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Advanced Reactor for Cogeneration of Heat & Electricity-R&D

Deliverable D-44.11: Review of SG operating / transient conditions from previous HTR reactor designs

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Summary

This document contains the available data's from former built HTR SG (High Temperature Reactor Steam Generator). The information's have been made available to the project by AREVA and ALSTOM.

Inputs and information was taken from actual / former built and operated SG.

Material and design for this HTR SG will be kept under review within the project, and maybe changed based under results from later investigations.

Approval

Rev.	Short description	First author	WP Leader	SP leader	Coordinator
01	First issue	D. Ponca (ALSTOM)	D. Ponca (ALSTOM)	D. Buckthorpe (AMEC)	S. Groot (NRG)

Distribution list

Name	Organisation	Comments
All technical participants	ARCHER	Through internet workspace
WP44 Participants	ARCHER	Through internet workspace

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1 Introduction

The HTR Steam Generator consists of a primary side with the Reactor Cooling Gas (normaly Helium) and the secundary system side (Water Steam).
The Steam Generator acts as a forced circulated Steam Generator.
Although a lot of information about the former built SG are available and has to be verified during followed investigation of this project.

2 Summary of Contents

2.1 Contribution from AREVA

Based on AREVA past design studies, the AREVA contribution contains a brief description of the HTR Module primary and secondary circuits. The contribution also summarizes the conduct of operations and the a description of the transients which have been used to design the SG component.
The different operating conditions accounted for are :
- Start-up
- Shut-down
- Power Operation
- Reactor scram
- Residual heat removal

For these cases, information on relative flow rates, temperatures of the primary gas and secondary feed water/steam temperature and the associated pressure evolutions and variations are plotted as function of time.

In the second part of the AREVA contribution the main heat transfer malfunction events have been described. For each case, a description of the event is discussed and an associated assessment of the leading consequences has been provided.

The AREVA contribution has been included in this report in Annex 2..

2.2 Contribution from ALSTOM

2.2.1 Introduction

The focus of this review has been to search for available information about the already built SG, which are in operation since years.
This present report concentrates on the available data about HTR SG, and particularly seeks to consider the mechanical and thermal design of a SG. As such it forms part of the Deliverable D44.11 of the ARCHER Project "Review of SG operating / transient conditions from previous HTR reactor designs"
The Steam Generator is designed according to a forced circulated system. The heating surface consists of so called helical coiled heat exchanger tubes. The tubes are arranged in concentric assembled Cylinders. The primary Gas flows from the bottom part of the SG in parallel flow through the low pressure part and afterwards in a cross flow through the high-pressure part. The evaporation takes place on the secondary system part of the SG.

2.2.2 Thermal data's

Operating / performance data's of one SG are:

Thermal Power	128 MW
Reactor Cooling Gas	Helium
Gas Temp. Inlet	750 °C
Gas Temp. Outlet	250 °C
Gas Pressure (Bef. Steam Gen.)	38.5 bara
Pressure loss (Gas)	0.4 bar
Massflow Gas	49.25 kg/s
Live Steam	550 °C, 187 bara

Reheater Inlet	365 °C, 55 bara
Reheater Outlet	535 °C, 49 bara
Feedwater	152 t/h

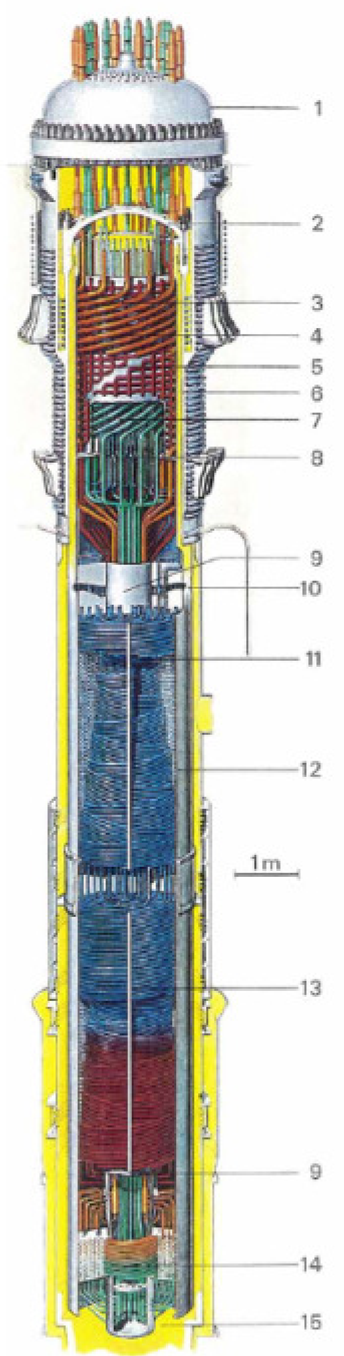
2.2.3 Key Requirements

Key product specific. items:

Pressure Vessel Code:	assuming CODAP 2000 or similar
Material Choices:	assuming free, i.e. result of concept studies
Lifetime Steam Generator:	assuming ~ 200’000 hours (EOH) except the “hot” parts ~ 100’000 hours
Plant Operation Type:	assuming Baseload & gradients < 3-4 K / min

Arrangement and max. Dimensions assuming free, i.e. result of concept study

2.2.4 Previous Concept

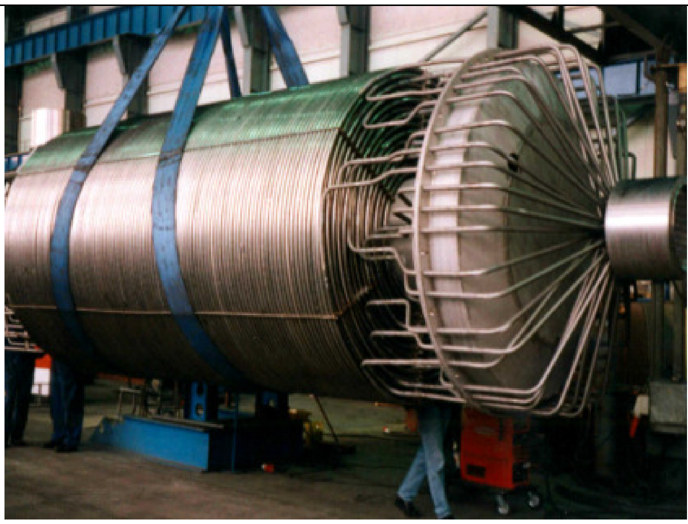
<table>
<tr>
<td>
1. Main pressure shell
2. Inner shell
3. Intermediate superheater steam outlet (535 °C, 49 bar)
4. Expansion zone
5. High pressure steam (550 °C, 186 bar)
6. Support
7. Intermediate superheater steam inlet (365 °C)
8. Feedwater inlet (180 °C)
9. Central pipe
10. Helium outlet (250 °C)
11. HP I Bundle
12. Steam generator shell
13. HP II Bundle
14. Intermediate superheater bundle
15. Helium Inlet (750 °C)
</td>
<td>


Picture 1 SG Arrangement
</td>
</tr>
</table>

2.2.5 Helix Design Principle

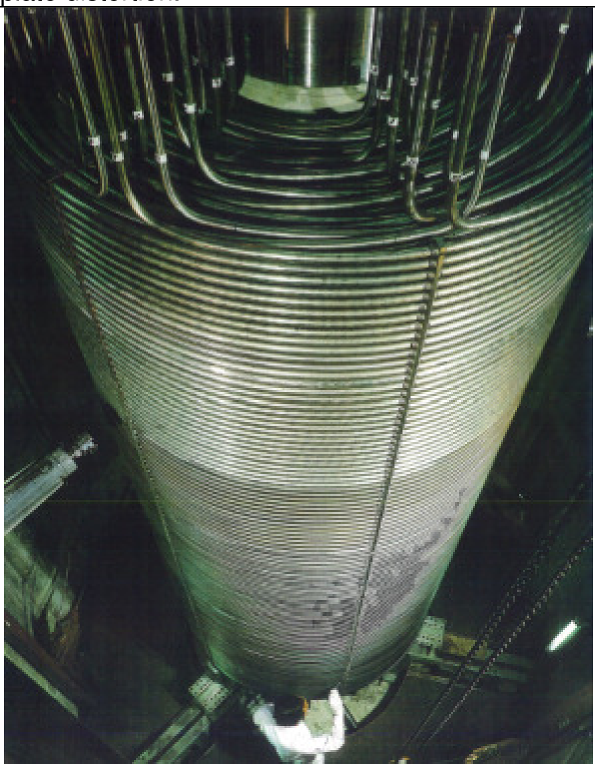
As a typical example an OTC (Once Through Cooler) from a Combine Cycle power plant can be used

LP OTC bundle
Total Length: 6300 mm
Diameter: 2050 mm
Hex Surface : 675 m^2
Helix design is field proven, robust, compact and very flexible in operation

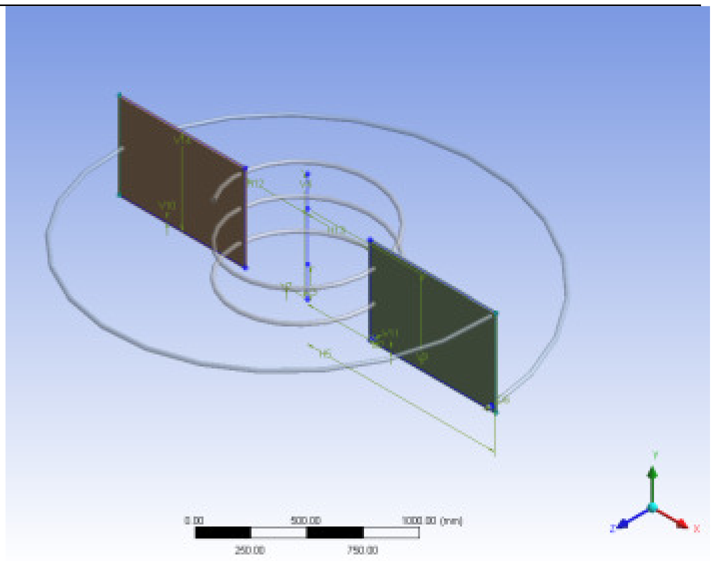


Picture 2 LP OTC tube bundle

Each Helix Cylinder is arranged in a clockwise ore in counter clockwise orientation in order to avoid support plate distortion.



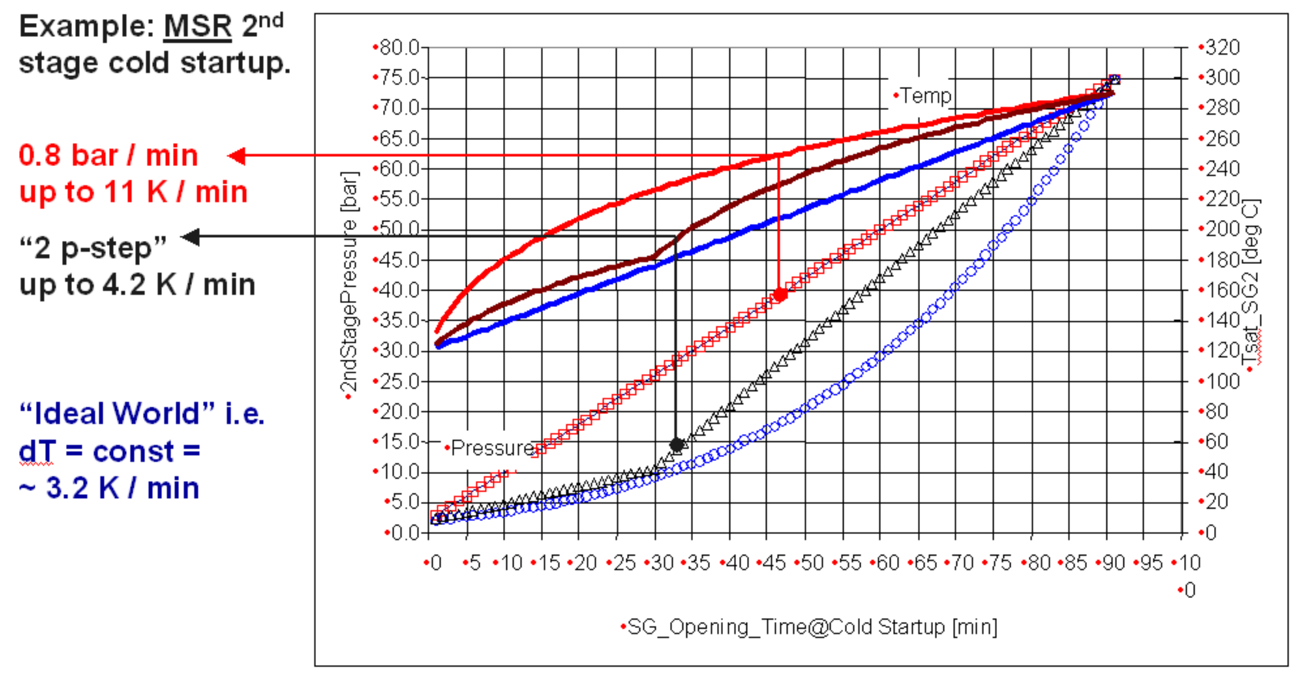
Picture 3 LP OTC tube bundle arrangement



Picture 4 LP OTC Helix coil arrangement

2.2.6 Gradients

In order to minimize and/or “Distribute” Startup Temperature Gradients a typical example from a MSR 2nd stage cold startup is taken.

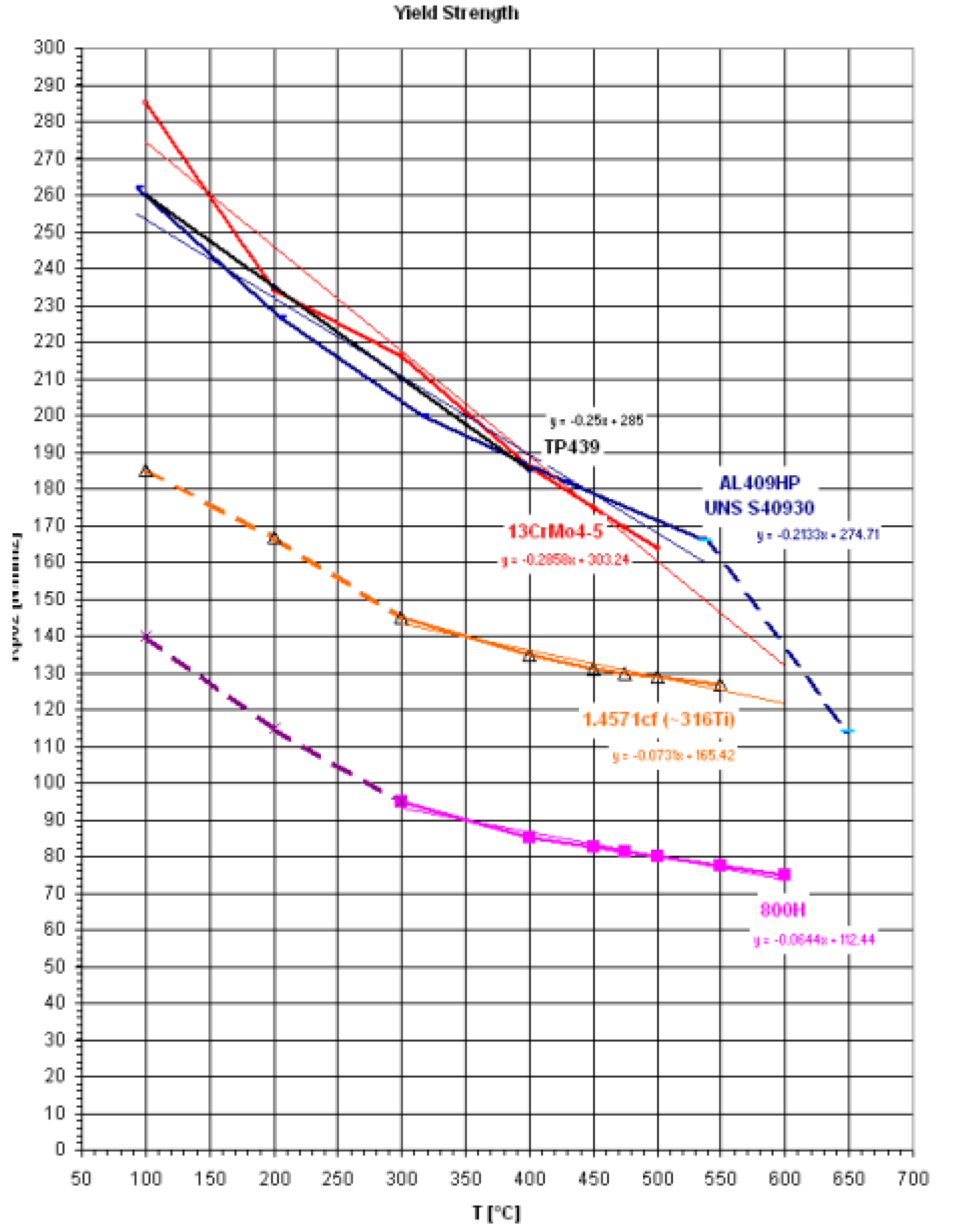


Picture 5 typical gradients

2.2.7 Material selection

Tube-side mechanical design parameters still tentative and therefore material selection on the tube side still under investigation.
The remaining parts (plates, forgings, etc) have to be selected by the required pressure vessel code. But the code itself is not sufficient.
Material selection should be the best compromise between:

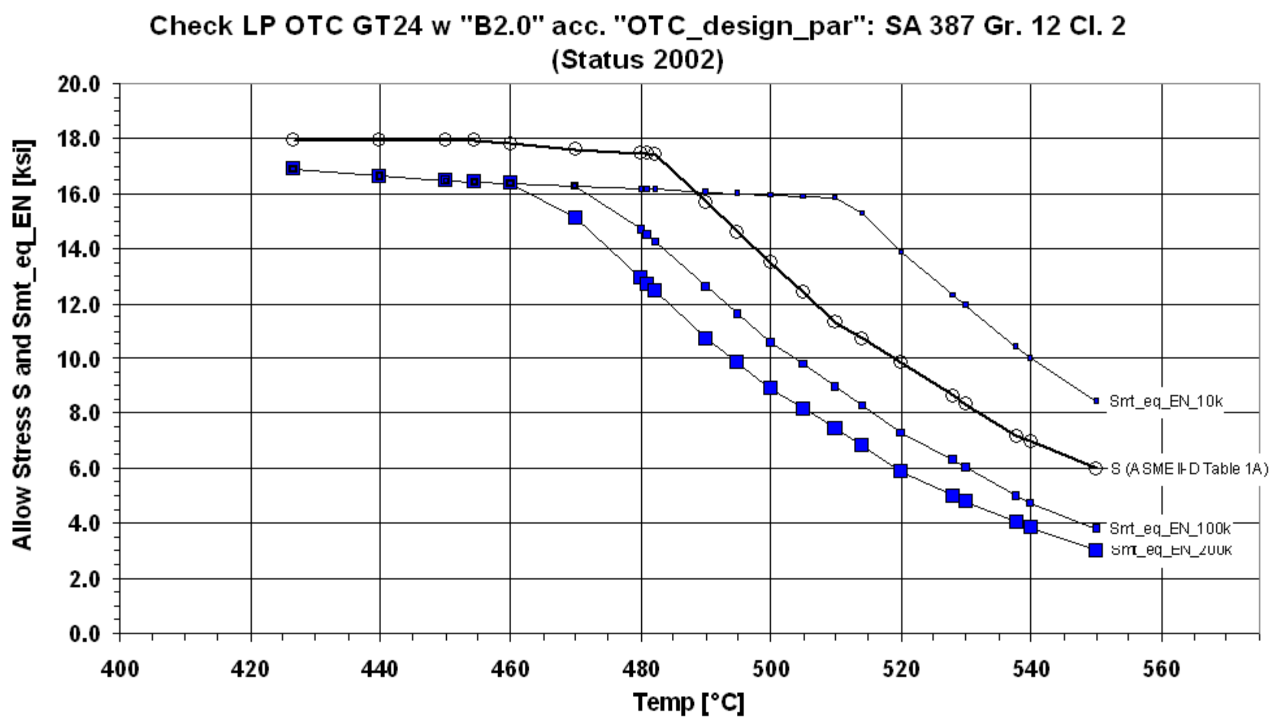
- Design parameters
- Material properties
- Code acceptance
- Manufacturing / Welding
- Lifetime
- Maintenance / Repair
- References / ERS
- Customer Specification
- First cost



Picture 6 Material properties

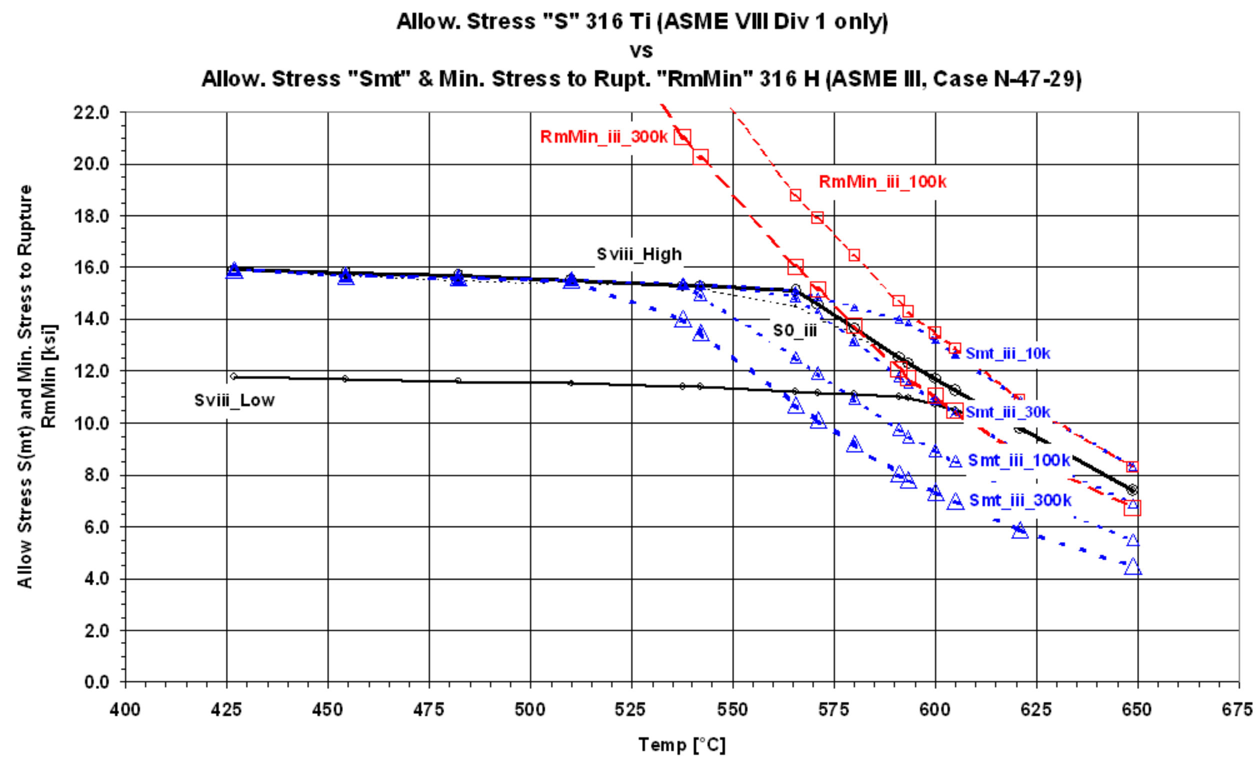
2.2.8 Lifetime

Typical values about allowable stress values related to an assigned temperature are taken for Grade 12 Material.



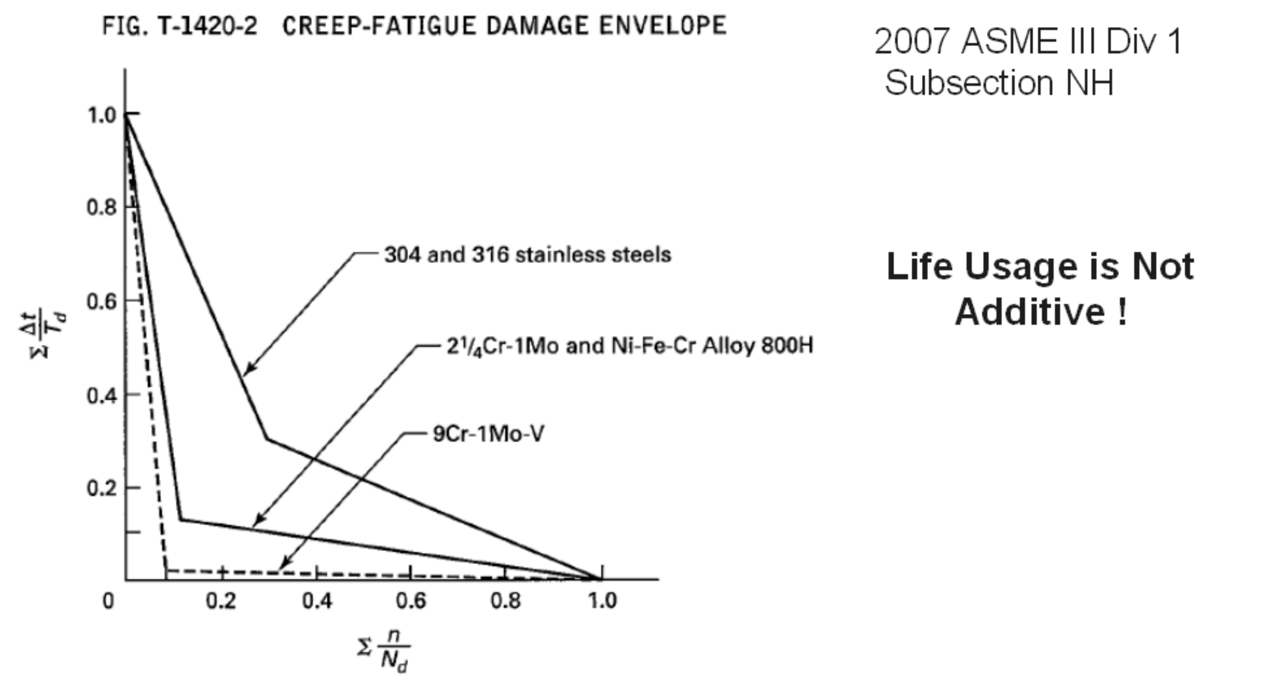
Picture 7 Material properties Grade 12

Typical creep values related to an assigned temperature are taken for Grade 316Ti Material.



Picture 8 Material properties Grade 316Ti

Based on ASME III Div. 1 Subsection NH the following interaction between creep and fatigue is given



Picture 9 Creep fatigue interaction acc ASME

3 Conclusion

The main conclusions arising from these investigations are as follows:
Design rules are available for SG based on former built HTR SG and from existing Combined Cycle power Plants. Design and materials for a SG are further defined by applicable Codes and Standards.
The temperature limit for the material for the SG is considered to be 750℃.
The remaining design parameters such as pressure, massflow, delta P are to be confirmed or defined with the ongoing process analysis.

The contribution from AREVA is based on the work performed on the Module (see Annex 2) and contains information on the relative flow rates, the temperatures of the primary gas and secondary feed water/steam and associated pressure variations. Information is also provided on the main heat transfer malfunction events with an assessment of consequences.

4 Recommendations

Concept design of a new SG based on the available data's is only partially possible. However there is sufficient information from actual and older built SG available in order to propose a new SG for latest HTR.
It is therefore recommended that the available SG operating design data provided by AREVA shall be used for the analysis and assessment SG design.
This situation will be kept under review within the project and may be changed should results from later reviews lead in some changes on the specific planned conceptual design.

5 Annexes

Annex 1 – Document approval by beneficiaries' internal QA

Annex 2 - AREVA contribution on operating/ transient conditions of the HTR Steam Generator.

5.1 Annex 1: Document approval by beneficiaries' internal QA

Fill involved beneficiaries as appropriate (mandatory for Milestones and Deliverables, but optional for other document type)

#	Name of beneficiary	Approved by	Function	Date
1				
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5.2 AREVA contribution on operating/ transient conditions of the HTR Steam Generator

AREVA

	Report	Report No. (Department/Year/Language/Sequential No.) PTE-G/2011/en/0001	Rev. A
Subject/Title **Operating / Transient Conditions of HTR MODULE Steam Generator**		Place Erlangen	Creation date 2011-11-11

Prepared by	Dieter Vanvor	Org.-Unit PESS-G	Signature
Reviewed by	Dr. Brinkmann	PTE-G	
Released by responsible dept.	Czech	Date PTE-G	
Released for external distribution	Bogusch	Date PTH-G	

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Summary — Report pages: 44 Appendices: -- Pages total: 44

This report contains a brief description of the HTR-MODULE, primary and secondary side. Also it summarizes the conduct of operations and the transients following main heat transfer malfunctions.

Distribution:

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List of revisions

Rev.	Date	Chapter/page	Scope of the revision
A	Corresponds to the last release date (see cover page)	-	First issue

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1. Introduction

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 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 bases for the subsequent PBR modular reactor designs and is the reference design as e.g. PBMR (South Africa) or HTR 10 and HTR PM (China).

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 1-1.

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 1-1 and Table 1-2. 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.

Figure 1-1: HTR-MODULE Plant System Schematic Representation

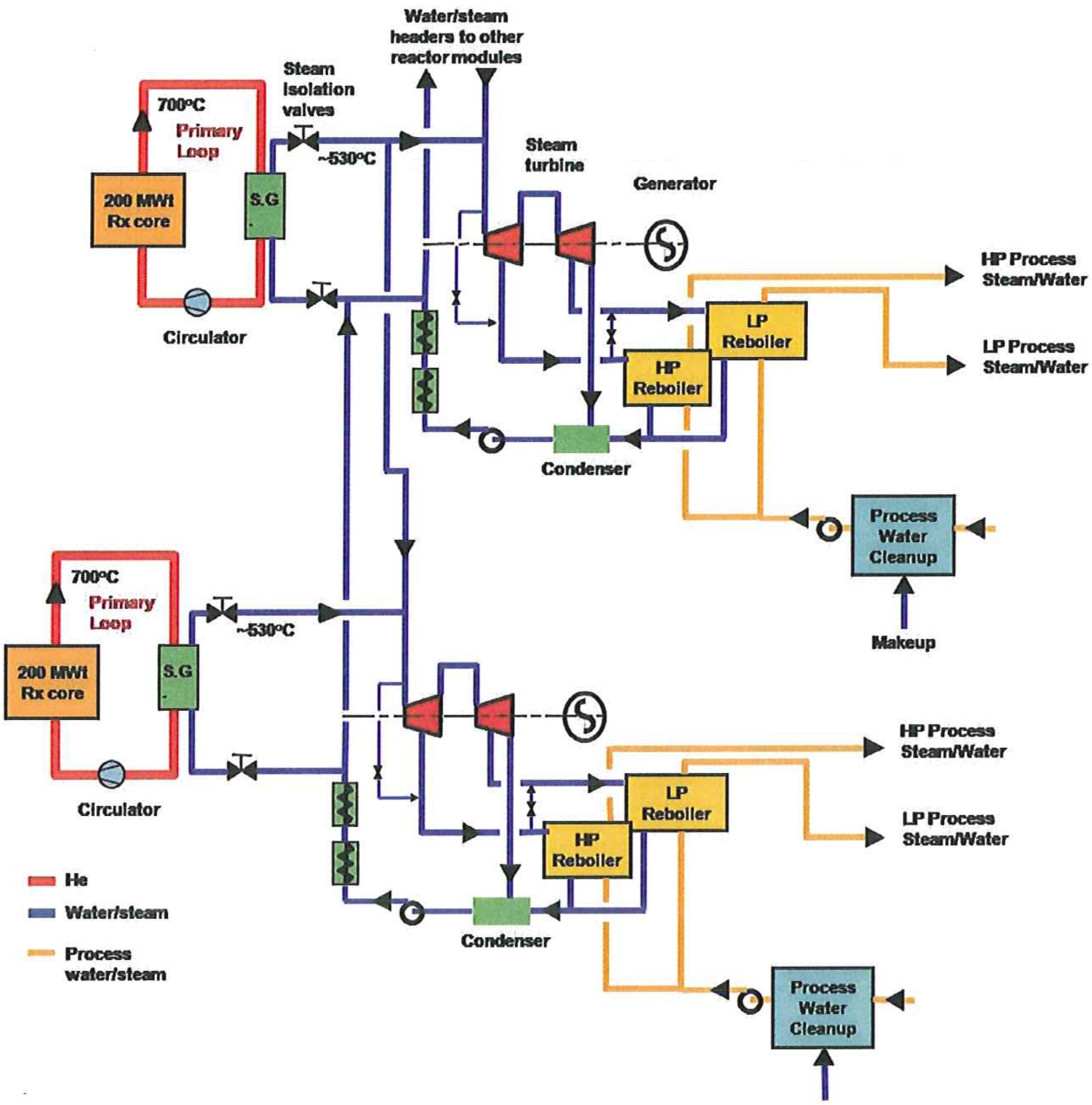


Table 1-1: General Plant Characteristics

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
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 1-2: 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

2. Description

2.1 Primary System

Each reactor unit, depicted in Figure 2-1 and Figure 2-2, 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 2-2 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.

2.2 Primary Cicurit 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 2-1 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.

2.3 Water Steam Cycle

The steam/power conversion system is designed for maximum operational reliability, availability and cost effectiveness with the simplest possible and manageable configuration.

To satisfy the availability requirements with respect to power and process steam generation, the steam/power conversion system consists of two identical, suitably interconnected half-duty units. This permits uninterrupted continuation of operation, possibly at reduced output, during overhauls or failure of plant equipment.

2.3.1 Main Steam System

One main steam line runs from each steam generator in the reactor building to the turbine building.

Each steam generator is protected by a safety valve and can be isolated by the main steam valves on actuation by the reactor protection system. A warm-up control valve is provided in parallel to the main steam valves. These valves are located in the reactor building.

In the turbine building, one supply line branches off each of the main steam lines to the separate sets of process steam pressure reducing stations assigned to each turbine generator.

The main steam lines in the steam turbine building and hence the two turbine generators are interconnected by means of connecting lines. The configuration chosen for normal operation is such that each turbine generator is aligned with one steam generator.

For reasons of availability, it is possible to connect either turbine generator set with either steam generator via the connecting lines. During malfunctions or regular overhauls of a turbine generator set or components of the water/steam cycle, it is also possible to maintain a symmetrical steam supply from both steam generators via the still available connecting lines.

Pressure, temperature and volumes are equalized between all trains and hence identical inlet pressures and temperatures are achieved ahead of the combined stop and control valves.

Steam is condensed in the associated condenser after expanding in the single-flow low-pressure section provided in each turbine.

2.3.2 Feedwater System

Each of the two steam generators is supplied from separate, dedicated feed water trains. Each train is provided with two electrically driven feed water pumps.

During normal operation the pumps supply feed water from the two feed water tanks to the steam generators via one control valve bank in each train. Both control valve banks are located in the reactor building annex and consist of one full-load and one part-load control valve.

On reactor scram, the feed water supply is interrupted by separate reactor protection system-actuated valves for each steam generator. These valves are located in the reactor building.

2.3.3 Steam Generator Isolation System

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

Whenever a reactor is tripped, the associated steam generator is isolated by the reactor protection system, which 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 startup and shutdown circuit whenever the reactor is tripped during startup and shutdown operation.

2.3.4 Steam Generator Relief (Dump) System

In the event of a steam generator tube break, the water or steam entering the reactor coolant system is limited by means of rapid steam generator relief. On completion of relief, primary system depressurization via the steam generator leak is prevented by closing the relief valves.

After the accident, the steam generator, together with the primary system to which it is connected through the break, is emptied completely into the helium purification system. Any remaining water vapor is condensed.

Steam generator relief is performed through the feedwater piping. No requirements are placed on the steam generator relief system regarding overpressure protection of the steam generator secondary side. This task is performed by a separate safety valve connected to the steam side.

The relief system of each steam generator consists of two parallel pressure relief lines, each of which is provided with two relief valves in series. The parallel and series connection of the relief valves ensures redundancy for reliable opening and closing. Both relief lines discharge to a flash tank. The depressurized feed water is collected in a relief tank connected to the flash tank.

The relief valves are actuated by the reactor protection system. The valves close once pressure equalizes between the secondary and primary side. The steam generator relief system is designed in such a way that 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.

2.3.5 Startup and Shutdown Circuit

The startup and shutdown circuit is like that of wet start once-through forced flow boiler. The system is purely operational and is not safety-related. Figure 2-3 presents a schematic of the startup and shutdown circuit.

The startup and shutdown circuit is used for operational startup and shutdown, for operational residual heat removal. The startup and shutdown system is kept at operating temperature as far as the valves ahead of the flash tank so as to be ready for operation on demand.

The startup and shutdown circuit has a heat sink in the form of a condenser/cooler presided by a flash tank.

In the various water-only phases of the startup and shutdown processes, the system discharges via a throttle valve into the flash tank and from there into the condenser/cooler. The water steam mixture and finally the steam at the end of startup operation or at the start of shutdown passes through a parallel steam conditioning valve provided with a spray attemperator into the flash tank and then into the condenser/cooler.

According to operating mode the condenser/cooler has the task of cooling or condensing. In water-only operation this heat sink can perform residual heat removal and reactor cool down.
The residual heat can be removed continuously in the recirculation mode.

The condenser/cooler is cooled by a closed cooling water system and a service water system.

The electric drives in the startup and shutdown circuit and in the associated closed cooling water and service water circuit are powered by the normal auxiliary power supply.

The isolation valves required for steam generator isolation – on reactor scram during startup and shutdown operation - are actuated by the reactor protection system.

The condenser/cooler block is designed to condense approximately 20% of the nominal steam output during startup and shutdown

Feedwater is sprayed into the superheated steam to reduce its pressure to approximately 7 bars.

2.3.6 Main Condensate System and Low-Pressure Feedwater Heating System

Each of the two turbines has a separate condensate system. The systems for both turbines are replicated so that the layout of only one condensate system needs to be described herein.

The main condensate passes from the condenser hotwell to the main condensate pump system.

The low pressure condensate pump transfers the condensate through two parallel runoff control valves, the single train low-pressure feedwater heating system and a header into the feedwater deaerator tank.

The level in the hotwell is maintained by means of the two parallel run-off control valves. Two in-series low-pressure feedwater heaters are supplied with separate steam extractions from the LP turbine; a bypass is provided.

The main condensate enters the feedwater deaerator tank via deaerating sprays. The feedwater is heated in the feedwater tanks by direct contact with steam from two extraction stages in the HP turbine.

On load shedding to auxiliary power level concurrent with 100% process steam generation, pegging steam is

Steam is fed into the feedwater deaerator tank under all conditions below water level and over the entire length of the tank (sparger pipes). The tank operates at approximately 7 bars. The feedwater is kept below the boiling point at all times. On entering, the main condensate is atomized by the automatic sprays which assure effective deaeration by the steam rising from the spargers. Vents in the area of the sprays enable the removal of noncondensable gases.

2.3.7 Process Steam System

The process steam system design described in this section differs from the reference HTR-MODULE configuration and was later on selected as a representative configuration for advanced design (see Figure 1-1).

Extraction steam from the HP turbine is used to generate HP and LP process steam in two reboilers. Extraction steam at 45.5 bars is fed to a process steam reboiler to generate HP process steam. The process steam is delivered to a HP process steam header.

Extraction steam at 34 bars is used to generate LP process steam in a separate reboiler.

Process feed water is supplied from the process system steam feed water system. Two feed water pumps supply the feed water from the process steam condensate system to the reboilers. Condensate from the reboilers is directed to the LP turbine condenser.

Figure 2-1: HTR-MODULE Primary System

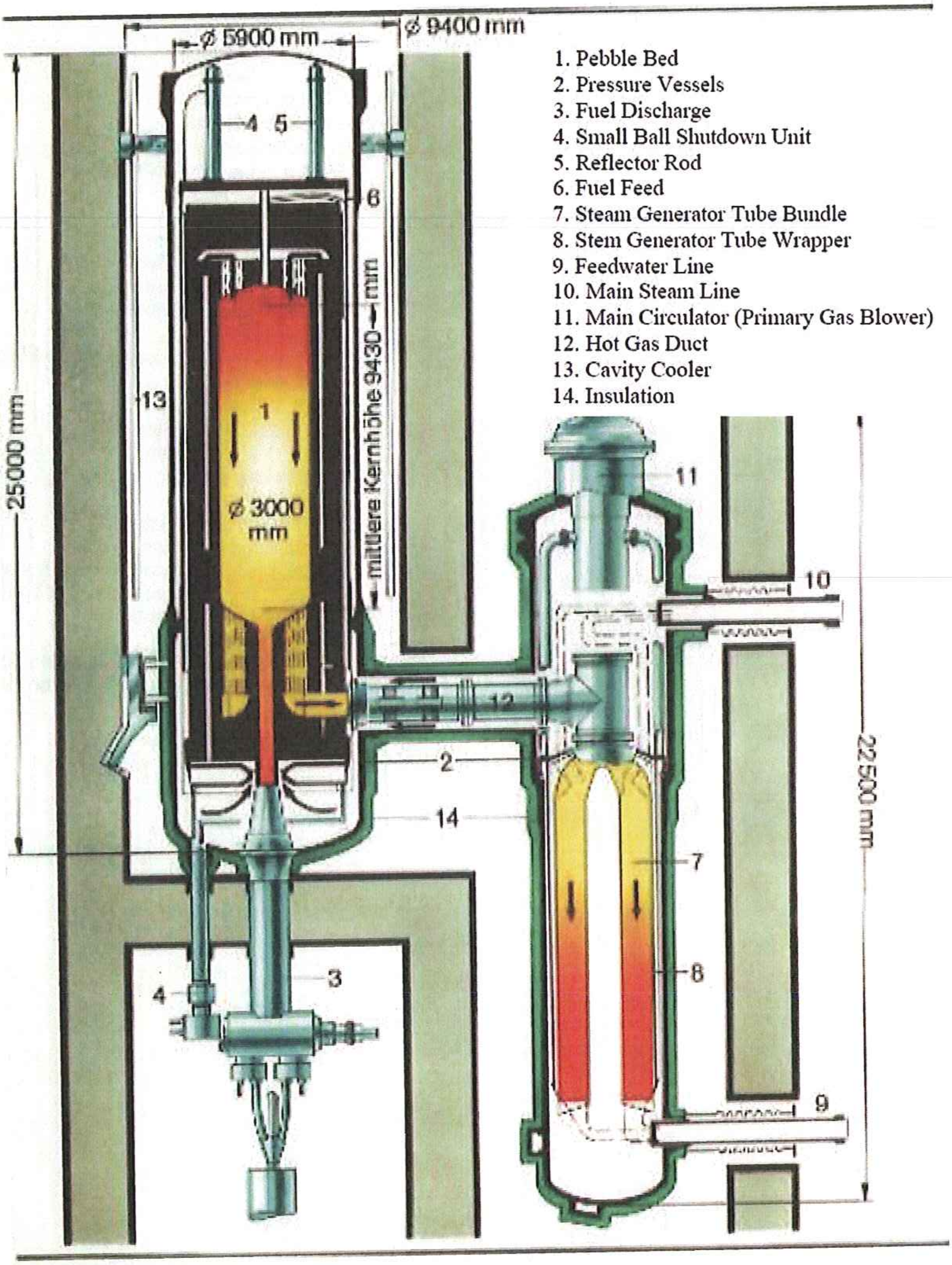
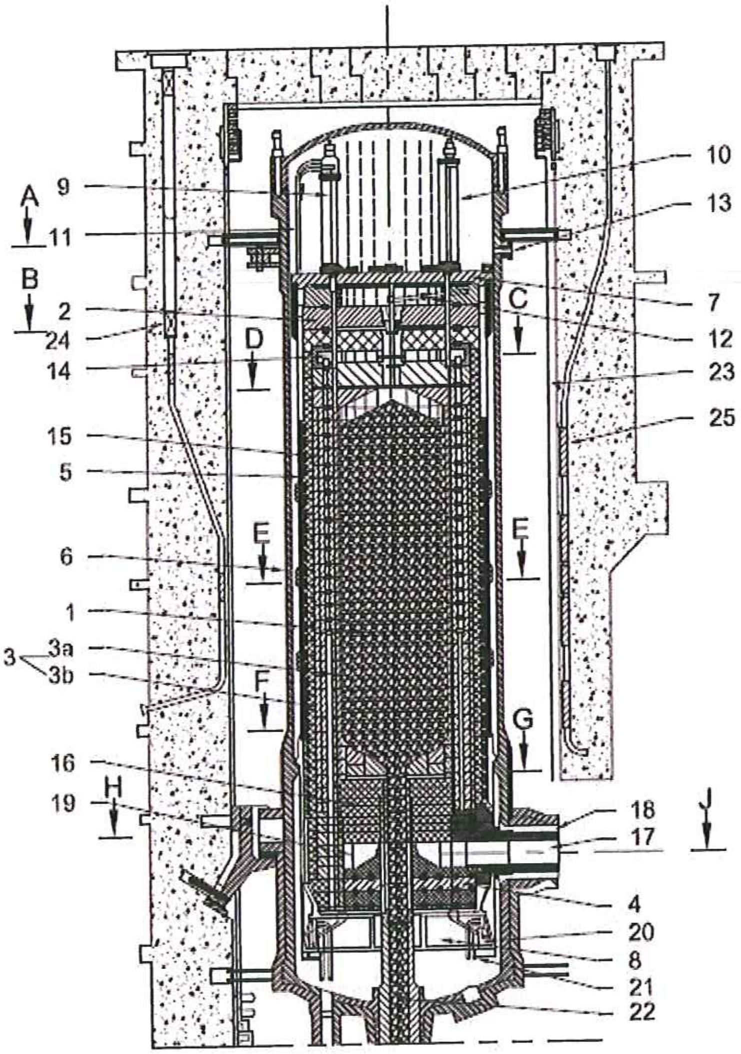
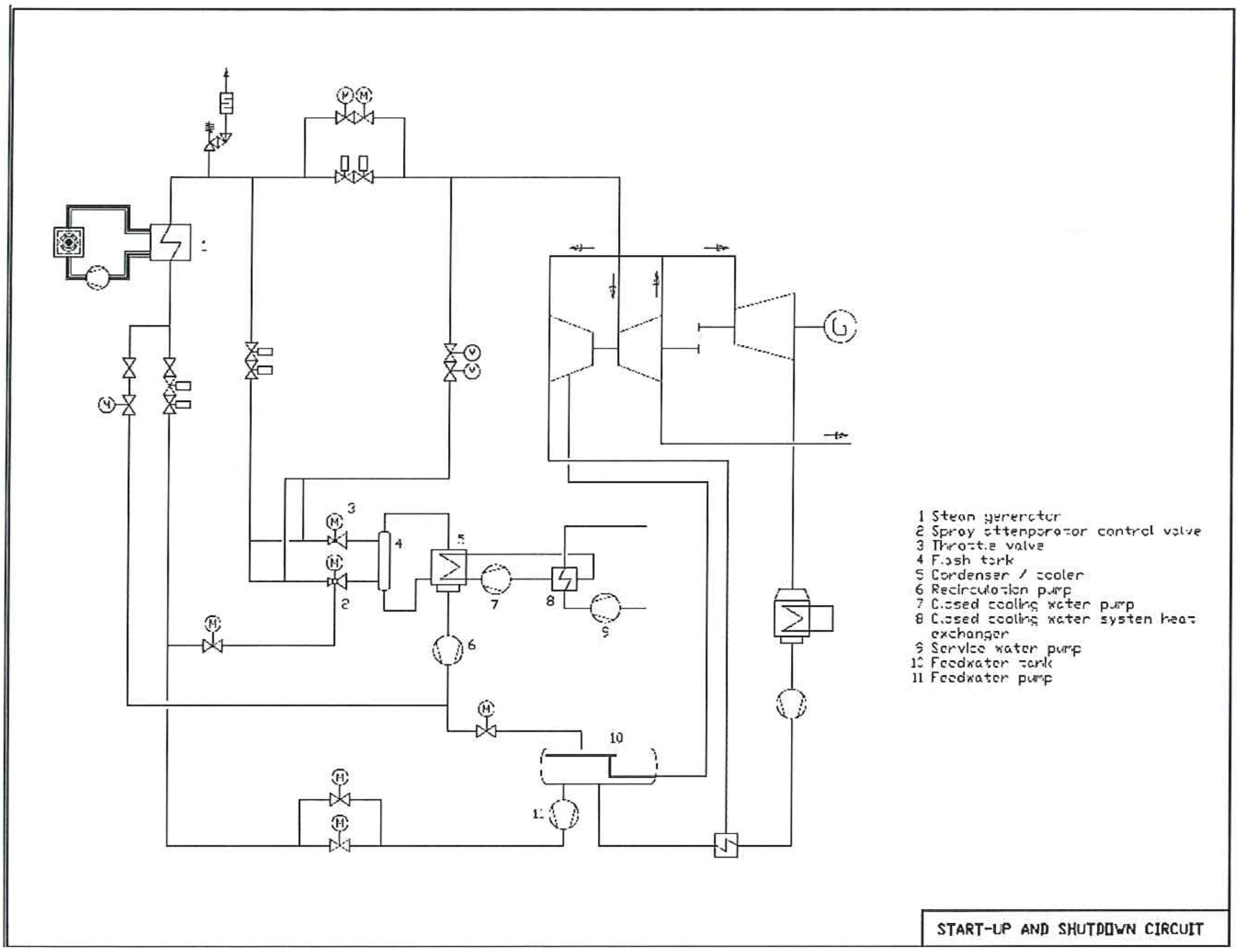


Figure 2-2: HTR-MODULE Longitudinal Cross Section



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)

Figure 2-3: Startup and Shutdown Circuit Schematic Diagram



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

3.1 Start-Up

Cold Start-Up (Figure 3-1)

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%.

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 (Figure 3-2)

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.

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 3-2, 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).

3.2 Shut-down (Figure 3-3)

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 bars.

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.

Cool down 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 Power Operation (Figure 3-4)

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.

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.

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 to) 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.

3.4 Reactor Scram (Figure 3-5 to 3-7)

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 shutdown 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 3-5 for an initial power level equivalent to 105% nominal power.

Mass flow patterns in the steam generator are governed by the simultaneous closure of the feedwater inlet valve and tripping of the primary gas circulator (see Figure 3-6). 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.

The primary coolant inlet temperature after reactor scram remains effectively constant, while the primary coolant outlet temperature decreases and adjusts to the water temperature in the cold area (see Figure 3-7). 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 4 (e.g., loss of auxiliary power supply). 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.

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.

3.5 Residual Heat Removal (Figure 3-8 to 3-10)

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 cool down 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.

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.

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. Figures 3-8 to 3-10 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 3.4.
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.

Figure 3-1: Cold Start-Up

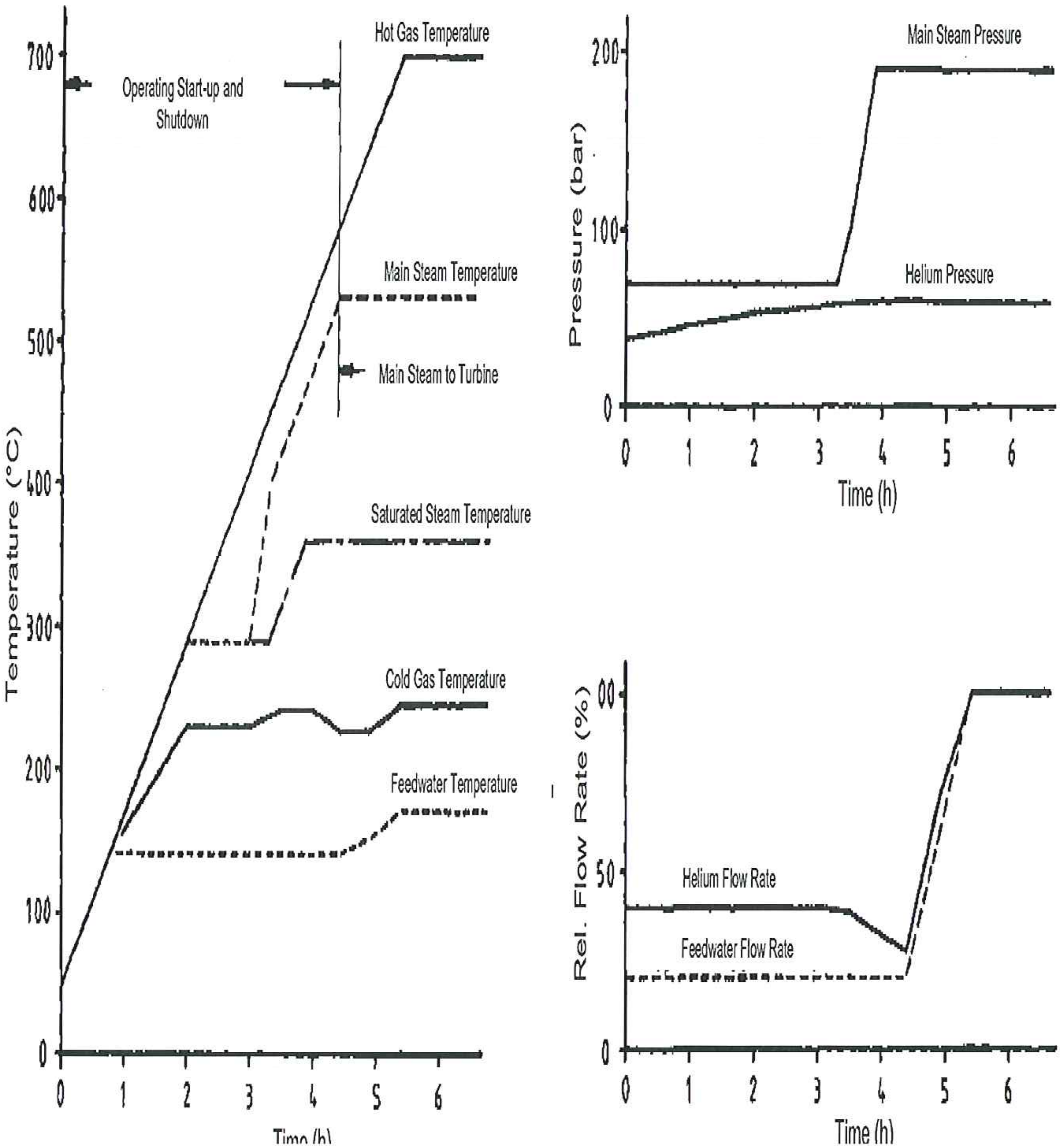


Figure 3-2: Hot Start-Up

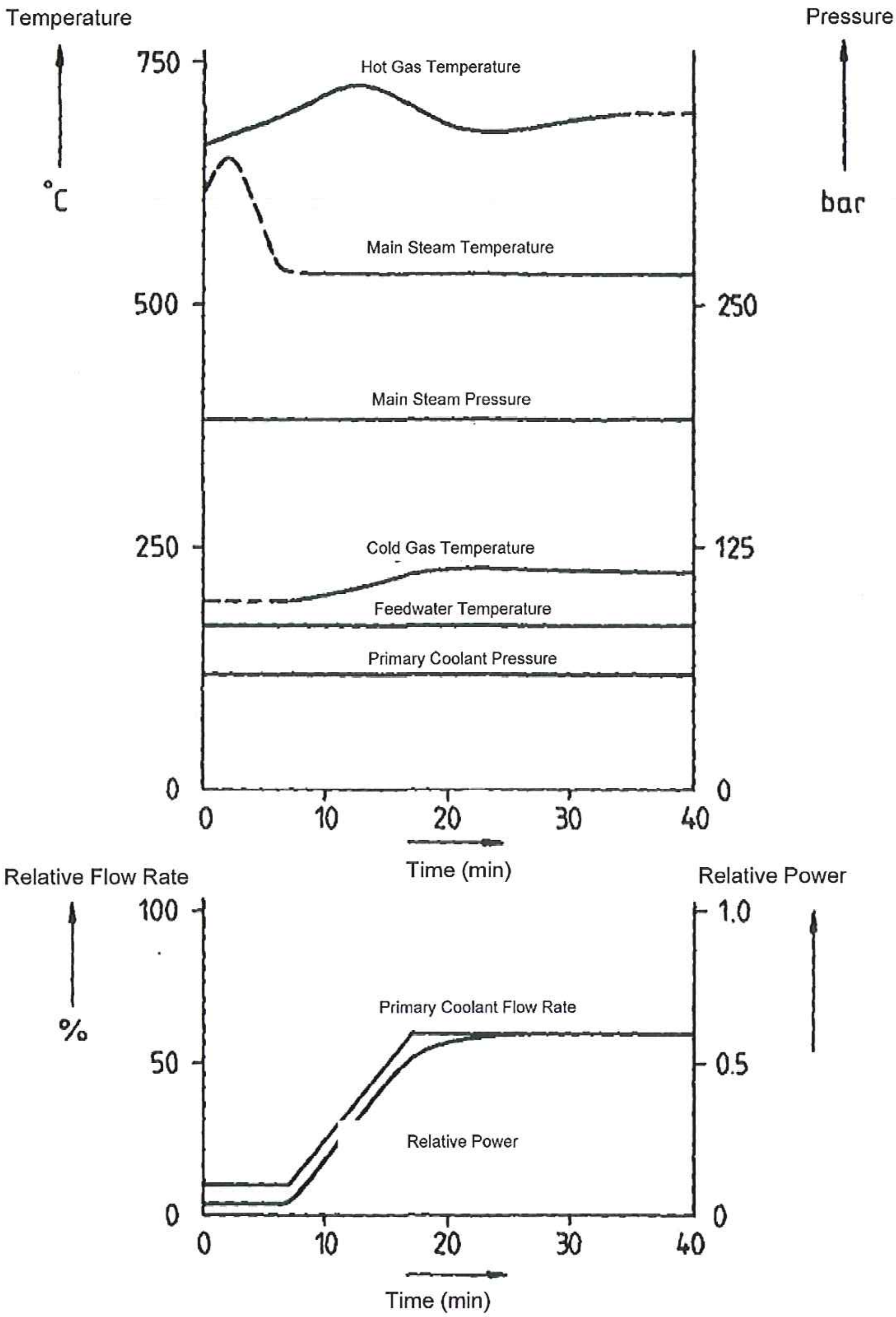


Figure 3-3: Shut Down

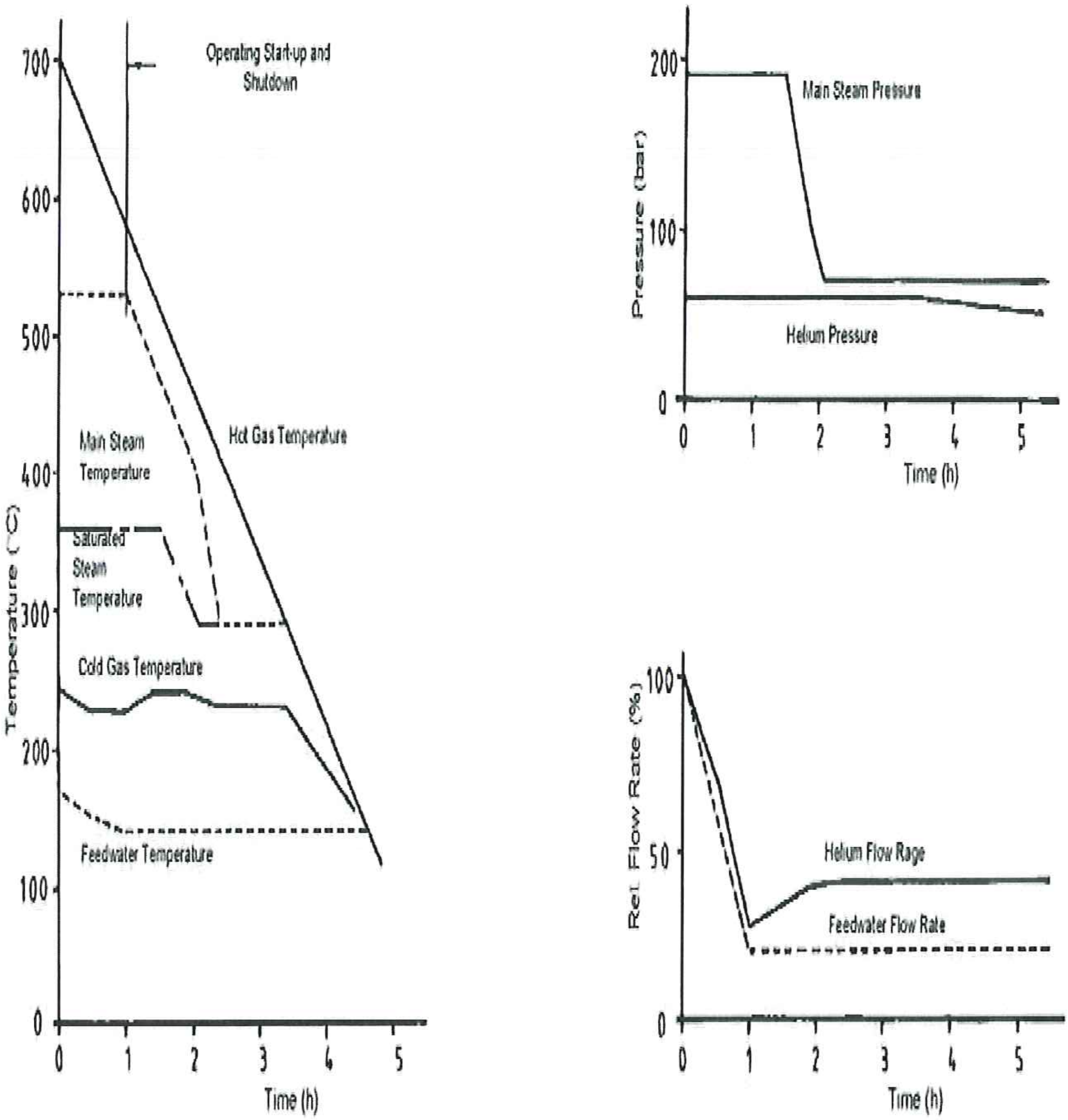


Figure 3-4: Part Load Diagram

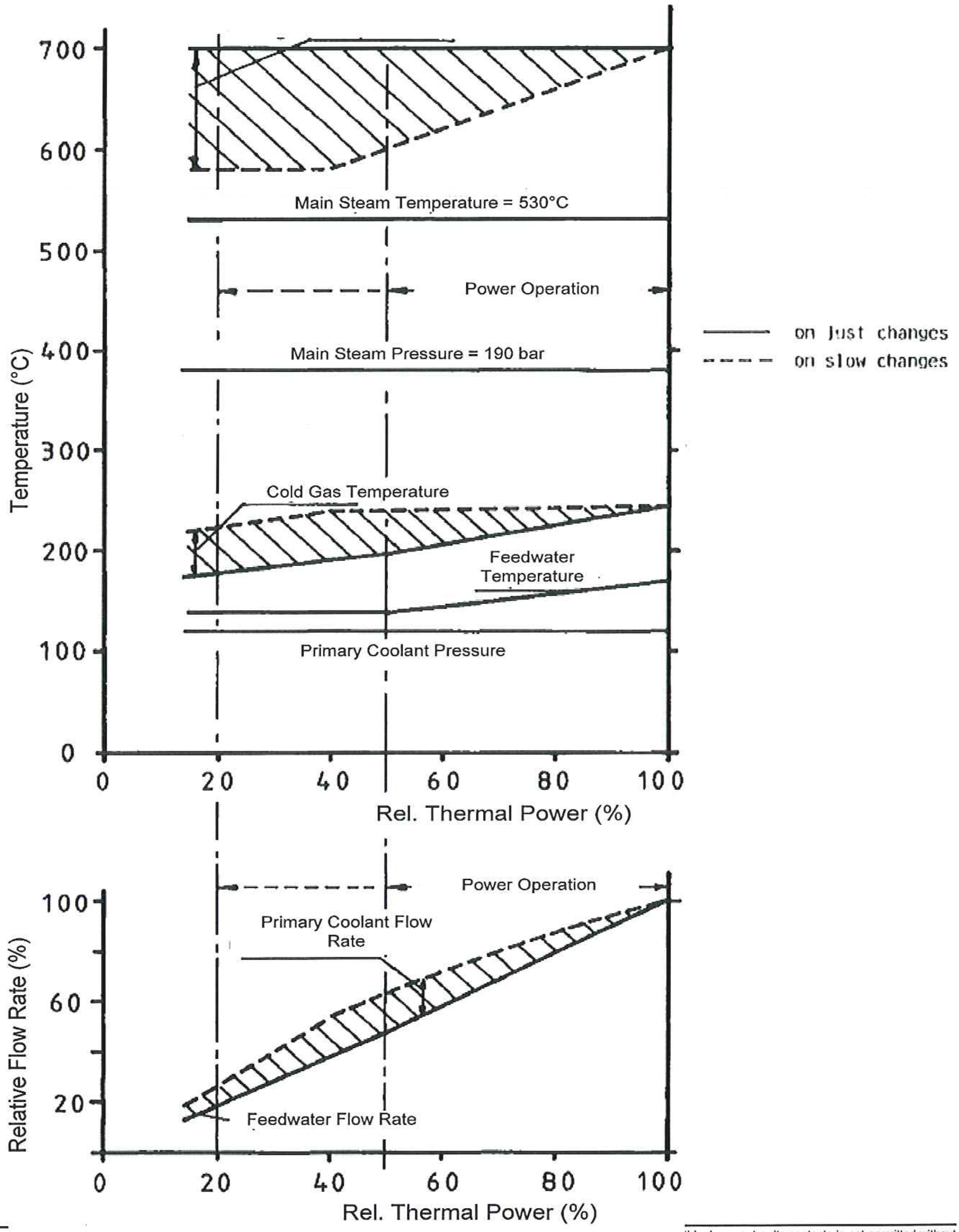


Figure 3-5: Maximum Fuel Temperature vs. Time Curve in Core after Reactor Scram, without Operation Residual Heat Removal

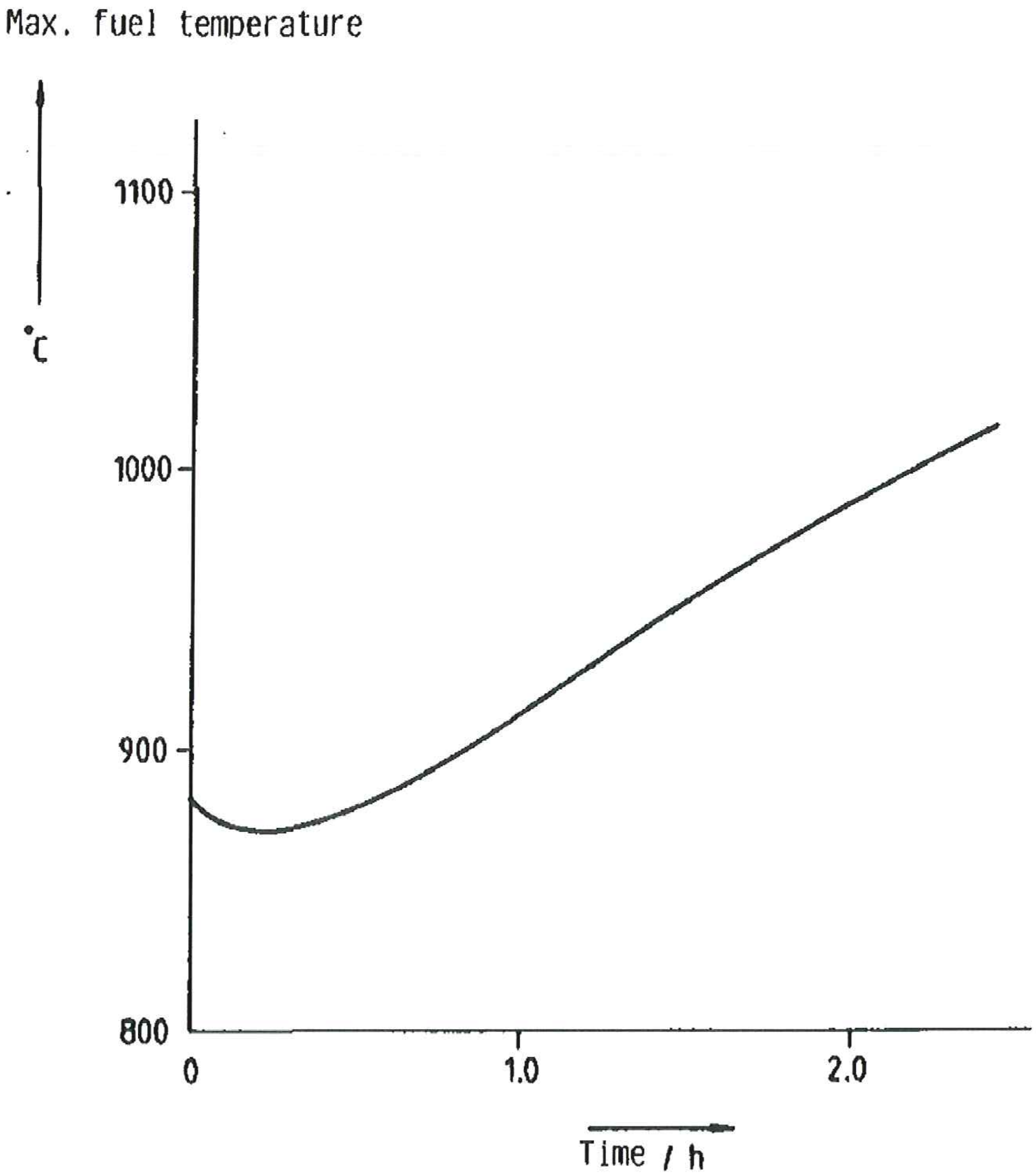


Figure 3-6: Relative Helium- /Water- /Steam- Flow Rates in Steam Generator after Reactor Scram

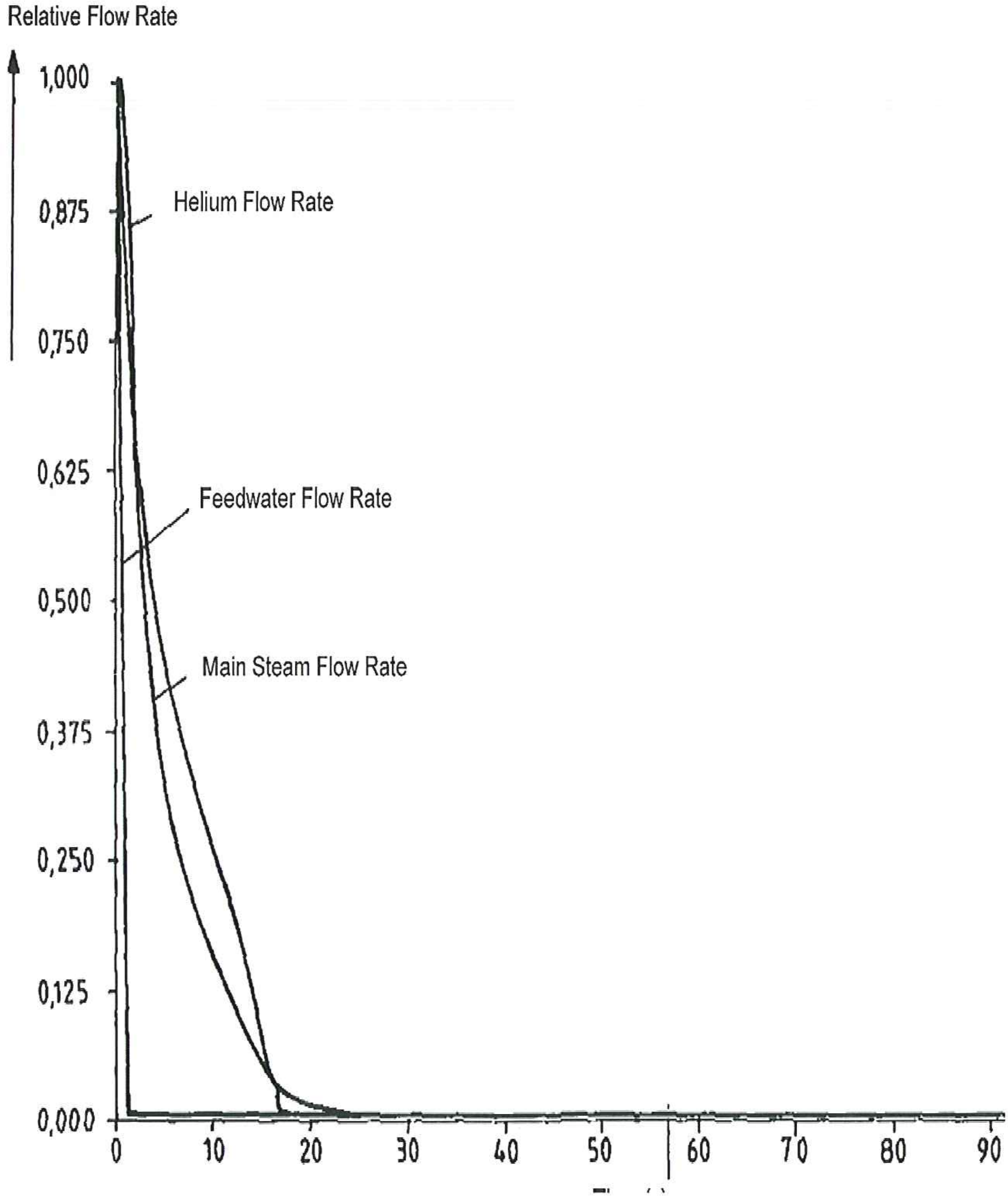


Figure 3-7: Temperature in Steam Generator after Reactor Scram

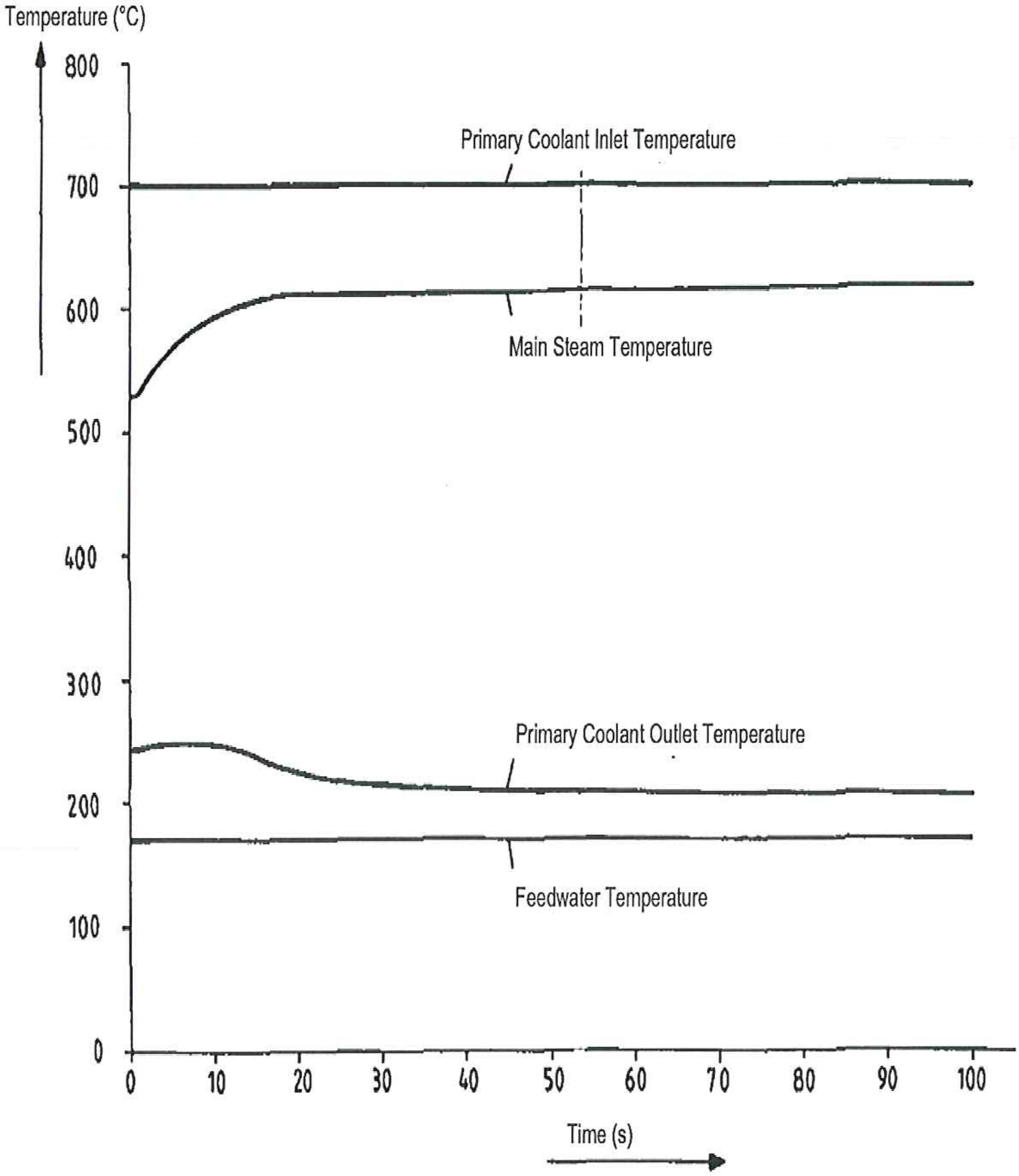


Figure 3-8: Peak Temperature vs. Time Curves

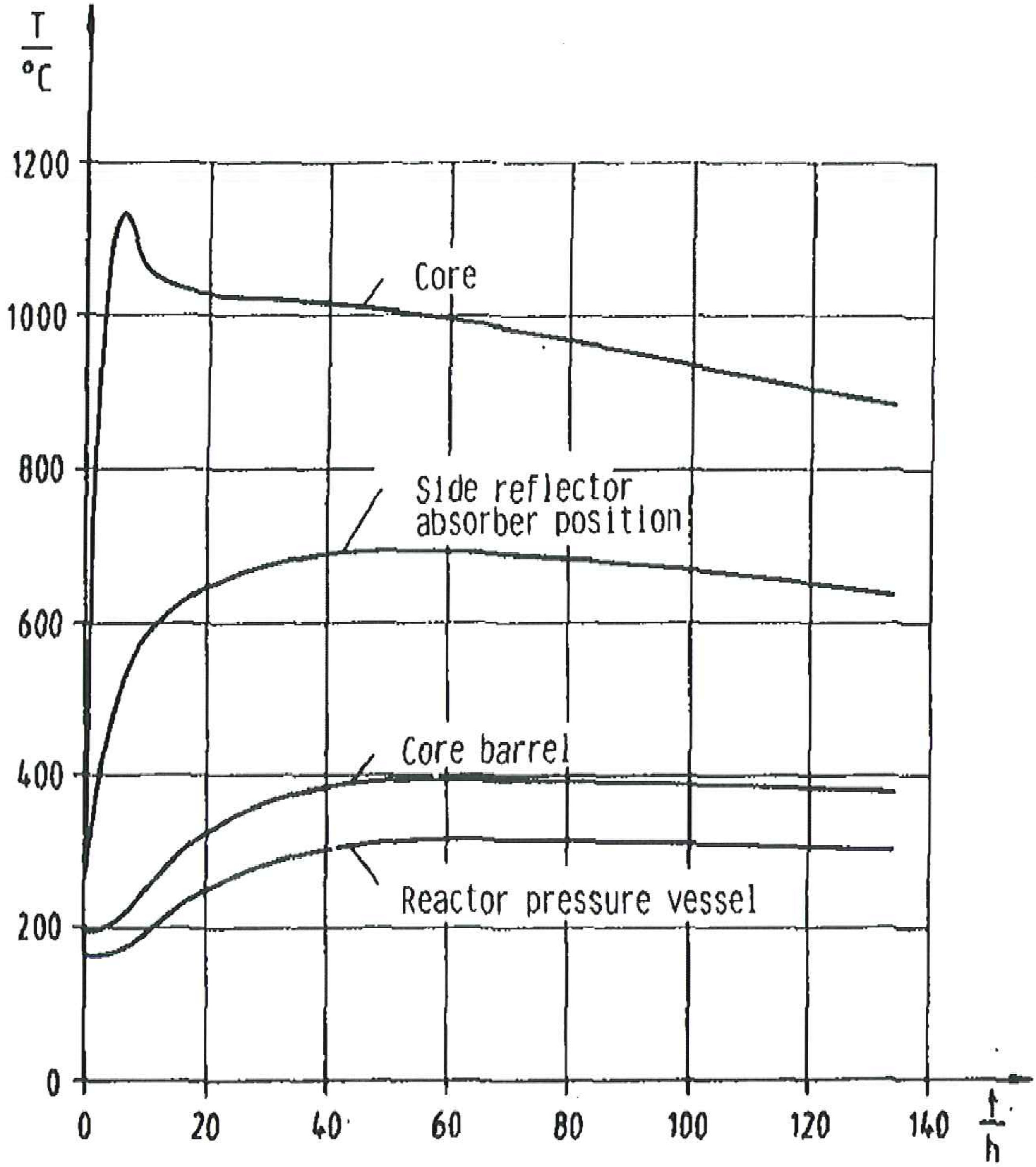


Figure 3-9: Radial Temperature Curves at Elevation of Peak Core Temperature

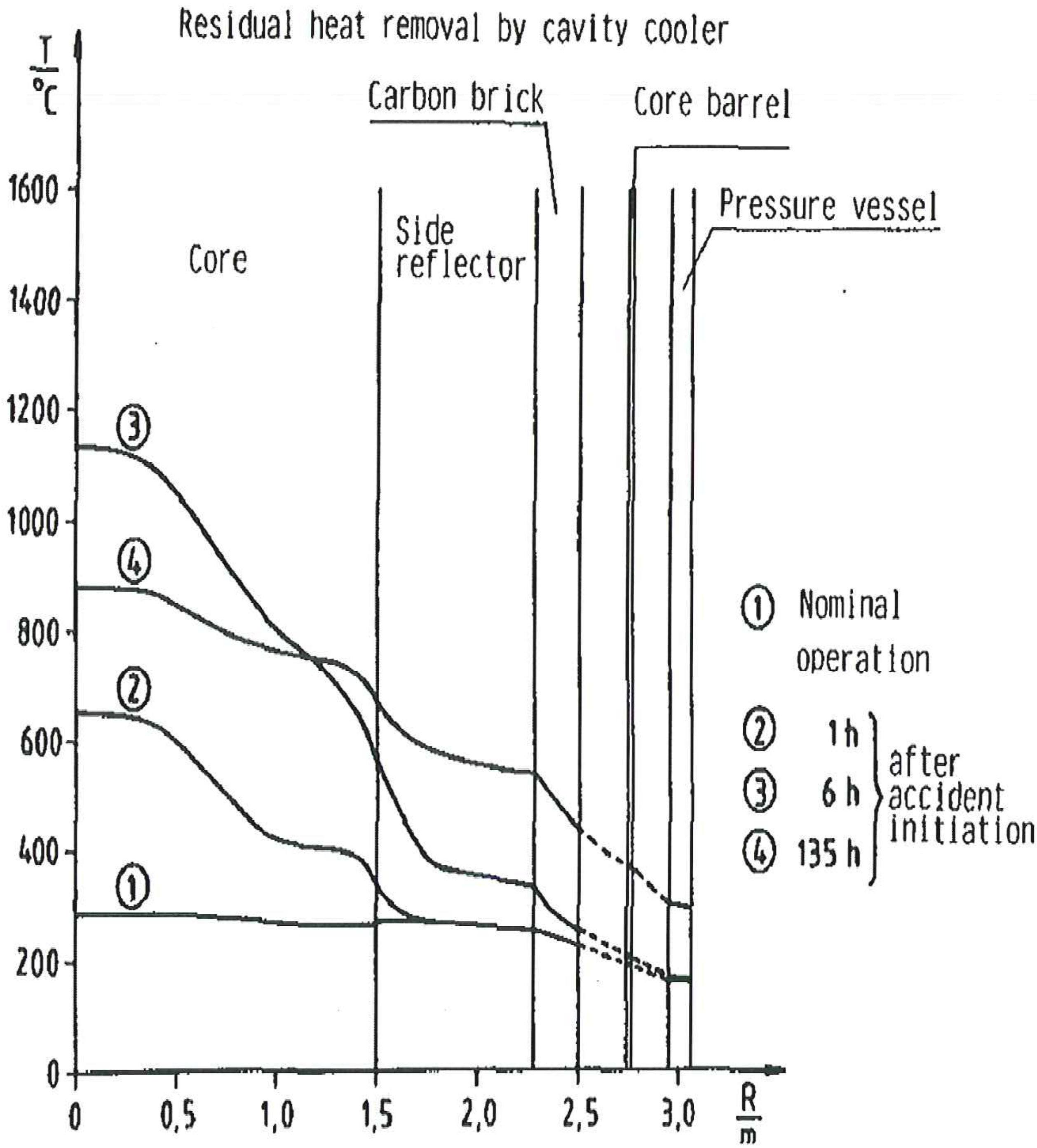
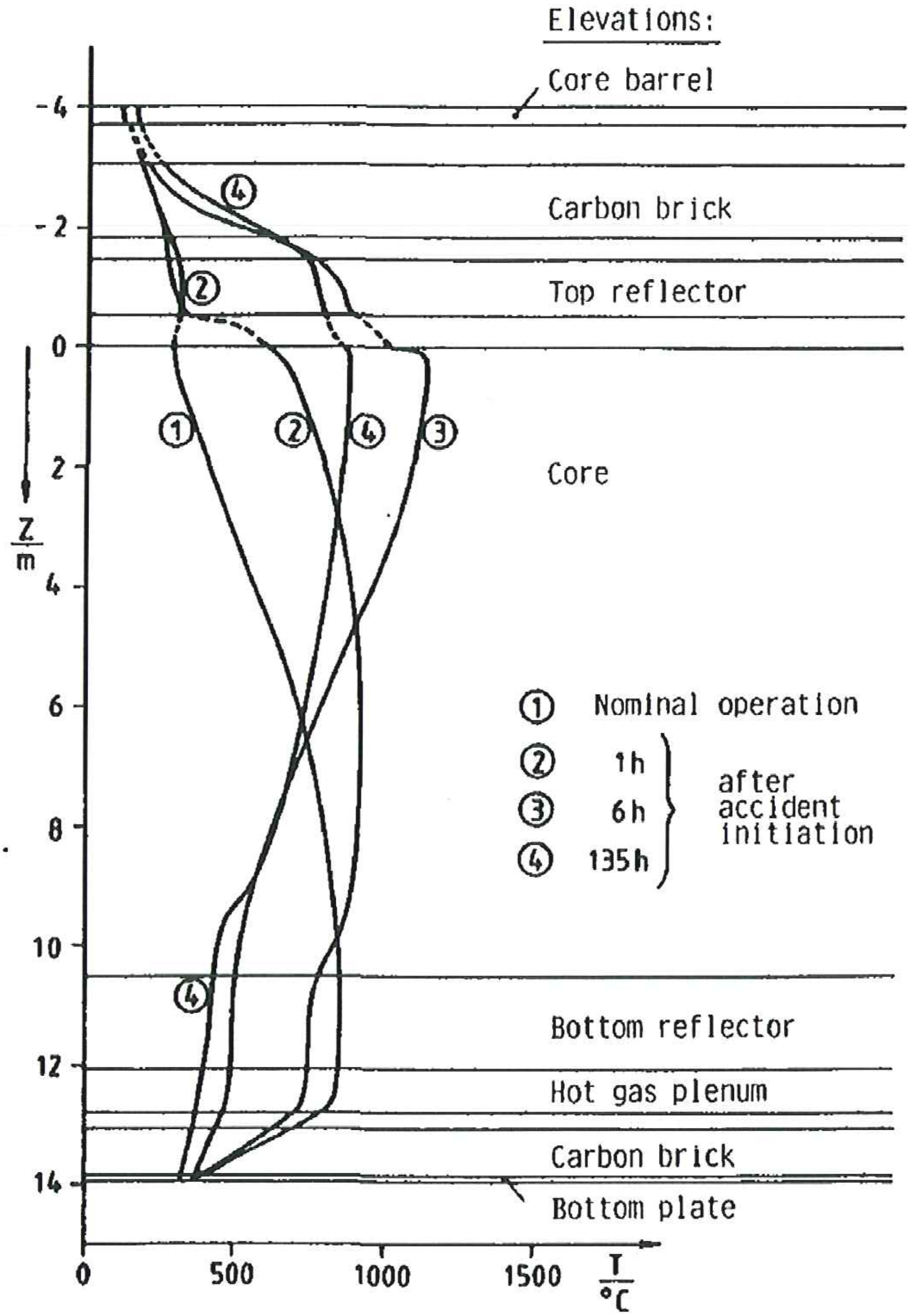


Figure 3-10: Axial Temperature Patterns Along Core Axis



Residual heat removal by cavity cooler

4. Main Heat Transfer Malfunction Events

4.1 Loss of Auxiliary Power Supply

4.1.1 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 3.4). 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 shortterm heat-up of the steam generator, that is further intensified, in contrast to the “manual scram” case, because the circulator damper (as postulated for DBE) 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.

4.1.2 Assessment

In the long term, the core reacts in the same way as for the "manual scram" case. Therefore, the performance is the same as described for the manual scram in Section 3.4.

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

4.2 Loss of Primary Coolant Mass Flow

4.2.1 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.

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.

4.2.2 Assessment

In the long term, the core reacts in the same way as for the "manual scram" case. Therefore, the performance is the same as described for the manual scram in Section 3.4.

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

4.3 Loss of Feedwater Flow

4.3.1 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.

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.

4.3.2 Assessment

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 3.4.

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

4.4 Inadvertent Closure of a Main Steam Valve

4.4.1 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 set point 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

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 cool down 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.

4.4.2 Assessment

In the long term, the core reacts in the same way as for the “manual scram” case. Therefore, the performance is the same as described for the manual scram in Section 3.4.

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

4.5 Main Steam Extraction Malfunction

4.5.1 Event Description

In addition to a main steam line break, 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, ist 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 is the same as closure of the main steam valve.

4.5.2 Assessment

In the long term, the core reacts in the same way as for the "closure of main steam valve" event. Therefore, the performance is the same as described for the closure of main steam valve event in Section 4.4..

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

4.6 Process Steam Extraction Malfunction

4.6.1 Event Description

Process steam extraction malfunctions affect main steam extraction that is covered by the malfunction events.

4.6.2 Assessment

This is covered by the barrier performance by the malfunction events.

This radiological consequences is covered by the malfunction events.

4.7 Inadvertent Opening of Valves in the Water/Steam Cycle

4.7.1 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 is covered by the malfunction events.

4.7.2 Assessment

In the long term, the core reacts in the same way as for the "loss of feedwater" event in Section 4.3.. Therefore, the performance is the same as described for the loss of feedwater is covered by the malfunction events.

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

4.8 Turbine Trip

4.8.1 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.

4.8.2 Assessment

There is very little change in the core response characteristics of temperature and pressure.

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