data		3. Plant type		4. Uprate data					
Country		2.1 Vendor NSSS	2.2 . Comm. Operation	3.1 Reactor type	3.2. Initial power	4.1 Uprate power	4.2 Year implemented	4.3 Uprate type & power increase		4.4 Technical solution	4.5 Equipment
					MWt	MWt		(E, S, MU)*	%		
USA	Brunswick 2/2002	GE	11.03.75	BWR	2436	2923	2002	E	20,0%	increasing steam production, holding liquid flow in the core, dome pressure and temperatures near current values. The increased steam production is achieved by "flattening" the core power profile, which involves increasing power generation in the outer regions of the core. There is an increase in feedwater flow to match the increased production of steam. performed in two steps	BOP modification
	Byron 1	WEC	09.16.85	PWR	3411	3587	2001	S	5,2%		
	Byron 2	WEC	08.21.87	PWR	3411	3587	2001	S	5,2%		
	Callaway	WEC	04.09.85	PWR	3411	3565	1988	S	4,5%		
	Calvert Cliffs 1/1977	CE	05.08.75	PWR	2560	2700	1977	S	5,5%		
	Calvert Cliffs 1/2006	CE	05.08.75	PWR	2560	2737	2006	MU	6,8%		
	Calvert Cliffs 2/1977	CE	04.01.77	PWR	2560	2700	1977	S	5,5%		
	Calvert Cliffs 2/2006	CE	04.01.77	PWR	2560	2700	2006	MU	6,8%		
	Clinton	GE	11.24.87	BWR	2894	3473	2002	E	20,0%	Revision of the design basis accident (higher steam and feedwater flows). The constant-pressure power uprate Done in two steps of 7 and 13%. the unit is operating within the power range of other BWR/6 nuclear steam supply systems.	New design of the fuel, BOP modification
	Comanche Peak 1	WEC	08.13.90	PWR	3411	3458	2001	MU	1,4%		
	Comanche Peak 2	WEC	08.03.93	PWR	3411	3445	1999	MU	1,0%		
	Comanche Peak 2					3458	2001	MU	1,4%		
	Crystal River 3	BW	03.13.77	PWR	2544	2568	2002	S	0,9%		
	D.C. Cook 1	WEC	05.07.85	PWR	3250	3304	2002	MU	1,7%		
	D.C. Cook 2	WEC	03.13.86	PWR	3411	3468	2003	MU	1,7%		
	Diablo Canyon 1	WEC	05.07.85	PWR	3338	3411	2000	S	2,2%		
	Dresden 2	GE	06.09.70	BWR	2527	2957	2001	E	17,0%	Increase of main steam and feedwater flows	modification of high-pressure turbine
	Dresden 3	GE	11.16.71	BWR	2527	2957	2001	E	17,0%	Increase of main steam and feedwater flows	modification of high-pressure turbine
	Duane Arnold/1985	GE	02.01.75	BWR	1593	1658	1985	S	4,1%		
	Duane Arnold/2001	GE	02.01.75	BWR	1593	1912	2001	E	19,4%	increasing steam production, holding liquid flow in the core, dome pressure and temperatures near current values. The increased steam production is achieved by "flattening" the core power profile, which involves increasing power generation in the outer regions of the core. There is an increase in feedwater flow to match the increased production of steam. performed in two steps	BOP modification
	Farley 1	WEC	12.01.77	PWR	2652	2775	1998	S	4.6%		
	Farley 2	WEC	07.30.81	PWR	2652	2775	1998	S	4,6%		
	Fermi 2	GE	01.23.88	BWR	3293	3430	1992	S	4,2%		
	Fitzpatrick	GE	07.28.75	BWR	2436	2536	1996	S	4,1%		
	Fort Calhoun/1980	CE	09.26.73	PWR	1420	1500	1980	S	5,6%		
	Fort Calhoun/2004	CE	09.26.73	PWR	1420	1525	2004	MU	7,4%		
	Fort Calhoun/2006	CE	09.26.73	PWR	1420	1547	2006	MU	8,9%		
	Ginna	WEC	07.01.70	PWR	1520	1775	2006	E	16.8%		

Country	1. Station name	2. Key data		3. Plant type		4. Uprate data					
Country		2.1 Vendor NSSS	2.2 . Comm. Operation	3.1 Reactor type	3.2. Initial power	4.1 Uprate power	4.2 Year implemented	4.3 Uprate type & power increase		4.4 Technical solution	4.5 Equipment
					MWt	MWt		(E, S, MU)*	%		
USA	Grand Gulf	GE	07.01.85	BWR	3833	3898	2002	MU	1,7%		
	H. B. Robinson/1979	WEC	03.07.71	PWR	2200	2300	1979	S	4,5%		
	H. B. Robinson/2002	WEC	03.07.71	PWR	2200	2339	2002	MU	6,3%		
	Hatch 1/1995	GE	01.01.76	BWR	2436	2558	1995	S	5,0%		
	Hatch 1/1998	GE	01.01.76	BWR	2436	2763	1998	E	13,4%		
	Hatch 1/2003	GE	01.01.76	BWR	2436	2804	2003	MU	15,1%		
	Hatch 2/1995	GE	09.05.79	BWR	2436	2558	1995	S	5,0%		
	Hatch 2/1998	GE	09.05.79	BWR	2436	2763	1998	E	13,4%		
	Hatch 2/2003	GE	09.05.79	BWR	2436	2804	2003	MU	15,1%		
	Hope Creek	GE	12.20.86	BWR	3293	3339	2001	MU	1,4%		
	Indian Point 2/2003	WEC	08.01.74	PWR	3071	3114	2003	MU	1,4%		
	Indian Point 2/2004	WEC	08.01.74	PWR	3071	3216	2004	S	4.7%		
	Indian Point 3/2002	WEC	08.30.76	PWR	3025	3067	2002	MU	1,4%		
	Indian Point 3/2005	WEC	08.30.76	PWR	3025	3216	2005	S	6.3%		
	Kewanee /2003	WEC	06.16.74	PWR	1650	1673	2003	MU	1,4%		
	Kewanee /2004	WEC	06.16.74	PWR	1650	1772	2004	S	7.4%		
	LaSalle 1	GE	01.01.84	BWR	3323	3489	2000	S	5,0%	Increase of main steam and feedwater flows	modification of BOP
	LaSalle 2	GE	01.01.85	BWR	3323	3489	2000	S	5,0%	Increase of main steam and feedwater flows	modification of BOP
	Limerick 1	GE	02.01.86	BWR	3293	3458	1996	S	5,0%	Increase of main steam and feedwater flows	modification of BOP
	Limerick 2	GE	01.08.90	BWR	3293	3458	1995	S	5,0%	Increase of main steam and feedwater flows	modification of BOP
	Millstone 2	CE	12.26.75	PWR	2560	2700	1979	S	5,5%		
	Monticello	GE	06.30.71	BWR	1670	1775	1998	E	6,3%	Increase of main steam and feedwater flows	modification of BOP, new design of fuel
	Nine Mile Point 2	GE	03.11.88	BWR	3323	3467	1995	S	4,3%	Increase of main steam and feedwater flows	modification of BOP, new design of fuel
	North Anna 1	WEC	06.06.78	PWR	2775	2893	1986	S	4,3%		
	North Anna 2	WEC	12.14.80	PWR	2775	2893	1986	S	4,3%		
	Palisades	CE	12.31.71	PWR	2530	2565	2004	MU	1,4%		
	Palo Verde 1/1996	CE	02.13.86	PWR	3800	3876	1996	S	2,0%		
	Palo Verde 1/2005	CE	02.13.86	PWR	3800	3990	2005	S	4,9%		
	Palo Verde 2/1996	CE	09.19.86	PWR	3800	3876	1996	S	2,0%		
	Palo Verde 2/2003	CE	09.19.86	PWR		3990	2003	S	5,0%		
	Palo Verde 3/1996	CE	01.15.88	PWR	3800	3876	1996	S	2,0%		
	Palo Verde 3/2005	CE	01.15.88	PWR	3800	3990	2005	S	4,9%		
	Peach Bottom 2/1994	GE	07.05.74	BWR	3293	3458	1994	S	5,0%		
	Peach Bottom 2/2002	GE	07.05.74	BWR	3293	3514	2002	MU	6,7%		
	Peach Bottom 3/1995	GE	12.23.74	BWR	3293	3458	1995	S	5,0%		
	Peach Bottom 3/2002	GE	12.23.74	BWR	3293	3514	2002	MU	6,7%		
	Perry	GE	11.18.87	BWR	3579	3758	2000	S	5,0%		
	Pilgrim	GE	12.09.72	BWR	1998	2028	2003	MU	1,5%		
	Point Beach 1	WEC	12.21.70	PWR	1519	1540	2002	MU	1,4%		
	Point Beach 2	WEC	09.30.72	PWR	1519	1540	2002	MU	1,4%		

Country	1. Station name	2. Key data		3. Plant type		4. Uprate data					
Country		2.1 Vendor NSSS	2.2 . Comm. Operation	3.1 Reactor type	3.2. Initial power	4.1 Uprate power	4.2 Year implemented	4.3 Uprate type & power increase		4.4 Technical solution	4.5 Equipment
					MWt	MWt		(E, S, MU)*	%		
USA	Quad Cities 1	GE	02.18.73	BWR	2511	2957	2001	E	17,8%	Increase of main steam and feedwater flows	modification of high-pressure turbine
	Quad Cities 2	GE	03.10.73	BWR	2511	2957	2001	E	17,8%	Increase of main steam and feedwater flows	modification of high-pressure turbine
	River Bend/2000	GE	06.16.86	BWR	2894	3039	2000	S	5,0%	two phases: 1.steam flow/feedwater flow increase in power (only flow) 2.flow increase/reactor pressure increase	modification of high-pressure turbine, new fuel design
	River Bend/2003	GE	06.16.86	BWR	2894	3091	2003	MU	6,7%		
	Salem 1/1986	WEC	06.30.77	PWR	3344	3411	1986	S	2,0%		
	Salem 1/2001	WEC	06.30.77	PWR	3344	3459	2001	MU	3,4%		
	Salem 2	WEC	10.13.81	PWR	3411	3459	2001	MU	1,4%		
	San Onofre 2	CE	08.08.83	PWR	3390	3438	2001	MU	1,4%		
	San Onofre 3	CE	04.01.84	PWR	3390	3438	2001	MU	1,4%		
	Seabrook1/2005	WEC	08.19.90	PWR	3411	3587	2005	S	5.2%		
	Seabrook1/2006	WEC	08.19.90	PWR	3411	3648	2006	S	6.9%		
	Sequoyah 1	WEC	07.01.81	PWR	3411	3455	2002	MU	1,3%		
	Sequoyah 2	WEC	06.01.82	PWR	3411	3455	2002	MU	1,3%		
	Shearon Harris	WEC	05.02.87	PWR	2775	2900	2001	S	4,5%		
	South Texas 1	WEC	08.25.88	PWR	3800	3853	2002	MU	1,4%		
	South Texas 2	WEC	06.18.89	PWR	3800	3853	2002	MU	1,4%		
	St. Lucie 1	CE	12.21.76	PWR	2560	2700	1981	S	5,5%		
	St. Lucie 2	CE	08.08.83	PWR	2560	2700	1985	S	5,5%		
	Surry 1	WEC	12.02.72	PWR	2441	2546	1995	S	4,3%		
	Surry 2	WEC	05.01.73	PWR	2441	2546	1995	S	4,3%		
	Susquehanna 1/1995	GE	06.08.83	BWR	3293	3441	1995	S	4,5%		
	Susquehanna 1/2001	GE	06.08.83	BWR	3293	3489	2001	MU	6,0%		
	Susquehanna 2/1994	GE	02.12.85	BWR	3293	3441	1994	S	4,5%		
	Susquehanna 2/12001	GE	02.12.85	BWR	3293	3489	2001	MU	6,0%		
	TMI-1	BW	09.02.74	PWR	2535	2568	1988	S	1,3%		
	Turkey Point 3	WEC	12.04.72	PWR	2200	2300	1996	S	4,5%		
	Turkey Point 4	WEC	09.07.73	PWR	2200	2300	1996	S	4,5%		
	Vermont Yankee	GE	11.29.72	BWR	1593	1912	2006	E	20.0%	Revision of Safety analyses, updated analysis methods to predict the load's experience by its steam dryers during power uprate operation.	Upgrading of existing eqquipment, component instrumentation of a steam dryer to collect actual pressure load data during extended power uprate operation,
	V. C. Summer	WEC	12.02.72	PWR	2775	2900	1996	S	4,5%		
	Vogtle 1	WEC	05.31.87	PWR	3411	3565	1993	S	4,5%		
	Vogtle 2	WEC	05.19.89	PWR	3411	3565	1993	S	4,5%		
	Waterford 3/2002	CE	09.24.85	PWR	3390	3441	2002	MU	1,5%		
	Waterford 3/2005	CE	09.24.85	PWR	3390	3665	2005	E	8,0%		
	Watts Bar	WEC	05.27.96	PWR	3411	3459	2001	MU	1,4%		
	WNP-2			BWR	3323	3486	1995	S	4,9%		

Country	1. Station name	2. Key data		3. Plant type		4. Uprate data				
Country		2.1 Vendor NSSS	2.2 . Comm. Operation	**3.1 Reactor type**	**3.2. Initial power**	4.1 Uprate power	4.2 Year implemented	4.3 Uprate type & power increase	4.4 Technical solution	4.5 Equipment
					MWt	MWt		(E, S, MU)* %		
USA	Wolf Creek	WEC	09.03.85	PWR	3411	3565	1993	S 4,5%		
Slovenia	Krsko	WEC	01.01.83	PWR	1882	2000	2000	S 6,3%	Use additional margin by new SG-higher heat transfer area	SGR

*
S- Stretch power (uprates typically up to 7 %, within the design capacity plant). Typically involve changes to instrumentation setpoints but do not involve major plant modifications)
MU-Measurement uncertainty (uprates are less than 2 %e achieved by implementing enhanced techniques for calculating reactor power)
E-Extended power (uprates greater than stretch. Require significant modifications to major BOP equipment)
**
Second thermal uprate
***
Third thermal uprate

PWR
BWR

Country	1. Station name	2. Key data		3. Plant type		5. Fuel				6. Annual liquid effluent releases		7. Annual gases effluent releases	
Country		2.1 Vendor NSSS	2.2 . Comm. Operation	3.1 Reactor type	3.2. Initial power	5.1 Fuel cycle	5.2 Average linear fuel rating BU	5.3 Average linear fuel rating AU	5.4 Fuel type	6.1 Before the uprate	6.2 After the uprate	7.1 Before the uprate	7.2 After the uprate
					MWt	months	kW/m	kW/m		GBq	GBq	GBq	GBq
Belgium	Doel 2	WEC	12.01.75	PWR	1192	11	22.22	22.22	FRA-AFA/KWU-AKA	4803 tritium 99,9% year 2003 (14% of total for 4 units)		208 tritium 52% year 2003 (14% of total for 4 units)	
	Doel 3	FRAM	10.11.82	PWR	2785	11	20	20	KWU-AKA/FRA-AFA/ABB-PAAD	15805 tritium 99.9% year 1992 (36% of total for 4 units)	11811 tritium 99.9% year 1994 (36% of total for 4 units)	9782 tritium 3% year 1992 (36% of total for 4 units)	1066 tritium 67% year 1994 (36% of total for 4 units)
	Tihange 1	ACLF	09.01.75	PWR	2665	18	22.15	22.15	FRA Std/W Std/Exxon Std	11041 tritium 99.9 % year 1994 (1/3 of total for 3 units)	14917 tritium 99.9 % year 1996 (1/3 of total for 3 units)	5617 tritium 29 % year 1994 (1/3 of total for 3 units)	6340 tritium 23 % year 1996 (1/3 of total for 3 units)
	Tihange 2/1995	ACLF	06.01.83	PWR	2785	15	17.85	17.85	FRA Std/ FRA AFA	11041 tritium 99.9 % year 1994 (1/3 of total for 3 units)	14917 tritium 99.9 % year 1996 (1/3 of total for 3 units)	5617 tritium 29 % year 1994 (1/3 of total for 3 units)	6340 tritium 23 % year 1996 (1/3 of total for 3 units)
	Tihange 2/2001	ACLF	06.01.83	PWR	2785	15	17.85	17.85	FRA Std/ FRA AFA	11033 tritium 99.9% year 2000 (1/3 of total for 3 units)	19867 tritium99.9% year 2002 (1/3 of total for 3 units)	3693 tritium 68% year 2000 (1/3 of total for 3 units)	4573 tritium 38% year 2002 (1/3 of total for 3 units)
Finland	Loviisa 1	AEE	05.09.77	PWR	1375	12	32.5* * peak linear fuel rating	32.5* * peak linear fuel rating		6000 tritium 99.9% year 1997 (1/2 of total for 2 units)	7000 tritium 99.9% year 1999 (1/2 of total for 2 units)	1825 tritium 7% year 1997 (1/2 of total for 2 units)	3040 tritium 3% year 1999 (1/2 of total for 2 units)
	Loviisa 2	ASEA	01.05.81	PWR	1375	12	32.5* * peak linear fuel rating	32.5* * peak linear fuel rating		6000 tritium 99.9% year 1997 (1/2 of total for 2 units)	7000 tritium 99.9% year 1999 (1/2 of total for 2 units)	1825 tritium 7% year 1997 (1/2 of total for 2 units)	3040 tritium 3% year 1999 (1/2 of total for 2 units)
	Olkiluoto 1/1984	ASEA	10.01.79	BWR	2000	12	17.90 4 MW/Assembly	17.90 4.32 MW/Assembly	8x8-1				
	Olkiluoto 1/1998	ASEA	10.01.79	BWR	2000	12	17.90 4.32 MW/Assembly	14.90 5 MW/Assembly	9x9-1/Altrium 10B "Novel BWR fuel"	655 tritium 99.9% year 1997 (1/2 of total for 2 units)	551 tritium 99.9% year 1999 (1/2 of total for 2 units)	700 tritium 21% year 1997 (1/2 of total for 2 units)	3160 tritium 8% year 1999 (1/2 of total for 2 units)
	Olkiluoto 2/1984	ASEA	07.01.82	BWR	2000	12	17.90 4 MW/Assembly	17.90 4.32 MW/Assembly	8x8-1				
	Olkiluoto 2/1998	ASEA	07.01.82	BWR	2000	12	17.90 4 MW/Assembly	13.10 5 MW/Assembly	SVEA-100/GE12 "Novel BWR fuel"	655 tritium 99.9% year 1997 (1/2 of total for 2 units)	551 tritium 99.9% year 1999 (1/2 of total for 2 units)	700 tritium 21% year 1997 (1/2 of total for 2 units)	3160 tritium 8% year 1999 (1/2 of total for 2 units)
Germany	Brokdorf	KWU	01.11.86	PWR	3765								
	Emsland	KWU	06.20.88	PWR	3765	12	16.67	16.67	KWU Convoy		8300 tritium 99.9% year 1991		780 tritium 86% year 1991
		KWU	06.20.88	PWR	3765								
	Grafenrheinfeld	KWU	06.17.82	PWR	3765								
	Grohnde	KWU	02.01.85	PWR	3765	12	21.10	21.10			16000 tritium 99.9% year 1991		1830 tritium 40% year 1991
		KWU	02.01.85	PWR	3765								
	Gundremmingen B	KWU	07.01.84	BWR	3840								
	Gundremmingen C	KWU	01.01.85	BWR	3840								
	Isar-1	KWU	03.21.79	BWR	2575								
	Isar-2/1991	KWU	04.09.88	PWR	3765	12		16.70	KWU	7200 tritium 99.9% year 1990	16000 tritium 99.9% year 1992	1120 tritium 79% year 1990	1580 tritium 83% year 1992
	Isar-2/1998	KWU	04.09.88	PWR	3765	12		17.10	WEST/KWU			1140 tritium 85% year 1997	980 tritium 49% year 1999
	Neckar-2/1991	KWU	04.15.89	PWR	3765	12				16740 tritium 99.9% year 1990 (62% of total for 2 units)	14880 tritium 99.9% year 1990 (62% of total for 2 units)	11960 tritium 6% year 1990 (62% of total for 2 units)	10168 tritium 5.5% year 1992 (62% of total for 2 units)
	Neckar-2/2005	KWU	04.15.89	PWR		12							
	Philippsburg-2.1991	KWU	12.17.84	PWR	3765	12		23.14	KWU 16x16-20	19000 tritium 99.9% year 1990	15001 tritium 99.9% year 1992	1710 tritium 94% year 1990	3300 tritium 45% year 1992
	Philippsburg-2/1992	KWU	12.17.84	PWR	3765	12		23.14	KWU 16x16-20	17000 tritium 99.9% year 1991	13000 tritium 99.9% year 1993	1880 tritium 75% year 1991	1560 tritium 77% year 1993
	Philippsburg-2/2000	KWU	12.17.84	PWR	3765	12		29.70	KWU/Fragema	18000 tritium 99.9% year 1999	13000 tritium99.9% year 2001	4510 tritium 24% year 1999	1001 tritium 30% year 2001
	Unterwester	KWU	10.01.79		3733	11	20.50	20.50		7700 tritium 99.9% year 1999	16000 tritium99.9% year 2001	4357 tritium 10% year 1999	3380 tritium 9% year 2001
Hungary	PAKS1	AEE	11.03.83	PWR	1375								
	PAKS2	AEE	09.21.84	PWR	1375								
	PAKS3	AEE	11.03.86	PWR	1375								
	PAKS4	AEE	11.19.87	PWR	1375								
Japan	Higasidory		05.03.05	BWR	?								
	Shhika	HIT	07.30.93	BWR	1593	15	14		8x8 high burnup				
Korea	Kori 3	WEC	10.01.86	PWR	2775	18	17.83		V-5H				
	Kori-4	WEC	04.29.86	PWR	2775	18	17.83		V-5H				
	YGN 1	WEC	08.25.86	PWR	2775	18	17.83		V-5H				
	YGN 2	WEC	06.10.87	PWR	2775	18	17.83		V-5H				

Country	**1. Station name**	**2. Key data**		**3. Plant type**		5. Fuel				**6. Annual liquid effluent releases**		7. Annual gases effluent releases	
Country		2.1 Vendor NSSS	2.2 . Comm. Operation	**3.1 Reactor type**	**3.2. Initial power**	5.1 Fuel cycle	5.2 Average linear fuel rating BU	5.3 Average linear fuel rating AU	5.4 Fuel type	6.1 Before the uprate	6.2 After the uprate	7.1 Before the uprate	7.2 After the uprate
					MWt	months	kW/m	kW/m		GBq	GBq	GBq	GBq
Mexco	LV 1	GE	07.29.90	BWR	1931	16	17.72 4.35 MW/Assembly	17.72 4.56 MW/Assembly					
	LV 2	GE	04.10.95	BWR	1931	22	17.72 4.35 MW/Assembly	17.72 4.56 MW/Assembly					
Spain	Almaraz-1	WEC	09.01.83	PWR	2696	18	17.30	17.30	"4.35"			9148 tritium 96% year 2002 (1/2 of total for 2 units)	
	Almaraz-2	WEC	07.01.84	PWR	2696	18	17.30	17.30	"4.35"			9148 tritium 96% year 2002 (1/2 of total for 2 units)	
	Asco-1/2000	WEC	12.01.84	PWR	2696	12	17.30 3.5 avg enrich	18.92 4.7 avg enrich	STD/ MAEF-IFM (Enusa)	43950 tritium 99.9% year 1999 (1/2 of total for 2 units)	24750 tritium 99.9% year 2001 (1/2 of total for 2 units)	12810 tritium 13% year 1999 (1/2 of total for 2 units)	2345 tritium 39% year 2001 (1/2 of total for 2 units)
	Asco-1/2003	WEC	12.01.84	PWR	2696	12		18.92	MAEF-IFM (Enusa)			2655 tritium 56% year 2002 (1/2 of total for 2 units)	
	Asco-2/1999	WEC	06.01.85	PWR	2696	12	17.30 3.5 avg enrich	18.92 4.7 avg enrich	STD/ MAEF-IFM (Enusa)	26106 tritium 99.9% year 1998 (1/2 of total for 2 units)	39050 tritium 99.9% year 2000 (1/2 of total for 2 units)	8560 tritium 12% year 1998 (1/2 of total for 2 units)	2100 tritium 38% year 2000 (1/2 of total for 2 units)
	Asco-2/2004	WEC	06.01.85	PWR	2696	12		18.92 4.7 avg enrich	MAEF-IFM (Enusa)			5251 tritium 19% year 2003 (1/2 of total for 2 units)	
	Cofrentes	GE	03.11.85	BWR	2894	18	4.64 MW/Assembly	4.73 MW/Assembly	Vale				
	Cofrentes	GE	03.11.85	BWR	2894	18	4.73 MW/Assembly	20.03 4.83 MW/Assembly	GE10/GE11	511 tritium 99.9% year 1997	227 tritium 99.9% year 1999	9180 tritium 13% year 1997	5567 tritium 18% year 1999
	Cofrentes	GE	03.11.85	BWR	2894	18	20.03 4.83 MW/Assembly	20.03 5.1 MW/Assembly	SVEA96/GE12/GE14/Altrium	1250 tritium 99.9% year 2001	1020 tritium 99.9% year 2003	21570 tritium 26% year 2001	16100 tritium 12% year 2003
	Cofrentes	GE	03.11.85	BWR	2894	18	20.03 5.1 MW/Assembly	20.03 5.2 MW/Assembly	SVEA96/GE12/GE14/Altrium			21036 tritium 19% year 2002	
	Vandellos-2/1999	WEC	03.08.88	PWR	2775	18		17.80 3.6 avg enric	OFA/ AEF	33509 tritium 99.9% year 1998	35727 tritium 99.9% year 2000	528 tritium 98% year 1998	15754 tritium 2% year 2000
	Vandellos-2/2002	WEC	03.08.88	PWR	2775	18		18.92 4.7 avg enrich	MAEF-IFM (Enusa)	10818 tritium 99.9% year 2001	32325 tritium 99.9% year 2003	442 tritium 62% year 2001	1978 tritium 10% year 2003
Sweden	Forsmark-1	ASEA	12.01.80	BWR	2711	12	4.01 MW/Assembly	106 ?? 4.33 MW/Assembly	8x8/9x9	759 tritium 76% year 1985 (1/2 of total for 2 units)	754 tritium 84% year 1987 (1/2 of total for 2 units)	71107 tritium insignificant, year 1985	333 tritium insignificant, year 1987
		ASEA	12.01.80	BWR	2711								
	Forsmark-2	ASEA	07.01.81	BWR	2711	12	4.01 MW/Assembly	106 ?? 4.33 MW/Assembly	8x8/10x10	759 tritium 76% year 1985 (1/2 of total for 2 units)	754 tritium 84% year 1987 (1/2 of total for 2 units)	232118 tritium insignificant, year 1985	198457 tritium insignificant, year 1987
		ASEA	07.01.81	BWR	2711								
	Forsmark-3	ASEA	08.01.85	BWR	3020	12	4.31 MW/Assembly	17.5 4.71 MW/Assembly	8x8/Altrium 10B	395 tritium 99% year 1988	339 tritium 90% year 1990	323470 tritium insignificant, year 1988	301412 tritium insignificant, year 1990
		ASEA	08.01.85	BWR	3020								
	Oskarshamn-2	ASEA	01.01.75	BWR	1700	12	3.83 MW/Assembly	39* peak linear fuel rating 4.05 MW/Assembly	8x8/Altrium 10	361 tritium 83% year 1981 (1/2 of total for 2 units)	399 tritium 89% year 1983 (1/2 of total for 2 units)	421231 tritium insignificant, year 1981	54845 tritium insignificant, year 1983
	Oskarshamn-3	ASEA	08.15.85	BWR	3020	12	4.31 MW/Assembly	41.5* peak linear fuel rating 4.71 MW/Assembly	8x8/10x10	438 tritium 98% year 1988	281 tritium 94% year 1990	109120 tritium insignificant, year 1988	69201 tritium insignificant, year 1990
		ASEA	08.15.85	BWR	3020								
	Ringhals-1	ASEA	01.01.76	BWR	2270	11	3.50 MW/Assembly	40 28 3.85 MW/Assembly	ABB 8 x8/ ABB SUEA64/SUEA100/ Altrium10B	669 tritium 95% year 1988	778 tritium 91% year 1990	489382 tritium insignificant, year 1988	56267 tritium insignificant, year 1988
		ASEA	01.01.76	BWR	2270								
	Ringhals-2	WEC	05.01.75	PWR	2440	12		22.02	W/ KWU/W FRA-ANP AFA 36		15691 tritium 0.5% year 1990 (32% of total for 3 units)		69760 tritium not measurd year 1990 (32% of total for 3 units)
	Ringhals-3	WEC	09.01.81	PWR	2783								
		WEC	09.01.81	PWR	2783								
	Ringhals-4	WEC	11.21.83	PWR	2783								

Country	1. Station name	2. Key data		3. Plant type		5. Fuel				6. Annual liquid effluent releases		7. Annual gases effluent releases	
Country		2.1 Vendor NSSS	2.2 . Comm. Operation	3.1 Reactor type	3.2. Initial power	5.1 Fuel cycle	5.2 Average linear fuel rating BU	5.3 Average linear fuel rating AU	5.4 Fuel type	6.1 Before the uprate	6.2 After the uprate	7.1 Before the uprate	7.2 After the uprate
					MWt	months	kW/m	kW/m		GBq	GBq	GBq	GBq
Switzerland	Leibstadt/1985	GE	12.15.84	BWR	3012	12	4.65 MW/Assembly	13.30 4.84 MW/Assembly	GE6				
	Leibstadt1998	GE	12.15.84	BWR	3012	12	13.30 4.84 MW/Assembly	13.30 5.13 MW/Assembly	GEA 6,7,8/SVEA 96				
	Leibstadt.1999	GE	12.15.84	BWR	3012	12	13.30 5.13 MW/Assembly	13.30 5.28 MW/Assembly	GEA 6,7,8/SVEA 96				
	Leibstadt/2000	GE	12.15.84	BWR	3012	12	13.30 5.28 MW/Assembly	13.30 5.42 MW/Assembly	GEA 6,7,8/SVEA 96				
	Leibstadt/2003	GE	12.15.84	BWR	3012	12	13.30 5.42 MW/Assembly	13.30 5.55 MW/Assembly	GEA 6,7,8,10/SVEA 96				
	Mühleberg/1976	BBC	11.06.72	BWR	947	12	3.95 MW/Assembly	4.15 MW/Assembly	GE				
	Mühleberg/1993	BBC	11.06.72	BWR	947	12	4.15 MW/Assembly	39.70 4.36 MW/Assembly	GE 11				
	Mühleberg/1994	BBC	11.06.72	BWR	947	12	4.36 MW/Assembly	39.70/ 14.6 4.57 MW/Assembly	GE 11/ GE14				
	Gösgen/1985	KWU	11.01.79	PWR	2808	12			RBU				
	Gösgen/1992	KWU	11.01.79	PWR	2808	12		22.6		12000 tritium 99.9% year 1991	13000 tritium 99.9% year 1993	5100 tritium not measurd year 1991	11000 tritium not measurd year 1993
USA	ANO-2	CE	03.26.80	PWR	2815	18		18.0					
	Beaver Valley 1/2001	WEC	10.01.76	PWR	2652	18	17.06	17.06	West Vantage 5	22904 tritium 99,99%, year 2000	13124 tritium 99.99%, year 2002	11359 tritium 84,5%, year 2000	5907 tritium 88.0%, year 2002
	Beaver Valley 1/2006	WEC	10.01.76	PWR	2652								
	Beaver Valley 2/2001	WEC	11.17.87	PWR	2652	18	17.06	17.06	West Vantage 5				1205 tritium 80.0%, year 2002
	Beaver Valley 2/2006	WEC	11.17.87	PWR	2652								
	Braidwood 1	WEC	07.29.88	PWR	3411	18	18.3	18.3		49103 tritium 99,99%, year 2000	43327 tritium 99,99%, year 2002	656 tritium 97,5%, year 2000	176 tritium 91,2%, year 2002
	Braidwood 2	WEC	10.17.88	PWR	3411	18	18.3	18.3		51487 tritium 99,99%, year 2000	43327 tritium 99,99%, year 2002	698 tritium 97,1%, year 2000	36 tritium 26,2%, year 2002
	Browns Ferry 2/1998	GE	03.01.75	BWR	3294	24	18.49 4.31 MW(Assembly	18.49 4.53 MW/Assembly	3E 98				
	Browns Ferry 2/2007	GE	03.01.75	BWR	3294	24	18.49 4.53 MW/Assembly	18.49 5.17 MW/Assembly	3E 98				
	Browns Ferry 3	GE	03.01.75	BWR	3294	24	17.5 4.31 MW(Assembly	17.5 4.53 MW/Assembly	GE 98				
	Browns Ferry 3/2007	GE	03.01.75	BWR	3294	24	17.5 4.53 MW/Assembly	17.5 5.17 MW/Assembly	GE 98				
	Browns Ferry 1	GE	01.08.74	BWR	3293	24	18.25 4.31 MW(Assembly	18.25 5.17 MW/Assembly	GE 7B				
	Brunswick 1/1996	GE	03.18.77	BWR	2436	24	4.35 MW/Assembly	18.42 4,56 MW/Assembly					
	Brunswick 1/2002	GE	03.18.77	BWR	2436	24	18.42 4,56 MW/Assembly	18.42 5.22 MW/Assembly		4514 tritium 99,99%, year 2001	2234 tritium 99,99%, year 2003	27861 tritium 19,4%, year 2001	9805 tritium 35,7%, year 2003
	Brunswick 2/1996	GE	11.03.75	BWR	2436	24	4.35 MW/Assembly	18.61 4,56 MW/Assembly					

Country	1. Station name	2. Key data		3. Plant type		5. Fuel				6. Annual liquid effluent releases		7. Annual gases effluent releases	
Country		2.1 Vendor NSSS	2.2 . Comm. Operation	3.1 Reactor type	3.2. Initial power	5.1 Fuel cycle	5.2 Average linear fuel rating BU	5.3 Average linear fuel rating AU	5.4 Fuel type	6.1 Before the uprate	6.2 After the uprate	7.1 Before the uprate	7.2 After the uprate
					MWt	months	kW/m	kW/m		GBq	GBq	GBq	GBq
USA	Brunswick 2/2002	GE	11.03.75	BWR	2436	24	18.61 4,56 MW/Assembly	18.61 5.22 MW/Assembly					
	Byron 1	WEC	09.16.85	PWR	3411	18	18.30	18.30		42809 tritium 99,9%, year 2000	35150 tritium 99,9%, year 2002	74 tritium 40,7%, year 2000	148 tritium 57,4%, year 2002
	Byron 2	WEC	08.21.87	PWR	3411	18	18.30	18.30		42809 tritium 99,9%, year 2000	35113 tritium 99,9%, year 2002	125 tritium 66,5%, year 2000	131 tritium 80,8%, year 2002
	Callaway	WEC	04.09.85	PWR	3411	18	19.13	19.13	Vantage 5				
	Calvert Cliffs 1/1977	CE	05.08.75	PWR	2560	24	20.62	20.62	C-E 14x14				
	Calvert Cliffs 1/2006	CE	05.08.75	PWR	2560								
	Calvert Cliffs 2/1977	CE	04.01.77	PWR	2560	24	20.56	20.56	C-E 14x14				
	Calvert Cliffs 2/2006	CE	04.01.77	PWR	2560								
	Clinton	GE	11.24.87	BWR	2894	24	18.85 4.63 MW/Assembly	18.85 5.57 MW/Assembly	GE6/GE7B/GE8B	insignificant, year 2000	insignificant, year 2004	1554 tritium 99,9%, year 2000	2072 tritium 57,3%, year 2004
	Comanche Peak 1	WEC	08.13.90	PWR	3411	18	17.81	17.81	SIEMENS	5609 tritium 99.9%, year 2000	51258 tritium 99.9%, year 2002	1133 tritium 96,6%, year 2000	10541 tritium 20.0%, year 2002
	Comanche Peak 2	WEC	08.03.93	PWR	3411	18	17.8	17.8	SIEMENS				
	Comanche Peak 2					18	17.8	17.8	SIEMENS				
	Crystal River 3	BW	03.13.77	PWR	2544	24	18.67	18.67	MARK B4Z	12366 tritium 99.9%, year 2001	25930 tritium 99.9%, year 2003	16078 tritium 3.3%, year 2001	6317 tritium 4.3%, year 2003
	D.C. Cook 1	WEC	05.07.85	PWR	3250	18	21.98	21.98		insignificant, year 2001	insignificant, year 2003	4908 tritium 85.0%, year 2001	5341 tritium 93.0%, year 2003
	D.C. Cook 2	WEC	03.13.86	PWR	3411	18	17.81	17.81					
	Diablo Canyon 1	WEC	05.07.85	PWR	3338	18	17.50	17.50	West Vantage 5		40515 tritium 99.9%, year 2001		9602 tritium 85.0%, year 2001
	Dresden 2	GE	06.09.70	BWR	2527	24	47.6* peak linear fuel rating 3.49 MW/Assembly	47.6* peak linear fuel rating 4.08 MW/Assembly					
	Dresden 3	GE	11.16.71	BWR	2527	24	47.6* peak linear fuel rating 3.49 MW/Assembly	47.6* peak linear fuel rating 4.08 MW/Assembly					
	Duane Arnold/1985	GE	02.01.75	BWR	1593	24	4.33 MW/Assembly	4.51 MW/Assembly	GE 8x8 barrier				
	Duane Arnold/2001	GE	02.01.75	BWR	1593	24	14.40 4.51 MW/Assembly	14.40 5.20 MW/Assembly					
	Farley 1	WEC	12.01.77	PWR	2652	18				17900 tritium 99.9% year 1997 (1/2 of total for 2 units)		4285 tritium 39% year 1997 (1/2 of total for 2 units)	
	Farley 2	WEC	07.30.81	PWR	2652	18				17900 tritium 99.9% year 1997 (1/2 of total for 2 units)		4285 tritium 39% year 1997 (1/2 of total for 2 units)	
	Fermi 2	GE	01.23.88	BWR	3293	18	37.73 4.31 MW/Assembly	37.73 4.49 MW/Assembly					
	Fitzpatrick	GE	07.28.75	BWR	2436	24	4.35 MW/Assembly	4.53 MW/Assembly					
	Fort Calhoun/1980	CE	09.26.73	PWR	1420	18			Standard				
	Fort Calhoun/2004	CE	09.26.73	PWR	1420	18	49.93* * peak linear fuel rating	49.93* * peak linear fuel rating		10900 tritium 99.9%, year 2003		873 tritium 18.6% year 2003	
	Fort Calhoun/2006	CE	09.26.73	PWR	1420								
	Ginna	WEC	07.01.70	PWR	1520								

Country	1. Station name	2. Key data		3. Plant type		5. Fuel				6. Annual liquid effluent releases		7. Annual gases effluent releases	
Country		2.1 Vendor NSSS	2.2 . Comm. Operation	3.1 Reactor type	3.2. Initial power	5.1 Fuel cycle	5.2 Average linear fuel rating BU	5.3 Average linear fuel rating AU	5.4 Fuel type	6.1 Before the uprate	6.2 After the uprate	7.1 Before the uprate	7.2 After the uprate
					MWt	months	kW/m	kW/m		GBq	GBq	GBq	GBq
USA	Grand Gulf	GE	07.01.85	BWR	3833	18	19.50 4.79 MW/Assembly	19.50 4.87 MW/Assembly	ANF 8x8/GE 8x8R	2238 tritium 99.9%, year 2001	861 tritium 99.9%, year 2003	4551 tritium 62,4%, year 2001	3639 tritium 62.1%, year 2003
	H. B. Robinson/1979	WEC	03.07.71	PWR	2200	18			ANF Standard				
	H. B. Robinson/2002	WEC	03.07.71	PWR	2200	18	43.96* * peak linear fuel rating	43.96* * peak linear fuel rating		12443 tritium 99.9% year 2001	6127 tritium 99.9% year 2003	422 tritium 99.0% year 2001	232 tritium 99.0% year 2003
	Hatch 1/1995	GE	01.01.76	BWR	2436	24	4.35 MW/Assembly	4.56 MW/Assembly					
	Hatch 1/1998	GE	01.01.76	BWR	2436	24	4.56 MW/Assembly	4.93 MW/Assembly					
	Hatch 1/2003	GE	01.01.76	BWR	2436	24	4.93 MW/Assembly	5.00 MW/Assembly		593 tritium 99.9%, year 2002	463 tritium 99.9%, year 2004	1166 tritium 44.2%, year 2002	1760 tritium 35.5%, year 2004
	Hatch 2/1995	GE	09.05.79	BWR	2436	24	4.35 MW/Assembly	4.56 MW/Assembly					
	Hatch 2/1998	GE	09.05.79	BWR	2436	24	4.56 MW/Assembly	4.93 MW/Assembly					
	Hatch 2/2003	GE	09.05.79	BWR	2436	24	4.93 MW/Assembly	5.00 MW/Assembly		351 tritium 99.9%, year 2002	741 tritium 99.9%, year 2004	1242 tritium 49.3%, year 2002	2168 tritium 41.3%, year 2004
	Hope Creek	GE	12.20.86	BWR	3293	18	17.52 4.31 MW/Assembly	17.52 4.37 MW/Assembly	GE7/GE9 barrier		144 tritium 99.9%, year 2002		819 tritium 59.3%, year 2002
	Indian Point 2/2003	WEC	08.01.74	PWR	3071	24	18.81	18.81					
	Indian Point 2/2004	WEC	08.01.74	PWR	3071								
	Indian Point 3/2002	WEC	08.30.76	PWR	3025	24	21	21	OFA/V-5	34258 tritium 99.9% year 2001	35309 tritium 99.9% year 2003	335 tritium 38.0% year 2001	674 tritium 4.6% year 2003
	Indian Point 3/2005	WEC	08.30.76	PWR	3025								
	Kewanee /2003	WEC	06.16.74	PWR	1650	18	20.82	20.82		3327 tritium 99.9% year 2002	17368 tritium 99.9% year 2004	105 tritium 99.9% year 2002	792 tritium 99.9% year 2004
	Kewanee /2004	WEC	06.16.74	PWR	1650					10360 tritium 99.9% year 2003		488 tritium 99.9% year 2003	
	LaSalle 1	GE	01.01.84	BWR	3323	24	44 4.35 MW/Assembly	44 4.57 MW/Assembly	GE6/Atrium 9B		insignificant, year 2001		101040 tritium 9%, year 2001
	LaSalle 2	GE	01.01.85	BWR	3323	24	44 4.35 MW/Assembly	44 4.57 MW/Assembly	GE6/Atrium 9B				
	Limerick 1	GE	02.01.86	BWR	3293	24	16.40 4.31MW/Assembly	16.40 4.53MW/Assembly	GE7B/GE8B/GE9B/GE11				
	Limerick 2	GE	01.08.90	BWR	3293	24	16.40 4.31MW/Assembly	16.40 4.53MW/Assembly	GE7B/GE9B/GE11				
	Millstone 2	CE	12.26.75	PWR	2560	18	18.5	18.5					
	Monticello	GE	06.30.71	BWR	1670	24	39 3.45 MW/Assembly	39 3.67 MW/Assembly	8x8/8x8R/P8x8R/BP8x8				
	Nine Mile Point 2	GE	03.11.88	BWR	3323	24	17.68 4.35MW/Assembly	17.68 4.54MW/Assembly	GE6B/GE11				
	North Anna 1	WEC	06.06.78	PWR	2775	18	18.59	18.59	West Standard				
	North Anna 2	WEC	12.14.80	PWR	2775	18	18.59	18.59	West Standard				
	Palisades	CE	12.31.71	PWR	2530	18	25.73	25.73	Siemens	7310 tritium 99.9%, year 2003		3227 tritium 25.5%, year 2003	
	Palo Verde 1/1996	CE	02.13.86	PWR	3800	18	18.14	18.14	C-E System 80	no data	no data	18633 tritium 78%, year 1995 (1/3 of total for 3 units)	no data
	Palo Verde 1/2005	CE	02.13.86	PWR	3800					no data		25726 tritium 96.7%, year 2004	
	Palo Verde 2/1996	CE	09.19.86	PWR	3800	18				no data	no data	18633 tritium 78%, year 1995 (1/3 of total for 3 units)	no data
	Palo Verde 2/2003	CE	09.19.86	PWR			18.21	18.21	C-E System 80	no data	no data	16098 tritium 62%, year 2002	49231 tritium 99%, year 2004
	Palo Verde 3/1996	CE	01.15.88	PWR	3800	18	18.37	18.37	C-E System 80	no data	no data	18633 tritium 78%, year 1995 (1/3 of total for 3 units)	no data
	Palo Verde 3/2005	CE	01.15.88	PWR	3800					no data		36881 tritium 98%, year 2004	
	Peach Bottom 2/1994	GE	07.05.74	BWR	3293	24	4.31MW/Assembly	4.53MW/Assembly					
	Peach Bottom 2/2002	GE	07.05.74	BWR	3293	24	16.37 4.53 MW/Assembly	16.37 4.60 MW/Assembly	GE8B/GE9B/GE11	1308 tritium 99.9%, year 2001	609 tritium 99.9%, year 2003	15911 tritium insignificant, year 2001	29726 tritium 2.4%, year 2003
	Peach Bottom 3/1995	GE	12.23.74	BWR	3293	24	4.31MW/Assembly	4.53MW/Assembly					
	Peach Bottom 3/2002	GE	12.23.74	BWR	3293	24	18.24 4.53 MW/Assembly	18.24 4.60 MW/Assembly	GE7B/GE8B/GE9B				
	Perry	GE	11.18.87	BWR	3579	15-18	19.85 4.78 MW/Assembly	19.85 5.02 MW/Assembly		no data for year 1999	1330 tritium 99.9%, year 2001	no data for year 1999	2093 tritium insignificant, year 2001
	Pilgrim	GE	12.09.72	BWR	1998	24	44* peak linear fuel rating 3.44 MW/Assembly	44* peak linear fuel rating 3.50 MW/Assembly		insignificant, year 2002	no data for year 2004	27842 tritium 92.0%, year 2002	12427 tritium 84.0%, year 2004
	Point Beach 1	WEC	12.21.70	PWR	1519	18	18.7	18.7	OFA	21331 tritium 99.9% year 2001	27683 tritium 99.9% year 2003	3007 tritium 98.0% year 2001	2276 tritium 99.9% year 2003
	Point Beach 2	WEC	09.30.72	PWR	1519	18	18.7	18.7	OFA				

Country	1. Station name	2. Key data		3. Plant type		5. Fuel				6. Annual liquid effluent releases		7. Annual gases effluent releases	
Country		2.1 Vendor NSSS	2.2 . Comm. Operation	3.1 Reactor type	3.2. Initial power	5.1 Fuel cycle	5.2 Average linear fuel rating BU	5.3 Average linear fuel rating AU	5.4 Fuel type	6.1 Before the uprate	6.2 After the uprate	7.1 Before the uprate	7.2 After the uprate
					MWt	months	kW/m	kW/m		GBq	GBq	GBq	GBq
USA	Quad Cities 1	GE	02.18.73	BWR	2511	24	43.96 3.47MW/Assembly	43.96 4.08 MW/Assembly		no data for year 2000	1636 tritium 99.9%, year 2002	no data for year 2000	19170 tritium 35.2%, year 2002
	Quad Cities 2	GE	03.10.73	BWR	2511	24	43.96 3.47MW/Assembly	43.96 4.08 MW/Assembly					
	River Bend/2000	GE	06.16.86	BWR	2894	18	18.86 4.64 MW/Assembly	18.86 4.87 MW/Assembly	GE/Siemens Power Corporation fuel	478 tritium 99.9%, year 1999	1721 tritium 99.9%, year 2001	29488 tritium 1.0%, year 1999	1237 tritium 34.0%, year 2001
	River Bend/2003	GE	06.16.86	BWR	2894	18	18.86 4.87 MW/Assembly	18.86 4.95 MW/Assembly	GE7B/GE8B	3404 tritium 99.9%, year 2002	3944 tritium 99.9%, year 2002	1581 tritium 35.4%, year 2002	2977 tritium 45.2%, year 2004
	Salem 1/1986	WEC	06.30.77	PWR	3344	18			West Standard				
	Salem 1/2001	WEC	06.30.77	PWR	3344	18	17.85	17.85			26344 tritium 99.9% year 2002		14116 tritium 52.2% year 2002
	Salem 2	WEC	10.13.81	PWR	3411	18	17.85	17.85	West Standard		8166 tritium 99.9% year 2002		39564 tritium 18.4% year 2002
	San Onofre 2	CE	08.08.83	PWR	3390	18	18.4	18.4			55078 tritium 99.9% year 2002		3989 tritium 41.0% year 2002
	San Onofre 3	CE	04.01.84	PWR	3390	18	18.4	18.4					
	Seabrook1/2005	WEC	08.19.90	PWR	3411	18							
	Seabrook1/2006	WEC	08.19.90	PWR	3411								
	Sequoyah 1	WEC	07.01.81	PWR	3411	18	17.85	17.85	West Vantage 5H	51164 tritium 99.9% year 2001		11292 tritium 20.5% year 2001	
	Sequoyah 2	WEC	06.01.82	PWR	3411	18	17.85	17.85	West Vantage 5H	no data for year 2001		8973 no data for tritium, year 2001	
	Shearon Harris	WEC	05.02.87	PWR	2775	18	40* * peak linear fuel rating	40* * peak linear fuel rating	West Standard		2612 tritium 99.9% year 2002		4143 tritium 98.6% year 2002
	South Texas 1	WEC	08.25.88	PWR	3800	18	17.3	17.3	West XL	11688 tritium 99.9% year 2001	43586 tritium 99.9% year 2003	20838 tritium 5.2 % year 2001	7370 tritium 23.8% year 2003
	South Texas 2	WEC	06.18.89	PWR	3800	18	17.3	17.3	West XL	33881 tritium 99.9% year 2001	15385 tritium 99.9% year 2003	6933 tritium 43.0 % year 2001	3963 tritium 27.0% year 2003
	St. Lucie 1	CE	12.21.76	PWR	2560	18	19.4	19.4	ANF				
	St. Lucie 2	CE	08.08.83	PWR	2560	18	14.5	14.5					
	Surry 1	WEC	12.02.72	PWR	2441	18	21.16	21.16	West SIF	18100 tritium 99.9%, year 1994 (1/2 of total for 2 units)	18350 tritium 99.9%, year 1996 (1/2 of total for 2 units)	5400 tritium 6%, year 1994 (1/2 of total for 2 units)	7800 tritium 5%, year 1996 (1/2 of total for 2 units)
	Surry 2	WEC	05.01.73	PWR	2441	18	21.16	21.16	West SIF	18100 tritium 99.9%, year 1994 (1/2 of total for 2 units)	18350 tritium 99.9%, year 1996 (1/2 of total for 2 units)	5400 tritium 6%, year 1994 (1/2 of total for 2 units)	7800 tritium 5%, year 1996 (1/2 of total for 2 units)
	Susquehanna 1/1995	GE	06.08.83	BWR	3293	24	4.31MW/Assembly	4.50MW/Assembly					
	Susquehanna 1/2001	GE	06.08.83	BWR	3293	24	19.32 4.50MW/Assembly	19.32 4.57MW/Assembly	Siemens 9x9-2		2446 tritium 99.9%, year 2002		5065 tritium 99.9%, year 2002
	Susquehanna 2/1994	GE	02.12.85	BWR	3293	24	4.31MW/Assembly	4.50MW/Assembly					
	Susquehanna 2/12001	GE	02.12.85	BWR	3293	24	14.32 4.50MW/Assembly	14.32 4.57MW/Assembly	Siemens 9x9-2				
	TMI-1	BW	09.02.74	PWR	2535	24	18.83	18.83	B&W Mark B8				
	Turkey Point 3	WEC	12.04.72	PWR	2200	18	18	18	OFA/LOPAR	13900 tritium 99.9%, year 1994 (1/2 of total for 2 units)	no data	572 tritium 5%, year 1994 (1/2 of total for 2 units)	no data
	Turkey Point 4	WEC	09.07.73	PWR	2200	18	18	18	OFA/LOPAR	13900 tritium 99.9%, year 1994 (1/2 of total for 2 units)	no data	572 tritium 5%, year 1994 (1/2 of total for 2 units)	no data
	Vermont Yankee	GE	11.29.72	BWR	1593		42.5 4.33 MW/Assembly	42.5 5.20 MW/Assembly	GE 7x7/ GE 98 P8DWB				
	V. C. Summer	WEC	12.02.72	PWR	2775	18	17.83	17.83	Vantage+/Performance+	27800 tritium 99.9%, year 1994	21400 tritium 99.9%, year 1996	6120 tritium 18%, year 1994	536 tritium 96%, year 1996
	Vogtle 1	WEC	05.31.87	PWR	3411	18	17.8	17.8		27400 tritium 99.9%, year 1992 (1/2 of total for 2 units)	19450 tritium 99.9%, year 1994 (1/2 of total for 2 units)	6045 tritium 65%, year 1992 (1/2 of total for 2 units)	3640 tritium 60%, year 1994 (1/2 of total for 2 units)
	Vogtle 2	WEC	05.19.89	PWR	3411	18	17.8	17.8		27400 tritium 99.9%, year 1992 (1/2 of total for 2 units)	19450 tritium 99.9%, year 1994 (1/2 of total for 2 units)	6045 tritium 65%, year 1992 (1/2 of total for 2 units)	3640 tritium 60%, year 1994 (1/2 of total for 2 units)
	Waterford 3/2002	CE	09.24.85	PWR	3390	18	17.52	17.52	CEA	12562 tritium 99.9% year 2001	49362 tritium 99.9% year 2003	4965 tritium 69.0 % year 2001	86593 tritium 3.0% year 2003
	Waterford 3/2005	CE	09.24.85	PWR	3390	18							
	Watts Bar	WEC	05.27.96	PWR	3411	18	17.88	17.88	V5H	no data for year 2000	22237 tritium 99.9%, year 2002	no data for year 2000	3085 tritium 60.2%, year 2002
	WNP-2			BWR	3323	12	43 4.35MW/Assembly	43 4.56MW/Assembly					

Country	1. Station name	2. Key data		3. Plant type		5. Fuel				6. Annual liquid effluent releases		7. Annual gases effluent releases	
Country		2.1 Vendor NSSS	2.2 . Comm. Operation	3.1 Reactor type	3.2. Initial power	5.1 Fuel cycle	5.2 Average linear fuel rating BU	5.3 Average linear fuel rating AU	5.4 Fuel type	6.1 Before the uprate	6.2 After the uprate	7.1 Before the uprate	7.2 After the uprate
					MWt	months	kW/m	kW/m		GBq	GBq	GBq	GBq
USA	Wolf Creek	WEC	09.03.85	PWR	3411	18	17.85	17.85	Standard/Vantage 5H	16700 tritium 99.9% year 1992	no data	12040 tritium 5%, year 1992	no data
Slovenia	Krsko	WEC	01.01.83	PWR	1882	15-18	17.62	17.62	Vantage5/Vantage+	10800 tritium 99.9% year 1999	7751 tritium99.9% year 2001	1850 tritium 24% year 1999	890 tritium 30% year 2001

*
S- Stretch power (uprates typically up to 7 %, within the design capacity plant). Typically involve changes to instrumen
MU-Measurement uncertainty (uprates are less than 2 %e achieved by implementing enhanced techniques for calcula
E-Extended power (uprates greater than stretch. Require significant modifications to major BOP equipment)
**
Second thermal uprate
***
Third thermal uprate

PWR
BWR

Country	1. Station name	2. Key data		3. Plant type		8. Occupational dose in operating year		9. Occupational dose in outage		10. Dose received during uprate
Country		2.1 Vendor NSSS	2.2 . Comm. Operation	3.1 Reactor type	3.2. Initial power	8.1 Before the uprate	8.2 After the uprate	9.1 Before the uprate	9.2 After the uprate	
					MWt	manSv	manSv	manSv	manSv	man-Sv
Belgium	Doel 2	WEC	12.01.75	PWR	1192	0.261 (ISOE) 0.28		0.237 (ISOE)		0.214 (outage)+0.195 (SGR)
	Doel 3	FRAM	10.11.82	PWR	2785	1.202 (ISOE 3 year)	0.879 (ISOE, 3 year)	1.03 (ISOE 3 year)	0.75 (ISOE 3 year)	3.30 (1993, ISOE annual)
	Tihange 1	ACLF	09.01.75	PWR	2665	1.282 (ISOE 3 year)	0.677 (ISOE, 3 year)	1.145 (ISOE 3 year)	0.57 (ISOE 3 year)	3.22 (1995, ISOE annual) 3.09 outage
	Tihange 2/1995	ACLF	06.01.83	PWR	2785	1.432 (ISOE 3 year))	0.986 (ISOE, 3 year)	1.140 (ISOE 3 year)	0.858 (ISOE 3 year)	1.41 (1995, ISOE annual) 1.21 outage
	Tihange 2/2001	ACLF	06.01.83	PWR	2785	0.488 (ISOE 3 year)	0.752 (ISOE, 2 year)	0.404 (ISOE 3 year)	0.658 (ISOE 2 year)	1.54 (2001, ISOE annual) 1.45 outage
Finland	Loviisa 1	AEE	05.09.77	PWR	1375	0.979 (ISOE 3 year)	1.095 (ISOE 3 year)	0.901 (ISOE 3 year)	1.042 (ISOE 3 year)	0.869 (1998, ISOE annual) 0.82 outage
	Loviisa 2	ASEA	01.05.81	PWR	1375	0.659 (ISOE 3 year)	0.489 (ISOE 3 year)	0.582 (ISOE 3 year)	0.430 (ISOE 3 year)	1.204 (1998, ISOE annual) 1.127 outage
	Olkiluoto 1/1984	ASEA	10.01.79	BWR	2000	0.502 (ISOE 3 year)	0.677 ISOE 3 year)	no data	no data	0.620 (1984, ISOE annual)
	Olkiluoto 1/1998	ASEA	10.01.79	BWR	2000	0.725 (ISOE 3 year)	0.611 (ISOE 3 year)	0.600 (ISOE 3 year)	0.495 (ISOE 3 year)	0.806 (1998, ISOE annual) 0.721 outage
	Olkiluoto 2/1984	ASEA	07.01.82	BWR	2000	0.410 (ISOE 3 year)	0.677 ISOE 3 year)	no data	no data	0.620 (1984, ISOE annual)
	Olkiluoto 2/1998	ASEA	07.01.82	BWR	2000	0.756 (ISOE 3 year)	0.669 (ISOE 3 year)	0.651 (ISOE 3 year)	0.591 (ISOE 3 year)	1.209 (1998, ISOE annual) 1.115 outage
Germany	Brokdorf	KWU	01.11.86	PWR	3765					
	Emsland	KWU	06.20.88	PWR	3765	0.078 (ISOE 2 year)	0.148 (ISOE 3 year)	0.068 (ISOE 2 year)	0.128 (ISOE 3 year)	0.149 (1990, ISOE annual) 0.130 outage
		KWU	06.20.88	PWR	3765					
	Grafenrheinfeld	KWU	06.17.82	PWR	3765					
	Grohnde	KWU	02.01.85	PWR	3765	0.657 (ISOE 3 year)	0.835 (ISOE 3 year)	no data	0.750 (ISOE 2 year)	0.619 (1990, ISOE annual)
		KWU	02.01.85	PWR	3765					
	Gundremmingen B	KWU	07.01.84	BWR	3840					
	Gundremmingen C	KWU	01.01.85	BWR	3840					
	Isar-1	KWU	03.21.79	BWR	2575					
	Isar-2/1991	KWU	04.09.88	PWR	3765	0.090 (ISOE 3 year)	0.236 (ISOE 3 year)	0.072 (ISOE 3 year)	0.176 (ISOE 3 year)	0.162 (1990, ISOE annual) 0.146 outage
	Isar-2/1998	KWU	04.09.88	PWR	3765	0.220 (ISOE 3 year)	0.167 (ISOE 3 year)	0.182 (ISOE 3 year)	0.118 (ISOE 3 year)	0.193 (1991, ISOE annual) 0.133 outage
	Neckar-2/1991	KWU	04.15.89	PWR	3765	0.093 (ISOE 2 year)	0.224 (ISOE 3 year)	0.053 (ISOE 2 year)	0.161(ISOE 3 year)	0.262 (1991, ISOE annual) 0.133 outage
	Neckar-2/2005	KWU	04.15.89	PWR		no data	no data	no data	no data	no data
	Philippsburg-2.1991	KWU	12.17.84	PWR	3765	0.336 (ISOE 3 year)	0.452 (ISOE 3 year)	0.183 (ISOE 1 year)	0.386 (ISOE 3 year)	0.305 (1991, ISOE annual) 0.266 outage
	Philippsburg-2/1992	KWU	12.17.84	PWR	3765	0.297 (ISOE 3 year)	0.451 (ISOE 3 year)	0.221 (ISOE 2 year)	0.375 (ISOE 23 year)	0.342 (1992, ISOE annual) 0.306 outage
	Philippsburg-2/2000	KWU	12.17.84	PWR	3765	0.225 (ISOE 3 year)	0.260 (ISOE 3 year)	0.142 (ISOE 3 year)	0.159 (ISOE 3 year)	0.334 (2000, ISOE annual) 0.227 outage
	Unterwester	KWU	10.01.79		3733	1.292 (ISOE 3 year)	1.054 (ISOE 3 year)	1.087 (ISOE 3 year)	0.877 (ISOE 3 year)	1.350 (2000, ISOE annual) 1..094 outage
Hungary	PAKS1	AEE	11.03.83	PWR	1375					
	PAKS2	AEE	09.21.84	PWR	1375					
	PAKS3	AEE	11.03.86	PWR	1375					
	PAKS4	AEE	11.19.87	PWR	1375					
Japan	Higasidory		05.03.05	BWR	?					
	Shhika	HIT	07.30.93	BWR	1593					
Korea	Kori 3	WEC	10.01.86	PWR	2775					
	Kori-4	WEC	04.29.86	PWR	2775					
	YGN 1	WEC	08.25.86	PWR	2775					
	YGN 2	WEC	06.10.87	PWR	2775					

Country	1. Station name	2. Key data		3. Plant type		8. Occupational dose in operating year		9. Occupational dose in outage		10. Dose received during uprate
Country		2.1 Vendor NSSS	2.2 . Comm. Operation	3.1 Reactor type	3.2. Initial power	8.1 Before the uprate	8.2 After the uprate	9.1 Before the uprate	9.2 After the uprate	
					MWt	manSv	manSv	manSv	manSv	man-Sv
Mexco	LV 1	GE	07.29.90	BWR	1931	6.808 (ISOE 3 year)	2.881 (ISOE 3 year)	0.995 (ISOE 3 year)	0.783 (ISOE 2 year)	6.202 (1999, ISOE annual) 0.690 outage
	LV 2	GE	04.10.95	BWR	1931	3.258 (ISOE 3 year)	2.460 (ISOE 3 year)	0.749 (ISOE 3 year)	0.284 (ISOE 2 year)	1.133 (1999, ISOE annual) 0.601 outage
Spain	Almaraz-1	WEC	09.01.83	PWR	2696	0.575 (ISOE 3 year)	no data	0.430 (ISOE 3 year)	no data	0.454 (2003, ISOE annual) 0.425 outage
	Almaraz-2	WEC	07.01.84	PWR	2696	0.339 (ISOE 3 year)	no data	0.225 (ISOE 3 year)	no data	0.363 (2003, ISOE annual) 0.334 outage
	Asco-1/2000	WEC	12.01.84	PWR	2696	0.705 (ISOE 3 year)	0.490 (ISOE 3 year)	0.574 (ISOE 3 year)	0.410 (ISOE 3 year)	0.595 (2000, ISOE annual) 0.605 ??outage
	Asco-1/2003	WEC	12.01.84	PWR	2696	0.457 (ISOE 3 year)	0.494 (only 2004)	0.431 (ISOE 3 year)	0.448 (only 2004)	0.669 (2003, ISOE annual) 0.543 outage 0.102 manSv modifications (mainly to the new access platforms in loop B) and 0.071 manSv head vessel replacement. replacement of the SVR (Radiological Surveillance System) has been performed.
	Asco-2/1999	WEC	06.01.85	PWR	2696	1.465 (ISOE 3 year)	0.350 (ISOE 3 year)	1.430 (ISOE 3 year)	0.340 (ISOE 3 year)	0.697 (1999, ISOE annual) 0.664 outage
	Asco-2/2004	WEC	06.01.85	PWR	2696	0.354 (ISOE 3 year)	no data	0.309 (ISOE 3 year)		0.716 (2004, ISOE annual) 0.614 outage
	Cofrentes	GE	03.11.85	BWR	2894	0.805 (ISOE 3 year)	3.329 (ISOE 3 year)	no dana	no data	3.424 (1988, ISOE annual)
	Cofrentes	GE	03.11.85	BWR	2894	1.812 (ISOE 3 year)	2.493 (ISOE 3 year)	1.478 (ISOE 3 year)	1.325 (ISOE 3 year)	0.469 (1998, ISOE annual) 0.032 unplanned outage
	Cofrentes	GE	03.11.85	BWR	2894	1.920 (ISOE 3 year)	3.085 (ISOE only year 2003)	1.325 (ISOE 3 year)	2.625 (ISOE only year 2003)	2.154/2.795 (2002)
	Cofrentes	GE	03.11.85	BWR	2894	1.966 (ISOE 3 year)	0.700 (no outage)	1.433(ISOE 3 year)		3.085 (2003, ISOE annual doze) 2.625 outage
	Vandellos-2/1999	WEC	03.08.88	PWR	2775	0.653 (ISOE 3 year)	0.650 (ISOE 3 year)	0.616 (ISOE 3 year)	0.571 (ISOE 3 year)	1.132 (1999, ISOE annual) 1.092 outage
	Vandellos-2/2002	WEC	03.08.88	PWR	2775	0.706 (ISOE 3 year)	0.591 (ISOE only year 2003)	0.647 (ISOE 3 year)	0.514 (ISOE only year 2003)	0.863 (outage)/0.964 (annual) 2002
Sweden	Forsmark-1	ASEA	12.01.80	BWR	2711	0.523 (ISOE 3 year)	0.551 (ISOE 3 year)	no data	no data	0.501 (1986, ISOE annual)
		ASEA	12.01.80	BWR	2711					
	Forsmark-2	ASEA	07.01.81	BWR	2711	0.598 (ISOE 3 year)	0.860 (ISOE 3 year)	no data	no data	0.505 (1986, ISOE annual)
		ASEA	07.01.81	BWR	2711					
	Forsmark-3	ASEA	08.01.85	BWR	3020	1.153 (ISOE 3 year)	0.864 (ISOE 3 year)	0.981 (ISOE 3 year)	0.719 (ISOE 3 year)	0.817 (1989, ISOE annual) 0.678outage
		ASEA	08.01.85	BWR	3020					
	Oskarshamn-2	ASEA	01.01.75	BWR	1700	0.739 (ISOE 3 year)	0.868 (ISOE 3 year)	0.548 (ISOE 3 year)	0.643 (ISOE 3 year)	0.491 (1982, ISOE annual) 0.305 outage
	Oskarshamn-3	ASEA	08.15.85	BWR	3020	0.602 (ISOE 3 year)	0.835 (ISOE 3 year)	0.463 (ISOE 3 year)	0.595 (ISOE 3 year)	0.556 (1989, ISOE annual) 0.454 outage
		ASEA	08.15.85	BWR	3020					
	Ringhals-1	ASEA	01.01.76	BWR	2270	3.009 (ISOE 3 year)	2.853 (ISOE 3 year)	2.275 (ISOE 3 year)	1.798 (ISOE 3 year)	1.667 (1989, ISOE annual) 1.196 outage
		ASEA	01.01.76	BWR	2270					
	Ringhals-2	WEC	05.01.75	PWR	2440	2.327 (ISOE 3 year)	1.267 (ISOE 3 year)	1.822 (ISOE 3 year)	1.074 (ISOE 3 year)	4.099 (1989, ISOE annual) 3.703 outage
	Ringhals-3	WEC	09.01.81	PWR	2783					
		WEC	09.01.81	PWR	2783					
	Ringhals-4	WEC	11.21.83	PWR	2783					

Country	1. Station name	2. Key data		3. Plant type		8. Occupational dose in operating year		9. Occupational dose in outage		10. Dose received during uprate
Country		2.1 Vendor NSSS	2.2 . Comm. Operation	3.1 Reactor type	3.2. Initial power	8.1 Before the uprate	8.2 After the uprate	9.1 Before the uprate	9.2 After the uprate	
					MWt	manSv	manSv	manSv	manSv	man-Sv
Switzerland	Leibstadt/1985	GE	12.15.84	BWR	3012	0.515 (ISOE only year 1984)	2.329 (ISOE 3 year)	no data	no data	1.484 (1985, ISOE annual)
	Leibstadt1998	GE	12.15.84	BWR	3012	1.651 (ISOE 3 year)	1.051 (ISOE 3 year)	1.132 (ISOE 3 year)	0.731 (ISOE 3 year)	1.095 (1998, ISOE annual) 0.725 outage
	Leibstadt.1999	GE	12.15.84	BWR	3012	1.463 (ISOE 3 year)	1.006 (ISOE 3 year)	1.034 (ISOE 3 year)	0.631 (ISOE 3 year)	1.165 (1999, ISOE annual) 0.793 outage
	Leibstadt/2000	GE	12.15.84	BWR	3012	1.197 (ISOE 3 year)	0.998 (ISOE 3 year)	0.790 (ISOE 3 year)	0.620 (ISOE 3 year)	0.979 (2000, ISOE annual) 0.691 outage
	Leibstadt/2003	GE	12.15.84	BWR	3012	1.051 (ISOE 3 year)	0.956 (ISOE only year 2003)	0.731 (ISOE 3 year)	0.657 (ISOE only year 2003)	1.029 (2002, ISOE annual) 0.492 outage
	Mühleberg/1976	BBC	11.06.72	BWR	947	1.838 (ISOE 3 year)	2.792 (ISOE 3 year)	no data	no data	3.475 (1976, ISOE annual)
	Mühleberg/1993	BBC	11.06.72	BWR	947	2.153 (ISOE 3 year)	1.602 (ISOE 3 year)	1.100 (ISOE 3 year)	0.982 (ISOE 3 year)	2.150 (1993, ISOE annual) 1.219 outage
	Mühleberg/1994	BBC	11.06.72	BWR	947					
	Gösgen/1985	KWU	11.01.79	PWR	2808	1.349 (ISOE 3 year)	1.813 (ISOE 3 year)	no data	no data	1.047 (1985, ISOE annual)
	Gösgen/1992	KWU	11.01.79	PWR	2808	1.467 (ISOE 3 year)	1.157 (ISOE 3 year)	no data	0.988 (ISOE 3 year)	0.957 (1992, ISOE annual) 0.776 outage
USA	ANO-2	CE	03.26.80	PWR	2815					year 2002
	Beaver Valley 1/2001	WEC	10.01.76	PWR	2652	0.898 (ISOE 3 year)	1.082 (ISOE 2 year)	0.697 (ISOE 3 year)	0.930 (ISOE 2 year)	1.822 (2001, ISOE annual) 1.513 outage
	Beaver Valley 1/2006	WEC	10.01.76	PWR	2652					
	Beaver Valley 2/2001	WEC	11.17.87	PWR	2652	0.735 (ISOE 3 year)	0.663 (ISOE only year 2002)	0.562 (ISOE 3 year)	0.636 (ISOE only year 2002)	0.022 (2001, ISOE annual)
	Beaver Valley 2/2006	WEC	11.17.87	PWR	2652					
	Braidwood 1	WEC	07.29.88	PWR	3411	1.177 (ISOE 3 year)	0.789 (ISOE 2 year)	1.061 (ISOE 3 year)	0.675 (ISOE 2 year)	0.901 (2001, ISOE annual) 0.797 outage
	Braidwood 2	WEC	10.17.88	PWR	3411	0.773 (ISOE 3 year)	0.886 (ISOE 2 year)	0.650 (ISOE 3 year)	0.794 (ISOE 2 year)	0.105 (2001, ISOE annual)
	Browns Ferry 2/1998	GE	03.01.75	BWR	3294	2.347 (ISOE 3 year)	2.439 (ISOE 3 year)	1.726 (ISOE 3 year)	1.700 (ISOE 3 year)	0.095 (1998, ISOE annual)
	Browns Ferry 2/2007	GE	03.01.75	BWR	3294					
	Browns Ferry 3	GE	03.01.75	BWR	3294	1.480 (ISOE 3 year)	1.011 (ISOE 3 year)	1.052 (ISOE 3 year)	0.343 (ISOE 3 year)	3.159 (1998, ISOE annual) 2.816 outage
	Browns Ferry 3/2007	GE	03.01.75	BWR	3294					
	Browns Ferry 1	GE	01.08.74	BWR	3293					
	Brunswick 1/1996	GE	03.18.77	BWR	2436	4.911 (ISOE 3 year)	1.293 (ISOE 3 year)	3.920 (ISOE only year 1995)	0.676 (ISOE 3 year)	3.178 (1996, ISOE annual) 2.110 outage
	Brunswick 1/2002	GE	03.18.77	BWR	2436	1.318 (ISOE 3 year)	0.768 (ISOE only year 2003)	0.622 (ISOE 3 year)	no data	2.100 (2002, ISOE annual)
	Brunswick 2/1996	GE	11.03.75	BWR	2436	3.604 (ISOE 3 year)	2.460 (ISOE 3 year)	no data	1.572 (ISOE 3 year)	3.985 (1996, ISOE annual) 2.916 outage

Country	1. Station name	2. Key data		3. Plant type		8. Occupational dose in operating year		9. Occupational dose in outage		10. Dose received during uprate
Country		2.1 Vendor NSSS	2.2 . Comm. Operation	3.1 Reactor type	3.2. Initial power	8.1 Before the uprate	8.2 After the uprate	9.1 Before the uprate	9.2 After the uprate	
					MWt	manSv	manSv	manSv	manSv	man-Sv
USA	Brunswick 2/2002	GE	11.03.75	BWR	2436	2.101 (ISOE 3 year)	1.983 (ISOE only year 2003)	1.483 (ISOE 3 year)	1.490 (ISOE only year 2003)	0.656 (2002, ISOE annual)
	Byron 1	WEC	09.16.85	PWR	3411	1.443 (ISOE 3 year)	0.667 (ISOE 2 year)	0.977 (ISOE 3 year)	0.751 (ISOE 2 year)	0.058 (2001, ISOE annual)
	Byron 2	WEC	08.21.87	PWR	3411	0.917 (ISOE 3 year)	0.775 (ISOE 2 year)	0.788 (ISOE 3 year)	0.660 (ISOE 2 year)	0.537 (2001, ISOE annual) 0.479 outage
	Callaway	WEC	04.09.85	PWR	3411	2.180 (ISOE 3 year)	2.487 (ISOE 3 year)	no data	no data	0.270 (1988, ISOE annual)
	Calvert Cliffs 1/1977	CE	05.08.75	PWR	2560	0.620 (ISOE 2 year)	3.303 (ISOE 3 year)	no data	no data	5.470 (1977, ISOE annual)
	Calvert Cliffs 1/2006	CE	05.08.75	PWR	2560					
	Calvert Cliffs 2/1977	CE	04.01.77	PWR	2560	no data	3.303 (ISOE 3 year)	no data	no data	no data (1977)
	Calvert Cliffs 2/2006	CE	04.01.77	PWR	2560					
	Clinton	GE	11.24.87	BWR	2894	1.090 (ISOE 3 year)	0.571 (ISOE only year 2003)	0.591 (ISOE 3 year)	no data	2.081 (2002, ISOE annual) 1.613 outage
	Comanche Peak 1	WEC	08.13.90	PWR	3411	0.882 (ISOE 3 year)	0.303 (ISOE 2 year)	0.749 (ISOE 2 year)	no outages (2 year)	1.105 (2001, ISOE annual) 1.060 outage
	Comanche Peak 2	WEC	08.03.93	PWR	3411	1.013 (ISOE 3 year)	0.704 (ISOE 3 year)	0.732 (ISOE 3 year)	0.551 (ISOE 2 year)	1.086 (1999, ISOE annual) 1.017 outage
	Comanche Peak 2					1.060 (ISOE 3 year)	1.156 (ISOE 2 year)	1.017 (ISOE 2 year)	0.853 (ISOE 2 year)	0.045 (2001, ISOE annual)
	Crystal River 3	BW	03.13.77	PWR	2544	1.995 (ISOE 2 year)	1.266 (ISOE only year 2003)	1.735 (ISOE 2 year)	1.210 (ISOE only year 2003)	0.050 (2002, ISOE annual)
	D.C. Cook 1	WEC	05.07.85	PWR	3250					no data year 2002
	D.C. Cook 2	WEC	03.13.86	PWR	3411					no data year 2003
	Diablo Canyon 1	WEC	05.07.85	PWR	3338	1.772 (ISOE 3 year)	0.055 (ISOE 2 year)	1.674 (ISOE 3year)	no outages (2 year)	0.904 (2000, ISOE annual) no data for outage
	Dresden 2	GE	06.09.70	BWR	2527	2.311 (ISOE 3 year)	1.785 (ISOE 2 year)	1.633 (ISOE 3 year)	1.283 (ISOE 2 year)	3.278 (2001, ISOE annual) 2.668 outage
	Dresden 3	GE	11.16.71	BWR	2527	1.709 (ISOE 3 year)	1.480 (ISOE 2 year)	1.045 (ISOE 3 year)	0.977 (ISOE 2 year)	0.610 (2001, ISOE annual)
	Duane Arnold/1985	GE	02.01.75	BWR	1593	5.176 (ISOE 3 year)	4.893 (ISOE 3 year)	no data	no data	11.120 (1985, ISOE annual)
	Duane Arnold/2001	GE	02.01.75	BWR	1593	1.723 (ISOE 3 year)	0.792 (ISOE 2 year)	1.133 (ISOE 3 year)	0.490 (ISOE 2 year)	0.920 (2001, ISOE annual) 0.880 outage
	Farley 1	WEC	12.01.77	PWR	2652	1.745 (ISOE 3 year)	0.979 (ISOE 3 year)	1.270 (ISOE 3year)	0.273 (ISOE 3 year)	2.306 (1998, ISOE annual) 1.958 outage
	Farley 2	WEC	07.30.81	PWR	2652	1.486 (ISOE 3 year)	1.923 (ISOE 3 year)	0.728 (ISOE 3year)	1.464 (ISOE 3year)	2.013 (1998, ISOE annual) 1.830 outage
	Fermi 2	GE	01.23.88	BWR	3293	1.887 (ISOE 3 year)	0.919 (ISOE 2 year)	no data	no data	2.450 (1992, ISOE annual) no data for outage
	Fitzpatrick	GE	07.28.75	BWR	2436	2.956 (ISOE 3 year)	1.759 (ISOE 2 year)	no data	0.965 (ISOE 3 year)	3.570 (1996, ISOE annual) 2.770 outage
	Fort Calhoun/1980	CE	09.26.73	PWR	1420	2.777 (ISOE 3 year)	3.693 (ISOE 3 year)	no data	no data	6.680 (1980, ISOE annual) no data for outage
	Fort Calhoun/2004	CE	09.26.73	PWR	1420	1.948 (ISOE 2 year)	no data	1.753 (ISOE2 year)	no data	no data (2004)
	Fort Calhoun/2006	CE	09.26.73	PWR	1420					
	Ginna	WEC	07.01.70	PWR	1520					

Country	1. Station name	2. Key data		3. Plant type		8. Occupational dose in operating year		9. Occupational dose in outage		10. Dose received during uprate
Country		2.1 Vendor NSSS	2.2 . Comm. Operation	3.1 Reactor type	3.2. Initial power	8.1 Before the uprate	8.2 After the uprate	9.1 Before the uprate	9.2 After the uprate	
					MWt	manSv	manSv	manSv	manSv	man-Sv
USA	Grand Gulf	GE	07.01.85	BWR	3833	1.421 (ISOE 3 year)	0.0313 (ISOE only year 2003)	1.022 (ISOE 3 year)	no outage in year 2003	1.764 (2002, ISOE annual) 1.525 outage
	H. B. Robinson/1979	WEC	03.07.71	PWR	2200	8.777 (ISOE 3 year)	13.370 (ISOE 3 year)	no data	no data	11.880 (1979, ISOE annual) no data for outage
	H. B. Robinson/2002	WEC	03.07.71	PWR	2200					year 2002
	Hatch 1/1995	GE	01.01.76	BWR	2436	3.478 (ISOE 3 year)	2.964 (ISOE 3 year)	no data	2.358 (ISOE 3 year)	1.817(1995, ISOE annual)
	Hatch 1/1998	GE	01.01.76	BWR	2436	3.393 (ISOE 3 year)	1.344 (ISOE 3 year)	2.358 (ISOE 3 year)	1.330 (ISOE only year 2001)	0.530(1998, ISOE annual)
	Hatch 1/2003	GE	01.01.76	BWR	2436	1.391 (ISOE 3 year)	no dana	1.022 (ISOE 3 year)	no data	0.380 (2003, ISOE annual) 1.525 outage
	Hatch 2/1995	GE	09.05.79	BWR	2436	3.478 (ISOE 3 year)	2.019 (ISOE 3 year)	no data	1.413 (ISOE 3 year)	3.080(1995, ISOE annual) 3080 outage ??
	Hatch 2/1998	GE	09.05.79	BWR	2436	2.123 (ISOE 3 year)	1.882(ISOE 3 year)	2.358 (ISOE 3 year)	1.615 (ISOE only year 2001)	2.770(1998, ISOE annual) 2.240 outage
	Hatch 2/2003	GE	09.05.79	BWR	2436	1.453 (ISOE 3 year)	no data	1.285 (ISOE 3 year)	no data	1.306 (2003, ISOE annual) 0.924 outage
	Hope Creek	GE	12.20.86	BWR	3293	1.639 (ISOE 3 year)	0.581 (ISOE 2 year)	1.219 (ISOE 3 year)	0.369 (ISOE 2 year)	1.562 (2001, ISOE annual) 0.935 outage
	Indian Point 2/2003	WEC	08.01.74	PWR	3071					no data year 2003
	Indian Point 2/2004	WEC	08.01.74	PWR	3071					
	Indian Point 3/2002	WEC	08.30.76	PWR	3025					no data year 2002
	Indian Point 3/2005	WEC	08.30.76	PWR	3025					
	Kewanee /2003	WEC	06.16.74	PWR	1650					no data year 2003
	Kewanee /2004	WEC	06.16.74	PWR	1650					
	LaSalle 1	GE	01.01.84	BWR	3323	1.933 (ISOE 3 year)	1.998 (ISOE 2 year)	0.693 (ISOE 3 year)	0.369 (ISOE 2 year)	0.585 (2000, ISOE annual) no outage
	LaSalle 2	GE	01.01.85	BWR	3323	2.037 (ISOE 3 year)	0.659 (ISOE 2 year)	no planned outages only forced the d collective dose	no planned outages only forced the d collective dose	2.018 (2000, ISOE annual) 1.433 outage
	Limerick 1	GE	02.01.86	BWR	3293	1.479 (ISOE 3 year)	1.338 (ISOE 3 year)	no outage in year 1995	0.757 (ISOE 3 year)	1.805 (1996, ISOE annual) 1.380 outage
	Limerick 2	GE	01.08.90	BWR	3293	1.350 (ISOE 3 year)	1.040 (ISOE 3 year)	0.752 (ISOE 3 year)	0.491 (ISOE 3 year)	2.041 (1995, ISOE annual) 1.483 outage
	Millstone 2	CE	12.26.75	PWR	2560	6.770 (ISOE 3 year)	8.600 (ISOE 3 year)	no data	no dana	4.270 (1979, ISOE annual) no data for outage
	Monticello	GE	06.30.71	BWR	1670	1.275 (ISOE 3 year)	1.690 (ISOE 3 year)	1.039 (ISOE 3 year)	1.159 (ISOE 3 year)	2.086 (1998, ISOE annual) 1.624 outage
	Nine Mile Point 2	GE	03.11.88	BWR	3323	3.293 (ISOE 3 year)	2.107 (ISOE 3 year)	no data	1.581 (ISOE 3 year)	4.020 (1995, ISOE annual) 3.250 outage
	North Anna 1	WEC	06.06.78	PWR	2775	5.748 (ISOE 3 year)	5.173 (ISOE 3 year)	no data	no data	3.610 (1986, ISOE annual) no data for outage
	North Anna 2	WEC	12.14.80	PWR	2775	6.770 (ISOE 3 year)	8.600 (ISOE 3 year)	no data	no data	4.270 (1986, ISOE annual) no data for outage
	Palisades	CE	12.31.71	PWR	2530	no data for year 2004	no data for year 2004	no data for year 2004	no data for year 2004	no data for year 2004
	Palo Verde 1/1996	CE	02.13.86	PWR	3800	1.718 (ISOE 3 year)	0.635 (ISOE 3 year)	1.502 (ISOE only year 1995)	0.510 (ISOE 3 year)	1.389 (1996, ISOE annual) 1.066 outage
	Palo Verde 1/2005	CE	02.13.86	PWR	3800					
	Palo Verde 2/1996	CE	09.19.86	PWR	3800	1.656 (ISOE 3 year)	0.543 (ISOE 3 year)	1.309 (ISOE only year 1995)	0.443 (ISOE 3 year)	1.589 (1996, ISOE annual) 1.463 outage
	Palo Verde 2/2003	CE	09.19.86	PWR			no data		no data	no data
	Palo Verde 3/1996	CE	01.15.88	PWR	3800	1.814 (ISOE 3 year)	0.806 (ISOE 3 year)	1.720 (ISOE only year 1995)	0.682 (ISOE 3 year)	1.389 (1996, ISOE annual) no outage
	Palo Verde 3/2005	CE	01.15.88	PWR	3800					
	Peach Bottom 2/1994	GE	07.05.74	BWR	3293	3,313 (ISOE 3 year)	1.336 (ISOE 3 year)	no data	0.421 (ISOE 3 year)	2.890 (1994 ISOE annual) no data outage
	Peach Bottom 2/2002	GE	07.05.74	BWR	3293	1.321 (ISOE 3 year)	no data	0.433 (ISOE 3 year)	no data	2.719 (2002 ISOE annual) 2.107 outage
	Peach Bottom 3/1995	GE	12.23.74	BWR	3293	2.720 (ISOE 3 year)	1.860 (ISOE 3 year)	no data	1.034 (ISOE 3 year)	2.995 (1995 ISOE annual) 2.010
	Peach Bottom 3/2002	GE	12.23.74	BWR	3293	1.321 (ISOE 3 year)	no data	no planned outages	no data	0.612 (2002 ISOE annual) no outage
	Perry	GE	11.18.87	BWR	3579	2.133 (ISOE 3 year)	1.639 (ISOE 2 year)	1.656 (ISOE 3 year)	0.970 (ISOE 2 year)	0.558 (2000 ISOE annual) no outage
	Pilgrim	GE	12.09.72	BWR	1998	3.325 (ISOE 3 year)	1.087 (ISOE 2 year)	2.489 (ISOE 3 year)	0.620 (ISOE 2 year)	0.506 (2000 ISOE annual) no outage
	Point Beach 1	WEC	12.21.70	PWR	1519	0.896 (ISOE 3 year)	no data	0.800 (ISOE 2 year)	no data	1.300 (2002, ISOE annual) 0.970 outage
	Point Beach 2	WEC	09.30.72	PWR	1519	0.648 (ISOE 3 year)	no data	0.452 (ISOE 2 year)	no data	0.607 (2002, ISOE annual) 0.507 outage

Country	**1. Station name**	**2. Key data**		**3. Plant type**		8. Occupational dose in operating year		9. Occupational dose in outage		10. Dose received during uprate
Country		2.1 Vendor NSSS	2.2 . Comm. Operation	**3.1 Reactor type**	**3.2. Initial power**	8.1 Before the uprate	8.2 After the uprate	9.1 Before the uprate	9.2 After the uprate	
					MWt	manSv	manSv	manSv	manSv	man-Sv
USA	Quad Cities 1	GE	02.18.73	BWR	2511	4.779 (ISOE 3 year)	11.263 (ISOE only year 2002)	3.463 (ISOE 3 year)	8.587 (ISOE only year 2002)	0,708 (2001 ISOE annual) no outage
	Quad Cities 2	GE	03.10.73	BWR	2511	1.296 (ISOE 3 year)	6.613 (ISOE only year 2002)	0.479 (ISOE 3 year)	4.860 (ISOE only year 2002)	0.663 (2001 ISOE annual) no outage
	River Bend/2000	GE	06.16.86	BWR	2894	2.249 (ISOE 3 year)	1.214 (ISOE 2 year)	1.387 (ISOE 3 year)	0.756 (ISOE 2 year)	2.016 (2000 ISOE annual) 1.329 outage
	River Bend/2003	GE	06.16.86	BWR	2894	no data	no data	no data	no data	no all data for 2003
	Salem 1/1986	WEC	06.30.77	PWR	3344	2.443 (ISOE 3 year)	2.402 (ISOE 3 year)	no data	no data	2.995 (1986, ISOE annual) no data for outage
	Salem 1/2001	WEC	06.30.77	PWR	3344	1.162 (ISOE 3 year)	1.438 (ISOE only year 2002)	0.640 (ISOE 3 year)	1.300 (ISOE only year 2002)	1.463 (2001, ISOE annual) 1.253 outage
	Salem 2	WEC	10.13.81	PWR	3411	0.892 (ISOE 3 year)	1.489 (ISOE only year 2002)	0367 (ISOE 3 year)	1.350 (ISOE only year 2002)	0.068 (2001, ISOE annual) no outage
	San Onofre 2	CE	08.08.83	PWR	3390	0.650 (ISOE 3 year)	0.612 (ISOE only year 2002)	no data	no outage in year 2002	0.578 (2001, ISOE annual) no outage
	San Onofre 3	CE	04.01.84	PWR	3390	1.067 (ISOE 3 year)	1.264 (ISOE only year 2002)	1.019 (ISOE 3 year)	1.164 (ISOE only year 2002)	0.069 (2001, ISOE annual) no outage
	Seabrook1/2005	WEC	08.19.90	PWR	3411					
	Seabrook1/2006	WEC	08.19.90	PWR	3411					
	Sequoyah 1	WEC	07.01.81	PWR	3411	no all data year 2002	no all data year 2002	no all data year 2002	no all data year 2002	no all data for year 2002
	Sequoyah 2	WEC	06.01.82	PWR	3411	no all data for year 2002	no all data for year 2002	no all data for year 2002	no all data for year 2002	no all data for year 2002
	Shearon Harris	WEC	05.02.87	PWR	2775	0.745 (ISOE 2year)	0.067 (ISOE only year 2002)	0.588 (ISOE 3 year)	no outage in year 2001)	2.522 (2001, ISOE annual) 2.412 outage
	South Texas 1	WEC	08.25.88	PWR	3800	no all data for year 2002	no all data for year 2002	no all data for year 2002	no all data for year 2002	no all data for year 2002
	South Texas 2	WEC	06.18.89	PWR	3800	no all data for year 2002	no all data for year 2002	no all data for year 2002	no all data for year 2002	no all data for year 2002
	St. Lucie 1	CE	12.21.76	PWR	2560	4.356 (ISOE 3 year)	7.025 (ISOE 3 year)	no data	no data	9.290 (1981, ISOE annual) no data for outage
	St. Lucie 2	CE	08.08.83	PWR	2560	6.315 (ISOE only year 1984)	3.422 (ISOE 3 year)	no data	no data	6.720 (1985, ISOE annual) no data for outage
	Surry 1	WEC	12.02.72	PWR	2441	2.173 (ISOE 3 year)	1.461 (ISOE 3 year)	no data	0.864 (ISOE 3 year)	2.227 (1995, ISOE annual) no outage
	Surry 2	WEC	05.01.73	PWR	2441	2.239 (ISOE 3 year)	1.027 (ISOE 3 year)	no data	0.778 (ISOE 3 year)	1.836 (1995, ISOE annual) 1.577 outage
	Susquehanna 1/1995	GE	06.08.83	BWR	3293	2.501 (ISOE 3 year)	1.401 (ISOE 3 year)	no data	0.578 (ISOE 3 year)	2.665 (1995 ISOE annual) 1.830 outage
	Susquehanna 1/2001	GE	06.08.83	BWR	3293	1.431 (ISOE 3 year)	2.059 (ISOE only year 2002)	0.575 (ISOE 3 year)	1.520 (ISOE only year 2002)	0.500 (2001 ISOE annual) no outage
	Susquehanna 2/1994	GE	02.12.85	BWR	3293	2.610 (ISOE 3 year)	1.860 (ISOE 3 year)	no data	0.578 (ISOE 3 year)	2.210 (1994 ISOE annual) no data for outage
	Susquehanna 2/12001	GE	02.12.85	BWR	3293	1.269 (ISOE 3 year)	0.540 (ISOE only year 2002)	0.467 (ISOE 3 year)	no outage in year 2002	2.390 (2001 ISOE annual) 1.890 outage
	TMI-1	BW	09.02.74	PWR	2535	2.637 (ISOE 3 year)	1.720 (ISOE 3 year)	no data	no data	2.100 (1988, ISOE annual) no data for outage
	Turkey Point 3	WEC	12.04.72	PWR	2200	1.738 (ISOE 3 year)	1.037 (ISOE 3 year)	no data	0.927 (ISOE 3 year)	0.137 (1996, ISOE annual) no outage
	Turkey Point 4	WEC	09.07.73	PWR	2200	1.738 (ISOE 3 year)	1.257 (ISOE 3 year)	no data	1.118 (ISOE 3 year)	1.733 (1996, ISOE annual) 1.585 outage
	Vermont Yankee	GE	11.29.72	BWR	1593					
	V. C. Summer	WEC	12.02.72	PWR	2775	2.110 (ISOE 3 year)	1.069 (ISOE 3 year)	no outage in year 1995)	0.954 (ISOE 3 year)	1.060 (1996, ISOE annual) 0.890 outage
	Vogtle 1	WEC	05.31.87	PWR	3411	2.090 (ISOE 3 year)	1.236 (ISOE 3 year)	no data	1.095 (ISOE years 95 and 96)	1.833 (1993, ISOE annual) no data for outage
	Vogtle 2	WEC	05.19.89	PWR	3411	2.090 (ISOE 3 year)	1.657 (ISOE 3 year) 1.947 (ISOE years 95 and 96)	no data	1.800 (ISOE years 95 and 96)	1.833 (1993, ISOE annual) no data for outage
	Waterford 3/2002	CE	09.24.85	PWR	3390	no all data for year 2002	no all data for year 2002	no all data for year 2002	no all data for year 2002	no data year 2002
	Waterford 3/2005	CE	09.24.85	PWR	3390					no data year 2005
	Watts Bar	WEC	05.27.96	PWR	3411	0.743 (ISOE 3 year)	0.936 (ISOE only year 2002)	0.471 (ISOE 2 year)	0.883 (ISOE only year 2002)	0.059 (2001 ISOE annual) no outage
	WNP-2			BWR	3323	6.493 (ISOE 3 year)	3.013 (ISOE 3 year)	4.690 (ISOE only year 1992)	2.539 (ISOE 3 year)	4.273 (1995 ISOE annual) 2.973 outage

Country	1. Station name	2. Key data		3. Plant type		8. Occupational dose in operating year		9. Occupational dose in outage		10. Dose received during uprate
Country		2.1 Vendor NSSS	2.2 . Comm. Operation	3.1 Reactor type	3.2. Initial power	8.1 Before the uprate	8.2 After the uprate	9.1 Before the uprate	9.2 After the uprate	
					MWt	manSv	manSv	manSv	manSv	man-Sv
USA	Wolf Creek	WEC	09.03.85	PWR	3411	2.013 (ISOE 3 year)	1.555 (ISOE 3 year)	no data	no data	1.680 (1993, ISOE annual) no data for outage
Slovenia	Krsko	WEC	01.01.83	PWR	1882	1.267 (ISOE 3 year)	0.837 (ISOE 3 year)	1.060 (ISOE 3 year)	0.760 (ISOE 3 year)	2.598 (2000, ISOE annual) 2.423 outage

*
S- Stretch power (uprates typically up to 7 %, within the design capacity plant). Typically involve changes to instrumen
MU-Measurement uncertainty (uprates are less than 2 %e achieved by implementing enhanced techniques for calcula
E-Extended power (uprates greater than stretch. Require significant modifications to major BOP equipment)
**
Second thermal uprate
***
Third thermal uprate

PWR
BWR

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SSI sätter gränser för stråldoser till allmänheten och för dem som arbetar med strålning, utfärdar föreskrifter och kontrollerar att de efterlevs. SSI håller beredskap dygnet runt mot olyckor med strålning. Myndigheten informerar, utbildar och utfärdar råd och rekommendationer samt stöder och utvärderar forskning. SSI bedriver även internationellt utvecklingssamarbete.

Myndigheten, som sorterar under Miljödepartementet, har 110 anställda och är belägen i Solna.

THE SWEDISH RADIATION PROTECTION AUTHORITY (SSI) is a central regulatory authority charged with promoting effective radiation protection for people and the environment today and in the future.

SSI sets limits on radiation doses to the public and to those that work with radiation. SSI has staff on standby round the clock to respond to radiation accidents. Other roles include information, education, issuing advice and recommendations, and funding and evaluating research.

SSI is also involved in international development cooperation. SSI, with 110 employees located at Solna near Stockholm, reports to the Ministry of Environment.

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