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Research

Development of an Input Model to MELCOR 1.8.5 for the Ringhals 3 PWR

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SKI Perspective

Background

Assessments of the source term for Swedish nuclear power reactors are based on calculations made by means of the severe accident analysis code MAAP. This code was developed for the US nuclear industry. MELCOR is another code for severe accident analysis, developed by Sandia National Laboratories for the US Nuclear Regulatory Commission (NRC). Both MAAP and MELCOR are integrated codes that simulate the whole course of events in a severe accident from the initiating event to the source term, i. e. radioactive releases to the environment. In Sweden, MELCOR has mostly been employed in some International Standard Problems (ISPs).

In consequence of plans to increase the power in Ringhals 3 there is demand for a MELCOR model that can be used for independent analyses of specific limiting transients and accidents.

Objective

The objective was to obtain an executable and complete input model for the Ringhals 3 that meets standard requirements for MELCOR 1.8.5. The input should work in steady-state calculations as well as in tentative accident cases. The intention was also to carry out calculations with the MELCOR model of certain accident sequences for comparison with results from earlier MAAP4 calculations. This work will be reported separately.

Results

Preliminary calculations were made in order to simulate the initial conditions at current nominal operating conditions in Ringhals 3 for 2775 MW thermal power. The results are presented in the report.

Project information

SKI reference: 14.7-040188-200407007
Coordinator: Ninos Garis, Departement of Reactor Technology and Structural Integrity.

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ABSTRACT

An input file to the severe accident code MELCOR 1.8.5 has been developed for the Swedish pressurized water reactor Ringhals 3. The aim was to produce a file that can be used for calculations of various postulated severe accident scenarios, although the first application is specifically on cases involving large hydrogen production.

The input file is rather detailed with individual modelling of all three cooling loops. The report describes the basis for the Ringhals 3 model and the input preparation step by step and is illustrated by nodalization schemes of the different plant systems.

Present version of the report is restricted to the fundamental MELCOR input preparation, and therefore most of the figures of Ringhals 3 measurements and operating parameters are excluded here. These are given in another, complete version of the report, for limited distribution, which includes tables for pertinent data of all components. That version contains appendices with a complete listing of the input files as well as tables of data compiled from a RELAP5 file, that was a major basis for the MELCOR input for the cooling loops.

The input was tested in steady-state calculations in order to simulate the initial conditions at current nominal operating conditions in Ringhals 3 for 2775 MW thermal power. The results of the steady-state calculations are presented in the report. Calculations with the MELCOR model will then be carried out of certain accident sequences for comparison with results from earlier MAAP4 calculations. That work will be reported separately.

SAMMANFATTNING

En indatafil har tagits fram till koden MELCOR 1.8.5 för Ringhals 3. MELCOR är ett beräkningsprogram för analys av svåra haverier i kärnkraftverk. MELCOR utvecklas av Sandia National Laboratories på uppdrag av USNRC och för CSARP (Cooperative Severe Accident Research Programme), där SKI deltar.

MELCOR är en s k integrerad kod som räknar på hela haveriförloppet från inledande händelse, t ex totalt elbortfall eller rörbrott, härdens uppvärmning, degradering och smältning, genomsmältning av reaktortankbotten och inverkan på inneslutningen.

Föreliggande uppdrag syftade till att ta fram en indatafil att användas i första hand för beräkning av förlopp som ger betydande vätgasbildning och därmed risk för antändning och stora tryckpåkänninger på reaktorinneslutningen. Avsikten var också att göra jämförelser med resultat från tidigare beräkningar med koden MAAP4. Arbetet med dessa jämförande beräkningar rapporteras separat.

En relativt detaljerad indatafil har tagits fram där alla tre kylkretsarna har modellerats var för sig. Som underlag har främst använts FSAR för R3. Data för reaktortank och kylsystem har till stor del kunnat hämtas från en detaljerad indatafil till RELAP5 som tagits fram av Studsvik på uppdrag av SKI.

Indatafilen till MELCOR har delats upp i 8 delfiler för olika system och kodpaket. Modellen för R3 omfattar totalt 117 controlvolymer, 180 flödesvägar och 161 värmestrukturer. Preliminära beräkningar har gjorts för att simulera initialtillståndet vid nuvarande nominella driftbetingelser för 2775 MW termisk effekt. Resultaten redovisas i rapporten.

I föreliggande rapportversion, som främst avser att visa principerna för indata-prepareringen till MELCOR, har siffervärden med Ringhals 3–data i text och bilagor med underlagstabeller samt listningen av indatafilerna utelämnats. Den kompletta rapporten med denna information och en elektronisk indatalista kan beställas från SKI (var god ange projektnummer 14.7-040188-200407007).

1. INTRODUCTION

Analyses of various hypothetical severe accident scenarios are continuously being done as a part of the ongoing safety research managed by the Swedish Nuclear Power Inspectorate (SKI). The safety research is carried out in cooperation with the nuclear industry and utilities, and has been formalised in a number of research projects such as FILTRA, RAMA, HAFOS and APRI. A new phase of the latter, APRI 5, has recently been initiated.

Safety assessment for postulated severe accidents in Swedish plants has in the past to a great extent been based on results from the MAAP 3 and more recently the MAAP 4 code, used by the industry and the utilities. MELCOR is another code for severe accident analysis, developed under the auspices of the US Nuclear Regulatory Commission (NRC) and made available to the SKI and the APRI partners through the CSARP project (Cooperative Severe Accident Research Programme). The MELCOR code offers an opportunity to carry out independent calculations and constitutes thus a basis for review of earlier results.

The aim with present project is to produce an input model for the Ringhals 3 PWR and recalculate two accident scenarios previously calculated with MAAP 4 [Ref. 1]. The accident cases are focused on hydrogen generation and pressure loads in the containment in case of hydrogen deflagration.

The MELCOR input has, however, been made as a general base model with the aim to facilitate simulation of a broader spectrum of severe accident scenarios for Ringhals 3.

In present version of the report most of the figures of Ringhals 3 measurements and operating parameters are excluded. The aim is here to demonstrate the fundamental MELCOR input preparation. There is also another, complete version of the report, for limited distribution, which includes tables for pertinent data of all components. That version contains appendices with a complete listing of the input files and tables of data compiled from a RELAP5 file, that was a major basis for the MELCOR input for the cooling loops. A copy of the complete report and the full input in electronic form can be obtained on request from SKI (Please, refer to the project No. 14.7-040188-200407007).

2. RINGHALS 3 PWR

Ringhals 3 (R3) is a three loop Westinghouse Pressurized Water Reactor (PWR) located on the West Coast of Sweden and operated by the Ringhals AB Company. Ringhals 3 has been in commercial operation since 1981. In 1995 the steam generators were replaced, due to excessive tube damage problems, by new ones manufactured by Framatome/ Siemens. The reactor containment is of the large dry type.

Some pertinent design parameters needed for the MELCOR input are listed below. Further details are found in following chapters describing the different systems.

- Reactor vessel, Total outside height
- Inner diameter
- Number of fuel assemblies
- Number of fuel rods and array
- Number of control rod clusters and array
- Total mass of UO_2 fuel
- Total mass of Zircaloy (Zr-4) in core
- Steam generator approx. height
- Number of heat exchanger tubes
- Heat exchange surface area per unit

The design thermal power is 3120 MW with the new steam generators, but R3 is presently operating at 2775 MW_{th}. This rating is also the basis for present MELCOR model concerning the targeted steady-state conditions. The following operating data are input according to information from the utility [Ref. 2]:

- Thermal power
- Operating pressure in primary system
- Primary coolant flow rate
- Temperature, cold leg, vessel inlet
- Temperature, hot leg, vessel outlet
- Feed water flow rate per loop accounting for bottom drainage
- Steam mass flow rate per loop
- Feed water temperature
- Steam dome pressure
- Circulation ratio

3. THE MELCOR 1.8.5 CODE

The MELCOR code [Ref. 3] is being developed at Sandia National Laboratory (SNL) under contract from the U.S. Nuclear Regulatory Commission (NRC). MELCOR is a successor to STCP (Source Term Code Package) and has thus a long evolutionary history.

MELCOR is a fully integrated code that models all phases of severe accident progression in a LWR plant. It is a so-called lumped parameter code, i.e. zero-dimensional with respect to the modelling of hydrodynamic volumes. Thermodynamic state properties are then given in one point in each cell of the volume and considered as constant within the volume. The spatial geometry is given in input as a volume/altitude table. Each volume can have a pool and an atmosphere fraction, the latter can consist of steam and a number of non-condensable gases. Two-phase models allow for steam bubbles in the pool and water droplets, “fog”, in the atmosphere. If the non-equilibrium thermodynamics option is chosen, different temperatures and phase conditions can exist between pool and atmosphere.

Through connections with flow paths having lengths and inclinations sort of a three dimensional model of the reactor system can be simulated. The code comprises a driver module and a large number of various model packages, such as the control volume hydrodynamics package, flow path, core description, heat structure, radionuclide package, etc. The number of packages, which are engaged in the execution, is dependent on the problem to be solved.

Various severe accident phenomena in both PWRs and BWRs can be treated by MELCOR. Version 1.8.5 include models for, among others, the following:

- Thermal-hydraulic response in the reactor coolant system, reactor cavity, containment and confinement buildings.
- Core heat-up, degradation and relocation.
- Core-concrete attack.
- Hydrogen production, transport and combustion.
- Fission product release, transport, and deposition.
- Impact of engineered safety features on thermal-hydraulic and radionuclide behaviour and on hydrogen combustion.
- Passive Autocatalytic Hydrogen Recombiner by the PAR package

MELCOR calculations are executed in two steps, MELGEN and MELCOR. MELGEN is the input processor to which the majority of the input is written. It carries out the input check, and if the input is accepted it writes preliminary initial conditions to a restart file. MELCOR then needs only a short input, mainly time-step and execution parameters for the transient calculations. Only few input data can be changed in the MELCOR input.

MELCOR runs produce a number of output files; a diagnostic (.DIA) file displaying input errors and non-fatal warnings to assist the user in debugging the input, an output text file (.OUT), a binary restart (.RST), a plot file (.PTF), and a message file (.MES). The latter announces major events in the accident progression and by means of control functions the user can flag for events of special interest such as valve actuations, containment failure, etc.

The code package also includes a graphics processor HISPLT, retained from earlier versions and which needs a special input. However, plotting is more conveniently made directly from the plot file by means of the general graphics code XMGR5, or its successor ACGRACE. Figure 1 below illustrates the relation between the various codes and files in the MELCOR package.

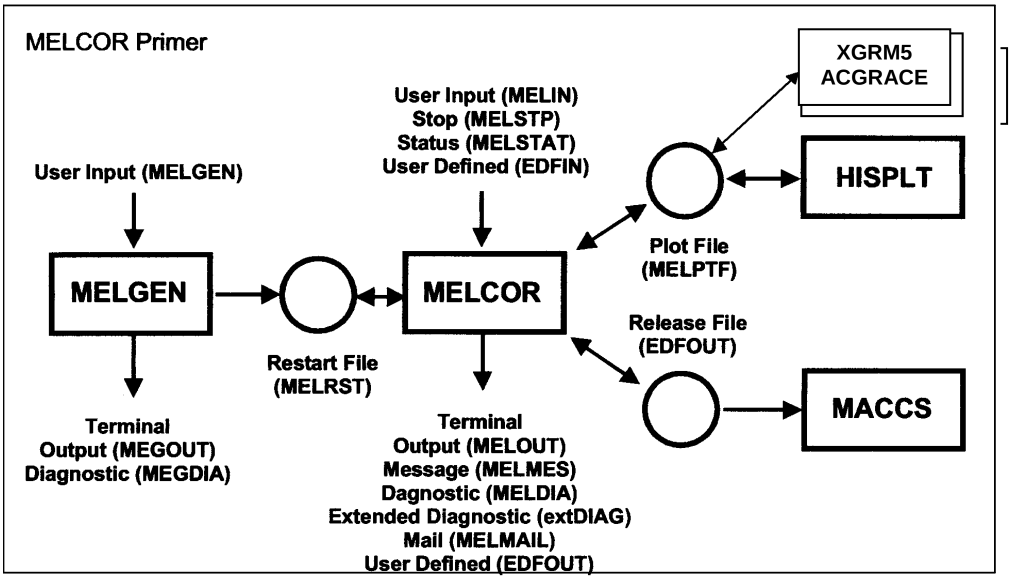

Figure 1. Relationship between the various codes and files in MELCOR

MACCS calculates off-site consequences of radioactive releases, i.e. from the source term, to the environment. The file names shown in the figure are default names which can be optionally changed by the user.

The latest released version, available to present project, was MELCOR 1.8.5, denoted “BASE CODE VERSION 1.8.5(A), SEP –25-2000, BASE CODE QZ”. This version was distributed at the 5th MELCOR User’s Workshop, May 10-15, 2001 in Bethesda, MD, USA.

The input manuals used here were printed from a CD-ROM handed out at the workshop.

4. BASIS OF INFORMATION USED FOR THE RINGHALS 3 MODEL

The main basis for the input data is the FSAR of Ringhals 3 [Ref. 4] of 1996-01-01 with a few later updates. Control systems and setpoint data are based on the PLS P-1 (Precautions, Limitations and Setpoints for Nuclear Steam Supply and Balance of Plant Systems), dated 2003-06-25 [Ref. 5].

The intention was to make a relatively detailed and general input model, that can be used for a range of severe accident calculations. However, one has to keep in mind that increasing the number of nodes and elements in the model also increases the execution time for the MELCOR calculations.

A valuable source of data that was utilized in the modelling of the reactor primary and secondary systems was the RELAP5 input produced for SKI by Studsvik. This includes a detailed model both for the original design [Ref. 6] and a revised one [Ref. 7] after replacement of the steam generators. The base RELAP5 model comprises 457 volumes, 518 flow junctions and 438 heat structures. The RELAP5 data were checked and found to be in good agreement with FSAR. Because the modelling principles are different in RELAP5 and MELCOR data had to be converted, and for that purpose tables were made up of relevant RELAP5 data.

The preparation of the input was made strictly following the MELCOR user’s guide. Default or recommended values are employed as far as possible for parameters to which optional values are offered.

5. INPUT MODELLING

5.1 Layout of the Input Files

Since an input for a full reactor model is rather extensive it was found convenient to split it into smaller subfiles, each representing separate parts or systems. This facilitates the overview and makes changes and updating easier.

The input to MELGEN is organised in eight individual files, of which one is a smaller conduct and initialisation file, which calls and includes the other files containing the majority of the input data. The MELCOR input is written to an own file. Editing was done in MS Windows with “WordPad” and the files stored as pure “text”-files. The MELGEN files are:

1) R3init — Initialisation, control and transient description file.
2) R3vessel; — Control Volume (cv) and Flow Path (fl) input for the reactor vessel.
3) R3vessel-hs; — Heat Structure (hs) input for the vessel.
4) R3core; — Core description (COR) and Transfer Process (TP) input.
5) R3dch-rn; — Decay Heat (DCH) and RadioNuclide (RN) input.
6) R3coolsyst; — cv, fl and hs input for the three primary and secondary cooling loops including Pressurizer, Pressurizer Relief Tank, Steam Generators, Feed Water Supplies and Steam Lines.
7) R3safesyst; — Input for Safety Systems, such as pressure control of primary and secondary loops and Safety Injection.
8) R3contmnt; — cv, fl and hs input for the containment, as well as necessary input to the Burn model (BUR), the Containment Spray Package (SPR), the Cavity interaction (CAV) and the Fuel Dispersal (FDI) models.

The basic input parameters are given by the Control Volume, the Flow Path and the Heat Structure data which together describe the major reactor system. Additional data for the reactor core are specified in the COR Package input.

The nodalization is shown in Figures 5 through 9 for the reactor vessel, core, cooling systems, primary and secondary sides, and for the containment. The MELCOR model for Ringhals 3 comprises totally 117 control volumes, 180 flow paths and 161 heat structures. The different input records ”cards” are characterized by a MELCOR package identifier (e.g. cv for the Control Volume package), followed by a component number and a record type identifier. Component numbers are assigned three digits for cv and fl and 5 digits for hs. The following series of numbers were chosen for the different parts of the reactor system:

001 - 099 Reactor vessel
101 - 149 Loop 1 of primary system

150 – 199	Loop 1 of secondary system
201 - 249	Loop 2 of primary system
250 – 299	” secondary system
301 - 349	Loop 3 of primary system
350 – 399	” secondary system
400 – 499	Pressurizer with spray and level control systems, primary system pressure control and relief tank
500 – 599	Main, common steam line, turbine simulation and feed water
600 – 699	Accumulators and Emergency Cooling Systems
700 – 799	Containment
800 – 899	MVSS Scrubber
900 – 999	Environment

The control volumes are specified as volume versus elevation. Only the free volume that can hold fluid is taken into account. MELCOR volumes have no actual length like a pipe component in RELAP5, which is one-dimensional code where wall friction pressure drop is attributed to the volume component. In MELCOR the total flow resistance pressure drop is instead based on the flow path data. No fluid volume is associated with the flow paths (similar to the junctions in RELAP5).

All elevations must refer to a common reference point, which in the MELCOR model was chosen to be at the bottom, inside of the reactor vessel. The elevations in FSAR refer to another level, probably at the sea level. In the RELAP5 model level zero was set at the centre of the vessel hot and cold leg outlet nozzles. This gives the following relation for the elevation coordinate z:

$$z(\text{MELCOR}) = z(\text{FSAR}) - 97.070 \text{ m} = z(\text{RELAP5}) + 8.093 \text{ m}$$

Main input data for control volumes, flow paths and heat structures are listed in Tables 1, 2 and 3, respectively. Table 4 presents a list of all Control Functions and Table 5 a list of the Table Functions.

5.2 Reactor Vessel Control Volumes and Flow Paths - File: ”R3vessel”

This file specifies input to the Control Volume Hydrodynamics and to the Flow Path packages for the reactor vessel.

Figure 5 shows the nodalization scheme of the reactor vessel. The total number of control volumes is 37 and the number of flow paths is 65. The total free volume in the vessel, i. e. the sum of all cv volumes, was checked to be correct. Table 1a and Table 2a show a list of the vessel control volumes and the flow paths. The following regions are identified:

<u>Downcomer</u>: The region between vessel inside and core barrel, from cold loop inlet nozzles to Lower Plenum (LP). The flow area is reduced at the core region by the space occupied by the neutron shields. The upper part of the downcomer (DC) was divided into three azimutal, 120 degree control volumes, each containing an inlet of a cold loop 1 to 3. Cross flow paths were established between the three upper DC parts. The lower part of

the DC from the elevation at the middle of the core and down to the LP forms one annular volume.

Lower Plenum: This goes from the lower head inside (elevation 0,0 m) up to the lower core plate and is axially divided into three control volumes. The lowermost part extends to the lower support plate (and lower end of the core barrel) and is modelled as one control volume, taking into account the varying flow area in half-spherically shaped lower head. The next control volume consists of the thick lower support plate, with a substantial volume, but most of it is steel and the free volume is only about 15%. The third cv constitutes the space between the lower support plate and the lower core plate.

The core region including the bypass channel is contained inside the core barrel and limited by the lower and upper core plates. Below and above the active core there are the inactive inlet and outlet parts, respectively. The core is divided into 4 radial so called “rings” and the active part into 4 equally long axial parts. Including the outer bypass between core barrel and baffle there are thus five parallel flow channels. The free flow area in the active region inside the core baffles was derived from the area inside the baffles minus the area occupied by fuel rods and minus the area of the CRD and instrument tubes. The total number of fuel assemblies, 157, was distributed into the four rings (see Fig. 2) as follows, and the flow area of the core rings was proportioned to the number of assemblies in each ring as:

Ring 1, centre	25 assemblies	15.92%
Ring 2	32 “	20.38%
Ring 3	44 “	28.03%
Ring 4, outermost	56	35.67%

The core cells are connected to each other through flow paths. In addition too the axial flow paths there are cross flow paths radially between control volumes at the same level. The opening height is set equal to the height of the cell and the flow path length equal to twice the assembly width from ring 1 to 2 and one assembly width from ring 2 to3 and from ring 3 to 4. The hydraulic diameter becomes equal to twice the gap width between the fuel pins.

The bypass channel flow area comprises the space between core barrel and baffle plus the flow path area inside the guide and instrument tubes.

Further details about the core is given in the section 5.4 describing the core package input.

The Upper Plenum (UP) region goes from the core exit up to, and including, the upper head. Its four axial parts correspond to the volumes in the RELAP5 model. The lowermost volume (cv60) includes the upper core plate. The next volume (cv70) is the space facing the outlet nozzles to the hot legs. Part three (cv80) ends below the upper support plate next to the flanges of the vessel, and part four consists of the upper head region. Parallel to cv 80 there is a bypass volume (cv85) above the inlet nozzles with openings to the DC, upper head and to cv 70 (Figure 5).

5.3 Heat Structures in Reactor Vessel - File: ”R3vessel-hs”

Solid materials are defined as “heat structures” and these data are specified on the hs-records. Heat structures are boundary constructions, which enclose fluids, like a pipe wall, or can be an internal component that either separates two control volumes, or is contained in a volume. Energy is transported through a heat structure if temperatures differ on in- and outside, and a heat structure acts as an energy sink or a source during a transient. Crucial parameters are thus, besides the location of an hs, the mass, material properties, surface area and heat transfer conditions. Convective Heat Transfer Coefficients (HTCs) are derived on the basis of input geometry, surface shape, orientation and boundary conditions using correlations described in the reference manual. In the input internal ‘INT’ or external ‘EXT’* flow has to be defined which affects the HTC correlations. HTC values can also be given explicitly, or specified by means of Control or Table Functions.

There are at present 17 default solid materials in the MELCOR Material Properties (MP) package which also has data for coolant and non-condensable gases. For these only the material mnemonic name has to be given. Additional materials can be specified with input of necessary properties, e. g. density, thermal conductivity and specific heat as functions of temperature.

Core structures in the vessel are specified separately in the Core Package input. There is, however, a coupling between core radial and axial boundary structures and the core. This is established by specifying the related core cell number on card hsccccc004 for the left-hand (inside) boundary fluid temperature option. There are two other options to obtain the boundary bulk fluid temperature, one is to have it calculated by the CVH Package, the other is to specify the temperature by means of a control function.

39 heat structures, listed in Table 3a, are specified for the vessel. The geometry of a structure is given by the type, which can be “rectangular”, “cylindrical”, “spherical” or “hemispherical” (top facing up or down). Other geometry data are the wall thickness and surface area. For cylindrical and spherical shapes this is given as inner and outer radii. The wall thickness is divided into a number of slabs, or mesh intervals, specified by the number of nodes, which has to be chosen with respect to the wall thickness and thermal diffusivity.

The orientation of a heat structure can be horizontal, vertical, or inclined at a certain angle. Careful attention has to be paid so that the structure is located within the boundaries of related control volumes, and this can be tricky to attain for inclined surfaces. Problems were met using the hemispherical geometry for e. g. vessel upper and lower heads, since they are not full half-spheres but somewhat truncated. To take this into account the multiplicity factor on record hsccccc003 was set to < 1.0. To manage this the location of the centre of the sphere had to be adjusted.

<u>The reactor vessel</u> is the major structure with a large mass. Measurements are taken from a FSAR drawing. The wall thickness varies and is larger at the cylindrical part than in the

* Generally, alphabetic quantities can be written in either lower or upper case, For “INT” and “EXT”, however, only upper case is accepted.

bottom. The flange part constitutes an additional substantial fraction of the vessel mass. The material is carbon steel with a stainless steel cladding on the inside. All other metal structures outside the core are specified as made of stainless steel.

The core barrel with the lower and upper support plates form an interior vessel enclosing the core bypass and the lower parts of the upper plenum. Since a MELCOR heat structure can have only two boundaries, a left and a right, structures covering several control volumes have to be split to fit the limits of neighbouring volumes.

The core baffle with the lower and upper core plates constitutes another internal container enclosing the actual core. The presence of two types of formers in the bypass region has to be accounted for. The core baffle is the radial boundary structure of the core. It is subdivided axially so that there is one heat structure per axial core level.

Internal structures outside the core include various components as control assembly guide tubes, instrument tubes, supporting structures, etc. These are considered to have an iso-thermal outer (right hand) boundary since they are wholly inside a volume. The neutron shields located in the downcomer region at the core level are other internal structures.

Heat transfer from the outside of the vessel, which is insulated, is set by a Table Function, which specifies an estimated heat transfer coefficient versus temperature on the outer surface. This gives the heat losses to the surrounding containment atmosphere.

Radiation heat transfer is supposed to be important for structures near the core where temperatures might be very high at core heat-up. Additional boundary input on cards hsccccc401 and –601 were therefore given for surface radiation data for those structures. An emissivity factor of 0.25 and the option ‘EQUIV-BAND’ was chosen for all surfaces.

Additional structure-to-structure radiation data were specified on the “hsrd” cards for the core boundary structures, according to the manual, page HS-UG-36, section 2.1.4. This input is also related on page COR-UG-70 in section 2.9.3 of the manual for the core package. It says there that melting of core outer ring boundary structures is modelled by the hsrd input, and “As any such structure melts, the molten steel will be added to core debris in the adjoining core cell.” *

* However, this input was said “ not recognised” by MELGEN.

5.4 Core Model - File: ”R3core”

The R3core file comprises input to the Core (COR) and Transfer Process (TP) packages.

The MELCOR Core package calculates the thermal response of the core and lower plenum internal structures, including the portion of the lower head directly below the core. The package also models the relocation of core and lower plenum structural materials during melting, slumping, and debris formation, including failure of the reactor vessel and ejection of debris into the reactor cavity.

The region in the vessel that is modelled in COR is bounded radially by the core baffle, and axially by the lower head at the bottom and by the upper core plate at the top. This region is here divided into four concentric radial rings, like the cv division, and into fifteen axial levels as shown by the nodalization scheme in Fig.6 (The number of rings can be chosen between 1 to 9 and number of levels from 1 to 99). Radial rings are numbered from centre and out and axial levels from the bottom and up. A particular radial ring and an axial level define a core cell. A three-digit number gives the cell number; the first digit represents the radial ring number and the last two digits the axial level number.

Each core cell is coupled to the thermal-hydraulics of the control volume in which it is contained. Fig.6, illustrates the relation between control volumes (to the left of the centre line) and the core cells (to the right). The height of a core cell is half of that of a control volume in the active core region, so there are twice as many core cells as control volumes. The radial rings correspond to the radial division used for the control volumes, and the distribution of fuel assemblies to the rings is illustrated by Fig. 2 below.

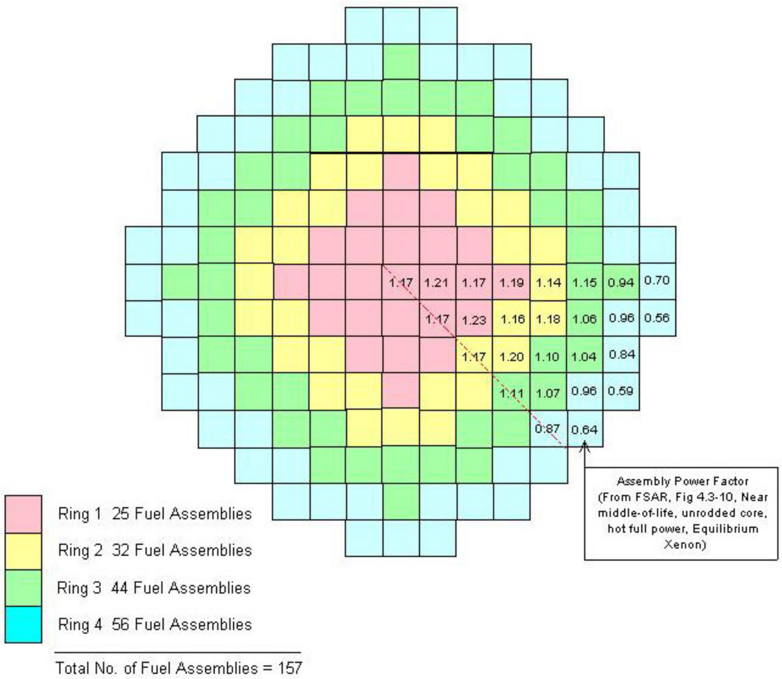

Figure 2. Cross section of core showing distribution of fuel assemblies into the radial rings and assembly power factors for 1/8 symmetry.

The input to the COR package defines the geometry, location and masses of the various components in the core, as well as boundary conditions, power distribution etc. The components being considered for a PWR in MELCOR 1.8.5 are:

- Fuel pellets
- cladding
- supporting structures
- non-supporting structures

The headings in the COR user guide gives the following blocks of input data to MELGEN:

1 General Core/Lower Plenum Input
2 Axial Level Input
3 Radial Ring Input
4 Specific Cell Input
5 Lower Head Input
6 COR Material Input
7 CVH Fluid Flow Interface Input

5.4.1 General Core/Lower Plenum Input

General data comprise, among others, number of rings, axial levels and nodes, fuel data and lower head failure parameters. Core and fuel rod data were taken from FSAR, Table 4.3-1, aware that these are from an earlier fuel cycle. Pertinent input data are:

- Number of fuel assemblies
- Rod array
- Rods per assembly
- Number of guide thimbles per assembly
- Number of grids per assembly
- Fuel rod outer (clad) diameter
- Pitch
- Fuel pellet diameter
- Thickness of gap
- Fuel weight
- Zircaloy (Zr-4) weight

Default values were used throughout for meltdown and transport of core material and for lower head failure parameters. Three different penetrations in lower head were modelled.

5.4.2 Axial Level Input

Axial level input includes elevation* and mesh for the 15 axial cell levels, radial boundary heat structure number and axial power distribution.

Many decimals in the input for the volumes of the core cv-s are sometimes required in order to avoid “Consistency errors”, which otherwise showed up in the MELGEN .DIA file.

The axial power distribution was taken from FSAR Fig. 4.3-17, the curve for the case "average, removed from control rods" (unrodded). The step function fitted to the curve for the eight active, axial core levels is shown in Fig. 3.

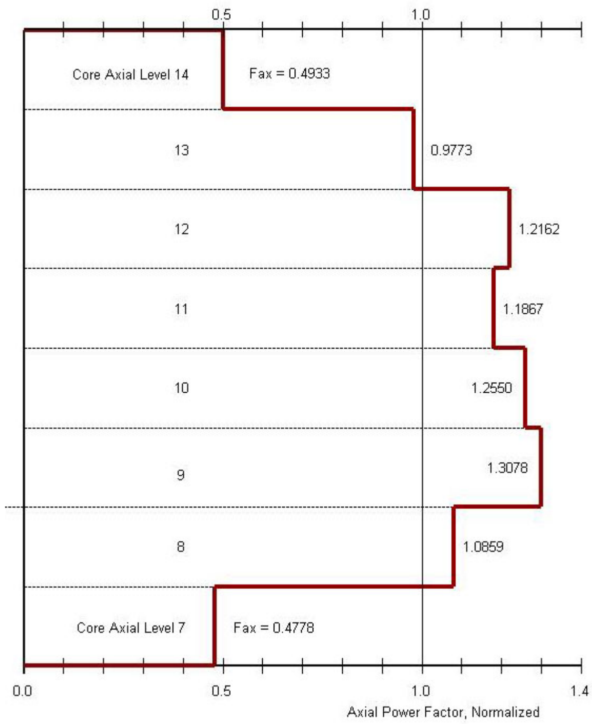

Figure 3. Axial power distribution used in the MELGEN COR input.

5.4.3 Radial Ring Input

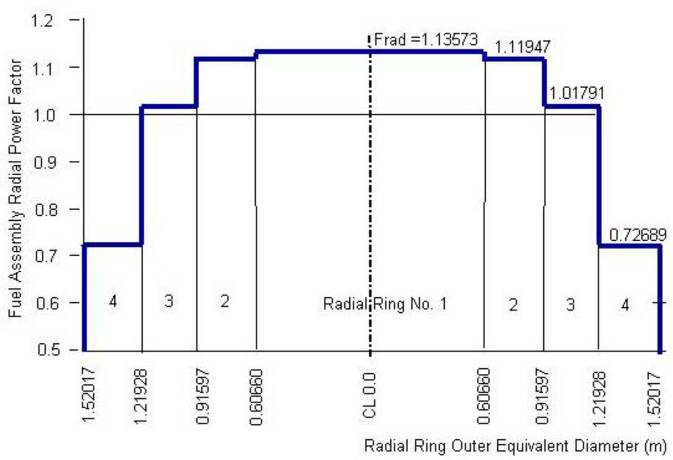

Figure 4. Radial power distribution per assembly in the radial rings, normalized.

The radial ring input comprise three sets of data; radial ring cross sectional area, upper boundary heat structure specification and relative power per radial ring, per unit mass.

The core diameter was obtained from FSAR, which gives the cross sectional area. The area per ring is proportioned to the number of assemblies (25, 32, 44 and 56 pcs. in ring 1, 2, 3, and 4, respectively). This area is the gross area and includes flow area and space occupied by solid fuel components.

The radial power factors, shown in Fig 4 above, are taken from FSAR Fig. 4.3-10, assuming full 1/8 symmetry with assembly power factors shown before in Fig. 2. These values have to be adapted to actual fuel conditions in other cases.

There are six more input records for the radial rings, by which values of some failure criteria can be modified from the default values set by the global input. These records were not entered.

5.4.4 Specific Cell Input

The following input parameters were employed for this group:

- Cell reference and fluid boundary control volumes
- Cell component masses
- Additional cell component masses
- Core initial temperatures
- Equivalent diameters
- Cell boundaries and flow areas
- Surface areas

This input block is the most comprehensive as it specifies detailed data for each of the 60 cells in the core. Here, the masses of all components as fuel, cladding, spacers, control rod material and all other materials within the core boundary are described. The recommended option to specify structures as “Supporting” (SS) and Non-Supporting” (NS) was adopted, and not the old option to classify structures as “Other Structures” (OS), which is retained to comply with MELCOR versions 1.8.4, and earlier. Data were taken from chapter 4 in FSAR and from the RELAP5 input [Ref.6].

The comments in the input file “R3core” are rather detailed concerning the origin of data and distribution of component masses, and will therefore not be repeated here.

5.4.5 Lower Head Input

Two kinds of records are given here, and include radii of the lower head rings with related cavity control volume number, and specification of lower head penetrations. Three penetrations, one in each one of the inner rings were included with preliminary data taken from a Surry file “BREAK” distributed at the 2001 MELCOR workshop.

The nodalization record is left out, which implies that the nodes given in the heat structure input are used (default).

It should be noted that the COR Package only considers discharge of core debris and melt if a break occurs in the lower head, and that it does not allow for outflow of coolant. A special flow path has therefore to be added for the CVH Package. This was here modelled as a valve flow path, initially closed and with the opening fraction determined by a control function based on the COR variable “COR-ABRCH”. The latter is the flow area of the lower head breach, in m^2, according to the manual.

5.4.6 COR Material Input

No input was given here since the default materials in the Materials Properties Package are adequate for the Ringhals 3 core.

5.4.7 Transfer Process Package Input

The Transfer Process, TP package provides an interface for a physics package to transfer mass and energy to other packages. The need for the TP package seems be as a temporary tool to handle imperfect interfaces between packages in present versions of MELCOR. In current case it is prescribed to be used by the COR, CAV, FDI and RN packages to facilitate transfer of radioactive mass from the core into the cavity.

The input is given as matrices for “In” and “Out” transfer, here to/from core and FDI (Fuel Dispersal package) and to/from FDI and cavity. The “In” transfer process number (011) for the core was input on card cor00004 in the COR package, and as “Out” transfer process number on FDI card fdinn00 for transfer from core to FDI (The FDI package input is located last in the file “R3contmnt”). The “OUT2 TP No. from FDI to cavity is given on record tpotnn00.

The options of input to the TP package are restricted to a few combinations of data sets. The input here was built from the examples in the manual, pages TP-UG-12—16. Note, that the transfer process numbers for radionuclide transfer must be COR and CAV numbers +500.

5.5 Decay Heat (DCH) and RadioNuclide (RN) Input - File: “R3dch-rn”

5.5.1 DCH Input

Input data specifies here the type of reactor, and give the basis for decay power of radioactive fission products after shut down. A scram trip number has to be specified. The standard ANS decay curve was chosen, the other option being ‘ORIGEN’. ANS means here ANSI/ANS-5.1-1979, which can be considered as conservative.

The total initial reactor power is given as a sum of power from thermal fission of U^{235} and Pu^{239} plus fast fission of U^{238}. The default values are for a PWR with a total thermal power of 3412 MW, and each of the contributing three fractions are here proportioned to the actual power in R3, 2775 MW_{th}.

There are built-in decay heat tables in MELCOR for a large number of elements, however not for CsI, which is present in the RN input. A new class, No. 16 has therefore to be defined on the “dchclsnnnm” records. The decay heat power as a function of time must then also be specified, which is made on the “dchnemnnmm” records.

5.5.2 RN Input

The RN package has models for calculation of radionuclide release from the fuel and debris, of transport, deposition and removal by engineered safety systems of radionuclides in the various reactor systems. The RN package operates on the basis of material classes, which are groups of elements that have similar chemical properties. Notice must be taken to the fact that some models in MELCOR use other grouping of elements, which is the case for VANESA. In these cases special mappings for transfer between the classes might be needed. The following groups of data were entered according to the manual (RN-UG section No. in brackets)

General control, Options and Mapping (3.1.1); Default values are employed as far as possible for definition of material classes. The number of sections (size bins) was, however decreased from 10 (default) to 5. The default number of classes is 16*, including CsI. The hygroscopic model is turned on. The default transfer fractions for mapping of non-radioactive masses to RN material classes is applied. Mapping of the new RN class 16 for CsI to a VANESA class must, however, be specified.

Initial Radionuclide Inventories (3.1.2) is specified for the fuel only, and must be input for each active core cell. Axial variation is proportional to the axial power distribution, normalised to 1.0 for each radial ring. The radial ring node multiplier is the normalised radial power distribution.

Release Model Parameters (3.1.3); The default CORSOR-M model with surface-to-volume ratio was selected, and the default for gap release temperature (1173 K). There are five additional CORSOR models to choose among. A new acceptor class (#16) had to be defined for CsI.

Aerosol Modelling Parameters (3.1.4); Default values were used for the aerosol size and density parameters. Only two other input groups were applied, these are for redefinition of vertical surfaces to horizontal (rndsxxx records), and for definition of intervolume settling (rnsetxxx). The RN models require all control volume to have at least one horizontal surface. Dummy heat structures must therefore be added in some control volumes. The rnsetxxx records facilitate settling of aerosols from volume to volume without bulk flow, which is in addition to, and independent of transport by intervolume flow. Preliminary “rnset” input data were here applied for vessel volumes only as a first step.

Aerosol Condensation Index (3.1.5); Default “ICOND” = 0 lets condensation take place on all aerosols.

Decay Heat Distribution (3.1.6); No redistribution applied.

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* In the manual on p. RN-UG-7 it says that the default is 16, while on p. RN-UG-15 the default is 15?

ESF Parameters (3.1.7); The Engineered Safety Features (ESF) applied here are the pool scrubbing in the PRT and MVSS scrubber, and the containment spray system. The pool-scrubbing model facilitates wash-out of aerosol and/or Iodine vapour provided that the SPARC model is activated in the flow path input. RN input parameters describe the vent pipe geometry. The containment spray input defines a spray partition coefficient, here set to 5000.0, which is the ratio of concentration of Iodine in the liquid droplets to that in the gas under equilibrium conditions.

Radionuclide Chemistry (3.1.8): No chemical reactions were defined now.

Chemisorption (3.1.9); Activated by letting ICAON = 1.

Iodine Pool Model (3.1.10); Not activated here (default).

5.6 Reactor Cooling Systems - File: R3coolsyst"

This file comprises control volume, flow path and heat structure input for the three coolant loops, both primary and secondary sides. It includes the Pressurizer (PRZ), Pressurizer Relief Tank (PRT), Main Steam Lines (MSL) and a turbine simulation model. Auxiliary systems like the Chemical and Control Volume System (CVCS), the Residual Heat Removal (RHR) system and the Component Cooling System (CCS) are not modelled. Only a few functions of the CVCS as the letdown flow from loop 3 and the charging flow to loop 1, and the RHR system related to safety injection are taken into account.

Main input data for all components of each loop for control volumes, flow paths and heat structures, respectively, were compiled as listed in Table 1b-d, 2b-d and 3b-d. Data were mainly based on the RELAP5 input [Refs. 6 and 7]. The reactor vessel, described before, belongs also to the primary cooling system. The sum of the volumes in reactor vessel, loop 1 – 3, surge line, PRZ and spray lines of the primary system was checked against available information.

In the secondary system the volume from and including the steam generators to the Main Steam Isolation Valves (MSIV) is the crucial fraction in cases when the MSIVs are closed initially in the transient. The largest volume is that of the secondary side of the SGs. Since the main steam lines have different lengths for loops 1, 2 and 3, the total volumes up to the MSIV differ somewhat.

The flow path length is generally set equal to the distance in the flow direction between the centres of adjoining control volumes. The sum of the segment lengths is meant to reflect the total pipe and component length, so that a correct frictional pressure drop is obtained along a loop. For each of loop 1, 2 and 3 in the primary system the total flow path length is counted from the hot leg outlet nozzle in the vessel to the cold leg inlet nozzle. The input for the cooling loops is arranged in the following order:

- For each loop 1 to 3 - Primary side control volume and flow path input
  - Secondary side control volume and flow path input
  - Primary and secondary side heat structures

- Steam Generator feed water supply and level control
- PRZ in Loop #2 and PRT
- Common Steam Line Header and Turbine simulation

The nodalization scheme is shown in Fig. 7 for the primary side of loop 1 and 2. Fig. 8 shows the nodalization of one loop of the secondary system. Piping and main components are identical for all three loops. However, some connecting systems and components differ in the primary system:

Loop 1; one of two tappings to the spray line and the inlet of the CVCS charging flow water is located in the pipe downstream the RCP.

Loop 2; PRZ surge line connects to the hot leg, and the other one of the two tappings to the spray line is located in the cold leg like loop 1.

Loop 3; Let-down to the CVCS from the bottom of the cross-over leg.

Differences on the secondary side concern the lengths of the main steam lines from the steam generators to the steam line header.

5.6.1 The Primary System Coolant Loop

This comprises the hot leg, steam generator in- and outlet plena, SG tube bundle, cross-over leg, RC pump and cold leg. Pipe dimensions were taken from FSAR:

The heat structures of the heat exchanger tubes are modelled for one U-tube, using the multiplicity option (HSMULT). New material data were entered for the wall metal, Inconel Alloy 690tt as Table Functions which are placed last in this file.

Heat losses to the containment are taken into account for components with large surface areas, i. e. the SGs and the PRZ. The same heat transfer coefficient, given by Table Function TF710 as a function of the outer wall temperature, is applied as for the reactor vessel. The heat structures of pipe walls are considered as perfectly insulated, i. e. at isothermal conditions and heat losses to the environment (the containment) are neglected.

The RCP pump characteristics is simply modelled by the ‘QUICK-CF’ control function option for the flow junctions. The pump pressure head is then specified by means of a control function.

5.6.2 Secondary System with Steam Generators and Main Steam Lines

Steam generator data comply with the design of the new Siemens/KWU SGs and input is based on RELAP5 data and FSAR. The secondary side is divided into four control volumes shown in Fig. 8. There is no steam separator model in MELCOR like in RELAP5. The “pool/atmosphere” approach in MELCOR is supposed to give an adequate simulation of the separator function provided that flow path arrangements are correct. Here the “pool-first” option was chosen for fl150 and fl151 representing fall-back lines from separators and dryers.

The downcomer shell is cylindrical with a conical transition to the dome region. The total external height includes the outlet nozzle. Although well insulated, wall heat losses might be noticeable since the total outer surface area is large for the surface above the tube plate level. A heat transfer coefficient according to Table Function TF710, as mentioned above, was applied to allow heat to be dissipated into the SG cubicles in the containment.

The Main Steam Lines goes from the SGs to the Steam Header. The placement of the steam generators results in different pipe lengths and different pipe volumes. The values were obtained from the RELAP5 input:

Each main steam line is equipped with relief and safety valves which are modelled by flow paths with valve control functions both given in the "R3safesyst" file.

5.6.3 Feed Water Supply and SG Level Control

The main feed water supply to the downcomer is modelled by means of a source added to the DC control volume. The flow as a function of the deviation from a nominal water level in the DC is governed by Control Functions. The level value for current operating conditions was obtained from [Ref. 2]. Feed water closes at reactor trip. Auxiliary feed water is not modelled separately.

5.6.4 Pressurizer and Pressurizer Relief Tank, PRZ Pressure and Level Control

The PRZ is connected by the surge line to the hot leg of loop2 as shown in Fig. 7. The surge line is vertical, but is partly inclined at an angle of 24 to 30 degrees. The PRZ consists of a vertical cylinder with half-spherical ends.

The Power Operating Valves (PORV) and the Safety Valves provide overpressure protection, and related input data are in the "R3safesyst" file. The valves are installed in a manifold valve pipe system connected to the PRZ. The totally six smaller pipes with the valves merge into a larger pipe that leads to the PRT located lower down in the containment at the reactor vessel elevation. The pipe exits under water in the PRT which is at atmospheric pressure. The system acts as a condensation pool, so the SPARC pool-scrubbing model in MELCOR is activated. To prevent overpressurization the PRT has a rupture disk opening to the containment.

Water spray nozzles at the top and electrical heaters in the lower part of the PRZ facilitate the pressure control. Spray water is taken from the cold legs of loop 1 and 2. The flow rate is controlled by a Control Function and is a function of the water level. Normal level gives a ratio of about 60% water and 40% steam in the PRZ. If the pressure goes below a certain set point the electrical heaters are turned on. There are two kinds of heaters, proportional heaters with a medium power, and back up heaters that can add power to a larger total rate. Heating power as a function of pressure is regulated by Control Functions.

A letdown flow and charging flow are used for volume control in the primary system and are in the MELCOR model based on the water level in the PRZ. These flows are simulated by means of the CVH “source” function. Water letdown is taken from bottom of the cross-over leg in loop 3 and the charging flow is added to the cold leg in loop 1. Water letdown and power to the PRZ heaters are shut off if the PRZ water level is low,. The maximum values of the charging flow and the letdown flow were taken from the PLS.

5.6.5 Steam Header, Turbine Steam Line and Turbine

A very simplified simulation is made of the steam exit to the turbine only to serve as a heat sink at steady-state and to be isolated at the start of transient calculations. Steam header data are from the RELAP5 file for the control volume. The turbine is simulated by a time-independent volume at atmospheric pressure, and the steam line between header and turbine as a flow path with a valve that is shut after trip. The valve area is adjusted to get the nominal steam flow to the turbines at steady-state operating power, 2775 MW.

5.7 Reactor Containment - File: ”R3contmnt”

The file R3contmnt includes cv, fl and hs input for the reactor containment and, in addition, input to the Containment Spray (SPR), the Burn (BUR), the Cavity (CAV) and the Fuel Dispersal (FDI) packages.

5.7.1 Containment cv, fl and hs Input

The reactor containment consists of large cylindrical building of prestressed concrete with a half-spherical dome. About 1/5 of the height of the building is below ground level. The containment is connected through discharge pipes to the MVSS (Multi-Ventury Scrubber System) located on the ground, outside of the containment building. The pipes are normally blocked by valves and a rupture disk, which are designed to open at a certain overpressure in the containment.

The lower part of the containment comprises a large number of compartments and structures. Much simplification has therefore to be done in the modelling. The nodalization includes 10 rooms, or control volumes and is shown in Fig. 9. In MELCOR one volume has to be defined as the cavity to interact with COR, and FDI models. The MVSS scrubber and the environment are additional, external control volumes. The containment model has 23 flow paths and 32 heat structures. Main input data for cv, fl and hs components are listed in Tables 1g, 2f and 3f, respectively.

The model is based on measurements on the drawings in FSAR, Fig. B.2-1 to B.2-13 showing cross sections and longitudinal sections of the containment and on information from chapter 3.8 and 9 (PMR). Some information was also obtained from part of an input file from SwedPower [Ref. 8] that was planned for an earlier Ringhals 3 model.

The cavity comprises the lowermost central part under the reactor vessel and the annular gap around the vessel up to the sealing at the flange against the refueling pool. The large space around the cavity consists of a number of compartments, that are combined into two annular rooms in the model, the Lower Containment and the Annular Containment. The two rooms are separated at “FSAR level” +100 m (MELCOR level +2.93 m) by the floor under the PRT and above which the RCPs and the SGs stand. The SG and the PRZ compartments form partly annular spaces around the components and are modelled by four individual rooms. The refueling pool is another room, actually consisting of two parts. One is the basin, with a certain water depth, reaching to the outer shell, and the other part is the space above the vessel.

The dome volume is the major part of the containment and is divided into two control volumes at the level just above the main steam lines. In the uppermost dome volume resides the polar crane and at the very top is the containment spray piping located.

Several flow paths with varying flow areas connect the volumes. For the large open areas, such as between Lower and Annular Containment, and between the Reactor Hall and the Dome, two parallel flow paths were defined in order to facilitate natural circulation. Some other paths are small, or even closed, but can be opened or enlarged if subjected to large pressure differences.

The heat structures comprise concrete walls and floors which separate the compartments, supporting structures and various system components, cables, the fuel handling machine the polar crane, etc. A substantial volume of the interior is occupied by the reactor system components. The outer boundary of the containment consists of the thick basement floor, and the containment shell and dome walls. A steel plate covered by concrete in the floor and being embedded in the cylindrical concrete shell seals the whole containment shell. In the cupola of the dome the steel plate is on the inside, uncovered. The thickness of interior walls and floors vary along the height. Material properties in the MELCOR MP package were used for concrete, carbon steel and stainless steel.

A roughly estimated heat transfer coefficient defined by Table Function 710, the same as for the reactor vessel, was applied on the outer wall above ground level. It will give a heat loss to the environment of about 120 kW at a temperature difference of 20 K.

The MVSS scrubber is placed outside the containment. It is assumed to be filled to 70 % of its height with water. Two parallel pipes which join into one common pipe lead from the containment to the scrubber. The discharge pipe system in the MVSS is simply modelled as one outlet pipe opening 5 m below the water level. The SPARC pool scrubbing function in MELCOR is activated.

Only the pipe from the containment to the MVSS containing the rupture disk, i. e. one of the parallel pipes is modelled (The other pipe has valves that can be opened manually). Breaking of the rupture disk is simulated by Control Functions. In the exit from the scrubber to the atmosphere there is another rupture disk which breaks at a low opening pressure. It shall prevent the nitrogen atmosphere in the scrubber from leaking out.

Containment leakage with an option to simulate break of the containment is modelled by means of a flow path and control functions. The initial leak area is put equal to 1 cm^2.

5.7.2 Burn (BUR) Package Input

The only input needed here is the single parameter to activate the BUR package. This will then apply MELCOR default values, such as ignition parameters, etc., in all calculations.

5.7.3 Containment Spray (SPR) Package Input

The first input record specifies the number of sprays (here one), its location and a Control Function to control the volumetric flow. Droplet initial size and distribution are also required input, but since the RN ESF is active (input on the rn2spr01 record, see section 5.5.2 above) only one size is recommended, according to the manual.

Other input parameters are for definition of flow junctions for carry over of droplets to other volumes, and definition of a sump volume. Here only one junction, between the upper Dome and the Reactor Hall is specified. The Lower Containment was chosen as the sump. Normal transport of liquid between control volumes should be handled by the CVH and FL packages.

5.7.4 Cavity (CAV) Package Input

Input defines here the cavity control volume (cv701) and activates the RN pool scrubbing function. The type of concrete was chosen as “BASALT”. Non-standard types of concrete can be specified through input of various compounds and their properties.

The geometry of the cavity is described using a coordinate system specific for CORCON which is separate from the system in other parts of MELCOR. A Transfer Function number must be given for transport to the FDI package.

5.7.5 Fuel Dispersal Interactions (FDI) Input

The FDI package models both the low-pressure molten fuel ejection from the reactor vessel into the cavity and high-pressure molten fuel ejection from the vessel with the possibility of dispersion of the debris over multiple containment volumes and surfaces.

Input is given only for the low-pressure model at this stage. Input comprises two records, the first specifies the cavity control volume, “In”- and “Out” Transfer Process number and the second record gives the bottom and top elevations of the interaction region.

5.8 Safety Systems - File: ”R3safesyst”

The following protection and safety systems are taken into account:

- Pressurizer Power Operated Relief Valves (PORVs)
- Pressurizer Safety Valves
- Accumulators
- HPSI and LHSI systems
- Steam Lines Isolation Valves (MSIV)

- Steam Lines Safety Valves
- Steam Lines Relief Valve

5.8.1 PORV Modelling

There are three power-operated valves, PORVs, which limit the pressure in the primary system through steam relief from the PRZ. The PORV function is modelled by a valve flow junction and the "HYST" option in the related Control Function. The valve flow area is set to sum of the three PORVs and the opening –closing pressure range is set according to the PLS data.

The PORVs are assumed to be able to operate on battery back-up as long as 2 hours after loss of AC power.

5.8.2 PRZ Safety Valves including PRT rupture disk

The PRZ safety valves are modelled similar to the PORVs, but this is a passive, pressure only controlled system. The flow area of the three valves is here almost twice as large as in the PORVs and the opening - closing pressures are higher.

Both the PORVs and Safety Valves blow to the Pressurizer Relief Tank, PRT, through a common larger pipe submerged under water in the PRT. The SPARC pool-scrubbing model is activated for the flow paths. The PRT is equipped with a rupture disk opening into the Annular Containment volume. The break pressure difference over the disk is taken from the FSAR and the flow area chosen to give a mass flow rate specified there .

5.8.3 Accumulators

The accumulators are connected by a line with a check valve to the cold legs downstream the RCP in each loop as shown in Fig. 7. Each accumulator tank holds a certain supply of cold water and is pressurized by a nitrogen atmosphere. This is a passive system that begins to operate as soon the pressure in the primary system goes below the accumulator pressure. A control function is used to stop the flow in order to avoid that nitrogen enters the primary system through the outlet in the bottom of the accumulator tank just before it is empty.

5.8.4 HPSI and LHSI Systems

High Pressure Safety Injection (HPSI) injection is provided by flow from 3 centrifugal pumps and Low Head Safety Injection (LHSI) from 2 centrifugal pumps. The latter is also used for Residual Heat Removal (RHR) in recirculation mode. Both systems take water from the large Refueling Water Storage Tank (RWST). When low level is reached the system is switched manually to cold leg recirculation mode.

Injection mode is automatically initiated upon S-signal on following conditions (FSAR):

- Low PRZ pressure

- High containment pressure
- High differential pressure between any two steam lines
- High steamline flow coincident with either low-low T_{avg} or low compensated steamline pressure
- Manual actuation

Only the injection mode is simulated here, by adding mass and energy to the cold leg of each loop, into volumes cv111, cv211 and cv311, respectively. A check is made so that the SI ends when the RWST is empty.

5.8.5 Steam Lines Isolation Valves (MSIV)

There are two modes of operation, a normal slow and a fast one. Only the fast mode is considered here. At present the MSIVs in the model closes after scram only, controlled by Control and Table Functions.

5.8.6 Steam Lines Safety Valves

The main steam line safety valves blow to the environment at overpressure. A valve flow path simulates the valves. There are 6 valves in each loop with different set points with the range given in PLS. Control Functions and a Table Function are used to control the opening area.

5.8.7 Steam Lines Relief Valve

The Steam Line Power Operated Valve blow to the environment, as well. There is one valve in each loop, the capacity and opening pressure is given by the PLS. A valve flow path simulates the valve function and the opening fraction is governed by a “HYST” Control Function.

5.9 Initialisation and Executive Control File - File: ”R3init”

In this file are some auxiliary data blocks placed, such as the “CVTYPE” to define control volume types and the NCG (Non-Condensible Gas) input needed to assign material type numbers for gases other than steam. The file also contains various Control Functions to initialise the calculation of steady state conditions at nominal operating conditions to form the starting point for the postulated accident. Lastly, there is a block of data describing the initiating event and the transient conditions of the severe accident. Only temporary data for a sequence with total loss of power are input for the transient in current file, but these have to be modified and adapted for other cases.

The transient description includes options for simulation of RCP pump leakage after loss of power and a model for surge line creep rupture.

The steady-state calculation is run with MELCOR after an initial MELGEN run. Since the initial thermal-hydraulic conditions are input in MELGEN only as rough estimates, and the steady-state initialisation flag (ISS on record hsccccc00) is set to zero, letting the code calculate the heat structure temperature distribution, a smooth start of the calculation is necessary. A temporary, additional, pressurizer with a large volume was therefore connected to the top of the reactor vessel. The connection was slowly closed after 50 s. *

The start up sequence included a slow increase of the reactor power to 2775 MW controlled by the Control Function for the fission power defined in the COR file (-ICFFIS=CF011). Simultaneously the RCP flow was increased to the full nominal value, and the turbine valve opened following the power increase. This initialisation was specified using several Control and Table Functions. A negative start time was chosen for the steady-state period in order to let the transient begin at time zero.

The transient input for the transient calculation comprises another set of Control and Table Functions. These are used to specify initiating events, e. g. loss of external power, scram trip and other trips for MSIV closure, RCP coast down, etc. The fission power decay curve is taken from the Surry file in the Workshop 2001 material. As mentioned before, the input for the transient conditions should be revised depending on the case to be calculated.

Simulation of a surge line creep rupture was introduced to account for the possibility of the surge line pipe to be superheated from steam release through the PORVs or the PRZ safety valves. A table for creep rupture stress as a function of temperature was introduced based on data for stainless steel SIS 2333.

The Control and Table Functions are listed in Table 4 and 5.

5.10 MELCOR Input for Steady-State and Transient Calculation Runs - File: ”R3melcor”

Input to MELCOR can be limited to a few lines of instructions to the EXEC code package, mainly specifying time-step and editing data for the output files. The MELCOR calculations start from a MELGEN restart file, which contains the complete database and initial conditions from the input processing with MELGEN. Restart can also be made from any restart file written by MELCOR.

Choice of time steps needs certain consideration. A small maximum time step must be chosen at start of the calculation and at transient start even if the minimum time step is small. During slow progression of the accident longer time steps can be applied. The parameter “SOFDTMIN” allows reduction of the time step for a specified number of times steps during unforeseen fast changes during the transient, e. g. at fuel melting and slumping, vessel break through, etc.

Only a few data can be entered or changed in the MELCOR input and only for some of the model packages. Useful changes can be made of the Control Functions multiplicity

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* Another means of initialisation is to use the “CVHTENDINI” record for control volumes through which selected volumes can be defined as “time-independent” for a certain period of time. This gave, however, noticeably larger pressure oscillations than the other method and lead often to numerical problems.

and additional constants, e. g. in order to control valve opening fractions, etc. There are also a large number of sensitivity coefficients for most of the packages, by which various constants and coefficients for e. g. melting temperature, heat transfer, and many model parameters can be changed. Some of these coefficients are related to the numerical solution, as time step control and convergence criteria. The almost infinite possibilities to combine sensitivity coefficients require much experience of the user, and advice from the code developers might be necessary.

6. TESTS AND RESULTS FROM STEADY STATE RUNS

Test runs were carried out during the input preparation of those files which can be run separately after addition of certain boundary conditions. This was done for the vessel cv and fl input, for the coolant loop and the containment input files.

Input checks were then made with the complete input. Still, after debugging a few non-fatal errors or comments came up with the MELGEN .DIA output file as

- “record hsrd001240 read but not processed”
- “Warning from subroutine coricg. core geometry suggests iaxsup = 1. rather than value of 5 inferred from ntlp input on record cor00000”

The “hsrd” record is for surface-to-surface radiation of heat structures facing each other. The input parameter “ntlp” is the number of axial nodes in lower plenum in the COR input, the true value is here equal to 5. No further explanations were given by the code. Since the input seemed to strictly follow the manual, the comments were ignored for the present. (Which, was correct according to later information from Sandia).

Steady-State runs were made in order to tune the input to the operating conditions obtained from the Ringhals plant. Values apply to present operating power of 2775 MW_{th}, and are listed on p.2. Since the aim was to keep the default values as far as possible, the number of parameters available for tuning was limited. The following parameters were varied:

- Pressure drop in the loops by varying loss coefficients and surface roughness
- RCP pressure head together with the above to adjust loop mass flow rates
- Area in turbine valve to get the steam flow to agree with feed water flows
- Pressure control in PRZ, control functions for spray and heaters

Loss coefficients are generally based on RELAP5 data. The default value for the surface roughness, 5.0E-5 m was used throughout, but resulted in too large total frictional pressure drop in the primary loops. The vessel pressure drop, 2.60 bar agreed with plant data. A too large contribution came from friction losses in the steam generator tubes. After reducing the wall roughness there to 1.0E-6 m better agreement was achieved.

Tuning was done as an iterative process by “trial-and-error”. The CPU time for a run became rather long, due to the detailed model. Steady-state runs were limited to a real time of 1000 s, after which conditions were relatively stable. With additional number of runs the deviations from the station operating values might have been reduced. However, present runs are only tentative and the initial conditions might be different for coming transients to be calculated.

The graphs in Figure 10 to 16 show the course of some selected parameters for which station nominal data were available. The pressures in primary and secondary systems are shown in Figure 10 and 11, respectively. Although a large dummy steam volume was introduced especially for the initialisation period there are still considerable oscillations, but the final values seem to be reasonable.

The mass flow rates in the primary loop plotted in Figure 12 show good agreement between all three loops. The final flow rate is 0.2% high. The turbine steam mass flow in Figure 13 is slightly decreasing towards the nominal value.

Figure 14 shows the temperatures in cold and hot loops, i. e. in vessel inlet and outlet, respectively. The temperature difference reflects the reactor power, although the average temperature exceeds the T_{avg} set point by approximately 4 K. The main reason is probably a too low heat transfer in the steam generators. By default the heat transfer coefficient is calculated by the code, but it can be modified by user input. Another way to increase the heat transfer in the SG-s is to increase the heat conductivity of the Inconel tube wall material. The high temperature level can act as a conservative condition for the severe accident calculations.

Water levels in pressurizer and in the steam generator downcomers are shown in Figure 15 and 16. No reference operating value of the PRZ level was available, only for the SGs. The deviation between the individual loops in the calculation is due to deviating initial levels in the input that will be corrected. Since the deviation was still within the allowed span the control function for the level did not have any effect on the value.

The steady-state values should be regarded as preliminary. New runs will be carried out before each transient case to follow. The initial conditions might then be different and will be dependent on the severe accident to be calculated.

Preliminary runs of the two transient cases, which are planned for the hydrogen studies, indicated that the execution times with the full base model would be inconveniently long, especially for a 24-hour real time case. Therefore, reduction of the number of control volumes was made for these cases. The reduction was made of the volumes in the vessel input (Figure 5), in four steps giving the following revised input models:

Revision 1:

- Core inlet volume cv010, cv020, cv030 and cv040 combined with lowermost active core volumes in each ring. In consequence, flow junctions fl11, fl21, fl31, fl41, fl7, fl8 and fl9 are eliminated. The combination of control volumes has also effect on related heat structures and core, in files R3vessel-hs and R3core, respectively, which are modified.
- The three upper downcomer control volumes cv1, cv2 and cv3 are merged into a new annular volume cv1, similar to the lower DC. The cross flow junctions fl51, fl52 and fl53 are thus left out. The triple parallel flow junctions at the bottom, fl1, fl2 and fl3 are summed into a new fl1, and, similarly, at the top fl94, fl95 and fl96 summed into a new fl94.
- The inlet junctions from the cold legs fl114, fl214 and fl314 now all end into cv1.
- Related heat structures in vessel wall and core barrel in file R3vessel-hs are changed consequently.

<u>Revision 2:</u>

- Further decrease of vessel control volumes by including upper inactive volumes cv15, cv25, cv35 and cv45 in upper active volumes cv14 through cv44, respectively.
- Related changes made in files 'R3vessel-hs' 'R3core' and 'R3dch-rn'.

<u>Revision 3:</u>

- Shortest control volume in Rev2, cv8, (Lower Support Plate) merged with cv9 (Lower Plenum, uppermost part), both cv8 and cv9 are inside the core barrel. Shortest volumes will now be the middle core volumes, 0.9125 m.

<u>Revision 4:</u>

- All core control volumes merged into one single volume combining all 24 original core cv-s. This eliminates all internal core flow paths, axial as well as cross flow paths. The new core control volume is now called cv010 with inlet flow path fl010 and outlet flow path fl016

The original detailed input file and the four revisions are available and have been tested. The saving in CPU time with Revision 4 was about 50 per cent of that with the original file. The latter, base input, is a good basis if further reductions of the file size are wanted, since it is easier to start with a detailed description of the reactor system than to go from a small input to a larger one. The level of detail can be chosen depending on the case to be simulated. Experience from the transient runs showed that the maximum time step, and thus the CPU time, was often limited by the Courant time step for the CVH Package.

Another temporary limitation could arise during the discharge from the accumulators. The temperature of the Nitrogen can then decrease down to 273.15 K at the adiabatic expansion of the gas. Since this is the lower limit of the property tables it might lead to very small time steps, or even a fatal error and termination of the calculation. One way to get around this problem could be to use the “equilibrium” flag for the accumulator control volumes rather than the normal “non-equilibrium” flag.

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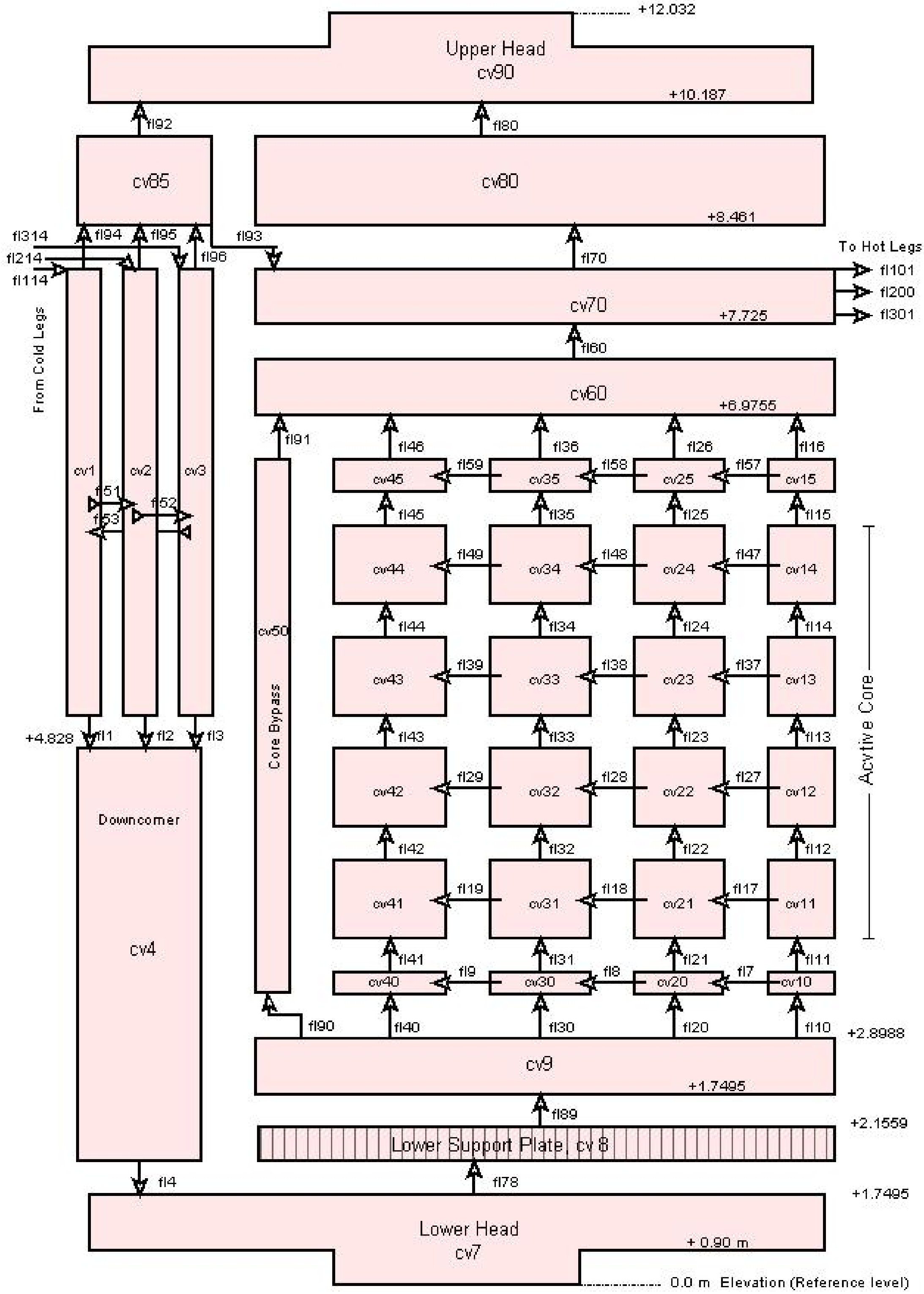


Figure 5. Nodalization scheme for the reactor vessel

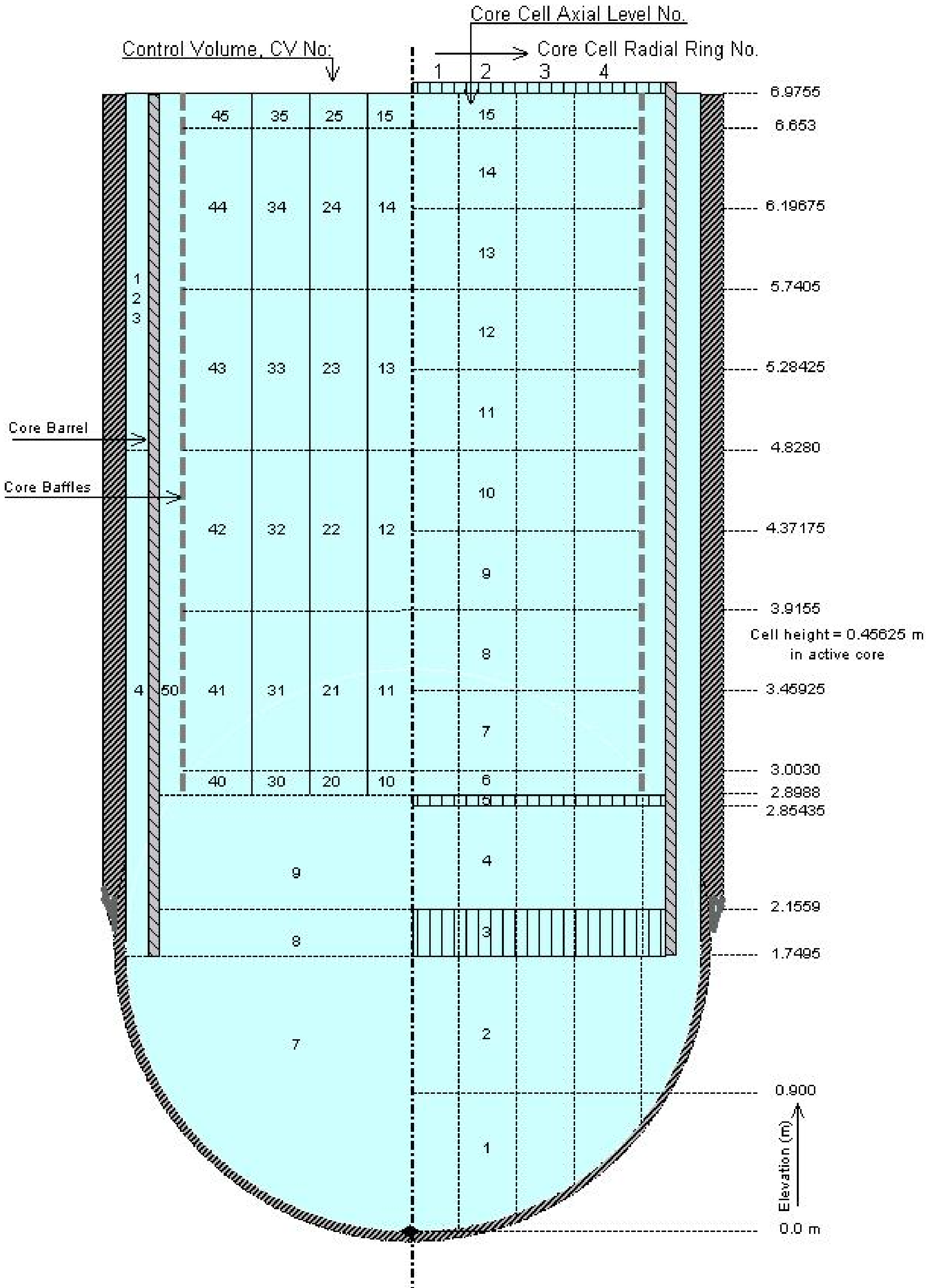


Figure 6. Nodalization of the core

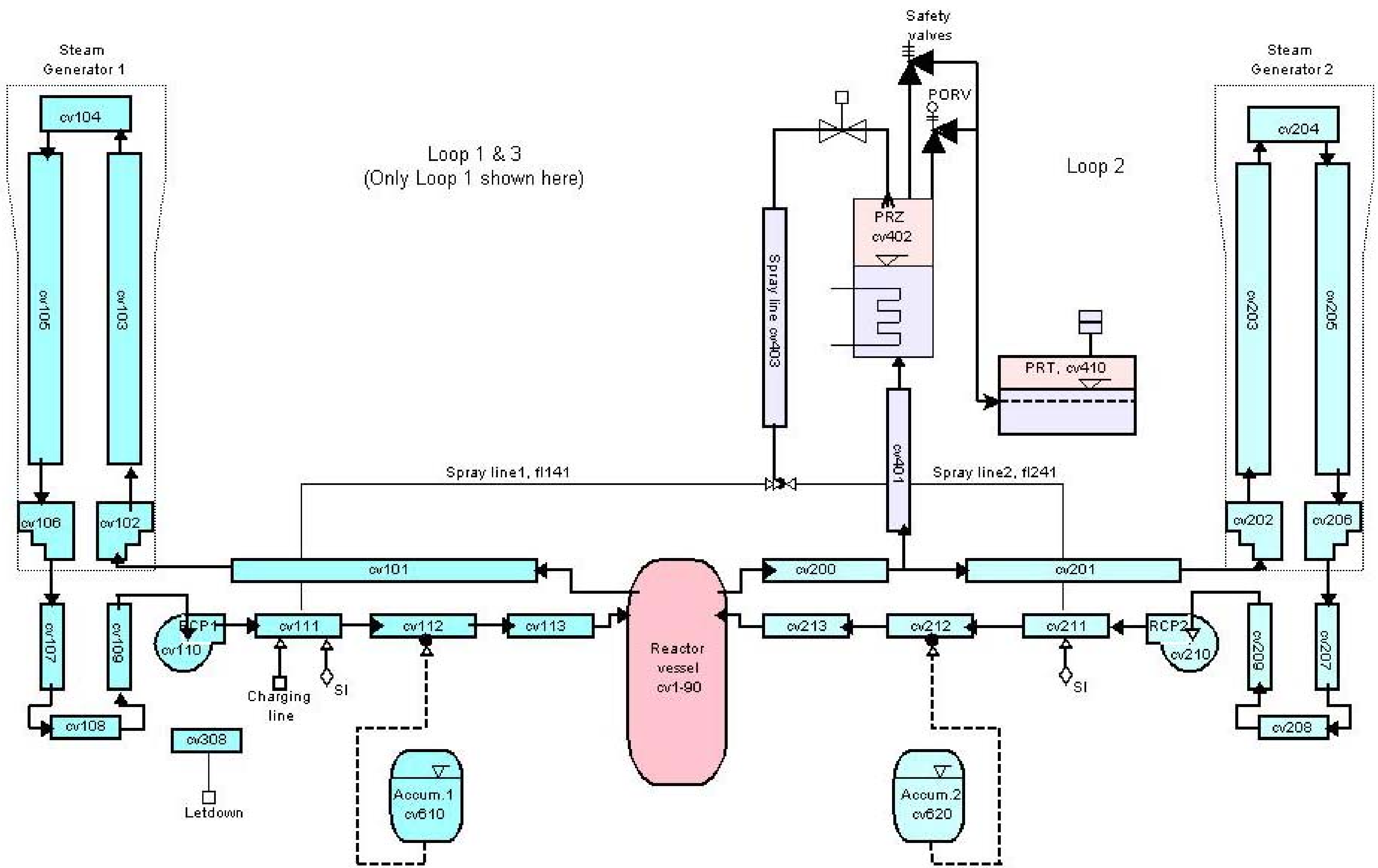


Figure 7. Nodalization scheme for the primary system cooling loops

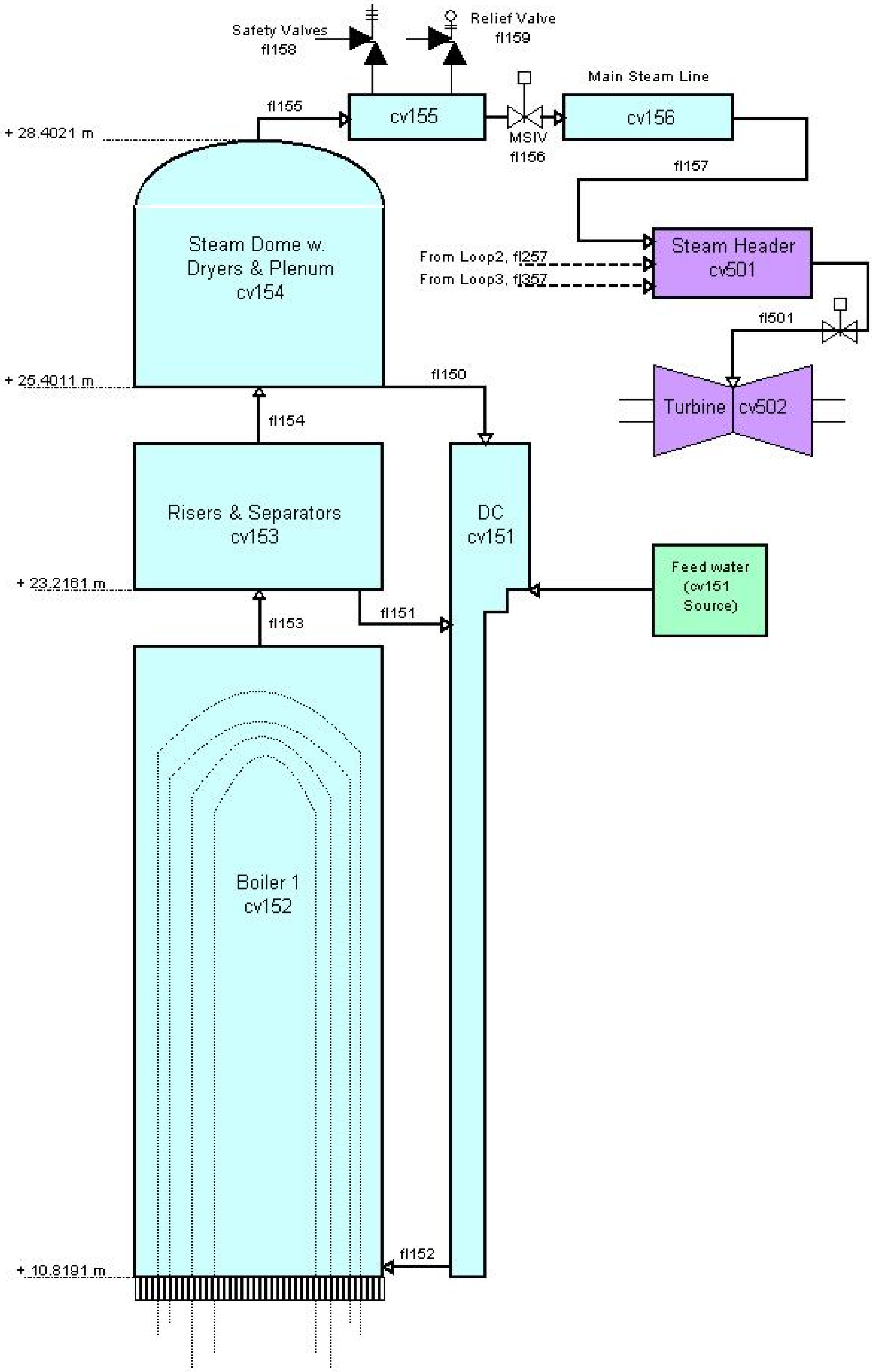
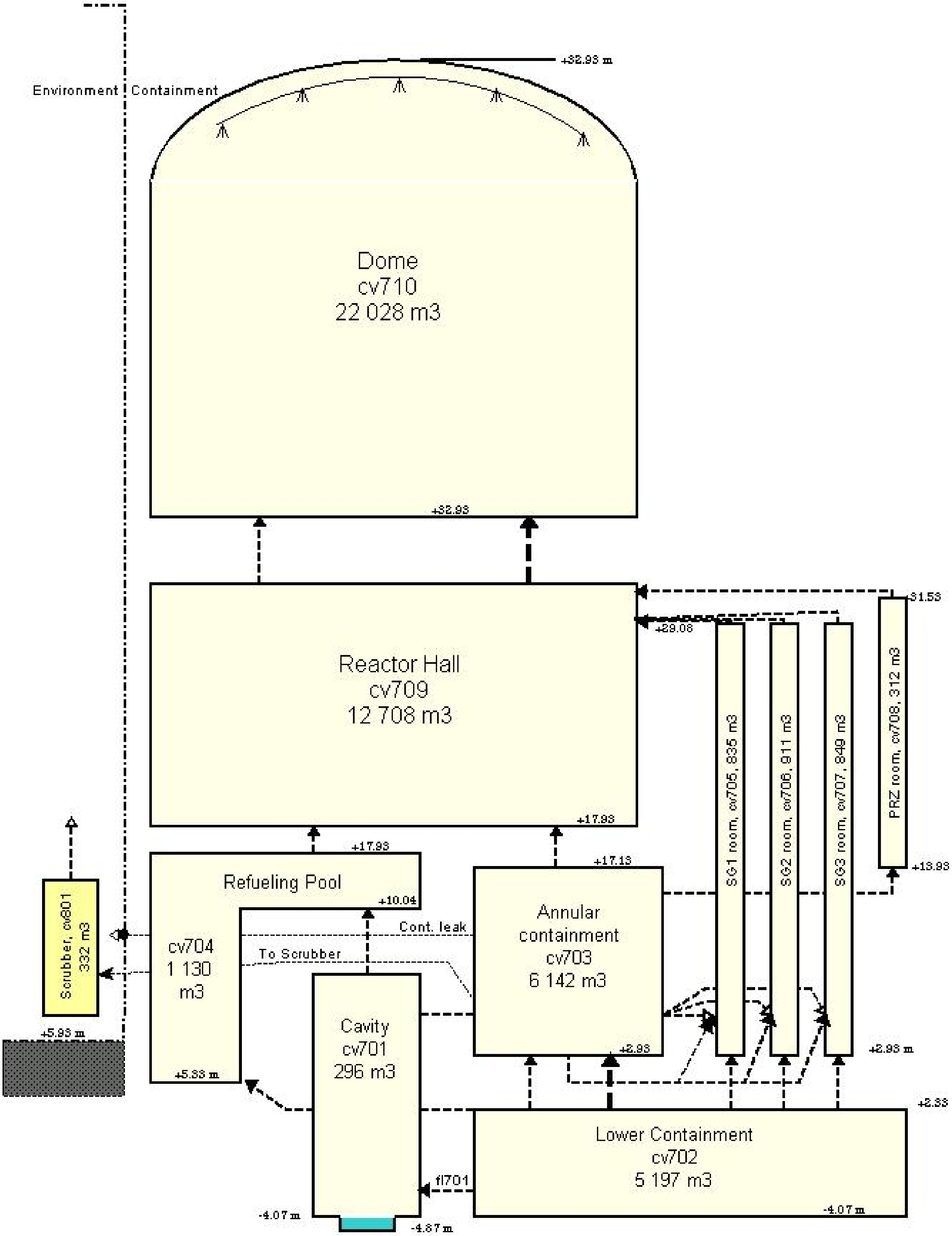


Figure 8. Nodalization of the secondary system

Figure 9. Nodalization of the containment



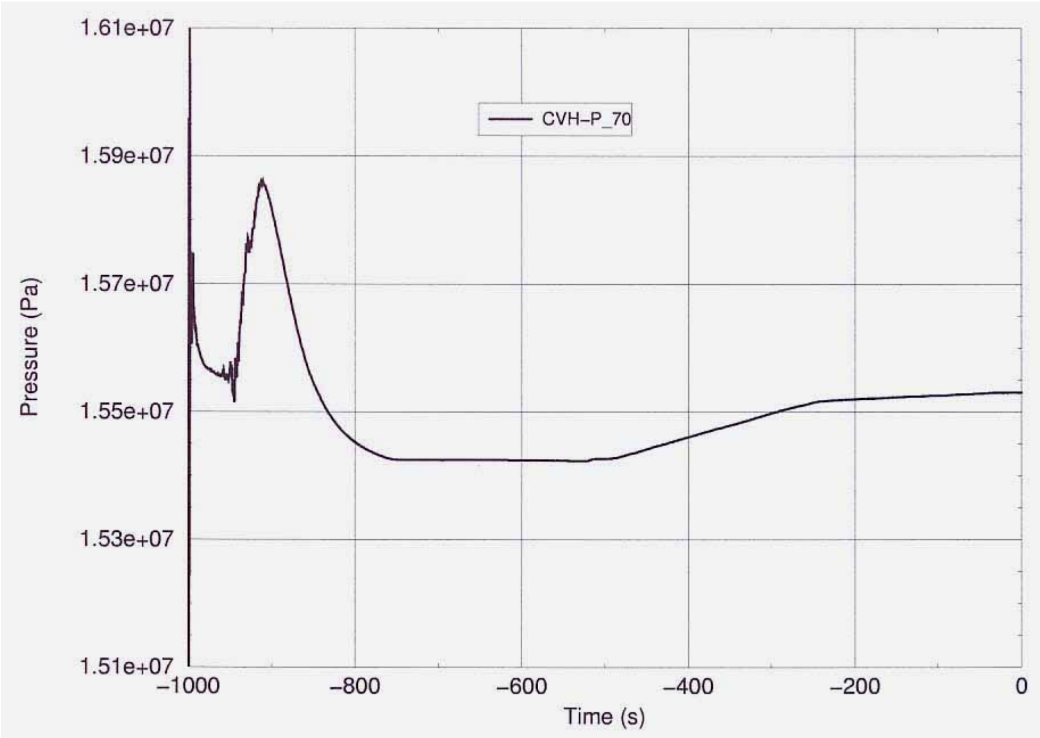


Figure 10. Pressure in Vessel Upper Plenum

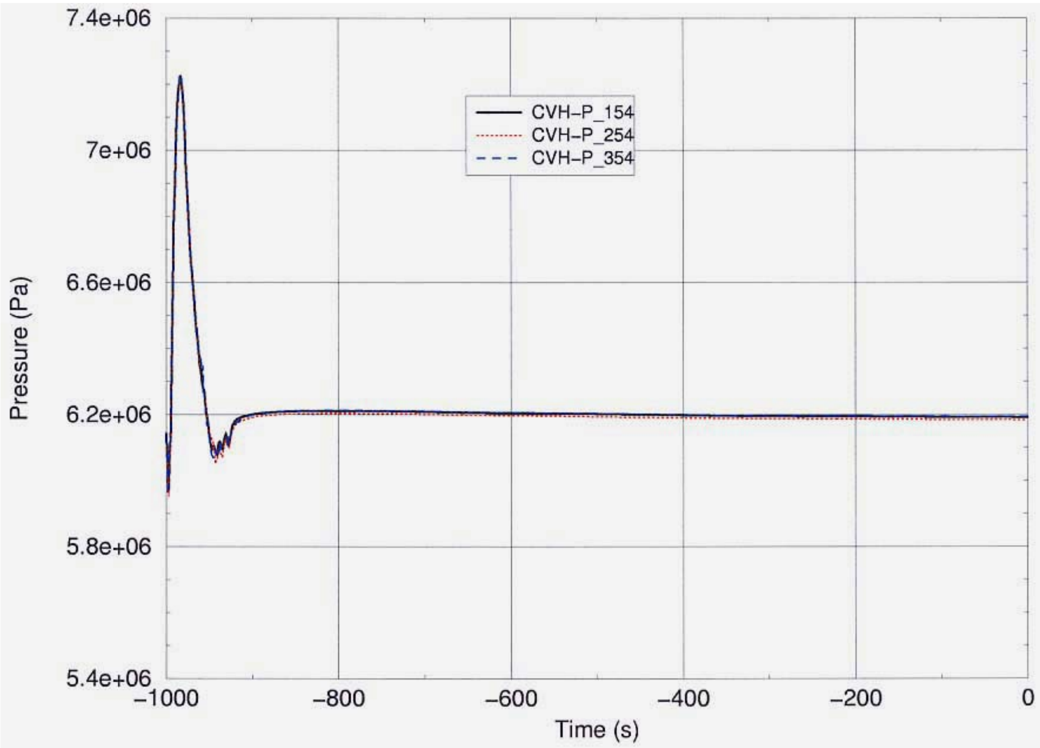


Figure 11. Pressure in Steam Dome, Steam Generator 1-3

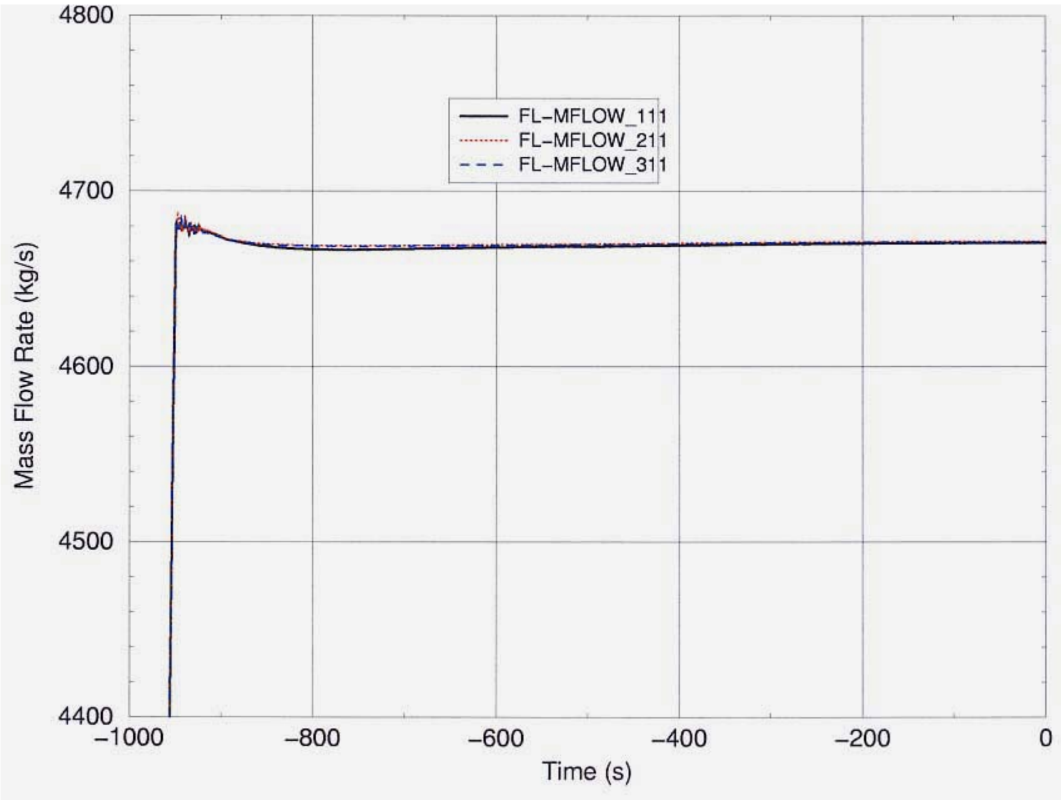


Figure 12. Recirculation Mass Flow Rate in Primary Loop

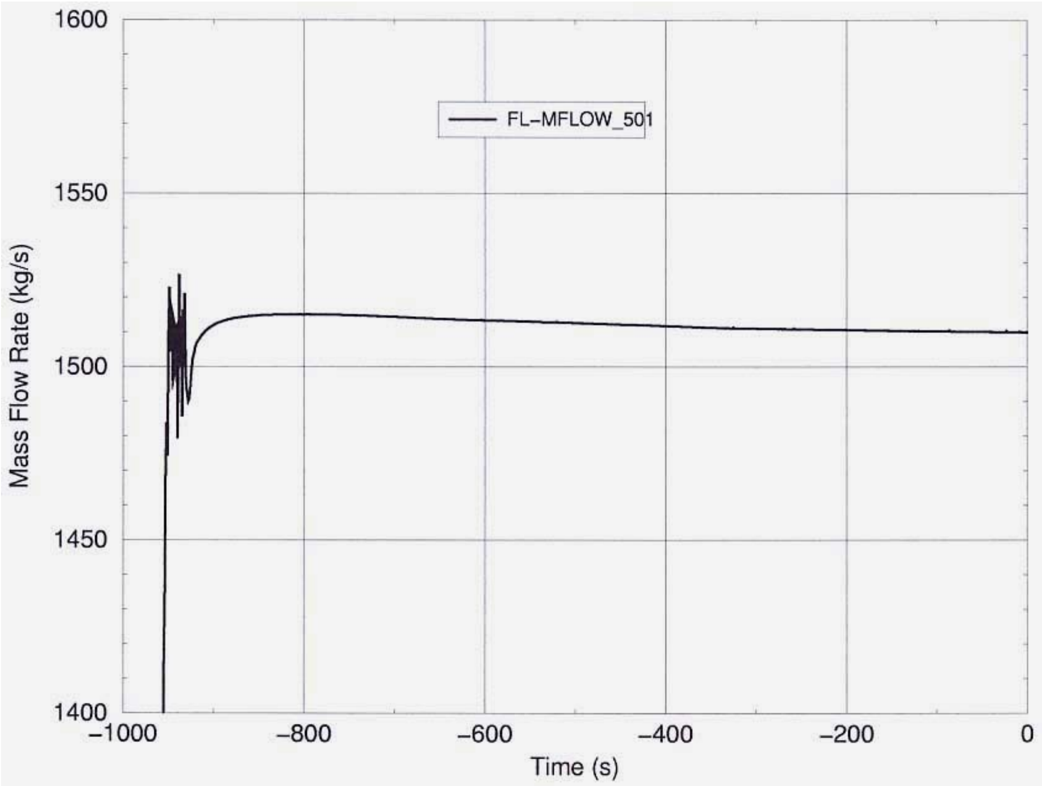


Figure 13. Steam Mass Flow Rate to Turbine

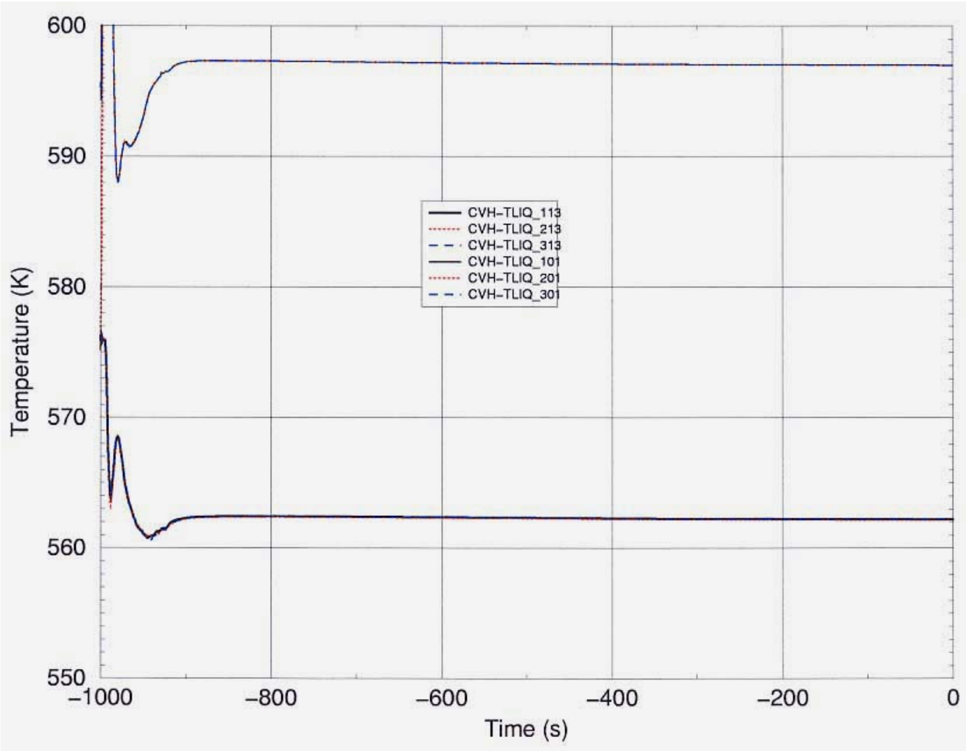


Figure 14. Primary Loop Temperatures in Cold and Hot Legs

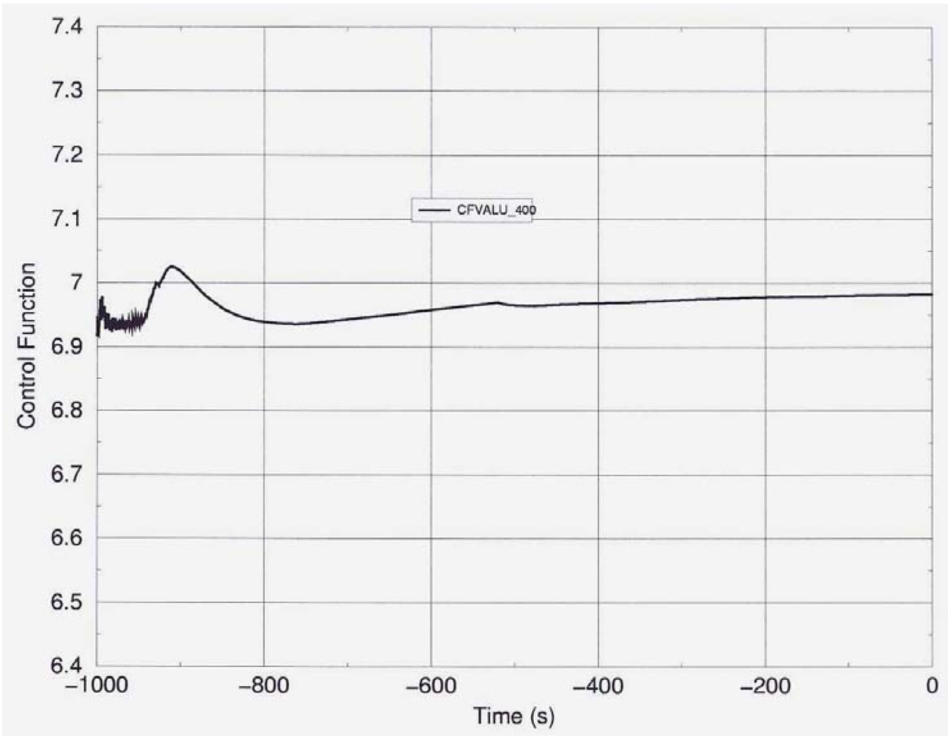


Figure 15. Water Level in Pressurizer (From PRZ bottom)

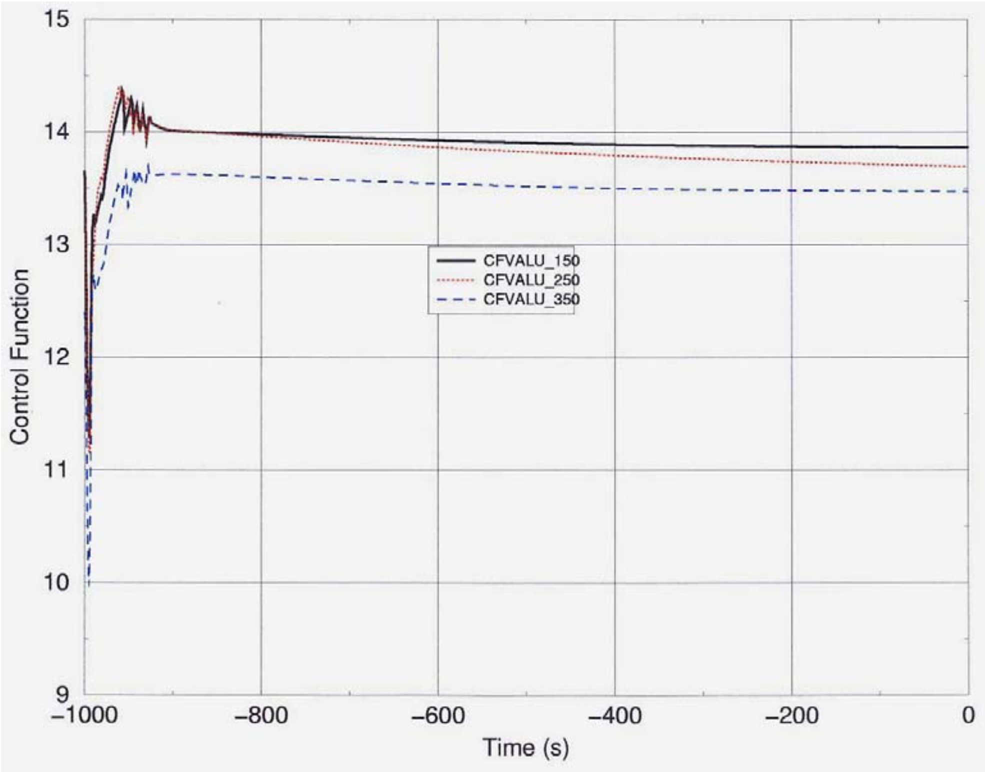


Figure 16. Water Level in Downcomer above Tube Plate, Steam Generator 1-3

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