Consequences of Power Uprating

Authors

Klas Lundgren
ALARA Engineering, Skultuna, Sweden

Rolf Riess
NPC, Neunkirchen, Germany

A.N.T. INTERNATIONAL®

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Advanced Nuclear Technology International
Krongjutarvägen 2C, SE-730 50 Skultuna
Sweden

info@antinternational.com
www.antinternational.com

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Acronyms and Explanations

AB	Auxiliary Building
ACEC	Alstom ACEC Energie
ALARA	As Low As Reasonably Achievable
AO	Axial Offset
AOA	Axial Offset Anomaly
ASEA	Allmänna Svenska Elektriska Aktiebolaget (General Swedish Electrical Limited Company)
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	Cofrentes nuclear power plant
CPSES	Comanche Peak Steam Electric Station
CRD	Control Rod Drive
CRUD	Chalk River Unidentified Deposits
CVCS	Chemical and Volume Control System
DB	Deep Bed
DG	Diesel Generator
DH	Dissolved Hydrogen
DO	Dissolved Oxygen
dP	Delta Pressure
DR	Dose Rate
DZO	Depleted Zinc Oxide
E	Extended power
EPU	Extended Power Up-rate
EBA	Enriched Boron Acid
EC	European Commission
ECP	Electrochemical (or corrosion) Potential
EOC	End Of Cycle
EPRI	Electric Power Research Institute (www.epri.com)
ERV	Electrometric Relief Valves
°F	Degrees Fahrenheit
F + DB	Filter + Deep Bed
FANP	Framatome ANP
FAC	Flow Accelerated Corrosion
FD	Filter Demineralizer
f_{FW}	Feedwater flow rate [kg/s]
FPHD	Forward Pumped Heater Drains
FRAM	Framatome
FRAMACECO	Framatome-ACEC-CO
FW	Feedwater
f_{RWCU}	Reactor water cleanup flow [kg/s]
GBq	Giga Becquerel
GE	General Electric
Gpm	Gallons per minute
HDCI	High Duty Core Index
HFT	Hot Functional Testing
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
I&C	Instrumentation and Control
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
LOCA	Loss of Coolant Accident
LP	Low Pressure
M5	Advanced Zircaloy quality developed by Areva
MP	Measuring Position for dose rate
MSL	Main Steam Line
MSLR	Main Steam Line Radiation
MOX	Mixed Oxide fuel
MU	Measurument Uncertainty
MUR	Measurument Uncertainty Recapture
MWe	MW electrical power
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
OEM	Original Equipment Manufacture
OL	Olkiluoto
OLNC	On Line Noble Chem
PCI	Pellet Cladding Interaction
PCMI	Pellet Cladding Mechanical Interaction
PCS	Power Conversion System
pH_{300}	pH at 300 °
PI	Performance Indicators
PLR	Primary Loop Recirculation (i.e. recirculation lines)
PRIS	Power Reactor Information System (IAEA)
PS	Pressure Suppression
PWR	Pressurized Water Reactor
R2	Ringhals 2
R3	Ringhals 3
R4	Ringhals 4
RB	Reactor Building
RCS	Reactor coolant system
REIRS	Radiation Exposure Information and Reporting System (NRC)
RFO	Refuelling Outage
RHR	Residual Heat Removal
Rpm	Rounds per minute
RPV	Reactor Pressure Vessel
RTD	Resistance Temperature Detector
RW	Reactor Water
RWCU	Reactor Water Clean-Up
Rx	Reactor
S	Stretch power

SCC	Stress Corrosion Cracking
SG	Steam Generator
SGR	Steam Generator Replacement
SHE	Standard Hydrogen Electrode
SNB	Sub-cooled Nucleate Boiling
SPU	Stretch Power Up-rate
SS	Stainless Steel
SSI	Swedish Radiation Protection Authority
STP	Standard Temperature and Pressure
STR	Special Topic Report
T_{Ave}	Average temperature
$T_{½}$	Half-life for radionuclide
TMI	Three Mile Island
TVO	Teollisuuden Voima Oy
VCT	Volume Control Tank
WANO	World Association of Nuclear Operators (www.wano.org.uk)
WEC	Westinghouse Electric Company
ZIRLO	Advanced Zircaloy quality developed by Westinghouse

Unit Conversion

TEMPERATURE

°C + 273.15 = K

°C*1.8 +32 = °F

T(K)	T(°C)	T(°F)
273	**0**	32
289	16	61
298	25	77
373	**100**	212
473	**200**	392
573	**300**	572
633	360	680
673	**400**	752
773	**500**	932
783	510	950
793	520	968
823	550	1022
833	560	1040
873	**600**	1112
878	605	1121
893	620	1148
923	650	1202
973	**700**	1292
1023	750	1382
1053	780	1436
1073	**800**	1472
1136	863	1585
1143	870	1598
1173	**900**	1652
1273	**1000**	1832
1343	1070	1958
1478	1204	**2200**

MASS

kg	lbs
0.454	**1**
1	2.20

Radioactivity

1 Sv = 100 Rem

1 Ci = 3.7 x 1010 Bq = 37 GBq

1 Bq = 1 s^{-1}

DISTANCE

x (μm)	x (mils)
0.6	0.02
1	0.04
5	0.20
10	0.39
20	0.79
25	0.98
25.4	**1.00**
100	3.94

PRESSURE

bar	MPa	psi
1	0.1	14
10	**1**	142
70	7	995
70.4	7.04	**1000**
100	10	1421
130	13	1847
155	15.5	2203
704	70.4	**10000**
1000	100	14211

STRESS INTENSITY FACTOR

MPa√m	ksi√inch
0.91	**1**
1	1.10

Contents

1 Introduction

Many nuclear power plants worldwide are planning power up-rates within the next few years. Several others have done it already. The permission to increase power is given by the responsible Government. The Swedish Radiation Protection Authority (*SSI)*, is one of the bodies to which the application for power up-rates 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 up-rates 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 to review the applications for power up-rates and in the assessment of the after-effects of these up-rates. This study by Zabric et al, 2007, is the main source of information for the compilation of this Special Topic Report (*STR*).

Available information shows that a relatively high percentage of all operating Nuclear Power Plants (*NPPs*) in the world have implemented, or are considering, some form of power increase (power up-rate). Such up-rates vary significantly. Small up-rates 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. Conversely, large up-rates, sometimes 20-30% of nominal power, may be undertaken which require substantial changes on the reactor side, including fuel, operating regime and limits, etc.

The majorities of power up-rates 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 up-rates require either major or minor modification to operating practices and conditions .Such modifications have to be evaluated especially under safety aspects. However, chemistry items are not involved in this kind of work, because they are availability issues and topics, which will be discussed throughout this *STR*. As an example, the occupational exposure 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 World association of Nuclear Operators (*WANO)* indicators and other sources of information. However, these had not been systematically analysed to determine which specific features of the up-rates were influencing the exposures.

2 Project description and objective

One objective of this *STR* is to identify the number of documents necessary to obtain the approval of the respective government or authority for performing a power up-rate. These documents shall be checked whether there is a connection to chemistry issues and whether these issues have to be adjusted in order to support the power up-rating.

Furthermore, the aim of the evaluation – as a typical example for the interaction of safety related and availability related issues - was to investigate what specific conditions and practices affect the occupational exposures received when reactor power is up-rated. Identification of these factors on a worldwide basis should then allow power up-rates to be planned in way that provides better exposure optimisation.

2.1 Project overview

The evaluation was divided into four tasks:

1) A compilation of power up-rates of light-water reactors worldwide. The compilation contains a technical description in brief of how the power up-rates were carried out.
2) The main emphasis of the evaluation 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 up-rate.
3) Experience from the reconstruction period with bearing on the radiation protection of workers.
4) A compilation of a typical set of documents which are necessary to get an approval for power up-rating.

This report is a compilation of all four tasks. Each task has its own chapter and for task 3 the analysis of the selected plants, are shown in three different subchapters. Task 4 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

The first task is to collect information on the implemented and planned up-rates on Pressurized Water Reactor (*PWR)* and Boiling Water Reactor (*BWR)* reactors internationally. In the second task the information was catalogued in accordance with criteria focusing on radiological impact. In task 3 a detailed analysis of plants selected for up-rates, was chosen according to established criteria, in line with the project requirements. For *BWR* two plants were picked out, one with 12% power increase and another with 25%. For the *PWR*s two up-rates in the range of 10% power increase were selected. The aim was a detailed analysis of causal relations between up-rates and the radiological content. The final task started with the safety issues typically requested by the authorities before approving the power up-rate. Thus the project was organized into four tasks.

Data collection for the detailed analysis was carried out through personal contacts. Data was specified which were 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 *PWR*s detailed radiological data was received but less technical and chemistry oriented. The *PWR* report is therefore not the detailed analysis that was hoped to be made, though a detailed analysis on radiological data has been performed.

2.3 Participants

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

- Tea Bilic Zabric and Bojan Tomic from Enconet Consulting, Vienna, Austria
- Klas Lundgren from *ALARA* Engineering, Skultuna, Sweden
- Mats Sjöberg ES-konsult, Solna, Sweden

The amendments to their report in order to become this *STR* were accomplished by:

- Klas Lundgren from *ALARA* Engineering, Skultuna, Sweden
- Rolf Riess from NPC, Neunkirchen, Germany

3 Compilation of power up-rates

3.1 Introduction

Quite a few *NPP*s have implemented, or are considering increasing the power level on which they operate (power up-rate). Those power up-rates vary significantly, from small ones of a few percentages to large up-rates above 30% of the nominal power. The alterations to the plants particularly those with larger power up-rates, require changes and modifications to the plant, in both hardware and operating arrangements (procedures, operating regime, and operating window). Apart from different requirements (on *PCS*) fuel remains an important (and also a limiting) issue during an up-rate. In addition to the safety impact of up-rates that is normally verified in depth before a licensee for an up-rate is issued the radiological aspects, the consequences of an up-rate are also of interest. Changes in operating regimes, but also changes and operation with new hardware might have an impact on the occupational exposures and their distribution. Increased power level may also have certain impact on the effluents, especially on tritium in *PWR*s.

The first task was to compile a database using worldwide sources. This database is 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 the second task relates to the worldwide up-rates performed (or planned) and which shows the radiological consequences and compares the data before and after the up-rates. The data were collected from 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 Information System on Occupational Exposure *(ISOE)* Program 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). OECD, Occupational exposures 1986-1996; 1998; 1999; 2001; 2002; 2003 and 2004.

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 International Atomic Energy Agency *(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 were used for reviewing information and proceedings from conferences.

NRC

The Nuclear Regulatory Commission *(NRC)* 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. '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, was used. The data are available from year 1998. The Organisation for Economic Cooperation and Development *(OECD)* developed the document 'Thermal Power Up-rating in Europe' which reflects the cooperation of many experts in Europe. In addition to the up-rating data, also included was some 'Plant data' and general information available from relevant countries, mostly based on the "*IAEA PRIS*" data. NUREG-0713, 2005.

EC

The European Commission *(EC)* 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, *EC*, Radioactive effluents 1995-1999 and 1999-2003.

OEMs and utilities

Based on personal contacts to *OEM*'s and utilities, a complete and detailed overview on all necessary documents for the approval of up-rating could be obtained. The information collected in this task is plant specific and represents the experience of two German *PWR*s.

Comments on the comparison of data across the sources of data

There is no systematic collection of data covering up-rates and related activities, nor is there any specific collection of radiological exposure information. Therefore, the information of relevance for radiological impact of up-rates 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 up-rates is not possible. The main source of data that was used for radiological releases covers the years after 1995. Therefore, it was not possible to assess the effects of earlier up-rates. 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.

Table 3-1 below compares *ISOE* and *WANO* data for the average recorded occupational exposures during outages (for a period before and after an up-rate) with the recorded occupational exposure for the outage during which an up-rate 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 3-1, 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.

Table 3-1: An example of comparison of dose data between ISOE and WANO

Examples	Occupational dose in operating year				Dose during up-rate (in the year of implementation)	
	Before the up-rate (manSv)		After the up-rate (manSv)		manSv	
	ISOE data	*WANO* data	*ISOE* data	*WANO* data	*ISOE* data	*WANO* data
Plant 1	1.202	1.14	0.879	0.83	3.30	1.49
Unit 1 / Plant 2	1.282	0.99	0.677	0.78	3.22	2.03
Unit 2 / Plant 2	1.432	0.99	0.986	0.78	1.41	2.03
Plant 3	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 up-rate activities. The printout of the data base containing all information collected is provided in Appendix A.

Table 3-2 describes the fields that are included in the data base and discusses the contents of each of those.

Table 3-2: Content of different fields in database, see Appendix 1

Field #	Title	Description/Comments	Main Reference
0	Country	Country where plant is located	
1	Plant name	Plant name with unit indication (if more up-rates were performed, year of the up-rate follow the unit designator, i.e. Tihange 2/2001)	*WANO*PI*
2.1	Vendor of nuclear steam supply system (*NSSS)*	Vendor (*GE, *WEC*, FRAM, etc.)	*WANO PI*
2.2	Commercial date of operation	Month, Day, Year	*WANO PI*
3.1	Reactor type	(*PWR, BWR*)	*OECD/NE*
3.2	Initial power	Original thermal power in *MWt*	*OECD/NE*
4.1	Thermal power	Up-rated thermal power in *MWt*	*OECD, IAEA, NRC*
4.2	Year implemented	Year when the up-rate was implemented (in some cases approved, data sometimes inconsistent)	*OECD, NRC*
4.3	Up-rate type and total power increase in the percentage for particular year	-*MU* – Measurement uncertainty (up-rates are less than 2%) - *S* – Stretch power (up-rates typically up to 7) - E – Extended power (up-rates greater than the stretch) Example: 1985 *S* 4.1 % 2001 *E* 19.4 % The first up-rate 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 up-rate. Example: Whether the up-rate was implemented by increase by **Rx* TAvg (with the same *FW* mass flow) OR T Avg remains same (but FW/*MSL* muss flow and pressures were increased) Data entered where available.	*OECD, NRC*
4.5	Equipment	Where available, list of main equipment modified / affected	*OECD, NRC*
5.1	Fuel cycle	Length of the fuel cycle – in months	*OECD*, NE
5.2	Average linear fuel rating (before up-rate)	Fuel rating in kW/m	*OECD*, NE
5.3	Average linear fuel rating (after up-rate)	Change in fuel rating in kW/m	*OECD*, NE
5.4	Fuel type	Type of fuel used. Sometimes different types used, depends on core design. Equilibrium cycle fuel entered (when available)	*OECD*, NE
6.1	Annual liquid effluents (before up-rate)	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 up-rate)	As above.	EC
7.1	Annual gaseous effluents (before up-rate)	Total Airborne releases in *GBq*	EC
7.2	Annual gaseous effluents (after up-rate)	As above	EC
8.1	Annual occupational dose (before up-rate)	Average value of the total collective dose over the three year period before the up-rate in manSv (when 3 years not available it is noted in the table)	*ISOE, NRC*
8.2	Annual occupational dose (after up-rate)	Average value of the total collective dose over the three year period after the up-rate in manSv (when 3 years not available it is noted in the table)	*ISOE, NRC*
9.1	Occupational dose during outage (before up-rate)	Average value of the collective outage dose over the three-years period before the up-rate – in manSv (when three years not available, it is noted in the table).	*ISOE*
9.2	Occupational dose during outage (after up-rate	Average value of the collective outage dose over the three-years period after the up-rate – in manSv (when three years not available, it is noted in the table).	*ISOE*
10	Occupational dose during up-rate	Doses received during the up-rate (if equipment was changed) – in manSv. Data not systematically available. In some cases data cover annual occupational dose for the year when the up-rate was implemented. Of no relevance for *MU*. Of limited relevance for S, except when major plant modifications are implemented. Sometimes involve *SGR*, if performed in parallel.	*ISOE*

* General Electric (*GE*); Westinghouse electric Company (*WEC*); Framatome (*FRAM*).
* Performance Indicators (*PI*); Reactor (Rx); Steam Generator Replacement (*SGR*)

Classification of up-rates

There are considerable economic benefits to up-rate 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 up-rates highly attractive to the utilities. Nevertheless, the complexity and significance of the safety and operational issues associated with up rates make additional gains anything but easy. Comprehensive safety analysis and, depending on the country, re-licensing by the regulator are important elements of every up-rate project. Changes in operating practices and methods of maintenance organisation following up rates lead to radiological consequences both in normal operation and during outages.

The aim of every up-rate is to increase the electrical power output available from the main generator. This can be achieved by modifying the *PCS* (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 Reactor coolant system (*RCS*) 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 Up-rate: up-rates of 1 to 2 percent power, typically achieved using more precise techniques for measuring Feed-water flow and/or performing analysis to reduce unnecessary conservatism.

- Stretch Power (*S*) Up-rate: up-rates 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*) Up-rate: up-rates of up to 30 percent power, achieved by major changes of core design and significant modifications to major plant equipment The majority of power up-rates 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 *PCS*. Some plants have performed different types of up-rate 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 up-rated thermal power. Of these, 24% were small up-rates of up to 2% increase, 49% were stretch and 27% were large power up-rates.

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

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

Brokdorf (*PWR*) 3.9% up-rate accomplished.
Emsland (*PWR*) is planning 4.9% up-rate (date not known).
Grafenrheinfeld (*PWR*) 4.9% up-rate accomplished.
Grohnde (*PWR*) is planning 4.5% up-rate (date not known).
Gundremmingen B and C (*BWR*) are planning 6.8% up-rate (date not known).
Isar-1 (*BWR*) is planning 7% up-rate (date not known).
PAKS 1, 2, 4 (*PWR*) are planning 9.1% up-rate in 2006.
PAKS 3 (*PWR*) is planning 9.1% up-rate in 2007.
Kori 3 & 4 (*PWR*) are planning 5% up-rate in 2006.
YGN 1 & 2 (*PWR*) are planning 5% up-rate in 2006.
Higasidory (*BWR*) is planning up-rate (% and date not known).
Shika (*BWR*) is planning up-rate (% and date not known).
Forsmark-1 (*BWR*) is planning 19.9% up-rate in 2010.
Forsmark-2 (*BWR*) is planning 19.9% up-rate in 2009.
Forsmark-3 (*BWR*) is planning 25.0% up-rate in 2011.
Oskarshamn 2 (*BWR*) is planning a 35.3% up-rate in 2011.
Oskarshamn-3 (*BWR*) is planning 29.1% up-rate in 2008.
Ringhals-1 (*BWR*) is planning 11.9% power up-rate (date not known).
Ringhals-3 (*PWR*) is planning 7.8% up-rate in year 2007 and 14.3% up-rate in 2009.
Ringhals -4 (*PWR*) is planning 18.6% up-rate in 2011.

The following table (Table 3-3) describing intended future power up-rates in the USA is based on information obtained from a survey of all licensees conducted in March 2006.

Table 3-3: Intended future power up-rates in the USA

Fiscal Year	Power Up-rates 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

*Measurument Uncertainty Recapture (*MUR*); *Extended Power Up-rate (*EPU*); *MW thermal power (*MWt*); Stretch Power Up-rate (*SPU*)

3.4 Conclusions

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

Relevance of yearly occupational doses

The data table provided the annual occupational doses for (usually) 3 years average before and after the up-rate 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 up-rate, 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 up-rate with any of the annual occupational 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 up-rate. 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 up-rate, then the comparison of the annual occupational dose before and after the up-rate is (almost) meaningless.

Recognising of the impact of up-rates

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 *NPP*s 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 International Commission on Radiological Protection *(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 up-rates and the occupational doses could be established.
- The occupational doses on some plants seem to be higher after the up-rate, 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 the removing of external effects.

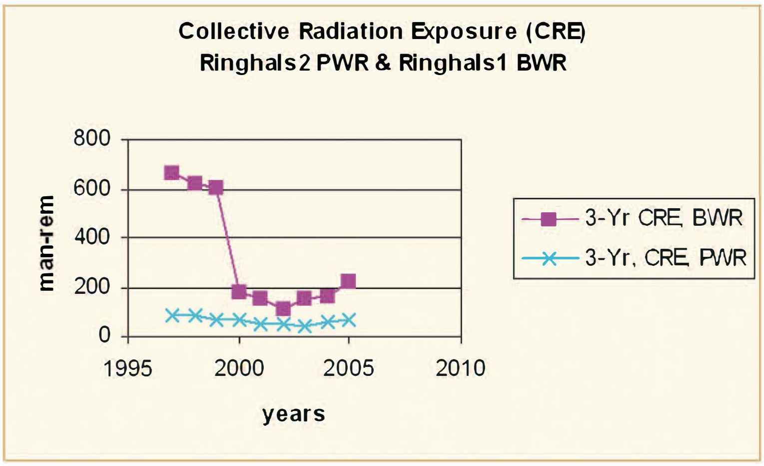

Figure 3-1: *PWR* to *BWR* variation.

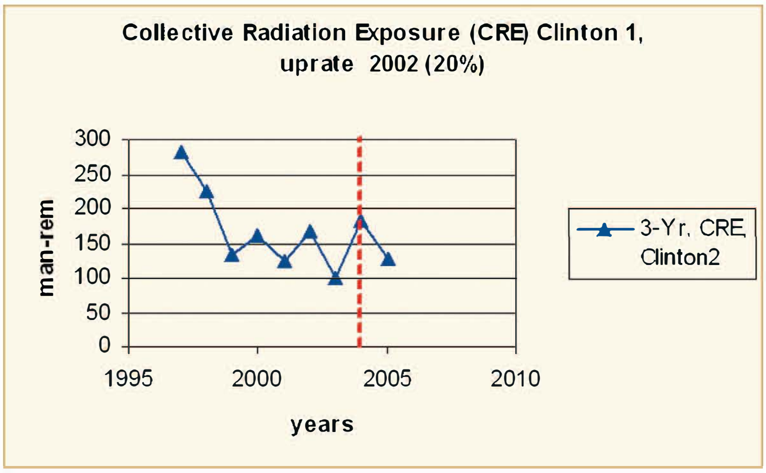

Figure 3-2: Overall trend in occupational doses and impact of an up-rate.

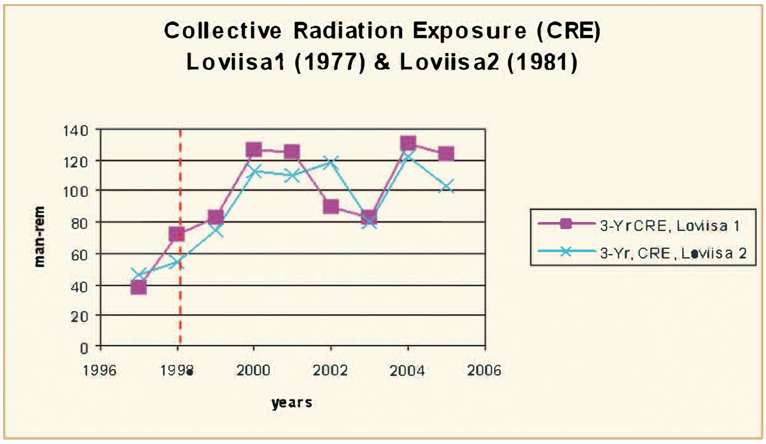

Figure 3-3: Same site variation among units.

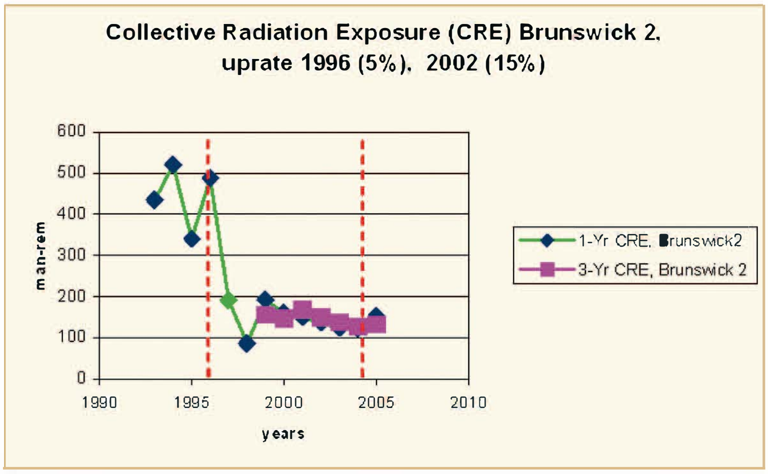

Figure 3-4: Effects of the averaging on presentation

Effects of the up-rate on fuel failures

The status of and the effects on fuel are one of the most important elements of the up-rates, in particular for extended up-rates, where originally established fuel margins are exceeded (requiring new design of fuel). This is the reason that the Up-rate 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 up-rates on fuel by evaluating fuel failures in relation to the time of up-rate. 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 *PWR*s and *BWR*s went in opposite directions, improving for *PWR* and deteriorating for *BWR*, although the failure rate (failed assemblies per 1,000 installed) for *PWR*s is still higher than for *BWR*s. It appears that whilst all fuel vendors have experienced fuel failures these failures are clustered on specific fuel models.

In *PWR*s, 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 *BWR*s, the dominant failures are debris fretting (five times higher than for *PWR*s) 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 *PWR*s, most failures are occurring on 11 units (16%). Among *BWR*s, there are 8 units (25% of total) where most failures occur.

The debris-related failures are hard to relate to an up-rate, 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 up-rate, but also of an extended fuel cycle. However, up-rates increased thermal duty in both *PWR*s and *BWR*s. Therefore, from a mechanistic point of view, power up-rates 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 up-rates offer a highly inconclusive picture. Only 17 US units did not perform any power up-rate at all. Of those, 14 experienced at least one fuel failure. On the other hand, 14 units performed large power up-rates (12 *BWR* and 2 *PWR*s). Of those only 9 experienced a fuel failure. This suggests that there is no obvious correlation of the power up-rate with fuel failures.

Review of recent operational events related to up-rates

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 up-rates. The aim of this activity was to help to 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 up-rates. 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 up-rates 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 up-rates.

Steam dryer damage at a *BWR*

After an extended power up-rate (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 up-rate was operated at power level greater than 100% because changes to the process-computer calibration constants for feed-water flow were not identified when the feed-water transmitters were replaced.

Unexpected feed-water heater Problems at a *BWR*

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

Turbine control system changes result in unanticipated operational challenges at a *PWR*

After a stretch up-rate 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 up-rate.

Power reduction at a *PWR*

Stretch up-rate 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 a *BWR*

In parallel with the extended up-rate, new fuel elements of GE11 type (9x9 fuel with part length rods) were introduced in a small *BWR*4 core, thus having a mixed core of GE11 and GE8 (8x8 fuel). During the performance of stability measurements, as part of an up-rate, 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 up-rate 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 (*ERV*s) at the end of the year. The licensee shut down the units to repair the *ERV*s and restarted the units with operation up to pre-up-rate power levels.

BWR plants had operated for several years at the extended power up-rate 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 were 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 up-rate conditions.

Abnormalities in ultrasonic flow meter instrumentation

Use of ultrasonic flow meter of the type used for *MUR* power up-rates 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 up-rate.

4 Analysis of the selected plants

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 up-rate of 25% compared to the initial thermal power level.
- Reactor design and operation conditions very close to most of the Swedish *BWR*s.
- A good availability of reactor data.

The following sections present the result of the in-depth review performed for the *OL1 & 2* plants. Data for the review had been obtained from the Teollisuuden Voima Oy *(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 up-rate

4.1.2.1 Power up-rate characteristics

On the west coast of Finland, in Eurajoki, *TVO* operates two 840 MWe boiling water reactors, *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 General Swedish Electrical Limited 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 (*RPV*), 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 up-rated twice since the commissioning. The thermal power of each reactor was increased from 2000 *MWt* to 2160 *MWt* in 1984 and to 2500 *MWt* 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 up-rate, 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. The latter up-rate 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.

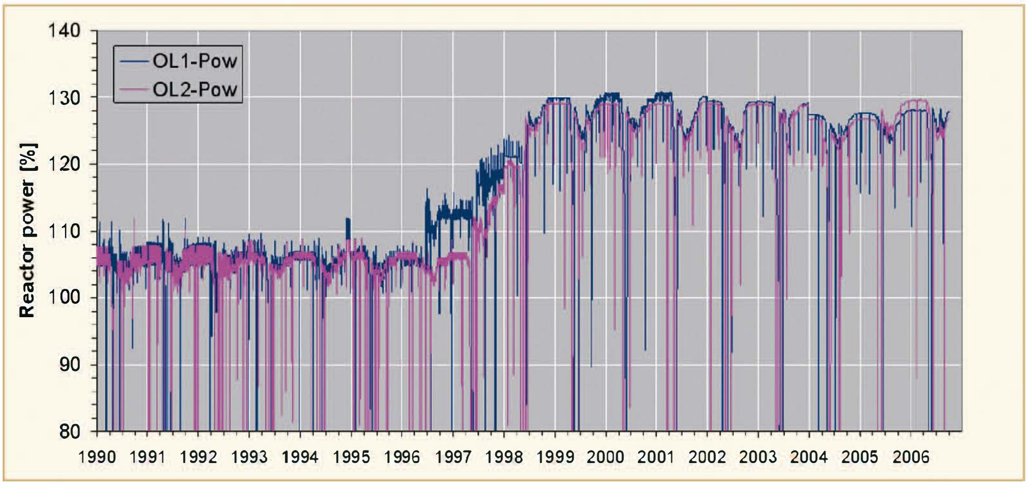

Figure 4-1: *OL1 & 2* – Relative reactor power based on net el. output (100% = 660 MWe).

The reactor building (Figure 4-2) is the highest and most dominant 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 a 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 the 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.

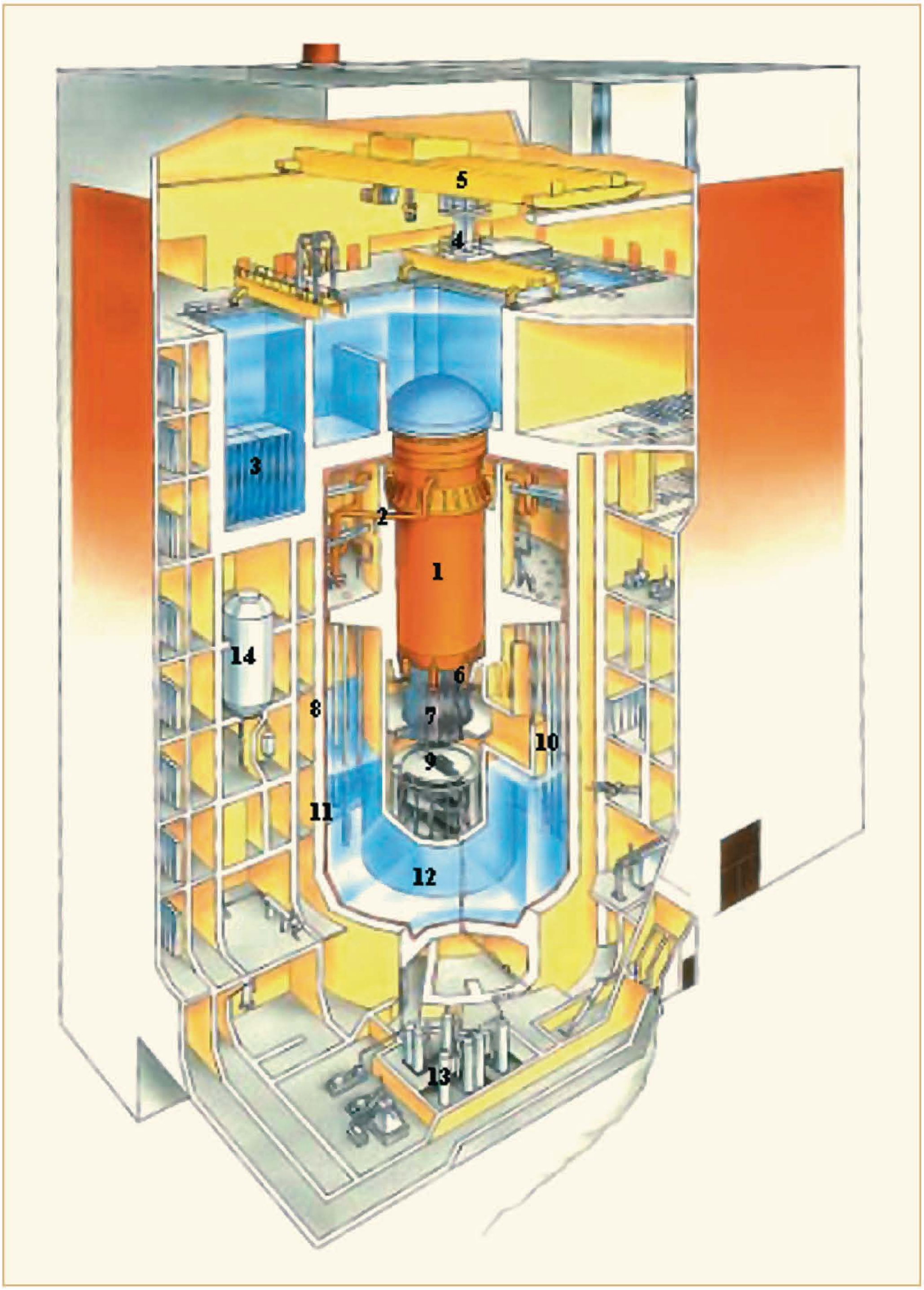

Figure 4-2: *OL1 & 2* - Section through the reactor building and the reactor containment vessel
1 – Reactor, 2 - Main steam lines, 3 - Fuel storage pool, 4 - Reactor service bridge,
5 - Reactor service room crane, 6 - Main circulation pumps, 7 - Control rod drives,
8 - Reactor containment vessel, 9 - Control rod service platform, 10 - Blow-down pipes,
11 - Embedded steel liner. 12 - Condensation pool, 13 - Scram system tanks,
14 – Venturi scrubber.

The *RPV* (Figure 4-3) 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 that the thermal insulation of the reactor cover is fastened to the inside of the containment dome, and is removed together with it when the reactor is to be opened. All external pipe connections to the reactor cover have been eliminated by making the connections inside the reactor, which means that the procedures for removing the reactor vessel cover have been considerably simplified.

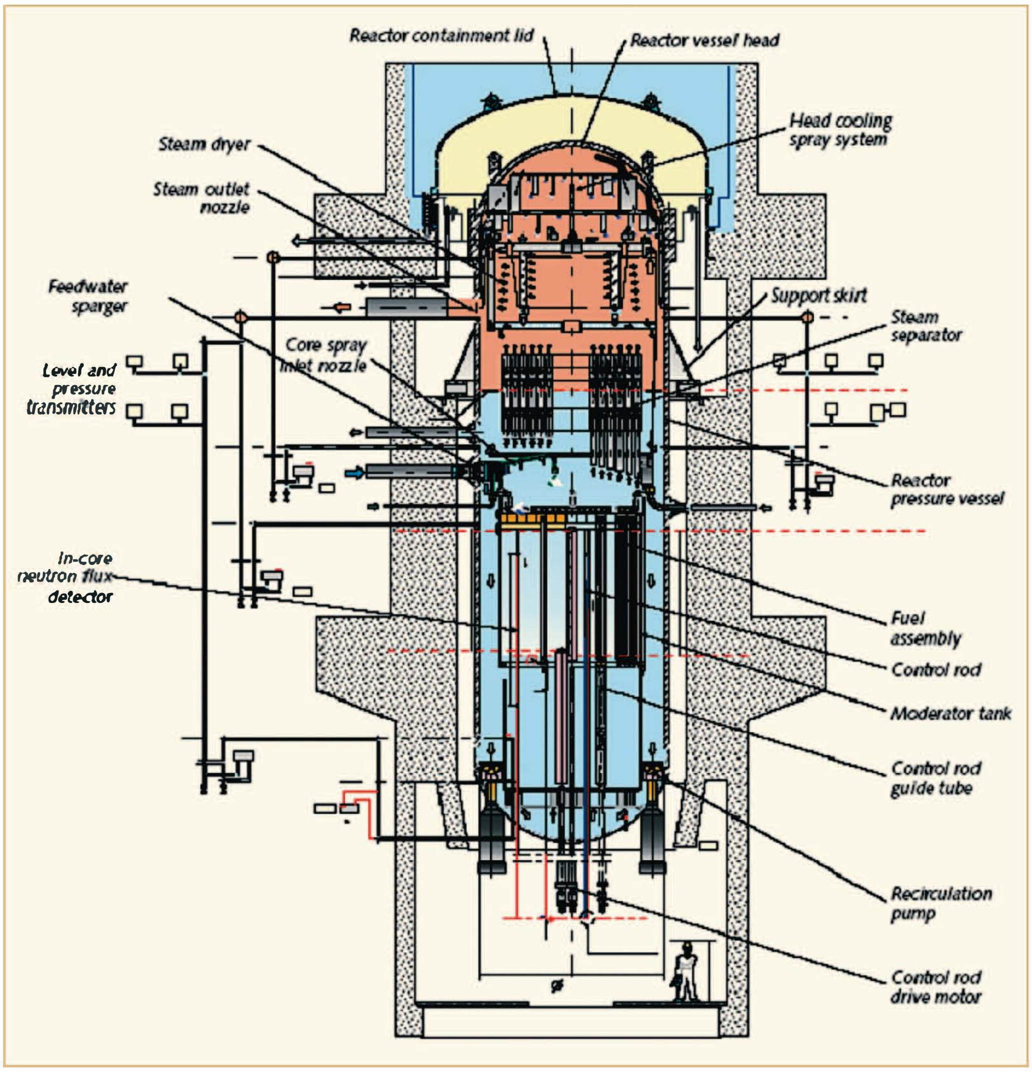

Figure 4-3: *OL1 & 2* - Sectional view of the *RPV*.

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 Rounds per minute (*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 4-4) 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.

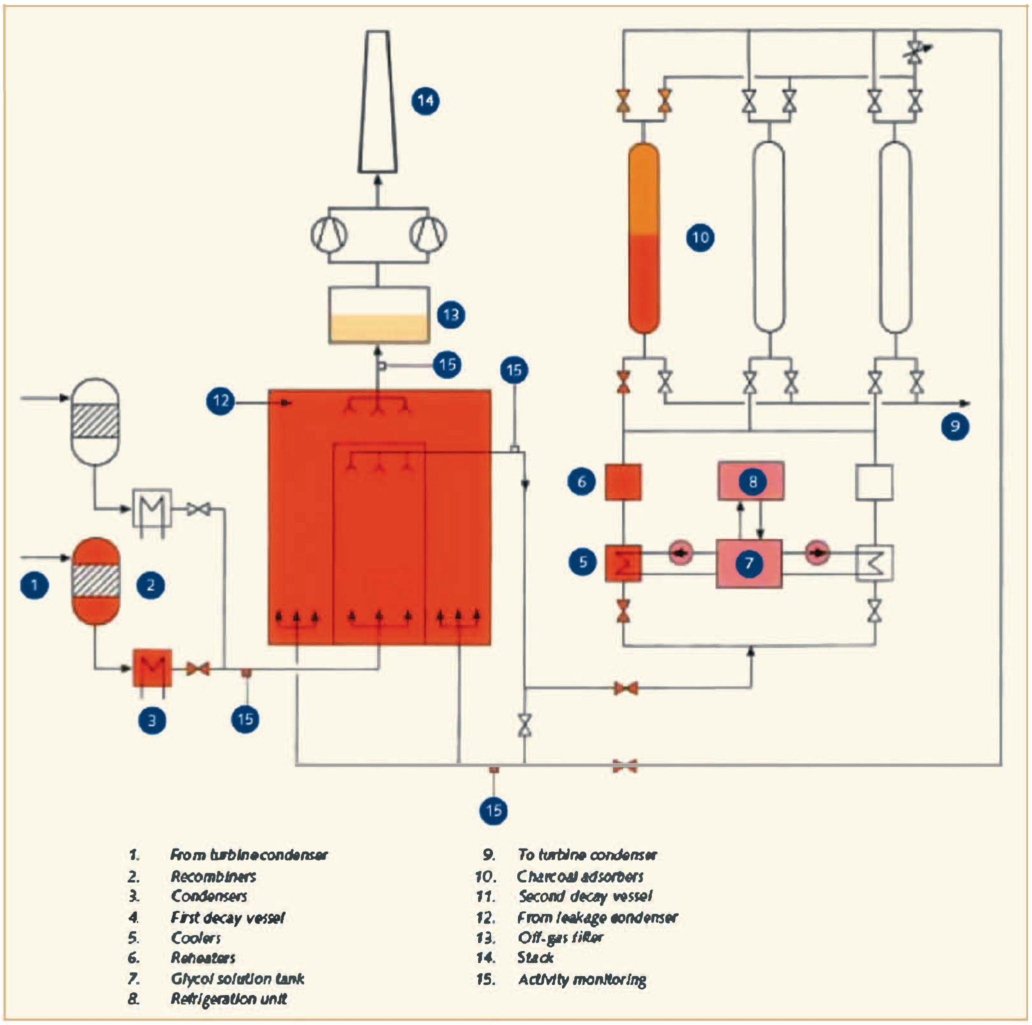

Figure 4-4: *OL1 & 2* – Offgas system.

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 *BWR*s. 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.

[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 *BWR*s in order to reduce radiation fields.

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 up-rate. 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 up-rate 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 *BWR*s and *PWR*s which affect radiation levels and occupational exposures in general, and especially at power up-rates. The following sections review technical factors important for radiation fields in *BWR*s and *PWR*s, and how these are affected by power up-rates.

5.1.2 *BWR* up-rates

5.1.2.1 Water chemistry issues

5.1.2.1.1 Corrosion product balance

The water chemistry control in *BWR*s to combat radiation build-up 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. 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.

The above general recommendations indicate different actions whether the plant is operating with *NWC*, *HWC* or has applied Noble Metal Chemical Addition *(NMCA)*.

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[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.

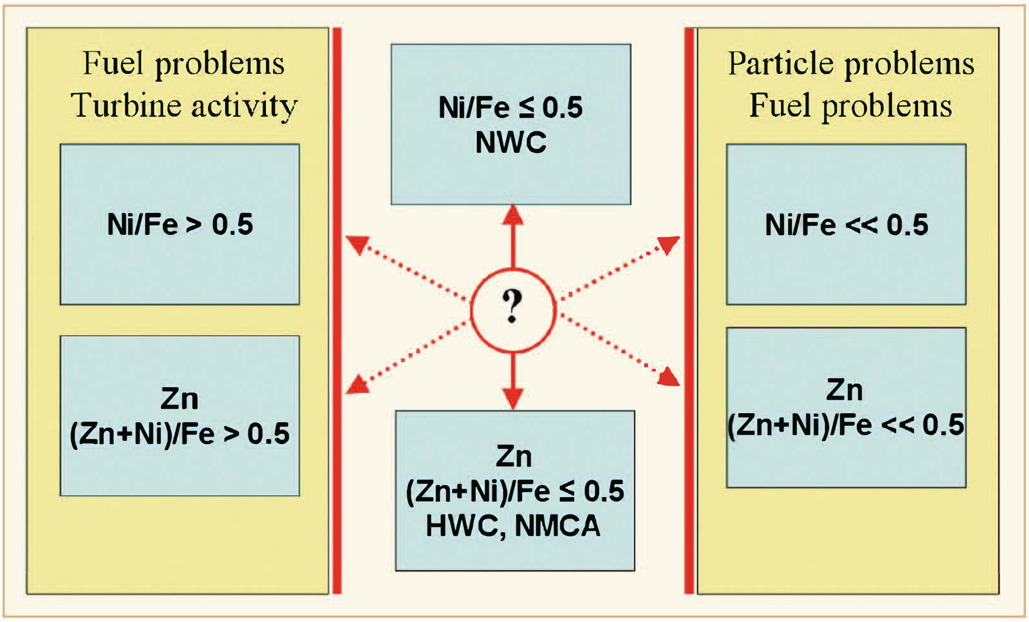

Figure 5-1: BWR Corrosion product balance Fe/Ni/Zn - Six different cases: Where to go?

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 *BWR*s 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 considerable 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. Factors affecting besides reactor design features are the degree of H_2 injection in the case of *HWC*, the degree of Noble Metal *(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 is accepted. 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 60Co and Zn is proposed instead (<2.0·10-5 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 nonmonitored 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:
FWCZn – Feedwater Zn concentration [ppb]
RWCZn – Reactor water Zn concentration [ppb]
MaxCZn – Maximum allowed reactor water Zn concentration [ppb]
fRWCU – Reactor water cleanup flow [kg/s]
fFW – 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, MaxCZn (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.

6 Conclusions

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

Compilation of power up-rates

Worldwide collection of information on the up-rates 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 *NPP*s 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 up-rates and the occupational doses could be established. The occupational doses on some plants seem to be higher after the up-rate, while on others seem to be lower.
- There is no obvious correlation of the power up-rate and fuel failures. However, performance of fuel for *PWR*s and *BWR*s went in opposing directions, improving for *PWR*s and deteriorating for *BWR*s.
- Through the data collection process events were identified that have occurred as a result of inadequate design or implementation of up-rates. 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 up-rates
- 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 up-rated 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 up-rate 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 up-rate) results in favourable water chemistry conditions that can be maintained, or even improved, after the power up-rate.
- 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 up-rate. 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 up-rate.
- 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 *RPV* 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 up-rate 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 up-rate 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 up-rates. 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 up-rate 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 up-rates. Radiation levels during operation indicate an average effect due to the power up-rates in the order +15% - +30%. This is probably due to the introduction of *HWC* with increased carryover 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 up-rates 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 up-rated 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 *SGR* was performed in 1995 but all safety analyses necessary for power up-rate were performed in 2000. The present study has focused on the first up-rate, resulting in a thermal power level of 8% compared to the initial power level. The latter up-rate was an up-rate of 1.5 %, achieved by using more precise techniques for measuring feedwater flow:

- *SGR* 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 up-rate.
- Dose rates, during outage, on reactor systems are maintained on a rather low and constant level after *SGR*.
- 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.

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Appendix A: Data base with *NPP* up-rates

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 MWt	4.1 Uprate power MWt	4.3 Year implemented	4.4 Uprate type & power increase (E, S, MU)*	%	4.5 Technical solution	4.6 Equipment
Belgium	Doel 2	WEC	12-01-75	PWR	1192	1311	2004	E	10%		SGR
	Doel 3	FRAM	10-11-82	PWR	2785	3064	1993	E	10%		SGR
	Tihange 1	ACLF	09-01-75	PWR	2665	2875	1995	E	8%		SGR
	Tihange 2/1995	ACLF	06-01-83	PWR	2785	2905	1995	S	4.3%		
	Tihange 2/2001	ACLF	06-01-83	PWR	2785	3064**	2001	E	10%		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
	Olkiluoto 1/1998	ASEA	10-01-79	BWR	2000	2500**	1998**	E	25%**		turbine, balance of plant equipment (pumps, steam dryer) Modifications in I&C
	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
	Olkiluoto 2/1998	ASEA	07-01-82	BWR	2000	2500**	1998**	E	25%**		turbine, balance of plant equipment (pumps, steam dryer)modifications in I&C
Germany	Emsland	KWU	06-20-88	PWR	3765	3850	1990	S	2,3%	Increase of average coolant temparature	
	Grohnde	KWU	02-01-85	PWR	3765	3850	1990	S	2.3%	Increase of average coolant temparature	
	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	

Country	1. Station name	5.Fuel				6. Annual liquid effluent releases		7. Annual gases effluent releases		8. Effluent Dose information	
Country		5.1 Fuel cycle months	5.2 Average linear fuel rating BU kW/m	5.3 Average linear fuel rating AU kW/m	5.4 Fuel type	6.1 Before the uprate GBq	6.2 After the uprate GBq	7.1 Before the uprate GBq	7.2 After the uprate GBq	8.1 Before the uprate microSv	8.2 After the uprate microSv
Belgium	Doel 2	11	22.22	22.22	FRA-AFA/KWU-AKA			206 tritium 52% year 2003 (14% of total for 4 units)			
	Doel 3	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 tri ium 67% year 1994 (36% of total for 4 units)		
	Tihange 1	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	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	15	17.85	17.85	FRA Std/ FRA AFA	11033 tritium 99.9% year 2000 (1/3 of total for 3 units)	19867 tritium 99.9% 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	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	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	12	17.90	17.90	8x8-1						
	Olkiluoto 1/1998	12	17.90	14.90	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	12	17.90	17.90	6x8-1						
	Olkiluoto 2/1998	12		13.10	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	Emsland	12	16.67	16.67	KWU Convoy		8300 tritium 99.9% year 1991		780 tritium 66% year 1991		
	Grohnde	12	21.10	21.10			16000 tritium 99.9% year 1991		1830 tritium 40% year 1991		
	Isar-2/1991	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	12		17.10	WEST/KWU			1140 tri ium 85% year 1997	980 tritium 49% year 1999		
	Neckar-2/1991	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	12									
	Philippsburg-2 1991	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	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	12		29.70	KWU/Fragema	18000 tritium 99.9% year 1999	13000 tritium 99.9% year 2001	4510 tritium 24% year 1999	1001 tritium 30% year 2001		
	Unterwester	11	20.50	20.50		7700 tritium 99.9% year 1999	16000 tritium 99.9% year 2001	4357 tritium 10% year 1999	3380 tritium 9% year 2001		

Country Country	1. Station name	9. Plant normal radiation levels 9.1 Before the uprate	 9.2 After the uprate	10. Occupational dose in operating year 10.1 Before the uprate	 10.2 After the uprate	11. Occupational dose in outage 11.1 Before the uprate	 11.2 After the uprate	12. Dose received during uprate
		manSv	manSv	manSv	manSv	manSv	manSv	man-Sv
Belgium	Doel 2			0.261 (ISOE) 0.28		0.237 (ISOE)		0.214 (outage)+0.195 (SGR)
	Doel 3			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			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			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			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			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			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			0.502 (ISOE 3 year)	0.677 ISOE 3 year)	no data	no data	0.620 (1984, ISOE annual)
	Olkiluoto 1/1998			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			0.410 (ISOE 3 year)	0.677 ISOE 3 year)	no data	no data	0.620 (1984, ISOE annual)
	Olkiluoto 2/1998			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	Emsland			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
	Grohnde			0.657 (ISOE 3 year)	0.835 (ISOE 3 year)	no data	0.750 (ISOE 2 year)	0.619 (1990, ISOE annual)
	Isar-2/1991			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			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			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			no data	no data	no data	no data	no data
	Philippsburg-2.1991			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			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			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			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