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LIST OF ABBREVIATIONS AND ACRONYMS

Abbreviation or Acronym	Description
2D	Two-dimensional
3D	Three-dimensional
AADQ	Annual Authorized Discharge Quantities
AB	Auxiliary Bearings
ac	Alternating Current
ach/h	Air Changes per Hour
ACS	Active Cooling System
AEC	Atomic Energy Corporation
AGC	Automatic Generation Control
ALARA	As Low As Reasonably Achievable
amsl	Above Mean Sea Level
AOO	Anticipated Operational Occurrences
APC	Atmospheric Pressure Change
ATP	Acceptance Test Procedure
AVR	Arbeitsgemeinschaft Versuchsreaktor (Jointly-operated Prototype Reactor)
BISO	Binary Coated Particles
BITE	Built-in Test Equipment
BOL	Beginning of Life
Bq	Becquerel
BS	Balancing Seal
BAS	Balancing Seal
C&I	Control and Instrumentation
CAS	Compressed Air System
CCDF	Complementary Cumulative Distribution Functions
CCGT	Closed Cycle Gas Turbine
CCS	Core Conditioning System
CCSV	Core Conditioning System Vessel
CDF	Core Damage Frequency
CFD	Computational Fluid Dynamics
CIS	Control and Instrumentation System
CMX	CCS Mixing Valve
CO	Carbon Monoxide
cps	Counts per Second
CRDM	Control Rod Drive Mechanism
CRF	Koeberg Circulating Water Pumpstation

Abbreviation or Acronym	Description
CRI	Certification Review Item
CRM	Continuous Radiation Monitors
CW	Cooling Water
DB	Dry Bulb
DBE	Design Basis Events
DBT	Design Basis Tornado
dc	Direct Current
DDP	Declaration of Design and Performance
DET	Decomposition Event Trees
DLOFC	Depressurized Loss of Forced Cooling
DOE	Department of Energy
DOP	Dispersed Oil Particulate
DQS	Designated Qualification Specialist
DS	Decontamination System
EFPD	Effective Full-power Days
EFPY	Effective Full-power Years
EHS	Equipment Handling System
EMB	Electromagnetic Bearings
EOL	End of Life
EPS	Equipment Protection System
EV	Explosion Valve
FDAS	Fire Detection and Alarm System
FHSS	Fuel Handling and Storage System
FIMA	Fissions per Initial Metal Atoms
FMEA	Failure Mode and Effects Analysis
FMECA	Failure Mode, Effects and Criticality Analysis
FP	Fission Product
FPD	Full Power Day
FPS	Fire Protection System
FPY	Full Power Years
FSAR	Final SAR
FSDP	Final Safety Design Philosophy
FTP	Fuel Technical Package
FZJ	Forschungszentrum Jülich
GDC	General Design Criteria
GmbH	Gesellschaft mit beschraenkter Haftung (German for Proprietary Limited)
GOR	General Operating Rules

Abbreviation or Acronym	Description
GRP	Glass Reinforced Plastic
GS	Generator System
GV	Generator Valve
GWd/t	Gigawatt Days per Tonne
h	Hour
H/D	Height/Diameter (ratio)
HAZOP	Hazard and Operability studies
HBS	Hardware Breakdown Structure
HCV	High Pressure Coolant Valve
He	Helium
HEPA	High Efficient Particulate Airfilter
HEU	Highly Enriched Uranium
HICS	Helium Inventory Control System
HLD	Heavy Lifting Device
HMI	Human-machine Interface
HMS	Helium Make-up System
HP	High Pressure
HPS	Helium Purification System
HPT	High-pressure Turbo-unit
HPTC	High-pressure Turbo-unit Compressor
HPTC-A	High Pressure Turbo Casing Top Volume
HPTC-B	High Pressure Turbo Casing Middle Volume
HPTC-C	High Pressure Turbo Casing Bottom Volume
HPTC-D	High Pressure Turbo Casing Double Wall Internal Volume
HPTV	High Pressure Turbo Vessel
HPTV-LPTV	High Pressure Turbo Vessel and Low-Pressure Turbo Vessel Connection Vessel
HTGR	High Temperature Gas Cooled Reactor
HTR	High Temperature Reactor
HTR-Modul	High Temperature Modular Reactor
HTTR	High Temperature Test Reactor
HV	High Voltage
HVAC	Heating, Ventilation and Air-conditioning
IAEA	International Atomic Energy Agency
ICS	Inventory Control system
ICV	Intercooler Valve
IE	Initiating Event

Abbreviation or Acronym	Description
ILSP	Integrated Logistic Support Plan
ILTI	Inner Low Temperature Isotropic
ISI	In-service Inspection
ISO	International Standardization Organization
KKS	Kraftwerk Kennzeichen System
KNPS	Koeberg Nuclear Power Station
kPa	Kilopascal
KSSR	Koeberg Site Safety Report
LBE	Licensing Basis Event
LCO	Limiting Conditions for Operation
LERF	Large Early Release Frequency
LEU	Low Enriched Uranium
LEU-TRISO	Low Enriched Uranium – Triple Coated Particle
LOCA	Loss of Coolant Accident
LOFC	Loss of Forced Cooling
LOI	Limiting Oxygen Index
LP	Low Pressure
LPC	Low-pressure Compressor
LPT	Low-pressure Turbo-unit
LPTC	Low-pressure Turbo-unit Compressor
LPTV	Low Pressure Turbo Vessel
LPTV-PTV	Low Pressure Turbo Vessel and Power Turbine Vessel Connection Vessel
LRU	Line Replaceable Unit
LSA	Logistic Support Analysis
LSAR	Logistic Support Analysis Records
LSSS	Limiting Safety Systems Settings
LTI	Low Temperature Isotropic
LV	Low Voltage
LWR	Light Water Reactor
MCB	Metallic Core Barrel
MCLR	Metallic Core Lateral Restraint
MCNP	Monte Carlo Nuclear Program
MCR	Maximum Continuous Rating
MECS	Method of Equivalent Cross-section
MES	Module Electrical System
MHTGR	Modular High Temperature Gas Cooled Reactor
min	Minute

Abbreviation or Acronym	Description
MOC	Means of Compliance
MPS	Main Power System
MRI	Master Record Index
MSL	Mean Sea Level
MSS	Main Support Systems
MTBF	Mean Time Between Failure (of an item)
MTTR	Mean Time to Repair (a function)
MWd/t	Megawatt Days per Tonne
MWh	Megawatt Hour
NAB	Nuclear Auxiliary Building
NB	Nominal Bore
NDE	Non-destructive Examination
NDRC	National Defence Research Council
NECSA	National Energy Council of South Africa
NNIT	Non-nuclear Integrated Heat Test
NNR	National Nuclear Regulator
NPP	Nuclear Power Plant
NUREG	Nuclear Regulations
OBE	Operating Base Earthquake
OCS	Operational Control System
OLTI	Outer Low Temperature Isotropic
OTS	Operational Technical Specification
p.a.	Per Annum
PB	Pressure Boundary
PBMR	Pebble Bed Modular Reactor
PBS	Product Breakdown Structure
PCU	Power Conversion Unit
PCUPV	Power Conversion Unit Pressure Vessel
PCV	Pre-cooler Vessel
PCV-RCV	Pre-cooler Vessel and Recuperator Vessel Connection Vessel
PEI	Post Event Instrumentation
PFSDP	PBMR Fundamental Safety Design Philosophy
PHST	Packaging, Handling, Storage and Transportation
PIDP	PBMR Integrated Design Process
PIE	Post-irradiation Evaluations
PLICS	Primary Loop Initial Clean-up System
PLOFC	Pressurized Loss of Forced Cooling

Abbreviation or Acronym	Description
POS	Phase One Scram
PPB	Primary Pressure Boundary
ppm	Parts per Million
PPRDMP	Post Pressure Relief Damper
PPSB	Power Plant Services Building
PQB	PBMR Qualification Board
PRA	Probabilistic Risk Assessment
PSID	Preliminary Safety Information Document
PT	Power Turbine
PTC	Power Turbine Casing
PTG	Power Turbine Generator
PTS	Phase Two Scram
PTV	Power Turbine Vessel
PTV-RPV-A	Power Turbine Vessel and Reactor Pressure Vessel Connection Vessel A
PTV-RPV-B	Power Turbine Vessel and Reactor Pressure Vessel Connection Vessel B
PWR	Pressurized Water Reactor
PWS	Potable Water System
PyC	Pyrolitic Carbide
QA	Quality Assurance
QAM	Quality Assurance Manager
QAP	Quality Assurance Procedure
QC	Quality Control
R/B	Release-to-birth
RAM	Reliability, Availability, Maintainability
RC	Reactor Cavity
RCCS	Reactor Cavity Cooling System
RCM	Reliability-centred Maintenance
RCS	Reactivity Control System
RCV	Recuperator Vessel
RCV-SBS	Recuperator Vessel and Start-up Blower System Vessel Connection Vessel
RH	Relative Humidity
R-M	Radioactive Material
RO	Reverse Osmosis
RP	Radiological Protection
RPL	Rupture Panel

Abbreviation or Acronym	Description
RPS	Reactor Protection System
RPV	Reactor Pressure Vessel
RPV-CCS-A	Reactor Pressure Vessel and Core Conditioning system Vessel Connection Hot Pipe
RPV-CCS-B	Reactor Pressure Vessel and Core Conditioning System Vessel Connection Vessel Cold Pipe
RPVCS	Reactor Pressure Vessel Conditioning System
RPV-HPTV	Reactor Pressure Vessel and High Pressure Turbo Vessel Connection Vessel
RSS	Reserve Shutdown System
RU	Reactor Unit
s	Second
SABS	South African Bureau of Standards
SAR	Safety Analysis Report
SAS	Small Absorber Spheres
SBS	Start-up Blower System
SCB	Software Control Board
SBSV	Start-up Blower System Vessel
SE	System Engineering
SF	Spent Fuel
SFCS	Spent Fuel Cooling System
SiC	Silicon Carbide
SIL	Safety Integrity Level
SIMPLE	Semi Implicit Pressure Linked Equations
SLO	Safety Limits for Operation
SR	Surveillance Requirements
SRU	Shop Repairable Unit
SSC	Systems, Structures and Components
SSE	Safe Shutdown Earthquake
Sv	Sievert
TBD	To be Determined
TEDE	Total Effective Dose Equivalent
THTR	Thorium High Temperature Reactor
TINTE	Time-dependent Neutronics and Temperatures
TRISO	Triple Coated Particles
TTR	Time to Repair (a function)
UF	Used Fuel
UFCS	Used Fuel Cooling System

Abbreviation or Acronym	Description
UPS	Uninterruptible Power Supply
URS	User Requirement Specification
USB	Unit Services Building
UV	Ultraviolet
Vac	Volts Alternating Current
Vdc	Volts Direct Current
VDU	Video Display Unit
vppm	Volume Parts per Million
VSOP	Very Superior Old Programs
W	Watt
WB	Wet Bulb
WHS	Waste Handling System

THE ROLE OF SAFETY ASSESSMENT IN THE SAFETY CASE

6.0.0 ROUTE MAP FOR SAFETY ASSESSMENT

The applicable fundamental safety principles are stated in the safety case **[2]** and the associated support document **[3]**. This chapter extracts from those documents, and provides a route map to other chapters of the SAR covering safety assessment, that link to additional detail of **what** is required and **how** and **when** it will be accomplished. All the points must be jointly satisfied to meet the requirements of the safety principles that apply in respect of safety assessment.

H-1.1 Appropriate safety assessments demonstrate that the PBMR design is in line with the PBMR Fundamental Safety Design Philosophy and meets the associated regulatory requirements. The assessments are made using probabilistic and deterministic methods and include design basis verification reviews against recognised good engineering practice. The safety assessments are undertaken to demonstrate that all events have been taken into account and, as appropriate, covered by studies. The systems or design features provided to prevent or mitigate the consequences of these analysed events are demonstrated to effectively ensure the safety objectives of the PBMR Fundamental Safety Design Philosophy are met.

H-2.1 Deterministic Assessment

H-2.1.1 Deterministic safety assessment consists of reviewing all licence base event sequences (including normal operation) from the assumed initial plant operating conditions through to the point of a safe stable state. This analysis takes into account, where necessary, the functioning of appropriate safety classified SSC's. For each Licence Basis Event analysed the initial plant conditions are chosen in order to penalise the possible consequences. This makes it possible to check compliance with the FDSP, and associated safety criteria and regulatory criteria. Specifically, it makes it possible to demonstrate that the plant design avoids the need for early operator intervention or the early functioning of any systems with moving mechanical parts in order to maintain nuclear safety.

H-2.1.2 This assessment also checks the ability to return the unit and maintain it in a safe state in the longer term, and is therefore used in the preparation of operating rules and procedures.

H-2.1.3 The assessment is extended to other areas where fuel is stored (new or used) other than the core to demonstrate that the same criteria in terms of adequate heat removal, limitation of chemical and physical attack and control of reactivity are met.

H-2.1.4 The assessment also checks the safety classification of SSC and defines their functional requirements necessary in the longer term for maintaining the unit at a safe state, including their contingent qualification.

H-2.1.5 The safety assessment is performed using:

- Verified and validated computer codes.
- Analysts with appropriate experience.
- Independent verification of key calculations.
- A set of Licence Basis Events (LBE), selected in accordance with regulatory criteria for deterministic analysis, are analysed in detail. LBE which can be shown to be of the same class but of a lesser consequence are regarded to be bounded by those LBE which are analysed.
- The single failure criterion in the analysis where applicable.
- Conservative assessment.
- Allowance for uncertainties.

H-2.1.6 The most penalising single failure is used in the analysis of events. Consideration is also given to the most penalising load combinations comprising for example the mechanical effects of earthquake and pipe breaks. The safety assessment also shows that LBE are not the cause or initiator of events in the next higher category. Thus the analysis considers SSC other than those responsible for initiating the event. The mechanical and other stresses brought to bear on these SSC is determined to demonstrate that they will not subsequently fail.

H-2.2 Probabilistic Risk Assessment

H-2.2.1 The probabilistic risk assessment of the PBMR design provides a systematic analysis to identify and quantify all risks that the plant imposes to the operators, general public and the environment and thus demonstrates compliance to regulatory risk criteria. The calculations of consequence are undertaken in the same manner as Deterministic Assessment (H-2.1) except that best estimate assumptions are used in all cases.

The SAR describes the methods used in the PRA, and summarizes the results. A preliminary estimate of fuel performance as a function of temperature and time is described and used to assess possible releases due to LBE analysed.

H-2.2.2 The PRA also identifies strong and weak points in the design and evaluates the levels of defence in depth that exist.

H-2.2.3 Unlike the structure of a PRA model for a typical LWR, which is based on the challenges to the integrity of successive barriers to the release of radioactivity, the structure of the PRA model for the PBMR is based on the challenge and the measures to protect the integrity of the coated fuel particles as the fission product barrier. Also the concept of core damage as used for the LWR's is not used. Rather fuel degradation is defined. The extent of fuel degradation, and therefore fission product release, is predominantly a factor of excessive fuel temperature and time at that temperature.

H-2.2.4 A demonstration that regulatory risk criteria are met is achieved through this focus on the challenges to fuel integrity. No credit is taken in the deterministic safety analysis of LBE for the role that civil structures may play in the confinement or limitation of radiological release from the fuel to the environment. However the status of systems, structures and components which may act as a further barrier or obstacle to the release of fission products, such as the primary circuit boundary and reactor building civil structure is modeled in the PRA. This best estimate approach provides a measure of the levels of defence-in-depth that exist in the design and operation of the PBMR and provides a tool for the optimisation of the design and operating programmes.

H-2.2.5 Initiating events are identified which could lead to a release of radioactive material and exposure to the workforce or general public. In particular initiating events are identified that may lead to challenges to the integrity of the fuel particles, either through inadequate heat removal, physical or chemical attack of the fuel or loss of control of reactivity. This includes areas where fuel is stored (new or old) other than the core. Specifically the PRA provides a basis for the classification of Licence Basis Events. It also provides a basis for judgement of plant personnel protection (particularly for very low frequency events where conservative deterministic analysis is inappropriate) to ensure that the risk is within specified limits.

H-2.2.6 The assessment also includes a hazard analysis to verify the adequacy of the plant layout. The assessment provides:

- a verification that no potential missiles risk damaging sensitive equipment as identified in the design phase.
- a confirmation that there are no sensitive plant locations where a pipe rupture could lead to the non-compliance with a safety objective.

- an analysis of internal flooding, covering the origin of the risk, the need for equipment to operate in the conditions in question, and relative layout of floodable rooms and possible pathways.

- a vulnerability analysis to verify the validity of the arrangements adopted for fire protection.

H-2.2.7 Structuring the overall approach according to the barriers and obstacles against fission product releases, in line with the deterministic analysis, ensures consistency between the probabilistic and deterministic safety analyses

The deterministic analysis as given in the SAR describes the LBE, and lists the obstacles that might prevent the progression of an event. In the majority of the cases investigated, it is assumed that most of the obstacles fail to function in order to arrive at a conservative enveloping result. However, the PRA assigns a probability of failure to each obstacle. This leads initially to only a few cases for which the consequences are determined analytically, and they are classed as enveloping events. Where obstacles are assumed to be in place for the deterministic analysis, they are in accordance with the description in the PRA.

H-2.3 Design Basis Verification

H-2.3.1 An engineering review of the design against proven engineering practice provides assurance that the design experience gained in other nuclear and related power plant technologies is used. The review focuses on the acceptability and compatibility of the design rules, codes and standards used and the adequacy of design margins.

Design reviews of safety classified SSC will be performed using international inputs, and a traceable record of these reviews will form part of the design verification documentation. The design review forms an integral part of the PIDP which specifies the design codes, standards and design rules. Design reviews are an ongoing activity.

H-2.3.2 Because of the uniqueness of certain aspects of the PBMR design it is difficult in some areas to confirm that the calculation techniques exactly represent the phenomena which come into play during an event. The design review verifies that appropriate conservatism caters for these uncertainties and inaccuracies. Results are only accepted from codes for which the design review shows to be sufficiently stringent as determined by their qualification testing. In areas where uncertainty exists, demonstration will be undertaken on the first module in tests which allow for adequate conservatism to ensure safety, but provides an appropriate benchmark. The degree of such conservatism will be confirmed by technical and safety reviews prior to the commencement of the test. The key principle of the testing programme is that

appropriate analysis is performed to ensure that nuclear safety is not affected by such testing.

H-2.3.3 The design base verification includes the review and acceptance of the qualification and the test and commissioning programme results.

H-2.3.4 All computer codes and software used in the design and analysis of the PBMR SSC are verified and validated to standards commensurate with the importance to safety of their application.

SH-2.1.1 LBE's are analysed to determine if the regulatory licensing criteria are met. Plotting the results on the Frequency – Consequence – Probability diagram (Figure SH-1) the dominant LBE's in terms of radiological consequences and frequency are identified and the degree of uncertainty in the results and their significance can readily be assessed.

SH-2.2.3 PBMR PRA

The PBMR Fundamental Safety Design Philosophy does not rely on active engineered safety features to maintain fuel integrity. The margin of safety and "defence-in-depth" is provided entirely by the single barrier embodied by the ceramic encapsulation of the fuel particles in each fuel ball. The interpretation of "defence-in-depth" can be understood in the sense that small particles of fuel are each encapsulated by the barrier, rather than large quantities of fuel being encapsulated by multiple barriers.

SH-2.2.4 to 2.2.7

No additional support information is required.

SH-2.3.1 Engineering review

The engineering review of the design is performed against review criteria derived from the PBMR Fundamental Safety Philosophy. This philosophy ensures that the review verifies that the principles of ALARA and defence-in-depth have been incorporated into the design.

SH-2.3.1.1 ALARA review

The ALARA activities accompanying the design of the PBMR reflect the actual engineering stage. They are thus conducted by following a stepwise approach.

The first ALARA review is of the engineering documents of the basic design phase, describing general principles of dose and source term optimisation and justifying the target value for the individual and collective dose.

A more detailed review of doses based on an analysis of maintenance tasks is performed on the engineering documents associated with the detailed design.

SH-2.3.1.2 Defence-in-depth review

The design review verifies that the defence in depth concept has been adopted.

The review verifies that detailed analysis and assessment of the design of the plant and of the various procedures have been performed to ensure that the lines of defence or barriers are of satisfactory quality and independence, taking into account all the plant provisions and operating procedures.

The safety philosophy is aimed primarily at the prevention of accidents but the review verifies that appropriate attention has been given to the mitigation of the consequences of accidents that could give rise to major releases of radioactivity.

REFERENCES

[1] Basic Licensing Requirements for the PBMR, document number LG-1037.

[2] PBMR Safety Case Philosophy, PNL-001, Rev. 3.

[3] Safety Assessment Philosophy Support Document, PNL-009, Rev. 2.

LICENSING BASIS EVENTS ASSESSMENT

6.0.1 EXECUTIVE SUMMARY

The Licensing Basis Events (LBE) are described phenomenologically using results from calculations performed with standard or proven reliable software. The results of the risk assessment and consequence analysis are given.

6.0.2 SOFTWARE USED IN ANALYSIS

Numerous software packages are used in the Safety assessment. Some software was developed in the time before strict Quality Assurance (QA) demands were placed on the development and use of software, but they generally come with a sufficient pedigree of successful application on other projects. This type of software is generally called legacy codes and is indicated as such in the text. Efforts have been made to find alternative software for some legacy codes in order to do independent checks on results obtained and in most cases this has been successful and is being applied. The list of software used is given in ***Table 6.1-1***. Some software is developed especially for PBMR applications and, where applicable, a full programme of verification and validation has been done or is in the process of being performed.

Table 6.1-1: SOFTWARE USED IN SAFETY ANALYSIS OF PBMR

PBMR Programme	Used For	Type
VSOP	Core calculations, control rod worths, heat transport	Legacy
TINTE	Core dynamics, DPLOFC heat transport etc	Legacy
FRESCO II	Fission product releases from fuel	Legacy
SPATRA	Fission product transport and plate out	Legacy
ORIGIN	Fission product & actinide production	Standard
MCNP	Monte Carlo neutronics, shielding, ex core criticality	Standard
STAR CD / FLUINT	Blow down forces on PCU, heat removal	Standard
FLOWNET	Depressurisation events, thermo-hydraulics simulation	Standard
Risk Spectrum	PRA event frequency calculation	Standard
PC COSYMA	Environmental exposure calculation	Standard
NASTRAN	Seismic analysis	Standard

6.0.3 COMMON INITIAL CONDITIONS USED IN THE SAFETY ANALYSIS

The aim of the safety analysis is to produce a set of results that have both a median value and a 95% upper confidence level. The calculations therefore use the nominal operating characteristics

of the reactor and thermo-hydraulic cycle as point of departure to calculate event progressions. Superimposed on these results are then the error margins as determined from a sensitivity analysis incorporating both the calculational errors and the uncertainties in material properties. The following initial conditions are common to all calculations, where applicable:

- Power output: 268 MW thermal.
- He inventory: 4600 kg in Power Conversion Unit (PCU).
- Coolant pressure: 70 MPa on high pressure side.
- Reactor outlet temperature: 900 ºC.
- Reactivity available in control: 1.3 Nile.
- Decay heat calculation to: DIN.
- Reactor core: equilibrium conditions.
- Fresh fuel enrichment: ~ 8.1 %.
- Spent fuel burn-up: 80 000 MWD/tonne.
- Failed fuel fraction: 6×10^{-5}.

6.0.4 DESIGN BASIS EVENTS

6.0.4.1 Loss of Power Conversion Unit (LBE-1)

The PCU is the main vehicle for removing energy from the reactor to the generator/heat sink. There are a number of reasons for the PCU to fail, i.e. Electromagnetic Bearing (EMB) failure, loss of load, vibration etc. The Operating Control System (OCS) will determine if the condition of the PCU allows the operation of the SBS. Should the SBS be available, the situation reverts to a standard operating state. The same applies if the CCS is started up on loss of PCU. The main interest for this event is when cooling is effected only by the RCCS or possibly when no RCCS is available either.

6.0.4.1.1 Loss of PCU with RCCS cooling (LBE-1a)

The reactor is shut down by the RCS and/or RSS and remains subcritical under all subsequent conditions. On cessation of cooling the decay heat is first distributed through the core cavity by natural circulation, radiation and conduction, heating up the centre graphite spheres. Natural

circulation continues to distribute heat in the core and transfer it to the reflector. With time the core temperature rises and a temperature gradient builds up across the core internals from the fuel to the reflector and across the core barrel to the RPV. From the RPV, the heat is transported by convection, radiation and conduction to the RCCS. Core temperatures continue to rise until the heat carried away by the RCCS exceeds the heat production. After that the core and the various component temperatures drop steadily in tandem with the reduction in decay heat production. Calculations of the fuel and component temperatures as a function of time were performed with TINTE and are reported in **[2]**. Note that a conservative margin of 10% was added to the decay heat production to account for any deviations in operating conditions and possible other uncertainties. These corrections will be quantified in further studies. In the calculation it was also assumed that prior to the event the control rods were being withdrawn and a reactor scram signal, either from neutron flux high or high outlet temperature was effective at the time the outlet temperature reached 950º. As can be seen in ***Figure 6.1-1***, the maximum fuel temperature stays below 1200 ºC as a result of natural convection in the pressurized core and no delayed fission product release is expected.

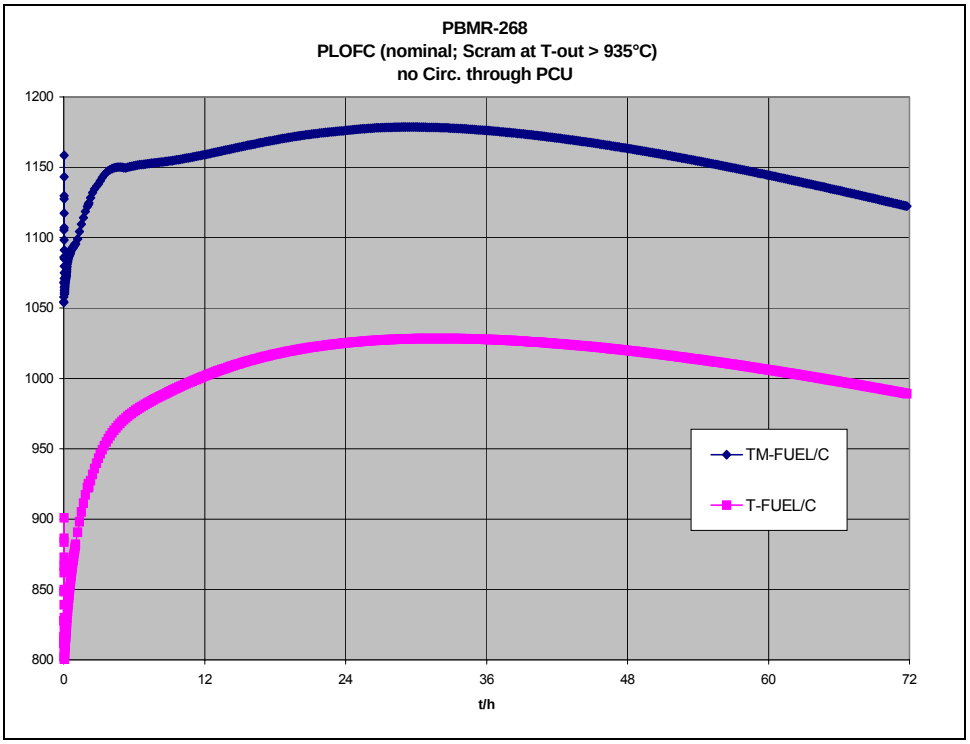


Figure 6.1-1: MAXIMUM AND AVERAGE FUEL TEMPERATURES FOR 72 HOURS AFTER EVENT

6.0.4.1.1.1 Plant end state

The core and gas in it are above normal operating values for at least the first 72 hours and must be reduced through the use of the CCS before operation can resume. There are no long term effects.

6.0.4.1.2 Loss of PCU without reactor trip (LBE-1b)

In this event, the reactor scram is assumed to fail altogether. As a result of loss of flow, the core temperature will increase immediately and cause the reactor to go subcritical. The reactor remains subcritical until the Xe has disappeared from the system and some cooling down has occurred. As a result of the cooldown the reactor will regain criticality after approx. 50 hours. In the ten hours following recriticality the reactor will oscillate between criticality and subcriticality as shown in ***Figure 6.1-2***. The reactor stabilizes at a power level of 0.5% of full power after about 72 hours. At this power level an equilibrium between power produced in the reactor and removed by the RCCS is established and no further fuel or vessel temperature increases will occur. The fuel temperatures are shown in ***Figure 6.1-3***. It can be seen that in this case the maximum fuel temperature will reach 1320 ºC for as long as the reactor remains critical. The likelihood that no action to cram the reactor is taken in 72 hours after the event is very small, but even so there is only a limited probability of delayed fission product release at these relatively low fuel temperatures.

6.0.4.1.2.1 Plant end state

The reactor must be tripped and cooled down by the CCS before normal operation can resume.

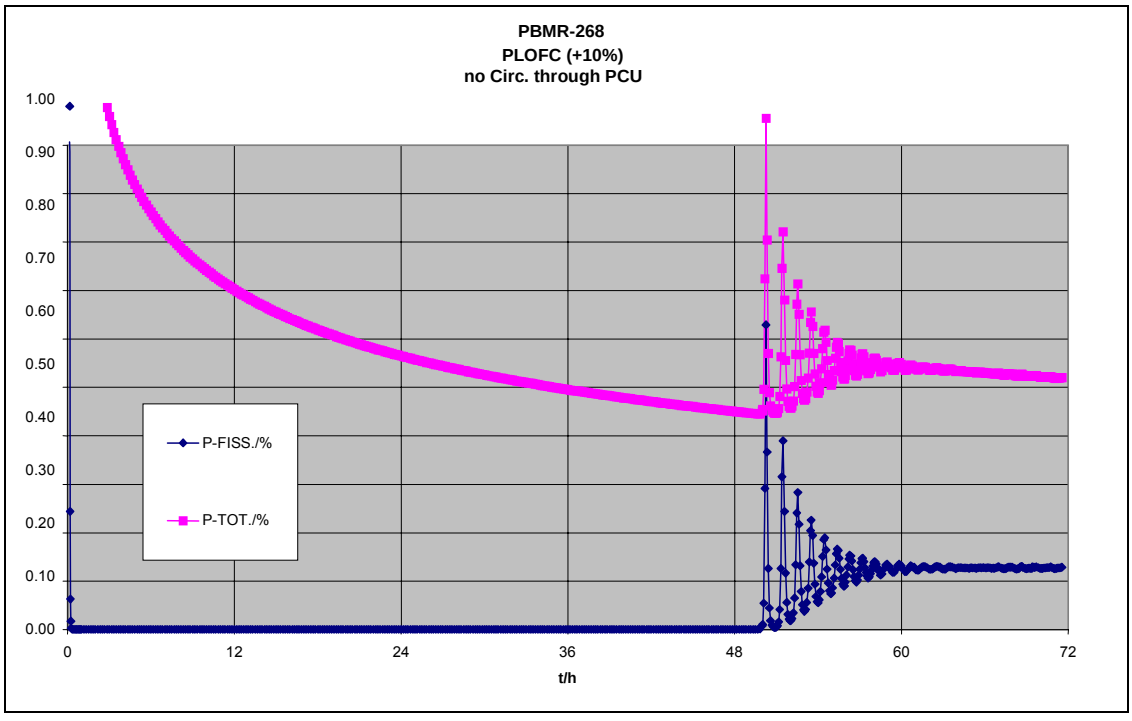


Figure 6.1-2: POWER LEVEL IN PBMR AFTER LOFC AND FAILURE TO SCRAM

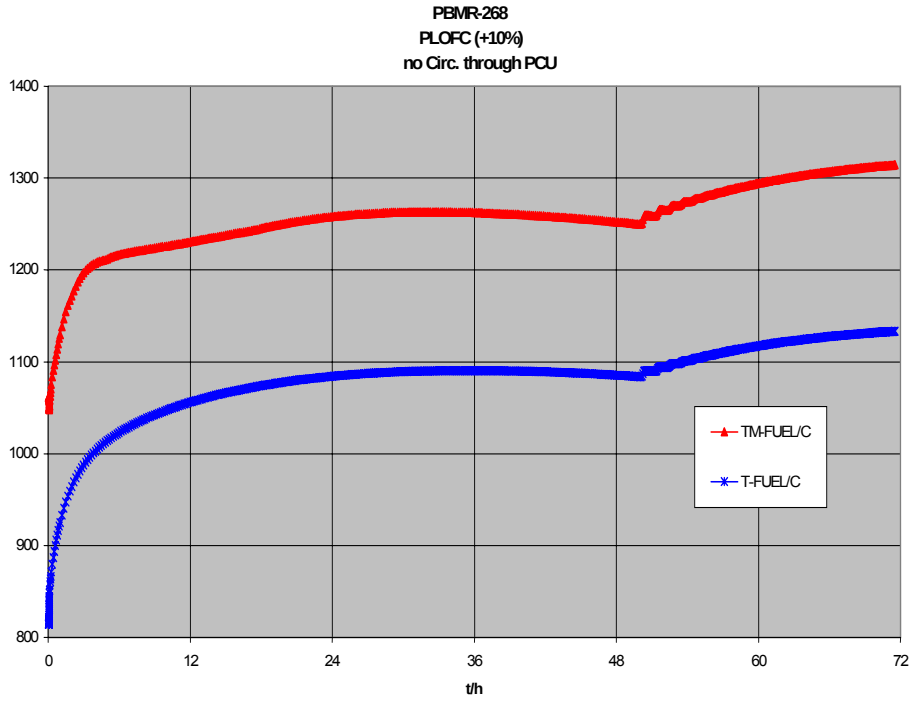


Figure 6.1-3: MAXIMUM AND AVERAGE FUEL TEMPERATURES FOR LOFC WITHOUT SCRAM

6.0.4.1.3 Loss of PCU without RCCS cooling (LBE-1c)

In this LBE, it is assumed that the RCCS is empty at the initiation of the event, due to an earthquake or similar external event. With the RCCS unavailable, heat is transferred from the RPV via the empty RCCS to the concrete wall surrounding the reactor cavity. Calculations show that the concrete will absorb and eventually transmit the decay heat generated by the core without a material rise in fuel temperature. The calculation was done for the case where there is a depressurization event before or accompanying the loss of cooling as this is the more penalizing event for the PBMR design. The results of the calculations are reported in **[1]**. The analysis shows that the concrete temperature will rise to unacceptable levels for continued use but that the core barel and RPV temperature will remain in the temperature range allowed by the design code as can be seen from ***Table 6.1-2***. The second column shows initial conditions after 1 hour with RCCS cooling. Columns 3 and 4 compare the components temperatures after 84 hours with and without the RCCS. The RPV reaches a maximum temperature after 84 hours and the concrete temperature is the same as the RCCS temperature. These results are in line with those calculated for the HTR-Modul.

Table 6.1-2: COMPONENT TEMPERATURES WITH AND WITHOUT RCCS OPERATING INITIAL TEMPERATURES

Component	Initial Temperatures (ºC)	T = 84 h with RCCS (ºC)	T = 84 h without RCCS (ºC)
CB	350	560	620
RPV	300	400	510
RCCS	50	50	380

6.0.4.1.3.1 Plant end state

The fuel and the core internals as well as the RPV will survive the event without serious damage. The structural material of the reactor cavity must be assumed to have deteriorated to the point where resumption of operation will be prohibited. The analysis shows no change in the maximum fuel temperature with or without the RCCS. The core can be defuelled and with time decommissioned.

6.0.4.2 Control Rod Group Withdrawals (LBE-2)

6.0.4.2.1 Control rod withdrawal with CCS cooling (LBE-2a)

As shown in ***Figure 6.1-4*** and ***Figure 6.1-5***, a group rod withdrawal does not result in a serious immediate fuel temperature rise due to the self limiting criticality caused by the negative temperature coefficient. The main effect is in elevated fuel temperatures when there is no forced cooling. This is described in LBE-2b. With cooling by the CCS present, cooling recommences after about 5 hours due to the time to condition the CCS, and by that time the fuel temperature is starting to rise but has not exceeded any temperatures where enhanced release of fission products will occur. There is no need to calculate this event as the immediate fuel temperature rise is enveloped by that for the uncooled case depicted in ***Figure 6.1-1***, and the long term effect is that the reactor is kept subcritical and cooled during the time it takes for the Xe to decay. If the reactor has not been scrammed by the time this has happened the CCS will be set to keep the reactor at a temperature where subcriticality is ensured. For this LBE, it is assumed that scram has taken place and the reactor remains subcritical under all cooling conditions.

6.0.4.2.1.1 Plant end state

As soon as the CCS has reduced the core temperature to normal operating conditions the reactor can be put into operation again, provided the cause of the event is known and repaired.

6.0.4.2.2 Control rod withdrwal with RCCS cooling (LBE-2b)

A group rod withdrawal can occur from any normal operating power level. The main consequence will be an increase in power level until the rods are fully withdrawn and the temperature coefficient stops the chain reaction. As the resulting maximum fuel temperature is mainly dependent on the decay heat, the worst case will be one where the reactor operates at full power before the event. In the equilibrium condition the maximum reactivity addition is limited to 1.3 Nile. In this situation the reactor power will increase until the reactivity addition has ended or until one of the scram signals causes the control rods to insert. Scram signals are generated in the following order by:

- Reactor period shorter than 20 seconds.
- Reactor maximum power level of 106% exceeded.
- Reactor coolant core outlet temperature exceeds 935 ºC.

For the analysis, it is conservatively assumed that reactor shutdown occurs only after the last scram signal activation.

In the normal operating position, the control rods are inserted to a depth of roughly 240 cm into the reflector. The maximum speed with which they can be removed is 1 cm/s, therefore it will take 240 seconds for full withdrawal. All three scram levels will be exceeded before this time has passed. In ***Figure 6.1-4***, the first 3 minutes of this events are depicted. It was assumed that the first two scram signals failed to function. Fuel and gas outlet temperatures for the first 6 minutes are shown in ***Figure 6.1-5***. The long-term fuel temperature is identical to that shown in ***Figure 6.1-1***.

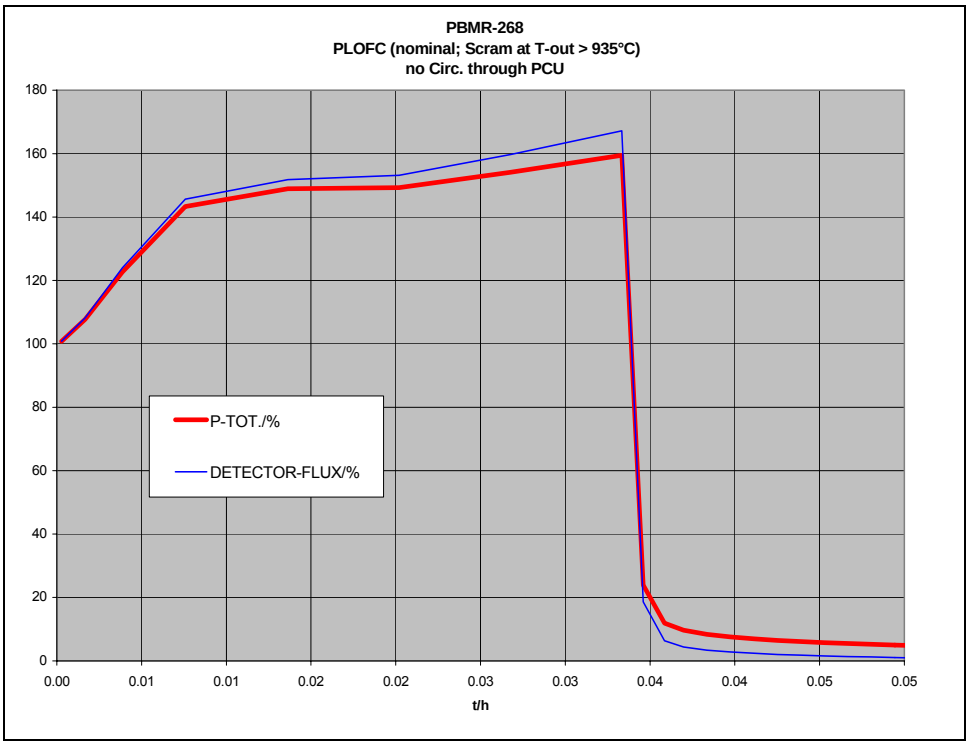


Figure 6.1-4: CORE POWER LEVEL AS A FUNCTION OF TIME DUE TO CONTROL ROD WITHDRAWAL WITH SCRAM

From the foregoing, it can be concluded that no significant consequences can be expected following a group rod withdrawal event.

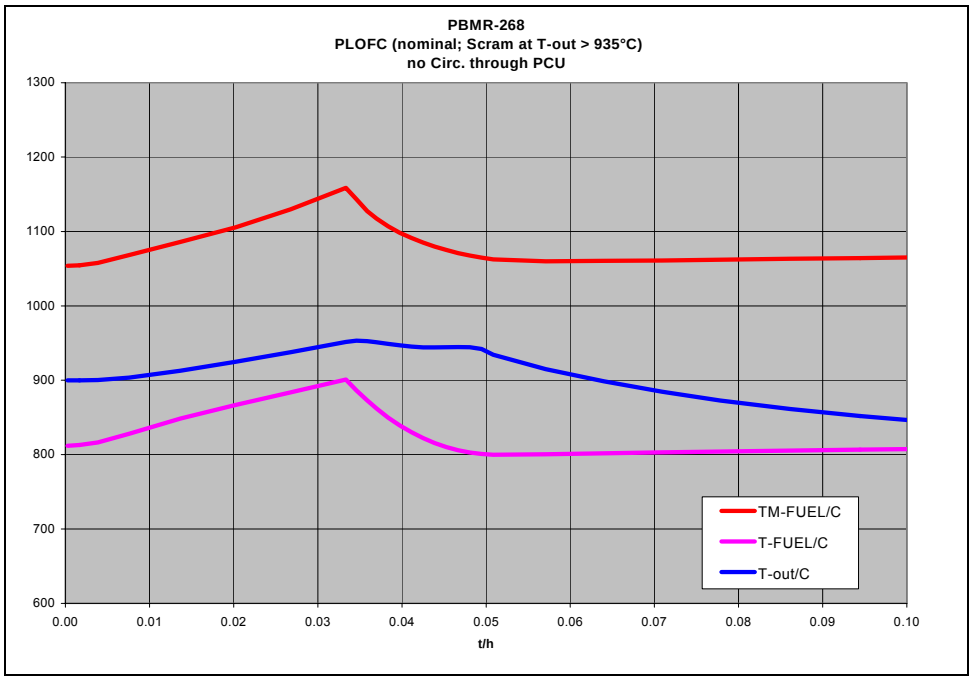


Figure 6.1-5: AVERAGE AND MAXIMUM FUEL TEMPERATURES AND GAS OUTLET TEMPERATURE FOR FIRST 6 MINUTES OF EVENT

6.0.4.2.2.1 Plant end state

For the event described, the long-term (>72 hrs) end state is that the reactor is shut down by the Reactivity Control System (RCS) without forced cooling. The RCCS is removing decay heat at the rate at which it is produced and the fuel temperatures have peaked and are slowly decreasing. Plant normal conditions can be restored by slowly introducing the CCS to bring the helium temperatures down to the operating level. As a result of the heat up of the fuel, enhanced release of fission products from the failed fuel present in the core will have been distributed through the Reactor Unit (RU), with much of it being entrained in the graphite parts. On restarting the circulation, some or most of these fission products will be transported to other parts of the MPS where they will plate out in the normal way, as happens during normal operation. A detailed analysis of releases over the time period and temperatures involved is at the moment beyond the scope of the present design and will be addressed in later issues of the SAR.

6.0.4.3 Primary coolant leak with isolation (LBE-3)

A small coolant leak is defined as a leak that is appreciably more than the normal expected daily loss of coolant, such that the plant has to be operated to a managed shutdown if isolation is not achieved. The leak size is such that the resulting building interior pressures and temperatures are within the capability of the Heating, Ventilation and Air-conditioning (HVAC) system and releases will thus be filtered. Isolation means that the leak is in a location that can be isolated from the MPS and if this is done early after detection, the system pressure may remain high enough for continued operation at normal or reduced power levels. The leak is assumed to be repairable while the plant is operating. The following subsets are investigated:

- LBE-3a: Small primary coolant leak with SBS cooling.
- LBE-3b: Small primary coolant leak with CCS cooling.
- LBE-3c: Small primary coolant leak with RCCS cooling.

6.0.4.3.1 Source term and releases

For a small leak, it can reasonably be assumed that only those fission products circulating with the coolant will be released from the system. The amount of the release is dependent on how soon isolation of the leak can be achieved and how much residual activity will leak from the unisolated components. As all the releases will be channelled through the HVAC and the exhaust stack filters, the only source term to be considered is the activity contained in the noble gases and the C-14. As this quantity is expected to be small, the calculation assumes that no isolation occurs until the

system is depressurized. This is a conservative assumption that will overestimate the dose to the public or the personnel. The resulting releases are given.

6.0.4.3.2 Plant end state

The plant end state is influenced by the location of the leak and the speed with which the system isolation has been done. As it will take many hours for the system to depresurize, the likelihood of successful isolation is high. In such a case the plant will continue to operate normally. If the leak is in the inventory control system, load following will be restricted until repairs have been effected. If the leak is in the Fuel Handling and Storage System (FHSS), normal operation will continue until non availability of the FHSS restricts power levels. Local contamination is very likely and this may affect the time it takes to repair the system, as local decontamination of surfaces in the affected area may be required. The three end states are:

- LBE-3a: Small leak with SBS cooling. Core is kept at normal operating temperature until repairs have been effected.
- LBE-3b: Small leak with CCS cooling. The delay in getting the CCS operational will result in core heat up until cooldown with the CCS is established.
- LBE-3c: Small leak with RCCS cooling. Core will heat up to standard long-term PLOFC temperatures until cooling is reestablished. Some delayed fission products release will take place. These fission products will plate-out in the system once normal operation resumes.

6.0.4.4 Primary Coolant Leak without Isolation with Pumpdown (LBE-4)

The main difference between this event and LBE-3 is the fact that no isolation occurs. This is due either to the leak being in the MPS where no isolation is possible, or the failure of isolation valves to close. In the latter case there is a high probability that late closure can be effected before all the helium has leaked from the system and that CCS operation may still be commenced. Of interest is the case where isolation is impossible. As the leak is small, operation will continue until it has been established that the leak cannot be isolated. This will then lead to a mandatory shutdown of the reactor in order to limit the amount of decay heat that can eventually heat up the core in the aftermath of the vent. When possible, the system will be configured to remove helium from the primary circuit for storage in the HICS. This removes some of the radioactivity available for release and preserves valuable helium needed to restart the reactor. As long as helium pressure in the system remains above atmospheric, the SBS can be used to remove decay heat. On reaching close to atmospheric pressure, the coolant circulation will cease and core heat up will commence. The following two cases are identified:

- LBE-4a: Small primary coolant leak with failed isolation with pumpdown.

- LBE-4b: Heat exchanger tube leak with pumpdown.

6.0.4.4.1 Source terms and releases

Core heat up will, in time, lead to enhanced release of fission products from already failed fuel particles. This constitutes an additional source term to that described in LBE-3. In this event, the released activity will be trapped by the filters of the HVAC and exhaust air.

6.0.4.4.2 Plant end state

The MPS is depressurized with two possible end states, that are applicable to LBE-4a. Only the last state applies to LBE-4b:

- The leak is easily repairable and the plant is ready for restart in a short time. A decision on decontamination can be made on the level of the contamination and the need to decontaminate for repairs or other access requirements. Decontamination may be postponed to a planned outage.

- The leak is not easily repaired. In this case, the action will be to close the leak temporarily to prevent air ingress during repair work. If the temporary repair is considered safe enough, cooling by SBS may be effected. If the RU can be isolated from the leak through the use of the maintenance valves, the CCS can be used to cool down the core and keep it at a designated temperature. If none of these is possible, i.e. the leak may be at the top of the reactor vessel, requiring special measures for access, the core will heat up and the decay heat is removed from the reactor cavity by the RCCS.

Note that for this LBE, the HVAC remains available to filter the helium expelled during the heat up phase that lasts approx. 48 hours. After that time cooldown may suck air into the reactor, although this will again depend on the size and location of the break.

The building itself and the major components within it will not be affected by this event.

6.0.4.5 Primary Coolant Leak without Isolation without Pumpdown (LBE-5)

LBE-5 is similar to LBE-4 excerpt that no pumpdown occurs. The only difference in the result is a higher source term to the environment as all the noble gases and C-14 released pass through the filter of the HVAC. Plant end states are as in LBE-4.

6.0.4.6 Primary Pressure Boundary Break with Isolation (LBE-6)

A medium size leak is defined as one that is too large for the HVAC to handle and that may result in the opening of the rupture panels leading to the pressure relief shaft. A timely isolation of the leak may prevent release of the leaking coolant through the pressure relief shaft to the environment. In that case the HVAC will be reactivated once the temperatures have reduced and cleaning of the air may recommence. It is assumed that isolation only takes place once the system is depressurized and the total contents of the system coolant are vented to the environment. A leak of this size will probably result in lift off of part of the plated out fission products, thereby enhancing the initial release. The following cases are identified:

- LBE-6a: Medium PPB break with isolation with SBS cooling.
- LBE-6b: Medium PPB break with isolation with CCS cooling.
- LBE-6c: Medium PPB break with isolation with RCCS cooling.

Isolation of the leak (in the Helium Inventory Control System (HICS) or FHSS) will probably occur at close to atmospheric pressure. This is still sufficient for the closed system to remove the decay heat of the reactor, shutdown by the protection system or by natural means, using the SBS or the CCS. Therefore no long-term core heat up will result.

6.0.4.6.1 Source term and releases

The source will include those isotopes that would otherwise be trapped by the filters. It will also include some desorbed radioactive material and possibly some dust. Calculations on the amount of material that will be desorbed depend heavily on the leak location and size. Detail calculations for a number of event scenarios in this respect will be performed once design details and source terms become available.

Source assumption is that all isotopes in circulation are released unfiltered. A small percentage of plated out activity will additionally be lifted off and released. The releases and consequences are described.

6.0.4.6.2 Plant end state

It can be assumed that isolation is successful before system pressure reaches atmospheric. This means that cooling by SBS or CCS can be achieved and no long-term core heat up will result. The pressure relief shaft system is provided with both an automatic and manual closure system and normal HVAC circulation can resume when closure is successful. The area around the leak will be contaminated and may need decontamination before the leak can be repaired and normal operation be resumed. The following states are possible:

- LBE-6a: Medium break with SBS cooling. Primary gas pressure sufficient to operate SBS after isolation. Core remains cooled and subcritical.

- LBE-6b: Medium break with CCS cooling. RU is isolated and cooling established after CCS conditioning. Slight core heat up until cooling is effected.

- LBE-6c: Medium break with RCCS cooling only. After isolation, natural convection will distribute temperatures as in the PLOFC event. Core heat up will lead to delayed release of fission products. These are distributed to the system and plated-out after recommencement of operation.

6.0.4.7 Primary Pressure Boundary Break without Isolation (LBE-7)

The major difference between this and LBE-4 is the fact that the HVAC is isolated and unable to filter the early release from the system. This makes for a larger source term to the environment which is enhanced by increased desorption from the plated out fission products. The increase in pressure and temperature will cause isolation of the HVAC with the result that one or more of the rupture panels will open. On closure of the pressure relief shaft after the event, either automatically or manually, normal HVAC circulation can be resumed. The following case is identified:

- LBE-7a: Medium non-isolatable PPB break.

6.0.4.7.1 Source term and releases

The full inventory of the MPS, including some percentage of the plated out fission products and dust, will be vented to the environment. The results of the analysis are given. As for the small pipe break, a heat up phase will follow and if not countered by recommencement of cooling by the CCS, additional material will be released from the failed fuel particles.

6.0.4.7.2 Plant end state

The MPS is depressurized and the core will heat up due to decay heat production. There are again two possibilities:

- The leak is not in the RU and maintenance valves can be closed and the CCS can be put into operation, thereby stopping core heat up and bringing the core to an acceptable final temperature;

- The maintenance valves cannot be used to isolate the reactor and the core heat up and subsequent cooldown will take place as long as the leak has not been fixed.

In the latter case, increased releases of fission products from the fuel will enter the core region and are partly distributed through the system and may enter the reactor cavity and citadel. If isolation of the pressure relief shaft was successful, the HVAC will remove any aerosols from the building atmosphere and no additional releases to the environment will take place. Failure to re-close the pressure relief shaft and restart the HVAC is at present regarded as a low probability event and not discussed in this report. However, the final release of fission products to the environment will be less than that described for the large break which can be regarded as the limiting event.

During the subsequent heat up phase, helium will be forced out of the reactor through the break. This will continue for about 30 hours, after which the core cooldown could suck air into the reactor. As long as no circulation of air through the core is possible, the little air that can enter will pose no corrosion problem. Air circulation as a result of two openings will result in corrosion of the graphite in places. Due to the fact that location and system resistance play a large role in this effect, an analysis of air ingress volume can only be done when detail piping layouts are available. The fact that so much time is available to close the opening makes this a low probability event that will be analysed and described in a separate report.

6.0.4.8 Beyond Design Basis Pressure Boundary Break with Isolation (LBE-8)

This event is similar to LBE-6 except that the faster depressurization will desorb more fission products and dust and the source term will be larger. There is no need for further description except that an additional calculation of the source term and releases is required. The following states are identified:

- LBE-8a: Large PPB break with isolation with SBS cooling.
- LBE-8b: Large PPB break with isolation with CCS cooling.

6.0.4.8.1 Plant end states

Due to isolation after depressurization, the following are three end states:

- LBE-8a: Large isolated leak with SBS cooling. Core is kept cooled to normal operating temperature and subcritical.
- LBE-8b: Large isolated leak with CCS cooling. Core is subcritical at normal operating temperature after CCS cooling is established.

6.0.4.9 Beyond Design Basis Pressure Boundary Break without Isolation (LBE-9)

Large non isolatable breaks can occur on the MPS or the HICS. Should the leak be due to a break

in the MPS and if the event is of such magnitude that the normal pressure relief system is unable to handle the overpressurization, the floor above the PCU will open up and panels in the outside wall will open to atmosphere to release the gas and relief the pressure. Wall panels in the helium storage area will open to atmosphere to release the gas and relief the pressure should a large break occur in the HICS. The case is identified as follows:

- LBE-9a: Large PPB break without isolation with RCCS cooling.

6.0.4.9.1 Source term and releases

The immediate effect of such a break in the MPS is the total release of all the coolant as well as a part of the plated out fission products and dust. The percentage of accumulated fission products that is desorbed is estimated to be 5% **[3]**. In the case of such a large break, it must be assumed that the damage cannot be repaired immediately. It is here assumed that closure of the maintenance valves is either impossible or impractical. Therefore there will be no active core cooling for some days after the event and core heat up with additional fission product release from the fuel to the core internals will take place. It is assumed that the vents have reclosed, but the building damage may persist for some time.

Migration of the fission products from the core to the opening and the release to the environment has not yet been analyzed, but data from previous studies indicate that even in this event the maximum dose to the public for category “B” events will not be exceeded.

6.0.4.9.2 Plant end state

A large break implies that part of the building is damaged and that repair or, at the least the restoration of initial conditions, of the floor of the laydown areas must be completed before the reactor can be restarted. Over and above that there is the possibility that replacing the damaged component, i.e. a pipe, may take some time. Operating procedures will detail mitigating actions to take to restore core isolation. These will depend on the location and size of the opening. At the minimum, the objective will be to close the opening in such a way that air ingress is minimized or stopped. This needs to be done before the core starts to cool down again, as this would imply that the air will be drawn into the core. As contamination within the area will be limited due to the low fission product inventory, such a closure appears to be feasible within a few hours after the event. The procedures will include the possibility that the event was due to an earthquake and that access to the area may be blocked.

As for the medium size break, the possibility of air ingress will be the subject of future studies to be performed on the detail design and on the premise that unlikely events also need to be analysed.

6.0.4.10 Large Earthquake (LBE-10)

Earthquakes are the major external initiating events.

- LBE 10a: Safe Shutdown Earthquake (0.3g) with SBS cooling.
- LBE 10b: Safe Shutdown Earthquake (0.3g) with CCS cooling.
- LBE 10c: Beyond Safe Shutdown Earthquake (0.4g) with intact PPB.

6.0.4.10.1 Safe Shutdown Earthquake (0.3g) with SBS cooling (LBE-10a)

The sequences and effects are described as applicable for a Safe Shutdown Earthquake (SSE). There is effectively no difference in the plant end state to that caused by any other intentional reactor shutdown with forced cooling. As there are no leaks and no core heat up, there are no immediate or delayed releases of radioactivity.

6.0.4.10.2 Safe Shutdown Earthquake (0.3g) with CCS cooling (LBE-10b)

The sequences and effects are described as applicable for a SSE. The difference with LBE-10a is that the cooling is effected with the CCS as the SBS is considered unavailable. Otherwise this is similar to a planned shutdown for maintenance. There is minimal core heat up during start-up of the CCS and there are no releases.

6.0.4.10.3 Beyond Safe Shutdown Earthquake (0.4g) with intact PPB (LBE-10c)

This is a beyond SSE as described. The SBS is unavailable for cooling and the CCS is assumed to be functional. The end state is the same as for LBE-10b, with no releases. The plant may only be restarted after inspection of key parts as described in the Technical Specifications and Procedures.

6.0.4.11 Large Earthquake with PPB Break (LBE-11)

This is a very low probability earthquake with consequences. Due to the damage to the MPS, the SBS and CCS are unavailable for cooling. The RCCS can continue to function in either the active or passive mode. The consequences and releases are identical to those described for LBE-9 and the public risk is described.

REFERENCES

[1] Basic Licensing Requirements for the PBMR; document number LG – 1037.

[2] Rüthen H.J, Haas H.A, Lauer A, Scherer W; Analysis on Neutronics and Thermal-hydraulics of

the Reference PO-Core in Accident Situations; document number FZJ-ISR-RC-5015/200.

[3] Moorman R, Vorläufiger Überblick zum Spaltproduktverhalten im Normalbetrieb sowie bei Druckentlastungen des PBMR-SA. Studies on the Safety Concept of the PBMR with PO-Core; Part B, June 1999.

PROBABILISTIC RISK ASSESSMENT AND CONSEQUENCE ANALYSIS

6.0.5 EXECUTIVE SUMMARY

The main purpose of the Probabilistic Risk Assessment (PRA) is to quantitatively demonstrate compliance of the Pebble Bed Modular Reactor (PBMR) design with the safety criteria of the National Nuclear Regulator (NNR) which are contained in **4.** At this stage of development, the PBMR meets regulatory criteria.

Other objectives of the PRA are to provide the basis for selecting Licensing Basis Events (LBE) and to aid in the design development process by identifying potential improvements of the design. The PRA is successively refined and updated as the design of the PBMR progresses. These updates will be reflected in future revisions of the Safety Analysis Report (SAR).

The scope of the PRA included the following:

- A systematic identification of initiating events.
- Event trees to define events sequences that result from the initiating events.
- Fault trees and data analysis to support event sequence frequency estimates.
- An uncertainty and sensitivity evaluation for the event frequency estimates.
- Common mode failures in fault tree models.
- Dose and risk to the public due to various licensing basis events.

A broad range of events were considered in this assessment to identify dominant sequences with potentially significant safety impact. The following initiating events families were utilized for selection of LBE:

- Group rod withdrawal.
- Loss of power conversion unit transients.
- Primary coolant releases.
- Earthquake.

From this list of events, only primary coolant releases including heat exchanger tube leaks and earthquakes were found to result in potential offsite consequences. These initiating events were

further used to generate a set of LBE.

6.0.6 INTRODUCTION

6.0.6.1 Background

The PBMR PRA has evolved over a period of two years starting in 1999. The process involved development of the methodology **5**, application and demonstration of the methodology using a limited number of licensing events **6**, and now recently performing a realistic PRA which reflect the state of the design **7**. An international review of the methodology and of the modelling of fault and event tree was also conducted in an effort to improve the quality of the PRA.

6.0.6.2 NNR Licensing Criteria

Licensing of nuclear installations in South Africa is based principally on the probabilistic approach to safety. In order to obtain a license for the PBMR, it must be demonstrated that the public is not exposed to unacceptably high risks as a result of its operation, by complying with the regulatory requirements laid down by the NNR in **4**. The criteria are applicable to both normal operations and accident conditions, and address the public. The regulatory requirements set out in **4** apply to all LBE that may lead to exposure. The LBE are classified into three categories, namely A, B, and C according to frequency. Category A events are all the events which occur with frequency greater than 10^{-2} per year. Category B events occur with a frequency greater than 10^{-6} per year but less than 10^{-2} per year. Category C events include all events, even those with a frequency less than 10^{-6} per year.

6.0.6.3 Analyses Required to Demonstrate Compliance

According to **4**, to demonstrate compliance, two types of calculations need to be performed, namely;

- deterministic dose calculations for category A and B events; and
- probabilistic calculations for Category C events[1].

The compliance criteria are summarized in ***Table 6.2-1***. The NNR has other criteria that must be complied with. One such criterion is associated with large releases and does not apply to the PBMR. This criteria is used in Light Water Reactors (LWR) to protect the public against the risks

[1] The risk criteria specified for Category C events must also be applied in Categories A and B events

from large accidents. The PBMR does not have any credible event which leads to fatalities. Another requirement of the NNR is compliance with risk criteria associated with plant personnel. This is performed separately.

Table 6.2-1: SUMMARY OF NNR LICENSING CRITERIA FOR THE PUBLIC

RISK TYPE	DOSE LIMITS		RISK (Fatalities per Year)
	Category A	Category B	
Average Risk			1×10^{-8}
Peak Risk	250 μSv.y^{-1}	50 mSv	5×10^{-6}

6.0.6.3.1 Deterministic dose calculations for Category A and B events

It must be demonstrated that the dose accumulated by the public for Category A and B events are less than 250 μSv per year and 50 mSv per year, respectively. These doses are determined at the site boundary, which represents the closest point to the plant where the public might be found.

6.0.6.3.2 NNR risk criteria for Category C events

In addition to the above deterministic calculations, risk calculations must be performed for Category C events. The NNR licensing criteria restricting the maximum number of yearly fatalities for the public arising from Category C events is 5×10^{-6} respectively. This value is referred to as the peak public risk. The average annual mortality risk for the public is 1×10^{-8}.

6.0.6.4 Scope of this Assessment

The scope of this assessment includes:

- the description of the methodology to perform the PRA;
- identification of Initiating Events (IE) and their quantification;
- development of the PBMR event trees to define event sequences resulting from each functional initiating event category;
- definition of end states for PBMR event sequences;
- analysis of the risk of the dominant sequences;
- treatment of common cause failures; and

- comparison of results to NNR criteria to determine compliance

6.0.6.4.1 General Assumptions

The following assumptions were made in this assessment:

- The technical details of the fuel, reactor and plant are based on the description given in this report and supporting plant diagrams.
- Where PBMR data were unavailable, generic information from the DOE High Temperature Gas-cooled Reactor (HTGR) PRA study was used **9**.
- The event trees are based on classical PRA methodology, but taking into account the unique features and characteristics of the PBMR.

6.0.7 METHODOLOGY

Generally, the objective of the PRA is to determine the frequencies of core damage and of fission product releases to the environment. Ultimately these are used to determine the health impact to the public. Traditionally, the process involves:

- identifying and quantifying initiating events;
- developing event sequences; and
- determining conditional probabilities and consequences of the event sequences.

In the traditional LWR PRA, the above are achieved through three levels of PRA, namely Level 1, Level 2, and Level 3. The Level 1 PRA assesses plant failures leading to determination of core damage frequency (CDF). The Level 2 PRA involves assessing the response of the containment leading to the determination of magnitude and frequency of fission product releases to the environment, including the estimation of Large Early Release Frequency (LERF). Level 3 determines the impact of the releases on the public health and safety.

While risk metrics such as CDF and LERF are key products of a typical PRA on LWR, they do not apply in any shape or form to the PBMR nor to any of its graphite moderated gas-cooled reactor predecessors such as the MHTGR. The inherent characteristics of the PBMR preclude core melting. The reactor core and heat removal systems have been designed to retain their physical and structural integrity, and to preclude failure of a significant number of fuel particle coatings over

a wide range of accident conditions that correspond to similar conditions in an LWR that would produce core melting. These inherent PBMR characteristics also eliminate fuel coolant interactions, reactivity induced pressure spikes, and core-coolant-containment interactions that are part and parcel of LWR accident progression phenomena.

The concept of a Loss of Coolant Accident (LOCA) also has no analogue to the PBMR because even though one can postulate depressurization accidents due to failures in the Primary Pressure Boundary (PPB), it is not possible to pull a vacuum on the primary system and, hence a total loss of coolant is physically impossible. Even if one could postulate such a condition, it is irrelevant to the capability of protecting the fuel elements in the PBMR because the core is designed to be cooled by conduction and radiation to the confinement without any active heat removal systems or without any forced or natural circulation of helium. Due to these characteristics the consequences of accident sequences which can be postulated by identifying functional initiating event classes, and various combinations of failures of active systems as is done in a PRA are small. Importantly, there is no single pinchpoint in accident progression, analogous to an LWR core melt event that leads to a big increase in potential consequences. Hence, one key challenge to developing a suitable risk informed process for the PBMR is the need to develop a different set of risk metrics to replace CDF, LERF, and CCFP.

In the PBMR, there is no discrete event or plant state such as “core damage” or “large early release” that produces large source terms from the fuel, from the PPB, or from the confinement. By contrast, the potential consequences of PBMR accidents span a continuum over a range of accidents that involve the release of very small sources of radioactivity over long periods of time. The timing and quantity of radioactive material releases in a PBMR accident vary over a limited range and involves participation by several small components of the total radionuclide inventory of the PBMR. A breakdown of the sources of radioactive material in the PBMR is as follows:

- Core inventory in Intact Fuel Particles.
- Core inventory in defective Fuel Particles.
- External to fuel particles in graphite matrix.
- Circulating Primary Coolant Activity.
- Plate-out activity.
- Dust activity.
- Spent fuel inventory in intact fuel particles.

- Spent fuel inventory in defective fuel particles.
- Helium purification, gas waste systems.

The vast proportion of the PBMR inventory of radioactive material resides within the TRISO coated fuel particles that are manufactured to a high degree of reliability with a strict quality control process to assure less than one defective particle in 10,000. These fuel particles are designed to remain intact over the full range of burn-up and normal and accident transient conditions. As such, the source term components that are potentially available for release during a PBMR accident are limited to such minor sources as the circulating primary system activity, the inventory in a very small fraction of the fuel particles that have manufacturing defects and a very limited increase in the failed fuel fraction from damage due to burn-up and thermal transients that may occur during an accident.

In the MHTGR PRA **9**, the most important component of the source term was found to be associated with that portion of the inventory in the failed or defective fuel particles at the beginning of the accident, and over the full spectrum of accidents considered in the PRA there was an insignificant source term identified from failure of any intact fuel particles during the accident. The PBMR is expected to exhibit similar behaviour. This should not be surprising as the core is designed to limit temperature increases during accidents to levels well below that needed to challenge the integrity of a significant number of the intact fuel particles. In the MHTGR PRA, a set of accident families or release categories were defined that delineated key variables of the accident sequence definition that determine the magnitude of the source term and offsite radiological consequences.

A similar set of accident families has been developed to support the ongoing PBMR PRA. These variables include:

- Status of the PPB (range from intact to different leak sizes).
- Status of actions to isolate pressure boundary leaks and to depressurize the primary system.
- Status of the core cooling systems (forced cooling on the SBS or CCS or conduction cool-down on the RCCS).
- Status of the RCCS (forced cooling mode or boil off mode).
- Degree of air or water ingress (whether any core oxidation will add to core temperatures or whether fuel hydrolysis will enhance the release from failed fuel particles.

- Response of the reactivity control systems.
- Response of the confinement systems.

The accident sequences in the event trees are assigned unique end states based on the outcomes of the above variables in order to group the sequences with similar potential for consequences and to develop the accident families for dose and consequence analysis. By contrast with the Levels 1,2, and 3 framework of a typical LWR PRA, the nature of the PBMR lends itself to a single integrated event sequence modelling framework that starts with initiating events and results in accident sequence families that encode sufficient information to support the offsite consequence and dose assessments.

The PBMR PRA adopts a methodology that differs from the classical PRA approach, because the PBMR design does not require engineered safety systems to mitigate the consequences of accidents **5**. The overall approach is based on a barrier/obstacle analysis for each major source of radioactive material in the PBMR.

The first step of the barrier/obstacle approach is to identify the barriers or obstacles containing (or potentially containing) fission products sources. The SiC coating of the fuel particle containing fission products is considered the primary barrier. Consideration is also given to another source of fission products associated with the primary barrier. These fission products are present in the graphite matrix in which the fuel particles and therefore the primary barrier are bonded. They result from the following:

- diffusion through an intact coating;
- diffusion through a chemically, thermally or irradiation induced degraded coating;
- diffusion through a small number of defective fuel particle coatings and
- contamination during manufacturing.

There are two types of secondary obstacles, namely:

- The PPB.
- The spent fuel storage containers.

The tertiary obstacle in the PBMR design is the confinement. The confinement completely houses the Main Power System (MPS) and spent fuel containers and as arranged to preclude a release path that bypasses the containment.

The second step in the barrier/obstacle approach is the identification of the *challenges* and *failure modes* of each barrier and obstacle. Two principal challenges to the primary barrier are identified:

- *Temperature* – events that lead to elevated fuel particle temperatures that may result in the release of fission products from any failed particle coatings. Due to the inherent features of the PBMR, fuel particles failures are essentially confined to these very small number of defects in the fuel manufacturing process.

- *Corrosion* – chemical and irradiation corrosion of the SiC coating.

There are several challenges to the secondary obstacles (the pressure boundary and spent fuel storage tanks) which may result in release of radioactive material to the environment. Challenges to the pressure boundary are mainly overpressure events, thermal shock, and mechanical failure of other Systems, Structures and Components (SSC). The spent fuel storage tanks, on the other hand, may be challenged by mechanical failure. Mechanical failure may be caused by internal or external events such as missiles, earthquakes, etc.

The challenges to the tertiary obstacle (confinement) include overpressure (venting), structural faults, external missiles, earthquakes, etc.

Practically, the methodology for identifying the challenges to the barrier and obstacles includes the following elements:

- identifying the initiating events that could lead to challenges to the barriers/obstacles;

- determining the responses of the barriers/obstacles; and

- determining the consequences of failure of the barriers/obstacles.

6.0.7.1 Initiating event selection

The starting point of a PRA analysis is the identification of initiating events. However, due to a large number of events that potentially could result in release of radioactive material, it would be impractical to analyze each initiating event. Therefore, a set of initiating events is determined that is representative of a wide spectrum of events. The selection of initiating events is achieved by following a systematic approach. This systematic approach provided a logical framework to guide the selection and grouping of initiating events, and ensure completeness.

The selection process involved:

- identifying all potential sources of fission products in the plant and barriers/obstacles to the

releases; and

- identifying the challenges (failure modes) that could lead to failure of the barriers/obstacles and subsequent release of fission products.

The identification of challenges that could potentially lead to radioactive material release was performed using a logic diagram.

The identification of events initiators is followed by the development of event sequences, each culminating in a particular type of release. Event sequences were identified using event trees. The estimates of the consequences of the various releases are compared, and those releases with dominant risks are then identified. The initiating events of these dominant event sequences are then selected for detailed analysis in the PRA. The methodology used to identify initiating events is described.

6.0.7.2 Event Tree Development

Once initiating events have been identified, the next step in PRA assessment is the identification of event sequences. These are identified by constructing event trees for each initiating event identified. Event trees model as top events those systems required to protect the various barriers to fission product release. Because of the absence of engineered active safety systems for decay heat removal and the protection of fission product barriers and obstacles, traditional event trees representing the subsequent successes or failures of these systems cannot be constructed. Also, in the traditional approach, fault trees are constructed that represent the logical combinations of redundant component failures that may lead to safety system failure.

In order to construct event trees, knowledge of plant systems is required, as is knowledge of plant transient response to the various initiating events. The branches in each event tree should reflect the capabilities of each system and their availabilities. The logic for developing the PRA is based on the need to resolve whether the plant achieves a safe stable end state or, if not, to resolve the appropriate consequence state.

Although the PBMR design does not contain classical safety systems, it does contain systems that could reduce the *magnitudes* of the consequences of initiating events. For completeness, these systems will also be included, and fault trees constructed where required.

6.0.7.3 Consequence States

The primary logic used in the development of the event trees is based on the need to resolve the key factors in the plant response to an initiating event that determine whether the plant will reach a

safe stable end state, or if not, the magnitude of the potential consequences. A unique consequence state is determined for each event tree sequence based on the outcome of key questions that are resolved in the delineation of event tree top events. The consequence end states are derived from a consideration of the time dependent response of each barrier to the release for each source of radioactive material.

For accident sequences involving the sources of radioactivity in the PPB, which constitute the greatest sources and largest potential for release, the following key factors determine the end states:

- PPB status.
- Core cooling status.
- Corrosion attack status.
- Reactor shutdown status.
- Confinement status.

6.0.7.3.1 Primary Pressure Boundary (PPB) Status

The status of the PPB at the time of the initiating event and any changes to this status during the accident are key factors to the determination of whether the PBMR can achieve a safe stable end state or the magnitude of any accident consequences. The PPB status determines whether and how much of the circulating primary coolant activity is released, and will help define the fission product transport of any releases from the fuel or graphite. The key variables in the PPB status are

- Whether or not there is a failure in the PPB as the initiating event.
- The size and number of the PPB leaks, if any at the time of the initiating event.
- Whether operator actions to isolate the leak during the sequence are successful or not if the leak is isolable.
- Whether actions to pump-down the primary system during small un-isolated PPB leaks are successful.

6.0.7.3.2 Core Cooling Status

A key factor in determining the potential for releases from the fuel elements and graphite during an accident is the time dependent temperature response of each location within the reactor core and

graphite. There are several modes of core cooling following an initiating event supported by different systems that help to resolve the detailed time dependent response of the core that is needed to calculate the source term. The key variables in the status of the core cooling system that help define the end states include:

- Whether there is continued forced circulation of Helium following the initiating event and whether the forced cooling is provided by continuation of the Brayton Cycle, use of the SBS blowers with heat removal via the precooler and/or intercooler to the ACS, or use of the CCS with heat removal to its ACS.

- For cases where there is no forced circulation of Helium in which case there is a conduction cooldown, whether the RCCS operates in the active mode with continued pumped circulation of cooling water, or whether the system operates in a passive boil off mode.

For sequences involving conduction cooldown using the RCCS, the resulting increase in temperature in some parts of the core is expected to contribute to the release of part of the source of radioactive material in the failed or defective fuel particle coatings.

6.0.7.3.3 Corrosion Attack Status

Depending on the initiating event and event sequence progression, there may be air or water ingress to the primary system and part of this may enter the active core. Air ingress to the core could result in graphite oxidation which in turn could contribute to local increases in temperature and elevation in releases from the failed fuel particles. Water ingress could contribute to hydrolysis reactions which could increase the source term from these same failed fuel particles. The states considered for this end state parameter include:

- Whether or not there is significant air ingress to the core.

- Whether or not there is significant water ingress to the core.

6.0.7.3.4 Reactor Shutdown Status

The negative temperature coefficient and small excess reactivity of the PBMR design provides for a passive means of reactor shutdown for core heat up transients independent of any active trip system actions. In addition to these features, there are two active trip systems the RCS which is automatically activated using control rods, and the RSS which is manually activated using borated pellets. These systems are designed to shutdown the reactor during transients that will terminate the Brayton cycle. The temperature response of the core is affected by the response of these systems, and in the event of the RSS the timing of its activation and are influenced by the nature of

the initiating event. The end state variables for this parameter include:

- Whether the initiating event produces a reactivity excursion or not.
- Whether there is negative reactivity inserted via the RCS or RSS.

6.0.7.3.5 Confinement Status

The confinement is designed to filter any release from the PPB for a range of accidents involving small PPB leaks, and for a larger set of leaks, is designed to operate a rupture disk to limit confinement pressures during PPB depressurization and to re-close a damper valve to limit the rate of egress of any fission products subsequently released from the PPB. The variables tracked for this end state parameter include:

- Whether there is any release from the PPB and hence, whether the confinement function is needed.
- Whether the confinement filtration function is maintained during the accident.
- Whether the confinement vent valves re-close or not when opened for larger PPB leaks.

A similar logic is used to define the end states for the other sources of radioactivity.

6.0.8 PBMR PRA EVENT FREQUENCY RESULTS

6.0.8.1 Initiating Events Analysis

6.0.8.1.1 Initiating event identification

Initiating events identification is the first step in the PRA. Initiating events are essentially events that cause the plant to deviate from its normal operating states and could potentially result in releases of radionuclides into the environment. There are generally two approaches that can be taken to identify initiating events. The first approach involves a comprehensive engineering evaluation taking into account information from previous risk assessments, documentation reflecting operating histories, and plant-specific design data. The information is evaluated and a list of initiating events is compiled based on engineering judgement. The second approach involves systematically searching for initiating events using a top level master logic model and then deducing the appropriate set of initiating events. Portions of each approach can be effectively used to determine the initiating events. Both approaches have been used in the PBMR.

The current initiating event list for the PBMR was generated using the second approach by constructing a master logic model. The process followed is described in detail in **7**. The master logic model essentially looks at all potential sources of radiation even if their potential contribution

to public risk is insignificant. It then looks at the various functions needed to maintain control of radioactive releases from these sources. Then the critical systems and structures required to perform these functions are identified. The set of initiating events that challenge these safety functions is then derived. The first approach was used to test for completeness by examining initiating events considered in the MHTGR design **9** and the HTR-Modul **10**.

6.0.8.1.1.1 The master logic diagram

The master logic was constructed to guide the selection of the accident initiating events and to ensure completeness. It ensures completeness by providing a logical framework to systematically identify ways in which radiological releases can occur from each source of radioactive material. Events which have initiated such releases are then the initiating events. The concept of safety functions forms the basis of the master logic. The master logic diagram used in the current assessment is given in

Figure 6.2-7 and

Figure 6.2-9. The following are the major steps of the master logic diagram:

Step 1: Identification of radionuclide sources in the plant and the barriers that retain them (for example, the SiC coating and the pressure boundary). This step also includes identification of safety functions responsible for ensuring radionuclide retention in the various barriers.

Step 2: Identification systems and structures which can mitigate the releases if they occurred.

Step 3: Identification of the failure mechanisms of the various systems and structures responsible for ensuring that the barriers are intact.

Step 4: Identification of the potential initiators of the various failure modes. Initiators can be both direct and indirect.

Step 5: Screening of the various initiators by consequence or frequency.

The complete process of identifying the initiating events using the master logic can be found in **7**. Potential initiating events derived from following this process are only summarised here in ***Table 6.2-2***. Many of these initiating events were screened out either on consequence or frequency. The reason for the screening can be found in **7**. The surviving list of initiating events used in further analysis is given in ***Table 6.2-3***. From the above list, only those events leading to breach of the pressure boundary will lead to any offsite consequences and hence their impact. The initiating events are grouped into functional categories according to safety function that would be challenged. The safety functions of relevance to the reactor core sources of radioactive material are heat removal, control of heat generation, and control of chemical attack.

Support system faults cannot influence the actual consequence of previously considered failure modes, but may result in increasing the frequency by which this failure mode may occur. The most likely support systems to affect these functions are the reactor protection instrumentation cooling water systems and electric power.

This assessment considers both internal and external initiating events. The list of external events consistent with that contained in **11** was used as a basis for the selection of external events. However, only one such event, earthquake, was considered in this study. Many other initiators were screened out because of site-specificity.

Table 6.2-2: SUMMARY OF ALL INTIATING EVENTS CONSIDERED

Uncontrolled Single Rod Withdrawal
Uncontrolled Group Rod Withdrawal
Uncontrolled Rod Insertion
Inadvertent SAS Insertion
Too Rapid SAS Removal
Top Reflector Drop
Earthquake
Aircraft Crash
External Flooding
Fuel Handling Loading Error
Mixing of Spheres
Inadvertent Bypass Valve Opening.
Inadvertent Group Rod Insertion.
Inadvertent Pump Down to HICS
Loss of Cooling Water
Station Blackout
Loss of Off-Site Power
Turbine Trip
Pressure boundary Break
Pipe Whip
Internal Missile
Internal Fire
Loss of forced cooling at Shutdown
Internal Flooding
Heat exchanger tube Leak
Helium Purification System Failure
Depressurised Flooding
Depressurised Air Ingress
Heavy Weight Drop
Pressure Relief System Failure
Primary pressure boundary break
Primary coolant leak
Fuel Handling Detection Error
Spent Fuel Compaction
Loss of Spent Fuel Cooling
Spent Fuel Tank Break
Loss of Used Fuel Cooling
Water Ingress
Used Fuel Tank Break
FHSS Boundary Break
FHSS Boundary Leak
HICS Boundary Break
HICS Boundary Leak
HPS Boundary Break
HPS Boundary Leak
HPS Failure
Waste Storage Tank Leak
Waste Storage Tank break
RCCS Water Chambers Leak
Loss of RCCS during nromal operation
Loss of RCCS during maintenance
HVAC Ducting Leak

Table 6.2-3: INITIATING EVENTS USED IN THE PBMR PRA

- Uncontrolled group rod withdrawal.
- Loss of power conversion unit transients.
    - Inadvertent by-pass valve opens.
    - Loss of cooling water (to pre-cooler, inter-cooler, and generator).
    - Station black-out.
    - Inadvertent pumpdown to HICS.
    - Turbine trip.
- PPB breaks
- Pressure coolant leaks.
- Earthquakes.

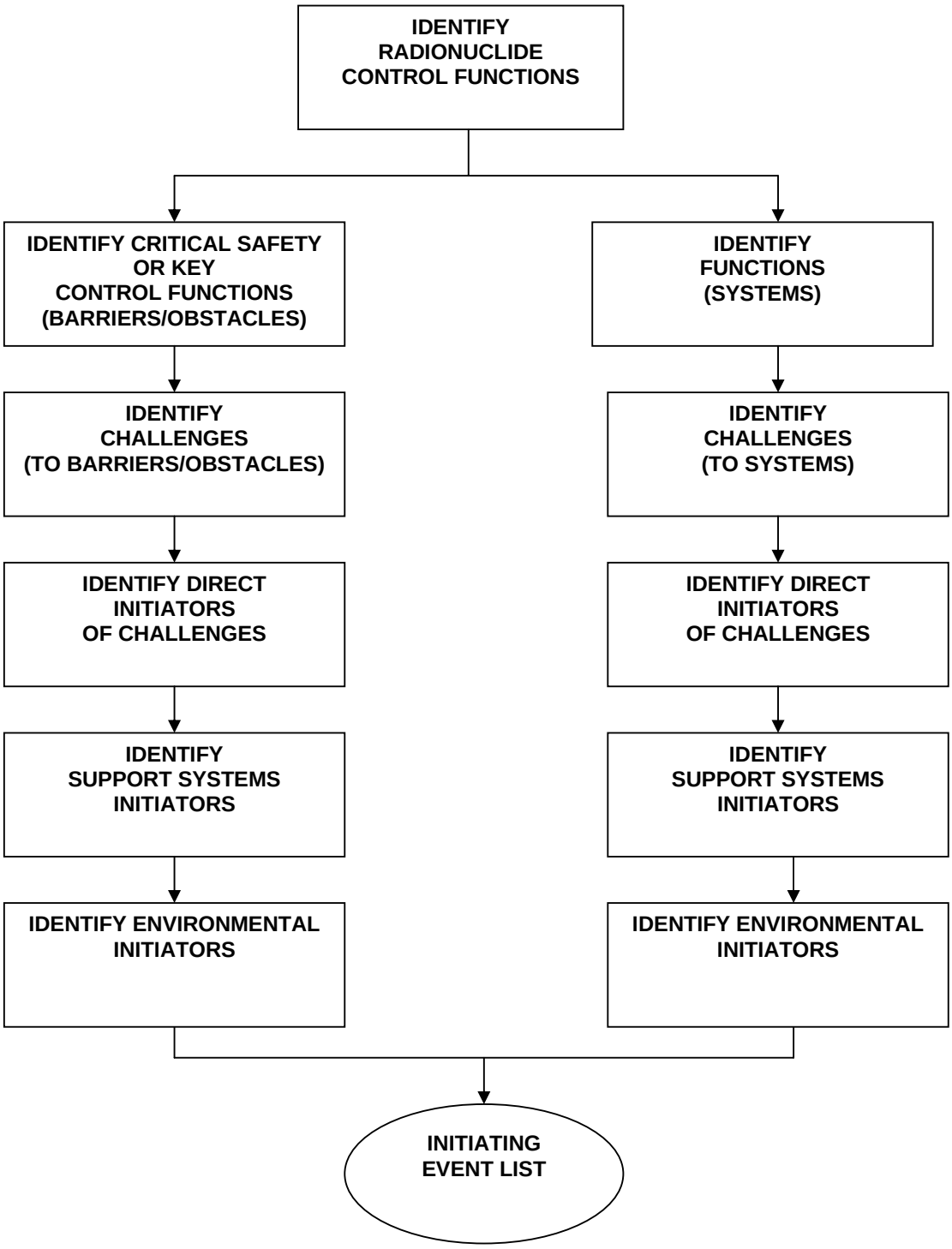


Figure 6.2-1: SYSTEMATIC BLOCK DIAGRAM APPROACH IDENTIFYING INITIATING EVENTS

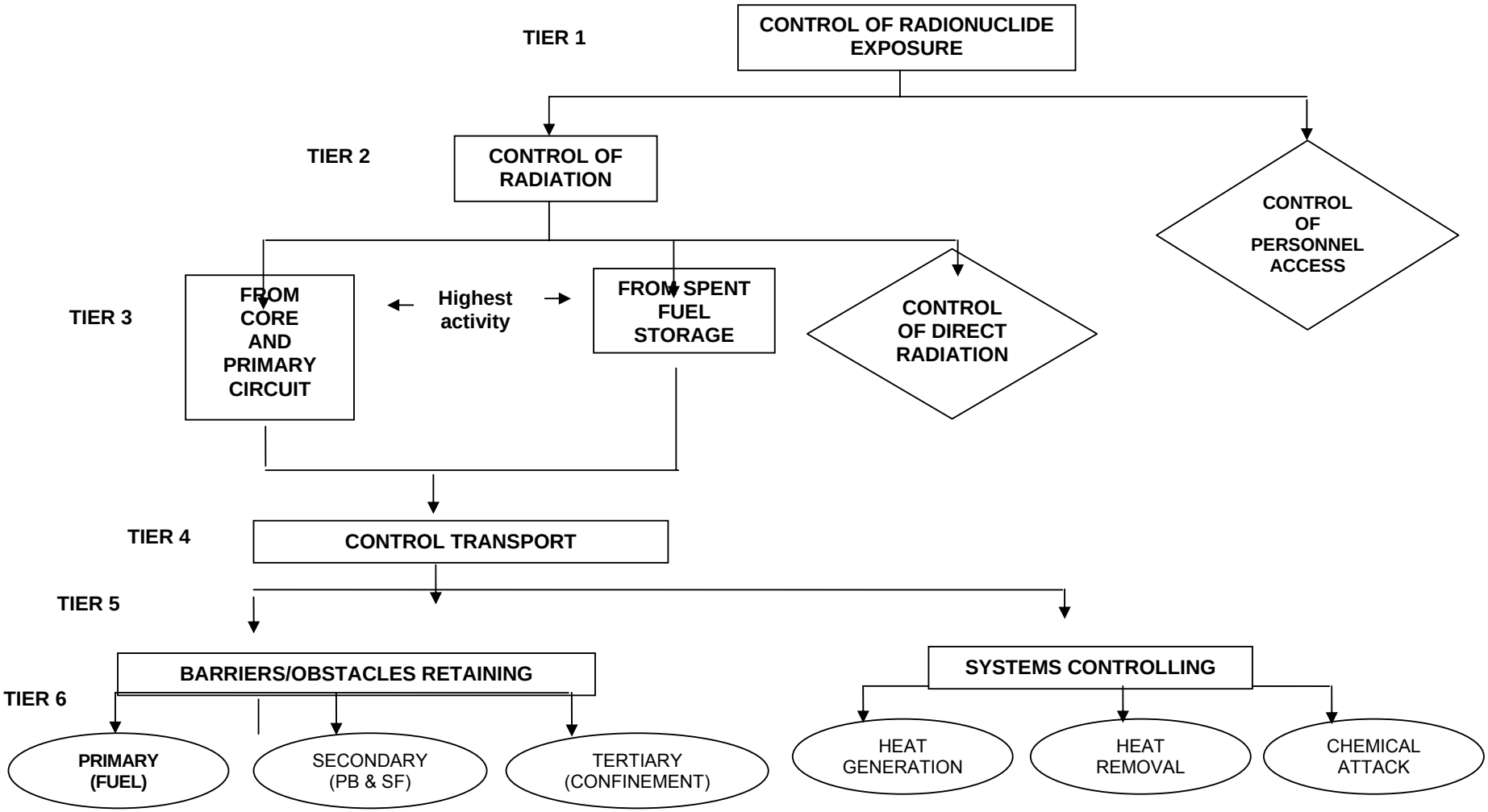


Figure 6.2-2: GENERIC LOGIC DIAGRAM – IDENTIFICATION CRITICAL RADIONUCLIDE CONTROL FUNCTIONS

Reference: SAR Rev 1 Section 6

6.0.8.2 Initiating Events Grouping

For each initiating event considered, an event tree must be constructed. Because the list of initiating event can be very long, resulting in a correspondingly long list of event trees, it is customary to group initiating events into functional groups which will have the same plant response. This reduces the number of event trees that need to be constructed. The initiating events are grouped into three distinct categories, namely;

- Transients.
- Pressure boundary leaks.
- Transients without control rod drop.

The initiating events considered in the current analysis and their groupings are given in ***Table 6.2-3***.

6.0.8.2.1 Transients

Two types of transients were considered, namely, those involving loss of power conversion unit and those resulting in rapid reactivity changes in the core. The loss of PCU transients can be caused by several initiators, namely:

- Inadvertent By-pass Valve Opens.
- Loss of Cooling Water to Pre-Cooler/Inter-Cooler and Generator.
- Station Blackout.
- Inadvertent Pump Down to HICS.

Transients involving changes in reactivity in the core are initiated by

- Uncontrolled Group rod withdrawal.
- Fuel Handling Loading Error
- Uncontrolled Group Rod Insertion.
- Inadvertent SAS Insertion.

Some initiating events may fall in more than one category. A choice was made as to which

category they should fall. For example, Uncontrolled Group Rod Insertion and Inadvertent SAS Insertion cause the Brayton cycle to terminate thus resulting in trip of the PCU. This events were forced into the reactivity transients group because they involve movement of reactivity control systems. In the PBMR, transients events do not result in any release to the environment. They are considered because they may lead to high temperatures resulting in additional fission products to be released from initially failed particles.

6.0.8.2.2 Pressure Boundary Leaks

These are events involving the rupture of the reactor coolant boundary including piping, valves, pressure vessel, and interconnecting systems resulting in leakage of helium from the PPB and into the confinement. The breaks are grouped indirectly according to rate of loss of coolant and directly according to break size. Three break sizes have been considered, namely small, medium, and large pressure boundary breaks. Small pressure boundary breaks are ruptures which are equivalent in area up to a 10 mm diameter pipe. Included in the small break category are breaks of the heat exchanger tubes. These have diameters of 15 mm. The size of the small breaks was chosen such that the releases from such leak sizes would be within the capacity of the confinement filtration system. Medium pressure boundary breaks are ruptures which are equivalent in area to a pipe of diameter between 10 and 65 mm inclusive. This size of leaks is too large for the capacity of the confinement filtration system and is thus relieved through the shaft vent. The vent is assumed to re-close after the initial puff, thus limiting delayed releases from the core. The large pressure boundary breaks have ruptures greater than 65 mm in diameter. This size breaks are assumed to preclude the rupture panel vent from re-closing. In this case, any subsequent release from the core is direct to the environment without filtration or release mitigation via vent re-closure.

6.0.8.2.3 Transients without rod drop

These are events which require the insertion of control rods but which do not materialise due to mechanical faults or malfunction of the control or protection system. These are not modelled as initiating events by themselves but are part of transient events. They are represented by branches in transient event trees where the rods fail to insert. These events do not result in any release to the environment.

Table 6.2-4: SUMMARY OF INITIATING EVENTS FOR THE PBMR

INITIATING EVENT GROUP	DESCRIPTION	INITIATING EVENTS IN THE GROUP
Group Rod withdrawal	**Events that challenge the reactor trip safety function**	**Group rod withdrawal**
Loss of Power conversion unit	**Events that challenge the reactor trip safety function with loss of forced cooling**	Inadvertent By-pass Valve Opens Loss of Cooling Water Station Blackout Turbine trip **Inadvertent Pump Down to HICS**
Small isolatable primary coolant leaks	**Primary coolant leak, equivalent in area to a 10mm diameter pipe or less which is isolatable**	**PPB leak in the HICS, FHSS**
Small non-isolatable primary coolant leaks	**Primary coolant leak, equivalent in area to a 10mm diameter pipe or less which is non-isolatable**	**PPB leak in the PMS**
Heat exchanger tube leaks	**Leaks in the heat exchanger tubes equivalent in area too much smaller than 15 mm diameter pipe.**	**Small leaks in the heat exchanger**
Heat exchanger tube breaks	**Breaks in the heat exchanger tubes equivalent in area to a 15 mm diameter pipe.**	**Tube rupture in the heat exchanger**
Medium isolatable pressure boundary breaks	**Primary pressure boundary break, equivalent in area to greater than 10mm to 65 mm diameter pipe which is isolatable**	**PPB break in the HICS, FHSS, FHS**
Medium non-isolatable pressure boundary breaks	**Primary pressure boundary break, equivalent in area to greater than 10mm to 65 mm diameter pipe or less which is non-isolatable**	**PPB break in the MPS**
Large isolatable pressure boundary breaks	**Primary pressure boundary break, equivalent in area to greater than 65 mm diameter pipe which is isolatable**	**PPB break in the HICS**
Large non-isolatable pressure boundary breaks	**Primary pressure boundary break, equivalent in area to greater than 65 mm diameter pipe or less which is non-isolatable**	**PPB break in the MPS**
Large Earthquake	**Earthquake**	**Safe shutdown earthquake** **Beyond safe shutdown earthquake**

6.0.9 EVENT SEQUENCE MODELLING AND QUANTIFICATION

This section presents the event tree models developed for each of the initiating event groups discussed in the preceding sections. Event trees define the sequences leading to fuel degradation. While extensive fuel damage is precluded in the PBMR design, additional fission products may be released from defective fuel particles if core temperatures are sufficiently elevated thus increasing the risk of exposure to the public in the event of a pressure boundary rupture. The event trees address safety functions required to prevent fuel degradation. There are three safety functions and are defined as follows:

- Reactivity control

  This function is modeled by terminating the fission process. This is achieved by inserting the control material (control rods or SAS) into the core. The node modeled is labeled RCS/RSS.

- Heat removal

  This function is modeled by including a forced cooling node on the event tree. Force cooling can be fulfilled by either the SBS or the CCS. The SBS is, however, the system of choice. Only when it is unavailable is the CCS required to perform the same function. In the absence of the SBS and CCS systems, the RCCS is relied upon to remove heat from the reactor cavity and thus cool the core. Three nodes model heat removal and these are SBS, CCS, and RCCS.

- Control of chemical attack

  This function is modeled by preventing water and air ingress into the reactor. This is achieved by preventing systems such as the SBS and CCS from operating in the event of non-isolatable breaks.

6.0.9.1 Event Tree Models and Nodes

- The event tree models for each initiating event are shown in
-
-
- Figure 6.2-3 to
-

Figure 6.2-14. These figures also show quantified sequences. Sequences with frequencies below the truncation limit of 10^{-8} per year are screened out, although they appear on the trees. The top events (branches) modeled for each event tree are shown in ***Table 6.2-5*** and are described below.

RCS/RSS

This event tree node addresses whether the control rods or the reserve shutdown spheres are successfully inserted into the core to stop the fission process. These do not necessarily have to be dropped in like in the case of a scram but can be driven in. This is because the event progression is generally very slow and therefore does not require any quick action.

SBS

This node models the availability of the SBS to provide forced circulation of helium and heat removal via the pre-cooler and inter-cooler to the ACS in order to remove decay heat in the event that the Brayton cycle has stopped.

CCS

This node fulfils the same objective as the SBS using independent means of helium circulation and heat removal. However, it requires at least 5 hours to condition before it becomes effective. The CCS is the preferred alternative to the SBS.

RCCS

This node models heat removal from the reactor cavity. Both passive and active modes are modelled. Also included is an operator action that is required in the event that the RCCS water level has reduced in which case the operator can refill them with water.

HEDETECT

This event tree node addresses whether a PPB leak is detected or not. In the event that the leak is not detected, helium will continue to be released into the confinement.

ISOLATION

This node models whether a PPB leak has been isolated or not. Isolation will allow for forced cooling to be used to cool the core while at the same time trying to fix the break without going to

cold shutdown.

HVAC

The releases from small leaks are within the capacity of the confinement filtration system. Any release, whether the break was detected and isolated or not, is discharged through the HVAC to the atmosphere.

DEISOLATE

This node is necessary as the HICS would need to be de-isolated in order to start pumping gas into one of its tanks.

PUMPDOWN

In the event of a non-isolatable leak, this top event models the function in which the helium pumped down to 40% storage so as to limit the amount of circulating activity that would be released into the confinement and to reduce the driving force for any fission products associated with the delayed release from the fuel.

COMPRESSOR

The pumpdown of the remaining 40% of the helium gas is done though the use of compressors. This node models their availability.

Table 6.2-5: TOP EVENTS MODELED IN THE EVENT TREES

Event Tree top event	Loss of PCU				Group Rod withdrawal	Primary Coolant Breaks					
	Loss of cooling water	Inadvertent by-pass valve opening	Station black-out	Turbine trip		Small Isolatable Break	Small Non-isolatable breaks	Heat exchanger tube leak	Heat exchanger tube rupture	Medium or large isolatable break	Medium or large non-isolatable break
PCU TRIP					X						
RCS/RSS	X	X	X	X	X		X	X	X	X	X
SBS		X		X	X					X	
CCS	X	X		X	X			X		X	
RCCS	X	X	X	X	X		X	X	X	X	X
HDETECT						X	X	X	X	X[2]	
ISOLATION						X	X				
HVAC						X	X	X	X		
DAMPER										X	X
DEISOLATE							X				
PUMPDOWN							X	X	X		
COMPRESSOR							X	X	X		
EARTHQUAKE											

[2] This node applies to large isolatable breaks only.

6.0.9.2 Fault Tree Development

Sequences were modelled and quantified using the PRA software known as Risk Spectrum. Top events of the event trees were quantified using applicable fault trees. The top events requiring fault trees are given below:

- Insertion of RCS and RSS.
- Core heat removal using SBS.
- Core Heat Removal using CCS.
- Core heat removal using RCCS.
- HICS and FHSS isolated.
- HICS de-isolated.
- PCU trip.
- Helium detected via room and HVAC detectors.
- MPS pumpdown to 40%.
- Compressors pumpdown balance of MPS.
- Confinement filter/vent open.
- Damper in the explosion relief shaft re-closes.

All the top events used required fault trees.

6.0.9.2.1 Component Failure Data

Since there is no operating PBMR plant yet, no plant specific failure data exist for systems and components. The failure data used in the PRA were obtained from a variety of source. A combination of data from LWR **12**, industrial sources and earlier gas-cooled reactors **9** were used. The pipe failure data came from **12**. The database used in the PRA is described in **7**.

When selecting appropriate data to use in the PRA, cognisance was taken of the different operating environments of particular components and systems. The most appropriate reliability data available were used. The reliability data used provides information on failure modes for systems and components, failure frequencies (λ, 1/h), demand failures (q, 1/demand), repair times (τ, h), and common mode beta factors (β).

6.0.9.2.1.1 Common Cause Considerations

Phase two PRA has considered failures of two or more redundant components which might arise from the same cause. The majority of components in the PBMR design are dominated by passive ones and so are not susceptible to common cause failure. However, there are active systems employing redundancy that would be susceptible to such failures. The PRA uses Multiple Greek Letter method to estimate effects of common mode. The Multiple Greek Letter approach uses parameters beta, gamma and delta.

6.0.9.2.1.2 Human Reliability

Currently, no procedures exist describing plant operations, maintenance interfaces, design of control systems, etc. All of these are required in order to perform human reliability analysis. However, the certain obvious human actions were incorporated in the event. Some examples of these are the manual insertion of reactivity control material (control rods and RSS), pumpdown of helium from the primary circuit to the HICS, and isolation of the PPB leaks. Due to a lack of the PBMR-specific identified data, values used for Koeberg were used for the PBMR human reliability.

6.0.9.3 Licensing Basis Events

The sequences generated by the events trees are used as a basis for selecting the licensing basis events. Only those sequences with point frequencies greater than 10^{-8} per year form part of the LBE if they are in the design basis region or they result in radioactivity releases. The analyses of these LBE are presented. LBE for the basis of the radiological assessment for purposes of demonstrating compliance with regulatory requirements.

Small isolatable pressure boundary breaks	Helium detected via room and HVAC detectors	HICS and FHSS isolated	Reactor building filter/vent open				
IE-DPE01.1	HEDETECT-DPE01.1	ISOLATION-DPE01.1	HVAC-DPE01.1	No.	Freq.	Conseq.	Code
				1	2.12E-02		
				2	4.04E-05		ISOLATION-DPE01.1
				3	4.06E-07		HEDETECT-DPE01.1

Figure 6.2-3: SMALL ISOLATABLE PRESSURE BOUNDARY BREAK EVENT TREE

Small non-isolatable PPB breaks	Helium detected via room and HVAC detectors	HICS and FHSS isolated	Insertion of RCS and RSS	HICS de-isolated	MPS pumped down to 40%	Compressor pumps down balance of MPS	Decay heat removal using Reactor Cavity Cooling System	Reactor building filter/vent open				
IE-DPE01.2	HEDETECT-DPE01.2	ISOLATION-DPE01.2	RCS/RSS-DPE01.2	DEISOLATE-DPE01.2	PUMPDOWN-DPE01.2	COMPRESSOR-DPE01.2	RCCS-DPE01.2	HVAC-DPE01.2	No.	Freq.	Conseq.	Code
									1	4.10E-02		
									2	6.87E-09		RCCS-DPE01.2
									3	1.71E-02		COMPRESSOR-DPE01.2
									4	5.42E-09		COMPRESSOR-DPE01.2-RCCS-DPE01.2
									5	8.29E-05		PUMPDOWN-DPE01.2
									6	1.96E-11		PUMPDOWN-DPE01.2-RCCS-DPE01.2
									7	2.29E-10		DEISOLATE-DPE01.2
									8	5.33E-17		DEISOLATE-DPE01.2-RCCS-DPE01.2
									9	5.51E-08		RCS/RSS-DPE01.2
									10	7.07E-05		ISOLATION-DPE01.2
									11	1.20E-11		ISOLATION-DPE01.2-RCCS-DPE01.2
									12	2.94E-05		ISOLATION-DPE01.2-COMPRESSOR-DPE01.2
									13	9.54E-12		ISOLATION-DPE01.2-COMPRESSOR-DPE01.2-RCCS-DPE01.2
									14	7.53E-06		ISOLATION-DPE01.2-PUMPDOWN-DPE01.2
									15	3.02E-12		ISOLATION-DPE01.2-PUMPDOWN-DPE01.2-RCCS-DPE01.2
									16	1.05E-10		ISOLATION-DPE01.2-RCS/RSS-DPE01.2
									17	7.85E-07		HEDETECT-DPE01.2

Figure 6.2-4: SMALL ISOLATABLE PRESSURE BOUNDARY BREAK EVENT TREE

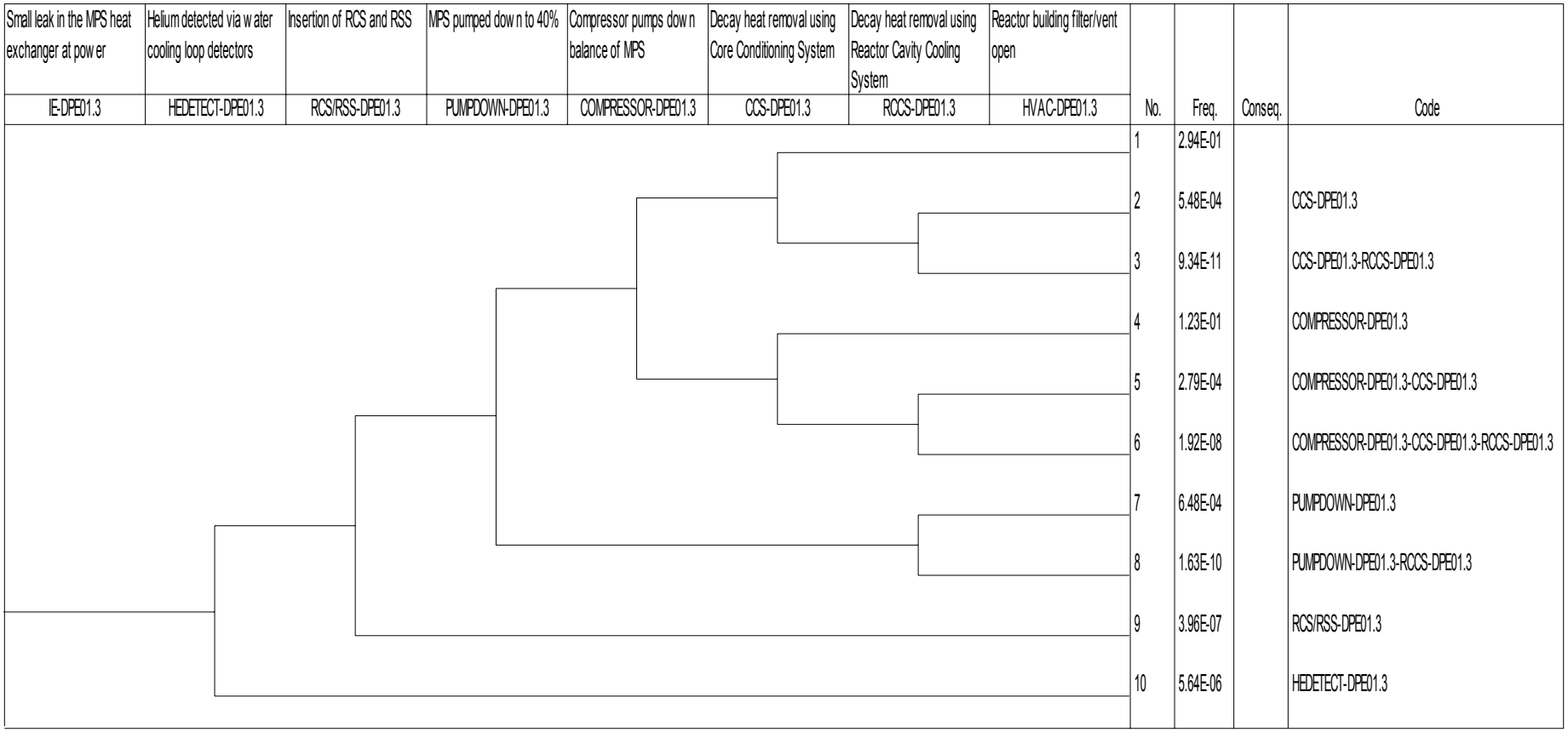


Figure 6.2-5: HEAT EXCHANGER TUBE LEAK EVENT TREE

Medium isolatable PPB breaks	Insertion of RCS and RSS	HICS and FHSS isolated	Decay heat removal using Start-up Blower System	Decay heat removal using Core Conditioning System	Decay heat removal using Reactor Cavity Cooling System	Damper in the explosion relief shaft re-closes				
IE-DPE02.1	RCS/RSS-DPE02.1	ISOLATION-DPE02.1	SBS-DPE02.1	CCS-DPE02.1	RCCS-DPE02.1	DAMPER-DPE02.1	No.	Freq.	Conseq.	Code
							1	2.15E-04		
							2	3.81E-06		SBS-DPE02.1
							3	1.86E-08		SBS-DPE02.1-CCS-DPE02.1
							4	1.42E-11		SBS-DPE02.1-CCS-DPE02.1-RCCS-DPE02.1
							5	4.09E-07		ISOLATION-DPE02.1
							6	8.31E-14		ISOLATION-DPE02.1-RCCS-DPE02.1
							7	1.36E-11		RCS/RSS-DPE02.1

Figure 6.2-6: HEAT EXCHANGER TUBE RUPTURE

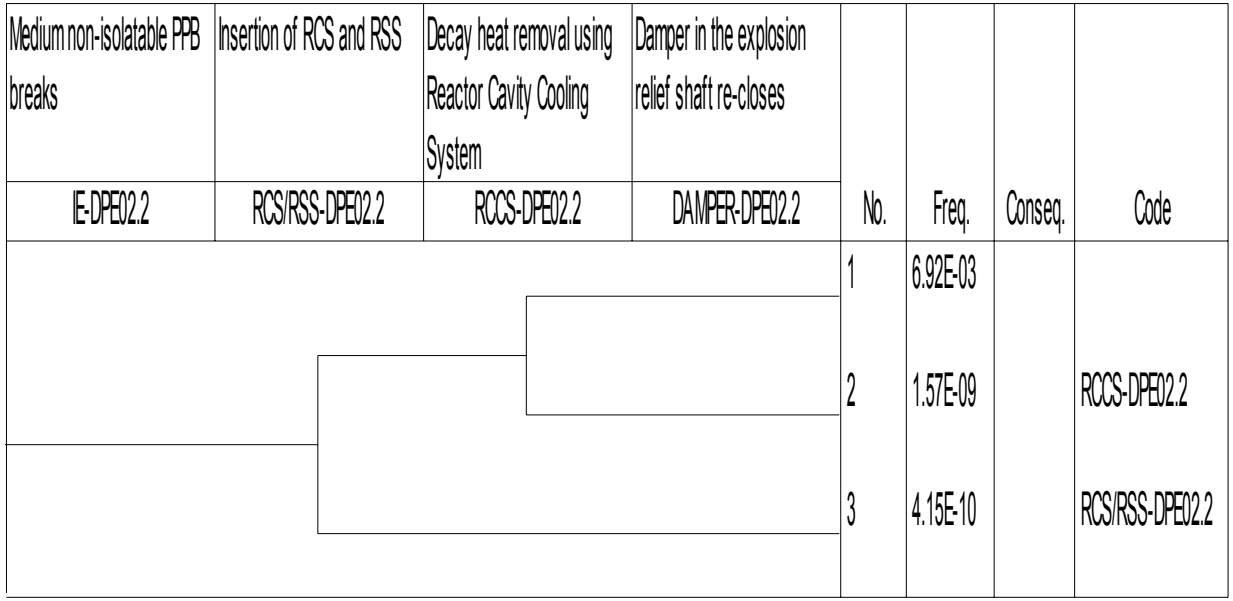
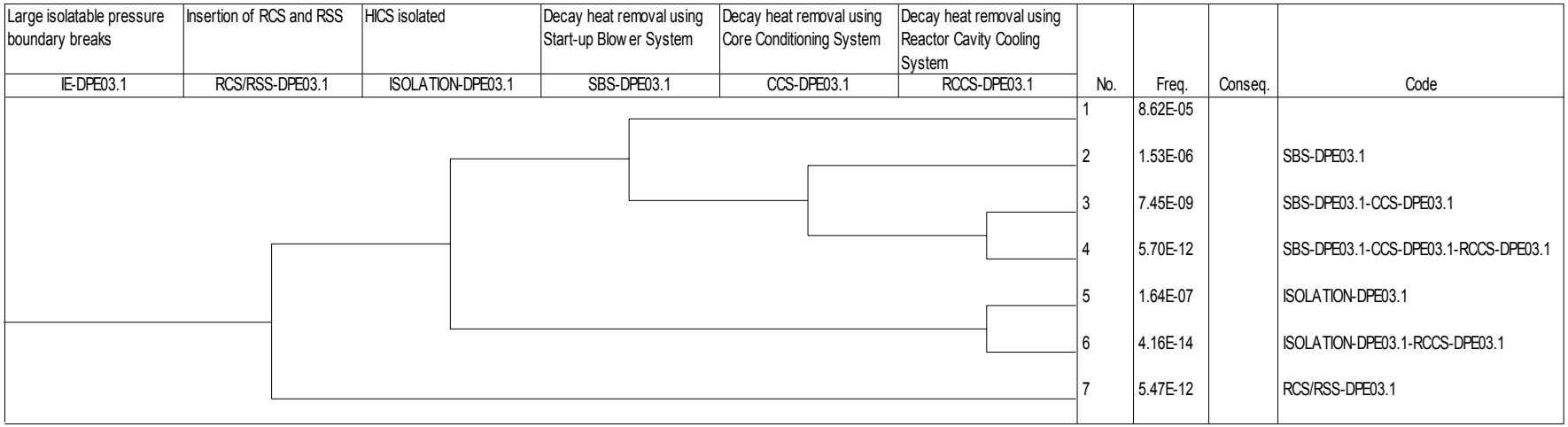
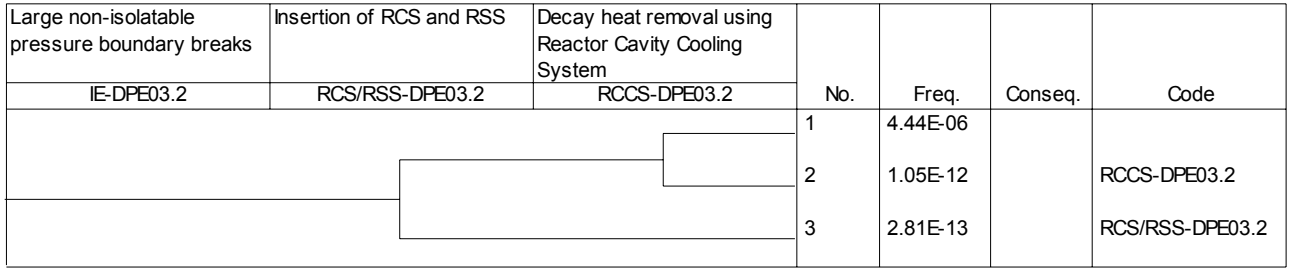


Figure 6.2-7: MEDIUM NONISOLATABLE PRESSURE BOUNDARY BREAK EVENT TREE



Large isolatable pressure boundary breaks	Insertion of RCS and RSS	HICS isolated	Decay heat removal using Start-up Blower System	Decay heat removal using Core Conditioning System	Decay heat removal using Reactor Cavity Cooling System	No.	Freq.	Conseq.	Code
IE-DPE03.1	RCS/RSS-DPE03.1	ISOLATION-DPE03.1	SBS-DPE03.1	CCS-DPE03.1	RCCS-DPE03.1				
						1	8.62E-05		
						2	1.53E-06		SBS-DPE03.1
						3	7.45E-09		SBS-DPE03.1-CCS-DPE03.1
						4	5.70E-12		SBS-DPE03.1-CCS-DPE03.1-RCCS-DPE03.1
						5	1.64E-07		ISOLATION-DPE03.1
						6	4.16E-14		ISOLATION-DPE03.1-RCCS-DPE03.1
						7	5.47E-12		RCS/RSS-DPE03.1

Figure 6.2-8: LARGE ISOLATABLE PRESSURE BOUNDARY BREAK EVENT TREE



Large non-isolatable pressure boundary breaks	Insertion of RCS and RSS	Decay heat removal using Reactor Cavity Cooling System	No.	Freq.	Conseq.	Code
IE-DPE03.2	RCS/RSS-DPE03.2	RCCS-DPE03.2				
			1	4.44E-06		
			2	1.05E-12		RCCS-DPE03.2
			3	2.81E-13		RCS/RSS-DPE03.2

Figure 6.2-9: LARGE NON-ISOLATABLE PRESSURE BOUNDARY BREAKS

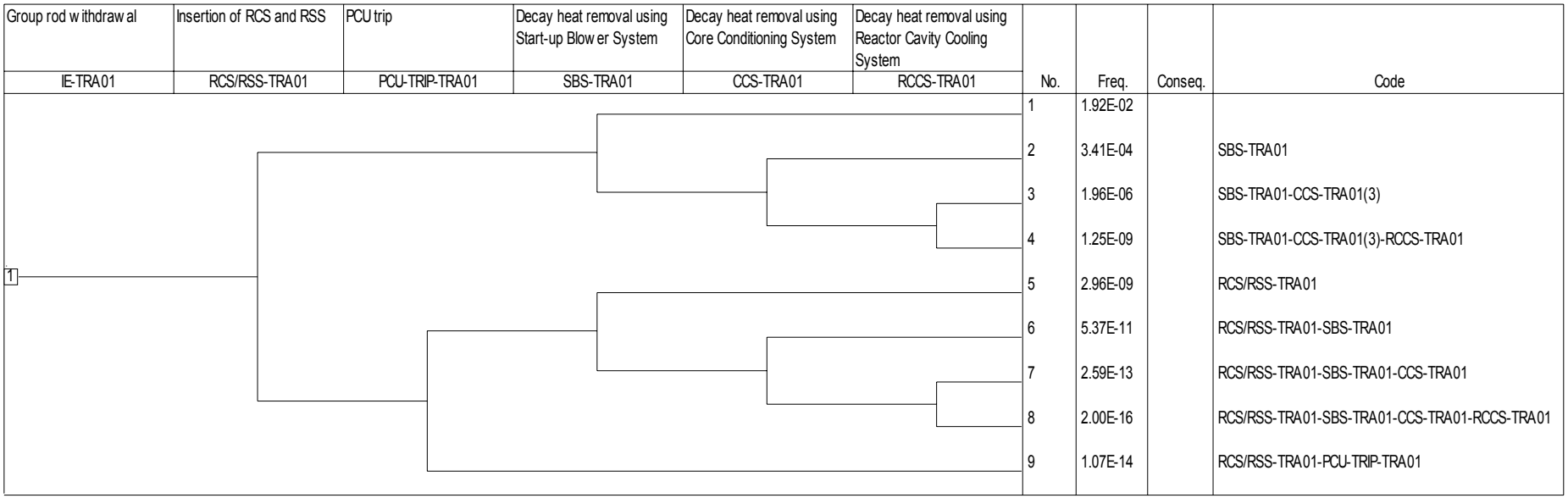


Figure 6.2-10: GROUP ROD WITHDRAWAL EVENT TREE

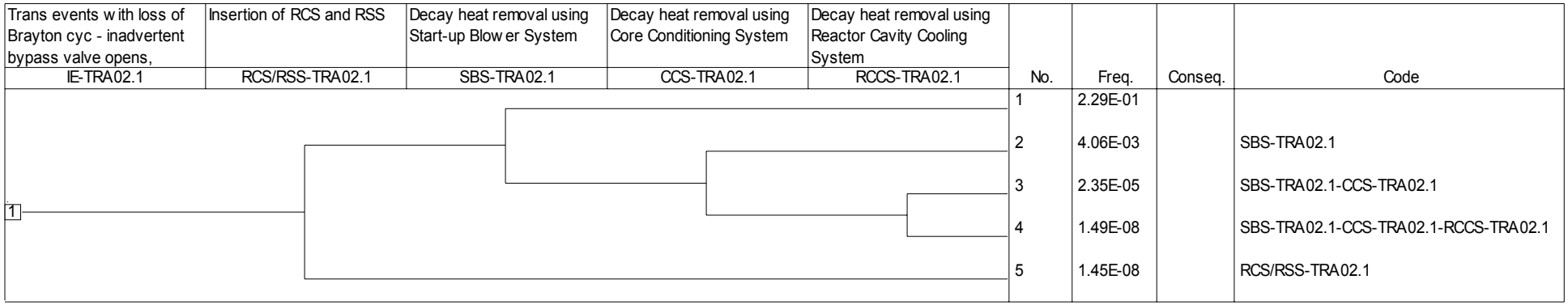


Figure 6.2-11: LOSS OF PCU EVENT TREE DUE TO INADVERTENT BY-PASS VALVE OPENING

Reference: SAR Rev 1 Section 6

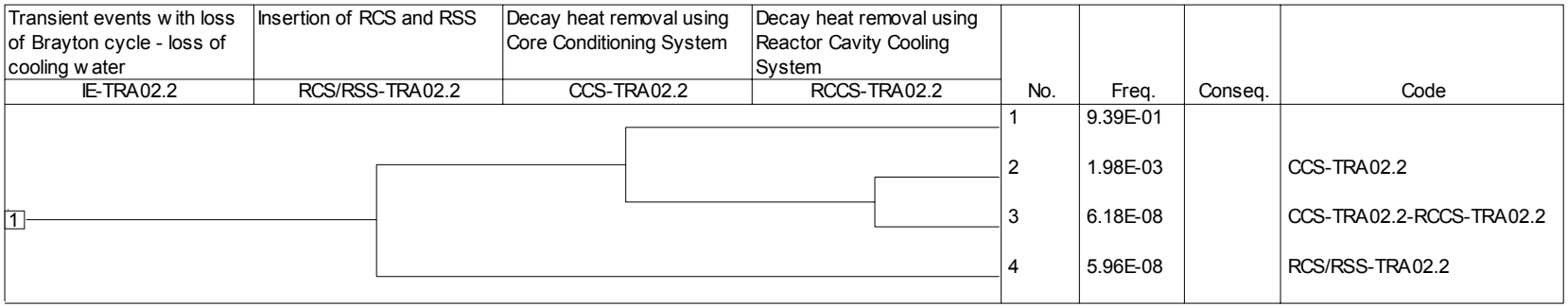


Figure 6.2-12: LOSS OF COOLING WATER EVENT TREE

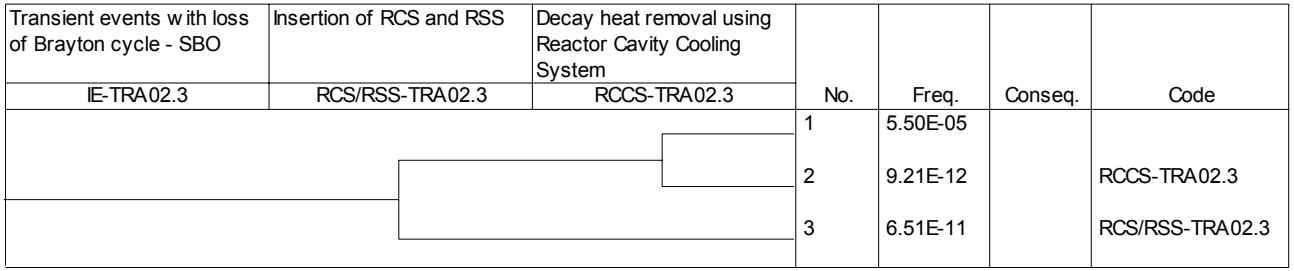
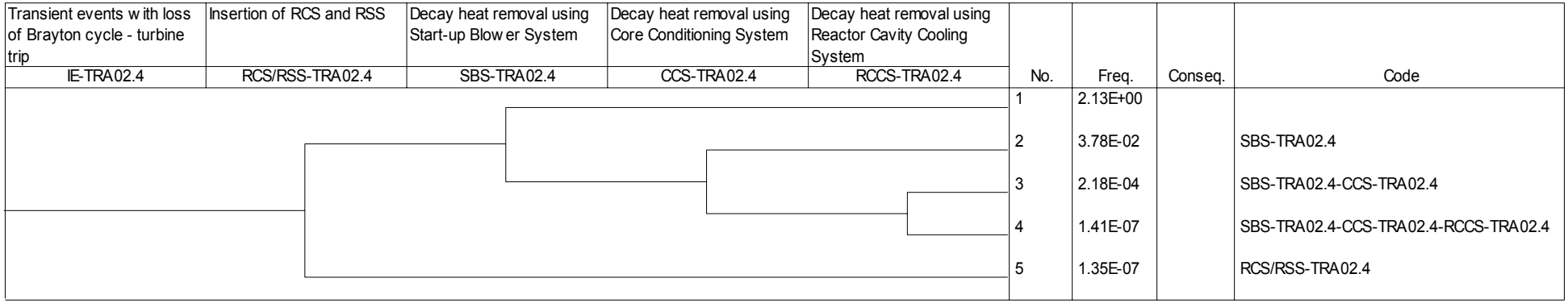


Figure 6.2-13: STATION BLACKOUT EVENT TREE



Transient events with loss of Brayton cycle - turbine trip	Insertion of RCS and RSS	Decay heat removal using Start-up Blower System	Decay heat removal using Core Conditioning System	Decay heat removal using Reactor Cavity Cooling System				
IE-TRA02.4	RCS/RSS-TRA02.4	SBS-TRA02.4	CCS-TRA02.4	RCCS-TRA02.4	No.	Freq.	Conseq.	Code
					1	2.13E+00		
					2	3.78E-02		SBS-TRA02.4
					3	2.18E-04		SBS-TRA02.4-CCS-TRA02.4
					4	1.41E-07		SBS-TRA02.4-CCS-TRA02.4-RCCS-TRA02.4
					5	1.35E-07		RCS/RSS-TRA02.4

Figure 6.2-14: TURBINE TRIP EVENT TREE

6.0.10 CONSEQUENCE ASSESSMENT

6.0.10.1 Introduction

The last step of the PRA is the assessment of consequences due to various event sequences and demonstrating compliance with regulatory criteria. The releases and their frequencies of occurrence form the basis for assessing the radiological consequences and risk to the public. The events considered here are limited to those involving breach of the pressure boundary as only these will result in any radiological exposure of the public.

To perform consequence assessment, a large amount of input data is required. Generally, the following main types of data must be specified:

- source term data;
- meteorological data;
- population distribution data; and
- agricultural production data.

These are discussed in more detail in following sections.

It is customary to group event sequences with similar consequences into groups called release categories. Each release category is characterized by a source term describing the quantity of radioactive material released, release time and duration, release frequency, release height, and thermal energy of the released gas. For each release category, a distribution of consequences is calculated. There are various endpoints for which distributions of consequences may be calculated. This assessment looks at dose and risk consequences to the public. The overall risk of the PBMR plant is obtained by summing over the distribution of consequences for each release category taking into account its frequency of occurrence. This risk is presented as a function of distance from the reactor.

Radiological consequences of each release are evaluated using the NRPB code PC COSYMA version 2.01. A description of PC COSYMA is presented.

6.0.10.2 PC COSYMA

Radiological consequences were evaluated using PC COSYMA version 2.01, which was released in 1995. The overall methodology of PC COSYMA has been described in detail in the PC COSYMA Report, **21**. Only a brief description of important parts is discussed in this section.

PC COSYMA is a highly modularized code, with each module addressing a particular part of the analysis. The various modules describe the transfer of released activity through the atmosphere, the dose and health consequences to the public, and also the economic impact of countermeasures that may be applied to mitigate the radioactive release. The modules are interlinked and can be run in combination by the system.

PC COSYMA describes consequences of accidental releases as end-points. The major end-points which can be calculated by the system are:

- concentration and deposition of nuclides at selected points;
- doses received at various distances away from site;
- numbers of early and late fatal and non-fatal health effects;
- individual and collective risk of early and late health effects;
- amounts of food banned; and
- economic costs, including contribution from application of countermeasures and health effects.

The first end-point provides information on the air concentration of activity in the air, and deposition of nuclides at various selected distances. Information on doses accumulated by different organs can also be obtained at each distance considered. The doses can be short-term or long-term. Short-term doses in this evaluation are doses accumulated over a maximum of 30 days, while long-term doses assume a dose integration period of 50 years. Detailed information on contributions to doses by different nuclides and different exposure pathways can also be obtained. Numbers of health effects, both long-term and short-term, can also be determined. In addition, Complementary Cumulative Distribution Functions (CCDF) can also be calculated. Of interest to this evaluation, are CCDF for health effects. The CCDF is the probability that a given consequence can be greater than or equal to a particular value, and is used to demonstrate compliance with the bias against larger accidents criterion of the NNR.

Health effects due to exposure to radiation are of two types, namely:

- early health effects; and
- late health effects.

Some health effects are fatal, while others are not. Early health effects are those which appear within a few weeks after the accident. They can only occur following very large releases over short

periods of time as there are dose thresholds that must be exceeded to produce these effects. This part of the calculation is not relevant to the PBMR as the range of possible dose consequences are one to two orders of magnitude lower than these thresholds. Early health effects can be fatal, but largely include loss or impairment of various vital organs. Late health effects, on the other hand, can surface even years after the event has occurred. They are largely cancers, fatal and non-fatal, and also include hereditary effects in the descendants of the exposed public. Further information on health effects can be obtained in terms of the risks of such effects. There are two types of risks of health effects, namely individual risk of early and late health effects at various distances from site.

For purposes of this assessment, the results presented only include:

- doses at distances away from the reactor;
- short-term and long-term risk of morbidity and mortality for various health effects; and
- CCDF for early and late health effects.

In a severe event, information on amounts of food produced in the area near site that may be banned from consumption can also be obtained. Also, after an accidental release, protective measures are generally taken to mitigate the release. These measures are referred to as countermeasures. These vary from distribution of stable iodine tablets, to relocation of the affected population. All forms of countermeasures have been excluded from the present assessment.

Two types of calculations can be carried out using PC COSYMA, namely probabilistic and deterministic calculations. Probabilistic calculations of consequences take into consideration the whole spectrum of atmospheric conditions, while deterministic calculations consider a single meteorological condition. Both of these have been performed to demonstrate compliance with NNR regulations.

The results of probabilistic calculations are in terms of conditional probability distributions of the end-points. The conditional probability of a consequence is the probability of the consequences, assuming that the event has happened. In other words, the probabilities obtained from PC COSYMA as output do not take into account the frequency of occurrence of a particular event. To obtain actual probabilities of consequences, the conditional probabilities must be multiplied by the frequency of occurrence of the particular event type.

6.0.10.2.1 Assumptions made in PC COSYMA

To perform the consequence assessment, the following conservative assumptions were made:

- People are assumed to be outdoors at the time of the event, and remain there for a period of 30 days which takes no account of emergency intervention. This is conservative since some emergency intervention would be undertaken in the real event.
- That the event takes place for a core at steady state and after the reactor had been operating for 32 years. This leads to maximum plated out activity. In reality, the event can occur at any time during the life of the plant.

6.0.10.2.2 Source Term Data

To calculate consequences of events, a source term is required. The source term describes the strength and the timing of the release. It is generally characterized by the following parameters:

- release fraction;
- frequency of occurrence;
- energy of release;
- release height; and
- timing of release.
- Release duration
- Exposure period

6.0.10.2.2.1 Release fraction

The release fraction describes the fraction of the initial available inventory that is released to the atmosphere. In general, to determine the release fractions, one must take into account transport, deposition and release in the primary system; transport, deposition, and release in the confinement system. The PRA has not considered any transport, deposition and release in the confinement system. It has been conservatively assumed that what is released from the PPB into the confinement system is released into the environment. No credit was taken for any hold-up in the confinement.

In Pressurized Water Reactor (PWR)-type reactors, the fraction of radioactive material released is usually specified with respect to core inventory at shutdown. For the PBMR, the approach is slightly different. Instead, for each release category, the amount of activity for each radionuclide available for release is determined. This is then specified as the source term available for release.

There are three sources of fission products in the PBMR core, namely, that contained in the helium circuit during normal operation, that plated-out in various surfaces in the primary circuit, including dust, and that contained in defective particles that may be released during heat-up events. Events may have one, two or all of these sources present.

6.0.10.2.2.1.1 Initial Core inventory and Activity in the primary circuit

It is assessed that no large-scale fuel damage occurs at fuel temperatures below fuel failure temperature that can lead to severe fission products release into the environment. There is extensive experimental evidence to support the expected performance of the intact particles. However, during normal operation, some fission gases may escape the fuel particles and reside in the helium coolant. The activity is generally released into the helium coolant from the fuel due to the following processes:

- diffusion of fission products through defective coatings;
- coating failure; and
- desorption (desorption refers to the remobilization of fission products from surfaces) of activity.

The activity available for release into the atmosphere will thus be the sum of the activity due to lifted off plate-out and activity contained in the helium circuit. In some cases, when local fuel temperatures are increased during certain conditions, additional activity from initially failed particles may be released. This is estimated from the fraction of initially failed particles and initial core inventory. Delayed releases will occur for events where forced cooling is not available. Releases form initially intact particles due to temperature increases is negligible due to the design basis of the reactor.

The initial core inventory and activity contained in the helium circuit are given in ***Table 6.2.A-5***. This table only shows the activity of radiologically significant fission products. The activities of any other radionuclides are far lower than these because of the high fission product retention capacity of the fuel coating. The remaining fission products are not expected to appreciably increase the predicted overall plant safety risk results. The inventory presented above is for an equilibrium core assuming the reactor has been operating continuously at full power.

The activities presented in ***Table 6.2.A-5*** were derived by adjusting the results of the High Temperature Modular Reactor (HTR-Modul). The activities were adjusted by scaling the activities according to installed capacity. Plated-out activity is presented in ***Table 6.2.A-6***. This was derived for an equilibrium core at the end of life assuming the reactor has been operating continuously at full power. This is equivalent to 32 Effective Full Power Years (EFPY). Events in which there is no

forced cooling will have a delayed release from initially failed particles. Exact values of delayed activity release for the PBMR are not available. However, for the HTR-Modul, it was found that during depressurization event, peak temperatures are reached after many hours (32 hours) and even then only a small fraction, less than 2%, of fuel elements experiences this temperature **22**. Thus, it is assumed that additional activity from initially failed release is approximated by 2%. This is assumed to happen after 32 hours after the depressurization. However, since all the gas has been lost, the only driving force for this delayed release is gas expansion as a result of high temperatures. An even small fraction of the activity released from the core into the reactor building is released into the environment with the exception of noble gases. Other fission products will deposit on the colder structures in the reactor building. This has not been taken credit of.

6.0.10.2.2.2 Frequency of occurrence

The frequency of occurrence (or release frequency) of an accident sequence leading to a particular release of radioactivity into the atmosphere describes the expected frequency of that particular accident sequence occurring per year. The frequencies of occurrence of the various release categories are given in ***Table 6.2-13***.

6.0.10.2.2.3 Energy of release

The energy content of the release is usually calculated from the sensible heat of the released primary helium and from the decay heat of radionuclides. This generally depends on the time at which the release from the pressure boundary occurs. The release of this stored heat directly influences radiological consequences, as it may cause the plume to rise substantially into the atmosphere before significant dispersion takes place. This will result in a reduction of exposure of the population located downwind. Fission product transport calculations in the reactor building have not been performed yet. As a result calculated release energies are not available. A low energy value has been used which yield conservative results. A value for energy corresponding to decay heat 2 hours after fission reaction has stopped was used. This was 1.06 MW. In depressurization events, the release is immediate and so the thermal energy of the released gas will still be high, much higher than the decay heat value after 2 hours.

6.0.10.2.2.4 Release height

The height of release is also very important as it influences the exposure of the population near the plant. There are two predetermined release routes in the event of a depressurization event. For small pressure boundary leaks which are assumed to be within the capacity of the confinement filtration system, the release is filtered through the system and exhausted to the atmosphere at the top of the reactor. The exhaust point is at a height 23 m. The second scenario involves medium

releases. These have a dedicated shaft which interconnects with other cavities and vents at the roof, as in the small leak case. Larger breaks release is through same route as the medium releases except that the blast panels, located at roof which is at a height of 23 m, are assumed to not re-close after the initial release.

6.0.10.2.3 Timing of release

This describes the time after nuclear fission stops at which the release occurs. This is important, as it allows for activity reduction through radioactive decay.

6.0.10.2.3.1 Release Duration

This parameter describes the duration of the release into the atmosphere. It influences horizontal dispersion of the released material. PC COSYMA models releases as puffs lasting for one hour. This is conservative as most releases will last longer than one hour. All releases are modelled as short duration releases except for those which have delayed release, in which case they are represented as two puffs lasting one hour each but separated by the time at which the delayed release is supposed to start.

6.0.10.2.4 Release Categories

In consequence analysis, it is often necessary to group event sequences that have similar consequences, together. These groups of event sequences are called release categories. The grouping of event sequences is necessary, as it is impractical to calculate the consequence of each event sequence, because of the large number of event sequences involved. A set of event sequences for the PBMR was compiled. For the PBMR, six release categories have been identified and are denoted RC-1, RCF-1, RCF-2, RCP-1, RCPF-1, and RCFP-2. The release categories, together with the release frequencies, are given in ***Table 6.2-6*** . The release frequency of each release category in ***Table 6.2-7*** was obtained by summing the sequence frequencies for all sequences in each release category.

6.0.10.2.4.1 Description of Release Categories

<u>RC-1</u>

This category groups all small breaks which are isolatable. The small breaks are breaks with equivalent area as a 10 mm diameter pipe which are assumed to be within the capacity of the confinement filtration system. Thus the release to the environment is filtered through the system. Only circulating fission products are available for release. About 10% of the gas will have escaped before isolation.

RCF-1

This release category consists of all small leaks where the leak is not isolatable. Pumpdown is assumed and since this is a small pipe break no revolatization of plated-out fission products is assumed. However, there is a delayed released due to the slow depressurization. In this event about 50% of the gas is assumed to have been released before pumpdown is complete.

RCF-2

This release category consists of those events where the leak was not isolatable but for which no pumpdown is assumed. Since this is a small pipe break no revolatisation of plated-out fission products is assumed. However, there is a delayed released due to the slow depressurization. In this event about 100% of the gas will have been released.

RCP-1

This category consists of Medium and large breaks that have been isolated. 100% of the circulating gas is assumed released. The released is unfiltered and is exhausted through vent into the atmosphere. Only circulating activity and plate-out form part of the source term.

RCPF-1

This category consists of medium breaks which are not isolatable. The whole circulating inventory and 5% of plated out material is released immediately. The vent is assumed to reclose after the initial puff. Delayed release is assumed to occur after 32 hours.

RCPF-2

This category consists of large breaks which are not isolatable. The whole circulating inventory and 5% of plated out material is released. The vent is assumed to not reclose after the initial puff. Delayed release occurs after 32 hours.

Table 6.2-6: RELEASE CATEGORIES FOR THE PBMR

Release Category	Description	Source Term
RC-1	Small leak; Isolated; Filtered vent; immediate circulating inventory released, no plate-out[3]; 10% gas released	Circulating
RCF-1	Small leak; Not Isolated; with Pumpdown; Filtered vent; immediate circulating inventory released; delayed fuel release; 50% gas released	Circulating and Fuel
RCF-2	Small leak; Not Isolated; without Pumpdown; Filtered vent; immediate circulating inventory released; delayed fuel release; 100% gas released	Circulating and Fuel
RCP-1	Medium and large breaks; Isolated; Unfiltered vent; immediate circulating inventory released, with immediate circulating and plate-out released. In the event of a medium break 90% of the gas is released. 100% is assumed released in the case of large breaks.	Circulating and Plate-out
RCPF-1	Medium break; Not Isolated; Unfiltered vent; immediate circulating inventory released, with immediate plate-out and delayed release; Vent Reclosed; 100% gas released	Circulating and Plate-out and Fuel
RCPF-2	Large break; Not Isolated; Unfiltered vent; immediate circulating inventory released, with immediate plate-out and delayed release; Vent Open; 100% gas released	Circulating and Plate-out and Fuel

[3] Plate-out activity is assumed to include dust.

Table 6.2-7: LICENSING BASIS EVENTS FOR THE PBMR

LBE ID	Description	Frequency (/plant yr)	Cat.	Radionuclide Retention in Fuel			Release Characterization			
				Control of Heat Generation	Heat Removal	Control of Chemical Attack	Source	Timing from Core	Timing from PPB	Path from RB
LBE 1	**Loss of Power Conversion Unit**						**none**	**n/a**	**n/a**	**n/a**
1a	Loss of PCU with RCCS cooling	2.1E-03	B	temperature coefficient & RCS / RSS	convection, conduction, & radiation to RCCS	no challenge				
1b	Loss of PCU without RCS/RSS trip	1.5E-08	C	temperature coefficient	convection, conduction, & radiation to RCCS	no challenge				
1c	Loss of PCU without RCCS cooling	1.5E-08	C	temperature coefficient & RCS / RSS	convection, conduction, & radiation to RB/ground	reactor vessel integrity				
LBE 2	**Control rod group withdrawal**						**none**	**n/a**	**n/a**	**n/a**
2a	Control rod withdrawal with CCS cooling	3.4E-04	B	temperature coefficient & RCS / RSS	CCS (ACS)	no challenge				
2b	Control rod withdrawal with RCCS cooling	2.0E-06	B	temperature coefficient & RCS / RSS	convection, conduction, & radiation to RCCS	no challenge				
LBE 3	**Primary coolant leak with isolation**						**Circulating**	**n/a**	**Immediate**	**HVAC filter**

Reference: SAR Rev 1 Section 6

LBE ID	Description	Frequency (/plant yr)	Cat.	Radionuclide Retention in Fuel			Release Characterization			
				Control of Heat Generation	Heat Removal	Control of Chemical Attack	Source	Timing from Core	Timing from PPB	Path from RB
3a	Small primary coolant leak with SBS cooling	2.1E-02	A	temperature coefficient & RCS / RSS	SBS (ACS)	leak isolation				
3b	Small primary coolant leak with CCS cooling	3.6E-04	B	temperature coefficient & RCS / RSS	CCS (ACS)	leak isolation				
3c	Small primary coolant leak with RCCS cooling	1.8E-06	B	temperature coefficient & RCS / RSS	conduction & radiation to RCCS	leak isolation				
LBE 4	**Primary coolant leak without isolation with pumpdown**						**Circulating Fuel**	**n/a Delayed**	**Immediate Delayed**	**HVAC filter HVAC filter**
4a	Small primary coolant leak with failed isolation with pumpdown	1.1E-04	B	temperature coefficient & RCS / RSS	conduction & radiation to RCCS	negligible air				
4b	Heat exchanger tube leak with pumpdown	4.3E-02	A	temperature coefficient & RCS / RSS	conduction & radiation to RCCS	negligible water & air				
LBE 5	**Primary coolant leak without isolation without pumpdown**						**Circulating Fuel**	**n/a Delayed**	**Immediate Immediate**	**HVAC filter HVAC filter**
5a	Small primary coolant leak with failed isolation without pumpdown	1.2E-05	B	temperature coefficient & RCS / RSS	conduction & radiation to RCCS	negligible air				
5b	Heat exchanger tube leak without pumpdown	4.3E-03	B	temperature coefficient & RCS / RSS	conduction & radiation to RCCS	negligible water & air				

Reference: SAR Rev 1 Section 6

LBE ID	Description	Frequency (/plant yr)	Cat.	Radionuclide Retention in Fuel			Release Characterization			
				Control of Heat Generation	Heat Removal	Control of Chemical Attack	Source	Timing from Core	Timing from PPB	Path from RB
LBE 6	**Primary pressure boundary break with isolation**						**Circulating** **Plate-out**	**n/a** **n/a**	**Immediate** **Immediate**	**Unfiltered Vent** **Unfiltered Vent**
6a	Medium PPB break with isolation with SBS cooling	2.2E-04	B	temperature coefficient & RCS / RSS	SBS (ACS)	break isolation				
6b	Medium PPB break with isolation with CCS cooling	3.8E-06	B	temperature coefficient & RCS / RSS	CCS (ACS)	break isolation				
6c	Medium PPB break with isolation with RCCS cooling	1.9E-08	C	temperature coefficient & RCS / RSS	conduction & radiation to RCCS	break isolation				
LBE 7	**Primary pressure boundary break without isolation**						**Circulating** **Plate-out** **Fuel**	**n/a** **n/a** **Delayed**	**Immediate** **Immediate** **Delayed**	**Unfiltered Vent** **Unfiltered Vent** **Closed Vent**
7a	Medium non-isolatable PPB break	6.9E-03	B	temperature coefficient & RCS / RSS	conduction & radiation to RCCS	limited oxidation				
LBE 8	**Beyond design basis pressure boundary break with isolation**						**Circulating** **Plate-out**	**n/a** **n/a**	**Immediate** **Immediate**	**Unfiltered Vent** **Unfiltered Vent**
8a	Large PBB break with isolation with SBS cooling	8.6E-05	B	temperature coefficient & RCS / RSS	SBS (ACS)	break isolation				
8b	Large PBB break with isolation with CCS cooling	1.5E-06	B	temperature coefficient & RCS / RSS	CCS (ACS)	break isolation				

Reference: SAR Rev 1 Section 6

LBE ID	Description	Frequency (/plant yr)	Cat.	Radionuclide Retention in Fuel			Release Characterization			
				Control of Heat Generation	Heat Removal	Control of Chemical Attack	Source	Timing from Core	Timing from PPB	Path from RB
LBE 9	**Beyond design basis pressure boundary break without isolation**						**Circulating** **Plate-out** **Fuel**	**n/a** **n/a** **Delayed**	**Immediate** **Immediate** **Delayed**	**Unfiltered Vent** **Unfiltered Vent** **Unfiltered Vent**
9a	Large PPB break without isolation with RCCS cooling	4.6E-06	B	temperature coefficient & RCS / RSS	conduction & radiation to RCCS	TBD				
LBE 10	**Large earthquake**						**none**	**n/a**	**n/a**	**n/a**
10a	Safe shutdown earthquake (.3g) with SBS cooling	1.0E-04	B	temperature coefficient & RCS / RSS	SBS (ACS)	PPB integrity				
10b	Safe shutdown earthquake with CCS cooling	1.8E-06	B	temperature coefficient & RCS / RSS	CCS (ACS)	PPB integrity				
10c	Beyond SSE (.4g) with intact PPB	1.0E-06	B	temperature coefficient & RCS / RSS	CCS (ACS)	PPB integrity				
LBE 11	**Large earthquake with primary pressure boundary break**						**Circulating** **Plate-out**	**n/a** **n/a**	**Immediate** **Immediate**	**Unfiltered Vent** **Unfiltered Vent**
11a	Beyond SSE (.4g) with PPB break	1.0E-08	C	temperature coefficient & RCS / RSS	CCS (ACS)	TBD				

Reference: SAR Rev 1 Section 6

6.0.10.2.5 Meteorological Data

Probabilistic runs require large amounts of meteorological data. It is standard practice to use hourly atmospheric data over a period of two years. The current data being used are the data used by Koeberg Nuclear Power Station (KNPS). The meteorological data is based on data for the years 1998 and 1999 **23**. The meteorological data file contains 17252 hours of data.

The meteorological parameters presented in these data are as follows:

- wind direction;
- atmospheric stability category;
- rainfall rate;
- wind speed; and
- mixing layer depth.

The meteorological data file used for the current evaluation can be found in **23**.

For probabilistic calculations, the large amount of data involved makes it impractical to calculate consequences for each hour of data. It is common practice to perform the calculations for a smaller representative set of weather data. To achieve this, a sampling technique called stratified sampling is used. In stratified sampling, weather data that will result in similar consequences are grouped together under one class. Several of these groups are created. Weather sequences are then randomly selected for analysis by randomly selecting one or more hours per group. In this manner, a full range of weather conditions is considered. The sampling scheme used for this evaluation is the default scheme that comes with PC COSYMA. The sampling scheme used is given in ***Table 6.2.A-1***. ***Table 6.2.A-2*** describes the groups and parameters used in the sampling scheme.

6.0.10.2.6 Population Distribution Data

The population data used for this evaluation represent the population distribution within 50 km of the site. The data used are similar to that used in KNPS. The data are based on the 1991 government census, but were adjusted to reflect 1996 conditions. The population data used in this analysis are given in ***Table 6.2.A-3*** **25**. The data are in the format used by PC COSYMA. The population is generally spatially distributed on a circular grid with 16 directions (each described by an angle of 22.5°). The radial grid used has 22 radial distances up to 50 km. The grid is centred on the facility from which radionuclides are assumed to be released.

6.0.10.2.7 Agricultural Production

The agricultural production data are also an important element of consequence analysis. Agricultural production is important, since the food grown in an area is affected by the release. In the current evaluation, we assume a total food ban, and so agricultural data will not be used. Agricultural data would generally be specified in PC COSYMA in the same format as the population data.

6.0.10.2.8 Assumptions

Some general assumptions were made when calculating risk due to releases to the environment.

6.0.10.2.8.1 Ingestion pathway

This analysis does not consider continued irradiation as a result of ingestion of contaminated food. This will, however, not impact significantly the overall health risks, because in the event of significant exposure, a total food ban could be instituted to limit the exposure via this route.

6.0.10.2.8.2 Emergency plan

For purposes of this evaluation, however, it is conservatively assumed that no countermeasures as per the emergency plan are instituted.

6.0.10.2.9 Inputs into PC COSYMA

The summary of input options chosen when performing PC COSYMA calculations is given in ***Table 6.2.A-4*** in ***Appendix 6.2.A***. The calculations assume that all the R-M in the helium circuit is released in the first hour of the event. Delayed releases from medium to large non-isolatable breaks are assumed to commence 32 hours after the pipe break. The release height used was 23 m from the ground. This value represents the top of the reactor building where the HVAC or the vents are located. These are the only release routes in the event of a pressure boundary breach. The energy of release was assumed to be 1.06 MW **26**. The value of 1.06 MW was obtained by multiplying the decay heat for a single fuel element two hours after fission has stopped by the number of fuel balls in the core (assumed to be 330 000). The decay heat for a single fuel element is 3.201 W two hours after termination of fission reaction.

6.0.10.2.9.1 Event source term

The initial inventory available for release used as input into PC COSYMA is the sum of the activity in the helium inventory at the time of the event, the accumulated plate-out activity over 32 EFPY of operation and the activity from the delayed release. It is assumed that only 5% of plated-out R-M is

released in the event of a medium and large pressure boundary break. No plate-out is assumed for small breaks. The reason for this is that a small pipe break will not generate high enough flow forces to dislodge plated-out activity. For purposes of demonstrating compliance with NNR regulations, it is conservatively assumed that the entire primary circuit inventory is released. Therefore, the initial inventory available for release is calculated by summing the activities in ***Table 6.2.A-5***, 5% of those in ***Table 6.2.A-6*** and delayed activity where applicable. The initial activity inventory for each release category is given in ***Table 6.2-8***. For medium and large releases, the whole inventory available for release is conservatively assumed to be released. However, for, a small fraction of the helium circuit activity is released with no contribution from plated-out fission products.

Table 6.2-8: TOTAL INITIAL INVENTORY AVAILABLE FOR RELEASE[4] [In Bq]

Nuclide	Half Life	RC-1	RCF-1		RCF-2		RCP-1	RCPF-1		RCPF-2	
			Immediate	Delayed	Immediate	Delayed		Immediate	Delayed[5]	Immediate	Delayed
Kr - 83m	1.83h	7.80E+09	3.90E+09	5.36E+10	7.80E+09	5.36E+10	7.80E+09	7.80E+09	5.36E+10	7.80E+09	5.36E+10
Kr - 85m	4.48h	2.40E+10	1.20E+10	1.21E+11	2.40E+10	1.21E+11	2.40E+10	2.40E+10	1.21E+11	2.40E+10	1.21E+11
Kr – 85	10.76a	1.00E+08	5.00E+07	1.97E+09	1.00E+08	1.97E+09	1.00E+08	1.00E+08	1.97E+09	1.00E+08	1.97E+09
Kr – 87	76.3m	2.90E+10	1.45E+10	2.46E+11	2.90E+10	2.46E+11	2.90E+10	2.90E+10	2.46E+11	2.90E+10	2.46E+11
Kr – 88	2.80h	5.60E+10	2.80E+10	3.47E+11	5.60E+10	3.47E+11	5.60E+10	5.60E+10	3.47E+11	5.60E+10	3.47E+11
Kr – 89	3.18m	1.20E+10	6.00E+09	4.44E+11	1.20E+10	4.44E+11	1.20E+10	1.20E+10	4.44E+11	1.20E+10	4.44E+11
Kr – 90	32.3s	5.00E+09	2.50E+09	4.84E+11	5.00E+09	4.84E+11	5.00E+09	5.00E+09	4.84E+11	5.00E+09	4.84E+11
Xe – 131m	12.0d	4.60E+08	2.30E+08	3.52E+09	4.60E+08	3.52E+09	4.60E+08	4.60E+08	3.52E+09	4.60E+08	3.52E+09
Xe – 133m	2.2d	4.60E+09	2.30E+09	2.18E+10	4.60E+09	2.18E+10	4.60E+09	4.60E+09	2.18E+10	4.60E+09	2.18E+10
Xe – 133	5.29d	9.10E+10	4.55E+10	7.33E+11	9.10E+10	7.33E+11	9.10E+10	9.10E+10	7.33E+11	9.10E+10	7.33E+11
Xe – 135m	15.3m	6.80E+09	3.40E+09	1.34E+11	6.80E+09	1.34E+11	6.80E+09	6.80E+09	1.34E+11	6.80E+09	1.34E+11
Xe – 135	9.17h	5.60E+10	2.80E+10	8.48E+10	5.60E+10	8.48E+10	5.60E+10	5.60E+10	8.48E+10	5.60E+10	8.48E+10
Xe – 137	3.83m	1.90E+10	9.50E+09	6.62E+11	1.90E+10	6.62E+11	1.90E+10	1.90E+10	6.62E+11	1.90E+10	6.62E+11
Xe – 138	14.1m	3.60E+10	1.80E+10	6.70E+11	3.60E+10	6.70E+11	3.60E+10	3.60E+10	6.70E+11	3.60E+10	6.70E+11
Xe – 139	39.7s	5.90E+09	2.95E+09	5.24E+11	5.90E+09	5.24E+11	5.90E+09	5.90E+09	5.24E+11	5.90E+09	5.24E+11
I – 131	8.04d	5.20E+05	2.60E+05	3.25E+11	5.20E+05	3.25E+11	3.50E+09	3.50E+09	3.25E+11	3.50E+09	3.25E+11
I – 132	2.38h	7.00E+06	3.50E+06	4.96E+11	7.00E+06	4.96E+11	5.57E+08	5.57E+08	4.96E+11	5.57E+08	4.96E+11
I – 133	20.8h	3.40E+06	1.70E+06	7.20E+11	3.40E+06	7.20E+11	2.50E+09	2.50E+09	7.20E+11	2.50E+09	7.20E+11
I – 134	52.0m	1.70E+07	8.50E+06	8.28E+11	1.70E+07	8.28E+11	5.67E+08	5.67E+08	8.28E+11	5.67E+08	8.28E+11
I – 135	6.59h	5.90E+06	2.95E+06	6.83E+11	5.90E+06	6.83E+11	1.41E+09	1.41E+09	6.83E+11	1.41E+09	6.83E+11
Rb – 88	17.8m	2.70E+08	1.35E+08	3.54E+11	2.70E+08	3.54E+11	3.95E+08	3.95E+08	3.54E+11	3.95E+08	3.54E+11
Sr – 90	28.5a	4.75E+02									
			2.37E+02	1.70E+10	4.75E+02	1.70E+10	2.20E+08	2.20E+08	1.70E+10	2.20E+08	1.70E+10

[4] This is conservatively assumed to be the source that is released to the environment

[5] All delayed source terms are conservatively assumed to be equal to that for RCPF–2

Nuclide	Half Life	RC-1	RCF-1		RCF-2		RCP-1	RCPF-1		RCPF-2	
			Immediate	Delayed	Immediate	Delayed		Immediate	Delayed[5]	Immediate	Delayed
Cs – 134	2.06a	7.80E+03	3.90E+03	1.94E+10	7.80E+03	1.94E+10	4.95E+09	4.95E+09	1.94E+10	4.95E+09	1.94E+10
Cs – 137	30.1a	1.60E+04	8.00E+03	1.87E+10	1.60E+04	1.87E+10	8.10E+10	8.10E+10	1.87E+10	8.10E+10	1.87E+10
Cs – 138	32.2m	9.40E+07	4.70E+07	5.06E+11	9.40E+07	5.06E+11	1.74E+08	1.74E+08	5.06E+11	1.74E+08	5.06E+11
Ag – 110	250.4d	5.92E+02	2.96E+02	1.91E+08	5.92E+02	1.91E+08	1.30E+08	1.30E+08	1.91E+08	1.30E+08	1.91E+08
H – 3	12.35a	1.10E+11	5.50E+10	7.80E+07	1.10E+11	7.80E+07	1.10E+11	1.10E+11	7.80E+07	1.10E+11	7.80E+07
C – 14	5736a	1.20E+09	6.00E+08	2.04E+01	1.20E+09	2.04E+01	1.20E+09	1.20E+09	2.04E+01	1.20E+09	2.04E+01

* Delayed filtered release

6.0.10.2.9.2 Population data

The population distribution used in PC COSYMA calculation is given in ***Table 6.2.A-3*** **25**

6.0.10.2.9.3 Further inputs into PC COSYMA

PC COSYMA requires two site-specific parameters to be specified when calculating the radiation dose from inhalation of radioactive aerosol, gases and vapours. These are the surface roughness (terrain type) and the breathing rate.

6.0.10.2.9.3.1 Surface roughness

Terrain irregularities will generally affect the wind and air temperatures. The irregularities usually include building structures, hills, valleys, and other obstacles. The presence of building structures and other line-of-sight obstructions can significantly attenuate the ground contamination dose rate. However, the effect of these irregularities is much smaller high up in the atmosphere.

PC COSYMA allows for two choices of surface roughness: smooth and rough surface. The smooth terrain choice is appropriate for rural areas, while the rough terrain is appropriate for sites with large forested areas in the vicinity of the release point, or for sites located near urban centres where there are relatively tall building structures. The current evaluation covers an area of 50 km radius around the site. This area includes both the big metropolitan areas such as Cape Town, and the rural farms in the vicinity of the site. Because most of the area near the site can be considered ‘smooth terrain’, a smooth terrain type for calculations is assumed.

6.0.10.2.9.3.2 Breathing rate

PC COSYMA requires the breathing rate to be specified to use for calculating both short-term and long-term consequences. For the current analyses, a value of 3.333×10^{-4} m^3/s is used.

6.0.10.2.9.3.3 Reactor Building Dimensions

The dimensions of the reactor building are also specified.

- Length is 54.65 m.
- Width is 32.9 m.
- Height is 42 m.

The roof is at level +23 m. PC COSYMA requires, in addition to release height, the diameter of the

reactor building to be specified. This is because it is designed for Light Water Reactors (LWR) which have cylindrical containment buildings. The dimensions of the building are important since the volume within which the gas will expand into must be known. For the PBMR, the above dimensions must be converted to that of cylindrical structure with the same volume. This gives a radius of 24 m. Thus, the building width is 48 m.

6.0.10.2.9.4 Miscellaneous parameters

In addition to site-specific data that the user must provide as input, PC COSYMA also requires values for various quantities, such as dose per unit intake, or concentration in food per unit deposit for calculations of various radiological consequences. This information is located in data libraries provided with the code. Some of the information contained in these data libraries that is relevant to the current assessment is listed in ***Table 6.2.A-7*** and ***Table 6.2.A-8***. This information includes, among others, parameters needed by the health effects model to calculate early health effects and default risk coefficients used for late health effects.

6.0.10.3 Results

This section presents the results of PC COSYMA calculations for various events. The consequences evaluated here are for the equilibrium core at the End Of Life (EOL) of the plant, this being the worst-case scenario.

Two types of results are presented in this section as required by the NNR, namely deterministic dose calculations and probabilistic results. Deterministic calculations are required for Category A and B events, while probabilistic calculations are done for C category events. Deterministic Dose Results.

Deterministic doses due to accidents were determined at the zone boundary and at a point inside the zone boundary.

To demonstrate compliance with the NNR criteria for Category A and B events, short-term doses were calculated for specific weather conditions. Calculations conservatively assumed a Pasquill Category F weather, no rain, no delay time, and no variability in wind direction. A wind speed of 2 m/s was assumed. The value of 2 m/s is the average wind speed when Category F weather occurs **23**. It was computed by summing all possible wind speeds in Category F weather, and dividing by the total number of wind speeds.

The whole body dose results presented here are plume centre-line effective doses, and therefore represent the maximum possible doses at distances away from the reactor. The results of deterministic calculations are presented in ***Table 6.2-9***.

Table 6.2-9: DETERMINISTIC EFFECTIVE WHOLE BODY DOSES FOR CATEGORY F WEATHER

Distance	RC-1	RCF-1	RCF-2	RCP-1	RCPF-1	RCPF-2
	[Sv]	[Sv]	[Sv]	[Sv]	[Sv]	[Sv]
0.15	2.72E-06	7.66E-04	7.66E-04	3.59E-05	7.89E-04	7.89E-04
0.4	1.23E-06	3.33E-04	3.33E-04	1.60E-05	3.43E-04	3.43E-04
1.5	2.39E-07	5.86E-05	5.86E-05	3.05E-06	6.05E-05	6.05E-05
3.75	6.40E-08	1.41E-05	1.41E-05	7.99E-07	1.46E-05	1.46E-05
6.25	3.00E-08	6.17E-06	6.17E-06	3.69E-07	6.39E-06	6.39E-06
8.75	2.24E-08	4.37E-06	4.37E-06	2.72E-07	4.53E-06	4.53E-06
11.25	1.82E-08	3.38E-06	3.38E-06	2.19E-07	3.51E-06	3.51E-06
13.75	1.54E-08	2.74E-06	2.74E-06	1.84E-07	2.84E-06	2.84E-06
16.25	1.33E-08	2.26E-06	2.26E-06	1.58E-07	2.35E-06	2.35E-06
18.75	1.18E-08	1.90E-06	1.90E-06	1.38E-07	1.98E-06	1.98E-06
21.25	1.07E-08	1.63E-06	1.63E-06	1.25E-07	1.70E-06	1.70E-06
23.75	9.72E-09	1.42E-06	1.42E-06	1.12E-07	1.48E-06	1.48E-06
26.25	8.86E-09	1.24E-06	1.24E-06	1.02E-07	1.30E-06	1.30E-06
28.75	8.16E-09	1.09E-06	1.09E-06	9.30E-08	1.14E-06	1.14E-06
31.25	7.59E-09	9.72E-07	9.72E-07	8.61E-08	1.02E-06	1.02E-06
33.75	7.06E-09	8.67E-07	8.67E-07	7.96E-08	9.11E-07	9.11E-07
36.25	6.62E-09	7.79E-07	7.79E-07	7.42E-08	8.20E-07	8.20E-07

Table 6.2-10: COMPLIANCE TEST FOR A AND B CATEGORY EVENTS

PUBLIC DOSE						
Criterion	**RC-1**	**RCF-1**	**RCF-2**	**RCP-1**	**RCPF-1**	**RCPF-2**
	[mSv]	**[mSv]**	**[mSv]**	**[mSv]**	**[mSv]**	**[mSv]**
Calculated	1.23E-03	3.33E-01	3.33E-01	1.60E-02	3.43E-01	3.43E-01
Target for A	2.5E-02	2.5E-02				
Target for B	50	50	50	50	50	50

6.0.10.3.1 Results of Probabilistic Calculations

Probabilistic calculations are required for Category A, B and C events.

Results of probabilistic calculations presented here are:

- short-term and long-term risk of mortality for various health effects; and
- CCDF.

The various consequences are presented in terms of the mean (expectation value) of the magnitude of each consequence. For each block in the radial grid, PC COSYMA determines the conditional mean individual risks of developing the various health effects from the dose within that block. The dose at a particular location depends on the source term of a particular event, the prevailing meteorological conditions in the vicinity of the site, and the dispersion model being used. To obtain the mean risk in each distance band, the risks in the various blocks within each band are averaged.

6.0.10.3.1.1 Risk of mortality

PC COSYMA determines, in each distance annulus and for each release category, the conditional mean individual risk of fatalities resulting from short-term (early effects) and long-term (late effects) exposure of a single organ, or from the total effect of irradiation of all organs. The total mean conditional individual risk of fatalities per distance band and per release category resulting from exposure to R-M is evaluated as the sum of the risks due to short-term and long-term exposure. That is, if $R_e(k)$ and $R_l(k)$ are the mean conditional risks of early and late health effects, respectively, at the kth annulus, then the total mean conditional risk, R(k), at the kth annulus, for each release category, is given by:

$$R(k) = R_e(k) + R_l(k). \quad \text{(6.2.1)}$$

The total risk per distance band due to the operation of a single PBMR module is thus the weighted sum (by release frequency) of the total mean conditional risk per distance band. This can be expressed mathematically as:

$$R_M(k) = \sum_{rc=1}^{6} F_{rc} * R_{rc}(k), \quad \text{(6.2.2)}$$

where R_M (k) is the total risk per distance band for a single PBMR module, F_{rc}, the release frequency for each release category, and R_{rc}(k), the mean conditional risk at the kth distance band and for release category rc, where rc runs from 1 to 6.

The public risk due to a standard PBMR plant is defined as the total risk averaged over the national population. This is represented as the weighted average (by proportion of population in each distance band) of the total risk per distance band. Thus, the public risk, R_P, for a standard PBMR plant can be determined by:

$$Rp = \left(\frac{1}{N} \sum_{rc=1}^{6} F_{rc} \sum_{k=1}^{n} Rm(k) * P(k) \right), \quad \text{(6.2.3)}$$

where P(k) is the population in the kth distance band, n the number of distance bands used and N the total national population. This equation represents the average risk to the national population. The national population based on the 1996 census is currently about 40.6 million **24**.

The mean conditional individual risk of early deaths for a single PBMR module due to all LBE is 0 as the maximum doses are only a small fraction of the dose thresholds for early health effects. The mean conditional individual risk of late deaths for a single PBMR module, and for all release categories as a function of distance is given in ***Table 6.2-11***. ***Table 6.2-12*** gives the population mortality risk. To demonstrate compliance with NNR criteria, only the risk of fatality as a result of the irradiation of all organs is required. This risk is in the column labelled ‘Total’ for risk of long-term fatalities. Thus, the total mean conditional risk for each release category is the sum of the risk of early and late health effects, and the values are displayed in ***Table 6.2-12***. This table lists the total conditional risk per band, the total population per band, and calculates the population risk for the standard PBMR station. In order to calculate the public risk over the national population, it is assumed that the last distance band is defined by distance greater than 50 km. This band is conservatively assumed to have the same risk as the band 45 to 50 km.

6.0.10.3.1.2 CCDF

The CCDF are used in this assessment to calculate the probability that a particular consequence is equalled or exceeded. The consequences relevant to the current assessment are the number of fatalities. PC COSYMA calculates the CCDF per band for each event. Much like in the discussion of the previous section, the outputs of PC COSYMA are conditional CCDF. The probability that greater than *n* or more fatalities will occur, CCDF(n), for the standard PBMR station can be determined using:

$$CCDF(n) = \left(\sum_{rc=1}^{6} F_{rc} * \left(CCDF_{e,rc}(n) + CCDF_{l,rc}(n) \right) \right), \quad \text{(6.2.4)}$$

where F_{rc} is the release frequency for release category rc, $CCDF_{e,rc}(n)$ and $CCDF_{l,rc}(n)$ are the CCDF for equalling or exceeding n early or late fatalities, respectively. The CCDF is used to demonstrate compliance with the NNR criteria which impose a bias against larger accidents.

Table 6.2-11: RISK OF LATE MORTALITY PER BAND DUE TO OPERATION OF PBMR

Distance [KM]	RC-1 [Per year]	RCF-1 [Per year]	RCF-2 [Per year]	RCP-1 [Per year]	RCPF-1 [Per year]	RCPF-2 [Per year]	TOTAL [Per year]
0.15	1.10E-06	3.97E-06	3.97E-06	2.56E-06	3.97E-06	3.97E-06	2.39E-07
0.4	**2.70E-07**	**9.67E-07**	**9.67E-07**	**5.44E-07**	**9.67E-07**	**9.67E-07**	**5.81E-08**
1.5	3.31E-08	1.16E-07	1.16E-07	6E-08	1.16E-07	1.16E-07	6.98E-09
3.75	8.10E-09	2.78E-08	2.78E-08	1.33E-08	2.78E-08	2.78E-08	1.68E-09
6.25	3.94E-09	1.33E-08	1.33E-08	6.02E-09	1.33E-08	1.33E-08	8.04E-10
8.75	2.39E-09	8.05E-09	8.05E-09	3.43E-09	8.05E-09	8.05E-09	4.86E-10
11.25	1.70E-09	5.72E-09	5.72E-09	2.21E-09	5.72E-09	5.72E-09	3.45E-10
13.75	1.40E-09	4.71E-09	4.71E-09	1.7E-09	4.71E-09	4.71E-09	2.84E-10
16.25	1.03E-09	3.42E-09	3.42E-09	1.37E-09	3.42E-09	3.42E-09	2.07E-10
18.75	7.79E-10	2.61E-09	2.61E-09	1E-09	2.61E-09	2.61E-09	1.58E-10
21.25	6.70E-10	2.24E-09	2.24E-09	7.34E-10	2.24E-09	2.24E-09	1.35E-10
23.75	6.00E-10	2E-09	2E-09	5.89E-10	2E-09	2E-09	1.21E-10
26.25	5.47E-10	1.81E-09	1.81E-09	4.87E-10	1.81E-09	1.81E-09	1.10E-10
28.75	4.98E-10	1.65E-09	1.65E-09	4.34E-10	1.65E-09	1.65E-09	9.95E-11
31.25	4.96E-10	1.63E-09	1.63E-09	4.21E-10	1.63E-09	1.63E-09	9.84E-11
33.75	4.86E-10	1.58E-09	1.58E-09	4.08E-10	1.58E-09	1.58E-09	9.58E-11
36.25	4.59E-10	1.49E-09	1.49E-09	3.9E-10	1.49E-09	1.49E-09	9.03E-11
38.75	4.37E-10	1.42E-09	1.42E-09	3.73E-10	1.42E-09	1.42E-09	8.60E-11
41.25	4.11E-10	1.33E-09	1.33E-09	3.43E-10	1.33E-09	1.33E-09	8.05E-11
43.75	3.82E-10	1.23E-09	1.23E-09	3.16E-10	1.23E-09	1.23E-09	7.47E-11
46.25	3.41E-10	1.1E-09	1.1E-09	2.66E-10	1.1E-09	1.1E-09	6.67E-11
48.75	3.11E-10	1.01E-09	1.01E-09	2.12E-10	1.01E-09	1.01E-09	6.10E-11

Table 6.2-12: POPULATION RISK FOR SINGLE PBMR MODULE

Distance [Km]	**Population per band**	**Total Risk [Per year]**	**(Risk)*(population)**
0.4	2815	5.81E-08	1.64E-04
1.5	0	6.98E-09	0.00E+00
3.75	2884	1.68E-09	4.85E-06
6.25	2916	8.04E-10	2.34E-06
8.75	1530	4.86E-10	7.44E-07
11.25	11441	3.45E-10	3.95E-06
13.75	34450	2.84E-10	9.78E-06
16.25	24584	2.07E-10	5.09E-06
18.75	18722	1.58E-10	2.96E-06
21.25	15073	1.35E-10	2.03E-06
23.75	42060	1.21E-10	5.09E-06
26.25	137094	1.10E-10	1.51E-05
28.75	199684	9.95E-11	1.99E-05
31.25	351300	9.84E-11	3.46E-05
33.75	338250	9.58E-11	3.24E-05
36.25	371141	9.03E-11	3.35E-05
38.75	285560	8.60E-11	2.46E-05
41.25	196646	8.05E-11	1.58E-05
43.75	343970	7.47E-11	2.57E-05
46.25	299795	6.67E-11	2.00E-05
48.75	190887	6.10E-11	1.16E-05
> 50 km	37729198	6.10E-11	2.30E-03
	4.06E+07		**2.74E-03**
	Peak Risk =	**2.39E-07**	
	Public Risk =	**8.09E-11**	

Table 6.2-13: PROBABILITY OF EXCEEDING A PARTICULAR NUMBER OF LATE FATALITIES

Fatalities	CCDF			Module Probability of Exceeding	NNR Limit Line (A/N)
	Early Fatalities	Late Fatalities	Total		
1.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	2.18E-02
1.26E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	1.73E-02
1.59E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	1.37E-02
2.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	1.09E-02
2.51E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	8.67E-03
3.16E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	6.89E-03
3.98E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	5.47E-03
5.01E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	4.35E-03
6.31E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	3.45E-03
7.94E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	2.74E-03
1.00E+01	0.00E+00	0.00E+00	0.00E+00	0.00E+00	2.18E-03
1.26E+01	0.00E+00	0.00E+00	0.00E+00	0.00E+00	1.73E-03
1.59E+01	0.00E+00	0.00E+00	0.00E+00	0.00E+00	1.37E-03
2.00E+01	0.00E+00	0.00E+00	0.00E+00	0.00E+00	1.09E-03
2.51E+01	0.00E+00	0.00E+00	0.00E+00	0.00E+00	8.67E-04
3.16E+01	0.00E+00	0.00E+00	0.00E+00	0.00E+00	6.89E-04
3.98E+01	0.00E+00	0.00E+00	0.00E+00	0.00E+00	5.47E-04
5.01E+01	0.00E+00	0.00E+00	0.00E+00	0.00E+00	4.35E-04
6.31E+01	0.00E+00	0.00E+00	0.00E+00	0.00E+00	3.45E-04
7.94E+01	0.00E+00	0.00E+00	0.00E+00	0.00E+00	2.74E-04
1.00E+02	0.00E+00	0.00E+00	0.00E+00	0.00E+00	2.18E-04

6.0.10.3.2 DEMONSTRATION OF COMPLIANCE

In this assessment, compliance with NNR criteria for public risk was considered. The deterministic dose value to be used and should be read at distance 400 m, this being the site boundary for the PBMR.

The criteria of the NNR for the public requires the demonstration of compliance deterministically by calculating the dose at the site boundary for Category A and B events, and probabilistically by determining peak individual and public risks.

6.0.10.3.3 Compliance with Deterministic Criteria for Category A and B Events

Table 6.2-9-gives the short-term effective dose as a function of distance for category F weather, assuming wind speeds of 2 m/s and no rainfall. The calculations assume a 30-day exposure. The 30-day exposure was used in order to be conservative. Realistically, people would not be expected to remain outside without clothing for prolonged lengths of time. More realistic values of exposure normally assumed in PRA are 6 to 24 hours. ***Table 6.2-10*** gives the maximum doses for each event category for a single PBMR module at the site boundary. Release categories RC-1 and RCF-1 have both A and B category events. From ***Table 6.2-10***, it is clear that Release category RCF-1 does not meet public exposure criterion for A events.

6.0.10.3.4 Compliance with Probabilistic Criteria

Probabilistically, the criteria of the NNR for Categories A, B and C events requires that:

- The peak risk due to the operation of the PBMR plant is less than 5×10^{-6} fatalities per year.
- The public risk is less than 1×10^{-8} fatalities per year.

6.0.10.4 Peak Risk

The peak risk refers to the maximum risk to which the public can be exposed. The risk is highest closest to the plant. Since the point closest to the plant is the site boundary, the peak risk is thus represented by the risk at the site boundary. The site boundary in for PBMR is located 400 m from the reactor.

From ***Table 6.2-11***, the peak risk for the standard PBMR plant is given by the total risk due to a single PBMR module at the 400 m distance. This yields a value of 5.8×10^{-8} fatalities per year. This value is well within the NNR limit of 5×10^{-6} fatalities per year.

6.0.10.4.1 Public Risk

From ***Table 6.2-11***, the public risk due to the standard PBMR plant is given by the weighted sum of the total risk per band times by the population per band and divided by the total national population of 40.6 million. This gives a value of about 6.7×10^{-11}. This also complies with NNR criteria.

6.0.11 CONCLUSION

This preliminary analysis despite the application of many conservative assumptions in the consequence modeling has successfully demonstrated compliance of the PBMR with the NNR safety criteria for the public. Analysis for compliance with criteria for plant personnel was not part of the scope of the current analysis. This will, however, be done in future revisions of the SAR.

The results indicate that no deaths result from the event sequences considered in the analysis. This is largely because of the small amount of R-M that is released into the environment.

The analysis assumed that no emergency actions over a period of 30 days are implemented in order to mitigate the consequences of the event.

The source term used was that of the HTR-Modul corrected for PBMR core, and was limited to a few nuclides which contribute significantly to health effects. The next revision of the SAR will use a PBMR-specific source term.

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Appendix 6.2.A: INPUTS TO PC COSYMA

MODELS USED TO PERFORM CALCULATIONS

To determine the doses incurred from a particular release of radioactivity from a nuclear reactor, the atmospheric concentration (measured in Bq/m^3) of each radioisotope and the ground concentration (measured in Bq/m^2) of each radionuclide deposited by the passing plume are often calculated. The ground concentration is needed to determine the long-term irradiation after the plume has passed, and in particular to estimate the effects of the doses from the ingestion of contaminated food. The air concentration is determined from an atmospheric dispersion model. Once the concentrations are known, the whole body dose due to various exposure routes can be estimated.

ATMOSPHERIC DISPERSION

PC COSYMA uses the Gaussian plume model to describe the movement of radionuclides through the atmosphere. The Gaussian Plume model used by PC COSYMA allows for hourly changes in meteorological conditions affecting the plume. It also takes into account the type of terrain (i.e., whether the terrain is rough or smooth) over which the plume travels.

The dispersion model allows for plume depletion by various processes and build-up of radioactive isotopes. It further allows for material released over a long period by describing the release as a series of hourly phases, which are treated separately. A description of the dispersion model used in PC COSYMA can be found in **1**.

EXPOSURE PATHWAYS AND DOSE CALCULATION MODELS

There are generally six different ways in which the public can accumulate a radiation dose after a release of R-M into the atmosphere. These exposure pathways are:

- External exposure to γ and β radiation from activity in the passing plume or cloud of R-M. This pathway is referred to as *cloudshine*.
- External exposure to γ and β radiation from activity deposited on the ground. This pathway is referred to as *groundshine*.
- Internal exposure to activity due to inhalation of activity directly from the passing plume.
- Internal irradiation due to inhalation of activity from re-suspended ground deposit.

- Internal exposure following ingestion of food contaminated by activity deposited from the plume.

- External γ and β irradiation from activity deposited onto the skin and clothing.

When determining the dose, PC COSYMA takes into account all these exposure routes. For the current study, however, the ingestion pathway has been ignored for reasons discussed later. The doses calculated by PC COSYMA are those for an average adult. For each pathway, the dosimetric models are used to evaluate not only the effective dose equivalent, but also the dose in each of a number of organs and tissues that are of significance for evaluating the health impact in the exposed population. These organs include the bone marrow, lung, thyroid, breast, liver, G I tract, skin, and the average dose in the remainder of the body.

Dosimetric models are described for both external and internal radiation exposures. The external pathways are γ irradiation from materials in the plume or deposited on the ground, and β irradiation of the skin from material deposited on the skin and clothes. The dosimetric model for external gamma dose from the passing plume assumes a semi-infinite plume of uniform concentration. The model used for external γ dose from material deposited on the ground takes into account material deposited on different surfaces (such as walls, roofs, grass, etc.) in the residential areas, and the movement of activity into the ground. The dose obtained is appropriate for people who are outdoors. The model used for external β dose to the skin from material deposited on the skin or clothes assumes that the dose is a fraction of the dose due to dry deposition on the ground at the same location. The contribution of the external β dose from material deposited on the skin and clothes is generally very small compared with other exposure pathways.

Internal exposure in PC COSYMA is due to inhalation of material in the plume or following re-suspension, and from ingestion of contaminated food. PC COSYMA calculates the inhalation dose by taking the product of the time integrated air concentration, breathing rate, and the dose per unit activity inhaled. The dose per unit activity is provided in the data library included with the system.

Calculation of External Dose

External pathways considered in PC COSYMA are γ irradiation from material in the passing radioactive cloud, or that deposited on the ground, and β irradiation from material deposited on the skin and clothes. The γ dose from the plume depends on the predicted distribution of airborne activity. The dose model for external γ irradiation assumes a semi-infinite plume of uniform

radionuclide concentration. The γ dose from material in the plume is calculated by:

γ_{plume} = χ .(Dose conversion Factor) .....
..... (6.2.5)

The dose conversion factor is defined as the dose per unit radionuclide concentration. The dose conversion factors for particular radionuclides are pre-calculated and are located in the data library of PC COSYMA. The dose calculated in this Equation is corrected to allow for the plume not being semi-infinite or of uniform concentration. The correction factor takes into account the shape and size of the plume.

The γ dose from material deposited on the ground is determined by:

γ_{ground} = A x D, ...
.....
(6.2.6)

where A is the amount of the material deposited and D the dose per unit deposition. The dose per unit deposition is obtained from the PC COSYMA data library.

The external β dose from materials deposited on the skin on clothes is calculated as follows:

β_{skin} = A x T x D, ...
.....
(6.2.7)

where A = the amount of the material deposited on the skin or clothes;

T = the time for which the material remains on the skin or clothes; and

D = the dose per unit deposit.

The deposition on the skin and clothes is taken as a fraction of the dry deposition on the ground at the same location. The user of PC COSYMA is able to specify the multiplication factor. The time for which the material remains on the skin is calculated using an effective half-life for material on the skin, and this can also be set by the user. The dose rate per unit deposit is determined from the data library. The dose calculated in this way refers to the area of the skin which is not covered by clothing. The dose to the skin covered by clothing is assumed to be much less because of the shielding effect of the clothing.

Calculation of Internal Dose

The internal exposure pathways considered in PC COSYMA include inhalation of R-M from the passing plume, or inhalation following re-suspension and ingestion of food. In the current study, ingestion dose is not considered. The dose from material inhaled from the passing radioactive cloud is calculated as:

$$Dose_{cloud} = \chi \,.\, B.(\text{dose per unit activity inhaled}), \quad (6.2.8)$$

where χ = the time integrated air concentration; and

B = the breathing rate.

The dose per unit activity inhaled is provided in PC COSYMA data library. The inhalation dose from re-suspended material is also determined using ***Equation 6.2.8***. However, the air radionuclide concentration at a time t due to re-suspended material after deposition is determined using a re-suspension factor, R, given by:

$$R = a.e^{-bt} + c, \quad (6.2.9)$$

where a, b, and c are parameters which can be set by the user, and t is the time since deposition. Since the air concentration of radionuclides inside and outside buildings may be different, PC COSYMA treats these appropriately. However, for the current analysis, the air concentrations calculated here are those for outdoor situations.

METHODS USED TO CALCULATE THE RISK OF HEALTH EFFECTS

The health effects that can be calculated with PC COSYMA were indicated earlier. This section describes the dose response relationships and the default parameter values which are used in deriving the risk of health effects from the calculated doses. As already mentioned, the health effects are of two kinds, namely early and late health effects.

Calculation of Risk of Early Health Effects

The models used to determine the risk of early health effects are described in **1**. The models used in PC COSYMA are exactly the same as those used in the mainframe version of PC COSYMA. The risk of experiencing early effects following irradiation is modelled using a dose-response relationship, which assumes that there is a threshold below which no effects occur. Above this dose threshold, the risk increases rapidly with dose in a non-linear fashion. This is modelled using a ‘hazard function’ in which the probability of an individual being affected, r, is given by:

$$r = 1 - e^{-H}, \quad (6.2.10)$$

where

$$H = \ln 2 \left(\frac{D}{D_{50}} \right)^{S}, \quad (6.2.11)$$

in which

D is the dose received in the appropriate period;

D_{50} is the dose which causes the effect in 50% of the exposed population; and

S is the shape parameter, which characterizes the slope of the dose-risk relationship.

Since doses which are accumulated at a low dose rate are much less effective in causing early health effects than doses delivered at high dose rates, the effective dose is calculated by summing overdoses in a number of different time periods, with appropriate values for D_{50}. The risk is then given by the formula above, but with the ratio D/D_{50} replaced by:

$$\frac{D}{D_{50}} = \sum \frac{D^{i}}{D_{50}^{i}}, \quad (6.2.12)$$

where

D^i is the dose received in the appropriate period; and

D^i_{50} is the dose which causes the effect in 50% of the exposed population, in the time period considered.

The model described here predicts that there is a non-zero risk of early health effects for any dose, however small. Experience shows that there is a threshold dose below which early health effects are not observed. This is incorporated in PC COSYMA by setting the risk to zero if the dose is less than the threshold value set by the user. A default value of 1% is used.

PC COSYMA assumes that the variation of D_{50} with dose rate is given by:

$$D_{50}^{i} = D_{\infty} + \frac{D_0}{DR^{i}}, \quad (6.2.13)$$

where

D_∞ is the value of D_{50} at high dose rate;

DR^i is the dose rate averaged over the period considered; and

D_0 is a parameter.

The first term in the equation is the D_{50} at high dose rate, while the second term describes the increase in D_{50} as the dose rate is reduced.

Organs have a different sensitivity for high and low Linear Energy Transfer (LET) irradiation. This is included in the model by multiplying the high LET dose by the Relative Biological Effectiveness (RBE). Different values of RBE are used for different organs. An allowance is made for irradiation of more than one organ when calculating the total individual risk of early health effects, although the model does not include synergism between risks and doses in different organs. The risks of death from irradiation of different organs are combined in a way which prevents the overall risk exceeding unity.

Calculation of Risk of Late Health Effects

The dose-risk model used to calculate risk of late health effects is linear without a threshold. PC COSYMA calculates the risk of late health effects as the product of the dose and a risk coefficient. The risk coefficients used to calculate the risk of late health effects are given in ***Table 6.2.A-8***. From this table, the risk coefficient used to calculate risk of late death due to total whole body dose is 0.05 fatalities per Sv. The dose from material deposited on the skin makes a large contribution to the skin dose calculated in PC COSYMA. The dose calculated is therefore that to a small area of skin which is not covered by clothing. The calculation of the risk of skin cancer requires the average dose to the whole of the skin. An allowance for partial skin irradiation is included in calculating the risk of skin cancer.

Table 6.2.A-1: THE STRATIFIED METEOROLOGICAL SAMPLING SCHEME

```
1
100
'Y'
IN-THE-POPULATION calculations input file: \PCCOSYM2\DATA\RC_ALL.MT1
&INPUT
 ISTART=1
 IFIN=17420
 IEXTRA=100
 NODIRB=6
 WOCOEF=8.0000E-05
 RNEXP=6.0000E-01
 BWIDTH=3.7000E+01
 BHITE=5.6000E+01
 HEIGHT=28
```

```
 ZREFA=10
 HEATP=1.6337E+03
 IROUGH=3
 WPE=7.0000E-02 ,1.3000E-01 ,2.1000E-01 ,3.4000E-01 ,4.4000E-01 ,4.4000E-01
 STANGL=0.0000E+00
&END
  'RNFL'
  'F'  5.0000E-05  1.0000E+01
  'RNFL'
  'F'  5.0000E-05  2.5000E+01
  'RNFL'
  'F'  5.0000E-05  5.0000E+01
  'RNFL'
  'F'  5.0000E-05  7.5000E+01
  'RNFL'
  'F'  5.0000E-05  1.0000E+02
  'RNFL'
  'F'  5.0000E-06  1.0000E+01
  'RNFL'
  'F'  5.0000E-06  2.5000E+01
  'RNFL'
  'F'  5.0000E-06  5.0000E+01
  'RNFL'
  'F'  5.0000E-06  7.5000E+01
  'RNFL'
  'F'  5.0000E-06  1.0000E+02
  'TIMDST'
  'F'  -2  0.0000E+00  0.0000E+00  2.0000E+01  6.0000E+01
  'TIMDST'
  'F'  5  2.0000E+00  4.0000E+00  2.0000E+01  6.0000E+01
  'TIMDST'
  'F'  5  2.0000E+00  6.0000E+00  2.0000E+01  6.0000E+01
  'TIMDST'
  'F'  5  2.0000E+00  0.0000E+00  2.0000E+01  6.0000E+01
  'TIMDST'
  'F'  5  0.0000E+00  6.0000E+00  2.0000E+01  6.0000E+01
  'TIMDST'
  'F'  5  0.0000E+00  0.0000E+00  2.0000E+01  6.0000E+01
  'TIMDST'
  'F'  0  1.0000E+00  3.0000E+00  2.0000E+01  6.0000E+01
  'TIMDST'
  'F'  0  1.0000E+00  0.0000E+00  2.0000E+01  6.0000E+01
  'TIMDST'
  'F'  0  0.0000E+00  3.0000E+00  2.0000E+01  6.0000E+01
  'TIMDST'
  'F'  0  0.0000E+00  0.0000E+00  2.0000E+01  6.0000E+01
```

Table 6.2.A-2: GROUPS AND PARAMETERS IN THE SAMPLING SCHEME

Bin	Description of Category
	Wet sequences
	$\Lambda/u > 5 \times 10^{-5}$ within 10 km
	$\Lambda/u > 5 \times 10^{-5}$ within 25 km
	$\Lambda/u > 5 \times 10^{-5}$ within 50 km
	$\Lambda/u > 5 \times 10^{-5}$ within 75 km
	$\Lambda/u > 5 \times 10^{-5}$ within 100 km
	$\Lambda/u > 5 \times 10^{-6}$ within 10 km
	$\Lambda/u > 5 \times 10^{-6}$ within 25 km
	$\Lambda/u > 5 \times 10^{-6}$ within 50 km
	$\Lambda/u > 5 \times 10^{-6}$ within 75 km
	$\Lambda/u > 5 \times 10^{-6}$ within 100 km
	Dry Sequences
	Categories A and B, all travel times
	Categories E and F, times < 2 and < 4 h

Categories E and F, times < 2 and < 6 h
Categories E and F, times < 2 and > 6 h
Categories E and F, times > 2 and < 6 h
Categories E and F, any other times
Categories C and D, times < 1 and < 3 h
Categories C and D, times < 1 and > 3 h
Categories C and D, times > 1 and < 3 h
Categories C and D, any other times

Note: *Each of the groups described above is combined with six groups of wind direction.*

Table 6.2.A-3: POPULATION DISTRIBUTION WITHIN 50 km AROUND KOEBERG

```
21      16
 THE KOEBERG POPULATION DISTRIBUTION DATA -- for 22.5 radial grid
       3.00E-01 0.50E+00 2.50E+00 5.00E+00 7.50E+00
       1.00E+01 1.25E+01 1.50E+01 1.75E+01 2.00E+01
       2.25E+01 2.50E+01 2.75E+01 3.00E+01 3.25E+01
       3.50E+01 3.75E+01 4.00E+01 4.25E+01 4.50E+01
       4.75E+01 5.00E+01
      0        44      188       41       42
      30       37       45      239     2624
      1958     148      106      117      130
      4196     1529     174      185      214
      215      206
      0        44      266       73       51
      43      10184    31159    11224     1689
      302     131      169      161      206
      229     265      294      341      374
      369     263
      0       44      226      111       62
      91     228      352      447      757
      839    904      727      469     1562
      3495   9460     6185      705      362
      309    330
     0        44      181      130      148
     91      172      296      286      234
     194     971      177      412      577
     562     781      890      888      520
     603     703
     0        44      266      282      297
     87      115      541      182      146
     150     317      695      486      579
     793     973     1532     1352     1675
     2417   16737
      0       44      307      440      325
      187     380      143      218      274
      332     547     1062     1330     3016
      8034    2686     1743     2939     3997
      7858    9846
      0       44      307      726      377
      470     406      364      303      434
      783     2079    23811    38192    64823
      70030   29740     7068    16897    36704
      29787   23365
      0       44      284     1043     1165
      397     151     1110    11536    12359
      10832   35088    63924    77618   249658
      200421  283361   178932   100112   222705
      240849  121707
      0       21        5       15      407
```

```
107     27      564        0        0
0      2510    50124    81130    38077
47445  45996    77854    76884    76722
11598  9711
 0        0        0        0        0
 0        0        0        0        0
 0        0        0        4        0
 0      486     1587     2532     4096
 0        0
 0        0        0        0        0
 0        0        0        0        0
 0        0        0        0        0
 0        0        0        0        0
 0        0
 0        0        0        0        0
 0        0        0        0        0
 0        0        0        0        0
 0        0        0        0        0
 0        0
 0        0        0        0        0
 0        0        0        0        0
 0        0        0        0        0
 0        0        0        0        0
 0        0
 0        0        0        0        0
 0        0        0        0        0
 0        0        0        0        0
 0        0        0        0        0
 0        0
 0        16       2        0        0
 3        3        1       18        6
 0        0        0        0        0
 0        0        0       77      489
 38       0
 0        44      163       32       40
 30      36       44       53       60
 57      64       82       93      105
 119     136      142      130      233
 98      100
```

Table 6.2.A-4: SUMMARY OF INPUTS CHOSEN IN PC COSYMA

```
 ***************************************************************************
 *                                                                         *
 *                               PC COSYMA                                 *
 *                              VERSION 2.01                               *
 *                                                                         *
 *                          RUN DATE :  20/05/01                           *
 *                                                                         *
 *                                                                         *
 *  RUN NAME           : RCP-1P                                            *
 *  DESCRIPTION OF RUN :                                                   *
 *                                RCP-1P                                   *
 *        PROBABILISTIC RUN; H=23M, T=1 HR; CIRC PLUS PLATEOUT  ACTIVITY   *
 ***************************************************************************


SUMMARY OF THE OPTIONS CHOSEN FOR THIS RUN
------------------------------------------

                      ***** PROBABILISTIC RUN *****

                *** RESULTS CALCULATED IN THE POPULATION ***


    HOURLY MET FILE :
        \PCCOSYM2\DATA\MET9899.MH2


    MET SAMPLING PROGRAM                  :         USED
    MET SAMPLING INPUT FILE :
        \PCCOSYM2\DATA\RPC-1P.MT1


    SOURCE TERM PROGRAM                   :         USED
    SOURCE TERM INPUT FILE :
        \PCCOSYM2\DATA\RCP-1P.ST3


    RADIONUCLIDE CONCENTRATIONS           :         YES

    COUNTERMEASURES CONSIDERED            :         NO
    COUNTERMEASURES AS ENDPOINT           :         NO
    COUNTERMEASURES - SHORT TERM
       -  PATTERN, POTENTIAL DOSES
          AND PROBABILITIES               :         NO
       -  NUMBERS AND AREAS               :         NO
    COUNTERMEASURES - LONG TERM
       -  PATTERN, POTENTIAL DOSES
          AND PROBABILITIES               :         NO
       -  NUMBERS AND AREAS               :         NO

    SHORT TERM INDIVIDUAL DOSES           :         YES
    LONG TERM INDIVIDUAL DOSES            :         YES
    SHORT TERM INDIVIDUAL RISKS           :         YES
    LONG TERM INDIVIDUAL RISKS            :         YES
    SHORT TERM HEALTH EFFECTS             :         YES
    LONG TERM COLLECTIVE DOSES            :         YES
    LONG TERM HEALTH EFFECTS              :         YES
    ECONOMIC CONSEQUENCES                 :         YES

    COSTS TO BE CALCULATED :
       EVACUATION COSTS                   :         NO
       RELOCATION COSTS                   :         NO
```

```
      DECONTAMINATION COSTS              :         NO
      FOOD BAN COSTS                     :         NO
      EARLY HEALTH EFFECTS COSTS         :         YES
      LATE HEALTH EFFECTS COSTS          :         YES



   SHORT TERM DOSES CALCULATED TO        :          30 DAYS
   ALL PHASES TRAVEL IN WIND DIRECTION OF 1st HOUR OF 1st PHASE


*****************************************************************************
                  SPATIAL DISTRIBUTION DATA
*****************************************************************************


   THIS RUN USES  22 DISTANCE BANDS
   THE OUTER LIMITS OF THE DISTANCES (IN KM) ARE


    3.0000E-01      5.0000E-01      2.5000E+00      5.0000E+00      7.5000E+00
    1.0000E+01      1.2500E+01      1.5000E+01      1.7500E+01      2.0000E+01
    2.2500E+01      2.5000E+01      2.7500E+01      3.0000E+01      3.2500E+01
    3.5000E+01      3.7500E+01      4.0000E+01      4.2500E+01      4.5000E+01
    4.7500E+01      5.0000E+01


   THE CUT-OFF DISTANCE FOR CALCULATIONS WHICH INVOLVE EARLY COUNTERMEASURES
   OR EFFECTS, IF REQUIRED, IS SET AT       5.0000E+01 KM.


   DETAILED INFORMATION IS GIVEN AT THE FOLLOWING  2 GRID POINTS
    4.0000E-01 KM      2.3750E+01 KM


   THIS RUN USES 16 SECTORS.


   SPATIAL DISTRIBUTION OF POPULATION USES NUMBERS FROM THE
   COSGRIDS PROGRAM, INPUT FILE: \PCCOSYM2\DATA\PBMR_225.GR2


*****************************************************************************
                            DOSE PARAMETERS
*****************************************************************************


   EFFECTIVE DOSE IS CALCULATED (ICRP-60)

   BREATHING RATE                 :      3.3330E-04 M**3/S

   RESUSPENSION PARAMETERS  WLAMR :      3.5000E-08 /S
                             RESE :      1.0000E-09 /M
                             RES0 :      5.0000E-08 /M


*****************************************************************************
                              MET SAMPLING
*****************************************************************************



THIS COSYMA RUN CONSIDERS MULTIPLE WEATHER SEQUENCES DETERMINED FROM
A STRATIFIED SAMPLING SCHEME FROM FILE : \PCCOSYM2\DATA\RPC-1P.MT2


*****************************************************************************
```

DISPERSION

****

INPUT LIST OF THE METEOROLOGICAL ZONE

HEIGHT OF MIXING LAYER (M) OF THE DIFFUSION CATEGORY

A B C D E F

READ IN FROM METEOROLOGICAL DATA FILE

SURFACE ROUGHNESS : SMOOTH TERRAIN (LOW PLANTS, RURAL AREAS; Z0 = 10 CM TO 1 M)

INPUT LIST OF THE PARAMETERS FOR THE WIND PROFILE

WIND PROFILE EXPONENT FOR THE DIFFUSION CATEGORY

A	B	C	D	E	F
7.0000E-02	1.3000E-01	2.1000E-01	3.4000E-01	4.4000E-01	4.4000E-01

THE FOLLOWING PARAMETERS ARE A FUNCTION OF THE FOLLOWING HEIGHTS

HEIGHT (M) 5.0000E+01 1.0000E+02 1.8000E+02

DISPERSION COEFFICIENTS FOR SMOOTH TERRAIN

CAT.	PY	QY	PY	QY	PY	QY
A	9.4600E-01	7.9600E-01	9.4600E-01	7.9600E-01	9.4600E-01	7.9600E-01
B	8.2600E-01	7.9600E-01	8.2600E-01	7.9600E-01	8.2600E-01	7.9600E-01
C	5.8600E-01	7.9600E-01	5.8600E-01	7.9600E-01	5.8600E-01	7.9600E-01
D	4.1800E-01	7.9600E-01	4.1800E-01	7.9600E-01	4.1800E-01	7.9600E-01
E	2.9700E-01	7.9600E-01	2.9700E-01	7.9600E-01	2.9700E-01	7.9600E-01
F	2.3500E-01	7.9600E-01	2.3500E-01	7.9600E-01	2.3500E-01	7.9600E-01

CAT.	PZ	QZ	PZ	QZ	PZ	QZ
A	1.3210E+00	7.1100E-01	1.3210E+00	7.1100E-01	1.3210E+00	7.1100E-01
B	9.5000E-01	7.1100E-01	9.5000E-01	7.1100E-01	9.5000E-01	7.1100E-01
C	7.0000E-01	7.1100E-01	7.0000E-01	7.1100E-01	7.0000E-01	7.1100E-01
D	5.2000E-01	7.1100E-01	5.2000E-01	7.1100E-01	5.2000E-01	7.1100E-01
E	3.8200E-01	7.1100E-01	3.8200E-01	7.1100E-01	3.8200E-01	7.1100E-01
F	3.1100E-01	7.1100E-01	3.1100E-01	7.1100E-01	3.1100E-01	7.1100E-01

HORIZONTAL STANDARD DEVIATION (DEGREE) OF WIND DIRECTION

A	2.0500E+01	2.0500E+01	2.0500E+01
B	1.3900E+01	1.3900E+01	1.3900E+01
C	1.0100E+01	1.0100E+01	1.0100E+01
D	6.9000E+00	6.9000E+00	6.9000E+00
E	4.0000E+00	4.0000E+00	4.0000E+00
F	2.0000E+00	2.0000E+00	2.0000E+00

****

SOURCE TERM AND DEPOSITION

****

INPUT LIST OF THE NUCLIDE DATA - USING SOURCE TERM PROGRAM

THE SOURCE TERM PROGRAM MAY NOT USE ALL THE NUCLIDES IN THIS FULL LIST,
DEPENDING ON THE CUT-OFF SPECIFIED. REFER TO THE OUTPUT FILE :
"\PCCOSYM2\RESULTS\SRC*.SRC"
WHERE * IS THE NAME OF THE *.ST1 FILE

NO.	NUCLIDE	HALF-LIFE (Y)	DEPOSIT. VELOCITY (M/S) *	WASHOUT COEFFICIENT FOR RAIN TERM A	TERM B

* CORRECTION FACTOR FOR CALCULATING THE GROUND CONTAMINATION OF

NOBLE GASES:	1.0000E+00
AEROSOLS:	1.0000E+00
ELEMENTAL IODINE:	1.0000E+00
ORG. BOUND IODINE:	1.0000E+00

EXPOSURE PATHWAYS FOR EACH NUCLIDE ARE DETERMINED
BY THE SOURCE TERM PROGRAM. SEE OUTPUT FILE "\PCCOSYM2\RESULTS\SRC*.SRC".
WHERE * IS THE NAME OF THE *.ST1 FILE.

GROWTH OF THE FOLLOWING DAUGHTERS DURING DISPERSION IS CONSIDERED:

DAUGHTER PRODUCT NO.	NAME	PARENT NO.	NAME	YIELD

INPUT LIST OF THE SOURCE TERM

NUMBER OF RELEASE PHASES: 1
SHIFT OF THE STARTING POINT: 0 H

	DELAY (H)	THERMAL ENERGY (MW)	INITIAL RELEASE HEIGHT (M)	WIDTH (M)	SOURCE HEIGHT (M)
1. PHASE:	1	1.0600E+00	23	4.8000E+01	2.3000E+01

LIST OF THE AMOUNT OF ACTIVITY RELEASED FOR EACH NUCLIDE FOR EACH PHASE IN BQ

NUCLIDE	SUM	PHASE 1 T = 1H	PHASE T =

THE NUCLIDES AND THE AMOUNTS RELEASED ARE CALCULATED
USING THE SOURCE TERM PROGRAM. A PRINT OUT OF
THESE QUANTITIES IS FOUND IN FILE "\PCCOSYM2\RESULTS\SRC*.SRC"
WHERE * IS THE NAME OF THE *.ST1 FILE.

****
COUNTERMEASURES
****

CALCULATIONS ARE MADE ASSUMING NO COUNTERMEASURES

THE PARAMETER VALUES LISTED IN THIS SECTION ARE THOSE PC COSYMA USES
TO MODEL THIS ASSUMPTION

SHELTER THEN EVACUATE AUTOMATICALLY

THIS COUNTERMEASURE IS NOT CONSIDERED.

SHELTER THEN EVACUATE, ON DOSE LEVEL

THIS COUNTERMEASURE IS NOT CONSIDERED.

SHELTERING ON A DOSE LEVEL, NO EVACUATION

THIS COUNTERMEASURE IS NOT CONSIDERED.

SHELTERING IN A CIRCLE, NO EVACUATION

THIS COUNTERMEASURE IS NOT CONSIDERED.

STABLE IODINE TABLETS

THIS COUNTERMEASURE IS NOT CONSIDERED.

RELOCATION AND RESETTLEMENT

THIS COUNTERMEASURE IS NOT CONSIDERED.

LOCATION FACTORS

THERE ARE 4 TYPES OF LOCATION FACTORS, NOT ALL OF WHICH MAY APPLY TO THE
PATTERN OF COUNTERMEASURES AND POPULATION BEHAVIOUR IN THIS CALCULATION.
THE CORRESPONDING GROUPS AND LOCATION FACTORS ARE:

	L O C A T I O N F A C T O R S				
	CL	GR	IH	IHR	SK
NORMAL ACTIVITY	1.000E+00	1.000E+00	1.000E+00	1.000E+00	1.000E+00
INITIAL DELAY	1.000E+00	1.000E+00	1.000E+00	1.000E+00	1.000E+00
SHELTERING (BUILDINGS)	1.000E+00	1.000E+00	1.000E+00	1.000E+00	1.000E+00
CARS	1.000E+00	1.000E+00	1.000E+00	1.000E+00	1.000E+00

****
SHORT TERM DOSES AND RISKS
****

THE FOLLOWING PATHWAYS ARE TAKEN INTO ACCOUNT
FOR EARLY DOSE AND RISK CALCULATIONS

- CLOUDSHINE : YES
- GROUNDSHINE : YES
- INHALATION : YES
- RESUSPENION : YES
- SKIN/CLOTHES : YES

PARAMETERS OF THE DOSE-RISK RELATIONSHIPS

MORBIDITY

EFFECT	SHAPE PARAMETER	THETA-1 GY**2/H	THETA-INFINITY GY
LUNG IMPAIRMENT	7.0000E+00	1.5000E+01	5.0000E+00
HYPOTHYROIDISM	1.3000E+00	3.0000E+01	6.0000E+01
CATARACTAE	5.0000E+00	1.0000E-02	3.0000E+00
MENTAL RETARD.	1.0000E+00	0.0000E+00	1.5000E+00
EFFECTS ON SKIN	5.0000E+00	5.0000E+00	2.0000E+01

MORTALITY

EFFECT	SHAPE PARAMETER	THETA-1 GY**2/H	THETA-INFINITY GY
PULMONARY SYNDR	7.0000E+00	3.0000E+01	1.0000E+01
HEMATOP.SYNDR.	6.0000E+00	1.0000E-01	4.5000E+00
GASTROINT.SYNDR	1.0000E+01	0.0000E+00	1.5000E+01
PRE/NEON. DEATH	3.0000E+00	0.0000E+00	1.5000E+00
SKIN BURNS	5.0000E+00	5.0000E+00	2.0000E+01

RATIO OF ACTIVITY CONCENTRATIONS ON SKIN TO DRY DEPOSITED ACTIVITY
CONCENTRATIONS ON GROUND SURFACE

```
              1  NOBLE GASES:         1.0000E+00
              2  AEROSOLS:            1.0000E+00
              3  ELEMENTAL IODINE:    1.0000E+00
              4  ORG. BOUND IODINE:   1.0000E+00


*************************************************************************
                  LONG TERM DOSES AND RISKS
*************************************************************************

   THE FOLLOWING PATHWAYS ARE TAKEN INTO ACCOUNT
   FOR LATE DOSE AND RISK CALCULATIONS

        - CLOUDSHINE   : YES
        - GROUNDSHINE  : YES
        - INHALATION   : YES
        - INGESTION    : NO
        - RESUSPENION  : YES
        - SKIN/CLOTHES : YES


   INTEGRATION TIME FOR SKIN DOSES  (DAYS) :  1.0000E+04
   FRACTION OF SKIN CONTAMINATED           :  1.0000E-01


   THE FOLLOWING PARAMETERS ARE USED TO CALCULATE THE
   STOCHASTIC HEALTH EFFECTS:

                   CANCER MORTALITY     CANCER MORTALITY
                   FACTORS (PER SV)         FRACTIONS

     BONE MARROW       5.1600E-03            1.0000E+00
     BONE SURF.        1.3300E-04            1.0000E+00
     BREAST            8.0000E-03            4.0000E-01
     LUNG              9.0000E-03            7.5000E-01
     STOMACH           9.0500E-03            8.5000E-01
     COLON             3.4300E-03            5.5000E-01
     LIVER             4.6700E-03            1.0000E+00
     PANCREAS          5.2600E-03            9.0000E-01
     THYROID           1.7700E-03            1.0000E-01
     REMAINDER         3.8600E-03            6.0000E-01
     SKIN              1.3800E-04            1.0000E-02

     HERED. EFF.       1.0000E-02


*************************************************************************
                              ECONOMICS
*************************************************************************



  HEALTH EFFECTS COSTS USE HUMAN CAPITAL APPROACH

  CURRENCY UNIT USED = DM


  EARLY HEALTH EFFECTS COSTS
  **************************


  IMOR =  1: COST CATEGORIES = MORBIDITY AND MORTALITY

  OPTION TO BE USED FOR COST CALCULATIONS:  OPTION 2  -  HUMAN CAPITAL APPROACH
```

```
   UNIT COSTS (ADJUSTED):

                                                    MEDICAL TREATMENT COSTS,
                                                    UEHTR(NOG,MOR)  (M.U./CASE)
                                                       (1,1) = 3.6000E+03
                                                       (2,1) = 7.6000E+03
                                                       (3,1) = 2.5000E+03
                                                       (4,1) = 0.0000E+00
                                                       (5,1) = 7.9000E+03
                                                       (1,2) = 1.3300E+04
                                                       (2,2) = 2.0200E+05
                                                       (3,2) = 1.0300E+05
                                                       (4,2) = 0.0000E+00
                                                       (5,2) = 1.0300E+05

                                                    LOSSES-TO-ECONOMY COSTS,
                                                    UCLOE = 3.3000E+04 (M.U./CASE-
YR)

   GENERAL INPUT VALUES:

                                                    EFF. DURATION OF LOSSES-TO-
ECONOMY,
                                                    DUREH(NOG,MOR)  (YR)
                                                       (1,1) = 1.0000E-01
                                                       (2,1) = 1.0000E-01
                                                       (3,1) = 1.0000E-01
                                                       (4,1) = 0.0000E+00
                                                       (5,1) = 1.0000E-01
                                                       (1,2) = 3.9000E+01
                                                       (2,2) = 3.9000E+01
                                                       (3,2) = 3.9000E+01
                                                       (4,2) = 0.0000E+00
                                                       (5,2) = 3.9000E+01

      DISCOUNT RATE,  DISHE = 3.0000E+00 (%/YR)


   LATE HEALTH EFFECTS COSTS
   *************************


   OPTION TO BE USED FOR COST CALCULATIONS:  OPTION 2  -  HUMAN CAPITAL APPROACH

                                                    FATAL AND NON-FATAL CANCERS
HEREDITARY EFFECTS

   UNIT COSTS (ADJUSTED):

                                                    MEDICAL TREATMENT COSTS,
                                                    ULHTC(NOG,MOR)  (M.U./CASE)
                                                       (1,1) = 9.4000E+04
                                                       (2,1) = 6.9000E+04
                                                       (3,1) = 1.1000E+04
                                                       (4,1) = 7.6000E+04
                                                       (5,1) = 7.6000E+04
                                                       (6,1) = 1.1000E+04
                                                       (7,1) = 9.4000E+04
                                                       (8,1) = 7.2000E+03
                                                       (1,2) = 9.4000E+04
                                                       (2,2) = 6.9000E+04
                                                       (3,2) = 1.1000E+04
                                                       (4,2) = 7.6000E+04
                                                       (5,2) = 7.6000E+04
                                                       (6,2) = 1.1000E+04
                                                       (7,2) = 9.4000E+04
                                                       (8,2) = 7.2000E+03
```

```
                                                    LOSSES-TO-ECONOMY COSTS,
                                                    UCLOE = 3.3000E+04(M.U./CASE-
YR)


MEDICAL TREATMENT COSTS,

ULHTH = 6.5000E+05 (M.U./CASE)


LOSSES-TO-ECONOMY COSTS

UCLOE = 3.3000E+04 (M.U./CASE-YR)

   GENERAL INPUT VALUES:

                                                    EFF. DURATION OF MED.
TREATMENT,
                                                    TTRCLH(NOG,MOR)  (YR)
                                                       (1,1) = 5.0000E-01
                                                       (2,1) = 5.0000E-01
                                                       (3,1) = 5.0000E-01
                                                       (4,1) = 5.0000E-01
                                                       (5,1) = 5.0000E-01
                                                       (6,1) = 5.0000E-01
                                                       (7,1) = 5.0000E-01
                                                       (8,1) = 5.0000E-01
                                                       (1,2) = 1.0000E+00
                                                       (2,2) = 1.0000E+00
                                                       (3,2) = 1.0000E+00
                                                       (4,2) = 1.0000E+00
                                                       (5,2) = 1.0000E+00
                                                       (6,2) = 1.0000E+00
                                                       (7,2) = 1.0000E+00
                                                       (8,2) = 1.0000E+00

EFF. DURATION OF MED. TREATMENT

+ NO. OF YEARS OF LIFE LOST,

TTLHLH = 4.0000E+01 (YR)

      DISCOUNT RATE,  DISHE = 3.0000E+00 (%/YR)
```

Table 6.2.A-5: INITIAL CORE INVENTORY AND STEADY STATE COOLANT ACTIVITY IN THE PRIMARY SYSTEM

Nuclide	Half–life	Initial Inventory [Bq]	PBMR-Adjusted [Bq] 2
Kr – 83m	1.83h	4.47E+16	7.80E+09
Kr – 85m	4.48h	1.01E+17	2.40E+10
Kr – 85	10.76a	1.64E+15	1.00E+08
Kr – 87	76.3m	2.05E+17	2.90E+10
Kr – 88	2.80h	2.89E+17	5.60E+10

Nuclide	Half–life	Initial Inventory [Bq]	PBMR-Adjusted [Bq] 2
Kr – 89	3.18m	3.70E+17	1.20E+10
Kr – 90	32.3s	4.03E+17	5.00E+09
Xe – 131m	12.0d	2.93E+15	4.60E+08
Xe – 133m	2.2d	1.82E+16	4.60E+09
Xe – 133	5.29d	6.11E+17	9.10E+10
Xe – 135m	15.3m	1.12E+17	6.80E+09
Xe – 135	9.17h	7.07E+16	5.60E+10
Xe – 137	3.83m	5.52E+17	1.90E+10
Xe – 138	14.1m	5.58E+17	3.60E+10
Xe – 139	39.7s	4.37E+17	5.90E+09
I – 131	8.04d	2.71E+17	5.20E+05
I – 132	2.38h	4.13E+17	7.00E+06
I – 133	20.8h	6.00E+17	3.40E+06
I – 134	52.0m	6.90E+17	1.70E+07
I – 135	6.59h	5.69E+17	5.90E+06
Rb – 88	17.8m	2.95E+17	2.70E+08
Sr - 90	28.5a	1.42E+16	474.7
Cs – 134	2.06a	1.62E+16	7.80E+03
Cs – 137	30.1a	1.56E+16	1.60E+04
Cs – 138	32.2m	4.22E+17	9.40E+07
Ag – 110m	250.4d	1.59E+14	591.6
H – 3	12.35a	6.50E+13	1.10E+11
C – 14	5736a	1.70E+07	1.20E+09

Table 6.2.A-6: TOTAL ACTIVITIES DEPOSITED IN THE HELIUM CIRCUIT AFTER 32 EFPY

Nuclide	Half-life	PBMR-Adjusted[6] (Bq) 2
I – 131	8.04d	7.00E+10
I – 132	2.38h	1.10E+10
I – 133	20.8h	5.00E+10
I – 134	52.0m	1.10E+10
I – 135	6.59h	2.80E+10

[6] These values are design values

Rb – 88	17.8 m	2.49e+09
Sr - 90	28.5 a	4.40E+09
Cs – 134	2.06a	9.90E+10
Cs – 137	30.1a	1.62E+12
Cs – 138	32.2 m	1.60E+09
Ag – 110m	250.4 d	2.59E+09

Table 6.2.A-7: DEFAULT MODEL PARAMETERS FOR EARLY HEALTH EFFECTS MODEL

Dose In Organ/Tissue				Parameters for Calculating Median Dose D_{50}			
External Irradiation	Internal Irradiation	Effect	Shape **parameter S**	D_{00} Gy	D_0 GY^2/hour	Times over which doses are integrated (days)	RBE
		Mortality					
Lung	Lung	Pulmonary syndrome	7.0	10.0	30.0	1, 7, 30, 365	7.0
Red bone marrow	Red bone marrow	Haematopoietic syndrome	6.0	4.5	0.1	1, 7, 30	2.0
Remainder[2]	-	Gastrointestinal syndrome	10.0	15.0	0.0	1, 7	-
Ovaries	Uterus	Pre-and neonatal death [3]	3.0	1.5	0.0	1, 30	20.0
Skin	-	Death from skin burns [4]	-	-	-	-	-
		Morbidity					
Lung	Lung	Lung function impairment	7.0	5.0	15.0	1, 7, 30, 365	7.0
Thyroid	Thyroid	Hypothyroidism	1.3	60.0	30.0	1, 7, 30	0.0
Skin	-	Skin burns	5.0	20.0	5.0	1, 7, 30	0.0
Eye [5]	-	Cataracts	5.0	3.0	0.01	1, 7, 30	0.0
Ovaries	Uterus	Mental retardation [3]	1.0	1.5	0.0	1, 30	20.0

Notes

1. In accidents involving releases of α-emitters, the absorbed dose should be multiplied by a factor to account for the relative biological effectiveness of acute α-irradiation.
2. Since the dose conversion factors for the GI-trace were not included in the GSF-data, the calculations are based on the dose to the remainder organs.
3. Risks predicted using these parameters have to be multiplied by 0.02 to be applicable to the general population, to allow for the fraction of that population who are pregnant at any one time.
4. Risk of death from skin burns is calculated assuming that a fraction of the people burned will die from those burns.
5. Dose to the eye is assumed to be that to the skin.

Table 6.2.A-8: DEFAULT RISK COEFFICIENTS FOR LATE HEALTH EFFECTS

Organ	Effect	Fraction of Cancers which are Fatal	Risk per Sv[1]	
			Based on GSF	Based on ICRP-60
Bone marrow	Leukaemia	1.0	5.16×10^{-3}	5.0×10^{-3}
Bone surface	Cancer	1.0	1.33×10^{-4}	5.0×10^{-4}
Breast	Cancer	0.4	8.00×10^{-3}	2.0×10^{-3}
Lung	Cancer	0.75	9.00×10^{-3}	8.5×10^{-3}
Stomach	Cancer	0.85	9.05×10^{-3}	1.1×10^{-3}
Colon	Cancer	0.55	3.43×10^{-3}	8.5×10^{-3}
Liver	Cancer	1.0	4.67×10^{-3}	1.5×10^{-3}
Pancreas	Cancer	0.9	5.26×10^{-3}	2.6×10^{-3} (2)
Thyroid	Cancer	0.1	1.77×10^{-3}	8.0×10^{-4}
Skin	Cancer	0.01	1.38×10^{-4}	2.0×10^{-4}
Remainder	Cancer	0.6	3.86×10^{-3}	9.3×10^{-3} (3)
Total			5.05×10^{-2}	5.0×10^{-2}
Gonads	Hereditary		2.0×10^{-2}	2.0×10^{-3}

Notes:

4. For leukaemia and cancer, these give the risk of a fatal effect. For hereditary effects, it gives the risk of seven effects in all future generations.

5. ICRP does not consider this organ. The value is obtained from the GSF value using a DDREF of 2.0.

6. This covers all organs not identified above, and is not the same as the ICRP remainder.

APPENDIX REFERENCES

1. J A Jones, P A Mansfield, and S M Haywood (NRPB); I Hassemann, C Steinhauer, J Ehrhardt and D Faude (KfK): *PC COSYMA Version 2: An Accident Consequence Assessment Package For Use on a PC*, EUR 16240 (1995).

2. Koeberg Nuclear Power Station, Safety Analysis Report, Rev. 1a, 1998, Figure F-II-3.4-1.

RADIOACTIVE MATERIALS AND RADIOLOGICAL PROTECTION

6.0.12 EXECUTIVE SUMMARY

This chapter presents a preliminary overview of the estimated quantities of radioactive material, on major components in the Pebble Bed Modular Reactor (PBMR). The values quoted within this document are based on the estimated activities stated in International guidelines **30** and German experience **29**, **33**, and where applicable on actual modelled estimates. As the design of the PBMR matures and further details are developed, the radiological evaluation will also be refined, and included in a detailed radiological evaluation. In the future revision, the radiation doses for the operator during normal and outage periods will be estimated. It is expected that the radiation doses will be well within the regulatory requirements **42**. Until this detailed evaluation is conducted, this chapter only meets the broad outline of the licensing documentation.

The Radiation Protection Programme and Waste Management Programme will be consistent with the basic licensing requirements for the PBMR, as described in the NNR Licensing Guide, LG-1037, and will ensure compliance with the fundamental safety requirements relating to a system to safeguard personnel and the public against radiological hazards for normal operation, based on the following principles and objectives:

- All exposures of site personnel and the general public shall be kept As Low As Reasonably Achievable (ALARA), taking into account the resulting Total Effective Dose Equivalent (TEDE), together with economic and social factors.

- The dose to individuals shall not exceed the effective and equivalent dose limits detailed in **44** To achieve this, radiological protection and radioactive waste management programmes shall be established to control occupational, public, potential, chronic and emergency exposures.

- The defence-in-depth concept described in **45** shall be applied in the operational radiological protection and the radioactive waste management programmes.

The component layout has not yet been fully finalized, hence equipment location cannot be quoted, and only the major components are referenced. This will be finalized in a later revision.

6.0.13 INTRODUCTION

This chapter addresses normal operating conditions. The high retention of radiologically significant fission products by the coated fuel particles has been studied extensively in the German fuel development programme **42**. The high degree of safety and the low source term of the PBMR are a consequence of this ability of the coated fuel particles to retain fission products, even at high temperatures.

For the purposes of identifying radiation fields and designing the radiation protection programme for the PBMR, the following nuclear systems have been considered as the main sources of radiation:

- reactor and Power Conversion Unit (PCU);
- fuel handling equipment;
- primary coolant-conveying systems; and
- water-carrying systems.

The strongest radiation field is that around the reactor during operation. It determines the design of the shielding around the reactor and the PCU. Apart from gamma radiation, neutron radiation is also significant. After reactor shutdown, only gamma radiation need be considered in shielding design. The same applies to all non-fuel-containing systems for all operating states.

The sources of all other radiation fields also originate from the reactor. They are products either of nuclear fission in the fuel, or of activation in the radiation field of the reactor.

Other radiation fields are primarily caused by:

- Fission products - in the fuel elements and as contaminants in the primary coolant.
- Activation products - in the structural material and surrounding systems around the reactor core.

Nearly 100% of the total quantity of fission products is retained in the fuel elements **42**, which form the main radiation source in the fuel handling equipment. The main components of the fuel handling equipment are located underneath the reactor cavity

in the fuel discharge compartment.

The noble fission product gases and highly volatile fission products form the basis of the primary coolant activity. These fission products primarily originate from the small fraction of failed particles caused by manufacturing and irradiation-induced defects. The main activity-carrying components of the helium purification system are the dust removal filters, the molecular sieves, and the helium storage tank. They are housed in the reactor building.

The Active Cooling System (ACS) can contain radioactivity during operation. As the Reactor Cavity Cooling System (RCCS) is located in the radiation field of the reactor, radioactive isotopes are produced by activation of the water and any impurities present in it, and by activation of the structural materials followed by corrosion. This then causes a radiation field in areas where this cooling water is located. Any leaks or deposition from this system may also cause areas of contamination.

6.0.14 SOURCE TERMS

6.0.14.1 The Reference Nuclide Mixture

Due to the very large fission product inventory of the equilibrium reactor core, it is necessary to select a smaller radionuclide mixture to model the migration of core nuclides to plant components.

There is no available standardized mixture for the evaluation of releases for High Temperature Gas Cooled Reactors (HTGR) **28**.

To establish a model, the findings from comparable cases, such as the operating experience from the German AVR and Thorium High Temperature Reactor (THTR) plants, have been used.

The nuclides were selected on the basis of nuclide specific data such as inventory, half-life, volatility and radiotoxicity. Plant-specific activity barriers and technical procedures for the treatment of gaseous waste are also considered.

To calculate the potential annual radiation exposure by radionuclide releases from the PBMR to the environment, the reference nuclide mixture in ***Table 6.3-1*** was used.

The following radionuclide groups are represented in the reference nuclide mixture:

- fission noble gases (Kr, Xe);
- halogens (I-131 to I-135);
- long lived aerosols (Co-60, Sr-90, Ag-110m , Cs-134 , Cs-137); and
- other important radionuclides (Ar-41, C-14 , H-3).

Due to the high retention property of the PBMR fuel element, the fission product releases remain limited to a small fraction of the core inventory. The fission product inventory for the PBMR equilibrium core has been calculated **27**. The fission product inventories listed in ***Table 6.3-2*** are obtained from **27** and show a good correlation to the fission product core inventory calculated for the HTR-Modul listed in **30**. The selection of radionuclides is based on the recommended reference nuclide mixture in **27**. The fission noble gases, halogens and solids (long-lived aerosols) have a relatively small contribution to exposure on the waste air route. Carbon-14 and tritium, on the other hand, play a more significant role.

Ar-41 which is produced by air activation is added as a dose regulating nuclide. Thus the mixture, H-3, C-14, Ar-41 plays a dominant role in the potential radiation exposure to the environment.

In addition to the reference radionuclide mixture compiled in ***Table 6.3-1***, it was found that the dose contribution by alpha-emitters is insignificant according to operational experiences on the AVR and THTR. This also applies to bromine-82, which was detected in the AVR waste air **28**.

TABLE 6.3-1: STANDARD NUCLIDE MIXTURE

Nuclide	Half-life
Ar-41	1.83 h
Kr-83m	1.83 h
Kr-85m	4.48 h
Kr-85	10.76 y
Kr-87	76.3 m
Kr-88	2.80 h
Kr-89	3.18 m
Xe-131m	12.0 h
Xe-133m	2.20d
Xe-133	5.29 d

Xe-135m	15.30 m
Xe-135	9.17 h
Xe-137	3.83 m
Xe-138	14.1 m
I-131	8.04 d
I-132	2.38 h
I-133	20.8 h
I -134	52.0 m
I-135	6.59 h
Co-60	5.27 y
Ag-110m	250.4 d
Cs-134	2.06 y
Cs-137	30.10 y
Sr-90	28.5 y
C-14	5 730 y
H-3 (HTO)	12.28 y

TABLE 6.3-2: SIGNIFICANT FISSION PRODUCT INVENTORIES IN THE EQUILIBRIUM CORE (331 000 FUEL ELEMENTS)

Radionuclide	Half-life	Core Inventory (Bq)
Kr-83m	1.83 h	4.47×10^{16}
Kr-85m	4.48 h	1.01×10^{17}
Kr-85	10.76 y	1.64×10^{15}
Kr-87	76.3 m	2.05×10^{17}
Kr-88	2.80 h	2.89×10^{17}
Kr-89	3.18 m	3.70×10^{17}
Kr-90	32.3 s	4.03×10^{17}
Xe-131m	12.0 d	2.93×10^{15}
Xe-133m	2.2 d	1.82×10^{16}
Xe-133	5.29 d	6.11×10^{17}
Xe-135m	15.3 m	1.12×10^{17}
Xe-135	9.17 h	7.07×10^{16}
Xe-137	3.83 m	5.52×10^{17}
Xe-138	14.1 m	5.58×10^{17}
Xe-139	39.7 s	4.37×10^{17}
I-131	8.04 d	2.71×10^{17}
I-132	2.38 h	4.13×10^{17}
I-133	20.8 h	6.00×10^{17}

Reference: SAR Rev 1 Section 6

Radionuclide	Half-life	Core Inventory (Bq)
I-134	52.0 m	6.90×10^{17}
I-135	6.59 h	5.69×10^{17}
Cs-134	2.06 y	1.62×10^{16}
Cs-137	30.1 y	1.56×10^{16}
Ag-110m	250.4 d	1.59×10^{14}
Sr-90	28.5 y	1.42×10^{16}
H-3	12.28 y	6.5×10^{13}
C-14	5 730 y	1.7×10^{7}

6.0.14.2 Radioactive Materials in the Primary Cycle

6.0.14.2.1 Release of radioactivity by the fuel elements

The release of activity in the coolant can be divided into the following according to **30** and **33**:

- activity release by intact particles;
- activity release due to particles damaged during manufacture;
- activity release due to particle failure as a result of irradiation; and
- activity release from contamination of the graphite matrix.

In intact particles, the coating of the particles forms an effective barrier under normal operating conditions, so that no relevant release of activity occurs at the surface of the fuel elements.

During manufacture of the fuel elements, it cannot be excluded that a small number of defective particles may occur. It is estimated that the failure fraction would be 3×10^{-5} **33**. A conservative design value of double this fraction, 6×10^{-5}, will be used.

If the particle coating is defective, then the barrier which is supposed to prevent activity release, is no longer effective. Fission and activation products are consequently transported by diffusion from their point of origin in the grains of the fuel kernels to the grain boundary. Subsequently, these products are directly transported by grain boundary diffusion into the graphite matrix, and from there to the surface of the fuel element. The release of gaseous fission products is only delayed because

the diffusion rate of these products in the fuel kernel grains is so slow. Diffusion along grain boundaries and in the graphite matrix is relatively fast. The diffusion rates of solid fission and activation products in fuel kernels and graphite matrix are relatively slow, which leads to a delay in release, and therefore results in a reduction in the quantity of released fission products, particularly of isotopes with a short half-life.

During utilization of the fuel elements in the reactor, additional particle defects have to be taken into consideration. These are caused by the effects of burn-up, fast neutron fluence or temperature. It is estimated that the fraction of irradiation-determined defective particles will be 2×10^{-5}. This value which represents the 95% confidence level, was derived from radiation tests which were carried out with approx. 200 000 particles. None of these particles exhibited any defects. For design purposes, a more conservative estimate which is 10 times higher, namely 2×10^{-4}, is used. The transport of the fission and activation products from the fuel kernels to the fuel element surface occurs in similar fashion to that which was described for particles damaged during manufacture.

The natural graphite in the matrix is contaminated with uranium. This uranium contamination which occurs naturally in graphite, causes a limited number of uranium fissions in the graphite matrix, i.e. outside the coated particles.

The fission and activation products of this uranium fission can also be transported from the graphite matrix via diffusion to the fuel element surface in analogous fashion to the above-mentioned mechanism.

A ‘free uranium inventory’ was determined by combining the uranium arising from matrix contamination, and from particles damaged during manufacture. This ‘free uranium inventory’ was used in relation to the maximum permissible amounts in fuel elements and for the calculation of the release rate. The design value of 6×10^{-5}, which was mentioned in connection with the fraction of manufacturing-determined defective particles, includes the uranium contamination of the graphite matrix. As the same release mechanism is responsible for the release from manufacturing-determined defective particles and from the graphite matrix, both source terms are combined in the concept of ‘free uranium’, which is used to calculate the release rate at the fuel element surface.

In addition to uranium contamination, the graphite matrix is also contaminated with Ag-109. Under the influence of neutron radiation, the isotope Ag-110m (among other

products), is generated from Ag-109 by the (n, γ)-reaction. Ag-110m is released in appreciable quantities by fuel elements in analogous fashion to fission product release due to uranium contamination.

The fission product release rates from fuel elements in the reactor core for a number of important nuclides, classified according to nuclide groups, are listed in ***Table 6.3-3***. The reason for the selected nuclides is that they are responsible for the most important, i.e. radiologically relevant, contribution to release of activity into the environment.

The release rates in ***Table 6.3-3*** are obtained by multiplying the HTR-Modul values in **29** by the factor 1.34. This factor is the ratio of the PBMR power of 268 MW divided by the HTR-Modul power of 200 MW.

The release rate of cesium-137 in ***Table 6.3-3*** shows a good comparison with the release rate calculated for PBMR by **35**.

Except for Ag-110m, which shows a higher release rate by an order of magnitude calculated by **35**, the values for Sr-90 and Cs-134 are lower than the expected values in ***Table 6.3-3***.

As soon as uncertainties in the input data of the Ag-110m calculations have been resolved, a further calculation will follow.

The fission and activation products which were not included, do not contribute significantly to radiation exposure due to their low activity and shorter half-life. They are retained by the fuel element particles and graphite, and have a less pronounced biological effect.

The release rates for the PBMR will be calculated using the Booth model **30** and FRESCO-II code **30, 31**. The Booth model is most commonly taken to determine the ratio of the Release rate (R) to the Birth rate (B), (R/B) of fission products. (R/B) is a function of the kernel diffusion coefficient, the decay constants of the fission product isotopes, and of the surface-to-volume ratio, (S/V = 3/a), of a sphere equivalent to the representative grain of fuel with radius a.

In practice, the kernel is often regarded as a solid sphere, and the diffusion coefficient derived from experimental data using the kernel radius. The Booth model is applicable only to the diffusive release of fission products from kernels and from matrix graphite grains (which contain the heavy metal contamination) as simulated by

equivalent spheres. The release rates of fission gases such as Kr, Xe and I are calculated by this model, and can also be used for gaseous precursor nuclides of radiologically-relevant, long-lived metallic fission products (Xe-137, Kr-89, Kr-90).

The FRESCO-II code calculates the transport of fission products in Pebble Bed Reactors on the basis that diffusion phenomena control the transport. Sorption of metallic fission products on graphite is also considered.

While the FRESCO-II code was specifically developed for spherical HTR fuel elements, that portion treating coated particles themselves applies to both pebble-bed and prismatic fuel elements. In general, the FRESCO-II code assumes that:

- diffusion phenomena controls transport;
- gases behave as ideal gases;
- reactions between fission products are not important;
- reactions between fission products and coolant impurities are not important;
- diffusion rates of all isotopes of a given nuclide are the same, and
- desorption and absorption of metallic fission products at the coolant-graphite surfaces can vary with fission product concentration, in accordance with empirical adsorption isotherms.
- adsorption on graphite applies to fission-product metals (fission gases such as noble gases and iodine are not absorbed on graphite in a practical sense).
- In the calculations, spherical symmetry is assumed, with the coated particles being homogeneously distributed in the fuel region of the fuel sphere.

Further assumptions are as follows:

- Temperature differences between fuel particles and the matrix graphite at a given location are very small relative to their influence on values of diffusion coefficients.
- The graphite matrix consists of graphite ‘crystals’ plus a graphite ‘binder’ material.
- A part of the uranium contamination in the graphite matrix is in the crystals, with

the remainder in the binder connecting the crystals.

- Fission products that escape the coated particles remain only in the binder, and do not enter the crystals (because diffusion of fission products through the binder is much more rapid than through the crystals).

- Fission-product transport from the crystals to the binder follow the relatively slow 'volume diffusion' process.

- Fission-product transport through the binder follows the relatively fast 'pore-diffusion' process.

TABLE 6.3-3: RELEASE RATE OF RADIONUCLIDES INTO THE PRIMARY SYSTEM

Radionuclide	Half-life	Design Bq/s	Expected Bq/s
Kr-83m	1.83 h	3.0×10^{6}	8.9×10^{5}
Kr-85m	4.48 h	4.4×10^{6}	1.3×10^{6}
Kr-85	10.76 y	6.0×10^{3}	1.5×10^{3}
Kr-87	76.3 m	1.6×10^{7}	4.8×10^{6}
Kr-88	2.80 h	1.5×10^{7}	4.6×10^{6}
Kr-89	3.18 m	1.3×10^{8}	4.2×10^{7}
Kr-90	32.3 s	3.4×10^{8}	1.0×10^{8}
Xe-131m	12.0 d	2.2×10^{4}	6.6×10^{3}
Xe-133m	2.2 d	2.8×10^{5}	8.1×10^{4}
Xe-133	5.29 d	4.8×10^{6}	1.4×10^{6}
Xe-135m	15.3 m	1.8×10^{7}	5.2×10^{6}
Xe-135	9.17 h	6.7×10^{6}	1.9×10^{6}
Xe-137	3.83 m	1.9×10^{8}	5.7×10^{7}
Xe-138	14.1 m	1.0×10^{8}	3.1×10^{7}
Xe-139	39.7 s	3.5×10^{8}	1.0×10^{8}
Total noble gases	**-**	**1.2×10^{9}**	**3.6×10^{8}**
I-131	8.04 d	2.8×10^{5}	7.1×10^{4}
I-132	2.38 h	3.6×10^{6}	8.9×10^{5}
I-133	20.8 h	1.8×10^{6}	$4.5x\ 10^{5}$
I-134	52.0 m	9.7×10^{6}	2.4×10^{6}
I-135	6.59 h	3.3×10^{6}	8.2×10^{5}
Total iodines	**-**	**1.9×10^{7}**	**4.6×10^{6}**
Sr-90	28.5 y	1.8×10^{1}	6.3×10^{0}
Cs-134	2.06 y	2.1×10^{3}	1.0×10^{3}
Cs-137	30.1 y	4.4×10^{3}	2.2×10^{3}

Radionuclide	Half-life	Design Bq/s	Expected Bq/s
Ag-110m	250.4 d	1.6×10^{2}	8.2×10^{1}
Total long half-life solids	**-**	**6.6×10^{3}**	**3.4×10^{3}**

6.0.14.2.2 Gaseous radioactive materials in the primary cycle

6.0.14.2.2.1 General

The radioactive materials which are released by the fuel elements into the coolant of the primary cycle or are formed in the primary cycle, are transported together with the primary cycle coolant from the core into the primary cycle.

During normal operation, a semi-stationary coolant activity is established, which can be calculated by using a balance equation which takes all the source and loss terms into account.

The release of activity from the fuel elements, which must be regarded as the most important source term, has already been described in the previous section.

An additional source of aerosol-bound radionuclides, is due to the decay of short-lived noble fission gases in the primary gas cycle. The importance of this source of radioactivity is secondary to that which arises from the fuel elements.

The most important losses are caused by radioactive decay, removal in the helium purification plant and removal via primary cycle leaks, and in the case of aerosol-bound fission products, by plate-out on the surface structures of the primary cycle. The main factors which reduce radioactivity are the helium purification plant in the case of the noble fission gases, while for iodine and aerosol-bound radionuclides, the metal surface structures of the primary cycle play a decisive role.

6.0.14.2.2.2 Noble gas primary coolant activity

For the noble fission gases, radioactive decay and removal by the helium purification plant are sink mechanisms. The PBMR design value for the purification rate according to **32** is 0.011 per hour, and for the HTR-Modul, 0.05 per hour. The loss term caused by release via primary gas leaks was not taken into consideration, because the time constant for leaks of 0.001 per day is so small in comparison to the purification constant of the helium purification plant of 0.011 per hour, that it can be

ignored.

Based on the release rates of noble fission gases from the fuel elements, the noble gas-primary coolant activity depicted in ***Table 6.3-4*** is obtained.

It must be noted here that the design activities in the following tables are from **33** and the expected activities from **29**. The HTR-Modul activities are adjusted by multiplying the activities by the factor 1.34. As mentioned earlier, this value is the ratio of 268 MW for the PBMR power and the 200 MW HTR-Modul.

It was necessary to follow the 'adjustment route', as a full migration model of radionuclides for the PBMR is not yet available.

TABLE 6.3-4: STEADY-STATE NOBLE GAS-COOLANT ACTIVITY

Radionuclide	Half-Life	Design Activity (Bq)	Expected Activity (Bq)
Kr-83m	1.83 h	2.5×10^{10}	7.8×10^{9}
Kr-85m	4.48 h	7.7×10^{10}	2.4×10^{10}
Kr-85	10.76 y	4.3×10^{8}	1.0×10^{8}
Kr-87	76.3 m	9.6×10^{10}	2.9×10^{10}
Kr-88	2.80 h	1.9×10^{11}	5.6×10^{10}
Kr-89	3.18 m	3.8×10^{10}	1.2×10^{10}
Kr-90	32.3 s	1.6×10^{10}	5.0×10^{9}
Xe-131m	12.0 d	1.6×10^{9}	4.6×10^{8}
Xe-133m	2.2 d	1.5×10^{10}	4.6×10^{9}
Xe-133	5.29 d	3.1×10^{11}	9.1×10^{10}
Xe-135m	15.3 m	2.3×10^{10}	6.8×10^{9}
Xe-135	9.17 h	1.9×10^{11}	5.6×10^{10}
Xe-137	3.83 m	6.4×10^{10}	1.9×10^{10}
Xe-138	14.1 m	1.2×10^{11}	3.6×10^{10}
Xe-139	39.7 s	2.0×10^{10}	5.9×10^{9}
Total noble gases		**1.2×10^{12}**	**3.5×10^{11}**

6.0.14.2.2.3 Coolant gas activities for metallic fission products and iodine

The fission products which are released during the normal operation of the service life of the reactor, or those which are formed by the decay of the short-lived noble gases, are preferentially deposited on the metal surfaces of the primary cycle, or are bound by graphite dust and then deposited in the dead areas of the primary cycle.

The primary gas-bound metallic fission products and iodine activities are small compared to the plated-out activity.

To determine the design activity in the primary gas of the HTR-Modul, a flat rate of 10% per circulation was adopted for the plate-out of all solids and iodine. For an HTR-Modul coolant throughput of approximately 85 kg/s and a circulating coolant mass of 1 450 kg, a plate-out constant of $6.2 \times 10^{-3} s^{-1}$ is used for the design scenario **29**, **33**.

Based on the above-mentioned assumptions, the coolant gas activity design values for the metallic fission products are shown in ***Table 6.3-5***, and those for iodine in ***Table 6.3-6***.

The expected steady-state noble gas and iodine coolant activities in ***Table 6.3-4*** and ***Table 6.3-6*** show a good agreement with the activities calculated for the PBMR by **34**.

The contribution for Sr-90, which is caused by the decay of Kr-90, was not shown separately, but is contained in the Sr-90 value depicted in ***Table 6.3-5***.

The primary gas activity of the noble gas decay products is depicted in ***Table 6.3-7***.

TABLE 6.3-5: STEADY-STATE SOLIDS-COOLANT ACTIVITY

Radionuclide	Half-life	Design Activity (Bq)	Expected Activity (Bq)
Cs-134	2.06 y	3.5×10^5	7.8×10^3
Cs-137	30.1 y	7.2×10^5	1.6×10^4
Ag-110m	250.4 d	2.5×10^4	5.8×10^2
Sr-90	28.5 y	2.8×10^3	4.7×10^1
Total solids	**-**	**1.1×10^6**	**2.4×10^4**

TABLE 6.3-6: STEADY-STATE IODINE-COOLANT ACTIVITY

Radionuclide	Half-Life	Design Activity (Bq)	Expected Activity (Bq)
I-131	8.04 d	4.6×10^7	5.2×10^5
I-132	2.38 h	6.0×10^8	7.0×10^6
I-133	20.8 h	2.9×10^8	3.4×10^6
I-134	52.0 m	1.5×10^9	1.7×10^7
I-135	6.59 h	5.2×10^8	5.9×10^6

Total iodine	**-**	**2.9 x 10^{9}**	**3.4 x 10^{7}**

TABLE 6.3-7: STEADY-STATE NOBLE GAS DECAY PRODUCTS-COOLANT ACTIVITY

Radionuclide	Half-life	Design Activity (Bq)	Expected Activity (Bq)
Rb-88	17.8 m	1.7 x 10^{10}	2.7 x 10^{8}
Rb-89	15.4 m	4.0 x 10^{9}	-
Rb-90	157 s	6.7 x 10^{9}	-
Sr-89	50.6 d	1.0 x 10^{5}	-
Cs-138	32.2 m	6.7 x 10^{9}	9.4 x 10^{7}
Cs-139	9.4 m	3.4 x 10^{9}	-
Ba-139	1.4 h	7.2 x 10^{7}	-
Total of noble gas decay products		**3.8 x 10^{10}**	**3.6 x 10^{8}**

6.0.14.2.2.4 Production and emission of carbon-14 (half life: 5 740 years)

The production and emission of carbon-14 by the PBMR is calculated and discussed in detail in **36** and **37**.

The following summary gives the nuclear reactions and release rates of C-14 described in the report.

In principle, the following C-14 source reactions take place in any High Temperature Reactor:

- N-14(n,p)C-14 in the coolant gas and in fuel spheres;
- O-17(n,α)C-14 in the UO_2 kernels of the fuel spheres; and
- C-13(n,γ)C-14 in the spheres and reflectors.

Due to the low isotopic abundance of O-17 (0.04%) and the high retention capability of TRISO coated particles (tritium release fraction < 10^{-5}), the O-17(n,α)C-14 reaction can be neglected. Equally the C-13 (n,γ) C-14 reaction does not contribute to the C-14 core release, because of its extremely low neutron cross section (3 10^{-4} barn) and the low corrosive weight loss of the fuel spheres (< 0.02%).

The only remaining reaction N-14 (n,p) C-14 takes place in the fuel spheres and in the coolant gas. The C-14 release from the fuel spheres is controlled by selective corrosion of C-14, which was produced by N-14 being sorbed at the inner surface of the matrix material.

The C-14 production rate caused by N-14 in the coolant is identical with its release rate.

In High Temperature Reactors, carbon-14 represents a unique position. It is the only radioactive nuclide that is quantitatively emitted to the environment, provided that it exists in the gaseous state. Therefore, the main source of gas-borne carbon, the nitrogen contamination of the coolant, has to be controlled, either by the Helium Purification System (HPS) (cryogenic unit), or by a strict reduction of the nitrogen ingress rate into the primary circuit. In this context, the containment function of the Main Power System (MPS) is inverted: the in-leakage of air must be minimized, not the out-leakage of helium.

The carbon-14 balance in the MPS is determined by:

The sources

a) neutron absorption of the nitrogen-14 (N-14) in the coolant gas; and

b) neutron absorption of N-14 in the fuel spheres followed by selective corrosion.

The sinks

c) molecular sieves of the HPS followed by stack discharge during regeneration cycles; and

d) leakage from the MPS.

Using expected input parameters, the C-14 emission rate grows continuously with time and attains 1.1×10^{5} Bq/s (3.5×10^{12} Bq p.a.) after 35 years.

This result is predominantly determined by the ingress of nitrogen into the MPS. The emission rate is reduced to 3.8×10^{4} Bq/s (1.2×10^{12} Bq p.a.) after 35 years, provided that the locks for fresh elements are evacuated in advance of the charging process, whereby the total nitrogen ingress rate is decreased by a factor of 3. The advantage of this measure is growing during the operation time of the reactor.

The C-14 emission rate after 35 years is reduced by an order of magnitude to:

1.0×10^4 Bq/s (3.2×10^{11} Bq p.a.), if the upper limit of the MPS leakage rate (0.1% per day), is assumed during the whole life of the reactor. In this case, the emission rate stays constant after approximately four years.

During the first two operation years, the C-14 emission rate is not significantly influenced by the above quoted variations of the leakage and the ingress rates.

The presently used expected nitrogen ingress rate of 2.3×10^{19} atoms/s must be reduced by at least a factor 10.This can be achieved by evacuating the FHSS locks and purging the primary circuit with helium evacuating the FHSS locks, and purging the primary circuit with helium after depressurizations (maintenance, initial going to power). A further strategy would be temporary nitrogen removal from the coolant with a mobile cryogenic unit once every six years.

6.0.14.2.2.5 Production and emission of tritium (half-life: 12.3 years)

The production and emission of tritium by the PBMR is calculated and discussed in detail in **37**. The following summary gives the nuclear reactions and release rates of tritium described in the report.

Five tritium sources are to be considered in a helium-cooled High Temperature Reactor:

- reaction He-3 (n, p) H-3 in the coolant;
- reaction Li-6 (n, α) H-3 in the fuel spheres and the reflector;
- ternary fission in the fuel spheres;
- reaction B-10 (n, 2α) H-3 in the control rods; and
- reaction B-10 (n, α) Li7 (n, nα) H-3 in the control rods.

The latter two sources are not treated here, because the tritium produced in the control rods will mostly be retained within the rods.

The He-3 (n,p) H-3 reaction rate is proportional to the He-3 inventory of the MPS. The MPS leak rate controls the tritium emission to the environment, and supplies the MPS with new feed atoms (fresh helium) for the further production of tritium.

The tritium generated by the lithium contamination of the fuel spheres can be retained within the matrix material to a certain degree. However, calculations have shown, that after 280 days under THTR conditions, already 84% of the tritium produced by the Li contamination is released from the fuel spheres. Therefore, it is assumed that all tritium produced by Li is released from the fuel spheres. On the other hand, it is conservatively assumed that the lithium contamination is entirely fixed in the fuel spheres. The maximum production rate occurs in the initial phase of the plant, when the Li contamination has not yet been burnt up. At equilibrium, the Li induced tritium production rate equals the supply rate of Li-6 with the fresh fuel spheres (after one year, approximately 70% of the Li-6 inventory of fresh fuel spheres have reacted to tritium).

The activation of Li-6 in the reflectors has been neglected. This is consistent with the general concept not to treat the interaction with the reflectors quantitatively at the present stage.

The most powerful tritium production takes place in the fuel by ternary fission. However, intact TRISO particles retain tritium practically 100%. Annealing experiments with irradiated pyrocarbon coated particles have shown that even BISO particles retain tritium at least at temperatures up to 1 000 °C efficiently Therefore, ternary fission tritium contributes to the core release only from failed coated particles and the uranium contamination. This source is therefore negligible.

The most effective tritium sources in the MPS are the neutron activation reactions with the He-3 contents of the coolant gas and with the lithium contamination of the fuel and graphite spheres. During the initial operation phase, the lithium contamination of the reflector also contributes significantly to the total tritium production. However, this contribution is to a large extent compensated by the sorption capacity of the graphite reflector. The interaction of the MPS tritium inventory with the reflector was not considered in the report.

Due to its relatively long radioactive half-life and its high mobility, tritium produced by the lithium contamination will be released from the spheres during their residence time in the core, more or less quantitatively. Therefore, on average, the lithium induced tritium production rate in the spheres equals its core release rate. The same holds true for the production rate of He-3 induced tritium.

In summary, the tritium balance in the MPS is determined by the following sources

and sinks:

- gas leakage from the MPS;
- extraction by the (HPS);
- sorption in graphite reflectors (not treated); and
- radioactive decay.

The expected initial tritium core release rate of 9.8 x 10^{14} atoms/s (1.7 x 10^{6} Bq/s or 5.4 x 1 0^{13} Bq p.a.) is caused to equal parts by the lithium contamination of the fuel spheres and by the He-3 content of the coolant gas. After 35 years of operation, this rate is reduced by a factor of 5.6.

The expected maximum tritium emission rate to the environment after one year (0.0015%/d leak rate) amounts to 1.6 x 10^{4} Bq/s or 5.0 x 1 0^{11} Bq p.a.

Due to the burn-up of the initial He-3 inventory of the coolant and the Li inventory of the reactor core, the expected emission rate after 35 years amounts to only 3.6 x 10^{3} Bq/s or 1.1 x 10^{11}Bq p.a. In the case of the expected leak rate, the MPS leakage contributes 16% to the total tritium emission and 84% is caused by the regeneration processes in the HPS.

Assuming the reference leak rate of 0.1%/d the emission rates increase from 1.7 x 10^{5} Bq/s or 5.05 x 1 0^{11} Bq p.a. after one year to 1.1 x 10^{5} Bq/s or 3.5 x 1 0^{12} Bq p.a. after 35 years, respectively. In this case, as much as 92% of the total emission is due to the leakage.

The most sensitive input parameter of the expected release rate of H-3 is the HPS regeneration loss. The above emission rate of 1.6 x 10^{4} Bq/s or 5.0 x 1 0^{11} Bq p.a. would be reduced by 42%, if only 50% of the given loss fraction of the regeneration cycles are used.

A tritium discharge rate of less than 4.3 x 10^{13} Bq p.a. is expected in the liquid waste storage of the HPS. After 35 years full power operation a total activity of 1.9 x 10^{14} Bq tritium (HTO) has accumulated in the waste water tank of the HPS. For a leak rate of 0.1%/d, this activity increases to 4.0 x 10^{14} Bq.

6.0.14.2.3 Plate-out of radioactive materials in the primary cycle

Source terms for surface activity are fission products which are released from the fuel

elements and are plated-out by adsorption on the metal surfaces of the primary cycle, particularly on the Recuperator, as well as dust which is formed by abrasion, and carburization which primarily forms deposits in the dead spaces of the primary cycle.

The code SPATRA in **30**, **31** is used by the PBMR to calculate the plate-out distribution of fission products in the primary circuit under normal operating conditions. The code combines ad-/desorption behaviour and mass transfer. The desorption rate is proportional to the lattice vibration frequency, and to an Arrhenius factor containing the desorption enthalpies. The assumption of an ad-/desorption equilibrium is a reasonable approach, because the desorption process is much faster than the diffusion process into the metal.

For the calculation of the fission product surface activity values of the HTR-Modul, a flat plate-out rate per circulation of 90% for the design scenario, and 10% for the estimated scenario was used. The corresponding plate-out constants equal 6.2×10^{-3} s^{-1}, for the estimated scenario, and 0.136 s^{-1}, for the design scenario.

The PBMR power adjusted design and expected plate-out values in ***Table 6.3-8*** show a very good comparison with the PBMR values, calculated using the SPATRA code, in ***Table 6.3-9*** and ***Table 6.3-10***. It must be noted that the source terms in this document after 32 FPY and 35 FPY are approximately the same (with the exclusion of Ag-110m in ***Table 6.3-9***).

The reason for this, is that in the design case ***Table 6.3-9***, the release rate of silver was taken from the German report in **46** as 1.56×10^{5} Bq/s.

For the calculation of the other plate-out values, the release rates in ***Table 6.3-3*** were used as inputs.

The design and expected activities after 35 years in some of the major components in the PCU are given in ***Table 6.3-9*** and ***Table 6.3-10***.

For HTGR designs, ^{137}Cs and ^{110m}Ag are the radiologically most relevant radioisotopes after 35 years. Plate-out depends, inter alia, on temperatures, flow paths, flow-rates and surface areas.

Turbo-compressor units

Two turbo-compressors are used in the PCU. The function of the Turbo-

compressor Unit is to provide a pressure increase in the thermo-dynamic cycle to overcome system pressure drop.

The design and expected activity in the Turbo-compressor Unit were calculated in **38**. ***Table 6.3-9*** and ***Table 6.3-10*** show the estimated activity in the Turbo-compressor Units for the significant nuclides.

Power turbine

The power turbine is used to convert part of the heat in the coolant gas to rotational energy for the purpose of driving the electric power generator.

The design and expected activities in the power turbine were calculated in **38**. ***Table 6.3-9*** and ***Table 6.3-10*** show the activity in the power turbine for the significant nuclides.

Recuperator

The Recuperator is used to transfer heat from the gas downstream of the power turbine to the reactor inlet gas. The activity levels in this unit were estimated for the PBMR in **38**. ***Table 6.3-9*** and ***Table 6.3-10*** the activity in the Recuperator for the significant nuclides.

Pre-cooler and Intercoolers

The PBMR utilizes a closed loop Brayton thermodynamic cycle. In the cycle, the Pre-cooler fixes the temperature of the low-pressure gas exiting the Recuperator and before the first compressor, while the Intercooler has the function of reducing the volumetric flow rate to the second compressor, causing a reduction in compressive work.

The design and expected values are provided in ***Table 6.3-9*** and ***Table 6.3-10*** as calculated in **38**.

TABLE 6.3-8: PLATE-OUT ACTIVITY ON THE PRIMARY CYCLE SURFACES AFTER 32 FPY

Radionuclide	Design Activity (Bq)	Expected Activity (Bq)
I-131	2.8×10^{11}	7.0×10^{10}
I-132	4.4×10^{10}	1.1×10^{10}
I-133	2.0×10^{11}	5.0×10^{10}

Radionuclide	Design Activity (Bq)	Expected Activity (Bq)
I-134	4.3×10^{10}	1.1×10^{10}
I-135	1.1×10^{11}	2.8×10^{10}
Total iodine	**6.8×10^{11}**	**1.7×10^{11}**
Rb-88	1.6×10^{11}	2.49×10^{9}
Rb-89	3.4×10^{10}	-
Rb-90	9.4×10^{9}	-
Sr-89	3.9×10^{9}	-
Cs-138	1.1×10^{11}	1.6×10^{9}
Cs-139	1.6×10^{10}	-
Ba-139	3.2×10^{9}	-
Total noble gas decay products	**3.4×10^{11}**	**4.0×10^{9}**
Sr-90	1.2×10^{10}	4.4×10^{9}
Cs-134	2.0×10^{11}	9.9×10^{10}
Cs-137	3.2×10^{12}	1.6×10^{12}
Ag-110m	5.0×10^{9}	2.5×10^{9}
Total of long half-life solids	**3.4×10^{12}**	**1.7×10^{12}**

Table 6.3-9: PLATE-OUT DISTRIBUTION IN THE PCU AFTER 35 FPY (DESIGN VALUES)

Description	Cs-134	Cs-137	Ag-110M Bq	I-131	I-133	Sr-90
* Core Outlet Pipe	4.6×10^{7}	7.9×10^{8}	1.6×10^{8}	6.7×10^{6}	4.6×10^{6}	7.8×10^{5}
High Pressure Turbine	1.6×10^{8}	2.7×10^{9}	5.6×10^{8}	2.3×10^{7}	1.6×10^{7}	2.7 x 106
Low Pressure Turbine	1.1×10^{8}	1.8×10^{9}	1.1×10^{9}	1.6×10^{7}	1.2×10^{7}	1.9×10^{6}
Power Turbine	9.6×10^{8}	6.6×10^{9}	7.5×10^{10}	4.9×10^{7}	7.9×10^{7}	1.4×10^{7}
Recuperator	1.9×10^{11}	3.1×10^{12}	5.5×10^{12}	2.6×10^{11}	1.8×10^{11}	1.3×10^{10}
Pre-cooler	1.1×10^{10}	1.9×10^{11}	3.3×10^{11}	2.0×10^{10}	1.1×10^{10}	7.8×10^{8}
Low Pressure Compressor	3.3×10^{8}	5.6×10^{9}	9.6×10^{9}	5.9×10^{8}	3.3×10^{8}	2.3×10^{7}
Inter Cooler	1.7×10^{9}	3.0×10^{10}	5.1×10^{10}	3.1×10^{9}	1.8×10^{9}	1.2×10^{8}
High Pressure Compressor	5.4×10^{7}	9.2×10^{8}	1.6×10^{9}	9.6×10^{7}	5.4×10^{7}	3.7×10^{6}
RX (HP)s	3.0×10^{8}	5.1×10^{9}	8.8×10^{9}	5.3×10^{8}	3.0×10^{8}	2.1×10^{7}
Core Inlet Pipes(e)	3.2×10^{5}	1.4×10^{6}	1.6×10^{7}	6.7×10^{3}	2.5×10^{4}	6.4×10^{3}
Total plate-out	2.0×10^{11}	3.4×10^{12}	5.9×10^{12}	2.8×10^{11}	1.9×10^{11}	1.4×10^{10}

Table 6.3-10: PLATE-OUT DISTRIBUTION IN THE PCU AFTER 35 FPY (EXPECTED VALUES)

Description	Cs-134	Cs-137	Ag-110m Bq	I-131	I-133	Sr-90
* Core Outlet Pipe	2.3×10^{7}	4.1×10^{8}	7.0×10^{4}	1.8×10^{6}	1.2×10^{6}	2.8×10^{5}
High Pressure Turbine	7.8×10^{7}	1.4×10^{9}	2.5×10^{5}	6.0×10^{6}	4.1×10^{6}	9.7×10^{5}
Low Pressure Turbine	6.0×10^{7}	1.0×10^{9}	5.5×10^{5}	4.5×10^{6}	3.3×10^{6}	7.3×10^{5}
Power Turbine	5.2×10^{8}	3.8×10^{9}	3.5×10^{7}	1.5×10^{7}	2.3×10^{7}	5.7×10^{6}
Recuperator	8.8×10^{10}	1.6×10^{12}	2.4×10^{9}	6.6×10^{10}	4.5×10^{10}	4.5×10^{9}
Pre-cooler	5.4×10^{9}	9.7×10^{10}	1.4×10^{8}	5.0×10^{9}	2.9×10^{9}	2.7×10^{8}
Low Pressure Compressor	1.6×10^{8}	2.8×10^{9}	4.2×10^{6}	1.4×10^{8}	8.2×10^{7}	7.9×10^{6}
Intercooler	8.3×10^{8}	1.5×10^{10}	2.2×10^{7}	7.7×10^{8}	4.4×10^{8}	4.2×10^{7}
High Pressure Compressor	2.6×10^{7}	4.6×10^{8}	6.9×10^{5}	2.4×10^{7}	1.4×10^{7}	1.3×10^{6}
RXs	1.4×10^{8}	2.6×10^{9}	3.8×10^{6}	1.3×10^{8}	7.6×10^{7}	7.3×10^{6}
Core Inlet Pipes(e)	1.5×10^{5}	6.6×10^{5}	7.2×10^{3}	1.6×10^{3}	5.9×10^{3}	2.1×10^{3}
Total plate-out	9.5×10^{10}	1.7×10^{12}	2.6×10^{9}	7.2×10^{10}	4.9×10^{10}	4.8×10^{9}

6.0.14.2.4 Dust in the PBMR

In a pebble-bed HTGR, graphite dust forms resulting from abrasion during fuel balls recirculating, and represents a major transport medium for fission products besides the coolant itself **41**.

A simulation programme with the goal of evaluating the behaviour of dust within the MPS and Fuel Handling and Storage System (FHSS) of the PBMR has been initiated. This simulation will make use of Computational Fluid Dynamics to determine the dust transportation and deposition within the MPS and the dust retention capability of the FHSS.

The following data were obtained from various sources and give an indication of quantities involved:

THTR Primary Circuit **39**

Licensing data:

Aerosols (dust) in the primary circuit are graphitic

Weight loss of fuel element by abrasion: 40 mg per cycle

Aerosol production in the fuel handling system: 54 kg/FPY

Aerosol production in the core: 6 kg/FPY

Aerosol input to the primary circuit: 16 kg/FPY (6 kg/FPY from core)

Uniform deposition of 8 kg/FPY dust on primary circuit surfaces

Aerosol input to gas purification plant: 8 kg/FPY

Mean dust concentration in the coolant

Expected value	1 µg/m^3
Design value	300 µg/m^3

Distribution of aerosols after 300 fpd

Dead water regions (heat exchanger exit, hot gas channel exit, etc)	50%

Surfaces with low He velocity (heat exchanger, cold gas channel) 20%

Gas purification plant 30%

Aerosol concentration of the coolant 1 to 5 μg/m^3

THTR expected design value: 1 to 300 μg/m3

AVR Primary Circuit **[14]**

Dust production rate per operating year 3 kg/y

Total dust inventory at the end of 1988 (˜22 years) 60 kg

Average dust concentration in cooling gas (steady state) 5 μg/m^3

HTR-Modul (Prediction by Siemens) **[4]**

Dust accumulation after 32 years of operation: 500 to 1 000 kg

Average dust production per year: 15.6 kg to 31.2 kg

Estimated abrasion per transit of a fuel element based on AVR experience: 10 mg

PBMR

Calculation of dust release

Number of spheres (331 000 fuel – plus 110 000 moderator graphite spheres) 441 000

10 transits per three- year period

Assume 10 mg abrasion per sphere (HTR Modul prediction)

441 000 (spheres) x 10 mg (abrasion) x 10 (transits, Three year period): 4.41 x 10^7 mg

Total dust production in three years: 44.1 kg

Average dust production per year: 15 kg

From the above information, the expected abrasion per transit of a fuel element

based on AVR experience can be taken as 10 mg per cycle, and the design value based on THTR experience as 40 mg per cycle for the PBMR. The dust distribution in the different regions of the PBMR is being studied and will be presented in a further revision.

Nuclide Distribution in Graphite Dust

In the report **[15]**, the relative specific activities of the dust taken from the housings of the moisture sensors of the THTR in 1988, after 1.16 years of full load power, averaged over the 18 samples, are summarized in ***Table 6.3-11***. The specific activity of Co-60 is normalized to one. In addition, the absolute values for Co-60 are stated. For comparison, the activities from the cold gas (filter cartridge of the gas pre-purification plant) of the AVR after two years of full load power are also listed in ***Table 6.3-11***.

TABLE 6.3-11: TYPICAL NUCLIDE DISTRIBUTION IN GRAPHITE DUST

Nuclide	Half-life	THTR (After 423 d)	AVR [1] (After 2 y)
Cr 51	27.70 d	0.007	0.5
Fe 59	44.6 d	0.018	2.8
Co 60	5.27 a	1 [2]	1 [2]
Hf 181	42.4 d	0.112	n.m.
Zr 95	64.0 d	0.397	n.m.
Nb 95	35.0 d	1.150	n.m.
Ru 103	39.4 d	0.091	n.m.
Ce 141	32.5 d	0.179	n.m.
Ce 144	284 d	1.210	n.m.
Cs 134	2.06 a	0.036	0.1
Cs 137	30.17 a	0.059	0.4
Pa 233	27.0 d	0.570	n.m.
Total:		*4.82*	*4.8*
incl. other nuclides:		*approx. 7* [3]	*13.3* [4]
specific Co-60 activity Bq/g		**$2.2 \cdot 10^7$** [5]	**$6.0 \cdot 10^5$**

n.m.: not measured

Notes:

1) In the cold gas (mechanical filter of the preliminary gas purification).

2) Standardization.

3) Completed by computation, as far as β-/γ-emitter could not be measured.

4) Incl. measured values not contained in the table.

5) For housing of moisture sensors; value for sensors 3.5 times larger.

It can be seen from ***Table 6.3-11***, that the specific Co-60 activity of the THTR-300 is 37 times larger than the value of the AVR. Furthermore, it is noticeable that the activity of the fission products is much lower for the AVR than for the THTR-300. This feature could not be explained sufficiently, since, in general, the nickel content of the stainless steel used in the primary circuit was limited to < 0.02 weight percent.

The high specific activity of fission products in the dust samples from the THTR-300 is caused by broken fuel elements. This is indicated by the presence of the isotope Pa-233 in the dust and by the fact, that the relative THTR activities presented in ***Table 6.3-11*** agree quite well with the results from depletion calculations. The relative specific α-activity of the dust obtained by these calculations is 2.6×10^{-4}. Thus, the β-activity was $7/(2.6 \times 10^{-4}) \approx 3 \times 10^{4}$ times larger than the α-activity.

Based on the expected amount of dust of 1 000 kg (max.) in the primary cycle of the HTR-Modul after 32 FPY, and the activity concentration on the fuel element surface while ignoring radioactive decay, the values for dust-bound surface activities shown in ***Table 6.3-12*** are found. If these values are compared with the surface activity values depicted in ***Table 6.3-8***, it is evident that the contribution of the dust-bound activity can be ignored.

TABLE 6.3-12: DUST-BOUND ACTIVITIES ON PRIMARY CIRCUIT SURFACE AREAS AFTER 32 FPY

Radionuclides	Dust-bound Surface Activity (Bq)
I-131	1.2×10^{9}
Cs-137	9.4×10^{7}
Sr-90	7.5×10^{7}
Ag-110m	1.7×10^{7}
Total	**1.3×10^{9}**

6.0.14.2.5 FHSS

The main contribution to radiation in the FHSS is due to the radiation emitted by the fuel pebbles and the activated de-fuelling pipe below the Reactor Pressure Vessel (RPV) and not by coolant or plated-out radionuclides.

6.0.14.2.6 Neutron and gamma flux near the core and activation of structures

The reactor core is the strongest radiation source in the PBMR. Thermal fission creates a neutron source over the whole core volume, which is the basis for all neutron shield calculations outside the reactor core. The neutron source strength distribution is derived from the power density distribution and the burn-up conditions of the individual fuel elements. Coolant and plate-out activities in the dose rate calculations play a minor role in this case.

Gamma sources result from:

- prompt fission gamma rays;
- gamma radiation caused by fission product decay; and
- capture gamma radiation occurring inside and outside the reactor core as a result of neutron capture.

The source strength of the prompt fission gamma rays is dependent on the instantaneous power level of the reactor while, in contrast, the source strength of the neutron capture gamma radiation and the gamma radiation caused by fission product decay is a function of the integrated power.

6.0.15 RADIOACTIVE MATERIALS IN THE MAIN SUPPORT SYSTEMS

6.0.15.1 Helium Purification Plant

The helium purification plant consists of the following:

- a dust filter which removes metallic fission products and dust-bound activity;
- a molecular sieve which adsorbs tritium in the chemically bound form of HTO, as well as C-14 in the chemically bound form of CO_2, and also retards noble fission gases.

The specified service life of the purification plant between two regeneration cycles is 1 000 h. For regeneration approximately 24 h are required. More details regarding the helium purification plant are given in **32**.

Tritium (HTO) in the regeneration gas which is formed during regeneration of the molecular sieve, is condensed and stored within the plant in a container for condensate, which is generated during operation, until disposal.

During regeneration, C-14 which is bound in the form of CO, as well as tritium which has not been removed, are transferred to the storage container for radioactively contaminated helium, and after intermediate storage, are removed via the exhaust chimney without filtering.

The activity inventories in one section of the helium purification plant shown in ***Table 6.3-13***, were determined by basing the calculation on the design activities in the primary coolant (***Table 6.3-4*** to ***Table 6.3-7***) and the specified purification constant for the HTR-Modul, of 0.05 per hour. Radioactive decay during the service life of the purification plant between two regeneration cycles lasting 1.000 h for noble gases, tritium and C-14, or 32 FPY for aerosols, was also taken into consideration in this conservative calculation. The retention factor in the purification plant for all the nuclides was estimated at 100%.

TABLE 6.3-13: ACTIVITY INVENTORY IN ONE OF THE HELIUM PURIFICATION SECTIONS AFTER 32 FPY AND 1 000 H DELAY TIME

Radionuclide	Design Activity (Bq)	Expected Activity (Bq)
Kr-83m	3.4×10^{9}	1.5×10^{9}
Kr-85m	2.5×10^{10}	9.5×10^{9}
Kr-85	2.1×10^{10}	5.4×10^{9}
Kr-87	8.8×10^{9}	5.0×10^{9}
Kr-88	3.6×10^{10}	1.5×10^{10}
Kr-89	1.5×10^{8}	7.9×10^{8}
Kr-90	1.0×10^{7}	3.2×10^{8}
Xe-131m	2.9×10^{10}	8.7×10^{9}
Xe-133m	5.9×10^{10}	1.7×10^{10}
Xe-133	2.8×10^{12}	8.2×10^{11}
Xe-135m	4.3×10^{8}	5.5×10^{8}
Xe-135	1.3×10^{11}	3.9×10^{10}
Xe-137	2.9×10^{8}	1.3×10^{9}

Radionuclide	Design Activity (Bq)	Expected Activity (Bq)
Xe-138	2.0×10^{9}	2.9×10^{9}
Xe-139	1.6×10^{7}	3.9×10^{8}
Total noble gas	**3.1×10^{12}**	**9.2×10^{11}**
I-131	5.9×10^{8}	7.2×10^{6}
I-132	$1.0x\ 10^{8}$	1.2×10^{6}
I-133	4.4×10^{8}	5.1×10^{6}
I-134	9.5×10^{7}	1.1×10^{6}
I-135	2.4×10^{8}	2.9×10^{6}
Total iodine	**1.5×10^{9}**	**1.7×10^{7}**
Rb-88	3.6×10^{10}	6.0×10^{8}
Rb-89	1.5×10^{8}	-
Rb-90	$1.O \times 10^{7}$	-
Sr-89	1.5×10^{8}	-
Cs-138	2.0×10^{9}	1.5×10^{8}
Cs-139	1.6×10^{9}	-
Ba-139	1.6×10^{9}	-

Distribution of the activity inventory in the molecular sieve and the dust filter of the helium purification system after 32 FPY and 1 000 h delay time is given in ***Table 6.3-14*** and ***Table 6.3-15***.

TABLE 6.3-14: MOLECULAR SIEVE

Radionuclide	Molecular Sieve Activity Inventory (Bq)
Noble gases:	
Kr-83m	1.1×10^{8}
Kr-85m	3.5×10^{8}
Kr-85	1.9×10^{6}
Kr-87	4.3×10^{8}
Kr-88	8.0×10^{8}
Kr-89	9.9×10^{7}
Kr-90	1.0×10^{7}
Xe-131m	3.5×10^{7}
Xe-133m	3.5×10^{8}
Xe-133	7.0×10^{9}
Xe-135m	3.1×10^{8}

Radionuclide	Molecular Sieve Activity Inventory (Bq)
Xe-135	4.3×10^{9}
Xe-137	2.9×10^{8}
Xe-138	1.5×10^{9}
Xe-139	1.6×10^{7}
Total noble gas activity	**1.6×10^{10}**
Noble gas decay products:	
Rb-88	8.0×10^{8}
Rb-89	9.9×10^{7}
Rb-90	1.0×10^{7}
Sr-89	9.9×10^{7}
Cs-138	1.5×10^{9}
Cs-139	1.6×10^{7}
Ba-139	1.6×10^{7}
Total of noble gas decay products	**2.5×10^{9}**
H-3*	3.5×10^{12}
C-14*	3.5×10^{12}

.....Note: *3

TABLE 6.3-15: DUST FILTER

Radionuclide	Dust Filter Activity Inventory (Bq)
I-131	6.3×10^{8}
I-132	1.0×10^{8}
I-133	4.4×10^{8}
I-134	9.5×10^{7}
I-135	2.4×10^{9}
Total noble gases	**1.5×10^{9}**
Rb-88	3.8×10^{8}
Rb-89	7.4×10^{7}
Rb-90	2.1×10^{7}
Sr-89	8.8×10^{6}
Cs-138	2.5×10^{8}
Cs-139	3.8×10^{7}
Ba-139	7.2×10^{6}
Total noble gas decay products	**7.8×10^{8}**

Radionuclide	Dust Filter Activity Inventory (Bq)
Cs-134	4.4×10^{8}
Cs-137	9.2×10^{9}
Ag-110m	1.1×10^{7}
Sr-90	2.8×10^{7}
Total solids	**7.7×10^{9}**

The activity inventory which was deposited on the dust filters should be regarded as a maximum value after 32 FPY. The highest contribution of 5.8×10^{9} Bq, is made by the fission products Sr-90, Cs-134, Cs-137 and Ag-110m. The contributions of iodine and of the noble gas products are five and ten times smaller, respectively.

For the distribution of the noble fission gas activity in the molecular sieve of the HTR-Modul, the following delay times in the molecular sieve were used for the Xe and Kr isotopes.

- Kr: 0.09 h;
- Xe: 0.45 h.

In the calculation of the tritium saturation activities it was assumed that 100% of the tritium which was in the form of HT would be removed by the molecular sieve. A saturation activity of 4.6×10^{12} Bq in the molecular sieve was calculated.

100% of the C-14 is expected to be retained by the molecular sieve. The activity inventories of short-lived noble gas decay products were calculated by using the saturation activities of the mother nuclides in the molecular sieve, and the mother/daughter nuclide transition probability. These activity inventories relate to a plant service life of 32 FPY. The values depicted in ***Table 6.3-7***, should be regarded as maximum values.

The activity inventory in the helium purification plant of the HTR-Modul was based on the design values in the primary coolant cycle, and they therefore represent a high estimate.

A sufficiently large difference compared to the expected activity inventories was selected with the factors:

- noble gases : factor 3.3
- iodine : factor 80
- long-lived solids : factor 3.2
- H-3 : factor 1.6
- C-14 : factor 10
- noble gas products (excluding Sr-90) : factor 3.2

This ensured that no problems associated with excess activity would be expected.

According to the design criteria, the purification section is regenerated every 1 000 h.

6.0.15.1.1 Evacuation system in the helium purification plant

The evacuation system in the helium purification plant is used for the regeneration of the purification plant. After the pressure in the helium cycle has been released into the pressure release containers, the low-pressure/vacuum system generates the vacuum that is required for the regeneration of the molecular sieve.

The evacuation system is switched over to the molecular sieve in the third phase of regeneration, i.e. after CO_2 and H_2O have been expelled, and the H_2O in the water cooler has been removed. The evacuation system removes any water and carbon dioxide which may have remained behind in the molecular sieve, by lowering the pressure of 100 kPa to 0.1 kPa.

The steps in the regeneration of the purification plant components, as well as details relating to the evacuation system, are described in **32**.

6.0.15.2 Evacuation System Used in Fuel Element Handling

The evacuation system is allocated to the fuel element handling section of the fuel element supply plant. It evacuates fresh fuel elements from the charge lock after the latter has been loaded, while preventing access of air into the primary cycle.

The evacuated air is purified in particle filters and transferred to the exhaust chimney.

If the expected dust generation in the fuel element supply plant of 1 000 kg after 32 FPY is used as a point of departure and it is kept in mind that the proportion of fresh

fuel elements equals approximately 7% of the total partially burnt fuel elements which have been circulated, approximately 2 kg of dust per year is generated due to abrasion of the fresh fuel elements.

Because of the proportion of free uranium in the fuel-free layer of the fuel element sphere of $\leq 6 \times 10^{-5}$, the uranium concentration in the dust equals approximately 9.2×10^{-6} g uranium per gram of dust. If the retention factor is 10^{-4}, which can usually be achieved with particle filters, the annual removal of dust-bound activity via the exhaust chimney is less than 10^3 Bq per Module. Compared to the other amounts which are removed, this quantity is so small that it can be ignored.

6.0.15.3 Primary Loop Initial Clean-up System (PLICS)

The PLICS consists of the following two subsystems:

- The PLICS vacuum subsystem; and
- The PLICS heater subsystem.

The PLICS will remove water, air and other gases from the RPV, PCU and CCS and thereby limit the levels of impurities in the primary loop. This operation will limit the levels of impurities in the primary loop before commissioning of the MPS, to that which will be maintained by the HICS in the MPS during normal operation.

The core structures and pebbles will be heated to a temperature between 250 ºC and 300 ºC, using the PLICS heater subsystem to increase the rate at which water is desorbed from it. The system will then be flushed with nitrogen or dry air to remove the desorbed water.

After this operation, the primary loop will be evacuated, using the PLICS vacuum subsystem, to a total pressure of 50 Pa at 300 ºC.

The gases and vapours shall be filtered for dust and scrubbed by the activated carbon filters of the PLICS vacuum subsystem before they are released to atmosphere.

The PLICS system will be contaminated with dust-borne radioactivity when used to evacuate the MPS after maintenance operation on the MPS. The PLICS design and construction shall facilitate the decontamination of contaminated parts. Other parts, which cannot be decontaminated, shall be designed for easy removal and replacement.

6.0.15.3.1 Pressure release system of the secondary helium cycles and the fuel element handling section

All the systems from which helium must be removed for operational purposes are connected to the pressure release system. The most important of these are:

- the helium purification plant;
- the evacuation system of the primary cycle and the helium purification plant;
- the fuel element handling systems;
- the sampling system (gas analysis system); and
- the water removal system of the secondary helium systems.

The pressure release system is designed in such a way that it is capable of removing all the exhaust gases simultaneously. Depending on the chemical or radioactive impurities of the exhaust gases contained, they are either:

- transferred into the storage container for radioactively contaminated helium; or
- fed back into the helium purification plant.

The evacuated air is filtered to stop potential aerosol transfer via the evacuation systems of the primary cycle and the fuel element handling section.

6.0.15.4 Nuclear and Secure Auxiliary Cooling System

The following cooling systems are connected to the auxiliary cooling systems:

- the primary cycle blower;
- the cooler and blower of the helium purification plant;
- the fuel element supply plant and discharge units;
- the waste water system;
- the surface cooler;
- the support of the pressurized container unit; and

- the connections at the bottom of the pressurized reactor container.

Deionized water is used in both systems as coolant. To prevent corrosion, the oxygen content of this system is limited.

Heat transferred from the RPV wall by radiation, conduction and convection is absorbed in water chambers surrounding the reactor in the reactor cavity, called the Reactor Cavity Cooling System (RCCS). Closed cooling water systems are connected to the chambers. Activation of the water in these cooling chambers by neutron capture has been considered. The possibility of a leakage of radioactivity from this closed cooling system has therefore also been considered.

The activity in these systems originates from activation of the structural materials which is followed by corrosion, by activation of water and the impurities it contains, as well as leaks from the systems that contain the primary coolant.

The auxiliary cooling systems are operated as closed units. No provision has been made for demineralizing these cycles. Consequently, activity is only released from these systems if leaks occur and containers are emptied. A maximum activity inventory of 3×10^8 Bq is expected in these systems. Sampling can also be used to monitor activity in these cycles.

The activity in the cycles of the auxiliary coolant systems is determined by activation of the structural materials in the region of the surface cooler followed by corrosion, and by activation of the coolant and its impurities.

Based on previous experience that was gained with operational reactors, the selection of suitable materials and corresponding conditioning of the coolant can be used to restrict corrosion that occurs in the cycles, and the formation of activation products to specific levels.

The expected activity concentration in the vicinity of the surface coolers caused by activation was estimated for reactor operation over 32 FPY. The most important activation products which play a role are Cr-51, Mn-54, Fe-59, Co-60 and Ta-182.

Since the amount of oxygen in the coolant is controlled to reduce corrosion, the corrosion rate of 10^{-3} mm p.a. can be regarded as a technically achievable value.

Based on the activity concentration in the surface cooler and the above mentioned corrosion rate, the activity inventory depicted in ***Table 6.3-16*** is found in the auxiliary

cycles at the end of 32 FPY of reactor operation. Co-60 makes the most important contribution with 4.0 x 10^8 Bq to these values. This was calculated using conservative assumptions. The contribution of Ta-182 is approximately 10 times lower, and that of the other nuclides is at least 100 times lower.

TABLE 6.3-16: ACTIVITY INVENTORY IN THE NUCLEAR AND SECURE INTERMEDIATE COOLING SYSTEMS CAUSED BY CORROSIVE ABRASION IN THE SURFACE COOLER AFTER 32 FPY OPERATION

Radionuclide	Activity of the Intermediate Cooling Systems (Bq)
Cr-51	9.4×10^5
Mn-54	4.3×10^6
Co-60	4.0×10^8
Fe-69	4.0×10^6
Ta-182	2.7×10^7
Total	**4.4×10^8**

Additional activity transfers into the secure auxiliary cooling system because of primary coolant leaks do not occur, because the system has no direct connection with the primary cycle. The activity of the individual sections is monitored by regular sampling.

In the nuclear intermediate cooling system, the possibility exists that activity from the primary cycle can be transferred due to leaks. How much activity such leaks will transfer depends on the impermeability of the systems, and it can, therefore, not be estimated in advance. Regular sampling ensures that possible leaks are detected rapidly so that relevant measures can be taken to reduce activity transfers to a minimum.

Because both systems are operated as closed cycles, activity releases from both systems are only possible if leaks should occur, and during container emptying. Adequate provision has been made in the design of the waste water pipe for the radioactive materials which are released during leaks and container emptying processes.

6.0.15.5 Nuclear and Secure Auxiliary Cold Water Systems

The nuclear and secure intermediate cooling systems transfer the heat taken up by

the intermediate coolers to the corresponding auxiliary coolant systems.

Re-cooling of nuclear auxiliary coolant water is carried out in a hybrid cooling tower.

A number of wet cooling cells are available for re-cooling of secure auxiliary cooling water.

The auxiliary cold water systems are usually free of activity. Due to of pressure gradients, leaks and activity transfers into the secondary cold water systems can occur if the auxiliary coolers are not leak-proof.

To check the leak-proofing of the auxiliary coolers, both cycles are continuously monitored with a gamma-sensitive detector, as well as regular sampling. In the secure auxiliary cold water system the samples are taken from the collection pipe, and in the nuclear auxiliary cold water system they are taken from the pipe which leads to the hybrid cooling tower.

6.0.16 RADIOACTIVE MATERIALS IN THE AIR OF THE REACTOR CAVITY

The following are the most important sources of radioactive materials in the air inside the reactor cavity:

- activation of air in the vicinity of the pressurized reactor vessel as a source of Ar-41;
- activation of metal surface on the outside of the pressurized reactor vessel and the surface cooler in the form of aerosols; and
- primary coolant leaks in the form of aerosols and noble gases.

The reactor cavity is screened from neighbouring chambers. The design value for leaks arising in neighbouring cells which penetrate the reactor cell, is 2 000 m^3/d. To maintain a low pressure of 0.1 kPa to 0.15 kPa, approximately 2 000 m^3 (free volume of air in the cell) is removed every day and transferred to the exhaust chimney without filtering.

6.0.16.1 Activation of Air

Components which occur in air, such as nitrogen, carbon, oxygen and argon, are activated by the neutron flux which occurs in the cell.

The most important reactions are:

- Ar-40 (n, γ) Ar-41;
- O-16 (n, p) N-16;
- O-18 (n, y) O-19; and
- N-14 (n, p) C-14.

Ar-41 makes the largest contribution to air activity. Based on two-dimensional neutron flux fields, the macroscopic reaction rates surrounding the air space between the pressurized reactor container and the cell wall were calculated, by using a grid point method and integrating the macroscopic reaction rates over the volume of the cell. An Ar-41 production rate of 1.1×10^{6} Bq/s is calculated. For the design case, a production rate which is twice as high, is used. If the air exchange factor of 1.16×10^{-5} s^{-1}, is taken into consideration, then an activity equilibrium of 1.9×10^{10} Bq adjusts itself in the reactor cell.

Direct radiation by N-16 which is formed by the O-16 (n, p) N-16 reaction is negligible, as the process takes place at neutron energy levels which exceed 9 MeV, where the formation rate of N-16, and consequently also the activity, can be ignored.

6.0.16.2 Aerosol Concentration in the Air Supply

Two sources are mainly responsible for aerosol contamination of the air supply:

- Formation of activation products on the metal surfaces of the reactor pressure vessel and the surface cooler caused by the neutron flux arising from the reactor core, and by abrasion of the surfaces. The most important activation products are Cr-51, Mn-54, Mn-56, Fe-59, Co-58, Co-60, and Ta-182.
- Aerosol releases which accompany helium leaks. The aerosol spectrum which is released in this case corresponds to that of the primary cycle.

The contribution to the aerosol concentration in the primary cell caused by helium leaks was estimated. It was assumed that the specified helium leak rate of 0.1% per day exclusively occurs in the primary cavity. If only the long-lived aerosols with $T_{1/2}$ $\approx$8 days are considered, then an equilibrium concentration of 0,45 Bq/m^3 adjusts itself in the cavity. The nuclide-specific proportions of this concentration are shown in

Table 6.3-17. In comparison to the activation products, this contribution can be ignored.

TABLE 6.3-17: AEROSOL-BOUND ACTIVITY CONCENTRATION IN THE REACTOR CAVITY CAUSED BY HELIUM LEAKAGE

Radionuclide	Activity Concentration in the Reactor Cavity due to Helium Leakage (Bq)
Cs-134	0.17
Cs-137	0.36
Ag-110m	0.012
Sr-90	1.5×10^3
Total solids	**0.55**
I-131	22.8
I-132	36.9
I-133	80.4
I-134	36.
I-135	73.7
Total iodine	**249.9**

Aerosol contributions which can possibly be made by activation products were estimated. The values for reactor full load operation for 32 years were used as the point of departure to calculate the activation concentrations of the activation products in the pressurized reactor container and the surface cooler. When the activation products of Co-60, Fe-59 and Ta-182 are taken into consideration, the aerosol-bound activity concentration of the activation products in the cell equal between $\approx$ 2 Bq/m^3 and 70 Bq/m^3, and when all the activation products with $T_{1/2} \approx$ 8 days are taken into consideration, values between $\approx$ 2,8 Bq/m^3 and 90 Bq/m^3 are found. A design value of 30 Bq/m^3 for the aerosol-bound activation concentration was used The design value can be achieved by filtering the expelled air before removal. The nuclide-specific composition of the aerosol-bound dust activity including all the activation products with $T_{1/2} \approx$ 8 days, is shown in ***Table 6.3-18***. Co-60 was selected as the representative nuclide for the calculation of activation products. The selection of Co-60 covers all the other radionuclides.

TABLE 6.3-18: DESIGN VALUES FOR THE ACTIVITY CONCENTRATION OF THE ACTIVATION PRODUCT AND AR-41 IN THE REACTOR CAVITY

Radionuclide	Activity Concentration in the Reactor Cavity (Bq)
Cr-51	2.4
Mn-54	0.9
Fe-59	6.0
Co-58	0.8
Co-60	15.1
Ta-182	15.1
Total activation products	**40.4**
Ar-41	1.3 x10^7

6.0.16.3 Noble Fission Gas Concentration in the Air Supply

If it is assumed that all the primary coolant leaks occur in the reactor cavity, then an equilibration concentration of 1.5 x 10^5 Bq/m^3 corresponding to the nuclide-specific classification in ***Table 6.3-19***, adjusts itself. The contribution of the noble gas products, particularly Sr-90 which is formed by decay of Kr-90, can be ignored. The depicted values should be regarded as maximum upper limit values, because in reality most of the helium leaks will probably occur outside the reactor cavity.

TABLE 6.3-19: MAXIMUM NOBLE GAS ACTIVITY CONCENTRATION IN THE REACTOR CAVITY CAUSED BY HELIUM LEAKAGE

Radionuclides	Activity Concentration of Noble Gases in the Reactor Cavity (Bq/m^3)
Kr-83m	1.3 x 10^2
Kr-85m	8.0 x 10^3
Kr-85	2.1 x 10^2
Kr-87	3.5 x 10^3
Kr-88	1.3 x 10^4
Kr-89	6.0 x 10^1
Kr-90	4.4 x 10^0
Xe-131m	7.4 x 10^3
Xe-133m	5.6 x 10^3
Xe-133	1.3 x 10^5
Xe-135m	1.7 x 10^2
Xe-135	3.4 x 10^4
Xe-137	1.2 x 10^2
Xe-138	8.7 x 10^2

Xe-139	6.7×10^{0}
Total noble gases	**2.0×10^{5}**

6.0.17 VENTILATION SYSTEMS IN THE RADIOLOGICAL CONTROLLED AREA

The ventilation system circulates and filters the air in the controlled zones and exhaust air is released through the Heating, Ventilation and Air-conditioning (HVAC) System. The present design foresees filtering in the HVAC System.

6.0.18 RELEASES OF RADIOACTIVE MATERIALS (WASTE)

6.0.18.1 Liquid Waste

6.0.18.1.1.1 Design objectives

The liquid waste systems are designed to collect potentially radioactive waste liquids for monitoring, treatment and/or discharge via the main cooling water outflow. The routing ensures no mixture of high and low active waste prior to any required treatment. The piping is such as to ensure minimal leakage from the circuit, and thus reduce the spread of contamination. To ensure that authorized discharge quantities are not exceeded, liquid waste is sampled and analysed prior to release, and if required, treated.

6.0.18.1.2 Estimated releases

Liquid waste generated in the controlled area of the PBMR plant is collected and processed. This liquid waste includes washroom and shower water, liquid waste from laboratories, water collected after decontamination operations, and sump water from the building drain system. ***Table 6.3-27*** lists the expected quantities and corresponding activity concentrations. The activity values in the table were obtained by adjusting the HTR-Modul activities by multiplying the HTR-Modul values by the power ratio of 1.34 **42**.

TABLE 6.3-20: ORIGIN OF LIQUID WASTE

Origin	Volume Produced	Activity (Bq/m^3)
Decontamination and laboratory water	1 to 3 m^3/d 480 m^3 p.a.	Up to 6×10^{7}

Origin	Volume Produced	Activity (Bq/m^3)
Sump and leak-off water	0.4 to 4 m^3/d 365 m^3 p.a.	Up to 6 x 10^6
Laundry water	1 to 4 m^3/d 500 m^3 p.a.	Up to 6 x 10^7
Shower and washroom water	0.3 to 0.6 m^3/d 100 m^3 p.a.	Up to 6 x 10^6

The volume produced per annum is based on the assumption that the standard quantity is produced in 300 days, and the maximum quantity in 60 days, i.e. for decontamination water: 1 m^3/d for 300 days and 3 m^3/d for 60 days, which equates to 480 m^3 p.a.

The nuclide mixture in ***Table 6.3-21*** was obtained from calculations of radioactive releases estimated for the German HTR-Modul, and considers possible fluctuations in the nuclide composition in a conservative manner. The activity values in the table were obtained by adjusting the HTR-Modul activities by multiplying the latter by the power ratio.

TABLE 6.3-21: RADIOACTIVE RELEASES IN LIQUID EFFLUENTS AND ACTIVITY CONCENTRATIONS AT THE POINT OF RELEASE 29

Nuclide	Fraction of Nuclide Mixture (%)	Release Based on Nuclide Mixture (Bq p.a.)	Activity Concentrations[1)] (Bq/m^3)
Co-60	24.0	2.3 x 10^9	42.9
Sr-90	0.5	4.9 x 10^7	0.92
I-131	5.0	4.9 x 10^8	9.14
Cs-134	15.0	1.4 x 10^9	26
Cs-137	55.0	5.2 x 10^9	97
Ag-110m	0.5	4.9 x 10^7	0.91
Total mixture	100	9.5 x 10^9	177
H-3	100	4.3 x 10^{13} **37**	802 000

Note:

1) Activity concentrations at the point of use for mixing with 1.7 m^3/s of average run-off of the discharge receiving water.

6.0.18.1.3 Release points

The liquid releases will be diverted to the sea water discharge of the Koeberg plant. The design will ensure that all releases to the environment are controlled and monitored. The impact on Koeberg releases is minimal, i.e. they will not adversely affect the ability of the Koeberg site to comply with the Annual Authorized Discharge Quantities (AADQ).

6.0.18.1.4 Dilution factors

The dilution factors calculated are based on the PBMR open circuit flow rate of 6 120 m^3/h cooling water.

Table 6.3-22 details the effect of the estimated liquid releases using the AADQ and Dose Conversion Factors calculated for the Koeberg site.

TABLE 6.3-22: EFFECT OF THE ESTIMATED LIQUID RELEASE ON THE KOEBERG AADQ

Nuclide	Release Based on Nuclide Mixture (Bq p.a.)	Annual Dose Estimate to the Public (μSv)
Co-60	2.3×10^{9}	1.3×10^{-2}
Sr-90	4.9×10^{7}	8.1×10^{-5}
I-131	4.9×10^{8}	4.3×10^{-4}
Cs-134	1.4×10^{9}	1.1×10^{-3}
Cs-137	5.2×10^{9}	2.3×10^{-2}
Ag-110m	4.9×10^{7}	3.7×10^{-2}
H-3	4.3×10^{13}	3.0×10^{-7}
Total Dose		3.7×10^{-2}

6.0.18.2 Gaseous Waste

The radioactive emissions via the exhaust chimney consist of the following:

- Noble gas, iodine, C-14, H-3 and aerosol emissions caused by leaks in the primary cycle and the systems that contain primary coolant. To calculate the annual emission, a primary coolant leak rate of 0.1% per day and per Module, as well as a mean air exchange factor of 1 h^{-1}, were used.

- Iodine, C 14 and H3 emissions from the storage containers for radioactively contaminated helium. According to the design criteria, 15 regenerations per year are used.

- Emission of activation products together with the air supply of the primary cells. The air exchange factor á = 1 d^{-1}.

For all three routes, it is assumed that the emissions occur continuously and without filtering via the exhaust chimney.

6.0.18.2.1 Emission caused by primary coolant leaks

The leak rate exhibited by the Peach Bottom was 1% and by the AVR and Dragon reactors was 0.2%). To achieve lower leak rates, very high demands will be made on the impermeability of components and systems. Special attention will have to be given to this aspect during construction planning of the components and systems.

By including reserves in the design of other components, it will also be possible to restrict the radioactive emissions to the design values, even if an unexpectedly high leak rate occurs.

To calculate the emission rates, it is assumed that the leaks occur inside the reactor building, and that the radioactive materials which are released will be removed at a rate corresponding to an air exchange of 1 h^{-1} (as is usual in such buildings). Leaks that could occur in the primary cells are not taken into consideration. In view of the emissions, this approach can be regarded as conservative.

The annual emissions via the exhaust chimney caused by primary coolant leaks are shown in ***Table 6.3-23***. It was assumed that 100% of the radioactive iodine was elemental.

TABLE 6.3-23: ANNUAL EMISSION VIA THE EXHAUST CHIMNEY FOR THE PBMR CAUSED BY PRIMARY COOLANT LEAKAGE

Radionuclide	Activity (Bq)	
	Design Value	Expected Value
Kr-83m	5.8×10^{9}	1.7×10^{9}
Kr-85m	2.1×10^{10}	6.7×10^{9}
Kr-85	1.3×10^{8}	3.4×10^{7}
Kr-87	1.9×10^{10}	6.0×10^{9}

Radionuclide	Activity (Bq)	
	Design Value	**Expected Value**
Kr-88	4.7×10^{10}	1.4×10^{10}
Kr-89	8.0×10^{8}	-
Kr-90	6.4×10^{7}	2.0×10^{7}
Xe-131m	5.0×10^{8}	1.6×10^{8}
Xe-133m	4.6×10^{9}	1.3×10^{9}
Xe-133	9.4×10^{10}	2.8×10^{10}
Xe-135m	1.9×10^{10}	5.5×10^{8}
Xe-135	5.4×10^{10}	1.6×10^{10}
Xe-137	1.7×10^{9}	5.0×10^{8}
Xe-138	9.4×10^{9}	2.8×10^{9}
Xe-139	1.0×10^{8}	-
Total noble gases	**2.6×10^{11}**	**7.9×10^{10}**
I-131	1.4×10^{7}	1.6×10^{5}
I-132	1.5×10^{8}	1.7×10^{6}
I-133	8.7×10^{7}	1.0×10^{6}
I-134	2.5×10^{8}	3.1×10^{6}
I-135	1.5×10^{8}	1.7×10^{6}
Total iodine	**6.5×10^{8}**	**7.4×10^{6}**
Cs-134	1.1×10^{5}	2.4×10^{3}
Cs-137	2.3×10^{5}	5.1×10^{3}
Ag-110m	8.0×10^{3}	1.8×10^{2}
Sr-90	8.7×10^{2}	1.5×10^{1}
Total solids	**3.4×10^{5}**	**7.4×10^{3}**
Rb-88	2.8×10^{10}	4.6×10^{8}
Rb-89	3.4×10^{8}	-
Rb-90	1.3×10^{8}	-
Sr-89	3.2×10^{4}	-
Cs-138	9.4×10^{8}	-
Cs-139	1.9×10^{8}	-
Ba-139	1.5×10^{7}	-
Total noble gas decay products	**3.0×10^{10}**	**4.6×10^{8}**
H-3	3.5×10^{12} **37**	1.1×10^{11}
C-14	3.5×10^{12} **36**	3.2×10^{11}

6.0.18.2.2 Activity emissions in air from the primary cavities

The activity emissions which are removed via the exhaust chimney are shown in ***Table 6.3-24***. They are based on the activity inventory in the primary cavity shown in ***Table 6.3-18***, and an air exchange factor of á = 1 d^{-1}.

TABLE 6.3-24: ANNUAL EMISSION OF RADIOACTIVE MATERIAL TOGETHER WITH EXPELLED AIR FROM THE REACTOR CAVITY

Radionuclide	Activity (Bq)
Cr-51	1.5×10^{6}
Mn-54	5.9×10^{5}
Fe-59	3.8×10^{6}
Co-58	5.0×10^{5}
Co-60	9.4×10^{6}
Ta-182	1.1×10^{7}
Total activation products	**2.4×10^{7}**
Ar-41	8.0×10^{12}

6.0.18.2.3 Expected release rates of gaseous effluents to the environment

The low activity inventory in the primary coolant results in the annual release due to primary coolant leaks being small. Iodine and aerosol-bound fission products are exclusively emitted into the environment via this route. The design value for a total annual iodine release is 6.5×10^{8} Bq, and for long-lived fission products, it is 3.4×10^{5} Bq in ***Table 6.3-23***. The annual activity emission rate in the form of aerosol-bound fission products which are formed by decay of short-lived noble gases, is 3.0×10^{10} Bq. Since the half-lives of these radionuclides, with the exception of Sr-89, are shorter than eight days, they can be added to the noble gases, so that only Sr-89 with 3.2×10^{4} Bq/a must be taken into consideration, together with the long-lived aerosols.

Most of the activity in noble fission gases, H-3 and C-14 is emitted during regeneration of the helium purification plant from the storage containers for radioactively contaminated helium. Annual releases of 2.6×10^{11} Bq for the noble fission gases, 3.5×10^{12} Bq for tritium, and 3.5×10^{12} Bq for C-14, must be reckoned with in the design scenario.

Expelled air from the primary cell is responsible for the emission of Ar-41 and most of the aerosol activity. Annual releases of 8×10^{12} Bq for Ar-41, and 2.4×10^{7} Bq for aerosols, must be reckoned with in the design scenario. Co-60 was selected as the representative nuclide for aerosol emissions.

In summary, it is important to consider the unfiltered emissions via the exhaust chimney given in ***Table 6.3-25*** (which is a combination of ***Table 6.3-22*** and ***Table 6.3-23***). Filtered emissions will decrease the released activities.

TABLE 6.3-25: GASEOUS RADIOACTIVE MATERIALS RELEASED ANNUALLY

Radionuclide	Activity Release (Bq)
Sum of noble fission gases[1]	2.6×10^{11}
Ar-41[4)]	8.0×10^{12}
I-131[4)]	1.4×10^{7}
Total of all iodines (I-131 included)[4)]	6.5×10^{8}
Co-60 (Aerosol)	2.4×10^{7}
Ag-110m (Aerosol)	8.0×10^{3}
Cs-134 (Aerosol)	1.1×10^{5}
Cs-137 (Aerosol)	2.3×10^{5}
Sr-90 (Aerosol)	8.7×10^{2}
Sum of long lived aerosols[5)] (half-life >10 d) :Co-60, Ag-110m, Cs-134, Cs-137, Sr-90	2.4×10^{7}
C-14[2)]	3.5×10^{12}
Tritium[3)]	3.5×10^{12}

Notes:

1. Sum of released noble gas activity calculated by multiplying the coolant activity in ***Table 6.3-4*** by 0.1%/d x365d

2. PBMR calculated value in **36**.

3. PBMR calculated value in **37**.

4. All other PBMR source terms calculated by multiplying the HTR Module source terms as in ***Table 6.3-6***.by the power ratio of 268 MW/200 MW x 0.5

Aerosol values obtained from **29** and adjusted as in the previous note (4).

- Release points

  All gaseous releases are routed via the reactor building ventilation system and released at a height of 23 m above ground level.

- Dilution factor

  Dilution of gaseous effluents is into the main ventilation flow of the reactor building.

- Effect on the Koeberg gaseous AADQ

  Table 6.3-26 details the effect of the estimated gaseous releases using the Koeberg AADQ and Dose Conversion Factors calculated for the Koeberg site.

TABLE 6.3-26: EFFECT OF THE ESTIMATED RELEASES

Nuclide	Release (Bq p.a.)	Annual Dose Estimate to the Public (μSv)
Noble gases	2.6×10^{11}	2.1×10^{-3}
Ar-41	8.6×10^{12}	4.3×10^{-2}
I-131	1.4×10^{7}	8.7×10^{-4}
Long-lived aerosols Half-life > 10 days	2.4×10^{7}	3.4×10^{-2}
H-3	5.4×10^{12}	8.2×10^{-2}
C-14	3.2×10^{11}	2.6×10^{-1}
Total Dose		4.2×10^{-1}

6.0.18.3 Solid Waste

During standard operation, 360 fuel elements are transferred every day. The burnt-up fuel elements are transferred via the filling station, and filled in loose piles into transport and storage containers. The filling level of the containers is controlled with the aid of counters. Once a container has been filled and conditioned, the lid is closed. Prior to removal of the containers, each container is checked for leaks and surface contamination. The containers are then stored in an intermediate fuel element store on the power plant site.

The space between the double lids of the containers is monitored for leaks with the aid of pipes which are connected to the outside world.

Other solid, radioactive materials, such as used-up filter materials and contaminated, disassembled components, are packed into iron-hooped drums and conditioned *in situ* or transported for external conditioning. Residues are fed back into the plant. Ninety iron-hooped drums are used every year. These drums are stored in the drum

store in the auxiliary reactor building until they are transported to an external store. The store has a capacity of 200 drums.

As provision has been made for the storage of solid, radioactive materials, no problems are experienced with environmental pollution.

6.0.18.3.1 Expected production volumes

Expected volumes are scaled up from HTR-Modul SAR. The total, excluding the spent fuel, is 52.5 m^3 per annum. During design, the target is to minimize waste to a volume of 10 m^3 per annum. The basis of this is that with a minimal need for liquid waste treatment, the volumes given are conservative. The anticipated distribution of solid waste for the PBMR is as shown in ***Table 6.3-27***.

TABLE 6.3-27: ANTICIPATED WASTE

Description	Number of Drums p.a. [2]
Solid operational waste	25 to 100
Filters	8 [1)]
Unserviceable activated and contaminated Systems, Structures and Components (SSC)	3 [1)]

Notes:

1. This preliminary information derived from the High Temperature Modular Reactor (HTR-Modul) Nuclear Power Plant (NPP).
2. Based on 210 litre drum.

6.0.18.3.2 Packaging

In general, 210 litre metal drums will be used, qualified to IP-2 and able to carry SCO-2 or LSA-2 material. Where waste cannot be drummed in 210 litre drums due to activity/dose rate or physical size, suitable containers will be purchased. The use of concrete drums is not envisaged.

6.0.18.3.3 Storage facilities

Spent fuel will be stored in spent fuel tanks as described in the FHSS. All other waste will be stored in the waste storage facility.

6.0.18.3.4 Shipment

Any shipments that are required will comply with the IAEA safety Series 6 and the NECSA acceptance criteria, once permission is obtained from the National Nuclear Regulator (NNR) to ship such waste.

6.0.19 PROCESS AND EFFLUENT RADIOLOGICAL MONITORING SYSTEMS

6.0.19.1 Design Objectives

The basis for selection of location of Continuous Radiation Monitors (CRM) is as follows **43**.

- on all identifiable radioactive release pathways to the environment;
- in major areas where dose rates and/or airborne radioactivity can vary significantly, and worker occupancy is possible;
- on all processes identified to be needing automatic action in the event of increasing radioactivity level; and
- on processes requiring radiation monitoring for operational reasons, and to inform operators of deteriorating conditions requiring action.

Each of these CRM will have pre-determined alarm and action thresholds. In addition, a calibration and surveillance programme will be developed to ensure the correct functioning of these instruments.

6.0.19.2 Radio Chemical Sampling

Sampling will be required for gaseous or liquid releases to the environment. Further, there will be a need to sample inputs to the gaseous and liquid waste systems such as:

- chemical laboratory;
- decontamination areas;
- laundry;
- water cooling circuits;
- HICS; and

- FHSS.

Where possible, sample locations will be designed to minimize exposure of personnel.

Detailed descriptions of the type of sampling and frequency for each location will be included once plant details become fully available.

6.0.20 OCCUPATIONAL EXPOSURE AND DOSE ASSESSMENT

6.0.20.1 Introduction

One of the aims of the design of the PBMR is to minimize the radiation exposure of personnel involved in work activities that have to be performed in radiation-controlled areas. Design aspects and an effective radiological protection programme will provide assurance that the dose to workers is in line with the principles of As Low as Reasonably Achievable (ALARA). This section describes aspects of the design to minimize dose, and the methodology that will be followed to arrive at an estimate of the radiation exposure to plant personnel.

6.0.20.2 Exposure Categories

Six typical exposure categories will be used to estimate the effective dose to plant personnel. These categories are as follows:

6.0.20.2.1 Routine operations

Routine operations consist of the following:

- routine patrols for general surveillance in normally accessible plant areas;
- periodic testing and monitoring of specific equipment or systems during normal plant or systems operations;
- control operations where personnel occupancy is continuous or where operator actions are required for routine system operation; e.g. control panel activities; and
- administration and office personnel in administration buildings, warehouses and engineering offices.

6.0.20.2.2 Preventative maintenance

Preventative maintenance includes all scheduled and routine maintenance that is required to be performed in radiation areas during normal plant power operation, shutdown or refuelling.

6.0.20.2.3 In-service inspection (ISI)

In-service inspection and testing and non-destructive examination are normally performed by plant personnel during plant outages or scheduled system shutdowns. The inspection and testing procedures are normally preplanned, and requirements are established to minimize personnel radiation exposure.

6.0.20.2.4 Refuelling

Refuelling operations include all activities that are necessary to prepare and support the actual replacement, transfer and storage of spent and new fuel. Refuelling is performed continuously during operation, and can result in significant radiation exposure to plant personnel because of the total time involved during an operating cycle, and the highly radioactive sources that are involved.

6.0.20.2.5 Waste processing

Waste processing includes all activities associated with the collection, processing, storage, sampling and disposal of radioactivity from process and leakage collection systems.

6.0.20.2.6 Corrective maintenance

Corrective maintenance and repairs are neither scheduled nor routine. Uncertainty in the frequency of these activities makes it difficult to assess the contribution of these activities to the total annual exposure estimate.

Maintenance and ISI operations are expected to be the dominant contributors to worker dose, since the most time could be spend in higher radiation areas.

6.0.20.3 Design Aspects to Minimize the Dose to Maintenance Personnel

Examples of important design aspects relating to the reduction of radiation are as follows:

6.0.20.3.1 Building design

The design of the building ensures accessibility of highly radioactive plant items, while simultaneously keeping radiation exposure low along access and exit routes. Furthermore, it ensures that the dose rate at the individual plant items is influenced to the smallest possible extent by adjacent plant items. The following examples illustrate these design principles. Next to the controlled access passage linking the module with the Services Building on level 0, is the feed and buffer line for fuel to the spent fuel tanks. The lines are highly active, and are therefore shielded to the passage by means of a 1 000 mm concrete wall to protect workers in the passage. The same thickness wall is provided on the other side of the piping to allow maintenance on the fuel distribution valve block and sampling of fuel for inspection and analysis.

The principle of providing sufficient space for the removal of complete components, is followed. Examples are laydown areas where main components that are radioactive can be parked, and if necessary, shielded for a limited time until they are reinstalled. An example is the removal of the HPT compressor that might be activated and contaminated, and can be placed and stored in the laydown area which is on floor level 9.5 directly above the PCU area. The item is then put in a shielded cask in preparation for transport and further inspection and maintenance.

6.0.20.3.2 Component design

Components are selected that, as far as is possible, are proof-tested and service-proven. This will ensure that the frequency of maintenance activities and the necessary effort and time involved are kept low. The aim of design is to minimize the scope of ISI that potentially have high associated exposure rates. An example of minimizing exposure times by design is the insulated parts of the pressure vessel that will be subject to in-service ultrasonic examinations. These are provided with lagging that can be removed in a short time.

6.0.20.3.3 Component arrangement

The component arrangement is such that the dose rates to multiple components inside any given compartment, is kept at a low level while simultaneously granting ease of access with minimum obstruction at the workplace. In the Helium Inventory Control System (HICS) purification system, for example, the filters are located in such a manner that one can remove them without influencing the maintenance

schedule of the compressor and valves in the same area. This is achieved by means of a shielding wall.

Permanently installed or mobile platforms will be provided for maintenance activities where the frequency and scope of the activity, as well as level of radiation exposure, necessitate such facilities in order to reach ALARA objectives. Rails, for example, are provided for ISI of the pressure vessel unit.

Free spaces are provided that will allow for encapsulating protective gear around radioactive components, where required. Ease of movement when respiratory protective equipment has to be worn by maintenance personnel, is also considered in the design of free space. A typical example is maintenance of the Start-up Blower System (SBS) valves. These are situated directly next to the recuperator vessel. The Recuperator will be shielded temporarily where these valves are to be maintained.

6.0.20.3.4 Auxiliary facilities

Provision is made for a space to erect a temporary decontamination compartment in the laydown area. Plant items can be decontaminated, should this be required, prior to maintenance.

The introduction and planning of auxiliary facilities will be weighted against the potential reduction of personnel dose, so that the work can be done cost-effectively in the shortest period of time, by the smallest team and/or at the lowest dose rate. The need for auxiliary systems will be analysed as part of the ALARA programme. Elements of this programme will, for example, include work pre-planning, as well as debriefing following maintenance, to optimize procedures, ALARA-oriented tools and familiarization mock-ups.

6.0.20.4 Dose Assessment Methodology

Two methods can be used to estimate exposures, depending on the work category.

- Preventative/corrective maintenance and ISI doses can be estimated using area-by-area and task-by-task methods. In this method, the maintenance and ISI tasks are assigned to the various plant areas, and occupancy times are developed based on task manpower, duration, and frequency. A general area dose rate for a particular plant area is then used to calculate estimated individual and collective

annual doses.

- Estimates are determined for the average times that workers typically spend in different radiation zones.

The second method is used for routine operations such as refuelling.

A structured and iterative process will be followed to radiologically classify areas as radiation zones, assign zone categories for each area in the PBMR Module, and to determine radiation doses associated with maintenance in these areas. The basic process is as follows:

Each area or room in the building is allocated a unique number, e.g. Z-23/A, Z-23/B, etc. up to Z+18/A, Z+18/B.

The plant equipment that is situated in each sub-area is identified.

The duration and frequency of maintenance/inspection/adjustment tasks, etc. on each of these items of equipment are determined. Initially this information is based on the data gathered during the Logistic Support Analysis (LSA) process as applied to each subsystem. The LSA is ongoing, and durations and frequencies were estimated, also using Failure Mode and Effects Analysis (FMEA) results available at a specific point in the design. The possibility of contamination/radiation when sealed pieces of equipment are opened, is taken into consideration.

The distribution of radioactivity in the Module systems is calculated and assigned to areas and equipment.

The areas/volumes/rooms are then classified in accordance with the definitions in the Radiological Protection Programme.

On completion of this process, the results will be tabulated that will include the consideration of the following factors:

- plant area number;
- area description;
- brief description of equipment in the area;
- maintenance tasks;

- frequency of task;
- mean time to maintain/repair;
- plant mode;
- area classification;
- dose rate on component;
- extremity dose;
- ambient area external dose rate;
- potential for contamination (surface and airborne);
- maximum annual dose per maintenance task; and
- collective dose.

These results will then be used to complete the prospective hazard assessment for workers before commissioning and operation commences.

Dose rate information estimated on the main components and German experience indicates that the radiation doses will be well within the regulatory requirements **42**.

The duration period indicates the number of days available for an activity. This number of days will not necessarily be the duration of the task.

6.0.21 SHIELDING

6.0.21.1 Design Objectives

The design of the PBMR power plant is based on the following principles intended to keep radiation exposure of the operating personnel as low as reasonably achievable:

- There is a clear division between different radiation areas.
- Plant equipment and shielding facilities are designed and installed in such a way as to maintain occupational radiation exposure of personnel as low as reasonably achievable and also below statutory limits.

The following measures are taken for this purpose:

Various barriers, such as the pyrocarbon and silicon carbide coatings of fuel (so-called TRISO coating), the graphite structure of the fuel element and the primary gas envelope prevent uncontrolled releases of radioactive materials to plant areas.

Fission product release from the fuel is very low because of:

- the smaller number of fuel particles in silicon carbide layers having manufacturing defects;
- the low irradiation-induced fraction of particle failure in normal operation;
- negligible fission product diffusion through intact silicon carbide layers; and
- retention of solids in the graphite matrix.

The resulting primary coolant activity is very low.

The following facilities are also employed to limit radioactivity:

- Systems for extraction of radioactive materials from the primary coolant and for storage of these materials.
- Where possible, design of plant equipment is such to avoid accumulation of solids. Where this is not possible, facilities for removal of such will be available.

Wherever reasonable, further radiation protection measures are taken during plant design:

- Shielding is, where possible, designed such that movement is not required.
- Shielding inside the controlled area is designed such that the dose rates in compartments containing radiation sources are not significantly affected by radiation from adjacent compartments.
- Shielding is designed to minimize streaming of high levels of radiation.
- Shielding of compartments which do not contain radiation sources is based on necessary accessibility of the compartment.
- Shielding of the controlled area to the outside during normal operation and anticipated operational occurrences is such as to ensure safe adherence to the limits for non radiation workers on the rest of the power plant site, and safe

adherence to the dose limits for the public off-site.

- The physical layout of the controlled area is selected in such a way that compartment configuration meets radiological protection requirements wherever possible.
- No compartment is to be entered through compartments in which local dose rates are expected to be higher than in the target compartment itself.
- Entrances are equipped with doors or traps where necessary for radiation protection reasons.
- Wall penetrations, e.g. for ventilation, cables and pipes, are positioned and designed such that radiation passing through them does not govern the design dose rates in adjacent compartments.
- At selected locations in the controlled area, the local dose rate is monitored by means of stationary or area-dedicated portable measuring instruments.
- Shielding is such as to allow access to control rooms for the maintenance of a safe plant state.

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