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Pebble Bed Reactor Scoping Safety Study

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Acronyms and Definitions

Acronym	Definitions
2A (break)	Double Ended Guillotine (break)
ALARA	As Low As Reasonably Achievable
ANSI	American National Standards Institute
ASME	American Society of Mechanical Engineers
AVR	Arbeitsgemeinschaft Versuchsreaktor (German for Jointly-operated Prototype Reactor)
BDBA	Beyond Design Basis Accident
BMI	German Federal Minister of the Interior
Bq	Becquerel
C&I	Control and Instrumentation
CAS	Compressed Air System
CB	Core Barrel
CFR	Code of Federal Regulations
CLS	Core Loading Subsystem
CRD	Control Rod Drive
CRDM	Control Rod Drive Mechanism
CW	Cooling Water
DBA	Design Basis Accident
DC	Direct Current
DFC	Depressurized Forced Cooling
DiD	Defense in Depth
DLOFC	Depressurized Loss of Forced Cooling
DOE	Department of Energy
DNxx	European Nominal Pipe Diameter ‘xx’, i.e., DN65 means 65 mm “nominal” pipe inside diameter, actual pipe dimensions depend on the pipe manufacturing type.
EAB	Exclusion Area Boundary
EPA	Environmental Protection Agency
EHS	Equipment Handling System
FCS	Fuel Handling Control Subsystem
FE	Fuel Element
FHSS	Fuel Handling and Storage System
FOAK	First-of-a-Kind
FPS	Fire Protection System
FS	Fuel Sphere
FZJ	Forschungszentrum Jülich GmbH (Jülich Research Centre)
HEPA	High Efficiency Particulate Air Filters
HEU	Highly Enriched Uranium
HFE	Human Factors Engineering

Acronym	Definitions
HLW	High-Level Waste
HLWS	High-Level Waste System
HPB	Helium Pressure Boundary
HTGR	High Temperature Gas-cooled Reactor
HTR	High Temperature Reactor
HVAC	Heating, Ventilation and Air-conditioning
IAEA	International Atomic Energy Agency
IEEE	Institute of Electrical and Electronics Engineers, Inc.
INL	Idaho National Laboratory
ISA	Instrumentation, Systems and Automation Society
ISI	In-Service Inspection
KTA	Kerntechnische Ausschuss
LBE	Licensing Basis Event
LEU	Low-Enriched Uranium
LOFC	Loss of Forced Cooling
LWR	Light Water Reactor
MM	Multi-Module
MPS	Main Power System
NGNP	Next Generation Nuclear Plant
NFPA	National Fire Prevention Association
NOAK	Nth-of-a-Kind
NPP	Nuclear Power Plant
NRC	Nuclear Regulatory Commission (U.S.A.)
NUREG	Nuclear Regulations (from NRC)
OD	Outside Diameter
OEM	Original Equipment Manufacturer
OHSA	Occupational Health and Safety Act
PB	Pressure Boundary
PBR	Pebble Bed Reactor
PBMR	Pebble Bed Modular Reactor
PGA	Peak Ground Acceleration
PLOFC	Pressurized Loss of Forced Cooling
PP	Power Plant
PPB	Primary Pressure Boundary
PRS	Pressure Relief System
PWR	Pressurized Water Reactor
QA	Quality Assurance
RC	Reactor Cavity

Acronym	Definitions
RCCS	Reactor Cavity Cooling System
RCS	Reactivity Control System
ROT	Reactor Outlet Temperature
rpm	Revolutions Per Minute
RPS	Reactor Protection System
RPV	Reactor Pressure Vessel
RS	Reactor Scram
RSS	Reserve Shut-down System
RU	Reactor Unit
SAR	Safety Analysis Report
SAS	Small Absorber Sphere
SiC	Silicon Carbide
SSC	Structures, Systems and Components
SSE	Safe Shut-down Earthquake
THTR	Thorium High Temperature Reactor
TRISO	Tri-isotropic Coated Particle
U^{235}	Uranium - 235
vpm	Volume Parts Per Million

1.0 SUMMARY

1.1 Purpose

This report provides a scoping safety assessment of the pebble bed reactor technology. The basic safety design characteristics of the pebble bed reactor (PBR) are reviewed. This review and evaluation is based on an existing pebble bed reactor safety analysis report that is not publicly available [1]. The PBR reactor design and safety analysis examined is the HTR-Module, an AREVA pebble bed high temperature gas-cooled reactor design. The purpose of this assessment is to: a) examine the safety features of the pebble bed reactor, b) review and assess the main accident sequences and the dose consequences associated with each scenario, and c) evaluate the key PBR technology issues that could impact safety.

This scoping safety assessment will provide technical supporting input for a decision by the DOE on future development of the PBR technology for the Next Generation Nuclear Plant (NGNP) project.

1.2 Pebble Bed Reactor Background

In the late 1980s a modular pebble bed reactor concept, the HTR-Module, was proposed in Germany. The HTR-Module is a high temperature pebble bed modular gas-cooled reactor with a cylindrical core and limited passive decay heat removal features as required by the German regulatory authority. The HTR-Module design was reviewed by the German regulatory agency (TÜV) and approved by the German reactor safety commission (RSK – expert commission of the federal government). At the time the design was considered ready for final design activities in Germany. This design formed the basis for the subsequent PBR modular reactor designs and is therefore selected as the reference design for this scoping safety assessment.

The HTR-Module reactor core is a cylindrical structure constructed from graphite reflector blocks. The pebble bed core contains approximately 360,000 fuel spheres (i.e., pebbles). Each fuel sphere contains approximately 11,600 coated fuel particles for a total heavy metal loading of seven grams of low enriched uranium (LEU) oxide. The pebbles are randomly packed in the vessel. The fuel spheres circulated through the core multiple times. The direction of the pebble movement is from top to bottom. As the pebbles are removed from the bottom they are examined for their structural integrity and burnup level. Once it is determined that the pebble is structurally sound and has not yet reached its burnup limit it is returned to the top of the pebble bed core. A total of 15 cycles are expected for the HTR-Module fuel sphere before its 80,000 MWd/MT burnup target is reached and it is discarded into the used fuel storage/transport facility. When the HTR-Module fuel sphere is discarded a fresh fuel sphere is introduced to the top of the pebble bed core.

The reactor uses helium gas as the heat transport media. The cold gas is blown in from the top of the core and forced through the packed bed of fueled spheres to carry off heat generated by the nuclear fission. The heat generated in the reactor core is carried by the gas to the steam generator where it transfers its heat to the water in the steam generator to produce super-heated steam. The primary circuit is then completed as the cooled gas is forced back into the core by the primary gas circulator.

Reactor control is achieved with control rods inserted into the side reflector. The core diameter is selected such that the geometry provides for sufficient negative reactivity worth in the radial absorber rods so that in-core reactivity control is not necessary.

A secondary reactor shut-down system is also provided. This system consists of neutron absorber spherical elements that are dropped into the dedicated channels in the graphite reflector to shut down the reactor. This system is available as a backup/secondary system to the control rods, but it is not used for power shaping or power maneuvering.

The reactor is designed to operate as "base-load" or "load-following" modes. Load-following mode of operation is achieved by varying the speed of the helium circulator thus controlling the primary coolant flow.

The plant is configured with two 200 MWt reactor and steam generator sets, each generating super-heated steam for independent turbine-generator sets for electricity production and reboilers to provide high temperature steam for industrial application steam heating.

1.3 Conclusions

The design and safety characteristics of the high temperature gas-cooled reactor and specifically the pebble bed concept of this technology are dramatically different from that of the well known light water reactors (LWRs). In the United States, before widespread use of this technology is realized, safety and licensing rules and regulations must be developed and exercised. Nevertheless, reactors with PBR concepts have been reviewed by regulators in other countries such as Germany, South Africa, and China.

This study's referenced PBR design, i.e., HTR-Module, has been reviewed extensively by the German regulator TÜV and the RSK. The design and the regulatory review process in Germany were based primarily on LWR safety and licensing experience, and was therefore considered deterministic; however, extensive probabilistic safety assessment (PRA in the U.S.A. context) insights were used to support the safety design case.

The HTR-Module safety characteristics and the original design basis accident analysis developed for the German HTR-Module licensing activities in the late 1980s were reviewed and assessed in the present study. This includes all heatup and cooldown event scenarios from either pressurized or depressurized state. In addition several beyond design basis events scenarios were reviewed and assessed including complete failure of cavity cooling. The specific phenomenon of air and water ingress was discussed. This included assessment of design basis accidents involving air and water ingress. In addition, the HTR-Module plant design includes features that protect the plant from internal and external hazards such as fires, flood, sabotage, explosions, and airplane crash. These features were reviewed and assessed.

The result of our assessment indicates that the HTR-Module can demonstrate adequate protection of public health and safety within the U.S.A. regulatory environment. There are extensive designs, experiments, and R&D results that support this safety claim. The highlights of the original HTR-Module safety analysis are discussed and reported in this document.

Several well known PBR technology issues that could have an impact on the safety of the HTR-Module were identified in course of this study and their impact on safety is discussed in this report. These include: a) the stochastic nature of the pebble bed reactor concept, b) sudden excess reactivity insertion as the result of core compaction, c) safety impact of graphite dust, and d) safety impact of lost or broken pebbles.

The results of our review and assessment of these technology issues indicate that none have adverse impact on plant nuclear safety in such a way that could limit the progress of the PBR technology deployment.

2.0 INTRODUCTION

2.1 NGNP Project

The High temperature gas-cooled reactor (HTGRs) can provide an important addition to the U.S.A. and the world's energy supply portfolio. Enabling commercial deployment of the HTGR technology has gained importance as environmental and energy security issues have become more apparent, and the national resolve to solve these issues has become stronger. The Next Generation Nuclear Plant (NGNP) project authorized by the Energy Policy Act of 2005 (EPAct) provides for a collaborative effort between government and industry to enable the commercialization of the HTGR technology.

To achieve this goal, the NGNP Project must develop and demonstrate the design, licensing, performance, operational capabilities, and economic viability of HTGR and associated process heat technologies. The Project must further enable development of the commercial vendor/owner/user infrastructure, and support the timely Design Certification of the commercial designs by the NRC to help assure subsequent deployment in the commercial market place.

Currently, the NGNP Project is a Government-sponsored project focused on the development, early design and licensing of an advanced HTGR and the associated advanced technologies to transport the high temperature process heat. The basis for the HTGR technology embodied in the NGNP was first developed over 40 years ago in the UK, the U.S.A. and Germany. Most of the previous work has focused on the generation of electricity. Seven experimental and demonstration reactors have been built world-wide, including a U.S.A. commercial scale demonstration of a specific HTGR concept for electric power generation at the Fort St. Vrain plant that operated from 1976 through 1989. Other HTGR system-related development efforts exist in South Africa, France, Japan, Russia, and China at the design stage or engineering pilot scale. Additionally, a commercial scale demonstration plant utilizing the pebble technology is currently under construction in China.

As currently envisioned, the NGNP Project will result in full scale First-of–a-Kind (FOAK) facilities that demonstrate the commercial potential of the HTGR and associated technologies. Definition of the specific NGNP facilities to be built as part of the Project will be established over the next several years. The conceptual design for two HTGR technologies are being developed as part of the initial phase of the NGNP project. The prismatic design concept is being developed under a DOE FOA funding by the General Atomics design team and the pebble bed HTGR reactor technology concept is being evaluated by the AREVA design team. As the conceptual design and technology assessment work progresses, the facility design is better defined, and the costs and the economics of the project are defined with more certainty.

2.1.1 NGNP Project Objectives

The primary goal of the NGNP Project is enabling the commercialization of the HTGR technology across new industrial and commercial markets previously not accessible to nuclear technology. The NGNP Project will create the option for deployment of the HTGR technologies for a range of applications and sites not traditionally served by nuclear energy.

Key objectives for achieving this goal include [2]:

- Fully characterizing the potential market through end-user collaborations and application studies in order to identify a wide range of viable candidate sites, applications and projects
- Providing guidance to design teams regarding the range of site and application requirements that could impact NGNP design and licensing
- Preparing, submitting, and acquiring one [3] or multiple Early Site Permits (ESPs) that envelop the range of potential sites and applications for deployment of HTGRs

- Performing the design activities necessary to prepare, submit, and eventually obtain a Combined License (COL) for one or both HTGR technologies
- Developing the regulatory framework for the licensing of the HTGR technologies
- Enabling the long-lead developmental activities for fuel, high-temperature materials, and methods that support licensing and subsequent construction of the FOAK facilities
- Securing the fuel fabrication capacity needed to support HTGR projects
- Completing the final design activities to allow construction, start-up, confirmatory testing, and operation of the FOAK facilities
- Acquiring the necessary government incentives to make the FOAK facilities economically viable investments for the private sector
- Construction, start-up, confirmatory testing, and completing a commercial operations run for the FOAK facilities
- Enabling the establishment of the supply chain infrastructure necessary for commercial build-out of the HTGR technologies
- Obtaining design certifications from the NRC to support the deployment of the initial fleet of commercial plants
- Capturing the lessons learned from FOAK construction and operations, and validating the assumptions for future plant construction costs and schedule

By meeting the objectives above, it is expected that the NGNP Project will establish an acceptable basis for commercial deployment of the HTGR technology in the broader energy sector. Completing the design, licensing, construction and initial operations of a FOAK plant provides a solid foundation for commercialization and commitment to the extensive deployment anticipated for the HTGR technology, end-user site requirements and hazards, and nuclear-industrial collocation conditions.

2.2 Pebble Bed Reactor Technology Status Assessment

The U.S. Department of Energy (DOE) has selected Idaho National Laboratory (INL) as the lead national laboratory for nuclear energy research. Per the terms of the EPAct, Title VI, Subtitle C, Section 662, INL, under the direction of DOE, will lead the development of the NGNP by integrating, conducting, and coordinating all necessary research and development activities, and by organizing all project participants, including industry. INL will also be responsible for conducting site and project related procurements, and coordinating project efforts within the industrial and international communities.

As required by the EPAct, the Nuclear Energy Advisory Committee (NEAC) will conduct a “first project phase review,” when the first phase of NGNP is nearly complete. The first phase of NGNP includes the research and development, technology, licensing, and conceptual design information derived from all Phase 1 activities. Two main technology options are under consideration for the NGNP: the prismatic block core modular HTGR, and the pebble bed reactor (PBR) modular HTGR. The evaluation of these two reactor concepts will form an important part of the Phase 1 review. Conceptual design information for the prismatic reactor concept is being developed under a separate work scope. The purpose of this work is to develop key information to support the review of the PBR technology option.

This effort will provide a limited assessment of the PBR concept that includes the basic design information and various assessments of the design concept needed to evaluate the maturity of the PBR design concept and its technical readiness to advance to the next level. This work did not intend to produce a conceptual design of the NGNP reactor with the PBR technology.

The bases for the PBR technology readiness status assessment is the AREVA HTR-Module design developed in Germany in the late 1980s plus enhancements that support current requirements, safety, and licensing.

Adjustments to the referenced plant design would be considered based on HTGR design experience since the HTR-Module was not originally developed to meet the NGNP requirements. The pertinent NGNP requirements are reactor outlet temperature of 750°C or greater, electricity production, and heat for other process applications.

An evaluation of the readiness of this design is made using trade studies and expert engineering judgments. The results of these assessments are documented in four deliverables:

1) The Plant Design Description report – PDD describes the reference PBR design that is based on the HTR-Module and identifies potential design enhancements. The PDD identifies key system requirements, describes the overall PBR plant and provides a description of each critical structure, system, and component (SSC). Engineering analyses and trade studies, such as a point design and steady-state plant analyses, shall be performed to adapt the previous designs to the NGNP requirements.

2) The PBR Technology Readiness Assessment report – The technology readiness assessment comments on the readiness status of various technologies necessary to build the NGNP with PBR technology. An existing set of design data needs (DDN) will also be reviewed and potential changes or modifications will be recommended. A study evaluating the overall PBR technology readiness for deployment was performed. This study performed the following: a) examined key PBR technology issues, b) identified technology needs by evaluating the existing design data needs (DDNs) for the PBR design and gaps in the identified needs, c) discussed fuel and graphite qualification and acquisition, and d) discussed the constructability and component transportability of the PBR design concept.

3) The PBR Scoping Safety Study report (this report) – In the safety study report the PBR safety case is presented and discussed, the original German HTR-Module accident analysis results are provided and discussion of key technical issues relevant to PBR safety case is presented. The scoping safety study is based on existing analyses; new analyses are not within the scope of this work. This work included review of prior HTR-Module safety analyses. The review included identification and assessment of the PBR plant safety issues and discussion/assessment of the expected outcomes for each major accident sequence. Considerations specific to the PBR technology, such as graphite dust and the requirement for a stochastic approach to the core design and analysis, are reviewed and discussed. The safety study also includes an evaluation and discussion of expected dose at the site boundary (about 400m) for accidents with dose releases using accepted U.S.A. dose calculation methodology and with the original accident source terms.

4) The Cost and Schedule report – This report provides an updated cost and schedule for the PBR FOAK and the NOAK plants. Cost and schedule estimates for deployment of the PBR are developed for the FOAK and NOAK plants. The cost estimate is based on historical information from previous PBR evaluations and similar components as appropriate with scaling, and adjusted as necessary to match the current PBR design concept. The cost estimate addresses a single plant for the FOAK plant and a multiple plant installation for the NOAK. The plan includes an overall project schedule covering detailed design, fabrication, and construction of the demonstration PBR plant.

2.3 Approach to Scoping Safety Study

Steps utilized to develop this Scoping Safety Study report are:

1) Review previous PBR safety assessments and analyses – the HTR-Module safety analysis report was used to develop the assessment. The HTR-Module safety analysis report was developed about twenty years ago and is not in general public domain. The report must be interpreted with care since design changes and regulatory reviews were being conducted concurrently.

2) Evaluate PBR safety characteristics in light of current technology – the safety characteristics of the HTR-Module are reviewed in light of the current regulations, considering that limited HTGR safety regulations exist in the U.S.A. regulations, comments are nevertheless made concerning acceptability of the HTGR

safety case in the current and anticipated U.S.A. regulatory environment for the new and advanced reactor concepts.

3) Identify and evaluate major safety issues and expected outcomes – the scoping safety study includes identification and assessment of the PBR technology issues that impact plant safety and provide comments on the expected outcomes for major accident sequences. Considerations specific to the PBR technology such as safety implications of graphite dust and the stochastic core characteristics of the PBRs are reviewed and discussed.

4) Finally, the scoping safety study includes an evaluation and discussion of expected dose at a plant site boundary for those accident sequences that have dose consequences. This includes calculation of site dose consequences using U.S. NRC accepted methodology for an exclusion area boundary of 400m using previously calculated HTR-Module source terms.

2.4 Document Structure

This document is organized as follows:

Chapter 1 provides a summary of the scoping safety study evaluations, assessments and conclusions. This section also summarizes the methodology used to develop these conclusions.

Chapter 2 provides an overview of the NGNP project and an introduction to the role the AREVA PBR technology readiness status assessment task and the scoping safety study.

Chapter 3 provides a short description of the HTR-Module primary and support systems to promote context and understanding of the safety evaluation. More detail PBR system description is provided in the Plant Design Description document that is a separate deliverable of the technology assessment document.

In Chapter 4 the HTR-Module safety case is discussed and major plant safety features and components are discussed and evaluated.

Chapter 5 provides discussions and analysis of the PBR safety events including discussions of design basis and beyond design basis events.

In Chapter 6 four major PBR technology issues affecting plant safety analysis are discussed. This includes the stochastic core characteristics, a sudden core compaction potential, graphite dust and its safety implications, and the safety implication of the broken or lost pebbles.

In Chapter 7 the HTR-Module accident sequences with radioactivity release were identified and the site boundary doses were recalculated using the U.S. NRC methodology and compared to the acceptable regulatory release limits.

References used to perform this study are listed in Chapter 8.

3.0 PEBBLE BED REACTOR DESCRIPTION

In the late 1980s the modular pebble bed reactor concept, the HTR-Module, was proposed in Germany. The HTR-Module is a pebble bed modular gas-cooled reactor with a cylindrical core and passive decay heat removal features. This design formed the bases for the subsequent PBR modular reactor designs and is the reference design for this scoping safety analysis.

The pebble bed reactor cylindrical core structure is constructed from graphite reflector blocks. The HTR-Module pebble bed core consists of approximately 360,000 fuel spheres (i.e., pebbles). Each fuel sphere contains approximately 11,600 coated fuel particles and seven grams of low enriched uranium (LEU) oxide. As the pebbles are automatically loaded into the cylindrical core cavity, they become randomly distributed with a packing density (or packing fractions) of 0.61.

The reactor uses helium gas as the heat transport media. The cold gas is blown in from the top of the core and forced through the packed bed of fueled spheres to carry off heat generated by the nuclear fission. The heat generated in the reactor core is carried by the gas to the steam generator where it transfers its heat to the water in the steam generator to produce steam. The primary circuit is then completed as the cooled gas is forced back into the core by the primary gas circulator.

The primary reactor control is achieved with control rods inserted into the side reflector. The core diameter is selected such that the geometry provides for sufficient negative reactivity worth in the radial absorber rods so that in-core reactivity control is not necessary.

A secondary reactor shut-down system is also available. This system consists of neutron absorber spherical elements that are dropped into the dedicated channels in the graphite reflector to shut down the reactor. This system is available as a backup/secondary system to the control rods, but is not used for power shaping or power maneuvering.

The reactor is designed to operate in “base-load” or in “load-following” modes. Load following mode of operation is achieved by varying the speed of the helium circulator thus controlling the primary coolant flow.

A representative plant schematic is presented in Figure 3-1.

The HTR-Module power plant consists of two reactor and steam generator sets. Each set is comprised of one high-temperature pebble bed reactor core, one steam generator and one primary gas circulator. The primary helium transports the reactor fission heat to the steam generator coils. Helium is flows from the reactor to the steam generator through a gas duct pressure vessel where the reactor inlet and outlet flow in opposite directions separated by an insulated circular duct.

The general plant characteristics and selected plant parameters are presented in Table 3-1 and Table 3-2. Values presented in these tables are representative of one reactor unit. The basic commercial plant power block, however, is based on a dual unit configuration that combines two reactor units into a single reactor building.

3.1 PBR Primary System

Each reactor unit, depicted in Figure 3-2, has a cylindrical core made from graphite blocks forming the core cavity. The graphite blocks also function as the neutron reflector, core heat sink and radial residual heat removal path when the active core heat removal system is not operating. During reactor operation the core cavity is filled with spherical fuel elements. Once critical core neutronic conditions are reached, nuclear heat is generated by uranium fission and transported to the steam generator by the circulating helium gas.

The reactor is designed for continuous refueling operation. Therefore, it operates with a low excess reactivity. Fresh fuel is loaded from the top of the core and used-fuel is discharged through the bottom. Each fuel element generates a small amount of fission heat as it passes through the core. Each fuel sphere makes multiple passages through the core before reaching maximum allowable burnup.

The primary circuit components shown in Figure 3-3 consists of the reactor pressure vessel with the core internals, shut-down systems and facilities for the charging and discharging fuel elements, and the gas duct pressure vessel conveying helium coolant to/from the once-through helical coil gas-to-water steam generator.

3.2 Primary Circuit Components

The reactor primary components consist of the reactor, the steam generator vessel with helical coil steam generator tubes, feedwater inlet and steam outlet nozzles, and a helium circulator. The reactor and the steam generator vessels are connected by a gas duct pressure vessel with internal concentric ducting that directs and separates hot and cold helium gas between the reactor vessel and the steam generator vessel. The steam is sent to a multistage steam turbine for electricity generation and/or circulated through one or more re-boilers to produce high temperature process steam for industrial applications.

Each reactor unit is installed in a separate concrete cavity that supports the weight of the primary system pressure vessels. Surface coolers are installed on the inside of the reactor cavity to remove dissipated heat during normal operation and decay heat during shut-down and accident conditions.

As shown in Figure 3-3 the reactor and the steam generator are in a side-by-side, staggered position offering the following design advantages.

- After a reactor shut-down, natural circulation of hot helium through the primary circuit is minimized by the thermal-hydraulic decoupling of heat source and heat sink. Therefore, there is no need to cool the steam generator after shut-down.
- The positioning of the steam generator beside and lower than the reactor permits simple, operationally-favorable upward evaporation.
- Water from a steam generator tube break will pool in the steam generator and not all enter the core.
- The substantial separation of the reactor core and the steam generator tubing by the concrete shielding walls of the reactor cavity allows easy access to the steam generator cavity after shut-down for inspection and maintenance.

The steam generator is designed as a once-through helical-coil tube and shell, with water/steam flowing inside the tubes. The downward flow of helium gas over the steam generator tubes is cooled from 700°C inlet to 250°C outlet, heating up the feedwater from 170°C and generating steam at 530°C at a secondary pressure of 19 MPa.

3.3 Conduct of Operations

The HTR-Module design can be tailored to the requirements imposed by the electric grid and the process steam system. The central consideration is the demand for high availability of process steam supply and electricity generation.

For this reason the power block essentially consists of two identical 50%-duty power plants that are suitably interconnected and share key operational support systems. This makes it possible to continue operation of one reactor, during overhauls in the other or on failure of plant items, without interruption of power operation, possibly at reduced power.

The description of plant operation is divided into sections on start-up, power operation and shut-down. The description of the modes of operation that follows is restricted to essential measures that have to be taken to transition the plant operating conditions from a specific initial condition to the required condition:

- Cold start-up
- Hot start-up after reactor scram
- Shut-down
- Shut-down after reactor trip

- Power operation
- Residual heat removal

3.3.1 Start-Up

Cold Start-Up

The plant must be in the following standby status prior to start-up:

- Primary system filled with helium (primary system pressure about 38 bar at 50°C)
- The steam/power conversion system (turbine-generator) and the auxiliary and supporting systems are on standby.

Start-up operation is initiated by starting the primary gas circulator. Start the start-up and shut-down pump system (water-only mode) if it is not in operation.

The helium flow is manually set to approximately 40%, the feedwater flow to approximately 20% and the pressure in the water/steam cycle to approximately 70 bar.

The reactor is then made critical and the hot gas temperature is raised at a rate of about 2 K/min.

As the temperature of the primary coolant and hence of the water/steam cycle increases, generation of steam commences in the water/steam cycle. The produced steam is fed via a steam transformer valve to the start-up and shut-down condenser.

At a hot gas temperature higher than approximately 450°C, the closed-loop controls of the hot gas temperature, main steam temperature, main steam pressure, and feedwater flow are placed into operation. Then the set point of the hot gas temperature is raised at 2 K/min.

At the same time the set points of the main steam temperature and pressure are raised to 530°C and 190 bar as a function of the hot gas temperature. The set point of the feedwater flow remains constant at 20% (Figure 3-4).

Start-up and loading of the steam turbine generator sets can be performed independently of start-up of the reactors provided they are generating sufficient steam to drive the turbines. Steam must be dry (at least 50K superheat) and have a higher temperature than the high-pressure turbine shaft.

Because the mean helium temperature and consequently the helium pressure increases only slowly due to system inertia, the desired helium pressure characteristic will be realized with the help of the pressure control of the primary system.

The start-up to nominal power continues with the help of the steam generator power control and the supervisory control unit that changes the set values of the hot gas temperature, circulator speed, and thermal steam generator power according to the part load diagram. A gradients limiter ensures that the set value of the hot gas temperature is not changing faster than 2 K/min.

The overall duration of start-up operation is about 6 – 8 hours.

Hot Start-Up After Reactor Scram

Hot start-up is possible for about one hour after a reactor scram provided the cause and the consequences of the scram can be traced and eliminated within this period. If it is not possible to initiate hot start-up within this time period, the rapid rise in xenon poisoning of the core prevents restarting within the next 24 hours. If operation was at a part-load level prior to reactor scram, the period during which a hot start-up is possible increases and the subsequent outage time decreases.

The following actions are initiated by reactor scram:

- Drop of all reflector rods

- Shut-down of the primary gas circulator
- Secondary-side isolation of the steam generator and shut-down of the feedwater pumps. The circulator damper is also closed.

The plant remains in this hot standby condition until it can either be restarted (returned to power) or residual heat removal operation is initiated (cold shut-down, see Section 3.3.5).

The restart from hot standby proceeds as follows:

- The primary gas circulator will be run up to the minimum speed (10% of the nominal speed), the steam generator isolation valves will be opened, and the feed water mass flow will be increased manually up to approximately 10% of the nominal value. In the event that the main steam temperature increases over 530°C, the feed water mass flow will be increased further up to the maximum 15%.
- The controls for the main steam pressure (set value 190 bar) and main steam temperature (set value 530°C) will be activated afterwards.
- The steam generator is then used to reverse the core temperature rise that occurs in the standby condition and to bring the main steam pressure to nominal conditions.
- The six reflector rods are simultaneously maneuvered from their lowest to uppermost positions. As shown in the example given in Figure 3-5, the core goes critical about 5 to 10 minutes after initiation of hot start-up.

Followed by the transition from the start-up and shut-down control circuits to the steam and power conversion control circuits where the hot gas temperature and the plant thermal power are controlled.

Afterwards the thermal power will be run up with approximately 5% nominal load/min up to 45% of the nominal value by the help of the supervisory control unit. The set values of the hot gas temperature, circulator speed and thermal steam generator power will be changed according to the part load diagram.

Primary system cold gas temperatures are in the range of 200°C – 230°C due to the feedwater temperature and heat transfer characteristic of the steam generator.

About 20 minutes after the initiation of hot start-up a repeated increase in xenon poisoning is curbed by the power increase. Subsequent xenon burn-out increases core reactivity that is compensated for by reactor control (by reflector rod insertion).

3.3.2 Shut-down

Shut-down will be initiated by using the supervisory unit control to shut down the affected reactor within one hour to a low part load set point of less than or equal to 20% of the nominal power. The set values of the hot gas temperature, primary gas circulator speed and thermal steam generator power will be changed according to the part load diagram. A gradients limiter ensures that the set value of the hot gas temperature is not changing faster than 2K/min.

By isolating the main steam valves on this power level, the switch from the turbine operation to the shut-down operation, with the start-up and shut-down condenser as heat sink, takes place. Furthermore the supervisory unit control and the thermal power control will be put out of operation.

The feedwater mass flow will be kept constant at 20%. The hot gas temperature set value will be decreased in the next step by 2K/min.

The set values of the main steam temperature and the main steam pressure will be decreased simultaneously as well, depending on the hot gas temperature. The set value of the main steam pressure will be decreased to the minimum value of 70 bar.

The main steam temperature control increases the primary gas circulator speed (during decreasing hot gas temperature) up to 40% of the nominal speed.

Once the primary gas circulator speed has reached this value it will be kept constant and the controls for the hot gas and main steam temperature will be put out of operation.

The further decrease of the hot gas temperature takes place manually. The steam production will be finished; the switch to the water operation takes place.

At this point and onwards, the nuclear shut-down of the reactor takes place. The long-term residual heat removal is made by the start-up and shut-down circuit during circulating operation.

After this the pressure control of the primary system will be put out of operation. In contrast to the expected temperature patterns on the inlet and outlet of the reactor, the mean primary coolant temperature and the primary system pressure decrease slower because of larger heat capacity inside the reactor.

Cooldown After Reactor Scram

If a restart is not possible within one hour of reactor scram the plant is cooled down.

For this purpose the main heat transfer system is put into operation in conjunction with the start-up and shut-down circuit. Steam produced in the steam generator is removed by the start-up and shut-down system. The primary coolant and feedwater flows are set to about 10% of nominal. The plant then cools down at a rate of 1 to 5K/min. Main steam temperature and pressure are lowered as a function of the hot gas temperature by the closed-loop controls until the water-only condition is reached.

3.3.3 Power Operation

The range of power capability for a single reactor is unrestricted between 50% and 100% of reactor nominal power. A single HTR-Module reactor can operate in a lower range of power of about 20% to 50% of the nominal power if the xenon poisoning has been reduced by the previous operating mode.

Operations at constant hot gas temperature is preferred for fast load changes so as to ensure low-amplitude temperature transients whereas the reactor outlet temperature is lowered for long term operation at a reduced power level.

Figure 3-7 shows the part load operating window (shaded area) in which requirements of the majority of operational parameters for the primary circuit components are met.

A change in thermal power of the reactor is performed by the plant control system. A power demand signal generated in the plant control system will modify the control set points of the lower tiered control circuits for the hot gas temperature, circulator speed and thermal power according to the plant part load diagram (Figure 3-7).

Power output is apportioned to the electric grid and the process steam system as follows: the process steam system extracts the required amount of process steam from the high pressure exhaust steam system. The control parameters are set by the steam and power conversion control circuits. The plant thermal power generated and not required by the process steam system is converted into electric power in the turbine generator sets at the highest possible efficiency level.

The two steam turbines are designed such that during operation with low process steam demand (e.g. 2 x 85 Mg/h, 1 Mg = 1 tonne) and 100% reactor thermal output the condensing turbines are just able to accept all leftover steam for power generation. When process steam demand is high (e.g. 2 x 200 Mg/h), the turbines generate power in straight backpressure operation.

This design makes it possible to operate both HTR-Module reactors at full power throughout the load cycles - if so desired.

3.3.4 Reactor Scram

If the reactor scrams, the reactor protection system also initiates the following actions:

- Reflector rod drop
- Shut-down of the primary gas circulator
- Closing of the feedwater and main steam side steam generator valves

Furthermore, the gas circulator damper is closed.

Reactor scram for each modular unit is triggered by a dedicated reactor protection system but can also be triggered manually.

Reactor scram in one of the two reactors results in a 50% drop main steam flow. This is compensated for by the controls of the heat-extracting system in such a way that the reactor that is still running and the steam-consuming system may continue operation (possibly at reduced levels in order to adjust to a new load demand after shut-down of selected loads).

Reactor scram is followed, in the long term, by temperature redistribution and a gradual temperature rise in the core that is separated from its operational heat sink (steam generator). During this phase a portion of the residual heat is removed by the cavity cooler. Temperatures in the core after reactor scram are shown in Figure 3-8 for an initial power level equivalent to 105% nominal power.

Mass flow patterns in the steam generator (Figure 3-9) are governed by the simultaneous closure of the feedwater inlet valve and tripping of the primary gas circulator. Closure of the circulator damper is delayed. Primary and secondary side closure and delay times are selected such that thermal loadings on primary system components from reactor scram remain within permissible limits. Main steam side isolation is performed such that response of the safety valve is prevented.

Figure 3-10 shows the temperature development in the steam generator after reactor scram. The primary coolant inlet temperature remains effectively constant, while the primary coolant outlet temperature decreases and adjusts to the water temperature in the cold area. This decrease will be damped further due to the subsequent flow through the primary system components (steam generator pressure vessel, primary gas circulator). The increase of the main steam temperature is due to the heat up in the hot tube bundle area by the continuing primary coolant mass flow and by the adjustment of the steam temperature to the temperature of the tube wall after isolation of the feed water supply. A short time after the mass flow stagnation, the temperatures of the now stagnating mediums have reached effectively steady final values at the tube ends.

Brief loads in the primary and water/steam cycle (primarily imposed by steam generator response) occurring upon manual reactor scram result in virtually the same flow disturbances discussed in Section 5.3 (e.g., loss of auxiliary power supply, failure of primary gas circulator). In these situations, reactor protection system actions are initiated quickly after onset of the disturbance that the flow transients, which cause the loadings sustained, are shifted a matter of only seconds as against the sequence following manual scram. Consequently the temperature transients are virtually congruent.

During the primary system standby phase that follows brief transients, the temperature distribution in the steam generator remains more or less steady-state. After this phase (about 1 hour) the decision is made whether to cool down the reactor or to restart.

The profiles of core and primary system variables occurring in the two possible subsequent modes of operation (hot start-up or residual heat removal) are discussed in Sections 3.3.1 and 3.3.5.

If the circulator damper fails to close, reverse-flow natural circulation occurs in the primary system with an initial mass flow of about 4 kg/s. Owing to the rapidly changing uplift conditions in the core and in the steam generator, mass flow drops within one hour to about 1.2 kg/s.

Primary coolant heated in the core flows upwards through the top reflector and then downward through the side reflector before it reaches the metallic lower structure and the reactor pressure vessel. The majority of the thermal energy carried out of the core by the primary coolant is absorbed by the side reflector; the coolant flows past the

RPV at a temperature at 300°C initially and about 360°C after one hour. Neither the RPV nor down-stream components such as the steam generator and primary gas circulator are subjected to unacceptable loadings.

All core and primary system components are designed to withstand the thermal loadings sustained after reactor scram for the number of occurrences specified for the plant's service life.

3.3.5 Residual Heat Removal

Reactor residual heat can be removed by the main heat transfer system and/or by the cavity cooler.

Operational Residual Heat Removal by the Main Heat Transfer System

The reactor system is cooled down after normal operation for extended outages by the primary helium circuit. Plant cooldown and long-term residual heat removal are achieved on the primary side by the primary gas circulator and the steam generator and on the secondary side initially by the water/steam cycle and later by the start-up and shut-down system as steam temperature falls and during water-only operation (see Section 3.3.2).

Residual Heat Removal by the Main Heat Transfer System After Reactor Scram

For reactor scrams due to malfunctions, active cooling of the reactor is initially not performed. This facilitates quick restarting of the power plant if the cause can be quickly remedied. During the interruption of heat removal by the heat transfer system, a portion of the residual heat is removed by natural convection, thermal conduction and radiation by the cavity cooler. A plant restart from the hot condition is possible for period of about 1 hour after reactor scram. Increasing xenon poisoning of the core prevents a restart for about 24 hours thereafter. In this case the reactor is then cooled down by the primary helium circuit, if available; assisted by the start-up and shut-down system (see Section 3.3.2).

Residual Heat Removal by the Cavity Cooler

The cavity cooler runs at all times even during normal reactor operation and continues to do so unchanged upon interruption of active heat removal to the main heat sink so that it is unnecessary to take measures to initiate residual heat removal.

Upon failure of forced circulation after reactor scram and with the reactor at pressure, the non-uniform uplift forces cause natural circulation in the core, bottom and top reflectors. Heated primary coolant rises in the core, cools as it comes into contact with the top reflector and the upper portion of the side reflector, and then flows downward around the core perimeter to the bottom reflector. Thermal conduction and radiation also contribute to thermal transport.

Heat transmitted in this manner to the structures adjacent to the core is transported by these core structures by conduction and through gaps by radiation and natural convection to the cavity cooler.

Natural circulation results in pronounced temperature redistribution in the core and adjacent structures. Figure 3-11, Figure 3-12 and Figure 3-13 show the temporal, radial and axial temperature patterns in the core and in various core internals.

After six hours a few fuel elements reach their maximum temperature of about 1130°C and temperatures then begin to fall slowly. Only about 5% of the fuel elements remain at temperatures in excess of 1000°C for 20 hours. The center of the underside of the top reflector briefly attains temperatures of about 1000°C. Its perimeter rises over a lengthy period to temperatures of about 700°C. Other core internals heat up only to a limited extent.

The reactor pressure vessel reaches a maximum temperature of about 320°C after some 75 hours. The maximum core barrel temperature is about 400°C. Under these conditions, the thermal energy removed by the cavity coolers peaks briefly at 815 KW.

If the circulator damper fails to close, natural circulation in addition to natural convection in the core reverses flow in the primary system. The development of natural circulation during the first hour after reactor scram is described in Section 3.3.4.

The mass flow later drops to below 0.5 kg/s after 3 hours and has reached a level of about 0.15 kg/s after 25 hours.

Thermal loadings on the ceramic portion of the core are not higher than if the circulator damper closes. The maximum core temperature is about 50K lower since the residual heat generated in the core spreads over larger portions of the primary system.

After 8 hours the temperature of the primary coolant leaving the side reflector at the bottom reaches a maximum of 390°C and then falls slightly. As the primary coolant moves towards the steam generator, it transfers heat to the steam generator pressure vessel so that it has a temperature that is not in excess of 350°C upon reaching the annular gap between steam generator pressure vessel and tube bundle. The flow of primary coolant, owing to its low mass flow, cannot heat the steam generator pressure vessel or other components to unacceptable levels.

However, if feedwater supply to the steam generator is stopped, natural circulation ceases after about 2-3 hours. In this case temperatures of the fuel elements and the pressure vessel unit are equivalent to those reached with the circulator damper closed.

Table 3-1: General Plant Characteristics

Description	Baseline Value
Nuclear Plant	
Number of Reactors	2
Reactor Type	High Temperature Gas-cooled Graphite Moderated Reactor
Construction	Partially Underground
Reactor Building	Vented/Filtered
Primary Coolant	Helium
Reactor Core	
Fuel Form	Fuel Spheres (Pebbles)
Configuration	Cylindrical Core
Moderator	Graphite
Reflector	Graphite Blocks
Fuel	
Fuel Design	TRISO Coated Particles in Spherical Fuel Elements
Fuel Kernel	UO2
Coating	TRISO
Fuel Enrichment	8 ± 0.5%
Basic Fuel Element	A mixture of coated particle fuel in graphite matrix shaped into a sphere with an out layer of fuel free zone
Primary Loop Configuration	
System Configuration	Conventional Rankine Steam Cycle w/Helium to Water/Steam Generation
No. of Loops	One
No. of Cross Ducts	One
No. of Steam Generators	One
No. of Circulators	One
Primary Fluid	Helium
Steam Generator	
Steam Generator Type	Once-through helical coil
Heat Transfer Medium	Helium to Water/Steam
Fueling System	
Fueling Cycle	Continuous / Multi-pass Recirculation

Table 3-2: Selected Plant Parameters

Description	Baseline Value
Reactor Parameters	
Core Power (MWt)	200
Core Diameter (m)	3.0
Core Height (m)	9.4
Mean Power Density (W/cm^3)	3.0
Number of Pebbles	~360,000
Number of Control Rods	6
Number of Absorber Ball Channels	18
Average Number of Passes per Pebble	15
Number of Particles per Pebble	around 11,600
Heavy Metal Loading (g/pebble)	~7
Discharge Burnup (MWd/MT)	80,000
Fuel Residence Time (days)	~1000
Reactor Pressure Vessel Unit	
RPV Height (m)	25
RPV Flange Outside Diameter (m)	6.7
RPV Inside Diameter (m)	5.9
Circulator	
Circulator (delta P - static head) (bar)	1.5
Steam Generator Pressure Vessel	
SG Pressure Vessel Height (m)	22
SG Inside Diameter (m)	
Upper	3.6
Lower	3.2
SG Tube Outside Diameter (mm)	23
Hot Duct Pressure Vessel	
Hot Duct Pressure Vessel Length (m)	2.9
Hot Duct Pressure Vessel ID (m)	1.5
Hot Gas Duct Inside Diameter (m)	0.75
Hot Gas Support Pipe Inside Diameter (m)	1.0
Insulation Thickness (mm)	100

Figure 3-1: HTR-Module Plant System Schematic Representation

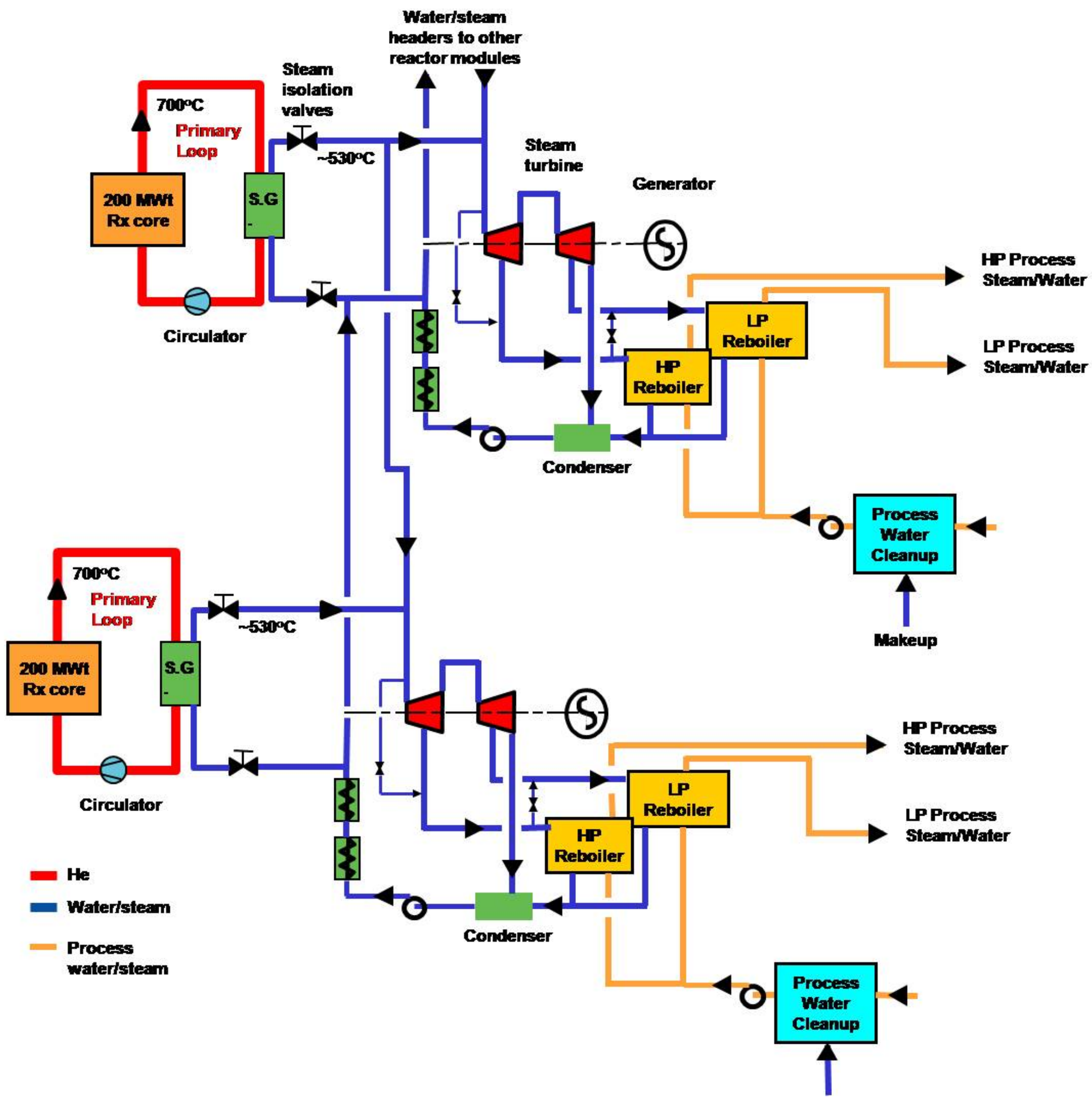
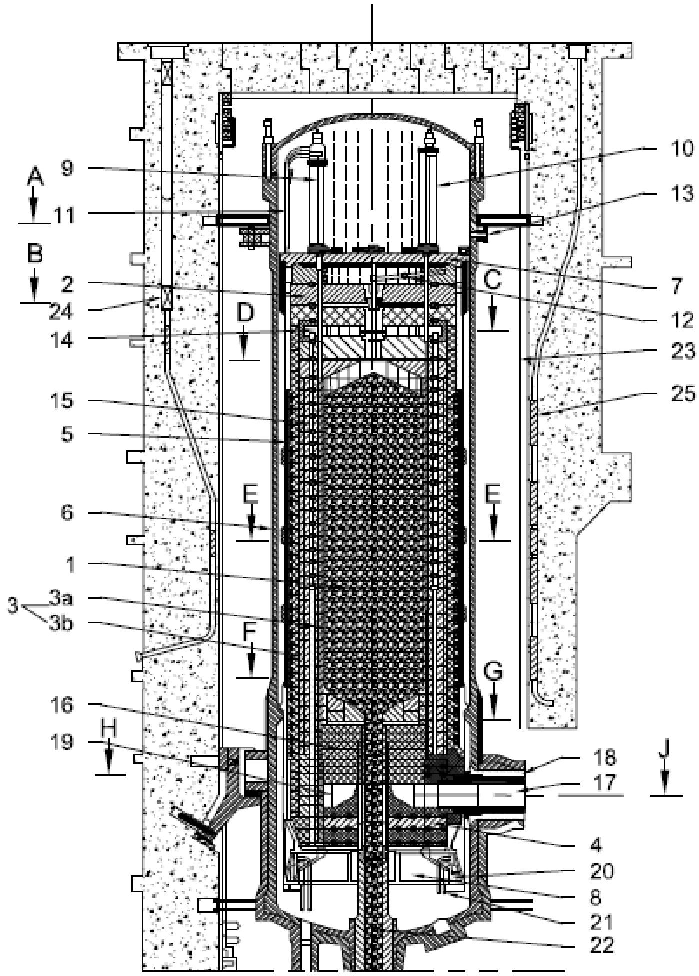


Figure 3-2: HTR-Module Longitudinal Cross Section



1. Pebble bed
2. Top reflector
3. Side reflector
3a. Graphite brick
3b. Carbon brick (borated)
4. Bottom reflector
5. Core barrel
6. Reactor pressure vessel
7. Top thermal shield
8. Lower support structure
9. Small ball shutdown unit
10. Reflector rod
11. Small ball shutdown element feed line
12. Fuel feed
13. Supply nozzle
14. Cold gas plenum
15. Cold gas column
16. Hot gas mixer
17. Hot gas duct
18. Cold gas inlet duct
19. Hot gas plenum
20. Discharge vessel to small ball shutdown system
21. Carrier gas line to small ball shutdown unit
22. Fuel discharge tube
23. Cavity cooler
24. Neutron flux instrumentation (source range)
25. Neutron flux instrumentation (intermediate and power range)

Pebble Bed Reactor Scoping Safety Study

Figure 3-3: HTR-Module Primary System

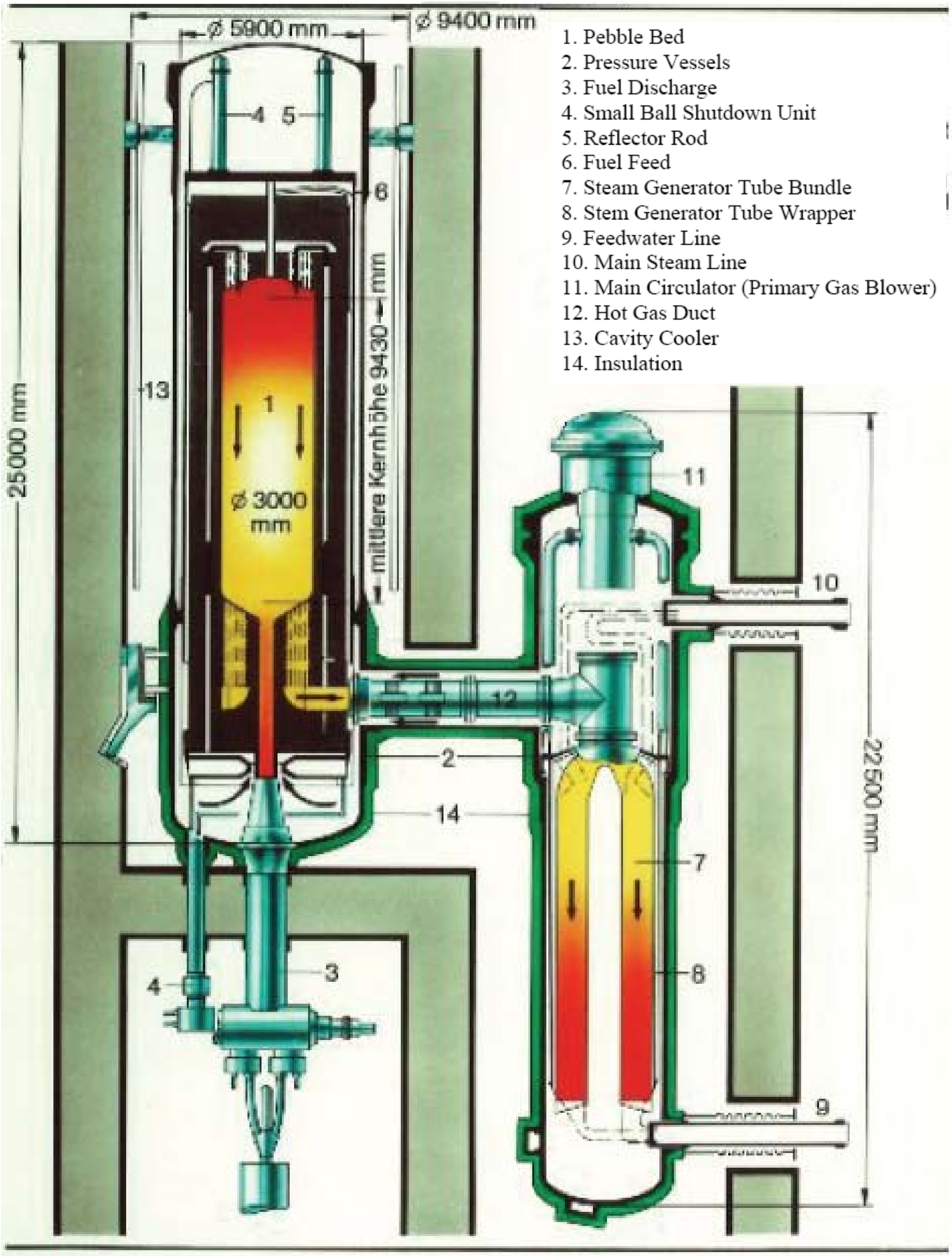


Figure 3-4: Cold Start-Up

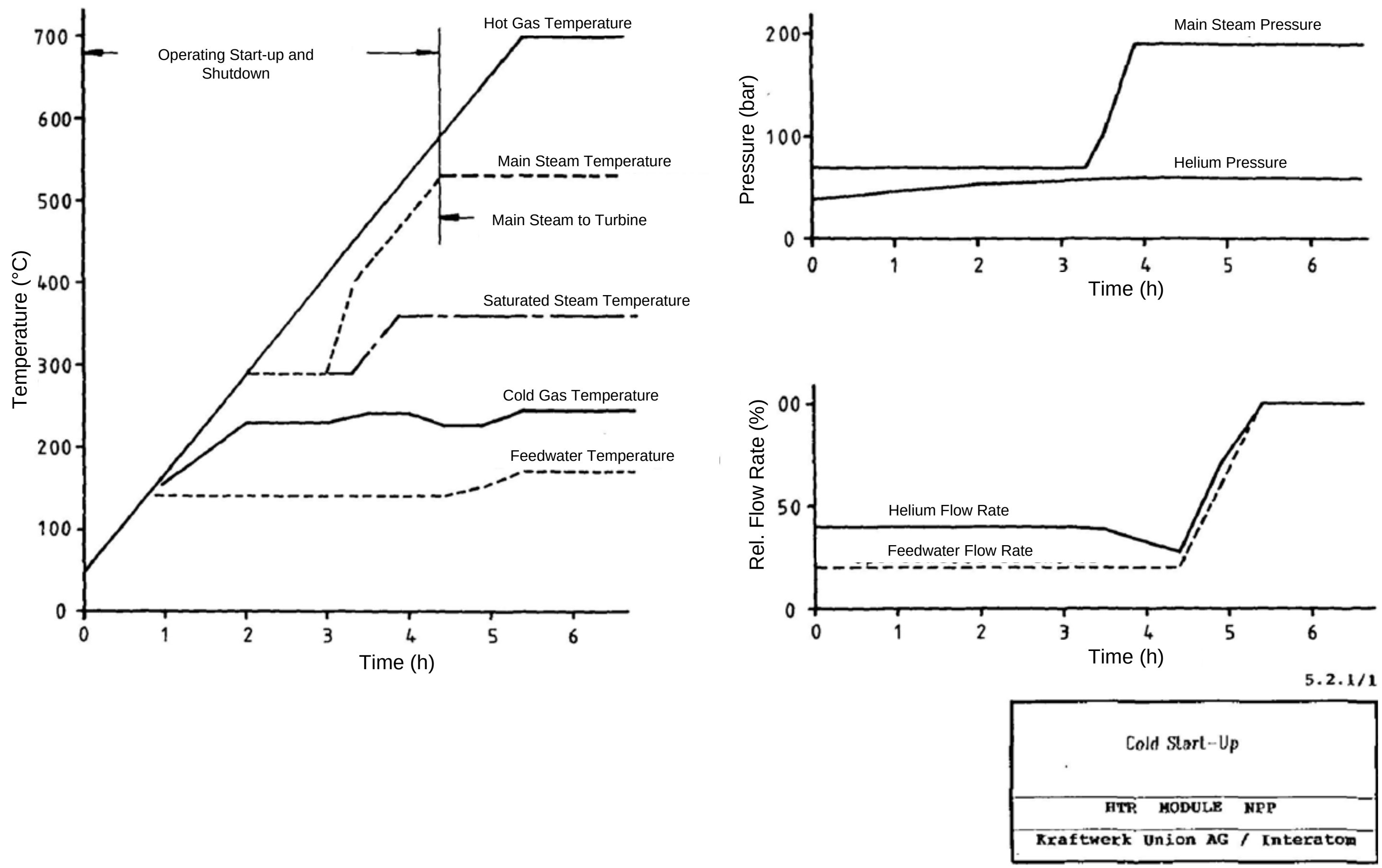


Figure 3-5: Hot Start-Up

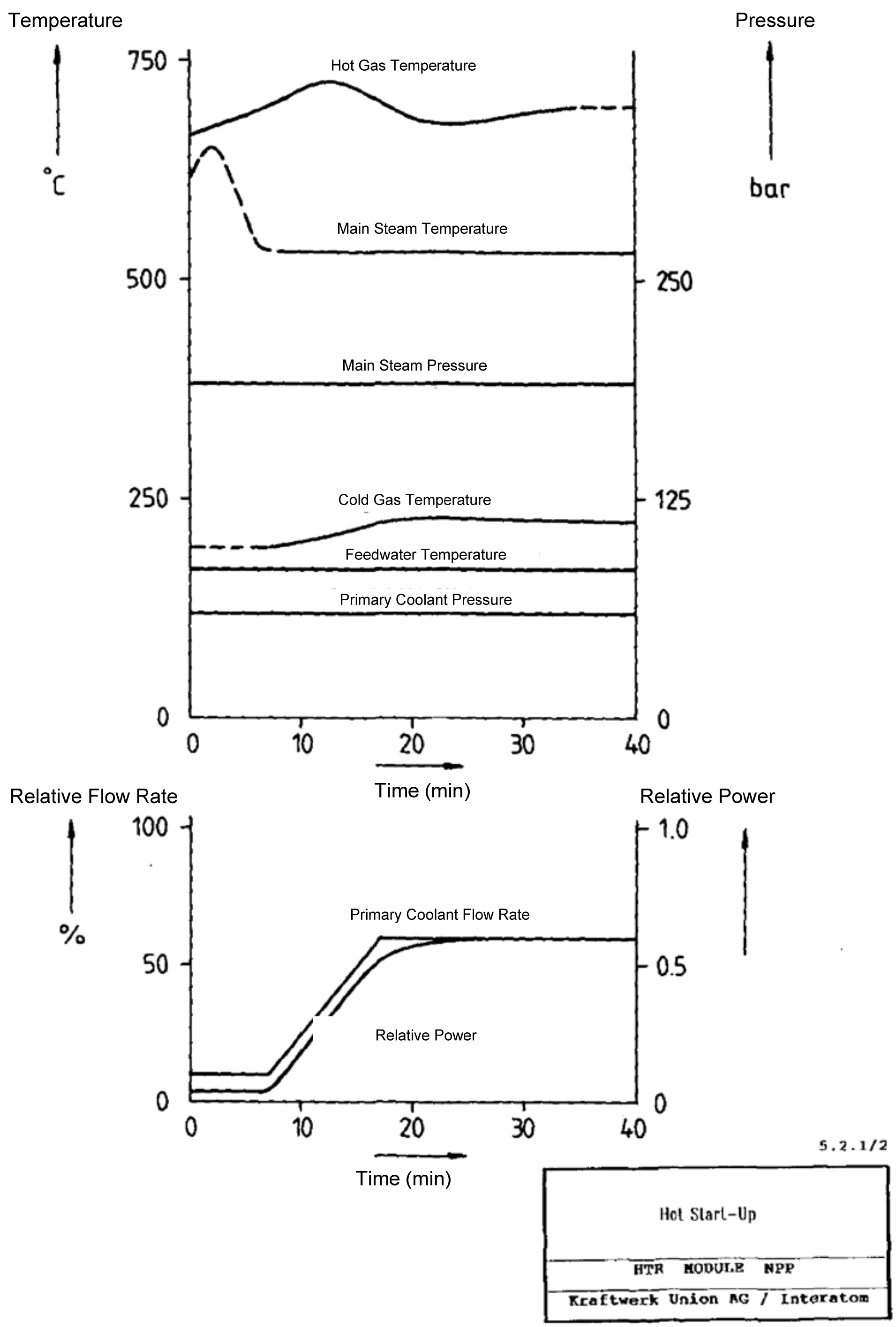


Figure 3-6: Shut-Down

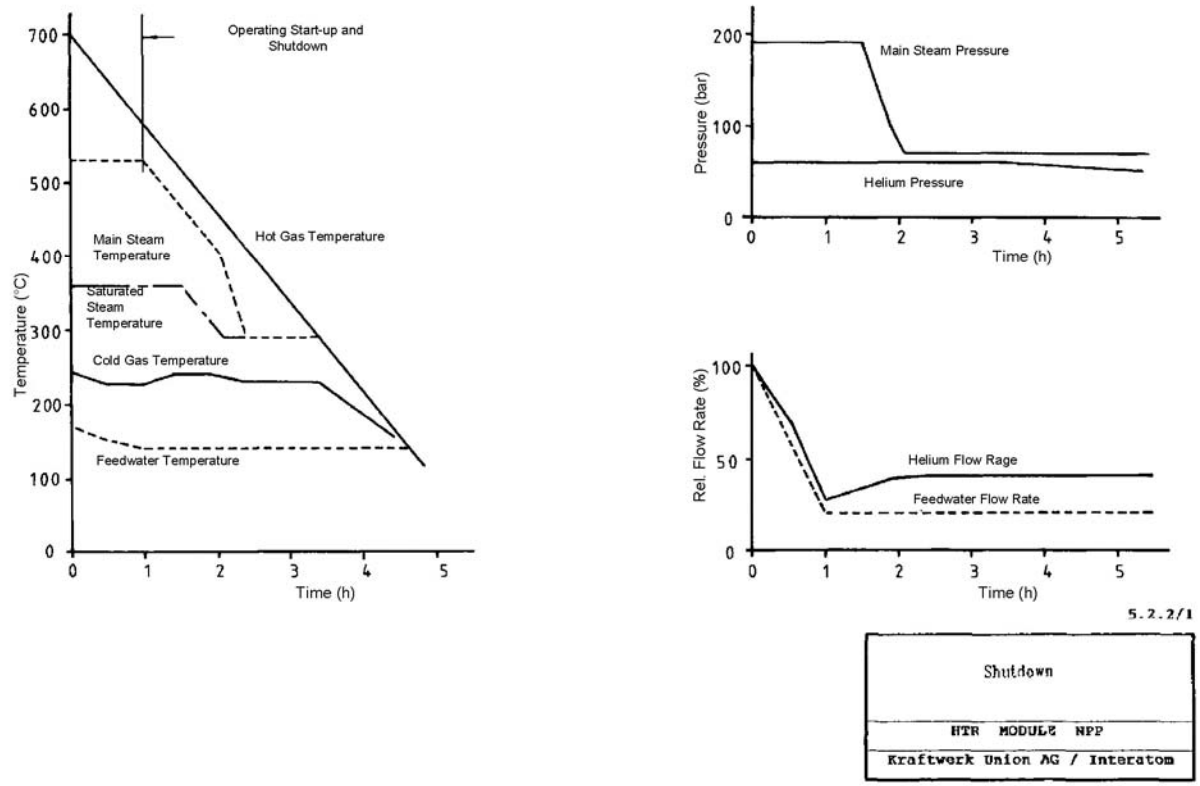


Figure 3-7: Part-Load Diagram

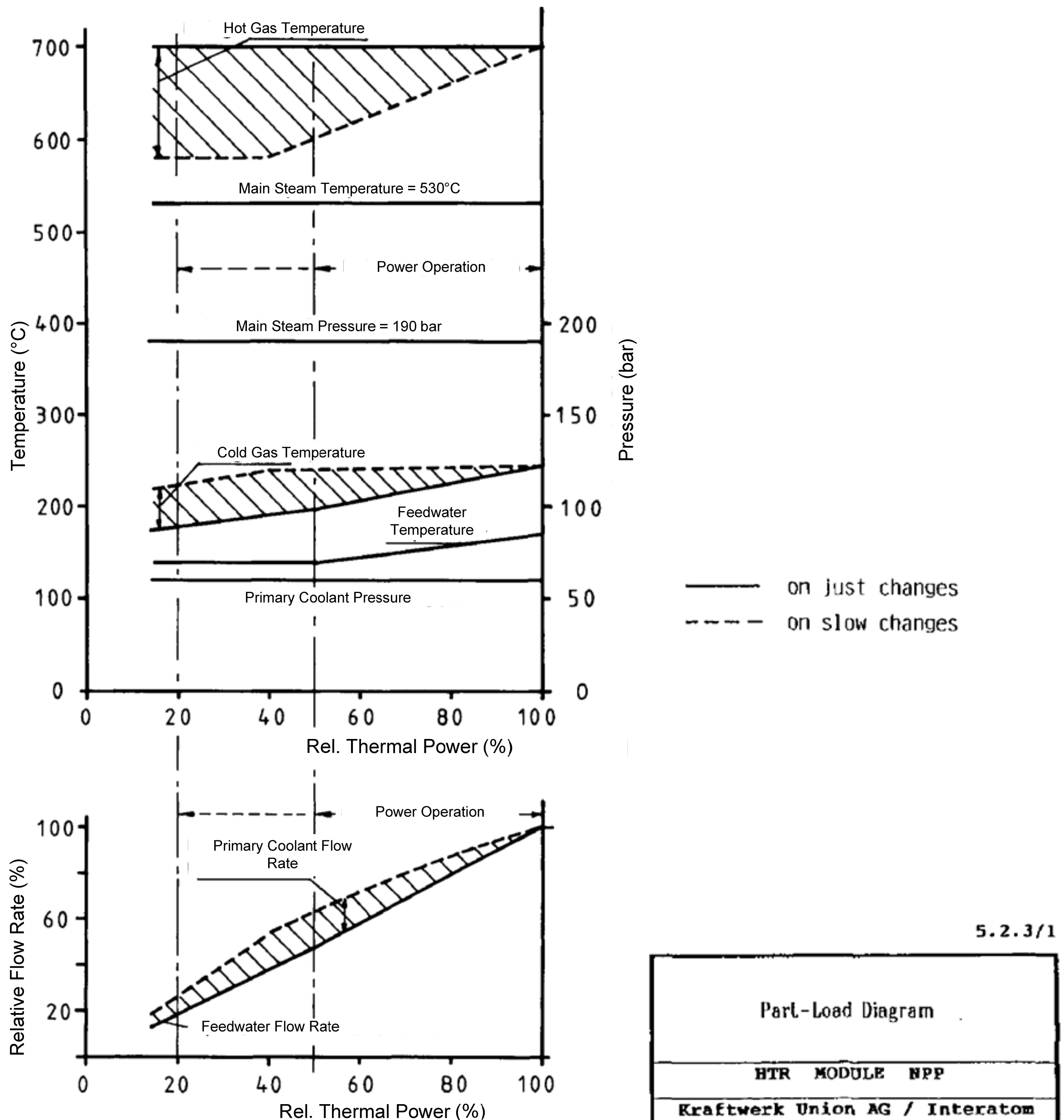
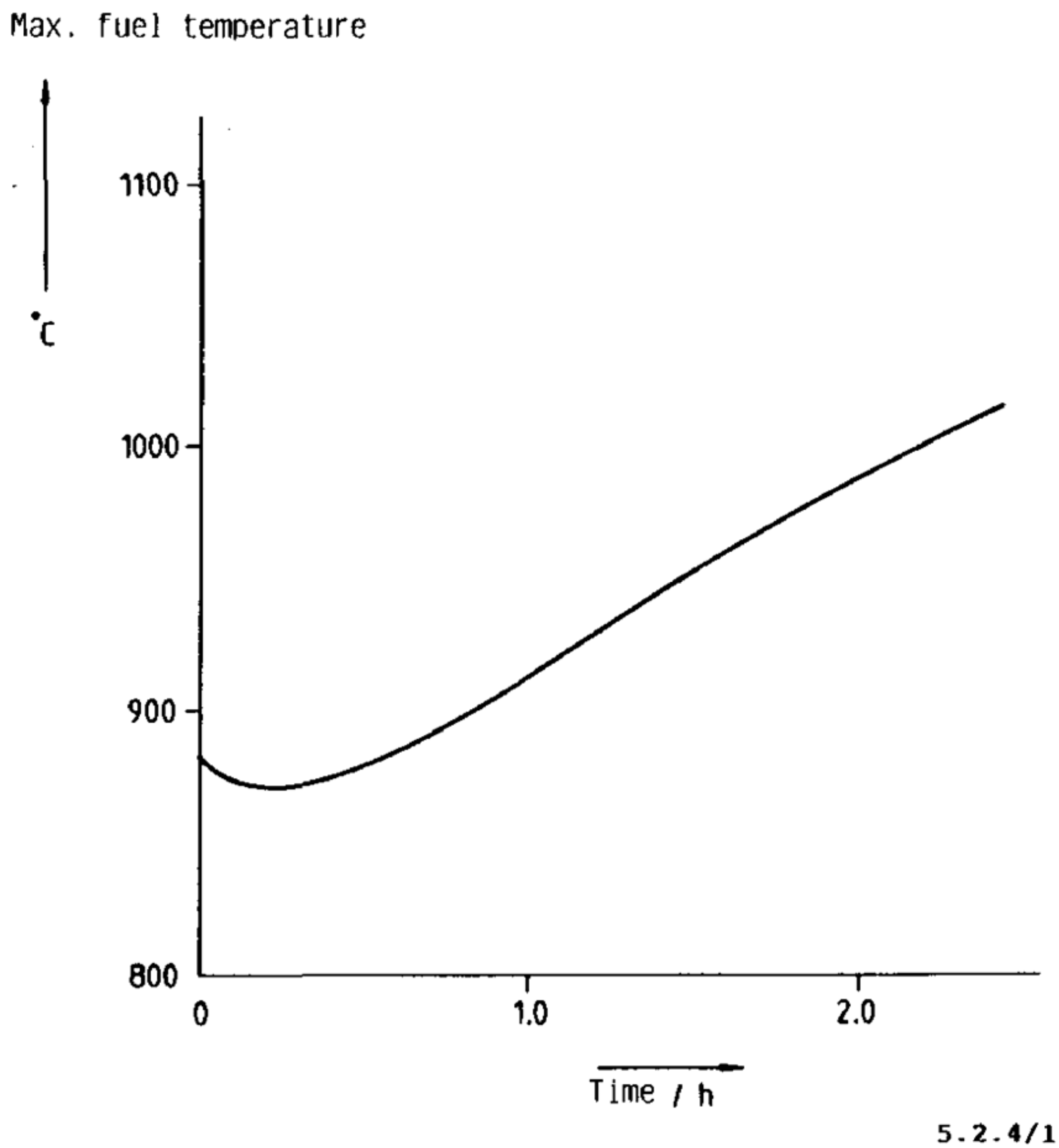


Figure 3-8: Maximum Fuel Temperature vs. Time Curve in Core after Reactor Scram without Operational Residual Heat Removal



Maximum Fuel Temperature
-Versus- Time Curve in Core after
Reactor Trip without Operational
Residual Heat Removal

HTR MODULE NPP

Kraftwerk Union AG / Interatom

Figure 3-9: Relative Helium -/Water-/Steam- Flow Rates in Steam Generator after Reactor Scram

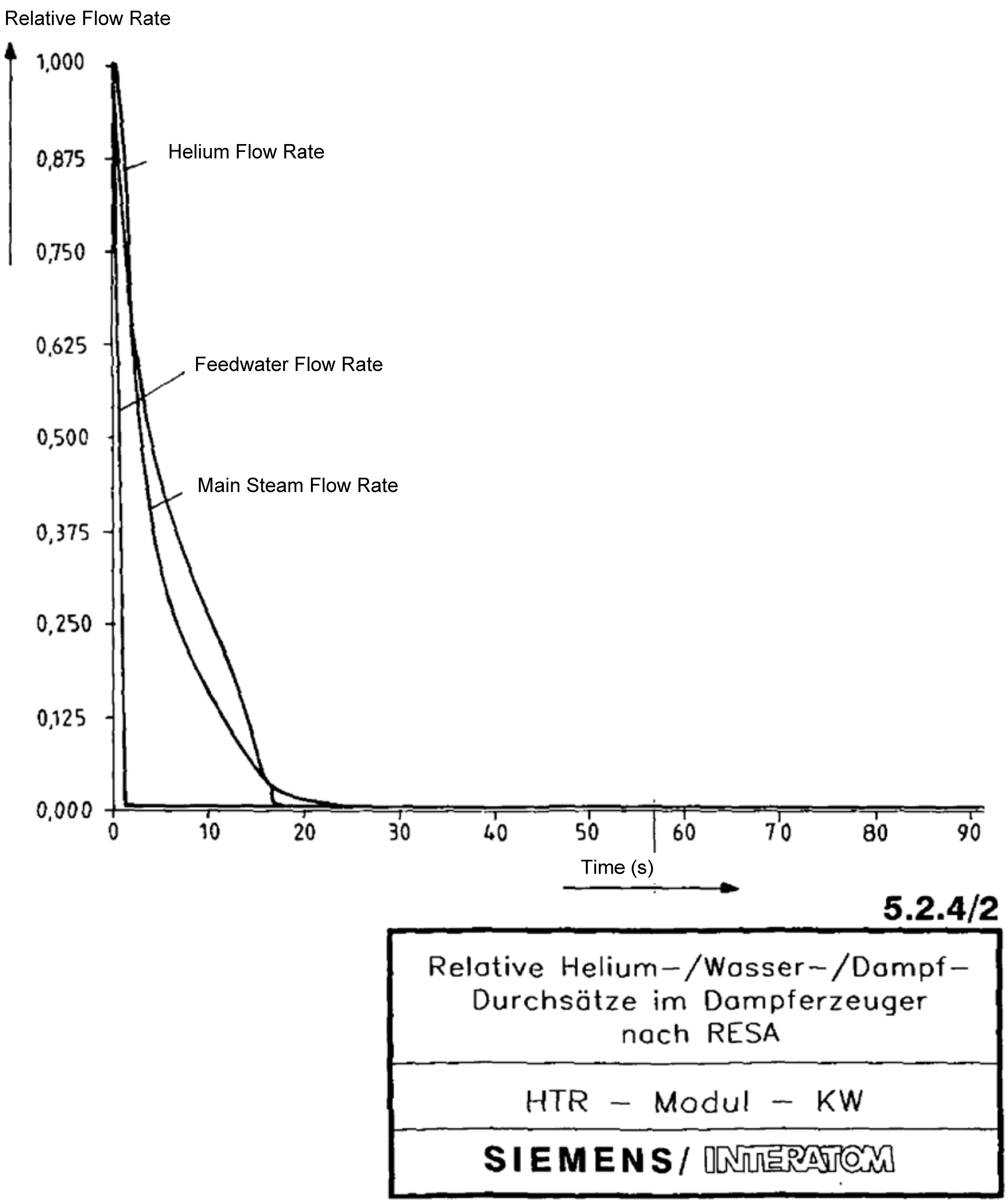


Figure 3-10: Temperature in Steam Generator after Reactor Scram

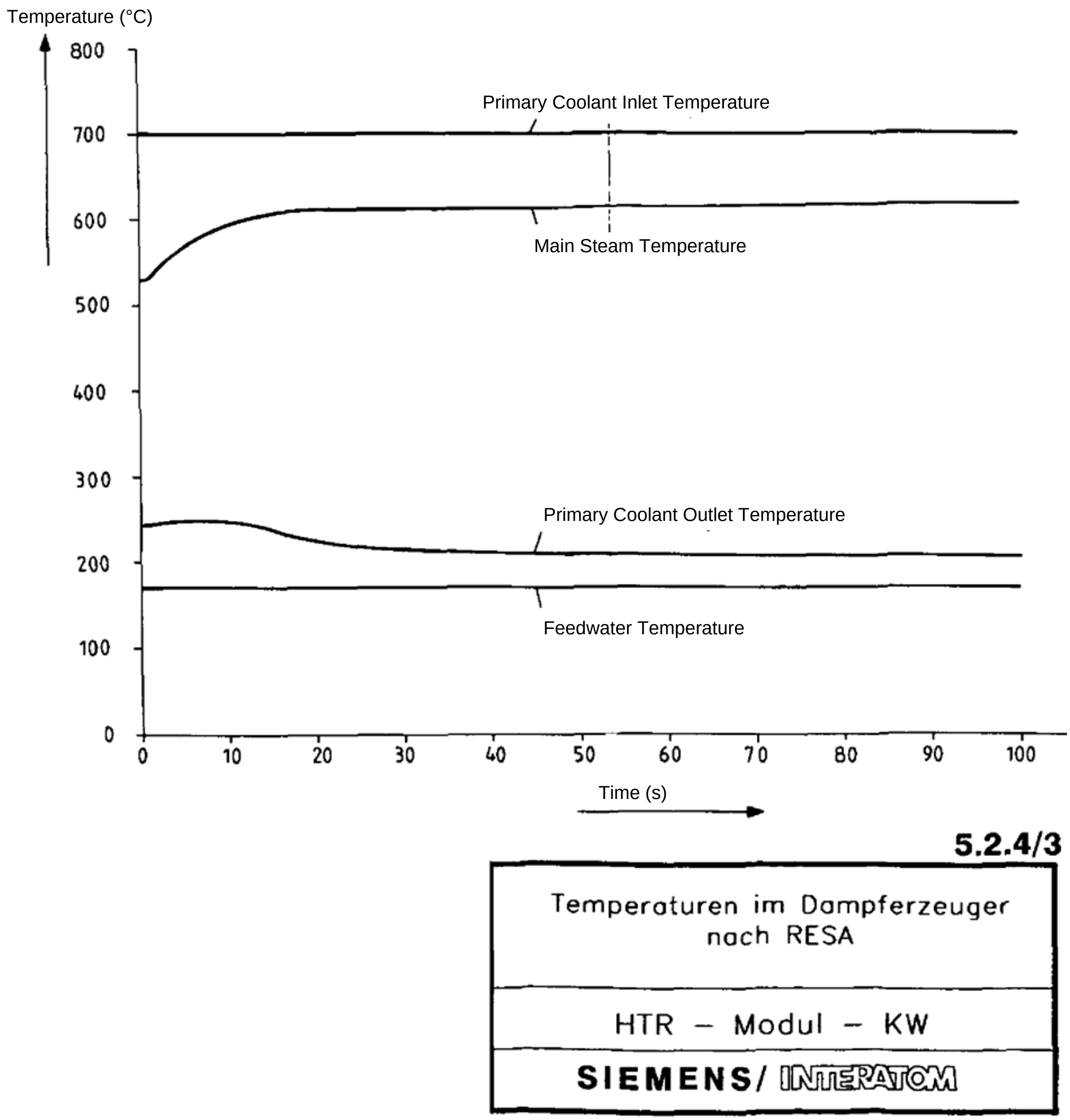


Figure 3-11: Peak Temperature vs. Time Curves

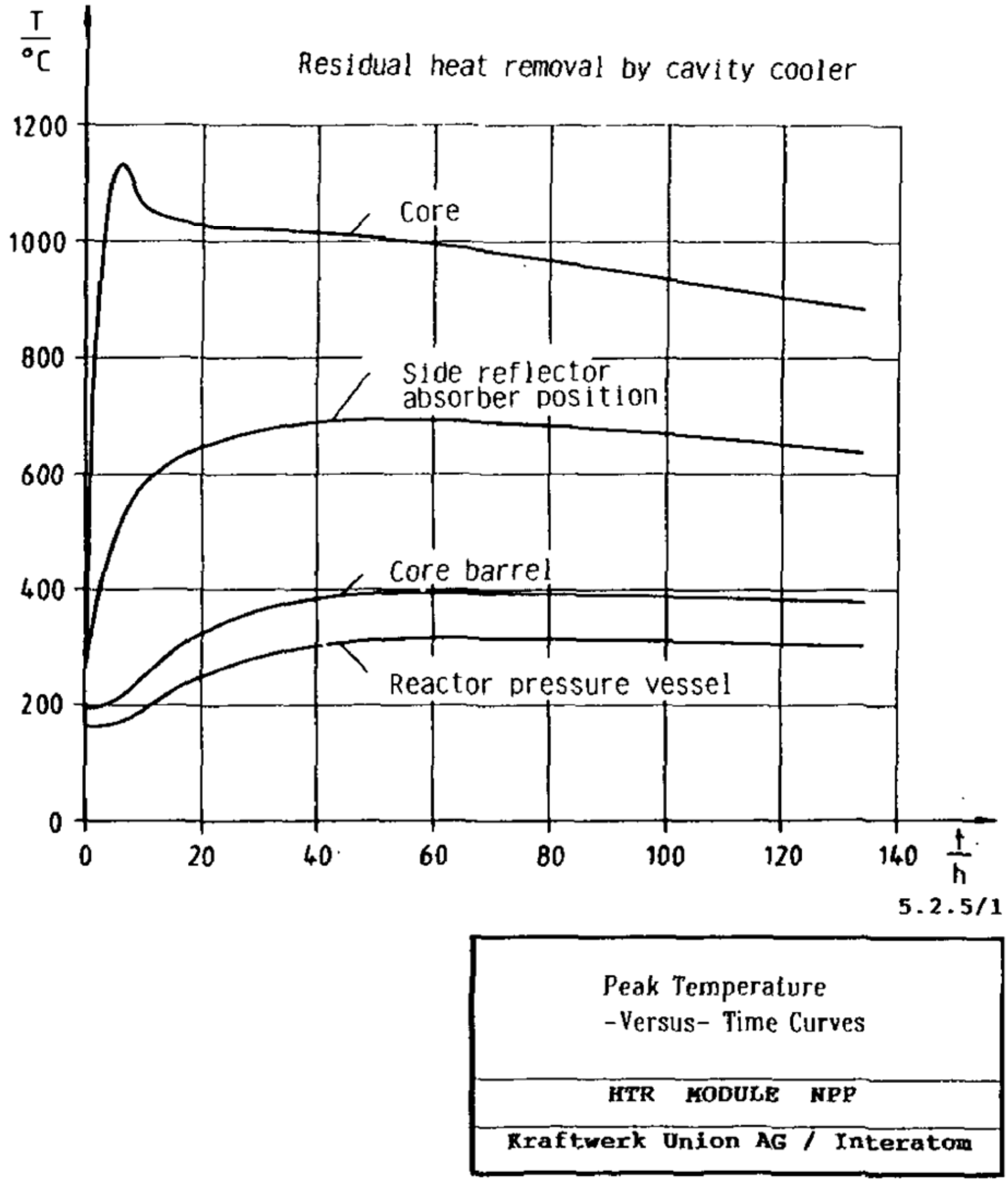


Figure 3-12: Radial Temperature Curves at Elevation of Peak Core Temperature

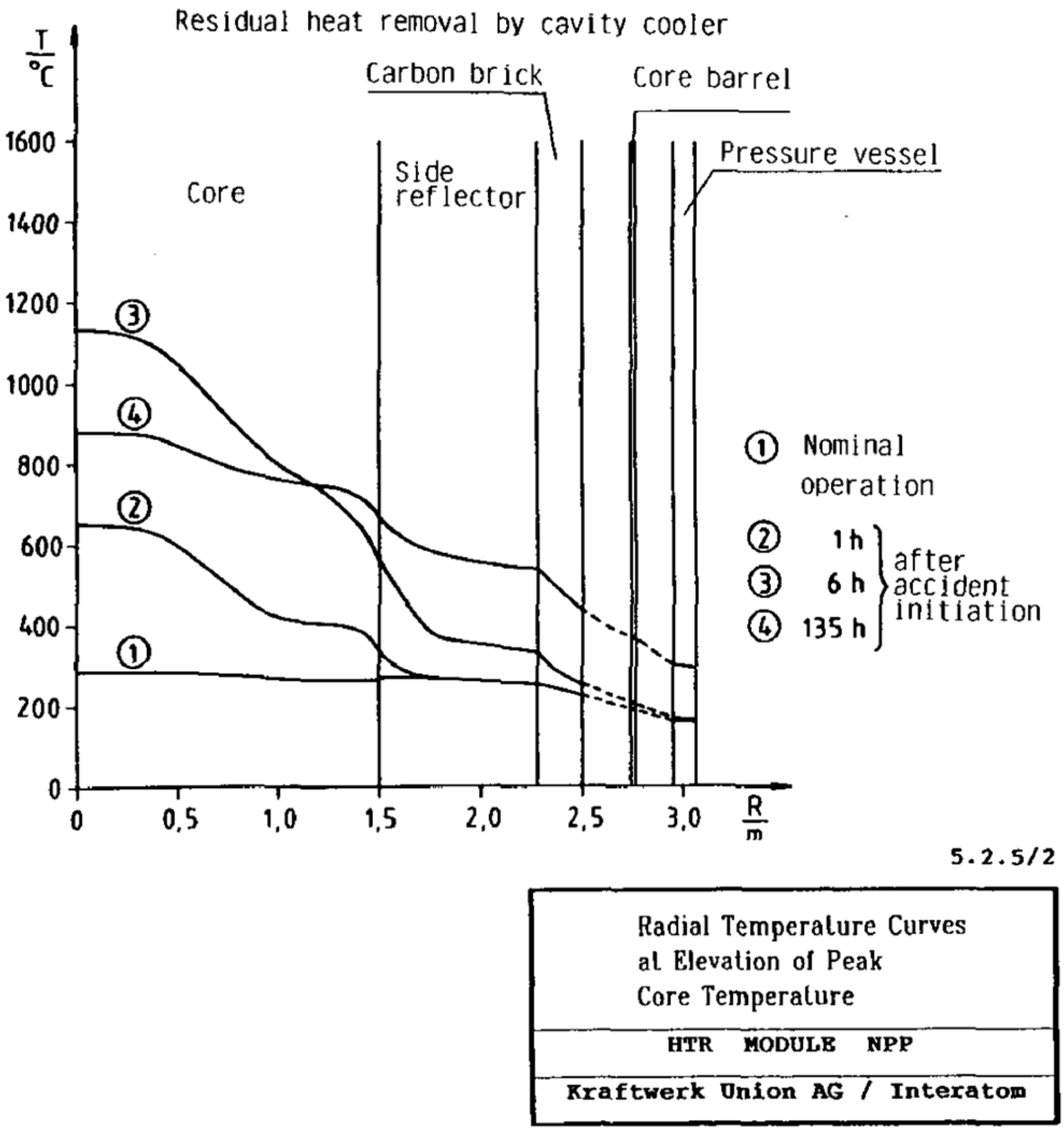
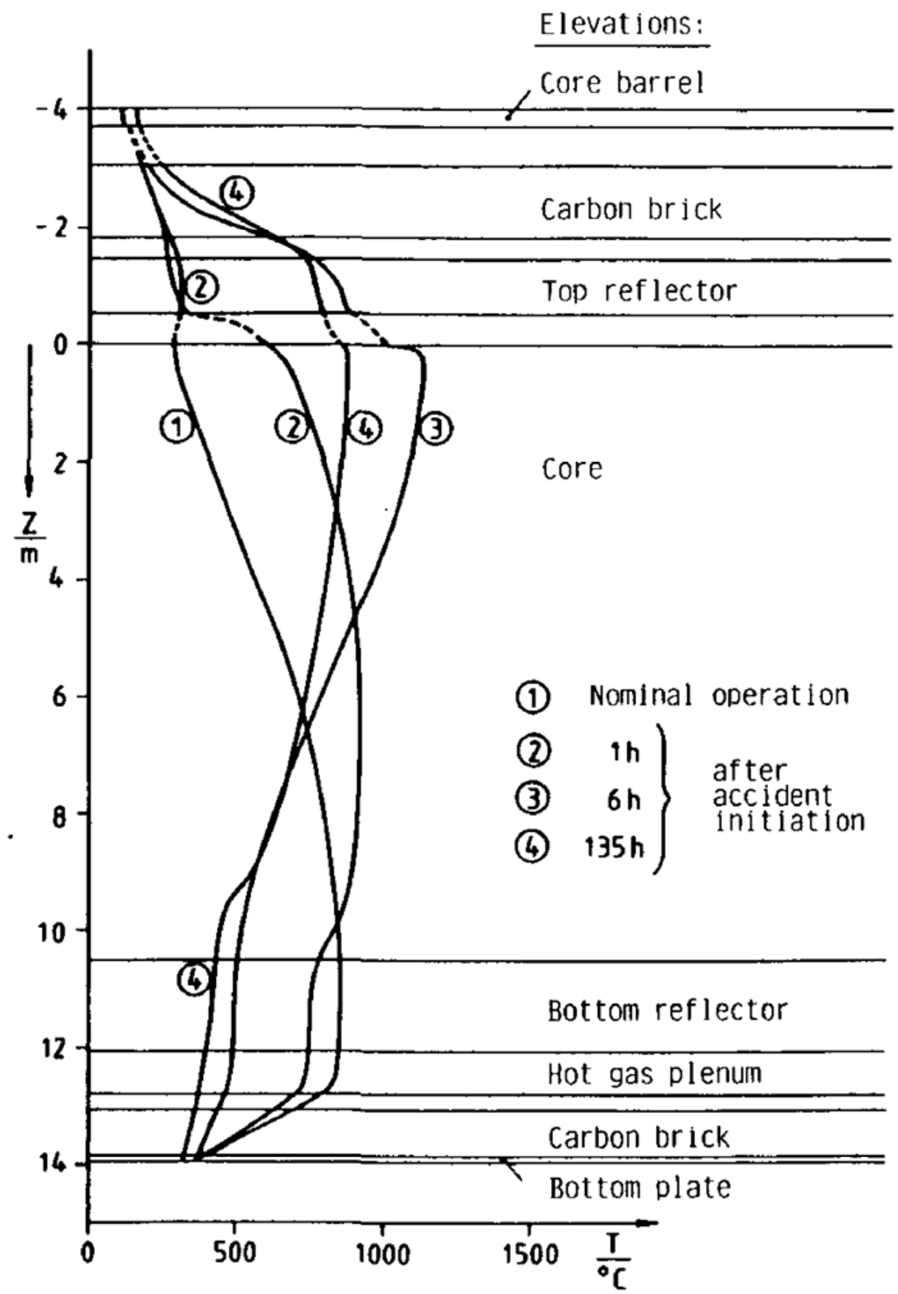


Figure 3-13: Axial Temperature Patterns Along Core Axis



Residual heat removal by cavity cooler

5.2.5/3

Axial Temperature Patterns along Core Axis
HTR MODULE NPP
Kraftwerk Union AG / Interatom

4.0 GENERAL PBR SAFETY CHARACTERISTICS

This chapter provides the key safety characteristics of the PBR technology and more specifically those that are unique to the HTR-Module reactor concept. These safety characteristics are the basis for the safety assessment for the design of the plant.

4.1 Barriers Against Release of Radioactivity

The PBR technology in general and HTR-Module in particular uses fuel elements in which the uranium fuel is distributed among many small fuel particles. Each particle is coated with two high-density layers of pyrocarbon and one layer of silicon carbide. The particles are embedded in a carbon matrix with an un-fueled outer zone. Refer to Figure 4-1.

The primary characteristic safety feature of the HTR-Module is that the majority of the radioactive substances produced during nuclear fission are confined within the fuel particle during all operating and accident conditions in such a way that there can be no significant release of radioactivity from these fuel particles. This safe confinement of radioactivity is assured by the design of the fuel particle coatings, and a temperature qualification limit of approximately 1600 °C, the maximum anticipated fuel temperature under accident conditions.

The design of the coated particle fuel also provides key safety characteristics. In particular, for temperatures up to at least 1600 °C, the silicon carbide layer keeps its integrity so that no radiologically significant quantities of gaseous or metallic fission products are released from intact particles.

For design purposes a fraction of coated fuel particles are assumed to have manufacturing or in-service radiation defects exposed by accident conditions. The HTR-Module design basis assumes a fraction of the defective particles could be at the maximum accident temperature of approximately 1600 °C. An average of about two defective particles is assumed to exist for each fuel element. This accounts for the manufacturing defects, the burnup induced failures and the core maximum temperature induced failure distribution in normal and accident conditions.

Some of the radioactive substances released from these defective particles are retained within the fuel element matrix. The portion that is not retained goes into the primary coolant and is distributed in the primary system.

The gas-borne activity in the primary system decreases as a result of radioactive decay, separation in the helium purification system and deposition on the surfaces of the primary system. The primary system pressure boundary thus forms the next barrier against the release of radioactive substances. The components of the HTR-Module pressure vessel unit are designed in such a way that through-wall cracks can be ruled out. Because of the quality assurance measures taken and periodic in service inspections made, the un-isolable breaks in the connecting piping are made highly improbable.

In the event of an un-isolatable break, that is nevertheless assumed, the radioactive inventory in the helium primary circuit is very small. This includes a portion of the activity deposited on the surfaces of the primary system that could be lifted off and released into the reactor building. Therefore, the HTR-Module design did not place any leak tightness requirements on the reactor building to comply with accident dose limits imposed by the German regulations (German Radiological Protection Ordinance). In order to assess acceptance in the U.S.A., the present study also calculated (see Chapter 7), using the same source term, the accident dose in accordance with the U.S. NRC methodology. This was possible mainly because of the high radioactive retention capacity of the fuel particles. However, to further minimize the impact on the environment, the reactor building is provided with a sub-atmospheric pressure control system and a pressure relief system with filtered release capability.

Figure 4-1 provides a pictorial representation of particle fuel fission product generation, transport paths, retention schemes and physical barriers for a pebble bed reactor such as the HTR-Module.

4.2 HTR-Module Plant Inherent Safety Characteristics

The engineered configuration and nuclear design of the HTR-Module is such that even in the event of assumed failures of all active shut-down and residual heat removal systems, the peak fuel temperature stabilizes at approximately 1600°C. Sufficient margin exists between the peak fuel temperature and the temperature at which addition fuel failure due to temperature is conservatively assumed to occur and that the core temperature coefficient has enough negative reactivity to shut down the reactor and stop further temperature increase. This safety margin is possible because a nominal temperature differential of approximately 750K exists between the maximum allowable fuel temperature and the maximum operating temperature of the fuel elements in the HTR-Module reactor cores. This is also true even in the presence of accident induced excess reactivity, e.g., water ingress.

Furthermore, residual core heat can be dissipated from the reactor to the surrounding components, structures and surfaces solely through natural physical processes. These include thermal conduction, radiation, and convection. The design choices of low mean power density in the reactor core, the geometric design of the core and the surrounding core internals, and the use of suitable materials make the passive core decay heat removal characteristic possible.

The HTR-Module also includes active operational systems for residual heat removal. These active systems are normally used to limit loadings on the passive heat removal components and structures. In addition, design margins are selected such that if active systems are inoperable for several hours the allowable design limits of for the components systems and structures (SSCs) are not exceeded and the passive engineered safety features will respond in ample time.

The HTR-Module primary systems, core design and material selections reduce the safety requirements on the water/steam cycle and the start-up and shut-down systems. Therefore, these systems are designed and operated as purely conventional plant systems.

4.3 Technical Design Features Important to Safety

4.3.1 Reactor

Table 4-1 provides the reactor core nominal data for the HTR-Module Equilibrium Core.

4.3.1.1 Fuel Elements

The HTR-Module fuel elements are spherical and have a diameter of 6 cm. The 5 cm-diameter inner zone of the sphere contains 7 g of low enrichment uranium in the form of coated fuel particles. The outer zone of the sphere is unfueled providing protection for the fuel particles during pebble movement.

The fuel particles for the HTR-Module are spherical kernels of low enriched uranium oxide (UO2) with a diameter of 0.5 mm, surrounded by a buffer of porous carbon, two pyrolytically deposited layers of carbon (IPy and OPy) and a layer of silicon carbide (SiC). This forms the TRISO coated fuel particle.

These TRISO coated fuel particles are evenly distributed in a carbon matrix (natural graphite, electro-graphite, artificial resin binder) with a particle-free edge zone to form a spherical fuel element (i.e., pebble).

The fuel elements feature the following characteristics:

- The retention of fission products within the coated fuel particles in order to fulfill the radiological release requirements for a predetermined plant design. This is achieved with low levels of:
  - Uranium contamination of the carbon matrix
  - Particle failure due to manufacturing defects
  - Irradiation-induced particle failure

  - Accident-induced particle failure.
  - Figure 4-1 shows the radioactivity release barriers and paths in a fuel element.
  - Figure 4-2 shows the limiting failure rate-versus-temperature curve of UO2 particle fuel fraction not surrounded by intact silicon carbide layers. The expected value of the sum of all such fractions lies below this limiting curve, which is used as the basis for the design.
- Dimensional stability under irradiation to assure transportability of the fuel elements in the fuel handling equipment,
- Mechanical strength to maintain fuel element integrity to assure that they can be transported and handled and to avoid fission product release due to mechanical damage to the coated fuel particles,
- Transportability even after corrosion due to a depressurization accident (air ingress) or rupture of a steam generator tube (water ingress).

The target burnup of the fuel elements is 80,000 MWd/MTU. Reprocessing of the spent fuel is not planned.

4.3.1.2 Reactor Core

The HTR-Module equilibrium core consists of approximately 360,000 spherical fuel elements in a loose pebble bed having a diameter of approximately 3 m and an average height of approximately 9.4 m and is cooled by helium. Mean power density during normal power operation of the core is limited to 3 MW/m^3 and the mean core outlet temperature to 700°C. In this single-zone core, axial power density distribution must be made sufficiently uniform by multiple recycling.

Under this multi-pass refueling scheme, each fuel element passes through the equilibrium core an average of approximately 15 times during its useful life.

Under all operational and accident conditions, core residual heat can be removed solely by thermal conduction, thermal radiation and natural convection to the surface coolers outside the reactor pressure vessel.

The primary heat transport system can be used as the active residual heat transport system. However, even without active residual heat removal from the core, the maximum fuel temperature of approximately 1600°C is not exceeded.

The HTR-Module core and its geometry are so designed that the reactor can be shut down by the insertion of neutron absorbers in the reflector columns surrounding the core.

Because of the core design, the total temperature reactivity coefficient is sufficiently negative such that the inadvertent withdrawal of all reflector rods is mitigated solely by primary gas circulator trip; again, the allowable fuel temperature of approximately 1600°C is not exceeded.

Furthermore, the core design is such that the reactivity effect caused by design basis accident-induced water ingress is less than that of an inadvertent withdrawal of all the reflector rods.

The core height is selected such that un-damped axial xenon oscillations are not possible thus are ruled out by design.

4.3.1.3 Core Internals

4.3.1.3.1 Ceramic Core Internals

The HTR-Module ceramic core structure, consisting mainly of side, bottom and top reflectors, surrounds the pebble bed core and also has the task of reflecting neutrons leaving the core back into the pebble bed.

The side reflector is constructed of 24 individual carbon brick and graphite stacks constituting a 24-sided core cavity with a width across the flats of approximately 3000 mm inside and approximately 5000 mm outside. For thermal insulation, the outer portion is made of approximately 250 mm thick carbon bricks. The 24 stacks are supported by metal rings and consist of individual blocks of graphite and carbon brick.

The key characteristics of the core ceramic structures are as follows:

- In order to provide unimpeded flow of fuel elements near the side reflector, by an ordered-structure in the pebble bed, the inward facing surfaces of the individual bricks are corrugated.
- The top reflector is made of several layers, each of which consists of 24 cantilevered segments that correspond to the stack segments of the side reflector and are joined to them with dowels. A plug in line with the core axis closes the gap in the upper layers.
- The cold gas plenum is located between the lower two graphite layers and the upper two carbon brick layers of the top reflector. The lower carbon brick layer is borated.
- The carbon bricks are partially or fully borated for radiation shielding purposes.
- The core bottom is constructed of several segmented layers. The lower two layers are of carbon brick, the upper layer of graphite. The segments of the individual bottom layers are arranged coincident with the stack segments, except for the lowermost layer. This results in a defined pattern of movement of the core bottom at azimuthally temperature differences. The segments of the lowermost carbon brick layer are offset from the stack segments to avoid continuous vertical gaps from the hot gas plenum to the metal bottom plate. The bottom layers are fastened to each other and to the metal bottom plate with dowels. The top of the core bottom stacks is formed by 30° beveled top bricks. The top bricks are held in place on the segments by positive locking joints.
- Parts of the core bottom are borated to avoid unacceptable activation of the reactor pressure vessel near the gas duct pressure vessel, the hot gas duct and the steam generator. The two lower layers of the bottom are made of carbon bricks for thermal insulation.
- The ceramic portion of the fuel discharge tube is constructed of several courses of graphite centered on and supported by the metal bottom plate. The inside diameter is approximately 500 mm, thus positively preventing plugging by the fuel elements. The transition from core bottom to fuel discharge tube is formed by the inner top bricks of the core bottom. Beneath the bottom plate, the fuel discharge tube material is metallic. The lower portion of the ceramic discharge tube is borated.
- The circular cross section of the hot gas outlet passage spans two individual stacks of the side reflector.
- Sealing in the flange region between hot gas and cold gas passages is flexible because of relative movement due to thermal expansion.

The core gas flow path starts with cold gas entering the core barrel through columns in the core barrel bottom plate and is conveyed to the cold gas plenum above the core through columns in the side reflector. In order to minimize the parasitic heat loss from hot to cold gas and thus keeping the radial outlet temperature profile as uniform as possible, graphite thermal insulation sleeves are inserted into the cold gas columns from the core barrel bottom plate to approximately the level of the lower third of the core. The graphite sleeves was also used to reduce the reverse flow (due to natural convection) during a hot standby where the blower damper does not close.

After the primary coolant has passed through the core from top to bottom and has been heated, it enters the coolant flow holes in the bottom reflector and collects in the annular hot gas plenum from which the primary coolant exits the core and enters the hot gas duct.

4.3.1.3.2 Metallic Core Internals

The cylindrical part of the metallic core barrel is constructed of individual courses. At various levels in the core region and above the top reflector, azimuthally guides prevent deflection of the individual stacks of the side reflector.

As part of the bottom structure the bottom plate contains openings in line with the cold gas columns in the side reflector for the passage of cold gas. The entire ceramic core structure is joined to this bottom plate with dowels.

The individual stacks of the side reflector are supported at top and bottom by spacers that are rigidly fixed to core barrel.

The core barrel is flanged at the top to form a leak-tight joint with the gray cast iron top thermal shield that, in conjunction with the upper radiation shield, thus allows access to the area above the core barrel when the reactor is shut down and the reactor pressure vessel is open. The core barrel is radially supported by guides in the reactor pressure vessel above the top thermal shield. This does not restrain relative changes in length between core barrel and reactor pressure vessel.

The core barrel is vertically supported in the reactor pressure vessel above the gas duct pressure vessel.

The space between the core barrel and the reactor pressure vessel is filled with stagnant helium, the pressure of which is matched to that of the primary system by a pressure equalizing system.

The metal core barrel and ceramic core internals are designed to last the service life of the reactor. Spot visual inspections and repairs to reactor pressure vessel internals are possible. For this purpose, the pressure equalizing system is closed after the pressure has been lowered, so that the space above the core barrel is accessible. In so doing, the reactor core is kept at low temperatures through heat removal by the main heat transfer system.

4.3.1.4 Control and Shut-Down Systems

Two independent and diverse reactor shut-down systems, the reflector rods and the small absorber ball shut-down systems are provided. The absorber elements are inserted into the side reflector for control and shut down of the reactor.

The shut-down systems are designed and arranged in such ways that, on demand, the absorber rods drop into their most effective position solely under the force of gravity.

Only the first shut-down system is controlled by the reactor protection system, while the second shut-down system is manually actuated, when needed.

One of the two shut-down systems is also used for reactor power control. In particular, the rod position of the first shut-down system is selected for the equilibrium core in such a way that load changes within a range of 50% to 100% of nominal power are possible at any time.

Reflector Rods

There are six reflector rods, each rod consist of several absorber elements held together by articulated joints. The absorber material in the form of sintered B_4C rings is located between two concentric tubes in each element. The absorber is cooled inside and outside by a stream of cold gas.

In order to integrate the rod drive mechanisms into the reactor pressure vessel, the absorber length is limited to approximately 4.8 m. A link chain driven by an electric motor through a planetary gearbox serves as the lifting element between the drive mechanism and the absorber proper. On reactor trip, the motor's power supply is cut off, which results in the reflector rod falling under gravity but damped to its lowest position (1 m below core center).

Because the rod drive mechanisms are fully integrated into the pressure vessel unit, depressurization accidents via the drive mechanism housing and hence rod ejection due to fluid flow forces can be ruled out.

Each rod is equipped with an analog position indicator that measures the position of the rod over its entire positioning range and with binary position indicators for the upper and lower limit positions.

Small Absorber Ball Shut-Down Units

There are eighteen small absorber ball shut-down units. They are distributed as evenly as possible in slotted holes in the side reflector around the perimeter of the core. The shut-down elements are graphite balls with a B_4C content of approximately 10% and a diameter of approximately 10 mm. The small ball shut-down elements are stored above the side reflector on the top thermal shield and fall under gravity into reflector columns on demand.

Each of the eighteen storage vessels has a closure that allows the discharge without causing ball breakage. The power supply to the closure solenoid is cut off on demand, causing the vessel closure to open by gravity and the small ball shut-down elements to fall freely into the slotted holes in the side reflector. Both limit positions of the closure are monitored.

A pneumatic suction system is used to return the shut-down elements from the reflector columns in controlled quantities to the top storage vessels. Measuring devices constantly monitor the fill level of the storage vessels, including the “vessel full” and “vessel empty” states.

The shut-down elements are drawn from vessels integrated into the metallic core support structure. These vessels are filled with absorber balls at all times. The transport fluid is cold gas that is extracted from the primary system beneath the metallic core support structure and is conveyed to a common carrier gas circulator outside the primary system. Sets of control valves for carrier and bypass gas supply serving groups of six small ball shut-down units are arranged in one valve bank each, in the reactor pressure vessel. Each of these valve banks also includes a manual isolation valve and a valve actuated by the reactor protection system (i.e., primary system isolation valve).

Servicing of the reflector rod drive mechanisms and components of the small ball shut-down units (e.g., vessel closure) is possible when the reactor is depressurized.

Shut-down by Shut-down Systems

Equilibrium Core

The first shut-down system (reflector rods) has sufficient negative reactivity that renders the reactor subcritical hot from all accident-induced conditions for a sufficient length of time, even if failure of the highest-worth reflector rod is postulated.

The second shut-down system (small ball shut-down system) is able to render the reactor subcritical down to the lowest possible operating temperature (50°C) from all operating states and for any length of time.

Both shut-down systems together render the reactor subcritical for any length of time down to the lowest possible operating temperature (50°C) from all operating states and accident conditions, even postulating the failure of the highest-worth small ball shut-down unit.

First Core

The requirements and functions of the control and shut-down systems in the first core are not fundamentally different from those of the equilibrium core. However, since the temperature coefficient is especially negative during the first months at full load, shut-down margins for the first core are lower than for the equilibrium core. Under accident conditions, on loss of function of one shut-down system and assuming most adverse uncertainties in the reactivity balance calculations, it would consequently be possible for the reactor to return to criticality at temperatures below 100°C. This condition does not present a problem from the safety standpoint, since the core can only generate an output corresponding to the small power loss and stabilizes itself at the low temperature. After the accident reactivity is removed or repairs to the shut-down system that has failed have been performed, the reactor can always be cooled down to a subcritical cold state.

Shut-Down by Cutting Off Primary Coolant Flow

Another way of shutting down the reactor is to cut off the primary coolant flow; this leads to a slight increase in the mean core temperature and renders the reactor subcritical because of the negative temperature coefficient for reactivity. As a rule, the primary coolant flow is stopped by shutting down the primary gas circulator following a reactor trip. In addition, the circulator damper is closed by an operational automatic control.

4.3.1.5 Core Instrumentation

The neutron flux measuring instrument is an ex-core instrumentation system in which probes in probe guide tubes are distributed radially and axially in the concrete structure of the reactor cavity in such a way as to:

- Monitor integral core power,
- Detect the general structure of axial power distribution, and
- Detect azimuthal asymmetry.

The ex-core flux measurement system consists of the following instrument channels:

- One instrument channel group for the start-up range (source range),
- One instrument channel group for the intermediate range,
- One instrument channel group for the power range,
- One instrument channel for the intermediate range with indicator at the remote shut-down station.

Altogether, the detectors monitor core power from the subcritical cold state up to 200% of nominal power, and also macroscopic power distribution. For this purpose, a neutron flux density range of about 11 decades has to be detected at the measurement point.

The source range and the logarithmic intermediate range together cover the total range up to double nominal power. The source range covers approximately the lower 6 decades, the logarithmic intermediate range approximately the upper 6 decades.

The linear power range detects the upper two decades of the neutron flux density. The constant values derived from the power range make it possible to detect incorrect positioning of the reflector rods or the inadvertent insertion of small ball shut-down elements, as well as to determine core power.

In order to assure an adequate background for the source range detectors, a neutron source is provided for initial reactor start-up. It is located in a column branching off a reflector rod column in the side reflector in the upper half of the core. No additional secondary source is provided since so many photo-neutrons are produced by fission product decay in the reactor core after a few months at full load that the plant can be safely started even without an external neutron source.

Thermocouples are installed at several locations in the ceramic internals of the core area as part of the operational instrumentation. Measurements from them serve to validate the analyses of temperature distributions in this area in various operating states and to quantify the relationships between important operating variables in the first years of plant operation. Because of the limited period of use, non-replaceable thermocouples are installed.

4.3.2 Primary Circuit Components

Refer to Figure 4-3 for primary helium flow path and Figure 3-1 for the schematic flow diagram of the primary circuit components.

Refer to Table 4-2 for principal data on the primary circuit components.

The HTR-Module power plant consists of two reactors facilities having a nominal thermal power of 200 MWt each and sharing some of the same systems.

Each reactor consists of:

- The reactor pressure vessel with core, core internals, shut-down systems, and systems for feeding and discharging fuel elements,
- The gas duct pressure vessel with hot gas duct,
- The steam generator with tube bundle and primary gas circulator.

The reactor, the heat source of the plant, is located at a higher elevation than the steam generator in this system. Natural circulation in the primary system is restricted in such a way that unacceptable temperatures cannot occur in the metallic portions of the system on loss of primary circulation.

Under all accident conditions, the reactor pressure vessel, including core internals and shut-down systems, and the concrete structure of the reactor cavity can be protected from unacceptable temperatures by the cavity cooler alone.

The primary system pressure vessels include:

- The reactor pressure vessel, including;
  - nozzles, closure head
  - fuel discharge tube with forged failed fuel separator block
  - fuel feed connection tube with valve bank
  - small absorber ball shut-down unit connection tubes with valve banks
  - electrical and I & C penetrations
- The steam generator pressure vessel, consisting of steam generator pressure vessel portion and circulator pressure vessel portion, including;
  - nozzles, cover
  - main steam and feedwater nozzles with connecting nozzles to secondary side systems
  - electrical and I &C penetrations
- The gas duct pressure vessel.

Because of the arrangement of the gas flow channels, hot gas does not come into contact with the pressure vessel unit at any point. It is thus possible to use pressure vessel technology proven in light water reactors.

The same applies to the feedwater nozzle. Only the main steam nozzle is exposed to high temperatures and is thus constructed in accordance with the technology used in conventional power plants together with extra requirements specific to high temperature reactor.

The pressure vessel unit is equipped with man-ways at suitable locations.

The primary gas envelope consists of:

- Pressure vessel unit, excluding components of the main steam and feedwater nozzles that carry only secondary coolant,
- Steam generator tube bundle with tube-sheets on the main steam and feedwater side,
- Primary-side connecting lines, including the valves actuated by the reactor protection system.

As a rule, joints in the pressure vessel unit and all adjoining piping systems are made with leak-tight welds.

Flanged connections in the pressure vessel unit are equipped with welded-lip seals or metal ring gaskets. The flanged connections for the reactor pressure vessel closure head and the circulator cover are sealed with double metal ring gaskets. Leak tests are conducted after every assembly to ensure that the system is leak-tight.

During power operation, the air in the primary cavity is monitored for helium and steam so as to detect any possible leaks at the pressure vessel unit.

In service pressure test with gas takes place at 8 year intervals, core off-load is not necessary.

During the inspection of one reactor (4 year intervals) the second reactor can be operated. The industrial safety of personnel is ensured during assumed malfunctions of the operating reactor with structural and organizational procedures.

4.3.2.1 Reactor Pressure Vessel

The reactor pressure vessel (RPV) consists of forgings welded together and is closed by a reactor pressure vessel closure head that consists of two welded forgings. The closure head and the body of the vessel are bolted together.

The reactor pressure vessel bottom head is penetrated in the middle by a large nozzle for fuel discharge. Suspended from this is the fuel element discharge tube with the forged failed fuel separator block. In addition, there are nozzles for fuel element supply and for the small ball shut-down system.

The shell course above the bottom head (support lug course) is reinforced. It includes the gas duct pressure vessel nozzle. The three RPV support lugs are fixed to the outside.

Above the support lug course is a support ring for internals with a sealing surface for the core barrel.

The RPV horizontal supports, the stops for the core barrel guide pads, and various nozzles are arranged in a reinforced course in the upper portion of the RPV.

Only that portion of the RPV beneath the pebble bed is insulated on the outside. The insulation can be removed for in-service inspections.

4.3.2.2 Gas Duct Pressure Vessel with Hot Gas Duct

The gas duct pressure vessel and the concentric hot gas duct serve to convey the primary coolant between the reactor pressure vessel and the steam generator pressure vessel.

Hot gas flows from the reactor outlet to the steam generator (SG) in the hot gas duct; cold gas flows from the gas outlet nozzle of the steam generator pressure vessel to the inlet nozzle of the RPV in the annulus between the gas duct pressure vessel and the hot gas duct.

The vessel consists of a seamless forged ring welded to the nozzle of the RPV and the steam generator pressure vessel. It is insulated on the outside.

Working radially outwards from the inside the straight hot gas duct is constructed as follows:

- Metal pipe to guide the hot gas flow and to protect the insulation material from direct fluid flow loadings,
- Wound fiber mats with radial convection barriers,
- Metal support pipe acting as load-bearing member and as seal between the hot and cold gas regions.

Bellows expansion joints compensate for displacements caused by thermal expansion of the hot gas duct.

In-service inspections of the hot gas duct are not planned.

4.3.2.3 Steam Generator

The steam generator pressure vessel consists of the steam generator pressure vessel portion and the circulator pressure vessel portion joined to the former by bolting. The steam generator vessel is located to the side of and lower than the reactor pressure vessel.

The steam generator pressure vessel portion consists of several forgings joined by welding.

The bottom torispherical head has a blanked-off nozzle for use as a man-way.

The bottom forged ring with the feedwater nozzle and seismic supports is reinforced. The nozzle is welded into the shell around a thermo-sleeve.

The lower cylindrical shell is forged as a seamless ring.

The reducer ring serves to support the cold gas header of the steam generator shell and as a transition between the different diameters of the lower and upper courses of the SG pressure vessel portion.

The upper shell bearing the nozzle for the gas duct pressure vessel, the main steam nozzle and the support lugs for the sliding support, is reinforced. The support lugs are mounted on the outside. The main steam nozzle is welded into the shell around a thermo-sleeve.

The flanged ring at the top of the steam generator pressure vessel portion receives the bolts for fastening the circulator pressure vessel portion.

The circulator pressure vessel portion consists of the following 3 forgings:

- A flange ring integral with an internal ring forming the supporting flange for the primary gas circulator,
- A cylindrical shell with various nozzles and an integrated cover flange,
- The circulator cover consisting of a flange ring and domed cover plate.

Steam Generator Internals

The steam generator tube bundle is of helical coil construction with water/steam flowing inside the tubes and the primary gas on the outside. The flow paths are as follows:

The hot primary coolant enters the tube bundle through an elbow from above. It is the function of this flow deflector assembly to redirect the primary coolant entering from the side, downwards and to ensure as uniform a flow through the tube bundle as possible.

The primary coolant flows from the flow deflector assembly down through the tube bundle between central pipe and the tube wrapper. The tube wrapper is welded at the bottom to the steam generator shroud, thus preventing bypass stream flows of gas around the tube wrapper. The tube wrapper and the SG shroud form the barrier between the flow through the tube bundle and the concentric return flow of the cold gas.

After the gas leaves the tube bundle, it is deflected through 180°. The cold primary coolant now flows upwards between the steam generator shroud and the wall of the pressure vessel; the temperature of the pressure vessel is thus governed by the temperature of the cold gas. To minimize parasitic heat loss between the tube bundle flow and the gas return flow, the steam generator shroud is thermally insulated.

As the primary coolant flows back upwards, it is collected in a cold gas header and passes from there to the primary gas circulator. From the circulator, the primary coolant flows back to the reactor through the top plenum of the SG pressure vessel and the annular gap of the gas duct pressure vessel.

The feedwater enters the SG at the bottom through the lateral feedwater nozzle. From there, it is distributed via a tube-sheet to the tubes leading to the tube bundle. The water flows through the helical tubes from bottom to top, where it is evaporated and then superheated. This steam reaches the main steam nozzle via an expansion tube bundle above the main tube bundle. The expansion tube bundle compensates for the differences in thermal expansion between the tubes and the SG pressure vessel. The expansion tube bundle is thermally insulated to minimize heat losses.

The helical tubes are individually connected to the feedwater and main steam tube-sheets by connecting tubes. The helical tubes are held by vertical supporting walls which are arranged radially around the central pipe. The weight of the bundle is borne by the support walls that stand on a support structure at the lower, cold end.

The helical tubes are arranged in cylindrical planes at constant radial distances from the central pipe. The number of tubes per cylinder and their longitudinal pitch are chosen in such a way that all the tubes are of almost the same length. The number of tubes is such that sufficient spare heat surface exchange capacity is available in case damaged tubes have to be plugged. Tube-sheets for feedwater inlet and main steam outlet are provided in order to assure accessibility for examinations or for plugging damaged tubes. These tube-sheets are accessible after the feedwater or main steam nozzle has been opened.

Primary Gas Circulator

The primary gas circulator consists of a single-stage centrifugal compressor. It is driven by a variable-speed asynchronous motor. The motor is cooled with helium that in turn is cooled by water. The circulator and its motor are arranged as an insertable unit with vertical shaft in the circulator pressure vessel portion. There is a circulator damper on the suction side of the circulator.

The circulator raises the pressure of the helium extracted from the SG outlet side by approximately 1.5 bar.

Motor shaft bearings are provided on both sides of the motor. The upper bearing is a combined thrust and journal bearing, while the lower bearing is only of the journal type.

The circulator motor is powered from the auxiliary power supply through a converter. The circulator is provided with diverse tripping.

Except for operational monitoring no special or in-service inspections are envisaged for the primary gas circulator.

Any necessary repairs in the region of the motor can be performed with the primary system depressurized and the circulator cover removed.

In addition, the circulator unit can be removed without allowing air into the primary system.

4.3.2.4 Support of the Pressure Vessel Unit

The supports for the RPV and the SG pressure vessel are located at the level of the central axis of the gas duct pressure vessel. Three radially guided support lugs set 120° apart around the RPV take the load and secure hold the vertical axis of the RPV in place. Below the closure head flange are stops that ensure the stability of the pressure vessel unit during external events.

The load of the steam generator pressure vessel is taken by sliding bearings that allow horizontal displacement in line with the gas duct pressure vessel that is welded to the RPV. Horizontal forces exerted by the gas duct pressure vessel are absorbed at this level. Likewise, the lower end of the steam generator can travel along a gas duct pressure vessel axis.

Shock loads due to external events are absorbed by snubbers.

4.3.2.5 Pressure Control and Pressure Relief System

Pressure Control System

Operation of the primary system pressure is controlled by feeding helium through the helium purification system to and from the purified gas store (helium supply and storage system).

Pressure Relief System

The pressure relief system protects the primary gas envelope from overpressure and has a two-train system configuration.

Each train contains an isolation valve, a blocking valve, a safety valve, and a rupture disc.

The actuation pressures of the two safety valves are staggered in such a way that the train with the smaller cross-section responds first. This train alone is sufficient to handle all design basis accidents.

In the postulated event of one safety valve being stuck open after actuation the train concerned is automatically closed by the blocking valve to minimize the escape of primary coolant.

The isolation valves upstream of these valves are closed only for maintenance, inspection and repairs, and are interlocked in such a way that the valve of only one train can be closed at a time.

The rupture discs downstream of the valves are provided merely for sealing against possible leaks in the affected train; they allow leakage monitoring in the area between safety valve and rupture disc.

Both trains of the pressure relief system have a joint letdown line to the helium purification system before the primary system isolation valve. When required, they blow-down into the reactor building (see Section 4.3.3).

Pressure Equalizing System

The pressure equalizing system protects the core barrel in the reactor pressure vessel from unacceptable differential pressure loadings and assures the integrity of the primary coolant envelope when the reactor pressure vessel is open.

To equalize the differential pressure across the core barrel, a connecting line that can be isolated externally is provided as an operational train. The isolation valve is closed only after primary gas pressure has been reduced to ambient pressure before opening the reactor pressure vessel, e.g., for in-service inspection.

To compensate for the differential pressure caused by pipe breaks of large cross-sections (up to DN65), an internal train is provided that opens an internal connection (inside diameter of approximately 200 mm) between the core barrel and the core barrel-to-RPV gaps containing stagnant helium on response of a rupture disc at a differential pressure of approximately 1 bar. After rupture disc response and reduction of primary gas pressure to ambient pressure occurs, this internal train can be closed manually with the isolation valve on the core barrel in order to re-isolate it from the atmosphere before the reactor pressure vessel closure head is opened.

4.3.2.6 Primary System Isolation

All primary coolant pipes connected to the pressure vessel unit have a cross-section less than or equal to DN65, or have suitably reduced cross-sections.

As a rule, they are provided with a combination of two series connected valves of which the one nearest the pressure vessel unit can be operated manually and the second (primary system isolation valve) is actuated by the reactor protection system.

Exceptions to this rule are:

- Lines for failed fuel discharge, that have only one valve actuated by the reactor protection system,
- The external line of the pressure equalizing system, that cannot be isolated in the event of a postulated break,
- Instrument lines, that have two isolation valves not actuated by the reactor protection system,
- Lines with blanked-off flanged ends and manually operated isolation valves.

Only the valves actuated by the reactor protection system have safety functions. The manually operated valves are provided solely for operational purposes such as repairs.

4.3.2.7 Steam Generator Isolation and Pressure Relief Systems

Steam Generator Isolation System

The steam generator isolation valve separates the primary system from the conventional water/steam cycle, that is not subject to any safety requirements in the event of any accident or malfunction in that the reactor is tripped.

Whenever the reactor is tripped, the steam generator is isolated by the reactor protection system that shuts off the main steam and feedwater sides by closing two series-connected isolation valves in each line. The reactor protection system also isolates the start-up and shut-down circuit whenever the reactor is tripped during start-up and shut-down operation.

Steam Generator Dump System

In the event of a steam generator tube break, the steam generator is quickly emptied in order to limit the amount of water that enters the primary system. The pressure relief valves are closed after relief in order to prevent the primary system from depressurizing through the steam generator tube break.

When the accident condition has been terminated, the steam generator and the primary system connected to it by the break are completely emptied through the helium purification system.

Steam generator relief is performed through the feedwater piping. The steam generator relief system is not subject to any requirements relating to overpressure protection of the steam generator secondary side. This is performed by a separate safety valve connected to the steam side.

The steam generator relief system consists of two parallel relief lines for each steam generator. Two relief valves are arranged in series in each relief train.

This parallel and serial arrangement of the relief valves assures redundancy for reliable opening and closing. The two relief lines discharge into a flash tank. The discharged feedwater is collected in a tank downstream of the flash tank.

The relief valves are actuated by the reactor protection system.

The steam generator relief system is designed in such the relief valves are closed after the steam generator is emptied. Therefore, no more than 600 kg of water can enter the primary system in the event of a steam generator tube break with only one of the two pressure relief trains operating.

4.3.3 Confinement Envelope

The confinement envelope of the HTR-Module acts in conjunction with other barriers to the release of radioactive substances to minimize the radiological impact on the environment caused by the accident conditions.

The following features fulfill confinement functions in the HTR-Module:

- Reactor building,
- Secured sub-atmospheric pressure system,
- Building pressure relief system, HVAC systems isolation.

Because the fuel elements themselves provide reliable retention of radioactive fission products and because helium is used as the primary coolant, no special demands are made of the confinement envelope since compliance with the dose limits as stipulated by the German Regulation in Article 28, Paragraph 3 of the Federal German Radiological Protection Ordinance is assured even without additional fission product retention facilities. Acceptability and dose limit compliance in other regulatory jurisdictions must be confirmed. Preliminary assessments of dose consequences using methodology used in the U.S.A are provided in Chapter 7.

In principle, it is acceptable to discharge the unfiltered primary coolant directly into the environment in the event of accident induced depressurization of the primary system.

Nevertheless, it is intended to minimize the radiological impact on the environment of a depressurization accident by filtering the escaping primary coolant in the secured sub-atmospheric pressure system before discharge from the vent stack. The design basis for the filtration system is a primary system break with a diameter less than or equal to 10 mm.

For larger postulated breaks that lead to a significant pressure build-up in the reactor building, building pressure is relieved by direct discharge of primary coolant into the environment. After depressurization, the pressure relief ports are closed; further discharges via the secured sub-atmospheric pressure system are filtered.

4.3.3.1 Reactor Building

No leak-tightness demands are made of the reactor building as part of the confinement envelope or of its penetrations and entrances. A sub-atmospheric pressure system and a pressure relief system are provided in the reactor building merely to minimize the impact on the environment after a postulated break in the primary system. The entrances to the building consist of interlocked doors, which assure that a directional control of air flow in the reactor building can be maintained at all times.

Since it is possible to relieve building pressure directly into the atmosphere in the event of a depressurization accident, it is unnecessary to design the building for high interior pressures.

The primary cavities and outside walls of the reactor building are designed for the pressures and temperatures that are postulated to occur in the event of a primary coolant pipe break and building pressure relief to the environment.

The building's design pressure is approximately 0.3 bar. The radiological shielding effect of the primary cavity provides reliable protection of the environment from direct radiation.

4.3.3.2 Secured Sub-atmospheric Pressure System

Areas containing primary coolant pipes are monitored for helium leakage. If specified limits are exceeded, the affected area of the building is switched from the unfiltered air exhaust system to the secured sub-atmospheric pressure system.

The two circulators of the secured sub-atmospheric pressure system are designed for a volumetric flow rate of 1.25 m^3/s each and receive emergency power back-up. The system is equipped with a HEPA filter and an activated-carbon absorber capable of handling total volumetric flow of 2.5 m^3 /s.

Design of the secured sub-atmospheric pressure system for postulated external events is not envisaged.

4.3.3.3 Building Pressure Relief System

The arrangement of the compartments in the reactor building and the HVAC systems assure that air flows from rooms with low airborne activity levels into rooms with higher airborne activity levels in order to limit any spread of radioactive materials in the building under normal operating conditions.

As far as possible, the secured sub-atmospheric pressure system is used to minimize radioactive releases in the event of accidents.

Relief ports fitted with rupture discs or dampers that connect the modular units with the reactor hall are provided to control depressurization accidents ($\leq$DN65 breaks).

Building pressure itself is relieved into the environment through two physically separate ports. Each port is equipped with:

- A pressure relief damper that opens at a response pressure of approximately 0.1 bar and closes automatically after pressure equalization
- An isolation damper that is normally open and can be closed manually after a depressurization accident in case the pressure relief damper sticks open.

Sub-atmospheric pressure can thus be restored after pressure relief.

A pressure relief system is also provided for the equipment compartments (helium tract) in the reactor auxiliary building.

4.3.4 Residual Heat Removal

The residual heat of the HTR-Module is determined by:

- directly formed fission products
- neutron capture in the fission products
- decay of the actinides

The HTR-Module's residual heat generation rate is calculated on the basis of the HTGR specific draft of DIN-Norm 25485 or on the almost identical ANS Standard 5.1/1979. These standards are fully applicable to the HTR-Module in regards to directly formed fission products and the Np239 fraction. The two reactor-specific fractions for neutron capture in fission products and the other actinides are determined on the basis of a reactor-specific reaction cross-section library using the ORIGEN program (except actinides (e.g., Np 239), that are taken directly from the core layout). Following the above mentioned DIN-Norm standard the calculated residual heat generation is applicable for the relevant time range up to approximately 40 hours with an average 2-sigma error.

4.3.4.1 Residual Heat Removal through the Main Heat Transfer System

In normal operation and during anticipated operational occurrences, the reactor is cooled down for lengthy outages through the main heat transfer system. Plant cool down and long-term residual heat removal are performed by the primary gas circulator and the steam generator on the primary side, and either the operational water/ steam cycle or the operational start-up and shut-down circuit on the secondary side, the latter running at decreasing steam temperatures and in water-only operation. Residual heat removal through the main heat transfer system is not safety-related.

4.3.4.2 Residual Heat Removal through Cavity Cooler

Under normal operating conditions, the cavity cooler installed in the reactor cavity serves as a heat sink for the heat dissipated by the reactor. It thus also protects the concrete structures from unacceptably high temperatures. On unavailability of the main heat transfer system or after a reactor scram not followed by reactor cooldown, the cavity cooler limits the temperature not only of the concrete structures but also of reactor structures especially that of the reactor pressure vessel.

Brief Outages

No active steps are taken to cool down the reactor during brief outages or after scrams caused by malfunctions. This makes it easier to restart the plant if the malfunction is quickly eliminated.

During the interruption of active heat removal by the primary system, a portion of the residual heat generated is transferred by natural convection, thermal conduction and radiation from the reactor core through the side reflector to the reactor pressure vessel and from there to the cavity cooler in the reactor cavity.

Failure of the Main Heat Transfer System

In the event of main heat transfer system unavailability (e.g., under accident conditions), the cavity cooler also assumes the task of long-term removal of residual heat. In this case, the temperatures in the core and in some structures will rise temporarily over a period of several hours until equilibrium is established between the rate at which residual heat is generated and the rate of removal through the reactor pressure vessel. The temperatures involved remain in all cases below the specified design values, both when the primary system is fully pressurized and when it is depressurized.

If repairs to the main heat transfer system can be made without substantial work having to be done on the primary system, the reactor can then be cooled down through the main heat transfer system with the primary system either fully pressurized or depressurized.

Cavity Cooler

The cavity cooler consists of water-carrying, vertical, welded membrane tubes arranged side-by-side to form a closed, cooled panel wall (water wall). It is of three-train configuration and designed for all postulated accidents within the plant as well as external events.

Two trains of the cavity cooler are served by the secured cooling system. This system has emergency power back-up and is of earthquake-proof design. The two secured cooling system heat exchangers are cooled by the secured service water system, which is also constructed in two trains. These systems also have emergency power back-up and are of earthquake-proof design. Connections, provided in the reactor building for feeding water to the secured cooling system from the outside, keep cavity cooler supplied in the event of an aircraft crash or explosion blast wave.

The third train of the cavity cooler is served by the operational component water cooling system and is not safety-related. The layout and design of the third cavity cooler train is the same as the two other trains.

The three trains of the cavity cooler are constructed in such a way that every third tube of the tube wall is connected to the same manifold and header. This assures that heat is removed around the entire circumference even in the event of failure of one train or two trains.

Hot air flow behind the cavity cooler is inhibited to such an extent as to prevent unacceptable heating of the concrete under all circumstances. Under normal operating conditions, the amount of heat to be removed is approximately 400 kW, maximum core residual heat is about 815 kW.

Visual inspections, leak tests and repairs to the cavity cooler are possible.

Cooling of RPV and Steam Generator Supports

Each of the support pads bearing the support lugs of the RPV and the steam generator are located at the middle support level at gas duct pressure vessel height also has a three-train cooling system to reduce heat flux into the concrete structure. Cooling is performed by the secured cooling system and the operational component cooling system in a manner similar to that of the cavity cooler.

4.3.5 Helium Purification System

The helium purification system has the following operational functions:

- Removal of gaseous contaminants from the primary coolant to maintain specified values, especially of H2O, CO, CO2, H2, N2, CH4
- Removal of radioactive gaseous contaminants from the helium before transfer to the purified gas store (H3, Xe, Kr, Ar)
- Removal of dust and other particles in suspension in helium
- Start-up purification of the primary system before initial start-up and after inspection and maintenance
- Purification of newly delivered helium
- Pressure control in the primary system together with purified gas store.

In the event of a steam generator tube break, the helium purification system has the function of removing the in-leakage water and any corrosion products from the primary system. This is not needed for accident control, but rather serves to minimize corrosion in the reactor core due to water-carbon reactions and prevents the response of pressure limitation systems in the primary system causing the release of a small amount of primary coolant into the reactor building.

The primary systems of the two modular units are connected to a common three-train helium purification system. Each reactor is assigned one purification train. The third purification train is available in the event of accident-induced failure, repairs, or regeneration of the train assigned to one of the reactors. Water separation equipment for the event of water in-leakage is built into the third purification train only (post-accident water separator, post-accident compressor, etc.).

On detection of moisture in the primary system by the reactor protection system, the primary system concerned can be switched manually to the purification train equipped with the post-accident water separator. The post-accident compressor is capable of passing the entire primary system inventory of one reactor through the post-accident moisture separator in one hour. There are possibilities for functional and economical optimization.

Power is supplied to the helium purification system from the auxiliary power supply. The portions of the helium purification system outside the primary gas envelope are not designed for external events.

4.3.6 Fuel Handling and Storage

4.3.6.1 Fuel Handling

Fuel elements are constantly added to and removed from the core during operation. The fuel elements are forwarded by gravity and pneumatically. Primary coolant at cold gas temperature serves as the carrier gas, in special situations, air or nitrogen is used. Both reactors share one charge station for new or used fuel elements and the same systems for storing spent or partially depleted elements.

Each reactor has a separate feed and discharge system:

- The fuel elements are forwarded to the core via the central fuel element feed tube, which can be extended when the core is partially filled. Breaking gas bled from the primary system in the region of the top reflector absorbs kinetic energy of the forwarded fuel element, thus decelerating it to an acceptable impact velocity.
- The burnup of discharged fuel elements is determined by measuring the characteristic gamma line of the nuclide Cs-137, that is proportional to burnup. If the target burnup has not been reached, the fuel element is returned to the core. Spent fuel elements are sent to the shipping casks (spent fuel storage).
- Fuel element fragments or out-of-shape fuel elements are extracted in a failed fuel separator and conveyed to a failed fuel cask. For availability reasons, each reactor is provided with two failed fuel separators.
- With the reactor depressurized, the feed tube and the failed fuel separator can be removed without allowing air into the reactor.

4.3.6.2 New Fuel Storage

New fuel elements are stored in the reactor auxiliary building in double-walled series-produced hooped drums (new fuel shipping casks) with enough capacity for about 1000 fuel elements per cask. The gap between the outer and inner walls of each drum is filled with fine-grained Ferro boron with a boron content of about 18 % by weight. The casks can then be stored in any number and arrangement. No additional complex technical or administrative measures are needed with respect to criticality. The fuel elements are stored in the sealed casks in an air atmosphere in the reactor auxiliary building.

The new fuel store holds a stock of shipping casks and is designed for one-year's requirements, i.e., about 200,000 fuel elements, equivalent to 200 casks. Suitable hoists and trucks are provided for handling and transporting the casks within the store. A buffer store is provided in the fuel charging station because of the physical separation of the new fuel store from the fuel charging station in the reactor building. Trucks are used to transport the casks to the reactor building and then to the buffer store, that is in one compartment with the fuel charging station. The casks are stacked as in the new fuel store. The buffer store is designed for a minimum of 10 casks.

4.3.6.3 Spent Fuel Storage

Spent fuel elements are stored in shipping casks that were developed for pressurized water reactor fuel assemblies. The design of the casks fulfils the following requirements:

- Shielding of fuel element radioactivity
- Retention of fission and activation products
- Subcriticality
- Residual heat removal by natural circulation of air around the cask
- Integrity under impacts such as dropping, collision, fire, earthquake, explosion, etc.

Each full power day 360 spent fuel elements are brought to the spent fuel storage area. Since the system is shared between two reactors, this means that 263,000 elements (for two reactors) per full power year are to be stored. Spent fuel elements are emptied from the filling station into shipping casks in a loose pile. The cask is ready for removal after it is filled and the technical operations completed (i.e., the cover system fully assembled, leakage test and contamination check performed).

The cask is transported from the fuel filling station on a rail-mounted truck into the area served by the reactor building crane. It is then lifted through a shaft to the transport floor by the reactor building crane and loaded onto a truck. The cask is placed in temporary hold in the spent fuel storage provided on the plant site.

4.3.6.4 Used Fuel Storage

For unusual repairs that require the complete unloading of the core, the partially used fuel elements are transferred to shipping casks as provided for spent fuel storage using the available fuel handling equipment. Special covers with an integral single-exit lock disc are fitted to these shipping casks before being filled with partially used fuel elements.

The shipping casks filled with partially used fuel elements are stored in the spent fuel storage until the reactor is reloaded. On completion of repairs, the core is reloaded using equipment similar to the fuel element discharge of the core (failed fuel separator, failed fuel cask) and via the fuel feed system.

4.3.7 Emergency Power Supply

The emergency power generation and distribution system supplies energy to those electric loads that are necessary to maintain plant parameters within predefined operating envelope (i.e., safe conditions) in the event of loss of auxiliary power. The HTR-Module power plant is designed so that the design range load requirements are also met during long-term loss of both auxiliary power and emergency power supplies. That is, if the power cannot be restored from the auxiliary power supply or from the emergency power supply within the required 15-hour time period, then external filling of the cavity cooler water supply through the fire brigade connections will ensure that the design temperature of the reactor pressure vessel is not exceeded.

Emergency Power Network in the Switchgear and Emergency Supply Building

A two-train network powered by 2 x 100%-duty emergency power supply corresponding to the 2 x 100%-duty redundancy requirements of the process loads is installed in the earthquake-proof switchgear and emergency supply building.

The emergency power network consists of the following sub-networks:

- Emergency three-phase AC network for loads that can accept a power cut after loss of auxiliary power until the diesel engines have started.
- A DC network for loads that must remain in operation without interruption or that must be switched on during the interruption before the diesel engines have started in the event of loss of auxiliary power.

The emergency power supply to these networks is assured by the following components:

- Emergency diesel generator sets designed for each train's maximum emergency power requirement. They are started either by an operational automatic controller or manually.
- Batteries designed to supply power immediately to the connected loads for at least 2 hour after loss of auxiliary power.

Emergency Power Network in the Reactor Building

An additional single-train emergency power network is installed in the reactor building for those loads that must remain operable for monitoring of the plant from the remote shut-down station on loss of the two-train emergency power supply in the switchgear and emergency supply building, e.g., by breakdown of the station service supply and non-availability of the emergency power diesels (greater than 2 hours), due to events such as aircraft crash and blast waves..

Consequently, this additional emergency power network is designed for aircraft crash and blast waves.

The single-train emergency power network in the reactor building is served by the emergency power supply in the switchgear and emergency supply building when the latter is available.

On loss of the emergency power supply in the switchgear and emergency supply building, the single-train network is powered by batteries.

The batteries for the remote shut-down station in the reactor building are designed for a 15-hour discharge period. A facility for connection of cables to external sources is provided.

4.3.8 Reactor Protection System

Each reactor is equipped with a dedicated protection system that is separate and independent from that of any other reactor. Under accident conditions, protective actions are thus initiated only in the reactor concerned. Accidents that affect both reactors are detected separately by each reactor protection system.

The reactor protection system monitors and processes essential process variables in order to detect malfunctions and accident conditions, and automatically initiates protective actions.

The following process variables are monitored and processed:

- Neutron flux
- Hot gas temperature
- Cold gas temperature
- Moisture in the primary system
- Pressure in the primary system
- Pressure in the secondary system
- Mass flow in the primary system
- Feedwater mass flow

All of the initiation criteria derived from these variables are used to actuate jointly the following protective actions regardless of the initiating event:

- Primary gas circulator trip
- Reflector rod drop
- Steam generator isolation

In addition, the primary system is isolated in the event of a pressure loss in the primary system, and the steam generator relief system is activated in the event of steam generator tube rupture and water leakage into the primary system.

Since only one process variable (i.e., moisture measurement) is available for initiating steam generator relief, the moisture instrumentation is accordingly designed to high standards such as or similar to KTA 3501.

The reactor protection system limitations and interlocks for the shut-down systems are described in Section 4.3.1.4.

The reactor protection system is installed in the switchgear and emergency supply building and is earthquake-proof. Additionally it is designed against air plane crash and explosion blast wave in the reactor building. So it is assumed that on demand, the following reactor protection actions will be initiated or not be prevented by an air plane crash or blast wave or by any consequential damage that these might cause (refer to Section 4.6):

- Reflector rod drop
- Primary gas circulator trip
- Steam generator isolation
- Primary system isolation
- Steam generator relief

The cabinets in each redundancy group are supplied with power from two 220 V DC system boards of the emergency power system; the feeders are diode-decoupled and incorporate DC/DC converters. A loss of power to the reactor protection systems actuates the reactor protection actions. The reactor protection actions remain actuated until voltage recovery. The power plant will maintain controlled safe condition; therefore no further monitoring with the reactor protection system is necessary. The reactor protection actions are removed manually when the power plant is again in a normal operating state.

The reactor protection system, the control room, and the plant computer system are climate controlled by a two-train air conditioning system with emergency power back-up. In normal operation and in emergency power mode, the coolers are supplied by the secured chilled water system in the switchgear and emergency supply building. Chilled water is supplied by two air-cooled water chillers.

4.3.9 Remote Shut-down Station

The remote shut-down station is housed in the reactor building; it is protected against external events and has unlimited accessibility; for this reason, it has a separate entrance from the outside.

It serves to monitor the plant on loss of control room function. For this purpose, accident instrumentation is provided in the remote shut-down station.

In essence, the accident instrumentation in the remote shut-down station acquires measured data giving information on operation of the shut-down circuit, the cavity cooler and the integrity of the pressure vessel unit; it also includes radiological measurements and equipment for documenting the measurements.

It is possible to trip the small absorber ball shut-down system from the remote shut-down station; this system ensures long-term sub-criticality. No further actions are initiated from the remote shut-down station.

The remote shut-down station is independent of the control room. The actuation equipment for tripping the small ball shut-down system and the indicators that are also installed in the central control room are non-interactive.

A single-train emergency power network is installed in the reactor building to supply power to the remote shut-down station (Section 4.3.7).

The remote shut-down station has a dedicated HVAC system with emergency power backup, which is independent of the nuclear HVAC system-of the reactor building.

4.3.10 Controlled Area

The controlled area of the HTR-Module power plant includes

- Reactor building, except remote shut-down station and its entrance
- Reactor building annex (area of closed cooling water systems)
- Reactor auxiliary building (most portions)
- Spent fuel store (set-down positions; central truck entrance only temporarily when handling shipping casks or for inspection).

The main entrance to that portion of the controlled area comprising the reactor building and the reactor auxiliary building is in the reactor auxiliary building.

The reactor building and reactor auxiliary building each have two closed stairwells with exits to the outside. These exits to the outside fulfill the requirements for plant security and escape routes.

During power operation, only a few persons are present temporarily in the reactor building for checks and maintenance purposes and for delivering new fuel elements and removing spent ones.

4.4 Nuclear Classification and Quality Requirements

The systems and components differ in functional and safety-related importance for the overall plant and are classified accordingly. This basically applies to

- Pressure and activity-carrying systems
- HVAC Systems
- Hoists and cranes
- Steelwork items (system and component-specific)

The quality requirements relating to integrity and operability are graded according to nuclear classification.

4.4.1 German Nuclear Classification

Refer to Figure 4-4 for the German nuclear quality classifications diagram. Parallel and similarities exists to U.S. NRC and ASME codes and standards (for LWRs) but no attempt has been made to develop a cross-walk table.

4.4.1.1 Pressure and Activity-Carrying Systems

The classification of pressure and activity-carrying systems relates to the integrity of the pressure-retaining walls. Component internals, such as RPV internals and steam generator pressure vessel internals, are not covered by this classification; their specifications are based on their functional requirements.

Quality Class MK 1

- Class MK 1 covers systems and components, postulated failure of which could lead to radioactive releases in excess of the dose limits stated in German regulations (Article 28, Paragraph 3 of the Federal German Radiological Protection Ordinance).
- This class also covers systems and components, postulated failure of which would lead to a non-isolable break in the primary gas envelope, even though the dose limits stated in German regulations (Article 28, Paragraph 3 of the Radiological Protection Ordinance) would not be exceeded.

In accordance with these criteria, the components of the pressure vessel unit and all connected primary coolant piping up to and including the primary system isolation valve are assigned to Quality Class MK 1.

Quality Class MK 2

Quality Class MK 2a

- Systems or system sections, which are necessary to limit loadings on components and structures for residual heat removal.
- Systems and components, which are required for reactor shut-down

Quality Class MK 2b

- Systems and components that limit the release of radioactivity to levels that do not exceed the specified limits for normal operation and anticipated operational occurrences

As connecting components between the Quality Class MK 2 to the Class MK 1 and NNK:

- Pipes and components, which connect to those of the class MK 1.
- Valves with isolating functions that connect non-nuclear-classified systems and components to class MK 2 items

Non-Nuclear-Classified (NNC)

All Systems that are not assigned to classes MK 1 or MK 2 are non-nuclear-classified.

The systems and system sections assigned to classes MK 1 and MK 2 are listed in Table 4-3 (also see Figure 4-4).

4.4.1.2 HVAC Systems

Quality Class ML

- HVAC components that have to be operated during or after accidents to minimize the radiological impact on the environment and thus function as activity-carrying components during such operation, and
- HVAC components that ensure heat removal from compartments containing systems or components for accident control or monitoring.

Systems performing the functions described above are:

- building pressure relief system,
- secured sub-atmospheric pressure system,
- secured air conditioning unit for control room, computer and electronic equipment rooms, including the secured chilled water system,
- secured air conditioning unit for the remote shut-down station.

Non-Nuclear-Classified (NNC)

HVAC components that are not assigned to class ML are non-nuclear-classified.

4.4.1.3 Hoists and Cranes

Hoists and cranes quality classifications are in accordance with KTA 3902. Two nuclear quality classes MH 1 and MH 2 and one not-nuclear specific quality class NNK are formed.

Quality Class MH 1

Hoists and cranes are assigned to class MH 1 if, during load transport, failure of the hoists or cranes

- could lead to radioactive releases in excess of the dose limits stated in Article 28, Paragraph 3, or
- would lead to a non-isolable break in the primary gas envelope, even though the dose limits stated in Article 28, Paragraph 3 would not be exceeded.

This applies only to the reactor crane and its load lifting member:

- RPV closure head transport rig,
- lift rig for shipping cask,
- slab lift rig,
- erection lugs, load attachment lugs in concrete slabs,
- hoist mechanisms and load attachment rigging for equipment casks.

Quality Class MH 2

Hoists and cranes are assigned to the class MH 2, if during load transport, failure of the hoists or cranes could lead to radioactive releases. The present concept does not provide hoists and cranes of the class MH 2.

Non-nuclear-classified (NNC)

Hoists and cranes that are not assigned to class MH 1 or MH 2 are non-nuclear-classified. These include the following cranes in the reactor building and reactor auxiliary building:

- maintenance equipment of the failed fuel separator,
- crane in the hot laboratory,
- crane in the decontamination area,
- crane in the reactor auxiliary building entrance,
- all other cranes outside of these building.

4.4.1.4 Steelwork (System and Component-Specific)

The steelwork classification applies to the following items, provided that they serve systems or components:

- Steel platforms, steel stairways and steel landings,
- Piping supports (including valve supports), e.g., piping anchors, guides, sliding supports, support frames as well as variable support spring hangers and variable spring supports, constant hangers and constant supports, snubbers and shock absorbers,
- Component supports, e.g., structural steel frames, pump frames, lugs, support skirts, saddle structures,
- Protective and special structures, e.g., bumpers and manipulator rails,
- Anchoring items.

The various steelwork items are classified according to the following criteria.

Quality Class MS 1

This class covers steelwork items, postulated failure of which

- could lead to radioactive releases in excess of the dose limits stated in Article 28, Paragraph 3 of the Federal German Radiological Protection Ordinance,
- would lead to a non-isolable break in the primary gas envelope, even though the dose limits stated in Article 28, Paragraph 3 of the Radiological Protection Ordinance would not be exceeded,
- can cause simultaneous failure of all trains of the cooling systems in the primary cavity.

Quality Class MS 2

This class covers steelwork items, postulated failure of which

- can degrade the operability of parts assigned to seismic class I,
- could lead to loss of integrity of MK 2 components in the reactor building.

Non-nuclear-classified (NNC)

Steelwork items that are not assigned to class MS 1 or MS 2 are non-nuclear-classified.

In the case of steelwork items supporting differently classified components or piping, the classification of the overall structure is basically governed by the component class with the greatest significance to safety.

The boundaries of quality classes MS 1 and MS 2 and non-nuclear-classified (NNC) items are apparent from Table 4-4.

4.4.2 Quality Requirements

Component families are established within the quality classes defined in Section 4.4.1 and quality characteristics and verifications are specified for these families.

The specifications for the nuclear classified systems (component families) are submitted to the authorized experts appointed by the licensing authority for approval.

Separate specifications based on the requirements imposed on similar components are written for individual components that cannot be assigned to any component family. The quality characteristics and verifications are governed by the rules for the associated quality class.

Specifications for components that are not assigned to system classes (internals) are written on the basis of their functional requirements.

4.4.2.1 Pressure and Activity-Carrying Systems

The pressure and activity-carrying systems are grouped in the following component families:

- Vessels/heat exchangers,
- Piping greater than DN80 (no MK 1 components),
- Piping less than or equal to DN80,
- Valves greater than DN80,
- Valves less than or equal to DN80,
- Pumps, circulators and compressors (no MK 1 components).

The quality characteristics of the component families conform to the rules of non-nuclear codes and standards (e.g., Federal German Pressure Vessel Code, Federal German Technical Code for Steam Generating Installations). Additional requirements are placed on class MK 1 and MK 2 components, for example:

- Use of KTA 3201 and KTA 1401 for class MK 1 vessels/heat exchangers and valves greater than 80 mm,
- Preparation of special process specifications and quality assurance programs.

Quality verifications (review of manufacturing documents) for MK 1 components and MK 2a components are carried out with the participation of the authorized expert appointed by the licensing authority.

For MK 2b components and some NNK components that are subject to seismic design requirements, if failure of those leads to follow up damages to safety related components, the authorized expert appointed by the licensing authority performs just a system evaluation.

4.4.2.2 HVAC Systems

To reflect the system and component-specific tasks, the application of DIN 25 414 (Ventilating Systems in Nuclear Power Plants) necessitates the following categorization:

- Building pressure relief system in category 3,
- Secured sub-atmospheric pressure system in category 3,
- Secured air-conditioning unit for control room, Computer and electronic equipment rooms and of the secured air-conditioning unit for the remote shut-down station in category 2.

Non-nuclear-classified HVAC systems are assigned to category 3 or 4 as appropriate to their functional requirements.

Quality verifications (review of manufacturing documents) for fans, filters (series A/S/R), filter housings (series A/S/R), back draft dampers, dampers and multi-blade dampers subject to class ML leak tightness requirements are carried out with the participation of the authorized expert appointed by the licensing authority.

4.4.2.3 Hoists and Cranes

The hoists listed in the class MH 1 and MH 2 are designed in compliance with the stringent requirements of KTA Safety Standard 3902, other hoists and cranes in compliance with nonnuclear codes and standards of the class NNK (Load Attachment Rigging for Hoist Operation; VBG 9a).

Quality verification is performed as appropriate to the items concerned in accordance with KTA 3903 for class MH hoists and cranes, and VBG 9a for NNC hoists and cranes.

4.4.2.4 Steelwork for Systems and Components

The quality characteristics of steelwork comply with rules of DIN 18 800 (Part 1: Steel Structures; Design and Construction - Part 7: Steel Structures; Fabrication, Verification of Suitability for Welding). Additional requirements are placed on class MS 1 and MS 2 steelwork (e.g., application of KTA 3205.1 for the pressure vessel unit support).

Quality verifications (reviews of manufacturing documents) for class MS 1 steelwork are carried out with the participation of the authorized expert appointed by the licensing authority.

4.5 Postulates and Measures for In-Plant Events

4.5.1 Postulated Failure of Pressure-Retaining Components or Components with Rotating Parts

Pressure-retaining components and components with rotating parts are designed to withstand all expected loadings with adequate safety margins, taking into account foreseeable changes in the material properties during the plant's service life. They are fabricated from materials that are suitable for the intended purpose and are manufactured, assembled, installed, tested and operated within the scope of an extensive quality assurance program.

Regardless, the design of structures and components of the HTR-Module power plant is based on postulated failure of pressure-retaining piping and vessels and components with rotating parts, insofar the consequences of failure have a bearing on the safety objectives. The failure postulates are governed by the energy content and quality of the respective component and by the potential impairment of safety-related systems in the event of postulated failure.

Piping

Water or steam-conveying pipes with an operating pressure greater than or equal to 20 bars and/or an operating temperature greater than or equal to 100°C, and gas-carrying pipes with an operating pressure greater than or equal to 20 bars are classed as high energy pipes.

Circumferential breaks or subcritical cracks at any location are postulated in high-energy pipes. The following are the local and global effects of postulated failure of high-energy pipes that are taken into account in the design of safety-related structures and components:

Local Effects

- Jet impingement forces,
- Reaction forces,
- Shock wave forces,
- Forces caused by whipping pipes.

Global Effects

- Pressure build-up, pressure differentials (not in the case of water-conveying pipes with fluid temperatures less than 100°C),
- Temperature rise (only in the case of pipes with fluid temperatures exceeding 100°C),
- Humidity increase (only in the case of water and steam conveying pipes with fluid temperatures exceeding 100°C),
- Flooding (only in the case of water-conveying pipes).

Subcritical cracks are postulated in moderate-energy, water conveying pipes. The only effect considered is the resulting flooding that is included in the design basis for safety-related structures and components.

Vessels

Design basis for safety-related buildings (see Table 4-5) is the postulated rupture of high-energy vessels in the form of a rupture shock wave. In addition, lift-off of vessel parts (e.g., vessel head) after a postulated circumferential break in a high-energy vessel is considered in the design of safety related buildings.

An exemption from vessel failure postulates is made for the pressure vessel unit (see Section 4.3.2). It is designed to comply with the requirements of KTA 3201 with HTGR specific modifications and is assigned to class MK 1. Through-wall cracks can be safely ruled out during the reactor service life as the result of extensive quality assurance measures taken with respect to planning, design, materials, manufacture and inspection. Rupture of a connecting pipe with a break cross section less than or equal to DN65 is the enveloping break postulated.

Missiles

Preventive measures reduce the probability of missile generation inside the plant to such an extent that, with a few postulated exceptions, it is not necessary to consider missile effects.

Where failure of pump rotors, compressors, circulators and fans, and hence the effects of such failures on safety-related systems and components cannot be ruled out, measures are taken to ensure that resulting missiles cannot penetrate the surrounding casings. This also applies to the rotors of electric motors and generators inside the stator winding and in principle to turbine rotors. An alternative measure that can be taken to prevent unacceptable effects of turbine failure is to install the turbines in line with the reactor building.

The few cases of postulated missile generation are limited to:

- Missiles resulting from postulated vessel failure,
- Missiles resulting from postulated failure of cable penetrations in the pressure vessel unit.

4.5.2 Protective Measures

Protective measures against postulated failure of pressure retaining components or components with rotating parts are:

- Physical separation or suitable physical arrangement,
- Design for resulting loads,
- Features for deflecting or retaining missiles.

4.5.2.1 Reactor Building and Annex

The following high-energy pipes are installed in the reactor building and annex:

- Gas-carrying pipes (e.g., of the primary gas envelope, helium purification system, fuel handling equipment),
- Secondary-side pipes (e.g., of the main steam and feedwater system).

The equipment layout in the reactor building and annex is based on the principle of physical separation of the

- Two modular units, and
- Secondary-side pipes from the gas-carrying systems and from the secured cooling system inside each modular unit.

This principle of physical separation is affected by designing the structures inside the reactor building and reactor building annex to withstand the local and global effects of postulated failure of high-energy pipes (see Section 4.3.3 for limitation of the pressure build-up in the event of postulated pipe failure).

Gas-Carrying Pipes

The local and global impacts of the postulated failure of pipes are considered in the plant layout.

Failure of the pressure vessel unit is not postulated because of the extensive quality assurance measures taken. Only the occurrence of missile generation resulting from postulated failure of cable penetrations is assumed. Effects on the cavity coolers can be ruled out as the result of suitably positioned and designed retaining features.

In the event of postulated failure of pipes of the fuel handling equipment, released fuel elements are captured in concrete ducts.

The failed fuel cask is located in a mobile shielding and shipping cask that is designed to withstand the rupture shock wave resulting from postulated failure of the failed fuel cask.

The nominal diameters of the inlet lines of the failed fuel cask are greater than DN65. The possible free escape cross-section is reduced to such an extent by the single-exit gate disc that the maximum attainable blowdown rate is that which would occur after a break in a DN65 pipe at the worst-case location.

Secondary-Side Pipes

Secondary-side pipes that are consistently under high energy are:

- The main steam line,
- The feedwater line,
- The safety valve line up to the safety valve,
- Pipes of the start-up and shut-down circuit up to the valve closed during power operation,
- Pipes of the steam generator relief system up to the valve closed during power operation.

Local and global effects of postulated failure are accounted for in the design basis for the structures.

The primary gas envelope is protected against the effects of postulated failure of secondary-side pipes:

- By physical separation where practicable,
- By designing the steam generator nozzles for reaction forces,
- By optimizing the secondary-side pipe routing so that impingement of pipe ends on the steam generator nozzles is geometrically impossible,
- By designing the steam generator tubes for shock waves and differential pressures.

The secured cooling system and the secured service water system are protected by physical separation.

Anchors that are designed to withstand reaction forces resulting from postulated failure outside the reactor building or inside the reactor building annex (e.g., due to external events) are installed in wall penetrations in the outside walls of the reactor building.

4.5.2.2 Reactor Auxiliary Building

The reactor auxiliary building does not contain any plant equipment that performs safety-related functions.

In the event of postulated failure of pipes or components, the safety objectives for the reactor auxiliary building are:

- Prevention of the escape of liquid radioactive materials into the ground water
- Global stability to prevent missile loads on the directly adjacent reactor building or the seal structure.

Protection against the escape of liquid radioactive materials into the ground water is provided by installing the reactor auxiliary building and the reactor building in a common seal structure (Section 4.6).

In order to ensure that the safety objectives are not violated, the main load-bearing structures are also designed for the pressure build-up resulting from postulated failure of high-energy pipes (e.g., helium purification system). The reactor auxiliary building is equipped with a pressure relief system to limit over pressure (see Section 4.3.3).

The reactor auxiliary building is not designed to withstand local loads (e.g., due to jet impingement forces, whipping pipes, rupture of small vessels, missiles).

4.5.2.3 Turbine Building, Central Gas Supply Systems Building

Although the turbine building and the central gas supply systems building house high-energy components, they do not contain any plant equipment that performs safety-related functions.

The feedwater tank and purified gas storage tank are taken as representative high-energy vessels. These tanks are installed in such a way that any parts (e.g., a vessel head) lifting off after a circumferential break cannot hit safety-related buildings (Table 4-5). Earthquake-proof building design is done by SRSS (square root of sum of squares) superposition of seismic loading and the rupture shock wave caused by vessel failure.

4.5.3 Fire

The HTR-Module fire protection concept is based on a combination of structural and equipment-based features and administrative measures. Adequate consideration is given to the aspects of reactor safety and the protection of persons and property.

Prime consideration is given in establishing fire protection features, to measures for the minimization and containment of fire loads and for the minimization of smoke development.

Structural fire protection, e.g., the formation of fire zones and fire suppression zones, takes precedence over equipment based protection. The design of the HTR-Module fire protection system is in accordance with the applicable German fire protection standards discussed below. The German standards are comparable and consistent with the NFPA standards; however, a detailed comparison or reconciliation of the HTR-Module fire protection design with the NFPA standards is outside the scope of this study.

Structural Fire Protection

In principle, noncombustible building materials of class A as specified in DIN 4102 or flame-resistant building materials of class B1 as specified in DIN 4102 are used (in particular for decontamination coatings and floor covering).

The buildings:

- Reactor building,
- Reactor building annex,
- Reactor auxiliary building,
- Switchgear and emergency supply building, and
- Safety-related cable ducts,

are positioned at an adequate distance from each other or are separated by fire walls or buildings with fire areas.

Considering system requirements, e.g., to minimize and limit damage, and building size, additional fire suppression zones may be designated. Especially areas with high fire loads, e.g., horizontal and vertical cable routes, cable spaces and oil storage areas are designed as fire suppression zones.

Components for separation of fire zones and fire suppression zones as well as corresponding closures and openings (e.g., cable breaks, doors, and fire dampers) are sufficiently designed regarding fire resistance, i.e., at least F90 specified in DIN 4102.

The rescue routes are designed in such a way that from each position in a room with constant work places, a secure area is reachable within a maximum distance of 35 meters and from other accessible areas within a maximum distance of 50 meters.

The fire zone reactor building is subdivided into the following fire zones:

- Remote shut-down station with access, and
- Reactor 1 + 2 with services area and reactor hall.

The reactor building is subdivided into the following fire suppression zones:

- Reactor 1,
- Reactor 2,
- Services area with reactor hall.

Joints in partitions of these fire suppression zones are sealed with incombustible materials.

To control the pressure relief accidents, all rooms in the reactor building (excluding remote shut-down station) and reactor auxiliary building (helium tract) are included in the pressure equalization. Thereby impinged fire protection components such as doors, fire dampers, closures for pressure equalize openings in walls and ceilings are sufficiently fire resistant and able to withstand the corresponding compression stress.

Plant Specific Fire Protection Measures

In order to detect fires in an early stage, a fire alarm system is installed in the plant.

A central fire alarm system with single detection recognition is designated in the switchgear and emergency supply building.

An undividable ring line system is designated for the fire water supply. The ring line system is supplied by two independent feed lines.

The fire water supply is designed in such a way that the largest stationary fire-fighting system and fireplugs outdoor or inside the buildings can be supplied simultaneously with an additional water demand of approximately 1600 l/min.

Fixed fire fighting systems are installed in rooms that contain significant fire loads and are hard to reach due to insufficient smoke and heat removal.

The cabling in the reactor building and in the cable ducts between reactor building and switchgear and emergency supply building is made by flame-resistant, halogen-free special cables (e.g., FRNC-Cable). Fixed fire protection systems are not required in this area.

In order to keep the rescue routes smoke-free, each of the following can be aerated sufficiently:

- both staircases of the reactor areas via the incoming and outgoing air supply from the reactor auxiliary building
- the staircases in reactor building, reactor auxiliary building and switchgear and emergency supply building via a ventilation that is independent from the building ventilation.

A smoke control system is designated for the cable, switchgear and electronic rooms in the switchgear building.

The cable rooms are protected additionally by sprinkler systems.

Administrative Fire Protection

Administrative fire protection provisions are recorded site-specific in accordance with chapter 6 of KTA 2101.1.

4.5.4 In-plant Explosions

Because of the limited use of hazardous materials, the gas mixtures selected, the concentrations that might form and the design boundary conditions selected, no special precautions are taken to protect the plant against in-plant explosions:

- Methane/argon mixtures in measuring equipment are not explosive.
- Hydrogen and oxygen are not handled in safety-related buildings.
- The handling of oil does not place any requirements on explosion protection.
- Battery compartments are ventilated; therefore a flammable mixture cannot form there either. Additionally, the pertinent regulations (VDE) are observed.

4.5.5 Dropping of Heavy Loads

The reactor building crane is designed to comply with the stringent requirements of KTA Safety Standard 3902 and is operated in accordance with KTA Safety Standard 3903. The dropping of a load on to safety-related systems with unacceptable consequences is therefore ruled out.

4.6 External Hazards

The HTR-Module is designed to protect against the following external hazards

- Earthquake
- Chemical explosion
- Aircraft crash

while accounting for the following loadings:

- Lightning
- Wind, storm

- Snow, rain, hail
- High water, low water
- Hazardous gases

The plant features designed to protect against these external hazards include a combination of equipment-based, structural and organizational protective measures that utilize the characteristic safety features of the HTR-Module power plant.

The external hazards listed above can be characterized as follows:

- The earthquake is a widespread event. Vibrations caused by an earthquake affect all plant equipment and the vicinity of the HTR-Module power plant.
- The blast wave caused by an external chemical explosion is a widespread impact. Loadings caused by it are primarily limited to the vicinity of the explosion source.
- Aircraft crash is a local impact. Loadings caused by aircraft crash, the impact of wreckage, fuel fire and fumes affect individual buildings of the HTR-Module power plant.

Special administrative, organizational and structural measures are taken to counteract sabotage.

4.6.1 Protection Functions

The plant is designed to prevent external hazards from disrupting plant operations or affecting safety-related plant equipment. Because any impairment of the plant results in a disturbance to system behavior or system conditions then no additional signals, other than the existing protection criteria of the reactor protection system are needed to actuate equipment required for external hazards.

To protect the environment from a release of radioactive materials, the following safety functions are assured to be operational during and after the above external hazards:

- Reactor shut-down and long-term subcriticality
- Residual heat removal
- Limitation of the radioactive release.

Shut-down and Long-Term Subcriticality

The following actions are initiated by the reactor protection system when required even if the power supply fails following the external hazard event: reflector rod drop and primary gas circulator trip. The reflector rods drop into the scram position due to gravity if the power supply to the motor is cut off and the primary gas circulator is tripped on loss of power. The reactor is then in the “subcritical hot” condition.

The purpose of the small ball shut-down system is to compensate the reactivity gain when the reactor cools down. In order to actuate this system, the power supply to the closure solenoids is cut off manually from the control room or the remote shut-down station causing the vessel closures to open from the force of gravity and the small ball shut-down elements to fall freely into the reflector columns. All equipment in the reactor pressure vessel required for shut-down and long-term subcriticality is designed to withstand external hazards and remain operational.

Residual Heat Removal

Residual heat is removed by the secured cooling system to protect the reactor pressure vessel, vessel internals, vessel supports, and the concrete structures of the primary cavity.

The secured cooling system and the secured service water system are designed for earthquake and remain operational.

In addition, the secured cooling system is designed to take water via the hose connections (fire brigade connections) for aircraft crash and explosion blast wave. If the cooling loops fail, the above cooling loads can be cooled via the hose connections. The time available before performance of this action becomes necessary is about 15 hours.

Limitation of Radioactive Release

By far the largest fraction of the radioactive inventory of the overall plant is contained in the fuel elements within the primary gas envelope.

The reactor building is structurally protected against external hazards. The relevant internals, e.g., primary gas envelope, are designed to withstand induced vibrations.

After external hazards, the reactor building is accessible for repair work. Totally encapsulating protective gear is necessary because of the possible presence of airborne activity.

The remote shut-down station is directly accessible from outside. In addition, it is equipped with a separate HVAC system that prevents any airborne activity from entering the station.

Earthquake

Plant system, structures and components required to assure the above safety functions are operable during and after an earthquake are assigned to Seismic Class I and designed to functionally survive an earthquake.

Plant system, structures and components not required for this purpose are assigned to Class II and are not designed to functionally survive earthquake, provided it is assured that failure of these plant features will not impair the operability of Class I system, structures and components.

The reactor building and reactor auxiliary building are located in a common seal structure. The seal structure of the reactor auxiliary building and the main load-bearing structures that support it are designed to functionally survive an earthquake to avoid having to conduct analyses of releases of radioactivity into the ground.

The other main load-bearing structures of the reactor auxiliary building are designed, in accordance with DIN 4149, to be stable at the intensity of the safe shut-down earthquake to prevent unacceptable effects on the seal structure and the adjacent reactor building. The reactor building and the switchgear and emergency supply building are designed against the effects of earthquakes (see Table 4-5).

The layout or arrangement of earthquake resistant designed structures is to avoid impact from the wreckage loads from non-earthquake resistance designed structures.

Inside the reactor building, the primary gas envelope and the equipment for failed fuel separation are designed for earthquake-induced vibrations and remain operational.

In the event of breaks in connecting systems, the primary system isolation valves are closed by the reactor protection system.

The steam generator is designed to survive earthquake-induced vibrations and not fail the steam generator tubes.

The integrity of the main steam and feedwater lines connected to the steam generator pressure vessel in the reactor building and the operability of the associated isolation valves are assured.

In the seismically designed switchgear and emergency supply building, the reactor protection system and the emergency power facilities and their auxiliary systems remain intact. The plant can be monitored and the current status checked from the control room that is also located in this building.

If the auxiliary power supply is cut off by an earthquake, the emergency diesel sets in the switchgear and emergency supply building start up automatically and supply power to the connected systems (e.g., secured cooling system, reactor protection system).

Residual heat is removed by the secured cooling system in order to protect the reactor pressure vessel, its internals, supports for the pressure vessel unit and the concrete structures of the primary cavity.

If damage to the lines of the fuel handling equipment that are not designed for earthquake, allows fuel elements to escape, they are retained and shielded in the concrete ducts through which the fuel handling lines are run.

Aircraft Crash and Explosion Blast Wave

Aircraft crash and explosion blast wave are infrequent events that are not classed as design basis accidents in the German regulations - "Guidelines for the Assessment of the Design of PWR Nuclear Power Plants for Accidents pursuant to Article 28, Paragraph 3 of the Radiological Protection Ordinance" (Accident Guidelines).

Neither a single failure nor an outage for repairs or maintenance is postulated for external hazards of low frequency.

The reactor building and the safety relevant plant sections inside the building are designed against the loads of aircraft crash or explosion blast wave. All other buildings and plant sections are not designed for these events.

The outside walls and roof of the reactor building are dimensioned in such as way as to provide complete protection against aircraft crash. The inside structure is uncoupled from the outside structure; the only link between the two is the foundation slab.

Inside the reactor building the primary gas envelope is designed to withstand induced vibrations.

If any disturbances arise within the plant as the result of an aircraft crash, they are detected by the reactor protection system and the reactor is rendered "subcritical hot", i.e., reflector rod drop, primary gas circulator trip, steam generator isolation. The primary system isolation valves are closed in the event of breaks in connecting systems. The steam generator tubes are designed to withstand vibrations. Therefore water ingress due to aircraft crash is not postulated. Inside the reactor building, the secondary-side pipes are designed to withstand induced vibrations.

In case of breakdown of power supply all reactor protection actions are actuated.

The secured cooling system up to and including the hose connections is designed to withstand the sustained loadings.

If parts of the secured cooling water supply (heat exchanger, pumps, service water pipes) located outside the reactor building are no longer available, water can be fed manually into the cavity coolers via the hose connections.

Installation of the hose connections near the remote shut-down station assures their accessibility.

Fuel elements may escape if the vibrations cause breaks in the lines of the fuel handling equipment. However, these fuel elements are retained in the concrete ducts and shafts through which the fuel handling lines are run. The concrete ducts shield the radiation emitted by the fuel elements.

The switchgear and emergency supply building, that also houses the control room, the reactor protection system and the emergency power supply, is not designed to withstand loadings due to aircraft crash. Partial or complete destruction of this building as the result of a crash is possible.

If the reactor protection system is also affected, the protective actions reflector rod drop, primary gas circulator trip and primary system isolation are nevertheless initiated and actuated, i.e., not prevented if required. This is due to the design and type of the equipment used (e.g., deenergize-to-trip principle).

If the control room has failed, the plant will be supervised by the remote shut-down station that is located in the reactor building and protected against the effects of an earthquake or airplane crash.

The remote shut-down station is equipped with a separate power supply that is capable of assuring plant monitoring and adequate ventilation of the remote shut-down station. It has a direct access from outside, which allows access after an aircraft crash or an explosion blast wave.

Immediate staffing of the remote shut-down station is not necessary for plant monitoring. A period time of at least 15 hours is available before this action becomes necessary. After the arrival of personnel, water can be fed into the cavity coolers via the hose connections, and mobile emergency power generating sets can be connected to the power supply of the remote shut-down station as required.

4.6.2 Postulated Loads

Earthquake

Pursuant to KTA 2201.1 (6/75) and KTA 2201.2 (11/82), design of the HTR-Module power plant for seismic events is based on: (1) the operating basis earthquake (OBE), the safe shut-down earthquake (SSE) and state-of-the-art correlations in site-specific surveys of geology, seismology and building lot, and 2) the regional free-field spectra input for the seismic load calculation (required for sites in the Federal Republic of Germany) unless site-specific peculiarities require otherwise. Therefore,

1) Definition of the OBE and SSE site intensities on the basis of the earthquake frequency. The frequency of occurrence to be used is 10^{-5}/yr for SSE and 5 x 10^{-3}/yr for OBE.

2) Definition of the spectra according to particular OBE and SSE site intensities for both horizontal and free-field components. Commensurate with the rare earthquake declared in (1) an average value spectra (50%-fractiles) have to be chosen, e.g., acceleration-response-spectra, and the dependence on local substrate conditions has to be taken into account.

3) The accelerations for the vertical design response spectra are to be defined as being 2/3 of those of the horizontal design response spectra.

4) The strong motion durations are to be selected as appropriate to local substrate conditions as intensity independent mean values for regions less than 60 km.

Table 4-5 Provide a cross reference listing of the building design and external hazards.

Aircraft Crash

The following postulated loads are used as the basis for aircraft crash:

Impact load time history (for impact on a rigid wall)

Impact time (ms)	Impact load (MN)
0	0
10	55
30	55
40	110
50	110
70	0

Impact Area

The impact area is assumed to be a circle with an area of 7 m2. The angle of impact is assumed as being perpendicular to the tangent plane at the point of impact. If it is impossible for the aircraft to impact perpendicular to the tangent plane (e.g., due to the building layout, embedment of the building, terrain), the loads are established on the basis of the realistically possible impact directions. Reductions of the impact load due to the arrangement of multiple walls and the protective effect of structures acting as barriers in the line of impact, as well as the embedment of the building, soil coverage and the terrain, are taken into account.

Chemical Explosion

Design for blast waves resulting from chemical explosions are based on the following pressure profile that is independent of the explosion location (the blast wave can come from any direction) and if site-specific special feature(s) do not require another design layout:

- Rise of overpressure to 0.45 bar in 0.1 sec,
- Drop to 0.3 bar in 0.1 sec,
- All-round quasi-static load of 0.3 bar for an additional 0.8 sec.

4.6.3 Precautions Against Other External Hazards

High water

In conformance with KTA Safety Standard 2207, the design of nuclear power plants must account for flood water as follows:

- Design-basis flood
- 10-year flood (only in combination with operating basis earth-quake)

The following protection against high water is provided for the buildings listed in Table 4-5:

- The outside walls are constructed in watertight concrete to DIN 1045 up to the design-basis water level.
- Stability is verified for the design-basis water level.
- All accesses to the protected buildings are located above the design-basis water level.
- All pipe and cable penetrations below the design-basis water level are constructed watertight.

Intake of Toxic Gases

Toxic gases do not cause disturbances in the plant vital control and operating areas protected by the secured air handling and area isolation, but can incapacitate the operating personnel not located in the control room. A simultaneous, independent disturbance in the plant is not postulated.

If the plant personnel outside of the vital control areas are incapacitated during power operation, the engineered safety features keep the plant in a safe condition.

If an alarm is given, isolation of the HVAC systems in the reactor building, reactor auxiliary building and switchgear and emergency supply building can be initiated. Respiratory protective equipment with a compressed air supply is stored in the control room. This provides the personnel with an air supply that is not dependent on the ambient air.

Intake of Explosive Gases

The reactor building and remote shut-down station are protected against the ingress of gases that can form an explosive atmosphere.

The reactor auxiliary building is protected from the ingress of explosive gases by the design of the HVAC Systems. The air intake ports of the reactor auxiliary building are equipped with isolating dampers that are closed in the presence of explosive gases.

The gas hazard alarm sounds if signaled by the gas detection system for explosive gases or in response to administrative actions. The presence of explosive or combustible gases does not impair the plant; simultaneous emergency power operation is therefore not postulated for events involving explosive gases.

Intake of Corrosive Gases

Corrosive gases do not cause any short-term damage in the plant. Therefore it is always possible to shut down the plant and carry out inspections. Provisions for detection are not necessary.

If an alarm is given, isolation of the HVAC systems can be initiated in the reactor building, reactor auxiliary building and switchgear and emergency supply building.

Wind, Snow and Ice Formation

The effects of these loadings are considered in the design of the buildings.

Fire in the Vicinity

The arrangement of the power plant buildings and their construction at an appropriate distance from the site fence rule out the possibility of direct fire impacts from fires in the vicinity. For this reason, no features are provided apart from the measures taken to protect the plant against fuel fires after aircraft crash or against explosive gases.

Lightning

All buildings of the nuclear power plant are equipped with lightning protection systems to protect them from lightning strikes. In addition to arresting the lightning current and conducting it into the ground, its function is to protect the instrumentation and control equipment from unacceptable over-voltages.

The lightning protection system consists of external and internal lightning protection features.

External lightning protection features substantially reduce the effects inside buildings of lightning strikes. The external lightning protection system includes the following equipment: arresters, lightning conductors and ground conductors or connections to the power plant's central grounding system. In addition, buildings containing central I&C equipment are constructed in the manner of a fine-meshed Faraday cage to provide protection against outside interference fields.

Further internal lightning protection measures are central grounding of the neutral conductor (defined potential to ground), shield connection and voltage-limiting measures.

Sabotage

Technical and administrative organizational measures are taken to provide protection against sabotage or any other form of external interference.

Pursuant to German regulations in Article 3, Paragraph (2) of the Federal German Atomic Licensing Procedure Code, information on measures to protect the plant and operation of the plant against sabotage or any other form of external interference must be submitted separately.

Table 4-1: Reactor Core Nominal Data (Equilibrium Core)

Basic Data	
Reactor power (thermal)	200 MW
Mean power density	3 MW/m3
Core diameter	3 m
Mean core height	9.4 m
Primary system pressure	60 bar
Flow direction	downwards
Primary coolant temperature (inlet/outlet)	250 / 700°C
Fuel	low enrichment uranium (LEU)
Feed method	multiple recycle (MEDUL)
Equilibrium Core Nuclear Data	
No. of radial enrichment zones	1
No. of fuel element recycles (avg.)	approx. 15
Heavy metal charge	7 g/fuel element
No. of fuel elements in core	360,000
Integral fuel element residence time	1,007 EFPD
Fuel element dwell time	67 EFPD
Enrichment	8wt% ± 0.5
Mean discharge burnup	80,000 MWd/MT-U
Mean fuel element output	0.6 kW/fuel element
Maximum neutron fluence at side reflector after 32 full-load years)	($e \geq 0.1$ Mev) 1.8×10^{22} cm-2
Fuel inventory	
heavy metal (excluding fission products)	2,396 kg
fissionable materials	107 kg
1st shut-down system (reflector rods)	
Rods	6
Absorber length	4,800 mm
Absorber diameter	100 mm
Maximum travel	6,750 mm
Normal speed	1 cm/s
Absorber material	B_4C
2nd shut-down system	
Shut-down units	18
Small ball shut-down element diameter	10 mm
No. of small ball shut-down elements per unit	approx. 2.4×10^5
Absorber material	B_4C in graphite (10% B_4C by vol)

Table 4-2: Primary Circuit Components Principal Data (Nominal Values)

Pressure vessel	
Design pressure	70 bar
Nominal operating pressure	60 bar
Reactor pressure vessel height	approx. 25 m
Reactor pressure vessel inside diameter	approx. 5.9 m
Steam generator pressure vessel height	approx. 22 m
Steam generator pressure vessel inside diameter,	
upper portion	approx. 3.6 m
lower portion	approx. 3.2 m
Gas duct pressure vessel length	approx. 4.5 m
Gas duct pressure vessel inside diameter	approx. 1.5 m
Materials	
Pressure vessels	20 MnMoNi 5 5
Main steam nozzle	X 20 CrNiMo 12 1
Feedwater nozzle	20 MnMoNi 5 5
Steam generator	
Heat transfer capacity	202 MW
Primary coolant mass flow	approx. 85 kg/s
Primary coolant inlet temperature	700°C
Primary coolant outlet temperature	approx. 250°C
Primary coolant inlet pressure	60 bar
Feedwater temperature	max. 200°C
Main steam temperature	max. 530°C
Main steam pressure at SG outlet	max. 190 bar
Tube outside diameter in tube bundle	approx. 23 mm
Materials	
Steam generator tubes	X 10NiCrAlTi 32 20
Feedwater tube-sheet	X 10NiCrAlTi 32 20
Main steam tube-sheet	X 10NiCrAlTi 32 20
Primary gas circulator	
Total static head	1.50 bar
Hot gas duct	
Gas pipe inside diameter	approx. 750 mm
Support pipe inside diameter	approx. 1,000 mm
Insulation thickness	approx. 100 mm
Materials	
Gas pipe	X 10NiCrAlTi 32 20
Insulation	Al_2O_3 Fiber mats

Table 4-3: Classification of Pressure and Activity-Carrying Systems

Class MK 1 Systems
• Reactor pressure vessel
• Steam generator pressure vessel
• Gas duct pressure vessel
• Pressure relief system for primary system [1]
• Pressure equalizing system
• Helium purification system up to and including the primary system isolation valve
Class MK 2 Systems
Class MK 2a Systems
• Secured cooling system (cooling system for cooling loads with emergency power backup)
• Operational component cooling system (cooling system for operational cooling loads) [3]
• Feedwater piping system (including SG relief system) [6]
• Start-up and shut-down system [6]
• Main steam piping system [6]
• Start-up and shut-down steam system [6]
• Secured Service water system (service water system for secured cooling loads)
Class MK 2b Systems
• Irradiated fuel store
• Fuel feed and discharge systems [2]
• Transport equipment
• Small ball shut-down element feed system
• Helium purification system
• Regeneration system for helium purification system
• Water extraction system for helium supporting systems
• Liquid waste treatment system [4]
• Liquid waste storage system [4]
• Gaseous waste storage system
• Gas evacuation system for primary system, helium purification System and fuel handling equipment [5]
• Dump system for helium supporting Systems and fuel handling equipment

1) The system sections downstream of the safety valves are non-nuclear-classified.
2) The fuel charging equipment and the inspection line are non-nuclear-classified.
3) System sections from the pressure vessel unit up to and including the isolation valves and system sections that belong to the component surface cooler.
4) The subsystems chemicals station, centrifuge unit, agitators and seal water supply are non-nuclear-classified.
5) The train for evacuation of the primary system is non-nuclear-classified as this train is not connected to the primary system during power operation.
6) System sections up to and including the isolation valves (system sections inside the reactor building).

Table 4-4: Classification of Steelwork Items Quality Class

Type of steelwork items	MS 1	MS 2	NNC
Steel platforms	inside primary cavity	[1)]	for class II items [2)]
Piping supports	inside primary cavity	for MK 2 systems in the reactor building	for NNC systems and MK 2 systems outside reactor building
Component support structures	inside primary cavity	for MK 2 systems in the reactor building	for NNC component and MK 2 components outside reactor building
Protective and special structures	Special Specifications		
Anchoring items	inside primary cavity for MK 1 systems (system sections) outside primary cavity	for MK 2 systems in the reactor building	for NNC systems and MK 2 systems outside reactor building

1) Steel platforms, postulated failure of that can degrade the operability of seismic Class I items.

2) Refer to Section 4.6.1

Table 4-5: Design of Power Plant Building against External Hazards

	Design Basis Earthquake (2)	Aircraft Crash	Explosion Blast Wave
Reactor building	X	X	X
Reactor building annex	X	-	-
Reactor auxiliary building	(1)	-	-
Switchgear and emergency supply building	X	-	-
Cable ducts	X	-	-
Secured induced-draft cooling towers	X	-	-

X means considered

1) Seal structure in ground and the main load-bearing structures supporting the seal structure; the other main load-bearing structures are designed in accordance with DIN 4149 to be stable at the intensity of the safe shut-down earthquake.
2) Consequential loads are taken into account in the analyses for safe shut-down earthquake (e.g., rupture shock wave).

Figure 4-1: HTR-Module Radioactivity Release Barriers and Paths

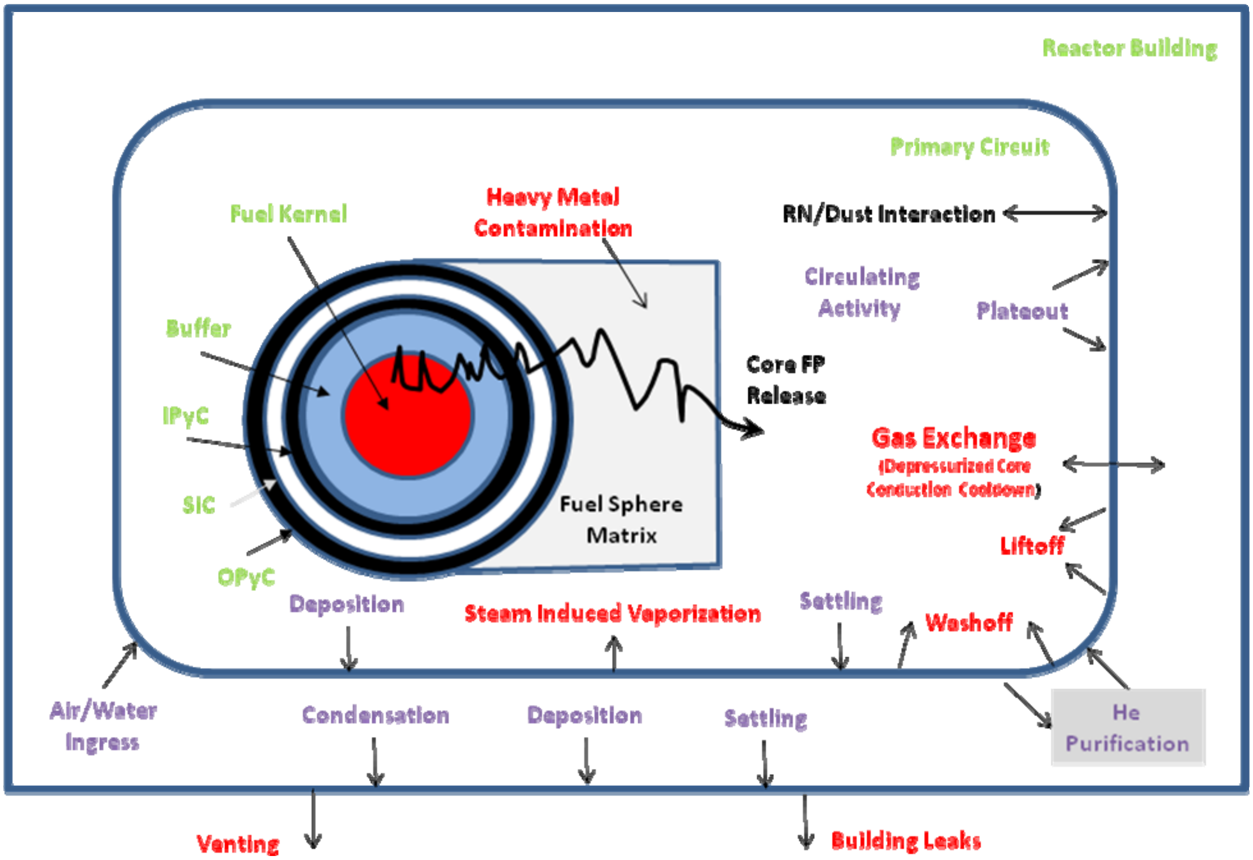


Figure 4-2: Maximum Failed Particle Fraction as a Function of Fuel Temperature

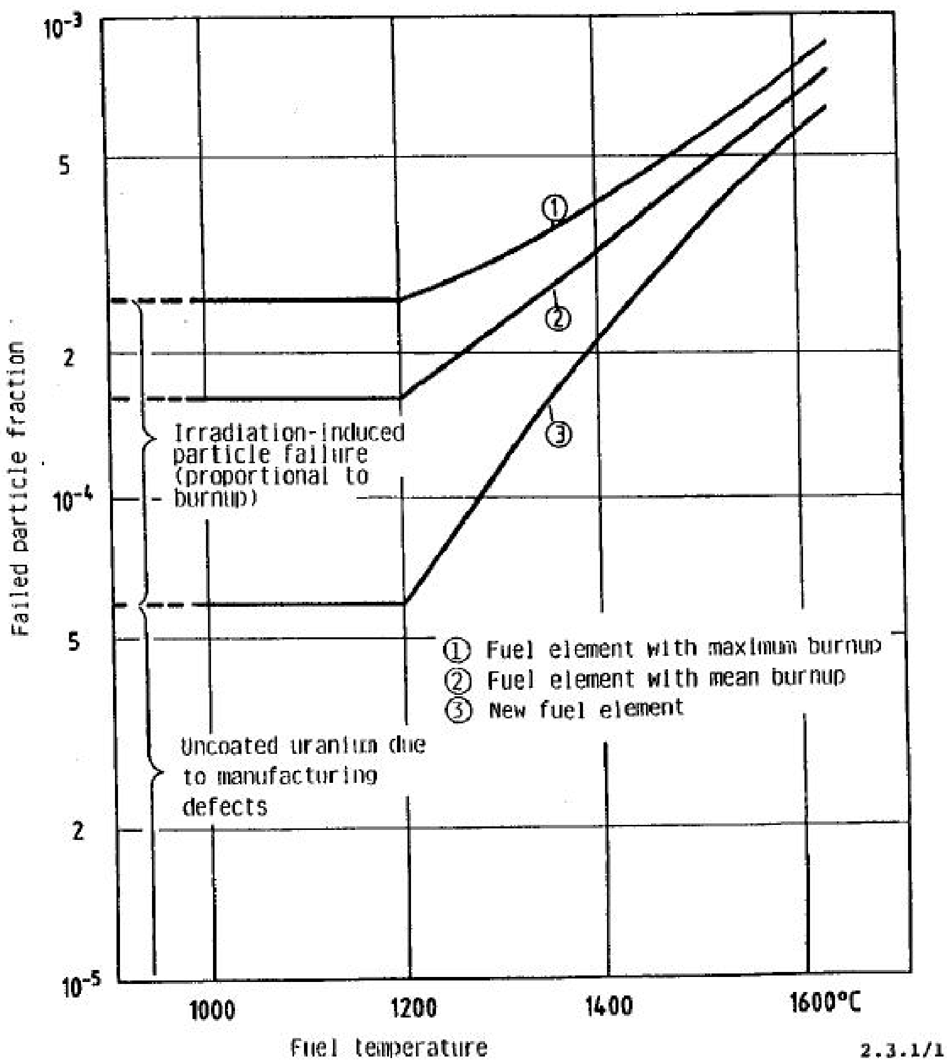


Figure 4-3: Primary Helium Flow Path

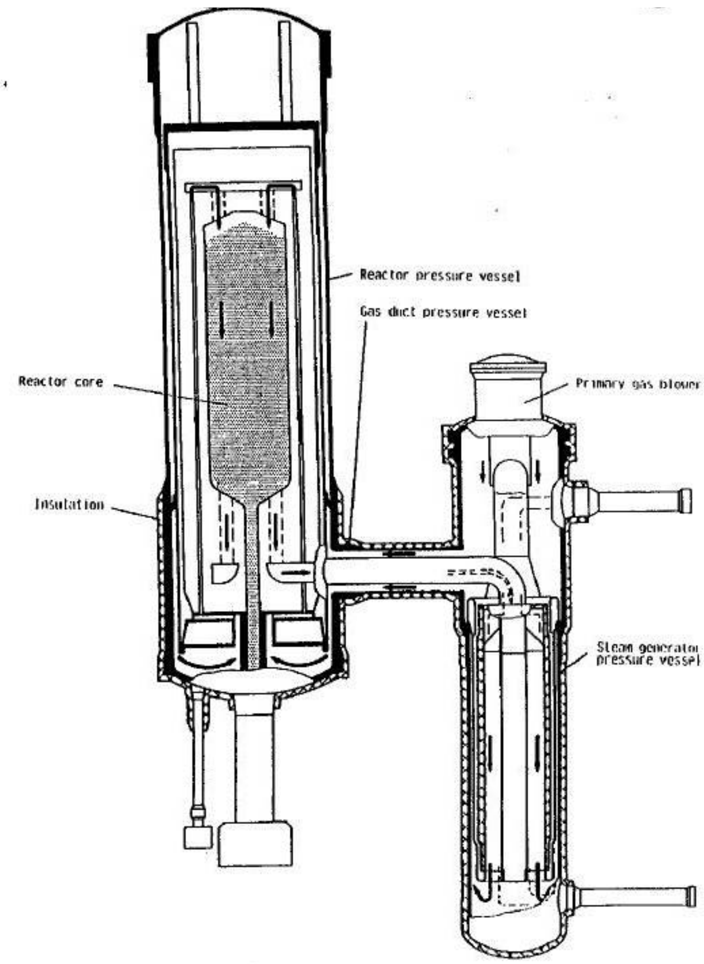
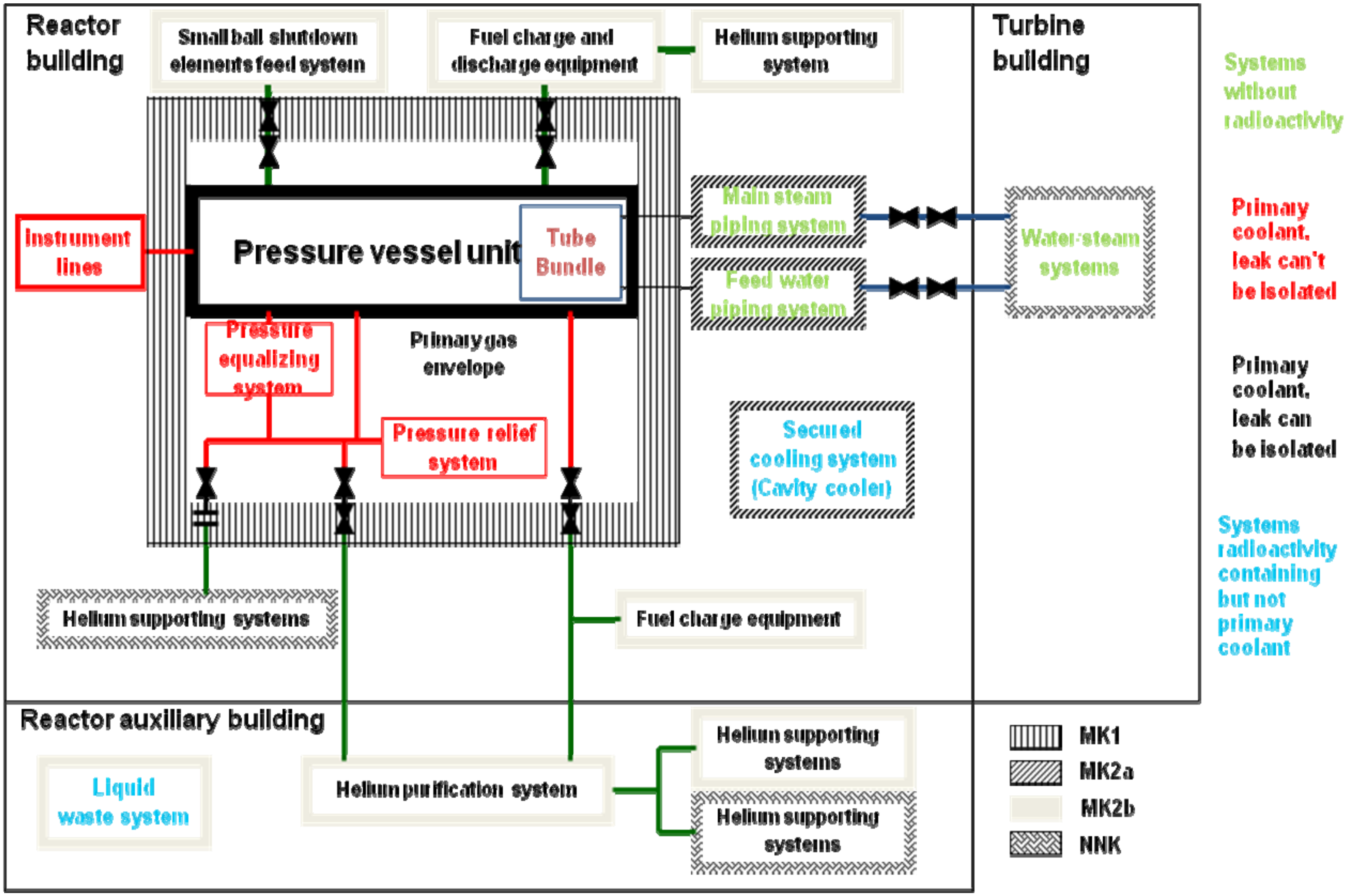


Figure 4-4: Nuclear Quality Classifications



5.0 TYPICAL PBR SAFETY EVENTS

5.1 Summary of Design Basis Events/Expected LBE Categories Defined

Nuclear power plants are designed for safe operation based on the concept of multiple layers of protection from accidents and their consequences. An accident is a sequence of events, the onset of which precludes further operation of the plant for safety reasons, but for which the plant has been designed (design basis accident). The multiple layers of protection are:

1) Accident prevention via a high quality plant, high quality conduct of operations and by control of operational malfunctions. Safety principles and the precautions taken to ensure safety require that the design and construction of the plant meet stringent technical requirements relating to quality.

2) Accident control by designing the plant to withstand a representative spectrum of postulated accidents. Regardless of the above requirements for quality, it is assumed that technical installations will still fail requiring supplementary engineered safety features to protect the public from the effects of accidents.

Accident analyses, originally performed for HTR-Module licensing in Germany, are discussed and described in the following sections. The actuating limits for the reactor protection system are discussed in Section 4.3.8.

The analyzed accidents selected for discussions in this study were chosen based on events and combinations of events that envelope the possible release of radioactive materials and the loadings sustained by components and systems for that category of events.

The spectrum of accident set selected for illustrations and discussion in this study is based on the HTR-Module original design basis accidents listed in Table 5-1, Table 5-2 and Table 5-3 . Also included are operational disturbances (upset operating conditions), that are relevant for component and system analyses and that trigger automatic counter measures.

Characteristic features of the HTR-Module power plant are that:

- the systems used for accident control are primarily used to keep radiological impacts below the allowable limits and to reduce component and system loadings, and
- sufficient time is generally available for repairs to be carried out or alternative actions to be taken.

This latter feature allows some systems to be activated manually for accident control, thus dispensing with initiation by the reactor protection system.

Some of the plant systems are consequently used for damage mitigation and are taken into account in the accident analysis if there is sufficient time to start up and, if necessary, repair these systems. The available time is apparent from the results of the accident analysis.

In the accident analyses, events and assumed failures are combined in compliance with the applicable rules and criteria (particularly the single failure criterion).

In the case of system malfunctions in which the parts are only exposed to temperature loadings, the analysis of these sequences of events primarily serves to establish service loading combinations, which are determined by the "frequent" events. Penalty factors are not taken into account in this context.

Taking penalty factors into consideration, loadings due to these events do not pose fatigue problems because such combinations of events are infrequent. Selection of a sufficiently ductile material together with appropriate strain limitation ensures that allowable limit loadings (inelastic strain due to temperature loadings) are not exceeded. The German regulatory authorities have accepted this position. The acceptance of this position by the U.S. NRC and the ASME is expected but has not been formally requested or granted.

The list of design basis accidents (originally postulated for the HTR-Module licensing - see Table 5-1 and Table 5-2) is based on the German Federal Accident Guidelines for Pressurized Water Reactor Nuclear Power Plants issued on October 18, 1983. As described later in this chapter, these are applied to the HTR-Module power plant where applicable and include specific plant information.

The calculation of the potential radiation exposure is based on the pertinent rules in the German "Incident Guidelines" for PWR nuclear power plants dated October, 18, 1983, with the necessary adjustments for the HTR-Module. The dose limits required in German regulation in Article 28, Paragraph 3 of the German Federal Radiological Protection Ordinance are not violated in the course of design basis accidents. In Chapter 7, results of an equivalent dose calculations using methodology used in the U.S.A. are presented.

Upset operating conditions that occur as a result of component or system malfunctions and that do not preclude further operation for safety reasons, are assigned to the category of "normal operation" and "anticipated operational occurrences" (pursuant to Article 45 of the German Federal Radiological Protection Ordinance).

Table 5-1 lists accidents that are of relevance because of their radiological impact on the environment and for which the plant must be designed. Accidents against which the letters RA are entered in this table are representative of the radiological impact on the environment and are analyzed by conducting calculations to verify compliance with the accident planning reference levels given in the German regulation - Article 28, Paragraph 3 of the Radiological Protection Ordinance.

Table 5-2 lists other design basis accidents for which the plant is designed. However, the measures taken render the radiological impact of these accidents on the environment as insignificant.

Table 5-3 lists events that are not considered as design basis accidents because of their low probability but for which measures are nevertheless taken to minimize risk.

The postulated accident and/or event groups are listed in the first column of the tables. The accident and/or event definitions given in the second column of the tables are governed by the design of the HTR-Module.

The third column of Table 5-1 and Table 5-2 states from which point of view the accident defined in the second column is considered:

- RA: Of all the radiologically significant accidents in Table 5-1, only the radiologically representative accidents are marked with "RA". The radiological impacts are calculated.
- AS: These accidents are analyzed for the purpose of designing engineered safety features or countermeasures.
- SI: The analysis of these accidents is used for the design of components or structural items for stability or integrity.
- VO: An accident analysis is not necessary due to the precautionary measures listed in the fourth column. The accident concerned is prevented or controlled by the precautionary measures.

5.2 Reactivity Events

In the HTR-Module, the following events cause an undesired increase of the reactivity (in some cases over an extended period of time) in the core thus impacting the reactor power:

- Inadvertent withdrawal of reflector rods or small absorber ball shut-down elements,
- Ingress of moderating materials into the core (water),
- Inadvertent increase in primary coolant mass flow (negative temperature coefficient of reactivity),
- Inadvertent decrease in of the cold gas temperature (negative temperature coefficient of reactivity),
- Compaction of the pebble bed due to earthquake.

It is also important to consider reactivity accidents caused by the inadvertent insertion of shut-down reactivity into the core.

The following are the criteria for initiation of the reactor protection system to control the above reactivity accidents or to minimize resulting additional component loadings.

The following will actuate reactor scram:

- Thermally corrected neutron flux greater than or equal to approximately 120%,
- Hot gas temperature greater than or equal to approximately 750°C,
- Cold gas temperature greater than or equal to approximately 280°C,
- Negative variable limit for thermally corrected neutron flux rate of change greater than or equal to approximately 20%/minute,
- Reactor period less than or equal to approximately 20 seconds,
- Intermediate-range neutron flux greater than or equal to maximum,
- Moisture in the primary system greater than or equal to approximately 800 vpm.

Reactor scram is actuated once the reactor protection limits are reached. As a result, the reflector rods drop into the lowest position, the primary gas circulator is tripped and the steam generator is isolated on the feedwater and steam side. These safety measures bring the plant to a subcritical operational mode.

Inadvertent withdrawal of reflector rods or small absorber ball shut-down elements is described in Section 5.2.1 for the equilibrium core and is shown to be an enveloping accident for other reactivity accidents.

5.2.1 Withdrawal of Reflector Rods or Small Ball Shut-down Elements

5.2.1.1 Event Description

Control system errors or mal-operation can cause withdrawal of shut-down elements at maximum speed resulting in a rapid reactivity increase at a rate that is dependent on the position and the worth characteristic of the shut-down element.

Withdrawal of shut-down elements is evaluated for the following initial conditions:

- Full load
- Part load
- Cold critical

Simultaneous withdrawal of all reflector rods is assumed for full load and part load, while withdrawal of the elements of one small absorber ball shut-down unit is assumed for the “cold critical” condition.

Full Load

An initial power of 210 MWt is assumed to be the maximum power level based on the limitation of the speed of the primary circuit circulator. Also, it is conservatively assumed that the primary coolant mass flow is at nominal. It is further assumed that the primary coolant mass flow and reactor inlet temperature do not change before the reactor protection system responds. This approach is conservative, since the main steam temperature control of the steam generator adjusts the mass flow downwards in the event of a power increase and hence has a damping effect on any further increase in reactor power through the negative temperature coefficient for reactivity.

During full-load operation, the position of the shut-down elements regulating the reactor depends on the load condition in which the reactor has previously been operated, i.e., on the extent to which the core has been influenced by the dynamic behavior of the neutron poison xenon 135. The magnitude of the reactivity feedback induced by a temperature increase in the fuel elements is also dependent on the xenon inventory of the core through the moderator temperature coefficient for reactivity.

The following are possible full-load operation initial conditions:

- Steady-state full-load equilibrium

  In this condition, the reactor core is in equilibrium with the fission products. This condition is attained after the reactor has been continuously operated at full load for several days. The 6 reflector rods provided for system control are positioned approximately 2.5 meters below the top edge of the pebble bed and bind 1.2% reactivity that is held in reserve for the requirements of part-load operation and hot start-up.
- Full load after return to power from part load

  After return to full power from part load, the reflector rods are temporarily inserted beyond the normal position in order to compensate for the burnout of the neutron poison xenon 135. The maximum reactivity to be compensated for occurs after the step load change from 50% part-load equilibrium, at which point the reflector rods temporarily bind a total of about 2.5% reactivity.
- Full load after zero load

  Maximum reactivity binding occurs here after raising the power of the xenon-free core to full load. In addition to the 1.2% reactivity bound by the reflector rods, the xenon reactivity that amounts to 3.2% is compensated for by small absorber ball shut-down units.

The reactor scrams due to thermally corrected neutron flux greater than or equal to approximately 120% or hot gas temperature greater than or equal to approximately 750°C. During the scram the reflector rods drop into their lowest position at an average speed of 50 cm/s. The primary gas circulator is tripped simultaneously.

Failure of one reflector rod to drop is conservatively assumed in the accident analysis. A very long period of 10 seconds is used for the response time of the hot gas temperature measurement. The effects of assumed failure of the first protective signal (thermally corrected neutron flux greater than or equal to approximately 120%) are demonstrated on this basis. As an example Figure 5-1and Figure 5-2 show the system response for the accident "withdrawal of all reflector rods at maximum speed at full-load equilibrium (210 MWt)."

Release of the reactivity bound in the reflector rods leads to an increase in reactor power) and, approximately 12 seconds after the 1st scram signal on "thermally corrected neutron flux greater than or equal to approximately 120%" has been reached, the reactor scrams (dashed curve).

In the event of assumed failure of the first scram signal, the neutron flux initially continues to rise. The fuel elements are heated up by that fraction of the reactor power that has not been removed so that the counter reactivity inserted on account of the negative temperature coefficient for reactivity dampens the power increase in the subsequent course of the accident. The second scram initiating criterion "hot gas temperature greater than or equal to approximately 750°C" is exceeded after about 80 seconds (see Figure 5-2) and a scram is actuated. During this period, the maximum fuel temperatures rise by about 70K to approximately 950°C (disregarding hot spots), and are therefore within a temperature range that has been validated for fuel elements on the basis of operating experience. The primary coolant outlet temperature remains below the design values for the metallic parts of the steam generator (see Figure 5-2).

Apart from some slight differences, reactivity accidents caused by withdrawal of all reflector rods or small absorber ball shut-down elements at the other full-load conditions described above have the same impacts as accidents. The second scram initiating criterion "hot gas temperature greater than or equal to approximately 750°C" limits the respective maximum fuel temperature increase to approximately 100K.

Slight differences only arise due to:

- radial deformation of the temperature profile for initial conditions with deeply inserted reflector rods, and
- higher power release in the lower core regions during the emptying of a small ball shut-down unit

The worst cases are "Withdrawal of all reflector rods at full load after return to full power from the equilibrium condition at 50% nominal load (xenon equilibrium)" and "Emptying a small ball shut-down unit in a Xenon free full load state after a cold start-up." The margins add to the maximum fuel temperature are approximately 20K respectively 50K.

Part Load

New xenon equilibrium sets in during long-term part load operation that is below the full-load level. An unrestricted load change is only possible between nominal load and 50% part load due to the restricted reactivity hold out of the nominal full load core. Out of all imaginable initial states, the xenon equilibrium state at 50% part load and a maximum decreased hot gas temperature is the worst regarding the failure of shut-down elements because:

- The reflector rods for compensation of the xenon and temperature reactivity are in their lowest entry position ($\Delta\rho = 2\%$).
- The operational parameters of the initial state (thermally corrected neutron flux, hot gas temperature) are furthermost from the limit values of the reactor protection system.

The results of the accident analysis for the case "withdrawal of all reflector rods" are shown for this initial state in Figure 5-3and Figure 5-4. The release of the bound reactivity in the reflector rods leads to an increase of the reactor power (see Figure 5-3) and after approximately 30 seconds to the scram of the reactor (dashed curve shape) by reaching the reactor scram criterion - "thermally corrected neutron flux greater than or equal to 120%".

On postulated failure of this signal, the reactor scram takes place after approximately 200 seconds when the second reactor scram criterion "hot gas temperature greater than or equal to 750°C" has been exceeded. The maximum fuel element temperatures increase up to approximately 1070°C and based on operating experience, are in an acceptable temperature range for fuel elements (see Figure 5-4).

In addition to the unrestricted load change it is possible with a certain operation to shut down the reactor to a part load level down to 20% of the nominal load by very slow load decrease in hours or days and with the help of the small ball shut-down units for reactivity compensation. The worst initial state is in the xenon equilibrium at 20% nominal power and a maximum decreased hot gas temperature of 575°C because:

- The maximum possible reactivity is bound by the shut down elements ($\Delta\rho = 2.5\%$).
- The operational parameters (thermally corrected neutron flux and hot gas temperature) are furthermost from the limit values of the reactor protection system.

The reactor protection system initiating criterion "hot gas temperature greater than or equal to 750°C" will be reached after approximately 600 seconds for this initial state. Then, the maximum fuel temperature increases up to a value of approximately 1200°C.

However, the first reactor protection system initiating criterion intervenes after approximately 11 seconds (period less than or equal to 20 seconds) and after approximately 40 seconds the second initiating criterion (thermally corrected neutron flux greater or equal to 120%) prevents the increase of the maximum fuel element temperature to values above 650°C (see Figure 5-5 and Figure 5-6).

Cold Critical (start-up accident)

Following an operational reactor shut-down or one initiated by the reactor protection system, the reactor core is cooled after approximately 1.5 hours by removing the generated decay heat via the main heat transfer system. Due to decay of the absorbing, unstable isotopes, reactivity is subsequently released because of the cold to hot margin and this, in turn, is bound by the small absorber ball shut-down elements. The maximum reactivity to be compensated for by the small absorber ball shut-down elements amounts to 6.6% for the reactor in the cold critical condition. For start-up from the cold condition it may be assumed that, one of the small absorber ball shut-down units that is still inserted when the core goes critical will continue to be withdrawn at maximum speed as a result of mal-operation or errors in the control system.

Up to a power of 1% the reactor scrams due to intermediate-range neutron flux greater than or equal to maximum and reactor period less than or equal to approximately 20 seconds.

In this event, the reflector rods drop into the lowest position and the primary gas circulator is tripped. Countermeasures of the reactor protection system were not assumed in this analysis to keep it as clear as possible. Figure 5-3 shows the power-versus-time curve for this case. The points during the event at which the scrams are initiated are shown in the figure.

5.2.1.2 Barrier Performance

For the full load case, the maximum fuel temperatures rise by about 70K to approximately 950°C (disregarding hot spots) and are therefore within a temperature range that has been validated for fuel elements on the basis of operating experience.

For the part load case, the maximum fuel element temperatures increase up to approximately 1070°C and, based on operation experience, are in an acceptable temperature range for fuel elements.

For the cold critical case, the fuel temperature response is bounded by the "full load" and "part load" cases.

In all cases the reactor vessel is not challenged because the main steam temperature control of the steam generator adjusts the mass flow downwards in the event of a power increase and hence has a damping effect on any further increase in reactor power through the negative temperature coefficient for reactivity.

The reactor building is not affected because there are no releases from the reactor.

5.2.1.3 Radiological Consequences

An evaluation of the radiological consequences is not required for this event since no radioactive material is released from the fuel.

5.2.1.4 Assessment

A review of the HTR-Module reactivity events analysis results indicates that:

Accident scenarios: A complete set of accident scenarios were examined

Dose assessment: The reactivity events do not have any dose consequences

Prevention and Mitigation: Adequate prevention and mitigation measure are in place to address HTR-Module reactivity events

Conclusions: The HTR-Module design meets the intent of the current and anticipated U.S.A. regulatory requirements for reactivity insertion accidents

5.2.2 Water Ingress into the Primary System

Refer to Section 5.6.

5.2.3 Inadvertent Operation of a Small Ball Shut-down Unit or a Reflector Rod

5.2.3.1 Event Description

Inadvertent Drop of Small Ball Shut-down Elements

Inadvertent drop of small ball shut-down elements causes a decrease in reactor power and, on fast filling of the side reflector column, leads to a reactor shut-down when scram is initiated due to "negative variable limit for thermally corrected neutron flux greater than or equal to approximately 20%/min" (Figure 5-7).

In the event of slow filling of the side reflector column with the controls intact the above scram criterion is not met, and inadvertent drop of the small absorber ball shut-down elements can be detected on the basis of the:

- off-standard reflector rod position,
- annunciation “vessel closure or storage vessel open”, and
- azimuthally unbalanced load monitor.

The reactor is shut down if the fault cannot be corrected shortly after detection.

Inadvertent Insertion or Withdrawal of One Reflector Rod

The accident sequences are the same for inadvertent insertion or inadvertent withdrawal of one reflector rod.

For rod insertion and simultaneously assumed failure of the control system, reactor shut-down results from a scram initiated due to “negative variable limit for thermally corrected neutron flux greater than or equal to approximately 20%/min”.

For rod withdrawal, scram is initiated on:

- Thermally corrected neutron flux greater than or equal to approximately 120%, or
- Hot gas temperature greater than or equal to approximately 750°C.

The above scrams do not occur if the control system remains operable. Assuming a completely inserted or withdrawn reflector rod, the resulting local hot gas temperature change is bout 10K and 30K respectively. The mixing facilities downstream of the core make the additional temperature gradients acceptable for the heat removing components. Reactor shut-down is not necessary.

5.2.3.2 Barrier Performance

There are no unacceptable temperature or pressure loadings that occur in the plant even if scram fails to initiate as required.

5.2.3.3 Radiological Consequences

An evaluation of the radiological consequences is not required for this event since no radioactive material is released from the fuel.

5.2.3.4 Assessment

A review of the safety analysis performed for the inadvertent operation of small ball shut-down units or reflector rod indicates that:

Accident scenarios: The event scenarios examined are complete

Dose assessment: The resultant reactivity insertion events do not have any dose consequences

Prevention and Mitigation: Adequate prevention and mitigation measure are in place to address these events

Conclusions: The HTR-Module design meets the intent of the current and the anticipated U.S.A. regulatory requirements for inadvertent rod insertion accidents

5.2.4 Inadvertent Over-speeding of the Primary Gas Circulator

5.2.4.1 Event Description

Inadvertent over-speeding of the primary gas circulator is possible during part-load operation, start-up and shut-down operations and hot start-up of a reactor. In this case, reactivity is inserted into the reactor core because of the negative temperature coefficient, leading to distinct power excursions that differ depending on the initial reactor condition.

As an example, assume this event occurs at the lowest feasible part-load condition (50% of nominal mass flow) and that the primary coolant mass flow increases within approximately 7 seconds to 85% and within an additional 7 seconds to the maximum of 103% of the nominal mass flow. After about 3 seconds scram occurs on "mass flow ratio (primary to secondary side) greater than or equal to 1.3". If scram should fail, the reactor power continues to rise (Figure 5-8) and the average hot gas temperature drops simultaneously due to the excessive primary coolant mass flow (Figure 5-9). The steam generator is unable to remove the increased heat supply. Consequently the cold gas temperature rises. After about 10 - 20 seconds, this leads to a reactor scram on cold gas temperature greater than or equal to 280°C."

5.2.4.2 Barrier Performance

Unacceptable fuel element and component pressure and temperature loadings do not occur even if there is a failure to initiate scram as required.

5.2.4.3 Radiological Consequences

An evaluation of the radiological consequences is not required for this event since no radioactive material is released from the fuel.

5.2.4.4 Assessment

A review of the safety analysis performed for the inadvertent over-speed of the primary gas circulator indicates that:

Accident scenarios: The event scenarios examined are complete

Dose assessment: The resultant reactivity insertion events do not have any dose consequences

Prevention and Mitigation: Adequate prevention and mitigation measure are in place to address these events

Conclusions: The HTR-Module design meets the intent of the current and anticipated U.S.A. regulatory requirements for inadvertent over-speed of primary gas circulator event

5.2.5 Maximum Decrease in Cold Gas Temperature

5.2.5.1 Event Description

Inadvertent increases in feedwater flow during part-load operation, start-up and shut-down operation, and hot start-up can cause the cold gas temperature to decrease. Consequently, reactivity is inserted into the reactor through the negative temperature coefficient for reactivity. This leads to an increase in reactor power in the event of concurrent postulated failure of the plant controls. This will result in a reactor scram on either:

- Mass flow ratio (primary to secondary side) is less than or equal to 0.75.
- Hot leg gas temperature is greater than or equal to approximately 750°C.

5.2.5.2 Barrier Performance

Effects of the inadvertent increase in feedwater flow events on the core power and temperature response (component loadings) are enveloped by the "Inadvertent Withdrawal of the Reflector Rods" design basis accident described in Section 5.2.1.

5.2.5.3 Radiological Consequences

An evaluation of the radiological consequences is not required for this event since no radioactive material is released from the fuel.

5.2.5.4 Assessment

A review of the safety analysis performed for the maximum decrease in cold gas temperature event indicates that:

Accident scenarios: The event scenarios examined are complete

Dose assessment: The resultant reactivity insertion event is bounded by the inadvertent withdrawal of the reflector rods event and does not have any dose consequences

Prevention and Mitigation: Adequate prevention and mitigation measure are in place to address these events

Conclusions: The HTR-Module design meets the intent of the current and anticipated U.S.A. regulatory requirements for the maximum decrease in cold gas temperature event

5.2.6 Pebble Bed Compaction Due to Earthquake

5.2.6.1 Event Description

An earthquake can cause an increase in the pebble bed fill factor, i.e., compact the pebble bed, resulting within a short time a reactivity insertion in the core due to a reduction of neutron leakage from the pebble bed and as a result of movement of the pebble bed surface relative to the reflector rods.

Figure 5-10 shows the reactivity increase from each of these two components on the pebble bed fill factor.

For an assumed horizontal earthquake acceleration of 0.5 g (g = 9.81 m/s2), the pebble bed packing fraction increases from 0.61 to 0.614 within approximately 6 seconds at constant excitation. The inserted reactivity amounts to 0.125% with 0.05% due to movement of the pebble bed surface relative to the reflector rods.

Scram is initiated on:

- Periods are less than or equal to 20 seconds
- Thermally corrected neutron flux greater than or equal to approximately 120%.

Figure 5-11 shows the reactor power-versus-time curve for this accident assuming scram is not initiated.

However, the points at which the scram would be initiated are indicated in the figure. The first scram initiating criteria is reached after approximately 2 seconds, the second criterion after approximately 4 seconds. The figure also shows that, even if a scram is not initiated, the long-term power increase would be extremely low. The hot gas temperature increase would be about 50K after approximately 500 seconds. The resulting maximum fuel temperature is in a temperature range that has been validated for fuel elements by operating experience. For the pebble bed density the change from standard sphere package density of 0.61 to 0.614 was confirmed.

5.2.6.2 Barrier Performance

The resulting maximum fuel temperature is in a temperature range that has been validated for fuel elements by operating experience.

5.2.6.3 Radiological Consequences

An evaluation of the radiological consequences is not required for this event since no radioactive material is released from the fuel.

5.2.6.4 Assessment

A review of the safety analysis performed for the pebble bed compaction due to an earthquake event indicates that:

Accident scenarios: The event scenario examined is complete

Dose assessment: The resultant reactivity insertion event does not have any dose consequences

Prevention and Mitigation: Adequate prevention and mitigation measure are in place to address these events

Conclusions: The HTR-Module design meets the intent of the current and the anticipated U.S.A. regulatory requirements for the reactivity excursion due to the pebble bed compaction caused by an earthquake

5.3 Main Heat Transfer Malfunction Events

5.3.1 Loss of Auxiliary Power Supply

5.3.1.1 Event Description

Loss of the auxiliary power supply leads to failure of the primary gas circulator and the feedwater pumps and results in a reactor scram due to either mass flow ratio (primary to secondary side) greater than or equal to 1.3, or negative variable limit for thermally corrected neutron flux greater than or equal to approximately 20%/min.

The first scram signal (mass flow ratio) is triggered because the feedwater mass flow decreases much more quickly than the primary coolant mass flow. Mass flow patterns of relevance for the sustained system behavior begin simultaneously; same as in "manual scram" (see Section 3.3.4). Differences may occur in the subsequent mass flow pattern because the feedwater decrease is faster in this case (due to failure of the feedwater pumps) than in the “manual scram” case (due to isolation valve closure). This leads to somewhat more intensive short-term heat-up of the steam generator, that is further intensified, in contrast to the “manual scram” case, because the circulator damper only closes when the emergency power supply becomes available, i.e., after diesel start-up. Consequently, after loss of the auxiliary power supply, steam generator behavior deviates slightly from the “manual scram” case in that the steam generator tends to heat up to a greater extent.

The failure of the primary-gas circulator triggers the second scram signal given above (thermally corrected neutron flux) due to the reaction of the core to the loss of primary coolant mass flow. Even if the malfunction is not detected until the second scram signal is triggered it does not alter the accident-induced loss of the two mass flows (primary, secondary). The mass flow patterns that are of significance for the temperature transients in the steam generator are the same or slightly different as the mass flow patterns detected for the first scram signal. Therefore, the temperature transients occurring in the primary system and water/steam system are the same or almost identical to the first scram signal case above.

If power supply from the grid is restored within one hour of loss of auxiliary power, hot start-up or residual heat removal via the main heat transfer system can be initiated. If auxiliary power is cut off for a longer period, the plant continues residual heat removal operation by the cavity cooler.

5.3.1.2 Barrier Performance

In the long term, the core reacts in the same way as for the “manual scram” case. Therefore, the barrier performance is the same as described for the manual scram in Section 3.3.4.

5.3.1.3 Radiological Consequences

An evaluation of the radiological consequences is not required for this event since no radioactive material is released from the fuel.

5.3.1.4 Assessment

A review of the safety analysis performed for the loss of auxiliary power supply event indicates that:

Accident scenarios: The event scenario examined is complete

Dose assessment: The resultant plant response does not have any dose consequences

Prevention and Mitigation: Adequate prevention and mitigation measure are in place to address this event

Conclusions: The HTR-Module design meets the intent of the current and anticipated NRC regulatory requirements for the loss of auxiliary power supply

5.3.2 Loss of Primary Coolant Mass Flow

5.3.2.1 Event Description

The loss of primary coolant mass flow event is caused by either failure of the primary gas circulator motor or inadvertent closure of the circulator damper resulting in a scram due to either mass flow ratio (primary to secondary side) less than or equal to 0.75 or negative variable limit for thermally corrected neutron flux greater than or equal to approximately 20%/min.

If an initial reactor scram is due to the mass flow ratio criteria, the scram and the accompanying protective actions take place so quickly that the mass flow patterns of relevance to the sustained loadings only experience shifts in the range of seconds as compared to the normal course of reactor scram. The steam generator temperature transients are therefore almost identical to the "manual scram" case described in Section 3.3.4.

If an initial reactor scram is due to the thermally corrected neutron flux criteria, the reactor scram and shut-down of the feedwater mass flow are significantly delayed. As a result, the steam generator is temporarily subcooled and simultaneously filled on the water side. In this phase until both cooling fluids have come to a standstill (after the protective actions have taken effect), the temperatures over the entire length of the steam generator and, in particular, the two outlet temperatures drop.

The cold gas outlet temperature adjusts to the feedwater temperature. Once flow has reached a standstill, the steam generator remains in an almost steady-state condition, apart from the temperature equalizing process between tubes and cooling fluids.

If the components causing the accident regain operability within one hour of failure, hot start-up or residual heat removal via the main heat transfer system can be initiated. If the fault cannot be eliminated within a short time, the plant continues in residual heat removal operation by the cavity cooler for a lengthy period of time.

5.3.2.2 Barrier Performance

In the long term, the core reacts in the same way as for the "manual scram" case. Therefore, the barrier performance is the same as described for the manual scram in Section 3.3.4.

5.3.2.3 Radiological Consequences

An evaluation of the radiological consequences is not required for this event since no radioactive material is released from the fuel.

5.3.2.4 Assessment

A review of the safety analysis performed for the loss of primary coolant mass flow event indicates that:

Accident scenarios: The event scenario examined is complete

Dose assessment: The resultant plant response does not have any dose consequences

Prevention and Mitigation: Adequate prevention and mitigation measure are in place to address this event

Conclusions: The HTR-Module design meets the intent of the current and the anticipated U.S.A. regulatory requirements for the loss of primary coolant mass flow

5.3.3 Loss of Feedwater Flow

5.3.3.1 Event Description

The loss of feedwater flow event is caused by either the failure or trip of the feedwater pumps, or the closure of feedwater valves resulting in a scram due to either mass flow ratio (primary to secondary side) greater than or equal to 1.3, or cold gas temperature greater than or equal to approximately 280°C.

If the initial reactor scram is due to the mass flow ratio criteria, then the scram and the accompanying protective actions take place so quickly that the mass flow patterns of relevance to the sustained loadings only experience shifts in the range of seconds as compared to the normal course of reactor scram. The steam generator temperature transients are therefore almost identical to the "manual scram" case described in Section 3.3.4.

If the initial reactor scram is due to the cold gas temperature criteria, the reactor scram and primary gas circulator trip are significantly delayed. As a result, the steam generator is temporarily overheated, i.e., it dries out. In this phase until both cooling fluids have come to a standstill (after the protective actions have taken effect), there is a rise in temperatures over the entire length of the steam generator, and in particular, for the two outlet temperatures. The main steam temperature adjusts to the hot gas temperature. Once flow has reached a standstill, the steam generator remains in an almost steady-state condition, apart from the temperature equalizing process between tubes and cooling fluids.

If the water supply to the steam generator can be restored within one hour of the onset of the malfunction, hot start-up or residual heat removal via the main heat transfer system can be initiated. If the fault cannot be eliminated within a short time, the plant continues in residual heat removal operation by the cavity cooler for a lengthy period of time.

5.3.3.2 Barrier Performance

In the long term, the core reacts in the same way as the "manual scram" case. Therefore, the barrier performance is the same as described for the manual scram in Section 3.3.4.

5.3.3.3 Radiological Consequences

An evaluation of the radiological consequences is not required for this event since no radioactive material is released from the fuel.

5.3.3.4 Assessment

A review of the safety analysis performed for the loss of feedwater flow to the steam generator event indicates that:

Accident scenarios: The event scenario examined is complete

Dose assessment: The resultant plant response does not have any dose consequences

Prevention and Mitigation: Adequate prevention and mitigation measure are in place to address this event

Conclusions: The HTR-Module design meets the intent of the current and the anticipated U.S.A. regulatory requirements for the loss of steam generator feedwater flow

5.3.4 Inadvertent Closure of a Main Steam Valve

5.3.4.1 Event Description

After inadvertent closure of a main steam valve the pressure in the affected steam generator increases from the operating point (approximately 190 bar) to the actuation setpoint of the main steam safety valve (208 bar) at which point the valve is actuated to blow down the total main steam flow intermittently.

There are practically no feedback effects on the primary system of the affected reactor as the steam generator initially continues to be fully cooled. The effect on the water/steam cycle (flow rate drop to 50% in the main steam supply system of the overall plant) and its reaction are the same as for the “manual scram” case (Section 3.3.4).

Automatic shut-down of the affected reactor does not take place directly. If it is not possible to reopen the valve, either “manual scram” or operational cooldown of the reactor is initiated.

Assuming that no manual actions are taken, the feedwater supply is pumped dry. This causes feedwater pump trip (by unit equipment protection circuits) and also automatic reactor scram when the initiating criterion “mass flow ratio (primary to secondary side) greater than/ equal to 1.3” is met.

5.3.4.2 Barrier Performance

In the long term, the core reacts in the same way as for the “manual scram” case. Therefore, the barrier performance is the same as described for the manual scram in Section 3.3.4.

5.3.4.3 Radiological Consequences

An evaluation of the radiological consequences is not required for this event since no radioactive material is released from the fuel.

5.3.4.4 Assessment

A review of the safety analysis performed for the inadvertent closure of a main steam valve event indicates that:

Accident scenarios: The event scenario examined is complete

Dose assessment: The resultant plant response does not have any dose consequences

Prevention and Mitigation: Adequate prevention and mitigation measure are in place to address this event

Conclusions: The HTR-Module design meets the intent of the current and the anticipated U.S.A. regulatory requirements for the inadvertent closure of the a main steam valve

5.3.5 Main Steam Extraction Malfunction

5.3.5.1 Event Description

In addition to a main steam line break (discussed in Section 5.7.2), excessive main steam loss events could be due to inadvertent opening of a turbine valve, or inadvertent opening of a main steam bypass valve.

In the inadvertent opening of a turbine valve event, the second turbine is throttled back by the controls, while in the inadvertent opening of a bypass valve both turbines are throttled back by the controls. The cooling of the steam generator is not impacted, so that the malfunction has practically no influence on the primary circuits.

Only at a much lower power level (less than 50%) is it possible for a reactor scram to occur due to “negative variable limit for steam pressure”.

The “decrease in main steam extraction event” can occur only when one turbine generator set is in operation, its condenser low vacuum trip gear responds and the resulting excess steam is not accepted by the process steam system. Depending on the available excess power, the main steam pressure increases either rapidly or slowly until actuation of the main steam safety valve. The continuation of the sequence of this event, closure of the main steam valve, is described in Section 5.3.4 – inadvertent closure of main steam valve.

5.3.5.2 Barrier Performance

In the long term, the core reacts in the same way as for the “closure of main steam valve” event. Therefore, the barrier performance is the same as described for the closure of main steam valve event in Section 5.3.4.

5.3.5.3 Radiological Consequences

An evaluation of the radiological consequences is not required for this event since no radioactive material is released from the fuel.

5.3.5.4 Assessment

A review of the safety analysis performed for the loss of main steam events indicates that:

Accident scenarios: The event scenario examined is complete

Dose assessment: The resultant plant response does not have any dose consequences

Prevention and Mitigation: Adequate prevention and mitigation measure are in place to address this event

Conclusions: The HTR-Module design meets the intent of the current and the anticipated U.S.A. regulatory requirements for the loss of main steam events

5.3.6 Process Steam Extraction Malfunctions

5.3.6.1 Event Description

Process steam extraction malfunctions affect main steam extraction that is covered by the malfunction events described in Section 5.3.5 – loss of main steam events.

5.3.6.2 Barrier Performance

This is covered by the barrier performance described in Section 5.3.5.

5.3.6.3 Radiological Consequences

This is covered by the radiological consequences described in Section 5.3.5.

5.3.6.4 Assessment

This is covered by the assessment described in Section 5.3.5

5.3.7 Inadvertent Opening of Valves in the Water/Steam Cycle

5.3.7.1 Event Description

If the inadvertent opening of valves in the water/steam cycle does not lead to loss of water from the water/steam cycle, then it will only result in efficiency losses. In the worst case, it results in the actuation of protective devices of machines or components.

Inadvertent opening of a valve through which the circuit can be drained leads to a feedwater deficiency that then causes operational tripping of all feedwater pumps.

The continuation of the loss of feedwater event is described in Section 5.3.3.

5.3.7.2 Barrier Performance

In the long term, the core reacts in the same way as for the "loss of feedwater" event. Therefore, the barrier performance is the same as described for the loss of feedwater event in Section 5.3.3.

5.3.7.3 Radiological Consequences

An evaluation of the radiological consequences is not required for this event since no radioactive material is released from the fuel.

5.3.7.4 Assessment

This is covered by the assessment described in Section 5.3.3

5.3.8 Turbine Trip

5.3.8.1 Event Description

Under unacceptable operating conditions that could jeopardize the turbine generator set, the turbine trip gear initiates a turbine trip and shuts down the turbine generator set.

The turbine trip can be actuated both hydraulically and electrically and quickly leads to closure of isolation valves on the turbine side.

In addition to being initiated by the automatic trip gear, turbine trip can be actuated by manually, either by operating the trip lever on the turbine or the remote electrical trip from the control room.

The main steam that is no longer required by the turbines is dumped through the main steam bypass systems.

Turbine trip does not normally lead to a reactor scram as this malfunction can be compensated for by the unit coordinating controls.

5.3.8.2 Barrier Performance

Not evaluated because there is very little change in the core response characteristics of temperature and pressure.

5.3.8.3 Radiological Consequences

An evaluation of the radiological consequences is not required for this event since no radioactive material is released from the fuel.

5.3.8.4 Assessment

A review of the safety analysis performed for the turbine trip event indicates that:

Accident scenarios: The event scenario examined is complete

Dose assessment: The resultant plant response does not have any dose consequences

Prevention and Mitigation: Adequate prevention and mitigation measure are in place to address this event

Conclusions: The HTR-Module design meets the intent of the current and the anticipated U.S.A. regulatory requirements for the turbine trip event

5.4 Primary System Depressurization Events

Possible initiating events for this accident can either be leaks or breaks in a system connected to the pressure vessel unit (either upstream or downstream of the primary system isolation valves) or inadvertent opening of a safety valve in the primary side pressure relief system.

Breaks in the pressure vessel unit itself are not assumed because of the extensive quality assurance measures taken (see Section 4.5).

Loss of primary coolant results in a decrease in primary system pressure that initiates a reactor scram due to "negative sliding limit for primary system pressure greater than or equal to approximately 180 mbar/min". This results in the following reactor protection actions:

- reflector rod drop,
- primary gas circulator trip,
- closure of the main steam and feedwater isolation valves, and
- closure of the primary system isolation valves.

In the event of small losses of coolant that does not result in the above reactor protection system actions because the initiating criterion is not met or there is an assumed failure of the above initiating criterion, the reactor is shut down due to "mass flow ratio (primary to secondary side) less than or equal to 0.75." Although the primary system isolation is not actuated in this case, the effects of the no primary system isolation event are covered by the effects resulting from non-isolable breaks described later in this section.

If activity is detected in the exhaust air, the plant is automatically switched over to the secured sub-atmospheric pressure system to ensure filtered release of small leakages. This measure is taken to minimize environmental impact.

On the whole, only a few pipes are connected to the pressure vessel unit as follows:

• Fuel forwarding line (one)	DN65
• Fuel carrier gas recirculation line (one)	~DN50
• Fuel discharge lines (two)	DN65
• Failed fuel discharge lines (two)	DN125
• Carrier gas lines of small ball shut-down system (three)	DN65
• Pressure equalization line (one)	DN65
• Oil supply lines for primary gas circulator (three)	DN50
• Connecting lines of helium purification system (two)	DN65

In addition, there are some instrument lines with inside diameters not exceeding 10 mm.

Apart from the failed fuel discharge lines, all primary system connections have diameters not exceeding DN65. The possible free cross section of the failed fuel discharge line is reduced to such an extent by the upstream single-exit gate disc that the maximum attainable blowdown rate is that which could occur for a break in a DN65 pipe at the worst position.

The lines of the fuel handling equipment and small ball shut-down system are connected to RPV nozzles in the fuel discharge compartment. Two isolation valves are installed directly at the end of each nozzle (one actuated by the reactor protection system); consequently the primary system can be directly isolated at the pressure vessel unit if breaks occur in connecting systems.

Only the connections of the pressure equalizing system, the helium purification system, the oil supply for the primary gas circulator and the instrument lines are located inside the primary cavity.

Outside the primary cavity, the inlet and outlet lines of the helium purification system and the oil supply are each equipped with one manual isolation valve and one primary system isolation valve actuated by the reactor protection system.

The isolation valves of the instrument lines are located outside the primary cavity. In the event of primary system depressurization, these valves are not closed by the reactor protection system; so that the instruments continue to supply information on the condition of the reactor after the accident.

All pipes connected to the pressure vessel unit are accessible for inspections and, if necessary, repairs. The lines connected to the nozzles in the fuel discharge compartment are accessible during normal operation, and those running through the steam generator cavity are accessible approximately one day after shut-down.

The most likely causes for primary system depressurization are breaks in systems downstream of the primary system isolation valves. The primary system is separated from the break by actions taken by the reactor protection system. The sequence and effects of these accidents are described in Sections 5.4.2 and 5.8.1.

Small leakages caused by breaks up to the size of an instrument line rupture (inside diameter less than or equal to 10 mm) are described in Section 5.4.3.

The consequences of inadvertent opening of safety valves of the pressure relief system are described in Section 5.4.4.

The consequence of an assumed break in a pipe upstream of the primary system isolation valves is depressurization of the primary system until the pressure reaches ambient. The occurrence of such a break is very unlikely, since the pipes between the pressure vessel unit and primary system isolation valves are designed to comply with the highest quality requirements.

Nevertheless, complete rupture of a DN65 pipe directly at the pressure vessel unit is assumed as a design basis accident. Such an accident results in maximum mass flow through the break and hence maximum loadings of the primary system and enveloping loadings of the reactor building.

Section 5.4.1.1.1 describes the effects on the pressure vessel unit, its internals and the reactor building as well as the radiological consequences of total depressurization.

Core heat-up starts after depressurization. The resulting temperature profiles and the radiological effects are discussed in Section 5.4.1.1.2.

Some countermeasures can be taken during the slow core heat-up phase to prevent or minimize releases to the environment.

Assuming a failure (single failure) of the primary system isolation valve actuated by the reactor protection system in the event of leaks and breaks downstream of the primary system isolation valve, the associated isolation valve can be closed manually. This isolates the primary system from the environment. There is no release to the environment during the core heat-up phase.

Certain further countermeasures can be taken against non-isolable breaks in the primary cavity.

The depressurized reactor can be cooled down by restarting the main heat transfer system thus preventing heat-up.

Regardless, the secured sub-atmospheric pressure system is connected to the affected redundancy section of the modular unit after depressurization.

In general, it is always possible to seal the break in situ since breaks outside the primary cavity are accessible at once, and breaks inside the steam generator cavity are accessible after about 1 day.

5.4.1 Break of a Large Connection Line between Pressure Vessel Unit and Primary System Isolation Valve

5.4.1.1 Depressurization Phase

5.4.1.1.1 Event Description

Effects on the Pressure Vessel Unit and its Internals

On rupture, the primary system is voided within minutes at a maximum depressurization rate of 0.8 bar/s. During the initial phase of the accident up to initiation of actions by the reactor protection system, the pressure differences and the mass flow only change slightly in that region of the primary system containing circulating coolant. The maximum change in mass flow as compared to operating conditions is less than approximately 20%.

In the case of double ended guillotine (2A) breaks in the outer pressure equalization line, the pressure difference between the stagnating primary coolant in the area between core barrel and pressure vessel and the gap between the core barrel and the graphitic internals changes more significantly than in other parts of the primary system. This differential pressure is limited to approximately 1 bar by the internal pressure equalization system. New pressure distribution and flow pattern develops in the primary system on primary gas circulator trip and closure of the circulator damper. As a result, primary coolant leaks from the hot gas region into the cold gas region for some break locations. The primary coolant temperature generally drops due to decompression. In the core and steam generator, the primary coolant temperature follows the temperatures of the fuel elements and the tube bundle respectively. These, however, change only slightly.

Only leakages from the cold gas regions of the primary system are possible if breaks occur between the pressure vessel unit and the primary system isolation valves.

The largest mass flow through the break is obtained for the assumed 2A break in the outlet line to the helium purification system at the steam generator pressure vessel.

The accident sequence for this break location is shown in Figure 5-12 and Figure 5-13.

5.4.1.1.2 Barrier Performance

Depressurization causes only insignificant loadings on the pressure vessel unit and its internals due to temperature changes.

Failure of the circulator damper to close has only an insignificant influence on the temperature profile.

Response of the Reactor Building

The escaping primary coolant enters the reactor building, where it causes a pressure and temperature increase in the building atmosphere. Various building zones are interconnected by relief ports that limit the pressure differences in the building.

If the differential pressure between building atmosphere and environment exceeds 0.1 bar, the outer pressure relief ports open and discharge the air, or rather a mixture of air and primary coolant, from the reactor building via the stack to the environment. As depressurization only lasts for a short time, the building structures are only heated to an insignificant extent. Following the discharge (after about 3 minutes), a temperature of about 100°C prevails at a depth of 25 mm in wall sections on which the discharging primary coolant jet has impinged. The support function is therefore not impaired. By and large the spread of the blown down primary coolant is restricted to the vicinity of the break and to a flow path to the outer pressure relief ports.

Large areas of the reactor building, principally the escape routes in the reactor hall and that part of the building housing the unaffected reactor, are not immediately permeated by primary coolant. The accident sequence is shown in Figure 5-13. The temperatures given in the figure are maximum values for the mentioned building regions.

5.4.1.1.3 Radiological Consequences

On depressurization of the primary system to ambient pressure, practically the total primary coolant inventory containing the gaseous and aerosol radioactive materials present in steady-state reactor operation is released into the atmosphere of the reactor building.

Since the flow velocities in the primary system increase no more than locally and are just slightly higher than under operating conditions, and this only in the initial phase of depressurization, liftoff of deposited activity can be ruled out. This applies especially to the steam generator that is the main trap for activity deposits. It is important to remember that, during blowdown, fractions of the solid and iodine activity bound by adsorption to primary system surfaces will be desorbed and, together with some of the stirred up radioactive dust that has accumulated in dead flow zones, will be taken up by the primary coolant flow.

Failure of the primary system isolation valves in the lines to the helium purification system to close causes these lines to be depressurized as well, and further radioactive materials are supplied primarily from the molecular sieve and possibly from the cryogenic adsorber due to desorption.

Most of the activity discharged into the reactor building with the primary coolant is released to the environment of the power plant via the pressure relief ports.

The following conservative assumptions are made to establish the activity release in the depressurization phase:

- Release of the total steady-state coolant activity (Table 5-23) including the noble gas, tritium and C^{14} activity deposited by adsorption in the helium purification system (Table 5-24).
- In order to estimate the maximum contribution of desorption in the period up to pressure equalization, the simplifying assumption is made that the partial pressure of the solid fission products and iodine isotopes in the primary coolant is always in adsorption/desorption equilibrium with the surface film. This causes mass transfer between surface and surrounding primary coolant at any rate and is thus equivalent to assuring that the primary-coolant-borne activity is constant throughout the blowdown process. On the basis of this conservative approach, desorption leads to the release from the primary system of approximately three times the steady-state coolant activity until pressure equalization is reached.
- Primary system surface activity as given in Table 5-23 (design values for 32 equivalent-power years).
- An enveloping value of 1 kg was assumed for the quantity of dust that is taken up by the primary coolant from dead flow zones during blowdown and is subsequently released from the primary system. For long-lived radionuclides, the specific activity of this dust is equivalent to the specific activity of the graphite on the surfaces of the fuel elements.
- Although deposition mechanisms, particularly with regard to particulate or aerosol activity, and the retention of a residual quantity of primary coolant in the reactor building are to be expected after pressure equalization with the environment, the escape factor from the reactor building is conservatively assumed to be 100%.
- Iodine is assumed as being in the chemical form which has the greatest radiological effects, i.e., elemental iodine.

The activity releases calculated with the assumptions specified above are listed in Table 5-4. This assumes that only the activity present in the primary circuit at the beginning of the depressurization is released outside of the primary circuit, therefore no significant additional activity is assumed released by the fuel during the depressurization.

The resulting potential radiation exposure in the environment of the power plant was calculated in compliance with Chapter 4 of the “Incident Calculation Bases for the Guidelines issued by the German Federal Minister of the Interior (BMI) for the Assessment of the Design of PWR Nuclear Power Plants pursuant to Sec. 28, paragraph (3) of the Radiological Protection Ordinance”, referred to as “BMI Incident Calculation Bases” in the following. The maximum doses are listed in Table 5-12.

Radiation exposure of the environment due to direct radiation from the reactor building is negligible due to the very low activity release into the reactor building and the thickness of the walls.

After pressure equalization with the environment, it is possible to enter the reactor building and, depending on the break location, to take actions to seal the break.

5.4.1.1.4 Assessment

A review of the safety analysis performed for the depressurization phase of a break of a large connection line between the pressure vessel unit and the primary system isolation valve indicates that:

Accident scenarios: The event scenario examined is complete and conservative

Dose assessment: The resultant dose consequences calculation are made with conservative assumption

Prevention and Mitigation: Adequate prevention and mitigation measure are in place to address these events

Conclusions: This phase of the primary system depressurization accident results in release of primary circulating activity to the environment. The offsite radiation exposure caused by this phase of the depressurization accident complies with the German and the U.S.A. (see Chapter 7) offsite exposure limits.

5.4.1.2 Depressurization Followed by Core Heat-Up

Temperature Development in the Core

Convection processes in the reactor core are of no significance when the primary system is depressurized. Heat is removed by the cavity cooler by radiation, and in the core, by conduction in the ceramic and metallic core internals and in the reactor pressure vessel, and by radiation and natural convection in the gas-filled spaces between these components.

The calculations to determine the effect on core temperature for the core heat-up phase of this event are based on the following assumptions:

- The reactor is abruptly depressurized while in long-term steady-state operation with a cold gas temperature of 280°C and at 105% of nominal power, close to the reactor protection system and the power limitations.
- The first scram signal due to the DN65 break, the “negative variable limit for primary circuit pressure” fails. Scram will then occur within 30seconds on the “negative variable limit for thermally corrected neutron flux”.

Temperature development during the accident is characterized by the large heat capacity of the core and the ceramic core internals in relation to the decay heat generation rate, by the steadily decreasing decay heat generation rate and by the thermal resistances between the centre of the core and the cavity cooler.

During the first hours, the central regions of the core undergo almost adiabatic heat-up until the temperature gradient across the core is large enough for a significant fraction of the residual heat to be removed, more than 90% leaving the core radially. Figure 5-14 shows the radial temperature curves from the core axis to the cavity cooler (0.65 meters below the middle of the core, location of peak core temperature) for different times. Figure

5-15 shows the axial temperature patterns along the core axis at different times and Figure 5-16 the temperature versus-time curves at different locations in the core and reactor pressure vessel.

5.4.1.2.1 Barrier Performance

The peak core temperature of ~1522°C is reached after about 30 hours. Later, the decay heat generation rate at the affected location drops below the heat removal rate and the temperature decreases again. After about 40 hours the perimeter zones of the core also start to cool down again, while the temperature peaks in the side reflector are only reached after 85 hours and those in the reactor pressure vessel at cavity cooler elevation after 100 hours. The core barrel reaches temperatures of up to 490°C (code limit for core barrel material), the pressure vessel no more than ~350°C. The maximum heat flux at the cavity cooler is ~720 kW.

Figure 5-14 and Figure 5-15 show that only a small part of the core reaches high temperatures. This is illustrated in Figure 5-17 by the plots of the volumetric components of the core that exceed certain temperatures versus the time. It is evident that less than 1% of the fuel elements attain temperatures of more than 1500°C to which they are only exposed for 20 hours at most.

For the temperature calculations, except for the above listed assumptions, nominal values were used for all input variables. The uncertainties relating to the input variables and statistical superposition produce an allowance of 98K that is added to the calculated fuel temperatures.

5.4.1.2.2 Radiological Consequences

As stated above, the maximum fuel temperatures on core heat-up following primary system depressurization is limited to ~1600°C even with allowance for uncertainties in the input variables. This is a temperature at which the retention capability and integrity of the coated particles are basically inviolate. Only the small fission product fraction, that is already present outside intact particle coatings before the accident due to manufacturing and irradiation-induced defects, and additional temperature- incurred particle failure that may possibly occur during core heatup make up the activity potential that can be released from the fuel elements.

An activity release from the fuel elements substantially exceeding that during normal operation is only to be expected some hours after onset of the accident. This is because the fuel temperatures increase relatively slowly and the fission products only diffuse gradually from the inside of defective fuel particles to the surface of the fuel elements.

The transport mechanism for an activity release from the depressurized primary system is temperature-induced gas expansion that is completed after about 160 hours when the volume has increased by approximately 9%. This is followed by global cooling, that causes the intake of primary cavity atmosphere into the primary system and terminates activity discharge into the reactor building. As verified by tests on mock-ups, natural convection or gas diffusion have hardly any significance as release mechanisms during the global cooling phase.

The activity leaving the core zone, with the exception of noble gases, is significantly less than that released from the fuel elements since the graphitic surfaces of colder fuel elements and reflectors act as activity traps. This also applies to the metallic surfaces that the primary coolant passes on its way to the break.

During the core heat-up phase sub-atmospheric pressure is maintained in the affected HVAC section of the reactor building, and the exhaust air is conveyed to the stack via the filter unit of the secured sub-atmospheric pressure system.

Calculations of the activity release to the environment due to core heat-up are based on the following data and assumptions:

- Fuel temperatures: All calculated fuel temperatures are raised by a total of 98K to cover uncertainties in the input variables for the temperature calculations.

- Accident induced particle failure: In heat-up experiments with LEU TRISO particles performed in the past, temperature induced defects were not observed below 1600°C. In the calculations, however, it was assumed that accident induced particle failure starts at 1200°C, rises exponentially with temperature and reaches a fraction of 5×10^{-4} at 1600°C.
- Diffusion data: Transport of metallic fission products through the diverse barriers of the coated particle and the fuel element was analyzed with the diffusion data from the "HBK-Standard Datensatz" (1984 status). The release data calculated with these data conservatively cover those measured for the reference LEU TRISO particle and for fuel elements in heat-up experiments.
- Only diffusion from the fuel of failed particles is of relevance to noble gas and iodine release in the core heat-up phase. The diffusion coefficient applied here has also been validated by up-to-date results of heat-up experiments.
- Release from the primary system: The time history of activity entrainment through the break into the reactor building is calculated by convolution of the release rate from the fuel elements and the expansion-induced primary coolant blowdown rate. Internal primary system retention mechanisms are considered globally.
- Efficiency of the filter unit in the secured sub-atmospheric pressure system:

    0% for noble gases

    99% for iodine

    99.9% for solids

The filter efficiency selected for iodine allows for the presence of organic iodine compounds and is extremely conservative for elemental iodine that is primarily to be expected.

The total quantity of iodine released to the environment is postulated as being in the chemical form which has the greatest radiological effects, i.e., elemental iodine.

Core heat-up is not an independent accident event. It only occurs in conjunction with preceding primary system depressurization. The radiological effects must therefore be established by addition of the activity released during both accident phases. The activity release during the depressurization phase has already been discussed (see Table 5-4).

The release of radiologically representative radionuclides analyzed for the core heat-up phase is shown for different time intervals in Table 5-5. Additionally the releases are listed without taking into account the filter system. These intervals were selected in such a way that they are compatible with the time intervals for the dispersion factors established in the "BMI Incident Calculation Bases", and that the radiation exposure in the environment is determined as an enveloping value for the entire accident sequence, i.e., depressurization followed by core heat-up.

The results of the potential radiation exposure calculated on the basis of the data and rules of Chapter 4 of the "BMI Incident Calculation Bases" are shown in Table 5-13.

The radiation impact on the environment due to direct radiation from the reactor building is also negligible during the core heat-up phase on account of the small activity release into the reactor building and the thickness of the walls.

The reactor building can be entered during and after core heat-up to perform repair work. Totally encapsulating protective gear is required because of airborne activity; the external radiation exposure is less than 100 µSv/h (10 mrem/h), assuming that the radioactivity released from the reactor during core heat-up remains in the reactor building.

5.4.1.3 Assessment

A review of the safety analysis performed for the core heat up phase of a break of a large connection line between the pressure vessel unit and the primary system isolation valve indicates that:

Accident scenarios: The event scenario examined is complete and conservative

Dose assessment: The resultant dose consequences calculation are made with conservative assumption

Prevention and Mitigation: Adequate prevention and mitigation measure are in place to address these events

Conclusions: This phase of the primary system depressurization accident results in release of primary circulating activity (unfiltered) and additional releases due to core heatup (filtered) to the environment. The offsite radiation exposure caused by this phase of the depressurization accident complies with the German and the U.S.A. (Chapter 7) offsite exposure limits.

5.4.2 Break of a Large Connecting Line (DN65) Downstream of Primary System Isolation Valve in the Reactor Building

5.4.2.1 Event Description

For breaks downstream of the primary system isolation valve, primary coolant escapes through the break until the primary system is isolated from the break by closure of the primary system isolation valve. If isolation valves fail to close the pressure in the downstream system equalizes with the reactor building.

Except for breaks at the fuel element discharge lines, the cold gas regions of the primary system are voided. If breaks occur in the fuel element discharge lines, the hot gas flowing from the core to the break is cooled by the fuel in the discharge tube.

By the time primary system isolation has been completed, the primary coolant temperature at this break location is ~300°C. On assumed failure of the primary system isolation valve, the primary coolant reaches a maximum temperature of 400°C at the break location.

Since the primary coolant discharge is restricted by primary system isolation, the trip pressure of the pressure relief system of the reactor building is not reached. The slight overpressure in the reactor building is lowered within a short time by the HVAC system. Activity is therefore primarily released to the environment via the vent stack. This release is initially unfiltered but later, after switchover to the secured sub-atmospheric pressure system, it is filtered by activated-carbon adsorbers and HEPA filters.

5.4.2.2 Barrier Performance

The temperature loading on the reactor building is only increased to an insignificant extent by the somewhat higher primary coolant temperature.

5.4.2.3 Radiological Consequences

Even when unfiltered these releases are enveloped by the radiological consequences of a break in a line of the helium purification system in the reactor auxiliary building with unfiltered activity release via the pressure relief system of the building (ground level). This is an enveloping case for all breaks or leakages downstream of primary system isolation valves with regard to both the source term and the release pathway.

The radiological effects of the break inside the reactor building remain below the doses listed in Table 5-19 because of the more favorable release conditions described above.

5.4.2.4 Assessment

A review of the safety analysis performed for a depressurization accident caused by a break of a large connection line downstream of the primary system isolation valve in the reactor building indicates that:

Accident scenarios: The event scenario examined is complete and conservative

Dose assessment: The resultant dose consequences calculation are made with conservative assumption

Prevention and Mitigation: Adequate prevention and mitigation measure are in place to address these events

Conclusions: This depressurization accident results in release of small amount of primary circulating activity. The dose consequence of this accident is bounded by the radiological consequence of a break in a helium purification line in the reactor auxiliary building. The offsite radiation exposure caused by this phase of the depressurization accident complies with the German and the U.S.A. (Chapter 7) offsite exposure limits.

5.4.3 Break of an Instrument Line and Small Breaks

5.4.3.1 Event Description

Break of an instrument line (inside diameter not exceeding 10 mm) or small breaks in other lines cause a pressure drop in the primary system. This is detected by the reactor protection system that then scrams the reactor.

The out flowing primary coolant is detected in the exhaust air resulting in switchover to the secured sub-atmospheric pressure system.

If it is impossible to isolate the break, the depressurization to ambient pressure that follows is relatively slow and primary coolant is released via the secured sub-atmospheric pressure system and the stack.

In the event of a very small break, the primary system pressure drop can be so slow in certain circumstances that it is not detected by the reactor protection system or the leakage flow is not compensated for by the pressure control system.

The out flowing primary coolant, however, is immediately registered by primary coolant activity detectors in compartments housing primary coolant carrying lines and in the exhaust air ducts. This is indicated by appropriate signals in the control room, thus giving the operators sufficient time for manual actions.

5.4.3.2 Barrier Performance

The long-term temperature response is covered by the calculations in Section 5.4.1.2.

5.4.3.3 Radiological Consequences

The radiological effects of this accident sequence were calculated as the activity release to the environment given in Table 5-6. This release consists of the steady-state primary coolant activity and the solid and iodine activity desorbing from the primary system surfaces during the blowdown process (see Section 5.4.1). Attention was also paid to the fact that the activity release from the fuel elements into the primary coolant continues after leak detection until the reactor has been scrammed and that credit can only be taken for the radioactive decay of short-lived noble gas and iodine isotopes after this.

The following data were assumed for the efficiency of the filter unit in the secured sub-atmospheric pressure system:

0% for noble gases

99% for iodine

99.9% for solids

The filter efficiency selected for iodine allows for the presence of organic iodine compounds and is extremely conservative for elemental iodine that is expected. The total quantity of iodine released to the environment is assumed as being in that chemical form of iodine that has the greatest radiological effects, i.e., elemental iodine.

The potential accident doses established on the basis of the rules and data given in Chapter 4 of the "BMI Incident Calculation Bases" are listed in Table 5-14 and Table 5-15. In both cases the calculated doses are far below the emergency reference levels pursuant to Article 2.8.3 of the German Radiological Protection Ordinance.

The long-term temperature development is covered by the calculations in Section 5.4.1.2.

In the event of a very small break, the primary system pressure drop can be so slow in certain circumstances that it is not detected by the reactor protection system or the leakage flow is not compensated for by the pressure control system.

5.4.3.4 Assessment

A review of the safety analysis performed for a depressurization accident caused by a break of an instrument line indicates that:

Accident scenarios: The event scenario examined is complete and conservative

Dose assessment: The resultant dose consequences calculation are made with conservative assumption

Prevention and Mitigation: Adequate prevention and mitigation measure are in place to address these events

Conclusions: This depressurization accident results in release of small amount of primary circulating activity. The dose consequence of this accident has been calculate and meets the radiological consequence limits established by the German and the U.S.A. regulation.

5.4.4 Inadvertent Opening of a Safety Valve

5.4.4.1 Event Description

The safety valves of the primary pressure relief system re-close automatically after a pressure drop of 8% below operating pressure. In the assumed event of this valve being stuck open, a blocking valve connected in series with the affected safety valve closes when a minimum pressure is reached.

Both safety valves blow down unfiltered into the reactor building. The primary coolant is detected in the exhaust air unit and is then released via filters of the secured sub-atmospheric pressure system.

Even without consideration of the filter system, the effects of this accident are covered by "Break of a large connecting line downstream of primary system isolation valve" (Section 5.4.2).

5.4.4.2 Barrier Performance

The effects of this accident are covered by "Break of a large connecting line downstream of primary system isolation valve" (Section 5.4.2).

5.4.4.3 Radiological Consequences

The effects of this accident are covered by "Break of a large connecting line downstream of primary system isolation valve" (Section 5.4.2).

5.4.4.4 Assessment

A review of the safety analysis performed for a stuck open primary system pressure relief valve indicates that:

Accident scenarios: The event scenario examined is complete and conservative

Dose assessment: The resultant dose consequences calculation are made with conservative assumption

Prevention and Mitigation: Adequate prevention and mitigation measure are in place to address these events

Conclusions: This depressurization accident results in release of small amount of primary circulating activity. The dose consequence of this accident is bounded by break of a large connecting line downstream of primary system isolation valve discussed in section 5.4.2.

5.5 Air Ingress

Air ingress is defined as the intrusion of outside air (from the reactor building) through the primary coolant boundary and into the reactor vessel. Air ingress events result from breaks in the primary coolant boundary such as ruptures of pipes connected to the primary system. A failure in any of these lines will result in primary system leakage ranging from the barely detectable (leaks) to rupture of the largest pipe connected to the reactor vessel. However, breaks in the primary system pressure vessels, including the gas duct pressure vessel and the steam generator vessel, are not postulated because of the extensive quality assurance measures taken and are considered to be beyond design basis events (Section 5.5.4).

The concern with air ingress events is the intrusion of oxygen into the reactor primary system could oxidize the graphite core structures and fuel element matrix. Unmitigated, this oxidation process could eventually degrade the structural integrity of the core or reduce of the fission product retention capability of the coated particle fuel (chemical attack). Depending on the size of the break, there is also a potential for the liftoff and release of plated-out radioactivity in the primary circuit and for off-normal differential pressure loads on structural components. Additionally, it has been shown that the reaction kinetics of the fuel elements from oxygen is considerably faster than corrosion in water vapor and that all oxygen entering the primary coolant system as airborne oxygen reacts with graphite at temperatures above 400°C. However, the reaction process from oxygen is self-limiting due to the lack of available air from leaks and from small and large breaks.

With the possible exception of very small breaks, such as “leaks” (Section 5.5.1), depressurization events leading to air ingress into the primary system have three distinct phases:

1) Initially depressurization phase – this phase occurs immediately following the break as coolant helium flows out of the reactor through the break and pressure balances between inside and outside of the primary system pressure boundary. For leaks and small breaks this could be the only phase because the event is terminated by leak isolation before complete depressurization could occur.
2) Air ingress phase - air enters the primary system; gradually by molecular diffusion or more rapidly by stratified flow, depending on the size of the break that controls the depressurization rate.
3) Natural circulation could eventually be established if thermo-hydraulic conditions are established for continuous circulation of outside air into the primary system.

The HTR-Module phenomenological sequences and the chemical and radiological consequences of air ingress accident assessment are based on the air ingress technical data and information in reference [4].

<u>*Depressurization*</u>

The initial rate of depressurization of the reactor vessel varies according to the break size. During depressurization, the reactor vessel leakage rate is affected by the rate of heatup of the gases in the vessel and by hydrostatic forces due to the density differences between air and helium.

Loss of primary coolant decreases the primary system pressure and initiates a scram on the negative sliding limit when primary system pressure $\leq$ ~180 mbar/min. The following reactor protection actions are also initiated: reflector rod drop, primary gas circulator trip, closure of the main steam and feedwater isolation valves and the primary system isolation valves.

In the event of instrument line breaks that do not meet the above shut-down criterion, the reactor will be shut-down due to mass flow ratio (primary to secondary side) less than or equal to 0.75. The isolation valves of the instrument lines are located outside the primary cavity and in the event of primary system depressurization these valves are not closed.

If primary coolant activity is detected in the exhaust air, the plant is automatically switched over to the secured sub-atmospheric pressure system to ensure filtered release of small leakages.

Initial Air Ingress

After the reactor depressurization has reached a pressure balance between the reactor vessel and the reactor building, hydrostatic forces will cause the helium in the reactor vessel to be displaced by reactor building air. Depending on conditions the air may enter gradually by molecular diffusion or rapidly via stratified flow. The air reacts with the graphite (graphite oxidation) and lessens the amount of air in the reactor. As the reactor core cools, thermal contraction of gases in the reactor vessel draws additional air into the vessel that also reacts with the graphite however cooling of the graphite and reduced oxygen reduces the potential for sustained graphite oxidation.

Natural Circulation

After the depressurization and initial air ingress, air entering the hot reactor will eventually reach an air concentration in the core sufficiently large to establish natural circulation. Tests have shown that a significant time period may be required for the air to displace helium in order to establish buoyancy-driven flow. The low power density core configuration limits temperature rise and core flow geometry provides significant resistance to limit natural convection air flow rates. Furthermore if air ingress occurs during core heats up, gas viscosity increases that reduces flow rate.

Fission Products Release

Fission products release can result from prolonged oxidization of the fuel element graphite matrix and particle coatings and exposes a damaged particle (damaged SiC coating). Oxidation does not damage intact SiC coating. The oxidation process also produces a mixture of carbon monoxide (CO) that is volatile and carbon dioxide (CO_2) that can be released into the reactor building. The amount of graphite that can be oxidized by air ingress is controlled by the temperature dependant graphite oxidation rate and by the amount of air available for air ingress into the reactor vessel.

Air ingress events are categorized and discussed below according to size of the break in the primary coolant boundary.

5.5.1 Small Breaks in the Primary Coolant Boundary

5.5.1.1 Event Description

Small break leakages result from instrument line or pipe breaks with inside diameter less than or equal to 10 mm. The reactor instrument lines connected to the vessel are $\leq$ 10 mm in inside diameter.

Small breaks have a relatively slow depressurization that will result in a reactor scram either due to a pressure drop in the primary system or due to mass flow ratio (primary to secondary side). The outflowing reactor coolant is detected in the exhaust air initiating a switchover to the secured sub-atmospheric pressure system and the stack. Primary coolant activity detectors in compartments housing primary coolant lines and in the exhaust air ducts will detect the leakage and signal the control room.

Assuming the small break cannot be isolated, after the initial depressurization hydrostatic forces will result in the ingress of air from the reactor building into the reactor vessel where some graphite oxidization will take place. Calculations have shown that several days after the event is initiated the air ingress will be limited to less than 5% of the reactor building volume and the oxidized graphite will be a fraction (< 10^{-4}) of the total graphite.

5.5.1.2 Barrier Performance

The long-term temperature development in the core is covered by Section 5.4.1.2.

The small fraction of total graphite mass that is oxidized results in negligible structural loss and does not affect safe shut-down and the ability to control heat generation and maintain afterheat removal.

The effect on the reactor building loadings from this relatively slow pressurization air ingress event is negligible and bounded by the design basis event large break event described in Section 5.5.3.

5.5.1.3 Radiological Consequences

The radiological consequences of the depressurization phase for this event are described in Section 5.4.3.

Calculations have shown that air ingress conditions (temperature, dry air) have negligible impact on fission product releases of TRISO fuel particles.

Due to limited size and the amount of air infiltration into the primary system, the air ingress phase of this event has insignificant release of fission products from the oxidized graphite and from the previously damaged or failed fuel particle. Therefore the radiological consequences provided in Section 5.4.3 for the depressurization phase will not be changed as a result of the air ingress.

5.5.1.4 Assessment

Primary coolant activity detectors will detect the leakage and signal the control room allowing sufficient time for manual actions. If it is not possible to isolate the break, the depressurization rate is relatively slow and the primary coolant is released out the sub-atmospheric pressure system and stack.

5.5.2 Large Breaks in the Primary Coolant Boundary

5.5.2.1 Event Description

Air ingress from large breaks result from pipes connected to the reactor vessel. All primary coolant pipes connected to the pressure vessel have a cross-section less than or equal to DN65. Rupture of a connecting pipe with a break cross section less than or equal to DN65, such as the fuel feed tube or the surge line, is considered to be the enveloping postulated break. The depressurization rate for a DN65 break is 0.8 bar/s,

Most of the circulating activities are released in the first few minutes of the initial depressurization phase. The hydrostatic displacement mechanism following the depressurization phase and during core heat-up will result in the release of fission products from the oxidized graphite and fuel particles with damaged SiC coating. For the fuel particles the release will be some hours later because the fuel temperatures increase relatively slowly and the fission products diffuse gradually from the inside of the defective fuel particles to the surface of the fuel elements

Assuming no mitigating actions, for the first 95 hours after the break the decompressed helium remaining in the primary coolant system would heat-up from an average of 210°C to a maximum of 290°C and leak 460 m^3 of helium into the reactor building. Therefore, of the original 11.45 kmol helium, 9.82 kmol remain in the primary coolant system.

From 95 to 500 hours, the helium gas cools down to an average of 233°C and a helium/air mixture will be drawn from the reactor building back into the primary coolant system. After 500 hours resulting from a combination of gas flowing back from the reactor building and chemical reactions, about 1.1 kmol of gas other than helium is introduced in the primary coolant system. Conservatively assuming that the induced gas consists of air and the complete conversion of the oxygen into carbon monoxide, then about 4-5 kg of graphite (out of the total of 450 tons) would be oxidized or about 10 g/h over 500 hours. Compared to above the relative percentage of air ingress due stratification and natural conviction are expected to be negligible.

In summary, for the large breaks, calculations have shown that the air ingress will be limited to a fraction of the reactor building volume (~ less than 5% of the reactor building volume) and the resulting oxidized graphite will be a very small fraction of the total graphite, i.e., < 5 kg out of 450 tons.

5.5.2.2 Barrier Performance

The long-term temperature development in the core is described in Section 5.4.1.2.

The small fraction of total graphite mass that is oxidized results in negligible structural loss and does not affect safe shut-down and the ability to control heat generation and maintain afterheat removal.

A detailed description of the effects of the depressurization phase on the pressure vessel is given in Section 5.4.1 and determined to be insignificant. Any further effect due to the hydrostatic transport mechanism displacing reactor helium with air will be negligible.

The effect on the reactor building loadings from the depressurization phase is described in Section 5.4.1 and determined to be insignificant.

5.5.2.3 Radiological Consequences

The effect of both the depressurization phase and the core heat-up phase of this event on radiological consequences is provided in Section 5.4.1.

The TRISO fuel particles have a ceramic coating layer (SiC) that is designed for very high temperatures. Calculations have shown that air ingress conditions (temperature, dry air) have negligible impact on fission product releases of TRISO fuel particles unless the air meets the SiC, which is nearly impossible due to the mass of graphite inside the core. Additionally, since there is very little oxidization of the graphite, there will be very little, if any, fission products released. The fission product releases are from the failed or failing fuel particles that are expected to be small fraction of the total number of particles that at temperature conditions that could result in failure.

Therefore the radiological consequences provided in Section 5.4.1 for the depressurization phase will not be changed as a result of the air ingress.

5.5.2.4 Assessment

There is no significant release of fission products for large break air ingress events due to oxidized graphite or from damage to the fuel particles for the following reasons:

1) Parametric studies show that graphite oxidization is not sensitive to rapid displacement (stratified flow) or slow displacement (molecular diffusion).

2) Studies have also shown that graphite oxidization is not very sensitive to time delays for onset of natural convection.

3) Experience has shown that self sustaining combustion is difficult to achieve in nuclear grade graphite requiring a sustained supply of oxygen. The heat generation from combustion must exceed the heat loss by conduction, convection, and radiation that is very difficult to achieve for high-purity, nuclear-grade graphite.

4) Ceramic fuel particles are designed for very high temperatures and calculations, supported by experience, have shown that air ingress conditions (i.e. high temperature, dry air) have negligible impact on fission

product releases and failure of TRISO fuel particles unless the air meets the SiC, that is nearly impossible due to the mass of graphite inside the core.

As a result, there will not be any significant release of fission products outside the vessel and the reactor surroundings will remain accessible so that intervention measures can easily be performed during the hours or days available. Section 5.4 discusses countermeasures the operator can take during the slow core heat-up phase to prevent or minimize releases to the environment and further countermeasures against no-isolable breaks in the primary cavity. Also, personnel will be able to enter the reactor building in full protective gear to take measures to terminate the air flow into the reactor core.

5.5.3 Large-Scale Rupture of Gas Duct Pressure Vessel and/or Reactor Vessel (Beyond Design Basis)

5.5.3.1 Event Description

The most significant and bounding source of air ingress is the "beyond design basis" large rupture in the primary coolant boundary, i.e., gas duct pressure vessel and/or reactor vessel, that allows a complete exchange of primary coolant and reactor building air. A rupture of the gas duct pressure vessel and/or reactor vessel could result in significant air ingress, graphite oxidation and fission product releases into the reactor building and subsequently to the environment.

5.5.3.2 Barrier Performance

A large rupture in the pressure vessel is a significant breach of the reactor pressure boundary allowing complete voiding of the primary system coolant into the Reactor Building. Pressure differences in the building are limited by interconnected relief ports but if the differential pressure between building atmosphere and environment should exceed a limit (0.1 bar), the outer pressure relief ports open and discharges the mixture of air and primary coolant via the stack to the environment. The spread of the blown down primary coolant is restricted to the vicinity of the break and to a flow path to the outer pressure relief ports.

Significant fuel particles failure and subsequent fission product release would occur from core heat-up phase only some hours after onset of the accident. This is because the fuel temperatures increase relatively slowly and the fission products only diffuse gradually from the inside of defective fuel particles to the surface of the fuel elements.

5.5.3.3 Radiological Consequences

On depressurization of the primary system to ambient pressure the total primary coolant inventory is released into the atmosphere of the reactor building. Therefore, all initial circulating activity is released to the reactor building and subsequently most of it released into the environment.

Following the depressurization phase oxidized graphite dominates as the primary source of fission products released into the reactor building until hours later when fission products resulting from already damaged fuel and fuel failures during the core heat-up would be released into the reactor building. However, the activity leaving the core zone is significantly less than that released from the fuel elements because the graphitic surfaces of colder fuel elements and reflectors and metallic surfaces in the coolant path act as activity traps. Additionally, during the core heat-up phase exhaust air is conveyed to the stack via the filter unit of the secured sub-atmospheric pressure system.

Radiation exposure of the environment due to direct radiation from the reactor building is negligible due to the very low activity release into the reactor building and the thickness of the walls.

5.5.3.4 Assessment

The plant's primary coolant boundary is designed for events of this type. Therefore, pressure vessel breaks are expected to be very unlikely and therefore are categorized as beyond design basis events and outside the licensing basis region. The reactor pressure vessel (including the cross-vessel) will be designed to ASME Class 1 standards and to the requirements of KTA 3201 with HTGR specific modifications and is assigned to class MK 1. Additionally, through-wall cracks can be safely ruled out due to the extensive quality assurance measures to be applied with respect to planning, design, materials selection, manufacture and inspection. This accident is not credible and is considered a beyond design basis event.

5.6 Water Ingress Events

Water ingress is defined as intrusion of water or water vapor into the primary circuit. Large scale water inleakage into the reactor coolant system of a high temperature reactor constitutes a significant hazard potential mainly for two reasons. Firstly, depending on the core design, the ingress of the neutron moderating steam can cause a power excursion of the core, and secondly, steam can react with fuel and hot graphite structures to form water gas, that when mixed with air may be flammable. During the design of the HTR-Module both mechanisms were considered, and special, radical measures were taken by utilizing physical properties to keep the impact of major inleakage into the reactor coolant system as low as possible, even in the event of failure of the active countermeasures that are specifically provided to minimize the effects of large scale inleakage accidents.

The negative fuel and moderator temperature coefficient in the pebble bed core prevents major excess reactivity increase by water/steam ingress (increased moderation) and thus reducing the heavy metal content of the fuel. The following discussion therefore refers only to the corrosion phenomena that can take place during the inleakage in the reactor coolant system, and the consequences thereof.

Inleakage into the reactor coolant system cannot be prevented in principle, because in spite of the tightest quality control procedures during the manufacturing of the steam generator tubes, leakages in the tube bundle cannot be ruled out completely. However the special geometric shape of the reactor coolant system enables the ingress of water or steam into the core, and thus the formation of water gas to be largely minimized or made difficult. Partly for this reason, the steam generator of the HTR-Module was placed in a separate pressure vessel and arranged laterally below the reactor core. It is therefore not possible in the case of a tube rupture for water to directly enter into the pebble-bed core or the graphite hot gas plenum, where it could evaporate on the hot graphite components and corrode to form water gas.

In addition, the specially chosen cold gas flow path in the reflector holes prevents formation of natural convection in the reactor coolant system, so that in the long-term the transport of water vapor from the steam generator pressure vessel into the core is suppressed, and thus in the long-term, the reaction of graphite and water vapor virtually comes to a halt solely for that reason. Ultimately, only a limited amount of water can be vaporized due to the limited heat capacity of the metallic structures in the steam generator pressure vessel, so that the maximum water gas production in the HTR-Module is limited, and does not depend either on how much water enters into the reactor coolant system, or how much water is postulated for the analysis of hypothetical accident sequences.

Water ingress in the HTR-Module reactor design is possible in case of a steam generator tube rupture. This section examines the steam generator tube rupture accident sequence relative to the HTR-Module reactor design.

5.6.1 Steam Generator Breaks/Tube Rupture

For HTR-Module the bounding water ingress event is a break or rupture of the steam generator tube(s).

A significant leak in the steam generator is capable of releasing significant quantities of water into the primary coolant circuit during normal power operation. The steam generator is the dominant source of water ingress because it can introduce water under all operating conditions at a more significant flow rate, and with a higher fraction of vapor, that is more readily transported around the primary circuit.

Water ingress has several potential adverse consequences, depending on the magnitude of the ingress and the plant operating state at the time of the event. The main water ingress consequences include:

- Pressure increases,
- Reactivity increase,
- Oxidation of graphite (i.e., chemical attack),
- Fuel hydrolysis, and
- Fission product mobilization.

5.6.1.1 Event Description

A steam generator tube break is detected after about 10 seconds by the moisture instrumentation in the primary system. The accident is detected due to moisture in the primary system greater than or equal to approximately 800 vpm. Of the steam generator water inventory, only about 600 kg of water can enter the primary system under the worst-case conditions [5]. After water ingress into the primary system due to steam generator breaks or tube rupture, the water/gas reaction, $H_2O + C = H_2 + CO$, results in a mild "chemical attack" causing a slight corrosion of the fuel elements.

Apart from triggering reactor scram, the reactor protection system also actuates steam generator relief (dump) to limit the quantity of water entering the primary system.

After steam generator relief, the steam generator relief valves are closed by spring action, if the pressure difference (primary to secondary) is approximately zero.

Residual heat is removed by the operational component cooling system and the secured cooling system. The water in-leakage is removed from the primary system via the third train of the helium purification system that contains the manually activated post accident water separator. In addition, the pressure control system is capable of preventing a pressure build-up in the primary system up to the response level of the pressure relief system.

The accident sequences and control of such sequences are described in greater detail in the following sections, both with and without superimposed failure of operational systems for mitigation of accident consequences.

Accident Sequence Assuming no Failure of Operational Systems

At full operating pressure, maximum 5.3 kg/s of water and steam enter the primary system during a 2A break of the steam generator tube (equal to 2 x 1.7 cm^2 at the steam side and 2 x 2.6 cm^2 at the water side). This small loss of mass as compared to the overall flow through the steam generator is compensated for by the control system. Consequently there is no significant pressure change in the steam generator prior to detection of the accident by the moisture instrumentation. Water or steam therefore enters the primary system at an almost constant leak rate until the accident is detected.

After detection of the accident by the moisture instrumentation, the main steam line (after a 30 second delay) and the feedwater line are isolated by isolation valves with closure times not exceeding less than or equal to 2 seconds. The isolation valves of the start-up and shut-down circuit will be closed as well. Steam generator relief is initiated simultaneously, and the steam generator is drained by two parallel trains, each equipped with two automatically opening valves. By only one train (single failure) the pressure relief takes place in less than 1 minute. During this period, the water or steam ingress rate into the primary system is dramatically reduced owing to the pressure decrease in the steam generator. When the pressure in the steam generator has dropped to approximately primary system pressure, steam generator relief is terminated by closing the valves. On completion of steam generator relief, the steam generator still holds about 290 kg of steam at a pressure of 60 bar and at various temperatures. Approximately 210 kg of water have entered the primary system up to closure of the relief lines; consequently about 500 kg of water in all can enter the primary system.

For the purposes of further analysis, the quantity of steam that has entered the primary system is assumed to be 600 kg. For the core temperature, a safety allowance of 100K is added to the calculated values. On reactor scram, the primary gas circulator is also tripped and the circulator damper closed. In order to protect the primary gas circulator, however, this damper does not close tightly during coastdown. This means that the flow through the primary system amounts to approximately 10% of the natural circulation flow through the open damper.

If it is assumed that the post-accident water separator is cut in manually about 30 minutes after onset of the accident, graphite corrosion and water/gas formation is limited to a fraction of the water that has entered the primary system.

Approximately 40 kg of water is converted and the maximum graphite corrosion is approximately 0.3%. This is acceptable both with regard to fuel corrosion as well as to the strength of the ceramic internals and fission product release.

The reaction is terminated after about 3 hours. The water gas that has formed and the remaining moisture are removed from the primary system by the helium purification system.

The pressure increase due to water ingress, water gas formation and temperature increase is relieved by the water separator, helium purification system and pressure control system in about one hour. This accident sequence therefore cannot lead to response of the pressure relief system.

Small steam generator breaks are also detected by the moisture instrumentation by the reactor protection system.

The smallest leakage, that leads long term (about hours) to a scram, are about 1g/s, this means the primary system contains approximately 4 kg of water when the moisture signal is given.

Steam generator leaks do not pose any threat to the plant, even over a lengthy period of time, and are detected by measurement of the impurities in the helium purification system.

The moisture instrumentation is designed and installed in such a way that detection of tube breaks is possible even if there is no flow through the primary system and if the quantities of steam entering the core are not larger than in the sequence described here.

Accident Sequence Assuming Failure of Operational Systems

The following are the operational measures for mitigation of accident consequences:

- Closure of the circulator damper,
- Operation of the post-accident water separator,
- Operation of the pressure control system and helium purification system.

Of the resulting failure combinations, the combination of operation of the circulator damper with non-availability of the post-accident water separator, the pressure control system and the helium purification system is the worst case with respect to maximum fuel corrosion and pressure build-up in the primary system.

If water ingress of 600 kg is assumed in this case as well, the reaction takes place as shown in Figure 5-21with a steep rise in the corrosion rate, due to its exponential temperature dependence, and a peak after about 2.5 hours. The subsequent decrease in the corrosion rate is due to the fact that most of the steam that has entered the core has now been converted and, because of the low natural circulation rate in the primary system only a small amount of steam is still being supplied to the core. The steam mass flow is increased by response of the pressure relief system after about 5 hours and consequently a further peak in the corrosion rate occurs at this time.

Owing to failure of the post-accident water separator, almost the entire water in-leakage is converted (Figure 5-22). Graphite corrosion attains approximately 3.3% locally. Figure 5-23 shows time curves for the volume fractions of the core exhibiting corrosion in excess of 1%, 2% and 3%. Only corrosion of more than 40% would erode the unfueled graphite shell of the fuel elements. But even then the fuel particles in the inner fuel element zone would still be undamaged because of their enclosure in multiple layers of pyrocarbon and silicon carbide and fission product retention would be assured.

The overall pressure response (Figure 5-24) starts at approximately 64 bar because of the water ingress and the temperature increase in the primary system and continues to rise as a result of water gas production and a further primary coolant temperature increase.

After approximately 5 hours, the safety valve of the pressure relief system opens at 69 bar and, for about 30 minutes, blows down a mixture of primary coolant, water gas and steam, the total quantity of which amounts to approximately 400 kg. The water gas fraction is around 9%. Such a gas mixture is not flammable in any mixture with air.

Even if the small quantity of gas mixture present in the pressure equalization line before blowdown were to have higher water gas content, this would not present any hazard for the components or the reactor building.

If the HVAC systems have not already been manually switched to the secured sub-atmospheric pressure system during the several hours long accident sequence, this switchover is affected on detection of activity in the reactor hall, and the blown down gas mixture is released via filters through the stack.

5.6.1.2 Radiological Consequences

In response of the pressure relief system, activity is initially discharged into the reactor building with the release of primary gas mixture. In this accident, as in the case of small primary system breaks, the HVAC system of the affected building area is switched to the secured sub-atmospheric pressure system that is equipped with activated-carbon adsorbers and HEPA filters.

The discharged activity consists of fractions of

- The steady-state primary coolant activity,
- The activity remobilized by water steam from primary system surfaces, primarily steam generator surfaces,
- The activity of the unfueled shell released with the graphite of the fuel elements (corrosion) due to conversion of the moisture entering the core region.

The activity released to the environment is calculated on the basis of the following conservative assumptions and data:

- Primary coolant design activity as given in Table 5-23,
- Primary system surface activity after 32 equivalent full-power years as given in Table 5-23,
- Fraction of the primary system activity deposited on the steam generator tubes:
  - 95% for cesium and strontium
  - 90% for silver
  - 80% for iodine
- Integral fuel graphite corrosion: approximately 0.3 percent by weight,
- Mean activity concentration of the corroded coating of the unfueled shell for radiologically representative radionuclides:

	Bq/g
I 131	8.9 E+3
Cs 137	7.0 E+2
Sr 90	5.6 E+2
Ag 110 m	1.3 E+3

- Remobilization mechanisms for the fission products deposited on the steam generator surfaces:
  a) Direct, 100% mechanical decontamination of the surface affected by the impinging water/steam jet (conservative assumption: 10 m^2)
  b) Reaction of water with the fission products deposited on the surface for the duration of the wetting period of approximately 60 seconds
  c) Reaction of steam with the fission products deposited on the surfaces inside the steam generator for a duration of approximately 5 hours

For the reaction rates of the water and steam with the fission products, experimentally established data were used for cesium, while data derived from these were employed for strontium, iodine and silver.

- 10% of the gas-borne or remobilized primary system activity is discharged until the safety valve is closed.
- No deposition of activity in the reactor building.
- Efficiency of the filter unit in the secured sub-atmospheric pressure system:
  - 0% for noble gases
  - 99% for iodine
  - 99.9% for solids

The filter efficiency selected for iodine allows for the presence of organic iodine compounds and is extremely conservative for elemental iodine, which is primarily to be expected. The iodine is released to the atmosphere in the radiologically most effective form, i.e., elemental iodine.

The activity release to the reactor environment and calculated on the basis of the boundary conditions specified above is listed in Table 5-7. Also the activity releases are listed without consideration of the filter system. Table 5-17 and Table 5-18 show the potential radiation exposure determined on the basis of the regulations and data stated in Chapter 4 of the "BMI Incident Calculation Bases." For the maximum exposed organ, with consideration of the filter system, it is four orders of magnitude below the emergency reference levels pursuant to Article 28.3 of the German Radiological Protection Ordinance. Even without consideration of the filter system the radiation exposure is two orders of magnitude below the emergency reference levels.

5.6.1.3 Assessment

A review of the safety analysis performed for the water ingress caused by steam generator tube rupture indicates that:

Accident scenarios: The event scenario examined is complete and conservative

Dose assessment: The resultant dose consequences calculation are made with conservative assumption

Prevention and Mitigation: Adequate prevention and mitigation measure are in place to address these events

Conclusions: The radiological impact to environment of a steam generator tube rupture accident is calculated on the basis of conservative boundary conditions. The activity releases are listed with and without consideration of the filter system. Table 5-17 and Table 5-18 show the potential radiation exposure determined on the basis of the German regulations. For the maximum exposed organ, with consideration of the filter system, the results are four orders of magnitude below the emergency reference levels established by the German regulations (See Chapter 7 for U.S.A. dose calculations – also acceptable). Even without consideration of the filter system the radiation exposure is two orders of magnitude below the emergency reference levels.

5.7 Secondary Side Leaks and Breaks Events

Leaks and breaks in the water/steam system (except for tube rupture discussed in 5.6) cause loadings in the steam generator region and loadings on the reactor building if the break location is inside the building.

These accidents result in a scram on:

- Mass flow ratio (primary to secondary side) less than or equal to 0.75,
- Mass flow ratio (primary to secondary side) greater than or equal to 1.3,
- Cold gas temperature greater than or equal to approximately 280°C,
- Negative variable limit for main steam pressure greater than or equal 8 bar/min.

The protective actions reactor scram, primary helium circulator trip and secondary system isolation are taken as countermeasures.

The feedwater is supplied to the steam generator tubes via the feedwater nozzles to which the tubes are joined by a tubesheet. Similarly, the main steam enters the main steam nozzles via a tubesheet. These nozzles are brought out of the steam generator cavity. The main steam line is connected to the main steam nozzles outside the steam generator cavity. The feedwater line and steam generator relief lines are connected to the feedwater nozzle.

The two steam generators can be isolated from the secondary system by the reactor protection system by means of two independent automatic quick-closing valves in both the feedwater and main steam lines. The valve closure times are less than 2 seconds. However, the main steam valves close after a delay of approximately 30 seconds to rule out response of the main steam safety valve.

The maximum loadings in the steam generator region and the maximum steam or water blowdown rates from the steam generator are obtained for 2A breaks in the largest connecting lines, 180 mm, directly at the nozzles.

The analyses do not take account of friction losses and flow restrictions during blowdown.

5.7.1 Feedwater Line Break

5.7.1.1 Event Description

A 2A break in the feedwater line interrupts feedwater flow and immediately leads to a reactor scram due to “mass flow ratio (primary to secondary side) greater than or equal to 1.3”. If that scram fails, then the reactor scrams on violation of the main steam pressure limit. Flow in the steam generator reverses and hot water enters the lower, cold regions of the steam generator. The temperature increases in all steam generator regions.

Figure 5-18 shows the cold gas temperature coolant flow through the primary system when the primary gas circulator is tripped. Depending on the break location, the steam generator is drained via the break or is isolated from it.

5.7.1.2 Barrier Performance

The pressure and temperature loadings on the reactor building are covered by the main steam line break.

5.7.1.3 Radiological Consequences

An evaluation of the radiological consequences is not required for this event since no radioactive material is released from the fuel.

5.7.1.4 Assessment

A review of the safety analysis performed for the main feedwater line break accident indicates that:

Accident scenarios: The event scenario examined is complete and conservative

Dose assessment: There is no dose consequence associated with this accident

Prevention and Mitigation: Adequate prevention and mitigation measure are in place to address these events

Conclusions: The HTR-Module design meets the intent of the current and the anticipated U.S.A. regulatory requirements for the feedwater line break accident.

5.7.2 Main Steam Line Break

5.7.2.1 Event Description

For a 2A break of the main steam line, the pressure in the steam generator drops abruptly within a few seconds. The reactor scram is initiated by fulfillment of the criterion "negative variable limit for main steam pressure". Loading of the steam generator due to the dryout process is slight.

The steam escaping from the defective steam generator enters the reactor building where it causes an increase in the pressure and temperature of the building atmosphere. Before the main steam isolation valves are closed, backflow of main steam into the reactor building compartment of the intact reactor is prevented by check valves.

The various reactor building zones are interconnected by relief ports that limit the pressure differences inside the building.

If the differential pressure between the building atmosphere and the environment rises above 0.1 bar, the outer pressure relief ports open, releasing air or a mixture of steam and air from the reactor building to the environment.

The building structures and components only heat up to an insignificant extent on account of the short accident duration and the low temperature during dryout of the steam generator. The temperatures given in Figure 5-20 are maximum values for the mentioned building regions.

The steam concentrates in the vicinity of the break and along a flow path to the outer pressure relief ports. Steam does not enter the greater part of the reactor building, in particular the area housing the unaffected reactor and the escape routes in the reactor hall and services tract.

For failure of the first level reactor scram, supply to the steam generator can be continued for a longer period thus resulting in blowdown of low-temperature steam into the reactor building.

5.7.2.2 Barrier Performance

For the case where the reactor scrams on the first protection level "mass flow ratio (primary to secondary side) less than or equal to 0.75" loading of the steam generator due to the dry-out process is slight.

For the case where the first level scram fails, loadings on the steam generator, the reactor building and safety-related plant equipment are acceptable.

5.7.2.3 Radiological Consequences

An evaluation of the radiological consequences is not required for this event since no radioactive material is released from the fuel.

5.7.2.4 Assessment

A review of the safety analysis performed for the main steam line break accident indicates that:

Accident scenarios: The event scenario examined is complete and conservative

Dose assessment: There is no dose consequence associated with this accident

Prevention and Mitigation: Adequate prevention and mitigation measure are in place to address these events

Conclusions: The HTR-Module design meets the intent of the current and the anticipated U.S.A. regulatory requirements for the main steam line break accident.

5.7.3 Steam Generator Breaks/Tube Ruptures

This accident is fully covered in Section 5.6 – Water ingress event.

5.8 Disturbances in Auxiliary and Supporting Systems

5.8.1 Breaks or Leaks in Primary Coolant Conveying Components - Components outside the Reactor Building

5.8.1.1 Event Description

The design basis accident involving the release of radioactive materials from primary-coolant-conveying components outside the reactor building is the rupture of a pipe of the helium purification system in the reactor auxiliary building. As explained in Section 5.4.1, the primary system is isolated from the defective purification train by the primary system isolation valves that are actuated by the reactor protection system.

In the reactor auxiliary building, the blown down primary coolant causes the pressure relief port of the building to release to the vent stack. It is assumed that the total released activity enters the power plant environment via the vent stack. In compliance with German regulation in the "BMI Incident Calculation Bases", the influence of the reactor building on the atmospheric dispersion was taken into account.

If failure of the primary system isolation valve to close is assumed, the affected reactor is also depressurized via the break in addition to the effects described above.

5.8.1.2 Barrier Performance

Not evaluated because there is very little change in the gross core characteristics of temperature and pressure.

5.8.1.3 Radiological Consequences

For calculation of the radiological impact outside the reactor building, it was conservatively assumed that primary system isolation is only effected when the primary system pressure has dropped to 50 bar and that primary coolant is blown down through the break up to this time. Furthermore, failure of isolation measures inside the helium purification system is assumed; consequently this system is totally depressurized. In addition to release of the activity borne by the primary coolant, entrainment of the noble gas, tritium and C14 activity adsorbed in the molecular sieve and cryogenic adsorber is also assumed. In contrast, desorption of solid fission products (primarily noble gas decay products) is negligible. Table 5-8 shows the calculated total activity that has been released.

The potential radiation exposure in the environment of the power plant is shown in Table 5-19. In compliance with the "BMI Incident Calculation Bases", the influence of the reactor building on the atmospheric dispersion was taken into account. If part of the activity releases enters the power plant environment via building leakages, the data of Table 5-19 will increase. In Table 5-20 the results are listed assuming that the whole activity release takes place via building leakages. The doses are clearly below the emergency reference levels given in Article 28.3 of the German Radiological Protection Ordinance.

5.8.1.4 Assessment

A review of the safety analysis performed for the breaks helium purification system accident indicates that:

Accident scenarios: The event scenario examined is complete and conservative

Dose assessment: The dose consequence associated with this accident are calculated with conservative assumptions

Prevention and Mitigation: Adequate prevention and mitigation measure are in place to address these events

Conclusions: The radiological release associated with this accident is in compliance with the German and U.S.A. regulations. The dose calculation also indicates that the exposure in the environment of the power plant (assuming building leakage) is also in compliance with the German and U.S.A. (see Chapter 7) radiological exposure regulations.

5.8.2 Breaks of a Vessel Containing Radioactively Contaminated Water

5.8.2.1 Event Description

A break in an evaporator concentrate vessel is postulated for this accident. Total discharge of the inventory from an evaporator concentrate vessel is assumed for calculation of the radiological impact of vessel failure in the reactor auxiliary building. The activity inventory of the vessel is given in Table 5-9.

5.8.2.2 Barrier Performance

Not evaluated because there is very little change in the gross core characteristics of temperature and pressure.

5.8.2.3 Radiological Consequences

In order to establish the activity entering the air from the discharged concentrate, it was assumed that the evaporated fraction of 1% contains a weight-related activity concentration amounting to 5% of the concentration in the concentrate. For airborne iodine, the elemental form was assumed for the sake of conservatism.

The activity is released to the environment of the power plant with the exhaust air via the stack. The calculated activity releases are listed in Table 5-10.

The values for potential radiation exposure in the environment of the power plant that were calculated on the basis of the above assumptions are shown in Table 5-21. The doses are much lower than the emergency reference levels of Article 28.3 of the German Radiological Protection Ordinance.

No radioactive material is released from the fuel as the result of this accident.

5.8.2.4 Assessment

A review of the safety analysis performed for the breaks waste concentrate vessel accident indicates that:

Accident scenarios: The event scenario examined is complete and conservative

Dose assessment: The dose consequence associated with this accident are calculated with conservative assumptions

Prevention and Mitigation: Adequate prevention and mitigation measure are in place to address these events

Conclusions: The radiological release associated with this accident is in compliance with the German regulations. The dose calculation also indicates that the exposure in the environment of the power plant (assuming building leakage) is also in compliance with the German radiological exposure regulations. Current NRC regulations treats this accident as an anticipated operational occurrence (AOO), therefore 10CFR Part 20 limits would be imposed. See Chapter 7 for worst case dose calculations under the NRC limits and regulations.

5.9 Seismic Effects on the Reactor Auxiliary Building

5.9.1 Effect of Seismic Event

5.9.1.1 Event Description

No seismic stability, integrity or functional analyses are carried out for the systems in the reactor auxiliary building.

In order to establish potential radiological impact on the environment, breaks in the activity-retaining systems and components and direct ground-level release are assumed.

5.9.1.2 Barrier Performance

Not evaluated because there is very little change in the gross core characteristics of temperature and pressure.

5.9.1.3 Radiological Consequences

The principal activity inventories that can be released via building leaks are only present in the components of the helium purification system, the storage tank for radioactively contaminated helium, the holding tank of the water extraction system and the evaporator concentrate vessels. The effects of ground-level release of the inventory of one train of the helium purification system are described in Section 5.8.1 (Table 5-20).

The effects of release of the inventory of one evaporator concentrate vessel via the stack are given in Section 5.8.2 (Table 5-21). Ground level release of this inventory only heightens these effects to an insignificant extent.

The activity release presented in Table 5-11 covers simultaneous breaks in all important activity-retaining systems in the reactor auxiliary building. The resulting radiation exposure in the environment of the plant is given in Table 5-22.

The values are therefore much lower than the emergency reference values stated in Article 28.3 of the German Radiological Protection Ordinance

No radioactive material is released from the fuel as the result of this accident.

5.9.1.4 Assessment

A review of the safety analysis performed for the breaks in auxiliary building activity retaining systems and components as the result of a seismic event indicates that:

Accident scenarios: The event scenario examined is complete and conservative

Dose assessment: The dose consequence associated with this accident are calculated with conservative assumptions

Prevention and Mitigation: Adequate prevention and mitigation measure are in place to address these events

Conclusions: The radiological release associated with this accident is in compliance with the German regulations. The dose calculation also indicates that the exposure in the environment of the power plant (assuming building leakage) is also in compliance with the German radiological exposure regulations. Current NRC regulations treats this accident as an anticipated operational occurrence (AOO), therefore 10CFR Part 20 limits would be imposed. See Chapter 7 for worst case dose calculations under the NRC limits and regulations.

5.10 Failure of the Auxiliary Power Supply and Unavailability of the Emergency Diesel

In failures of the auxiliary power supply a distinction is drawn between the short-term failure (< 2 hours) and the long-term failure (< 15 hours). See Section 5.12 for discussions for the hypothetical case of long term loss of cavity cooler. For accident conditions it is assumed that one emergency diesel is out-of-service in maintenance or repair and that a single failure has resulted in a second emergency diesel failure to start.

5.10.1 Short-term Failure of Auxiliary Power and Unavailability of Emergency Diesel

The effects on the primary and secondary system due to failure of auxiliary power and unavailability of an emergency diesel are described in Section 5.3.

In case of unavailability of both emergency diesels all users, except for those connected to the continuous battery power supply, are supplied by the auxiliary supply and emergency power network that include the users in the secured sub-atmospheric pressure system and the closed-circuit cooling chain when the cavity cooler malfunctions.

The control room, the reactor protection system and the operational instrumentation and control are supplied by the 220 V batteries that are available for at least 2 hours.

The users of the remote shut-down station are supplied by 24 V batteries with a capacity of 15 hours.

The plant is in a secure state after the execution of the following reactor protection measures:

- Reflector rod drop
- Primary gas circulator trip
- Steam generator isolation

No further measures are required until the auxiliary power supply is restored.

All reactor protection measures are performed without the availability of electrical power.

Failure of the cavity cooler is acceptable for a timeframe up to 15 hours without exceeding design limits (see Section 5.10.2)

Since the reactor protection system is still intact, the accident specific reactor protection measures function as designed even when this event is assumed to occur as a result or in conjunction with design basis accidents (e.g., the primary system is isolated when depressurized; the steam generator relief is actuated from water ingress due to SG tube rupture).

There are radiological effects from combining this event with a depressurization accident. The environmental impact in accordance with the listed values in Section 5.4.1 during the depressurization phase will result in a maximal 55 mrem thyroid exposure of an infant.

Within 2 hours, the maximum fuel temperature increases by 50°C up to approximately 900°C and no additional release of fission products can occur.

Design basis accidents in conjunction with this event will not affect accident boundary conditions for the analyses described in Section 5.2 because the reactor protection system performs the required reactor protection measures.

The main heat transfer system and the cavity cooler are available again for cooling after 2 hours so that the plant can be cooled down.

5.10.2 Long-term Failure of Auxiliary Power and Unavailability of Emergency Diesel

Long-term failure of auxiliary power (≤ 15 h) is unlikely due to the high network quality, therefore the long-term failure of auxiliary power is not assumed in conjunction with other design basis accidents.

Failure of auxiliary power actuates the following reactor protection measures:

- Reflector rod drop
- Primary gas circulator trip
- Steam generator isolation

The plant is thereby in a secured state.

The 220 V batteries are assumed to be not available after 2 hours. Therefore, except for the remote shut-down station required for plant monitoring, other systems such as the reactor protection system, the control room and other instrumentation and controls will no longer have an available electrical power supply.

If not yet actuated, as a consequence of the electrical power failure the reactor protection system and selected control equipment and switch gear will trigger isolation of the primary system and steam generator relief. After these protection measures, the reactor protection system has no further protection functions.

The core and the core internals heat up slowly due to the unavailability of the main heat transfer system.

The plant is designed so that design limits are not exceeded due to unavailability of the cavity cooler for up to 15 hours. After this time, the temperature of the reactor pressure vessel increases to a maximum of 310°C , the cavity cooler to 220°C and of the reactor cavity concrete to 150°C (Figure 5-19).

The closed cooling water systems are designed for the increased pressures and temperatures so there will be no safety valves actuated.

After about 10 hours it can be expected that the pressure in the primary system increases up to the response pressure of the primary system safety valve of 69 bar that will blow down for about 30 minutes. For each blow down during the pressure relief phase, about 10% of the discharged fission products that would be released due to the break of a large connection line between the pressure vessel unit and the primary system isolation will be released into the reactor building.

Ceramic fuel particles are designed for very high temperatures and so the temperature increase of the fuel elements up to approximately 1130°C will have negligible contribution to the fission products released.

It is conservatively assumed that the relief port of the building does not open and the release takes place near the ground. Therefore, the environmental exposure impact is less than for the case of seismic effects in the reactor auxiliary building (Section 5.9), where a near-ground release takes place as well.

After the auxiliary power is restored the accident will be terminated by cooling down the plant through the start-up and shut-down circuit and operation of the cavity cooler.

In case that the auxiliary power is still not available after 15 hours, the external feed-in into the cavity cooler through the fire brigade connection prevents the design values of the reactor pressure vessel from being exceeded long-term failure of auxiliary power and unavailability of both emergency diesels, see Table 5-3.

5.11 Postulated Transient without Reactor Scram (ATWS)

For the HTR-Module, the reactor shut-down under all accident conditions and transients that lead to response of the reactor protection system is performed by the reflector rod drop.

In case of rods fail to drop the increase of the fuel element temperature resulting from the negative temperature coefficient of the reactivity will result in reactor shut-down. In any case the reactor protection system shuts down the primary gas circulator, simultaneously closing the circulator damper (either of which interrupts reactor cooling), causing the increase in the fuel element temperature. The long-term shut down is performed by initiation of the small ball shut-down system that is performed by manual action.

The peak fuel element temperature remains far below the maximum tolerable values.

5.12 Failure of Cavity Cooler (a Hypothetical Event Assessment)

The HTR-Module plant cavity cooler (reactor cavity cooling system –RCCS) surrounds the reactor pressure vessel at a distance of approximately 1.5 meters. It is installed as a closed pipe wall in front of the concrete walls of the reactor cavity at a distance of approximately 10 cm. Cooling water is circulated upwards at a temperature of 40 °C and at a pressure of 5 bar.

The cavity cooler function is to discharge the reactor heat loss (approximately 400 kW during normal operation) and to remove the residual heat of the reactor core of up to 850 kW (design capacity of the cavity cooler) during failure of the main heat transfer system. The cavity cooler is constructed of three trains.

Two trains are connected to a separate secured service water system, whereby the electrical loads are connected to the double-trained emergency power supply of the plant. The ultimate cooling of these two trains takes place via separate wet cooling towers.

The third train of the cavity cooler is connected to the nuclear service water system for nuclear cooling points. This system is not secured by emergency power and its ultimate cooling is performed by the plant cooling tower.

The cooling points of the support brackets of the reactor pressure vessel and the cooling points of the fuel element discharge are also constructed of three trains and are cooled in parallel to the cavity cooler.

The two trains of the secured service water system that are equipped with permanently available hose connections that are easily accessible in the reactor building, are a special safety precaution, in order to supply the cavity cooler externally even in the case of failure of all cooling systems.

An evaluation of the safety characteristics of the HTR-Module is the behavior of the plant after total long-term failure of all active equipment. In this case only the natural and physical design characteristics of the plant will react to mitigate and limit the consequences of an accident.

An evaluation of the cavity cooler failure has been performed (reference [6]) where a hypothetical failure of cavity cooler is assumed and the thermal consequences evaluated. For this hypothetical case only the natural and physical phenomena such as thermal conduction and thermal radiation remove the accruing residual heat of the fuel elements out of the reactor core. Based on the results of this evaluation and for an extreme case of a primary system depressurization accident concurrent with total failure of the cavity cooler it has been shown that:

- The maximum fuel temperature is at 1540 °C it does not depend on the functioning of the cavity cooler therefore the fission product release is bounded by the depressurization accident releases and is fully acceptable.
- The reactor pressure vessel will maintain its design integrity.
- The reactor support system will maintain their function.
- The stability of the concrete cavity will be retained for any length of time.
- Effects of dust on plant safety are discussed in Section 6.3.

5.13 Effects of the Considered Events on the Overall Plant

The HTR-Module power plant is designed to permit independent operation of the two reactors.

Each reactor has a separate, dedicated reactor protection system that monitors the operating parameters and, in the event of accidents, only shuts down the affected reactor.

The design ensures that, for all reactor accidents (e.g., reactivity accidents, loss of flow events and breaks in the primary system or steam generator) only the affected reactor is shut down automatically, while the other can be kept in operation.

All accidents lead to uncoupling of the affected reactor from the overall plant by shut down and secondary-side isolation of the steam generator.

Likewise, steam generator relief, that is additionally required in the event of tube breaks, does not prevent operation of the other reactor.

Depending on their size, primary-side breaks can cause a short-term pressure build-up in the reactor building. In principle, it is possible to continue operation of the unaffected reactor.

The effects of events affecting the overall plant (e.g., emergency power operation or-external events) are detected by the reactor protection system assigned to each reactor and cause shut-down of the affected reactor.

No safety requirements are placed on the water/steam cycle and the start-up and shut-down circuits. They are designed and operated as non-nuclear plant items only.

For reasons of availability, there are two complete water/steam cycles. These are uncoupled from each other to such an extent that the majority of the anticipated malfunctions can be handled by the controls. Therefore in most cases there is no need to shut down both reactors.

In addition to being equipped with a separate water/steam cycle, each reactor is provided with a start-up and shut-down circuit. This allows each reactor to be started up and shut down independently of the water/steam cycle.

5.14 Summary of Calculated Radiation Doses

The following accidents described in Chapter 5 are enveloping accidents for the overall plant from the radiological standpoint:

- 5.4.1 Break of a large connecting line between pressure vessel unit and primary system isolation valve,
- 5.4.3 Break of an instrument line and small breaks,
- 5.6.1 Steam generator breaks/tube rupture,
- 5.8.1 Breaks or leaks in primary-coolant-conveying components outside the reactor building,
- 5.8.2 Break of a vessel containing radioactively contaminated water,
- 5.9 Seismic effects on the reactor auxiliary building.

Radiation exposure in the environment after these accidents is calculated in accordance with the German regulations listed in Chapter 4 of the “Incident Calculation Bases for the Guidelines issued by the German Federal Minister of the Interior (BMI) for the Assessment of the Design of PWR Nuclear Power Plants pursuant to Sec 28, paragraph (3) of the Radiological Protection Ordinance” (Bundesanzeiger No. 245 a, dated 31.12.83).

The Table 5-12 to Table 5-22 gives the radiation exposures for relevant organs for the critical population group.

Table 5-1: Accidents for which Radiological Impact on the Environment is Relevant

Accident Group I		Accident Definition	Analysis	Remarks
I.1	Primary system depressurization			Catastrophic failure of the pressure vessel unit (RPV, gas duct pressure vessel. SG pressure vessel) can be ruled out thanks to quality assurance measures.
I.1.1		Break in a connecting pipe between reactor pressure vessel and primary system isolation valve	RA AS SI	Calculation of the radiological impact and the analysis for design of the building pressure relief system is based on a break size equal to the cross-sectional area of the largest connecting pipe (DN65). The release of primary coolant is unfiltered in the depressurization phase and in the core heat-up phase over the vent stack. The analysis is used to design measures for limiting air ingress into the primary system.
I.1.2		Break in a connecting pipe downstream of the primary system isolation valve inside the reactor building	AS	The radiological impacts are enveloped by accident I.1.1. The effects of malfunctions of the auxiliary and supporting systems inside the reactor building are therefore covered.
I.1.3		Break in a primary coolant conveying instrument line	RA AS	Analysis to verify the adequacy of the secured sub-atmospheric pressure system is based on a break size equal to the cross sectional area of a connecting pipe with a diameter of 10 mm. The release of activity is unfiltered via the vent stack.
I.1.4		Inadvertent opening of a safety valve and that valve stuck open	AS	Radiological Impacts are enveloped by accident I.1.1
I.2	Damage to steam generator tubes			
I.2.1		Rupture of a steam generator tube (2F)	AS	The analysis is used to design measures for limiting water ingress into the primary system.

Accident Group I		Accident Definition	Analysis	Remarks
I.2.2		Rupture of a steam generator tube concurrent with long term failure of water separation equipment and the primary system pressure control system	RA	The radiological impacts result from response of the primary side pressure relief system. The release of activity is unfiltered via the vent stack.
I.3	Malfunctions of the auxiliary and supporting systems outside the reactor building with radiological impact - Helium purification system - Liquid waste system			
I.3.1		Break in the largest primary coolant-conveying line outside the reactor building and closure of the primary system isolation valves	RA AS SI	Break in a line of the helium purification system is postulated for analysis of the radiological impact and design of the reactor auxiliary building. The release of activity is near ground via the building leakages.
I.3.2		Break in a vessel containing radioactively contaminated water	RA	The vessel selected is radiologically representative of all other vessels. The release of activity is unfiltered via the vent stack.
I.4	Earthquake	Effects on reactor building - switchgear and emergency supply building - service water systems - reactor auxiliary building	AS SI RA SI	Design of structural items, systems and components according to seismic requirements. Breaks in the components of the helium purification system and in the evaporator concentrate tanks are postulated for analysis of the radiological impact. The release of activity is near ground via the building leakages.

Table 5-2: Accidents for which Radiological Impact on the Environment is not Relevant

Accident Group II		Accident Definition	Analysis	Remarks
II.1	Reactivity accidents			
II.1.1	Malfunction of control and shut-down elements	Withdrawal of all reflector rods at different initial conditions	AS	Power rise, scram when limits reached
II.1.2	Steam generator breaks	Water ingress into the primary system	AS	Enveloped by II.1.1 with regard to reactivity feedback
II.2	Malfunction of the main heat transfer system	Loss of auxiliary power supply	AS	Failure of operational heat sink, scram, residual heat removal via cavity coolers
II.2.1		Failure of the auxiliary power supply	AS	Failure of operational heat sink, scram, residual heat removal via cavity coolers, operation of the emergency diesel.
II.2.2		Short-term failure of the auxiliary power supply (≤ 2 hours) and unavailability of both emergency diesels.	AS	This analysis serves as proof that on postulation of a single failure during maintenance of emergency power supply, all required protection measures can be initiated and no limit values are going to be exceeded. Failure of all heat sinks. Initiation of the reactor protection measures according to II.2.1. No sub-atmospheric pressure system inside the building. On network return after ≤ 2 hours, operational shut-down of the power plant. On superposition of design accidents, initiation of accident-related reactor protection measures. Radiological effects are covered with the corresponding cases in Table 5-1.

RA: The radiological impacts are calculated.
AS: These accidents are analyzed for the purpose of designing engineered safety features or countermeasures.
SI: The analysis of these accidents is used for the design of components or structural items for stability or integrity.
VO: An accident analysis is not necessary due to the precautionary measures listed in the fourth column.

Accident Group II		Accident Definition	Analysis	Remarks
II.2.3		Long-term failure of the auxiliary power supply (≤ 15 hours) and unavailability of both emergency diesels.	AS	In the long term, stability and integrity of the primary circuits will be maintained and no limit values exceeded, on postulation of single failure during maintenance of emergency power supply. Failure of all heat sinks. Initiation of the reactor protection measures according to II.2.1. After 2 hours, failure of DC supply and as a result automatic initiation of the remaining protection measures, if not initiated manually already. Simultaneous failure of control room is valid, monitoring by remote shut-down station. After 10 hours, response of the safety valves of the primary circuits; no sub-atmospheric pressure system inside the building. On network return after ≤ 15 hours, operational shut-down of the plant.
II.3	Secondary side leaks and breaks			
II.3.1		Break of feedwater line	AS SI	Failure of operational heat sink, scram, residual heat removal via cavity coolers Stability of steam generator, tube integrity and effects of reaction and jet impingement forces, flooding, loadings on building
II.3.2		Break of main steam line	AS, SI	cf. event II.3.1 Backflow from the adjacent reactor is prevented by a check valve.
II.4	In-plant fires and explosions	In-plant fires and explosions	VO	These accidents and/or unacceptable effects of these accidents are prevented by active and passive fire protection measures, e.g., minimization of fire loads, installation remote from ignition sources, fire zones, fire dampers in HVAC systems, and by explosion protection measures.

Accident Group II		Accident Definition	Analysis	Remarks
II.5	Failure of large components	Effect of turbine failure	VO	Unacceptable effects are prevented by installing the turbines in line with the reactor building or by means of an adequately dimensioned turbine casing.
II.6	Other external events	External fire	VO	Unacceptable effects of external fires are prevented by the measures taken against aircraft crash, blast waves due to chemical reactions and hazardous materials.
		High water	VO	Unacceptable effects are prevented by structural provisions
		Lightning	VO	Unacceptable effects are prevented by suitable lightning protection
II.7	Load drop	Failure of reactor building crane	VO	Design to tight requirements in compliance with KTA 3902

RA: The radiological impacts are calculated.
AS: These accidents are analyzed for the purpose of designing engineered safety features or countermeasures.
SI: The analysis of these accidents is used for the design of components or structural items for stability or integrity.
VO: An accident analysis is not necessary due to the precautionary measures listed in the fourth column.

Table 5-3: Non DBAs* but against which Measures are Taken to Minimize Risk

*Due to low probability

Accident Group III		Accident Definition	Analysis	Remarks
III.1	External man-induced events	Aircraft crash External events hazardous materials External blast waves due to chemical reactions	-	Design in compliance with Section 4.6 Design in compliance with Section 4.6 Design in compliance with Section 4.6
III.2	Postulated transient without reactor scram (ATWS)	Operational transient without reflector rod drop	-	Will be controlled by interruption of primary coolant mass flow via the negative temperature coefficient of the reactivity
III.3	Long-term failure of power supply	Long-term failure of auxiliary power supply (>15 hours) and unavailability of emergency diesel	-	Course of events up to 15 hours according to II.2.3. After 15 hours external feed of the cavity cooler and supply of the remote shut-down station.

Table 5-4: Activity Release in the Depressurization Phase after Break of a Large Connecting Line (DN65)

Nuclide	Activity Release in Bq (short-term release)
Kr83m	2. 1 E+10 [1)]
Kr85m	7.6 E+10
Kr85	1.6 E+10
Kr87	7.8 E+10
Kr88	1.6 E+11
Kr89	2.8 E+10
Kr90	1.2 E+10
Xe131m	2.3 E+10
Xe133m	5.5 E+10
Xe133	2.3 E+12
Xe135m	1.8 E+10
Xe135	2.4 E+11
Xe137	4.8 E+10
Xe138	9.3 E+10
Xe139	1.5 E+10
Total noble gases	3.3 E+12
I131	1.5 E+8
I132	1.8 E+9
I133	8.9 E+8
I134	4.7 E+9
I135	1.6 E+9
Total iodine	9.1 E+9
Sr90	5.6 E+5
Ag110m	2.0 E+5
Cs134	1.6 E+6
Cs137	2.9 E+6
Total long-lived solids	5.3 E+6
Rb88	5.3 E+10
Cs 138	2.0 E+10
Total noble gas decay products	7.3 E+10
Tritium (H3)	5.6 E+12
C14	6.0 E+10

1) 2.1 E+10 = 2.1×10^{10}

Table 5-5: Activity Release in the Core Heat-Up Phase after Break of a Large Connecting Line

With consideration of the filter system

	Activity Release in Bq in the Time Interval				
	0-8h	8-24h	24-72h	72h	Total
Xe133	2.6 E+8	1.9 E+10	1.6 E+11	8.5 E+10	2.7 E+11
I131	1.3 E+5	9.2 E+6	8. 1E+7	4. 1E+7	1.3 E+8
Sr89	3.0 E+1	5.5 E+3	1.1 E+5	1.1 E+5	2.2 E+5
Sr90	1.4 E+0	2.6 E+2	5. 1E+3	5.0 E+3	1.0 E+4
Ag110m	1.4 E+2	5.8 E+3	4.0 E+5	8.4 E+5	1.2 E+6
Cs134	1.0 E+3	5. 1 E+4	8. 1E+5	8.7 E+5	1.7 E+6
CS137	1.2 E+3	5.9 E+4	9.4 E+5	1.0 E+6	2.0 E+6

Without consideration of the filter system

	Activity Release in Bq in the Time Interval					
	0-34 h	34-42 h	42-58 h	58-106h	> 106 h	Total
Xe133	5.8 E+10	3.4 E+10	5.8 E+10	9.7 E+10	2.7 E+10	2.7 E+11
J131	2.7 E+9	1.6 E+9	2.8 E+9	4. 6 E+9	1.3 E+9	1.3 E+10
Sr89	2.1 E+7	1.9 E+7	4.1 E+7	1.0 E+8	3.7 E+7	2.2 E+8
Sr90	1.0 E+6	8.8 E+5	2.0 E+6	4.7 E+6	1.8 E+6	1.0 E+7
Ag110m	3.6 E+7	3.2 E+7	1.6 E+8	6.7 E+8	3.4 E+8	1.2 E+9
Cs134	1.7 E+8	1.4 E+4	3. 1E+8	7.8 E+8	3.3 E+8	1.7 E+9
CS137	2.0 E+8	1.6 E+8	3.6 E+8	9.1 E+8	3.8 E+8	2.0 E+9

1) 2.6 E+8 = 2.6 x 10^8

Table 5-6: Activity Release after an Instrument Line Break

Nuclide	Activity Release in Bq (short-term release)	
	I	II
Kr83m	1.2 E+10[1)]	1.2 E+10
Kr85m	4.6 E+10	4.6 E+10
Kr85	3.2 E+8	3.2 E+8
Kr87	4.0 E+10	4.0 E+10
Kr88	9.7 E+10	9.7 E+10
Kr8g	5.9 E+9	5.9 E+9
Kr90	2.2 E+9	2.2 E+9
Xel31m	1.2 E+9	1.2 E+9
Xe133m	1.1 E+10	1.1 E+10
Xe133	2.3 E+11	2.3 E+11
Xe135m	4.6 E+9	4.6 E+9
Xe135	1.3 E+11	1.3 E+11
Xe137	1.0 E+10	1.0 E+10
Xel38	2.7 E+10	2.7 E+10
Xe139	2.8 E+9	2.8 E+9
Total noble gases	**6.2 E+11**	**6.2 E+11**
I131	1.3 E+6	1.3 E+8
I132	6.8 E+6	6.8 E+8
I133	7.7 E+6	7.7 E+8
I134	8.3 E+6	8.3 E+8
I135	1.0 E+7	1.0 E+9
Total Iodine	**3.4 E+7**	**3.4 E+9**
Sr90	8.5 E+0	8.5 E+3
Ag 110m	7.7 E+1	7.7 E+4
Cs134	1.0 E+3	1.0 E+6
Cs137	2.2 E+3	2.2 E+6
Total long-lived solids	**3.3 E+3**	**3.3 E+6**
Rb88	4.6 E+6	4.6 E+9
Cs138	2.5 E+6	2.5 E+9
Total noble gas decay products	**7.1 E+6**	**7.1 E+9**

1) 1.2 E+10 = 1.2 x 10^{10}

I: with consideration of the filter system

II: without consideration of the filter system

Table 5-7: Activity Release after a SG Tube Break with Response of the Relief System

Nuclide	Activity Release in Bq (short-term release)	
	I	II
Kr 83 m	2.8 E+8 [1)]	2.8 E+8
Kr 85 m	2.7 E+9	2.7 E+9
Kr 85	3.4 E+7	3.4 E+7
Kr 87	5.0 E+8	5.0 E+8
Kr 88	3.6 E+9	3.6 E+9
Xe 131 m	1.1 E+8	1.1 E+8
Xe 133 m	1.1 E+9	1.1 E+9
Xe 133	2.3 E+10	2.3 E+10
Xe 135	9.8 E+9	9.8 E+9
Total noble gases	**4.1 E+10**	**4.1 E+10**
I 131	4.1 E+6	4.1 E+8
I 132	8.4 E+5	8.4 E+7
I 133	5.2 E+6	5.2 E+8
I 134	3.4 E+5	3.4 E+7
I 135	3.4 E+6	3.4 E+8
Total Iodine	**1.4 E+4**	**1.4 E+7**
Sr 90	1.9 E+4	1.9 E+7
Ag 110 m	3.0 E+3	3.0 E+6
Cs 134	1.2 E+5	1.2 E+8
Cs 137	1.7 E+6	1.7 E+9
Total long-lived solids	**1.8 E+6**	**1.8 E+9**

1) 2.8 E+8 is equivalent to 2.8×10^8

I: with consideration of the filter system

II: without consideration of the filter system

Table 5-8: Activity Release after a Pipe Break in the Helium Purification System Outside the Reactor Building

Nuclide	Activity Release in Bq (short-term release)
Kr83m	6.3 E+9
Kr85m	3.0 E+10
Kr85	1.6 E+10
Kr87	2.1 E+10
Kr88	5.5 E+10
Kr89	5.7 E+9
Kr90	2.4 E+9
Xe131m	2.2 E+10
Xe133m	4.8 E+10
Xe133	2. 2 E+12
Xe135m	3.8 E+9
Xe135	1.2 E+11
Xe137	9.8 E+9
Xe138	2.0 E+10
Xe139	3.0 E+9
Total noble gases	**2.5 E+12**
I131	6.8 E+6
I132	9.0 E+7
I133	4.4 E+7
I134	2.3 E+8
I135	7.7 E+7
Total Iodine	**4.5 E+8**
Sr90	4.3 E+2
Ag110m	3.8 E+3
Cs134	5.1 E+4
Cs137	1.1 E+5
Total long-lived solids	**1.6 E+5**
Rb88	2.6 E+9
Cs138	9.9 E+8
Total noble gas decay products	**3.6 E+9**
H3	5.6 E+12
C14	6.0 E+10

Table 5-9: Activity Inventory of the Evaporator Concentrate Vessel

Nuclide	Activity Inventory in Bq
Co 60	8.9 E + 10 [1)]
Sr 90	1.9 E + 08
I 131	1.9 E + 07
Cs 134	4.4 E + 09
Cs 137	5.6 E + 10

1) 8.9 E + 10 is equivalent to 8.9 x 10^{10}

Table 5-10: Activity Release after Failure of a Vessel Containing Radioactively Contaminated Water

Nuclide	Activity Inventory in Bq (short term release)
Co 60	4.4 E + 07 [1)]
Sr 90	9.3 E + 04
I 131	9.3 E + 03
Cs 134	2.2 E + 06
Cs 137	2.8 E + 07

1) 4.4 E + 07 is equivalent to 4.4×10^7

Table 5-11: Activity Releases to the Environment after Seismic Effects on the Reactor Auxiliary Building

Nuclide	Activity Release in Bq
Kr83m	1.3 E+l0 [1)]
Kr85m	6.0 E+l0
Kr85	3.2 E+l0
Kr87	4.2 E+l0
Kr88	1.1 E+11
Kr89	1.1 E+l0
Kr90	4.7 E+9
Xe131m	4.4 E+l0
Xe133m	9.2 E+l0
Xe133	4.3 E+12
Xe135m	7.6 E+9
Xe135	2.5 E+11
Xe137	2.0 E+10
Xe138	4.6 E+10
Xe139	6. 1 E+9
Total noble gases	**5. 1 E+12**
I131	1.4 E+7 [2)]
I132	1.8 E+8
I133	8.7 E+7
I134	4.6 E+8
Total iodine	**8.9 E+8**
Co60	8.9 E+7
Sr90	1.9 E+5
Ag110m	7.5 E+3
Cs134	4.6 E+6
Total long-lived solids	**1.5 E+8**
Rb88	5. 1 E+9
Cs138	2.0 E+9
Total noble gas decay products	**7. 1 E+9**
H3	1.9 E+13
C14	1.2 E+11

1) 1.3 E+10 is equivalent to 1.3 x 10^{10}
2) 1.4. E+10 is equivalent to 1.4 x 10^{10}

Table 5-12: Accident Doses in the Environment as a Result of the Depressurization Phase after Break of a Large Connecting Pipe (DN65)

(The accident is described in Section 5.4.1)

Organ	Critical Population Group	Accident Doses		Emergency Reference Levels Complying with Art. 28.3 RPO	
		Sv	rem	Sv	rem
Bone	Adult	5.57 E-5**	5.57 E-3	0.30	30
Liver	Adult	7.98 E-6	7.98 E-4	0.15	15
Whole body	Adult	9.76 E-6	9.76 E-4	0.05	5
Thyroid	Infant	5.45 E-4	5.45 E-2	0.15	15
Kidney	Adult	7.93 E-6	7.93 E-4	0.15	15
Lung	Adult	7.98 E-6	7.98 E-4	0.15	15
G.I.T.*	Adult	1.37 E-5	1.37 E-3	0.15	15
Skin	Adult	5.23 E-5	5.23 E-3	0.30	30

Emission height: H = 60 m, unfiltered release
Effective emission height: H_{eff} = 50 m (due to effect of building)
Reference point: 2001 m for thyroid; 100 m and 200 m for other organs
Predominant exposure pathway: Ingestion
Main nuclide: I 131, 1.5 E+8 Bq are released

*Gastrointestinal tract
**5.57 E-5 is equivalent to 5.57 x 10^{-5}

Table 5-13: Accident Doses in the Environment as a Result of Depressurization with Subsequent Core Heat-Up after Break of a Large Connecting Line (DN65) without consideration of filter system

(The accident is described in Section 5.4.1.2)

Organ	Critical Population Group	Accident Doses		Emergency Reference Levels Complying with Art. 28.3 RPO	
		Sv	rem	Sv	rem
Bone	Infant	5.26 E-4**	5.26 E-2	0.30	30
Liver	Infant	4.97 E-4	4.97 E-2	0. 15	15
Whole body	Adult	2.97 E-4	2.97 E-2	0.05	5
Thyroid	Infant	1.23 E-2	1.23 E-0	0. 15	15
Kidney	Adult	2.75 E-4	2.75 E-2	0.15	15
Lung	Infant	2.58 E-4	2.58 E-2	0.15	15
G.I.T.*)	Adult	2.99 E-4	2.99 E-2	0.15	15
Skin	Adult	2.97 E-4	2.97 E-2	0.30	30

Emission height: H = 60 m, unfiltered release/

Effective emission height: H_{eff} = about 50 m (due to effect of building)

Reference point: 2001 m

Predominant exposure pathway: Ingestion

Main nuclide: I 131, 1.3 E+10 Bq are released

*Gastrointestinal tract

**5.26 E-4 is equivalent to 5.26×10^{-4}

Table 5-14: Accident Doses in the Environment as a Result of Depressurization with Subsequent Core Heat-Up after Break of a Large Connecting Line (DN65) with Consideration of Filter System

(The accident is described in Section 5.4.1.2)

Organ	Critical Population Group	Accident Doses		Emergency Reference Levels Complying with Art. 28.3 RPO	
		Sv	**rem**	**Sv**	**rem**
Bone	Adult	5.57 E-5**	5.57 E-3	0.30	30
Liver	Adult	8.03 E-6	8.03 E-4	0.15	15
Whole body	Adult	9.80 E-6	9.80 E-4	0.05	5
Thyroid	Infant	6.22 E-4	6.22 E-2	0.15	15
Kidney	Adult	7.97 E-6	7.97 E-4	0.15	15
Lung	Adult	8.02 E-6	8.02 E-4	0.15	15
G.I.T.*	Adult	1.39 E-5	1.39 E-3	0.15	15
Skin	Adult	5.23 E-5	5.23 E-3	0.30	30

Emission height: H=60 m, unfiltered release during depressurization
H=60 m, filtered release during core heat-up

Effective emission height: H_{eff}= 50 m (due to effect of building)

Reference point: 100 m and 2001 m for thyroid; 100 m, 500 m and 2001 m for other organs

Predominant exposure pathway: Ingestion

Main nuclide: I 131, 2.8 E+8 Bq are released

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*Gastrointestinal tract

**5.57 E-5 is equivalent $5.57x10^{-5}$

Table 5-15: Accident Doses in the Environment as a Result of the Depressurization Phase after Break of an Instrument Line without Consideration of Filter System

(The accident is described in Section 5.4.3)

Organ	Critical Population Group	Accident Doses		Emergency Reference Levels Complying with Art. 28.3 RPO	
		Sv	rem	Sv	rem
Bone	Infant	4.39 E-6**	4.39 E-4	0.30	30
Liver	Infant	4.48 E-6	4.48 E-4	0.15	15
Whole body	Adult	2.43 E-6	2.43 E-4 3 E-5	0.05	5
Thyroid	Infant	4.88 E-4	4.88 E-2	0.15	15
Kidney	Adult	2.31 E-6	2.31 E-4	0.15	15
Lung	Infant	2.44 E-6	2.44 E-4	0.15	15
G.I.T.*	Adult	5.25 E-6	5.25 E-4	0.15	15
Skin	Adult	2.43 E-6	2.43 E-4	0.30	30

Emission height: H = 60 m, unfiltered release
Effective emission height: H_{eff} = 50 m (due to effect of building)
Reference point: 2001 m for thyroid; 100 m for other organs
Predominant exposure pathway: Ingestion
Main nuclide: I 131, 1.3 E+8 Bq are released

*Gastrointestinal tract
**4.39 E-6 is equivalent 4.39×10^{-6}

Table 5-16: Accident Doses in the Environment as a Result of the Depressurization Phase after Break of an Instrument Line with Consideration of Filter System

(The accident is described in Section 5.4.3)

Organ	Critical Population Group	Accident Doses		Emergency Reference Levels Complying with Art. 28.3 RPO	
		Sv	rem	Sv	rem
Bone	Infant	8.47 E-7**	8.47 E-5	0.30	30
Liver	Infant	8.49 E-7	8.49 E-5	0.15	15
Whole body	Infant	8.42 E-7	8.42 E-5	0.05	5
Thyroid	Infant	5.70 E-6	5.70 E-4	0.15	15
Kidney	Infant	8.39 E-7	8.39 E-5	0.15	15
Lung	Infant	8.42 E-7	8.42 E-5	0.15	15
G.I.T.*)	Adult	8.61 E-7	8.61 E-5	0.15	15
Skin	Adult	1.29 E-6	1.29 E-4	0.30	30

Emission height: H = 60 m, filtered release
Effective emission height: H_{eff} = 50 m (due to effect of building)
Reference point: 100 m
Predominant exposure pathway: Ingestion
Main nuclide: I 131, 1.3 E+6 Bq are released

*Gastrointestinal tract
**8.47 E-7 is equivalent to 8.47 x 10^{-7}

Table 5-17: Accident Doses in the Environment after a Steam Generator Tube Break with Response of the Pressure Relief System without Consideration of Filter System

(The accident is described in Section 5.6.1)

Organ	**Critical Population Group**	**Accident Doses**		**Emergency Reference Levels Complying with Art. 28.3 RPO**	
		Sv	**rem**	**Sv**	**rem**
Bone	Adult	3.19 E-3**)	3.19 E-1	0.30	30
Liver	Infant	2.23 E-3	2.23 E-1	0.15	15
Whole body	Adult	1.36 E-3	1.36 E-1	0.05	5
Thyroid	Infant	1.69 E-3	1.69 E-1	0.15	15
Kidney	Adult	1.17 E-3	1.17 E-1	0.15	15
Lung	Infant	1.08 E-3	1.08 E-1	0.15	15
G.I.T.*)	Adult	1.24 E-3	1.24 E-1	0.15	15
Skin	Adult	1.36 E-3	1.36 E-1	0.30	30

Emission height: H = 60 m, unfiltered release

Effective emission height: H_{eff} = 50 m (due to effect of building)

Predominant exposure pathway: Ground radiation

Main nuclide: Cs 137, 1.7 E+9 Bq are released

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*Gastrointestinal tract

**3.19 E-3 is equivalent to 3.19 x 10^{-3}

Table 5-18: Accident Doses in the Environment after a Steam Generator Tube Break with Response of the Pressure Relief System with Consideration of Filter System

(The accident is described in Section 5.6.1)

Organ	Critical Population Group	Accident Doses		Emergency Reference Levels Complying with Art. 28.3 RPO	
		Sv	rem	Sv	rem
Bone	Adult	8.13 E-7**	8.13 E-5	0.30	30
Liver	Infant	5.84 E-7	5.84 E-5	0.15	15
Whole body	Adult	3.57 E-7	3.57 E-5	0.05	5
Thyroid	Infant	1.26 E-5	1.26 E-3	0.15	15
Kidney	Adult	3.08 E-7	3.08 E-5	0.15	15
Lung	Infant	2.91 E-7	2.91 E-5	0.15	15
G.I.T.*	Adult	3.44 E-7	3.44 E-5	0.15	15
Skin	Adult	3.57 E-7	3.57 E-5	0.30	30

Emission height: H = 60 m, filtered release

Effective emission height: H_{eff} = 50 m (due to effect of building)

Reference point: 2001 m for thyroid; 100 m for other organs

Predominant exposure pathway: Ingestion

Main nuclide: I 131, 4.1 E+6 Bq are released

*Gastrointestinal tract

**8.13 E-7 is equivalent to 8.13 x 10^{-7}

Table 5-19: Accident Doses in the Environment after Break of a Pipe of the Helium Purification System, Unfiltered Release via Vent Stack

(The accident is described in Section 5.8.1)

Organ	Critical Population Group	Accident Doses		Emergency Reference Levels Complying with Art. 28.3 RPO	
		Sv	rem	Sv	rem
Bone	Adult	1.68 E-6**	1.68 E-4	0.30	30
Liver	Adult	6.69 E-6	6.69 E-4	0.15	15
Whole body	Adult	6.68 E-6	6.68 E-4	0.05	5
Thyroid	Infant	2.63 E-5	2.63 E-3	0.15	15
Kidney	Adult	6.68 E-6	6.68 E-4	0.15	15
Lung	Adult	6.69 E-6	6.69 E-4	0.15	15
G.I.T.*	Adult	6.96 E-6	6.96 E-4	0.15	15
Skin	Adult	4.93 E-5	4.93 E-3	0.30	30
Tissue	Adult	1.15 E-5	1.15 E-3	0.15	15

Emission height: H = 60 m, unfiltered release

Effective emission height: H_{eff} = 50 m (due to effect of building)

Reference point: 2001 m for thyroid; 100 m for other organs

Predominant exposure pathway: Ingestion

Main nuclide: I 131, 6.8 E+6 Bq are released

*Gastrointestinal tract

**1.68 E-6 corresponds to 1.68 x 10^{-6}

Table 5-20: Accident Doses in the Environment after Break of a Pipe of the Helium Purification System, Release via Building Leakages

(The accident is described in Section 5.8.1)

Organ	Critical Population Group	Accident Doses		Emergency Reference Levels complying with Art. 28.3 RPO	
		Sv	rem	Sv	rem
Bone	Adult	1.36 E-5**	1.36 E-3	0.30	30
Liver	Adult	6.52 E-5	6.52 E-3	0.15	15
Whole body	Adult	6.52 E-5	6.52 E-3	0.05	5
Thyroid	Infant	2.46 E-4	2.46 E-2	0.15	15
Kidney	Adult	6.52 E-5	6.52 E-3	0.15	15
Lung	Adult	6.52 E-5	6.52 E-3	0.15	15
G.I.T.*	Adult	6.81 E-5	6.81 E-3	0.15	15
Skin	Adult	5.05 E-4	5.05 E-2	0.30	30
Tissue	Adult	1.15 E-4	1.15 E-2	0.15	15

Emission height: H = 0 m, unfiltered release
Effective emission height: H_{eff} = 17.5 m (due to effect of building)
Reference point: 100 m
Predominant exposure pathway: Ingestion
Main nuclide: I 131, 6.8 E+6 are released
I 133, 4.4 E+7 are released

*Gastrointestinal tract
**1.36 E-5 corresponds to 1.36 x 10^{-5}

Table 5-21: Accident Doses in the Environment after Failure of a Vessel Containing Radioactively Contaminated Water

(The accident is described in Section 5.8.2)

Organ	Critical Population Group	Accident Doses		Emergency Reference Levels Complying with Art. 28.3 RPO	
		Sv	rem	Sv	rem
Bone	Infant	6.33 E-5**	6.33 E-3	0.30	30
Liver	Infant	6-31 E-5	6.31 E-3	0.15	15
Whole body	Adult	4.61 E-5	4.61 E-3	0.05	5
Thyroid	Adult	4.61 E-5	4.61 E-3	0.15	15
Kidney	Adult	4.56 E-5	4.56 E-3	0.15	15
Lung	Infant	4.43 E-5	4.43 E-3	0.15	15
G.I.T.*	Adult	4.65 E-5	4.65 E-3	0.15	15
Skin	Adult	4.08 E-5	4.08 E-3	0.30	30

Emission height: H = 60 m, unfiltered release
Effective emission height: H_{eff} = 50 m (due to effect of building)
Reference point: 100 m
Predominant exposure pathway: Ground radiation
Main nuclide: Co 60, 4.4 E+7 Bq are released

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*Gastrointestinal tract
**6.33 E-5 corresponds to 6.33 x 10^{-5}

Table 5-22: Accident Doses in the Environment as a Result of Seismic Effects on the Reactor Auxiliary Building

(The accident is described in Section 5.9)

Organ	Critical Population Group	Accident Doses		Emergency Reference Levels Complying with Art. 28.3 RPO	
		Sv	rem	Sv	rem
Bone	Infant	1.75 E-4**	1.75 E-2**	0.30	30
Liver	Adult	2.48 E-4	2.48 E-2	0.15	15
Whole body	Adult	2.44 E-4	2.44 E-2	0.05	5
Thyroid	Infant	5.52 E-4	5.52 E-2	0.15	15
Kidney	Adult	2.43 E-4	2.43 E-2	0.15	15
Lung	Adult	2.45 E-4	2.45 E-2	0.15	15
G.I.T.*	Adult	2.51 E-4	2.51 E-2	0.15	15
Skin	Adult	1.68 E-3	1.68 E-1	0.30	30
Tissue	Adult	3.77 E-4	3.77 E-2	0.30	30

Emission height: H = 0 m, unfiltered release

Effective emission height: H_{eff} = 17.5 m (due to effect of building)

Reference point: 100m

Predominant exposure pathway: Inhalation

Main nuclide: H3, 1.86 E+13 Bq are released

*Gastrointestinal tract

**1.75 E-4 corresponds to 1.75 x 10^{-4}

Table 5-23: Radioactive Materials in the Primary System of one Reactor (design data)

Nuclide	Half-life	Release rate in Bq / h.MW_t	Steady-state coolant activity in Bq [7)]	Surface activity after 32 EFPY in Bq
Kr83m	1.83 h	4.0E+7 [1)]	1.9E+10	
Kr85m	4.48 h	5.9E+7	5.8E+10	
Kr85	10.76 a	8.1E+4	3.2E+8	
Kr87	76.3 m	2.1E+8	7.2E+10	
Kr88	2.80 h	2.0E+8	1.4E+11	
Kr89	3.18 m	1.8E+9	2.8E+10	
Kr90	32.3 s	4.6E+9	1.2E+10	
Xe131m	12.0 d	3-0E+5	1.2E+9	
Xe133m	2.2 d	3.7E+6	1.1E+10	
Xe133	5.29 d	6.4E+7	2.3E+11	
Xe135m	15.3 m	2.4E+8	1.7E+10	
Xe135	9.17 h	9.0E+7	1.4E+11	
Xe137	3.83 m	2.6E+9	4.8E+10	
Xe138	14. 1 m	1.4E+9	9.1E+10	
Xe139	39.7 s	4.7E+9	1.5E+10	
Total noble gases		1.6E+10	8.8E+11	
I131	8.04 d	3.8E+6	3.4E+7	2.1E+11
I132	2.38 h	4.9E+7	4.5E+8	3.3E+10
I133	20,8 h	2.4E+7	2.2E+8	1.5E+11
I134	52.0 m	1.3E+8	1.1E+9	3.2E+10
I135	6.59 h	4.4E+7	3.9E+8	8.5E+10
Total iodine		2.5E+8	2.2E+9	5.1E+11
Rb 88	17.8 m [2)]		1.3E+10	1.2E+11
Cs138	32.2 m[3)]		4.8E+9	8.5E+10
Total short-lived solids			1.8E+10	2.1E+11
Sr90	28.5 a	2.4E+2	2.2E+3	9-3E+9
Cs134	2.06 a	2.8E+4 [4)]	2.6E+5	1.5E+11
Cs137	30.1 a	5.9E+4	5.4E+5	2.4E+12
Ag110m	250.4 d	2.1E+3 [5)]	1.9E+4	3.7E+9
Total long-lived solids		8.9E+4	8.2E+5	2.6E+12
H3	12.346 a	2.8E+7	1.1E+11	

Nuclide	Half-life	Release rate in Bq / h.MW_t	Steady-state coolant activity in Bq [7)]	Surface activity after 32 EFPY in Bq
C14	5736 a	3.0E+5 [6)]	1.2E+9	

1) 4.0E+7 is equivalent to 4.0 x 10^{+7}
2) Decay product of Kr88
3) Decay product of Xe138
4) Activation of Cs133
5) Activation of Ag109
6) Primarily activation of N14
7) Relative to 15000 m^3 of helium (20 °C, 1 bar)

Table 5-24: Radioactive Materials in one Helium Purification Train (Design data)

Nuclide	Saturation activity in Bq of: Dust removal filter	Molecular sieve	Cryogenic absorber	Total activity in Bq of one purification train
Kr83m		8.3 E+7 1)	2.4 E+9	2.5 E+9
Kr85m		2.6 E+8	1.8 E+10	1.9 E+10
Kr85		1.4 E+6	1.6 E+10	1.6 E+10
Kr87		3.2 E+8	6.3 E+9	6.6 E+9
Kr88		6.0 E+8	2.7 E+10	2.7 E+10
Kr89		7.4 E+7	3.3 E+7	1.1 E+8
Kr90		7.6 E+6	7.3 E+3	7.6 E+6
Xe131m		2.6 E+7	2.2 E+10	2.2 E+10
Xe133m		2.6 E+8	4.3 E+10	4.4 E+10
Xe133		5.2 E+9	2.1 E+12	2.1 E+12
Xe135m		2.3 E+8	9.4 E+7	3.2 E+8
Xe135		3.2 E+9	9.1 E+10	9.5 E+10
Xe137		2.2 E+8	1.7 E+6	2.2 E+8
Xe138		1.1 E+9	4.1 E+8	1.5 E+9
Xe139		1.2 E+7	-	1.2 E+7
Total noble gases		1.2 E+10	2.3 E+12	2.3 E+12
I131	4.7 E+8			4.7 E+12
I132	7.7 E+7			7.7 E+7
I133	3.3 E+8			3.3 E+8
I134	7.1 E+7			7.1 E+7
I135	1.8 E+8			1.8 E+8
Total iodine	1.1 E+9			1.1 E+9
Rb88	2.7 E+8	6.0 E+8	2.7 E+10	2.7 E+10
Cs138	1.9 E+8	1.1 E+9	4.1 E+8	1.5 E+9
Total noble gas decay products	4.7 E+8	1.7 E+9	2.7 E+10	2.9 E+10
Sr90	2.1 E+7 2)			2.1 E+7 2)
Cs134	3.3 E+8			3.3 E+8
Cs137	5.4 E+9 2)			5.4 E+9 2)
Ag110 m	8.2 E+6			8.2 E+6
Total long-lived solids	5.8 E+9 2)			5.8 E+9 2)

Nuclide	Saturation activity in Bq of: Dust removal filter	Molecular sieve	Cryogenic absorber	Total activity in Bq of one purification train
H3		4.6 E+12	9.3 E+11	5.6 E+12
C14		5.9 E+10	3)	5.9 E+10

1) 1.7 E+8 is equivalent to 1.7 x 10^{+8}
2) After 32 equivalent full-power years
3) Not calculated, 100 % of the C14 activity in the molecular sieve

Table 5-25: Accumulated Activity in the Storage tanks of Radioactively Contaminated Helium after one Regeneration cycle (Integrated Decay Period 12-h)

Nuclide	Accumulated activity in Bq after one regeneration cycle
Kr83m	2.7 E + 7 [1)]
Kr85m	2.9 E + 9
Kr85	1.6 E + 10
Kr87	9.5 E + 6
Kr88	1.4 E + 9
Xe131m	2.1 E + 10
Xe133m	3.7 E + 10
Xe133	2.0 E + 12
Xe135	3.9 E + 10
Total noble gases	2.1 E + 12
H3	1.1 E +11
C14	5.9 E +10

1) 2.7 E+7 is equivalent to 2.7 x 10^{+7}

Figure 5-1: Power vs. Time Curve on Withdrawal of All Reflector Rods

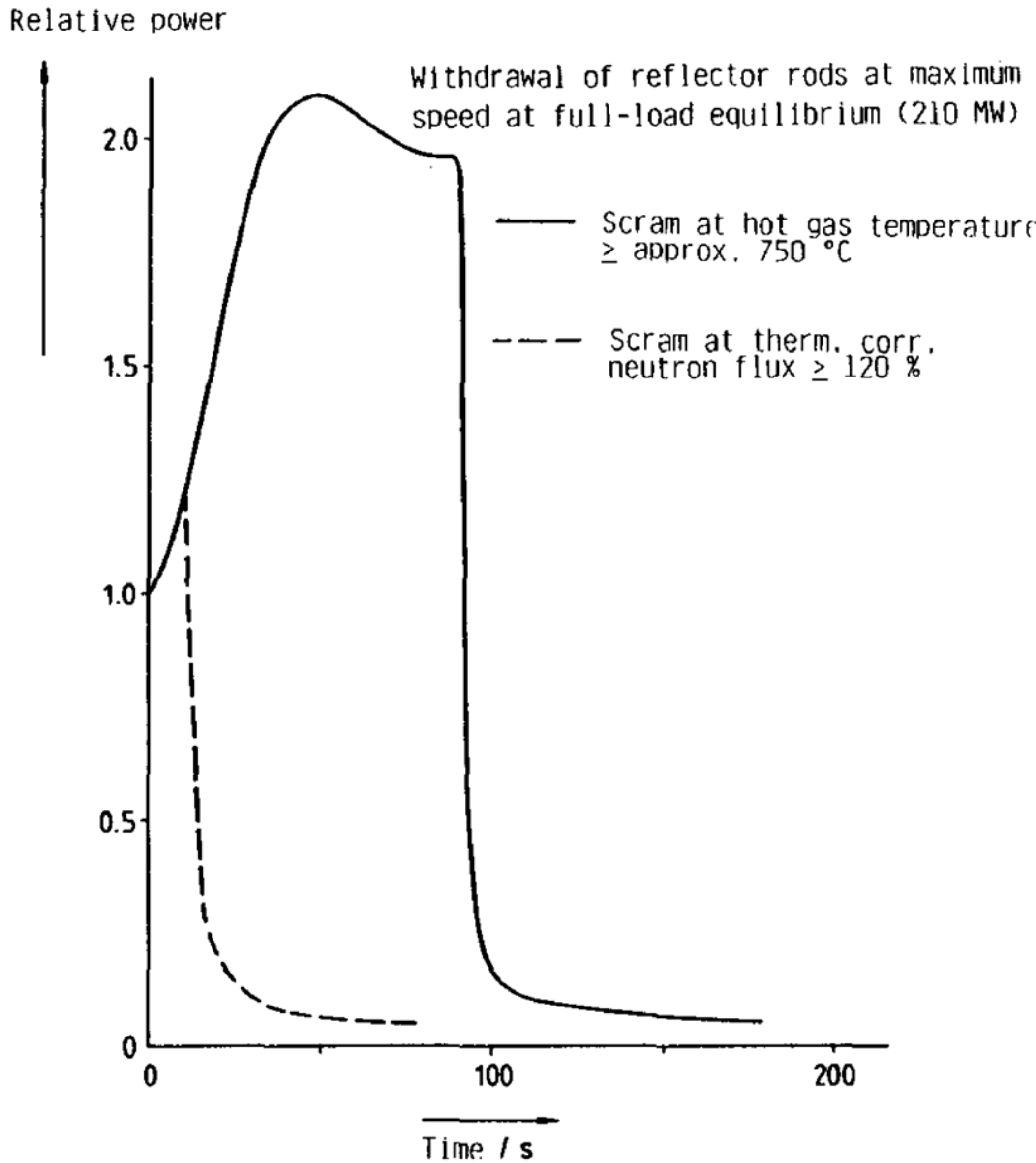

Figure 5-2: Maximum Fuel Temperature and Average Coolant Temperature vs. Time Curves on Withdrawal of All Reflector Rods

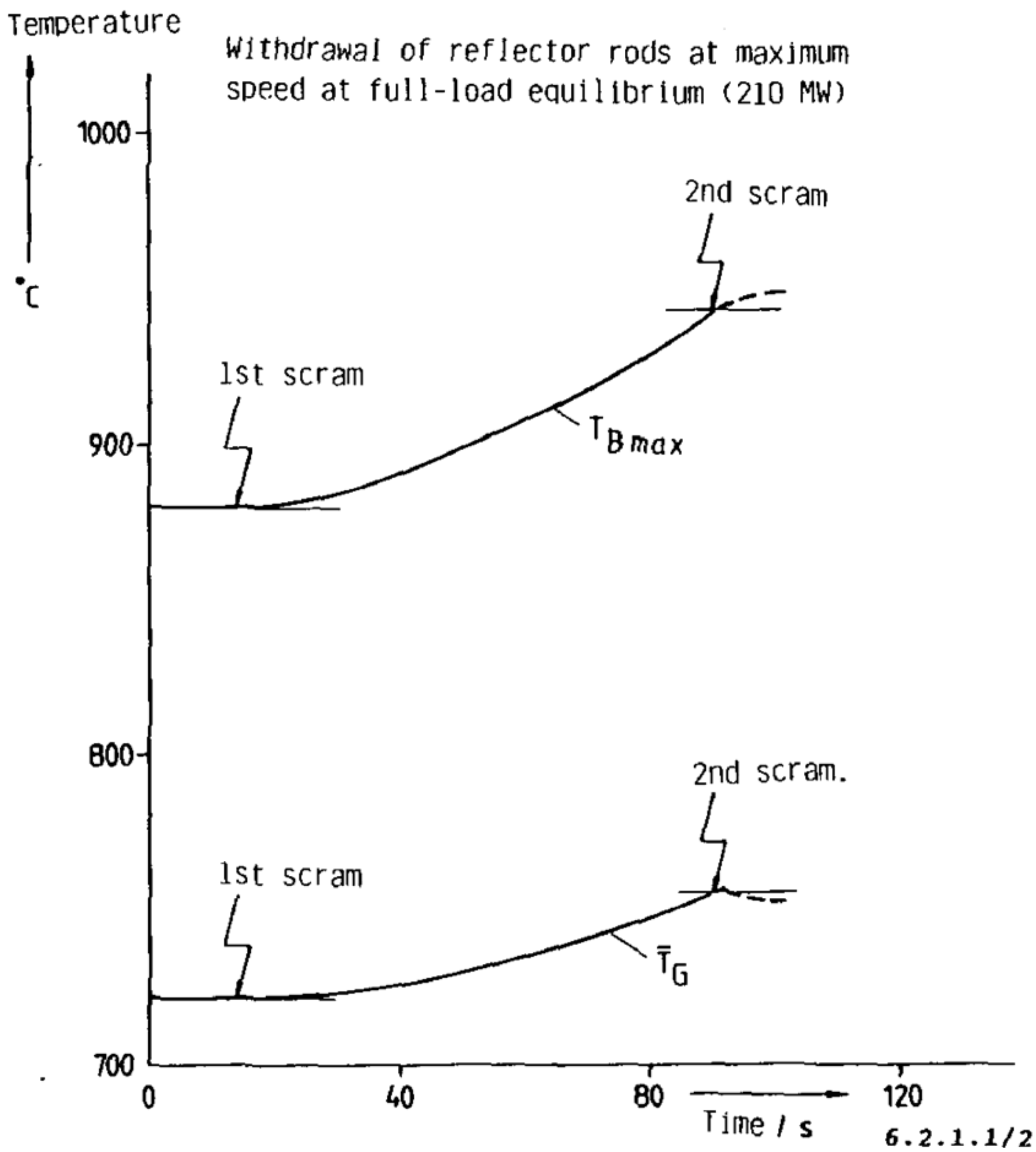

Figure 5-3: Power vs. Time Curve on Withdrawal of All Reflector Rods

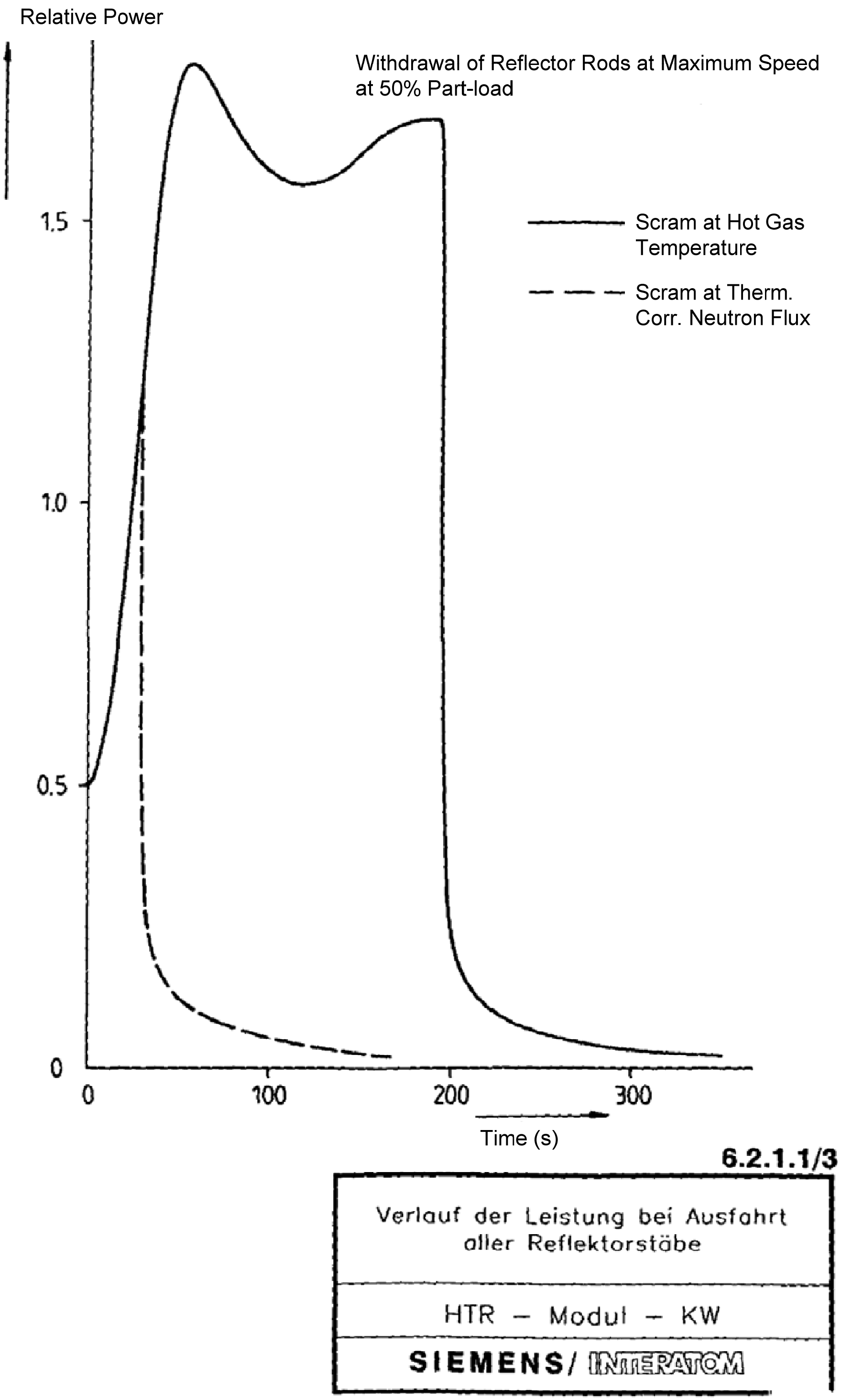

Figure 5-4: Temperature vs. Time Curve of Maximum FE-Temperature and Average Primary Coolant Temperature at Withdrawal of All Reflector Rods

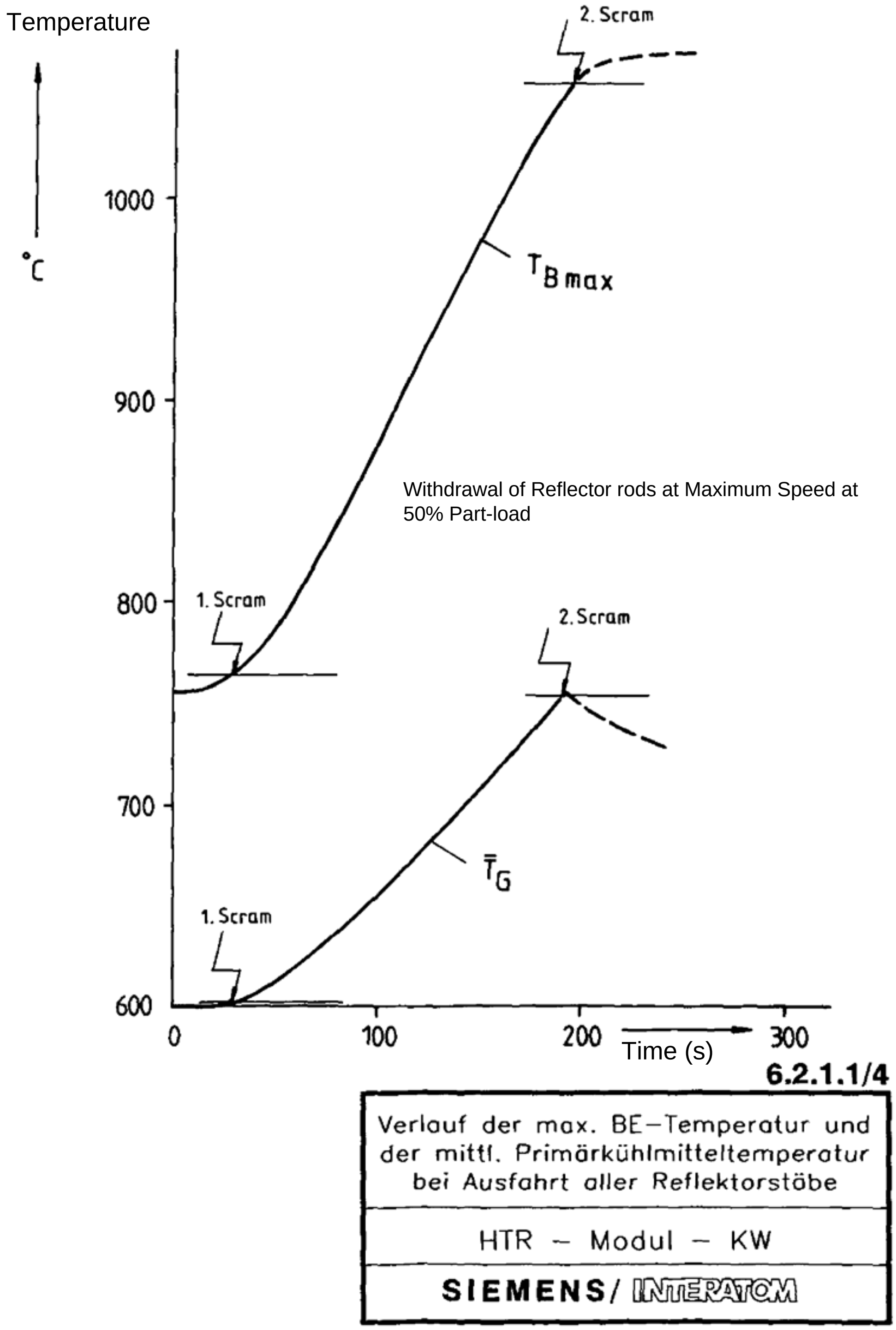

Figure 5-5: Power vs. Time Curve on Withdrawal of All Reflector Rods

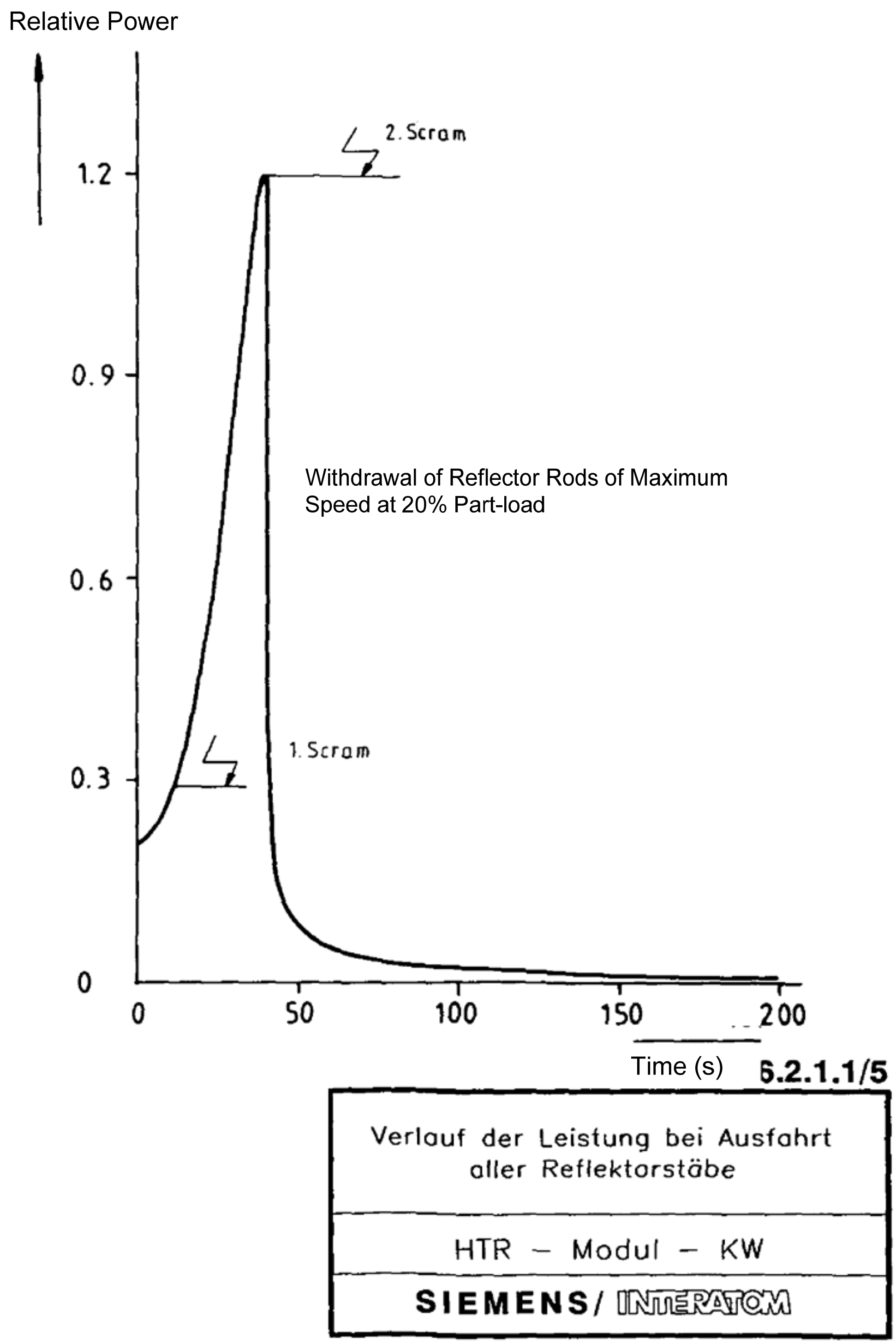

Figure 5-6: Temperature vs. Time Curve of Maximum FE-Temperature at Withdrawal of All Reflector Rods

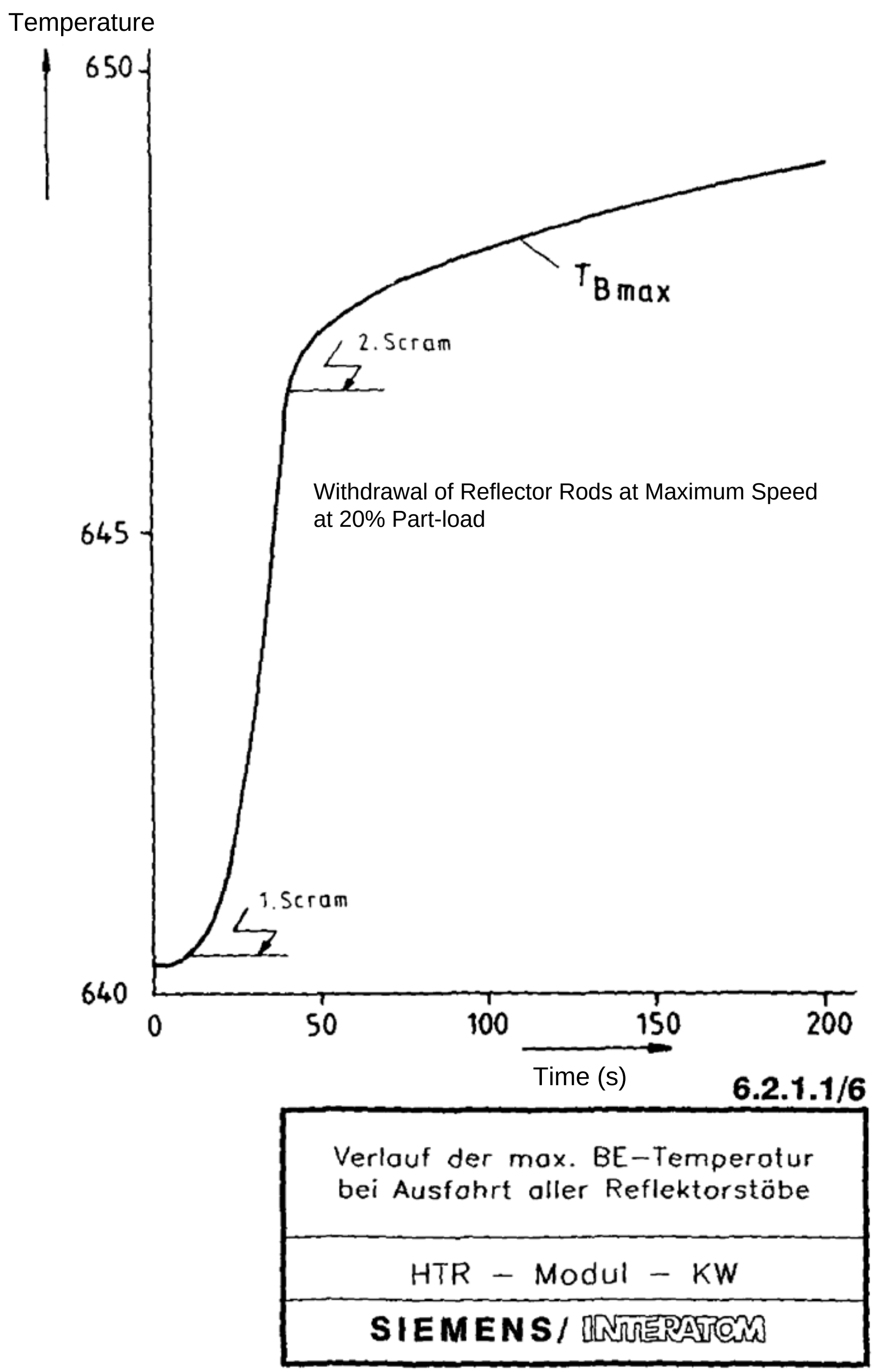

Figure 5-7: Power vs. Time Curve on Drop of One Small Ball Shut-down Unit

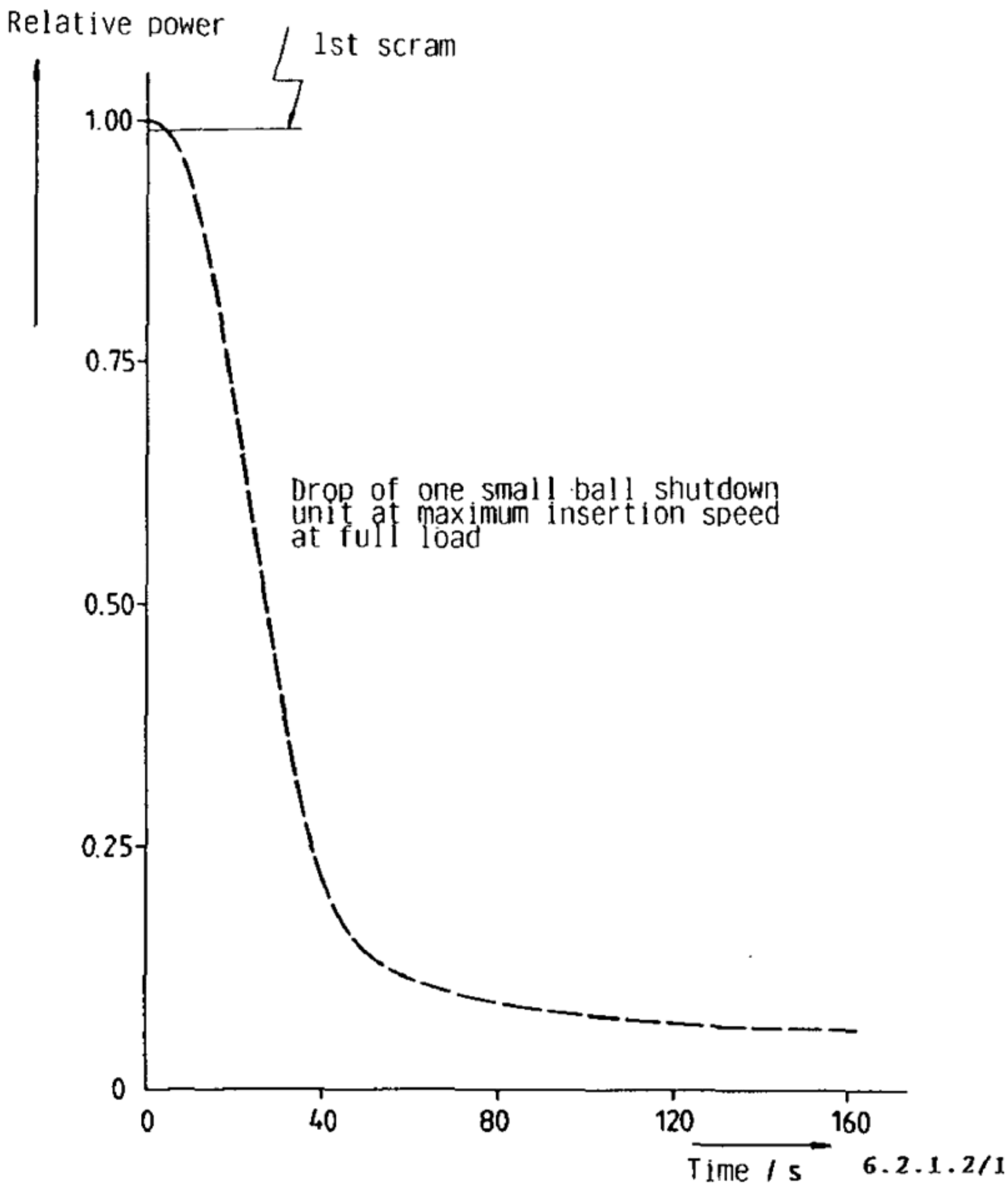

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Figure 5-8: Power vs. Time Curve on Primary Gas Circulator Overspeed

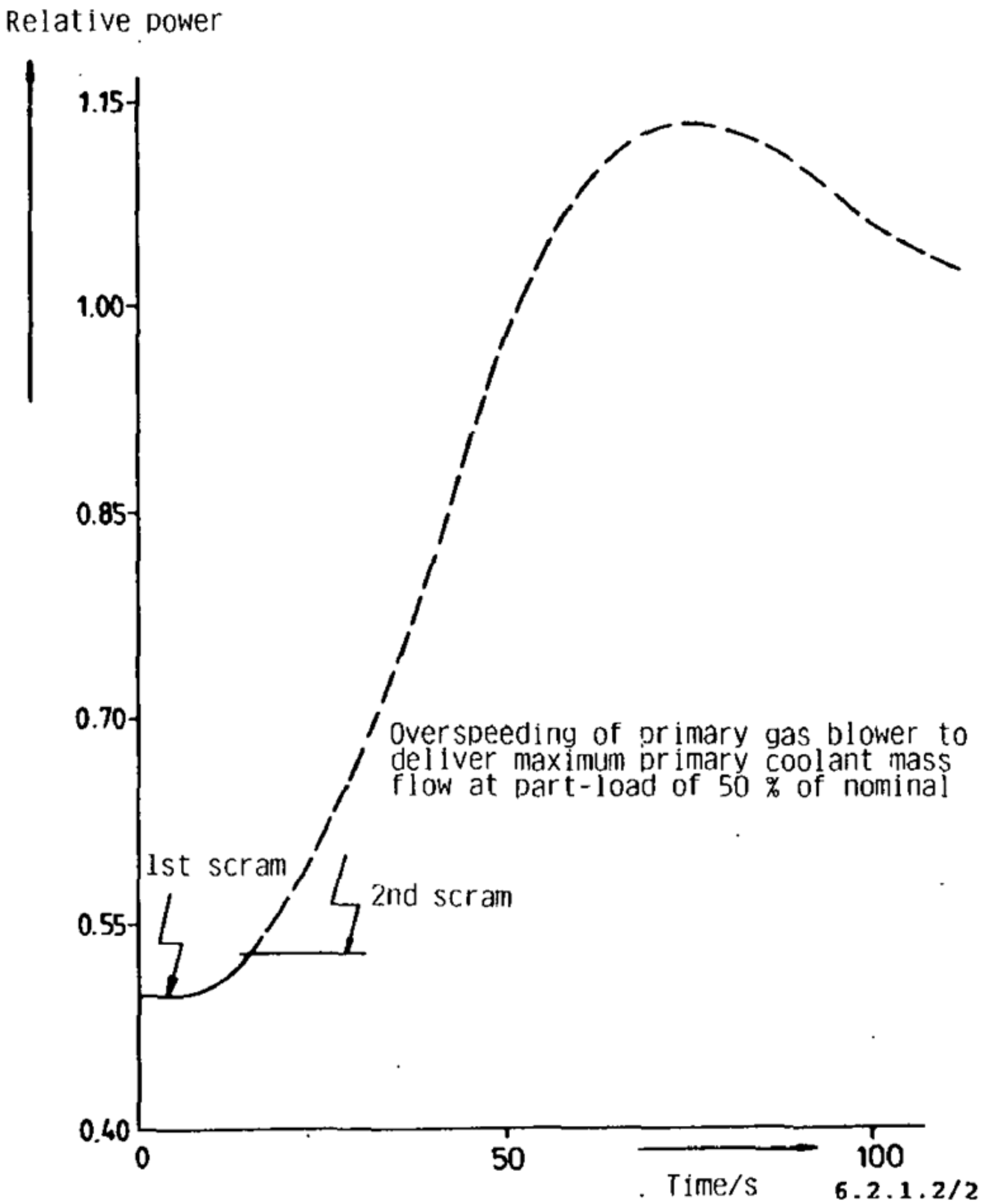


Figure 5-9: Average Hot Gas Temperature vs. Time Curve on Primary Gas Circulator Overspeed

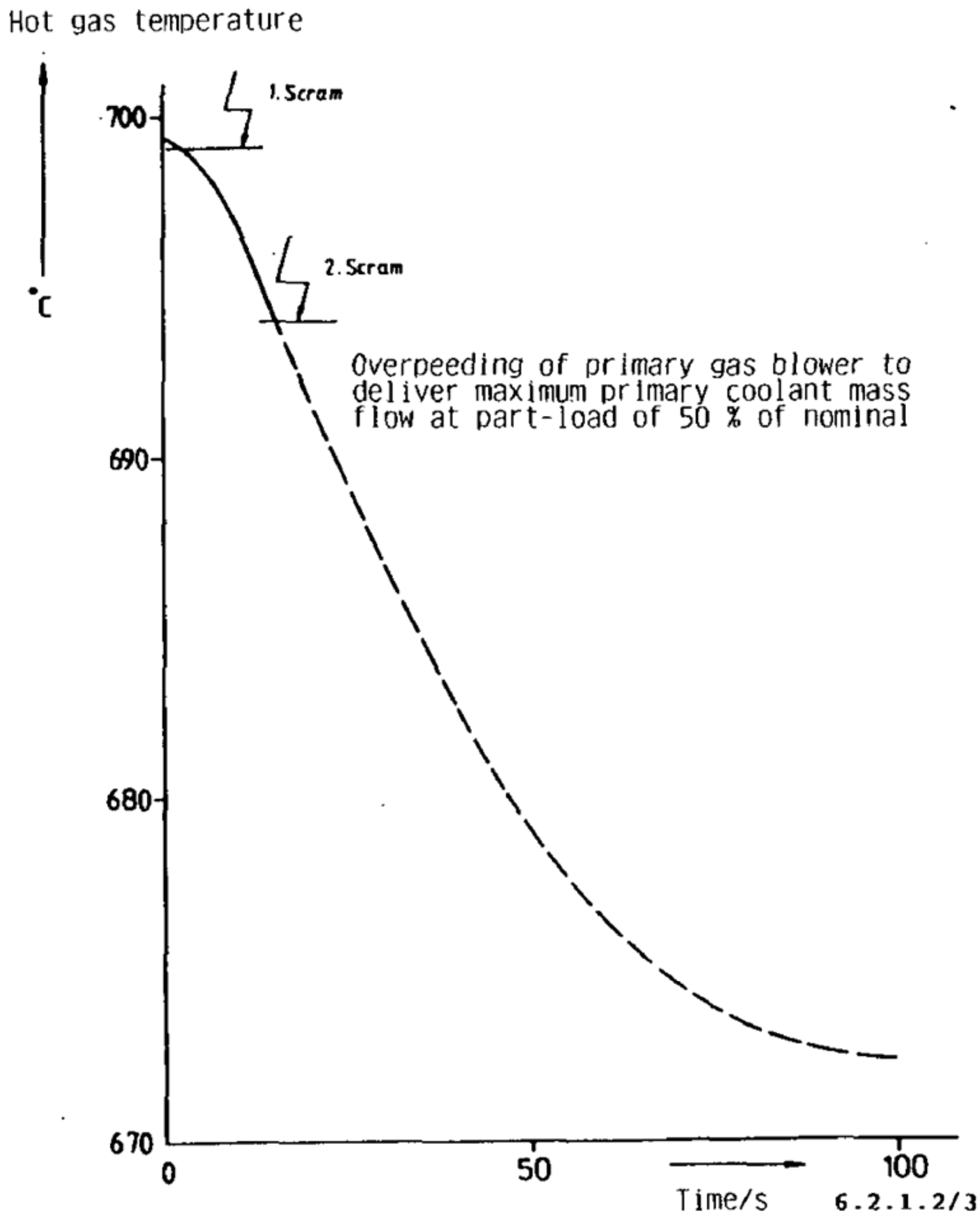


Figure 5-10: Reactivity Change Due to Earthquake as a Function of Fill Factor

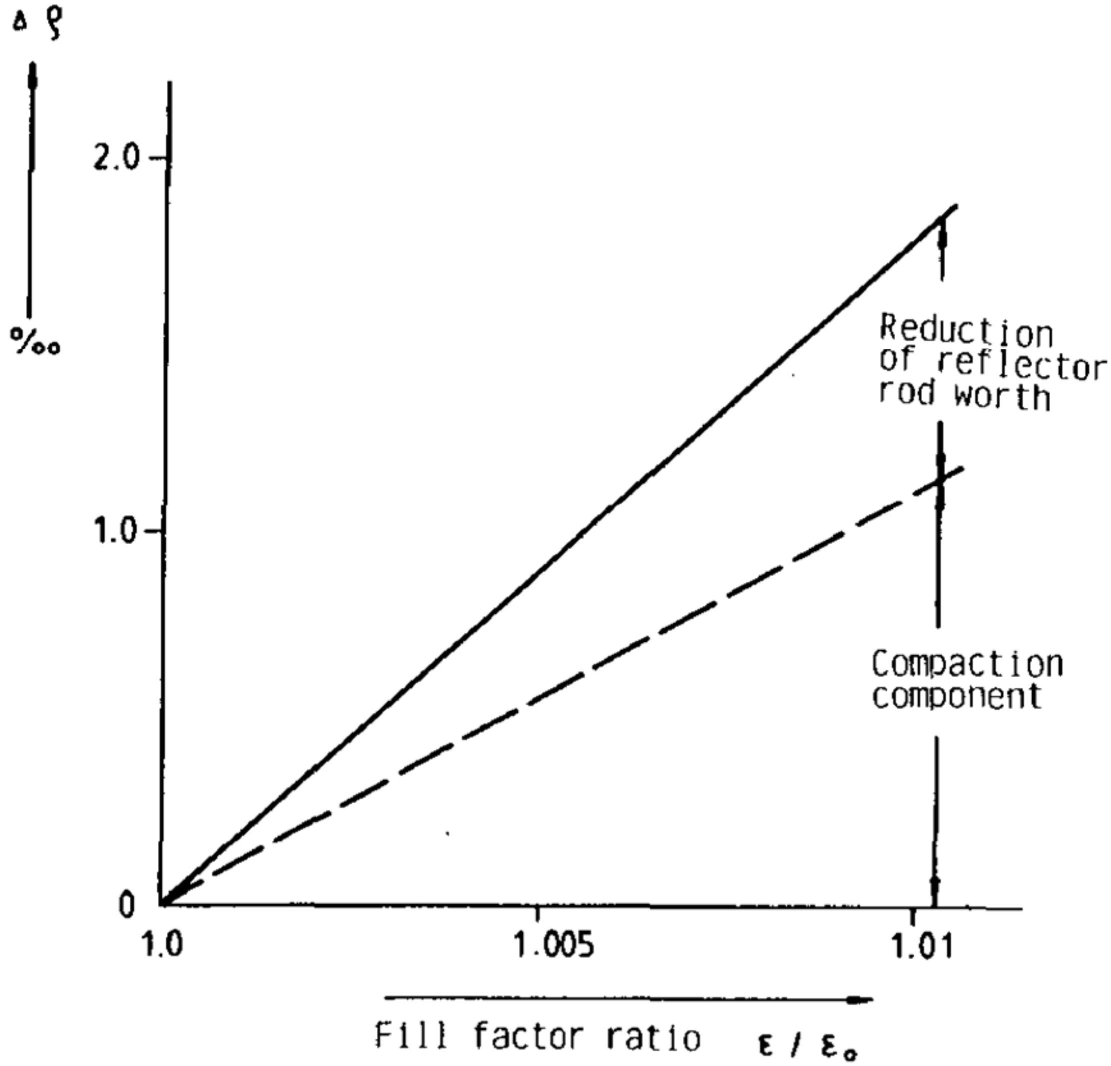


Figure 5-11: Power vs. time Curve for Core Compaction Due to Earthquake (Horizontal Acceleration: 0.5g)

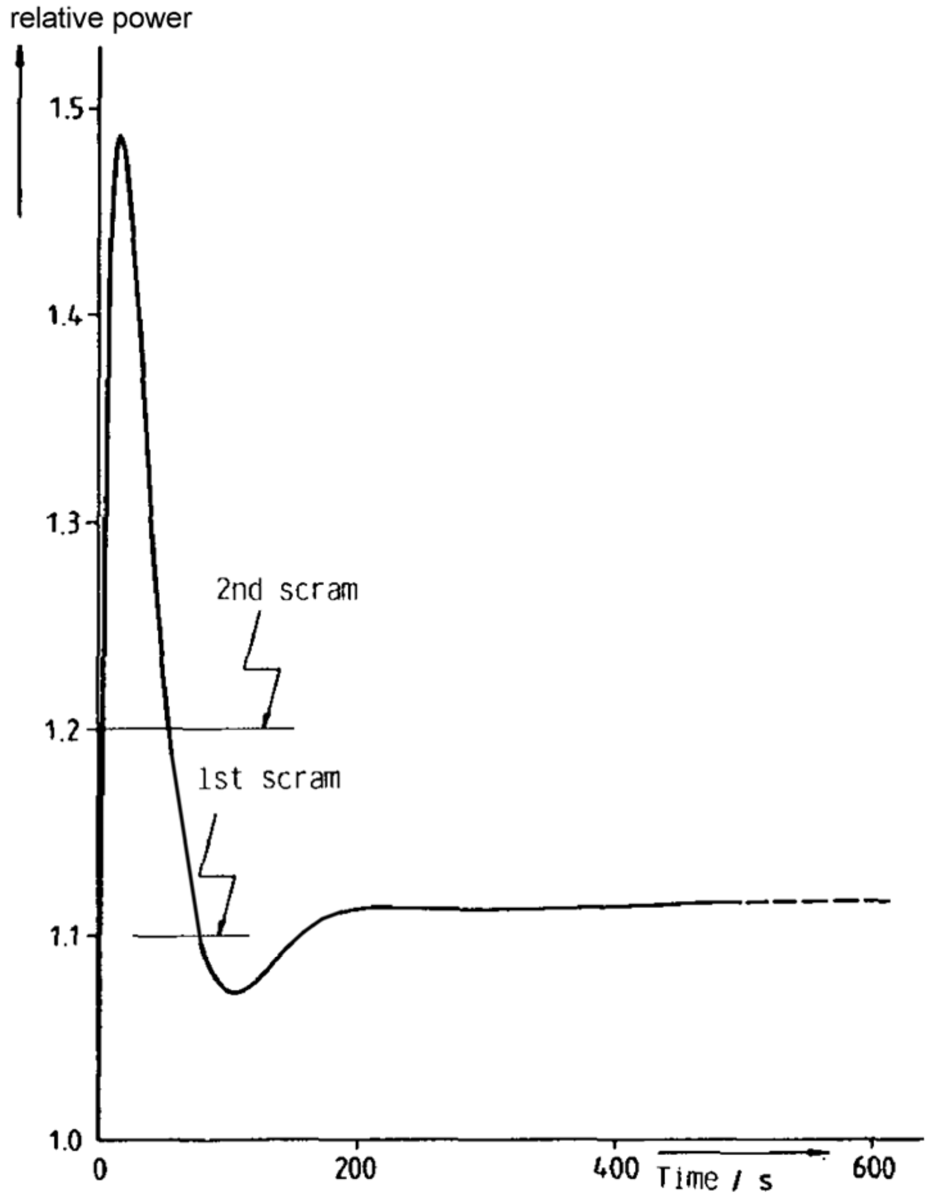


Figure 5-12: Pressure Response in Primary System on 2A Break of Outlet line to Helium Purification System

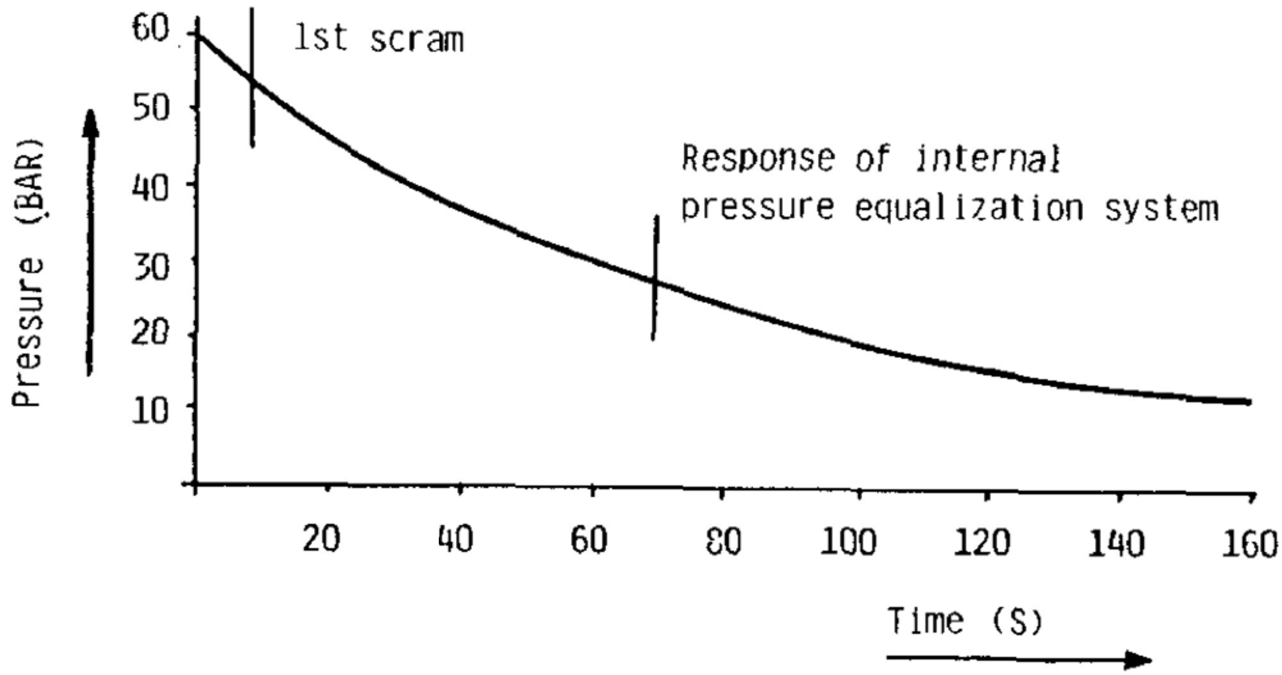


Figure 5-13: Accident Sequence in Reactor Building on 2A Break of Outlet Line to Helium Purification System

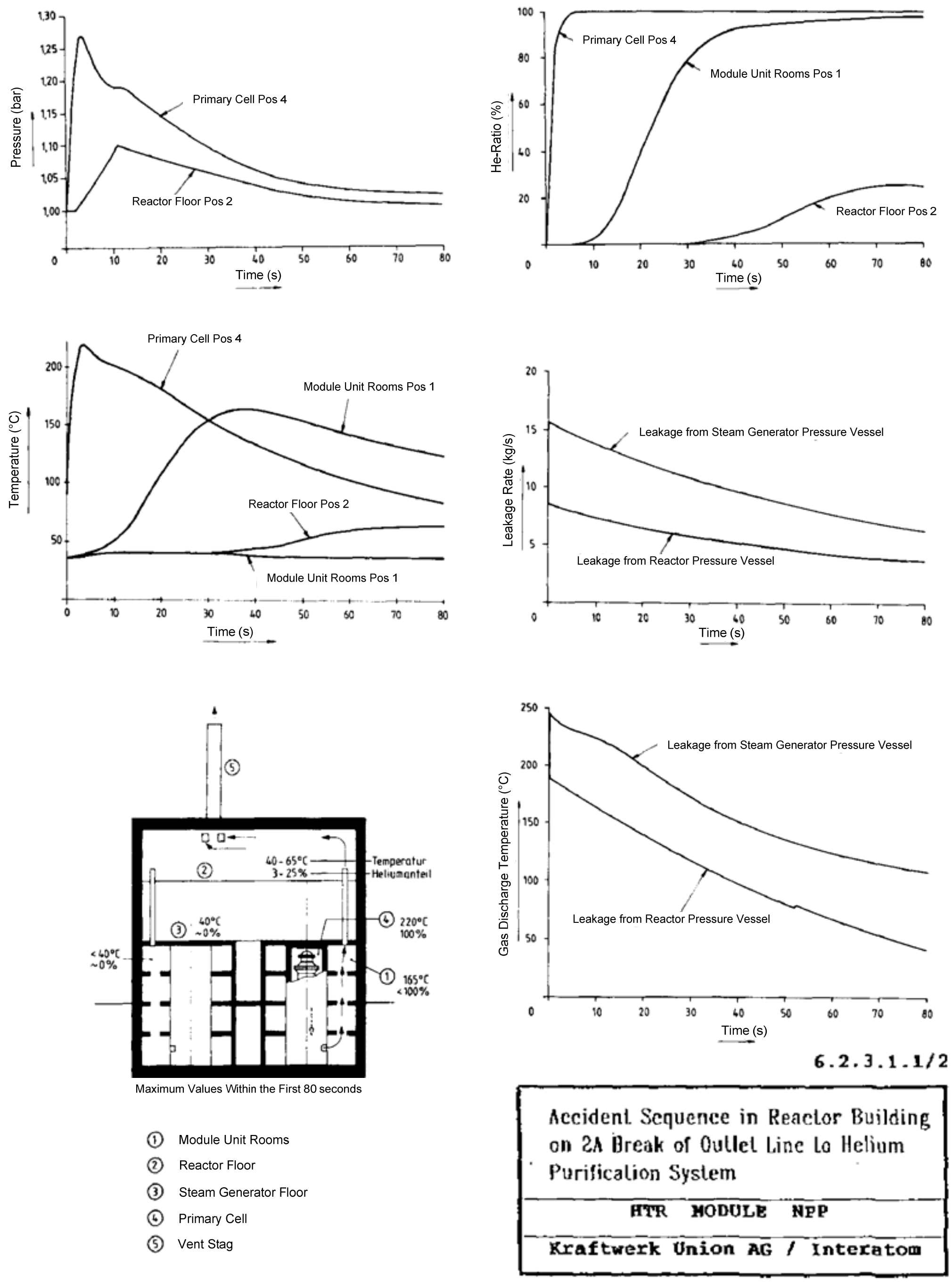


Figure 5-14: Radial Temperature Curves at Elevation of Peak Core Temperature

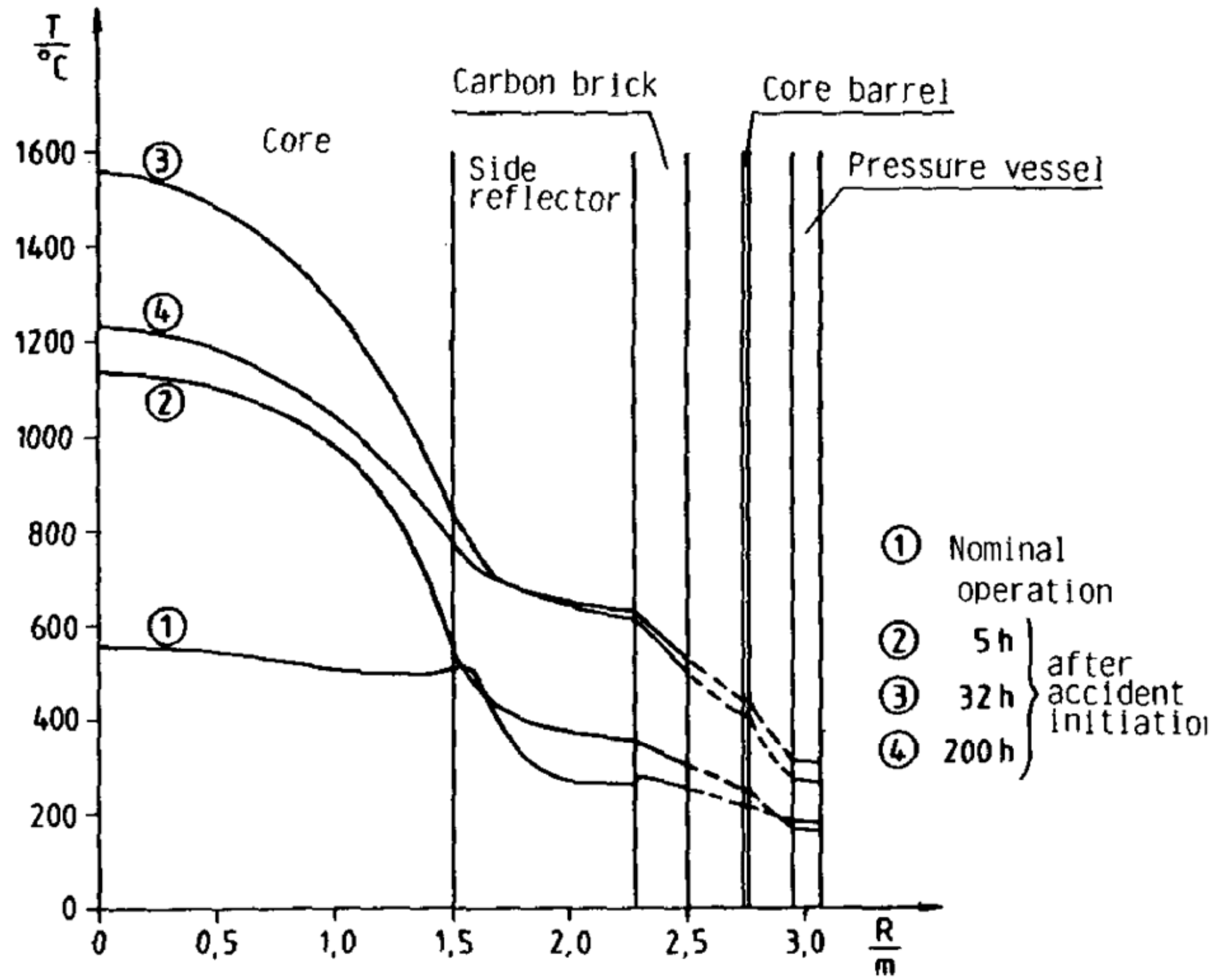


Figure 5-15: Axial Temperature Patterns Along Core Axis

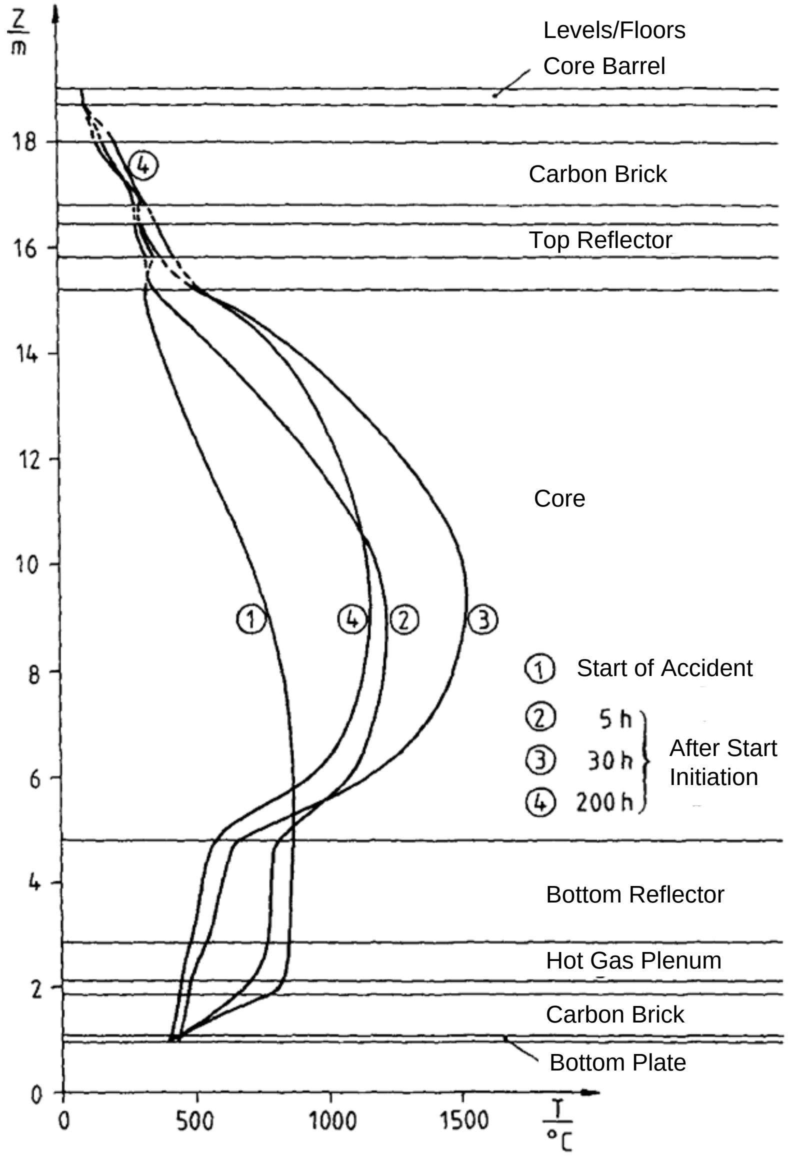


Figure 5-16: Peak Temperature vs. Time Curves

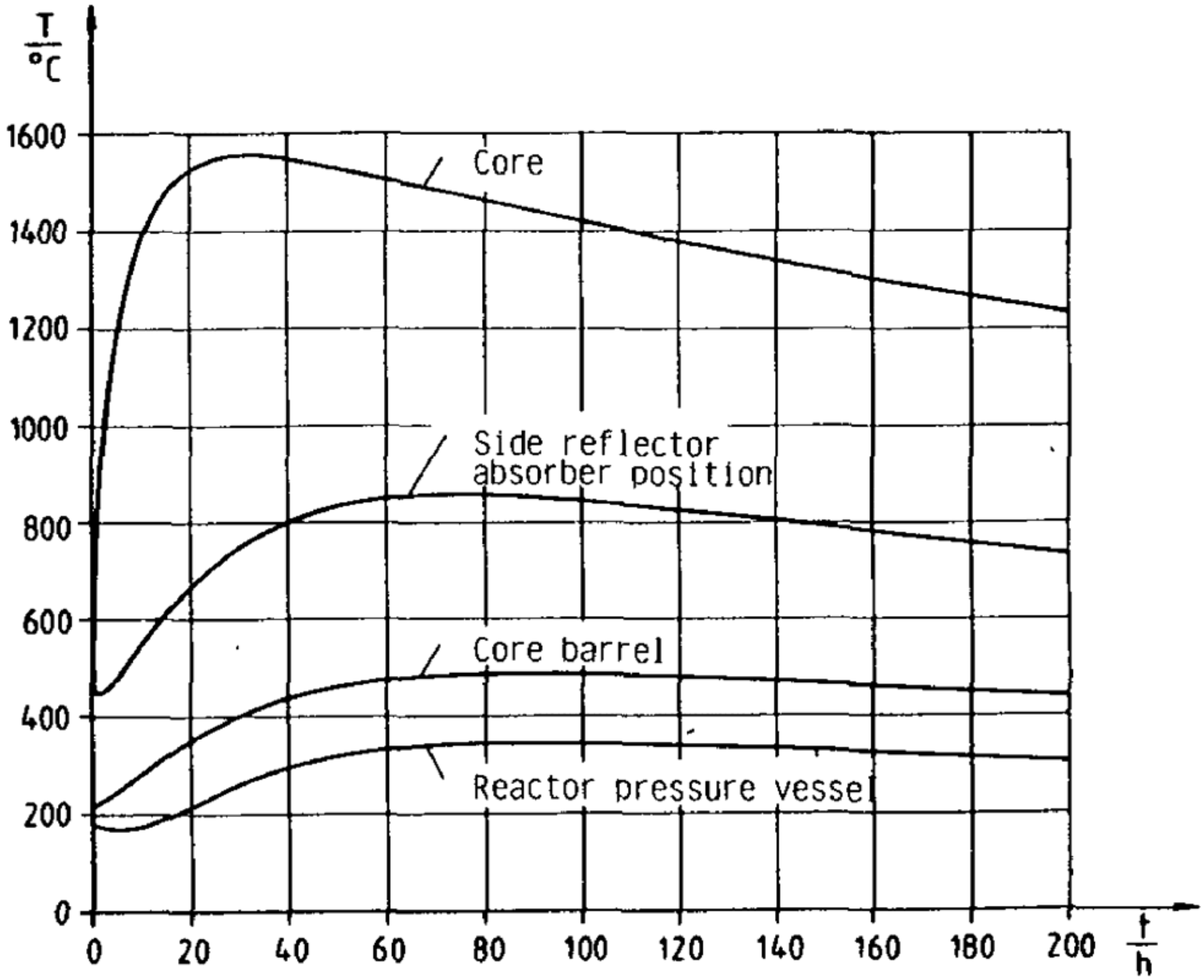


Figure 5-17: Plot of Volumetric Fractions of Core at Certain Temperatures

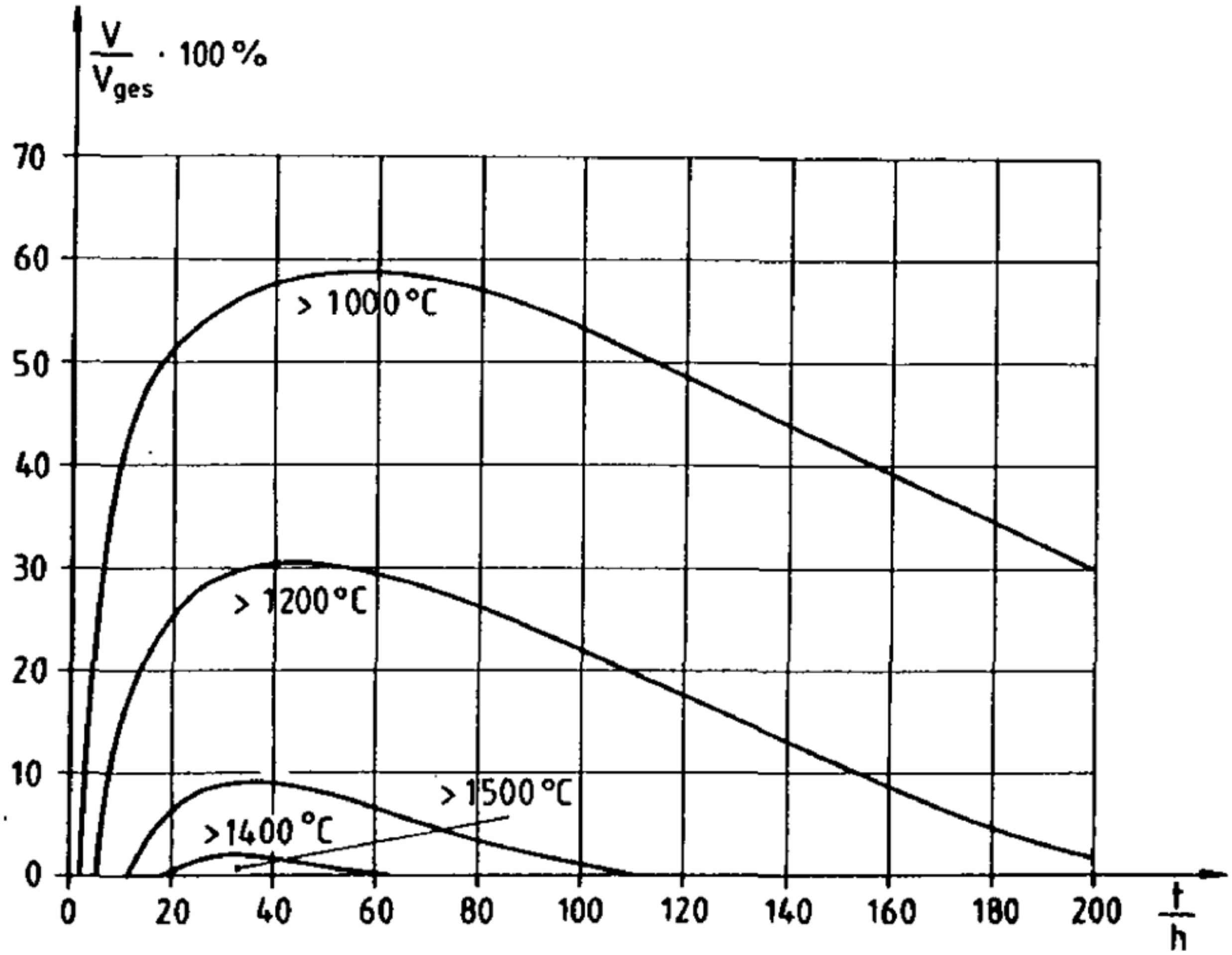


Figure 5-18: Cold Gas Temperature vs. Time Curve at SG Outlet on 2A Break of Feedwater Line

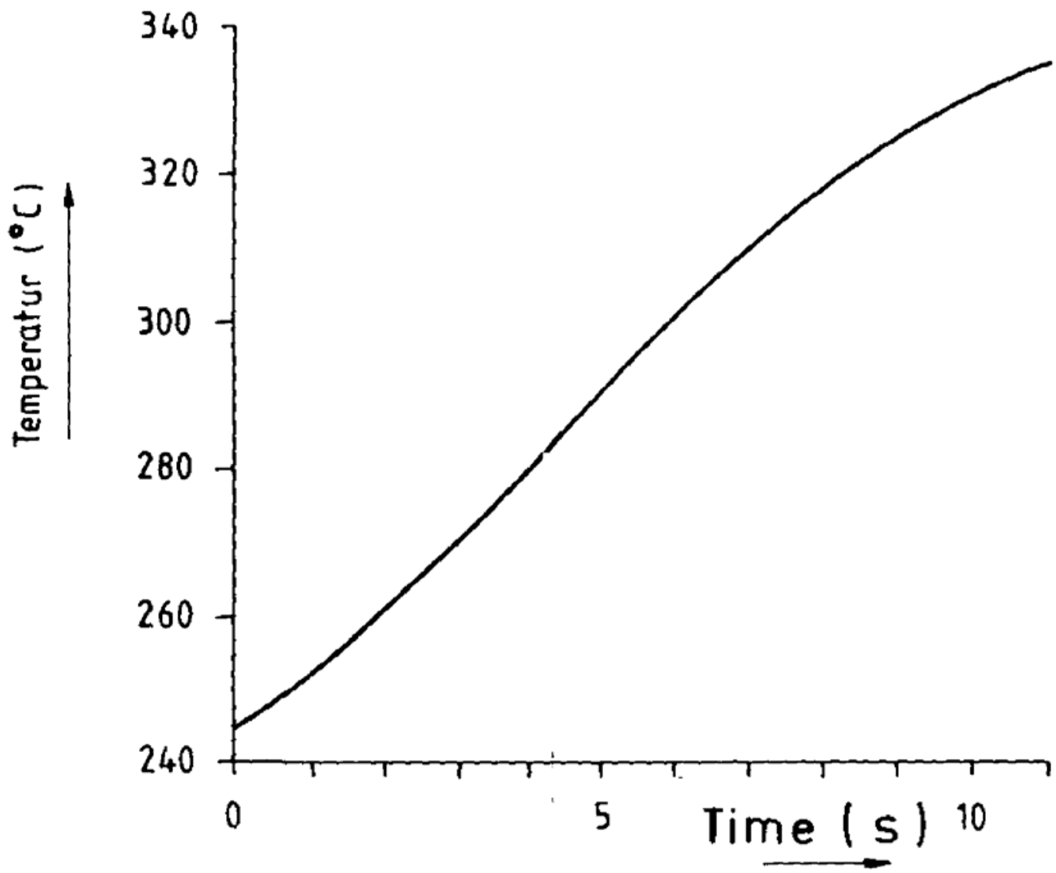


Figure 5-19: Temperature Time History Curves in the Primary Cell After Failure of Surface Cooler

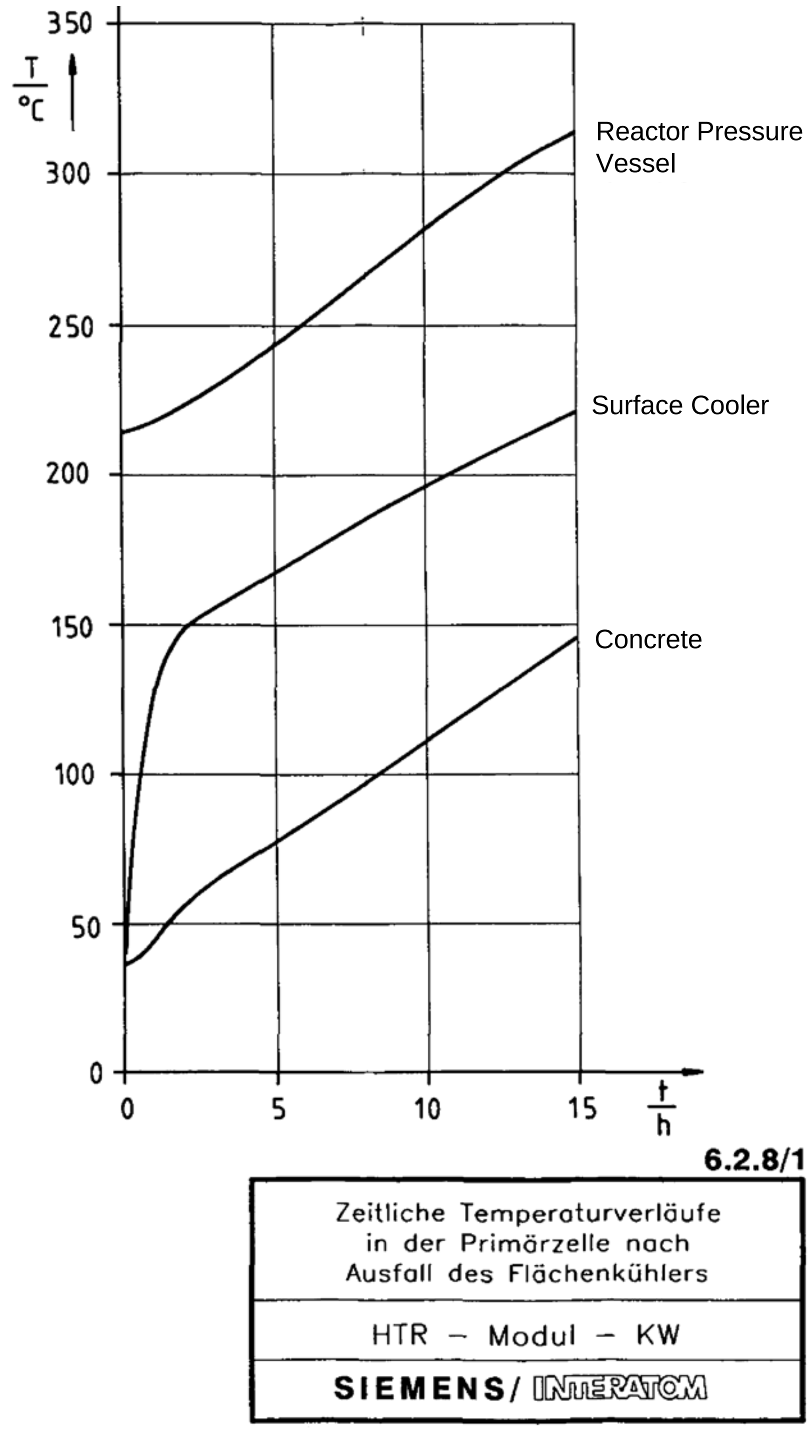


Figure 5-20: Non-Isolable 2A Break of Main Steam Line with Counteractions After 3s

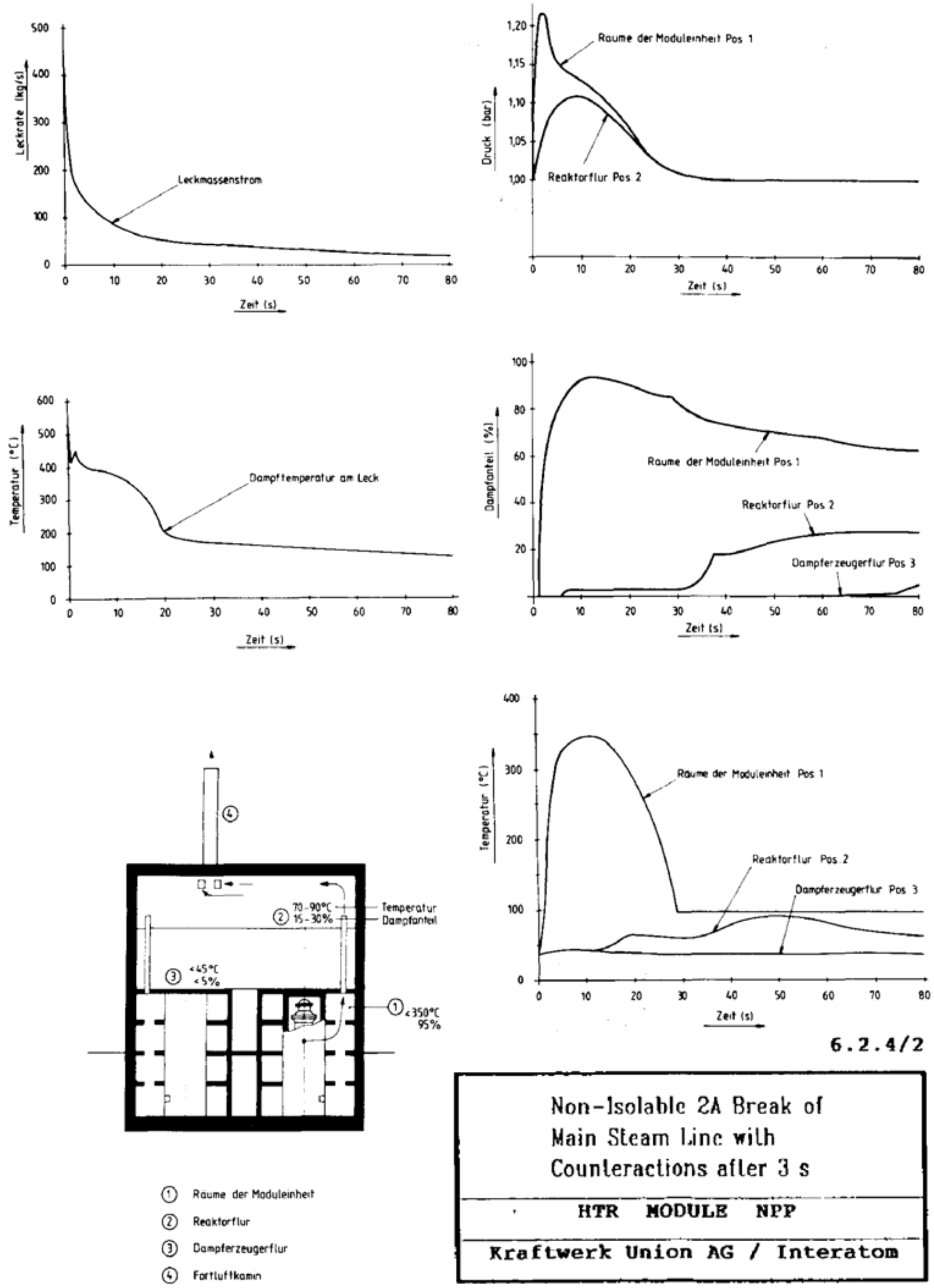


Figure 5-21: Integral Graphite Corrosion

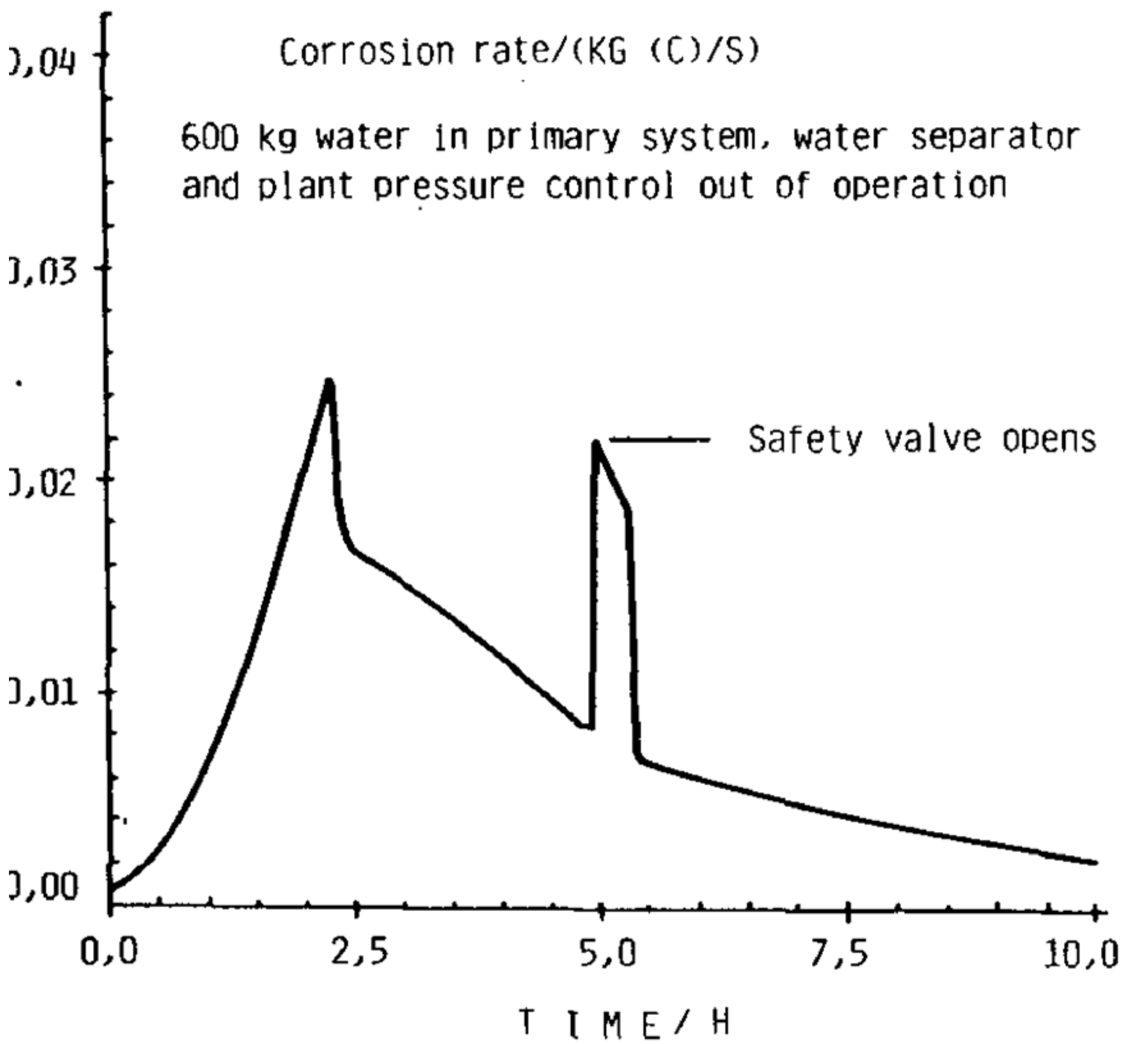


Figure 5-22: Quantities of Water and Graphite Converted

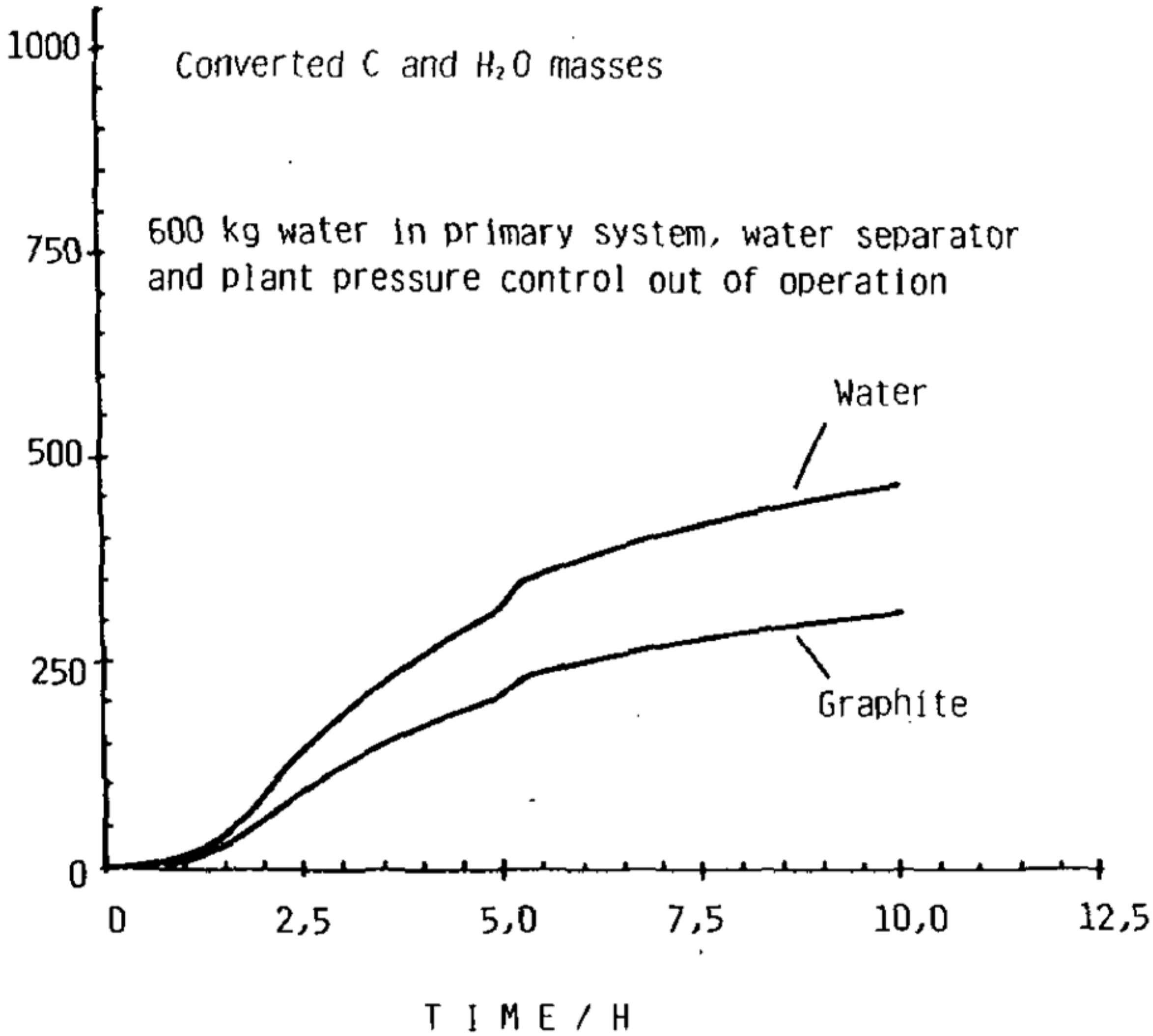


Figure 5-23: Relative Fraction of Corroded Fuel Elements

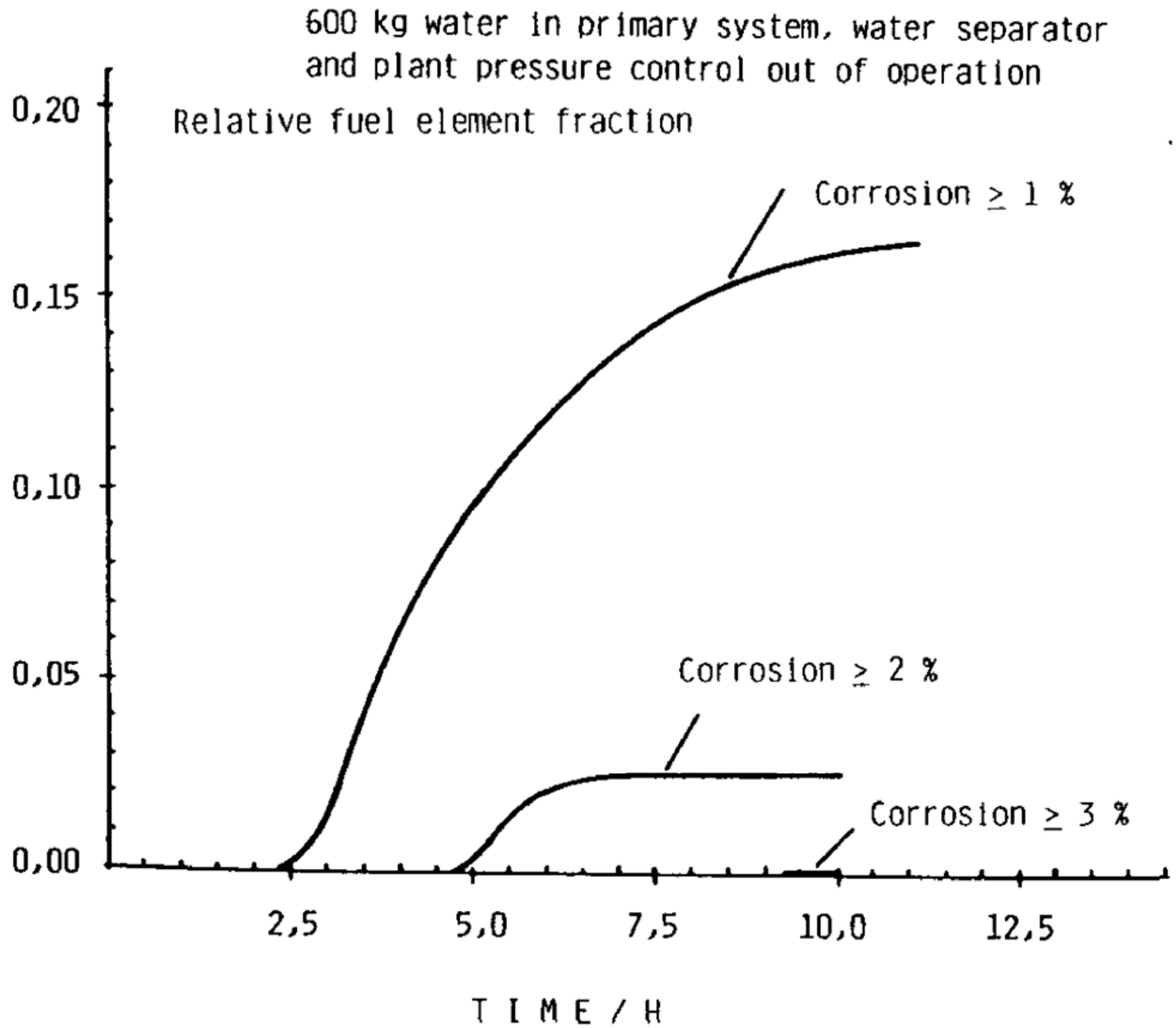
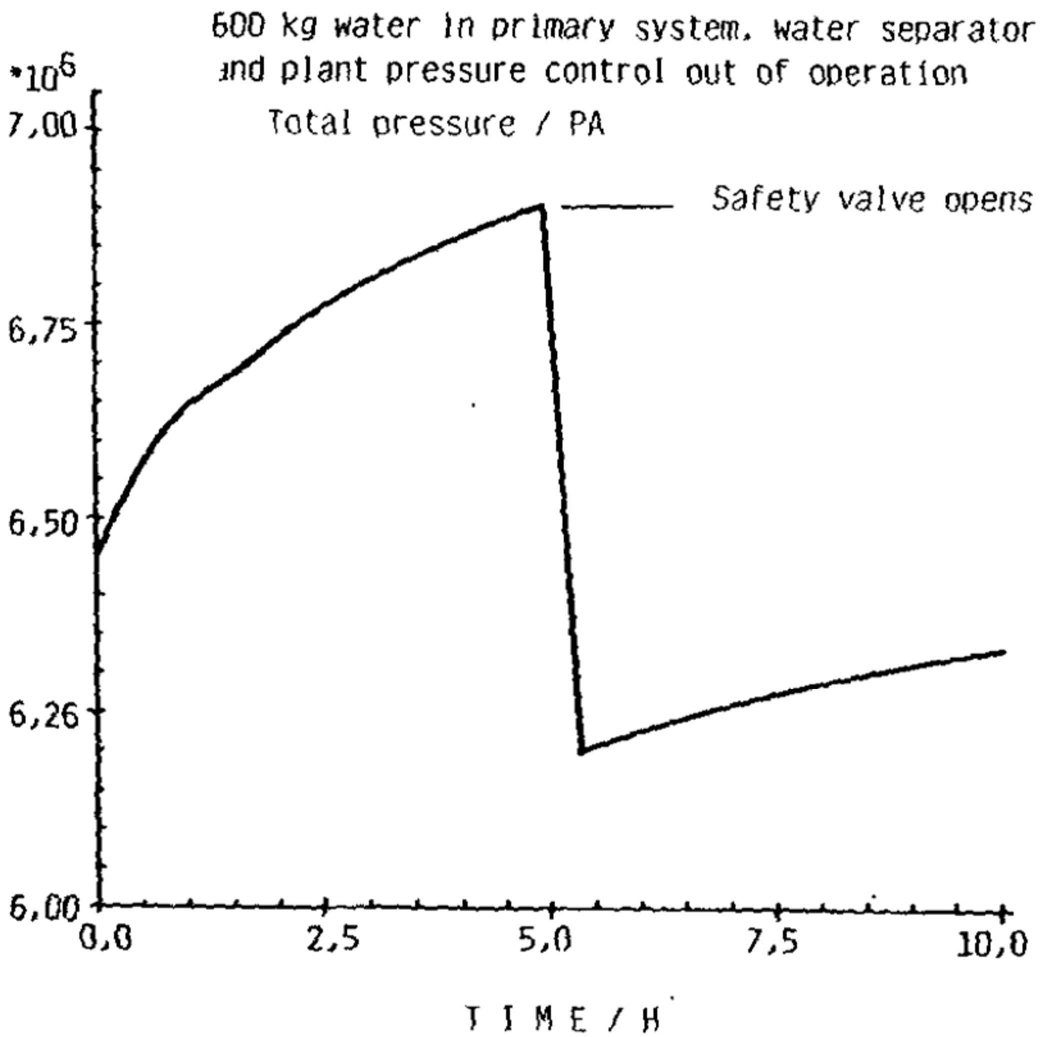


Figure 5-24: Pressure Response in Primary System



6.0 SAFETY IMPACT OF PBR TECHNOLOGY ISSUES

6.1 Safety Impact of Stochastic Core

For each reactor the HTR-Module refueling system continuously feeds about 5,360 fresh and partially-burned fuel pebbles daily into the top of the reactor vessel that contains approximately 360,000 pebbles also at various burned up states. The newly fed pebbles mix with the existing fuel pebbles and slowly move downward through the reactor core eventually exiting at the reactor vessel bottom and enter the fuel handling system where the pebble burnup measurement is made. If the measured burnup value exceeds a set threshold, the pebble is sent to the used pebble storage; otherwise the pebble is returned to the top side of the reactor core. The threshold is controlled so that the number of discarded pebbles is equal to the number of fresh pebbles added to the core, at least in the long run. Pebbles are small units of fuel and are used in thousands. They are not given an explicit identity, they are just counted. In this, a PBR is comparable to a liquid-core reactor but in contrast, the pebbles gain a temporary identity by their measured and recorded results. Because of this fueling and pebble accounting, the operator can discharge only pebbles with highest burnup – a unique advantage of PBR technology.

The disadvantage of the PBR fueling techniques is that the trajectories of the pebbles in the core cannot be known deterministically but must be estimated statistically. This adds uncertainties on how the materials and the nuclides are distributed in the core and thus to the prediction of neutronics and thermal hydraulics of the reactor core.

The continuous fuel sphere cycling allows for limited excess reactivity in the core and allows for the small measure of control rod worth for power control. The pebble flow and burnup history is important for the fuel temperature and flux distribution within the core. However, the core power peaking and average core temperature must be statistically calculated accounting for the uncertainties of the stochastic core.

General safety significances of this issue are discussed in the next section followed by an assessment relative to the HTR-Module design.

6.1.1 Issue Characteristics

The key issue is the additional uncertainties in predicting power peaking and fuel temperature distribution of the PBR core as a result of the stochastic nature of pebble movement. The "characteristics" that define this issue for the HTR-Module are the uncertainties of the "pebble movement" and "power and temperature profiles" in the core. Excluded from this issue can be uncertainties about the maximum fuel temperature when operating a PBR since it can be determined, based on the measured coolant core-outlet temperature, with higher accuracy than in most other reactor types due to the comparatively small temperature difference between fuel and surrounding coolant. References [7], [8], [9], [10], [11], [12], [13], [14], [15], [16], [17], [18], [19] and [20].

Uncertainty of Pebble Movement in the Core

The flow behavior of the pebble bed has been investigated [17] since the initial pebble bed HTGR development. Extensive AVR and model experiments with confirmation from numerical simulations were performed to investigate the pebble flow behaviors in the PBR. These investigations have determined that the motion of the pebbles can be grouped into feeding pattern of pebbles and in-pile behaviors. Also evaluated in these studies, and addressed below, are the effects that "stagnation zones," "crystallization" and "dome formation or bridging" have on pebble movement.

<u>Feeding of pebbles</u>

A pebble is introduced into the top of the reactor core. After landing at the top of the fuel pile, the fuel pebble can potentially, but not necessarily, roll away partly toward the reflector, away from the center of the core where it is dropped. Since this is a random process, the initial position of a fuel pebble is not known deterministically.

However, investigations have shown that the initial radial position of a fuel pebble on the surface of the pebble bed follows the normal distribution of the pebble flow on the surface (principal of flow continuity).

The AVR design, with one central fueling pipe and four satellite pipes feeding pebbles into the core, resulted in a more complex pebble bed surface with a large central fuel pile surrounded by four smaller fuel piles. Moreover, since the central fueling did not drop the pebbles fully vertically, there was a slight asymmetry in the core build-up. Since the HTR-Module uses a single central fueling tube it is expected to have a pure vertical loading of fuel and result in one central fuel pile on the surface of the core.

In-Pile Behavior

The important parameters that describe pebble flow behavior inside the fuel pile are pebble path and relative pebble flow velocity through the bed. Excessive time spent in a high power region of the pebble bed may result in higher irradiation than pebbles are exposed to on the average.

The positions of the pebbles can be determined by the flow lines and the random dropping from the top of core. However, because fuel pebbles pass through the core several times, there are many possible combinations of flow trajectories resulting from each core pass with varying probabilities. Because of the stochastic nature of pebble dropping and the in-pile flow behavior there is some probability that a fuel pebble goes several times straight through the hottest region (i.e. highest power) of the core.

The following are some of the major findings of different studies on the in-pile flow behavior of fuel pebbles:

- The pebble flow is laminar, that means the pebble flow "streamlines" do not cross each other. Nevertheless, the motion of individual pebbles is random on a small extent. The pebble flow is organized and not chaotic. That means that if the initial radial position of a pebble is fixed, the residence (i.e. through-put) time can be predicted with high accuracy.
- There is very little interference between pebble flow paths.
- Pebble flow velocities are small near the top cone, but increase sharply towards the bottom section.
- Velocity decreases near the reactor wall also as a result of wall friction, hence, the inner flow could be considered as a slug flow.
- Pebble flow is essentially vertical in the upper parts of the core with pebbles moving only a few pebble diameters in the radial direction.
- Mixing of pebbles is only noticeable in the lowermost defueling zone, a more neutronically insignificant region of the pebble-bed core.
- Pebbles located within 1½ pebble diameters of the reflector will touch the side reflector wall during their flow paths down through the core; this could have some additional slow-down effect relative to the other pebbles.
- The continuous sphere cycling limits excess reactivity in the core and allows for the small measure of control rod worth for power control

Additionally, pebble flow behavior on the bottom of the core becomes more uniform with increasing core floor "cone" inclination, a cone inclination of 30° has been determined to be an optimum between neutron-physics demands and pebble flow behavior demands. For core heights greater than 0.8 times core diameter, the influence of the core height is negligible. Since the HTR-Module design meets both of these criteria, a uniform pebble flow velocity profile can be assumed for the HTR-Module core.

Stagnation Zones

Regions of a pebble bed core that do not have pebble flow in any direction are considered as stagnation zones. The stagnation zone is normally caused by irregular core internal surface geometry. A stagnation zone, depending on where it occurs, could be crucial for the source term of the pebble bed reactor, since prolonged residence time could possibly cause fuel particles to fail thus releasing fission products. However, based on experiments at AVR [17] it was concluded that in the case of HTR-Module core geometry no stagnant zones are expected.

Crystallization

At very slow discharge rates, the pebbles can start to arrange in a perfectly ordered manner called crystallization, which can block local pebble flow. In order to avoid the crystallization of pebbles along the reflector, the reflector is slotted with indentations to enhance random motion of the pebbles near the reflector wall. These perturbations force the pebbles to move slightly in a radial direction on the way down. Due to the indentations in the reflectors, crystallization is not expected in the HTR-Module core.

Dome Formation

The pebble porosity driven flow exhibits a tendency of dome void formation or "bridging" over the discharge tube. This could cause pebble flow blockage, disruption of pebble flow and result in excessive burnup. This phenomenon can be avoided when the fuel discharge tube-to-pebble diameter ratio is 5.0 or greater. Given this rule of thumb, there was no dome formation detected during the operation of AVR that had a tube-to-pebble diameter ratio of 8.3 and that the tube-to-pebble diameter ratio of the HTR-Module is 10. Therefore, no dome formation is expected to occur in the HTR-Module core.

Uncertainty of Core Power and Temperature Profiles

In the HTR-Module:

- Fresh fuel is introduced at the top, circulated and reloaded until a set threshold burnup is reached; therefore, the core has a burnup profile with more fuel in the upper part of the core.
- The coolant flows from top to bottom so that the upper region of the core is cooler than the lower. This leads to a power profile peaking in the upper region of the core.
- In a cylindrical core, the maximum power density occurs along the centerline axis of the core.
- The average coolant gas core outlet temperature is generally specified as the reference parameter for the hot coolant temperature.

However, there are analytical uncertainties associated with the average coolant gas core outlet temperature and with the coolant flow distribution in the core. At AVR, with its insufficient temperature instrumentation of the hot coolant area, the outlet temperature was calculated from the core power, coolant gas inlet temperature and helium mass flow. Direct measurement of the gas temperature in the core is not possible and the determination of the mass flow, since no direct measurement was available, was by correlating the pressure drop over the circulator using their characteristic curve. The flow distribution in the core may be influenced by the shape of the core, porosity distribution in the core, and the inlet and outlet conditions.

A number of experiments and numerical simulations were performed in order to quantify these uncertainties in terms of statistical temperature distributions or deterministic hot-spot calculations. Discussed below is the evaluation of the AVR melt-wire experiment and its statistical results on temperature distributions and hypothetical simulations of clustered fresh fuel pebbles as the deterministic bounding values for the hot-spot analysis in PBR.

AVR Melt-Wire Experiment

In 1986 (after first time in 1974), an improved experiment was conducted in AVR to determine the radial distribution of the maximum hot gas temperature in the core. Pebbles containing fusible wires with different melting points were selectively loaded into top of core. After discharge and post-examination, the results showed that 21 out of 144 pebbles experienced maximum temperatures of at least 1264°C, far higher than the nominal core coolant outlet temperature of 950°C.

The unexpected differences in the maximum temperatures of monitor pebbles and the accepted nominal core outlet temperature can be attributed to the special design features of the AVR, as the four graphite buttresses protruding into the core, and in particular the strong effects of large core bypass flows, that due to an insufficient temperature instrumentation of the hot coolant area, were not recognized.

While AVR temperatures were higher than expected, the experiment has provided reliable information on the temperature distribution that AVR elements have seen, namely around 1070 °C when going through the central region of the inner core and at least around 1290 °C when going through the hottest zones of the outer core. References [21], [22], [23], [24], [25], [26], [27], [28], [29], [30], [31], [32], [33] and [34].

Hot Spots

In a pebble bed reactor, the loading and movement of fuel pebbles through the core follow a somewhat random process. These stochastic processes could, at first glance generate concern of the possible development of "hot spots" resulting from clustering of low burnup fuel pebbles that may form in the regions of high thermal neutron flux generating higher local thermal power and fuel temperatures than average.

In a circulating pebble bed, formation of clusters of any specific burnup is unlikely. Circulating pebble bed normally disorders any ordered cluster formation. Nevertheless, a cluster of low burnup pebbles does not suddenly appear but must gradually develop. Should it start developing and begin to have a more than negligible effect on power and temperature the reactor operator will observe the asymmetry of power in the core coolant outlet temperature, a rise in coolant activity, and probably also a higher demand for fresh fuel to compensate for reactivity losses long before any inadmissible temperature could be reached.

Yet, several studies were performed investigating the effect of accumulation of fresh (or highly reactive) fuel pebbles on their power and temperature loads during normal reactor operation and in accident scenarios such as a pressurized loss-of-flow cooldown event. Two of these studies found the following:

1. In a study on the South African PBMR core peaking power [16], a batch of 20 fresh fuel pebbles were introduced into the region with the highest power where the maximum volume averaged power peak occurs. The effect of the introduction of 20 fresh pebbles was very small and no significant effect could be seen on the maximum power.
2. A study performed at Idaho National Laboratory (INL)/PBMR analyzed [12] the consequences of the formation of clusters of different sizes combined with an estimation of the probability for their occurrence in a pebble-bed reactor of an annular core. This study determined that:

   a. The peak fuel temperature after a complete and sudden loss of pressure and forced cooling (bounding DLOFC) is unchanged from the nominal unperturbed value
   b. The data shows a 30C jump in this peak DLOFC temp from the nominal (complete mixing - no fresh pebbles adjacent to each other) to the case in which 2 fresh pebbles are adjacent in the location of a hotspot. This particular result is not significant because of the crude way in which the calculation was performed. (Analytically a uniform, partly burned region of the core was replaced with all fresh fuel) The initial jump in temp is more of a computational artifact than a physical phenomenon. The signficant result is the trend observed as the size of the fresh fuel cluster is analytically increased. There is very little rise in the peak DLOFC temperature even when large clusters are assumed.
   c. A probability value of 9.95E-3 was reported. This is the fraction of the core volume comprising the hotspot and was computed from the dimensions of the region somewhat arbitrarily chosen by the authors. The data in the paper, based upon basic probability theory, show that about 19 Two-pebble clusters of fresh fuel can be expected to exist in a hotspot of this volume. The probablility of finding even one Four-pebble fresh fuel cluster in this hotspot is, however, much lower.

   Therefore, a reasonable interpretation of the INL/PBMR study is that the existence of clusters of low-burnup fuel does not signficantly increase the peak severe accident temperature.

6.1.2 Affect on Safety Analyses

The following provides an assessment of the potential effect of the stochastic core in the HTR-Module on the safety analyses described in Chapter 5. This assessment is addressed in accordance with the issue's critical characteristics discussed above.

Based on experimental and analytical results, the stochastic nature of fuel pebble movement in the PBR core is well understood. Due to the simple continuous geometry of the HTR-Module and the chamfered edge at the upper end of the "cone" problems that were observed in the past, such as the presence of a stagnant zone, will not occur. And pebble flow paths through the core do not cross readily and are predictable with statistical methods.

The AVR melt-wire experiment provides valuable information on the maximum fuel temperature distributions as fuel pebbles pass through the core. Also studies have shown that fuel temperature in the outer core region is higher than the inner core, and pebbles move with propensity more towards the inner region as they flow downward. Although the AVR temperature measurements appeared to be higher than expected, studies show that the temperature differences were mainly due to coolant bypass flows and radial power distributions that were not included in the original analysis. PBR core flow bypasses as such do not pose an insurmountable problem as long as the effects on core temperature distribution are taken into account. As one of the lessons learned from AVR, in future PBRs the coolant temperature should be measured close to the core exit, i.e., for a down-flow reactor design these measurement points are in the coolant channels of the bottom reflector.

Hot spot studies by PBMR and INL show that although the maximum power delivered in a fuel pebble may increase due to clustering, the maximum fuel temperatures increases only moderately in normal operation.

Because margins have been incorporate in the safety analyses calculations that account for these conditions, it can be concluded that the results of the HTR-Module safety analyses (presented in Chapter 5) adequately account for the stochastic nature of the pebbles either in terms of pebble movement or core power and temperature profiles.

6.2 Safety Impact of Potential Core Compaction

The pebble bed core is filled with spherical fuel elements. Each fuel element is either resting on one or more elements below, is supporting the fuel elements above, and is in contact with one or more fuel elements to the side. Furthermore, the pebble bed core is in a porosity driven flow as the fuel elements move from top to bottom. This dynamic nature of the pebble bed core sets up a unique neutronic and thermo hydraulic characteristics as it reaches equilibrium condition.

The core equilibrium compaction could be disturbed by an external event such as an earthquake. The potential impacts of an earthquake on a pebble bed core are:

- Increased fuel element packing fraction,
- Change in coolant flow pattern.

General safety significances of these issues are discussed in the next section followed by an assessment relative to the HTR-Module design.

6.2.1 Issue Characteristics

The phenomenon of a dynamic core conditions and the potential unexpected compaction ratios were experienced in the THTR300 experience.

Increased core compaction as a result of core design or external effects could have the following characteristics:

a) Sudden increase in core packing fraction will inject positive reactivity into the core causing a power excursion.

b) Change in coolant flow patterns as the result of increased core packing fraction could increase core by-pass flow and result in a general core temperature increase.

6.2.2 Affect on Safety Analyses

In a cylindrical core such as HTR-Module a core packing fraction of 0.61 is expected in the equilibrium condition. This means 61% of the core volume is taken up by the fuel elements.

This natural packing fraction could be altered by external effects such as earthquakes that could shake-up the reactor vessel and suddenly increase the core packing fraction. The HTR-Module core fuel element compaction due to an earthquake event was discussed, and evaluated in section 5.2.6.

The safety analysis results indicate that following an earthquake the reactor is scrammed by two diverse reactor protection system scram signals. Following the earthquake, the first scram signal initiating criteria is reached after approximately 2 seconds and the second scam signal initiating criterion after approximately 4 seconds. The analysis also indicated that, even if a scram is not initiated, the long-term power increase due to sudden reactivity insertion would be extremely low. The hot gas temperature increase would be about 50 K after approximately 500 seconds.

6.3 Safety Impact of Graphite Dust

The HTR-Module Pebble Bed Reactor is designed for continuous refueling, via a fuel handling system, of about 5,360 fuel pebbles daily into the reactor vessel that contains approximately 360,000 pebbles in various degrees of motion and position in the core. Graphite dust is generally created by the rubbing of fuel pebbles (with each other, with the fuel handling equipment and the side reflector walls) in a porosity driven flow as the pebbles move through the reactor and in the fuel handling system (friction dust). There are other possible sources of graphite dust formation such as from the decomposition of structural graphite due to ingress of foreign fluids into the primary circuit but these other sources are considered insignificant based on the AVR experience. Additionally, tests have shown that the major source of friction dust generation is from the fuel handling system, in particular from the lift lines, and that the pebble bed produces only a minor portion of the total dust generated.

The graphite dust is a concern because of its fission product retention qualities and its mobility. The potential impacts of graphite dust are:

- Depressurization accidents releasing radioactive dust outside the reactor coolant barrier,
- Contaminated dust could possibly limit or prevent event mitigation measures during depressurization accident,
- Dust released into confined areas could possibly result in dust explosions,
- Contaminated dust could affect worker dose rate white dismantling or maintaining of components.

General safety significances of these issues are discussed in the next section followed by an assessment relative to the HTR-Module design. References [35] [36] [37] [38] [39] [40] [41] [42] [43] [44] and [45].

6.3.1 Issue Characteristics

As noted above a key issue is the potential for contaminated mobile graphite dust in the primary circuit that could be released outside the reactor coolant boundary during the depressurization phase of a design basis event. The “characteristics” that define this potential safety issue for the HTR-Module are the “mobility”, “mass” and “fission product retention capabilities” of the graphite dust.

HTGR Testing Program

The assessment of this issue relies heavily on the results of the German HTGR (Testing) Program that performed experimental investigations on dust generation, dust deposition and dust remobilization. The most interesting and relevant experiments were performed in the AVR because of the real operating conditions, length of operation and that the AVR was contaminated with significant dust-bound metallic fission products (strontium-90, caesium-137). Also note that all plant designs and concepts on which these experiments and measurements are based included dust filters in the helium purification systems, however, direct measurements of dust generation were not possible.

From 1973 to 1988 there were 141 experiments and investigations performed at AVR. Each change of an experiment required a shut down and restart of the reactor. Of particular note and usefulness to this assessment were the "Helium Flow Transient Experiments" performed starting in 1986. The helium circulator was used to introduce primary flow transients by suddenly increasing the circulator speed to remobilize settled graphite dust and capturing it in prepositioned dust filters.

In 2008/2009 the "Dust Experiments on AVR Components" were performed at AVR to gather data on dust and fission products in order to reduce uncertainties in nuclear safety calculations for the South African pebble bed reactor program. Experiments were performed on two sections of piping; one was operated under turbulent condition and the other under laminar conditions. Experimental data from both sections of piping were selected to cover both ends of the flow regime existing within the primary circuit design.

Dust Mobility - Normal Operation

The dust produced during normal operation is transported by the coolant until it is either deposited on a surface or remains in continuous recirculation. This process has been shown to result in very low helium borne dust concentration during normal operation. For example, the average concentration of helium borne dust in the stationary locations at AVR as measured in over 21 years of operation was 5µg/m^3 (STP).

The relevant deposition mechanisms are inertial impaction, diffusion, thermo-phoresis turbulent deposition and gravitational settling. Several successive deposition/re-suspension steps may occur before the particle reaches its final location. This effect ensures that all particles except for a small amount currently participating in the deposition/re-suspension equilibrium with the fluid have sufficient adhesive forces to the surface to remain fixed during normal operating conditions resulting in crust-like multi-layer deposits. Also, it has been confirmed that dust is deposited in areas subject to high gas velocities and that the dust in those regions comprises at least a significant major fraction of the overall amount of dust.

Dust in dead-zone regions (non-turbulent) that are part of the actively circulated primary circuit, such as pipe bends and the sides of steam generator tubes facing into the flow direction, would be minimal due to a lack of dust transport during normal operation. Preliminary simulations of the deposition/re-suspension processes that determine the dust distribution in the whole primary circuit of the HTR-Module confirm that the major part of the dust is deposited on the steam generator.

Dust Mobility - Depressurization Event

Dust deposited on a surface in the primary circuit can be re-mobilized during accidents involving depressurization. The re-suspended dust particles can be carried out of the primary circuit and, in part, out of the confinement.

In the "transient experiments" at AVR starting in 1986 the resulting helium borne dust was captured in dust filters until the accumulation reached a stationary value. The decrease of the activity collection in time in the dust filter was measured online and considered as a measure for the dust concentration of helium borne dust. This resulted in a concentration of helium borne dust of 1050 µg/m^3 (STP).

This was followed up by "Dust Experiments on AVR Components" in 2008/2009 that revealed a strong binding of the dust as a closed layer to the inner pipe walls such that dust was recovered only by forced scraping of the walls with a scalpel. No loose dust was found and loose dust could not be obtained by hammering on the pipe walls. It was concluded that the amount of remobilized dust during a depressurization event would be very low confirming as reasonable the results from the earlier circulator transient experiments.

Note that for HTR-Module those parts of the primary circuit normally filled with stagnant helium, especially the annulus between RPV and core barrel are not expected to produce any significant additional dust re-mobilization because:

- Those compartments normally do not exchange gas with the rest of the primary circuit and there is no way significant amounts of dust can be transported into them.
- There are two dominant ways of dust deposition in stagnant compartments: Gravitational settling and retention of the dust carried by those fractions of gas that creep through tiny gaps and porosities in those small spaces by effects similar to filtering. These gaps act as total sinks for the dust because even during fast depressurizations, no remobilization-relevant velocities can occur there. Gravitational settling in the RPV/core barrel annulus moves the dust to surfaces that are away from the flow paths during depressurization due to their dead-end location, so no significant remobilization can be expected here, either.

The AVR dust experiment provided an unexpected result in one section of the laminar flow pipe and that was the presence of rust dust. The rust may have been produced in that part of the pipe because this pipe part had been used for the dehumidification of the AVR reactor after a water ingress event in 1978. The procedure consisted of the evacuation of wetted gas that previously was pumped in as dry gas.

Mass of Graphite Dust

For the transient experiments starting in 1986 at AVR it was estimated that the total dust contained in the reactor primary circuit was about 60 kg. Based on the 60 kg value, it was determined that the mass of helium borne dust at stationary conditions would be 8 mg and at a depressurization phase it would be 1.73 grams.

In the 2008/2009 experiments at AVR the total dust mass was determined to be 70 kg based on surface specific scraped-off dust mass assumed to be representative of other parts of the primary circuit. Up-scaling of the 2008/2009 AVR value to the HTR-Module resulted in the dust generated for the HTR-Module over 32 full power years as follows:

- Total mass of graphite dust in the primary circuit: 727 kg
- Total mass of helium borne dust at stationary conditions: 210 mg
- Total mass of helium borne dust at depressurization phase: 45 grams

Note that this up-scaling does not take into account the deviation from normal operation flow patterns in those parts of the primary circuit normally filled with stagnant helium that, as discussed above, is expected not to be significant.

The most important unexpected result of these recent measurements is that there was no loose dust in turbulent flow regions of the primary circuit and a in the laminar flow regions most of the dust was stuck to the surfaces and very little was remobilized, noting that this was about 20 years after the end of operation.

Fission Product Retention Capabilities

The graphite dust is likely to cover all surfaces in a pebble bed reactor and the fission products on structural surfaces can be expected to be associated with this dust (rather than the underlying alloy) and be relatively uniform.

Two general mechanisms can be identified that lead to a contamination of the dust:

1. If the particles are produced by abrasion of fuel elements, any radio-nuclides present in the fuel element matrix will lead to the same specific activity of the dust.
2. Volatile fission products in the coolant can be adsorbed by dust particles that are either circulating with the helium or deposited on primary circuit structures.

The process of fission product adsorption competes with the plate-out on colder metallic structures and also with adsorption on solid graphitic structures like the side reflector. Therefore, not all of the free activity of the primary circuit can be considered dust-borne. Moreover, adsorption of coolant-carried fission products can only take place on surfaces that are in contact with circulating helium. Therefore, the activity of dust deposited in regions where helium is stagnant during normal operation will be limited to the radio-nuclides present in the fuel element matrix if the source of the dust particles is fuel element abrasion.

6.3.2 Affect on Safety Analyses

The following provides an assessment of the potential effect of the contaminated graphite dust generated in the HTR-Module on the safety analyses described in Chapter 5. This assessment is addressed in accordance with the issue's critical characteristics discussed above.

Releases During Depressurization Phase of a DBA

Most of the total quantity of estimated dust generation, i.e., 727 kg, will not be remobilized for the depressurization phase of a DBA, even in the case of a large break, because most of the dust is plated out on the primary circuit walls with very strong binding to the walls (it had to be scraped very strongly with a scalpel to recover the dust).

The total dust that can actually be released out of the primary circuit was estimated to be:

- Permanent helium borne dust: 210 mg
- Remobilized dust during depressurization: 45 g.

The safety analyses for depressurization of the HTR-Module (Section 5.4) used an enveloping value of 1kg of remobilized graphite dust that is 22 times the estimated available remobilized dust of 45 grams (the permanent helium borne dust being relatively insignificant).

The dust amount at the end of life does not present a release problem, even for larger amounts, because of the formation of closed layers at the inner surfaces. It is also not a volume problem, because the layer thickness is in the range of the roughness of the inner surfaces.

Event Interventions

Credit is taken in the analyses of the depressurization events for intervention by plant operators (Section 5.4). This intervention requires access to the reactor surroundings. Contaminated dust releases to the surroundings may negatively impact these interventions. However, the safety analyses for depressurization of the HTR-Module (Section 5.4) used an enveloping value of 1kg of remobilized graphite dust that is 22 times the estimated available remobilized dust of 45 grams. Therefore, credit for the event interventions accounts for the expected contaminated dust release, with substantial margin, and therefore the interventions should not be affected.

Dust Explosions

Fine dust is theoretically combustible and possibly could result in dust explosions during the depressurization phase of accidents. The explosion limit of graphite dust is given at 70 g/m^3 in air for 4 µm sized particles (the upper limit in particle size found in the AVR experiments). Given an upper limit for the total re-suspended dust mass of the 22g, just the distribution in the helium gas puts the concentration below the explosion limit. Additionally, the combustion of dust would require the presence of oxygen, and the mixing of the helium-carried dust with the air in the reactor building will lower the dust concentration even further. Moreover, although it is sufficient to assume perfect mixing for the calculation with a zero-dimensional containment code (the term is used for the sake of consistency, although in the case of HTR-Module, the structure to be analyzed is the confinement), in reality stratification effects would prevent an easy interaction of the dust with oxygen.

Interference with Radiant Heat Transfer

The absorption and scattering of thermal radiation due to presence of particulate or dust in atmosphere plays an important role in the overall energy transfer between the RPV and RCCS, especially under a DBA scenarios involving depressurization. Thermal radiation heat transfer can be either enhanced or attenuated, depending upon the size and concentration of particles, the temperature distribution, and the radiative properties of the dust particle. When a beam of radiation is incident upon a dust cloud, some of it is transmitted, some absorbed and some scattered. The scattered radiation includes that which is diffracted, refracted and reflected by the dust particles. The absorbed radiation is retransmitted at a wavelength corresponding to the temperature of the dust particle and not the source.

The attenuation of thermal radiation within a dilute cloud of pulverized coal and ash has been investigated experimentally and theoretically in the ranges of 1.6 to 30 μm particle size. An empirical expression has been developed for obtaining the absorptivity and emissivity of a coal/ash cloud [46]. Based on these empirical data the effective absorption and emission coefficients of dust are found to be negligible in the ranges of dust particle size and concentration of the HTR-Module.

Dismantling/Maintenance of Contaminated Components

The dust amount at the end of life, because of the formation of closed layers at the inner surfaces, will result in heavily contaminated components. For AVR a lot of components were dismantled, decontaminated and recycled without any problem. This is also expected to not be a significant problem for the HTR-Module.

6.4 Safety Impact of Broken and Lost Pebbles

The HTR-Module Pebble Bed Reactor contains approximately 360,000 pebbles in constant motion through the vessel of which about 5,360 a day are taken from the vessel bottom. These are then processed through the refueling system where they are checked against a set burnup threshold and if they exceed it they are discarded in a number appropriate to the number of fresh pebbles that the reactor needs (about 360). Due to this constant movement of a large number of pebbles the possibility that some pebbles are damaged or broken, particularly by the refueling system, cannot be excluded. The concern is the possibility that broken fuel pebble pieces could get stuck in the core or in the graphite core assembly outlet structure (essentially permanently) where they gradually accumulate fluence eventually resulting in particle failures.

General safety significances of this issue are discussed in the next section followed by an assessment relative to the HTR-Module design.

6.4.1 Issue Characteristics

The key issue is the possibility of damaged or broken pebbles permanently stuck in the core or the graphite core assembly outlet structure where they could gradually accumulate fluence eventually resulting in fuel particle failures. The conditions that must be evaluated that define this issue for the HTR-Module are: a) extent of stuck damaged pebbles, b) effect on pebble movement, c) effect on coolant flow disturbance and d) extent of over-irradiated stuck fuel pebbles.

Extent of Stuck Damaged and Broken Fuel Particles

Experience with the German PBRs, the AVR and THTR, have shown that the refueling system can be instrumental in damaging fuel pebbles. These damaged pebbles then can become transported and stuck in the core; however, experience has shown that such pebble damage can be minimized by proper refueling system design and operation. The damaged pebbles are detected and separated from the main fuel flow stream and are collected in the damaged fuel (scrap) collection container.

AVR Experience

The AVR pebble transport system was the first of its kind and in the first years of operation many changes were incorporated to its system components and control techniques. It then operated for more than 18 years with system failures accounting for only about 3% of the plant unavailability. Over this period the AVR experienced a variety of pebble damage as the operator gained experience with the system operation and made improvements. Some of the most significant causes for increase in scrap production and resulting fixes were as follows:

1. The initial AVR core used shell-type pebbles that, although less sensitive to surface damages compared to the later pressed-type pebbles, showed a higher breaking rate due to the 40 mm diameter gap between the inner fuel sphere and the graphite shell.
2. The increase in the use of pressed pebbles increased the scrap rate because the pebbles were surface damaged when falling on sharp-edged ribs of the scrap separator. A re-design of the scrap-separator waltz solved the problem.
3. In 1975, a new design of the “dosing wheel” - a pressure lock in the pebble line between the scrap separator and the pebble distribution and lifting device - resulted in an increase number of scraped pebbles. By rotating the wheel, a pebble was transported from the incoming pebble pipe to the outgoing one, while maintaining a barrier function for helium. A number of pebbles were usually collected in front of the dosing wheel before the pebbles were further sent on their way, one by one. Fresh pebbles, too, locked into the primary system in batches of 10, were lying as a column of 10 pebbles in front of the wheel. It was found that pebbles could jam in front of the wheel and that this could affect the proper position of the pebbles in the wheel in the moment it was rotated. The effect depended also on the number of pebbles in the column. A limitation of this number to 5 reduced the scrap rate to approximately 3 to 4 damaged pebbles in 100,000 circulated.
4. However, loading fresh pebbles in batches of 5 prolonged the time operators had to stay in the pebble charging room where all the fresh pebbles were stored and this increased the personnel dose uptake. Later on, as an experiment, the fresh fuel loading was returned to batches of 10 but again the scrap production rate rose. As a consequence, design and control of the dosing wheel were modified in 1986. With the new design, the scrap rate went down to approximately 1 to 2 pebbles in 100,000 circulated and remained that low until the end of AVR operation.

Analyse of the contents of the AVR pebble scrap showed that the total amount of scrap removed from the system was equivalent to about 220 pebbles that is low considering 2.4 million pebbles circulated.

More pebble scrap fell through the coolant channels in the core bottom reflector, bottom carbon structure, and core support structure. This scrap has not been analysed and is still present in the now grouted reactor vessel. The amount of scrap accumulated there is assumed to be smaller than what was removed from the scrap container. All in all, the total amount of pebble scrap produced in 21 years of AVR operation is assumed to be equivalent to some 300 to 400 pebbles out of nearly 300,000 fuel pebbles and approximately 70,000 non-fuel pebbles used in the AVR and also compared to some 2.4 million circulated pebbles.

THTR Experience

There were a lot of pebble scraps produced by the THTR. This has been primarily attributed to the shut-down rods that were directly forced into the pebble bed core. The problem was minimized later by lubricating the control rod tips with methane. The concept of direct core rods is no longer considered. The THTR pebble transportation system is an improved design consideration compared to that of the AVR. In the short time that the THTR operated, it processed the same amount of pebbles that were circulated in the AVR and there were no significant problems.

Effect on Pebble Movement

The issue is the possibility that stuck pieces of broken pebble scraps in the coolant channels of the core bottom structure and the conical surfaces of the bottom reflector that could trap the pieces and impact pebble movement and the coolant flow.

Throughout the AVR operational history there was never any evidence that pebble scrap transported in the core influenced pebble flow behavior or the coolant temperature. There were a number of occasions to experimentally get insights into the real pebble flow behavior of the reactor. All obtained data revealed a very stable and uniform pebble flow behavior over time. Moreover, in 1999, after the AVR had been defueled, the bottom reflector was video-inspected and no such scrap pieces were detected.

Effect on Coolant Flow

The issue is the possibility that stuck pieces of scrap in the coolant channels of the core bottom structure could reduce coolant flow through the channels thus impacting the fuel and or the coolant temperature.

The previously mentioned AVR video pictures shows pebble scrap sticking in the coolant channels. This "scrap", however, is uniform and consists exclusively of inner spheres of the above mentioned shell-type pebbles, nominally 40 mm diameter, pressed into a 35 mm wide core bottom reflector slit. There are altogether 12 such stuck inner spheres visible. However, they did not disturb plant operations or the coolant flow.

The video pictures also reveal damages in the bottom reflector structure. There were various cracks in the upper graphite-block layer and blocks have shifted sideways. As a consequence, some coolant channels are narrowed or even closed, while others became wider so that intact pebbles could enter. These structural changes had a much stronger impact on coolant penetration than the stuck scrap. Yet, there was always a stable relation between coolant circulator speed and measured core pressure drop. The latter included the bottom and top coolant penetration channels. In other words, the pressure drop across the pebble bed was so predominant that some changes in the bottom channel structure were negligible.

Extent of Over-irradiated Stuck Fuel Pebbles

The issue is the possibility that stuck pieces of damaged fuel pebbles in the coolant channels of the core bottom structure could, in the long-term, be over-irradiated possibility resulting in high rate of fuel burnup or fast-neutron damage. Note that this assumes that damaged pebbles in the core are too remote from high flux region to be a concern.

<u>Stuck Pebble Burnup</u>

High level of burnup of stuck damaged pebbles could result in increased levels of particle failures and fission product releases. As discussed above the AVR did have stuck fuel pebbles scraps, at least the inner spheres of the shell-type pebbles, belonging to the first core that were irradiated for the whole AVR lifetime. Their BISO-coated particles did not show any important fission product release primarily due to the low neutron flux field at the bottom reflector region. This conclusion was further demonstrated by the new, precise pebble measurement system, installed at the end of 1981, that showed that after the core was cleaned the coolant activity went down to low values and remained there for the rest of the AVR operation which included the stuck pebbles at the bottom of the core.

Note, however, that there were high levels of fuel pebble burnup in the AVR. The TRISO-coated fissile particles of the GFB (feed/breed) type pebbles achieved a very high burnup but still showed excellent fission-product retention despite the very high AVR temperatures.

<u>Fast Neutron Damage</u>

In the AVR, for test purposes, some 2000 THTR pebbles (with BISO coatings) intentionally were not discharged, as required, when they reached high levels of burnup but were left in the reactor for about 13 years and went through the core more than 10 times on the average. As a rough estimate, the fast-neutron dose was at the same order of magnitude as is expected for damaged stuck pebbles accumulated in the PBR vessel bottom in some 40 full-power years. No additional or unexpected fission product releases were observed due to this extended cycle time experiment.

6.4.2 Affect on Safety Analyses

The following provides an assessment of the potential effect of the stuck broken and damaged pebbles in the HTR-Module on the safety analyses described in Chapter 5. This assessment is addressed in accordance with this issue's critical characteristics discussed above.

Due to the continuous movement of a large number of pebbles in the HTR-Module some damaged or broken pebbles are expected. Therefore, in any PBR the fuel handling system components that perform separation and collection of damaged pebbles are indispensable. The experience with the German AVR and THTR has shown that the damaged pebble production rate can be kept very small and that it does not represent a problem for a safe and continuous operation. The experience has also shown that a pebble damaged in the core is very unlikely and that the pebble damaging rate in the fuel handling system is a direct function of how pebbles are processed by that system.

Therefore, the HTR-Module refueling system was designed based on lessons learned and experiences gained from the German PBR programs to avoid damaging or retaining damaged fuel scraps or pieces and to include removal of damaged fuel to a damaged fuel cask. Also the HTR-Module bottom reflector design includes design features that prevent stuck scrap pieces if any are present in that location.

Finally, experience and tests with stuck fuel pebbles at AVR have shown that the effect of stuck pebbles on pebble movement, coolant flow, fission product releases from high burnup and fast neutron damage are not a concern for PBRs.

Based on the above it can be concluded that the safety analyses presented in Chapter 5 are not impacted by this issue.

7.0 ACCIDENT EXPECTED DOSE FOLLOWING U.S. REGULATIONS

Dose assessment presented in Chapter 5 was originally performed according to German regulatory requirements. This section utilizes the HTR-Module source terms and calculates doses using U.S. A. dose calculation methodologies and practices. Title 10 of the U.S. Code of Federal Regulations (CFR) and accompanying regulatory guidance documents primarily address light water reactor licensing. There is currently no U.S. regulatory guidance for HTGR accident evaluation or regulatory acceptance criteria. The German HTR-Module Safety Analysis Report (SAR), Version 1988, was used as the basis for the present radiological assessment.

The radioactivity released from the HTGR reactor due to design basis accidents were documented in the SAR. The SAR dose assessments are based on ingestion pathway whereas for the U.S.A. accident doses are based on air immersion and inhalation.

The recalculation of the doses addressed in this section used the radioactivity released and calculated two-hour immersion and inhalation doses for an exclusion area boundary (EAB). The release of radioactivity was conservatively assumed to be instantaneous.

The atmospheric dispersion assumed a concentration factor of 3.35E-03 s/m3 corresponding to an EAB of 0.249 miles (400 m). Dose conversion factors are derived from ICRP-30 consistent with the current regulations.

The basic radiological acceptance criteria for the offsite receptors associated with the alternative source term (AST) methodology for light water reactors are found in 10 CFR 50.34(a)(1) with a limit of 25 rem (0.25 Sv) total effective dose equivalent (TEDE). This criterion, however, is used for evaluating potential light water reactor accidents of exceedingly low occurrence probability and low risk of public exposure to radiation. For events with higher probability of occurrence, the acceptance criteria for the offsite receptors are more stringent.

It is noted the U.S. regulations were changed to re-classify the rupture of a radioactive waste storage tanks as not being a DBA and applying a different acceptance criteria from 10CFR Part 20. This change in the regulation is found in the Standard Review Plan (SRP) Section 11.2 and Branch Technical Position (BTP) 11-6. BTP 11.6 applies the effluent concentration limits of 10CFR Part 20, Appendix B, and Table 2 as acceptable limits.

7.1 SUMMARY

The EAB air immersion and inhalation doses are conservatively calculated and found to be well within the current U.S. accident dose acceptance criteria (Table 7-17). In fact, for a site exclusion area boundary (EAB) of 0.249 miles (400 m), all dose results are within the 10 CFR Part 20 annual dose limits for normal operations of 100 mrem (10 CFR 20.1301). The accidents involving the waste disposal tanks in the auxiliary building (non DBA accidents) with tank failures require dilution factors to meet the 10 CFR Part 20, Appendix B, Table 2 limits. Alternatively, the total activity allowed to be stored can be reduced by administrative limits.

7.2 EVALUATIONS

7.2.1 DBA Case 1 (Section 5.4.1.1)

Large Break of Connection Pipe Between Pressure Vessel Unit and Isolation Valve of Primary Circuit

The bounding scenario for a large break accident, between the reactor pressure vessel and the primary circuit isolation valve, is a 2A pipe break of the Steam Generator discharge routing to the helium purification system. This accident scenario is bounding because of the highest release flow rate.

The release of primary coolant causes a pressure and temperature increase in the reactor building. The individual reactor building sections are connected to each other by wall openings. If the differential pressure between reactor building atmosphere and environment exceeds 0.1 bar, vents open and pressure is relieved by routing accident atmosphere to the stack.

During depressurization of the primary system to ambient pressure, the complete primary coolant inventory, containing the gaseous and aerosol radioactive materials, is released into the reactor building. During blowdown, fractions of the solid and iodine activity bound, by adsorption to primary system surfaces, will be desorbed and together with some of the re-entrained radioactive dust that has accumulated in dead flow zones, will be taken up by the primary coolant flow.

A failure of the primary isolation valves to close in the lines to the helium purification system is conservatively assumed, causing these lines to be depressurized as well. Further radioactive materials are released due to desorption; primarily from the molecular sieve and from the cryogenic adsorber. Most of the activity discharged primary coolant into the reactor building, is released to the environment via the stack.

To calculate the activity releases following conservative assumptions are made:

- Release of the total stationary coolant activity, including stripped out noble gases, tritium and C-14 activity of the helium purification activity,
- Maximum desorption of fission products and iodine isotopes from primary circuit surfaces,
- Bounding value of 1 kg of dust was assumed that is released by the primary coolant from dead flow zones during blowdown and subsequently released from the primary system. The specific dust activity of long-lived radionuclides is equivalent to the specific activity of the graphite on the fuel elements surface. The following activities per nuclide are assumed in the 1 kg dust:
    - I-131: 8.9 E+06 Bq
    - CS-137: 7.0E+05 Bq
    - Sr-90: 5.6E+05 Bq
    - Ag110m: 1.3E+05 Bq
- 100 % aerosol release from reactor building to the environment,
- Iodine is postulated as being in the chemical form which has the greatest radiological effects, i.e., elemental iodine.

Table 7-1: Activity Release of Case 1

Nuclides	Activity Release in Bq (short-term release)
Kr83m	2. 1 E+10
Kr85m	7.6 E+10
Kr85	1.6 E+10
Kr87	7.8 E+10
Kr88	1.6 E+11
Kr89	2.8 E+10
Kr90	1.2 E+10
Xe131m	2.5 E+10
Xe133m	5.7 E+10
Xe133	2.4 E+12
Xe135m	1.8 E+10
Xe135	2.4 E+11
Xe137	4.8 E+10
Xe138	9.3 E+10
Xe139	1.5 E+10
Total noble gases	**3.3 E+12**
I131	1.5 E+8
I132	1.8 E+9
I133	8.9 E+8
I134	4.7 E+9
I135	1.6 E+9
Total iodine	**9.2 E+9**
Sr90	5.6 E+5
Ag110m	2.0 E+5
Cs134	1.6 E+6
Cs137	2.9 E+6
Total long-lived solids	**5.3 E+6**
Rb88	5.3 E+10
Cs 138	2.0 E+10
Total noble gas decay products	**7.3 E+10**
Tritium (H3)	5.6 E+12
C14	5.9 E+10

The Table 7-2 below lists the resulting immersion and inhalation TEDE, based on the initial activity release of Case 1 listed in Table 7-1 above.

Table 7-2: Case 1 TEDE Dose

	Dose at EAB (Sv) [mrem] TEDE
Unfiltered Release	**1.361E-04 [13.6]**
Filtered Release	N/A

7.2.2 DBA Case 2 (Section 5.4.1.2)

Pressure Relief with Subsequent Core Heat-Up

Core heat-up is not an independent accident event. It only occurs in conjunction with preceding primary system depressurization (short-term phase). The radiological effects must therefore be established by addition of the activity released during both accident phases. The activity release during the depressurization phase has already been discussed in Case 1. The calculations for the heat-up phase (long-term) are based on the following assumptions at the start of the accident:

The reactor is abruptly depressurized while in steady-state operation at 105% of nominal power. Depressurization takes place quickly and does not affect the long-term temperature response.

Thirty (30) seconds after reaching the negative sliding boundary value of the thermal neutron flux, the reactor automatically shuts down.

Temperature response during the accident is characterized by the large heat capacity of the core and the ceramic core internals in relation to the decay heat generation rate, by the steadily decreasing decay heat generation rate and by the thermal resistances between the center of the core and the cavity cooler. During the first hours, the central regions of the core undergo almost adiabatic heat-up until the temperature gradient across the core is large enough for a significant fraction of the residual heat to be removed, more than 90% of the energy leaving the core radially. The peak core temperature (approx. 1550°C) is reached after about 32 hours.

The maximum fuel temperatures on core heat-up following primary system depressurization are limited to approx. 1600°C. This is a temperature where the retention capability and integrity of the coated particles are basically intact. Only the small fission product fraction, that is already present outside intact particle coatings (before the accident), can be released from the fuel elements.

An activity release from the fuel elements substantially exceeding that during normal operation is only to be expected some hours after onset of the accident. This is because the fuel temperatures increase relatively slowly and the fission products only diffuse gradually from the inside of defective fuel particles to the surface of the fuel elements.

The transport mechanism for the activity release of the depressurized primary circuit is temperature-induced gas expansion. This is followed by global cooling, that causes intake of primary atmosphere into the primary circuit and terminates activity release into the reactor building. Activity leaving the core zone, with the exception of noble gases, is significantly less than that released from the fuel elements since the graphitic surfaces of colder fuel elements and reflectors act as activity traps. This also applies to the metallic surfaces that the primary coolant passes on its way to the break.

During the core heat-up phase, sub-atmospheric pressure is maintained in the affected HVAC section, and the exhaust air is conveyed to the stack via the filtration unit of the secured sub-atmospheric pressure system.

To calculate the activity releases following conservative assumptions are made:

- All calculated fuel temperatures are increased by 98 K to cover uncertainties in the input variables.
- Accident-induced particle failure: In heat-up experiments with LEU TRISO particles performed in the past, temperature-induced defects were not observed below 1600°C. In the calculations, however, it was postulated that accident-induced particle failure starts at 1200°C, rises exponentially with temperature and reaches a fraction of 0.0005 at 1600°C.
- Only diffusion from the fuel of failed particles is relevant to noble gas and iodine release in the core heat-up phase. The diffusion coefficient applied here has also been validated by results of heat-up experiments.
- Release from the primary system: The time history of activity entrainment through the break into the reactor building is calculated by convolution of the release rate from the fuel elements and the expansion-induced primary coolant blowdown rate. Internal primary system retention mechanisms are considered globally.
- Efficiency of the filter unit in the secured sub-atmospheric pressure system:
  - 0 % for noble gases, 99% for iodine, and 99.9% for solids.
  - The filter efficiency selected for iodine allows for the presence of organic iodine compounds and is extremely conservative for primarily expected elemental iodine.
- The total quantity of iodine released to the environment is postulated as being in the chemical form which has the greatest radiological effects, i.e., elemental iodine.

The Table 7-3: and Table 7-4 list the nuclide specific activity releases in the core heat-up phase of Case 2, with and without consideration of the filter system. Table 7-5 lists the resulting immersion and inhalation TEDE

Table 7-3: Release of Case 2 with Consideration of the Filter System

(Activity release in Bq in the time interval)

	0 – 8 h	**8 – 24 h**	**24 – 72 h**	**> 72 h**	**Total**
Xe133	2.6 E+8	1.9 E+10	1.6 E+11	8.5 E+10	2.7E+11
I131	1.3 E+5	9.2 E+6	8.1 E+7	4.1 E+7	1.3 E+8
Sr89	3.0 E+1	5.5 E+3	1.1 E+5	1.1 E+5	2.2 E+5
Sr90	1.4 E+0	2.6 E+2	5.1 E+3	5.0 E+3	1.0 E+4
Ag110m	1.4 E+2	5.8 E+3	4.0 E+5	8.4 E+5	1.2 E+6
Cs134	1.0 E+3	5.1 E+4	8.1 E+5	8.7 E+5	1.7 E+6
CS137	1.2 E+3	5.9 E+4	9.4 E+5	1.0 E+6	2.0 E+6

Table 7-4: Activity Release Case 2 without Consideration of Filter System

(Activity release in Bq in the time interval)

	0 – 34 h	34 -42 h	42 – 58 h	58 – 106 h	> 106 h	Total
Xe133	5.8 E+10	3.4 E+10	5.8 E+10	9.7 E+10	2.7 E+10	2.7 E+11
J131	2.7 E+9	1.6 E+9	2.8 E+9	4.6 E+9	1.3 E+9	1.3 E+10
Sr89	2.1 E+7	1.9 E+7	4.1 E+7	1.0 E+8	3.7 E+7	2.2 E+8
Sr90	1.0 E+6	8.8 E+5	2.0 E+6	4.7 E+6	1.8 E+6	1.0 E+7
Ag110m	3.6 E+7	3.2 E+7	1.6 E+8	6.7 E+8	3.4 E+8	1.2 E+9
Cs134	1.7 E+8	1.4 E+8	3.1 E+8	7.8 E+8	3.3 E+8	1.7 E+9
Cs137	2.0 E+8	1.6 E+8	3.6 E+8	9.1 E+8	3.8 E+8	2.0 E+9

Table 7-5: TEDE Results of Case 2 (Immersion and Inhalation) based on the released activity

	U.S. EAB Dose (Sv) [mrem] TEDE
Unfiltered Release – Short Term	1.361E-04 [13.6]
Unfiltered Release – Long Term	2.214E-04 [22.1]
Filtered Release – Long Term	2.858E-06 [0.30]
Total - Unfiltered	**3.575E-04 [35.6]**
Total - Filtered	1.389E-04 [13.9]

7.2.3 DBA Case 3 (Section 5.4.3)

Break of an Instrument Line and Small Leakages (< DN 10)

An instrument line break (inside diameter < 10 mm) or small breaks in other lines causes a pressure drop in the primary circuit. This is detected by the reactor protection system that causes a reactor shut-down. Leakage of primary coolant is detected in the exhaust air, whereupon the switch to the sub-atmospheric pressure system occurs. If it is impossible to isolate the leakage, the slow primary coolant depressurization to ambient pressure is initiated. The leakage inventory will be released via the secured sub-atmospheric pressure system and the stack.

This release consists of the steady-state primary coolant activity and the solid and iodine activity desorbing from the primary system surfaces during the blowdown process (compare to Case 1). The activity release from the fuel elements into the primary coolant continues after leak detection until the reactor has been shut-down and radioactive decay of short-lived noble gas and iodine isotopes is credited.

To calculate the activity releases following conservative assumptions are made:

- Efficiency of the filter unit in the secured sub-atmospheric pressure system: 0 % for noble gases, 99% for iodine, and 99.9% for solids.
- The filter efficiency selected for iodine allows for the presence of organic iodine compounds and is extremely conservative for primarily expected elemental iodine.
- The total quantity of iodine released to the environment is postulated as being in the chemical form which has the greatest radiological effects, i.e., elemental iodine.

Table 7-6: Activity Release of Case 3

Nuclide	Activity Release in Bq (short-term release)	
	Filtered	**Unfiltered**
Kr83m	1.2 E+10	1.2 E+10
Kr85ro	4.6 E+10	4.6 E+10
Kr85	3.2 E+8	3.2 E+8
Kr87	4.0 E+10	4.0 E+10
Kr88	9.7 E+10	9.7 E+10
Kr8g	5.9 E+9	5.9 E+9
Kr90	2.2 E+9	2.2 E+9
Xel31m	1.2 E+9	1.2 E+9
Xe133m	1.1 E+10	1.1 E+10
Xe133	2.3 E+11	2.3 E+11
Xe135m	4.6 E+9	4.6 E+9
Xe135	1.3 E+11	1.3 E+11
Xe137	1.0 E+10.	1.0 E+10
Xel38	2.7 E+10	2.7 E+10
Xe139	2.8 E+9	2.8 E+9
Total noble gases	**6.2 E+11**	**6.2 E+11**
I131	1.3 E+6	1.3 E+8
I132	6.8 E+6	6.8 E+8
I133	7.7 E+6	7.7 E+8
I134	8.3 E+6	8.3 E+8
I135	1.0 E+7	1.0 E+9
Total iodine	**3.4 E+7**	**3.4 E+9**
Sr90	8.5 E+0	8.5 E+3
Ag11Om	7.7 E+1	7.7 E+4
Cs134	1.0 E+3	1.0 E+6
Cs137	2.2 E+3	2.2 E+6
Total long-lived solids	**3.3 E+3**	**3.3 E+6**
Rb88	4.6 E+6	4.6 E+9
Cs138	2.5 E+6	2.5 E+9
Total noble gas decay products	**7.1 E+6**	**7.1 E+9**

Table 7-7: TEDE Results of Case 3 (Immersion and inhalation) based on Activity Release

	U.S. EAB Dose (Sv) [mrem] TEDE
Unfiltered Release	**5.782E-05 [5.80]**
Filtered Release	5.183E-05 [5.20]

Table 7-6 provides the activity releases of Case 3 and Table 7-7 provides the TEDE results.

A core heat-up will be prevented by reactor shut down. In case this is not possible and the leakage cannot be isolated, the long-term temperature development is discussed in Case 1.

In the event of a very small break, the primary system pressure drop can be so slow in certain circumstances that it is not detected by the reactor protection system or the leakage flow is compensated for by the pressure control system.

The primary coolant release, however, is immediately detected by activity detectors in compartments housing the primary coolant lines and in the exhaust air ducts. This is indicated by appropriate signals in the control room, thus giving the operators sufficient time for manual actions.

7.2.4 DBA Case 4 (Section 5.6.1)

Radiological Consequences of the SGTR with Responding Pressure Relief System

The Steam Generator Tube Rupture (SGTR) with responding pressure relief scenario assumes an accident sequence with superimposed failures of operational systems.

The maximum fuel element corrosion and pressure increase in the primary circuit takes place by the combination of operation of the circulator damper and failure of the post-accident water separator, the pressure control system and the helium purification system. This superimposed scenario, in combination with the SGTR and pressure relief is the most limiting and bounding SGTR accident.

Due to steam generator breaks or tube rupture, water ingresses into the primary system. The following water-gas reaction may cause fuel corrosion:

$$H_2O + C \leftrightarrow H_2 + CO$$

Performed tests show that there is only slight corrosion of the fuel elements and that response of the pressure relief system and hence primary coolant release only occurs in the event of a combination of operational system failures and only after several hours.

The accident is detected by the initiating criterion: Moisture in the primary circuit ≥ approx. 800 vpm (volume parts per million).

Apart from triggering reactor scram, the reactor protection system also actuates steam generator pressure relief to limit the quantity of water entering the primary circuit. After pressure equalization between primary and secondary side, the steam pressurize relief valves close by a spring operated mechanism.

Residual heat is removed by the operational component of reactor cavity cooling system and the secured cooling system. The water inleakage into the primary system is removed from the primary circuit via the third train of the helium purification system; that contains the manually activated post-accident water separator.

On response of the pressure relief system, activity is initially discharged into the reactor building. As in case 3 the HVAC system of the affected building area is switched to the safety-related sub-atmospheric pressure system, which is equipped with activated-carbon and aerosol filters.

The released activity consists of the following:

- Initial primary coolant activity
- Remobilized activity products of primary circuit surface (including remobilized SG activity)
- Activity from corrosion of the fuel elements (reaction of humidity and fuel)
- Primary coolant design activity as given in Table 5-23
- Primary system surface activity after 32 equivalent full-power years as given in Table 5-23
- Fraction of the primary System activity deposited on the steam generator tubes: 95 % for cesium and strontium, 90 % for silver, and 80 % for iodine.
- Integral fuel graphite corrosion: approx. 0.3 percent by weight
- Mean activity concentration of the corroded un-fuel containing shell for representative radionuclides:
    - I 131 8.9 E+31 Bq/g
    - Cs 137 7.0 E+2 Bq/g
    - Sr 90 5.6 E+2 Bq/g
    - Ag 110 m 1.3 E+3 Bq/g
- Remobilization of fission products deposited on the steam generator surfaces
- A maximum of 10% of the primary circuit gas carried and remobilized activity will be released before the safety valve closes.
- The unfiltered release doesn’t credit the filtration system and the sub-atmospheric pressure system.
- Efficiency of the filter unit in the secured sub-atmospheric pressure system: 0% noble gases, 99% iodine, and 99.9% solids. The filter efficiency selected for iodine allows for the presence of organic iodine compounds and is extremely conservative for elemental iodine, which is primarily to be expected.
- Iodine is released to the atmosphere in the radiological most effective form, i.e., elemental iodine.

Table 7-8: Activity Release of Case 4

Nuclide	Activity release in Bq (short-term release)	
	Filtered	**Unfiltered**
Kr 83 m	2.8 E+8	2.8 E+8
Kr 85 m	2.7 E+9	2.7 E+9
Kr 85	3.4 E+7	3.4 E+7
Kr 87	5.0 E+8	5.0 E+8
Kr 88	3.6 E+9	3.6 E+9
Xe 131 m	1.1 E+8	1.1 E+8
Xe 133 m	1.1 E+9	1.1 E+9
Xe 133	2.3 E+10	2.3 E+10
Xe 135	9.8 E+9	9.8 E+9
Total noble gases	4.1 E+10	4.1 E+10
I 131	4.1 E+6	4.1 E+8
I 132	8.4 E+5	8.4 E+7
I 133	5.2 E+6	5.2 E+8
I 134	3.4 E+5	3.4 E+7
I 135	3.4 E+6	3.4 E+8
Total iodine	1.4 E+4	1.4 E+7
Sr 90	1.9 E+4	1.9 E+7
Ag 110 m	3.0 E+3	3.0 E+6
Cs 134	1.2 E+5	1.2 E+8
Cs 137	1.7 E+6	1.7 E+9
Total long-lived solids	1.8 E+6	1.8 E+9

Table 7-9: TEDE results of Case 4 (Immersion and Inhalation) based on the activity release.

	U.S. EAB Dose (Sv) [mrem] TEDE
Unfiltered Release	**3.464E-05 [3.50]**
Filtered Release	2.245E-06 [0.23]

Table 7-8 provides the activity releases of Case 4 and Table 7-7 provides the TEDE results.

7.2.5 DBA Case 5 (Section 5.8.1)

Break or Leakage of Primary Coolant Routing Components Outside the Reactor Building

The bounding accident for primary coolant activity release outside the reactor building is the pipe break of the helium purification system inside the reactor auxiliary building. In that case, the reactor protection system isolates immediately the affected purification train from the primary circuit. Conservatively this scenario assumes that the primary circuit gets isolated after reaching 50 bar (primary circuit pressure). Primary coolant discharges, through the leakage, until isolation occurs. It is also assumed that the complete helium purification system inventory discharges without isolation. Additionally, the release of primary coolant and its activity and the activity release from the molecular sieve and the low temperature adsorber are determined. This results in a release of noble gases, tritium and C^{14}. The desorption of solid fission products mainly noble gas daughter products are neglected.

The pressure raise in the nuclear auxiliary building causes the opening of the pressure relief flaps, which routes the unfiltered gas mixture and aerosols to the stack.

Table 7-10: Activity Release of Case 5

Nuclide	Activity release in Bq (short-term release)
Kr83m	6.3 E+9
Kr85m	3.0 E+10
Kr85	1.6 E+10
Kr87	2.1 E+10
Kr88	5.5 E+10
Kr89	5.7 E+9
Kr90	2.4 E+9
Xe131m	2.2 E+10
Xe133m	4.6 E+10
Xe133	2.2 E+12
Xe135m	3.8 E+9
Xe135	1.2 E+11
Xe137	9.8 E+9
Xe138	2.0 E+10
Xe139	3.0 E+9
Total noble gases	2.5 E+12
I131	6.8 E+6
I132	9.0 E+7
I133	4.4 E+7
I134	2.3 E+8
I135	7.7 E+7
Total iodine	4.5 E+8
Sr90	4.3 E+2
Ag110m	3.8 E+3
Cs134	5.1 E+4
Cs137	1.1 E+5
Total long-lived solids	1.6 E+5
Rb88	2.6 E+9
Cs138	9.9 E+8
Total noble gas decay products	3.6 E+9
H3	5.6 E+12
C14	6.0 E+10

Table 7-11: TEDE Results of Case 5 (Immersion and Inhalation), based on Activity Release

	Dose U.S. - EAB (Sv) TEDE
Unfiltered Release	**1.975E-04 [19.75]**
Filtered Release	N/A

Table 7-10 provides the activity releases of Case 5 and Table 7-11 provides the TEDE results.

7.2.6 DBA Case 6 (Section 5.8.2)

Leakage of Vessel Containing Radioactive Contaminated Water

This case assumes a failure of one 10 m^3 evaporator concentrate vessel and a complete draining of its content in the reactor auxiliary building. Table 7-12 provides the release activity.

Table 7-12: Total Activity of the Evaporator Concentrate

Nuclide	Bq	µCi
Co-60	8.90E+10	2.41 E+06
Sr-90	1.90E+08	5.14 E-03
I-131	1.90E+07	5.14 E-04
Cs-134	4.40E+09	1.19 E-01
Cs-137	5.60E+10	1.51 E+00

In order to establish the activity entering the air (see table below) from the discharged concentrate (table above), it is assumed that the evaporated fraction of 1 % contains a weight-related activity concentration amounting to 5 % of the concentration in the concentrate. For airborne iodine the elemental form was postulated for the sake of conservatism. Table 7-13 and Table 7-14 provide the airborne release and the resulting TEDE dose.

Table 7-13: Airborne Activity Release of Case 6

Nuclide	Activity release in Bq (short-term release)
Co 60	4.4 E+07
Sr 90	9.3 E+04
I 131	9.3 E+03
Cs 134	2.2 E+06
Cs 137	2.8 E+07

Table 7-14: TEDE Result for Case 6 (Immersion and Inhalation Dose)

(based on the airborne releases)

	Dose U.S. - EAB (Sv) [mrem] TEDE
Unfiltered Release	**2.443E-06 [0.24]**
Filtered Release	N/A

7.2.7 DBA Case 7 (Section 5.9)

Seismic Effects on the Reactor Auxiliary Building

No seismic stability, integrity or functional analyses are required for the reactor auxiliary building systems.

In order to establish potential radiological impact to the environment, breaks of the activity-retaining components and direct ground-level release are postulated. The activity inventories that can be released via building leaks are only present in the components of the helium purification system, the storage tank of radioactive contaminated helium, the water extraction system retaining tank, and the evaporator concentrate vessels. The accident scenario failure of one train of the helium purification system is described in Case 5 already. The airborne TEDE result of one evaporator concentrate vessel inventory release via the stack is described in Case 1.

To cover simultaneous breaks in all important activity-retaining systems in the reactor auxiliary building the same assumptions where taken as for Case 6 (failure evaporator concentrate vessel). Liquid release in the ground has not been considered, since the seal structure of the reactor auxiliary building is designed for earthquake loads. The total volume of all vessels is 80 m^3. Table 7-15 provides the airborne activity release.

Table 7-15: Airborne Activity Release of Case 7

Nuclide	Activity release in Bq (short-term release)
Kr-83m	1.3 E+10
Kr-85m	6.0 E+10
Kr-85	3.2 E+10
Kr-87	4.2 E+10
Kr-88	1.1 E+11
Kr-89	1.1 E+10
Kr-90	4.7 E+09
Xe-131m	4.4 E+10
Xe-133m	9.2 E+10
Xe-133	4.3 E+12
Xe-135m	7.6 E+09
Xe-135	2.5 E+11
Xe-137	2.0 E+10
Xe-138	4.0 E+10
Xe-139	6.1 E+09
I-131	1.4 E+07
I-132	1.8 E+08
I-133	8.7 E+07
I-134	4.6 E+08
Co-60	8.9 E+07
Sr-90	1.9 E+05
Ag-110m	7.5 E+03
Cs-134	4.6 E+06
Rb-88	5.1 E+09
Cs-138	2.0 E+09
H-3	1.9 E+13
C-14	1.2 E+11

The unfiltered immersion and inhalation TEDE, as listed in the Table 7-16 , covers simultaneous breaks in all important activity-retaining systems in the reactor auxiliary building and airborne release. Elemental iodine was assumed for conservatism.

Table 7-16: TEDE result of Case 7 (Immersion and Inhalation Dose)

(based on airborne release)

	Dose U.S. - EAB (Sv) TEDE
Unfiltered Release	**3.961E-04 [39.61]**
Filtered Release	N/A

7.3 Assessment of Results

The radiological assessment of the HTR-Module design basis accidents (DBA) has been performed by referencing the radioactive source term documented in the HTR-Module safety analyses report. The German HTR-Module SAR dose assessments were based on ingestion pathway whereas for the U.S. DBA accidents, doses are based on air immersion and inhalation. The radiological assessment used the tabulated nuclide released and calculated the two-hour immersion and inhalation exclusion area boundary (EAB) doses for the following event cases:

Case 1 - Large Break of Connection Pipe between pressure vessel and primary circuit isolation valve

Case 2 - Pressure Relief with Subsequent Core Heat-up

Case 3 - Break of an Instrument Line and Small Leakages (< DN10)

Case 4 - Radiological Consequences of the SGTR with Responding Pressure Relief System

Case 5 - Break or Leakage of Primary Coolant Routing Components Outside the Reactor Building

Case 6 - Leakage of Vessel Containing Radioactive Contaminated Water, and

Case 7 - Seismic Effects on the Reactor Auxiliary Building

The release of radioactivity was conservatively assumed to be instantaneous for all cases. All cases meet the most stringent dose criteria for an EAB of 400m. In fact, all cases meet the 10CFR20 annual dose limit of 100 mrem. A summary of the dose results is provided in Table 7-17.

Table 7-17: Summary of HTR-Module Accident Doses

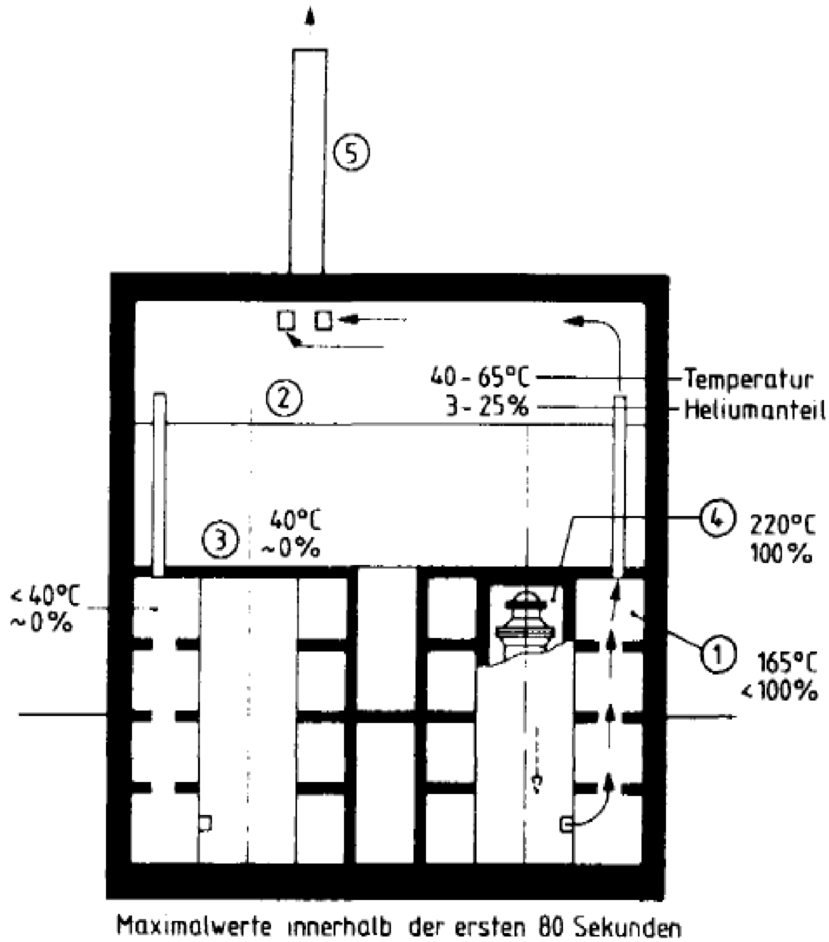
Design Basis Accidents	Case #	Dose (TEDE) Sv EAB (2 hour) [1]	Dose (TEDE) Sv NRC Limit [2]
Break of a Large Connecting Pipe (DN 65) - LBLOCA short-term unfiltered release	1	unfiltered: 1.361E-04 (13.61 mrem)	2.5E-1 6.3E-2 2.5E-2
Break of a Large Connecting Line (DN 65) - LBLOCA long-term unfiltered release with core heat up	2	filtered: 2.858E-06 unfiltered: 2.214E-04 (22.14 mrem)	
Instrument Line Break Pressure release phase (DN <10)	3	filtered: 5.183E-05 unfiltered: 5.782E-05 (5.782 mrem)	
Steam Generator Tube Rupture with response of the Pressure Relief System	4	filtered: 2.245E-06 unfiltered: 3.464E-05 (3.464 mrem)	
Helium Purification System Pipe Break release via stack	5	unfiltered: 1.975E-04 (19.75 mrem)	
Non-Design Basis Accidents			
Leakage of Vessel Containing Radioactive Contaminated Water	6	Unfiltered: 2.443E-06 (0.2443 mrem)	10CFR20
Seismic Effects on the Reactor Auxiliary Building	7	Unfiltered: 3.961E-04 (39.61 mrem)	

Notes:

(1) The worst two hour window is used with an atmospheric dispersion factor (X/Q) value of 3.35E-03 s/m^3 corresponding to an EAB distance of 0.249 miles (400 m).

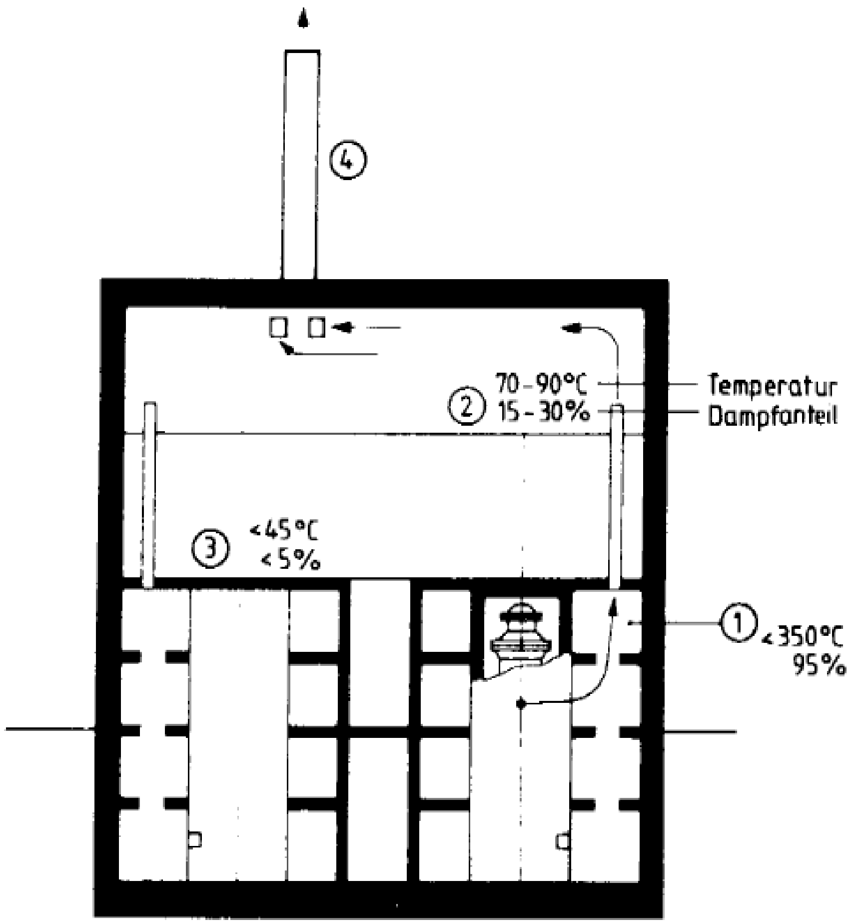
(2) The 0.25 Sv criterion is used for evaluating design basis accidents of exceedingly low occurrence probability and low risk of public exposure to radiation. The criterion for events of moderate frequency is 25% of the 0.25 Sv, or 0.063 Sv. The criterion for events of higher probability of occurrence, the acceptance criterion is10% of the full limit, or 0.025 Sv

Figure 7-1: Break of a Large Connecting Pipe (DIN65) – Unfiltered Release



1. Räume der Moduleinheit
2. Reaktorflur
3. Dampferzeugerflur
4. Primärzelle
5. Fortluftkamin

Figure 7-2: Radiological Consequences of a SGTR with Responding Pressure Relief System



1. Räume der Moduleinheit
2. Reaktorflur
3. Dampferzeugerflur
4. Fortluftkamin

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