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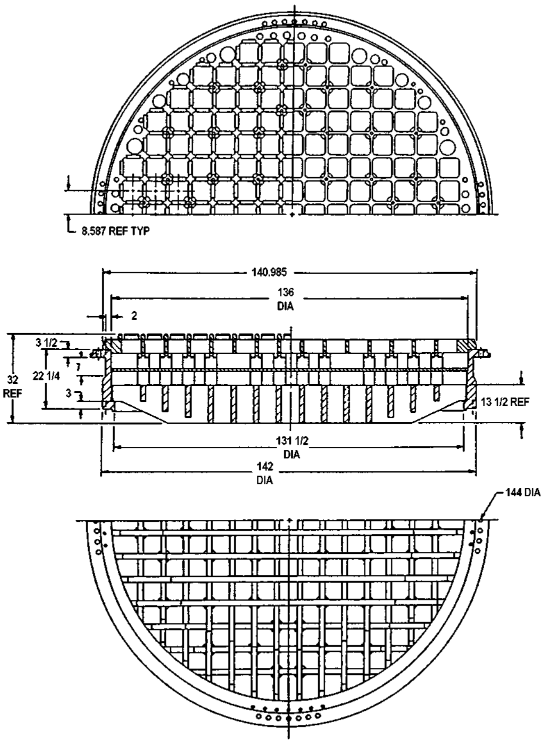
PRESSURISED WATER REACTOR MAIN STEAM LINE BREAK (MSLB) BENCHMARK

Volume I: Final Specifications



April 1999

US Nuclear Regulatory Commission

OECD Nuclear Energy Agency

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NEA NUCLEAR SCIENCE COMMITTEE
NEA COMMITTEE ON SAFETY OF NUCLEAR INSTALLATIONS

PRESSURISED WATER REACTOR MAIN STEAM LINE BREAK (MSLB) BENCHMARK

Volume I: Final Specifications

by

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ORGANISATION FOR ECONOMIC CO-OPERATION AND DEVELOPMENT

Pursuant to Article 1 of the Convention signed in Paris on 14th December 1960, and which came into force on 30th September 1961, the Organisation for Economic Co-operation and Development (OECD) shall promote policies designed:

- to achieve the highest sustainable economic growth and employment and a rising standard of living in Member countries, while maintaining financial stability, and thus to contribute to the development of the world economy;
- to contribute to sound economic expansion in Member as well as non-member countries in the process of economic development; and
- to contribute to the expansion of world trade on a multilateral, non-discriminatory basis in accordance with international obligations.

The original Member countries of the OECD are Austria, Belgium, Canada, Denmark, France, Germany, Greece, Iceland, Ireland, Italy, Luxembourg, the Netherlands, Norway, Portugal, Spain, Sweden, Switzerland, Turkey, the United Kingdom and the United States. The following countries became Members subsequently through accession at the dates indicated hereafter; Japan (28th April 1964), Finland (28th January 1969), Australia (7th June 1971), New Zealand (29th May 1973), Mexico (18th May 1994), the Czech Republic (21st December 1995), Hungary (7th May 1996), Poland (22nd November 1996) and the Republic of Korea (12th December 1996). The Commission of the European Communities takes part in the work of the OECD (Article 13 of the OECD Convention).

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The OECD Nuclear Energy Agency (NEA) was established on 1st February 1958 under the name of OEEC European Nuclear Energy Agency. It received its present designation on 20th April 1972, when Japan became its first non-European full Member. NEA membership today consists of all OECD Member countries, except New Zealand and Poland. The Commission of the European Communities takes part in the work of the Agency.

The primary objective of the NEA is to promote co-operation among the governments of its participating countries in furthering the development of nuclear power as a safe, environmentally acceptable and economic energy source.

This is achieved by:

- *encouraging harmonization of national regulatory policies and practices, with particular reference to the safety of nuclear installations, protection of man against ionising radiation and preservation of the environment, radioactive waste management, and nuclear third party liability and insurance;*
- *assessing the contribution of nuclear power to the overall energy supply by keeping under review the technical and economic aspects of nuclear power growth and forecasting demand and supply for the different phases of the nuclear fuel cycle;*
- *developing exchanges of scientific and technical information particularly through participation in common services;*
- *setting up international research and development programmes and joint undertakings.*

In these and related tasks, the NEA works in close collaboration with the International Atomic Energy Agency in Vienna, with which it has concluded a Co-operation Agreement, as well as with other international organisations in the nuclear field.

FOREWORD

The original PWR MSLB Benchmark specifications, defining three phases as benchmark exercises, was issued by OECD/NEA in October 1997 (NEA/NSC/DOC(97)15). A small team at Pennsylvania State University (PSU) – Kostadin N. Ivanov, Tara M. Beam, and Anthony J. Baratta – is responsible for authoring the final specification, co-ordinating the benchmark activities, answering questions, analyzing the solutions submitted by benchmark participants, and providing reports summarizing the results for each phase. In performing these tasks the PSU team has been collaborated with Adi Irani and Nick Trikouros of GPU Nuclear Inc.

Since the beginning of benchmark activities two benchmark workshops took place. The first was held in Madrid, Spain (June 1998), and the second in Garching near Munich, Germany (March 1999). It was agreed that in performing this series of exercises participants are working at the edge of present developments in coupling of neutronics and thermal-hydraulics and this benchmark leads to a common background understanding of the key issues. Based on preliminary comparisons of the participant's results and the follow-up discussions at the aforementioned workshops changes were suggested to the original specification. It was therefore agreed to issue a final version of the specification, covering the three exercises and incorporating these changes. In addition it was decided that in this final edition the required output information is specified for each phase. A list of calculated and used parameters must be filled by participants for each exercise. This list must be supplemented by a description (including graphs where useful) of the models used in enough details so that the compliance with specification can be verified. The list of deviations from the specification, if any, must be provided and the specific assumptions stated. It is up to the benchmark co-ordinators and report reviewers to decide whether the solution provided models with sufficient precision the system model provided. Solutions that deviate in the modelling in ways not compatible with the specifications will not be graphically compared with those complying. These other solutions will be described in separate sections of the report, where sensitivity effects are investigated or effect of deviation from standard parameters. Participants with strongly deviating solutions will be requested to carefully review their solutions, and if necessary withdraw them from publication.

It was also agreed that the PWR MSLB Benchmark would be published in four volumes:

- Volume 1: PWR MSLB Benchmark: Final Specifications (Phase I, II and III)
- Volume 2: Summary Results of Phase I (Point Kinetics)
- Volume 3: Summary Results of Phase II (3D Kinetics/Core T-H Boundary Model)
- Volume 4: Summary Results of Phase III (Best Estimate Coupled Simulation)

It is the intention to prepare also a CD-ROM with the four reports and the transient boundary conditions, decay heat values, as a function of time, cross-section libraries and supplementary tables and graphs not published in the paper version. The transient boundary conditions, decay heat values and the cross-section libraries can be found also at the benchmark **ftp** site:

Address: tashayar.nuce.psu.edu; **Id:** mslb; **Password:** MslB97

Acknowledgements

This report is dedicated to the students of Penn State, the next generation of nuclear engineers, who are the reason that we are here.

The authors would like to thank Dr. H. Finnemann of Siemens and Dr. S. Langenbuch from GRS, whose support and encouragement in establishing this benchmark were invaluable.

This report is the sum of many efforts, the participants, the funding agencies and their staff, the students at Penn State and the staff of the Organisation of Economic Co-operation and Development, Penn State and General Public Utilities. Many suggestions, comments and most of all much encouragement made this effort possible.

Of particular note are the efforts of Farouk Eltawila assisted by David Ebert, both of the US Nuclear Regulatory Commission. Through their efforts, funding was secured enabling this effort to proceed. We also thank them for their invaluable technical advice and assistance. Special appreciation is also given to the members of the TRAC Users' Group for their financial support.

The authors wish to express their sincere appreciation for the outstanding support offered by Dr. Enrico Sartori, who not only provided efficient administration, organising and valuable technical advice, but more importantly provided friendly council and advice.

Finally, we are grateful to Amanda McWhorter for having devoted her competence and skills to the final edit of this report.

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OECD MSLB BENCHMARK SCHEDULE

1.	15 December 1996	**	Specifications
2.	14 February 1997	**	Distribution of first draft to potential participants from NSC and CSNI/PWG2 for comments and feedback
3.	15 March 1997	**	Comments and feedback from NRC, GPUN and NSC and TG-THA, OECD
4.	7 April 1997	**	Second revised draft of specifications *(this is the version to be distributed by NSC to the OECD participants)*
5.	23-25 April 1997	**	First workshop, Washington, D.C. *(objective is to finalise the second draft taking into account the comments and feedback from NRC, GPUN, NSC and TG-THA*)
6.	9-11 June 1997	**	NSC meeting in Paris *(short presentation on status of the benchmark at NSC, review of comments and changes to final draft)*
7.	1 October 1997	**	Final draft of specifications
8.	15 January 1998	**	Deadline for submitting results for the first benchmark exercise *(point kinetics*)
9.	June 1998	**	First draft of analysis *(point kinetics*)
10.	22-23 June 1998	**	Second benchmark workshop – Europe (Spain) *(present point kinetics results and discuss how 3-D results are coming along)*
11.	15 July 1998	**	Distribute specification of additional parameters for Phase I and summary of second meeting
12.	1 August 1998	**	Distribute agreed boundary conditions for Phase II
13.	October 1998	**	Ad-hoc meeting in connection with the Reactor Physics Conference at Long Island, USA
14.	1 December 1998	**	Deadline for submitting Phase I results
15.	March 1999	**	Issue "final" draft for Phase I results, distribute to participants for comments

** *Completed tasks.*

16.	24-25 March 1999	**	Third meeting at GRS, Garching, presentation of "final" Phase I report and discussion of Phase II results
17.	20 April 1999	**	Deadline for issuing final PWR-MSLB benchmark specification
18.	20 June 1999		Deadline for sending in Phase I results
19.	20 July 1999		Deadline for submitting final results for the second benchmark exercise *(3-D kinetics/core T/H boundary model)*
20.	Mid August 1999		Issue final draft of Phase I report for review by subgroup *(A. Irani, K. Ivanov, A. Knoll, S. Langenbuch)*
21.	Mid September 1999		Final check and approval of Phase I report participants *(via mail and e-mail)*
22.	15 September 1999		Issue final draft of Phase II report for review by subgroup *(J.M. Aragonés, H. Finnemann, K. Ivanov, S. Langenbuch)*
23.	20-22 September 1999		Approval of final draft for Phase I by NSC Members and CSNI/PWG2 on coolant system behaviour
24.	27-30 September 1999		Ad-hoc benchmark participants meeting and presentation of progress in the PWR-MSLB benchmark study during the "Mathematics and Computation" '99 conference in Madrid, Spain
25.	Mid October 1999		Publication of Phase I Report
26.	1 November 1999		Deadline for submitting results of third benchmark exercise *(best estimate coupled simulation)*
27.	5 January 2000		Draft of Phase III report *(best estimate coupled simulation)*
28.	27-28 January 2000		Fourth PWR-MSLB benchmark meeting, Paris, France *(approve final Phase II report, discuss draft Phase III results and report)*
29.	March 2000		Submit final Phase II report to NSC and CSNI/PWG2 for approval
30.	March 2000		Issue final draft of Phase III report to subgroup for review
31.	April 2000		Approval by participants of Phase III report
32.	May 2000		Issue Phase II report
33.	June 2000		Submit final Phase III report to NSC and CSNI/PWG2 for approval
34.	Summer 2000		Issue Phase III report

Chapter 1

INTRODUCTION

The incorporation of full three-dimensional (3-D) modelling of the reactor core into system transient codes allows "best-estimate" simulations of interactions between reactor core behaviour and plant dynamics. Recent progress in computer technology makes the development of such coupled code systems feasible. As a result, considerable efforts in this direction have been made in numerous countries. The Nuclear Science Committee (NSC)* of the Nuclear Energy Agency (NEA)/Organisation for Economic Co-operation and Development (OECD) has recently released a set of computational benchmark problems for the calculation of reactivity transients in pressurised water reactors (PWR) [1]. These benchmarks use coupled code systems to verify the data exchange and to test the neutronics coupling to the fuel-rod heat-conduction solution methodology [2]. To further verify the capability of these codes to analyse complex transients with coupled core-plant interactions, and to fully test the thermal-hydraulic coupling, a plant transient benchmark which uses a three-dimensional neutronics core model must be defined. This benchmark is a natural continuation of previous NSC Core Transient benchmarks. It was approved by both the NSC and the Task Group on Thermo-Hydraulic Behaviour (TG-THA), as well as the Principle Working Group 2 (PWG2) of the Committee on Safety of Nuclear Installations (CSNI) of the NEA OECD for the purpose of validating system best-estimate analysis codes.

1.1. Objective

The reference problem chosen for simulation in a PWR is a Main Steam Line Break (MSLB), which may occur as a consequence of the rupture of one steam line upstream of the main steam isolation valves. This event is characterised by significant space-time effects in the core caused by asymmetric cooling and an assumed stuck-out control rod during reactor trip. Simulation of the transient requires evaluation of core response from a multi-dimensional perspective (coupled three-dimensional (3-D) neutronics/core thermal-hydraulics) supplemented by a one-dimensional (1-D) simulation of the remainder of the reactor coolant system [3]. The purpose of this benchmark is three-fold:

- To verify the capability of system codes to analyse complex transients with coupled core-plant interactions.
- To fully test the 3-D neutronics/thermal-hydraulic coupling.
- To evaluate discrepancies between predictions of coupled codes in best-estimate transient simulations.

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* The Nuclear Science Committee was previously called the Reactor Physics Committee.

1.2. Definition of three benchmark exercises

In addition to being based on a well-defined problem, with reference design and data from the Three Mile Island Unit 1 Nuclear Power Plant (TMI-1 NPP), the benchmark includes a complete set of input data, and consists of three exercises. These exercises are discussed below.

1.2.1. Exercise 1 – Point kinetics plant simulation

The purpose of this exercise is to test the primary and secondary system model responses. Compatible point kinetics model inputs, which preserve axial and radial power distribution and tripped rod reactivity from Exercise 3, are provided.

1.2.2. Exercise 2 – Coupled 3-D neutronics/core thermal-hydraulics response evaluation

The purpose of this exercise is to model the core and the vessel only. Inlet and outlet core transient boundary conditions are provided.

1.2.3. Exercise 3 – Best-estimate coupled core plant transient modelling

This exercise combines elements of the first two exercises in this benchmark and is an analysis of the transient in its entirety.

Chapter 2

CORE AND NEUTRONIC DATA

2.1. General

The reference design for the PWR is derived from the reactor geometry and operational data of the TMI-1 NPP [4]. The data provided in the following paragraphs, and in the pertinent tables and figures, completely defines the three PWR benchmark exercises. Two versions of the MSLB transient scenario, described in Section 5.1, are specified.

2.2. Core geometry and fuel assembly (FA) geometry

The radial geometry of the reactor core is shown in Figure 2.2.1. Radially, the core is divided into cells 21.811 cm (0.7156 ft) wide, each corresponding to one fuel assembly (FA), plus a radial reflector (shaded area) of the same width. There are a total of 241 assemblies, 177 FA and 64 reflector assemblies. Axially, the reactor core is divided into 24 layers with a height (starting from the bottom) of: 14.88 cm (0.4882 ft); 4.71 cm (0.1545 ft); 10.17 cm (0.3514 ft); 8×14.88 cm; 2×29.76 cm (0.9764 ft); 8×14.88 cm; 12.266 cm (0.4024 ft); 2.614 cm (0.0858 ft); and 14.88 cm, adding up to a total active core height of 357.12 cm (11.717 ft). Both the upper and lower axial reflector have a thickness of 21.811 cm (0.7156 ft). The axial nodalisation scheme accounts for material changes in the fuel design and for the exposure and moderator temperature (spectral history) variations.

The mesh used for the calculation is up to the participant, and should be chosen according to the numerical capabilities of the code. Output should, however, give volume-averaged results on the specified mesh in the format described in Chapter 6.

Fuel assemblies with different ^{235}U enrichments and different numbers of burnable absorber rods are present in the core. The axial and radial distributions of the enrichment and absorbers can be found in Section 2.4; the geometric data for the FA is given in Table 2.2.1.

The available gap width is 0.00955 cm. For the neutronic problem, each of the FAs is considered to be homogeneous.

The radial arrangement of the control assemblies (CA) is shown in Figure 2.2.2. Sixty-one of these CAs, grouped into seven groups, consist of full-length control rods. These rods contain a strong neutron absorber over a length that spans most of the active core region. In addition to the radial arrangement, on the position of control rod insertion in units of cm is given from the bottom of the lower reflector. The total CA length, which coincides with the absorber length, is 342.7055 cm (11.244 ft). No tip of control rods is defined. The position of the lower CA absorber edge from the bottom of the lower reflector is 36.2255 cm (1.189 ft) for a completely inserted CA, and 378.931 cm (12.4323 ft) for a completely withdrawn CA. Measured in units of steps, complete insertion and withdrawal of a CA correspond to 0 and 971 steps, respectively. Each step is 0.3531 cm (0.139 in). If one multiplies 971×0.3531 cm and adds 36.2255 cm, the result is 379.0856 cm. The top edge of the core region is 378.931 cm; therefore, there is a difference of 0.1546 cm which is not modelled in this benchmark. Here, the definition completely withdrawn means withdrawn from the active core, i.e. out of the core. The actual completely withdrawn position in terms of steps is 1 000 steps, or 36.2255 + 353.1 = 389.3255 cm from the bottom of the lower reflector, which is in the region of the upper reflector and is not modelled in this benchmark. In addition, eight of the CA (Group 8) consist of part-length control rods (axial power shaping rods, or APSR) whose presence is accounted for in the cross-section tables.

2.3. Neutron modelling

Two prompt and six delayed neutron groups are modelled. The energy release per fission for the two prompt neutron groups is 0.3213×10^{-10} and 0.3206×10^{-10} W-s/fission, and this energy release is considered to be independent of time and space. Table 2.3.1 shows the time constants and fractions of delayed neutrons.

It is recommended that ANS-79 be used as a decay heat standard model. In total 71 decay-heat groups are used: 69 groups are used for the three isotopes ^{235}U, ^{239}Pu and ^{238}U with the decay-heat constants defined in the 1979 ANS standard; plus, the heavy-element decay heat groups for ^{239}U and ^{239}Np are used with constants given in Table 2.3.2. It is recommended that the participants also use the assumption of an infinite operation at a power of 2 772 MW_t. For participants who are not capable of using the ANS-79 decay heat standard, a file of the decay heat evolution throughout the transient for both scenario versions is provided on CD-ROM and at the benchmark **ftp** site under the directory **Decay-Heat**. These predictions are obtained using the Pennsylvania State University (PSU) coupled code TRAC-PF1/NEM [3]. The effective decay heat energy fraction of the total thermal power (the relative contribution in the steady state) is equal to 0.07 143.

2.4. Two-dimensional (2-D) assembly types and three-dimensional (3-D) composition maps

Thirty assembly types are contained within the core geometry. There are 438 unrodded and 195 rodded compositions. The corresponding sets of cross-sections are provided. Each composition is defined by material properties (due to changes in the fuel design) and burn-up. The burn-up dependence is a three-component vector variables: exposure (GWd/t), spectral history (T_{mod}) and burnable poison (BP) history. The definition of assembly types is shown in Table 2.4.1. The radial distribution of these assembly types within the reactor geometry is shown in Figure 2.4.1. The 2-D assembly type map is shown in a one-eighth core symmetry sector together with the assembly exposure values at the end of cycle (EOC). The axial locations of compositions for each assembly are shown in Table 2.4.2.

2.5. Cross-section library

A complete set of diffusion coefficients and macroscopic cross-sections for scattering, absorption, and fission as a function of the moderator density and fuel temperature is defined for each composition. The assembly discontinuity factors (ADFs) are taken into consideration implicitly by incorporating them into the cross-sections in order to minimise the size of the cross-section tables. The group inverse neutron velocities are also provided for each composition. Dependence of the cross-sections on the above variables is specified through a two-dimensional table look-up. Each composition is assigned to a cross-section set containing separate tables for the diffusion coefficients and cross-sections, with each point in the table representing a possible core state. The expected range of the transient is covered by the selection of an adequate range for the independent variables shown in Table 2.5.1. A linear interpolation scheme is used to obtain the appropriate total cross-sections from the tabulated ones based on the reactor conditions being modelled. Table 2.5.2 shows the definition of a cross-section table associated with a composition. Table 2.5.3 shows the macroscopic cross-section table structure for one cross-section set. All cross-section sets are assembled into a cross-section library. The cross-sections are provided in separate libraries for rodded **(nemtabr)** (numerical nodes with a CA) and unrodded compositions **(nemtab)**. The format of each library is as follows:

- The first line of data is used to show the number of data points used for the independent thermal-hydraulic (T-H) parameters. These parameters include fuel temperature, moderator density, boron concentration and moderator temperature.

- Each cross-section set is in the order shown in Table 2.5.3. Each table is in the format described in Table 2.5.2. More detailed information on this format is presented in Appendix B. First, the values of the independent thermal-hydraulic parameters (fuel temperature and moderator density) used to specify that particular set of cross-sections are listed, followed by the values of the cross-sections**. Finally, the group inverse neutron velocities complete the data for a given cross-section set.

- The dependence on fuel temperature in the reflector cross-section tables is also modelled. This is because the reflector cross-sections are generated by performing lattice physics transport calculations, including the next fuel region. In order to simplify the reflector feedback modelling the following assumptions are made for this benchmark: an average fuel temperature value equal to 600 K is used for the radial reflector cross-section modelling in both the initial steady-state and transient simulations, and the average coolant density for the radial reflector is equal to the inlet coolant density. For the axial reflector regions the following assumptions are made: for the bottom – the fuel temperature is equal to the inlet coolant temperature (per T-H channel or cell) and the coolant density is equal to the inlet coolant density (again per channel); for the top – the fuel temperature is equal to the outlet coolant temperature (per channel) and the coolant density is equal to the outlet coolant density (per channel).

- There is a second rodded library, generated for a hypothetical return to power scenario with the 3-D kinetics model; named **nemtabr.rp**. The magnitude of the return to power is small; however, this is the maximum one can obtain by modifying the rodded thermal absorption cross-sections for control rod groups 1 through 6. This additional non-realistic scenario (even with accounting for all the conservative assumptions used in the licensing practice) is defined at the request of participants for more comprehensive testing of coupled 3-D kinetics/thermal-hydraulics models.

All cross-section data, along with a program for linear interpolation, are supplied on CD-ROM and at the benchmark **ftp** site under the directory **XS-Lib** in the format described above.

** Please note that the provided absorption cross-sections already take the xenon thermal cross-sections into account; however, at the participants' request, the xenon cross-sections are listed in the cross-section sets.

Table 2.2.1. FA geometry data

Parameter	Value
Pellet diameter	9.391 mm/0.3697 in
Clad diameter (outside)	10.928 mm/0.43 in
Clad wall thickness	0.673 mm/0.0265 in
Fuel rod pitch	14.427 mm/0.568 in
Guide tube diameter (outside)	13.462 mm/0.53 in
Guide tube diameter (inside)	12.650 mm/0.498 in
Geometry	15×15
Number of fuel pins	208
Number of guide tubes	16
Number of in-core instrument positions per fit	1

Table 2.3.1. Decay constant and fractions of delayed neutrons

Group	Decay constant (s^{-1})	Relative fraction of delayed neutrons in %
1	0.012818	0.0153
2	0.031430	0.1086
3	0.125062	0.0965
4	0.329776	0.2019
5	1.414748	0.0791
6	3.822362	0.0197

Total fraction of delayed neutrons: 0.5211%

Table 2.3.2. Heavy-element decay heat constants

Group no. (isotope)	Decay constant (s^{-1})	Available energy from a single atom (MeV)
70 (^{239}U)	4.91×10^{-4}	0.474
71 (^{239}Np)	3.41×10^{-6}	0.419

Table 2.4.1. Definition of assembly types

Assembly	Characteristics		
1	4.00 w/o	No BP	No Gd pins
2	4.95 w/o	3.5% BP	4 Gd pins
3	5.00 w/o	3.5% BP pulled	4 Gd pins
4	4.95 w/o	3.5% BP	4 Gd pins
5	4.40 w/o	No BP	No Gd pins
6	5.00 w/o	3.5% BP	4 Gd pins
7	4.85 w/o	No BP	4 Gd pins
8	4.85 w/o	No BP	4 Gd pins
9	4.95 w/o	3.5% BP pulled	4 Gd pins
10	4.95 w/o	3.5% BP	4 Gd pins
11	4.85 w/o	3.5% BP pulled	4 Gd pins
12	4.95 w/o	3.5% BP	4 Gd pins
13	5.00 w/o	3.5% BP pulled	4 Gd pins
14	5.00 w/o	No BP	8 Gd pins
15	4.95 w/o	No BP	8 Gd pins
16	4.95 w/o	3.5% BP pulled	4 Gd pins
17	4.95 w/o	3.5% BP	4 Gd pins
18	4.95 w/o	3.5% BP pulled	4 Gd pins
19	5.00 w/o	3.5% BP	4 Gd pins
20	4.40 w/o	No BP	No Gd pins
21	4.85 w/o	3.5% BP pulled	4 Gd pins
22	4.40 w/o	No BP	No Gd pins
23	4.95 w/o	3.5% BP	No Gd pins
24	4.95 w/o	3.5% BP pulled	4 Gd pins
25	5.00 w/o	No BP	8 Gd pins
26	5.00 w/o	No BP	4 Gd pins
27	5.00 w/o	No BP	No Gd pins
28	4.95 w/o	3.5% pulled	4 Gd pins
29	5.00 w/o	No BP	4 Gd pins
30		Radial reflector	

Table 2.4.2. Composition numbers in axial layers for each assembly type

Bottom

	1	2	3	4	5	6	7	8	9	10	11	12	13	14	15	16	17	18	19	20	21	22	23	24	25	26	27	28	29	30
1	24	24	24	24	24	24	24	24	24	24	24	24	24	24	24	24	24	24	24	24	24	24	24	24	24	24	24	24	24	24
2	1	16	34	49	64	79	94	109	124	139	154	169	184	199	214	229	244	259	274	289	304	319	334	349	364	379	394	409	424	25
3	2	17	35	50	65	80	95	110	125	140	155	170	185	200	215	230	245	260	275	290	305	320	335	350	365	380	395	410	425	25
4	3	18	36	51	66	81	96	111	126	141	156	171	186	201	216	231	246	261	276	291	306	321	336	351	366	381	396	411	426	25
5	4	19	37	52	67	82	97	112	127	142	157	172	187	202	217	232	247	262	277	292	307	322	337	352	367	382	397	412	427	25
6	5	20	38	53	68	83	98	113	128	143	158	173	188	203	218	233	248	263	278	293	308	323	338	353	368	383	398	413	428	25
7	6	21	39	54	69	84	99	114	129	144	159	174	189	204	219	234	249	264	279	294	309	324	339	354	369	384	399	414	429	25
8	7	22	40	55	70	85	100	115	130	145	160	175	190	205	220	235	250	265	280	295	310	325	340	355	370	385	400	415	430	25
9	7	22	40	55	70	85	100	115	130	145	160	175	190	205	220	235	250	265	280	295	310	325	340	355	370	385	400	415	430	25
10	7	22	40	55	70	85	100	115	130	145	160	175	190	205	220	235	250	265	280	295	310	325	340	355	370	385	400	415	430	25
11	7	22	40	55	70	85	100	115	130	145	160	175	190	205	220	235	250	265	280	295	310	325	340	355	370	385	400	415	430	25
12	8	23	41	56	71	86	101	116	131	146	161	176	191	206	221	236	251	266	281	296	311	326	341	356	371	386	401	416	431	25
13	8	23	41	56	71	86	101	116	131	146	161	176	191	206	221	236	251	266	281	296	311	326	341	356	371	386	401	416	431	25
14	8	23	41	56	71	86	101	116	131	146	161	176	191	206	221	236	251	266	281	296	311	326	341	356	371	386	401	416	431	25
15	8	23	41	56	71	86	101	116	131	146	161	176	191	206	221	236	251	266	281	296	311	326	341	356	371	386	401	416	431	25
16	9	27	42	57	72	87	102	117	132	147	162	177	192	207	222	237	252	267	282	297	312	327	342	357	372	387	402	417	432	25
17	9	27	42	57	72	87	102	117	132	147	162	177	192	207	222	237	252	267	282	297	312	327	342	357	372	387	402	417	432	25
18	9	27	42	57	72	87	102	117	132	147	162	177	192	207	222	237	252	267	282	297	312	327	342	357	372	387	402	417	432	25
19	9	27	42	57	72	87	102	117	132	147	162	177	192	207	222	237	252	267	282	297	312	327	342	357	372	387	402	417	432	25
20	10	28	43	58	73	88	103	118	133	148	163	178	193	208	223	238	253	268	283	298	313	328	343	358	373	388	403	418	433	25
21	11	29	44	59	74	89	104	119	134	149	164	179	194	209	224	239	254	269	284	299	314	329	344	359	374	389	404	419	434	25
22	12	30	45	60	75	90	105	120	135	150	165	180	195	210	225	240	255	270	285	300	315	330	345	360	375	390	405	420	435	25
23	13	31	46	61	76	91	106	121	136	151	166	181	196	211	226	241	256	271	286	301	316	331	346	361	376	391	406	421	436	25
24	14	32	47	62	77	92	107	122	137	152	167	182	197	212	227	242	257	272	287	302	317	332	347	362	377	392	407	422	437	25
25	15	33	48	63	78	93	108	123	138	153	168	183	198	213	228	243	258	273	288	303	318	333	348	363	378	393	408	423	438	25
26	26	26	26	26	26	26	26	26	26	26	26	26	26	26	26	26	26	26	26	26	26	26	26	26	26	26	26	26	26	26

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Table 2.5.1. Range of variables

T Fuel (°K)	Void	T Mod (°K)	Pressure (psia/MPa)	Rho M. (kg/m^3 / lb/ft^3)	Boron (ppm)
500.00	0.00	605.22	2200.0/15.17	641.40/8.2385	5.00
760.22	0.00	605.22	2200.0/15.17	641.40/8.2385	5.00
867.27	0.00	605.22	2200.0/15.17	641.40/8.2385	5.00
921.88	0.00	605.22	2200.0/15.17	641.40/8.2385	5.00
1500.00	0.00	605.22	2200.0/15.17	641.40/8.2385	5.00
500.00	0.00	579.35	2200.0/15.17	711.43/9.1379	5.00
760.22	0.00	579.35	2200.0/15.17	711.43/9.1379	5.00
867.27	0.00	579.35	2200.0/15.17	711.43/9.1379	5.00
921.88	0.00	579.35	2200.0/15.17	711.43/9.1379	5.00
1500.00	0.00	579.35	2200.0/15.17	711.43/9.1379	5.00
500.00	0.00	551.00	2200.0/15.17	769.47/9.8834	5.00
760.22	0.00	551.00	2200.0/15.17	769.47/9.8834	5.00
867.27	0.00	551.00	2200.0/15.17	769.47/9.8834	5.00
921.88	0.00	551.00	2200.0/15.17	769.47/9.8834	5.00
1500.00	0.00	551.00	2200.0/15.17	769.47/9.8834	5.00
500.00	0.00	545.00	1300.00/8.963	772.44/9.9216	5.00
760.22	0.00	545.00	1300.00/8.963	772.44/9.9216	5.00
867.27	0.00	545.00	1300.00/8.963	772.44/9.9216	5.00
921.88	0.00	545.00	1300.00/8.963	772.44/9.9216	5.00
1500.00	0.00	545.00	1300.00/8.963	772.44/9.9216	5.00
500.00	0.00	538.71	1000.00/6.895	781.31/10.035	5.00
760.22	0.00	538.71	1000.00/6.895	781.31/10.035	5.00
867.27	0.00	538.71	1000.00/6.895	781.31/10.035	5.00
921.88	0.00	538.71	1000.00/6.895	781.31/10.035	5.00
1500.00	0.00	538.71	1000.00/6.895	781.31/10.035	5.00
500.00	0.00	520.00	800.00/5.516	810.10/10.405	5.00
760.22	0.00	520.00	800.00/5.516	810.10/10.405	5.00
867.27	0.00	520.00	800.00/5.516	810.10/10.405	5.00
921.88	0.00	520.00	800.00/5.516	810.10/10.405	5.00
1500.00	0.00	520.00	800.00/5.516	810.10/10.405	5.00

Table 2.5.2. Key to macroscopic cross-section tables

T_{f1}	T_{f2}	T_{f3}	T_{f4}	T_{f5}
ρ_{m1}	ρ_{m2}	ρ_{m3}	ρ_{m4}	ρ_{m5}
ρ_{m6}	Σ_1	Σ_2	...	
		...	Σ_{29}	Σ_{30}

Where:

– T_f is the Doppler (fuel) temperature (°K)

– ρ_m is the moderator density (kg/m^3)

Macroscopic cross-sections are in units of cm^{-1}

Table 2.5.3. Macroscopic cross-section tables structure

```
*************************************************************************
        NEM – Cross-Section Table Input
*
*       T Fuel          Rho Mod.                 Boron ppm.        T Mod.
          5                6                        0                 0
*
***********     X-Section Set #
  #
*************************************************************************
        Group No. 1
*
*************       Diffusion Coefficient Table
*
*************       Absorption X-Section Table
*
*************       Fission X-Section Table
*
*************       Nu-Fission X-Section Table
*
*************       Scattering From Group 1 to 2 X-Section Table
*
*************************************************************************
        Group No. 2
*
*************       Diffusion Coefficient Table
*
*************       Absorption X-Section Table
*
*************       Fission X-Section Table
*
*************       Nu-Fission X-Section Table
*
*************       Xenon Absorption Cross-Section Table
*
*************************************************************************
*************       Inv. Neutron Velocities
*
```

Figure 2.2.1. Cross-section of the reactor core

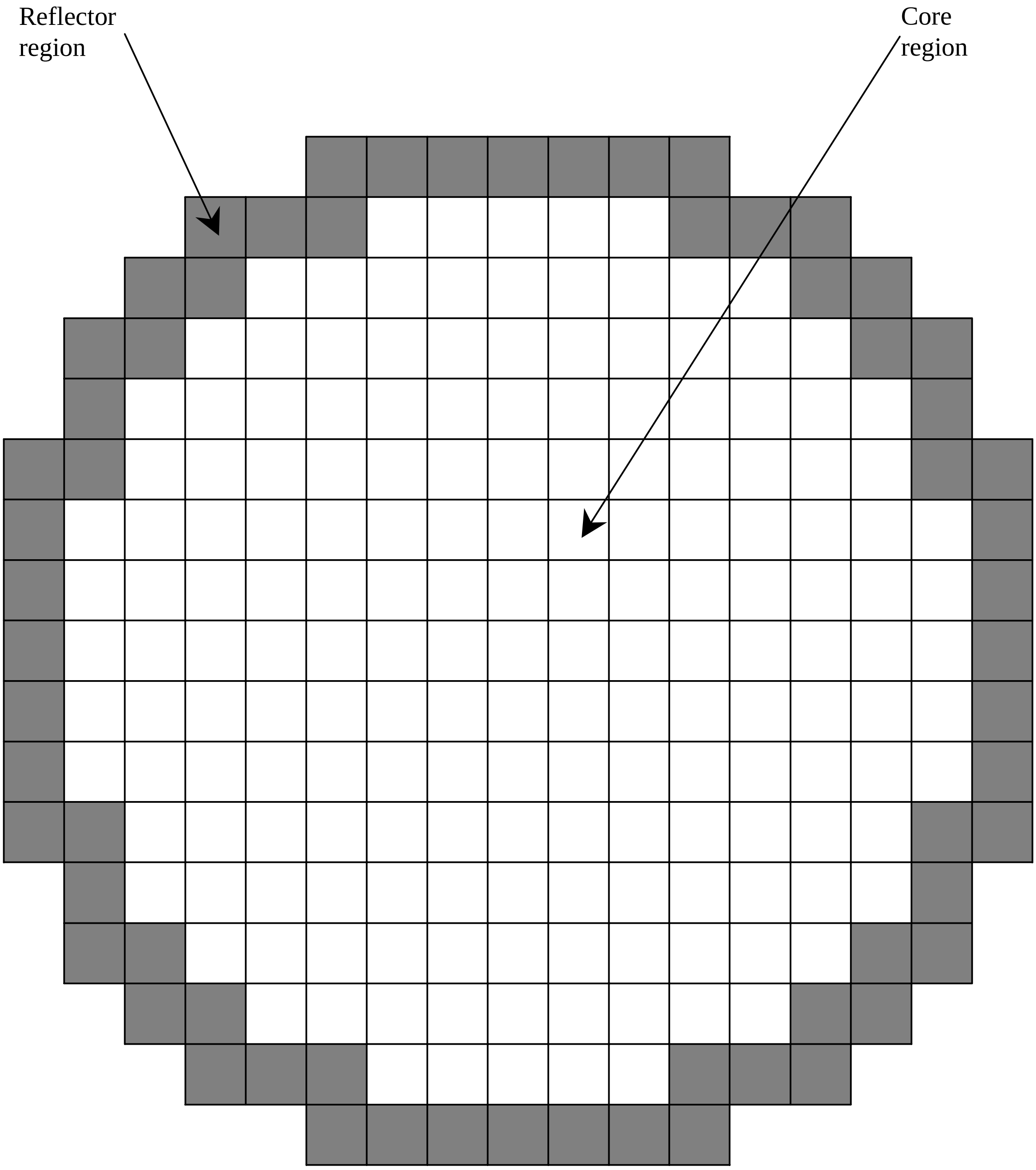


Figure 2.2.2. Arrangement of control rods

					1		6		1					
			3		5		5		3					
			7		8		7		8		7			
		3		5		4		4		5		3		
	1		8		6		2		6		8		1	
		5		4		2		2		4		5		
	6		7		2		7		2		7		6	
		5		4		2		2		4		5		
	1		8		6		2		6		8		1	
		3		5		4		4		5		3		
			7		8		7		8		7			
				3		5		5		3				
					1		6		1					

Bank	No. rods	Purpose
1	8	Safety
2	8	Safety
3	8	Safety
4	8	Safety
5	12	Regulating
6	8	Regulating
7	9	Regulating
8	8	APSR

Figure 2.4.1. Two-dimensional assembly type map

	8	9	10	11	12	13	14	15
H	1 52.863	2 30.192	3 56.246	4 30.852	5 49.532	6 28.115	7 53.861	8 55.787
K		9 57.945	10 30.798	11 55.427	12 29.834	13 53.954	14 25.555	15 49.166
L			16 57.569	17 30.218	18 54.398	19 27.862	20 23.297	21 47.300
M				22 49.712	23 28.848	24 52.846	25 40.937	
N					26 48.746	27 23.857	28 41.453	
O						29 37.343	A B	
P								
R								

A – Type of fuel assembly
B – Assembly average burn-up in GWD/T

Chapter 3

THERMAL-HYDRAULIC DATA

A TMI-1 RETRAN [4] thermal-hydraulic (T-H) skeleton input deck, as well as the TMI-1 RETRAN two-loop model nodalisation and steam line nodalisation diagrams, are provided in Appendix A.

3.1. Component specifications for the full thermal-hydraulic system model

3.1.1. Reactor vessel

The following tables provide all of the necessary data about the TMI-1 Reactor Vessel. Table 3.1.1.1 contains all of the design data, and Table 3.1.1.2 contains the volume data. Please note that the volumes and elevations in the skeleton input deck, given in Appendix A, are based on the nodalisation used for that deck as shown in Figure 3.1. They involve combining different physical regions together. In contrast, the values given in the tables are the actual physical volumetric data, as shown in Figure 3.1.1.1. For example, the core volume in the sample input deck consists of three regions of 198.6 ft^3 (5.624 m^3), for a total of 590.6 ft^3 (16.72 m^3). The value in Table 3.1.2 includes the regions included in the input deck calculation plus the core region below the active fuel of 132.16 ft^3 (3.742 m^3) for a total of 722 ft^3 (20.4 m^3). See Figure 3.1 for the way in which the core regions are defined in the RETRAN skeleton input deck. Figure 3.1 also shows the flow paths to the reactor vessel (RV) head. These are specified by Junction 8, which goes from the upper plenum to the RV head and has an initial flow of 3 104 lb/sec (1 409 kg/sec) and an area of 24.1 ft^2 (2.24 m^2), and Junction 9, which goes from the RV head to the outlet plenum, and has the same initial flow rate and a junction area of 8.564 ft^2 (0.797 m^2). Figures 3.1.1.1 through 3.1.1.10 provide more specific details about all of the components of the pressure vessel.

3.1.1.1. Characteristics of vent valves

Internal vent valves are installed in the core support shield to prevent a pressure imbalance which might interfere with core cooling following a postulated inlet pipe rupture. Under all normal operating conditions, the vent valve will be closed. In the event of a pipe rupture in the cold leg of the reactor loop, the valve will open to permit steam generated in the core to flow directly to the leak, and will permit the core to be rapidly recovered and adequately cooled after emergency core coolant has been supplied to the reactor vessel. Figure 3.1.1.3 provides additional information about the internal vent valves, including the pressure difference which they open.

Each valve assembly consists of a hinged disc, valve body with sealing surfaces, split-retaining ring and fasteners. Each valve assembly is installed into a machined mounting ring integrally welded in the core support shield wall. The mounting ring contains the necessary features to retain and seal the perimeter of the valve assembly. Also, the mounting ring includes an alignment device to maintain the correct orientation of the valve assembly for hinged-disc operation. Each valve assembly will be remotely handled as a unit for removal or installation. Valve components parts, including the disc, are of captured design to minimise the possibility of loss of parts to the coolant system, and all operating fasteners include a positive locking device. The hinged-disc includes a device for remote inspection of disc function. The vent valve materials were selected on the basis of their corrosion resistance, surface hardness, anti-galling characteristics and compatibility with mating materials in the reactor coolant environment.

The arrangement consists of eight 14 inch (35.56 cm) inside diameter vent valve assemblies installed in the cylindrical wall of the internals core support shield (refer to Figures 3.1.1.3 and 3.1.1.4). The valve centres are coplanar and are 42 inches (106.68 cm) above the plane of the reactor vessel coolant nozzle centres. In cross-section, the valves are spaced around the circumference of the core support shield wall.

The hinge assembly consists of a shaft, two valve body journal receptacles, two valve disc journal receptacles and four flanged shaft journals (bushings). Loose clearances are used between the shaft and journal inside diameters, and between the journal outside diameters and their receptacles. The valve disc hinge journal contains integral exercise lugs for remote operation of the disc with the valve installed in the core support shield.

The hinge assembly provides eight loose rotational clearances to minimise any possibility of impairment of disc-free motion in service. In the event that on a rotational clearance should bind in service, seven loose rotational clearances would remain to allow unhampered disc-free motion. In the worst case, at least four clearances must bind or seize solidly to adversely affect the valve disc-free motion.

In addition, the valve disc hinge loose clearances permit disc self-alignment so that the external differential pressure adjusts the disc seal face to the valve body seal face. This feature minimises the possibility of increased leakage and pressure-induced deflection loadings on the hinge parts in service.

The external side of the disc is contoured to absorb the impact load of the disc on the reactor vessel inside wall without transmitting excessive impact load to the hinge parts as a result of a loss-of-coolant accident.

3.1.2. Reactor coolant system

This section provides the participant with the data for the reactor coolant system (RCS). Table 3.1.2.1 summarises the basic RCS parameters, such as pressure drop. Table 3.1.2.2 provides the elevations for the primary system components, while Table 3.1.2.3 provides all of the data on the system piping. Tables 3.1.2.4 and 3.1.2.5 contain the flow areas and volume data for the system, respectively. Table 3.1.2.6 gives the system flow data, and Tables 3.1.2.7 and 3.1.2.8 list the system pressure settings and all of the safety valve data, respectively. Figures 3.1.2.1 through 3.1.2.5 provide specific detail about the RCS arrangement, the hot and cold leg pipes, the surge line and high pressure injection (HPI).

The components which contribute to the pressure difference in the RCS are described below:

1) *Flow distributor*

   The flow distributor is a perforated dished head with an external flange which is bolted to the bottom flange of the lower grid. The flow distributor supports the in-core instrument guide tubes and distributes the inlet coolant entering the bottom of the core.

2) *Thermal shield*

   A cylindrical stainless steel thermal shield is installed in the annulus between the core barrel cylinder and reactor vessel inner wall. The thermal shield reduces the incident gamma absorption internal heat generation in the reactor vessel wall and thereby reduces the resulting thermal stresses. The thermal shield upper end is restrained against inward and outward vibratory motion by restraints bolted to the core barrel cylinder. The lower end of the thermal shield is shrunk-fit on the lower grid flange and secured by 120 high strength bolts.

3) *Surveillance specimen holder tubes*

   Removed during the first refuelling due to flow vibration damage. The specimens are in the Crystal River reactor.

4) *In-core instrument guide tube assembly*

   The in-core instrument guide tube assemblies guide the in-core instrument assemblies from the instrument penetrations in the reactor vessel bottom head to the instrument tubes in the fuel assemblies. Horizontal clearances are provided between the reactor vessel instrument penetrations and the instrument guide tubes in the flow distributor to accommodate misalignment. Fifty-two in-core instrument guide tubes are provided and are designed so they will not be affected by the core drop.

The pressure differences in the RCS (based on RETRAN-3D and TRAC-PF1/NEM results) are shown below:

- Across the core (from the lower plenum to the upper plenum) – 29 psia (0.200 MPa).
- From the reactor vessel to steam generator (SG) inlet – 40 psia (0.2758 MPa).
- From the SG upper plenum to the SG lower plenum – 7 psia (0.048265 MPa).
- From the SG outlet to the reactor vessel – 77 psia (0.530915 MPa).

3.1.3. Steam generator

This section contains information about the once-through steam generator (OTSG). Table 3.1.3.1 gives the volume data and Tables 3.1.3.2 and 3.1.3.3 provide the design data. Figure 3.1.3.1 shows the OTSG in more specific detail.

The area of aspirator junction is equal to 10.33 ft^2 (0.961 m^2). The loss coefficient across the tube support plate is k = 5.3. The differential pressure across the OTSG is approximately 6 psi (0.041 MPa) and the pressure difference (based on the RETRAN-3D and TRAC-PF1/NEM results) in the steam line from the SG outlet to the turbine header is 41 psia (0.283 MPa).

3.1.3.1. Steam lines

The details of steam lines and steam header as modelled in RETRAN are provided in Appendix A (see Figure A.3). Table 3.1.3.4 gives the description of main steam safety valves (MSSV) per OTSG. In this benchmark they are modelled only on the intact steam line.

3.1.3.2 Feedwater system

An additional feedwater mass of 35 500 lbs (16 103 kg) in the piping between the feedwater isolation valve and the OTSG is simulated in this benchmark only for the affected OTSG. This additional feedwater is modelled as an extended feedwater flow rate vs. time boundary condition while preserving the specified mass (see Table 5.4.2).

The feedwater nozzle details are provided in order to calculate the loss coefficients associated with the feedwater nozzles. The feedwater enters the OTSG through 32 spray nozzles, each containing 82 holes 0.188 inch (0.478 cm) in diameter, connected to the 14 inch (35.56 cm) outside diameter (OD) main feedwater header. The arrangement of these nozzles can be seen in Figures 3.1.3.2 through 3.1.3.5.

3.2. Definition of the core thermal-hydraulic boundary conditions model

The full TMI-1 thermal-hydraulic (T-H) model can be converted to a core T-H boundary condition model by defining an inlet condition at the vessel bottom and an outlet condition at the vessel top. The boundary conditions (BC) for this model are provided on the CD-ROM and at the benchmark **ftp** site under the directory **3D-BC**. The BC are calculated using the TRAC-PF1/NEM best-estimate core plant system code. Radial distributions are provided for 18 T-H cells (from 1 to 18) as shown in Figure 3.2.1. These 18 T-H cells are coupled to the neutronic core model in the radial plane as shown in Figure 3.2.3. This mapping scheme follows the spatial mesh overlays developed for the TMI-1 TRAC-PF1/NEM model.

The TMI-1 TRAC-PF1 model is a 3-D vessel model in cylindrical geometry. A majority of the existing coupled codes model the core thermal-hydraulically using parallel channel models, which leads to difficulties in the interpretation of the mass flow BC. In order to avoid this source of modelling uncertainty, BC have been generated where mass flows are corrected for direct use as input data in the multi-channel core models. A geometrical interpolation method is used to process the TRAC-PF1/NEM BC in order to obtain inlet conditions for each assembly. The detailed mapping scheme (see Figure 3.2.3) shows how the provided 18 BC values are distributed per assembly. For the central row of assemblies, it is recommended that the average values of the upper and lower halves be used.

There are several files of data, available at the ftp site and on the CD-ROM, that are used for definition of the core T-H boundary condition model (Exercise #2). This data is taken from the best-estimate core plant system code calculations performed with the PSU version of TRAC-PF1/NEM:

A) File TEMP.BC
The transient inlet radial distribution of liquid temperatures from 0 s to 100 s during the transient. The values are extracted from the TRAC vessel axial layer #3 (see Figure 3.2.2) which is mapped to the bottom reflector of the core neutronics model. For each time interval the first number is the time, followed by 18 numbers corresponding to the liquid temperatures (°K) in the 18 azimuthal sectors that make up the core region.

B) File MASS_FLOWS.BC
The same as above for the inlet radial distribution of axial mass flows (kg/s).

C) File Press_Inlet.BC
The same as above for the inlet radial distribution of pressure (Pa).

D) File Press_Outlet.BC
The same as above for the outlet radial distribution of pressure (Pa). The values in the file Press_Outlet.BC are extracted from the TRAC vessel axial layer #10, which is mapped to the top reflector.

The average pressure drop between the third and tenth layers (i.e. across the core) at time = 0 s is about 0.136 MPa. In the TRAC nodalisation (Figure 3.2.2) the lower plenum is represented with the first, second and third layers. The upper plenum is represented with the eleventh and twelfth layers. The average pressure drop between the plenums at t = 0 s is 0.199 MPa.

At the request of some participants; a new file, Cold Temp.BC, is provided. The file contains cold leg temperatures for the two loops (the first column is for Loop A – broken loop – and the second column is for Loop B – intact loop) as a function of time.

3.3. Thermal-physical and heat-transfer specifications

The Doppler temperature, T_f, is found from the fuel temperature at the fuel rod centre $T_{f,c}$ and the fuel rod surface $T_{f,s}$ via the relation:

$$T_f = (1-\alpha)T_{f,c} + \alpha T_{f,s}$$

where α is equal to 0.7.

The UO_2 density without dishing is 10.412 g/cm^3 (650.02 lb/ft^3 – 95% of the theoretical density) at a temperature of 20°C (68°F). The pellet dishing amounts to 1.956%. The cladding material is zircaloy-4 with a density of 6.6 g/cm^3 (412.03 lb/ft^3).

Participants should use the following reference relations for the heat conductivity λ(W/m°K) and specific heat capacity c_p(J/kg°K) of fuel and cladding:

$$\lambda_{UO_2} = 1.05 + 2150/(T - 73.15)$$

$$\lambda_{zircaloy-4} = 7.51 + 2.09\ 10^{-2}\ T - 1.45\ 10^{-5}\ T^2 + 7.67\ 10^{-9}\ T^3$$

$$c_{p,UO_2} = 162.3 + 0.3038\ T - 2.391\ 10^{-4}\ T^2 + 6.404\ 10^{-8}\ T^3$$

$$c_{p,zircaloy-4} = 252.54 + 0.11474\ T$$

where T is the temperature (°K).

Expansion effects of fuel and cladding will not be considered in this benchmark.

The conductance of the helium filled gap between fuel and cladding (K_{gap}) is assumed to be constant:

$$K_{gap} = 11\ 356\ W/m^2 {}^\circ K\ (3\ 601\ BTU/ft^2 {}^\circ K)$$

The heat transfer coefficient between cladding and moderator has to be calculated using code specific correlations.

Table 3.1.1.1. Reactor vessel design data

Item	Data
Operating temperature (°F/°C)	605/318
Overall height of vessel and closure head (ft/m)	40.74/12.42
Straight shell minimum thickness (in/cm)	8.438/21.43
Water volume (core and internals in place) (ft^3/m^3)	4 010/113.6
Thickness of insulation (in/cm)	4.00/10.2
Flange ID (in/cm)	167.50/425.50
Shell ID (in/cm)	171/434
Inlet nozzle ID (in/cm)	28.0/71.1
Outlet nozzle ID (in/cm)	36.0/91.4
Core flooding water nozzle ID (in/cm)	11.50/29.21
Coolant operating temperature – inlet (°F/°C)	555/291
Coolant operating temperature – outlet (°F/°C)	605/318
Reactor coolant flow (lb/hr /kg/hr)	$139.7 \times 10^6/63.37 \times 10^6$
Closure head minimum thickness (in/cm)	6.625/16.83
Lower head minimum thickness (in/cm)	5.00/12.7
Control rod drive nozzles ID (in/cm)	2.76/7.01
Axial power shaping rod drive nozzles ID (in/cm)	2.76/7.01

Table 3.1.1.2. Reactor vessel volume data

Description	Value
Lower plenum (ft^3/m^3)	292/8.27
Core (ft^3/m^3)	722/20.4
Down comer (ft^3/m^3)	1 225/34.69
Upper plenum (ft^3/m^3)	776/21.97
Upper head (ft^3/m^3)	508/14.4

Table 3.1.2.1. Reactor coolant system steady-state parameters

Parameter	Value
Total core power output (MWt)	2 772
Design core flow available for heat transfer (10^6 lb/hr / 10^6 kg/hr)	129.5/58.74
Core flow area available for heat transfer (ft^2/m^2)	49.2/4.57
Core pressure drop (psi/kPa)	29.0/200
Reactor coolant system pressure drop (psi/kPa)	107/738
Unrecoverable core pressure drop (psi/kPa)	18.7/129
Average core coolant velocity (ft/sec / m/sec)	16.5/5.03
Cold leg coolant velocity (ft/sec / m/sec)	48.2/14.69
Hot leg coolant velocity (ft/sec / m/sec)	63.8/19.45

Table 3.1.2.2. Primary system component elevations

Component	Elevation (ft-in / m-cm)
Reactor outlet piping	0-0 (reference point)/0-0
Reactor vessel lower head*	(-) 24-0/7.32-0
Steam generator lower head*	(-) 29-0/8.84-0
Pressuriser lower head*	(-) 3-8.5/0.914-21.6
RC pump discharge piping*	(+) 3-6/0.914-15.2

* Elevations referenced to reactor outlet piping centreline

Table 3.1.2.3. Reactor coolant system piping design data

Parameter		Value
Reactor inlet piping	Pipe, ID (in/cm)	28.0/71.1
	Minimum thickness (in/cm)	2.25/5.72
	Coolant volume (hot-system total) (ft^3/m^3)	950/26.9
Reactor outlet piping	Pipe, ID (in/cm)	36.0/91.4
	Minimum thickness (in/cm)	2.875/7.303
	Coolant volume (hot-system total) (ft^3/m^3)	938/26.6
Pressuriser surge piping	Pipe size (in/cm)	10.0/25.4
	Coolant volume, hot (ft^3/m^3)	20.0/0.566

Table 3.1.2.4. Flow areas

Component	Location	Flow area (ft^2/m^2)
Hot leg	Flowmeter	7.07/0.657 typical (minimum of 6.6 at flowmeter)
Cold leg		4.30/0.399
Steam generator	RC inlet nozzle	7.07/0.657
Reactor vessel	RC outlet nozzle	14.1/1.31 (7.0756 ft^2/loop $\times$ 2)
RC pump	Outlet	4.30/0.399

Table 3.1.2.5. Reactor coolant system volume data

Description	Value
Pressuriser (at 220 in. (558.8 cm) water level)	
Water volume (ft^3/m^3)	800/22.7
Steam volume (ft^3/m^3)	700/19.8
Cold leg – each (ft^3/m^3)	237.5/6.73
Hot leg – each (ft^3/m^3)	469/13.3
Reactor coolant pumps (ft^3/m^3)	56.0/1.59
Surge line (ft^3/m^3)	20.0/0.566
Containment	
Free volume (ft^3/m^3)	$2.116 \times 10^6/5.99 \times 10^4$
Sprayed volume (ft^3/m^3)	$1.629 \times 10^6/4.61 \times 10^4$

Table 3.1.2.6. Reactor coolant system flow data

Description	Value
Total reactor flow (lb/hr / kg/hr)	$139.7 \times 10^6/63.4 \times 10^6$
Average flow path lengths (ft/m)	
Hot leg	66/20.1
Cold leg	53/16.2
Steam generator	70/21.3
Reactor vessel reactor	70/21.3
Coolant pump	18/5.49
Reactor coolant pumps	
4 pumps (gpm/pump / kg/sec/pump)	93 720/5 668
Maximum letdown flow (gpm / kg/sec)	140/8.47
High pressure injection	
3 pumps (gpm/pump / kg/sec/pump)	300/18.14
Low pressure injection	
2 pumps (gpm/pump / kg/sec/pump)	3 000/181.4

Table 3.1.2.7. Reactor coolant system pressure settings

Reactor trip	Pressure (psig/MPa)
High pressure reactor trip[a]	2 355/16.3
Low pressure reactor trip[a]	1 930/13.4

[a] At sensing nozzle on reactor outlet pipe

Table 3.1.2.8. Safety valve data

Description	Value
Pressuriser code safety valves	
Pressure set point (psig/MPa)	2 500/17.0
Capacity (lb/hr / kg/hr) total	690 000/312 979

Table 3.1.3.1. Steam generator volume data

Description	Value
Lower plenum (ft^3/m^3)	277/7.84
Upper plenum (ft^3/m^3)	281/7.96
Secondary side (ft^3/m^3)	3 412/96.62

Table 3.1.3.2. Steam generator design data

Item	Data per steam generator
Steam conditions at full load, outlet nozzles	
Steam flow (lb/hr / kg/hr)	$6.04 \times 10^6/2.74 \times 10^6$
Steam temperature (°F/°C)	571/299 (35°F/20°C superheat)
Steam pressure (psig/MPa)	915/6.41
Feedwater temperature (°F/°C)	460/238
Reactor coolant flow (lb/hr / kg/hr)	$69.85 \times 10^6/31.68 \times 10^6$
Reactor coolant side	
Operating pressure (psig/MPa)	2 155/14.96
Operating temperature (°F/°C)	
Outlet	605/318
Inlet	555/291
Coolant volume – hot (ft^3/m^3)	2 017/57.12

Table 3.1.3.3. Steam generator design data

Item	Data per steam generator
Secondary side	
Operating pressure (psig/MPa)	915/6.41
Net volume (ft^3/m^3)	3 412/96.62
Dimensions	
Tubes – OD/min. wall (in/cm)	0.625/0.034 / 1.59/0.086
Number of tubes	15 500
Overall height – including skirt (ft/m)	73.208/22.314
Shell – OD (in/cm)	151.125/383.858
Shell minimum thickness – at tube sheets and feedwater connect (in/m)	60 625/1539.9
Shell minimum thickness (in/cm)	4.1875/10.636
Tube sheet – thickness (in/cm)	24.0/60.96
Dry weight (lb/kg)	1 144 500/519 136
Exposed tube length (ft/m)	53.375/16.269
Nozzles – reactor coolant side	
Inlet nozzle ID (in/cm)	36.0/91.4
Outlet nozzle ID (in/cm)	28.0/71.1
Drain nozzle (in/cm)	1.00/2.54, Sch 160
Manway ID (in/cm)	16.0/40.6
Handholes (in/cm)	5.00/12.7
Nozzles – secondary side	
Steam nozzle OD (in/cm)	48.5/123.2
Vent nozzle (in/cm)	1.50/3.81, Sch 80
Drain nozzle (in/cm)	1.50/3.81, Sch 80
Drain nozzle (in/cm)	1.00/2.54, Sch 80
Level sensing nozzle (in)	1.00/2.54, Sch 80
Thermowell ID (in/cm)	0.375/0.953
Manway ID (in/cm)	16.0/40.6
Feedwater nozzle (in/cm)	14.0/35.6, Sch 80
Auxiliary feedwater nozzle (in/cm)	6.00/15.2, Sch 80
Handholes diameter (in/cm)	5.00/12.7

Table 3.1.3.4. Description of MSSVs per OTSG

Description of safety valves	Open setpoint		Close setpoint		Rated flow per valve at 3% accumulation	
	psia	Bar	psia	Bar	LBM/HR	Kg/Hr
Small safety (1 valve)	1 055.0	72.73	1 012.5	69.8	194 900	88 407
Safety bank 1 (1 valve)	1 065.0	73.42	1 022.0	70.46	824 265	373 887
Safety bank 2 (2 valves)	1 065.0	73.42	1 022.0	70.46	792 610	359 528
Safety bank 3 (2 valves)	1 075.0	74.11	1 031.5	71.12	799 990	362 875
Safety bank 4 (2 valves)	N/A	N/A	N/A	N/A	N/A	N/A
Safety bank 5 (1 valve)	N/A	N/A	N/A	N/A	N/A	N/A

Figure 3.1. RETRAN vessel nodalisation

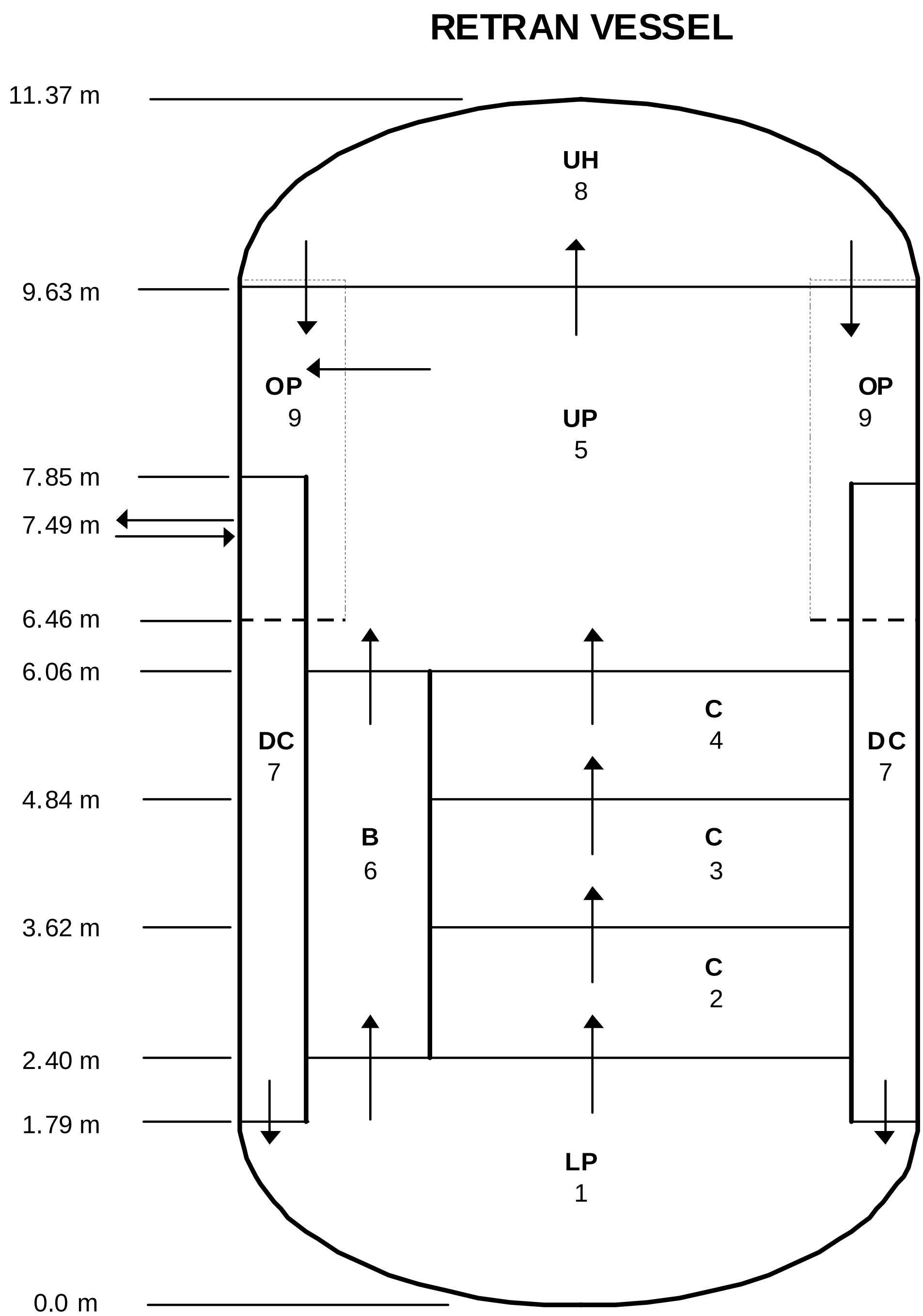


Figure 3.1.1.1. Pressure vessel

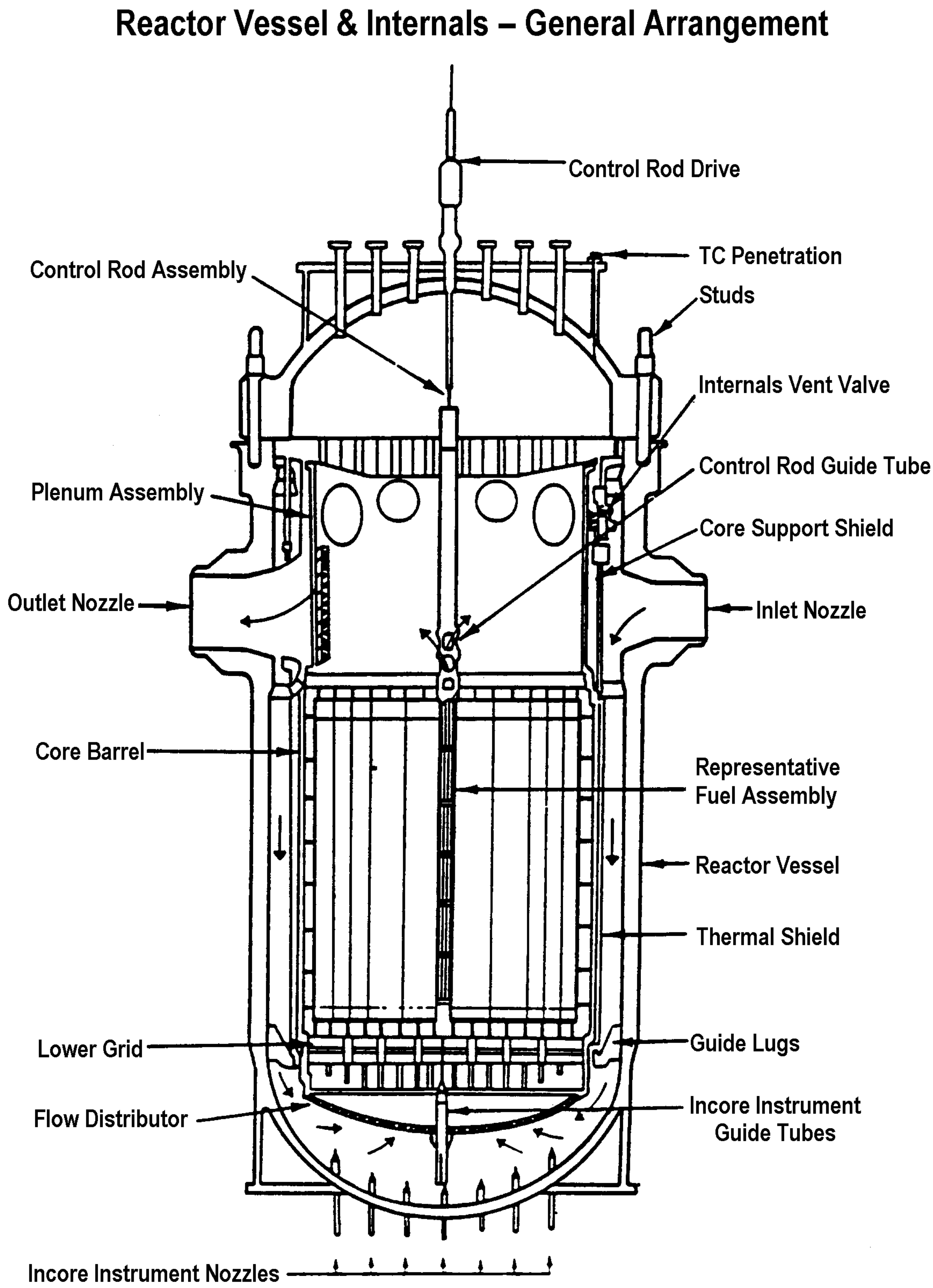


Figure 3.1.1.2. Upper plenum

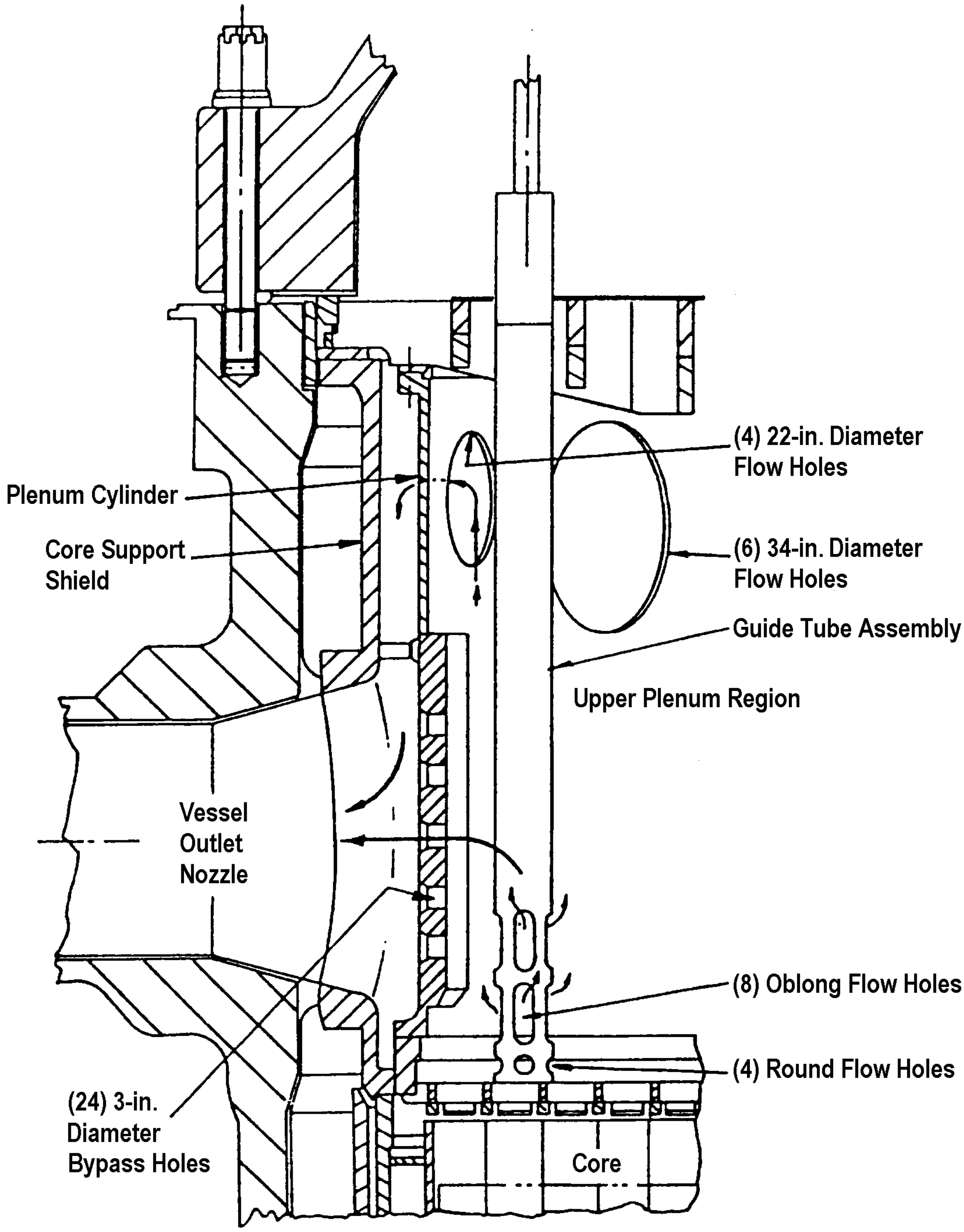


Figure 3.1.1.3. Internal vent valves

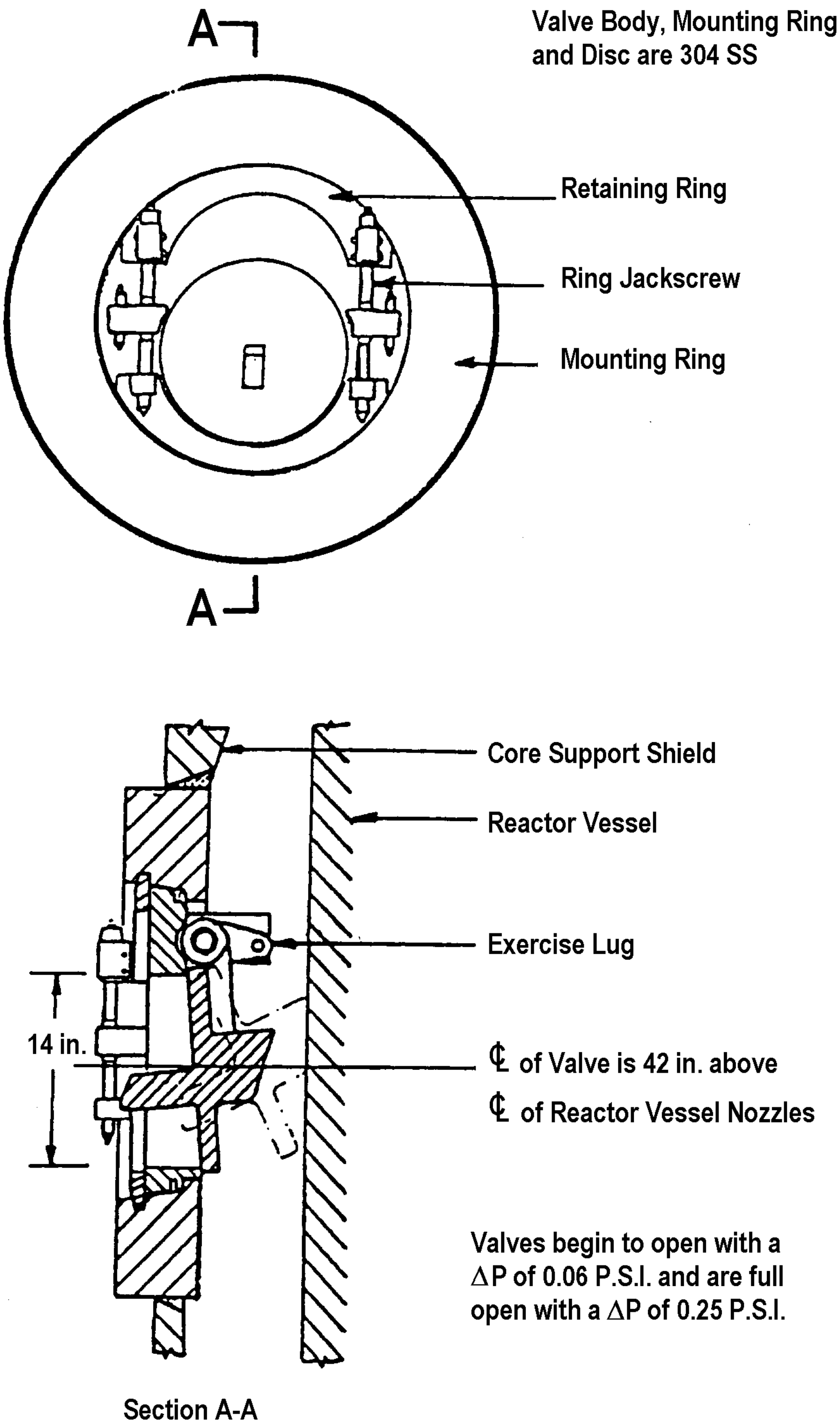


Figure 3.1.1.4. Core flooding arrangement

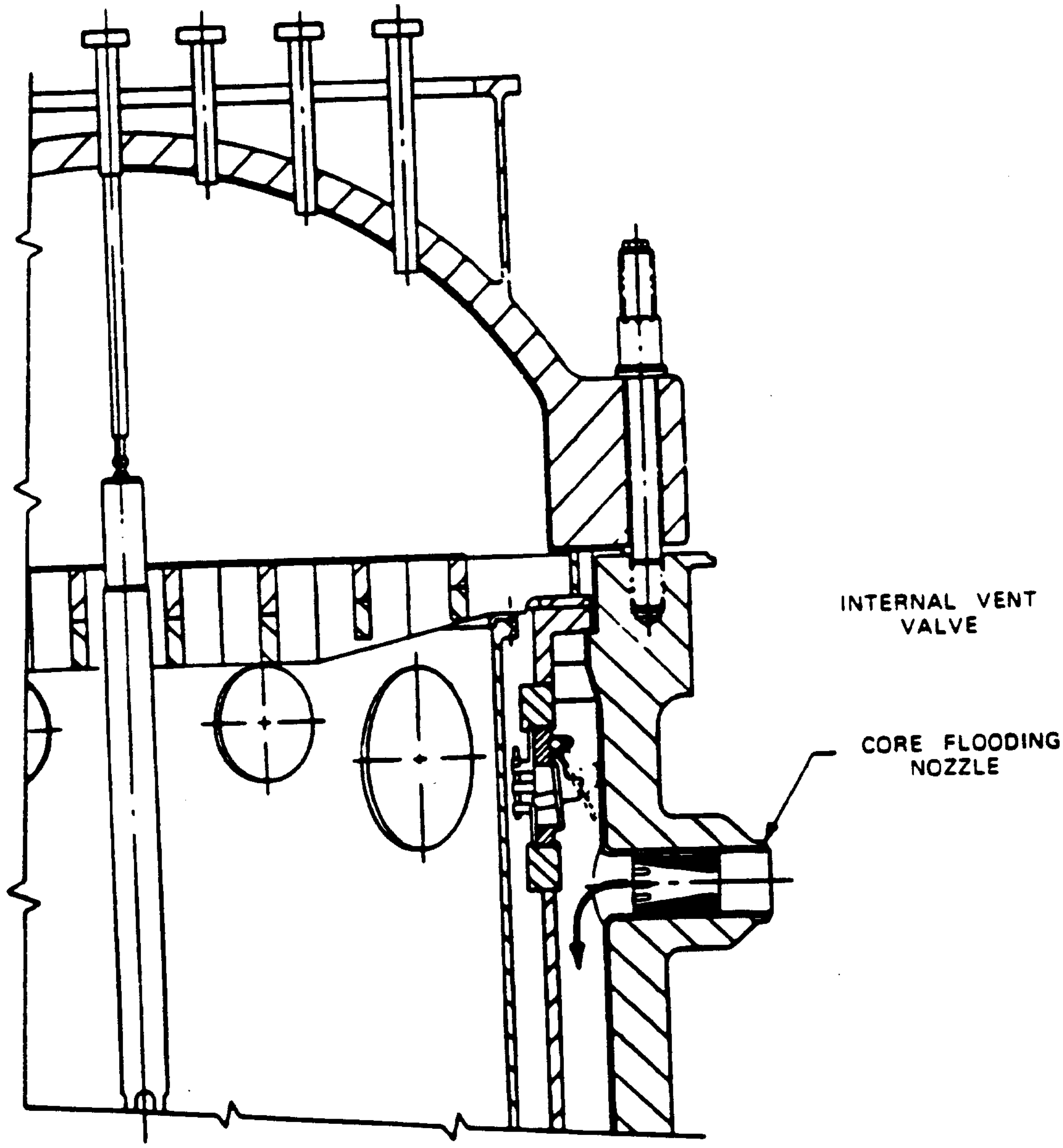


Figure 3.1.1.5. Upper plenum cover

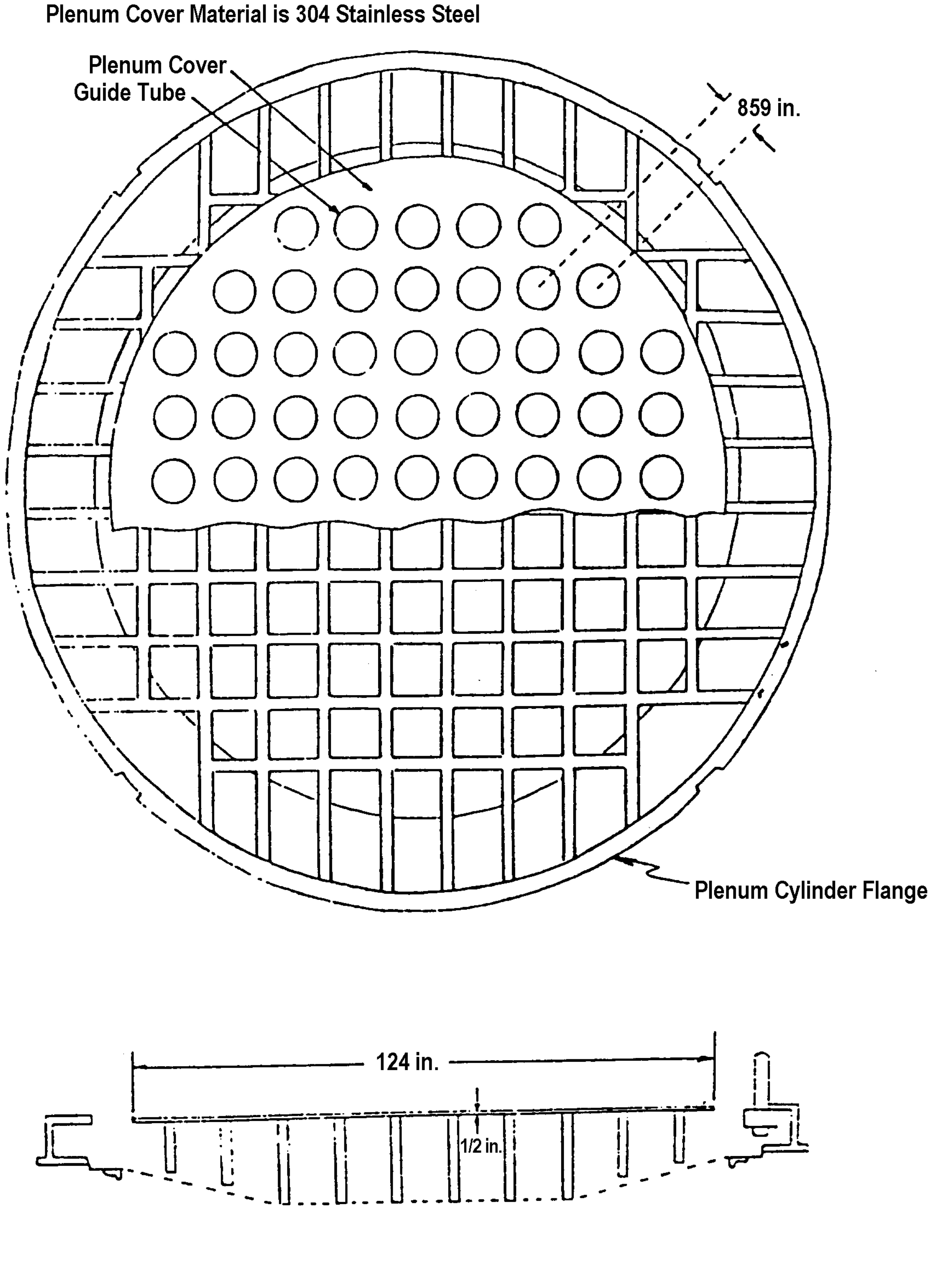


Figure 3.1.1.6. Plenum cylinder

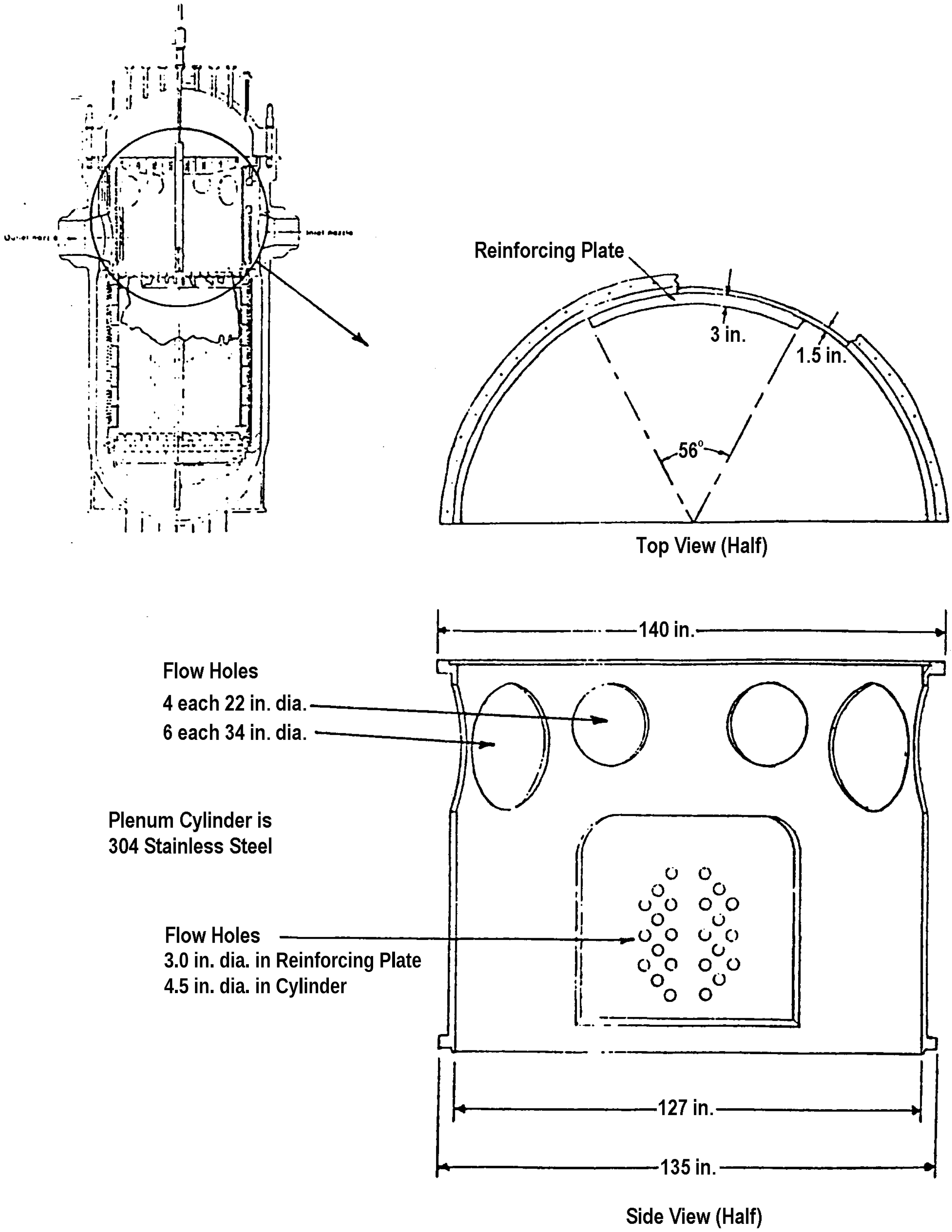


Figure 3.1.1.7. Control rod guide tube

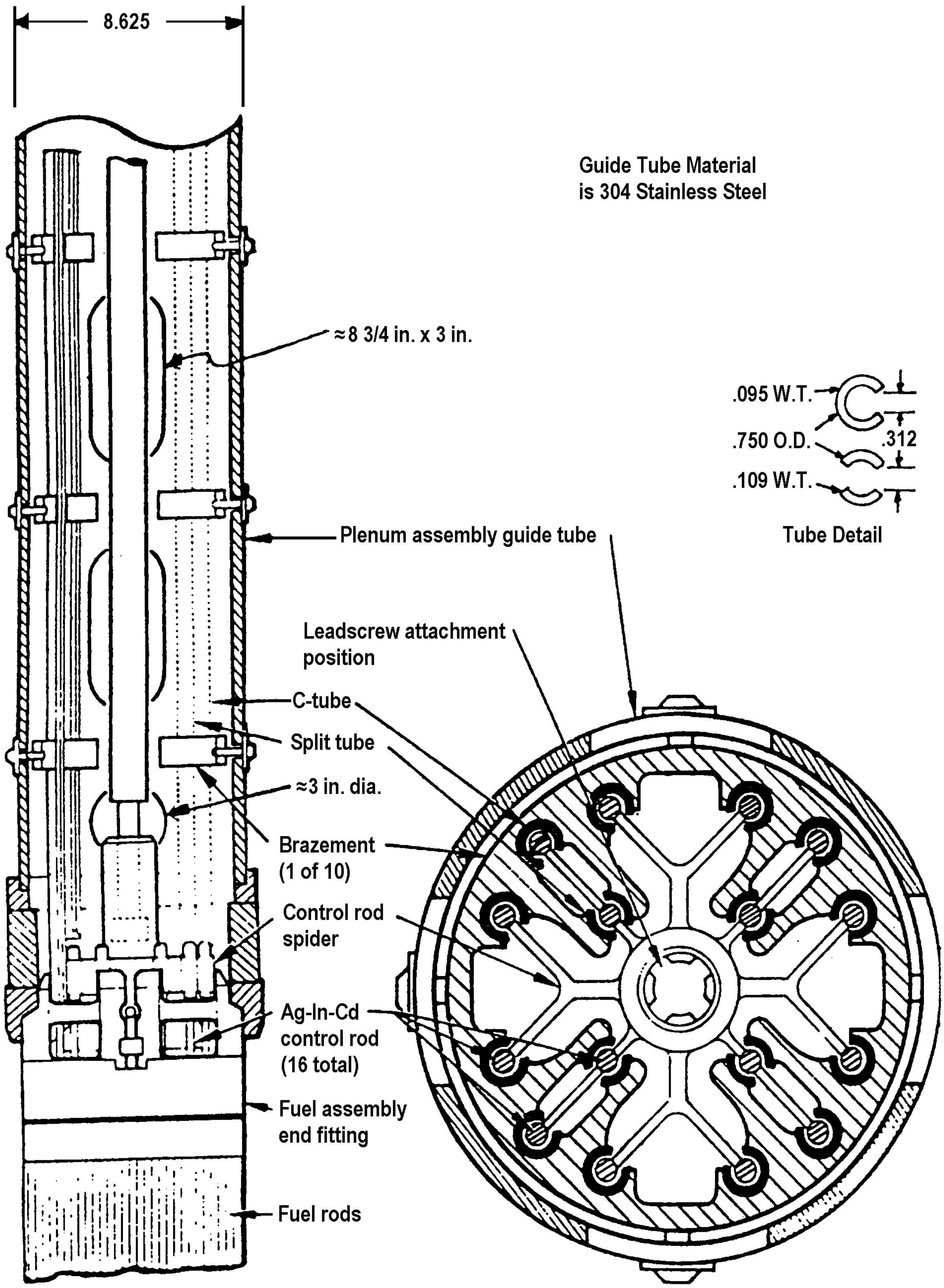


Figure 3.1.1.8. General arrangement of core support structure

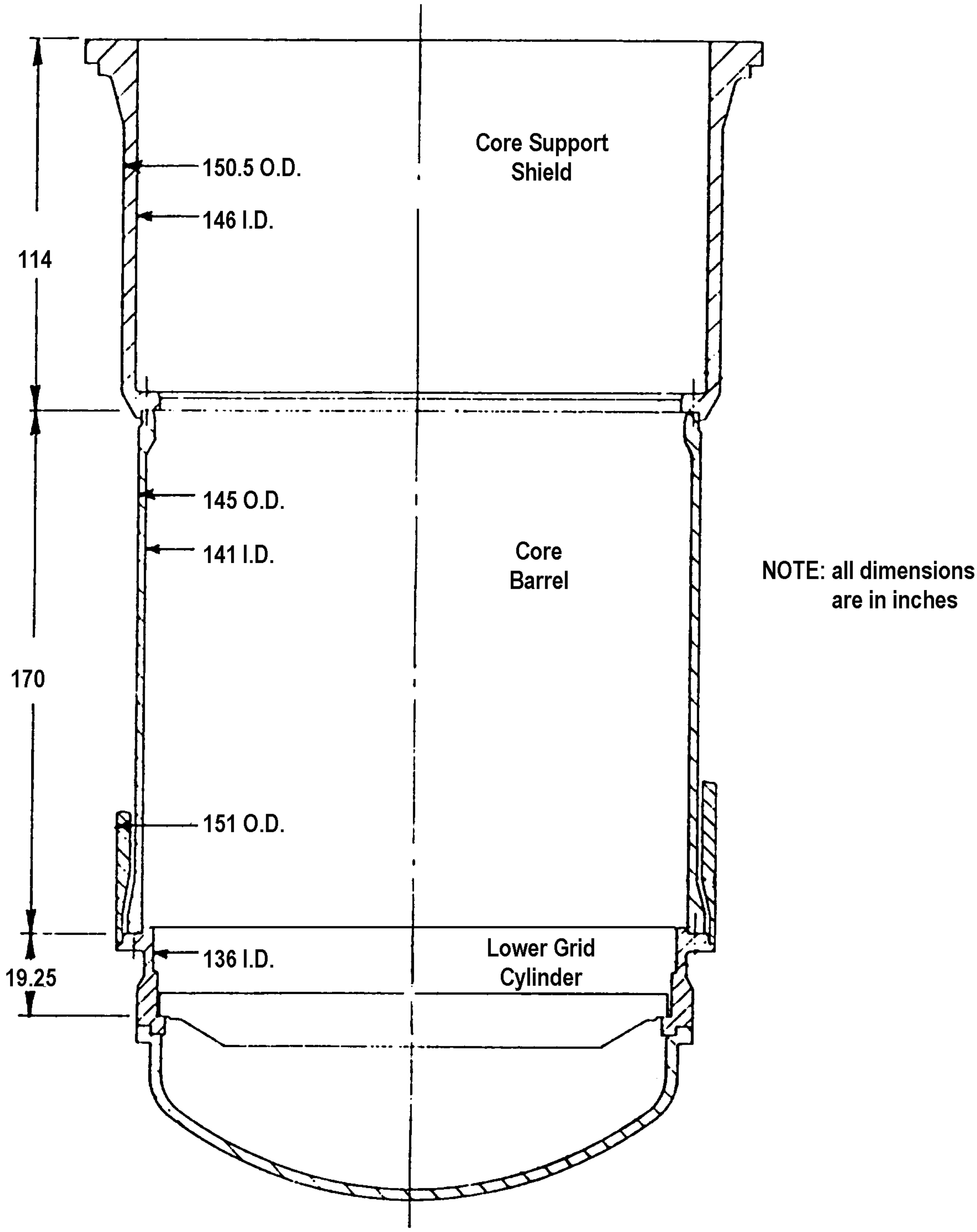


Figure 3.1.1.9. Lower grid assembly

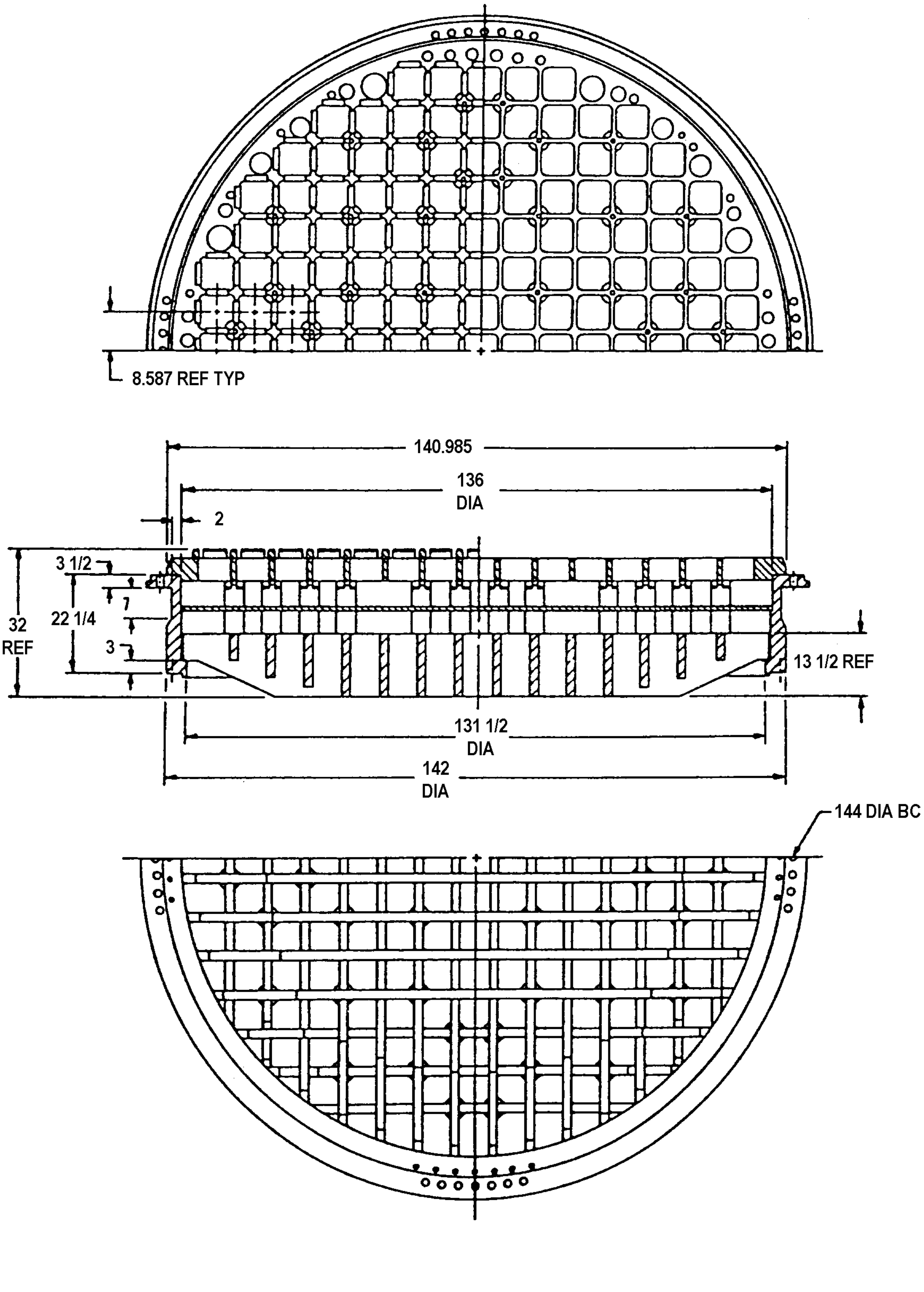


Figure 3.1.1.10. Lower plenum cross-section

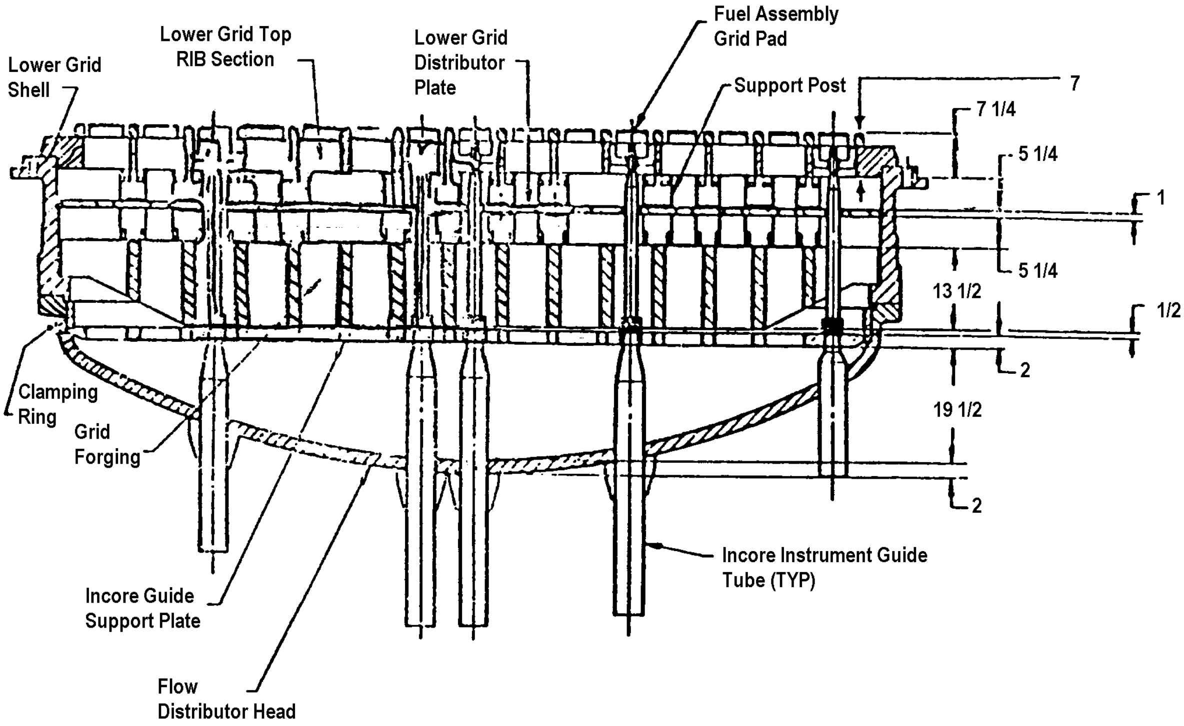


Figure 3.1.2.1. Reactor coolant system arrangement – plan

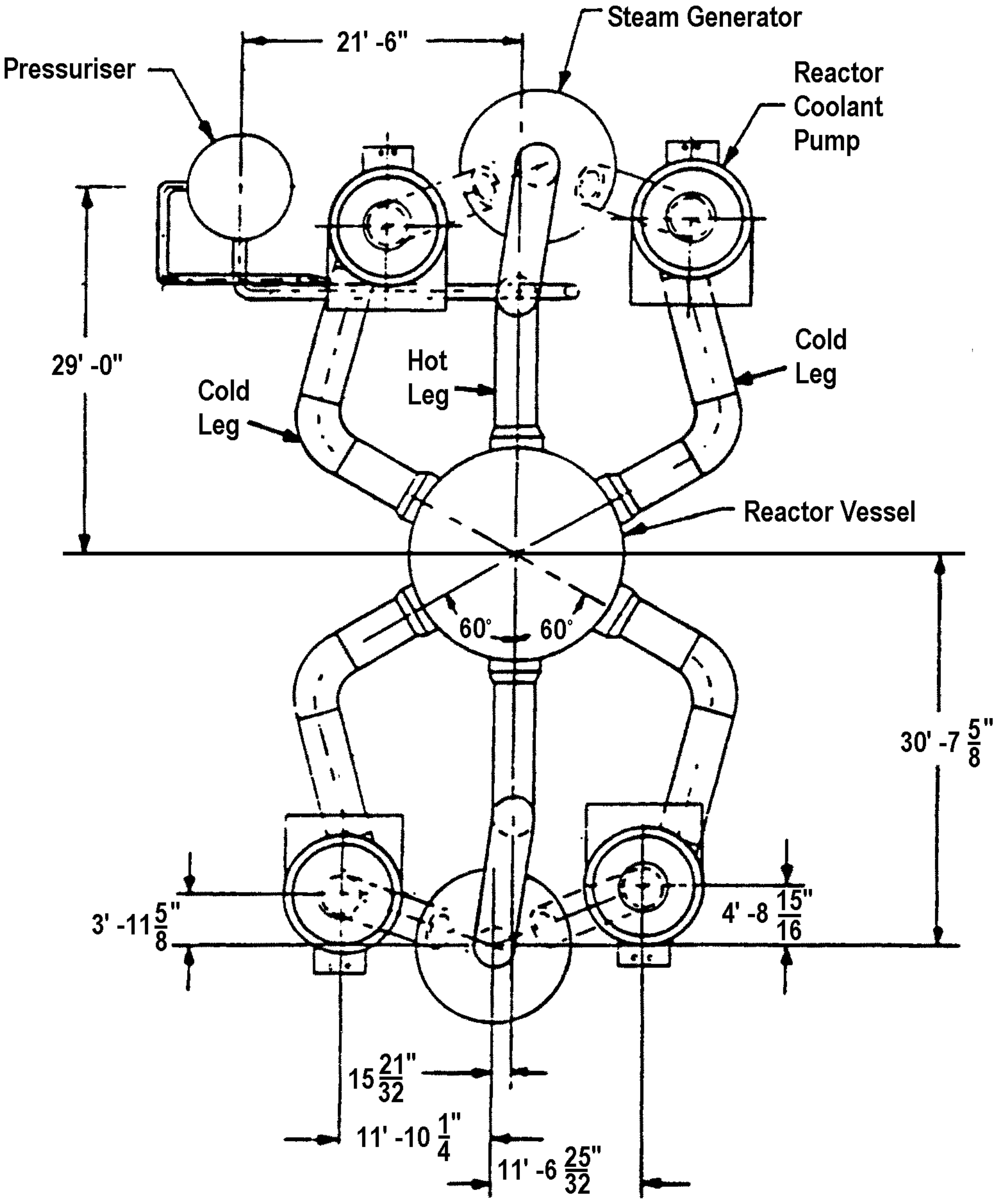


Figure 3.1.2.2. Reactor coolant system arrangement – elevation

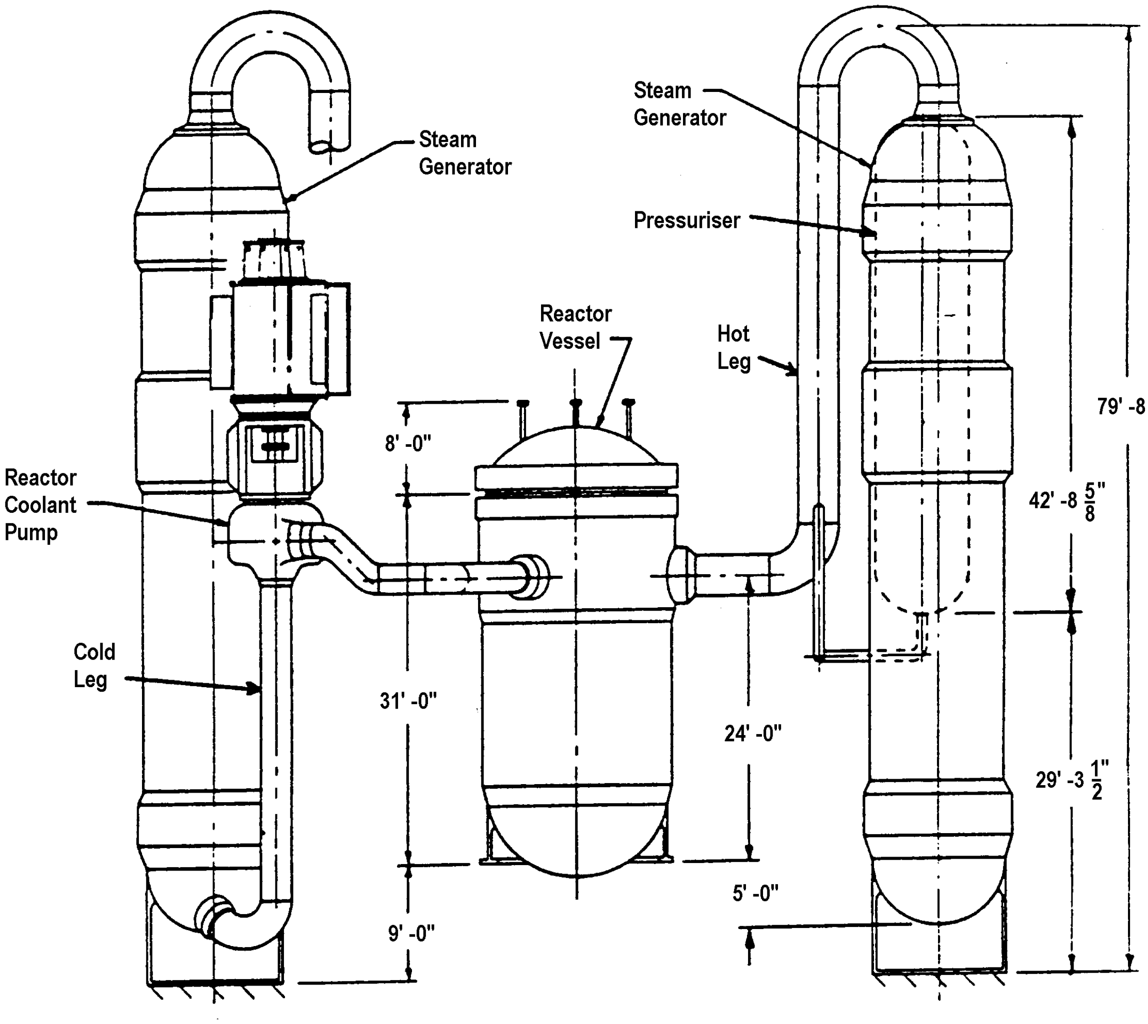


Figure 3.1.2.3. A & B loop hot leg pipe

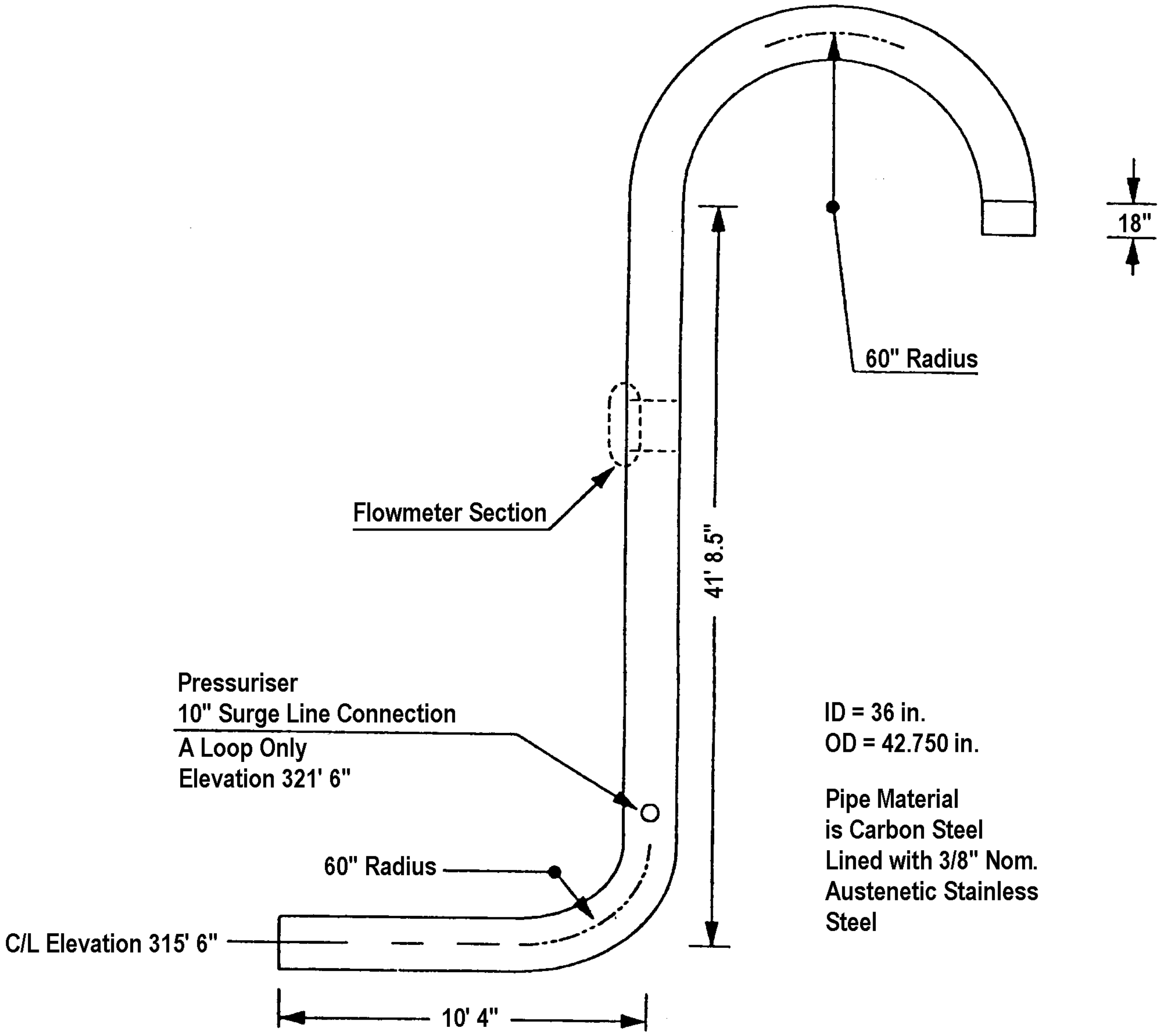


Figure 3.1.2.4. A & B loop cold leg pipes

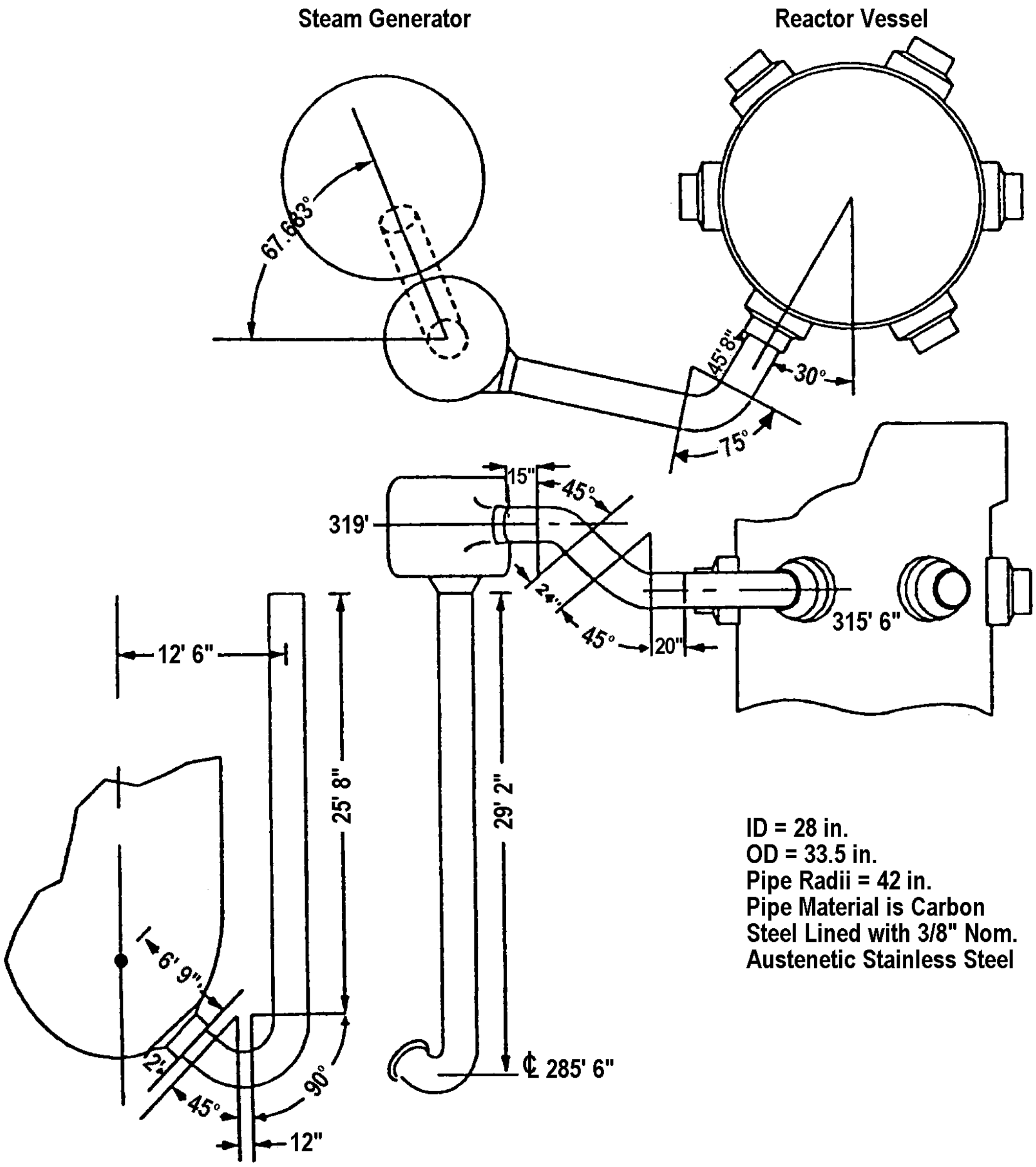


Figure 3.1.2.5. Surge and spray line details

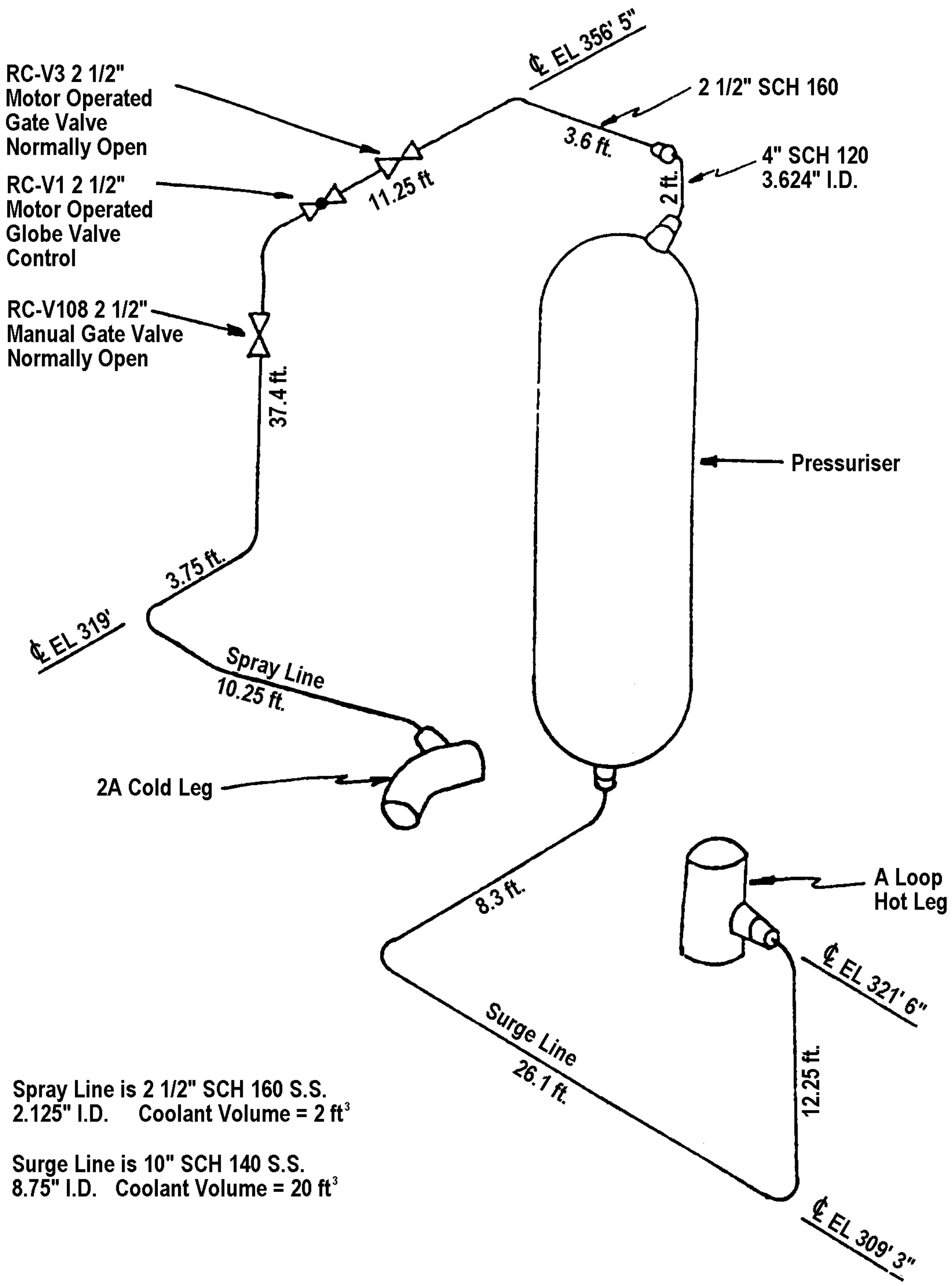


Figure 3.1.3.1. Once-through steam generator

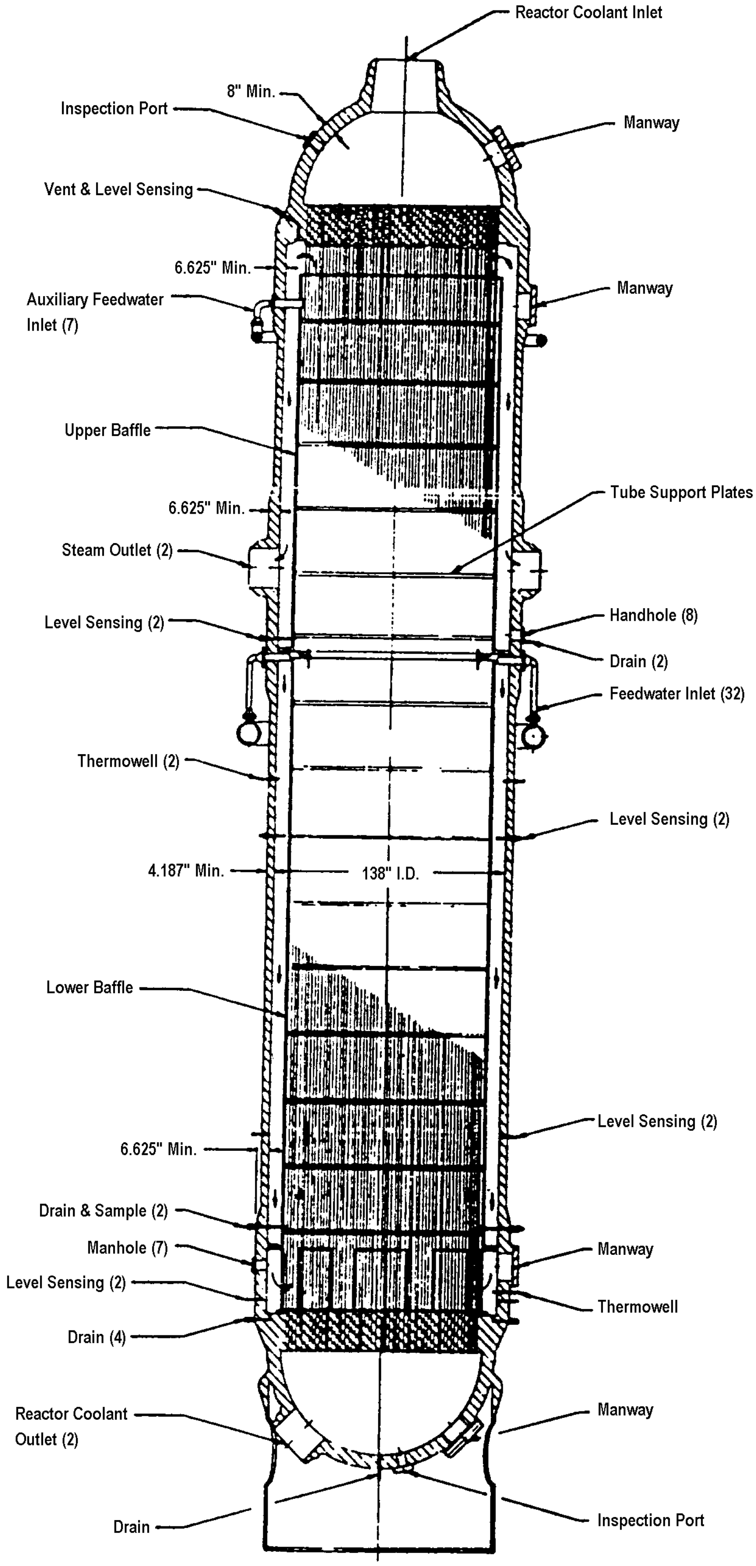


Figure 3.1.3.2. Main feedwater header and nozzle

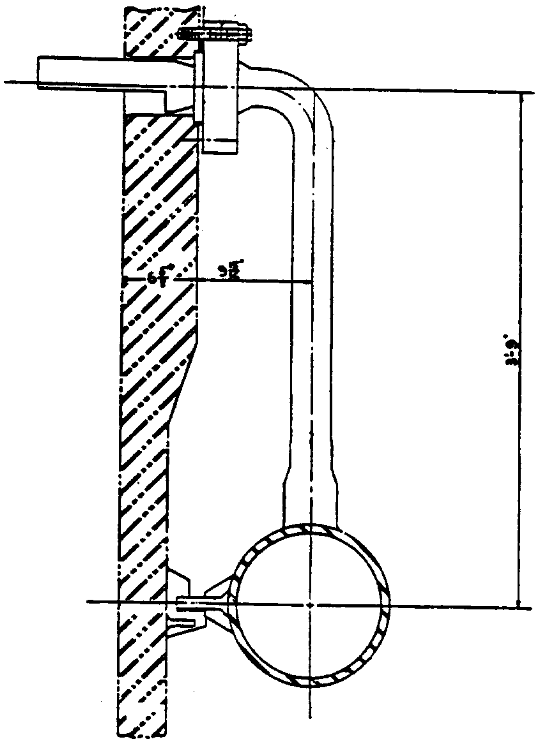


Figure 3.1.3.3. Feedwater spray nozzle

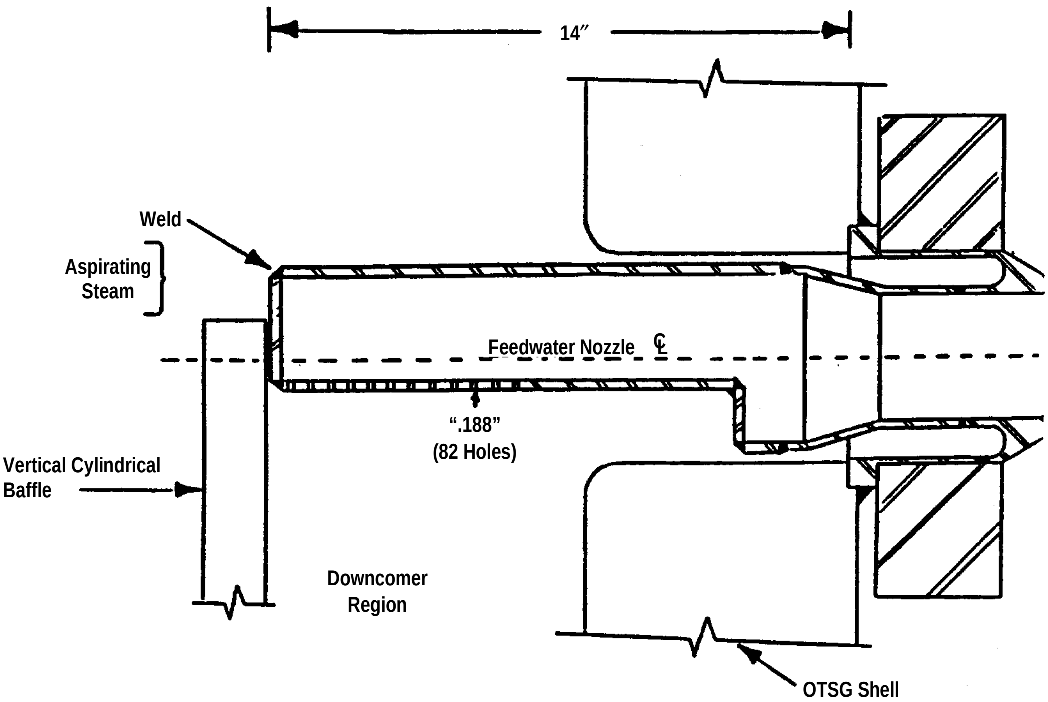


Figure 3.1.3.4. OTSG feedwater headers

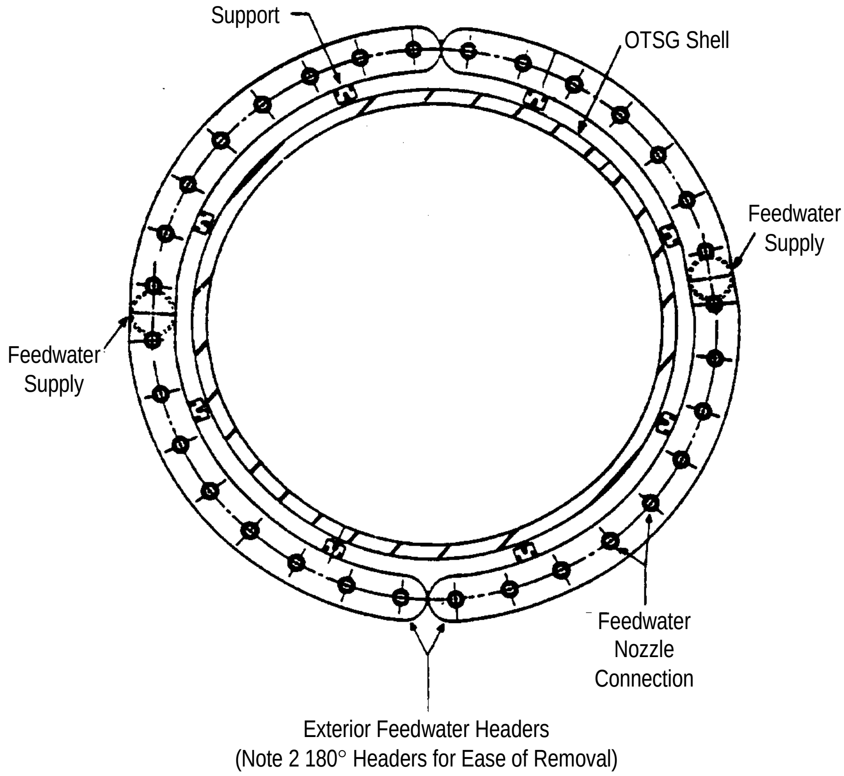


Figure 3.1.3.5. Emergency feedwater nozzle

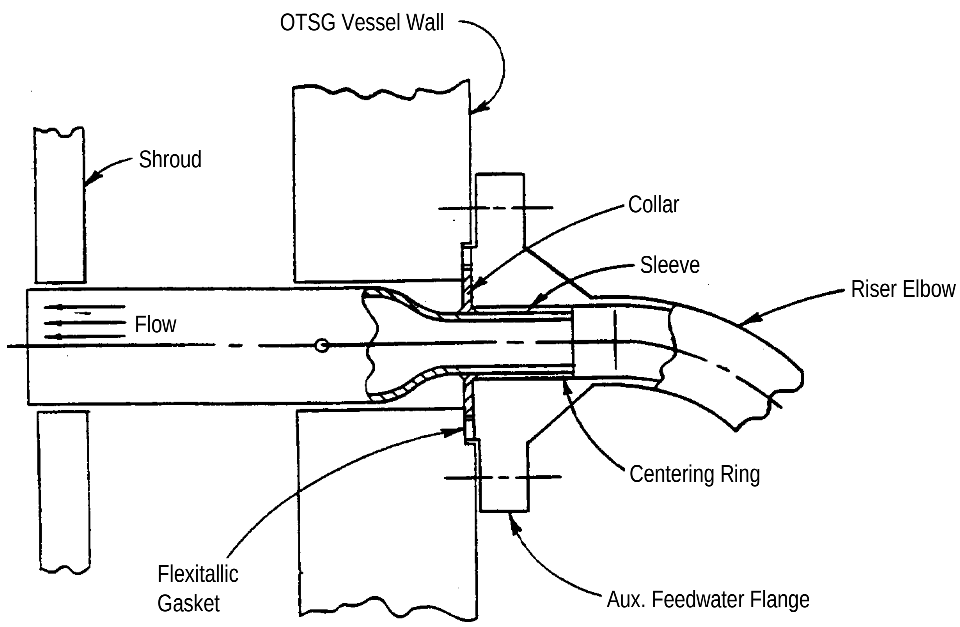


Figure 3.2.1. TRAC-PF1 vessel radial and azimuthal nodalisation

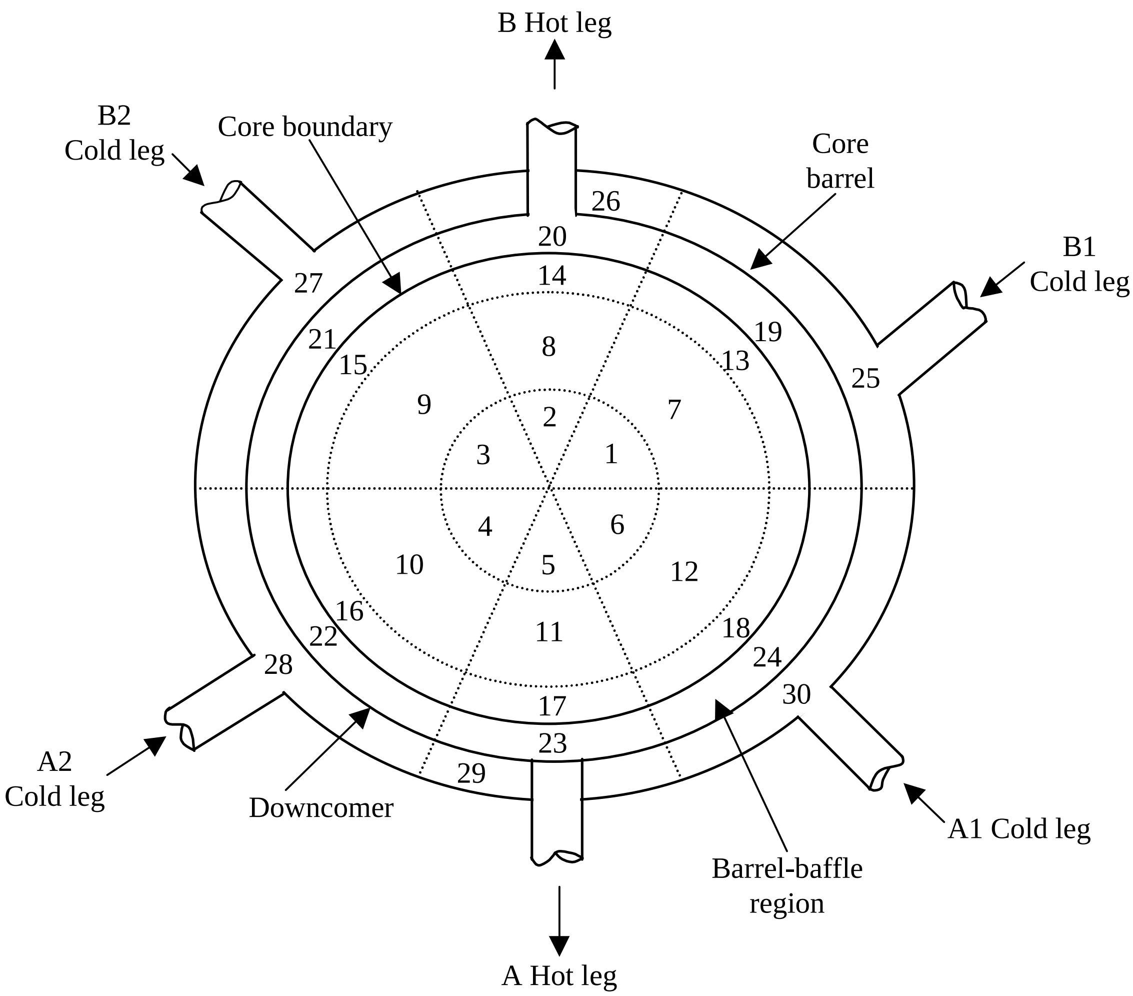


Figure 3.2.2. TRAC-PF1 vessel axial nodalisation

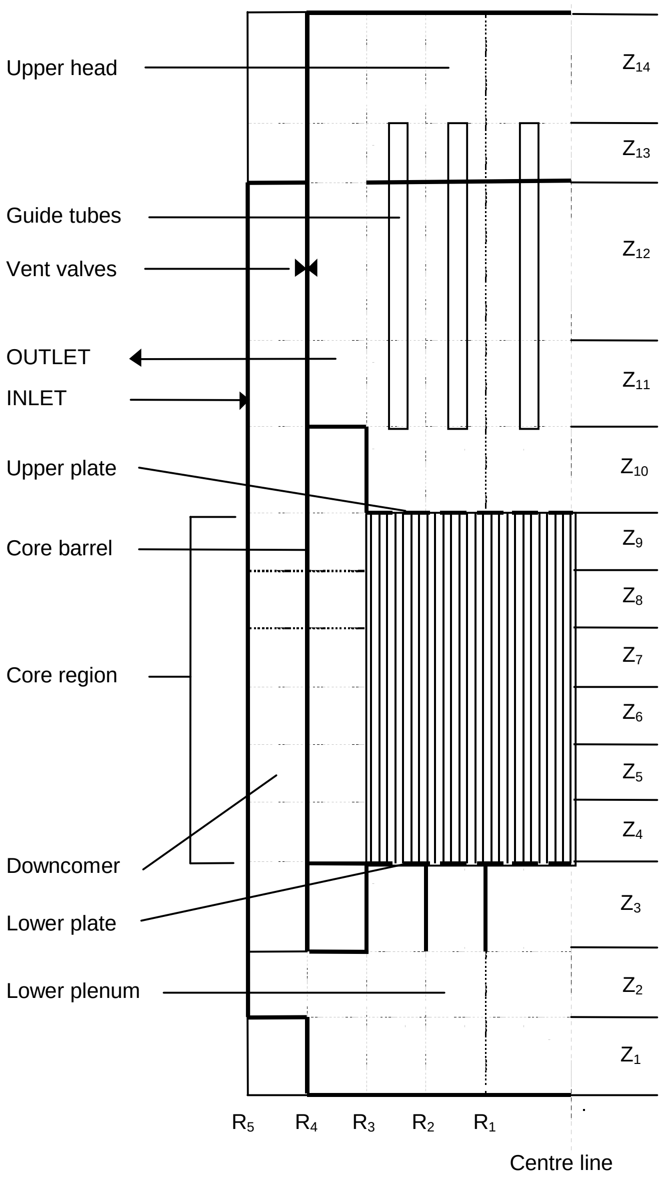
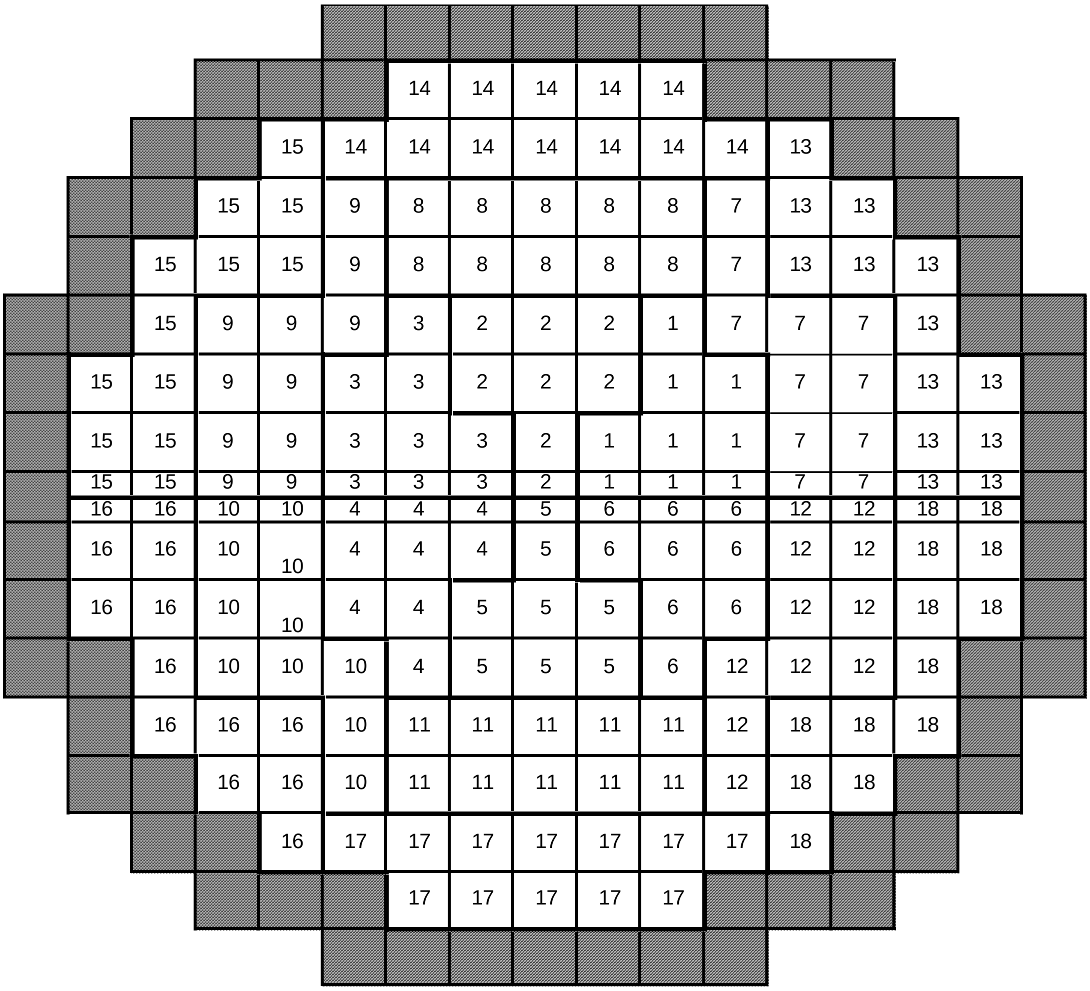


Figure 3.2.3. Transient core boundary conditions mapping scheme for the second exercise



Chapter 4

NEUTRONIC/THERMAL-HYDRAULIC COUPLING

The feedback, or coupling, between neutronics and thermal-hydraulics (T-H) can be characterised by choosing user supplied mapping schemes (spatial mesh overlays) in the radial and axial core planes.

Some of the inlet perturbations (inlet disturbances of both temperature and flow rate) are specified as a fraction of the position across the core inlet. This requires either three-dimensional (3-D) modelling of the vessel or some type of a multi-channel model.

For the purposes of this benchmark (Exercises 2 and 3), it is recommended that an assembly flow area of 40.061 in^2 (0.0262 m^2) be used in the core T-H multi-channel models.

Chapter 5

HOT FULL POWER (HFP) MSLB PROBLEM

5.1. Description of MSLB scenario

The transient being analysed is a main steam line break (MSLB) in a PWR, which may occur as a consequence of the rupture of one steam line upstream of the main steam isolation valves (MSIVs). This event is characterised by significant space-time effects in the core caused by asymmetric cooling and an assumed stuck-out control rod after the reactor trip. One of the major concerns for the MSLB accident is the return to power and criticality in the latter half of the transient. Because of this concern, the MSLB scenario was based on assumptions that conservatively maximise the consequences for a return to power. For the purpose of this benchmark problem two versions of this scenario were specified. The first version is the original scenario, which is employed in the current licensing practice. For this scenario, point kinetics models usually predict a return to power, while 3-D models do not despite conservative assumptions. The second scenario was specified upon request of the participants in order to test better the predictions of coupled 3-D kinetics/thermal-hydraulic codes. In this scenario, 3-D models are also expected to predict a return to power. Both scenario versions have the same initial steady-state conditions (as described in Section 5.2) and follow the same sequence of events (as described below in this section). The difference is in the value of the tripped rod worth. In point kinetics simulations the Table 5.3.2 specifies the rod worth versus time after trip for both versions. In 3-D kinetics simulations the point kinetics tripped rod worth values are matched by two different rodded cross-section libraries, to be used in each scenario version – **nemtabr** and **nemtabr.rp**. The cross-section libraries are provided on CD-ROM and at the benchmark **ftp** site. Table 5.3.2 specifies the rod worth versus time after trip for both scenario versions. This table is used for the point kinetics simulation, while the cross-section libraries provided at the benchmark **ftp** site and on the CD-ROM are used for the 3-D simulations. For the 3-D simulations, the point kinetics tripped rod worth values are matched by a rodded cross-section library for each scenario – **nemtabr** (version 1) and **nemtabr.rp** (version 2).

The limiting MSLB for TMI-1 (a B&W designed PWR) is at hot full power (HFP) because the steam generator (SG) liquid inventory increases with increasing power level. The worst case overcooling occurs at the maximum power level, which corresponds to the maximum liquid inventory in the SG.

The double-ended rupture of one steam line is assumed to occur upstream of the cross-connect (see Figures A.3 and A.4 provided in Appendix A). Figure A.3 shows the detailed RETRAN steam line nodalisation, while Figure A.4 shows a simplified steam line modelling to be used for this MSLB simulation. The 24 inch (60.96 cm) main steam line and 8 inch (20.32 cm) cross-connect rupture results in the highest break flow assumption and maximises the RCS cool-down. The worst single failure is the failure in the open position of the feedwater regulating valve to the broken SG. This failure in the open position causes feedwater flow from the intact SG to cross over to the broken

SG across the common header and maximises the feedwater flow to the broken SG. The feedwater flow is eventually terminated by closure of the feedwater block valve, which is conservatively assumed to close 30 seconds after the break occurs.

Subsequent to break initiation, and following reactor scram, the steam line turbine stop valves are assumed to slam shut, isolating the intact SG. The 8 inch (20.32 cm) cross-connect between the two steam lines of the broken SG remains open.

All four RCS pumps are assumed to operate during the event since maximising the primary to secondary heat transfer will cause maximum RCS cool-down. No credit is taken for pressuriser heater operation. This conservative assumption enhances the RCS depressurisation.

Reactor trip is modelled to occur when the neutron power reaches 114% of 2 772 MWt or when the primary system pressure reaches 1 945 psia (13.41 MPa) – 1 900 psig + 30 psi error + 15 psi absolute – at the hot leg pressure tap. A delay of 0.4 seconds is used for the high neutron flux trip; while the low RCS pressure delay is modelled as 0.5 seconds. These values represent the delay from the time the trip condition is reached to the time the control rods are free to fall and bound the actual delays for TMI-1.

High pressure injection (HPI) occurs when the primary system pressure drops to 1 645 psia (11.34 MPa) with a 25 second delay. HPI is expected to activate because of the large overcooling which occurs during this transient. No credit is taken for negative reactivity insertion from boron addition. No other emergency core coolant system (ECCS) action is expected.

Since the primary-to-secondary heat transfer is the driving force behind the RCS cool-down and the depressurisation, the steam generator inventory is maximised to provide the largest cool-down capacity. An initial steam inventory of 57 320 lbm (26 000 kg) was assumed. One can obtain the desired mass by either decreasing the aspirator flow until the downcomer quality is just saturated, or adjusting the initial void fraction in the bundle region of the SG. In addition to the initial inventory, the mass of the feedwater between the feedwater isolation valve and the downcomer of the broken steam generator, which was calculated to be 35 500 lbm (16 103 kg), is modelled and contributes to the overcooling and depressurisation of the RCS. For the purposes of this benchmark the additional feedwater is modelled as an extended boundary condition of feedwater rate vs. time in Table 5.4.2.

Vessel mixing is based on test data from Duke Power Company's Oconee Plant, also a B&W designed plant. These tests define the amount of mixing that occurs within the vessel as a ratio of the difference in hot leg temperatures to the difference in cold leg temperatures:

$$\text{Ratio} = [T_{hot}(\text{intact}) - T_{hot}(\text{broken})][T_{cold}(\text{intact}) - T_{cold}(\text{broken})]^{-1}$$

There is 20% mixing in the lower plenum and 80% mixing in the upper plenum, and the ratio (dT_{hot}/dT_{cold}) was conservatively chosen to be equal to 0.5. In order to be able to simulate this mixing, the reactor vessel in the RETRAN model was modified to include two equal parallel flow paths by splitting the downcomer, the lower plenum, the core and the upper plenum as shown in Figure A.2 in Appendix A. For the most part, these parallel flow paths behave independently, with the exception of common connections with the bypass and upper head volumes. These common flow paths keep the loop pressures in balance but contribute little to mixing of loop flows. The desired amount of loop flow mixing is obtained by exchanging energy between the two lower plenum volumes and the two upper plenum volumes with non-conducting heat exchangers. A proportional-integral control system is used to compute the ratio each time step from the hot and cold leg temperatures, compare the instantaneous value of the ratio to the target value, and adjust the energy exchange between the plenum volumes to match the target RATIO of 0.5. Consistent with the experimental results, 20% of the energy was exchanged between lower plenum volumes and 80% between upper plenum volumes.

Different code predictions will be compared and evaluated in regard to:

- Time and value of the power peak before reactor trip.
- Time and value of a power peak after reactor trip.
- Whether the system remains critical after the momentary return to power (if it occurs) for the transient duration.

5.2. Initial steady-state conditions

The reactor is assumed to be at end of cycle (EOC), 650 EFPD (24.58 GWd/MT average core exposure), with a boron concentration of 5 ppm, and equilibrium Xe and Sm concentration. Control rod Groups 1 through 6 are completely withdrawn (wd). Group 7 is 90% wd, or 900 steps (354.0155 cm) from the bottom of the lower reflector except the rod at position N12, (which is stuck throughout the transient) and is out of the core. The position of Group 7 in terms of insertion from the top of the core region is 26.9155 cm. The position of Group 8 is modelled implicitly through the cross-sections, and the participants do not have to account for it in their control rod models.

Five steady-states at EOC are defined for initialisation of the 3-D core neutronics models for the second and third exercises. The initial steady-state for the MSLB simulation, as described above, is to be calculated at both hot zero power (HZP) conditions (used to evaluate the scram worth and stuck rod worth) and HFP conditions (used as initial steady-state for the transient simulation). The steady-states at HZP for both versions of the MSLB scenario are defined as follows: all the control rods (Groups 1 through 7) are completely inserted except the stuck rod at position N12, which is completely wd. In state 0, all groups are completely wd. The HFP conditions are defined in Table 5.2.1 and the HZP conditions are defined as follows: the power level is 2 772 Wt; the fuel and moderator temperatures are 532°F (551°K); and the moderator density is 766 kg/m^3. The scram and stuck rod worths are evaluated using calculated eigenvalues for states 1 and 3 for the first scenario and states 1 and 4 for the second scenario. The scram worth is defined as:

$$\%\frac{dk}{k} = \frac{k_m - k_n}{k_m k_n} \times 100$$

where m = 3,4 and n = 1.

The initial RCS pressure is at the nominal operating value of 2 170 psia (14.96 MPa). The initial pressuriser liquid level is set to the typical HFP pressuriser level of 220 temperature-compensated inches (558.8 cm). The initial cold leg temperature is 555°F (564°K). The initial HFP steady state conditions are defined in Table 5.2.1.

The five steady-states described in Table 5.2.2 and briefly discussed above will be used for detailed comparisons of results for both the second and third exercises. For the purposes of the second exercise, one additional steady-state (2A) is defined in order to assess better the T-H modelling effects when making code-to-code comparisons: this additional state is the same as the second steady-state (HFP) but with uniform BC.

The calculated effective multiplication factor, K_{eff}, for the second steady-state will be used to divide the number of neutrons produced per fission, in order to obtain a critical initial steady-state for the transient calculations. This adjustment procedure should be applied to the provided cross-section tables.

5.3. Point kinetics model inputs

Point kinetics model inputs, which preserve axial and radial core power distributions, as well as scram reactivity obtained with the 3-D nodal core model, must be specified in order to make both simulations compatible. The following parameters for the point kinetics model and the 3-D neutronic transient core model should be consistent:

- Scram worth.
- Radial power distribution.
- Axial power distribution.
- Moderator temperature coefficient.
- Doppler temperature coefficient.
- Other kinetics parameters.

All other initial and boundary conditions also have to be identical between the two cases. The scram worth and maximum stuck rod worth at position N12 were calculated with the 3-D nodal core model at the following EOC hot zero power (HZP) conditions:

- Core power is equal to 2 772 Wt.
- Full flow and operating pressure.
- 5 ppm boron concentration.
- All control rods inserted.
- Xe distribution fixed at HFP conditions.
- Moderator temperature of 532°F (551°K)

Based on the calculated values, and including a 10% rod worth uncertainty, rod nvt effect and reduction in rod worth at HZP, an estimate for tripped rod worth (TRW) was calculated to be used as an input parameter in the point kinetics simulation. A summary of the point kinetics analysis input values is found in Table 5.3.1. EOC HFP radial and axial relative power distributions (based on 24 equal nodes of 14.88 cm axial height) are shown in Figures 5.3.1 and 5.3.2, respectively.

A scram reactivity table for use with the point kinetics simulation is provided in Table 5.3.2. For the first scenario the scram worth in % dk/k should match the PK tripped rod worth value of -4.526% dk/k, while for the second scenario a value of -3.040% dk/k is used.

5.4. Transient calculations

The key assumptions for performing the point and 3-D kinetics MSLB analyses are summarised in Table 5.4.1. Several of the assumptions that require additional explanation are described below:

- *Steam flow and break modelling*. The rupture of one steam line is assumed to occur upstream of the cross-connect. This resulted in the highest break flow assumption and maximises the reactor coolant system (RCS) cool-down. The main steam line piping length of about 146.98 ft (44.8 m) is included in the model (see Figure A.3 in Appendix A).

  Main steam flow to turbine. As shown in Figure A.3 each SG has two steam lines to the turbine. For the intact SG, the steam lines are combined for modelling purposes, while for the broken SG, they are modelled separately. The initial steam flow from the intact SG to the turbine is 1 679 lbs/sec (761.59 kg/sec), and from each of the steam lines on the broken SG the flow is half of this value, i.e. 839.5 lbs/sec (380 795 kg/sec). At the time of the break, the steam flow from the broken steam lines will all go to the break. The steam flow from the intact SG will continue to the turbine until the turbine stop valves close. The steam flow from the intact SG will not go to the break, because there are check valves in the steam lines which prevent reverse flow.

  Break size. The main steam line is specified as a 24 inch (60.96 cm) pipe. This corresponds to the outside diameter of the pipe. The break area should be based on the pipe inside diameter, which is 22.06 in (56.03 cm). The 8 inch (20.32 cm) cross-connect corresponds to the inside pipe diameter.

  Break location. The steam line nodalisation and break location can be identified in Figure A.3. The steam line break is the double-ended rupture of one 24 inch steam line, and is represented by junctions 178 and 179 below. The 8 inch cross-connect is represented by junction 360. The simplified main steam line modelling, shown in Figure A.4, is developed for the purposes of this benchmark.

  Main steam safety valves. The MSSVs are modelled for the intact SG only, and the small safety plus three (271-273) safety valve banks are modelled (compare Figures A.3 and A.4). The flow is ramped from zero to rated flow between the open set point and 3% above the open set point. The flow area for the small safety valve is approximately 0.0276 ft^2 (2.567×10^{-3} m^2) and 0.111 ft^2 (1.03×10^{-2} m^2) for the other valves.

- *Main feedwater and emergency feedwater flows.* The mechanical failure of the feedwater regulating valve in the broken SG in the open position is assumed. The feedwater flow (FW) is eventually terminated by closure of the feedwater block valve, which is conservatively assumed to close 30 seconds after the break occurs. The FW in the broken SG is assumed to be ramped to the new value over the time indicated (see Table 5.4.3). After 30 seconds an extended boundary condition models the additional feedwater between the feedwater isolation valve and the downcomer of the broken SG. The mass of feedwater between the isolation valve and the broken steam generator nozzle, which is specified as 35 500 lbm (16 103 kg) is approximated as a constant flow of 3 000 lbm/sec (1 362 kg/sec) over 12 seconds (see Table 5.4.3). The main FW to the intact SG should be kept constant until reactor trip and then ramped to zero in 10 seconds (see Table 5.4.4). It is anticipated that for the purposes of this benchmark, 100 seconds of transient time will be sufficient for a return to power to be seen if it occurs. Consequently, emergency feedwater flow (EFW) need not be modelled as liquid levels in the broken SG will only reach the 10 inch (25.4 cm) actuation set point just prior to 100 seconds.

- *Reactor trip*. For the first and third exercises, reactor scram occurs at a high neutron flux of 114% of 2 772 MWt (with a 0.4 second delay) or at a low RCS pressure of 1 945 psia (13.41 MPa), with a 0.5 second delay. For the second exercise, reactor scram is modelled to occur at 6.65 seconds into the transient. The high neutron flux set point is reached at 6.61 seconds and the insertion of control rod groups begin after the specified 0.4 second delay. Subsequent to the reactor trip signal, the most reactive control rod (the position of the rod is N12) is assumed stuck in its fully withdrawn position. The control rod movement during the scram in the second and third exercises is modelled with a speed of 155.773 cm/sec.

- *High pressure injection*. The high pressure injection (HPI) system is assumed to activate when primary system pressure drops to 1 645 psia (11.34 MPa) with a 25 second delay. The HPI is modelled taking credit for two pumps; the total flow versus pressure is shown in Table 5.4.5. No credit is taken for the negative reactivity insertion from boron addition.

- *SG conductors*. The secondary side heat conductors on the SG, downcomers, and steam annulus should be given zero thickness and zero heat transfer coefficient. This will increase the cool-down for this event.

- *Containment modelling*. The containment response is not modelled and is assumed to stay at atmospheric pressure during the transient.

Table 5.2.1. Initial conditions for TMI-1 at 2 772 MWt

Parameter	Value
Core power	2 772.00 MWt
RCS cold leg temperature	555°F/563.76°K
RCS hot leg temperature	605°F/591.43°K
Lower plenum pressure	2 228.5 psia/15.36 MPa
Outlet plenum pressure	2 199.7 psia/15.17 MPa
RCS pressure	2 170.00 psia/14.96 MPa
Total RCS flow rate	38 806.2 lb/sec / 17 602.2 kg/sec
Core flow rate	35 389.5 lb/sec / 16 052.4 kg/sec
Bypass flow rate	3 416.7 lb/sec / 1 549.8 kg/sec
Pressuriser level	220 inches/558.8 cm
Feedwater/steam flow per OTSG	1 679 lb/sec / 761.59 kg/sec
OTSG outlet pressure	930.00 psia/6.41 MPa
OTSG outlet temperature	571°F/572.63°K
OTSG superheat	35°F/19.67°K
Initial SG inventory	57 320 lbm/26 000 kg
Feedwater temperature	460°F/510.93°K

Table 5.2.2. Definition of steady-states

Number	T-H conditions	Control rod positions	Scenario version
0	HZP	Groups 1-7 ARO*	
1	HZP	Groups 1-6 ARO Group 7 is 90% wd N12 is 100% wd	
2	HFP	Groups 1-6 ARO Group 7 is 90% wd N12 is 100% wd	
3	HZP	Group 1-7 ARI N12 is 100% wd	1
4	HZP	Group 1-7 ARI N12 is 100% wd	2

* ARO – all rods out; ARI – all rods inserted

Table 5.3.1. Summary of point kinetics analysis input values

Parameter	Value
HFP EOC MTC (pcm/°F / pcm/°K)	-34.64/-62.35
HFP EOC DTC (pcm/°F / pcm/°K)	-1.43/-2.57
HFP EOC delayed neutron fraction (β_{eff})	0.5211E-02
HFP EOC prompt neutron lifetime	0.18445E-04
EOC TRW (% dk/k)	4.526

Table 5.3.2. Rod worth versus time after trip (version 1 and 2)

Time after reactor trip (sec)	Per cent of reactivity insertion (%)	Rod worth inserted** (% dk/k)	Rod worth inserted*** (% dk/k)
0.0	0.0	0.000	0.000
0.2	0.58	-0.026	-0.018
0.3	0.99	-0.045	-0.030
0.4	1.83	-0.083	-0.056
0.6	5.29	-0.239	-0.161
0.8	12.33	-0.558	-0.375
1.0	21.41	-0.969	-0.651
1.2	33.09	-1.498	-0.101
1.4	50.75	-2.297	-1.428
1.6	72.96	-3.302	-2.218
1.8	91.30	-4.132	-2.776
2.0	99.26	-4.493	-3.018
2.2	99.99	-4.526	-3.040
2.3	100.00	-4.526	-3.040
10^6	100.00	-4.526	-3.040

** Based on 4.526% dk/k TRW-version 1
*** Based on 3.040% dk/k TRW-version 2

Table 5.4.1. MSLB analysis assumptions

Parameter	**Value**
Vessel mixing	$dT_{hot}/dT_{cold} = 0.5$
Boron injection	No credit taken for boron injection
Steam line break	24 inch rupture (60.96 cm)
Critical flow model	Moody, cont. coeff. = 1.0
High flux trip set point	114%
High flux trip delay time	0.4 sec
Decay heat multiplier	1.0
Turbine stop valve closure time	0.5 sec
MFW flow	Flow vs. time
EFW flow	No credit
HPI flow	Flow vs. pressure
High pressure trip set point	2 370 psia/16.34 MPa
High pressure trip delay	0.6 sec
Low pressure trip	1 945 psia/13.41 MPa
Low pressure trip delay	0.5 sec

Table 5.4.2. Description of MSSVs per OTSG

Description of safety valves	Open setpoint		Close setpoint		Rated flow per valve at 3% accumulation	
	psia	Bar	psia	Bar	Lbm/Hr	Kg/Hr
Small safety (1 valve)	1 055.0	72.73	1 012.5	69.8	194 900	88 407
Safety bank 1 (1 valve)	1 065.0	73.42	1 022.0	70.46	824 265	373 887
Safety bank 2 (2 valves)	1 065.0	73.42	1 022.0	70.46	792 610	359 528
Safety bank 3 (2 valves)	1 075.0	74.11	1 031.5	71.12	799 990	362 875

Table 5.4.3. Main feedwater flow boundary conditions to broken SG

Time (sec)	Flow (lb/sec / kg/sec)
0	1 679.0/761.59
10	4 000.0/1 814.4
30	3 000.0/1 360.8
42	3 000.0/1 360.8
45	0.0/0.0

Table 5.4.4. Main feedwater flow boundary conditions to intact SG

Time (sec)	Flow (lb/sec / kg/sec)
0	1 679.0/761.59
Reactor trip	1 679.0/761.59
10 seconds after reactor trip	0.0/0.0
35	0.0/0.0

Table 5.4.5. HPI flow versus pressure

Flow (gpm / kg/sec)	Pressure (psia/MPa)
470.0/28.43	15/0.103
455.2/27.53	615/4.24
390.0/23.59	1 215/8.38
360.0/21.77	1 515/10.45
345.0/20.87	1 615/11.14
315.0/19.05	1 815/12.51
190.0/11.49	2 415/16.65

Figure 5.3.1. EOC HFP radial power distribution

Assembly relative radial power distribution (quarter symmetry)

Core Centre

0.918	1.253	1.057	1.285	1.031	1.248	0.805	0.439
1.253	1.023	1.270	1.051	1.278	1.048	1.124	0.496
1.057	1.270	1.039	1.278	1.022	1.254	1.051	0.476
1.285	1.053	1.278	1.048	1.273	0.952	0.767	
1.031	1.282	1.022	1.271	1.035	1.093	0.580	
1.248	1.043	1.254	0.952	1.093	0.740		
0.805	1.121	1.051	0.767	0.580			
0.439	0.493	0.475					

Core Reflector/ Boundary

Figure 5.3.2. EOC HFP core average axial power relative power distribution

Core average axial relative power distribution

Bottom

0.8008	0.98178	1.05563	1.06437	1.05347	1.03940	1.02745	1.01800	1.00775	1.00160	0.99907	0.99798
0.99785	0.99857	1.00041	1.00391	1.00980	1.01896	1.03230	1.05048	1.05834	1.03893	0.94526	0.79778

Top

Chapter 6

OUTPUT REQUESTED

- Results should be presented on paper ***and*** diskette (format details are given in Section 6.3).
- All data should be in SI units (kg, m, sec).
- For time histories, data should be at 0.1 second intervals.

6.1. Initial steady-state results

No results will be compared for **Exercise 1**.

The following results will be compared for **Exercise 2**:

- For the initial HZP states (0, 1, 3 and 4), the following parameters will be compared: K_{eff}; 2-D normalised power (NP) distribution; core averaged axial power distribution; the power peaking factors F_{xy}, F_z and axial offset.
- For the initial HFP steady-states (2 and 2a), the same information as for the HZP states plus: 2-D maps for inlet coolant temperature, inlet flow rate and outlet coolant temperature; axial distributions for the stuck rod (position N12) – relative power (normalised to the core power level), coolant density, mass flow rate and Doppler temperature. The spatial distributions should follow the format of the radial and axial power distributions.
- Scram and stuck rod worths for both scenarios.

The following results will be compared for the initial HZP steady states (0 to 4) for **Exercise 3**:

- K_{eff}.
- Radial power distribution – 2-D assembly NP distribution – axially averaged radial power distribution for 177 radial nodes (full core) normalised to core average power (relative radial power distribution).
- Axial power distribution – core average axial shape – radially averaged axial power distribution for 24 axial nodes (each 14.88 cm in length), normalised to core average power (relative axial power distribution).
- Power peaking factors – $F_{xy,}$ $F_{z,}$ and axial offset.

- Scram and stuck rod worths (in % dk/k) – for HZP states.

- Axial distributions for the stuck rod (position N12) – relative power (normalised to the core power level), coolant density, mass flow rate and Doppler temperature.

- Primary system pressure, temperatures and mass flow rates – core inlet and outlet pressure values; core average axial temperature and axial velocity distributions; core inlet radial coolant temperature and flow rate distributions; core outlet radial coolant temperature distributions. The spatial distributions should follow the format of the radial and axial power distributions respectively.

6.2. Transient results

Exercise 1

- Sequence of events.

- Transient core average results (time histories): total core power; fission power; steam line pressure in broken and intact loop; RCS pressure – core average, loop A (loop A is associated with the broken steam generator) and loop B (loop B is associated with the intact steam generator); core average coolant temperature; hot and cold leg liquid temperatures in both loops (last cell before/after vessel); break flow rates (24 inch, 8 inch and total); steam generator masses; reactivity edits; core average fuel temperature; and exchanged power between steam generators plus integrated water and steam break flow.

Exercise 2

- Snapshots at time of maximum power before reactor trip, at time of maximum power after reactor trip, and at 100 seconds – the same data as for the HFP steady-states except the total and fission power levels will be compared instead of $K_{eff.}$

- Time histories (core volume averaged without the reflector region): total power, fission power; coolant density; and Doppler temperature. In addition, the maximum nodal Doppler temperature vs. time will be compared.

Exercise 3

- Sequence of events

- Transient average results (time histories): steam line pressure in broken and intact loop; RCS pressure – core average, loop A (loop A is associated with the broken steam generator) and loop B (loop B is associated with the intact steam generator); core average coolant temperature; hot and cold leg liquid temperatures in both loops (last cell before/after vessel); break flow rates (24 inch, 8 inch and total); steam generator masses; reactivity edits; and exchanged power between steam generators plus integrated water and steam break flow.

- Time histories (core volume averaged without the reflector region): total power; fission power; coolant density; and Doppler temperature. In addition, the maximum nodal Doppler temperature vs. time will be compared.

- Snapshots at time of maximum power before reactor trip, at time of maximum power after reactor trip, and at 100 seconds – the same data as for the HFP states except the total and fission power levels will be compared instead of $K_{eff.}$

6.3. Output format

Results may be sent via E-MAIL to **benchmrk@nuce.psu.edu** or on diskette to Tara M. Beam, Nuclear Engineering Program, 231 Sackett Building, University Park, PA, USA, 16802-1408.

Diskette should be in PC 3.5" (720 KB or 1.44 MB) format, containing one text (ASCII) file for each case, respectively named RESULTS.A, RESULTS.B, RESULTS.C and RESULTS.D. Contents should be typed as close as possible to sample format.

Remarks:

- Time histories consist of data pairs (time, value), one per line, starting at 0 seconds, up to 100 seconds. Please provide the units on the first line of each time history.

- A plot of time histories would be appreciated for a first comparison of the transient results.

- Radial and axial profiles should be given according to the form shown in Figures 6.3.1 and 6.3.2.

- Please do not use tabs in the data files.

- **Start each line in column one and end each line with a carriage return <CR>**.

Output sample:

A) OECD PWR MSLB BENCHMARK
MSLB at Hot Full Power
RESULTS FROM CODE "XXXXXXXX"
EXERCISE 1: Point Kinetics

B) STEADY STATE RESULTS

B1) K_{eff} = 1.0000

B2) Radial power distribution (full core) – start each line in column one, leave a blank space in between each number, and use a total of six spaces per number):

0.9999 0.9999 0.9999 0.9999 0.9999
0.9999 0.9999 0.9999 0.99990.9999 0.9999 0.9999 0.9999 0.9999
0.9999 0.9999 0.9999 0.9999 0.9999 0.9999 0.9999 0.9999 0.9999 0.9999 0.9999
0.9999 0.9999 0.9999 0.9999 0.9999 0.9999 0.9999 0.9999 0.9999 0.9999 0.9999 0.9999 0.9999
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0.9999 0.9999 0.9999 0.9999 0.9999

B3) Axial power distribution – place all data starting in column one, leave a blank in between each number, and use a total of six spaces per number:

0.9999 0.9999 0.9999 0.9999 0.9999 0.9999 0.9999 0.9999 0.9999 0.9999 0.9999 0.9999
0.9999 0.9999 0.9999 0.9999 0.9999 0.9999 0.9999 0.9999 0.9999 0.9999 0.9999 0.9999

B4) Scram worth: 1.000% dk/k
Stuck rod worth: 1.000% dk/k

B5) Core inlet pressure: 9999.99E+99 Pa
Core outlet pressure: 9999.99E+99 Pa
Distributions of temperature, etc.

C) SEQUENCE OF EVENTS

Event description	**Time (seconds)**
Break opens	0.001
Reactor trip	
Turbine valves close	
High pressure injection starts	
Transient ends	100

D) TRANSIENT CORE AVERAGED RESULTS (TIME HISTORIES)

For each of the results in Section D, the first column of numbers should be the time covering a time interval of 0-100 seconds, with data taken every 0.1 second. The second column should the data at that time, with a space between the first and second columns.

D1) Fission power (W):

0.0000 9.9999E+99
99.999 9.9999E+99

D2) Coolant (liquid) temperature (core average, hot and cold leg) (K):

0.0000 9.9999E+99
99.999 9.9999E+99

D3) Pressure (core average, loop A, and loop B) (Pa):

0.0000 9.9999E+99
99.999 9.9999E+99

D4) Steam line pressure (broken and intact loops) (Pa):

0.0000 9.999E+99
99.999 9.999E+99

D5) Break flow rate (24 inch, 8 inch, total) (Kg/s):

0.0000 9.999E+99
99.999 9.999E+99

D6) Steam generator mass (broken and intact) (Kg):

0.0000 9.999E+99
99.999 9.999E+99

D7) Reactivity edits (dk/k):

Total core reactivity and reactivity components (contributions from changes in moderator density, fuel temperature and neutron flux distribution – optional):

0.0000 9.999E+99
99.999 9.999E+99

D8) Core average fuel temperature (K):

0.0000 9.999E+99
99.999 9.999E+99

D9) Exchanged power between steam generators plus integrated water and steam break flow (intact and broken):

0.0000 9.999E+99
99.999 9.999E+99

E) SNAPSHOTS:

– At time of maximum power before reactor trip.

– At time of maximum power after reactor trip.

– At the end of the transient (100 seconds).

E1) Fission power:

9.9999E+99

E2) Radial power distribution (full core) – start each line in column one, leave a blank space in between each number, and use a total of six spaces per number):

0.9999 0.9999 0.9999 0.9999 0.9999
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0.9999 0.9999 0.9999 0.9999 0.9999 0.9999 0.9999 0.9999 0.9999 0.9999 0.9999
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0.9999 0.9999 0.9999 0.9999 0.9999 0.9999 0.9999 0.9999 0.9999
0.9999 0.9999 0.9999 0.9999 0.9999

E3) Axial power distribution – place all data starting in column one, leave a blank in between each number, and use a total of six spaces per number:

0.9999 0.9999 0.9999 0.9999 0.9999 0.9999 0.9999 0.9999 0.9999 0.9999 0.9999 0.9999
0.9999 0.9999 0.9999 0.9999 0.9999 0.9999 0.9999 0.9999 0.9999 0.9999 0.9999 0.9999

Figure 6.3.1. Form for axial power distribution

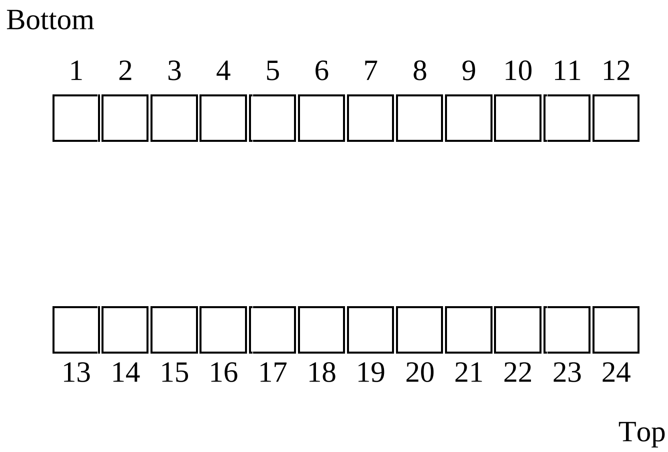
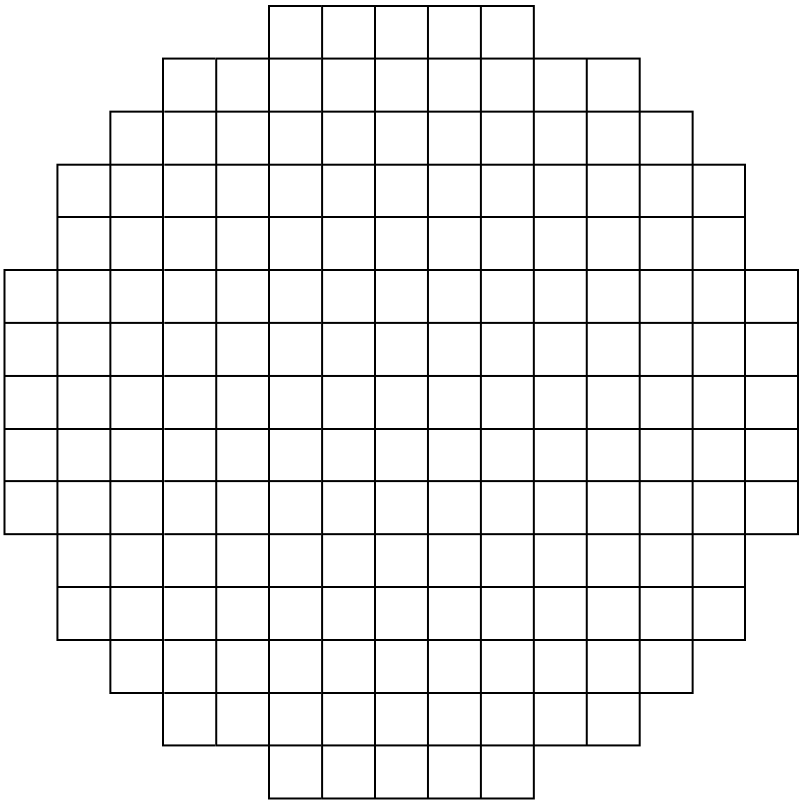


Figure 6.3.2. Form for radial power distribution



Chapter 7

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[2] C. Jackson and H. Finnemann, "Verification of the Coupled RELAP5/PANBOX System with the NEACRP LWR Core Transient Benchmark", Proc. of the International Conference on Mathematics and Computations, Reactor Physics and Environmental Analyses, p. 297, Portland, Oregon, 30 April-1 May 1995.

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APPENDIX A

Skeleton input deck

SKELETON INPUT DECK

NVOL	NJUN
92	120

Reactor Vessel

	Volume	Z Volume	FlowA	DiamV	Elev	
001	621.64	7.8665	38.95	3.84	-24.5833	Lower Plenum
002	196.862	4.00	49.2154	0.04277	-16.7168	Core - Lower Sect
003	196.862	4.00	49.2154	0.04277	-12.7168	Core - Middle Sect
004	196.682	4.00	49.2154	0.04277	-8.7169	Core - Upper Sect
005	877.90	11.7187	81.4	0.5783	-4.7168	Upper Plenum
006	193.23	12.00	17.11	0.0349	-16.7168	Core Bypass
007	689.37	19.865	35.13	1.67	-18.698	Reactor Downcomer
008	492.57	5.73	104.58	7.11	7.0019	Upper Head
009	247.96	10.65	23.29	1.29	-3.40	Outlet Plenum

Loop A

Primary RCS Volumes

	Volume	Z Volume	FlowA	DiamV	Elev	
108	526.3	49.661	7.07	3.00	-1.50	Hot Leg Piping
109	308.21	6.96	16.66	0.04642	30.78	SG UPR Plenum
110	113.99	4.343	26.25	0.04642	26.437	SG Tubes
111	113.99	4.343	26.25	0.04642	22.094	SG Tubes
112	113.99	4.343	26.25	0.04642	17.751	SG Tubes
113	113.99	4.343	26.25	0.04642	13.408	SG Tubes
114	113.99	4.343	26.25	0.04642	9.065	SG Tubes
115	113.99	4.343	26.25	0.04642	4.722	SG Tubes
116	113.99	4.343	26.25	0.04642	0.379	SG Tubes
117	113.99	4.343	26.25	0.04642	-3.964	SG Tubes
118	113.99	4.343	26.25	0.04642	-8.307	SG Tubes

119	113.99	4.343	26.25	0.04642	-12.65	SG Tubes
120	113.99	4.343	26.25	0.04642	-16.993	SG Tubes
121	113.99	4.343	26.25	0.04642	-21.336	SG Tubes
122	308.21	6.96	17.40	0.04642	-28.296	SG LWR Plenum
123	154.97	28.563	4.276	2.33333	-30.92	Pump Suct Pipe
124	56.00	7.022	1.00E+10	1.00E+10	-2.357	RC Pump 124
125	119.265	5.83	4.276	2.33333	-1.165	Pump DSCH pipe
126	154.97	28.563	4.276	2.33333	-30.92	Pump Suct Pipe
127	56.00	7.022	1.00E+10	1.00E+10	-2.357	RC Pump 127
128	119.265	5.83	4.276	2.33333	-1.165	Pump DSCH pipe
130	1550.00	40.28	38.4845	7.00	-3.02	Pressurizer

Secondary Volumes

	Volume	ZVolume	FlowA	DiamV	Elev	
151	758.90	32.526	23.331	1.375	-21.336	SG LWR DWNCMER
152	187.41	3.343	43.1515	0.06686	-21.336	SG Tube Region
153	187.41	3.343	43.1515	0.06686	-16.993	SG Tube Region
154	187.41	3.343	43.1515	0.06686	-12.65	SG Tube Region
155	187.41	3.343	43.1515	0.06686	-8.307	SG Tube Region
156	187.41	3.343	43.1515	0.06686	-3.964	SG Tube Region
157	187.41	3.343	43.1515	0.06686	0.379	SG Tube Region
158	187.41	3.343	43.1515	0.06686	4.722	SG Tube Region
159	187.41	3.343	43.1515	0.06686	9.065	SG Tube Region
160	187.41	3.343	43.1515	0.06686	13.408	SG Tube Region
161	187.41	3.343	43.1515	0.06686	17.751	SG Tube Region
162	187.41	3.343	43.1515	0.06686	22.094	SG Tube Region
163	187.41	3.343	43.1515	0.06686	26.437	SG Tube Region
164	461.88	19.80	23.331	1.375	10.983	SG UPPR DWNCMR
165	392.27	10.48	2.65	1.838	14.10	MSL to RB
365	392.27	10.48	2.65	1.838	14.10	MSL to MSIV
182	1.00E+06	1.00E+04	1.00E+03	0.00	-50.00	Turbine Sink Volume

Loop B

Primary RCS Volumes

	Volume	Z Volume	FlowA	DiamV	Elev	
208	526.30	49.661	7.07	3.00	-1.50	Hot Leg Piping
209	308.21	6.96	16.66	0.04642	30.78	SG UPR Plenum
210	113.99	4.343	26.25	0.04642	26.437	SG Tubes
211	113.99	4.343	26.25	0.04642	22.094	SG Tubes
212	113.99	4.343	26.25	0.04642	17.751	SG Tubes
213	113.99	4.343	26.25	0.04642	13.408	SG Tubes
214	113.99	4.343	26.25	0.04642	9.065	SG Tubes
215	113.99	4.343	26.25	0.04642	4.722	SG Tubes
216	113.99	4.343	26.25	0.04642	0.379	SG Tubes
217	113.99	4.343	26.25	0.04642	-3.964	SG Tubes
218	113.99	4.343	26.25	0.04642	-8.307	SG Tubes
219	113.99	4.343	26.25	0.04642	-12.65	SG Tubes
220	113.99	4.343	26.25	0.04642	-16.993	SG Tubes
221	113.99	4.343	26.25	0.04642	-21.336	SG Tubes
222	308.21	6.96	17.40	0.04642	-28.296	SG LWR Plenum
223	154.97	28.563	4.276	2.33333	-30.92	Pump Suct Pipe
224	56.00	7.022	1.00E+10	1.00E+10	-2.357	RC Pump 224
225	119.265	5.873	4.276	2.33333	-1.165	Pump DSCH Pipe
226	154.97	28.563	4.276	2.33333	-30.92	Pump Suct Pipe
227	56.00	7.022	1.00E+10	1.00E+10	-2.357	RC Pump 227
228	119.265	5.83	4.276	2.33333	-1.165	Pump DSCH Pipe

Secondary Volumes

	Volume	Z Volume	FlowA	DiamV	Elev	
251	758.90	32.526	23.331	1.375	-21.336	SG Downcomer
252	187.41	4.343	43.1515	0.06686	-21.336	SG Tube Region
253	187.41	4.343	43.1515	0.06686	-16.993	SG Tube Region
254	187.41	4.343	43.1515	0.06686	-12.65	SG Tube Region
255	187.41	4.343	43.1515	0.06686	-8.307	SG Tube Region
256	187.41	4.343	43.1515	0.06686	-3.964	SG Tube Region

257	187.41	4.343	43.1515	0.06686	0.379	SG Tube Region
258	187.41	4.343	43.1515	0.06686	4.722	SG Tube Region
259	187.41	4.343	43.1515	0.06686	9.065	SG Tube Region
260	187.41	4.343	43.1515	0.06686	13.408	SG Tube Region
261	187.41	4.343	43.1515	0.06686	17.751	SG Tube Region
262	187.41	4.343	43.1515	0.06686	22.094	SG Tube Region
263	187.41	4.343	43.1515	0.06686	26.437	SG Tube Region
264	461.88	19.80	23.331	1.375	10.983	SG UPPR Downcomer
265	752.30	10.48	5.30	1.838	14.10	MSL to RB

Junctions

Reactor Vessel

	Frm	To	JnArea	JunElev	Location
001	1	2	49.215	-16.7168	Frm Lwr Plnm to Core Btm
002	2	3	49.215	-12.7168	Frm Core Btm To Core Mid
003	3	4	49.215	-8.7168	Frm Core Mid to Core Upr
004	4	5	49.215	-4.7168	Frm Core to Upper Plenum
005	1	6	17.110	-16.7168	Frm Lwr Plnm to Core Bypa
006	6	5	17.110	-4.7168	Frm Core Bypass to Upr Plnm
007	7	1	35.130	-18.698	Frm RV Dwncomer to Lwr Plnm
008	5	8	24.100	7.0019	Frm Upper Plnm to Upr Head
009	8	9	8.564	7.0019	Frm Uppr Head to Out Plnm
010	5	9	50.750	4.2519	Frm Up Plenum to Out plen

Loop A

Primary Junctions

	Frm	To	JnArea	NunElev	Location
108	9	108	7.07	0.00	At RV Outlet
109	108	109	7.07	37.74	At SG Inlet
110	109	110	26.25	30.78	Frm SG tube Sht To Tubes
111	110	111	26.25	26.437	SG Primary Tubing
112	111	112	26.25	22.094	SG Primary Tubing
113	112	113	26.25	17.751	SG Primary Tubing

114	113	114	26.25	13.408	SG Primary Tubing
115	114	115	26.25	9.065	SG Primary Tubing
116	115	116	26.25	4.722	SG Primary Tubing
117	116	117	26.25	0.379	SG Primary Tubing
118	117	118	26.25	-3.964	SG Primary Tubing
119	118	119	26.25	-8.307	SG Primary Tubing
120	119	120	26.25	-12.65	SG Primary Tubing
121	120	121	26.25	-16.993	SG Primary Tubing
122	121	122	26.25	-21.336	Sg Tubes to SG Tube Sht
123	122	123	4.276	-26.836	SG Lwr HD to Piping to PP
124	123	124	4.276	-2.357	At Pump Inlet (1)
125	124	125	4.276	3.50	At Pump Outlet (1)
126	122	126	4.276	-26.836	SG Lwr Hd to Piping to PP
127	126	127	4.276	-2.357	At Pump Inlet (2)
128	127	128	4.276	3.50	At Pump Outlet (2)
130	108	130	0.4176	-1.50	Surge Line With No Vol
132	128	131	0.0246	4.665	Cold Leg to Spray Line
133	125	7	4.276	0.00	At RV Cold Leg Inlet
134	128	7	4.276	0.00	At RV Cold Leg Inlet
142	0	125	10.00	0.00	HPI System Loop A

Secondary Junctions

	Frm	To	JnArea	NunElev	Location
150	159	151	10.33	11.02	Aspirator Flow
151	0	151	1.00	10.64	Fdw Inlet to SG
152	151	152	4.28	-19.805	SG Lwr Dwcm to SG Second
153	152	153	43.1515	-16.993	SG Secondary
154	153	154	43.1515	-12.65	SG Secondary
155	154	155	43.1515	-8.307	SG Secondary
156	155	156	43.1515	-3.964	SG Secondary
157	156	157	43.1515	0.379	SG Secondary
158	157	158	43.1515	4.722	SG Secondary
159	158	159	43.1515	9.065	SG Secondary
160	159	160	43.1515	13.408	SG Secondary
161	160	161	43.1515	17.751	SG Secondary
162	161	162	43.1515	22.094	SG Secondary
163	162	163	43.1515	26.437	SG Secondary
164	163	164	23.331	30.03	Tubes to Up Dcmr

165	164	165	2.65	15.02	Up Dwcmr to MSL
365	164	365	2.65	15.02	Up Dwcmr to MSL
166	0	165	2.65	15.02	MSL at Break
300	0	166	10.00	34.00	MS to Turbine

Loop B

Primary Junctions

	Frm	To	JnArea	NunElev	Location
208	9	208	7.07	0.00	At RV Outlet
209	208	209	7.07	37.74	At SG Inlet
210	209	210	26.25	30.78	Frm SG Tube Sht to Tubes
211	210	211	26.25	26.437	SG Primary Tubing
212	211	212	26.25	22.094	SG Primary Tubing
213	212	213	26.25	17.751	SG Primary Tubing
214	213	214	26.25	13.408	SG Primary Tubing
215	214	215	26.25	9.065	SG Primary Tubing
216	215	216	26.25	4.722	SG Primary Tubing
217	216	217	26.25	0.379	SG Primary Tubing
218	217	218	26.25	-3.964	SG Primary Tubing
219	218	219	26.25	-8.307	SG Primary Tubing
220	219	220	26.25	-12.65	SG Primary Tubing
221	220	221	26.25	-16.993	SG Primary Tubing
222	221	222	26.25	-21.336	SG Tubes to SG Tube Sht
223	222	223	26.25	-26.836	SG Lwr Hd to Piping to PP
224	223	224	4.276	-2.357	At Pump Inlet (3)
225	224	225	4.276	3.50	At Pump Outlet (3)
226	222	226	4.276	-26.836	SG Lwr Hd to Piping to PP
227	226	227	4.276	-2.357	At Pump Inlet (4)
228	227	228	4.276	3.50	At Pump Outlet (4)
233	225	7	4.276	0.00	At RV Cold Leg Inlet
234	228	7	4.276	0.00	At RV Cold Leg Inlet
242	0	225	10.00	0.00	HPI System Loop B

Secondary Junctions

	Frm	To	JnArea	NunElev	Location
250	259	251	10.33	11.02	Aspirator Flow
251	0	251	1.00	10.64	FW Inlet to SG
252	251	252	4.28	-19.805	SG Lwr Dwcm to SG
253	252	253	43.1515	-16.993	SG Secondary
254	253	254	43.1515	-12.65	SG Secondary
255	254	255	43.1515	-8.307	SG Secondary
256	255	256	43.1515	-3.964	SG Secondary
257	256	257	43.1515	0.379	SG Secondary
258	257	258	43.1515	4.722	SG Secondary
259	258	259	43.1515	9.065	SG Secondary
260	259	260	43.1515	13.408	SG Secondary
261	260	261	43.1515	17.751	SG Secondary
262	261	262	43.1515	22.094	SG Secondary
263	262	263	43.1515	26.437	SG Secondary
264	263	264	23.331	30.03	Tubes to Up Dwcmr
265	264	265	5.30	15.02	Up Dwcmr to MSL
266	0	266	5.30	15.02	MSL at Break
271	0	266	0.111	33.09	STM Sfty Bank 1 Fill
272	0	266	0.222	33.09	STM Sfty Bank 2 Fill
273	0	266	0.222	33.09	STM Sfty Bank 3 Fill
274	0	266	0.222	33.09	STM Sfty Bank 4 Fill
275	0	266	0.111	33.09	STM Sfty Bank 5 Fill

Pump Curve Set Data

Number of Curve Sets

16

Pump Description

	001	**002**	**003**	**004**
Pump Curve Set Sequence:	1	1	1	1
Trip ID Number:	31	32	33	34
Pump Reversal:	No Reversal	No Reversal	No Reversal	No Reversal
Two-Phase Option:	No 2-Phase	No 2-Phase	No 2-Phase	No 2-Phase
Motor Torque Curve:	No Motor Torque	No Motor Torque	No Motor Torque	No Motor Torque
Pump Speed (RPM);	1190	1190	1190	1190
Pump Speed Ratio:	1.00	1.00	1.00	1.00
Rated Flow (gal/min):	88500	88500	88500	88500
Rated Head (ft):	350	350	350	350
Rated Torque (lbf-ft):	33710	33710	33710	33710
Moment of Inertia (lbm-ft^2):	700000	700000	700000	700000
Density (lbm/ft^2);	46.48	46.48	46.48	46.48
Motor Torque (lbf-ft):	0.00	0.00	0.00	0.00
Frictional Torque:	0.00	0.00	0.00	0.00

Head Curves

	Number Points	**Points (X,Y)**						
101	5	(0.0,1.42)	(0.227,1.29)	(0.435,1.18)	(0.73,1.085)	(1.0,1.0)		
102	12	(0.0,-0.89)	(0.17,-0.694)	(0.286,-0.51)	(0.41,-0.31)	(0.492,-0.174)	(0.592,0.0)	
		(0.65,0.15)	(0.68,0.21)	(0.73,0.31)	(0.82,0.51)	(0.89,0.69)	(1.0,1.0)	
103	6	(-1.0,2.97)	(-0.571,2.04)	(-0.405,1.826)	(-0.256,1.644)	(-0.123,1.524)	(0.0,1.42)	
104	6	(-1.0,2.97)	(-0.75,2.23)	(-0.53,1.78)	(-0.375,1.563)	(-0.17,1.235)	(0.0,1.0)	
105	7	(0.0,1.0)	(0.31,0.31)	(0.42,0.36)	(0.58,0.44)	(0.75,0.557)	(0.92,0.69)	(1.0,0.783)
106	11	(0.0,1.0)	(0.095,0.91)	(0.182,0.83)	(0.345,0.74)	(0.424,0.718)	(0.5,0.69)	
		(0.655,0.67)	(0.763,0.683)	(0.85,0.718)	(0.956,0.756)	(1.0,0.783)		
107	10	(-1.0,-1.644)	(-0.833,-1.235)	(-0.73,-1.0)	(-0.55,-0.694)	(-0.455,-0.574)	(-0.37,-0.47)	
		(-0.325,-0.39)	(-0.294,-0.346)	(-0.1,0.0)	(0.0,0.1)			
108	5	(-1.0,-1.644)	(-0.875,-1.563)	(-0.455,-1.29)	(-0.213,-1.132)	(0.9,-0.89)		

	Torque Curves Number Points	Points (X,Y)					
109	8	(0.0,0.98)	(0.099,0.98)	(0.196,0.96)	(0.49,0.96)	(0.59,0.98)	(0.7,1.0)
		(0.8,1.0)	(1.0,1.0)				
110	14	(0.0,-0.89)	(0.123,-0.59)	(0.193,-0.44)	(0.247,-0.346)	(0.295,-0.25)	(0.342,-0.148)
		(0.42,0.0)	(0.54,0.174)	(0.59,0.25)	(0.622,0.31)	(0.675,0.39)	(0.743,0.51)
		(0.85,0.69)	(1.0,1.0)				
111	8	(-1.0,2.295)	(-0.79,1.731)	(-0.59,1.384)	(-0.43,1.156)	(-0.313,1.085)	(-0.204,1.04)
		(-0.1,1.0)	(0.0,0.98)				
112	7	(-1.0,2.295)	(-0.87,2.1)	(-0.685,1.876)	(-0.526,1.73)	(-0.375,1.563)	(-0.114,1.291)
		(0.0,1.18)					
113	8	(0.0,-1.0)	(0.093,-0.857)	(0.192,-0.694)	(0.31,-0.592)	(-.392,-0.427)	(0.494,-0.31)
		(0.83,0.0)	(1.0,0.29)				
114	10	(0.0,1.18)	(0.104,1.085)	(0.19,1.0)	(0.367,0.842)	(0.513,0.731)	(0.635,0.63)
		(0.735,0.541)	(0.88,0.396)	(0.936,0.342)	(1.0,0.29)		
115	6	(-1.0,-3.845)	(-0.75,-2.78)	(-0.9,-2.04)	(-0.263,-1.563)	(-0.111,-1.235)	(0.0,-1.0)
116	6	(-1.0,-3.845)	(-0.656,-2.69)	(-0.423,-1.98)	(-0.244,-1.487)	(-0.11,-1.132)	(0.0,-0.89)

Properties of Inconel

Iconel-600

Thermal Conductivity Table

Temperature (Deg. F)	Thermal Conductivity (BTU/ft^3*F)
212.00	10.00
752.00	11.00
1832.00	18.00

Volumetric Heat Capacity

Temperature (deg. F)	Volumetric Heat Capacity BTU/ft^3*F)
200	57.26
400	60.14
450	60.63
500	61.12
550	61.62
600	52.06
800	64.15
1000	67.29

Figure A.1. RETRAN nodalisation diagram

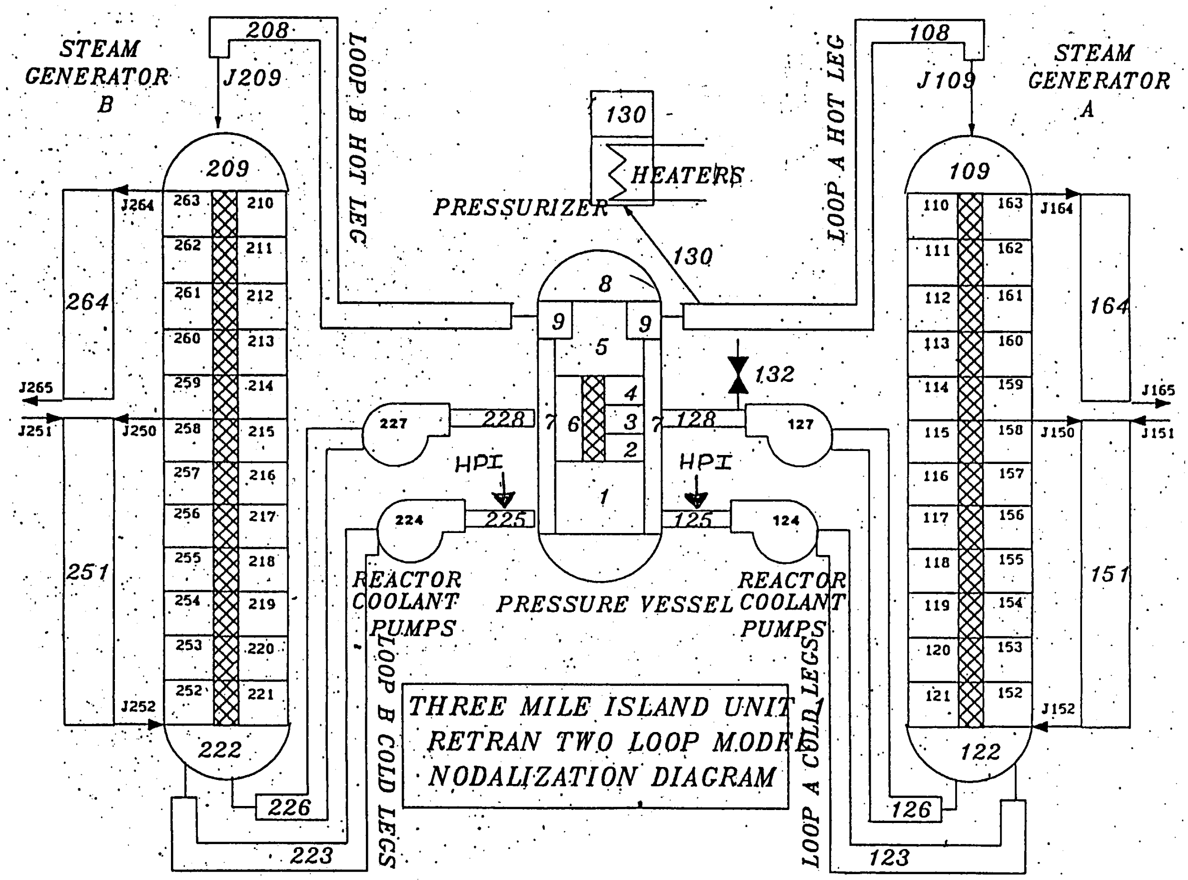


Figure A.2. RETRAN two channel model

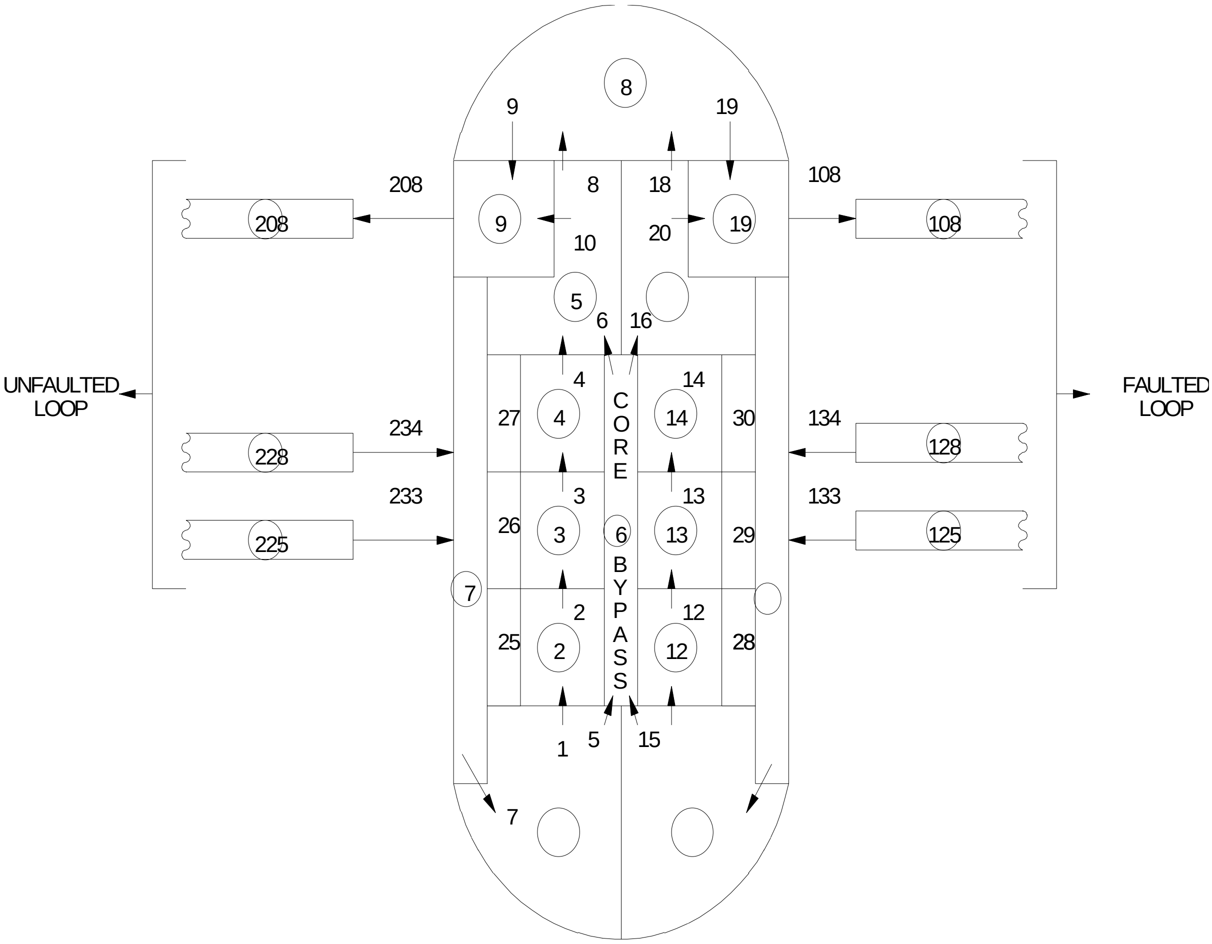
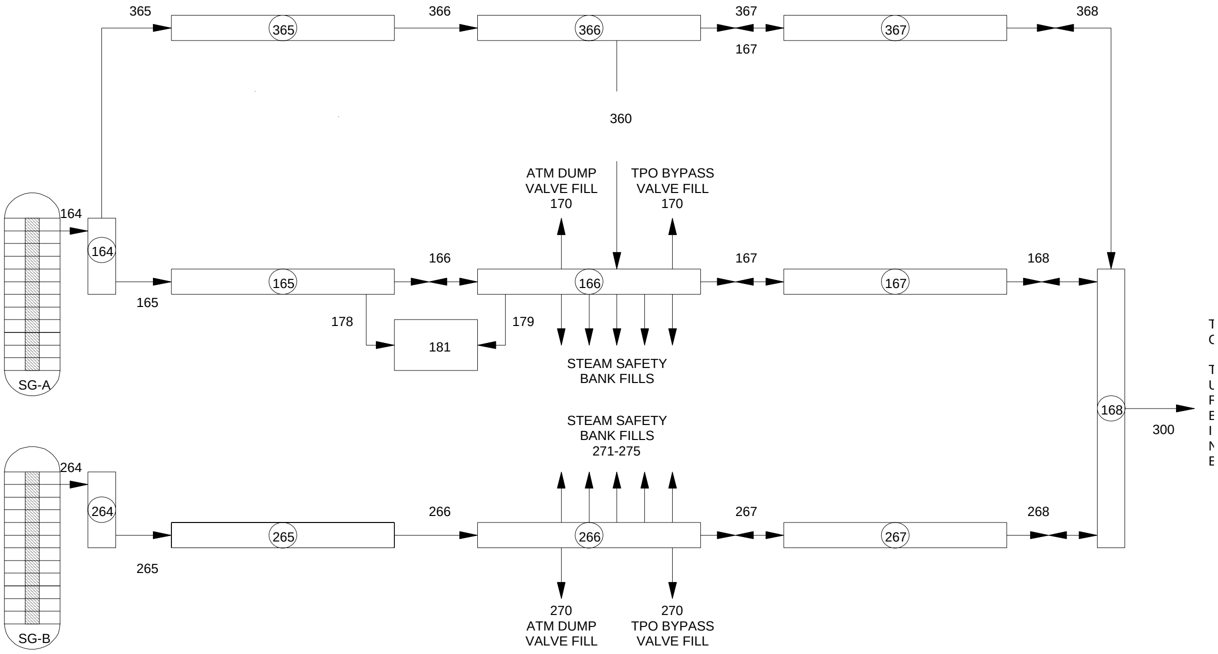
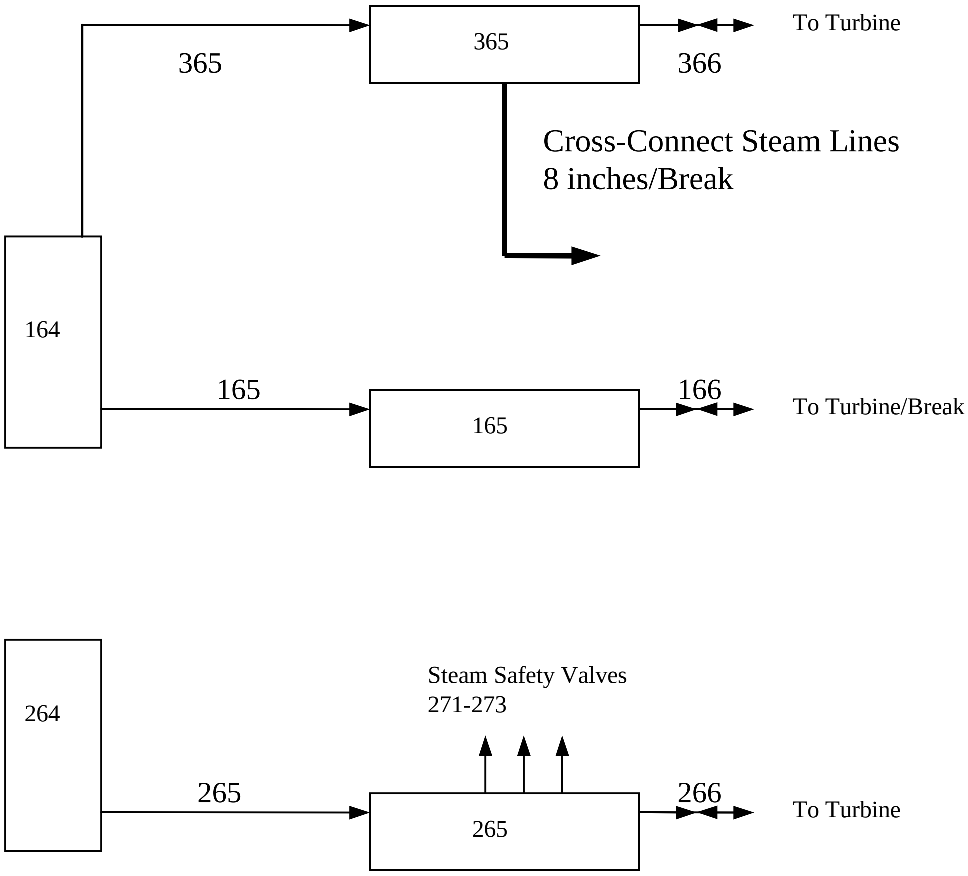


Figure A.3. RETRAN steam line nodalisation



	Volume	Z Volume	Flow A	Diam V	Elev	
165	392.27	10.48	2.65	1.838	14.1	MSL TO RB
365	392.27	10.48	2.65	1.838	14.1	MSL TO RB
166	166.83	10.34	2.65	1.838	22.75	MSL TO MSIV
366	166.83	10.34	2.65	1.838	22.75	MSL TO MSIV
167	420.35	4.01	2.65	1.838	32.16	MSL TO TSV
367	420.35	4.01	2.65	1.838	32.16	MSL TO TSV
168	95.72	2.5	4.9	2.5	34	STEAM HEADER
265	752.3	10.48	5.3	1.838	1.838	MSL TO RB
266	308.44	10.34	5.3	1.838	1.838	MSL TO MSIV
267	1134.46	4.01	5.3	1.838	1.838	MSL TO TSV

Figure A.4. MSLB steam line modelling



APPENDIX B

Sample cross-section table

```
*  NEM-Cross Section Table Input
*
*    T Fuel        Rho Mod.      Boron ppm.    T Mod.
       5              6             0             0
*
*******            X-Section set #        1
*
*      Group No.  1
*
***************              Diffusion Coefficient Table
*
  .5000000E+03  .7602200E+03  .8672700E+03  .9218800E+03  .1500000E+04
  .6413994E+03  .7114275E+03  .7694675E+03  .7724436E+03  .7813064E+03
  .8100986E+03  .1504700E+01  .1507100E+01  .1508200E+01  .1508700E+01
  .1514300E+01  .1437800E+01  .1440100E+01  .1441000E+01  .1441500E+01
  .1446600E+01  .1388600E+01  .1390800E+01  .1391700E+01  .1392100E+01
  .1397000E+01  .1387100E+01  .1389300E+01  .1390200E+01  .1390700E+01
  .1395500E+01  .1380300E+01  .1382500E+01  .1383300E+01  .1383800E+01
  .1388600E+01  .1356600E+01  .1358700E+01  .1359600E+01  .1360000E+01
  .1364700E+01
*
***************         Total Absorption X-Section Table
*
  .5000000E+03  .7602200E+03  .8672700E+03  .9218800E+03  .1500000E+04
  .6413994E+03  .7114275E+03  .7694675E+03  .7724436E+03  .7813064E+03
  .8100986E+03  .1055500E-01  .1070800E-01  .1076300E-01  .1079000E-01
  .1103800E-01  .1074100E-01  .1089800E-01  .1095500E-01  .1098300E-01
  .1123800E-01  .1087400E-01  .1103400E-01  .1109300E-01  .1112100E-01
  .1138300E-01  .1087900E-01  .1104000E-01  .1109800E-01  .1112700E-01
  .1138900E-01  .1089800E-01  .1105900E-01  .1111800E-01  .1114600E-01
  .1141000E-01  .1096300E-01  .1112600E-01  .1118500E-01  .1121400E-01
  .1148100E-01
*
***************              Fission X-Section Table
*
  .5000000E+03  .7602200E+03  .8672700E+03  .9218800E+03  .1500000E+04
  .6413994E+03  .7114275E+03  .7694675E+03  .7724436E+03  .7813064E+03
  .8100986E+03  .1977861E-02  .1970091E-02  .1966873E-02  .1965171E-02
  .1948015E-02  .2007931E-02  .2000111E-02  .1996925E-02  .1995221E-02
  .1978074E-02  .2030571E-02  .2022701E-02  .2019584E-02  .2017878E-02
  .2000667E-02  .2031764E-02  .2023930E-02  .2020737E-02  .2019030E-02
  .2001854E-02  .2035185E-02  .2027312E-02  .2024119E-02  .2022486E-02
  .2005230E-02  .2046539E-02  .2038770E-02  .2035537E-02  .2033827E-02
  .2016630E-02
```

```
*
***************           Nu-Fission X-Section Table
*
  .5000000E+03  .7602200E+03  .8672700E+03  .9218800E+03  .1500000E+04
  .6413994E+03  .7114275E+03  .7694675E+03  .7724436E+03  .7813064E+03
  .8100986E+03  .5342400E-02  .5322200E-02  .5313900E-02  .5309500E-02
  .5264900E-02  .5418000E-02  .5397900E-02  .5389700E-02  .5385300E-02
  .5340800E-02  .5473200E-02  .5453000E-02  .5444800E-02  .5440400E-02
  .5396000E-02  .5475400E-02  .5455300E-02  .5447100E-02  .5442700E-02
  .5398200E-02  .5483400E-02  .5463200E-02  .5455000E-02  .5450600E-02
  .5406100E-02  .5510100E-02  .5490000E-02  .5481700E-02  .5477300E-02
  .5432800E-02
*
***************           Scattering X-Section Table
*
****   From group 1 to 2
  .5000000E+03  .7602200E+03  .8672700E+03  .9218800E+03  .1500000E+04
  .6413994E+03  .7114275E+03  .7694675E+03  .7724436E+03  .7813064E+03
  .8100986E+03  .1335500E-01  .1326700E-01  .1323400E-01  .1321800E-01
  .1306600E-01  .1510700E-01  .1501300E-01  .1497800E-01  .1496100E-01
  .1480100E-01  .1650500E-01  .1640800E-01  .1637200E-01  .1635400E-01
  .1618700E-01  .1654500E-01  .1644800E-01  .1641200E-01  .1639400E-01
  .1622600E-01  .1674700E-01  .1665000E-01  .1661300E-01  .1659500E-01
  .1642700E-01  .1746700E-01  .1736700E-01  .1733000E-01  .1731200E-01
  .1714000E-01
*
*      Group No.  2
*
***************           Diffusion Coefficient Table
*
  .5000000E+03  .7602200E+03  .8672700E+03  .9218800E+03  .1500000E+04
  .6413994E+03  .7114275E+03  .7694675E+03  .7724436E+03  .7813064E+03
  .8100986E+03  .4046000E+00  .4052200E+00  .4054800E+00  .4056100E+00
  .4069800E+00  .3693900E+00  .3699500E+00  .3701800E+00  .3703000E+00
  .3715600E+00  .3440200E+00  .3445600E+00  .3447800E+00  .3448900E+00
  .3460800E+00  .3427700E+00  .3433100E+00  .3435300E+00  .3436400E+00
  .3448300E+00  .3390900E+00  .3396200E+00  .3398400E+00  .3399500E+00
  .3411400E+00  .3267900E+00  .3273100E+00  .3275200E+00  .3276300E+00
  .3287900E+00
*
***************      Total Absorption X-Section Table
*
  .5000000E+03  .7602200E+03  .8672700E+03  .9218800E+03  .1500000E+04
  .6413994E+03  .7114275E+03  .7694675E+03  .7724436E+03  .7813064E+03
  .8100986E+03  .9849700E-01  .9829800E-01  .9821600E-01  .9817500E-01
  .9774600E-01  .9955400E-01  .9934400E-01  .9925900E-01  .9921600E-01
  .9876400E-01  .1005400E+00  .1003200E+00  .1002300E+00  .1001800E+00
  .9970400E-01  .1006500E+00  .1004300E+00  .1003400E+00  .1002900E+00
  .9980900E-01  .1008400E+00  .1006100E+00  .1005200E+00  .1004800E+00
  .9998600E-01  .1014700E+00  .1012300E+00  .1011400E+00  .1010900E+00
  .1005800E+00
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*
***************              Fissin X-Section Table
*
  .5000000E+03  .7602200E+03  .8672700E+03  .9218800E+03  .1500000E+04
  .6413994E+03  .7114275E+03  .7694675E+03  .7724436E+03  .7813064E+03
  .8100986E+03  .5145311E-01  .5135296E-01  .5130844E-01  .5128803E-01
  .5106745E-01  .5171034E-01  .5160071E-01  .5155243E-01  .5153199E-01
  .5130000E-01  .5196260E-01  .5184540E-01  .5179896E-01  .5177478E-01
  .5152942E-01  .5200564E-01  .5188840E-01  .5184004E-01  .5181586E-01
  .5156494E-01  .5206176E-01  .5194078E-01  .5189240E-01  .5187013E-01
  .5161721E-01  .5224335E-01  .5211676E-01  .5206832E-01  .5204040E-01
  .5177802E-01
*
***************              Nu-Fission X-Section Table
*
  .5000000E+03  .7602200E+03  .8672700E+03  .9218800E+03  .1500000E+04
  .6413994E+03  .7114275E+03  .7694675E+03  .7724436E+03  .7813064E+03
  .8100986E+03  .1389800E+00  .1387300E+00  .1386200E+00  .1385700E+00
  .1380200E+00  .1395300E+00  .1392600E+00  .1391400E+00  .1390900E+00
  .1385100E+00  .1400600E+00  .1397700E+00  .1396500E+00  .1395900E+00
  .1389800E+00  .1401500E+00  .1398600E+00  .1397400E+00  .1396800E+00
  .1390500E+00  .1402700E+00  .1399700E+00  .1398500E+00  .1397900E+00
  .1391600E+00  .1406600E+00  .1403400E+00  .1402200E+00  .1401500E+00
  .1394900E+00
*
***************              Xe X-Section Table
*
  .5000000E+03  .7602200E+03  .8672700E+03  .9218800E+03  .1500000E+04
  .6413994E+03  .7114275E+03  .7694675E+03  .7724436E+03  .7813064E+03
  .8100986E+03  .3332000E-02  .3322400E-02  .3318300E-02  .3316200E-02
  .3293900E-02  .3327400E-02  .3317700E-02  .3313600E-02  .3311500E-02
  .3289200E-02  .3335000E-02  .3325100E-02  .3321000E-02  .3318800E-02
  .3296200E-02  .3340400E-02  .3330500E-02  .3326300E-02  .3324200E-02
  .3301400E-02  .3344100E-02  .3334100E-02  .3329900E-02  .3327800E-02
  .3304900E-02  .3354300E-02  .3344100E-02  .3339800E-02  .3337600E-02
  .3314500E-02
*
***************              Inv. Neutron Velocities
*
  .5456850E-07  .2491910E-05
*
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