Joint IAEA-GIF Technical Meeting
Vienna, 9-11.12.2019

HTR-MODULE
Siemens Design of the 80ies
HTR-MODULE - Power Plant for Cogeneration of Electrical Power and Process Heat

1. High Temperature Reactor
2. Steam Generator
3. Primary Circuit Blower
4. High Pressure Turbine
5. Medium Pressure Turbine
6. Low Pressure Turbine
7. Gear Unit
8. Electr. Generator
9. Process Steam Superheater
10. Condenser
11. Main Condensate Pump
12. Feedwater Tank
13. Main Feedwater Pump
14. Cooling Tower
15. Main Cooling Water Pump
**Time Schedule of the Licensing Procedure / Safety Concept Review (I)**

**Apr 87**  Application for initiation of concept licensing procedure pursuant to Art. 7 a of the Atomic Energy Act docketed with Lower Saxony Ministry for the Environment (licensing authority) on the basis of safety analysis submitted by Siemens/Interatom

**May 87**  Lower Saxony Ministry for the Environment retains TÜV Hanover to conduct safety review of HTR Module concept

**Sep-Dec 87**  Technical consultations with experts and licensing authority; appr. 100 technical documents generated for this purpose

**Feb 88**  Experts call for more supplementary technical documents

**Sep 88**  Revision of safety analysis report completed; submission to licensing authority and expert

**Dec 88**  Start of RSK consultations

**Feb 89**  Report on fire protection concept completed

**Mar 89**  Report on plant security concept completed
Time Schedule of the Licensing Procedure / Safety Concept Review (II)

April 89  Application for concept licensing procedure withdrawn by applicant and proceedings suspended by Lower Saxony Ministry for the Environment

May 89   Review continued by TÜV Hanover on behalf of BMFT

July 89  Draft review report submitted by TÜV Hanover

Sep 89   Final meeting of RSK Subcommittee for HTRs

Oct 89   Final meeting of RSK Subcommittee for Electrical Engineering

Dec 89   Completion of final review report

Mar 90   Recommendation on the HTR Safety Concept by RSK
Contents of Chapter 2 (SAR HTR-MODULE)

> General design features of the HTR module power plant
> Introductory remarks
> Characteristic safety features
  Barriers against release of radioactivity
  Inherent safety
> Technical design features
  Reactor
  Nuclear steam supply system
  Confinement envelope
  Residual heat removal
  Helium purification system
  Fuel handling and storage
  Accidental power supply
  Reactor protection system
  Remote shutdown station
  Controlled area
> Nuclear classification and quality requirements
> Summary of design basis events
> Postulates and measures for in-plant events
> Postulates and measures for external events
Residual heat removal
Provided by secondary system, cavity coolers, helium purification system
On loss of active cooling, residual heat removed from core to cavity coolers solely by thermal conduction, radiation and natural convection
Secured component cooling system, two-train
With cavity coolers intact and loss of core cooling, core can run hot for lengthy period of time (15 h) without design limits for RPV and concrete of reactor cavity being violated (Today: 72 hours)
External supply can be connected to cavity coolers in the event of severe accident conditions
Dr. Brinkmann, BriVaTech, IAEA-GIF Technical Meeting, Vienna, 9-11.12.2019
Reactivity Accidents

In the HTR-MODULE, the following events cause an undesired increase over time of the reactivity in the core and of the reactor power:
- Inadvertent withdrawal of reflector rods or small ball shutdown elements
- Ingress of moderating materials into the core (water)
- Inadvertent increase in primary coolant mass flow (negative temperature coefficient of the reactivity)
- Inadvertent decrease in of the cold gas temperature (negative temperature coefficient of the reactivity)
- Compaction of the pebble bed due to earthquake
Reactor Scram

• Reactor scram is actuated once the reactor protection limits have been reached. As a result, the reflector rods drop into the lowest position, the primary gas blower is tripped and the steam generator is isolated on the feedwater and steam side. This safely renders the plant subcritical.
Withdrawal of all Reflector Rods-Assumptions

**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 approx. 2.5 m below the top edge of the pebble bed and bind 1.2 % reactivity which is held in reserve for the requirements of part-load operation and hot start-up.*

An initial power of 210 MWth is assumed and by the limitation of the speed of the primary circuit blower it is ensured that it will not be exceeded. At the same time, 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.
Reactivity due to Withdrawal of all Reflector Rods

![Diagram showing reactivity changes over time](image)

- Scram at hot gas temperature ≥ approx. 750 °C
- Scram at therm. corr. neutron flux ≥ 120 %

Withdrawal of reflector rods at maximum speed at full-load equilibrium (210 MW)
Water Ingress

A steam generator tube break is detected after about 10 seconds by the moisture instrumentation in the primary system. Of the steam generator water inventory, only about 600 kg of water can enter the primary system under the worst-case conditions.

If one assumes that this quantity of water is distributed homogeneously as steam in the primary system, the resulting steam density in the hot core is approx. 0.7 kg per m³ of the pebble bed.

Due to use of the 7 gram fuel element, the pebble bed is only slightly undermoderated. Consequently steam ingress into the pebble bed only leads to a slight reactivity increase of 0.4 %.

In the context of the design basis accidents, the reactivity effects due to water ingress into the primary system are of subordinate importance and are enveloped by withdrawal of the reflector rods or small ball shutdown elements.
Compaction of the Pebble Bed due to Earthquake

An earthquake can cause the pebble bed fill factor to increase and hence, within a short time, to reactivity to be inserted:

- By reducing the neutron leakage from the pebble bed, and
- As a result of movement of the pebble bed surface relative to the reflector rods.

For a postulated horizontal earthquake acceleration of 0.5 g (g = 9.81 m/s²), the fill factor increases from 0.61 to 0.614 within approx. 6 seconds at constant excitation. The inserted reactivity amounts to 1.25 o/oo with 0.5 o/oo due to movement of the pebble bed surface relative to the reflector rods.
Reactivity Change due to Earthquake as a Function of Fill Factor
Reactivity due to Earthquake

![Graph showing reactivity changes over time due to an earthquake. The graph displays two scram events with corresponding relative values and time in seconds.]
The Aim of Design using Barriers

• The minimization of radioactive releases during normal operation and accidents.

• Knowing the design of an HTR the accident analyses have to demonstrate with adequate safety margins that the radioactive releases are far below the governmental dose limits.

• So the question is: What is the quality of each barrier and the quality of all together against radioactive releases?

  (Here: the confinement question is not a question of barriers against external events)
Fuel particles:

More or less a safe confinement of radioactivity (depending on burn up and temperature)

Quality of this first barrier is given by the failed Particle Fraction Curve

In general during licensing procedures the data for calculation are about one magnitude higher than the expected values.

So this curve is the starting point for the discussion confinement/containment.

For HTR Module (Siemens design) the curve was based on the German fuel tests.
HTR-Module Siemens Design - Maximum Failed Particle Fraction as a Function of Fuel Temperature

![Graph showing Maximum Failed Particle Fraction as a Function of Fuel Temperature](image)

- ① Irradiation-induced particle failure (proportional to burnup)
- ② Fuel element with maximum burnup
- ③ Fuel element with mean burnup
- ④ New fuel element
- Uncoated uranium due to manufacturing defects

Fuel Temperature vs. Maximum Failed Particle Fraction

Dr. Brinkmann, BriVaTech, IAEA-GIF Technical Meeting, Vienna, 9-11.12.2019
HTR-Module Siemens Design - Peak Temperature versus Time Diagram
HTR-Module Siemens Design - Volumetric Fractions of Core at certain Temperatures, Core Heat-up Phase
Primary gas envelope (primary boundary):

More or less a good confinement of radioactivity (depending on the quality of the components and piping)

Using the quality of LWRs (nuclear grade class 1 - whatever that means in different countries):
You can use the break postulations (state of the art):
  • no through wall cracks in vessels
  • 2A breaks in piping with known probability

If you change the quality of the second barrier, you have to strengthen the first or the third barrier.
HTR-Module - Second Barrier

For the HTR Module it was postulated (like PWRs in Germany)

- small breaks (DN10) - unisolable (Quality class for small piping)

- large breaks (DN65) - isolable (Quality class 2)

- large breaks (DN65) - unisolable (Quality class 1)
  - low probability
  - only a few connections to the vessel
Event Classification for HTR-Module Power Plants
Depressurization Phase:

Assumptions:
Direct counter measures of reactor protection system.
Depressurization time of about 3 minutes.
Pressure build up in the building.
Relief dampers to the stack open.

Radioactive Release
- Content of the primary circuit
- Content of helium purification system incl. desorption of filters
- Desorption of surface activity in the primary circuit
- Dust (1 kg)
- No plate out in the building
### HTR-Module - Large Breaks (DN 65) - unisolable (cont.)

Factor: design / expected

<table>
<thead>
<tr>
<th></th>
<th>3.3 E+12 Bq</th>
<th>~ 3</th>
</tr>
</thead>
<tbody>
<tr>
<td>Noble gases</td>
<td>9.1 E+09 Bq</td>
<td>~ 100</td>
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<tr>
<td>Iodine</td>
<td>5.3 E+06 Bq</td>
<td>~ 50</td>
</tr>
<tr>
<td>Sr, Ag, Cs</td>
<td>5.6 E+12 Bq</td>
<td>~ 2</td>
</tr>
<tr>
<td>Tritium</td>
<td>6.0 E+10 Bq</td>
<td>~ 10</td>
</tr>
</tbody>
</table>

**Radiological impact (dose):** unfiltered limit

- **whole body (child)**: 9.76 E-06 0.05 (Sv)
- **thyroid (child)**: 5.45 E-04 0.15 (Sv)
Core heat-up phase

Assumption:

> All operating parameters very high (temperature, power)
> Failure of the first protection signal (single failure in reactor protection system - not normal in licensing)
> Combination (addition) of uncertainties result: about 130°K to the calculated values.
> Gas expansion in the vessel due to temperature increase will end after 160 hours with 9% vol. increase, end of release into the building
> No plate out in the building
HTR-Module - Large Breaks (DN65) - unisolable (cont.)

Radioactive Release in the Building

<table>
<thead>
<tr>
<th></th>
<th>Design</th>
<th>expected</th>
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</thead>
<tbody>
<tr>
<td></td>
<td>T (lic.)</td>
<td>T (nom.)</td>
</tr>
<tr>
<td>Iodine</td>
<td>1.3 E+10</td>
<td>5.3 E+09</td>
</tr>
</tbody>
</table>

Radioactive Release to Environment

<table>
<thead>
<tr>
<th></th>
<th>filtered</th>
<th>unfiltered</th>
</tr>
</thead>
<tbody>
<tr>
<td>Xe 133</td>
<td>2.7 E+11</td>
<td>2.7 E+11 (Bq)</td>
</tr>
<tr>
<td>I 131</td>
<td>1.3 E+08</td>
<td>1.3 E+10 (Bq)</td>
</tr>
<tr>
<td>Ag 110m</td>
<td>1.2 E+06</td>
<td>1.2 E+9  (Bq)</td>
</tr>
<tr>
<td>Cs 137</td>
<td>2.0 E+06</td>
<td>2.0 E+9  (Bq)</td>
</tr>
<tr>
<td>Sr 90</td>
<td>1.0 E+04</td>
<td>1.0 E+7  (Bq)</td>
</tr>
</tbody>
</table>
Radiological Impact (Dose) Depressurization and Heat-up Phase:

<table>
<thead>
<tr>
<th></th>
<th>filtered</th>
<th>unfiltered</th>
<th>limit</th>
</tr>
</thead>
<tbody>
<tr>
<td>whole body (adult)</td>
<td>9.8 E-06</td>
<td>2.97 E-04</td>
<td>0.05 (Sv)</td>
</tr>
<tr>
<td>thyroid (child)</td>
<td>6.22 E-04</td>
<td>1.23 E-02</td>
<td>0.15 (Sv)</td>
</tr>
</tbody>
</table>

**Result:**

Impact far below the limits.

**Decision:**

Filters are not safety related, so in the licensing procedure the reference case is the unfiltered one.
Confinement Envelope consists of:

> Reactor Building (leak tightness 50Vol.%/day)
> Building pressure relief system
> HVAC system isolation
> Subatmospheric pressure system (filter) as operational system

This concept was accepted in the 80ies by TUeV Hanover (expert of Government Lower Saxony) and by RSK (expert of German Federal Government)
HTR-Module Siemens Design - Cross Section of Reactor Building
HTR-Module Siemens Design - Calculations with American Regulations in 2011

<table>
<thead>
<tr>
<th>Design Basis Accidents</th>
<th>SAR (1988)</th>
<th>Dose (TEDE) Sv</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td>EAB (2 hour)</td>
<td>NRC Limit</td>
</tr>
<tr>
<td></td>
<td>(1)</td>
<td>(2)</td>
</tr>
<tr>
<td>Break of a Large Connecting Pipe (DN 65) - LBLOCA short-term unfiltered release</td>
<td>6.2.3.1.1</td>
<td>unfiltered: 1.361E-04 (13.61 mrem)</td>
</tr>
<tr>
<td>Break of a Large Connecting Line (DN 65) - LBLOCA long-term unfiltered release with core heat up</td>
<td>6.2.3.1.2</td>
<td>filtered: 2.858E-06 unfiltered: 2.214E-04 (22.14 mrem)</td>
</tr>
<tr>
<td>Instrument Line Break Pressure release phase (DN &lt;10)</td>
<td>6.2.3.3</td>
<td>filtered: 5.183E-05 unfiltered: 5.782E-05 (5.782 mrem)</td>
</tr>
<tr>
<td>Steam Generator Tube Rupture with response of the Pressure Relief System</td>
<td>6.2.5.2</td>
<td>filtered: 2.245E-06 unfiltered: 3.464E-05 (3.464 mrem)</td>
</tr>
<tr>
<td>Helium Purification System Pipe Break release via stack</td>
<td>6.2.6.1</td>
<td>unfiltered: 1.975E-04 (19.75 mrem)</td>
</tr>
<tr>
<td>Non-Design Basis Accidents</td>
<td></td>
<td></td>
</tr>
<tr>
<td>Leakage of Vessel Containing Radioactive Contaminated Water</td>
<td>6.2.6.2</td>
<td>unfiltered: 2.443E-06 (0.2443 mrem)</td>
</tr>
<tr>
<td>Seismic Effects on the Reactor Auxiliary Building</td>
<td>6.2.7</td>
<td>unfiltered: 3.961E-04 (39.61 mrem)</td>
</tr>
</tbody>
</table>

Notes:
(1) The worst two hour window is used with an atmospheric dispersion factor (X/Q) value of 3.35E-03 s/m² 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 is 10% of the full limit, or 0.025 Sv.