Duke Power Company

OCONEE NUCLEAR STATION

UNITS 1, 2 AND 3

Final Safety Analysis Report

Volume 4

LIST OF EFFECTIVE PAGES
TABLE OF CONTENTS
VOLUME 4

Page	Revision
Cover Sheet, Volume 4	Original
List of Effective Pages ...	Rev. 31
I	Original
II	Original
III	Rev. 9
IV	Rev. 10
IV-a	Rev. 18
V	Rev. 21
VI	Rev. 21
VII	Rev. 5
VIII	Rev. 19
IX	Rev. 21
IX-a	Rev. 18
X	Rev. 21
XI	Rev. 18
XII	Rev. 22
XIII	Rev. 31

Rev. 29. 6/29/73
Rev. 31. 2/15/74

TABLE OF CONTENTS

APPENDIX 1A

TABLE OF CONTENTS - (Continued)

Rev. 9. 8/11/70

TABLE OF CONTENTS - (Continued

Rev. 10. 8/28/70
Rev. 18. 3/10/72

TABLE OF CONTENTS - (Continued)

TABLE OF CONTENTS - (Continued)

Rev. 10. 8/28/70
Rev. 21. 7/26/72

TABLE OF CONTENTS - (Continued)

Rev. 3. 3/16/70
Rev. 5. 5/25/70

TABLE OF CONTENTS - (Continued)

Rev. 4. 4/20/70
Rev. 5. 5/25/70
Rev. 18 3/10/72
Rev. 19 5/5/72

TABLE OF CONTENTS - (Continued)

Rev. 9. 8/11/70
Rev. 18. 3/10/72
Rev. 21. 7/26/72

TABLE OF CONTENTS - (Continued)

TABLE OF CONTENTS - (Continued)

Rev. 4. 4/20/70
Rev. 5. 5/25/70
Rev. 18. 3/10/72
Rev. 21. 7/26/72

TABLE OF CONTENTS - (Continued)

Rev. 4. 4/20/70
Rev. 5. 5/25/70
Rev. 18 3/10/72

TABLE OF CONTENTS - (Continued)

22. Replacement Pages for Units 1 and 2 Technical Specifications
Dockets 50-269, -270

Replacement Pages for Units 1, 2, and 3 Technical Specifications
Dockets 50-269, -270, -287

Rev. 4. 4/20/70
Rev. 5. 5/25/70
Rev. 13. 9/17/70
Rev. 14. 12/14/70
Rev. 16. 7/30/71
Rev. 18. 3/10/72
Rev. 22. 8/25/72

TABLE OF CONTENTS - (Continued)

FSAR SUPPLEMENTS (Located at rear of Volume 4)

LIST OF EFFECTIVE PAGES
FSAR SUPPLEMENT 15

DUKE POWER COMPANY

OCONEE NUCLEAR STATION

UNITS 1, 2, AND 3

APPLICATION FOR LICENSES

Dockets 50-269, -270, and -287

FSAR SUPPLEMENT 1

Submitted with FSAR Revision 4
April 20, 1970

LIST OF EFFECTIVE PAGES

FSAR SUPPLEMENT 1

Page	Revision
List of Effective Pages ..	Rev. 24
Cover Sheet Supplement 1 .	Rev. 4
1-1	Rev. 5
1-2	Original
1-3	Original
Fig. 1	Original
1-5	Original
1-6	Original
1-7	Original
1-8	Original
1-9	Original
1-10	Original
1-11	Original
1-12	Original
1-13	Original
1-14	Original
1-15	Original
1-16	Rev. 5
1-17	Rev. 5
1-18	Original
1-19	Rev. 5
1-20	Original

In response to questions contained in Dr. Peter A. Morris's letter of November
5. 28, 1969, the following information is submitted, tabulated by question number.

QUESTION-1 Provide a discussion of relevant experimental and operational data on sustained DNB available in the literature and in B&W's experimental programs.

ANSWER The heat transfer mechanism after reaching the "critical heat flux" (CHF) is either transition boiling or film boiling. The transition boiling regime varies between nucleate boiling with attendant high heat transfer coefficients, and film boiling, with much lower heat rate transfer coefficients. Data on post CHF heat transfer are somewhat meager compared to other heat transfer data. However, experimentors have investigated this boiling phenomenon within certain parameters and conditions.

Visual studies of film boiling[1,2] at MIT have shown that when the fluid is subcooled or very low in quality the flow is annular with the liquid in the center of the flow channel and the vapor forms a film around the heated surface. At high qualities the liquid core breaks up and the flow regime is one where liquid droplets are carried along in the vapor matrix. Sufficient data are available to develop correlations applicable to post critical heat flux operation in this regime of film boiling. Bishop, et al[3] report film boiling data obtained under the following conditions:

Heat Flux	$.20 - .61 \times 10^6$	BTU/hr-ft^2
Mass Velocity	$.5 - 2.49 \times 10^6$	lbm/hr-ft^2
Pressure	2430 - 3120	psia
Coolant inlet temp.	661 - 701 °F	
Test section I.D.	.1 and .2 in.	
Test section length	2, 6, and 9 ft.	

The data obtained during this investigation were combined with the data of Miropolski[4] to develop the correlation applicable to post CHF operation that was used in BAW-10014. The data of Miropolski were obtained from a .315 in. dia., 59 in. long test section. The pressure range was 294 psia to 2940 psia.

The General Electric Co. performed an extensive study of transition boiling and film boiling heat transfer under AEC sponsorship. The results of this program are summarized in reference 5. This study included data from flow inside tubes and in rod bundles. However, the results are not directly applicable to PWR conditions since the pressure range was 600 - 1400 psia.

While the correlation reported in reference 2 was developed from data that covered the pressure range of interest to PWR designers, the largest amount of data were for pressures of 2600 psi or greater. In order to assure ourselves that there were no major pressure effects omitted in the correlation, B&W conducted a sustained CHF experimental program with most of the data obtained at a pressure of 2200 psia. Two runs were made at 2600 psia so that a comparison could be made with the Westinghouse data. The range of test parameters investigated were:

Mass velocity	$0.5 - 2 \times 10^6$ lbm/hr-ft^2
Heat flux	$0.2 - 0.35 \times 10^6$ BTU/hr-ft^2
Test section ID	0.444 in.
Heated length	72 in.

Thermocouples were spaced on 3 inch centers.

This testing covered the range of interest for sustained CHF as analyzed in BAW-10014. For example, the condition analyzed in BAW-10014 was for mass velocity of 1.6×10^6 lbm/hr-ft^2 and a heat flux of 258,000 BTU/hr-ft^2.

At 2600 psia the data were in good agreement with the data of Reference 2. However, at 2200 psi the correlation of Reference 2 tended to under-predict the wall temperature. This is shown in Figure 1 in which the B&W data is compared with predictions made with the Reference 2 correlation. Figure 1 shows the test conditions that were closest to the conditions analyzed in BAW-10014 with both the heat flux and mass velocity being slightly larger than for the analyzed point. The maximum temperature difference between that predicted with Reference 2 and measured is about 150 F. With this increase factored into the analysis presented in BAW-10014, the primary conclusions are still the same. These are:

1. A DNB which occurs on a fuel rod due to a flow blockage will not propagate to adjacent fuel rods.

2. The maximum fuel temperature which would occur in a fuel rod in sustained DNB due to a flow blockage is well below the melting point for UO_2.

3. A corrosion reaction sufficiently rapid to cause a sudden energy release would not occur and the mode of cladding failure due to corrosion - erosion would probably be a slow local failure.

4. Short-term cladding strength is sufficient to prevent cladding burst, even with considerably more internal pressure than calculated.

REFERENCES

1. R.S. Dougall "Film Boiling on the Inside of Vertical Tubes with Upward Flow of the Fluid at Low Qualities", MIT Department of Mechanical Engineering, EPL Report No. 9079-26, September 1963.

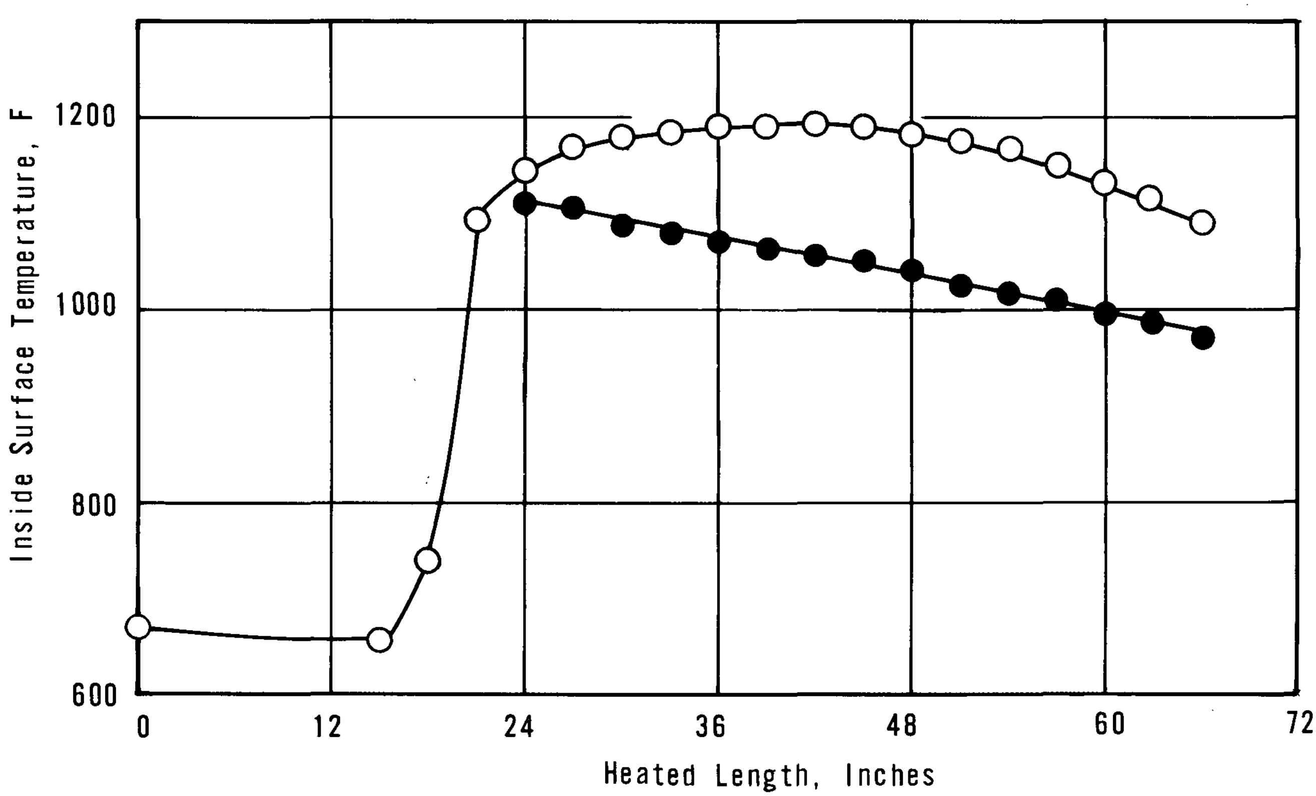
2. W.F. Laverty, W.M. Rohsenow "Film Boiling of Saturated Nitrogen Flowing in a Vertical Tube." Transactions of the ASME, 1967.

3. Bishop, A.A., Sandberg, R.O., and Tong, L.S., "Forced Convection Heat Transfer at High Pressure After the Critical Heat Flux," ASME 65-HT-31 (1965).

4. Z.L. Miropolski, "Heat Release During the Film Boiling of a Steam Water Mixture in Steam Generating Tube." Teploenergelika, Vol. 10, 1963.

5. E.E. Polomik "Transition Boiling Heat Transfer Program (Final Summary Report), AEAP-5563, October 1967.

MASS FLOW, $G = 2.0 \times 10^6$ LBM/HR-FT^2
HEAT FLUX - 350,000 BTU/HR-FT^2
PRESSURE - 2200 PSIA
SOLID SYMBOLS - CORRELATION PREDICTIONS (REF.2)
OPEN SYMBOLS - TEST DATA



OCONEE NUCLEAR STATION

FSAR Supplement 1

Figure 1

QUESTION-2 Provide a list of the mechanisms that might cause high heat flux leading to sustained DNB operation in the Oconee reactors (e.g. misplaced fuel assemblies). Include a discussion of the location and magnitude of these high heat fluxes and an analysis of the precautionary measures and design features of the Oconee reactors which limit the possibility and consequences of high heat fluxes leading to sustained DNB.

ANSWER The most credible mechanisms that could lead to high heat fluxes which may or may not result in a DNB are:

1. Fuel assembly loaded in wrong position.

2. Fuel rod or fuel pellet positioned improperly.

3. Malpositioned control rod assembly.

With respect to the first item, gross error would have to occur for a fuel assembly to be loaded in the wrong position. A fuel loading procedure will be issued to specify how and where each fuel assembly is to be loaded. The loading procedure for all 177 fuel assemblies will follow a predetermined sequence with each assembly being placed in the grid with a prescribed orientation. An independent check will verify that the fuel assembly identification and core position are correct.

For the second credible mechanism, it is extremely improbable that a fuel rod or pellet could be positioned in the wrong assembly or fuel rod. The fabrication process requires that a prescribed sequence be followed with concurrent independent quality assurance inspections. Fuel rod end caps are carefully stamped to identify the individual fuel rods and nominal enrichment in each rod. As received fuel pellets are given special prescribed attention throughout the entire fabrication process. Inspection procedures are carried out at the fuel pellet loading stations to assure that the correct pellet stack length and spacers are loaded to each pin prior to the helium back fill and final weld. Even if one assumes that improper loading can occur, and further that the error is sufficient in magnitude to result in a DNB, the condition would occur at beginning of life. At this time there would be essentially no internal fuel rod pressure. The failure would be one in which the cladding collapses onto the fuel. This would not result in an undetected condition that could lead to progressive failures. The situation would remain localized until the cladding had corroded away to permit detection as discussed in the answer to question 4.

The probability of occurrence of this type of DNB is further reduced by the magnitude of error that would be necessary. For example, the minimum DNBR for the hot rod, hot spot, is 2.0 at rated power. An error in locating the pellet would have to be sufficiently large to result in an increase in local heat flux by a factor of two. Further, this pellet would have to be placed at the hot spot in the hot rod. The combination of events in conjunction with the failure of B&W fuel loading Quality Assurance and QA inspection makes the probability of such an occurrence extremely small.

The third mechanism that could lead to high heat fluxes is related to the position of the control rods. Position measurement is accomplished by two separate methods. One method employs a series of reed switches located in a tube adjacent to the lead screw pressure housing tube. As the lead screw moves up in its housing, the reed switches are actuated in sequence giving a voltage that is an analog indication of the position of the rod. This indication is read out as a continuous indication on the operator control panel. The second method of position measurement is by a miniature motor, similar to the control rod drive motor, which drives a position potentiometer. The miniature motor responds to pulses of power to each rod group thus providing position information. A gross indication of rod position is supplied by the upper and lower limit switches. This information is supplied in a form that tells the operator whether a given rod is at upper limit, lower limit, or somewhere in between.

The 69 control rods of the core are separated into groups containing 8 to 12 control rods. If any one rod of a group deviates from the group average position by more than 5 inches, the outward motion of all control rods is stopped and the reactor power is runback to 60% of full power. Thus a single rod of a group cannot be greater than 5 inches from the group average position when the reactor power is above 60% of rated power.

Groups 1 through 4 of the control rods make up the safety bank. The safety bank must be withdrawn and placed on the holding buses before group 5 can be withdrawn. Groups 5, 6, and 7 are, respectively, the transient, Doppler and xenon override rod groups. The integrated control system imposes a restriction that these rod groups be moved in sequence. There are no criticality or power level restrictions (other than administrative restrictions) concerning rod groups 5, 6, and 7; however, the reactor could not remain at full power for a long period of time with group 5 inserted and group 6 less than 95% withdrawn. The reasoning here is that without moderator dilution the reactor cannot remain at full power for long periods of time, and dilution can occur only after rod group 6 is greater than 95% withdrawn.

QUESTION-3 Provide additional analysis and documentation of the local consequences to irradiated fuel elements caused by sustained DNB operation (e.g., accelerated cladding corrosion, fatigue, rod swelling, rod bowing). Include an analysis of the possibility and extent of DNB propagation caused by associated cladding failure.

ANSWER If a condition developes to cause a departure from nucleate boiling during full power operation, the resulting film boiling heat transfer mechanism will, of course, yield high clad temperatures. The assumed sustained condition requires that the higher temperature persist until failure occurs. The analysis presented in BAW-10014 indicates fuel rod internal pressure to be less than system pressure. Therefore, rod swelling would not occur at these conditions. The maximum calculated temperature in BAW-10014 (which was used to evaluate the fuel rod internal pressure) is considerably lower than that required to produce a fuel rod internal pressure equal to the reactor coolant system pressure. The power peaking history for the various core regions indicates that maximum nuclear peaking will occur early in core life. At later times in life, the nuclear factors are lower. This indicates that if a DNB did occur, it would be early in core lifetime. At this time in life the internal pressure would be considerably less than the value reported in BAW-10014 which was calculated for end-of-life conditions. Thus, the cladding would collapse on the fuel rather than swell or burst, and the resulting failure would be from corrosion.

Operation at sustained DNB will result in an acceleration of the corrosion rate. The magnitude of the rate change is, of course, highly dependent on the metal temperature. At the 1100 F reported in BAW-10014, the corrosion rate is low enough to take a matter of days to result in failure. However, if the metal temperature were in the 1600 F range, a corrosion rate could occur which would result in failure in a matter of hours. The failure would not be expected to be of violent nature because of the time involved. The type of failure that one would expect also supports the conclusion that a DNB condition in one channel would not propagate to surrounding channels. There are several reasons for this conclusion:

1. The most probable fuel rod to fail under normal operation is the fuel rod with the highest peaking factor. This rod is surrounded by rods with lower nuclear peaking factors and lower coolant flow requirements.

2. The relatively cold control rod channels and guide tubes with their dispersed arrangement throughout the assembly provide an effective heat sink.

3. A DNB would result in an increase in the void fraction in the channel in which the DNB occurs. A corresponding reduction in moderating hydrogen atoms will produce a very large negative local moderator coefficient and a reduction in fuel rod power. A concurrent rise in fuel temperature will also produce a negative doppler coefficient.

4. Any channel blockage that occurred would also promote local turbulence which would enhance the heat transfer capability.

5. Any reduction in flow surrounding a fuel rod undergoing a DNB will result in a local increase of flow to other fuel rods in the vicinity.

The favorable physical arrangement and inherent local power self-regulating characteristics would retard any attempt at propagation of DNB conditions. Consequently, the analysis reported in BAW-10014 and the above conclusions support the contention that sustained DNB will not propagate.

QUESTION-4 Provide an analysis of the methods available in the Oconee reactors to detect operations at sustained DNB.

ANSWER Operation at sustained DNB conditions would probably go undetected until the fuel rod failed. It is very doubtful that the power peaking change (as a result of local increase in voids caused by the DNB condition) could be detected by the in-core monitors. Therefore, the detection of sustained DNB could only be accomplished if the condition persisted long enough to allow the fuel rod to fail.

Gross fuel rod failure at the Oconee Nuclear Station will be detected by means of RIA-36, a sodium iodide scintillation detector that monitors gross gamma activity in the reactor coolant letdown line upstream of the purification demineralizers. Studies by B&W and Duke have shown that the gross gamma method can provide one of the earliest indications of fuel failure after startup (of the several methods analyzed) and is an effective monitor at equilibrium conditions. The other methods included individual nuclides such as Cs^{137}, I^{131}, and Mo^{101}, as well as delayed neutrons.

RIA-36 is a part of the process radiation monitoring system and reads out in the respective control rooms. The actual detector is shielded by 6" of lead and therefore the external gamma background effect on sensitivity is not significant, both in the case of operation with no defective fuel or in the 1% failed fuel condition. A removable sample pipe within the shield eliminates interference from contamination.

As to sensitivity, the system is capable of detecting a small fraction of the activity from one fuel rod at anytime during operation in the presence of the highest allowable tramp uranium activity. The activity released from a fuel rod rupture is assumed to be that associated with the release of 1% of the fission products in the ruptured rod. The instrument can detect this level of activity either early in the cycle or at equilibrium when 0.01% or less defective fuel and tramp uranium is contaminating the coolant. In the presence of 1% defective fuel, the instrument is capable of detecting the activity associated with approximately 50 fuel rod ruptures early in the cycle and that associated with approximately 100 fuel rod ruptures at equilibrium before the instrument is overranged. As a result of actual calibration performed by the manufacturer, Victoreen Instrument Division, the instrument approaches being overranged by activity at about 1000 uCi/ml of composite fission product activity. This activity is equivalent to about a 3% defective fuel condition.

Fuel failures can be detected promptly since the delay time from the core outlet to the detector is approximately one (1) minute and fifteen (15) seconds (which also allows for the decay of interfering N^{16} activity.)

QUESTION-5 Provide a discussion of how accidents, transients, and the function of engineered safeguards would be affected by operation at sustained DNB.

ANSWER The present accident analysis will not be affected by the assumption that a small portion of the core is initially in film boiling. The accidents which could possibly be affected by this additional assumption are limited to those which are initiated from high power levels. The most severe of the hypothesized reactivity transients, the rod ejection accident, is already analyzed assuming a perfectly insulated fuel rod -- a worse assumption than film boiling. Other transients, such as rod withdrawal and loss-of-coolant flow, are shown to be terminated at DNB ratios above 1.3 and therefore no DNB other than the assumed steady state sustained DNB can be expected. The damage criteria may be exceeded for these assumed initial conditions (depending on the magnitude of the assumption) but the analysis itself is unchanged.

The only consideration that might be added to the above is fuel failure propagation. Such a phenomenon has not been demonstrated and, in fact, recent tests in the Capsule Drive Core at the National Reactor Testing Station have demonstrated lack of propagation in severe reactivity transients, as reported in recent monthly progress reports. Finally, any hot channel condition worse than that normally assumed would cause negative reactivity addition in the vicinity of this channel, and subsequent accident transients would therefore further shutdown such regions.

Operation of the ECCS will not be affected by sustained DNB. Since sustained DNB does not propagate, the effectiveness of the ECCS will not be diminished.

In response to the questions contained in Dr. Peter A. Morris' letter of February 13, 1970, the following tabulation, in the same format as the Director's letter, gives the information requested or the reference where the information is contained in the FSAR.

2.0 SITE AND ENVIRONMENT

2.1 FSAR - 2.2.5, Page 2-2a

2.2 FSAR - 2.4.2, Page 2-7 to 2-7a

2.3 FSAR - Appendix 2A

3.0 REACTOR

3.1 Reactivity Calculations

3.1.1 FSAR - 3.2.2.2.1d, Pages 3-19c to 3-19d

3.1.2 FSAR - 3.2.2.2.1d, Pages 3-19c to 3-19d

3.1.3 FSAR - 3.2.2.2.1c, Pages 3-19c and Figure 3-6a, 3-6b, and 3-6c

3.1.4 FSAR - Table 3-2, Page 3-9 and 3.2.2.2.1e, Page 3-19d

3.1.5 FSAR - 3.2.2.2.1f, Pages 3-19d through 3-19f and Figures 3-6d, 3-6e, 3-6f, and 3-6g

3.2 Reactivity Coefficients

3.2.1 FSAR - 3.2.2.1.4a, Pages 3-15 and 3-16

3.2.2 FSAR - 3.2.2.1.4d, Pages 3-16a through 3-17c

3.2.3 FSAR - 14.2.2.3.4a, Pages 14-14 and 14-14a

3.2.4 FSAR - 3.2.2.1.4d, Page 3-17b and Table 3-7c

3.2.5 FSAR - 3.2.2.1.4d, Pages 3-16a through 3-17c

3.2.6 FSAR - 3.2.2.1.4e, Pages 3-17c and 3-17d

3.3 Shutdown Margin and CRA Worth

3.3.1 FSAR - 3.2.2.1.3, Pages 3-13 through 3-14b

3.3.2 (Later)

3.3.3 FSAR - 3.2.2.1.3, Pages 3-13 through 3-14b, Figures 3-4c, 3-4d, 3-4e, and 3-4f

3.4 <u>Xenon Stability</u>

FSAR - 3.2.2.2.2, Page 3-24, and Figures 3-6i and 3-6j

3.5 <u>Detection and Control of Power Maldistributions</u>

3.5.1 FSAR - 3.2.2.2.1f and g, Pages 3-19d through 3-20

3.5.2 FSAR - 3.2.2.2.1g, Pages 3-19f and 3-20

3.5.3 FSAR - 3.2.2.2.1g, Pages 3-19f and 3-20

3.5.4 FSAR - 3.2.2.2.1f, Page 3-19f; and 7.3.1.3, Page 7-33

3.5.5 FSAR - 3.2.2.2.1g, Pages 3-19f and 3-20

3.6 <u>Thermal - Hydraulic Design</u>

3.6.1 FSAR - 14.1.2.6.2, Pages 14-11 and 14-11a

3.6.2 FSAR - 3.2.3.2.3e, Pages 3-43 and 3-43a

3.6.3 (Later)

3.6.4 BAW-10021 - B&W Mixing Code, Temp. to be submitted about May 1, 1970.

3.6.5 FSAR - 3.2.3.1.1j, Page 3-32

3.6.6 FSAR - 3.2.3.1.1b2, Pages 3-24 and 3-24a, and Figure 3-6k

3.6.7 ANSWER: The overpower design criteria is based on the assumption that steady-state operation at 114% power occurs with design reactor coolant flow (131.32 x 10^6 lb/hr) at a constant average reactor coolant temperature of 579°F. With this assumption the reactor coolant outlet temperature increases with power and the reactor coolant inlet temperature decreases with power as shown in Figure 4-9 of the FSAR. The steam system flow rates are compatible with reactor coolant system operation at 114% power and at a constant average temperature of 579°F provided that:

a) the feedwater temperature is lower than normal, or
b) the turbine bypass system is flowing steam, or
c) combination of a) and b).

Reactor protection and DNBR limits do not depend on operation at 579°F constant reactor coolant temperature. Figure 15-2 of the FSAR shows the protection envelope for reactor coolant temperature and pressure. The reactor outlet temperature boundaries between 585°F and 620°F are based on 114% power and a minimum DNBR limit of 1.3. The following tabulation indicates the DNB conditions at 114% power.

Power %	T_{out} °F	Avg. R.C. Temp. °F	T_{in} °F	R.C. Pressure PSIA	DNBR
114	608	579	551	2200	1.55
114	620	593	566	2220	1.30

The tabulation indicates that reactor protection does not depend on constant average temperature in that:

a) the reactor average temperature can be elevated about 14F before reactor trip at the DNB limit.

b) the reactor inlet temperature can be elevated about 15F before reactor trip at the DNB limit.

c) the reactor outlet temperature can be elevated about 16F before reactor trip at the DNB limit.

3.6.8 FSAR - 3.2.3.2.3g3, Page 3-46, and Figure 3-32a.

3.6.9 FSAR - 3.2.4.2.1b, Page 3-64.

3.7 Internal Vent Valve

3.7.1.1 FSAR - 3.2.4.1.2h, Page 3-62a, and Table 3-16a.

3.7.1.2 FSAR - 3.2.4.1.2h, Page 3-62b, and Table 3-16b.

3.7.1.3 FSAR - 3.3.4, Page 3-94c

3.7.1.4 FSAR - 3.2.3.2.4, Pages 3-57 through 3-57c.

3.7.1.5 FSAR - 3.2.3.2.4, Pages 3-57 through 3-57c.

3.7.2 FSAR - 3.2.3.2.4, Pages 3-57 through 3-57c.

3.7.3 FSAR - 3.2.3.2.4, Pages 3-57a and 3-57b.

3.7.4 FSAR - 3.2.3.2.4, Pages 3-57 through 3-57c.

3.7.5 FSAR - 3.3.4, Page 3-94b.

3.7.6 FSAR - 3.3.4, Page 3-94c; and Page 13-17.

3.8 Reactor Internals

FSAR 3.3.4, Pages 3-94 through 3-94c.

4.0 REACTOR COOLANT SYSTEM

4.1 ANSWER: With regard to brittle fracture control of the reactor coolant pressure boundary, piping with wall thickness less than 1/2 inch need not have material property tests (such as charpy, V-notch) because they are austenitic stainless steel.

In accordance with Figures 15-3, 15-4, 15-4a, 15-4b, and 15-4c of the Oconee Technical Specification, at no time during heatup and cooldown of the reactor coolant system during the plant life will any component including piping be pressurized in excess of a reactor coolant system pressure of 20% of design pressure while the system is below DTT (NDTT + 60°F).

4.2 (Later)

4.3 FSAR - 4.2.6, Page 4-19; and Appendix 4B (submitted with Amendment 10, 3/16/70.

4.4 FSAR - 4.1.2.3, Page 4-2.

4.5 FSAR - 4.3.3, Page 4-24 and Figure 4-11.

4.6 FSAR - Appendix 4B (submitted with Amendment 10, 3/16/70).

4.7 (Later)

4.8 Reactor Vessel

4.8.1 ANSWER: There are no additional reactor vessel design requirements beyond ASME Section III, imposed by the State of South Carolina for the Oconee reactor vessels.

4.8.2 FSAR 4.3.3, Page 4-24.

4.8.3 FSAR 4.2.2.1, Page 4-7, and Figures 4-4, 4-4a, 4-4b, and 4-4c.

4.8.4 FSAR - 4.3.3, Page 4-22, and Figure 4-4c.

4.8.5 FSAR - 4.3.3, Page 4-22, and Tables 4-8 and 4-15.

4.8.6 FSAR - 4.2.2.1, Page 4-7

4.9 Steam Generator

4.9.1 through 4.9.11 (Later)

4.10 (Later)

4.11 (Later)

4.12 Other Class I Systems and Components

4.12.1 (Later)

4.12.2 FSAR Appendix IC (Submitted with Amendment 9 dated 2/9/70).

4.12.3 FSAR Appendix IC-3.4.1, Page 1C-4a.

4.12.4 (Later)

4.12.5 (Later)

4.13 Pipe Whip and Missile Protection

4.13.1 (Later)

4.13.2 (Later)

Sup. #2

4.13.3 FSAR - 4.2.2.6, Page 4-12, 4-12a, 4-12b, 4-12c.

4.13.4 FSAR - 4.2.2.6, Page 4-12, 4-12a, 4-12b, 4-12c

4.14 Inservice Inspection

4.14.1 ANSWER: The basis for the Inservice Inspection Program is Section XI of the ASME Boiler and Pressure Vessel Code as explained in Appendix 4A, Revision 4, 4/20/70.

4.14.2 ANSWER: By definition in Appendix 1C, System Design Criteria for Natural Phenomena, Section 1C.3, there are no Class I systems outside the primary system pressure boundary.

The inspection schedule for the reactor coolant pump flywheels is given on page 4-12a included in Revision 4. The primary vessel support inspection program is as given in Appendix 4A, Inservice Inspection.

All of the remaining equipment in the plant will be inspected as required consistent with past experience and good engineering judgment.

4.15 Leak Detection

4.15.1 FSAR - 4.3.10.3, Pages 4-32 and 4-32a.

4.15.2 FSAR - 4.3.10.3, Page 4-32a.

5.0 Structures

5.1 FSAR - Figure 5-4, Sheets 1 and 2.

5. 5.2 FSAR - 5.1.3.1, Pages 5-12

5.3 FSAR - 5.1.3.1, Pages 5-12 and 5-12a.

5.4 FSAR - 5.7.1.2, Page 5-61a.

5.5 FSAR - 5.7.1.2, Page 5-61a.

5.6 ANSWER: We understand the purpose and scope of a containment design report to be as follows:

1) Present design basis
2) Present design criteria
3) Present design analysis
4) Implementation of design criteria
5) Description of construction and testing procedures

Items 1, 2, 3 and 4 are covered by FSAR Section 5 and 5A and in answers to questions 5.1, 5.2, 5.3, 5.7, 5.8, 5.9, 5.11 and 5.12. Item 5 is covered in FSAR Sections 1B and 5B.

The detail analysis and design calculations, including computer printouts, have been made as described and results summarized in FSAR.

Quality Assurance of Engineering and Construction has been in accordance with FSAR Section 1B. Quality Control for containment materials and construction is described in FSAR Section 5B.

5.7 FSAR - 5.1.2.1, Page 5-3 and Table 5-0, Pages 5-3a and 5-3b.

5.8 FSAR - 5B.2.7, Pages 5B-8 through 5B-8b.

5.9 (Later)

5.10 FSAR - 5.7.2.2, Pages 5-62 through 5-62b, and Figure 5-22; and 1C-3.4.2, Page 1C-4c (Later).

5.11 (Later)

5.12 FSAR - 5.1.3.2, Page 5-19

5.13 FSAR - 1C-3.4.1 and 1C-3.4.2 (Later), Pages 1C-4b and 1C-4c.

5.14 FSAR - 5.7.2.2, Pages 5-62 through 5-62b and Figure 5-22.

5. 5.15 FSAR - 5.6.2.2, Page 5-59a

6.0 ENGINEERED SAFETY FEATURES

6.1a (Later)

6.1b (Later)

6.1c (Later)

6.1d FSAR 6.4.3, Page 6-29.

6.1e (Later)

6.1f (later

6.1g (Later)

6.2 (Later)

6.3 FSAR - 6.1.2.12, Page 6-12, and Tables 6-2a and 6-2b, Pages 6-12b and 6-12c.

7.0 INSTRUMENTATION AND CONTROL

7.1 FSAR - 7.1.1.8, Pages 7-2a and 7-2b; and 8.2.3.5, Page 8-18.

7.2 FSAR Appendix 1B.5.10, Pages 1B-11 through 1B-11b.

7.3 FSAR - 7.1.2.3.5, Pages 7-10 and 7-10a; 7.1.3.3.2, Page 7-15; 7.1.3.3.3, Pages 7-15 and 7-16; 8.2.2.12, Page 8-10.

7.4 FSAR - 7.1.1.7, Pages 7-2 and 7-2a.

7.5 FSAR - 7.5, Pages 7-44 and 7-44a.

7.6 FSAR - 7.4.5, Pages 7-43 through 7-44.

7.7	FSAR - 7.4.4, Page 7-43.
7.8	FSAR - 8.2.4, Pages 8-18 and 8-18a.
7.9	FSAR - 7.1.3.3.6, Page 7-16a.
7.10	FSAR - 7.1.2.2.4, Page 7-6a; 7.1.2.3.8, Pages 7-10a and 7-10b; 7.1.3.2.5, Page 7-13; 7.1.3.3.4, Page 7-16
7-11a	FSAR - Table 7-1, Page 7-4 and 7-5; Table 15-1, Page 15-11; 15.2.3, Page 15-9 and 15-10
7.11b	FSAR - Figure 7-6; Figure 7-7.
7.12	FSAR - 7.2.2.3.4, Pages 7-22 through 7-23c.
7.13	FSAR - 7.2.2.3.4, Pages 7-22 through 7-23c.
7.14	FSAR - 7.1.3.3.4, Page 7-16
7.15	FSAR - 7.2.2.1.2, Pages 7-17 and 7-17a.
7.16	(Later)
7.17	FSAR - 7.2.3.3.1, Page 7-29.
7.18	FSAR - 7.3.2.2.1, Pages 7-34 through 7-35a; Table 7-6, Page 7-35; Table 7-7, Page 7-35a.
7.19	FSAR - 7.3.1.2.1, Page 7-32 and 7-32a.
7.20	FSAR - 7.1.2.3.2, Page 7-8.
7.21	FSAR - 7.1.2.2.3, Page 7-5 and 7-6.
7.22	(Later)
11.0	<u>RADIOACTIVE WASTES AND RADIATION PROTECTION</u>
11.1	(Later)
11.2	(Later)

11.3 FSAR - 11.1.2.3.2, Page 11-14 and Fig. 11-3.

11.4 FSAR - 11.1.2.5.2, Pages 11-26 to 11-26a, and Fig. 11-3.

11.5 FSAR - 5.3.1.2, Page 5-44; and 5.3.2.1, Page 5-45.

11.6 (Later)

5. | 11.7 FSAR - 2.4.2, Page 2-8.

12.0 CONDUCT OF OPERATION

12.1 FSAR - 12.3.2, Pages 12-8 through 12-8h.

12.2 FSAR - 12.2.3, Pages 12-7 and 12-7a.

12.3 FSAR - 13.1, Pages 13-1 through 13-3; Appendix 12A; and Fig. 12-4A.

14.0 SAFETY ANALYSIS

14.1 (Later)

14.2 Design Basis Loss-Of-Coolant Accident

14.2.1 through 14.2.7 (Later)

Docket 50-269, -270, and -287
FSAR Supplement 1
April 20, 1970

The following information is voluntarily submitted in response to informal questions ask by the Division of Reactor Licensing:

QUESTION-1 In Section 14.2.2.1.2 of the FSAR, the fuel handling accident 2-hr dose at the site boundary is analyzed assuming 56 rods fail in a fuel assembly and iodine removal in water affords a factor of 100 reduction. On this basis integrated dose at the 1 mile exclusion distance is calculated as 0.027 Rem whole body and 0.43 Rem thyroid. Recalculate these integrated doses based on the following assumptions:

All 208 rods fail
20% of the noble gas in the assembly released
10% of the iodine in the assembly released
90% retention of iodine in pool water
72 hour shutdown decay preceding accident
2 hour ground release X/Q of 7.41×10^{-5}

ANSWER Even assuming the extremely unrealistic conditions proposed, which apply a factor of conservatism of 100 to 500 to the calculation presented in 14.2.2.1.2, the 2-hr integrated doses at the exclusion distance would be 0.66 Rem whole body and 174 Rem thyroid, well within the 10 CFR 100 criteria used by the Commission in its evaluation of proposed sites for power reactors.

QUESTION-2 At what value of stress in the concrete was the non-linear stress/strain relationship assumed to begin? Under what loading conditions, in what areas, and to what value was the modulus of elasticity corrected? For a sample area under a given loading condition, tabulate the magnitude of stress before and after the modulus of elasticity revision.

ANSWER FSAR-5.1.3.1, Pages 5-14 and 5-14a

QUESTION-3 It is stated in the FSAR that the loads and stresses at transfer of prestress will be compared to those allowed by ACI 318-63. However, the seating stresses shown for the three tendon types exceed the ACI value. Explain why this is acceptable.

ANSWER FSAR-5.1.4.2, Pages 5-34, 5-34a, and 5-34b

QUESTION-4 On Page 5-14, it is stated that the liner was treated as an integral part of the structure. Does this mean that it was included in the finite element mesh of the containment structure? If so, please provide a detailed sketch of the mesh.

ANSWER FSAR-5.1.3.1, Pages 5-14 and 5-14a, and Fig. 5-4

QUESTION-5 For which loading cases do the isostress plots shown on Figures 5-6 and 5-7 apply?

ANSWER FSAR-5.1.3.1, Pages 5-14 to 5-14a

DUKE POWER COMPANY

OCONEE NUCLEAR STATION

UNITS 1, 2, AND 3

APPLICATION FOR LICENSES

Dockets 50-269, -270, and -287

FSAR SUPPLEMENT 2

Submitted with FSAR Revision 5
May 25, 1970

LIST OF EFFECTIVE PAGES

FSAR SUPPLEMENT 2

Docket 50-269, -270, and -287
FSAR Supplement 2
May 25, 1970

In response to the questions contained in Dr. Peter A. Morris' letter of February 13, 1970, the following tabulation, in the same format as the Director's letter, gives the information requested or the reference where the information is contained in the FSAR for the questions marked "Later" in FSAR Supplement 1 dated April 20, 1970:

3.3 Shutdown Margin and CRA Worth

3.3.2 FSAR - 15.3.5.2, Pages 15-33 and 15-34; and 15.4.7, Page 15-61.

3.6 Thermal-Hydraulic Design

3.6.3 ANSWER: Production assembly measurements are not yet available for calculation of engineering hot spot factors. These factors will be provided when available.

3.7 Internal Vent Valve

3.7.1.4 (Additional information supplementing that listed in Supplement 1) FSAR - 14.2.2.3.5, Pages 14-61a and 14-62, and Figures 14-63d and 14-63e.

3.7.1.5 (Additional information supplementing that listed in Supplement 1) FSAR - 14.2.2.3.4, Page 14-52.

4.0 REACTOR COOLANT SYSTEM

4.2 FSAR - 4.1.3.3, Pages 4-5, 4-5a, 4-5b, and Tables 4-20 and 4-21 (Pages 4-61 and 4-62).

4.7 FSAR - 4.2.2.2, Pages 4-9 and 4-9a; 4.2.2.3, Page 4-10; and 4.2.2.5, Pages 4-11 and 4-11a.

4.9 Steam Generator

The non-proprietary summary of the once-through steam generator development is given in FSAR - 4.3.4, Page 4-28. Questions 4.9.1 through
21. 4.9.10 are answered by B&W Topical Report BAW-10027.

21. 4.10 FSAR - 4.1.2.4, Page 4-2; and Section 8.8 of BAW-10027.

4.11 ANSWER: The feasibility of in-service monitoring for vibration and the detection of loose parts is being explored by B&W. We have investigated the application of such sensors as accelerometers, strain gages and load cells to monitor vibration of internals, and of inertially loaded force pickups to monitor for loose parts. We plan additional discussion with consultants and instrumentation vendors in order to determine the feasibility and practicality of such systems in operating PWR systems.

4.12 <u>Other Class I Systems and Components</u>

4.12.1 FSAR - 1C.3.3, Page 1C-4a, and Table 1C-2 (Pages 1C-6, 7, 8, 9, and 10).

4.12.4 FSAR - 1C.3.3 and 1C.3.4, Pages 1C-4, and 1C-4b; Table 1C-2; and Appendix 4B.

4.12.5 FSAR - 4.3.6, Pages 4-30 and 4-30a.

4.13 Pipe Whip and Missile Protection

4.13.1 FSAR - 4.3.2, Pages 4-21 and 4-21a.

4.13.2 FSAR - 4.2.7, Pages 4-20 through 4-21a, and Tables 4-22 through 4-28 (Pages 4-63 through 4-71).

5.0 Structures

5.9 FSAR - 5.6.1.2.2, Pages 5-53 and 5-54, and Figure 5-21 (sheets 1, 2, and 3).

5.11 FSAR - Appendix 4B.

6.0 ENGINEERED SAFETY FEATURES

6.1 FSAR - 6.4.2 and 6.4.3, Pages 6-27 through 6-29, and Figures 6-5 through 6-5f.

6.2 FSAR - 6.1.2.12, Pages 6-12 and 6-12a, and Tables 6-2a and 6-2b (Pages 6-12b through 6-12e).

7.0 INSTRUMENTATION AND CONTROLS

7.16 FSAR - 7.2.2.2.1 and 7.2.2.3, Pages 7-19 through 7-22, and Figure 7-6a.

7.22 ANSWER: The analysis presented in the FSAR covered a spectrum of rupture sizes from 0.4 ft^2 to 14.1 ft^2 (double-ended rupture of the 36" pipe). For all of these cases, the actuation of the ECCS came as a result of a RCS pressure signal. Also, it was assumed in all of these cases that heat removal by the steam generators ceased at the time of the rupture. This causes a delay in the time to reach the RCS pressure signal (1500 psig) which was used to actuate the HPI system. In all cases, the 4 psig reactor building pressure signal occurs either before the 1500 psig RCS pressure or within one second of that signal as can be seen from the table below. Therefore, either of the signals provides the required signal for ECCS actuation. The 1500 psig signal is designed to provide the earliest possible ECCS initiation for emergency conditions commensurate with allowing an adequate margin to bypass the ECCS initiation during normal cooldown.

7.22 (Continued)

Table 7.22-1
Time to ECCS Actuation vs Break Size

Break Size, ft^2	Time to 1500 psig RCS Press., Sec	Time to 4 psig R.B. Press., Sec
14.1	< 1	< 1
8.5	< 1	< 1
5.0	1.6	< 1
3.0	4.0	< 1
1.0	24.0	3.5
.4	52.0	∿ 9

The instrumentation used to provide reactor trip has been tested in an environment that could result should a loss-of-coolant accident occur and has been shown to function as specified. The reliability of the low reactor coolant pressure signal makes a reactor building pressure signal unnecessary.

The subject of common mode failures is continuing to be evaluated.

11.0 RADIOACTIVE WASTES AND RADIATION PROTECTION

11.1 FSAR - 11.1.3, Pages 11-26a and 11-26b.

11.2 FSAR - 11.1.2.5.2, Page 11-26a.

11.6 ANSWER:

a. Those channels monitoring routine releases should remain on scale for releases up to technical specification limits:

AIRBORNE WASTES

The Technical Specifications permit effluent concentrations in a Unit Vent up to the following:

2.18×10^{-2} uCi/ml for gaseous activity;
2.34×10^{-5} uCi/ml for particulate activity; and
8.05×10^{-6} uCi/ml for iodine activity.

Using data from FSAR table 11-5, Process Radiation Monitors, for these concentrations of activity:

The Unit Vent Gas monitors, RIA-45 and RIA-46, would read 8.6×10^5 cpm and 93 cpm respectively.

11.6 (Continued)

The Unit Vent Particulate Monitor, RIA-43 would read 3.9 X 10^5 cpm.

The Unit Vent Iodine Monitor, RIA-44 would read 9.8 X 10^3 cpm.

Since the upper limit of the range of all these instruments is 10^6 cpm, they will all therefore remain on scale for releases up to (and beyond) the Technical Specifications limits.

LIQUID WASTE

Liquid waste could be pumped out of the Low Level Waste Tank at 50 gpm in concentrations up to 1.5 X 10^{-2} uCi/ml, with both hydro units operating, and still meet the normal effluent release limit of 10^{-7} uCi/ml. Under these most extreme conditions, RIA-33, the Waste Disposal Liquid Monitor (normal), would read 1.0 X 10^6 cpm and RIA-34, the Waste Disposal Liquid Monitor (high), would read 100 cpm.

b. Those channels monitoring the consequences of accidents should remain on scale during the postulated accident:

Two process radiation monitoring channels, RIA-51, the Penetration Room Gas Monitor, and RIA-31, the Low Pressure Service Water Discharge Monitor, are designed to remain on scale during the postulated accident (LOCA and MHA).

RIA-31 is described in Section 11.1.2.4.2(a). The detector for this monitor is located in the Turbine Building. Radiation levels at the detector location have been calculated for LOCA and MHA, and are found to be less than 2 mR/hr. As indicated in FSAR Table 11-5, the detector is shielded for 10 mR/hr and the sensitivity is as shown. Therefore, this monitor is capable of remaining on scale and performing its intended function during and after either the postulated LOCA or MHA.

RIA-51 described in Section 11.1.2.4.2(n) is designed to remain on scale for MHA conditions. The upper range limit on this monitor is based upon the following conditions:

1. Noble gas release from MHA (FSAR Table 14-5).
2. Fifty percent of design Reactor Building leakage into the Penetration Room.
3. One Penetration Room Fan in operation.

11.6 (Continued)

Actually, RIA-51 will remain on scale for noble gas concentrations at least 2 times greater than those calculated as resulting from the MHA.

c. Those channels providing a control function for an engineered safety feature should not have their function denied by the dose consequences of an accident:

There are no channels providing a control function for an engineered safety feature.

14.0 SAFETY ANALYSIS

14.1 FSAR - Appendix 14A.

14.2 Design Basis Loss-Of-Coolant Accident

14.2.1 FSAR - 14.2.2.3.5, Page 14-61.

14.2.2 FSAR - 14.2.2.3.3a (5), Page 14-39.

14.2.3 FSAR - 14.2.2.3.5, Page 14-59; 6.2.3.6, Page 6-20 and Figures 6-11 and 6-11a.

14.2.4 FSAR - 14.2.2.3.8 through 14.2.2.3.8.4, Pages 14-64 through 14-64c.

14.2.5 FSAR - 14.2.2.3.5, Page 14-59; and Table 14-13 (Page 14-60).

14.2.6 FSAR - 14.2.2.3.5, Table 14-13b (Page 14-61a).

14.2.7 FSAR - 14.2.2.3.5, Page 14-61; and Figures 14-63a through 14-63e.

Docket 50-269, -270, and -287
FSAR Supplement 2
May 25, 1970

In response to Dr. Peter A. Morris's letter of March 3, 1970, the answers to questions 11.8, 14.3.1, and 14.3.2 are tabulated below, giving the information requested or the reference where the information is contained in the FSAR.

11.8 FSAR - 11.1.2.4.2(e), Page 11-16; Table 11-5 (Pages 11-18 through 11-21); and FSAR Supplement 1, Page 9.

14.3.1 FSAR - 10.4, Pages 10-6 and 10-7.

14.3.2 ANSWER: As a result of reactor trip, two independent and redundant "Reactor Trip Confirmed" signals in the form of contact closures from the control rod drive system will energize two independent turbine stop valve close channels, A & B.

Channel A will operate the EHC master trip solenoid valves and close
10. all four of the turbine stop valves in approximately 220 milli-seconds
from initiation to valve closure (20 milli-second leeway). The 120 V.A.C. power for the EHC control system is fed from an auxiliary panelboard supplied from a static inverter (1KX), which has redundant power feeds from the Unit 1 battery, or from a regulated power panelboard (1KRA).

Channel B will energize solenoid operated test valves on each turbine stop valve which will close those turbine stop valves in approximately 12 seconds. The 120 V.A.C. power to energize those test valves is
9. fed from an AC Vital Instrumentation Power Panelboard (1KVIC) which
is supplied from a static inverter (1DIC) which has redundant feeders from Unit 1 and 2 batteries.

The above two channel circuitry provides two physically separated independent methods for closing the four turbine stop valves within a time period of twelve (12) seconds after initiation from the "Reactor Trip Confirmed" signal.

The following information is voluntarily submitted in response to informal questions asked by the Division of Reactor Licensing:

QUESTION-1 Cracking is noted to reduce the thermal moment, but unless the change in section stiffness is used to reanalyze the system, the new reduced moment is unknown. Explain in detail, with a numerical example, for the loading of $D + F + P + T_A + E$ at Section G-G, how the total moment, stress distribution, and strain distributions were obtained and used for the proportioning of this section.

ANSWER FSAR-5.1.3.1, Pages 5-14a through 5-14d

QUESTION-2 Insufficient information is presented in the FSAR with regard to the results of the containment analysis. Please provide for both structural concrete and liner:

a. A summary of allowable stresses used in the design.

b. A tabulation at actual locations of principal, meridional, hoop and radial stresses; the allowable stresses at these locations; and the factors of safety comparing critical, allowable and computed stresses for each of the loading combinations.

c. The complete finite element grid system used in the analysis.

d. Stress plots for maximum, minimum, hoop, meridional, and radial stresses for each of the loadings considered in the design taken separately and in combination with the other considered loads (at pertinent factors). Where one dimensional loads have been applied to the structure (such as from the seismic analysis), an explanation of how these loads have been translated into stress values will be included.

ANSWER FSAR-5.1.4.1, Page 5-32; and Table 5-3a (Pages 5-32a through 5-32f)

QUESTION-3 What was the maximum radial shear stress in the concrete under the controlling design load case and what was the radial shear stress under $D + F + P + T_A + E$? Define the value of shear for each component of loading forces of the load combination.

ANSWER FSAR-5.1.4.1, Table 5-3a (Pages 5-32a through 5-32f)

DUKE POWER COMPANY

OCONEE NUCLEAR STATION

UNITS 1, 2 and 3

APPLICATION FOR LICENSES

Dockets 50-269, -270, and -287

FSAR SUPPLEMENT 3

Submitted with FSAR Revision 6

June 22, 1970

LIST OF EFFECTIVE PAGES

FSAR SUPPLEMENT 3

Page	Revision	Page	Revision
List of Effective Pages ..	Rev. 24	3-24	Original
Cover Sheet Supplement 3 .	Rev. 6	3-25	Original
3-1	Original	3-26	Original
3-2	Rev. 21	3-27	Original
3-3	Rev. 21	3-28	Original
3-4	Rev. 21	3-29	Original
3-5	Rev. 21	3-30	Original
3-6	Rev. 21	3-31	Original
3-7	Rev. 21	3-32	Original
3-8	Rev. 21	3-33	Original
3-9	Original	3-34	Original
3-10	Original	3-35	Original
3-11	Original	3-36	Original
3-12	Original	3-37	Original
3-13	Original		
3-14	Original		
3-15	Original		
3-16	Original		
3-17	Original		
3-18	Original		
3-19	Original		
3-20	Original		
3-21	Original		
3-22	Original		
3-23	Original		

In response to the questions contained in Dr. Peter A. Morris' letter of March 3, 1970, the following tabulation in the same format as the Director's letter, gives the information requested or the reference where the information is found.

The general question in the body of the letter regarding the summary of the non-proprietary aspects of BAW-10008 Part 2 is answered in FSAR Appendix 4C, "Summary of Fuel Assembly Stress and Deflection Analysis Due to LOCA and Seismic Excitation" (submitted with FSAR Revision 6).

3.8 Reactor Internals (The following questions apply to B&W Report BAW-10008, Part 1.)

3.8.1 Par. 3.1.1 (BAW-10008, Part I (Rev. 1).)

3.8.2 Answer:

Two additional components would be subjected to buckling - type loads, the control rod guide tubes and the lower grid columns. Their analysis is explained in Para. 3.2.2.3 and 3.2.1, respectively. (BAW-10008, Part I (Rev. 1).)

3.8.3 Answer:

The dynamic load factor approach used for the reactor internals is simply the calculation of an equivalent static force, which accounts for the dynamic response of the structure to a transient applied force. The equivalent static force is used for the stress analysis of the component. The equivalent static force is usually determined as a multiplier (the dynamic load-factor) times the peak transient force.

The dynamic load factor is calculated from an analysis of the particular component subjected to the specified transient load using standard dynamic analysis methods.

For example, the calculation of the dynamic load factor for the plenum cylinder reinforcing plate is given in Appendix H, Para. 1.1. (BAW-10008, Part I (Rev. 1).) The dynamic load factors used for the bell-mode responses of the core support structure shells are given in Appendix H, Para. 4.1. (BAW-10008, Part I (Rev. 1).)

3.8.4 Questions as modified in AEC letter dated 4/22/70.

a. Figure F-1 (BAW-10008, Part I (Rev. 1).)

b. Tables F-1 and F-2 (BAW-10008, Part I (Rev.1).)

c. SHOCK code description, Appendix G (BAW-10008, Part I (Rev.1).)

d. Table F-3 (BAW-10008, Part I (Rev. 1).)

e. Appendix F and Para. 3.1.6 (BAW-10008, Part I (Rev.1).)

3.8.5 Appendix F (BAW-10008, Part I (Rev. 1).)

3.8.6 Appendix H, Para.1 (BAW-10008, Part I (Rev. 1).)

3.8.7 Para. 3.2.2.3 (BAW-10008, Part I (Rev. 1).)

3.8.8 a. Appendix H (BAW-10008, Part I (Rev. 1).)

b. Table 1 (BAW-10008, Part I (Rev. 1).)

3.8.9 Appendix A, Pg. A-7 (BAW-10008, Part I (Rev. 1).)

3.8.10 Appendix B, Pgs. 5 & 7 (BAW-10008, Part I (Rev. 1).)

3.8.11 Appendix C, Pg. C-2 (BAW-10008, Part I (Rev. 1).)

3.8.12 Appendix D (BAW-10008, Part I (Rev. 1).)

3.9 Fuel Assembly Structural Design (The following questions apply
21. to BAW-10035, Non-proprietary version of BAW-10008, Part 2, Rev. 1.

Answer: Parts (a) (b) (c)

The fuel assembly is a very complex structure composed of many components; such as spacer grids, fuel cladding, etc. Due to this complexity, it is quite difficult to accurately calculate the stresses in these individual components. B&W has therefore established load or deflection limits for the individual components of the fuel assembly, rather than stress limits. A description of these limits is given in Section 2.4, BAW-10035. A description of these tests
21. conducted to establish these limits is given in Section 5, BAW-10035. The actual numerical limits are shown in Table 2 of BAW-10035.

3.9.1.2 Combined LOCA and MHE

Answer:

(a) Refer to answer to 3.9.1.1.

(b) The components which contribute to the stability of the guide tubes, due to their geometry, and the end spacer grid assembly.

(c) The allowance of 85% of the critical buckling load was chosen, based upon engineering judgment, so as not to design to failure. The Euler formula was used to calculate the critical buckling load.

3.9.2 ANSWER:

Figure 3 is a combination of figure 6 (Δp across core for a 36" outlet break) and figure 10 (shear force on core for a 36" ID outlet break) BAW-10008, Part 1. This is also discussed in section 3.1,
21. BAW-10035.

3.9.3 ANSWER:

The LOCA Thrust force acting at the vessel's outlet nozzle was analyzed using the FLASH computer code and the relationship

$$\text{Thrust} = \text{Pressure} \times \text{Area}$$

Testing associated with the LOFT program tends to confirm that the horizontal thrust can be calculated by the above relationship. The FLASH program has been used to correlate the vessel pressure and therefore the thrust for some of the semi-scale blowdown tests. This
21. is also discussed in revised section 3.2, BAW-10035.

3.9.4 ANSWER:

(a) Engineering sketches of the structural features of importance is given in Figures A-2 and A-3, Appendix
21. A, BAW-10035.

(b) A precise description of the location of and basis for the computation of masses and section properties/boundary conditions is given in sections 2.1 and 2.3, Appendix A,
21. BAW-10035.

(c) Details on the manner in which the flexibility coefficients have been computed and the results achieved are given in
21. section 2.3, Appendix A, BAW-10035.

3.9.5 ANSWER:

(a) A picture of the complete digitalized record used in the analysis
21. is given in figure A-5, BAW-10035.

(b) Refer to answer to 3.9.1.1.

(c) A general description of the manner of digital-to-analog conversions of data; an estimate of the accuracy of the process and a description of the input techniques are given in section 2.4, Appen-
21. dix A, BAW-10035.

(d) Complete acceleration response spectra comparisons at 1 and 10 per-
21. cent critical damping are given in figures A-6 and A-7 BAW-10035.

(e) The vertical component of the earthquake was considered in the analysis. However, due to the high stiffness of the reactor internals in the vertical direction, the vertical seismic contribution to the displacement of the core is negligible (on the order of .002" to .003") with respect to the horizontal motion.

3.9.6 ANSWER

The dynamic equations of motion of the reactor vessel, drive service structure, internals, and core were programmed for solution on an analog computer. Actual earthquake time histories and the LOCA thrust force time history were used as input to the analog. The response of the internals were then recorded and used as input to a more detailed fuel assembly model. This more detailed model solved for the deflections and loads on an individual assembly. Details are given in Appendix A,
21. BAW-10035.

3.9.7 ANSWER:

21. Sketches of the second segment model are presented in Figures A-8 and A-9 BAW-10035. The second segment model consists solely of fuel assemblies. To determine their response, the motion of the upper and lower grid plates must be known. The response of the grid plates is obtained from the first segment model, which also contains the mass of the entire core to provide its influence on these motions.
21. This is also discussed in Section 2.7, Appendix A, BAW-10035.

3.9.8 ANSWER:

The spring constant variation for the fuel assembly in relation to its location within the core for that part of the load deflection curve which occurs after the gap is closed is presented in Figure 7,
21. BAW-10035.

3.9.9 ANSWER:

(a) By using the Oconee response spectra displacements vs. frequency, an estimate of the maximum amplitude for a fuel assembly was established. Tests were performed over a large range of amplitude (0" - .300") and a plot of frequency vs. amplitude was obtained. Fuel assembly frequencies were not established, but were part of the data obtained as a result of the tests.

(b) Both pluck (initial displacement tests and excitation tests were performed. The fuel assemblies were supported both top and bottom in tests fixtures which were made to duplicate the restraint provided by the upper and lower grid plates, respectively. Further
21. details are provided in Appendix B, BAW-10035.

(c) Only full-sized specimens were tested. Actual production materials were used, with the exception of the fuel, for which brass pellets were substituted to represent actual fuel weight. A list of materials
21. is given in Table 1, BAW-10035.

(d) (e) The description of the test data obtained and the analysis and interpretation of the results are discussed in Appendix B,
21. BAW-10035.

3.9.10 ANSWER:

A sketch showing the test specimen is presented in figure 10, BAW-10035. An explanation of how corrections were made for tempera-
21. ture effects is given in section 5.2 BAW-10035.

3.9.11 ANSWER: (Refer to answer to 3.9.1.1.)

LOCA and seismic loads rather than stresses are combined. For the horizontal analysis, the LOCA time-history force is applied to the reactor system in conjunction with the earthquake time history. The start of the LOCA relative to the start of the earthquake is such that the maximum load on the fuel assembly spacer grid is obtained.

3.9.12 ANSWER:

In the vertical contact analysis the affects of vertical and horizontal seismic excitation was considered. For the following reasons, however, their contributions were determined to be negligible.

1. Because of the high vertical stiffness of the vessel and internals, the core will experience essentially unamplified ground-motion in the vertical direction. The resulting forces in a fuel assembly will be less than 2% of those imposed by LOCA alone.
2. Horizontal seismic excitation of the fuel assembly will cause lateral displacement of the assembly. This was considered in reducing the allowable Euler buckling load of the guide tubes since the guide tubes would have some initial curvature due to horizontal assembly displacement. It was found that this small curvature of the span between spacer grids reduced the allowable buckling load of an assumed straight tube less than 2%.

In light of the above comments, the margin of safety quoted in Paragraph 6.2.1.2 of BAW-10035 reflects the effects of combined LOCA and seismic
21. loading; and seismic loads, although negligible, were considered in the vertical contact analysis.

3.9.13 ANSWER

21. For the upper end spacer grid, all major loading is caused by the axial motion of the fuel rods relative to the grid. The direction of this loading is normal to the plane of the grid as shown in Figure 9, BAW-10035. Since each fuel rod is restrained in the axial direction by a frictional force exerted by the end spacer grid, the maximum load that any rod can exert on the grid is limited to this frictional force. This is true regardless of the source of loading whether it be from normal operation, LOCA and/or earthquake. For this reason, the above conclusion regarding the addition of loads due to normal operation, LOCA and/or earthquake is considered valid.

3.10 Control Rod Drive System

21. 3.10.1 thru 3.10.6 are answered in Appendix A of BAW-10029 with questions listed as 1 through 6.

11.8 FSAR Supplement 2, Page 7 (dated 5/25/70)

14.3 Steam-Line-Rupture Accident

14.3.1 FSAR Supplement 2, Page 7 (dated 5/25/70)

14.3.2 FSAR Supplement 2, Page 7 (dated 5/25/70)

14.3.3 ANSWER:

The steam-line-rupture accident has been analyzed assuming no integrated control system action. The list below contains the major assumptions which were made for the analysis.

a. The plant was initially operating at rated power.

b. Following the rupture no feedwater valves were repositioned in any way, although normal control system action would close the startup and main feedwater valves because of reactor trip.

c. No operator action occurs, although it would be apparent to the operators present that a break had occurred in a particular steam line because of the rapid (and persistent) pressure decrease in that line. The operators would close the feedwater valves.

d. The maximum negative moderator coefficient corresponding to end-of-life conditions of the equilibrium cycle was used, although this is a value occurring only at time periods shortly before refueling.

e. The minimum tripped rod worth was used corresponding to the technical specification limit for the minimum shutdown margin; this was a very conservative approach.

f. The maximum-worth rod was assumed to stick out. This assumption, together with b through e above, results in a combination of events so improbable that it represents an unrealistic approach.

g. The increased capacity of the low pressure and high pressure injection systems with decreased reactor coolant system pressure has been neglected; design flow rates have been used throughout the transient. Although two (2) HP and two (2) LP pumps receive actuation signals, credit is taken for only one HP and one LP pump operating.

h. The cooldown rate of the reactor coolant system was assumed to be independent of any core power or any stored heat in the reactor coolant system components, although these would significantly improve the results of the analysis.

i. The boron injection is assumed to be perfectly mixed with all the reactor coolant before entering the core, although the injection occurs at the reactor vessel inlet and so would have the highest concentration in the core region.

j. Perfect heat transfer is assumed in the affected steam generator after the initial part of the transient; that is, the time constant for heat transfer is zero with no stored energy accounted for.

The steam line rupture causes an increase in the heat transfer from the reactor coolant to the feedwater. This initiates a cooldown of the reactor coolant system which in turn starts a pressure decrease in the system. The reactor quickly trips on low pressure (about 6 sec). Although the control system would try to maintain T_{avg} by withdrawing control rods, this system has been assumed not to function. By the time the reduction in T_{avg} is sensed, the reactor is tripped and the control rods would not have been withdrawn. A large negative moderator coefficient will also cause an increase in power, but the reactor trips on low pressure before the high flux trip is reached.

Reactor trip initiates turbine stop valve closure, isolating the unaffected steam generator on the steam side. This isolated loop continues to supply steam to drive the associated main feedwater pump. For the cooldown part of the calculation, it is assumed that all the feedwater from this pump goes to the affected steam generator; the equilibrium flow rate is about 135% of the rated flow in one steam generator. With the above assumptions, the resulting reactor coolant system temperature decrease causes initiation of high pressure injection at 30 sec and the core flooding tanks initiate injection in just over one minute following the rupture. These injections will keep the core subcritical during the ensuing cooldown until just before initiation of low pressure injection (about 100 sec). The neutron power peaks at less than 8% of rated power before going subcritical again, at 166 sec after the break; subcriticality is caused by injection of borated water from the low pressure injection system.

The reactor trips shortly after steam line failure and essentially remains shut down for the ensuing cooldown of the reactor coolant system. The peak power generated in the core will not cause any thermal limits to be approached and therefore the reactor damage criterion (Page 14-17 of the FSAR) is met. If credit were taken for operation of the Integrated Control System, the consequences of this accident could be further reduced.

14.3.4 ANSWER:

The steam line rupture accident has been analyzed using a proprietary hybrid computer program, SECRUP. This program solves time-dependent differential equations for the steam generator and the reactor coolant system. No numerical approximations have been made because the analog computer solves all differential equations; the digital computer is used for logic control and tabular information. The general layout of the program is shown in Figure 14.3.4-1. (attached)

The ordinary kinetics equations with six groups of delayed neutrons is solved in the kinetics model. Reactivity can be input due to moderator and fuel temperature feedback and any external source such as injection of borated water or control rod movement. These equations are:

$$\frac{dN}{dt} = \frac{\rho-\beta}{\Lambda} N + \sum_{i=1}^{6} \lambda_i C_i$$

$$\frac{dC_i}{dt} = \frac{\beta i}{\Lambda} N - \lambda_i C_i$$

Where

N = neutron power

t = time

ρ = reactivity

β = total delayed neutron fraction

Λ = neutron generation time

λ_i = decay constant for i^{th} delayed neutron group

C_i concentration of i^{th} delayed neutron group percursor

This model also contains a curve of decay heat versus time following shutdown; this curve can be scaled to reflect any known previous operating history.

The output from the kinetics routine is power (neutron power or/and decay heat), which is fed to a lumped parameter model of the fuel pin. This simulation contains nodes for the average fuel, gap, clad, and moderator temperatures. The thermal power, which is the instantaneous heat flux from the clad to the moderator, is also calculated. The equations used for the j^{th} node are the normal heat balance equations:

$$(MC_p)_j \frac{dT_j}{dt} = f_j Q + (UA)_{j-1} (T_{j-1}-T_j) - (UA)_j (T_j-T_{j+1})$$

where $(MC_p)_j$ = mass times specific heat of the j^{th} node.

T_j = average temperature of j^{th} node

f_j = fraction of the total heat being generated (Q) which goes to the j^{th} node

$(UA)_j$ = overall heat transfer coefficient times the effective surface area between the j and j+1 nodes.

The change in fuel and moderator temperatures are fed to the reactivity model for calculation of the temperature feedbacks. This model also includes a detailed simulation of the trip logic. Reactor trip can be initiated on high flux or high or low reactor coolant system pressure. Each of these three trip modes has a given trip delay time. Following this delay, there is a simulation of rod insertion as a function of time and a simulation of reactivity versus insertion. These reactivity feedbacks and control rod motions are then added to any other reactivity effect such as a change in boric acid concentration to obtain the net reactivity for the kinetics equation.

The core outlet moderator temperature from the fuel pin model is fed to a loop model which simulates all of the transport delays from the core to the pressurizer, to the steam generator, to the pumps, and back to the core. This part of the model also calculates the average reactor coolant temperature in the several parts of the hot and cold legs and determines the net reactor coolant system expansion and contraction based on water properties fed to it from the steam table tabulation. This net change in system specific volume results in a surge flow which is used in the pressurizer model.

The pressurizer model includes a simulation at the junction of the surge line and hot leg to account for mass and energy transfer. There are two distinct models for the pressurizer to account for pressure increase and pressure decrease. The pressure increase model is dependent on the compression of the steam space and the appropriate steam properties. The pressure decrease model depends on the total heat content in the pressurizer and the water and steam properties from the tables. No valves are simulated in the model because no valve set points are exceeded in any transient. The heaters and sprays are not simulated because they are control functions and would also be too slow to have a significant transient effect.

The feedwater system part of the model is used to simulate the properties of the feedwater entering the steam generator. This includes the valve arrangements for normal and emergency feed flow, the closing times for each valve, the options for automatic or operator action (or neither), and the enthalpy of the feedwater. This model consists entirely of input curves, tables, and logic circuits.

The feedwater flow and enthalpy are then fed to the inlet region of the steam generator where the downcomer region and the aspirator flows are simulated. This part of the model is an integral part of the main steam generator model where two variable length regions are simulated in each of the three physical parts of the steam generator: reactor coolant, steam generator tubes, and secondary. The two variable length regions in each part correspond to the boiling and superheat regions on the secondary side. The equations for the conservation of mass, energy, and momentum are solved in each steam generator region; these equations are, respectively:

$$\oint_{cs} \rho(\bar{V}\cdot\bar{n})\ dA + \frac{\partial}{\partial t}\int_{cv} \rho dv = 0$$

$$\oint_{cs} \rho h\ \bar{V}\cdot\bar{n} dA + \frac{\partial}{\partial t}\int_{cv} \rho u dv = Q - p\frac{dv}{dt}$$

$$\Sigma F_s + \int_{cv} \frac{g}{g_c}\rho dv = \frac{1}{g_c}\oint_{cs} \rho(\bar{V}\cdot\bar{n})\ \bar{V} dA + \frac{1}{g_c}\frac{\partial}{\partial t}\int_{cv} \rho\bar{V} dv$$

Where

ρ = density in control volume

$\bar{V}$ = fluid velocity

v = control volume

h = enthalpy in control volume

Q = total change in heat content of the control volume

p = pressure in control volume

F_s = forces acting on the control volume surface

The solution of the continuity equation can be illustrated by considering the superheat region on the secondary side. Five terms emerge representing the aspirator and steam flows leaving, the incoming steam flow, and the change in length of both regions:

$$W_a + W_{so} - W_{bo} + \rho_{bo} A\frac{dL_B}{dt} + A\frac{\partial}{\partial t}\int_{L_B}^{L_T} \rho dx = 0$$

Where L_B and L_T are the boiling length and total length. Using an average density in the superheat region (ρ_{si}), we can rewrite this equation to read:

$$W_a + W_{so} - W_{bo} + \rho_{bo} A\frac{dL_B}{dt} + A\frac{d\rho_{si}}{dt}(L_T - L_B) = 0$$

This equation is solved for W_{so}, the exit steam flow, on the analog computer.

Similarly, the "superheat region" on the reactor coolant side can be considered in order to demonstrate a typical solution for the conservation of energy equation. The basic equation for this region is

$$h_{po}W'_p - h_{ph}W_p + \rho A\frac{\partial}{\partial t}\int_{L_B}^{L_T} u_p dx = pA\frac{dL_B}{dt} - UP(L_T - L_B)(T_{pi} - T_{mi})$$

The first two terms are for the exiting and entering coolant respectively; the next two terms account for the change in region lengths; and the last term is the change in total heat content, where

U = overall heat transfer coefficient from the reactor coolant to the tubes

P = the effective tube perimeter

T_{pi} = average reactor coolant temperature in superheat region

T_{mi} = average tube temperature in superheat region

This expression can be simplified somewhat by making five substitutions:

(1) The flow leaving is equal to the incoming flow plus an effect due to the length change:

$$W'_p = W_p + \rho A \frac{dL_B}{dt}$$

(2) The average reactor coolant system temperature in the "superheat region" is a function of the inlet and outlet temperatures of that region:

$$T_{pi} = \frac{T_{ph} + T_{po}}{2}$$

(3) The rate of change in the enthalpy with respect to time is the same as the rate of change of the internal energy:

$$\frac{\partial h_p}{\partial t} = \frac{\partial u_p}{\partial t}$$

(4) The enthalpy difference between inlet and outlet in this region is equal to the temperature difference times an appropriate average specific heat:

$$(h_{po} - h_{ph}) = (C_p)_p (T_{po} - T_{ph})$$

(5) The enthalpy is a function of the internal energy, the pressure, and the specific volume:

$$h = u + pv$$

Where these expressions are used in the energy balance and the resulting equation solved for the outlet temperature change with time, the final analog expression can be written:

$$\frac{dT_{po}}{dt} = \frac{UP\left(T_{mi} - \frac{T_{ph} + T_{po}}{2}\right)}{\rho A (C_p)_p} + \frac{W_p (T_{ph} - T_{po})}{\rho A \, (L_T - L_B)}$$

Returning to the superheat region on the secondary, the expression for the conservation of momentum can be written:

$$(P_{bo} - P_{so})A - \frac{f(L_T - L_B)W_{so}^2}{2g_c \, DA \, \rho_{si}} + \frac{g}{g_c} \rho_{si} A(L_T - L_B) = \frac{1}{g_c A}\left(\frac{W_{so}^2}{\rho_{so}} + \frac{W_a^2}{\rho_a} - \frac{W_{bo}^2}{\rho_{bo}}\right) + \frac{(L_T - L_B)}{g_c} \frac{dW_{si}}{dt} - \frac{W_{bo}}{g_c} \frac{dL_a}{dt}$$

This equation is programmed on the analog to solve for P_{bo}.

These three equations are solved in a similar fashion for all seven regions in the steam generator. The loop model supplies the inlet reactor coolant temperature for the steam generator model, which in turn computes the outlet reactor coolant temperature for the loop model. Steam tables are used to provide appropriate steam and water properties for the several equations for the fluids.

The exiting steam flow goes to two more models (steam system or rupture) depending on the type of transient being run and depending on whether it is the good or affected steam generator. Normally the flow will go to the steam system which consists of a set of bypass and relief valves with pressure set points and operating times. Following an assumed steam line break, flow can go from the good steam generator to the break until this steam generator is isolated on the steam side. The affected steam generator continues to supply steam to the rupture until it is isolated on the feedwater side. Nearly all of this simulation is logic rather than equations except for the break itself which assumes critical flow of saturated steam.

This is an extremely complex model and therefore requires a good deal of input. Only the principal input parameters are discussed here. The neutron properties are input using the actual calculated values based on EOL conditions for the equilibrium cycle. The maximum negative moderator temperature coefficient and the minimum Doppler coefficient are used for the feedback calculation. The minimum tripped rod worth with a stuck rod is input. All trip delay times are greatly larger than those specified for the equipment. The trip set points are set to account for all sensor and instrumentation errors.

Only the minimum fuel pin initial energy content is assumed and no other source of stored energy in the reactor system (such as the reactor vessel and the reactor coolant system piping) is accounted for. Pressurizer level can be varied by initializing the steam and water volumes at different values. Most runs are done with normal level because the delayed depressurization effect of a high level is essentially balanced by the increased energy content of the system as it enters the reactor coolant system. The initial reactivity transient is dependent on the gross heat balance of the system and therefore the initial level is relatively insignificant.

The valve arrangement is input as logic for both the feedwater and steam systems. The capacities and closing (or opening) times for these valves is also input. The feedwater enthalpy is read from a function generator as a function of elapsed time. The rupture size must be specified.

The steam generator model itself requires considerable input such as heights, number of tubes, fluid inventories in the several regions, thermal properties, and the initial conditions on the pressures, temperatures, and flows for each region.

Values of some of the input parameters which are not given in answer to Question 14.3.5 or in the 1967 ASME Steam Tables are listed in Table 14.3.4-1.

Typical output is also given in the answer to Question 14.3.5 and includes (as a function of time): reactivity, average moderator temperature, reactor coolant system outlet temperature from the steam generator, downcomer height, boiling height, both flow and pressure at the rupture and entering the boiling region , and the flows entering the steam generator and entering the superheat region. Although not included in Question 14.3.5, neutron and thermal power are also typical output parameters.

The analog computer on which these differential equations were modeled is typical of current-generation computers with amplifiers that have a maximum gain error of 0.025%. The digital computer part of the model is used only for logic and tabulation of data. Therefore no numerical approximations are used, nor is the concept of time steps employed. The accuracy of the results, then, is only a function of the equations and the input and not the modeling or the computer facility.

The simulation of the once-through steam generator also served to clarify some important differences between this type of generator and a typical U-tube boiler. For a nuclear steam system rated at about 2568 MW, the OTSG has an inventory of 55,000 pounds on the secondary side with an associated reactor coolant system inventory of 514,000 pounds. The corresponding values using a saturation boiler are approximately 84,500 and 407,000 pounds. The values at the hot shutdown conditions are even more disparate between the two boilers, with the OTSG having an inventory of 20,000 pounds and the U-tube boiler 160,000 pounds.

Table 14.3.4-1

Parameter	Value
Core heat transfer coefficient (clad to moderator)	2600 Btu/hr-ft^2-F
Steam generator heat transfer coefficient	
a. Reactor coolant side	No film resistance
b. Secondary - nucleate boiling region (initial only)	See separate submittal
c. Secondary - superheat region	See separate submittal
Discharge coefficient	1.0
Effective delayed neutron fraction	0.0053
Doppler coefficient	$-1.2 \times 10^{-5} (\Delta k/k)/F$
Average fuel temperature	1400 F
Upper limit for high flux trip	114%
Main feedwater valve closing time: actual:assumed	7:10 sec.
Startup valve capacity (% of rated flow for one steam generator): actual:assumed	15:35%

Question 14.3.4
DRL Letter of 3/3/70

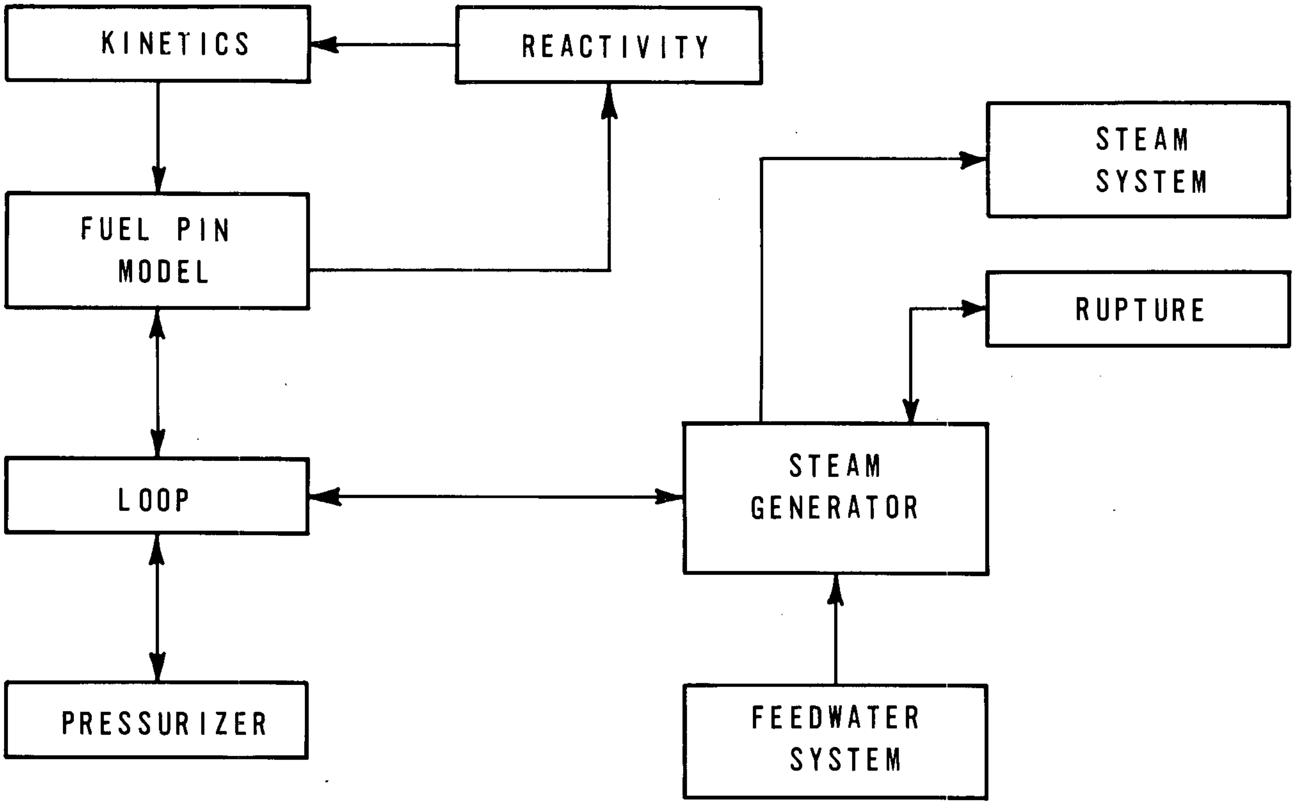


Figure 14.3.4-1
Question 14.3.4 DRL Letter of 3/3/70
FSAR Supplement 3 - 19

14.3.5 ANSWER:

(a) The worst case steam-line-rupture accident is based on the plant initially operating at rated power with the maximum end-of-life moderator coefficient. The minimum shutdown margin with the maximum worth stuck rod, full operation of the integrated control system, and no operator action are also assumed.

The rupture analysis includes the following input quantities and conservatisms:

1. The fluid inventory of the reactor coolant system was taken as 537,000 lb. The secondary side of each steam generator was calculated to have 55,000 lb. of water inventory at rated power. The fluid masses were calculated from actual design data for the reactor coolant and OTSG systems in the clean condition.

2. A 0.3 sec. time delay in the start of control rod drop was assumed for reactor trip after which 2/3 insertion of the control rods would take 1.4 seconds. The delay in the rod drop is about 25% greater than the sum of estimated electrical and mechanical delays. The time for 2/3 insertion of the control rods is greater than the actual drop time which has been realized after completion of life-testing at the rated power reactor coolant temperature and pressure.

3. No control instrumentation time delays were assumed.

4. The tripped rod worth provided the negative reactivity insertion. The minimum tripped rod worth was used corresponding to the technical specification limit for the minimum shutdown margin. This was a conservative approach. The maximum-worth rod was assumed to be stuck out which added further conservatism to the tripped rod worth.

5. A reactor coolant flow rate of 131.32×10^6 lb/hr was used. This rate is based on operation at rated power and normal design capacity operation of the four reactor coolant pumps. A secondary flow rate of 5.6×10^6 lb/hr per steam generator was calculated for steam flow at rated power.

6. Heat transfer coefficients have been calculated to produce a conservatively high rate of cooldown of the reactor coolant system from the affected steam generator. The film coefficient on the reactor coolant side and in the boiling region on the secondary side are kept constant during the transient when actually it will decrease with cooldown of the reactor coolant system. The heat transfer coefficient in the superheat region has no effect on accident results.

(b) The steam line rupture causes a sudden decrease in steam pressure which results in the increase of heat transfer from the reactor coolant to the steam generator feedwater. The large negative moderator temperature coefficient causes the reactor power to increase with the cooldown of the reactor coolant system such that the reactor trips on high flux or low coolant pressure in about 7 seconds.

When the reactor trips the turbine stop valves close, isolating the unaffected steam generator on the steam side. The reactor trip also results in the closing of all feedwater valves (main control, startup, and isolation valves) to both steam generators by the integrated control system. Feedwater flow ceases 10 seconds after reactor trip.

With the turbine stop valves closed, the pressure increases in the unaffected steam generator. Nine seconds after isolation, this steam generator has regained steam pressure to levels which require the turbine bypass valve to open. The turbine bypass valve remains open for about 6 sec. until the reactor coolant temperature goes below 550F. The isolated loop continues to supply steam to drive its associated main feedwater pump. Feedwater addition to the unaffected generator is made through the startup and main control valves to maintain a 2-foot minimum downcomer fluid level during cooldown if it is needed.

The steam generator with the rupture continues to blow down since it cannot be isolated. Approximately 50 seconds after the rupture, the water level in the downcomer is 2 feet.

The feedwater control system operates the feedwater startup and main control valves to maintain a 2-foot minimum level. Since the affected steam generator is unable to supply steam for support of its main feedwater pump, it too depends on the feedwater supply for the unaffected steam generator. The operation of the one main feedwater pump causes the emergency feedwater system to remain inactive throughout the transient.

The cooldown of the reactor coolant by the ruptured steam generator will actuate the high pressure injection system in about 23 seconds after the rupture and the core flooding tanks are actuated in just over 75 seconds following the rupture. These injections will keep the core subcritical during the ensuing cooldown until returning to about 1% rated power just before the low pressure injection system is actuated (about 170 sec). The core will then return subcritical and stay there through the remainder of the cooldown.

(c) The following process variables are considered the most important quantities as a function of time after a single steam line rupture and are presented in Figures 14.3.5-1 through 14.3.5-3. (attached)

1. Steam flow, pressure, and temperature for both the affected and normal steam generators.

2. The boiling and downcomer lengths for the affected steam generator.

3. The liquid and vapor mass inventories for the affected steam generator.

4. Reactor coolant system pressure.

5. Pressurizer level.

6. Reactor coolant system temperature.

7. Reactivity.

Other quantities are either limited in the analog model for analysis of the steam line breaks or are inconsequential as a function of time after rupture. These quantities include:

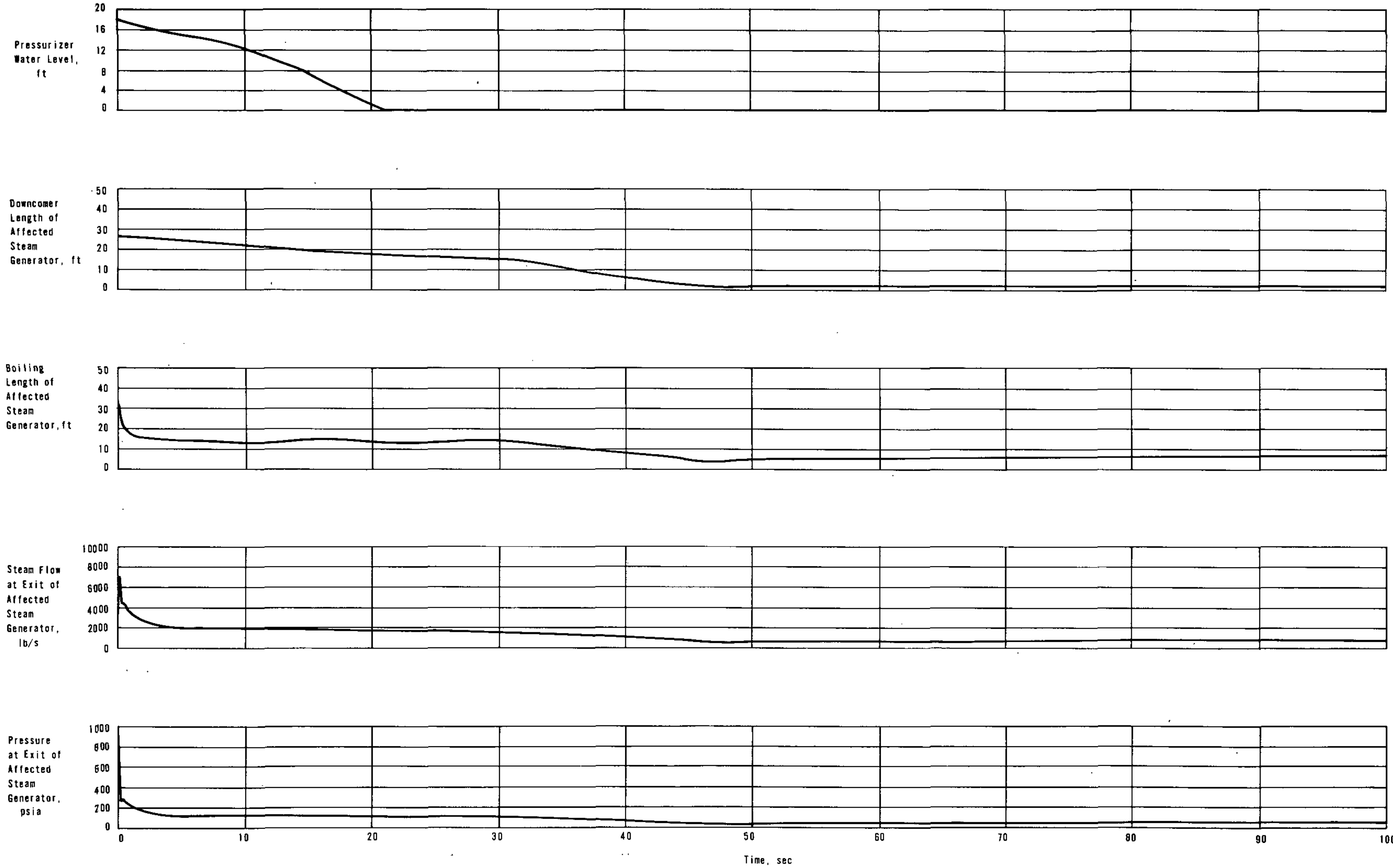
1. Feedwater pressure and temperature, which are input as constants of 950 psia and 460F for each steam generator. The feedwater temperature will drop to about 90F several minutes after the steam line rupture.

2. Liquid levels and liquid and vapor mass inventories for the unaffected steam generator which are not a part of the analog model output but are controlled in the program in maintaining a 2-foot minimum downcomer level.

3. A peak tensile stress of 13,770 psi for a maximum tube-to-shell temperature difference (the shell temperature being higher) of 50F and a maximum pressure difference of 2200 psi. This tensile stress is less than half of the code allowable of 42,000 psi at approximately 600F and represents a conservative estimate for this analysis since the calculated maximum temperature difference is 35F.

4. The reactor building pressure will not exceed the design value if a steam line failure resulting in the blowdown on one steam generator occurs within the reactor building. Within 30 seconds following the steam line rupture, the pressure rise in the building will initiate the 3 building fan coolers. These fans start the removal of heat from the building. As backup to the operation of these fans, the reactor building spray system will operate to provide an equivalent and separate cooling capacity. Approximately 50 seconds after the rupture, the water level in the affected steam generator reaches the 2 ft. control minimum

level. This level will be maintained until the reactor operator isolates the feedwater supply. With feedwater continuing, the affected steam generator continues to blow down, resulting in the lowering of the reactor coolant system temperature as it releases both decay and sensible heat to the reactor building. In about 250 seconds after the break, the reactor building has pressurized to approximately 38 psig. This amounts to a back pressure on the ruptured system which limits the temperature in the reactor coolant system to a minimum of 284F. Thus, the sensible heat flow from the reactor coolant system is stopped and beyond this time only decay heat can be released. The decay heat raises the reactor coolant system temperature so that the back pressure effect on the rupture is overcome. The mass and energy released from this time until the time when the cooling rate exceeds the decay heat generation rate is less than that required to reach the design building pressure. The heat removal capacity of the building coolers (240×10^6 Btu/hr.) equals the decay heat generation rate at 6 min. following the rupture. Thus, cooling capacity is available to provide adequate building integrity following a steam line failure within the reactor building.

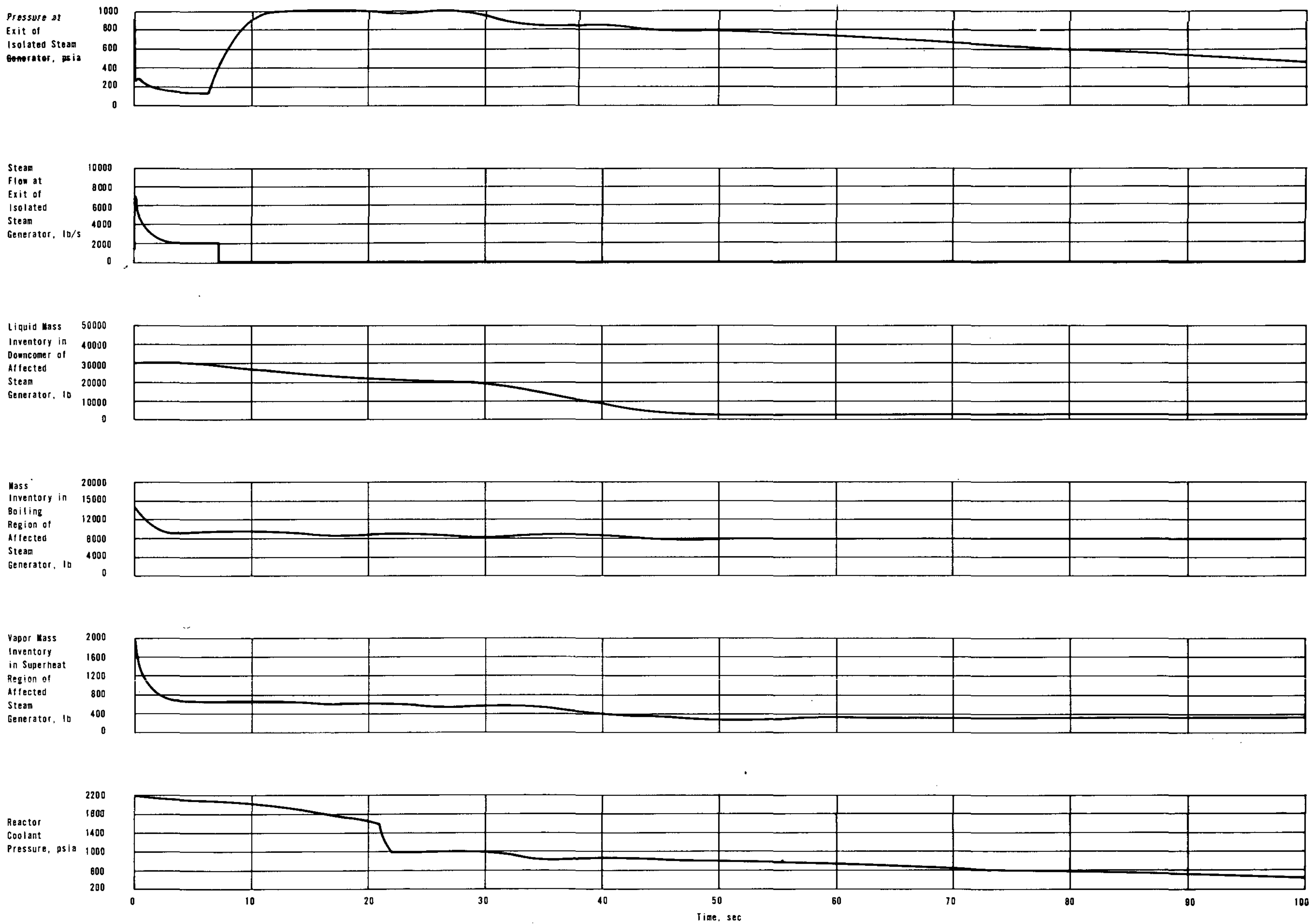


WORST CASE STEAM LINE RUPTURE

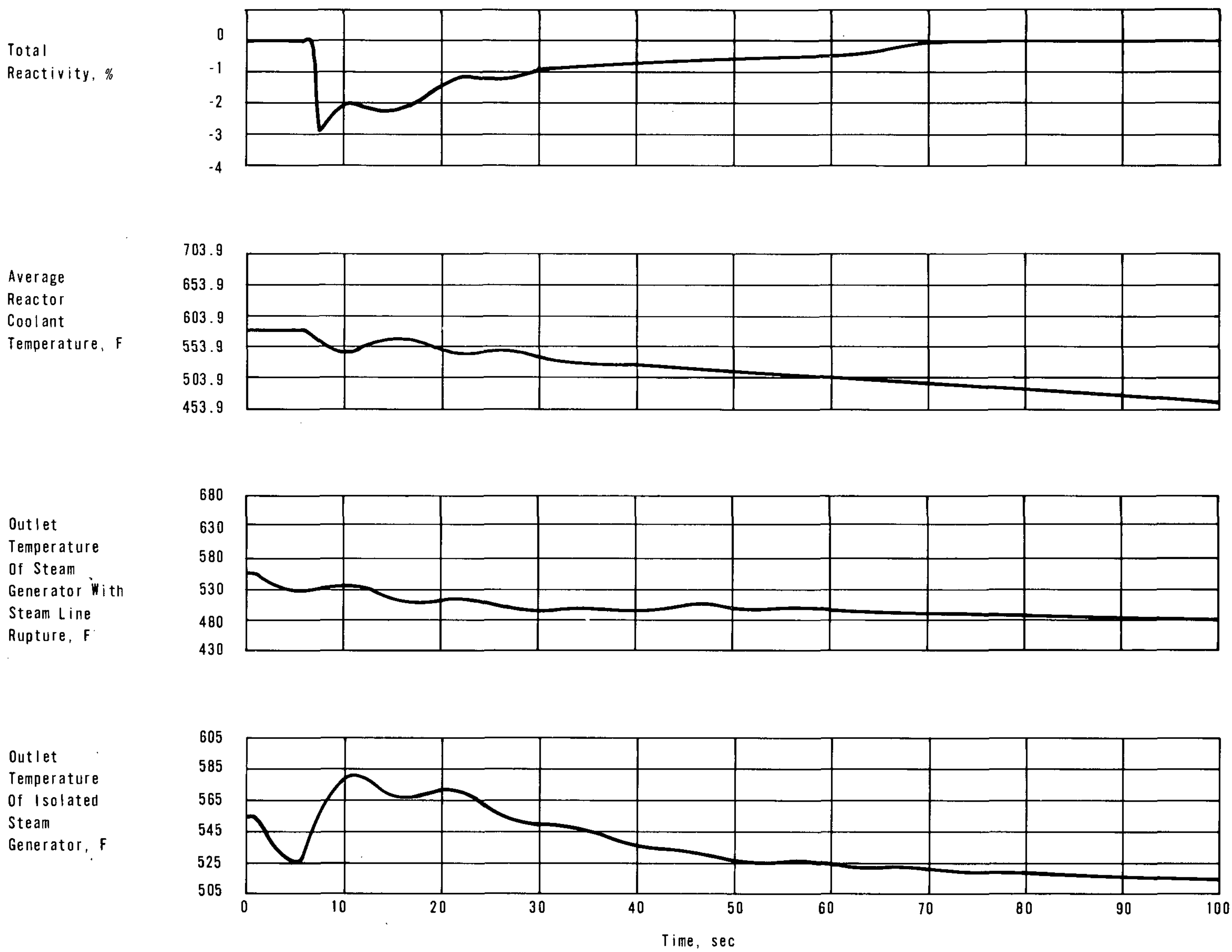
Figure 14.3.5-1

Question 14.3.5 DRL Letter of 3/3/70

FSAR Supplement 3 - 24



WORST CASE STEAM LINE RUPTURE
Figure 14.3.5-2
Question 14.3.5 DRL Letter of 3/3/70
FSAR Supplement 3 - 25



WORST CASE STEAM LINE RUPTURE
Figure 14.3.5 - 3
Question 14.3.5 DRL Letter of 3/3/70
FSAR Supplement 3 - 26

14.3.6 ANSWER:

The rate of heat removal from the reactor coolant following a steam line break accident is determined by the steam generator fluid inventory available for cooling. Since the steam generator inventory increases with power level, the larger inventory (55,000 lb) occurs at rated power. Thus, greater mass is available for cooling at 100% reactor power. The steam generator inventory will decrease linearly to 20,000 lb at 15% of rated power; below which the inventory remains constant.

Therefore, steam line breaks occurring at hot shutdown or at less than 100% rated power, by the very nature of steam generator operation, will result in less initial heat removal from the reactor coolant. The extended cooldown period will then lessen the adverse consequences of the steam line rupture.

A similar conclusion can be made for loss of offsite power concurrent with the steam line rupture. In this case, the electric motor driven hotwell and booster pumps in the feedwater system are lost. This will lead to cavitation of the main feedwater pumps and leave only the auxiliary feedwater system in operation. The reduced feedwater addition results in a low steam generator inventory and thus retards heat removal from the reactor coolant.

14.3.7 ANSWER:

The analysis of the worst case steam-line rupture accident as presented in Question 14.3.3 was extended to ten minutes following the rupture. At this time the reactor coolant had cooled below 250F which permits the operator to select two courses of action. He can isolate the feedwater flow to both steam generators immediately. This will conserve the remaining feedwater supply from the hotwell and upper storage tanks and allow the coolant temperature to rise back to 250F where heat removal by the unaffected steam generator will be more effective. The operator can then hold at 250F while the plant conditions are assessed and use manual feedwater control to maintain the liquid inventory in the unaffected steam generator. Additional feedwater is available to the operator, if needed, from the condensate storage tank.

The operator could also elect to isolate the feedwater supply to both steam generators and complete the cooldown using the decay heat cooling system. If the operator fails to isolate the feedwater supply, it will be drained in approximately 13 minutes after the steam line rupture.

14.3.8 ANSWER:

Based upon the results of testing and analysis done by B&W, steam generator tubes will not rupture concurrently with rupture of the main steam line as a result of blowdown loadings. Data taken from a steam-line-rupture test on a 37-tube model OTSG show that during the transient, the tubes become 100°F cooler than the shell. Although this temperature difference is greater than that reported in the answer to 14.3.5, this seeming inconsistency is explainable; and, in any case, the higher tube to shell Δt does not result in exceeding allowable stress levels.

The steam line rupture test had no feedwater flow after approximately 80 seconds even though the worst case accident assumed continuous feedwater flow controlling on minimum level. The lack of feedwater flow causes the steam pressure to drop rapidly with an associated drop in the saturation temperature. This rapid drop in steam temperature causes the 100°F tube to shell temperature difference.

The calculations used to arrive at the tube stress intensity for an assumed worst case degraded tube are attached. The pressure and temperature conditions are as follows:

Primary Pressure - 2185 psi
Secondary Pressure - 0 psi
Mean Shell Temperature - 553°F
Mean Tube Temperature - 453°F
Power Level - 100% prior to rupture.

The shell temperature is not assumed to change as rapidly as tube temperature during the steam line rupture transient.

The pressure and temperature conditions produce axial tensile tube stresses as follows:

A)	Design size tube (no degradation)	tube bundle @ edge	15.2 KSI
		tube bundle @ center	7.6 KSI
B)	Degraded tube with assumed 1/2 original wall thickness		
		tube bundle @ edge	31.5 KSI
		tube bundle @ center	15.3 KSI

For the assumed worst case (B) the maximum stress instensity is calculated as follows:

Stress Intensities

For Edge Tube (Worst Case)

$\sigma_L = 31.5$ KSI

$\sigma_R = -2.2$ KSI

$\sigma_C = 37.0$ KSI

$$\sigma_C = \frac{2185\ (.574)}{0.017\ (2)} = 37.0 \text{ KSI}$$

Mean Dia. = 0.557 + 0.017 = .574

$$\sigma_C = \frac{P\ r_M}{t}$$

$$r_M = \left(\frac{.574}{2}\right)$$

$\sigma_C - {}_R = 39.2$ KSI

$\sigma_L - {}_R = 33.7$ KSI

$\sigma_L - {}_C = 5.5$ KSI

As shown above, the circumferential-radial stress combination yielded the maximum stress intensity which was with the allowable stress limits as calculated below:

$$\sigma \text{ circumferential} - \sigma \text{ radial} = 39.2 \text{ KSI} < 1.5\ (1.2)\ S_m$$

$$39.2 \text{ KSI} < 42.0 \text{ KSI}$$

The degraded tube primary plus secondary stress intensity is less than the Oconee Nuclear Station FSAR Case III stress limit for primary stresses resulting from design loads plus pipe rupture loads.

The axial stress in the presumed degraded tube is at the yield point. However, the nature of thermal restraint stresses limits the amount of tube deformation during the steam line rupture transient.

The assumption that the tube wall thickness is reduced to one-half the original value is used to demonstrate that the chance of rupture is slight even when extreme effects of erosion, corrosion, vibration or leakage are considered. Results of tests conducted by B&W show that the tubes are not degraded to such an extent by those effects.

The steam line blowdown loads have been analyzed empirically and by simulation, and the results indicate these loads will not cause tube failure.

14.3.9 ANSWER:

The steam generator level indication system is a part of the Integrated Control System (ICS) and is not required to function during a steam line rupture accident. Failure of the ICS does not diminish the safety of the reactor. None of the functions provided by the ICS are required for reactor protection or for actuation of the ESPS. The reactor protection criteria, used in the analysis of the steam line rupture accident presented in Section 14 of the Oconee Nuclear Station FSAR can be met irrespective of ICS action.

14.4 PRESSURIZER LEVEL

14.4.1 ANSWER

The redundancy of pressurizer level transmitters in combination with independent computer alarms and indication effectively provide the reliability although without the complete separation which is the intent of IEEE-279. The letdown storage tank level indication represents further redundant information available to the operator to indicate abnormal pressurizer level. A sketch of the pressurizer level indicating and alarm system is attached.

Calculations have been performed for a range of rod withdrawal rates extending from very slow rates up to greater than $4 \times 10^{-4}\ \Delta k/k/s$, a value more than four times as great as the maximum rod withdrawal rate for one rod group. These calculations show that the pressurizer will not fill due to insurge from the primary system because the high-pressure trip will terminate the transient well before all the steam in the pressurizer has been expelled.

The normal level of the pressurizer is 18.33 feet of water at normal operating conditions. This value is used in the accident analysis.

The secondary system is assumed to remove no extra heat from the primary system during a start-up accident or a rod withdrawal from power. Thus, no secondary relief occurs. This very conservative assumption assures that the calculated insurge to the pressurizer is the maximum possible reactor coolant system volume expansion for the transient investigated.

In the unlikely event that either a start-up accident or a rod withdrawal from power were to occur with the pressurizer water at an abnormally high level, the high-pressure trip would occur much faster than normal. Only a very samll amount of energy would be required to raise the system pressure to the trip point under these circumstances. The pressure relief valves, should they be required to do so, are capable of relieving approximately 1,080,00 lb/h of water at 2500 psia.

A pressurizer level trip is not required. It is not justified to assume that either a rod withdrawal accident from power or a start-up accident should be compounded by a full pressurizer. To provide the basis for this belief, it is necessary to discuss the two accidents separately because of the different circumstances related to each accident.

The pressurizer contains a total volume of 1500 cu ft, its inside diameter is 7 ft and the distance between its upper and lower level sensing nozzles is approximately 35 ft. The pressurizer normally will contain 800 cu ft of water and 700 cu ft of steam. The volume in the upper portion of the pressurizer (even when the level is at the high level alarm point) is 400 cu ft. If the pressurizer were full to the point where the upper level indicating nozzle were covered, there would still be approximately 150 cu ft of steam in the pressurizer.

During normal operation, the pressurizer level is automatically maintained at approximately 18 ft. It is not considered credible that the pressurizer could become full during power operation and then be followed by a rod withdrawal accident. This is based upon the fact that several malfunctions and operator errors must occur to achieve a full pressurizer.

If the normal level control system fails such that full makeup is provided through the "2" flow control valve, the flow rate would be limited to approximately 140 gpm by the valve characteristics. With only one high-pressure injection pump running against a head of 2050 psia, the total available flow is 320 gpm of which approximately 180 gpm is required for the primary pump seal system - thus leaving a remainder of 152 gpm gross makeup including 12 gpm for pump in-seal leakage. Normally, there would be some letdown so that the net makeup would be considerably less than 152 gpm.

If two high-pressure injection pumps are operating, the normal makeup valve capacity would become controlling at 140 gpm.

In order to postulate more than 140 gpm, two failures must be assumed. Failure of the normal makeup control plus a failure which caused an emergency injection valve to open could provide 468 gpm gross makeup, if two high-pressure injection pumps were assumed running prior to the accident.

Abnormal makeup would produce a high pressurizer level alarm from any one of three independent sensors when the water inventory reached approximately 2240 gallons above normal.

After approximately 2990 gallons total had been pumped from the letdown storage tank, a low level alarm would be actuated in that tank.

If it is assumed that the pressurizer level controller fails, with the flow rates already postulated, the times required to reach points with the pressurizer initially at normal level and the letdown tank at high operating level are tabulated below:

No. of Pumps Operating	Flow Gpm	Failure Mode & Water Source	Time To Alarm (Min) Hi-Press Level	Lo-Letdown Tank Level	Time to Fill Pressurizer (Min)
1	148	Single - Letdown Tank	15	20	35
2	468	Double - Letdown Tank	4.8	6.40	11.2

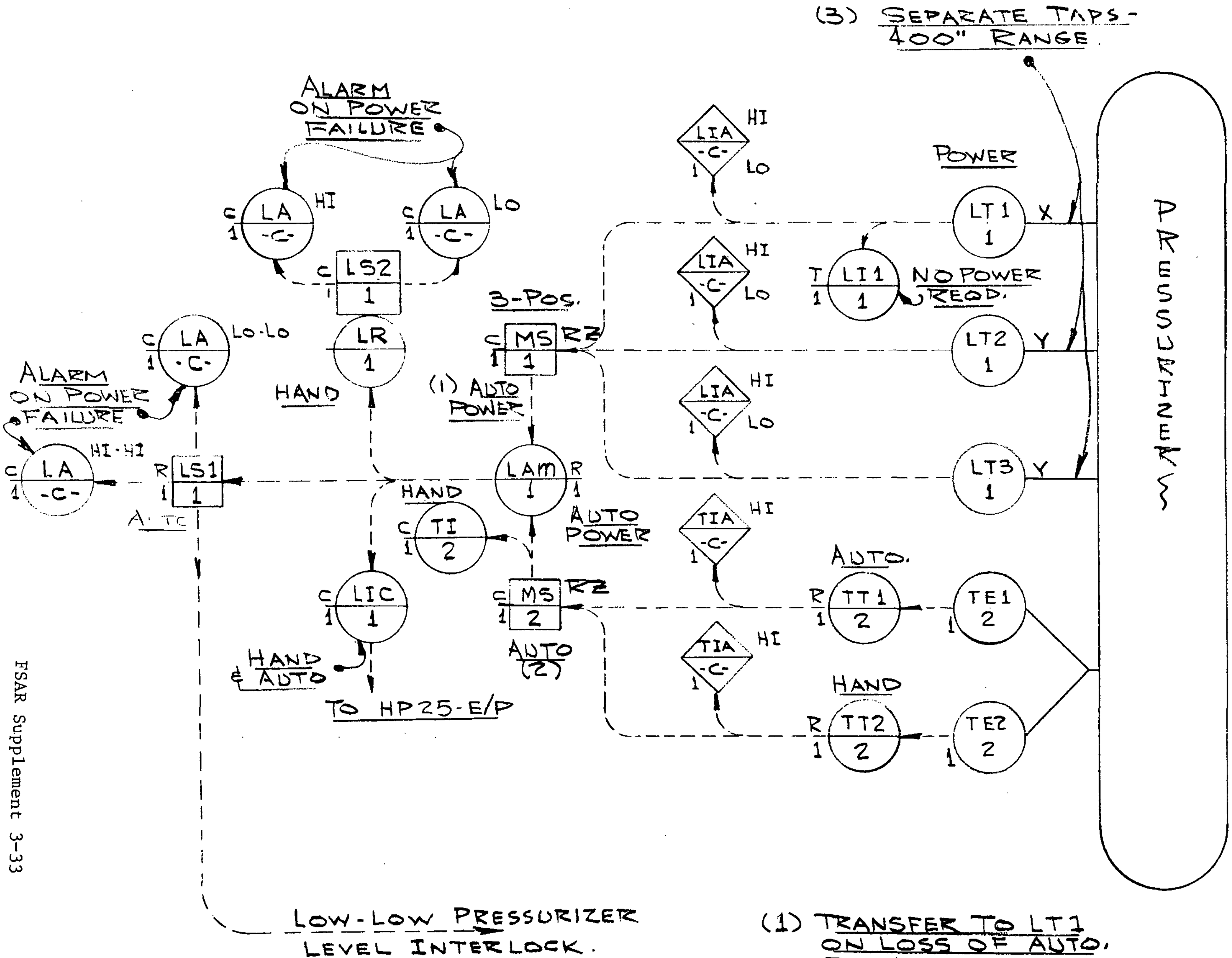
In conclusion, compounding the assumption of a rod withdrawal accident from power with a full pressurizer is not justified on the basis of the single failure criterion.

In connection with the discussion of the start-up accident and why a high pressurizer level trip is not required to protect against that accident, it should be pointed out that there are actually very few situations that involve filling the pressurizer completely. One of the few situations which does involve a solid system is a plant hydrotest.

Normal operating practice will call for a known pressurizer level to be established prior to performing any rod withdrawal operations. For example, when the reactor is being prepared for start-up, the written procedures will call for a minimum pressurizer level rather than a normal or above normal level. The procedures will be written this way to accommodate the coolant system expansion and to minimize the heatup time for the pressurizer.

If the system is to be hydrotested, the pressurizer is first filled solid by venting at low pressure. Then the reactor coolant system is isolated and pressure in the system is increased. During the low-pressure portion of the hydrotest, a low-pressure trip input to the reactor protection system exists and, during the high-pressure portion of the test, a high-pressure trip input to the reactor protection system exists.

In view of the above, it is not considered credible to compound a start-up accident with a full pressurizer.



X-118V. BUSS
Y-118V. BUSS
AUTO-118V. BUSS
HAND-118V. BUSS
} SEPARATE POWERED BUSSES

(1) TRANSFER TO LT1 ON LOSS OF AUTO. POWER.

(2) TRANSFER TO TT2 ON LOSS OF AUTO.

14.4.2 (Later)

In response to the questions contained in Dr. Peter A. Morris' letter of March 27, 1970, Babcock & Wilcox have submitted to you their Topical Report BAW-10023, "Computer Codes and Methods Used in Performing LOCA Analyses," dated June, 1970.

In response to the questions contained in Dr. Peter A. Morris' letter of April 22, 1970, the following tabulation, in the same format as the Director's letter, gives the information requested or the reference where the information is found:

3.8 Reactor Internals (The following request applies to B&W Report BAW-10008, Part 1.)

3.8.4 (See answer to 3.8.4 in reply to DRL letter of 3/3/70 (FSAR Supplement 3-2.)

3.8.13 a. Appendix H (BAW-10008, Part I (Rev. 1).)

b. Revised Fig. 23 and sketches in Appendix H (BAW-10008, Part 1 (Rev.1).)

3.8.14 Appendix H, paragraph 5.

3.8.15 Paras. 4.1 and 4.2 (BAW-10008, Part I (Rev. 1).)

3.8.16 Para. 4.3 (BAW-10008, Part I, (Rev. 1).)

3.9 (The following request applies to B&W Report BAW-10008, Part 2.)

3.9.14 ANSWER:

Since time history/modal analysis is not used in Part 2, but is used in Part 1, it is assumed that this question actually applies to Part 1.

(a) BAW-10008, Part 1, Figure 24.

(b) BAW-10008, Part 1, Appendix G (description of DYNAM computer code).

(c) Answer: No damping is used.

(d) BAW-10008, Part 1, Paragraphs 3.17 and Appendix H.4.4.

3.9.15 ANSWER:

(a) The mathematical models for the Phase 1 and Phase 2 analyses are presented in sections 2.1, 2.2, 3.1, and 3.2, Appendix A, BAW-10008, Part 2, Revision 1 (Proprietary).

(b) The engineering basis for and the validity of the decoupling assumed between the Phase 1 and Phase 2 models is given in section 2.7, Appendix A, BAW-10008, Part 2, Revision 1 (Proprietary).

(c) The analog diagrams for the two phases with accompanying explanation of symbols used are shown in Figures A-8, A-17, and A-18, BAW-10008, Part 2, Revision 1 (Proprietary).

(d) A complete discussion of the damping coefficients used is shown in Section 3.4, Appendix A, BAW-10008, Part 2, Revision 1 (Proprietary).

(e) A discussion of the gap and stiffness coefficients values selected is given in Section 3.5, Appendix A, BAW-10008, Part 2, Revision 1, (Proprietary).

(f) A complete force balance is given in Section 2.6, Appendix A, BAW-10008, Part 2, Revision 1 (Proprietary).

(g) A discussion of the output results is given in Section 3.7, Appendix A, BAW-10008, Part 2, Revision 1 (Proprietary).

Additional supplementary information in response to questions contained in Dr. Peter A. Morris' letter of February 13, 1970, is included with FSAR revision 6 (dated 6/22/70). The following tabulation gives the question number and the reference where the information is contained in the FSAR.

4.9 FSAR - 4.3.4, Pages 4-28 through 4-30b, and Table 4-29 (Page 4-72).

4.12.4 FSAR - 1C.3.4, Pages 1C-4a through 1C-4i and Figures 1C-20 through 1C-30.

4.12.5 FSAR - 1C.3.4.1, Pages 1C-4a through 1C-4C, and Table 1C-3, Page 1C-11; Table 1C-2, Pages 1C-6 through 1C-10; and 1C.3.4.3.2, Pages 1C-4j through 1C-4L.

5.10 FSAR - 1C.3.4.1, Pages 1C-4a through 1C-4C; and 1C.3.4.3, Pages 1C-4j through 1C-4L.

The following information is voluntarily submitted in response to informal questions asked by the Division of Reactor Licensing:

QUESTION An unreinforced buttress detail has been indicated in the FSAR on the basis of Taylor's tests to be unsatisfactory, but the capability of the added mild steel reinforcing to provide the necessary anchorage strength has not been demonstrated. In addition, Taylor's approach as used does not consider horizontal tensile stresses. As a result, the following information is requested for anchor zones at the mid height of the buttresses and the dome and vertical tendon anchor zones at the ring girder.

a. A drawing of the reinforcing in these zones.

b. The method of anchoring or splicing the reinforcing in such an area of biaxial tension combined with uniaxial compression.

c. The thermal gradients used in the analysis, with special consideration given the effect of transient thermal gradients due to startup or shutdown.

d. The calculated triaxial stress levels.

The following design criteria have been used on similar plants. Please indicate whether and how the Oconee design satifies these criteria.

ANCHORAGE ZONE DESIGN OF PRESTRESSING TENDONS

The design of concrete and rebars in the anchorage zones of prestressing tendons considers two main problems:

a. Evaluation of bearing stresses under the anchor bearing plate.

b. Determination of the transverse tensile forces (bursting forces) and the design of the corresponding reinforcing bars.

To provide an adequate margin of safety, the design of the anchorage zone shall be in accordance with the following criteria:

a. The tendon anchorage zone will be defined as a Class I element, as is the structure itself. Its design will be in accordance with the general design criteria for the containment structure and with the requirements of the current ACI Code.

b. The following will be established on a conservative basis by analytical, and/or experimental evidence:

(1) That there is no danger of delayed rupture of the concrete, under

sustained load, due to local overstress and microcracking.

(2) That reinforcing bars located in the anchorage zone are adequate to carry the tension stresses existing in this zone, with a safety factor similar to the safety factors provided in the design of the containment structure in general, and that the cracking in this zone will be safely controlled.

(3) That the possibility of concrete breaking along shear planes is excluded.

c. The most unfavorable loads and load combinations will be considered. Transient thermal gradients will be used in all cases where the use of steady state gradients under estimates the stress and strains. The design will cover not only the accident condition along, but also other cases such as: startup during very cold weather, after protracted shutdown; accident happening towards the end of the useful life of the structure, etc.

d. The design analyses will determine the three dimensional stress distribution in the anchor zones in sufficient detail to permit the rational evaluation of stress concentrations and of maximum tension and shear stresses. Standard design methods (by Guyon, Leonhardt, Taylor, etc.) will be modified to take care of three dimensional stress distribution existing in this zone. If experimental evidence is offered, it will include a three dimensional stress distribution similar to the most unfavorable stress distribution existing in the actual structure. Three dimensional creep and shrinkage of concrete will be considered.

ANSWER FSAR 5.1.3.2 (Pages 5-15, 5-17, 5-17a) and Figures 5.9A and 5.9B.

3.3.6 Exceptions to 3.3.5 shall be as follows:

(a) Both core flooding tanks shall be operational above 800 psig.

(b) Both motor-operated valves associated with the core flooding tanks shall be fully open above 800 psig.

(c) One pressure instrument channel and one level instrument channel per core flood tank shall be operable above 800 psig.

(d) One reactor building cooling fan and associated cooling unit shall
be permitted to be out of service for seven days provided both
30. reactor building spray pumps and associated spray nozzle headers
are in service at the same time.

3.3.7 Prior to initiating maintenance on any of the components, the duplicate (redundant) component shall be tested to assure operability.

Bases

30. The requirements of Specification 3.3 assure that, before the reactor can be made critical, adequate engineered safety features are operable. Two high pressure injection pumps and two low pressure injection pumps are specified. However, only one of each is necessary to supply emergency coolant to the reactor in the event of a loss-of-coolant accident. Both core flooding tanks are required as a single core flood tank has insufficient inventory to reflood the core.(1)

The borated water storage tanks are used for two purposes:

(a) As a supply of borated water for accident conditions.

(b) As a supply of borated water for flooding the fuel transfer canal during refueling operation.(2)

Three-hundred and fifty thousand (350,000) gallons of borated water (a level of 46 feet in the BWST) are required to supply emergency core cooling and reactor building spray in the event of a loss-of-core cooling accident. This amount fulfills requirements for emergency core cooling. The borated water storage tank capacity of 388,000 gallons is based on refueling volume requirements. Heaters maintain the borated water supply at a temperature to prevent freezing. The boron concentration is set at the amount of boron required to maintain the core 1 percent subcritical at 70^{o}F without any control rods in the core. This concentration is 1,338 ppm boron while the minimum value specified in the tanks is 1,800 ppm boron.

The spray system utilizes common suction lines with the low pressure injection system. If a single train of equipment is removed from either system, the other train must be assured to be operable in each system.

When the reactor is critical, maintenance is allowed per Specification 3.3.5 and 3.3.6 provided requirements in Specification 3.3.7 are met which assure operability of the duplicate components. Operability of the specified components shall be based on the results of testing as required by Technical Specification 4.5. The maintenance period of up to 24 hours is acceptable if the operability of equipment redundant to that removed from service is demonstrated immediately prior to removal. The basis of acceptability is a likelihood of failure within 24 hours following such demonstration.

It has been shown for the worst design basis loss-of-coolant accident (a 14.1 ft^2 hot leg break) that the reactor building design pressure will not
30. be exceeded with one spray and two coolers operable. Therefore, a maintenance period of seven days is acceptable for one reactor building cooling fan and its associated cooling unit.(3)

In the event that the need for emergency core cooling should occur, functioning of one train (one high pressure injection pump, one low pressure injection pump, and both core flooding tanks) will protect the core and in the event of a main coolant loop severence, limit the peak clad temperature to less than 2,300°F and the metal-water reaction to that representing less tha 1 percent of the clad.

Three low pressure service water pumps serve Oconee Units 1 and 2. One low pressure service water pump per unit is required for normal operation. The normal operating requirements are greater than the emergency requirements following a loss-of-coolant accident.

A single train of reactor building penetration room ventilation equipment retains full capacity to control and minimize the release of radioactive materials from the reactor building to the environment in post-accident conditions.

REFERENCES

(1) FSAR, Section 14.2.2.3
(2) FSAR, Section 9.5.2
30. (3) FSAR, Section 14.2.2.3.5
(4) FSAR, Section 6.4

c. Except for physics tests or exercising control rods, the control rod withdrawal limits are specified on Figures 3.5.2-1-1 (for up to 435 full power days of operation) and 3.5.2-1-2 (for after 435 full power days of operation) for four pump operation and on Figure 3.5.2-2 for three or two pump operation.

d. Reactor Power Imbalance shall be monitored on a frequency not to exceed two hours during power operation above 40 percent rated power. Imbalance shall be maintained within the envelope defined by Figure 3.5.2-3. If the imbalance is not within the envelope defined by Figure 3.5.2-3, corrective measures shall be taken to achieve an acceptable imbalance. If an acceptable imbalance is not achieved within four hours, reactor power shall be reduced until imbalance limits are met.

3.5.2.6 The control rod drive patch panels shall be locked at all times with limited access to be authorized by the superintendent.

Bases

The power-imbalance envelope defined in Figure 3.5.2-3, is based on LOCA analyses which have defined the maximum linear heat rate (see Figure 3.5.2-4) such that the maximum clad temperature will not exceed the Interim Acceptance Criteria. Operation outside of the power imbalance envelope alone does not constitute a situation that would cause the Interim Acceptance Criteria to be exceeded should a LOCA occur. The power-imbalance envelope represents the boundary of operation limited by the Interim Acceptance Criteria only if the control rods are at the withdrawal limits as defined in Figures 3.5.2-1 and 3.5.2-2 and if a 5 percent quadrant power tilt exists. Additional conservatism is introduced by application of:

a. Nuclear uncertainty factors
b. Thermal calibration uncertainty
c. Fuel densification effects
d. Hot rod manufacturing tolerance factors

The 30 percent overlap between successive control rod groups is allowed since the worth of a rod is lower at the upper and lower part of the stroke. Control rods are arranged in groups or banks defined as follows:

Group	Function
1	Safety
2	Safety
3	Safety
4	Safety
5	Regulating
6	Regulating
7	Xenon transient override
8	APSR (axial power shaping bank)

The minimum available rod worth provides for achieving hot shutdown by reactor trip at any time assuming the highest worth control rod remains in the full out position.(1)

Inserted rod groups during power operation will not contain single rod worths greater than 0.5% $\Delta k/k$. This value has been shown to be safe by the safety analysis of the hypothetical rod ejection accident.(2) A single inserted control rod worth of 1.0% $\Delta k/k$ at beginning of life, hot, zero power would result in the same transient peak thermal power and therefore the same environmental consequences as a 0.5% $\Delta k/k$ ejected rod worth at rated power.

Control rod groups are withdrawn in sequence beginning with Group 1. Groups 5, 6, and 7 are overlapped 25 percent. The normal position at power is for Groups 6 and 7 to be partially inserted.

The quadrant power tilt limits set forth in Specification 3.5.2.4 have been established within the thermal analysis design base using the definition of quadrant power tilt given in Technical Specifications, Section 1.6. These limits in conjunction with the control rod position limits in Specification 3.5.2.5c ensure that design peak heat rate criteria are not exceeded during normal operation when including the effects of potential fuel densification.

REFERENCES

[1]FSAR, Section 3.2.2.1.2

[2]FSAR, Section 14.2.2.2

DUKE POWER COMPANY

OCONEE NUCLEAR STATION

UNITS 1, 2, and 3

APPLICATION FOR LICENSES

Dockets 50-269, -270, and -287

FSAR SUPPLEMENT 4

Submitted with FSAR Revision 7

July 9, 1970

LIST OF EFFECTIVE PAGES

FSAR SUPPLEMENT 4

In response to Question 14.4.2 of Dr. Peter A. Morris' letter of March 3, 1970 the following answer is submitted.

ANSWER:

Uncovering energized direct immersion heaters does not immediately harm the heaters. Three heaters, one for each bundle assembly, are tested in air to provide an accelerated life test as follows:

(1) Tested for 100 hours at a sheath temperature of 600F to 1600F with a watt density of 85 watt/in.^{2}.

(2) Cycled 400 times with a cycle time of 15 minutes on and 15 minutes off with a watt density of 65 watt/in.^{2}.

The heaters successfully completed this test, which simulated a total of 200 hours "on" time for the heaters in an uncovered environment while in an energized condition. Moreover, the heater sheath is designed for 2500 psig and 670F with the heater terminal also designed for these same conditions. Therefore, the heater sheath could fail and the pressurizer vessel integrity would be maintained. This conclusion has been substantiated in tests conducted by the heater vendor for a similar design.

In response to the questions contained in Dr. Peter A. Morris' letter of April 15, 1970, the following tabulation in the same format as the Director's letter, gives the information requested or the reference where the information is found.

14.5 REACTOR COOLANT PUMP LOCKED ROTOR ACCIDENT

FSAR - 14.1.2.6, Pages 14-10a through 14-12, and Figures 14-15a, 14-16, 14-17, 14-17a, and 14-17b.

14.6 REACTOR COOLANT PUMP SHEARED SHAFT ACCIDENT

ANSWER:

The sheared-shaft accident has been analyzed under the same conditions as are outlined for the locked rotor accident (Question 14.5). The calculational techniques are essentially identical except for the calculation of flow versus time. The sheared shaft was simulated by changing the inertia in a stepwise fashion from its normal value (70,000 $lb\text{-}ft^2$) to a very low value (2400 $lb\text{-}ft^2$). The calculated flow versus time was found to be within $\pm$ 2% of the locked rotor cases, therefore, the calculated power, pressure and other plant parameters were nearly the same. Figure 14.6-1 (attached) shows the result of the thermal calculations. The minimum DNB ratio of 1.145 is reached at about 1.5 seconds after the initiation of the accident. The maximum hot spot centerline temperature was found to be 4066°F. Since the DNBR did not go below 1.0, the cladding temperature did not vary appreciably from its normal value.

14.7 OPERATION WITH LESS THAN FOUR REACTOR COOLANT PUMPS RUNNING

Part a. ANSWER:

The following statements summarize system flow conditions for less than four pumps operating:

Three Pumps Operating

a. Flow in the normally operating loop - 71.1 x 10^6 lb/hr.

b. Steam generator flow in the one pump loop - 29.5 x 10^6 lb hr.

c. Flow through the operating pump in the one pump loop - 43.6 x 10^6 lb/hr.

d. Back flow through the down pump - 14.1 x 10^6 lb/hr.

One Pump Each Loop

a. Flow through each operating pump - 44.5 x 10^6 lb/hr.

4 REACTOR COOLANT SYSTEM

4.1 DESIGN BASES

4.1.1 PERFORMANCE OBJECTIVES

4.1.1.1 Steam Output

The reactor coolant system is designed to operate at a core power level of 2,568 MWt and transfer a total of 2,584 MWt (including 16 MWt input from reactor coolant pumps) to the steam generators. The system will produce a total steam flow of 11.2 million lb/h.

4.1.1.2 Transient Performance

The reactor coolant system will follow step or ramp load changes under automatic control without relief valve or turbine bypass valve action as follows:

a. Step load changes - increasing or decreasing load steps of 10% of full power in the range between 20% and 90% full power.

b. Ramp load changes - increasing load ramps of 10% per minute in the range between 20% and 90% full power are acceptable, or decreasing load ramps of 10% per minute from 100% to 15% full power. From 15% to 20% and from 90% to 100% full power, increasing ramp load changes of 5% per minute are acceptable.

The combined actions of the control system and the turbine bypass system permit a 40% load rejection or a turbine trip from 40% full power without safety valve action. The combined actions of the control system, the turbine bypass valves, and the main steam safety valves are designed to accept separation of the generator from the transmission system without reactor trip.

4.1.1.3 Partial Loop Operation

The reactor coolant system will permit operation with less than four reactor coolant pumps in operation. The steady-state operating power levels for combinations of reactor coolant pumps operating are as follows:

Reactor Coolant Pumps Operating	Rated Power, %
4	100
3	75
2 in same loop	46
1 pump each loop	49

9. For single loop operation with two reactor coolant pumps, certain reactor trip setpoints must be adjusted while the reactor is shutdown. These adjustments are described in Section 7.1.2.2.3.

4.1.2 DESIGN CONDITIONS

4.1.2.1 Pressure

The reactor coolant system components are designed structurally for an internal pressure of 2,500 psig.

4.1.2.2 Temperature

The reactor coolant system components are designed for a temperature of 650 F with the exception of the pressurizer, surge line, and a portion of the spray line piping which are designed for 670 F.

4.1.2.3 Reaction Loads

Reactor coolant system components are supported and interconnected so that stresses resulting from combined mechanical and thermal forces are within established code limits. Equipment supports are designed to transmit piping rupture reaction loads to the foundation structures.

The reactor coolant system supports are on an eight foot six inch (8'-6")
thick, heavily reinforced concrete slab which rests on a solid rock subgrade.
4. The minimum ultimate crushing strength of rock cores tested was 720 kips per
square foot and the maximum applied dynamic gross load is 30 kips per square
foot. Based on the subgrade, the ratio of applied load to bearing capacity
of the subgrade, and the monolithic nature of the base slab, differential
settlement of the foundation is not anticipated.

4.1.2.4 Cyclic Loads

All reactor coolant system components are designed to withstand the effects of cyclic loads due to system temperature and pressure changes. Design transient cycles are shown in Table 4-8.

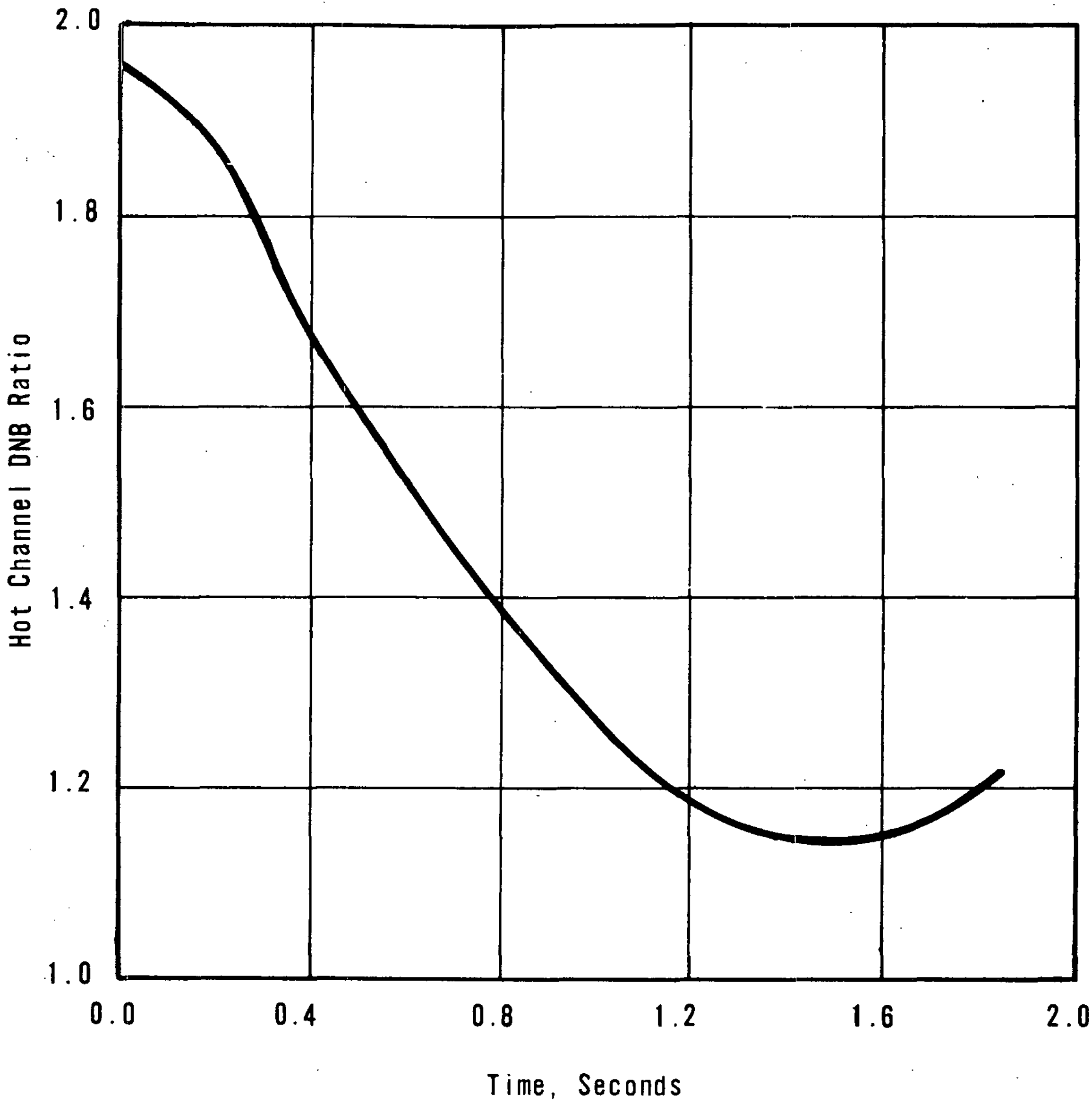
Flow-induced vibration analyses have been performed for the fuel assembly,
including fuel rods, and for the reactor internals components. The analyses
23. and design criteria for the thermal shield, flow distributor assembly, sur-
veillance holder tubes and shroud tubes, and the U baffles are given in B&W
Topical Report BAW-10051.

Components subjected to cross flow are checked for response during design, so
that the fundamental frequencies associated with cross flow are above the
vortex-shedding frequencies. It has also been conservatively determined that
5. the flow induced pressure fluctuations acting on the disc of the vent valve
are such that for normal operation there is always a positive net closing force
acting on the disc. Emergency operational modes are covered in Topical Report
21. BAW-10008, Part 1, and BAW-10035.

4.1.2.5 Seismic Loads and Loss-of-Coolant Loads

Reactor coolant system components are designated as Class I equipment and are

Rev. 4. 4/20/70
Rev. 5. 5/25/70
Rev. 21. 7/26/62
Rev. 23. 9/15/72



DNBR VERSUS TIME FOR A SHEARED
SHAFT FROM RATED POWER

Question 14.6 DRL Letter
of 4/15/70

b. Flow through each steam generator - 32.6×10^6 lb/hr.

c. Back flow through each down pump - 11.9×10^6 lb/hr.

Two Pumps in One Loop

a. Flow in operating loop - 73.6×10^6 lb/hr.

b. Back flow through down loop - 11.9×10^6 lb/hr.

One Pump Operating

a. Flow through operating pump - 45.0×10^6 lb/hr.

b. Back flow through idle pump in operating loop - 10.6×10^6 lb/hr.

c. Back flow in idle loop - 5.5×10^6 lb/hr.

For the configurations with both loops in operation the temperature in the cold legs will be the core inlet temperature (about 554°F). The hot leg fluid will be at about 604°F.

For the single loop, two pump case, with the idle loop steam generator isolated, the cold leg temperatures will be at the core inlet temperature (about 556°F). The idle hot leg will also be at the core inlet temperature. The core outlet will be at about 609°F and the reactor outlet will be approximately 6°F lower.

The reactor will not be operated at power in the single loop mode unless the steam generator in the idle loop is isolated, i.e., all feedwater valves closed, and both pumps are operating in the operating loop. The safety analysis for the startup of two idle pumps in a single loop without steam generator isolation is presented in 14.7e.

Part b. ANSWER:

1. The FSAR, Rev. 4, 4/20/70, Section 13 includes abstracts of the following tests which will measure reactor coolant system flows, flow coastdown, and temperatures.

   Reactor Coolant Flow Test (200/12)

   Reactor Coolant Flow Coastdown Test (200/13)

   Unit Startup Test (800/1)

   Unit Load Steady-State Test (800/28)

2. In the first two tests explicit flow measurements with 4 pump and

representative 3, 2, and 1 pump combinations will be made, with the reactor shut down, as follows:

(a) Loops A and B, 36" coolant legs, forward and reverse flow.

(b) Reactor coolant pumps 1A1 and 1A2 ΔP with flow derived from pump characteristic.

(c) Loops A and B 36" legs, loop flow during coastdown for representative one, two, three pump trips and four pump trip.

3. In the latter two tests above, reactor inlet and outlet temperatures (and other parameters) will be measured at various reactor power levels with four reactor coolant pumps operating, and representative three and two pump operating conditions.

Part c. ANSWER:

A qualitative analysis has been performed for various configurations where fewer than four coolant pumps are in operation. The effects of events that could occur during four pump operation are less severe for partial loop operation since steady-state DNB ratios are greater for partial pump modes of operation than for four pump operation.

There is, however, a type of accident that could occur during single loop operation that is not possible for two loop operation. Specifically, the idle loop startup is examined in detail in 14.7e.

Part d. ANSWER:

Present criteria with respect to operation with less than all reactor coolant pumps in operation calls for equalizing return temperatures to the core inlet and maintaining an average temperature of 579°F in the reactor coolant loop with the greatest number of pumps running.

Control modes consist of automatic, manual, load-tracking, and startup. These are defined in the following manner:

1. Automatic - All control stations on automatic control. The Unit Load Demand (ULD) is utilized as the feed-forward demand to the reactor control, steam generator control, turbine control, and the turbine bypass control. The ULD receives and conditions the load demand from the customer's automatic dispatch system (ADS) or the computer to make it compatible with the state of the unit and the unit's ability to change load. Load limiting and runback functions, if required, are initiated by the ULD.

2. Manual - The ULD control station is on hand control. In this mode, the operator is responsible for establishing the unit demand.

3. Load Tracking - Mode of operation whereby the ULD causes the system to track a component that is limited or on manual control. The following conditions will cause load tracking:

   (a) Unit under cross limits
   (b) Steam generator/reactor master in hand control
   (c) Control rods in manual
   (d) Reactor tripped
   (e) Reactor master in manual control
   (f) Power/load unbalanced
   (g) Turbine control station on manual
   (h) Both generator breakers tripped
   (i) Both Loop A and B feedwater demand selector stations on manual.

4. Startup - The reactor is at hot standby conditions (i.e., 532°F, 2200 psia) with the ULD on manual control.

The Steam Generator Feedwater Control portion of ICS is designed to maintain the feedwater flow equal to the unit load demand in steady state and parallel with the reactor power demand. The steam generator feedwater demand develops signals for automatic positioning of flow control devices based on the capability of the reactor, turbine, and feedwater system. Each feedwater positioning signal operates in parallel to each steam generator's feedwater control valve and is sequenced to respective valves to deliver the required feedwater flow for producing steam for reactor guarantee.

Low reactor coolant flow limits the feedwater flow to the affected steam generator consistent with the BTU availability of the primary system. When one steam generator becomes BTU limited (or high level, the remaining steam generator feedwater control is responsible for satisfying the reactor coolant inlet temperature deviation.

A. <u>Three Reactor Coolant Pumps Running</u>

1. Automatic Operation

The Unit Load Demand limits the reactor to a maximum of 75% load. A load distribution ratio of 2.44 to 1.0 between the steam generators is predicted as normal for two reactor coolant pumps in one loop and one reactor coolant pump in the other. This ratio scheme automatically controls the reactor inlet temperatures to maintain, under transient and steady-state conditions, no more than a 5°F difference between loops.

The steam generator with two primary pumps in operation produces steam at a minimum of 570°F while the steam generator with one pump produces steam at a maximum of 600°F resulting in an imbalance of 30°F maximum in the secondary outlet temperatures of the steam generators.

2. Manual Operation

The ICS functions as for the automatic case except that its load demand is now established by the operator.

3. Load Tracking

For a Type I load tracking condition, the operation of the ICS, with the exception of the ULD, is identical to the automatic case.

For a Type II load tracking condition, the ULD limits the load to 75% and the operator is responsible for maintaining proper loop flows to maintain equal return temperatures from each loop to the reactor core inlet. The load control is not functional with both steam generator feedwater selector stations on hand control.

4. Startup

With the system at hot standby condition and prior to loading, the reactor and turbine will be on hand control. Initially, both steam generators are on level limit control. The operator will manually increase the reactor power until both units come off level control. The steam generator with two pumps running will be off level control at approximately 11% power, the remaining steam generator will be off level control at approximately 26%. Until this power level is attained, the level control overrides the reactor inelt temperature control. From 26% power to full load, the operator may place the unit on automatic control and the resultant operation is similar as in the automatic case.

B. Two Pumps in One Loop

With two inoperative pumps in one primary loop, reverse flow occurs in the primary loop without pumps.

With respect to operation of the steam generator with backflowing primary water, two possibilities are considered:

1. Isolated steam generator. In this case, the shell side of the unit will be come filled with steam which is slightly superheated.

2. Load the steam generator by maintaining a preset downcomer level: Under this condition (backflow) the inlet temperature to the steam generator without primary pumps corresponds to the outlet temperature of the steam generator with pumps.

The outlet of the steam generator without pumps discharges into the reactor vessel outlet, bypasses the core, mixes with core outlet flow, and enters the inlet of the steam generator with pumps. The maximum loading of the steam generator without pumps would be that which causes the primary outlet temperature to approach saturation temperature at 900 psia (throttle pressure). This temperature is 532°F.

If the steam generator with no primary pumps is not loaded the temperature in the primary loop without pumps is equal to the outlet temperature of the steam generator with pumps.

If the steam generator without pumps is loaded to the maximum as described above, the effect will be to raise the core outlet temperature and lower the inlet temperature to the steam generator with two pumps. The maximum loading of the steam generator without pumps would increase with decreasing total load because the inlet temperature to this steam generator increases with decreasing load.

1. Automatic Operation

   The Unit Load Demand limits maximum load to 46%. The feedwater demand selector station for the steam generator with no reactor coolant pumps operating will be on manual and set to zero output. The feedwater flow to the remaining steam generator will be automatically controlled to meet the total feedwater demand. Further, there will be no automatic reactor inlet temperature control and the steam generator with no pumps running will be on minimum level control.

2. Manual Operation

   Operation is identical to 3 pump running case discussed in A.2.

3. Load Tracking

   The situation is identical to the 3-pumps running case discussed in A.3.

4. Startup

   Initially, both units will be on minimum level control. The steam generator with no pumps running will remain on minimum level control throughout the entire power range. The feedwater system for both units will be on full automatic control.

C. Two Pump Operation (1 per Loop)

This is a symmetrical situation and imbalances in the primary and secondary temperatures are not expected. Operations in this mode are identical to 4 pump operation except that the ULD limits maximum load to approximately 49%.

D. One Pump Operation

The general flow pattern in the reactor coolant system will be similar to that of two pumps in one loop. The flow rates, however, at all points except in the subloop of the one operating pump are reduced.

With only a single reactor coolant pump in operation, the reactor is in the tripped mode. With respect to the ICS, operation in this mode is not an allowable case.

Part e. ANSWER:

This information is included as a revision to Section 14.1.2.5, Pages 14-10 and 14-10a of the FSAR. (Submitted with Revision 7 dated July 9, 1970).

14.8 RESTART OF A TRIPPED PUMP

Part a. ANSWER:

Inasmuch as no power operation will occur with only one pump or with a non-isolated steam generator, the potential for a cold water accident will be limited to the case when subsequent to operation with three pumps, the pump in the one pump loop is shut off and then restarted. The potential for a cold water accident will be quite small, however, since the idle steam generator will be isolated within 30 minutes and only the time between shutting off the pump and steam generator isolation will be available for this incident.

In answer to Question 14.73, the worst cold water accident is presented that could occur with this reactor system and the results are shown to be acceptable even with very conservative assumptions. It is therefore concluded that the potential for cold water insertion is an acceptable one for this reactor system.

Part b. ANSWER:

Measurements of reactor inlet and outlet temperatures, reactor coolant flows, core power, reactivity and other parameters will be made during the Reactor Coolant Pump Restart test. This test will consist of reactor coolant pump restarts with the reactor at selected powers. The above variables will be monitored during each phase of the test in order to verify proper system operation and to evaluate the effects of transients on system performance.

14.9 STARTUP ACCIDENT

ANSWER:

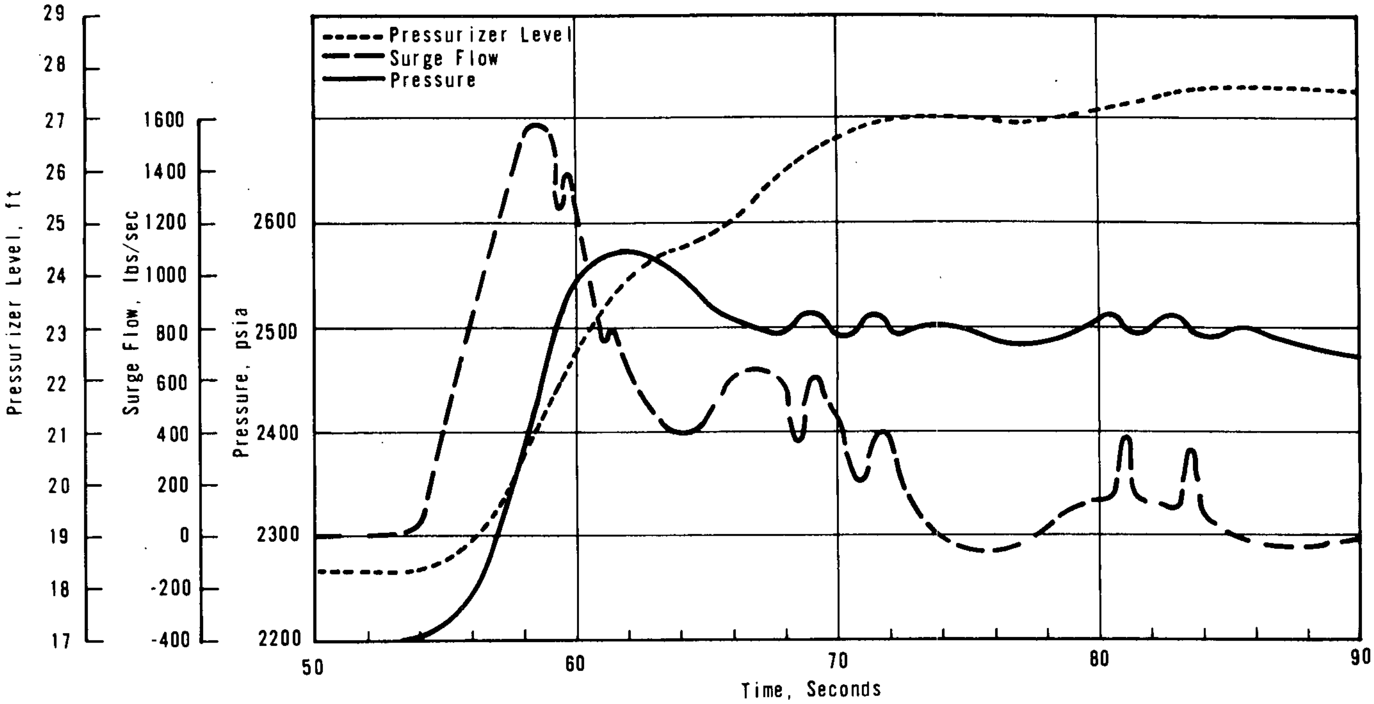
Figure 14.9-1, attached, shows the pertinent results of a startup accident calculation for a ramp reactivity insertion of 2.15×10^{-4} $(\Delta k/k)/s$. This reactivity insertion is approximately the maximum reactivity insertion rate that will not result in a high flux trip.

As seen from the data the pressure begins to increase about 54 seconds after the initiation of the accident and reaches its maximum value of 2575 psia at 62 seconds. The pressure then decreases to the safety valve setpoint and is approximately constant at this value for the remainder of the calculation. The pressure does not decrease below this value because NO heat is assumed to be removed on the secondary side.

The pressurizer level, initially at 220 inches (18.33 ft), increases to about 27.5 ft before becoming constant.

The system expansion rate, shown as the surge flow in the figure, reaches its maximum value of 1600 lb/s at about 58 seconds. The safety valves are capable of preventing pressure rises above 2515 psia up to surge flow rates of about 1000 lb/s, therefore the pressure increases slightly above 2515 psia and reaches 2575 psia. The surge flow begins to decrease at 58 seconds but does not go significantly below zero flow because of the assumption of no secondary heat removal. Normally the surge flow would go negative and remain negative until the pressurizer level returns to normal.

Docket 50-269, -270, and -287
FSAR Supplement 4
July 9, 1970



PRESSURE BEHAVIOR FOR A STARTUP ACCIDENT
AT 2.15 X 10^{-4} $\Delta k/k/s$

Question 14.9, DRL Letter
of 4/15/70

Additional supplemental information in response to questions 4.11 and 7.22 of Dr. Peter A. Morris' letter of February 13, 1970 is given as follows:

23. 4.11 ANSWER: (The response to this question has been superseded by the program given in B&W Topical Report BAW-10038)

First-of-a-kind instrumentation which will measure flow induced vibrations at specific locations during pre-operational testing will be installed on the Oconee 1 internals. Confirmatory measurements will be made on Oconee 2 and Oconee 3 internals.

General

The directions and velocities of the coolant flow are controlled by the design of the reactor internals and are primary criteria used to determine what internals components should or should not be measured. Consequently, a brief description of the coolant flow through the reactor as indicated in Figure 4.11-1 is given below.

Coolant for the core enters through the four reactor inlet nozzles. It is then directed downward in an outside annulus defined by the inside surface of the vessel, the core support shield, and the thermal shield. Approximately 99.6% of the downward flow enters an outside annulus at approximately 23 ft/sec. The remaining 0.4% enters an inside annulus between the inside surface of the thermal shield and the outside surface of the core barrel. The flow velocity in this annulus is limited to less than 1 ft/sec. by orifices located in the bottom of the core barrel cylinder.

Flow in the outside annulus enters the plenum region in the bottom of the vessel, turns and then flows upward through the core. Approximately 1.5% of the upward flow passes through an annulus between the core barrel inside surface and the back side of the baffle plates. Velocity in this annulus is also limited to less than 1 ft/sec.

As the coolant exits from the core, it enters the plenum assembly. The plenum cylinder maintains the coolant flow parallel to the outside of the guide tube assemblies. Flow passes from the plenum to the two outlet nozzles through 34" and 22" diameter holes in the upper section of the plenum. The maximum flow velocity across the guide tube assemblies adjacent to the plenum outlets is approximately 19 ft/sec. At the two locations where a small amount of outlet flow passes through a cluster of 24 3" diameter holes, the flow across the adjacent guide tube assemblies is only 8 ft/sec.

The flow direction and velocity control where chosen to reduce the possibility of developing forces which would result in damaging vibrations in all regions of the core. The resulting velocities are low enough to preclude the necessity of measuring motions of the core barrel, control rod guide tube assembly (a part of the plenum assembly), and other upper plenum assembly components, as can be seen from the following:

A. The 19 ft/sec. flow velocity across the guide tube assemblies adjacent to the outlets in the plenum results in a vortex shedding frequency of only 6 cps. Since this shedding frequency is much lower than the 50 cps fundamental of the guide tube assembly, it was concluded that the assemblies will not have significant vibratory motion from the cross flow.

B. The flow velocity in the annulus between the core barrel and the thermal shield is less than 1 ft/sec. At this extremely small velocity, the vibratory motion of the shell modes will be negligible. Beam type motions of the core barrel can be measured by the upper accelerometer in the surveillance holder tube assembly. (The accelerometer instrumentation is described later.)

C. The plenum cover assembly is an extremely stiff assembly. Flow across the plenum cover occurs only at the outer edge of the assembly at a low velocity of 5 ft/sec. The force on the assembly due to flow is insignificant.

D. Since the coolant at 100% power operation is subcooled at the discharge of the fuel assembly, no steam bubbles exist which might induce vibration of the control rod guide tubes, plenum cylinder or plenum cover assembly.

Pre-operational testing will yield results which are comparable to or more conservative than during operation for the following reasons:

A. The total flow is slightly greater during hot functional testing when the reactor core is not in place than during operation. This is particularly true for pump combinations of less than 4 pumps.

B. The velocities in areas of concern are not significantly influenced by the flow differences with or without the core.

Oconee 1

Instrumentation

The internals components which will be measured during pre-operational testing are the surveillance specimen holder tube, the thermal shield and the plenum cylinder. Details of the instrumentation follow.

A set of two accelerometer assemblies will be installed in each of two surveillance specimen holder tubes. The location of the holder tubes is shown in Figure 4.11-2. The accelerometer transducers will be located in the perforated section of the holder tube assembly as shown in Figure 4.11-3. In addition, two weights which simulate the surveillance capsules will be installed in each perforated tube.

The location of the lower accelerometer was selected to measure the midspan vibratory motions of the perforated tube. The perforated section of the surveillance holder tube is expected to have the largest flow induced vibratory amplitudes relative to the other sections of the holder tube assembly.

The upper end of the perforated tube is connected to the thermal shield. Consequently, the upper accelerometer will measure the thermal shield mid-plane vibratory amplitudes.

The 1" penetrations in the reactor vessel head permit the addition of three accelerometers to measure the shell mode vibrations of the plenum cylinder. One accelerometer will be located at the lower end of each of three tubes which are welded to the outside of the cylinder adjacent to the outlet holes as shown in Figure 4.11-4.

Each of the four accelerometers in the surveillance holder tube is biaxial. Therefore, there will be eight separate channels, four channels for measuring the acceleration amplitudes of the lower section of the surveillance holder tube and four channels for detecting the accelerations of the thermal shield. The uniaxial accelerometers for the plenum cylinder will provide three channels for measuring the acceleration amplitudes of the cylinder.

The accelerometers, specially designed for the components, will be capable of measuring the frequency of the components over a range of 2 to 300 Hz at acceleration up to 30g's.

Analysis

The acceleration signals from the various components will be recorded on tape by a FM tape recorder. After the signals are recorded, the information on the tape will be digitized by use of a mini-computer which samples the data at preset time intervals. The digitized time history record will then be used as input to a computer program which will analyze the record.

A B&W proprietary computer program will be used to plot the time history of the fluctuating accelerations, determine the predominant frequencies, the autocorrelation of the signal and phase differences between signals.

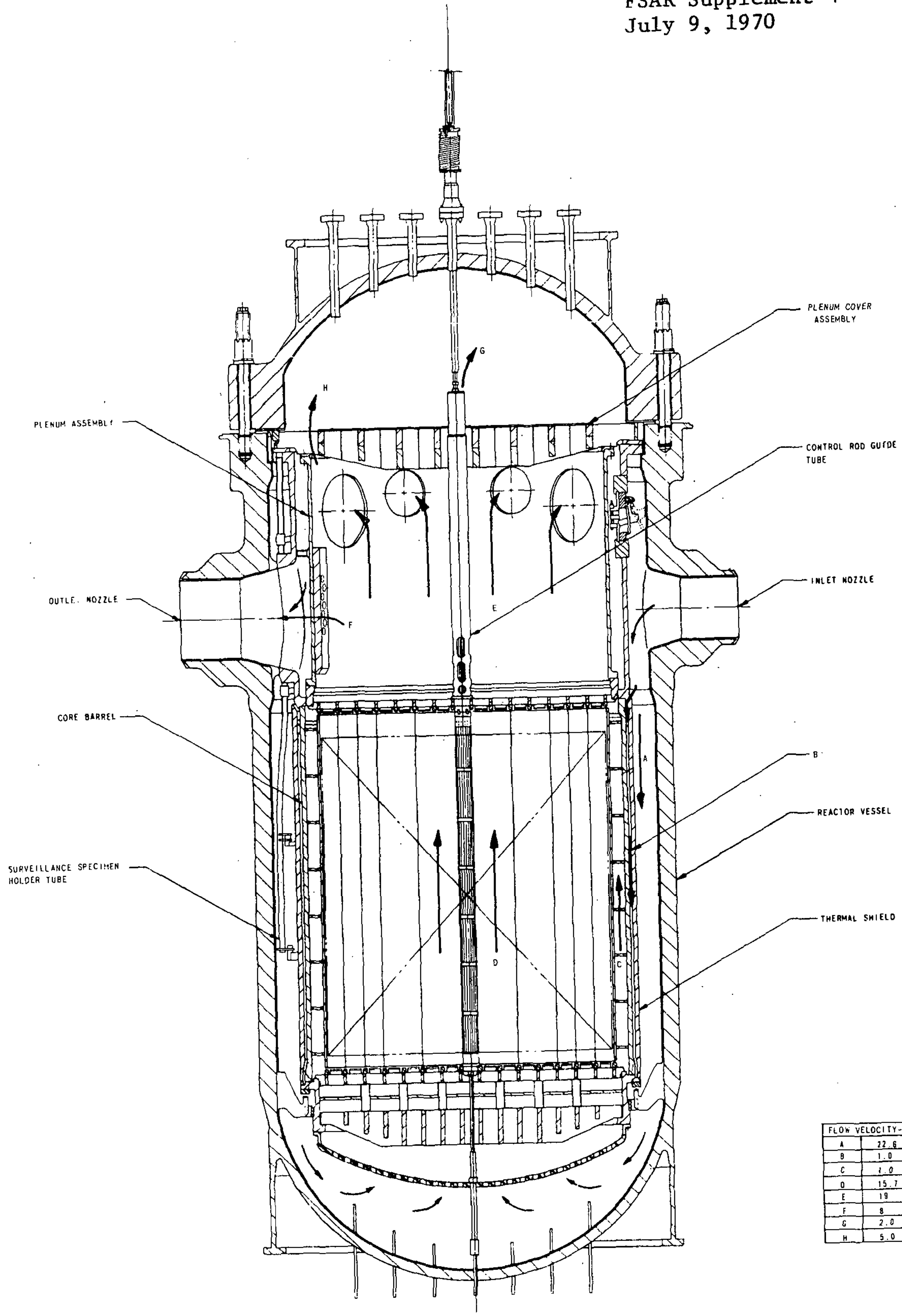
Cyclic stress values will be determined from the measured acceleration amplitudes, frequency and mode shapes. These dynamic stresses will be combined with normal operational stresses. The combined stresses will be judged acceptable if they are less than the endurance limit for the materials used to manufacture the components.

Oconee 2 & 3

The reactor vessels and internals designed for Oconee 2 and 3 are essentially identical to Oconee 1. To confirm that the fabrication process has not altered the characteristics of the internals, one surveillance holder tube for Oconee 2 and one for Oconee 3 will be instrumented like Oconee 1. Measurements will be made as described for Oconee 1. The instrument cables will go through a control rod nozzle (requiring the removal of a control rod drive mechanism) because the reactor vessel heads for these units do not have the 1" penetrations. The results from each of these tests will be compared to those for Oconee 1 to confirm that the vibration characteristics are similar.

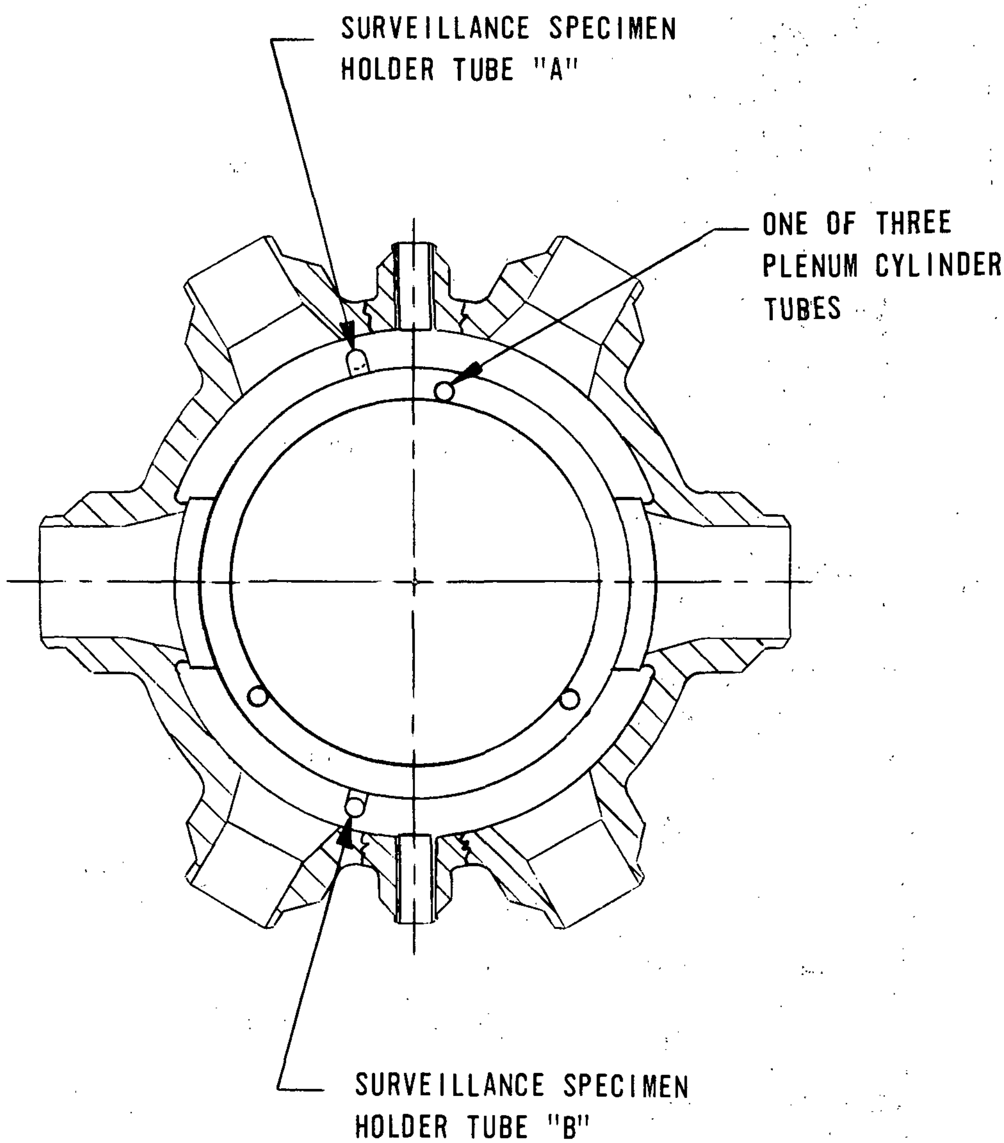
In-Service Monitoring

The feasibility of in-service monitoring for vibration and the detection of loose parts is being explored. If and when instrumentation for in-service monitoring for vibration and detection of loose parts inside the reactor vessel prove to be feasible and practical, giving reliable results, Duke Power Company will make every reasonable effort to install and keep in service these instruments in each of the Oconee reactors. Additional discussion with consultants and instrumentation vendors is planned in order to determine the feasibility and practicality of such systems in operating PWR systems.



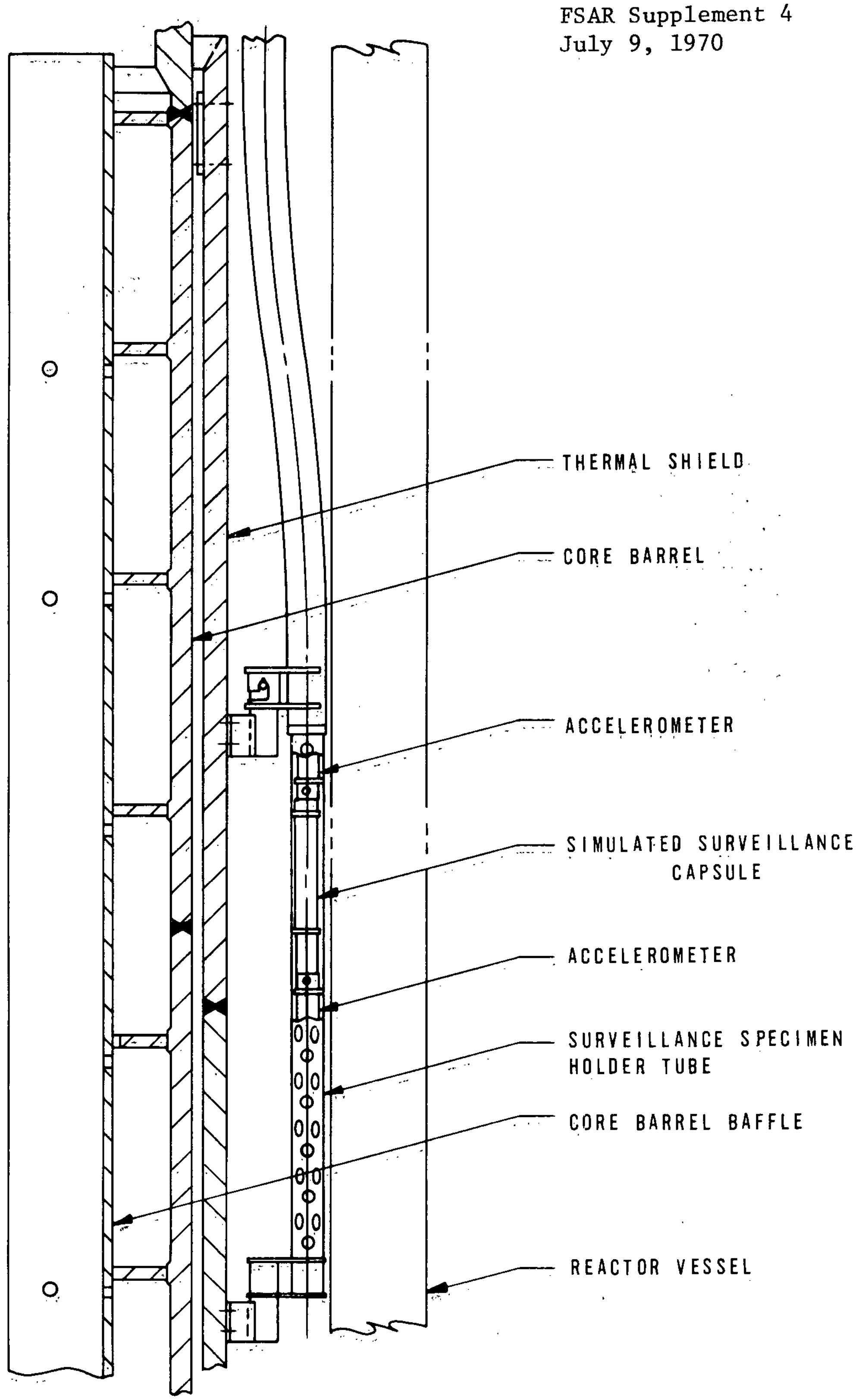
FLOW VELOCITY-FT/SEC	
A	22.6
B	1.0
C	2.0
D	15.7
E	19
F	8
G	2.0
H	5.0

DIRECTIONS AND VELOCITIES OF THE COOLANT FLOW IN THE REACTOR
Figure 4.11 - 1



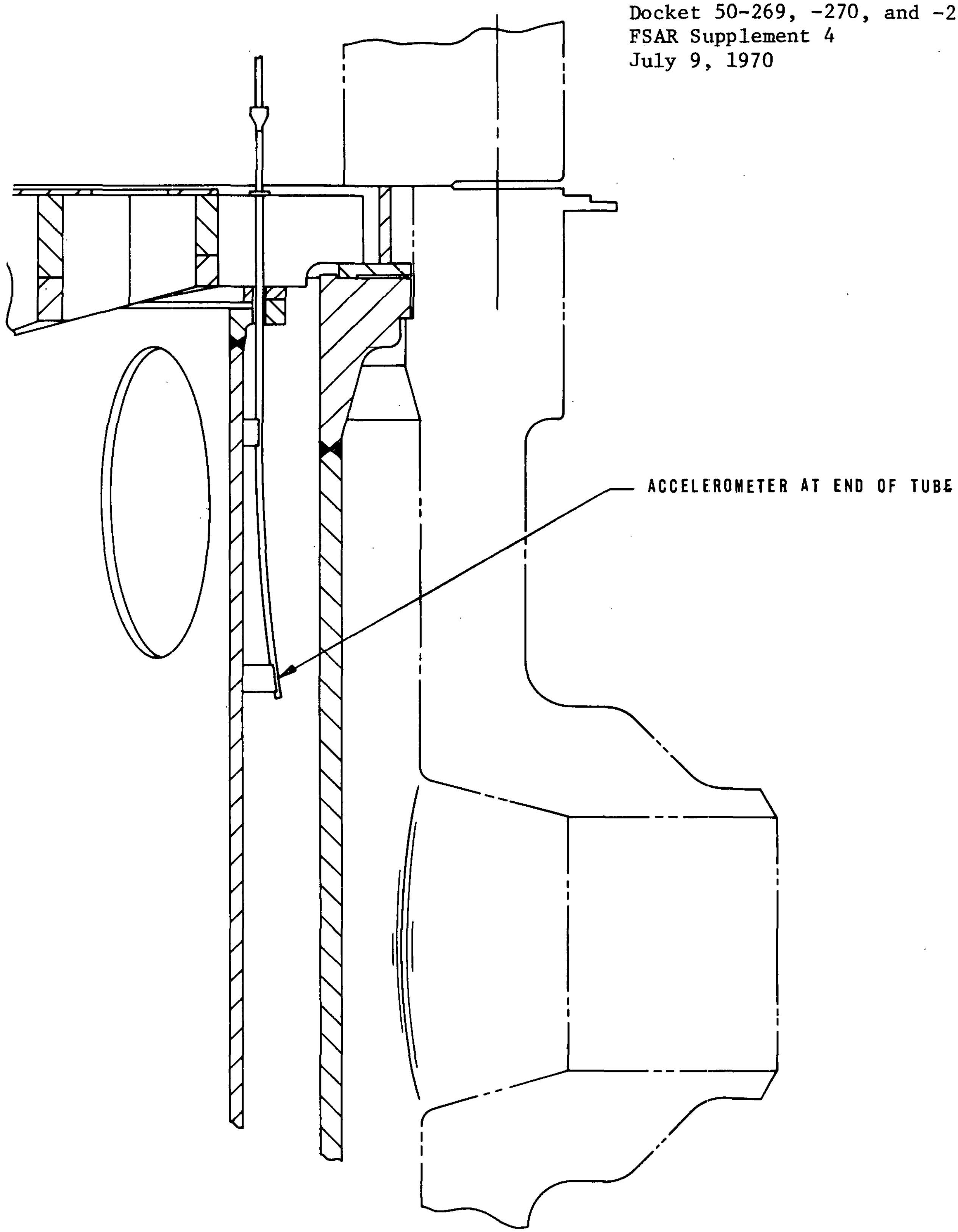
LOCATION OF INSTRUMENTED SURVEILLANCE SPECIMEN HOLDER TUBES AND THE PLENUM CYLINDER TUBES

Figure 4.11-2



LOCATION OF THE INSTRUMENTATION IN THE SPECIMEN HOLDER TUBE

Figure 4.11-3



LOCATION OF ACCELEROMETER IN THE
PLENUM CYLINDER TUBE

Figure 4.11-4

7.22 ANSWER:

Several break sizes and locations for the loss-of-coolant accident have been investigated with an assumed systematic failure of the low reactor coolant system pressure trip signal. Although this failure is not considered credible, the analysis has shown that either the void shutdown mechanism or the power/flow comparator should provide backup to shut down the reactor and render the ECCS effective. The final results of the analysis of the LOCA will be documented in a topical report on systematic failures, which is currently scheduled for submittal in July, 1970.

The following information is voluntarily submitted in response to informal questions asked by the Division of Reactor Licensing:

QUESTION 1

How would under frequency operation of the reactor coolant pumps effect the analysis of flow coastdown?

ANSWER:

The grid frequency will be monitored by Duke Power Company on a continuous basis, and automatic switch gear will remove the plant from the grid should the grid frequency reach 57 HZ. Thus it is highly improbable that the maximum coastdown accident (a four pump coastdown) could occur with a pump frequency below 57 HZ. Such a frequency reduction would result in approximately a 5 percent flow reduction. This flow reduction transient, as a result of frequency drop, was treated as the early portion of a flow coastdown. With the coastdown calculations being done for 102 percent rated power, normal coolant inlet temperature plus two degrees F, normal core pressure minus 65 PSI and a trip delay time of 0.62 seconds. The minimum DNDR thus attained differs from the minimum DNDR due to a steady state flow reduction. The result of this calculation with the 5 percent flow reduction was that the minimum DNDR reached during the transient would be 1.585, a value well above the criterion value of 1.3.

QUESTION 2

Explain how the 1.10 power-to-flow ratio results from the pump coastdown analysis.

ANSWER:

The procedure for determining the allowable power-to-flow ratio is as follows:

1. From a plot of minimum DNDR vs. time and parameters of power, find the time that yields a DNDR of 1.3 for the maximum power level of interest consistent with the number of pumps assumed operational.

2. The time found from Step 1 minus a conservative value of the sensing instrumentation delay time is the maximum coastdown time prior to trip.

3. From a plot of flow vs. time, find the minimum flow for the maximum coastdown time.

4. The maximum allowable flux-flow ratio is the maximum power level

of interest minus any error in the power level measurement divided by the minimum flow.

QUESTION 3

Justify the use of AISI C-1045 HR Steel for Reactor Building bearing plates and associated pieces.

ANSWER:

FSAR-5.6.1.1.2, Pages 5-49 and 5-50.

QUESTION 4

Supply supplemental data in support of the site meteorology with FSAR Appendix 2A.

ANSWER:

FSAR-2A.1.7, Page 2A-5, 2A-11 through 2A-73, and Figures 2A-5 and 2A-6.

QUESTION 5

Describe the demineralizers used in the condensate or secondary system.

ANSWER:

FSAR-10.2.6 and 10.2.7, Page 10-4.

QUESTION 6

Give the volume of water contained in the secondary system for normal operation, one unit, steam generator secondary side volume not included.

ANSWER:

Normal Circulating Volume	295,585 gal.
Storage Volume	
(Upper Surge Tank)	70,000
(Condensate Storage Tank)	7,500
Total	373,085 gal.

QUESTION 7

Describe the procedure followed by Duke Power Company to assure that required components are designed to withstand seismic loadings.

ANSWER:

FSAR-1C.3.4.4, Page 1C-4m.

DUKE POWER COMPANY

OCONEE NUCLEAR STATION

UNITS 1, 2, and 3

FSAR SUPPLEMENT 5

(Entire Supplement Deleted with Revision 18, 3/10/72)

Revision 18

March 10, 1972

LIST OF EFFECTIVE PAGES

FSAR SUPPLEMENT 5

Page	Revision
List of Effective Pages ...	Rev. 24
Cover Sheet Supplement 5 ..	Rev. 18

DUKE POWER COMPANY

OCONEE NUCLEAR STATION

UNITS 1, 2, and 3

APPLICATION FOR LICENSES

Dockets 50-269, -270, and -287

FSAR SUPPLEMENT 6

Submitted with FSAR Revision 9

August 11, 1970

LIST OF EFFECTIVE PAGES

FSAR SUPPLEMENT 6

DUKE POWER COMPANY
OCONEE NUCLEAR STATION
FINAL SAFETY ANALYSIS REPORT
SUPPLEMENT 6

This Supplement 6 to the Final Safety Analysis Report amends the application to incorporate a revised design reactor coolant pump into Oconee 1. The coolant pumps being installed in Oconee 1 are Westinghouse Model 93A pumps similar in design to the pumps being provided by Westinghouse on a number of stations for which construction permits have been issued. The revised Oconee 1 pumps are rated at 88,000 gpm each, resulting in the design flow of the Oconee reactors as described in the FSAR. Thus, the only changes required in the FSAR are the minor effects on partial loop operation resulting from lower expected backflow through an idle pump and the physical changes required in the plant to accommodate the revised pump.

Except as described below, all provisions of the FSAR, as heretofore amended and supplemented, remain valid for Oconee 1.

The following sections will briefly describe the significant changes to Oconee 1 resulting from the change of reactor coolant pump. The specific changes will be described and referenced to correspond to sections of the FSAR as previously amended and supplemented.

1 INTRODUCTION AND SUMMARY

Portions of Table 1-2, shown on page 1-14, are herewith changed to show revised values for the pump total developed head, and hydrostatic test pressure. These values are listed below:

Table 1-2

8. Principal Design Parameters of the Reactor Coolant Pumps

Total Developed Head, ft	350
Hydrostatic Test Pressure (cold), psig	4100

2 SITE AND ENVIRONMENT

No change.

3 REACTOR

The thermal and hydraulic analysis in Section 3 is applicable to Oconee 1 since the design reactor flow (131.32 x 10^6 lb/hr) corresponds to the design flow for the replacement pumps.

The design of the replacement pump offers greater resistance to backflow when a pump is not operating than does the original pump. Therefore, the analysis for operation with less than four (4) pumps as described in Section 3.2.3.1.1.k and 3.2.3.2.3.k is conservative for Oconee 1 with the replacement pumps. The results given in Table 3-13 as well as Figures 3-15 and 3-16 are valid for the replacement Oconee 1 pumps.

4 REACTOR COOLANT SYSTEM

The reactor coolant piping was modified slightly to accommodate the replacement pump. Both the original pump and the replacement pump are bottom suction and side discharge allowing installation of the replacement pump on the same centerlines as the original pump. The original motor will be utilized with the replacement pumps. The following specific changes to Section 4 are required for the replacement pumps:

4.1.3 CODES AND CLASSIFICATIONS

4.1.3.3 Reactor Coolant Pumps

The replacement pumps are designed and fabricated in accordance with the ASME Boiler and Pressure Vessel Code, Section III for Class A vessels through Summer 1967 Addendum. The code analysis was performed by Westinghouse and the code design report demonstrates compliance with allowable stress and fatigue limits.

4.2.2 MAJOR COMPONENTS

4.2.2.4 Reactor Coolant Piping

Figures 4-2a and 4-3a show the revised arrangement of the reactor coolant piping for Oconee 1. Two modifications are made to the original design to accommodate the replacement pump:

1. Install a 28" ID x 31" ID stainless steel transition section between the existing 28" ID coolant piping and the 31" ID pump suction.

2. Install a 28" ID small angle elbow section between the pump discharge nozzle and the reactor inlet pipe to account for the radial discharge of the replacement pump. The original pump had a tangential discharge nozzle. This elbow section is carbon steel with a section of stainless for welding to the pump casing nozzle.

The piping modifications for the replacement pump are designed so there will not be any furnace sensitized stainless steel in the pressure boundary material.

4.2.2.5 Reactor Coolant Pumps

Each reactor coolant loop contains two vertical single stage centrifugal-type pumps which employ a controlled leakage seal assembly. A cutaway view of the pump is shown in Figure 4-7c and the principal design parameters for the pumps are listed in Table 4-7a. The estimated reactor coolant pump performance characteristic is shown in Figure 4-8a.

Westinghouse pumps have operated more than 21,000 hours each at Haddam Neck and over 18,000 hours each at San Onofre. The pumps at Beznau and Zorita have operated approximately 9,000 snd 13,500 hours, respectively. Size 93 pumps have operated 5,000 to 6,000 hours at Ginna Station. No operating experience has been obtained on the new Size 93A concept. The first pumps of this configuration were shipped in May 1970. However, the fact that the rotating
10. assembly is very similar to the pumps in operation indicates that experience should be compatible with that obtained to date.

Monitoring of power input to the pumps at Haddam Neck indicates that the pumps are performing within 2-1/2 percent of theoretical values. In summary, overall experience indicates that pumps of similar design to that proposed for the Oconee 1 application have operated satisfactorily in nuclear service, and these pumps are expected to reflect this operational experience.

Reactor coolant is pumped by the impeller attached to the bottom of the rotor shaft. The coolant is drawn up through the bottom of the impeller, discharged through passages in the guide vanes and out through a discharge in the side of the casing. The motor-impeller can be removed from the casing for maintenance or inspection without removing the casing from the piping. All parts of the pumps in contact with the reactor coolant are constructed of austenitic stainless steel or equivalent corrosion resistant materials. A list of pressure containing materials is given in Table 4-9a.

The pump employs a controlled leakage seal assembly to restrict leakage along the pump shaft, as well as a secondary seal which directs the controlled leakage out of the pump, and a vapor seal which minimizes the leakage of vapor from the pump into the containment atmosphere.

A portion of the high pressure water flow from the H.P. Injection pumps is injected into the reactor coolant pump between the impeller and the controlled leakage seal. Part of the flow enters the Reactor Coolant System through a labyrinth seal in the lower pump shaft to serve as a buffer to keep reactor coolant from entering the upper portion of the pump. The remainder of the injection water flows along the drive shaft, through the controlled leakage seal, and finally out of the pump. A small amount which leaks through the secondary seal is also collected and removed from the pump.

Component cooling water is supplied to the thermal barrier cooling coil.

Table 4-7a
Reactor Coolant Pump - Design Data

Number of Pumps	4
Design Pressure/Operating Pressure, psig	2500/2185
Hydrostatic Test Pressure (cold), psig	4100
Design Temperature (casing), °F	650

Table 4-7a (Cont'd)

Operating Speed, rpm	1190
Suction Temperature, °F	554
Developed Head, ft	350
Capacity, gpm	88,000
Seal Water Injection, gpm	8
No. 1 Seal Water Leakoff, gpm	3
Pump Discharge Nozzle ID, in.	28
Pump Suction Nozzle ID, in.	31
Overall Unit Height, ft-in.	30
Weight (dry), lb	97,200
Coolant Volume, ft^3	56
Pump-motor moment of inertia, $lb\text{-}ft^2$	72,000
Injection Water Temperature, °F	125
Cooling Water Temperature, °F	105

Table 4-9a
Pressure Containing Materials
(for use in pump casing and pressure housings including main flange bolts)

Forging - Stainless Steel - SA182, 304

Static Casting - Stainless Steel - SA-351, Gr. CF8

Tubing and Pipe - Stainless Steel - SA-213, Type 316 or 304 and SA-376 or 312 (Seamless) Type 304 or 316

Bolting Material - SA-193

Welding Filler Metals - SA-298 or SA-371

Plate, Sheet and Strip - SA-240

Table 4-20 Pump Casing - Code Allowables

Table 4-21 Summary of Maximum Stresses - Casing

The designer of the replacement pumps (Westinghouse) has made a code analysis similar to Tables 4-20 and 4-21 to demonstrate compliance with code allowable stress limits.

4B STRESS ANALYSIS - REACTOR COOLANT SYSTEM

4B.2 SUMMARY AND CONCLUSIONS

The stress analysis of the Reactor Coolant System is being reviewed to confirm the adequacy of the analysis for the revised reactor coolant pump design for Oconee 1.

5 STRUCTURES

No change.

6 ENGINEERED SAFEGUARDS

16. The engineered safeguards functions are not affected by the change to the reactor coolant pumps. Because the High Pressure Injection System is also utilized for normal reactor coolant pump seal water services, there are minor modifications to the system. These modifications will be described under Section 9 of this Supplement. Figure 9-2a shows these modifications.

16.

16.

7 INSTRUMENTATION AND CONTROL

No change.

8 ELECTRICAL SYSTEMS

No change.

9 AUXILIARY AND EMERGENCY SYSTEMS

16.

The specific changes to the appropriate sections of the FSAR are as follows:

9.1 HIGH PRESSURE INJECTION SYSTEM

9.1.2 SYSTEM DESCRIPTION AND EVALUATION

9.1.2.1 Mode of Operation (Add the following paragraph)

16. When the leakage rate past the No. 1 face seal on any operating reactor coolant pump is less than 1 gpm, the isolation valve in the seal bypass line is opened allowing flow of injection water past the lower radial pump bearing for cooling and lubrication. Provision is also made for filling the No. 3 (vapor) seal standpipe from the No. 1 seal water return line to the Letdown Storage Tank.

The following information has been revised in Table 9-1:

Table 9-1
High Pressure Injection System Performance Data

Seal Flow to Each Reactor Coolant Pump (excluding makeup), gpm	8
Seal Inleakage to Reactor Coolant System per Reactor Coolant Pump, gpm	5

16.

9.3 COMPONENT COOLING SYSTEM

The following information has been revised in Table 9-7:

Table 9-7
Component Cooling System Component Data
(Component Data on a Per Unit Basis)

Component Coolers	
Type	Shell and Tube
Capacity, Btu/h	20.4×10^6
Component Cooling Water Inlet Temp, F	150
Component Cooling Water Outlet Temp, F	105
Code	ASME Section VIII

9.5 LOW PRESSURE INJECTION SYSTEM

16.

9.5.2.1 Mode of Operation

Two pumps and two coolers normally perform the decay heat cooling function for each reactor unit. The steam generators will reduce the reactor coolant temperature to approximately 250 F, and the reactor coolant system pressure will be reduced to a minimum of approximately 255 psig. This minimum pressure is required to meet the NPSH requirements of the reactor coolant pumps. When these temperature and pressure conditions are reached decay heat removal will be initiated by aligning one decay heat cooler to the reactor coolant outlet line.
10. This cooler discharges into the suction of one pump which returns the fluid to the reactor vessel. This method of alignment is required to prevent overpressurizing the decay heat cooler (design pressure = 370 psig). The former method of alignment permitted the decay heat pump to discharge through the cooler, and with this mode of operation the combined effect of the reactor coolant system pressure and the pump pressure rise would allow the cooler design pressure to be exceeded. Consequently, the decay heat removal system has been redesigned to permit this alternative method of operation while still retaining the original components. After the reactor coolant system pressure has reduced to approximately 150 psig, the system is realigned so that two pumps take suction from the reactor outlet line and discharge through two coolers. The equipment utilized for decay heat cooling is also used for low pressure injection during accident conditions.

16.

10 STEAM AND POWER CONVERSION SYSTEMS

No change.

11 RADIOACTIVE WASTE AND RADIATION PROTECTION

No change.

12 CONDUCT OF OPERATIONS

No change.

13 INITIAL TESTS AND OPERATION

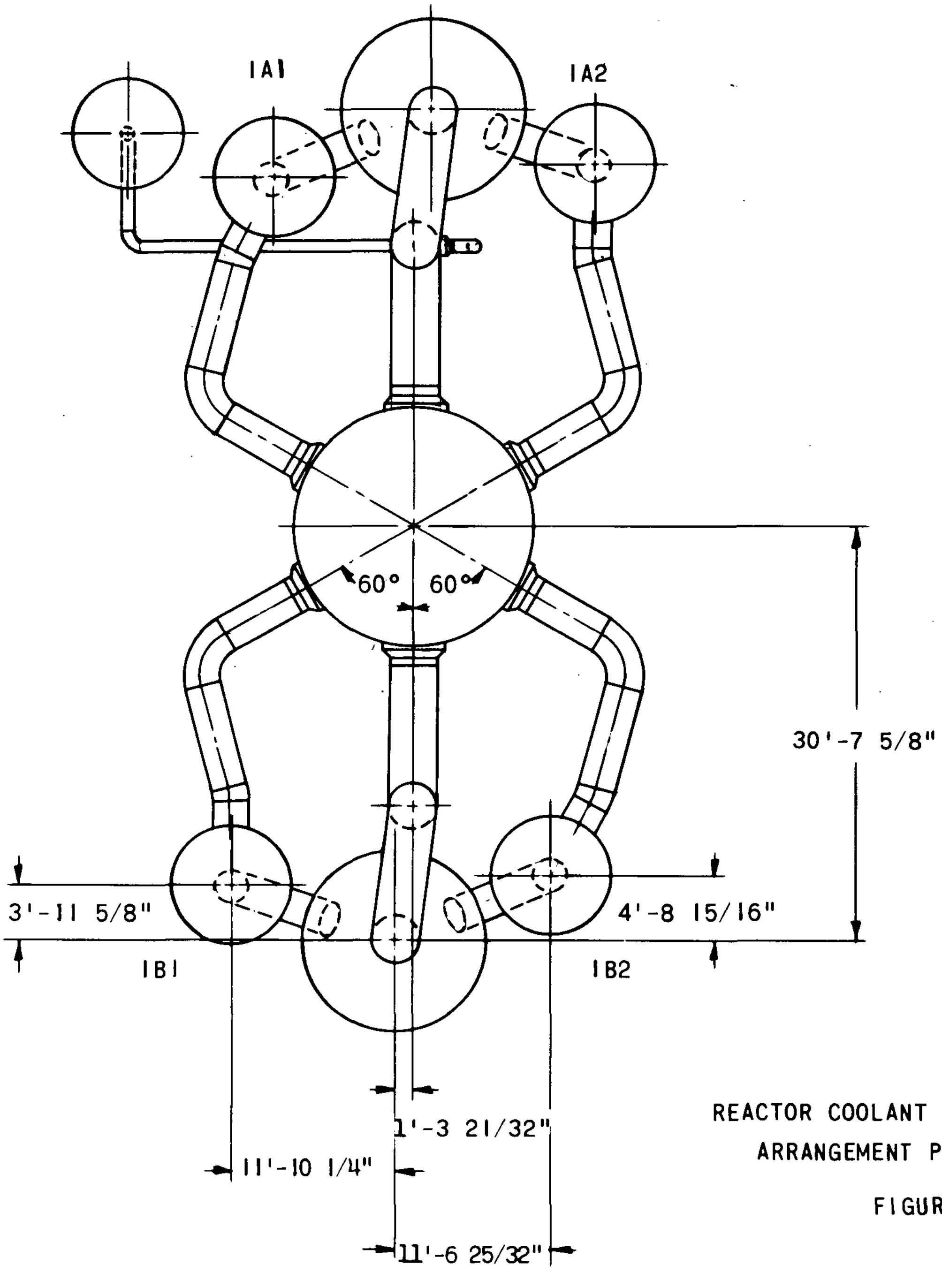
No change.

14 SAFETY ANALYSIS

No change.

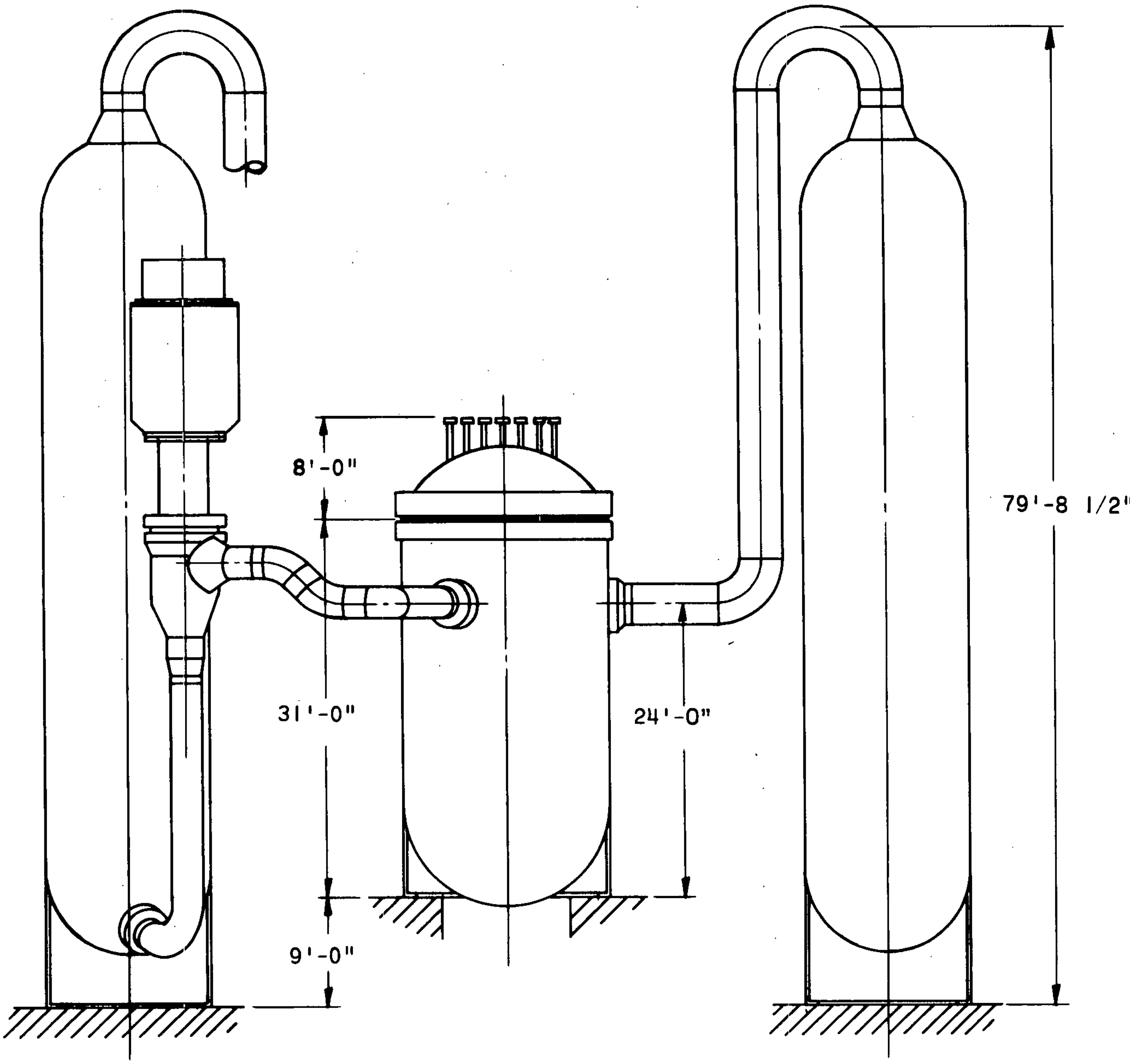
15 TECHNICAL SPECIFICATIONS

16. No change.



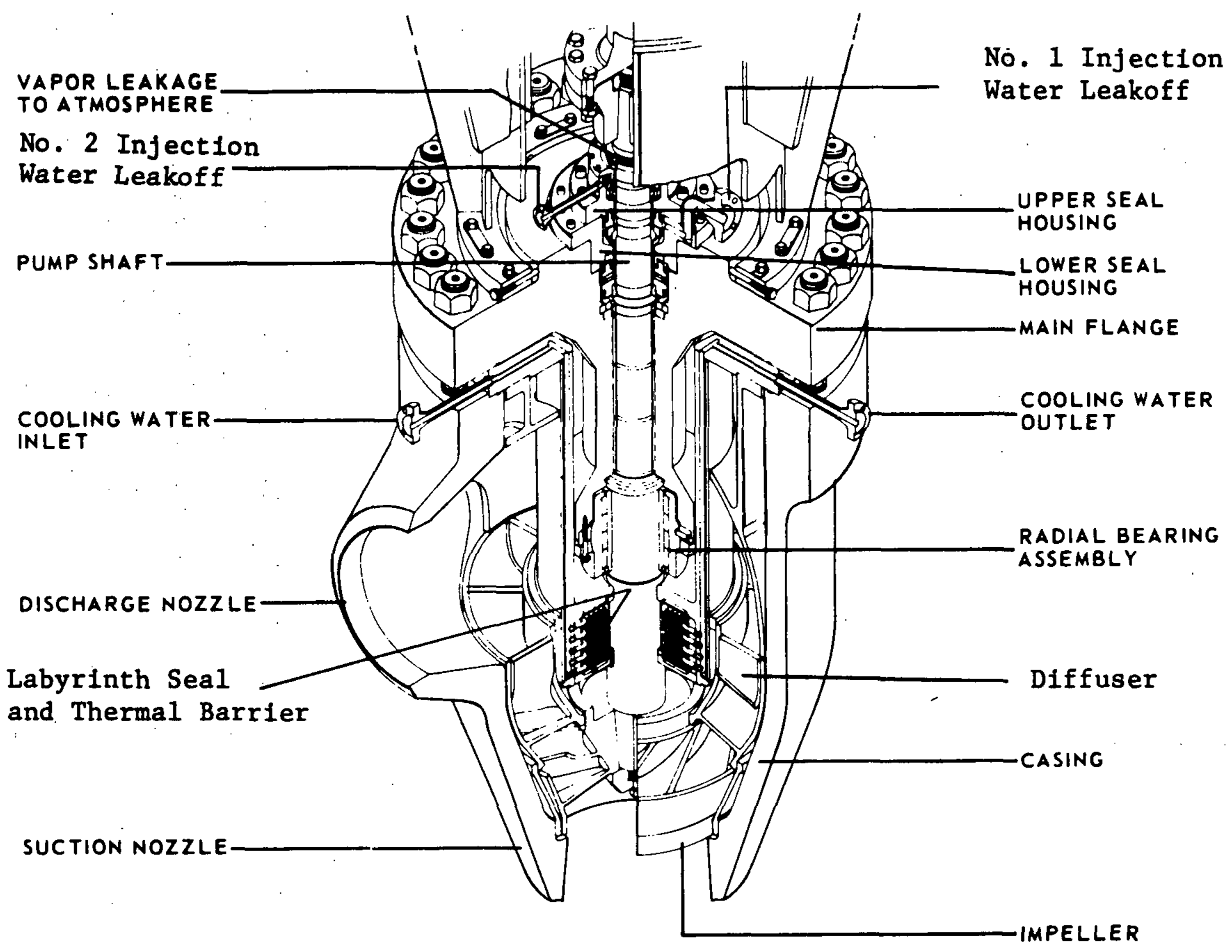
REACTOR COOLANT SYSTEM
ARRANGEMENT PLAN

FIGURE 4-2a



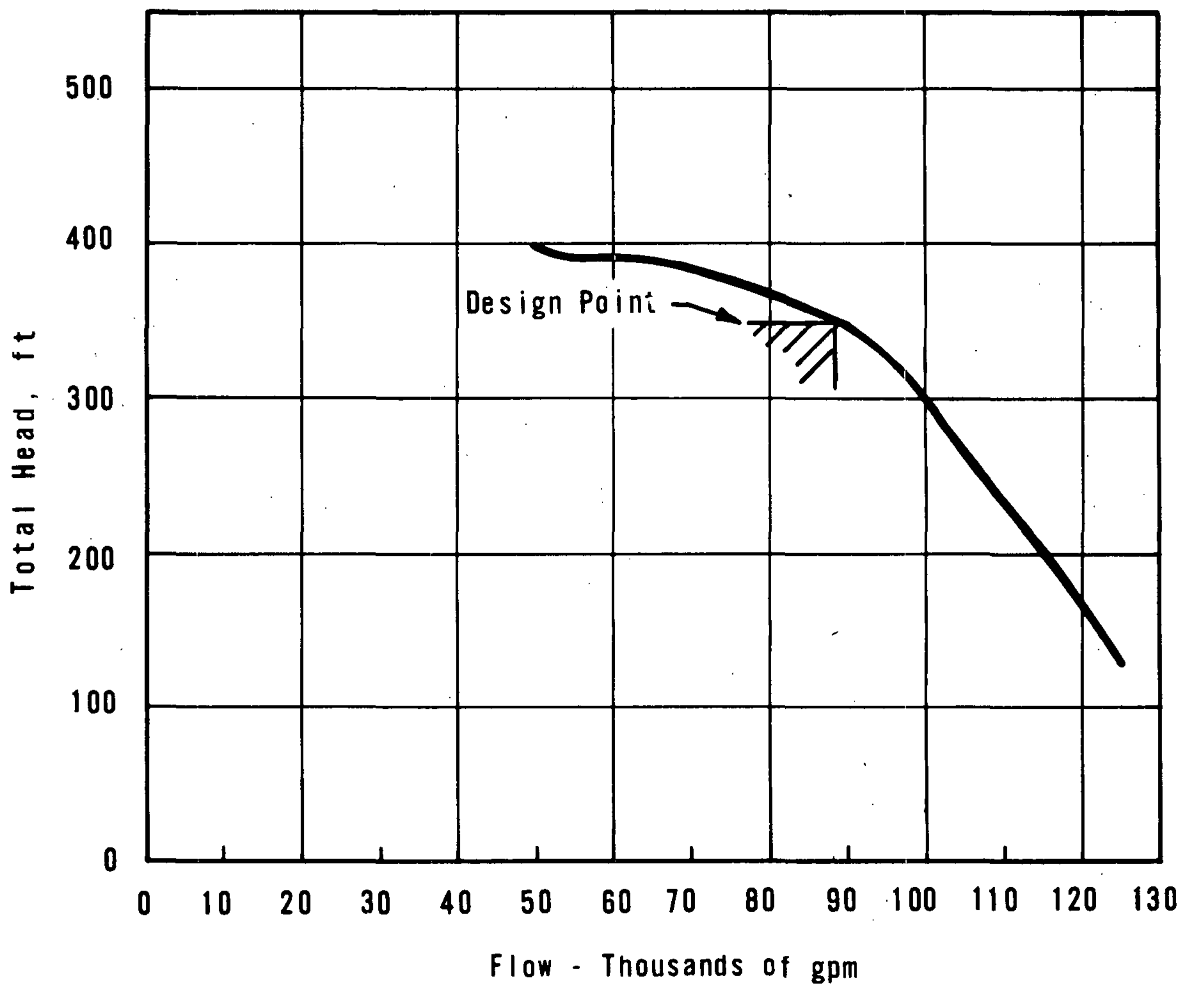
REACTOR COOLANT SYSTEM
ARRANGEMENT ELEVATION

FIGURE 4-3a



REACTOR COOLANT CONTROLLED LEAKAGE PUMP

Figure 4-7c



REACTOR COOLANT PUMP ESTIMATED
PERFORMANCE CHARACTERISTIC

Figure 4-8a

The following information is voluntarily submitted in response to informal questions asked by the Division of Reactor Licensing:

QUESTION 1. Provide a failure analysis for break of common flow instrumentation lines used in the plant.

ANSWER:

Flow Tube Failure Analysis

1.0 Failure in RC Flow Tube Instrument Piping

1.1 Reactor Coolant Flow Indication

In each primary loop, reactor coolant flow is detected by measuring the ΔP developed across a flow tube that is an integral part of the outlet piping of the loop. As illustrated in Figure 1, each flow tube has a high pressure (HP) tap and a low pressure (LP) tap. As illustrated in Figure 2, connections to the taps are made with 1-inch lines. The 1-inch lines are terminated at root valves located inside the secondary shield wall. From the root valves, 1/2-inch tubing runs through the secondary shield wall to HP and LP headers. Five ΔP transmitters are connected between the two headers. Four are used to provide flow information to the reactor protective system. The fifth is used to provide the operator with flow indication and alarms at the control console and to provide the ICS with flow information.

Each of the four reactor protective system channels receives a ΔP signal from a different one of the four ΔP transmitters. In other words, one transmitter is exclusively assigned to one protective channel. The identical arrangement and assignment of transmitters is used for each of the two primary reactor coolant loops.

Within each reactor protective system channel, the square roots of the ΔP signals from each loop are extracted to obtain loop flow signals. The loop flow signals are summed to obtain a total reactor coolant flow signal. The three flow signals are displayed within the channel's cabinets and monitored by the plant computer.

The reactor operator can read the individual loop flows and total flow at the control console, within each reactor protective channel's cabinets, and from the plant computer.

1.2 Failures Considered

The following failures are considered:

(a) Break in one of the 1-inch instrument lines.

(b) Break in one of the 1/2-inch instrument lines.

(c) A leak in one of the instrument lines.

(d) Rupture of ΔP transmitter bellows.

1.2.1 Break in 1-inch Instrument Line

A break of a 1-inch instrument line will result in a reactor trip due to low RC pressure. If the break occurs in a HP line, the reactor will trip due to a high power/flow ratio if the power/flow limit is exceeded.

The operator will receive at least the following alarms and indications:

Alarms:

(1) Break in 1" HP Instrument Line

(a) Low RC flow.

(b) Plant computer alarm and printout for low flow.

(c) Letdown storage low level.

(d) Pressurizer low level.

(e) Low reactor coolant pressure.

(f) Plant computer alarm and printout for low RC pressure.

(2) Break in a 1" LP Instrument Line

Identical alarms as listed for HP line break except RC flow is alarmed on high value.

Indication:

(1) Break in a 1" HP Instrument Line

(a) Control room indication of the reactor building atmosphere particulate and gas radioactivities increases.

(b) Loop flow indication on console falls to zero.

(c) Loop flow indication in each RPS channel falls to zero.

(d) Total flow indication on console falls approximately 50%.

(e) Total flow indication in each RPS channel falls approximately 50%.

(f) Makeup flow goes to maximum value.

(g) RC pressure falls on console indicators and within each RPS channel.

(h) Reactor building pressure and temperature indication rises.

(2) Break in a 1" LP Instrument Line

Identical indication as listed for HP line break except all loop flow indication goes full scale, total flow indication increases above normal.

1.2.2 Break in a 1/2-inch Instrument Line

A break of a 1/2-inch instrument line will result in a reactor trip due to low RC pressure. If the break occurs in a HP line, the reactor will trip due to a high power/flow ratio if the power/flow limit is exceeded.

The operator will receive the same alarms and indications as described for the 1-inch instrument line break.

1.2.3 Leak in One of the Instrument Lines

If the leak occurs in a HP line the operator will receive a low flow alarm for a 5% change in indicated flow and a high flow alarm for a similar leak in the LP line. At this alarm point, the leakage is in excess of 1 gpm, hence reactor building radiation monitors will readily detect such a condition and result in leak evaluation, and subsequent action as required by Technical Specifications.

Depending on the size of the leak, alarms and indication described in paragraph 1.2.1 may occur.

1.2.4 Rupture of ΔP Transmitter Bellows

If the bellows of a ΔP transmitter ruptures, the pressure between the HP and LP headers to which the transmitter is connected will be equalized. Since zero ΔP corresponds to zero flow, the output of all five ΔP transmitters for the affected loop will drop to zero. This will result in an immediate reactor trip if the power/flow limit is exceeded.

The operator will receive the following alarms and indication:

Alarms:

(a) Low RC flow.

(b) Plant computer alarm and printout for low flow.

Indication:

(a) Loop flow indication on console falls to zero.

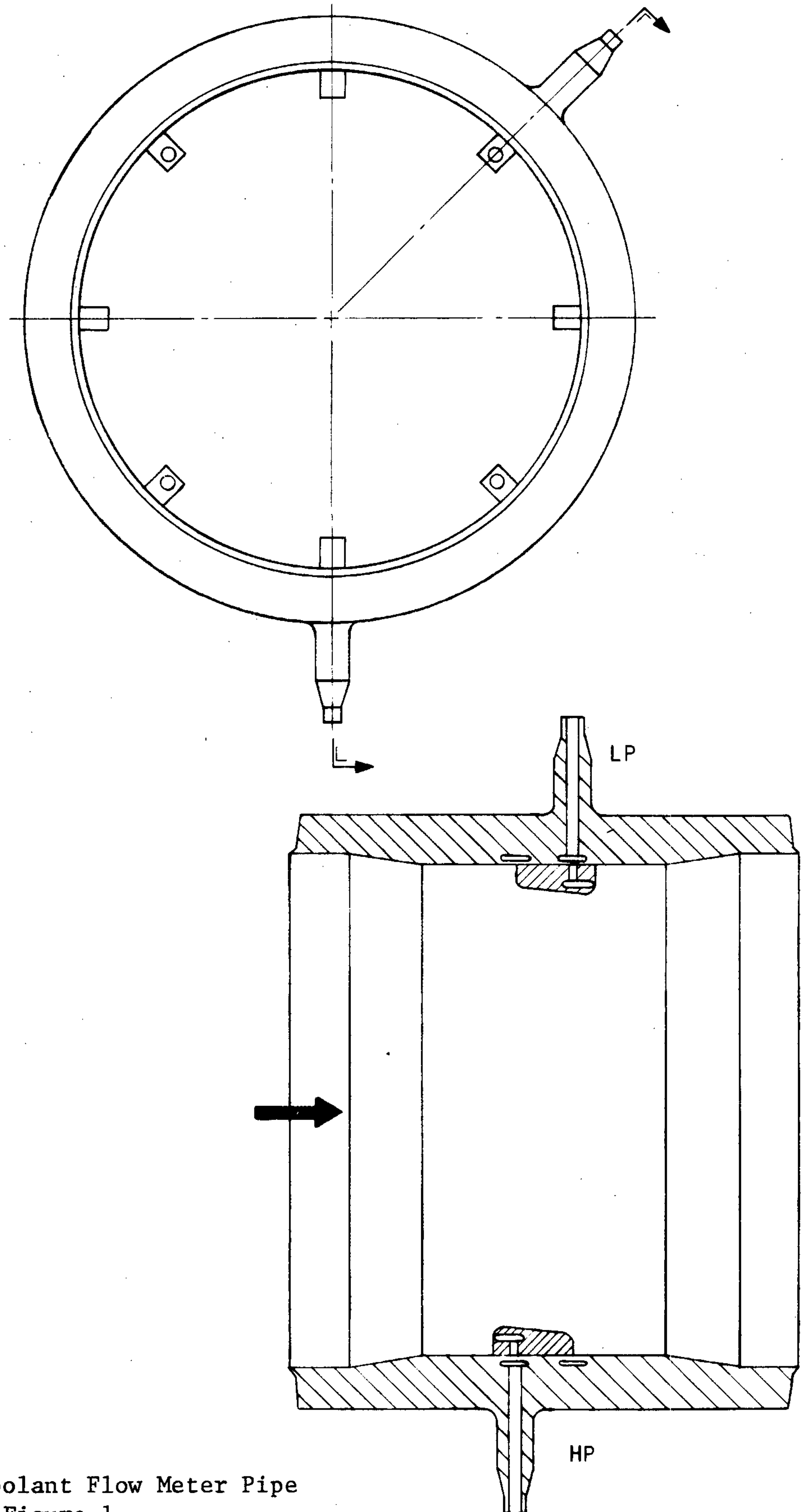
(b) Loop flow indication in each RPS channel falls to zero.

(c) Total flow indication on console falls approximately 50%.

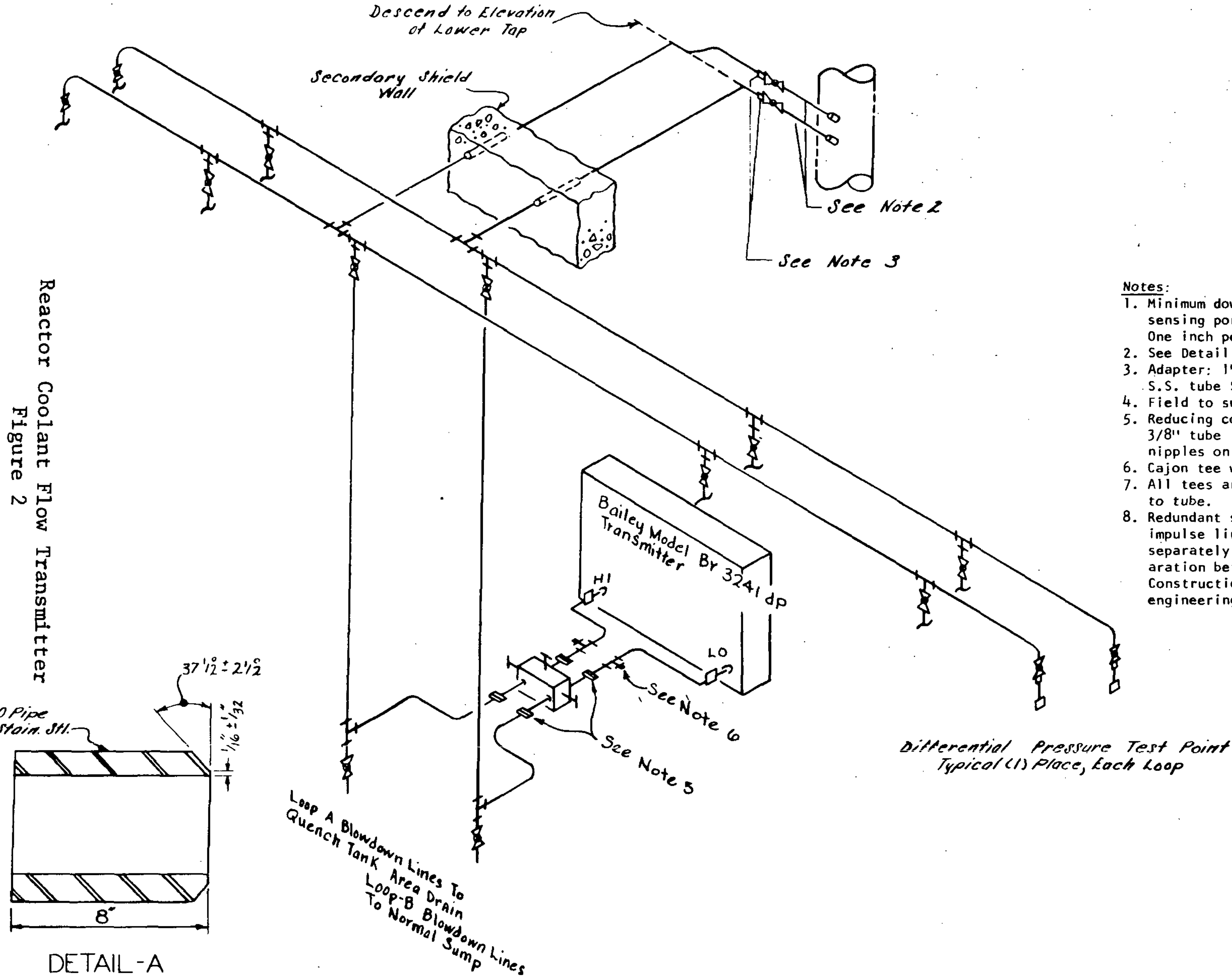
(d) Total flow indication in each RPS channel falls approximately 50%.

1.3 Conclusion

The conclusion of this analysis is that the operator has adequate indication and alarm facilities to quickly recognize a common mode failure in the flow instrumentation for the reactor protection system. Corrective action would therefore be positive and prompt.



Reactor Coolant Flow Meter Pipe
Figure 1



Notes:
1. Minimum downward slope from sensing port to instrument. One inch per foot.
2. See Detail A.
3. Adapter: 1" S.S. Pipe S.W. to ½" S.S. tube S.W. (316).
4. Field to support all lines.
5. Reducing coupling: ½" S.S. tube to 3/8" tube (for connecting to 3/8" nipples on manifold).
6. Cajon tee with plug.
7. All tees are ½" S.W. tube to tube to tube.
8. Redundant safety system instrument impulse lines are to be routed separately maintaining as much separation between them as practical. Construction to field route with close engineering department support.

Reactor Coolant Flow Transmitter
Figure 2

Docket 50-269, -270, and -287
FSAR Supplement 6
August 11, 1970

QUESTION 2.

Provide the results of a reactor building pressure analysis for the rupture of one of the steam generator feedwater headers occurring simultaneously with the worst reactor coolant system LOCA.

ANSWER:

A 5 ft^2 rupture of a hot leg pipe causes the highest pressure in the reactor building. The resulting pressure is 53.9 psig. In order to perform the simultaneous blowdown of the reactor coolant system and one of the steam generators, the following assumption was made.

Even though the pipe is not free to whip since it has only a 5 ft^2 rupture in it, the reactor coolant pipe was assumed to move into the feedwater piping, causing it to shear. This causes the blowdown of the coolant stored in the feedwater piping as well as that stored in the steam generator and the piping from the steam generator to the turbine. This combined secondary energy release is shown in Table I.

Using the same heat transfer coefficients on the steel and concrete surfaces as those used in the calculating results previously shown in the FSAR, a building pressure of 61.8 psig was obtained. For this work, it was assumed that 3 air coolers and 3000 gpm sprays were functioning. The initial heat transfer coefficient used in the FSAR was a value of 620 BTU/hr-ft^2-F on the steel until 110 BTU/ft^2 had been transferred. The heat transfer coefficient then stepped to a value of 40 BTU/hr-ft^2-F. Concrete surfaces used a constant value of 40 BTU/hr-ft^2-F.

In order to assess the conservatism in the above results and to assure that the peak pressure calculated on a realistic basis does not exceed 59 psig, a more realistic assumption of exponential decay from 620 BTU/hr-ft^2-F to 40 BTU/hr-ft^2-F was used. This assumption is described by the following relationship:

$$h_E = 40 + 580e^{-0.05(t-t^1)}$$

where t^1 is the time when 110 BTU/ft^2 has been transferred. A peak pressure of 58.8 psig was obtained. The constant value of 40 BTU/hr-ft^2-F was still used on the concrete surfaces.

Using the B&W fit to the TAGAMI data*, a pressure of 59 psig was obtained. The equations which were used for predicting the heat transfer coefficients are shown below. These equations provide a conservative fit to the data.

For

$$E/Vt_G \geq 10.0,$$

$$h_c = 24.6(E/Vt_G)^{1.2} \text{ for } 0 \leq t \leq t_G,$$

$$h_E - h_{stag} + (h_c - h_{stag})e^{-0.05(t-t_G)} \text{ for } t > t_G,$$

For

$$E/Vt_G < 10.0$$

$$h_s = 24.6(E/Vt_G)^{1.2}(t/t_G)^{0.4} \text{ for } 0 \leq t \leq t_G$$

$$h_E = h_{stag} + (h_c - h_{stag})e^{-0.05(t-t_G)} \text{ for } t > t_G$$

For both cases of E/Vt_G,

$$h_{stag} = 0.6 + 69.7(G_s/G_a) \text{ for } 0 \leq G_s/G_a \leq 2$$

$$h_{stag} = 140 \quad \text{for } G_s/G_a > 2$$

Where

t = time from beginning of blowdown, seconds,

t_G = blowdown duration**, seconds,

E = total blowdown energy, Btu,

V = containment volume, ft^3,

h_c = constant value during blowdown for steel surface, Btu/ft^2-h-°F,

h_s = parabolic-increase value during blowdown for steel surface, Btu/ft^2-h-°F,

G_s/G_a = steam-to-air weight ratio in containment atmosphere

h_E = exponential decay value after t_G, BTU/ft^2-h-°F

h_{stag} = stagnation heat transfer coefficient, Btu/ft^2-h-°F.

---

*Interim Report on Safety Assessments and Facilities Establishment Project in Japan for Period Ending June, 1965 (No. 1), Takashi Tagami, February 28, 1966, NSIC Accession No. 10701.

**For Tagami's experiments, this period ended when the pressure in the containment vessel equalled the pressure in the pressure vessel.

For concrete surface, use 40% of the values for steel surfaces.

Using the more realistic time dependent value of h, in the TAGAMI and exponential decay models, the building design pressure is not exceeded for the simultaneous rupture of the feedwater header and the 5 ft^2 hot leg pipe rupture.

Furthermore, using the heat transfer coefficient on the steel surfaces that decays exponentially from 620 to 40 BTU/hr-ft^2-F and assuming a 14.1 ft^2 pipe rupture instead of the 5.0 ft^2 rupture, occuring simultaneously with the feedwater header rupture, a peak pressure of only 58.0 psig was obtained.

Table I

Mass and Energy release rates resulting from the rupture of one of the main feedwater headers.

Time Interval (Seconds)	Mass Rate (lb_m/second)	Average Enthalpy (Btu/lb_m)	Total Mass To Reactor Building (lb_m)	Total Energy To Reactor Building (Btu x 10^{-6})
0-5	5404	528.497	27020	14.28
5-10	2846	667.604	41250	23.78
10-15	606	1201.320	44280	27.42
15-20	514	1206.226	46850	30.52
20-25	444	1202.703	49070	33.19
25-30	382	1209.424	50980	35.50
30-25	334	1197.605	52650	37.50
35-40	288	1201.389	54090	39.23
40-50	239	1200.837	56480	42.10
50-60	187	1192.513	58350	44.33
60-80	142.5	1196.491	61200	47.74
80-100	79	1183.544	62780	49.61
100-120	15	1166.667	63080	49.96

QUESTION 3.

Provide an analysis of missile protection in the area where two tendons are missing from the reactor building.

ANSWER:

FSAR - 5.6.1.1.2, pages 5-50 and 5-51, and Figure 5-23.

QUESTION 4.

Give an analysis of the damage to the fuel pool resulting from accidental dropping of the spent fuel cask.

ANSWER:

FSAR - 5.7.1.2, page 5-61a.

QUESTION 5.

Include the following information relating to the penetration room filter system:

(a) Indicate that the valves on the inlet side of the filters are normally locked open.

(b) Give the analysis that shows that there is sufficient time to restore air flow to a filter that has been cut off after it is fully loaded with fission products.

(c) State frequency at which operator will monitor filter instrumentation.

ANSWER:

(a) FSAR - Figure 6-5.

(b) FSAR - 6.4.3, page 6-29a.

10. (c) Operator will monitor filter instrumentation following the accident once every four hours.

QUESTION 6.

Provide a reliability analysis on parallel engineered safety feature buses.

ANSWER:

FSAR - 8.2.3.3, pages 8-14.

QUESTION 7.

Verify that the power supply from the 600 volt MCC's to the reactor building coolers is switched manually and that each fed from a separate bus.

ANSWER:

FSAR - 8.2.2.5, 8.2.2.6, pages 8-5, and Figures 8-3 and 8-4.

QUESTION 8.

Provide revised data on environmental testing of instruments for accident conditions.

ANSWER:

FSAR - Table 6-26, pages 6-12e and 6-12e(a); Table 7A-1, page 7A-2.

QUESTION 9.

Supplement answer to Question 4.10 of DRL's letter of February 13, 1970, which was answered in FSAR Supplement 2 relating to metal fatigue limits for core internals.

ANSWER:

FSAR - 3.2.4.1, pages 3-59.

QUESTION 10.

Provide assurance of seismic design acceptability of the hydrogen purge equipment.

ANSWER:

23. FSAR - 14A.5, page 14A-18.

QUESTION 11.

Provide assurance that low pressure service water valves outside the containment cannot fail closed.

ANSWER:

This valve is locked open as shown on FSAR Figure 9-9.

QUESTION 12.

Provide assurance that the reactor building spray system will be capacity tested prior to startup and periodically throughout the 40-year life.

ANSWER:

FSAR - Table 13-1, item 13, pages 13-24, and 15.4.2.2, pages 15-48.

QUESTION 13.

Provide assurance that the core internals will be inspected following Hot Functional Test.

ANSWER:

FSAR - 13.2.2, page 13.5

QUESTION 14

For partial pump operation at nominal conditions in the Oconee Plant, what is the associated hot channel mass velocity minimum DNBR quality at the minimum DNBR point, in channel exit quality? Also, what is the expected vessel outlet temperature?

ANSWER:

The partial pump nominal operating conditions are given in the following tables for three pumps, two pumps - two loops, and two pumps - one loop operation. The results are given for a margin of +2F on inlet temperature in the hot channel for an operating pressure of 2135 PSIA. Table 1A contains the results for all vent valves closed and the results in Table 1B assume an internal vent valve is open. The maximum design over power conditions for the pump combinations in Table 1A and 1B are given in Sections 3.2.3.1.1K and 3.2.3.2.3K respectively.

Table 1A
All Internal Vent Valves Closed

Parameter	3 Pumps	2 Pumps (2 Loops)	2 Pumps (1 Loop)
Assumed Power Condition, % of Rated Power (2568 MWt)	75.0	50.0	50.0
Reactor Coolant Flow, % of Rated Flow	74.7	49.0	45.8
Hot Channel Mass Velocity G x 10^{-6} (Lbm/hr - ft.^{2})	1.766	1.182	1.067
Hot Channel Minimum DNBR	2.50	3.37	3.00
Hot Channel Quality at Minimum DNBR Point, %	-9.20	-9.73	-3.77
Hot Channel Exit Quality, %	0.32	0.51	5.60
Reactor Vessel Outlet Temperature, °F	603.0	603.4	600.9

Table 1B
One Internal Vent Valve Open

Parameter	3 Pumps	2 Pumps (2 Loops)	2 Pumps (1 Loop)
Assumed Power Condition, % of Rated Power (2568 MWt)	75.0	50.0	50.0
Reactor Coolant Flow, % of Rated Flow	70.1	44.4	41.2
Hot Channel Mass Velocity G x 10^{-6} (Lbm/hr - ft.^{2})	1.645	1.103	1.002
Hot Channel Minimum DNBR	2.34	3.21	2.88
Hot Channel Quality at Minimum DNBR Point, %	-6.70	-8.23	-3.73
Hot Channel Exit Quality, %	2.71	2.76	7.57
Reactor Vessel Outlet Temperature, °F	602.9	603.3	600.7

QUESTION 15

Describe the pH of building spray solution following a LOCA and its effects.

ANSWER:

The nominal pH of the recirculated spray solution at BOL is 4.65. The pH of the solution is a function of the temperature, the time in the core life, and the boron concentrations assumed in the components of the emergency core coolant. The pH increases with increasing temperature, so it was assumed that the recirculated spray solution had cooled to arrive at the above value. The boron concentration in the reactor coolant system is greatest at BOL before equilibrium xenon is established; this was also accounted for. The concentration of boron in the borated water storage tank and core flood tanks was assumed to be 1800 ppm, the Technical Specification's limit. Even if an unusually high concentration of 2,000 ppm were used, the resulting pH is only 4.58. The pH as a function of time following a LOCA should be relatively constant. The only effects should tend to raise pH above this minimum value. The reactor coolant system ordinarily operates with a pH in this range, and system materials have therefore been selected for compatibiliby with such an environment. No deleterious effects on ECCS system materials as a result of pH values in the expected range are anticipated.

QUESTION 16

Does Duke intend to install filters in the ventilation discharge ducts from the Spent Fuel Building?

ANSWER:

No, however, we believe the analysis of the fuel handling accident as given in FSAR Section 14.2.2.1.2 and as amended by voluntarily submitted Question 1, Supplement 1, page 20, is conservative if a higher petition factor were used for the removal of iodine in pool water. If suitable documentation has not been provided on the public record to justify higher petition factors prior to moving irradiated fuel into the Spent Fuel Building, Duke Power Company agrees to install suitable filters for the removal of iodine in the Spent Fuel Building ventilation system.

QUESTION 17

Provide assurance that all plant revisions relating to safety will be reviewed by Duke Design Engineering prior to making the change.

ANSWER:

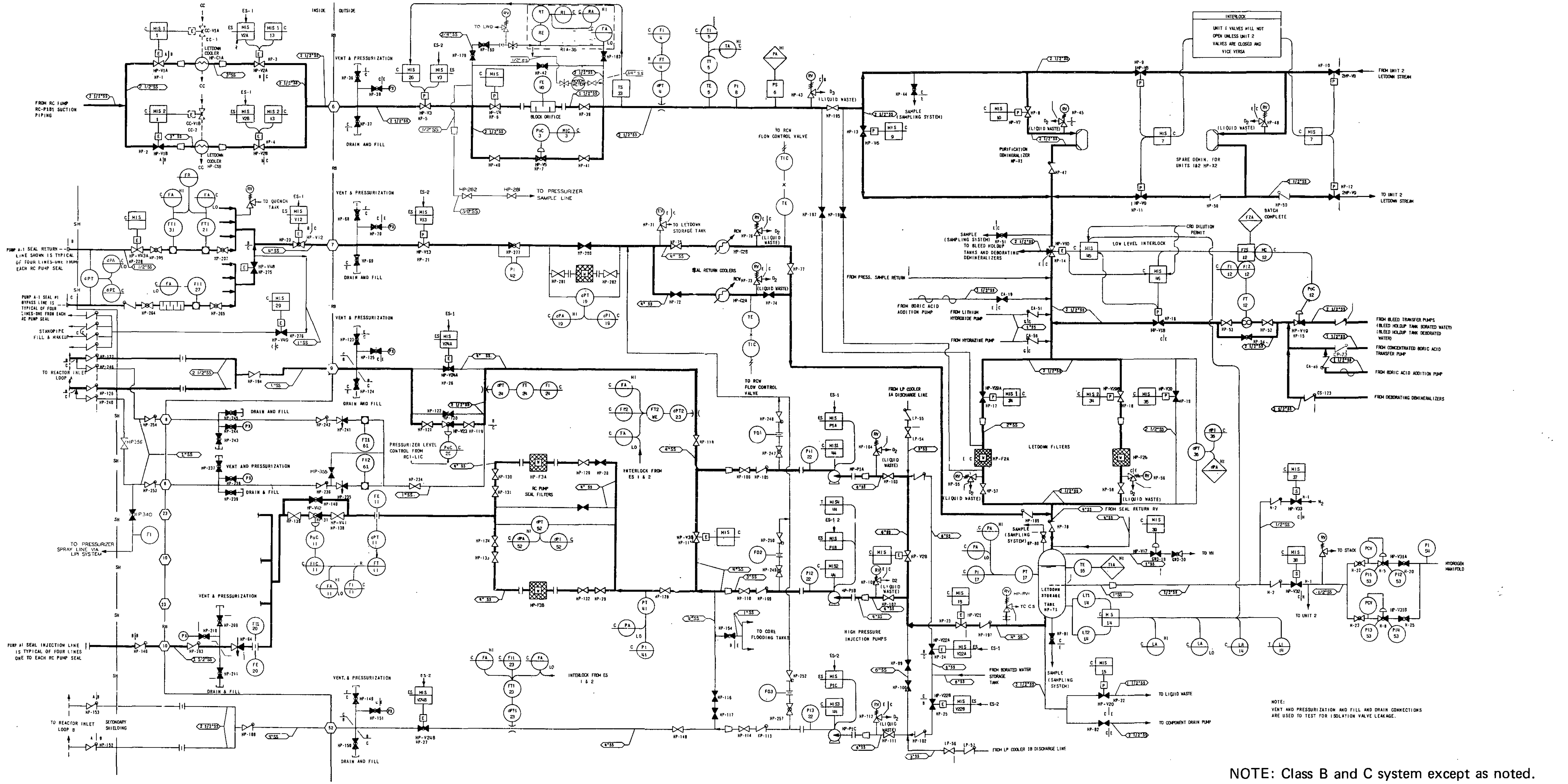
FSAR - 12.5, Page 12-11.

QUESTION 18

Provide assurance that a loose parts vibration monitor will be used during operation if a suitable one can be found.

ANSWER:

FSAR - Question 4.11, Page FSAR Supplement 4-12.



NOTE: Class B and C system except as noted.

HIGH PRESSURE INJECTION SYSTEM

(Oconee 1)

OCONEE NUCLEAR STATION

Figure 9 - 2A

(Supplement 6)

Rev. 9 8/11/71

Rev. 16 7/30/71

Rev. 18 3/10/72

DUKE POWER COMPANY

OCONEE NUCLEAR STATION

UNITS 1, 2, and 3

APPLICATION FOR LICENSES

Dockets 50-269, -270, and -287

FSAR SUPPLEMENT 7

Submitted with FSAR Revision 10

August 28, 1970

LIST OF EFFECTIVE PAGES

FSAR SUPPLEMENT 7

Page	Revision
List of Effective Pages..	Rev. 24
Cover Sheet Supplement 7.	Rev. 10
7-1	Original
7-2	Original
7-3	Original
7-4	Original
7-5	Original
7-6	Original
7-7	Original

The following is voluntarily submitted in response to informal questions asked by the Division of Reactor Licensing.

QUESTION 1

Discuss the effects of possible power peaking in the top of the core as a result of maloperation of APSR assemblies which might amplify axial peaking.

ANSWER:

The effects of axial peaking resulting from maloperation of APSRs are limited due to protection provided by a power imbalance trip. Axial power imbalance is limited by this trip to percentage values which assure that DNBR values greater than 1.3 are maintained in the core.
(Refer to 7.1.2.2.3, Table 7-1, and Fitures 7-1 and 7-2a.)

QUESTION 2

Document in the FSAR where the purge equipment will be stored. Describe the personnel precautions to provide access and hook-up of this equipment. What is the expected dose to personnel? How long are installation times? How close is the equipment stored to the charcoal filters?

ANSWER:

The mobile portions of the purge system will be securely stored in the Unit 1 east penetration room. The remainder of the system equipment consisting of permanent suction and discharge piping are located in the east penetration room of each unit.

Access and hook-up will be performed under a Radiation Work Permit which will require measurement of radiation levels and observance of working time limits, use of protective clothing and air supplied respiratory equipment.

The expected dose to personnel for hook-up is 1.3R.

Installation time within accident unit penetration room is 20 minutes.

The purge equipment is stored on El 809+3. The penetration room charcoal filters are located on El 838 which results in an elevation difference of approximately 29 feet. This elevation difference includes the floor at El 838 which consists of 10 inches of concrete.

The purge equipment storage area is 42 feet and 77 feet longitudinally, respectively from the two penetration room charcoal filters.

QUESTION 3 Document the backflow resistance of the replacement reactor coolant pumps for Unit 1 and show the resultant effects on core flow for partial pump operation.

ANSWER:

The following changes are referenced to applicable sections of the Supplement 6 description of the replacement reactor coolant pumps.

3. REACTOR

Expected Flow Rate with Westinghouse Pumps

The minimum expected flow rates shown in Table 1-1 are based on the replacement pump head capacity data which is currently available. These minimum values are based on a nominal head capacity curve.

The flow rates listed with a vent valve open are obtained as discussed in Paragraph 3.2.3.2.3 k1 of the FSAR, i.e., a constant net leakage of 4.6% is subtracted from the flow with the vent valves closed.

Pump Reverse Flow

The reverse flow coefficients (LB/hr/psi) for the Westinghouse pumps are higher than those for the Bingham pump. The absolute value of the reverse flow depends upon the system pressure drop and the head capacity curve.

Table 1-1
Comparison of Design Reactor Coolant Flow Rates
with Minimum Expected Flow Rates for Westinghouse R.C. Pumps

	Design Reactor Coolant Flow Values from FSAR Table 3-13 and Paragraph 3.2.3.2.3 k1 & k2		Minimum Expected Reactor Coolant Flow with Replacement R.C. Pumps	
R.C. Pump Combination Operating Pumps	% Rated Reactor Coolant Flow, % of 131.32 x 10^6 LB/hr	% Rated Reactor Coolant Flow, % of 131.32 x 10^6 LB/hr - One Vent Valve Open	% Rated Reactor Coolant Flow, % of 131.32 x 10^6 LB/hr	% Rated Reactor Coolant Flow, % of 131.32 x 10^6 LB/hr One Vent Valve Open
3 Pumps	74.7	70.1	75.3	70.7
2 Pumps - 2 Loops	49.0	44.4	49.8	45.2
2 Pumps - 1 Loop	45.8	41.2	46.0	41.4
1 Pump	21.9	17.4	22.2	17.6

To be consistent, the flow rates at the core inlet shown in Table 1-2 for the Westinghouse pump should be compared with the design flow rates shown in Table 1-1. The data shown in Table 1-2 should be compared with the data in Question 14.7 of Supplement 4 only on the basis of expected minimum flows for the original and replacement pumps.

Table 1-2
Reactor Coolant Flow Distribution
with Westinghouse R.C. Pumps

3 Pumps	
Flow in loop with two pumps	68.92×10^6 LB/hr
Flow in loop with one pump	29.95×10^6 LB/hr
Flow of pump in one pump loop	43.50×10^6 LB/hr
Idle pump reverse flow	13.55×10^6 LB/hr
Net reactor flow at core inlet	98.88×10^6 LB/hr
2 Pumps - 2 Loops	
Pump flow each loop	44.49×10^6 LB/hr
Steam generator flow each loop	32.67×10^6 LB/hr
Reverse flow each idle pump	11.82×10^6 LB/hr
Net reactor flow at core inlet	65.34×10^6 LB/hr
2 Pumps - 1 Loop	
Operating loop flow	71.22×10^6 LB/hr
Idle loop reverse flow	10.82×10^6 LB/hr
Net reactor flow at core inlet	60.40×10^6 LB/hr
One Pump	
Operating pump flow	45.06×10^6 LB/hr
Operating loop idle pump reverse flow	10.65×10^6 LB/hr
Idle loop reverse flow	5.23×10^6 LB/hr
Net reactor flow at core inlet	29.18×10^6 LB/hr

QUESTION 4

Give assurance that the pH of the borated water circulating in the Reactor Building following a LOCA will be controlled.

ANSWER:

Following the LOCA, there is a possibility that some small amount of chlorides may be added to the recirculated coolant. There is concern that low pH solutions containing substantial chloride concentrations could result in stress corrosion cracking of austenitic stainless steel piping.

Provisions are made in the Oconee design to obtain samples from the recirculated coolant following a LOCA to determine the pH and chloride concentration. If chlorides are present in excess of 10 ppm, the pH can readily be adjusted by adding a solution of sodium hydroxide via the Caustic Mix Tank. (Figure 9-3)

The adjustment of pH to a neutral condition will not result in generation of excess hydrogen other than what has been accounted for in Appendix 14A. The maximum additional water added to the recirculated coolant for pH control is less than 0.4 percent of the initial volume so that the boron concentration required for proper shutdown is not affected.

QUESTION 5

Describe the instrumentation used to follow the course of a LOCA and MHA.

ANSWER:

A wide range transmitter of the same type as those environmentally qualified in Tables 6-26 and 7A-1 is provided to monitor the Reactor Building pressure in the range - 15 psig to +75 psig. The transmitter is located outside the Reactor Building and is read out on a recorder and indicator in the control room.

Each line from the Reactor Building emergency sump is temperature monitored by a pneumatic temperature transmitter which indicates in the control room and a thermocouple which is read out on the computer (Reference Figure 6-9). These transmitters whose ranges are 0 to 300°F and thermocouples who ranges are 0 to 390°F are located outside the Reactor Building and are not subject to the environmental conditions of a LOCA.

Several channels of the Area Radiation Monitoring System described in Section 11.2.2 will be utilized for primary indication and backup in evaluating the extent of fission product release involved in both the LOCA and the MHA. These accidents are described in Sections 14.2.2.3 and 14.2.2.4, respectively.

RIA-4 which is located just within the Reactor Building entrance is environmentally protected against temperature, pressure, and moisture. This monitor would indicate an initial dose rate of approximately 3.5 X 10^6 mR/hr. following the LOCA but would be overranged by the MHA. In order to provide a backup to RIA-4 for the LOCA and to provide a primary indication of the MHA, RIA-15 located in the Auxiliary Building near the Reactor Building spray pump discharge lines will be used. The dose rate from the Reactor Building spray line as indicated by RIA-15 is approximately 2.0 X 10^3 mR/hr. for the LOCA and 1.3 X 10^5 mR/hr. for the MHA. RIA-1, located in the control room, would indicate an initial dose rate of approximately 2 X 10^2 mR/hr. following the MHA, providing a backup to RIA-15 in detecting and evaluating the magnitude of the MHA.

Each of these instruments has a range of 10^{-1} to 10^7 mR/hr. and therefore would provide adequate indication and backup for evaluating the extent of fission product releases to the Reactor Building (for LOCA, and from 0-100% of MHA releases according to TID 14844).

QUESTION 6

What happens if crud filled one of the two instrument lines from the flow annulus to the flow transmitter?

ANSWER:

No mechanism can be postulated which would completely block one of these lines. The reactor coolant system is a very clean system and is continuosuly filtered to assure that no significant particulate matter is circulated. The boric acid in the coolant is in concentrations about a factor of two below its solubility limit at 70 F and no precipitation would occur. The entire flow monitoring system is essentially stagnant because it is a pressure-sensitive device. There is no flow in the sensing lines to induce material into these lines. Any matter of sufficient size to block the instrument lines would have to penetrate the annulus which is of a smaller size than the instrument lines. Blockage of less than four entry ports to the annulus does not significantly impare the flow reading.

If the assumption is made that the line did become blocked, however, two possible situations would arise. The blockage of the high-pressure line would cause the average flow to appear high as flow decreases. Similarly, if the low pressure line is blocked, the average flow will appear higher than normal as flow is decreased. In both cases, the loss of one pump will not cause trip based on flux-flow if the power is constant as rated power. The results of a single pump coastdown from rated power was analyzed without trip or power runback. The minimum DNBR reached when the flow has settled to the three-pump steady state values is 1.34.

If power runback from the ICS is assumed, the reactivity added by control rod insertion is sufficient to reduce the power to 89 percent by the time the flow has reached its new value. Therefore, the hypothetical blocking of the instrument line would not cause the core thermal design limit to be exceeded as a result of the loss of one pump from rated power.

QUESTION 7

Confirm that a "back check" has been made to be sure that all purchased equipment meets the specific seismic requirements as developed by Bechtel for the applicable locations in the building.

ANSWER:

All items in Table 1C-2 have been reviewed for compliance with the latest specific seismic requirements as developed by Bechtel for the applicable locations in the plant. All equipment necessary for safe shutdown of the unit and all engineered safeguards meed these requirements.

QUESTION 8

Document the equivalent static analysis on hydro structures.

ANSWER:

FSAR - 5.7.3, Pages 6-62c and 5-62d

QUESTION 9

Provide the seismic analysis of the reactor coolant system purge line.

ANSWER:

FSAR - 4B.3.2.4, Pages 4B, 4B-6, 4B-41, 4B-43

QUESTION 10

Provide analysis associated with the stresses produced in the floor of the spent fuel pool floor from dropping the fuel cask.

ANSWER:

FSAR - 5.7.1.2, Page 5-61a

QUESTION 11

Provide analysis and times associated with the heatup of the penetration room filter following an accident.

ANSWER:

FSAR - 6.4.3, Pages 6-29a and 6-29b

Additional supplementary information in response to Question 3.8.14 contained in Dr. Peter A. Morris' letter of April 22, 1970, is given below:

ANSWER:

All bolts in the core barrel to core support shield joint are held captive by a mechanical locking device. The locking device prevents loosening by rotation and prevents displacement in the unlikely event of a bolt failure. Examination of reactor internals to assure that bolting and locking devices are intact will be in accordance with Section XI of the ASME Boiler and Pressure Vessel Code, "Rules for Inservice Inspection of Nuclear Reactor Coolant Systems."

DUKE POWER COMPANY

OCONEE NUCLEAR STATION

UNITS 1, 2, and 3

APPLICATION FOR LICENSES

Dockets 50-269, -270, and -287

FSAR SUPPLEMENT 8

Submitted with FSAR Revision 12

September 14, 1970

LIST OF EFFECTIVE PAGES

FSAR SUPPLEMENT 8

The following is voluntarily submitted in response to informal requests by the Division of Reactor Licensing.

REQUEST NO. 1

Submit additional information relating to the program for removal of reactor vessel material surveillance capsules.

RESPONSE:

Each Oconee reactor has six surveillance capsules as described in BAW-10006, "Reactor Vessel Material Surveillance Program." The six capsules allow for withdrawal and examination of four capsules from each reactor to establish the predicted NDTT shift of the reactor vessel material.

The specific time schedule for the withdrawals will be described in Section 15, Technical Specifications.

REQUEST NO. 2

What procedure will be followed to assure that the Core Flooding Tank isolation valves will not be accidentally closed while the reactor is operating at power?

RESPONSE:

The circuit breaker supplying power to the Core Flooding Tank (CFT) isolation valves will be open and tagged out under administrative control whenever the reactor is at power. Power to the starter controls comes from this same circuit breaker through a control transformer and will also be disconnected when the circuit breaker is open.

Lights are provided in the Control Room to indicate valve position (open or closed). These lights have a power supply separate from the circuit breaker serving the CFT isolation valves and are operated from limit switches on the valve operator.

Another limit switch on the valve operator will cause an annunciator alarm in the Control Room anytime a CFT isolation valve is away from the wide open position. The annunciator system has a power supply separate from that used to operate the valve or the indicating lights.

The unit computer also alarms and documents the position (open or closed) of the CFT isolation valve. The computer has a power supply separate from that used to operate the valve or the indicating lights.

REQUEST NO. 3

Submit additional information relating to the plans for recover from a major accident.

RESPONSE:

Recovery plans are described in Section 12.3.2.4.1 of the FSAR under the Emergency Plan. These call for comprehensive plans for recovery from any major accident to be formulated as a result of agreement between Duke Power Company, the AEC, and Division of Radiological Health of the South Carolina State Board of Health and the heads of local agencies concerned with public health and safety (before any re-entry or recovery is attempted). These plans will be formulated by the Superintendent of Oconee representing Duke Power Company.

Prior to that, as soon as radiological conditions permit, the site security fence gates across both approach roads would have been closed and guards posted to prevent unauthorized entry. The immediate affected area within the security fence would have been measured and delineated as a result of health physics surveys. Personnel exposure control measures within the affected area will be established by administrative control based on results of these surveys. The Emergency Plan describes the protective measures that will be taken to protect the health and safety of the public elsewhere within the Exclusion Area and beyond the Exclusion Area boundary.

Instruments and methods for measuring the magnitude of the LOCA or MHA are given in FSAR Supplement 7, Page 4, submitted August 28, 1970.

Measurements of radiological conditions within the reactor building (activity in air and water and radiation level) will be made by remote sampling and measurement before any recovery plans are made. Dose limits will be established for re-entry and recovery work and controls will also be established for protective clothing, respiratory protective equipment, radioactive waste disposal and handling of materials before this work is done.

It is also stated in Section 12.3.2.4.2 of the Emergency Plan in the FSAR that, "The principal criterion for these recovery plans is that they shall present no hazard to the health and safety of the public. Furthermore, and insofar as possible, recovery plans shall not result in exposure of the general public to doses of radiation or concentrations of radioactivity in effluents in excess of those permitted by 10CFR20 for normal operations."

REQUEST 4

Give details of a fuel surveillance program involving pre-pressurized fuel pins which expands the scope of the program shown in Paragraph 3A.5 of Appendix 3A.

RESPONSE:

21. The fuel surveillance program is described in Section 4 of the Technical Specifications.

REQUEST 5

What qualification testing has been performed on the new modules for the power/imbalance/flow trip circuitry?

RESPONSE:

Three new modules are required. These new modules are the Function Generator Module, the Power Range Test Module, and the Sum/Difference Amplifier Module.

The new modules have been tested in the same manner as the previously tested modules. The modules meet or surpass the environmental specifications of Table 7A-1 and seismic specifications of Paragraph 7A.2.3.1.2 in Appendix 7A of the FSAR. The new Power Range Test Module will replace the previously tested Power Range Test Module.

REQUEST 6

The topical report BAW-10019 on the systematic failure analysis does not cover three-pump operation. Analyze the appropriate transients with the systematic failure of the flux-flow trip for this condition.

RESPONSE:

The failure of the flux-flow trip during three-pump operation would be similar in result to the failure of the overpower trip at rated power. The consequences of transients initiated during three-pump operation would be somewhat milder, however, because the operating point is further from all of the safety limits.

The analysis has been performed in a manner similar to that reported in BAW-10019. The primary assumption is that the Flux-Flow Monitor does not function, but that the remainder of the Reactor Protection System (RPS) can be depended upon to mitigate the consequences of any abnormality that might occur. The three-pump mode of operation accentuates the consequences of failure of the Flux-Flow Monitor because the termination of many of the accidents that are assumed to occur is dependent on the Flux-Flow Monitor as the primary mode of protection. The transients that are normally terminated by the Flux-Flow Monitor have been analyzed and the results presented in Sections 1 through 5.

The reactor is assumed to be operating at normal conditions with three pumps operating and the fourth idle. The power level is assumed to be 75% of 2568 MW(t). All other plant parameters were assumed to be at their normal values; i.e., pressure, 2200 psia and average moderator temperature, 579 F. The design overpower for the three-pump mode of operation at which flux-flow trip occurs is 88.5% of 2568 MW(t); this will form a useful reference value against which to judge the results of the calculations. This overpower is based on the maximum real power attainable without causing trip by the flux-flow monitor, and does not represent a DNBR limit. It is 1.1 times the power level (75%) plus a maximum instrumentation error of 6%. The maximum allowable system pressure is 1.1 times the design pressure, or 2750 psia.

1. Startup Accident

The startup accident, or rod withdrawal from zero power, has been analyzed over a range of rod withdrawal rates from low values (3.0×10^{-5} $\Delta k/k/sec$) up to 2.15×10^{-4} $\Delta k/k/sec$. The latter value corresponds to about two times the maximum withdrawal rate expected. Figure 1 shows the peak reactor power versus ramp rate for these transients. It can be seen from this figure that no problems would be encountered due to thermal effects, since the maximum thermal power does not exceed 88.5% for any transient, even though the reactivity insertion rate may be as high as twice the maximum expected. Figure 2 shows the maximum pressure and surge flow for these startup accidents. It is seen that the pressure does not exceed a safety limit even for the maximum insertion rate.

2. Rod Withdrawal Accident

The effects of a systematic failure on the behavior of a rod withdrawal from power have been investigated. Figure 3 shows the important reactor parameters as a function of ramp rate over the range of interest. It is readily seen that no problem exists due to system pressurization. The surge flows encountered for this study are well within the capacity of the pressure relief valves. The maximum neutron power reaches 109% for the fastest rod withdrawal, so any faster ramp rate will actuate the high flux trip. This infers that the thermal power is a maximum for the 2.15×10^{-4} $\Delta k/k/s$ transient. Higher values of ramp rate will result in lower values of maximum thermal power since the thermal power lags the neutron power and hence will have less time to build to higher values. The minimum DNBR for the 2.15×10^{-4} $\Delta k/k/s$ case has been investigated, and at the time the power is a maximum the pressure is 2470 psia and the inlet temperature is approximately 556 F. This yields a DNBR of greater than 1.5. It is therefore concluded that the rod withdrawal from 75% power represents no problem even with a systematic failure of the Flux-Flow Monitor, the primary trip mode for these transients.

3. Rod Ejection Accidents

Calculations have been performed for the rod ejection accident for a variety of initial conditions. A nominal rod worth of 0.4% was used throughout the analysis. The results of the analysis are presented in Table 1. The largest pressure rise results from the low power beginning-of-core-life case, however, this transient does not exceed the capacity of the pressure relief valves at any time and hence does not approach the maximum allowable pressure of 2750 psig. This transient is tripped on high pressure and the flux-flow monitor is not required. The transients initiated from 75% were essentially equivalent in terms of thermal energy generated and neither reached the design overpower of 88.5%.

4. Idle Pump Startup

For the situation where one or more pumps are idle and a large negative moderator coefficient exists, it is possible to introduce reactivity into the core by increasing the core flow and thus reducing the average moderator temperature. This situation has been analyzed for three-pump operation and instantaneous acceleration of the fourth pump with an assumed failure of the Flux-Flow Monitor. The results of this accident proved to be quite acceptable. No secondary trip was reached for this incident. The maximum pressure rise was approximately 70 psi and the maximum neutron power reached was 96%. The maximum thermal power was about 84%. These values occurred well after the full 4-pump flow was established and therefore no safety limit is reached during this accident.

5. Loss-of-Coolant Flow

Calculations have been performed to determine the behavior of the hot channel in the event of a loss-of-coolant flow with systematic failure of the Flux-Flow Monitor. The Pump Monitor would normally trip the reactor upon loss of a second pump with the neutron power higher than 55%. However, since one pump monitor may be failed and undetected, the loss of two pumps would not necessarily be detected. For the loss of more than two pumps the pump monitor would, however, detect the situation (even with one failed monitor) and an immediate trip would result. For the purpose of this analysis, it is therefore clear that the worst coastdown that might occur with one pump idle would be a one pump coastdown. This calculation has been performed and the results are presented in Figure 4. Figure 4 shows the minimum departure from nucleate boiling ratio as a function of time after the coastdown begins. The reactor trips after approximately 37 seconds due to high outlet temperature. An allowance of 5 seconds instrument delay time is included in the transient time. The minimum DNBR reached during the transient is 1.36, which occurs just prior to the time of control rod trip.

No safety limit is exceeded for this accident even with both primary protective devices, the Flux-Flow Monitor and the Flux-Pump Monitor, assumed to be failed.

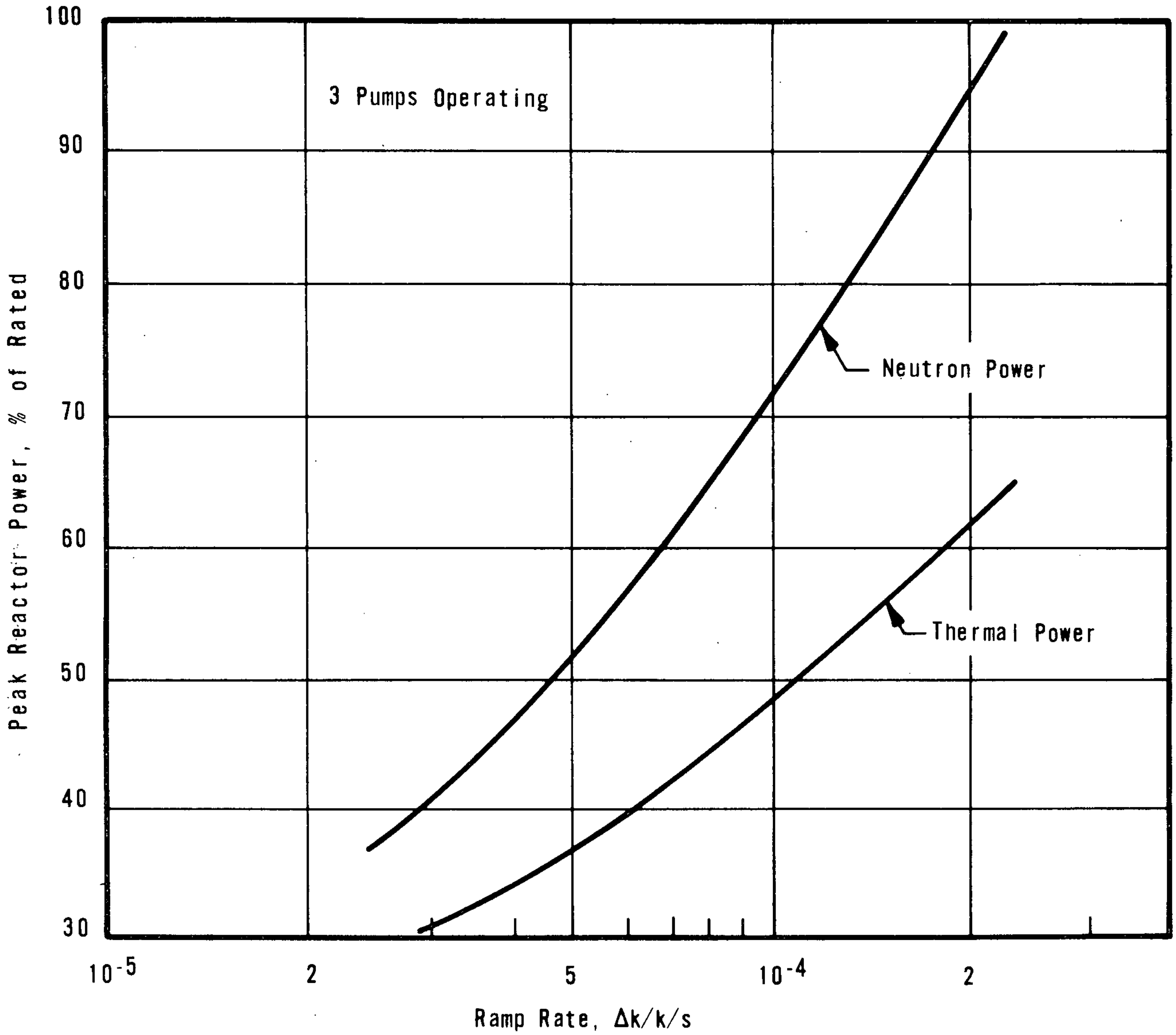
6. Summary and Conclusions

Transients that can occur with three pumps operating at 75% of rated power are no different, basically, than accidents that can occur with all pumps operating at rated power. Calculations involving these transients have been performed with an assumed systematic failure of the Flux-Flow Monitor. The results clearly show that the remainder of the RPS forms an adequate backup to the Flux-Flow Monitor.

Table 1

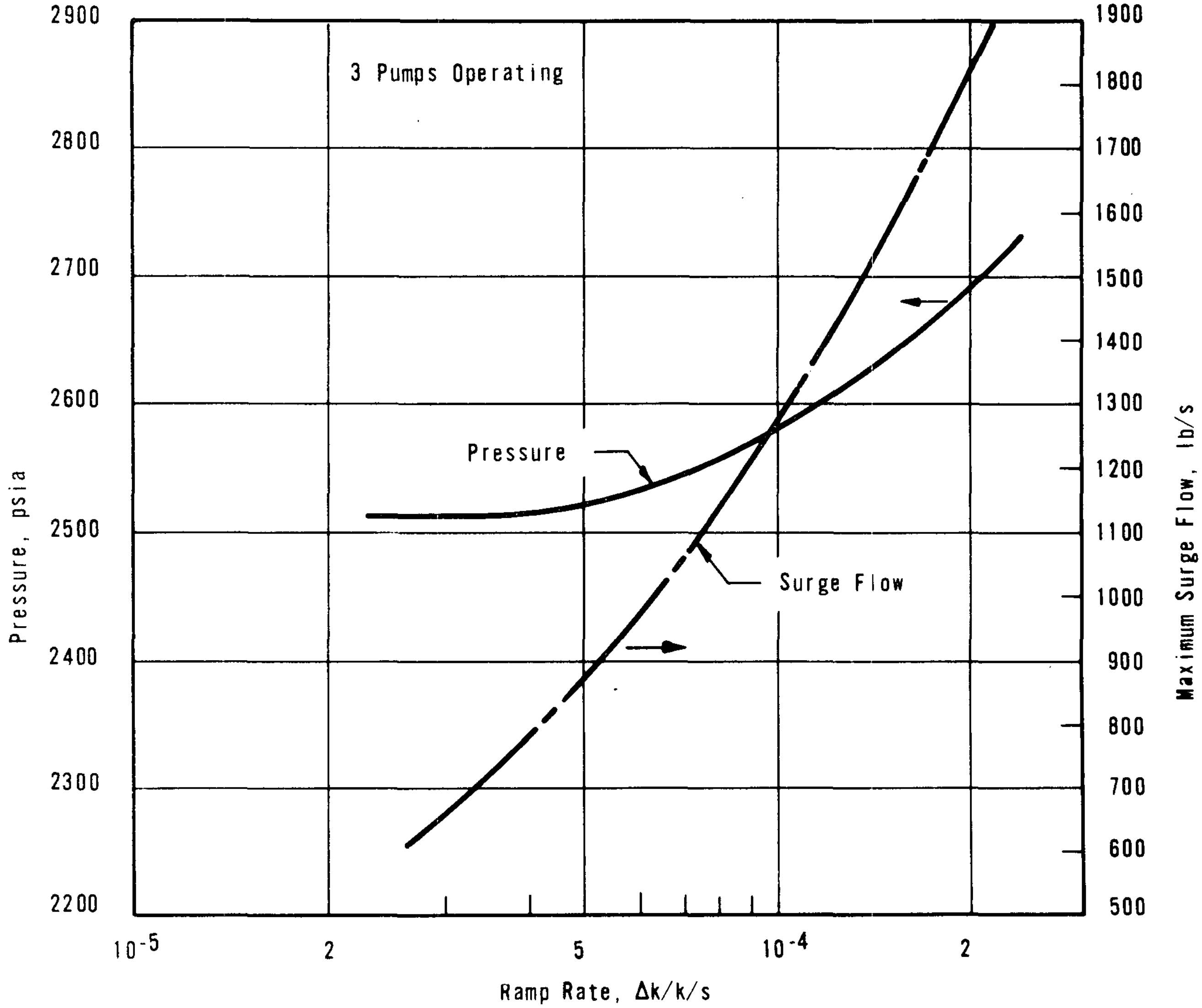
Results of Rod Ejection Accidents Assuming Systematic
Failure of Flux-Flow Monitor for Ejected Rod Worth of 0.4% $\Delta k/k$

Initial Power %	Time In Core Life	Maximum Neutron Power, %	Maximum Thermal Power, %	Maximum Pressure, psia	Type of Backup Trip
75	BOL	172.8	88.3	2291	Flux
75	EOL	265.4	88.3	2271	Flux



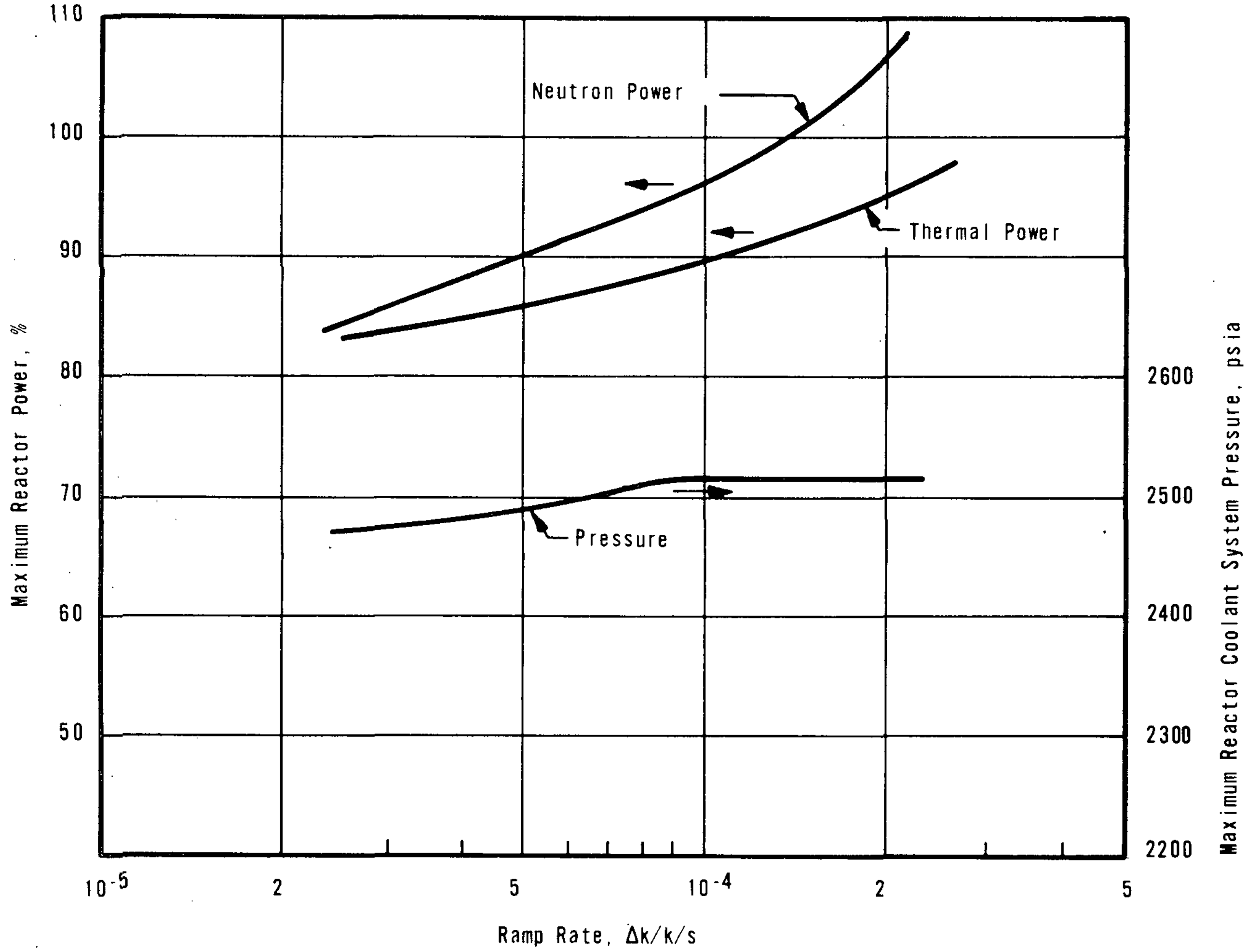
REACTOR POWER VS RAMP RATE FOR SEVERAL STARTUP ACCIDENTS DURING THREE-PUMP OPERATION WITH SYSTEMATIC FAILURE OF THE FLUX-FLOW TRIP

Figure 1



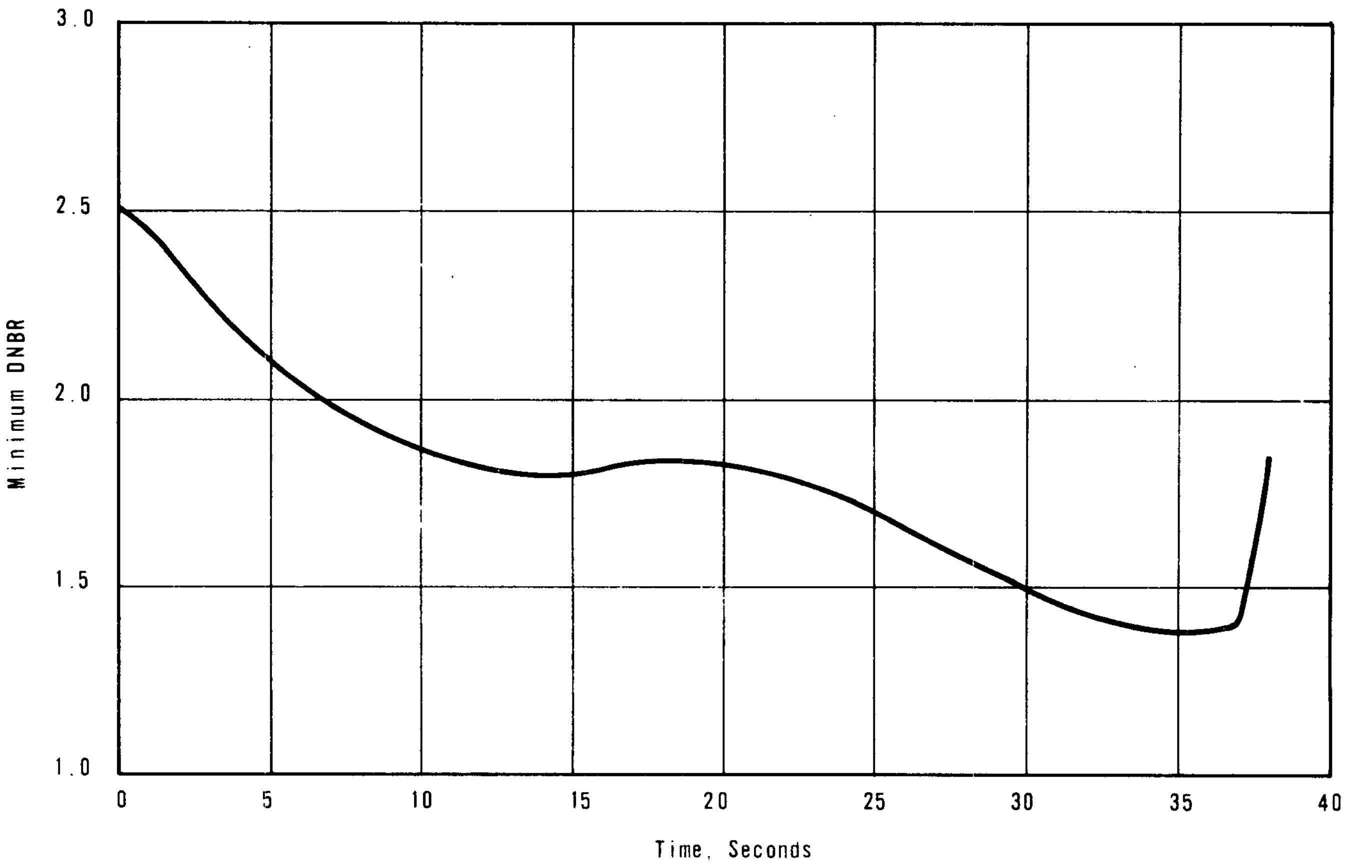
PRESSURE AND SURGE FLOW VS RAMP RATE FOR SEVERAL STARTUP ACCIDENTS DURING THREE-PUMP OPERATION WITH SYSTEMATIC FAILURE OF THE FLUX-FLOW TRIP

Figure 2



SEVERAL PLANT PARAMETERS VS RAMP RATE FOR THE ROD WITHDRAWAL FROM POWER DURING THREE-PUMP OPERATION WITH SYSTEMATIC FAILURE OF THE FLUX-FLOW TRIP

Figure 3



MINIMUM DNBR VS TIME FOR LOSS OF ONE PUMP DURING THREE-PUMP OPERATION WITH SYSTEMATIC FAILURE OF THE FLUX-FLOW MONITOR

Figure 4

REQUEST 7

Provide the results of a reactor building pressure analysis for the rupture of one of the steam generator feedwater headers occurring simultaneously with the worst reactor coolant system LOCA.

RESPONSE:

Please revise the response given in Revision 11 to the Oconee FSAR, Supplement pages 6-22, 6-22a, and 6-22b to the request noted above, as follows:

A 5 ft^2 rupture of a hot leg pipe causes the highest pressure in the reactor building. The resulting pressure is 53.9 psig. In order to perform the simultaneous blowdown of the reactor coolant system and one of the steam generators, the following assumption was made.

Even though the pipe is not free to whip since it has only a 5 ft^2 rupture in it, the reactor coolant pipe was assumed to move into the feedwater piping, causing it to shear. This causes the blowdown of the coolant stored in the feedwater piping as well as that stored in the steam generator and the piping from the steam generator to the turbine. This combined secondary energy release is shown in Table I.

Using the same heat transfer coefficients on the steel and concrete surfaces as those used in the calculating results previously shown in the FSAR, a building pressure of 62.7 psig was obtained. For this work, it was assumed that only the 3 air coolers were functioning. The initial heat transfer coefficient used in the FSAR was a value of 620 BTU/hr-ft^2-F on the steel until 110 BTU/ft^2 had been transferred. The heat transfer coefficient then stepped to a value of 40 BTU/hr-ft^2-F. Concrete surfaces used a constant value of 40 BTU/hr-ft^2-F.

In order to assess the conservatism in the above results, a more realistic assumption of exponential decay from 620 BTU/hr-ft^2-F to 40 BTU/hr-ft^2-F was used. This assumption is described by the following relationship:

$$h_E = 40 + 580e^{-0.05(t-t^1)}$$

where t^1 is the time when 110 BTU/ft^2 has been transferred. A peak pressure of 59.9 psig was obtained. The constant value of 40 BTU/hr-ft^2-F was still used on the concrete surfaces.

Using the B&W fit to the TAGAMI data*, a pressure of 59.3 psig was obtained.

The equations which were used for predicting the heat transfer coefficients are shown below. These equations provide a conservative fit to the data.

For

$$E/Vt_G \geq 10.0,$$

$$h_c = 24.6(E/Vt_G)^{1.2} \text{ for } 0 \leq t \leq t_G,$$

$$h_E - h_{stag} + (h_c - h_{stag})e^{-0.05(t-t_G)} \text{ for } t > t_G,$$

For

$$E/Vt_G < 10.0$$

$$h_s = 24.6(E/Vt_G)^{1.2}(t/t_G)^{0.4} \text{ for } 0 \leq t \leq t_G$$

$$h_E = h_{stag} + (h_c - h_{stag})e^{-0.05(t-t_G)} \text{ for } t > t_G$$

For both cases of E/Vt_G,

$$h_{stag} = 0.6 + 69.7(G_s/G_a) \text{ for } 0 \leq G_s/G_a \leq 2$$

$$h_{stag} = 140 \text{ for } G_s/G_a > 2$$

Where

t = time from beginning of blowdown, seconds,

t_G = blowdown duration**, time of first peak, sec

E = energy release at time t_G, BTU

*Interim Report on Safety Assessments and Facilities Establishment Project in Japan for Period Ending June, 1965 (No. 1), Takashi Tagami, February 28, 1966, NSIC Accession No. 10701.

**For Tagami's experiments, this period ended when the pressure in the containment vessel equalled the pressure in the pressure vessel.

V = containment volume, ft^3,

h_c = constant value during blowdown for steel surface, $Btu/ft^2\text{-}h\text{-}^oF$,

h_s = parabolic-increase value during blowdown for steel surface, $Btu/ft^2\text{-}h\text{-}^oF$,

G_s/G_a = steam-to-air weight ratio in containment atmosphere

h_E = exponential decay value after t_G, $BTU/ft^2\text{-}h\text{-}^oF$

h_{stag} = stagnation heat transfer coefficient, $Btu/ft^2\text{-}h\text{-}^oF$.

For concrete surface, use 40% of the values for steel surfaces.

Using the more realistic time dependent value of h, in the TAGAMI and exponential decay models, the building design pressure is not exceeded for the simultaneous rupture of the feedwater header and the 5 ft^2 hot leg pipe rupture.

Table I

Mass and Energy release rates resulting from the rupture of one of the main feedwater headers.

Time Interval (Seconds)	Mass Rate (lb_m/second)	Average Enthalpy (Btu/lb_m)	Total Mass To Reactor Building (lb_m)	Total Energy To Reactor Building ($Btu \times 10^{-6}$)
0-5	5404	528.497	27020	14.28
5-10	2846	667.604	41250	23.78
10-15	606	1201.320	44280	27.42
15-20	514	1206.226	46850	30.52
20-25	444	1202.703	49070	33.19
25-30	382	1209.424	50980	35.50
30-25	334	1197.605	52650	37.50
35-40	288	1201.389	54090	39.23
40-50	239	1200.837	56480	42.10
50-60	187	1192.513	58350	44.33
60-80	142.5	1196.491	61200	47.74
80-100	79	1183.544	62780	49.61
100-120	15	1166.667	63080	49.96

REQUEST 8

Give the results and conclusions of your multi-node analysis of the LOCA and of ECCS effectiveness for the Oconee Reactors.

RESPONSE:

In a letter from Dr. P. A. Morris, Director of the Division of Reactor Licensing, to Duke Power Company relating to the Oconee reactors dockets,[1] the results of a multi-node LOCA analysis were requested. The Babcock & Wilcox Company has had under active development over the past two and one-half years, a modification of the multi-node FLASH 2[2] computer code. Although the code was not operational at the time of receipt of Dr. Morris' letter, its development was accelerated, and further modifications were made as a result of initial discussions with the AEC Division of Reactor Licensing (DRL). The proprietary B&W code as revised is capable of representing up to 40 control (volume) regions describing the reactor coolant system, the steam generators, the reactor building, and the core flood tanks. Flow between control volumes is represented through appropriate connecting flow paths. The model permits multi-node representation of transient parameters within the core during LOCA, and integrates consideration of the hot channel within the core into the transient analysis. The B&W program has been described to and discussed with DRL and its consultants and preliminary results including the parameters requested in Dr. Morris' referenced letter have been informally described. Similarly, the B&W program and its capabilities and uses have been described to the ACRS Subcommittee currently reviewing the Oconee application.

(1) Ltr. of Dr. P. A. Morris to Duke Power Co., Attn. Mr. Austin C. Thies, of July 15, 1970; Dockets 50-269, 50-270, 50-287; Enclosure, Additional Information Request of July 15, 1970.

(2) Redfield, J. A., et al, FLASH 2: A FORTRAN IV Program for the Digital Simulation of a Multi-Node Reactor Plant During Loss of Coolant; WAPD-TM-666, April, 1967.

B&W has completed initial calculations with the modified FLASH 2 computer code for rupture of the reactor inlet and outlet pipes. The results of these calculations indicate maximum clad temperatures less than 2300 F in the period following pipe rupture. The highest temperature (2260F) occurs following a 6.0 square foot rupture of the 28-inch reactor inlet pipe.

A proprietary topical report will be prepared describing the details of the modified FLASH 2 program, the assumptions used in the calculations and the results of the calculations. The report will be filed with the AEC prior to fuel loading for Oconee 1.

REQUEST 9

Discuss the design considerations and the test programs planned or completed relating to possible flow-induced vibration of steam generator tubes.

RESPONSE:

Flow-induced vibrations were considered in the design of the Oconee steam generators. A series of development tests and analytical studies were factored into the final design of the steam generators.

The tube support plates in the steam generator have been spaced in an irregular pattern to prevent standing vibration waves and to create damping loads. Plucking tests and forced vibration tests show that a high degree of vibration damping does exist. The overall average of test values obtained showed a value of 5 to 6 percent of critical damping with a maximum spread of 3 to 10 percent. Flow-induced vibrations from Von Karman vortex shedding is avoided by the small amount of flow transverse to the tube axis. Cross flow exists only in the top and bottom span of the tube near the tubesheets. At the bottom span the entering velocity is less than two feet per second. This velocity is below the Reynolds Number which produces flow-induced vibration. Also, the entering fluid at the bottom of the generator is a two-phase mixture which is not likely to cause flow-induced vibration. At the top tube span, nearest the upper tubesheet the steam velocity is 55 feet per second. This results in a Von Karman vortex shedding frequency of 211 cycles per second. This frequency is almost seven times greater than the natural tube frequency. The tube support baffle holes are sized smaller than TEMA standards and the plate thickness of 1 1/2" provides lateral constraint.

To investigate the effects of tube vibrations, a forced vibration test was conducted. Ten million vibration cycles with an amplitude of 0.050 inches were imposed at the tube fundamental frequency. This test was used to compare the influence of different support plate surface finishes. The results were extrapolated to the 40 year

design life time and did not result in an unacceptable tube wall thickness. The plucking vibration test was performed to confirm the analytical method of determining a fundamental frequency with varying axial loads. The test results correlated well with the theoretical analysis. The tube vibration response for both experimental and theoretical methods is given in Figure 16 of Topical
21. Report BAW-10027; "Once-Through Steam Generator Research and Development Report". The results are reported for both dry and water-filled tube conditions.

Confirmatory testing to assure that production steam generators provide results consistent with the pluck test results noted above is planned by B&W. A production unit in the process of being tubed will be tested in the B&W Barberton shops by pluck test methods similar to those described in BAW-10027. Tubes will only be plucked in the dry
21. condition. Vibration response will be measured to assure that the production unit tube characteristics correlate well with the experimental and analytical results discussed above. It is expected that this test will be completed prior to the end of calendar year 1970, and results will be reported to the DRL staff.

Consideration has been given to possible means of pre-operational testing to monitor for tube vibration in one of the Oconee Unit 1 steam generators, but no practical and reliable method of carrying out such a program has been found. This study will continue and if such a practical and reliable method of instrumenting and testing a steam generator unit is found prior to startup of Oconee Unit 1, and if this method will not affect the vibratory characteristics of the tubes involved, such a program will be carried out on Oconee Unit 1 steam generators.

REQUEST 10

Discuss the reanalysis of the primary piping performed to show the effects of the Westinghouse pumps used for Oconee I.

RESPONSE:

A reanalysis of the reactor coolant system primary piping was performed considering the effects of the substitution of the Westinghouse pumps. The piping meets the requirements of USAS B31.7.

Only slight modifications to the primary piping configuration were required to incorporate the replacement pumps as described in Section 4.22.4 of Supplement 6. No changes were made to the 36" ID hot leg piping.

The revised piping configuration was first compared to the dynamic model originally used for the dead weight and seismic analysis of the piping described in Appendix 4B. Either pump has a large stiffness in comparison to the piping. The addition of the transition section increased the stiffness of that portion of the loop. The effect of the small angle elbow on overall seismic response would be negligible.

The masses used in the analysis were unchanged because the pump weights are approximately the same. Therefore, in view of the above and since the contribution of the seismic and dead load stresses to the total stress is small, the piping primary stress results presented in Appendix 4B are still valid.

However, the thermal stress portion of the analysis was redone for the revised cold leg piping, primarily because of the addition of the stainless steel transition section with its higher coefficient of thermal expansion. A model similar to Figures 4B-1 and 4B-2 was employed. The revised primary plus secondary stresses for the cold leg are given in Table 10-1, which is comparable to Table 4B-5b of Appendix 4B-1 and meet the requirements of USAS B31.7, 1968 edition.

TABLE 10-1

PIPE STRESS SUMMARY

(For 28-Inch Pipe)

BRANCH POINT NUMBER	BRANCH NUMBER	MAXIMUM PRIMARY PLUS SECONDARY STRESS INTENTSITY RANGE,psi	ALLOWABLE PRIMARY PLUS SECONDARY, psi
133	17	54,100.0(b)	56,100.0
123	11	53,600.0(b)	56,100.0
133	17	46,100.0	51,960.0
123	11	44,700.0	51,960.0
131	11	29,200.0	56,100.0
132	17	25,800.0	51,960.0
915	14	48,300.0	51,960.0
908	9	49,700.0	51,960.0
230	12	53,800.0	59,100.0
119	10	47,100.0	56,100.0
122	10	54,400.0 Max.	56,100.0
145	13	32,500.0(b)	56,100.0
141	17	48,200.0	56,100.0

(b) Final Stress was calculated from a detailed analysis based on Paragraph 1-705.1 of USAS B31.7.

REQUEST 11

Describe your plans for measuring the core support shield vibrations during pre-operational testing.

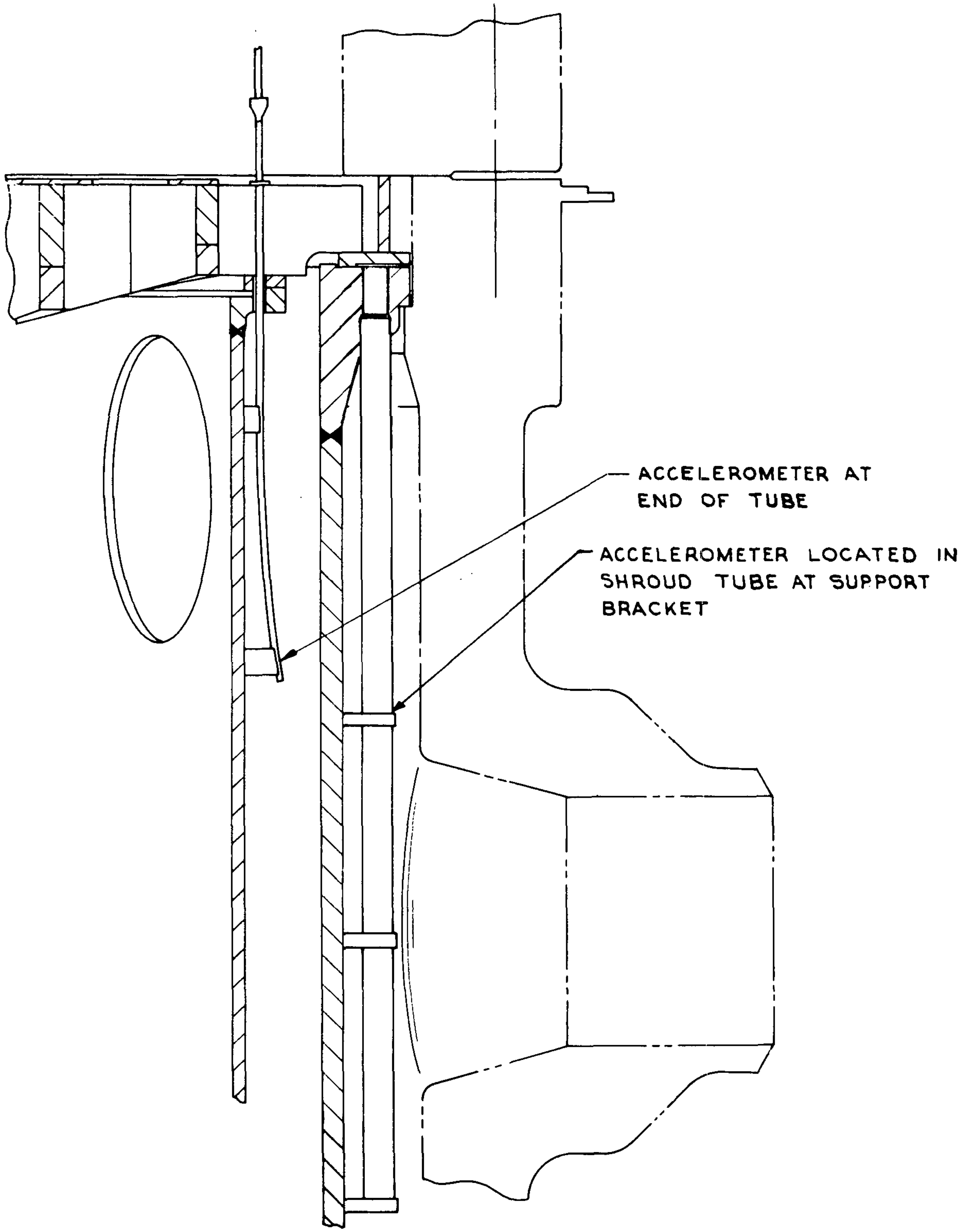
23. RESPONSE: (The response to this request has been superseded by the program given in B&W Topical Report BAW-10038)

In addition to the instrumentation described in answer to Question 4.11, one accelerometer will be located in the shroud tube which is welded to the core support shield as shown in Figure 1.

An acceptance criteria for the core support shield including a basis and model for the acceptance will be developed and reviewed with the DRL staff prior to the hot functional test for Oconee 1. The acceptance criteria will be included in the test procedure.

If the test results exceed the acceptance limits, additional analysis or vibration testing will be performed to demonstrate that no operating problems exist prior to fuel loading.

Prior to hot functional testing, a vibration test will be made on a set of internals which are essentially identical to the internals for Oconee. This test will be done when the internals are assembled in a fit-up stand. Excitation will be applied to the internals to determine the fundamental frequency of the various components which will be measured during hot functional testing.



LOCATION OF ACCELEROMETERS IN THE PLENUM CYLINDER TUBE AND CORE SUPPORT SHIELD SHROUD TUBE

FIGURE - 1

REQUEST 12

Discuss the analysis which shows that primary pipe whip will not cause failure of the secondary system.

RESPONSE:

A detailed study of the primary loop was performed to determine potential pipe break locations which could possibly cause either fluid impingement or pipe impact forces on the secondary system. The results of this evaluation indicated the most credible break locations which could cause either of these effects are:

1. a guillotine break at the pump discharge in the cold leg piping;

2. a longitudinal split in the vertical pump suction segment of the cold leg piping; or,

3. a longitudinal split in the vertical segment of the hot leg piping.

All of the above breaks could potentially affect the generator because of their prosimity to it. The main steam lines, however, are shielded from the effects of pipe breaks by the generator.

The primary piping and steam generator were analyzed for each of the above breaks and supports provided to restrain the pipe from whipping into the generator. In addition, the stresses in the generator shell due to the fluid impingement forces were calculated and found to be within acceptable limits.

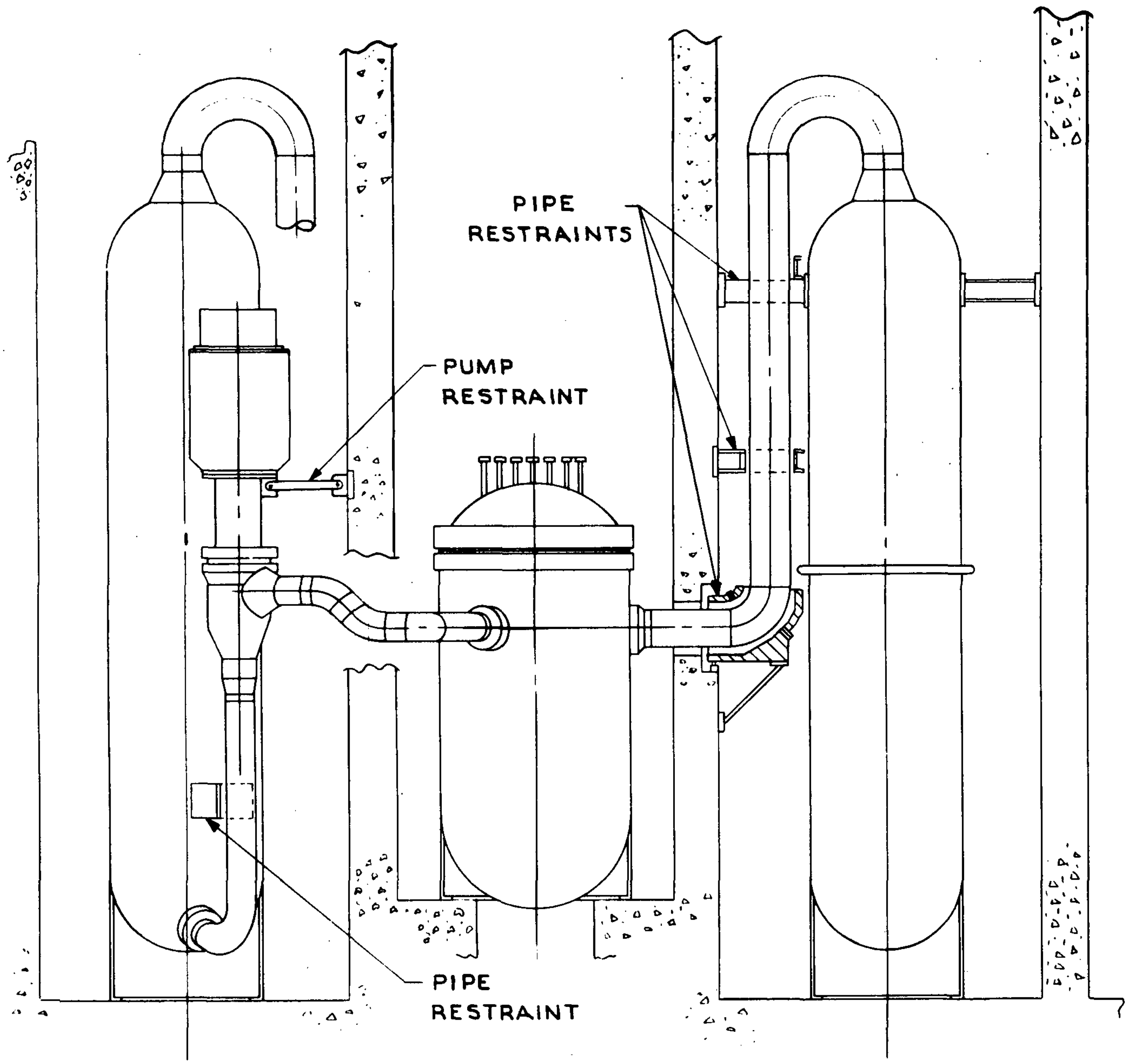
The restraints on the primary loop are shown in Figures 12-1 and 12-2. The coolant pump is restrained by steel supports from the primary shield wall. The hot leg piping is restrained by the concrete support at the primary cavity penetration, an intermediate steel support from the primary wall, and another steel support near the generator upper tube sheet. The vertical segment of the cold leg piping is restrained by a steel support midway along its length, which would spread any rupture load over a larger area of the generator shell.

To verify the location and size of the piping supports, the piping was analyzed for rupture loads occurring at the worst point along its length. The rupture thrust force was assumed equal to Px A, where P is the coolant pressure and A the flow cross-sectional area of the pipe. The thrust was applied as an equivalent static force using a dynamic load factor of 2.0. Assuming the force to be a point load acting at the midpoint of the span between supports, the piping stresses were calculated using beam models. The supports are located so as to prevent the formation of plastic hinges in the piping, which would lead to an unstable linkage-type structure and possible impacting against the generator.

To evaluate the effect of fluid jet impingement on the generator, an equivalent static pressure load on the shell was calculated. A break of 14 ft^2 for the hot leg or 8.5 ft^2 for the cold leg was assumed. The maximum initial mass velocity was computed using the methods outlined in the report "Maximum Two-Phase Vessel Blowdown From Pipes, APED-4827," by F. J. Moody. It was assumed that the fluid leaves the break in a direction normal to the pipe and that its velocity undergoes a 90° change in direction upon impinging on the OTSG. The resulting shell pressure loading was calculated to be 1300 psi.

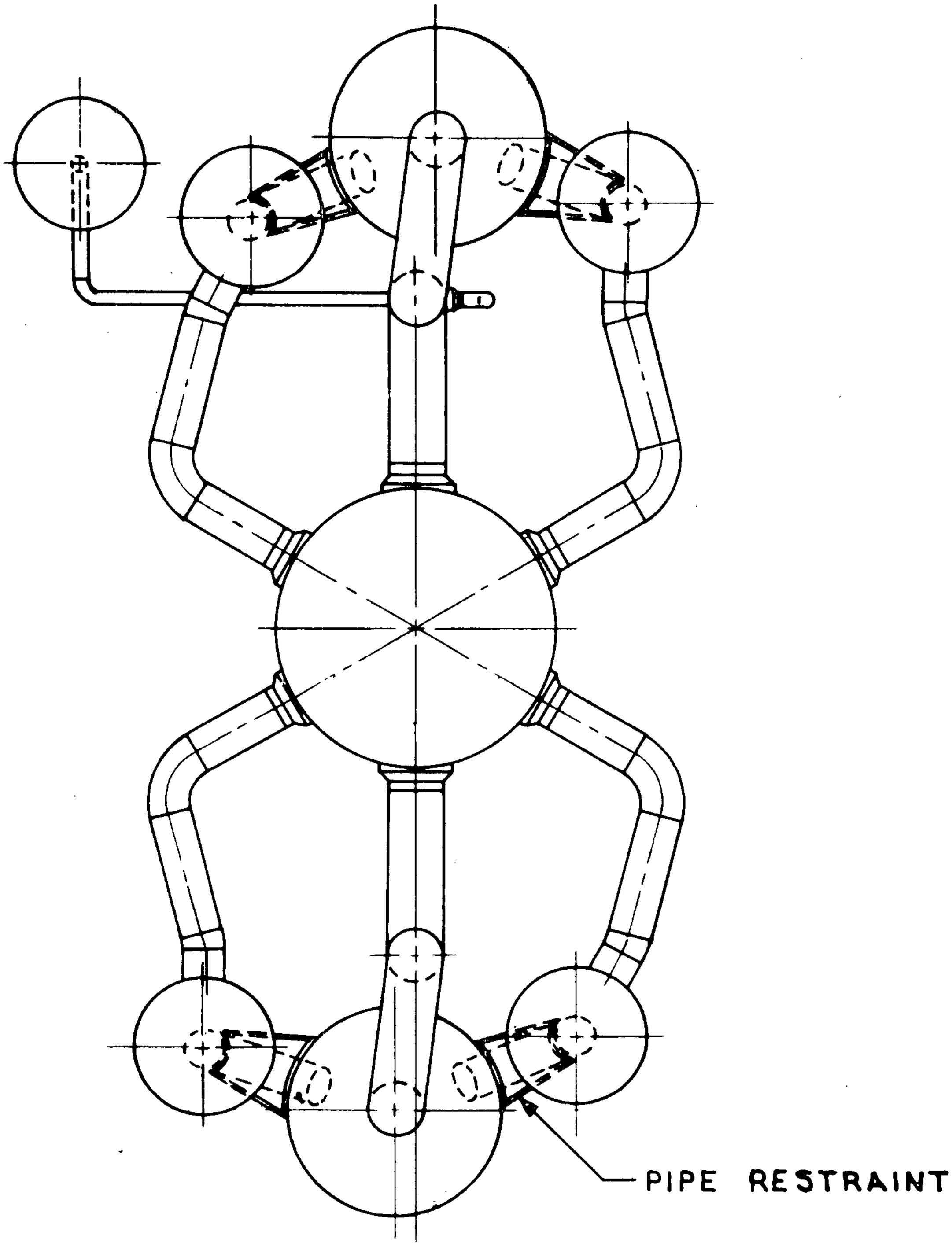
A shell analysis was performed on the OTSG to determine the stress intensity due to the above loading. A B&W proprietary digital computer code, which considers two-dimensional shells with asymmetric loading, was utilized. The loading distribution and stress model are shown in Figures 12-3 and 12-4.

The maximum stress intensity was computed to be 38,600 psi. This is less than the allowable stress of 46,670 psi. Based on these results for the 36" ID pipe break, it was concluded that the OTSG shell could also withstand the reduced loading which would be generated by a 28" ID break.



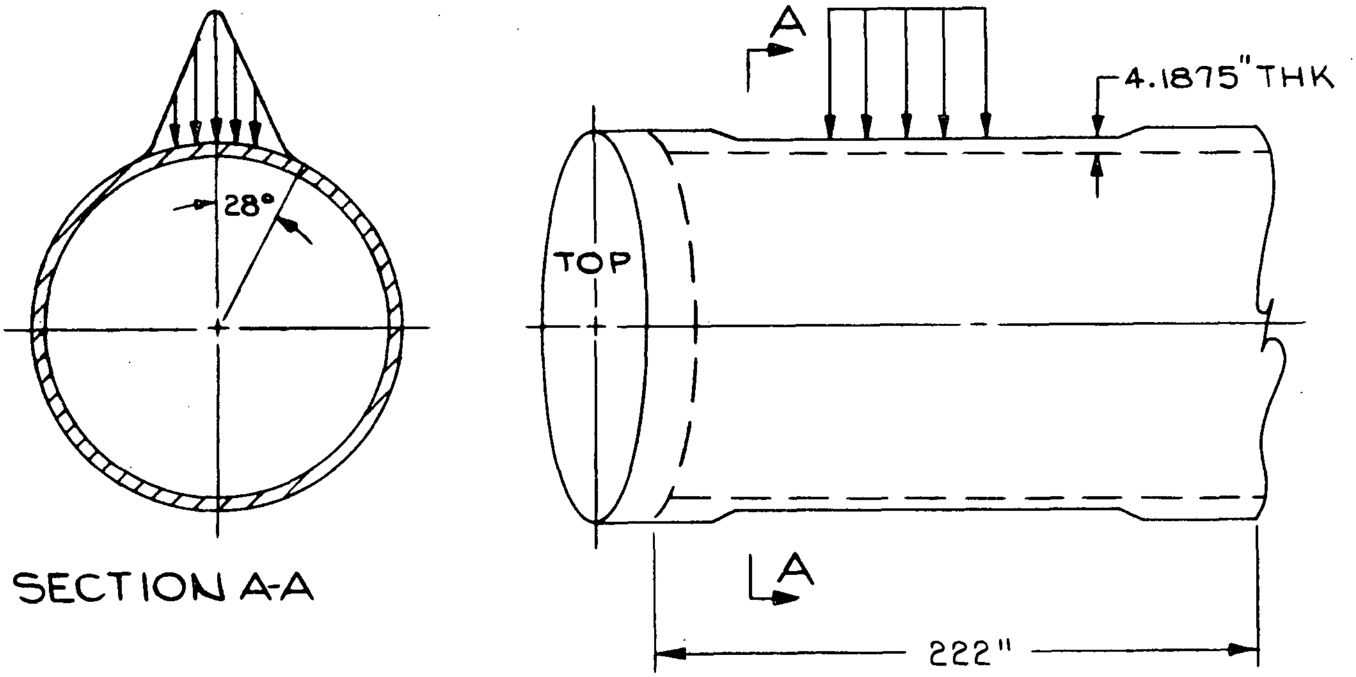
REACTOR COOLANT SYSTEM
ARRANGEMENT-ELEVATION

FIGURE 12-1

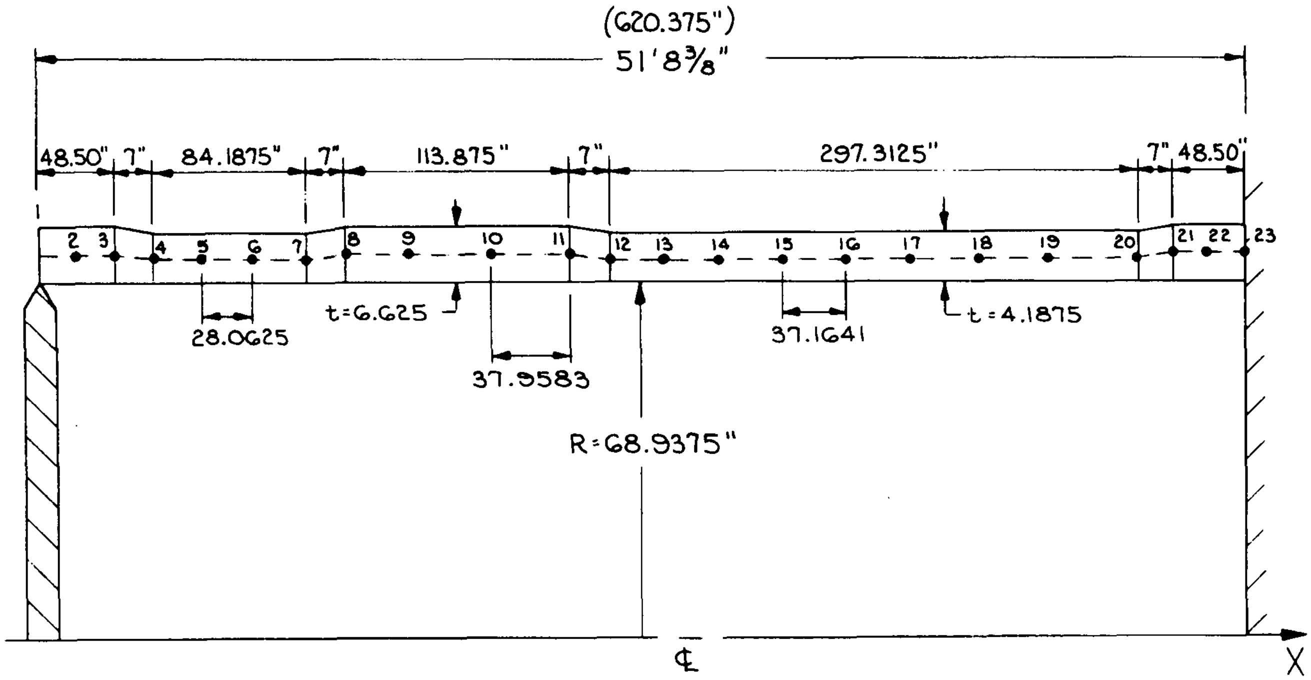


REACTOR COOLANT SYSTEM
ARRANGEMENT - PLAN

FIGURE 12-2



JET IMPINGEMENT LOAD ON THE STEAM GENERATOR
FIGURE 12-3



STRESS MODEL - STEAM GENERATOR
FIGURE 12-4

DUKE POWER COMPANY

OCONEE NUCLEAR STATION

UNITS 1, 2, and 3

APPLICATION FOR LICENSES

Dockets 50-269, -270, and -287

FSAR SUPPLEMENT 9

Submitted with FSAR Revision 16

July 30, 1971

LIST OF EFFECTIVE PAGES

FSAR SUPPLEMENT 9

Page	Revision
List of Effective Pages ...	Rev. 24
Cover Sheet Supplement 9 ..	Rev. 16
9-1	Rev. 22
9-2	Rev. 22
9-3	Rev. 21
9-4	Rev. 21
9-5	Original
9-6	Original
9-7	Original
9-8	Original
9-9	Original
9-10	Original
9-11	Original
Fig. 3-4H	Rev. 22
Fig. 3-4I	Rev. 22
Fig. 3-4J	Rev. 22
Fig. 3-4K	Rev. 22
Fig. 3-56A	Rev. 16
Fig. 4-5B	Rev. 16
Fig. 7-2B	Rev. 16
Fig. 9-6A	Rev. 18
Fig. 14-1A	Rev. 16
Fig. 14-11A	Rev. 16
Fig. 14-17C	Rev. 16
Fig. 14-29A	Rev. 16
Fig. 14-30A	Rev. 16

DUKE POWER COMPANY
OCONEE NUCLEAR STATION
FINAL SAFETY ANALYSIS REPORT
SUPPLEMENT 9

Supplement 9 to the Final Safety Analysis Report describes safety related differences in the design of Oconee 2 and/or 3 as compared to material contained in sections one through fourteen of the Final Safety Analysis Report. The following differences in design are annotated to show unit applicability and are organized to correspond to sections and paragraphs of the FSAR.

3 REACTOR

22. Units 2 and 3 are different from Unit 1 because of differences in fuel enrichments, the use of burnable poison rods and design modifications to the control rod drive assembly on Unit 3. Much of the information noting these differences is already described in the FSAR. The following specific changes are required:

Table 3-2

Nuclear Design Data

22.

Control Data	Oconee 2 and 3
Number of Burnable Poison Rod Assemblies (BPRA's)	68
BPRA Cladding Material	Zircaloy-4, cold worked
BPRA Poison Material	B_4C in Al_2O_3

3.2.2.1.3.1 Control Rod Groups for Operation

22. Figure 3-4H shows the position, function, and reactivity worth of the control rod groups for Oconee Units 2 and 3, Cycle 1, BOL. Figure 3-4i shows the same information for these units after 435 days of full power operation.

The worth of the transient control rod banks for Units 2 and 3 are as follows:

First transient bank BOL	= 0.99% $\Delta\rho$
First transient bank 250 days	= 1.35% $\Delta\rho$
Second transient bank 250 days	= 1.24% $\Delta\rho$
Second transient bank 435 days	= 1.37% $\Delta\rho$

The first transient bank BOL and the second transient bank EOL were examined for potential ejected rod worths. The calculations were performed in two dimensions (X-Y) and the APSR's were considered as full length rods inserted and withdrawn. The ejected rod worths are shown in Figures 3-4J and 3-4K.

22.

Table 3-10

Coefficients of Variation

(Values for Units 2 & 3)

CV No.	Description	Standard Deviation of Variable (σ)	Mean Value of Variable ($\bar{X}$)	Coefficient of Variation ($\sigma/\bar{X}$)
4	Local Fuel Loading			
	Oconee 2 & 3			0.00704
	Subdensity			
	Oconee 2 & 3	0.00647	0.925	0.00699
5	Average Fuel Loading			
	Oconee 2 & 3			0.00562
	Subdensity			
	Oconee 2 & 3	0.00485	0.925	0.00524
6	Local Enrichment			
	Oconee 2	0.00421	2.06	0.00204
	Oconee 3	0.00421	2.60	0.00162
7	Average Enrichment			
	Oconee 2	0.0042	2.06	0.00204
	Oconee 3	0.00421	2.60	0.00162

Enrichment values are for worst case normal assay batch;
Maximum variation occurs for minimum enrichment.

3.2.4.3 Control Rod Drives

Oconee 3 uses the Type C control rod drive mechanism in contrast to Oconee 1 and 2 which used Type A mechanisms. Both types are sealed, reluctance motor-driven screw units and the design requirements are identical. The Type C mechanism is described on Figure 3-56A. The Type C mechanism is described in
21. more detail in Topical Report BAW-10029, Control Rod Drive System Test.

Shim Safety Drive Mechanism

The Type C shim safety drive mechanism consists of a motor tube which houses a torque tube, a leadscrew, its rotor assembly, and a snubber assembly. The top end of the motor tube is closed by a closure and vent assembly. An external motor stator surrounds the motor tube (a pressure housing) and position indication switches are arranged outside the motor tube extension.

Those parts of the Type C CRDM subassemblies which are different from the Type A CRDM as described in the FSAR, are described below:

a. Motor Tube

That portion of the motor tube wall between the rotor assembly and
21. the stator is constructed of martensitic stainless steel.

b. Motor

The stator is a 48-slot four-pole arrangement with water cooling coils in the outside casing. The stator is varnish impregnated after winding to establish a sealed unit.

g. Torque Tube and Torque Taker

The torque tube is a separate tubular assembly containing a key that extends the full length of the leadscrew travel. The tube assembly is secured in elevation and against rotation at the lower end of the closure assembly by a retaining ring, keys and the insert closure. The lower end of the torque tube houses the snubber assembly and is the down stop. The leadscrew contacts the insert closure assembly for the upper mechanical stop.

The torque taker assembly consists of the position indicator per-anent magnet, the snubber piston and a positioning keyway. The torque taker assembly is attached to the top of the leadscrew and has a keyway that mates with the key in the torque tube to provide both radial and tangential positioning of the leadscrew.

h. Snubber Assembly

The total snubber assembly is composed of a piston that is the lower end of the torque taker assembly and a snubber cylinder and belleville spring assembly which is attached to the lower end of the torque tube. The snubber cylinder is closed at the bottom by the snubber bushing and leadscrew. The snubber cylinder has a twelve-inch active length in which the free-fall tripped leadscrew and control rod assembly is decelerated without applying greater than ten times gravitational force on the control rod. The damping characteristics of the snubber is determined by the size and position of a number of holes in the snubber cylinder wall and the leakage at the snubber piston and bushing. Leakage reduction at the snubber piston and bushing can only be reduced to a minimum amount caused by practical operating clearances. Therefore, at the end of the snubbing stroke, there is kinetic energy from a five foot per second impact velocity that is absorbed by the belleville spring assembly by a slight instantaneous overtravel past the normal down stop.

3.3.3.1 Prototype Testing

The Type C prototype drive mechanism was tested at Diamond Power Specialty Corporation, Lancaster, Ohio. This consisted of component testing, a 100% misalignment life test (equivalent to 20 year operation), and motor performance tests. Throughout these tests the drive components were examined for material fretting, wear and vibrational fatigue. The test results are described in
21. Topical Report BAW-10029, Control Rod Drive System Test.

3.3.3.4.1 Control Rod Drive Developmental Tests

The Type C prototype drive was tested at the Diamond Power Specialty Corporation, Lancaster, Ohio. The test program results are described in Topical Report BAW-
21. 10029, Control Rod Drive System Test.

4 REACTOR COOLANT SYSTEM

Unit 3 once-through steam generator has an internal auxiliary feedwater header, a reduced number of handholes, and minor material changes. The following specific changes are required:

4.2.2.2 Steam Generator

Unit 3 steam generator general arrangement with internal auxiliary feedwater header is shown in Figure 4-5B.

Table 4-4

Steam Generator Design Data
(Unit 3 Only)

Function	No.	ID, in.	Material
Auxiliary Feedwater Connection	1	6, Sch 80	Carbon Steel
Handholes	9	5	Carbon Steel

6 ENGINEERED SAFEGUARDS

The Unit 3 Low Pressure Injection System is different from that described in the FSAR because some components have been upgraded in pressure rating, and consequently provision for taking pump suction from the cooler discharge is not required.

These modifications will be described in Section 9 of this supplement. The following specific changes are required in Section 6 for Unit 3:

Table 6-6
Engineered Safeguards Piping Design Conditions
(Oconee 3 Only)

Low Pressure Injection System	Temp. (°F)	Press. (psig)
a. From the borated water storage tank to upstream of the borated water storage tank outlet valves.	150	Static
b. From upstream of the borated water storage tank outlet valve to upstream of the check valves in the borated water feed lines.	200	100

Table 6-6 (Cont'd)

Low Pressure Injection System	Temp. (°F)	Press. (psig)
c. From upstream of the check valves to upstream of the motor operated valves in the borated water feed lines.	300	200
d. From upstream of the electric motor operated valves in the borated water feed lines to upstream of the valves at the pump inlets.	300	388
e. From upstream of the system inlet valves at the pump inlets to upstream of the reactor building isolation valves.	300/250	470/505
f. From upstream of the system inlet valves to upstream of the check valves in the core flooding lines.	300	2,500
g. From upstream of the check valves in the core flooding lines to the reactor vessel.	650	2,500
h. From the reactor building emergency sump to upstream of the valves in the recirculation lines.	300	59

The following data has been revised in Table 6-9.

Table 6-9
Leakage Quantities to Auxiliary Building
(Oconee 3 only)

		Estimated Quantities	
Leakage Source	No. of Sources	Leakage Per Source (drops/min.)	Total Leakage (cc/h)
Valve Seats at Boundaries		(**)	950
			2,054

(**) Assuming 10 cc/h/in. of nominal disc diameter.

7 INSTRUMENTATION AND CONTROL

The instrumentation and control design is identical for all three units; however, use of the Type C CRDM on Unit 3 results in a change in one of the overpower trip set-points. This difference requires the following specific changes:

7.1.2.2.2 Summary of Protective Functions

Table 7-1
Reactor Trip Summary (Unit 3 Only)

Trip Variable	No. of Sensors	Steady State Normal Range	Trip Value or Condition for Trip
Nuclear Overpower Based on Flow and Imbalance	4 Two-Section Flux Sensors 8 ΔP Flow	NA	1.08 times flow minus reduction due to imbalance

7.1.2.2.3 Description of Protective Channel Functions

2. Overpower Trip Based on Flow and Imbalance

   The use of the Type C CRDM on Unit 3 requires a reduction in the overpower trip to a value of 1.08. This change results in new power - imbalance boundaries for Unit 3 as shown in Figure 7-2B.

9 AUXILIARY AND EMERGENCY SYSTEMS

9.5 Low Pressure Injection System

The decay heat removal coolers of Unit 3 have been redesigned and now have a higher pressure-temperature rating than that described in the FSAR. Provision is no longer made to take pump suction from the cooler discharge, and remotely operated bypass flow control valve has been added to each cooler for improved temperature control. The Unit 3 Low Pressure Injection System is shown on Figure 9-6A.

The following specific change is required to Section 9:

Table 9-11
Low Pressure Injection System Component Data
(Oconee 3 Only)

Cooler (each)	
Type	Shell and tube
Capacity (at 140 F), Btu/h	30×10^6
Reactor Coolant Flow, gpm	3,000
Low Pressure Service Water Flow, gpm	3,000
Low Pressure Service Water Inlet Temp, F	75
Material, Shell/Tube	CS/SS
Shell Design Pressure, psig	150
Tube Design Pressure, psig	470/505
Shell Design Temperature, F	300
Tube Design Temperature, F	300/250
Code	ASME Section III-C III and VIII

14 SAFETY ANALYSIS

The only safety-related difference among Units 1, 2 and 3 which affects the plant safety analysis is the longer trip delay time of the Type C control rod drive mechanism which is used in Unit 3. The effect of this on the trip delay time is as follows:

Trip Parameter	Type A Trip Delay time, sec	Type C Trip Delay time, sec
Power-to-Flow	0.65	0.75
Pump Monitors	0.62*	0.62*
Overpower	0.30	0.40
Pressure	0.50	0.60
Temperature	5.00*	5.00*

A time to 2/3 insertion of 1.4 seconds is used for both the Type A and Type C mechanism.

The items affected by the longer trip delay time are:

1. Power-to-Flow Trip ratio

2. Accident analyses involving a reactor trip

14.1.2.6 Loss-of-Coolant Flow

The longer trip delay time requires a lower power-to-flow trip ratio value to limit the transient DNBR to the same value. Therefore, the power-to-flow trip ratio for Unit 3 is 1.08 while the power-to-flow trip ratio for Units 1 and 2 is 1.10. The only accident analysis affected by the new power-to-flow trip ratio is the locked rotor analysis. Figure 14-17C shows the locked rotor accident DNB ratio versus time for both the 1.10 and new 1.08 Power-to-Flow trip ratio. The minimum DNBR does not go below 1.0 for either case therefore the protection criteria are met.

* The trip delay time for these trip parameters were not changed because the value used for the Type A mechanism was conservative by at least 0.1 sec.

14.1.2.2 Start-Up Accident

14.1.2.3 Rod Withdrawal Accident

14.2.2.2 Rod Ejection Accident

For those accidents where the trip delay time is an important parameter a sensitivity analysis was performed to show the effect of varying the trip delay time. Figure 14-1A, Figure 14-11A and Figure 14-29A show the effect of the Type A and Type C control rod drive mechanism trip delay time on the startup accident, the rod withdrawal accident and the rod ejection accident, respectively. Only for the rated power BOL rod ejection does the thermal power exceed 114 percent of rated thermal power. For all other cases, the thermal power does not exceed 114 percent and the peak pressure never exceeds code allowable limits. Since the Type C control rod drive trip delay time does result in a higher peak thermal power, the percent core experiencing DNB for the rated power BOL rod ejection was re-evaluated. Figure 14-30A shows the percent core experiencing DNB versus ejected control rod worth for both the 0.3 and 0.4 second trip delay time. The environmental consequences of the rod ejection accident using the Type C control rod drive trip delay time were calculated assuming that all fuel rods undergoing DNB release all of their gap activity to the reactor coolant. Subsequently, the gap activity and the activity in the reactor coolant from operation with 1% defective fuel pins is released to the reactor building. For the case of a 0.65% $\Delta k/k$ rod ejection from rated power at BOL, 4.9 percent of the core volume is in DNB and 22.7 percent of the fuel rods are assumed to fail, releasing activity to the reactor building as follows:

Activity Released to Reactor Building

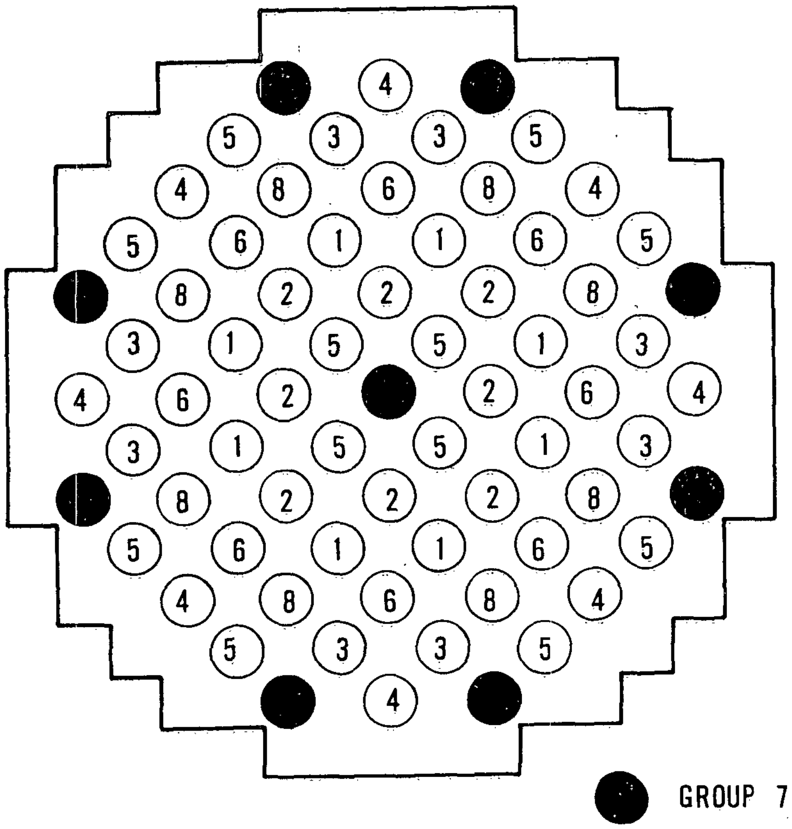
Isotope	Activity (Curies)
Kr-83m	2.04×10^3
Kr-85m	1.16×10^4
Kr-85	1.59×10^5
Kr-87	6.35×10^3
Kr-88	2.08×10^4
Xe-131m	1.91×10^4
Xe-133m	2.24×10^4
Xe-133	2.01×10^6
Xe-135m	6.56×10^3
Xe-135	9.41×10^3

Activity Released to Reactor Building

Isotope	Activity (Curies)
I-131	1.49×10^5
I-132	2.21×10^4
I-133	3.27×10^4
I-134	2.08×10^3
I-135	1.05×10^4

Fission product activities for this accident are calculated using the methods discussed in Section 11.1.1.3 of the FSAR. Using environmental models and dose rate calculational methods discussed under the loss-of-coolant accident (Section 14.2.2.3 of the FSAR), the total integrated 2-hour dose at the 1-mile exclusion distance is 2.0 Rem thyroid and 0.004 Rem whole body. The total integrated thyroid dose at the 6-mile low population zone distance is 2.2 Rem for 30-day exposure. These doses are well below the guideline values of 10 CFR 100.

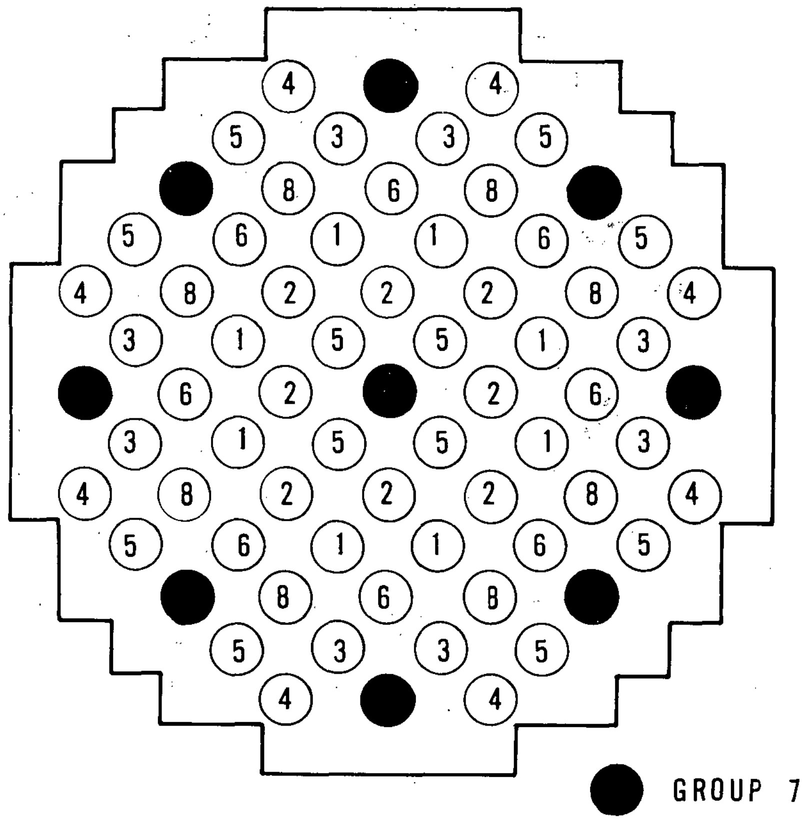
For all other accidents considered in Section 14 of the FSAR, the analysis is either insensitive to the trip delay time or there is no reactor trip.



GROUP	NO. RODS	WORTH (%Δρ)	PURPOSE
1	8	1.82	Safety
2	8	3.19	Safety
3	8	.86	Safety
4	8	1.28	Safety
5	12	1.42	Doppler
6	8	1.58	Doppler
7	9	.99	Transient
TOTAL WORTH	61	11.14	

ROD GROUPS AND WORTH OF GROUPS WITHDRAWN IN ORDER
(Oconee 2 & 3, Cycle 1, BOL)

OCONEE NUCLEAR STATION
Figure 3 - 4H
(New) Rev. 16 7/30/71
Rev. 22. 8/25/72

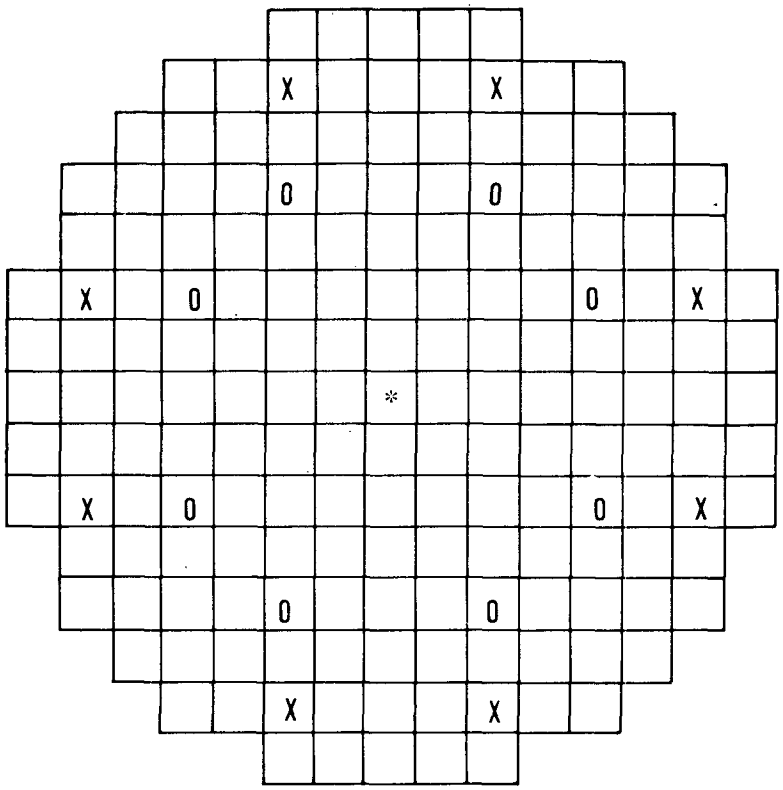


GROUP	NO. RODS	WORTH, (% $\Delta\rho$)	PURPOSE
1	8	1.04	Safety
2	8	2.62	Safety
3	8	0.96	Safety
4	8	1.32	Safety
5	12	1.71	Doppler
6	8	1.17	Doppler
7	9	1.37	Transient
TOTAL WORTH	61 RODS	10.19	

ROD GROUPS AND WORTH OF GROUPS
WITHDRAWN IN ORDER
(Oconee 2 & 3, Cycle 1, EOL)

OCONEE NUCLEAR STATION

Figure 3 - 41
(New) Rev. 16 7/30/71
Rev. 22. 8/25/72



EJECTED ROD WORTH

RODS IN	WORTH OF EJECTED ROD ($\% \Delta \rho$)
Transient Bank	.27
Transient Bank & APSR'S	.49

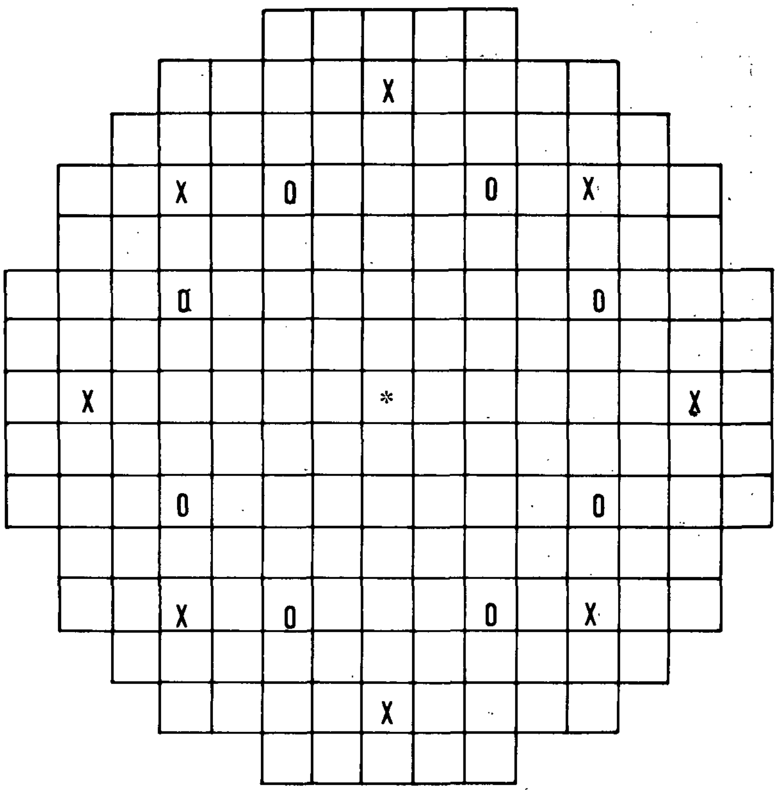
X - ROD POSITION

0 - APSR POSITION

* - EJECTED ROD

TRANSIENT BANK AT BOL, HOT FULL POWER
(Oconee 2 & 3, Cycle 1)

OCONEE NUCLEAR STATION
Figure 3 - 4J
(New) Rev. 16 7/30/71
Rev. 22. 8/25/72



EJECTED ROD WORTH

RODS IN	WORTH OF EJECTED ROD (%Δρ)
Transient Bank	.37
Transient Bank & APSR'S	.67

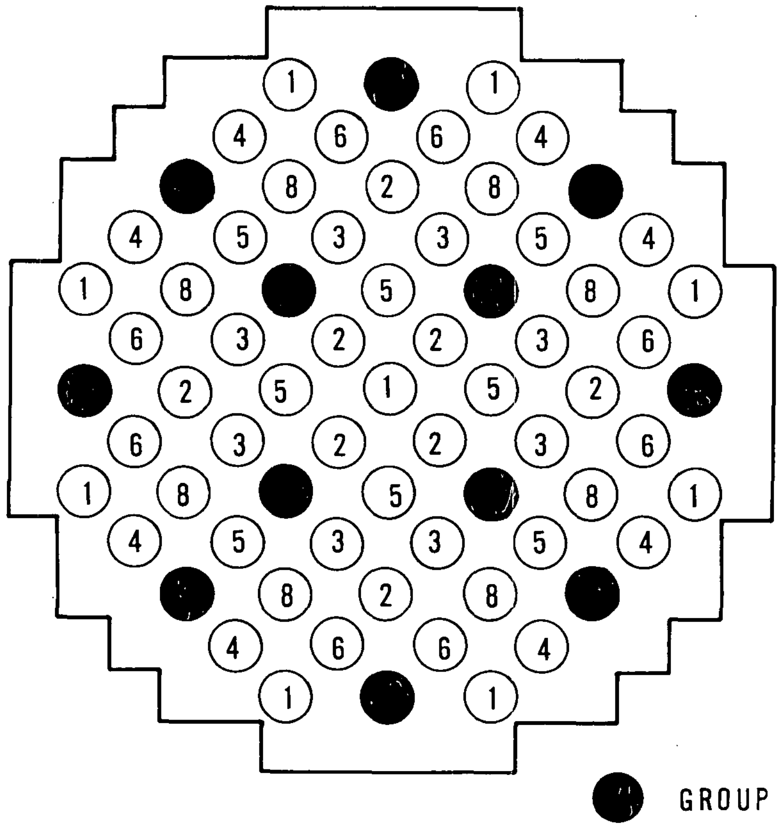
X - ROD POSITION

0 - APSR POSITION

* - EJECTED ROD

TRANSIENT BANK AT 435 DAYS
(Oconee 2 & 3, Cycle 1)

OCONEE NUCLEAR STATION
Figure 3 - 4K
(New) Rev. 16 7/30/71
Rev. 22. 8/25/72



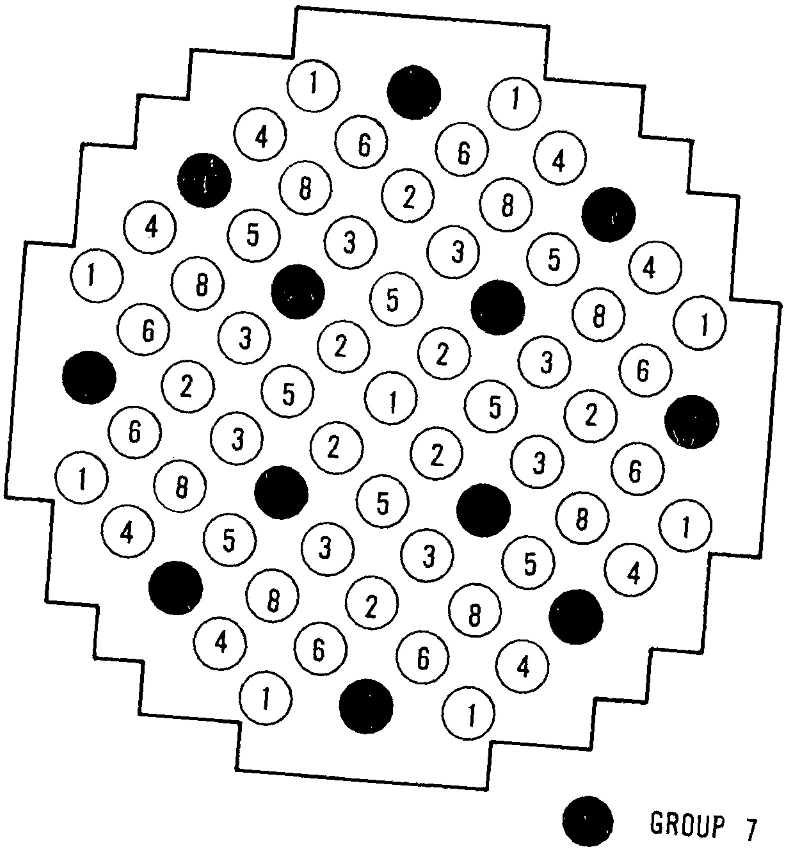
GROUP	NO. RODS	WORTH ($\%\Delta\rho$)	PURPOSE
1	9	2.81	Safety
2	8	.25	Safety
3	8	.74	Safety
4	8	1.33	Safety
5	8	.90	Doppler
6	8	.88	Doppler
7	12	1.45	Transient
TOTAL WORTH	61	8.36	

ROD GROUPS AND WORTH OF GROUPS
WITHDRAWN IN ORDER
(Oconee 3, Cycle 1, BOL)

CITIZENSHIP DUKE POWER SERVICE

OCONEE NUCLEAR STATION
Figure 3 - 4L
(New) Rev. 16 7/30/71

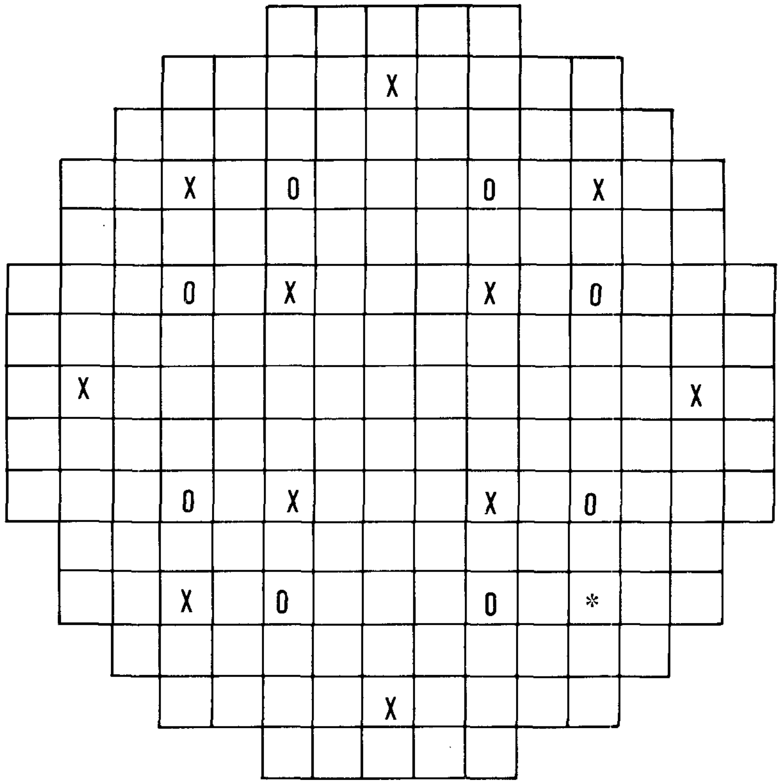
Docket 50-269, -270, and -287
FSAR Supplement 9
July 30, 1971



GROUP	NO. RODS	WORTH (%$\Delta\rho$)	PURPOSE
1	9	2.52	Safety
2	8	.27	Safety
3	8	.84	Safety
4	8	1.36	Safety
5	8	1.14	Doppler
6	8	.89	Doppler
7	12	1.84	Transient
Total Worth	61	8.86	

ROD GROUPS AND WORTH OF GROUPS
WITHDRAWN IN ORDER
(Oconee 3, Cycle 1, EOL)

OCONEE NUCLEAR STATION
Figure 3 - 4M
(New) Rev. 16 7/30/71



EJECTED ROD WORTH

RODS IN	WORTH OF EJECTED ROD ($\%\Delta\rho$)
Transient Bank	.24
Transient Bank & APSR'S	.21

X - ROD POSITION

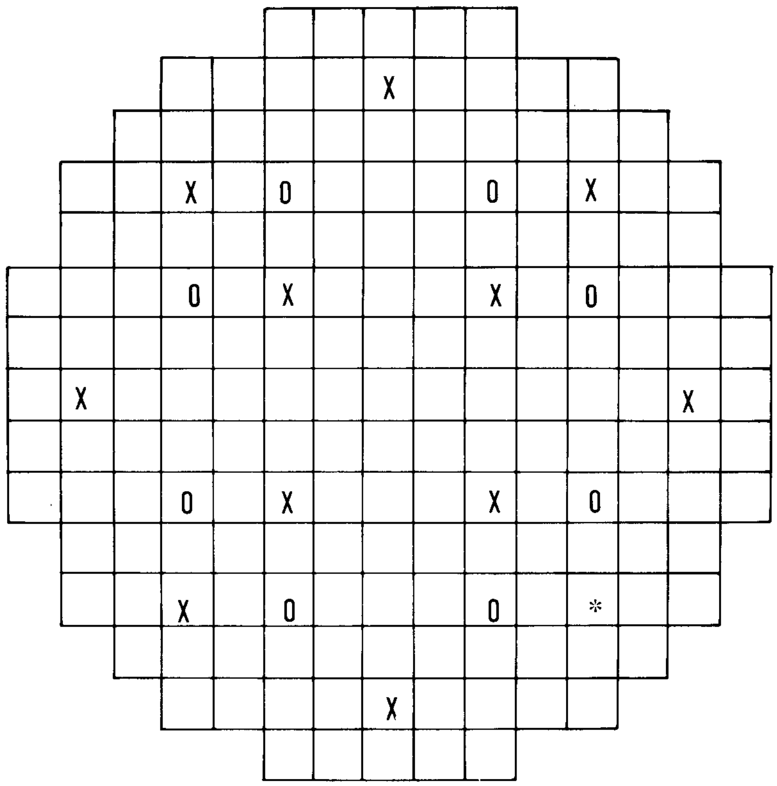
0 - APSR POSITION

* - EJECTED ROD

TRANSIENT BANK AT BOL, HOT FULL POWER
(Oconee 3, Cycle 1)

DUKE POWER CITIZENSHIP SERVICE

OCONEE NUCLEAR STATION
Figure 3 - 4N
(New) Rev. 16 7/30/71



EJECTED ROD WORTH

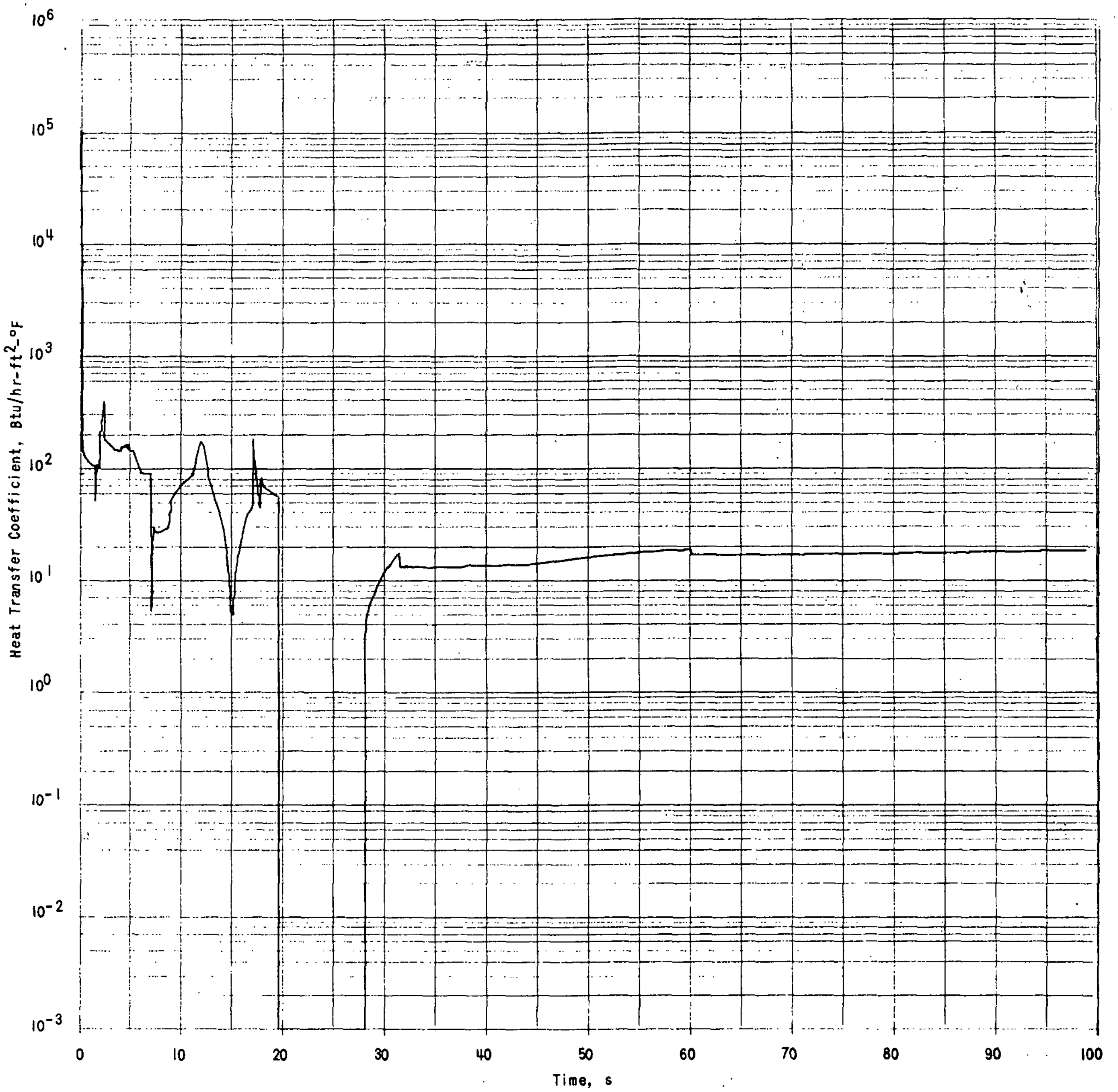
RODS IN	WORTH OF EJECTED ROD (%Δρ)
Transient Bank	.34
Transient Bank & APSR'S	.33

X - ROD POSITION

0 - APSR POSITION

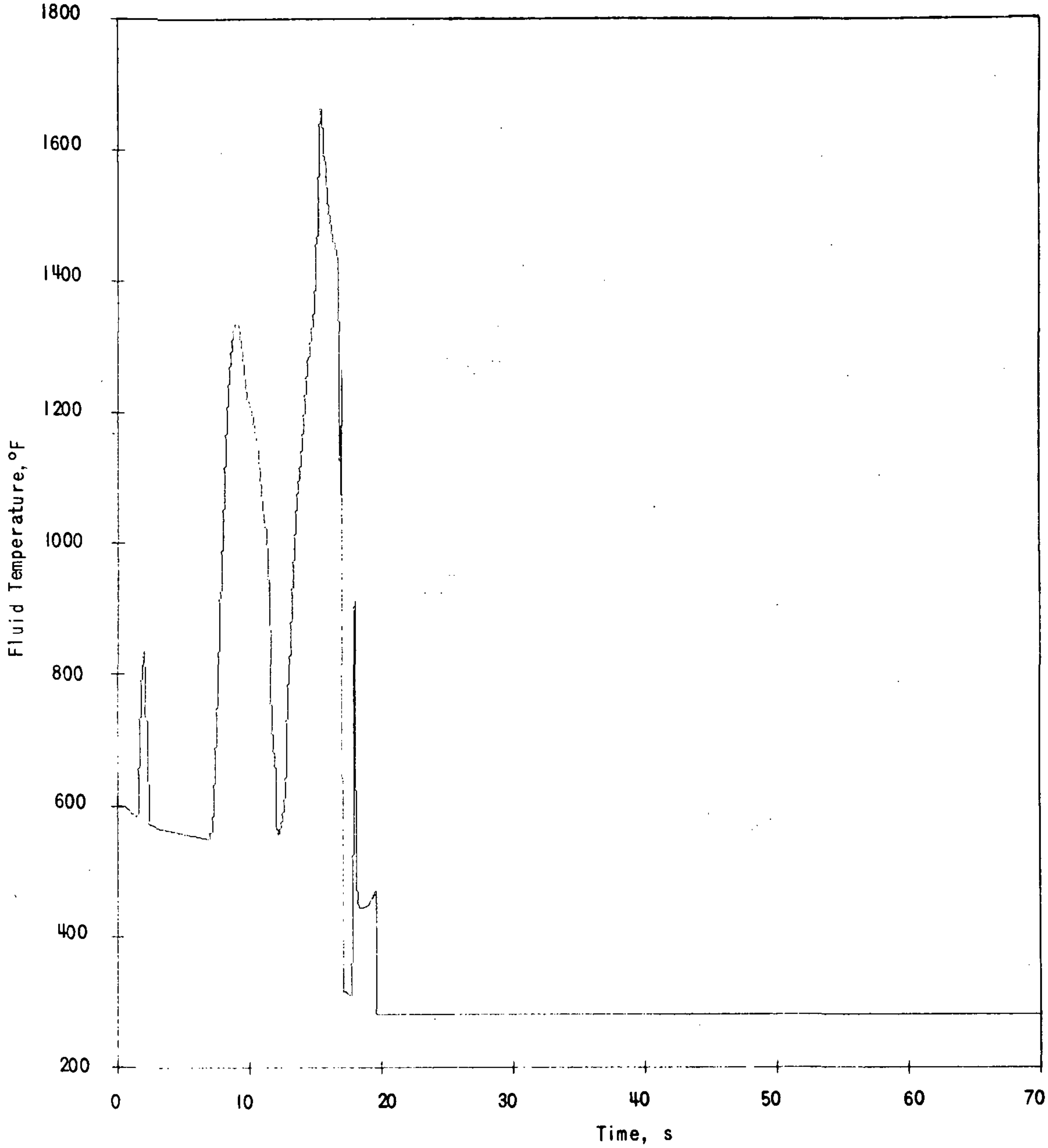
* - EJECTED ROD

TRANSIENT BANK AT 285 DAYS
(Oconee 3, Cycle 1)



HOT SPOT HEAT TRANSFER COEFFICIENT FOR 8.55 FT^2 SPLIT IN COLD LEG PIPE AT PUMP DISCHARGE WITH A SYMMETRICAL POWER SHAPE

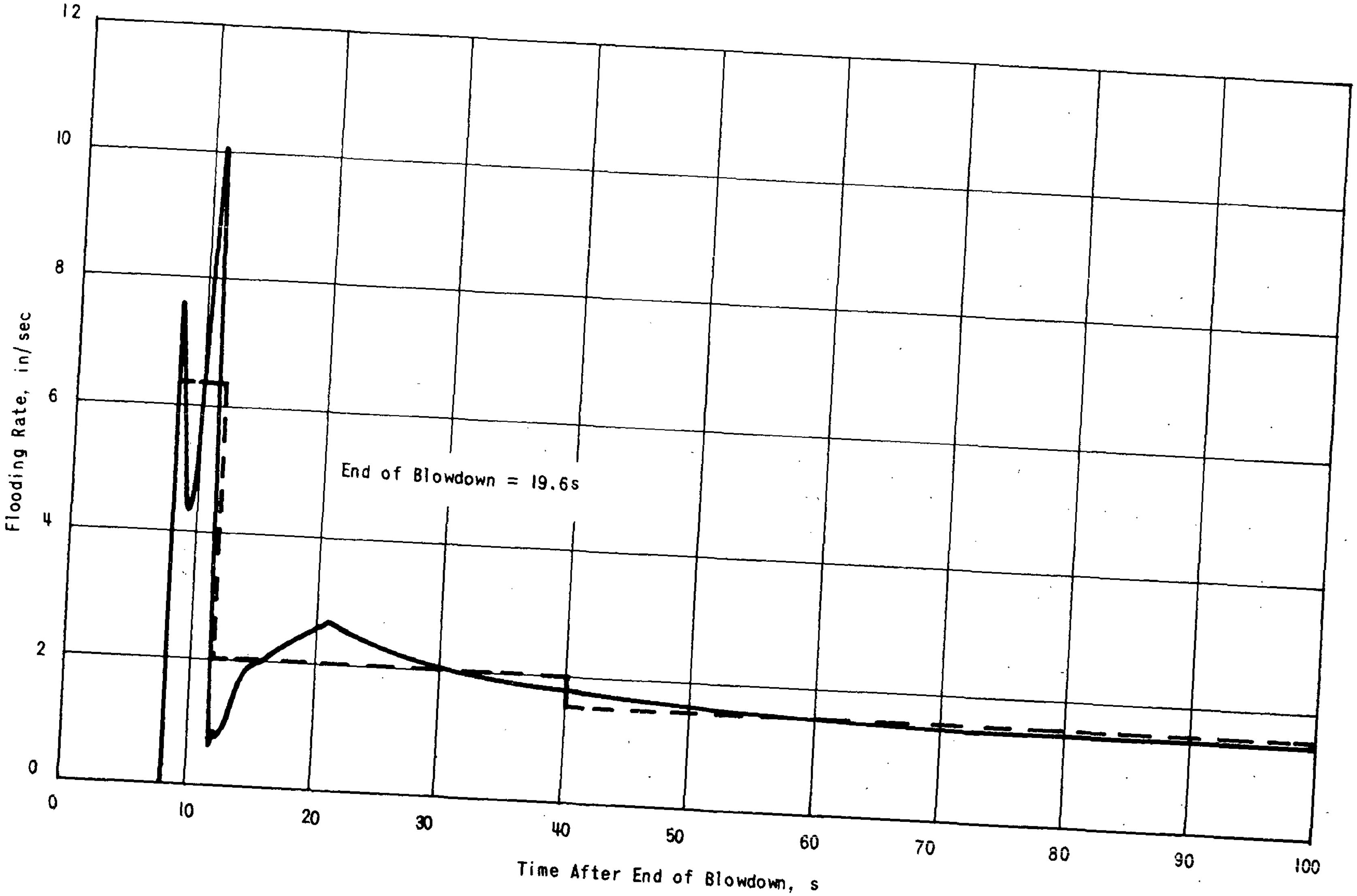
Figure 14.2-26



HOT SPOT FLUID TEMPERATURE FOR 8.55 FT^2
SPLIT IN COLD LEG PIPE AT PUMP DISCHARGE
WITH A SYMMETRICAL POWER SHAPE

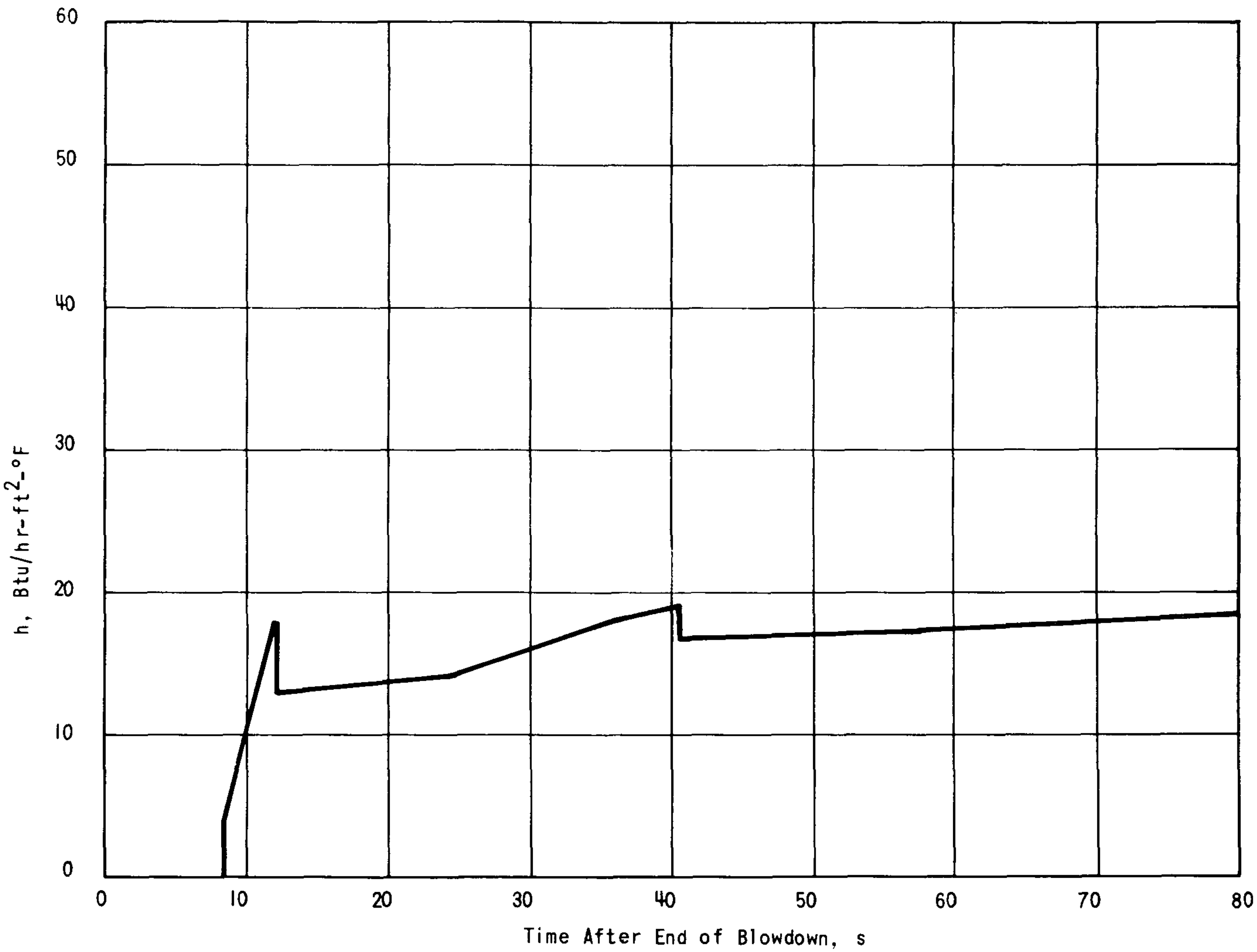
Figure 14.2-27

FSAR Supplement 13-63



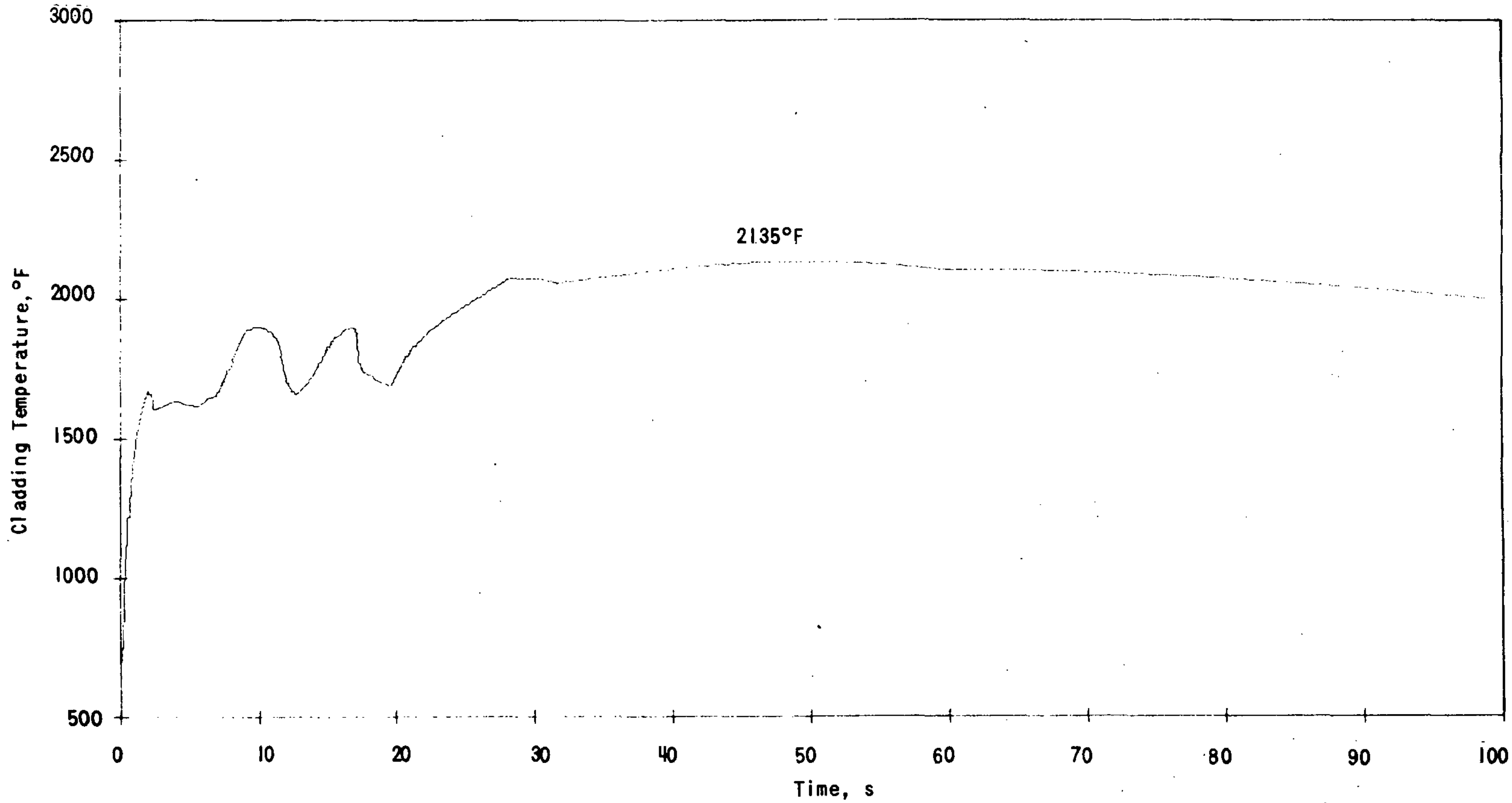
CORE FLOODING RATE FOR 8.55 FT^2 SPLIT IN COLD LEG PIPE AT PUMP DISCHARGE WITH A SYMMETRICAL POWER SHAPE

Figure 14.2-28



POST BLOWDOWN HOT SPOT HEAT TRANSFER COEFFICIENT FOR 8.55 FT^2 SPLIT IN COLD LEG PIPE AT PUMP DISCHARGE WITH A SYMMETRICAL POWER SHAPE

Figure 14.2-29



HOT SPOT CLADDING TEMPERATURE FOR 8.55 FT^2
SPLIT IN COLD LEG PIPE AT PUMP DISCHARGE
WITH A SYMMETRICAL POWER SHAPE

Figure 14.2-30

Docket 50-270 and -287
FSAR Supplement 13
January 29, 1973

Question - 14.3 For small break analysis calculate the case for a 0.5 ft^2 break using the same method as used for the 0.3 ft^2 break.

Answer - BAW-10052[1] presents the results of an analysis of loss-of-coolant accidents resulting from small breaks in the reactor coolant system. The validity of the small leak evaluation model was established by analyzing the 0.5 ft^2 break during the blowdown phase of the transient and comparing the results to those predicted by the large model. To facilitate this comparison, assumptions pertaining to the bubble rise model and pump performance were made such that consistency with the large model was maintained. In brief, both evaluation models predicted similar results with only slight disagreements occurring in terms of the timing of events.

To implement the comparison of the two evaluation models, the 0.5 ft^2 cold leg break has now been evaluated using the analytical methods and assumptions which are strictly applicable to the small leak evaluation model. Section 3 of BAW-10052 presents the method of analysis used and justification thereof.

For the 0.5 ft^2 break, the reactor trips in less than 0.1 second at which time pump coastdown is initiated. The core flow, Figure 14.3-1, exhibits a gradual decline in flow rate until the pump cavitates at 55 seconds. Flow is then nearly stagnate because of the formation of a steam bubble in the hot leg. However, at this time a two phase mixture is maintained in the core by flow from the loops and from the high pressure injection pump. Adequate cooling is demonstrated to exist by using pool boiling heat transfer.

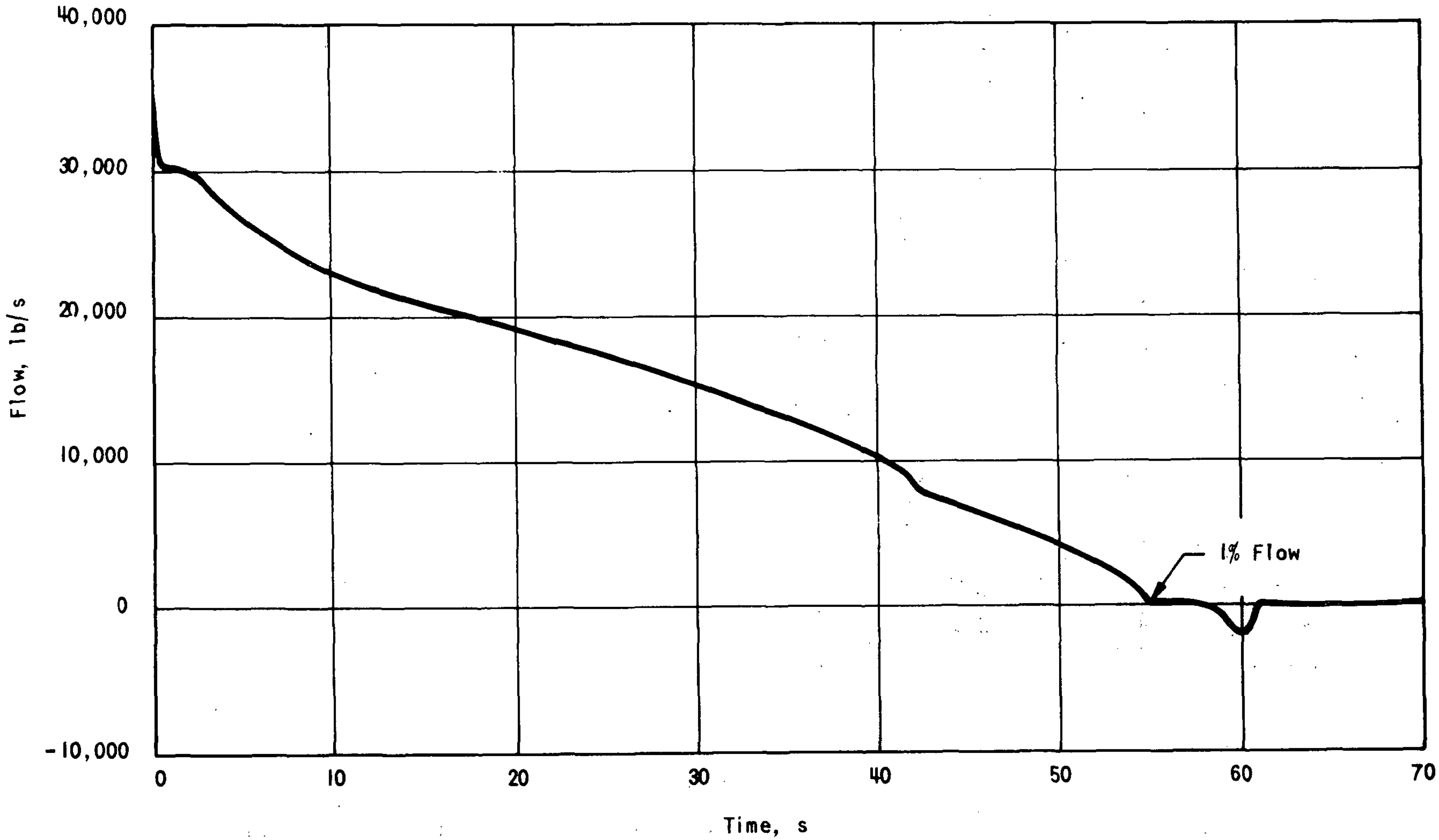
The power transient, pressure history, inner vessel mixture volume, vessel liquid volume, hot spot heat transfer coefficient, and hot spot cladding temperature responses are shown in Figures 14.3-2 through 14.3-7. The cladding temperature decreases initially due to the loss of power after trip without a substantial loss of flow. Then at 55 seconds the pumps cavitate and the core flow falls to 1% of its initial value. The heat transfer mode is then assumed to be pool film boiling and the heat transfer coefficients are based on Morgan's correlation. As a result, the cladding temperature increases to a new equilibrium value of approximately 700 F. The temperature falls slightly over the next 100 seconds, but then increases again as the effect of pressure on the heat transfer coefficient becomes evident. Since the Morgan heat transfer coefficient decreases with decreasing pressure, loss of pressure causes a rise in temperature to a maximum value of 710 F. The temperatures then decrease as further reductions in the heat transfer coefficient are matched by reductions in the decay heat rate.

At 400 seconds, the system pressure has decayed to a steady value, the core is covered with mixture, and the engineered safeguard systems are providing more makeup than is being leaked. Therefore, the transient is terminated.

---

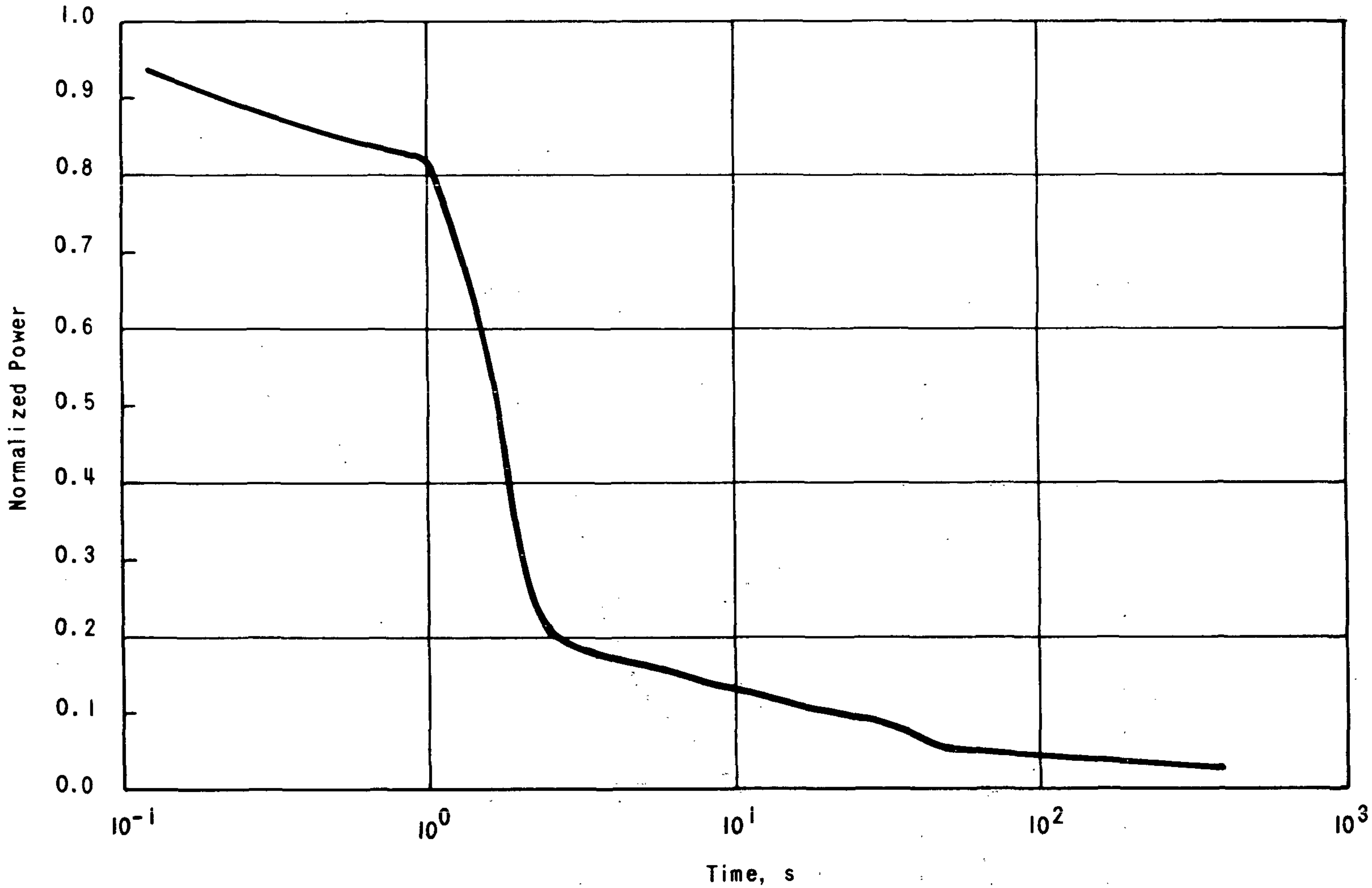
[1] C. E. Parks, et al., Multinode Analysis of Small Breaks for B&W 2568-MWt Nuclear Plants, BAW-10052, Babcock & Wilcox, Lynchburg, Virginia, Sept., 1972.

LIST OF FIGURES
APPLYING TO QUESTION
14.3



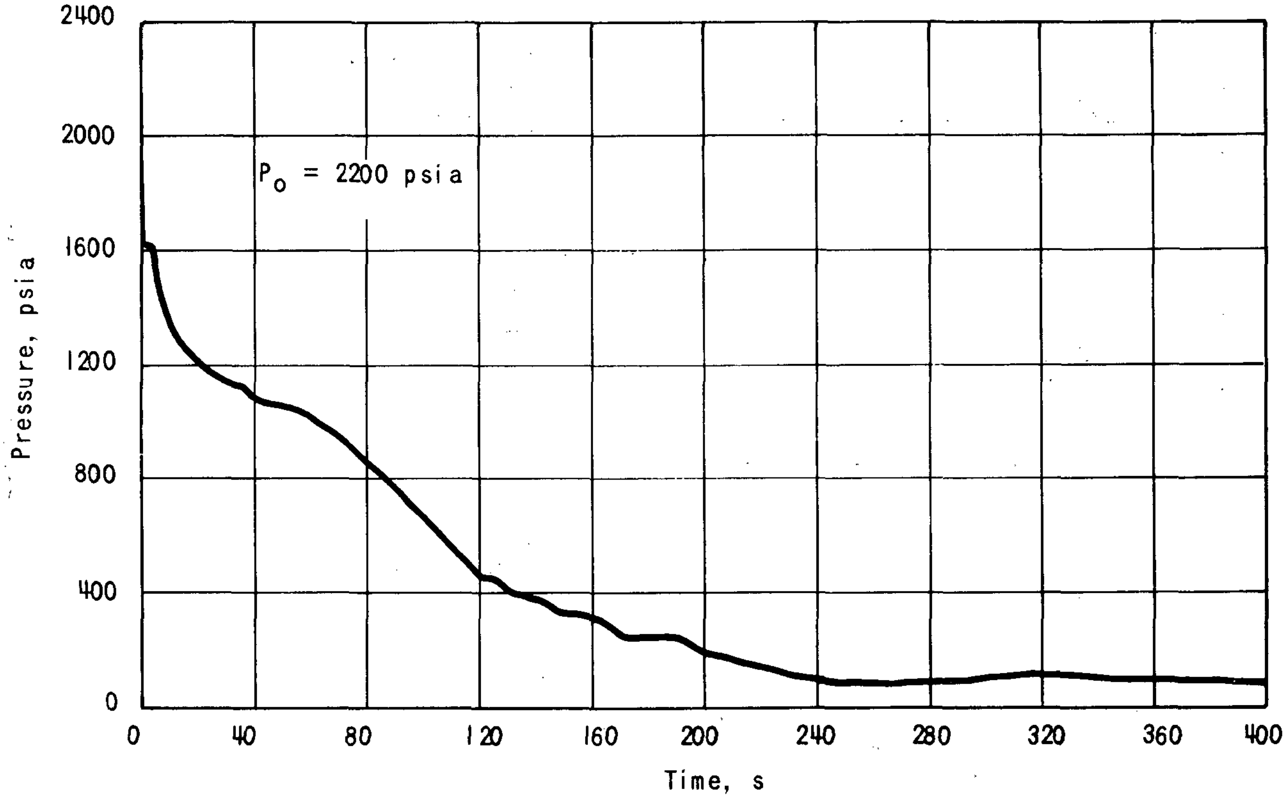
CORE FLOW FOR 0.5 FT^2 SPLIT IN COLD LEG PIPE AT PUMP DISCHARGE

Figure 14.3-1



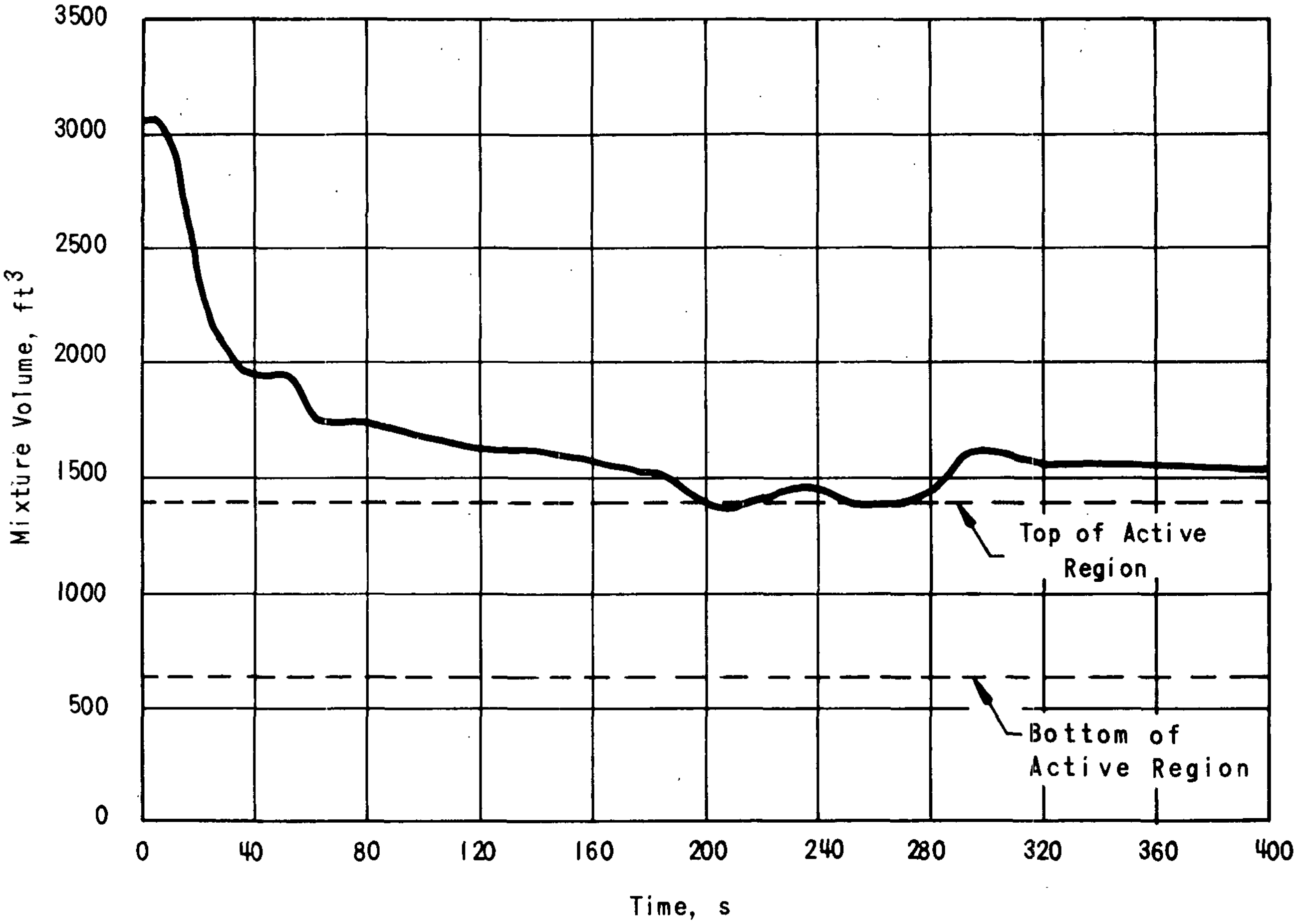
CORE THERMAL POWER FOR 0.5 FT^2 SPLIT IN COLD LEG PIPE AT PUMP DISCHARGE

Figure 14.3-2



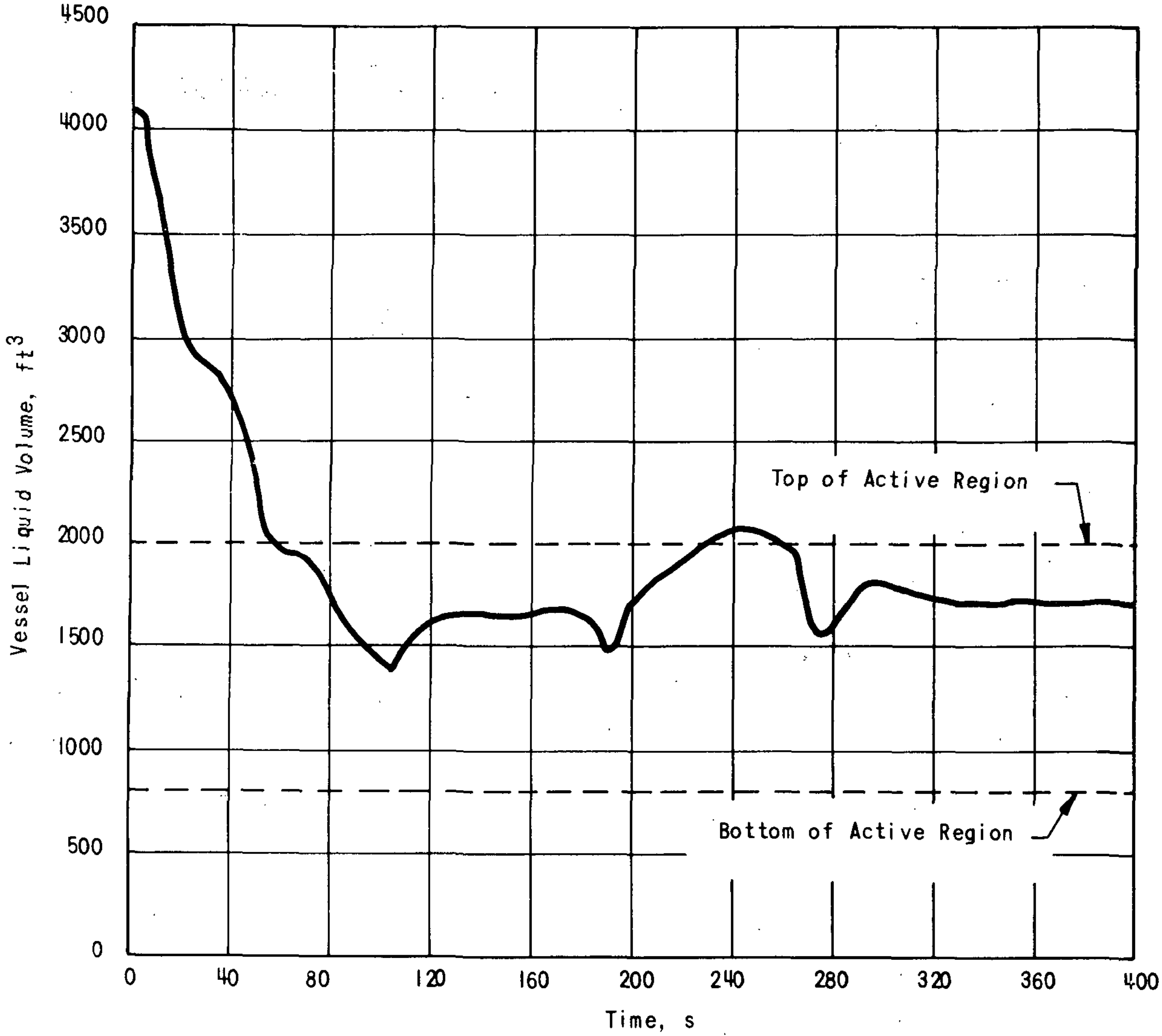
PRESSURE TRANSIENT FOR 0.5 FT^2 SPLIT IN COLD LEG PIPE AT PUMP DISCHARGE

Figure 14.3-3



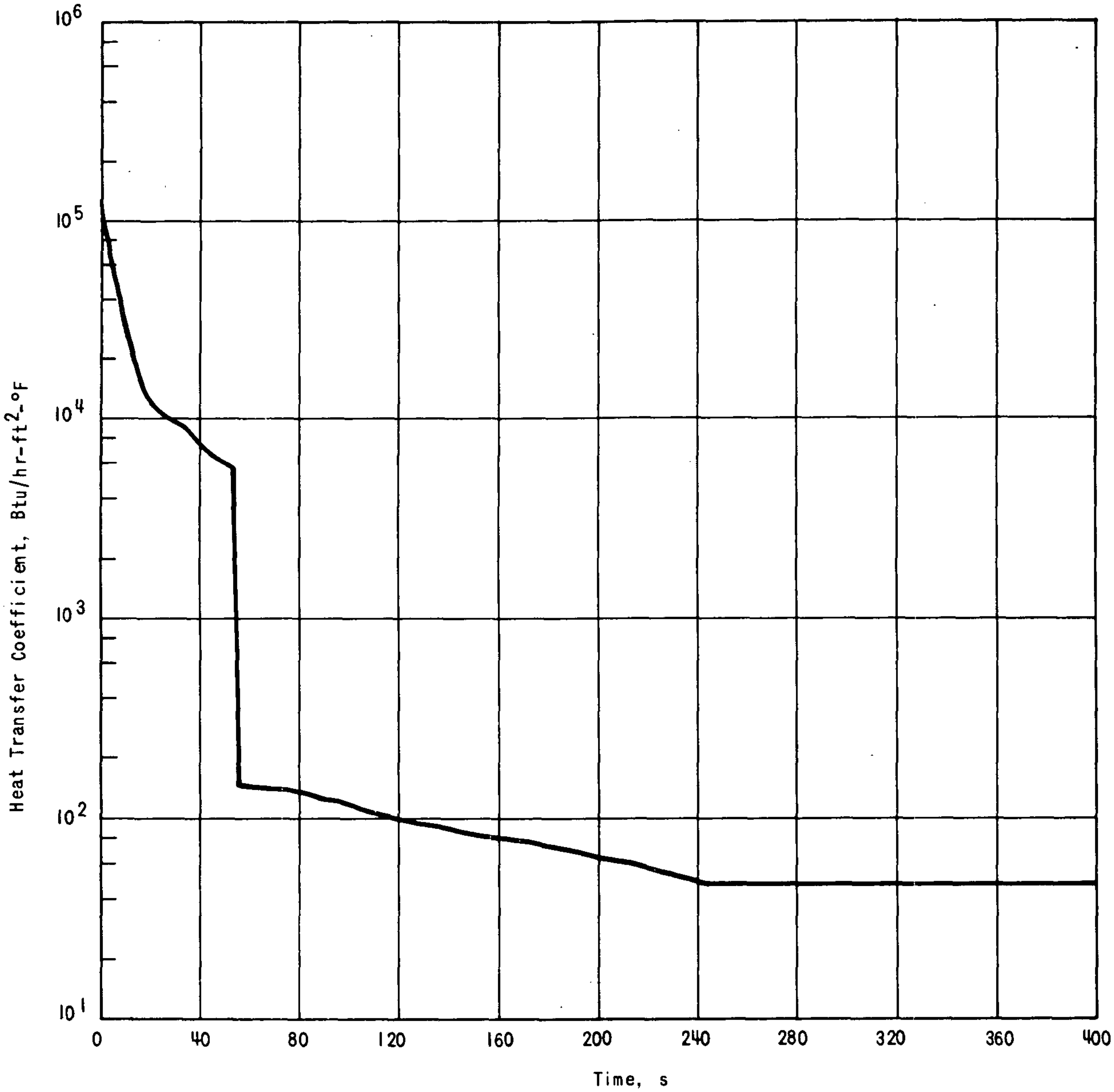
INNER VESSEL MIXTURE VOLUME FOR
0.5 FT² SPLIT IN COLD LEG PIPE
AT PUMP DISCHARGE

Figure 14.3-4



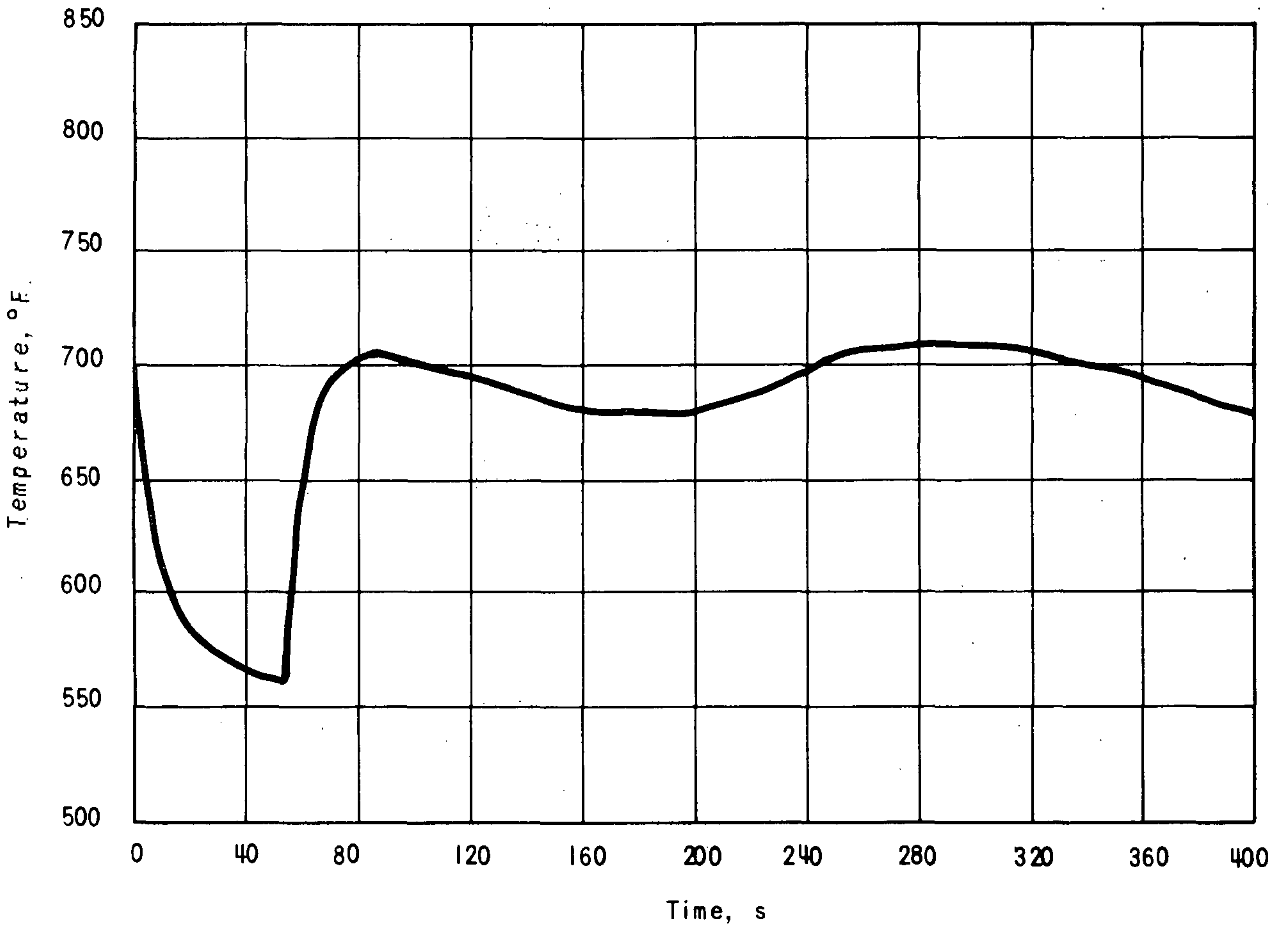
VESSEL LIQUID VOLUME FOR 0.5 FT^2 SPLIT IN COLD LEG PIPE AT PUMP DISCHARGE

Figure 14.3-5



HOT SPOT HEAT TRANSFER COEFFICIENT FOR 0.5 FT^2 SPLIT IN COLD LEG PIPE AT PUMP DISCHARGE

Figure 14.3-6



HOT SPOT CLADDING TEMPERATURE FOR 0.5 FT^2 SPLIT IN COLD LEG PIPE AT PUMP DISCHARGE

Figure 14.3-7

The following is voluntarily submitted by the applicant in response to concerns expressed by the staff on January 17, 1973.

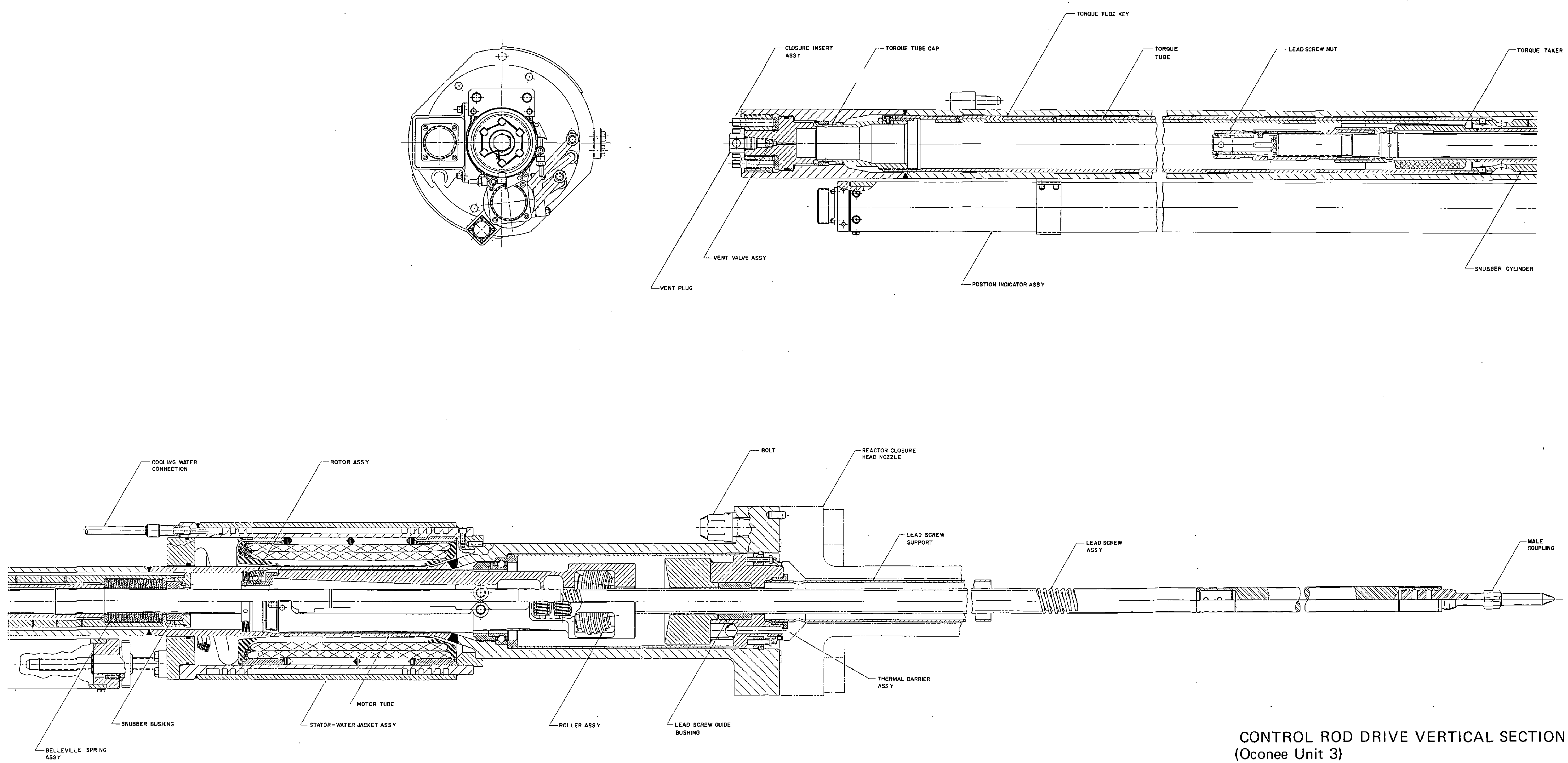
The analysis presented in BAW-10034 shows the consequences of breaks in the reactor coolant piping from a power level of 102% of 2568 MWt. That analysis established the 8.55 ft^2 cold leg break as the case resulting in the highest peak cladding temperature.

A further analysis has been performed for operation with one idle loop (2 pumps operating in other loop) at 55% power. Using BAW-10034 as a guide, the 8.55 ft^2 double-ended rupture was studied. Decay heat was determined from the ANS formulation plus 20%. Peaking factors were the same as those used in BAW-10034.

Two cases were studied. Case 1 was the double-ended rupture at the pump discharge in the loop with pumps running and Case 2 was the double-ended rupture at the pump discharge in the loop with idle pumps. This combination of pump operation results in flow backwards through the idle pumps or away from the reactor vessel. For Case 1, the CRAFT model was that as depicted on Figure 3-1 of BAW-10034. For Case 2, the model was adjusted by separating the noding into two pipe paths where they had formerly been combined into one path. For Case 1, the peak cladding temperature was 1265F occurring at + 45 sec. For Case 2, the peak cladding temperature was slightly higher at 1305 F occurring again at + 45 sec. By 70 seconds after the break occurred, the cladding temperature was down below 900 F. The B&W correlation of the FLECHT carryout rate fraction data was used in the REFLOOD code. By using all the ground rules in Appendix A, Part 4, of the Interim Acceptance Criteria, the analysis clearly shows that all portions of the Interim Criteria are met. The low temperatures experienced were not unexpected due mainly to the lower stored heat in the fuel and the lower power level and therefore decay heat.

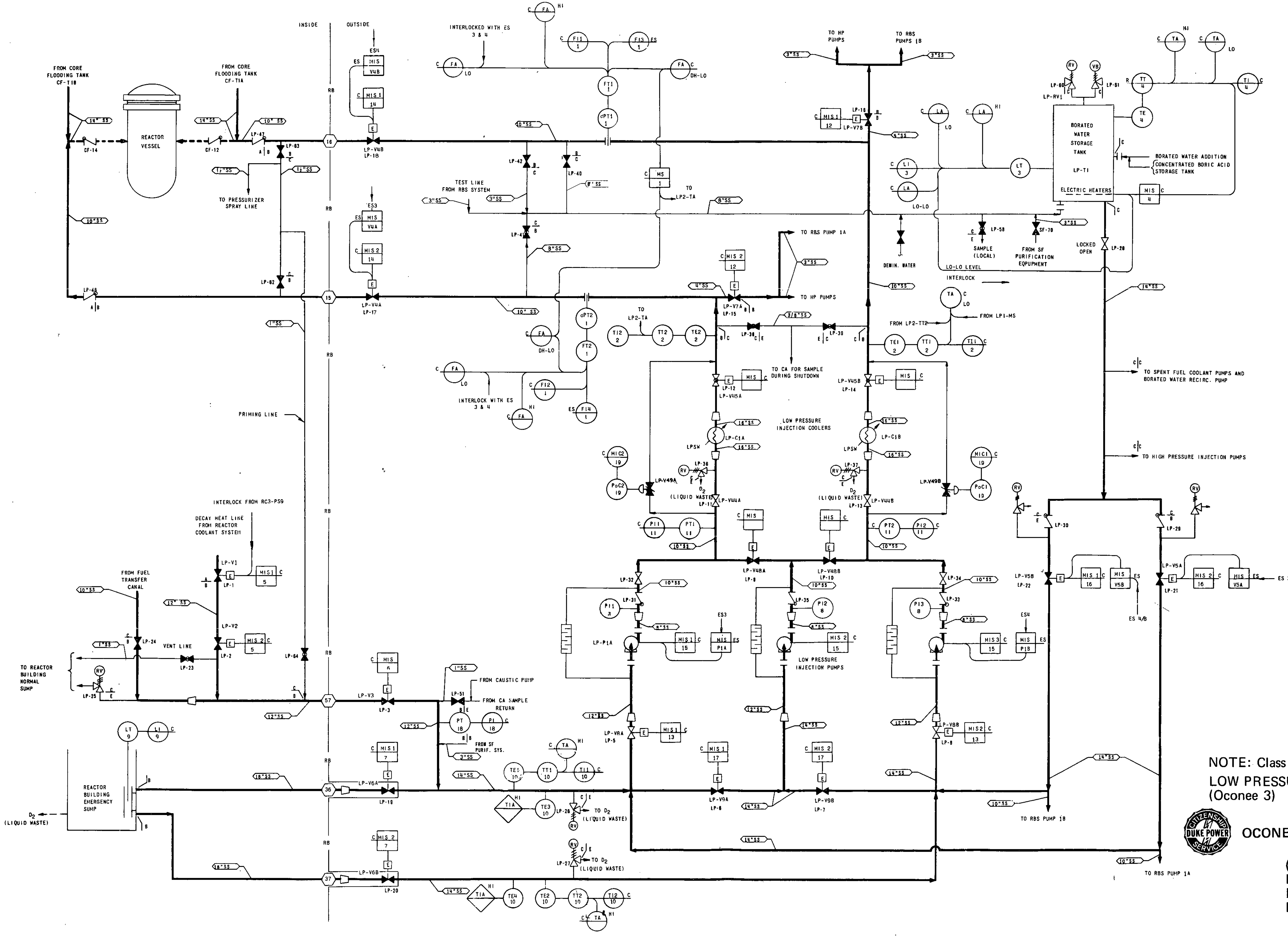
The analysis presented in BAW-10034 and this analysis span all proposed operating conditions of the plant. A LOCA occurring during operation at reduced power with three pumps running would be expected to give cladding temperatures somewhere between the values reported in BAW-10034 and the later analysis performed with an idle loop at 55% power.

Docket 50-269, -270, and -287
FSAR Supplement 9
July 30, 1971



CONTROL ROD DRIVE VERTICAL SECTION
(Oconee Unit 3)

OCONEE NUCLEAR STATION
Figure 3 - 56A
(New) Rev. 16 7/30/71

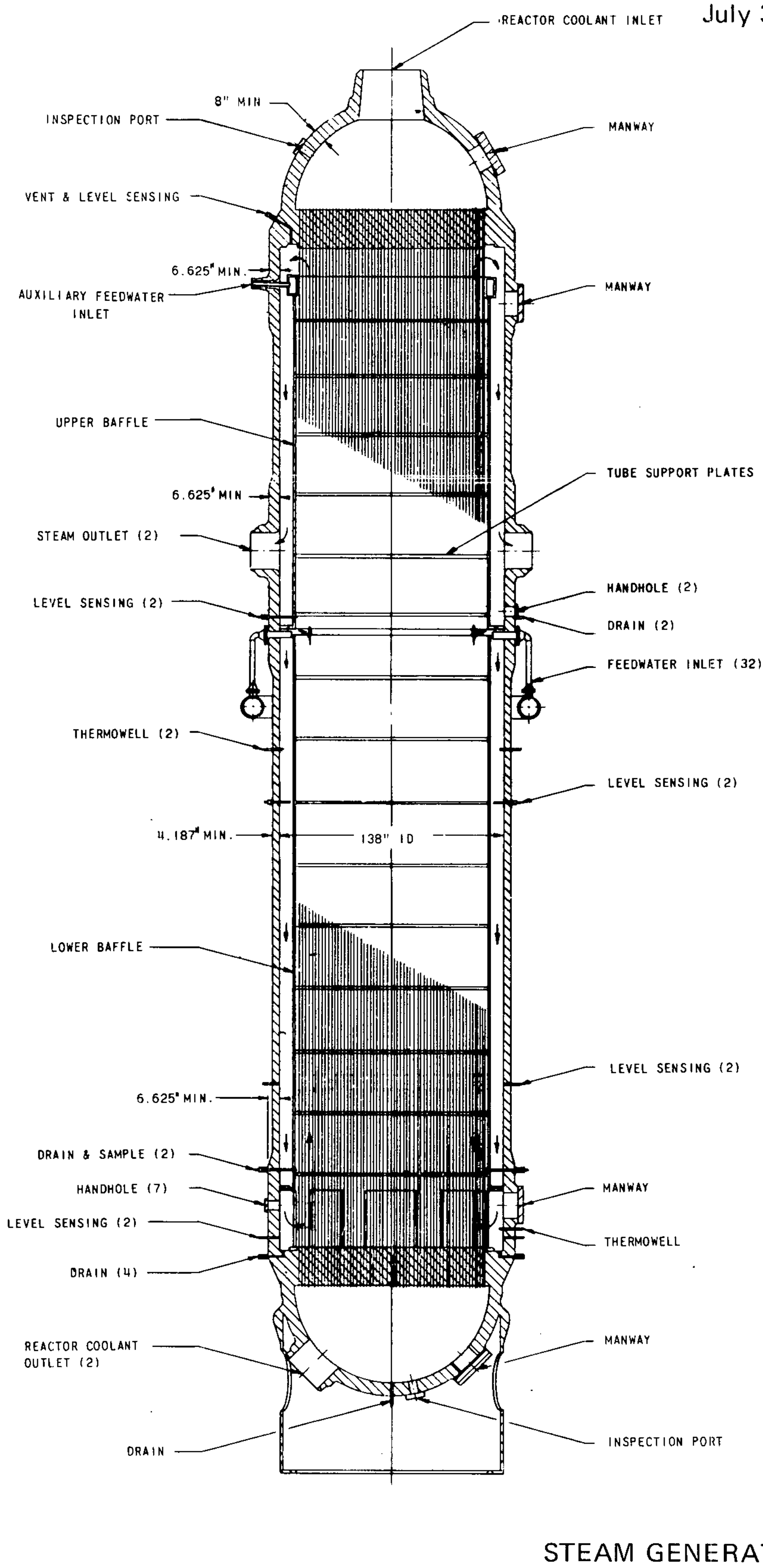


NOTE: Class B system except as noted.

LOW PRESSURE INJECTION SYSTEM (Oconee 3)

OCONEE NUCLEAR STATION

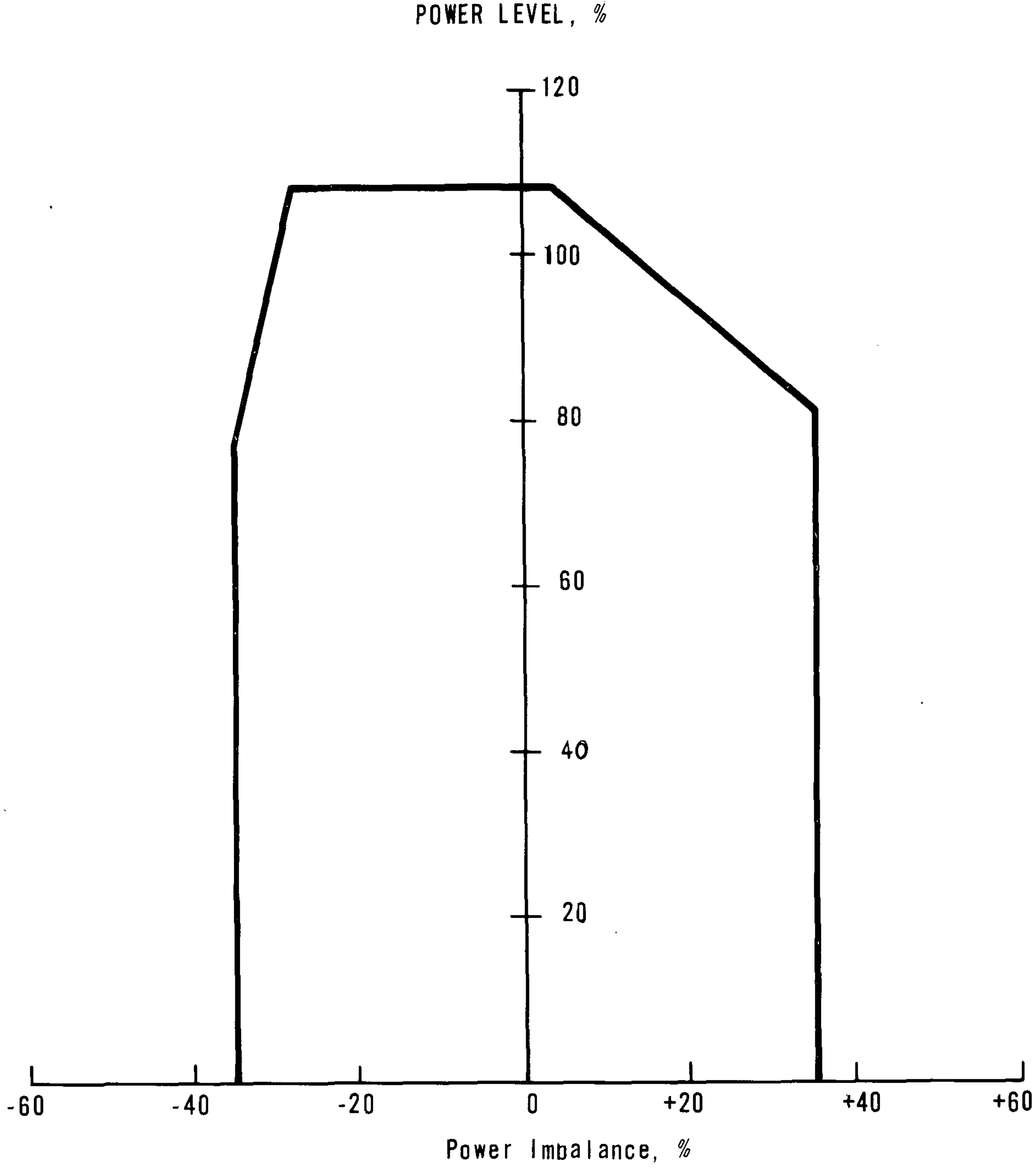
Figure 9 - 6A
(Supplement 9)
Rev. 9 8/11/70
Rev. 16 7/30/71
Rev. 18 3/10/72



STEAM GENERATOR
(OCONEE 3)

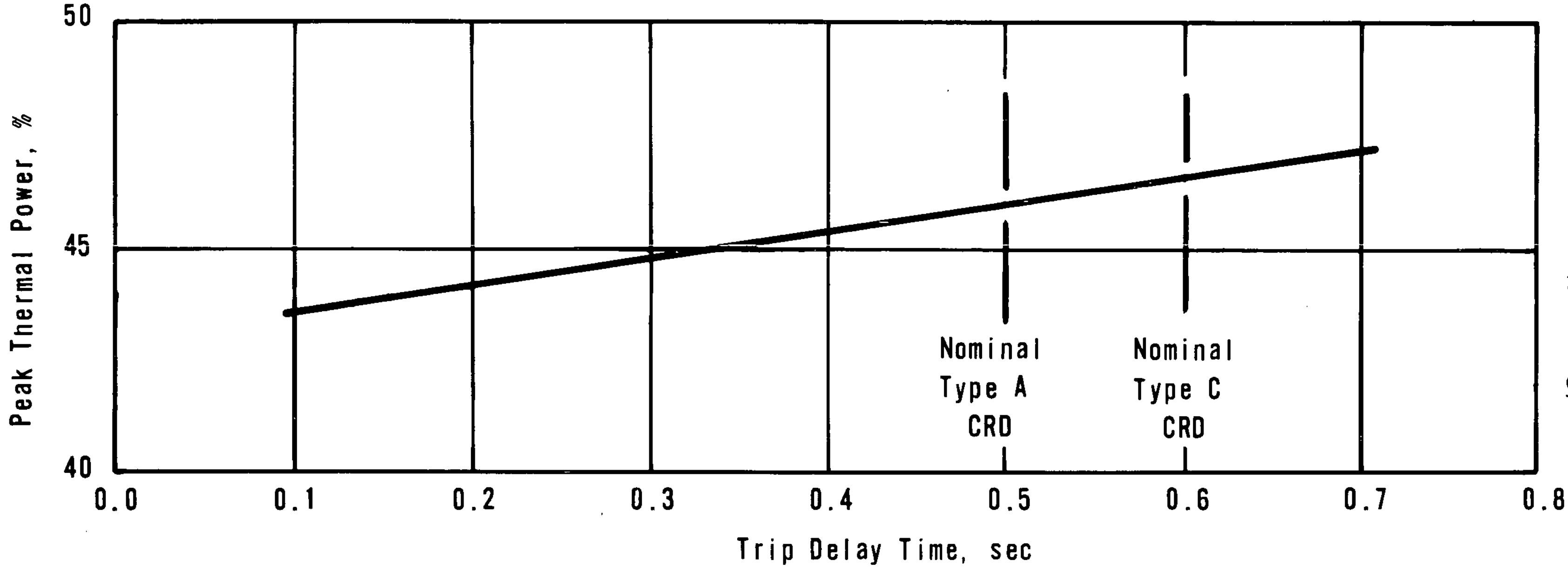
OCONEE NUCLEAR STATION
Figure 4 - 5B
(New) Rev. 16 7/30/71

Docket 50-269, -270, and -287
FSAR Supplement 9
July 30, 1971



POWER IMBALANCE BOUNDARIES
(Oconee 3)

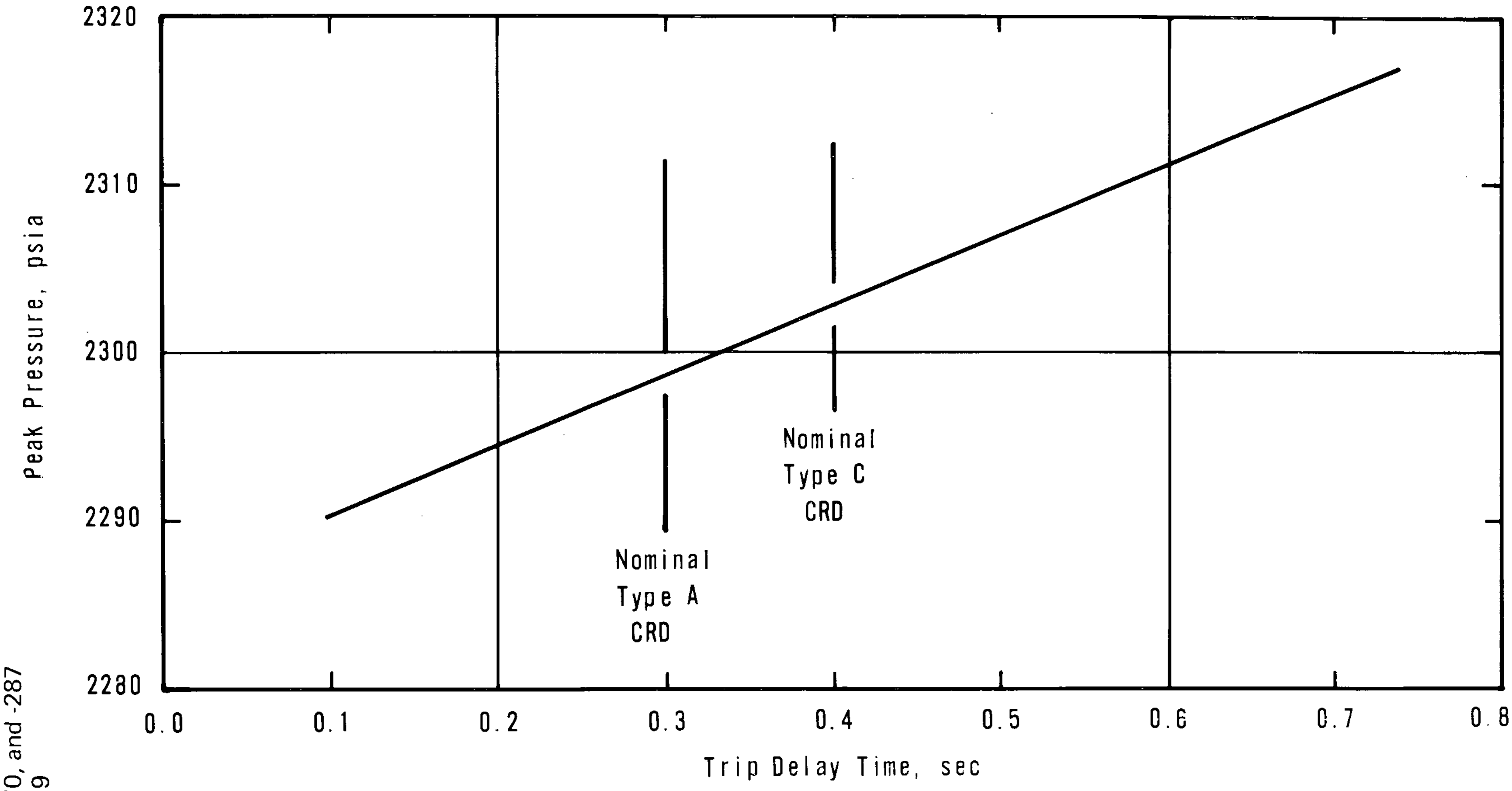
OCONEE NUCLEAR STATION
Figure 7 - 2B
(New) Rev. 16. 7/30/71



PEAK THERMAL POWER VERSUS TRIP DELAY TIME FOR A STARTUP ACCIDENT USING A 1.5% $\Delta K/K$ ROD GROUP AT 1.09 x 10^{-4} $(\Delta K/K)$/SEC FROM 10^{-9} RATED POWER

(Oconee 3)

OCONEE NUCLEAR STATION
Figure 14 - 1A
(New) Rev. 16 7/30/71

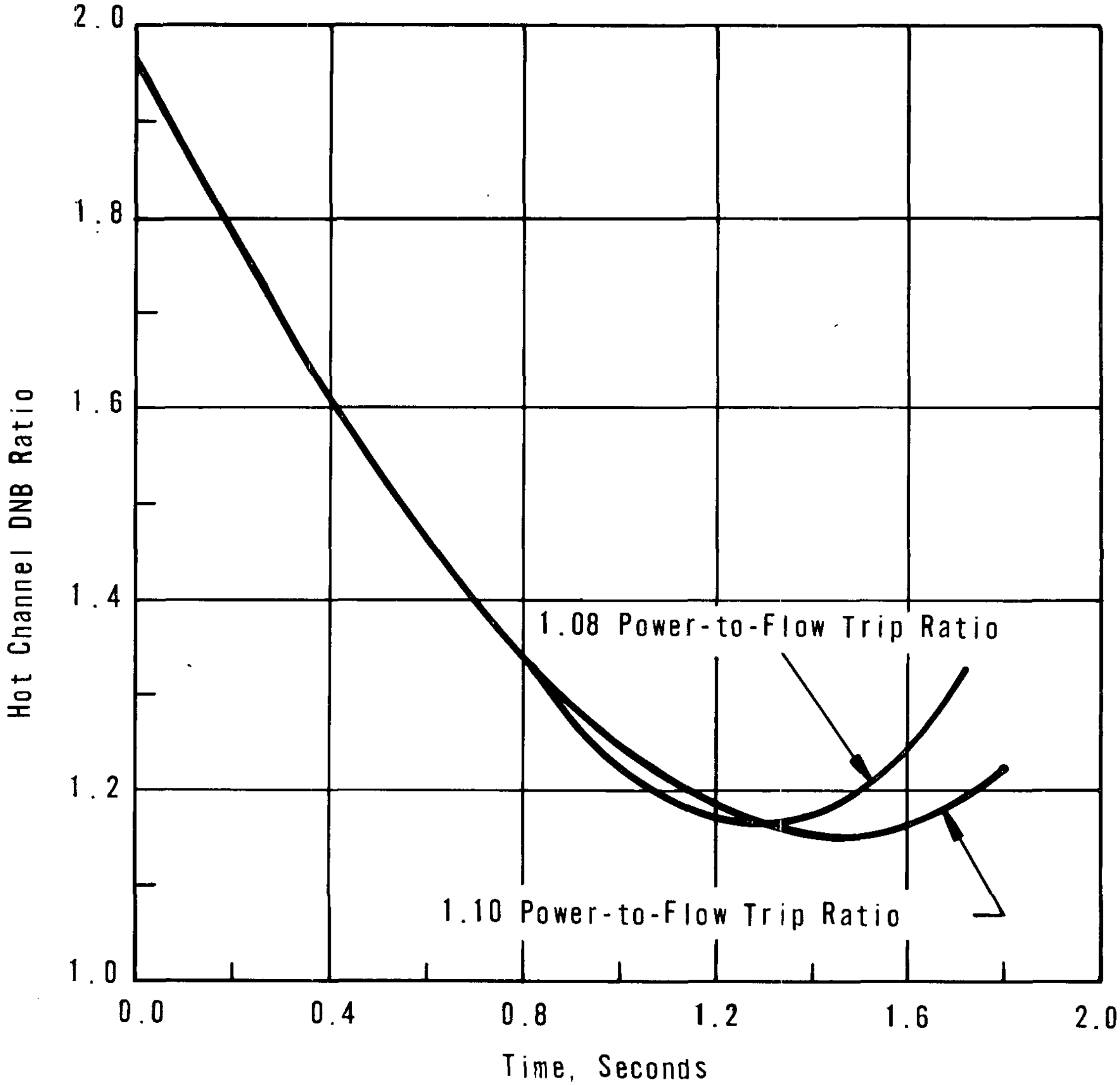


PEAK PRESSURE VERSUS TRIP DELAY TIME FOR A ROD WITHDRAWAL ACCIDENT FROM RATED POWER USING A 1.5% $\Delta k/k$ ROD GROUP

Docket 50-269, -270, and -287
FSAR Supplement 9
July 30, 1971

(Oconee 3)

OCONEE NUCLEAR STATION
Figure 14-11A
(New) Rev. 16 7/30/71



DNB RATIO VS. TIME FOR A LOCKED ROTOR ACCIDENT FROM 102% RATED POWER
(Oconee Unit 3)

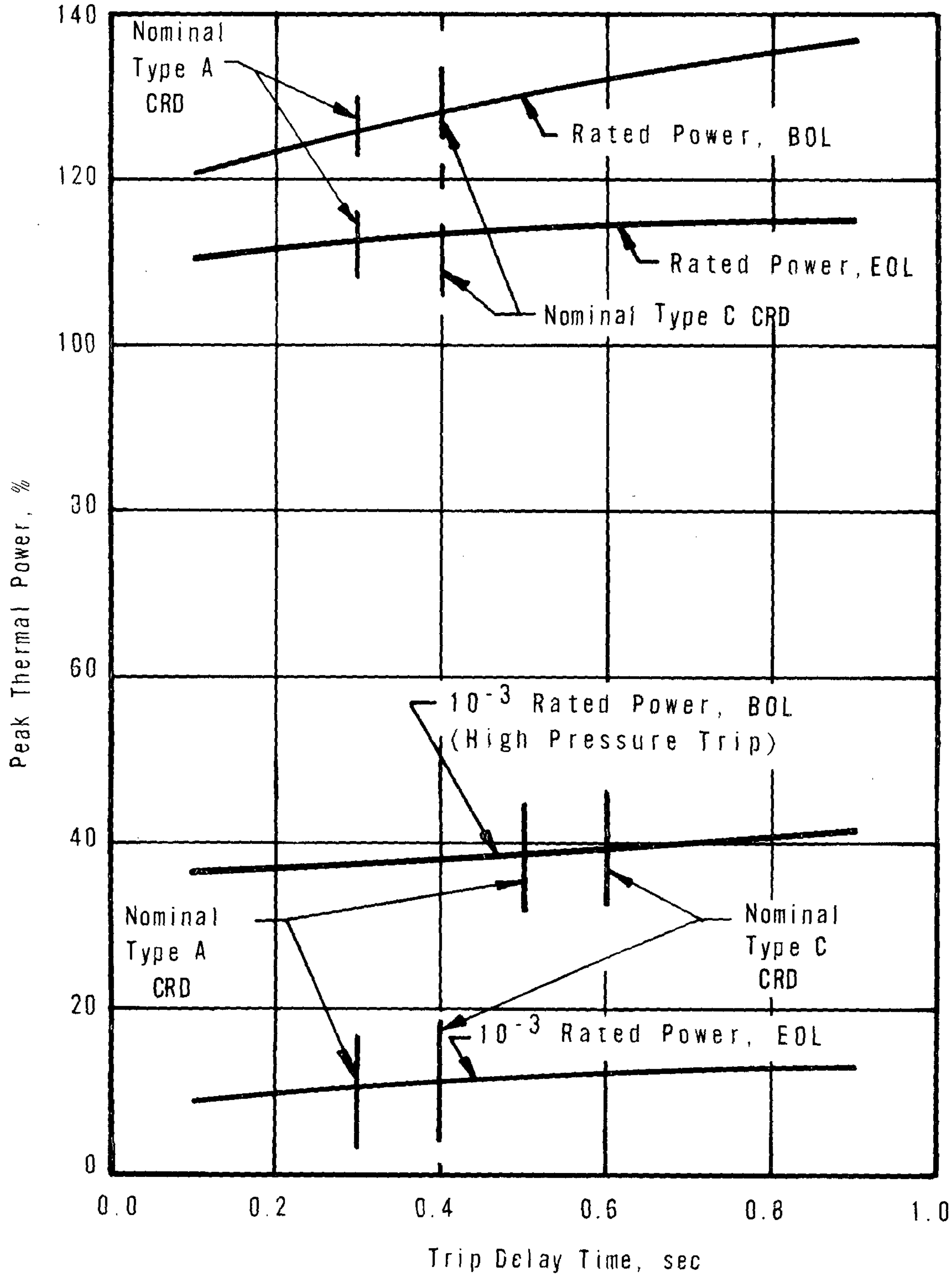
CITIZENSHIP DUKE POWER SERVICE

OCONEE NUCLEAR STATION

Figure 14 - 17C

(New) Rev. 16 7/30/71

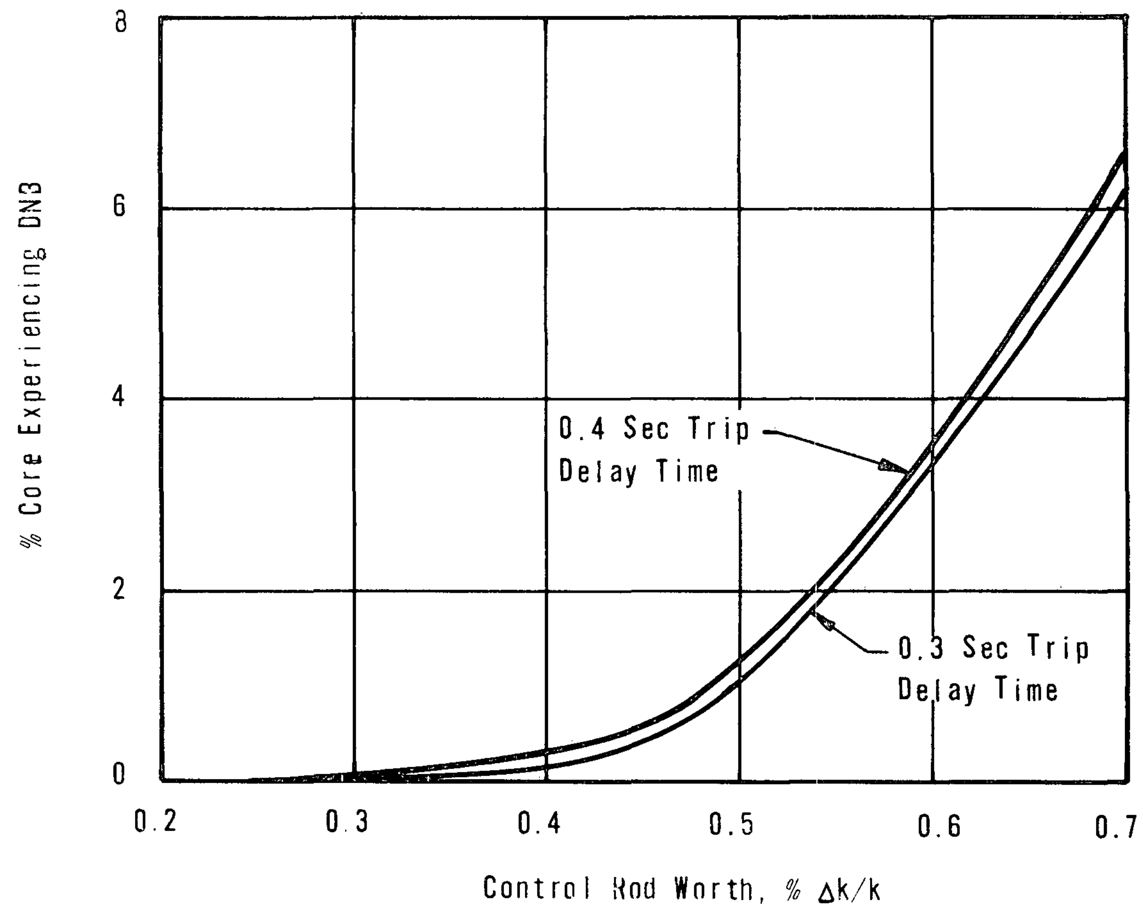
Docket 50-269, -270, and -287
FSAR Supplement 9
July 30, 1971



EFFECT ON PEAK THERMAL POWER OF VARYING THE TRIP DELAY TIME FOR AN EJECTED ROD WORTH OF 0.56% $\Delta k/k$ AT 10^{-3} RATED POWER AND 0.46% $\Delta k/k$ AT RATED POWER

(Oconee 3)

OCONEE NUCLEAR STATION
Figure 14 - 29A
(New) Rev. 16 7/30/71



PER CENT CORE EXPERIENCING DNB AS A FUNCTION OF EJECTED CONTROL ROD WORTH AT RATED POWER (BOL)
(Oconee 3)

OCONEE NUCLEAR STATION
Figure 14 - 30A
(New) Rev. 16 7/30/71

D U K E   P O W E R   C O M P A N Y

OCONEE NUCLEAR STATION

UNITS 1, 2, AND 3

APPLICATION FOR LICENSES
DOCKETS 50-269, -270, and -287

FSAR SUPPLEMENT 10

Submitted with FSAR Revision 17

December 17, 1971

Entire Supplement Revised

Rev. 24. November 15, 1972

November 15, 1972

Supplement 10 to the Final Safety Analysis Report, submitted with FSAR Revision 17 on December 17, 1971, supplemented the information presented in B&W Topical Report BAW-10034 in order to evaluate the effectiveness of the emergency core cooling system for unpressurized fuel pins, since portions of the Oconee Unit 1 and 3 cores were originally intended to employ unpressurized pins.

All three Oconee cores will now employ pressurized fuel pins; therefore, all of the results presented in BAW-10034 are applicable to Oconee Units 1, 2, and 3, and Supplement 10 is rescinded.

In addition to presenting the ECCS results for unpressurized fuel pins, Supplement 10 answered three questions which resulted from the AEC review of B&W's evaluation model. These questions were answered using unpressurized fuel pin models, but the results are also applicable to pressurized fuel pins and are being retained in the supplement to assure a complete record.

Request 1

Report the analysis to show that the assumption of a "locked rotor" condition during blowdown results in conditions less severe than those calculated for the "free spinning" case.

Response:

Section 5.5 of BAW-10034 explains how the pump model used in the CRAFT program was selected. To show that the free spinning rotor assumption is more conservative than the locked rotor case, a case was run assuming that the rotors on the pumps locked after the pump head went to zero due to cavitation. The split in the cold leg pipe with an area equivalent to a double-ended pipe break was the case selected. The resulting cladding temperature for this case is shown on Figure 1-1. The peak temperature is 1880F, which is 404F lower than the same case run with a free spinning rotor shown on Figure 1-2.

Request 2

Discuss the sensitivity of peak cladding temperature to variations in reflooding rate and the consequent variations in steaming rate and entrainment. The present analysis predicts flooding rates of at least 4 in/sec. The information to be provided should include rates down to 2 in./sec.

Response:

As stated in BAW-10034, the heat transfer coefficients used in the cladding heatup code during the reflood period are based on the FLECHT results. In order to use the FLECHT results, the core flooding rate must be known. The reflood model assumed that all of the steam generated in the core had to be relieved through the internals vent valves in the core support shield. As stated in Appendix B of BAW-10034, the internals vent valves have an average loss coefficient of 3.6. If this value is used, assuming no entrainment liquid by the steam, the core flooding rate shown as Curve A on Figure 2-1 is obtained for the 8.55 ft^2 split in the cold leg pipe. The flooding rate used in this supplement and in BAW-10034 assumed that the vent valves had to relieve 20% of the core region inlet flow as entrained liquid. This flooding rate is shown as Curve B on Figure 2-1. Both of these cases assumed that steam generation started when water started entering the core. The curves are almost identical after the initial surge of water into the core. The higher surge obtained by assuming no entrainment would cause a slightly lower (7F) peak cladding temperature than the one shown on Figure 1-2.

Using the reflood method which assumes 20% liquid entrainment, another run was made for the 8.55 ft^2 split in the cold leg piping which assumed that only 7 of the 8 vent valves opened. The flooding rates for these two cases are shown on Figure 2-2. The peak cladding temperature for the 7 vent valve case goes up to 2297F which is only 13F higher than the 8 vent valve case.

A further study was made which assumed that the flooding rate decreased to

2 in./sec. after the initial insurge. This assumed curve is shown as the dotted line on Figure 2-2. The temperature transient for this case is shown on Figure 2-3 along with the base temperature curve shown on Figure 1-2. As can be seen, the peak temperature is unaffected and the temperature is decreasing but at a slower rate.

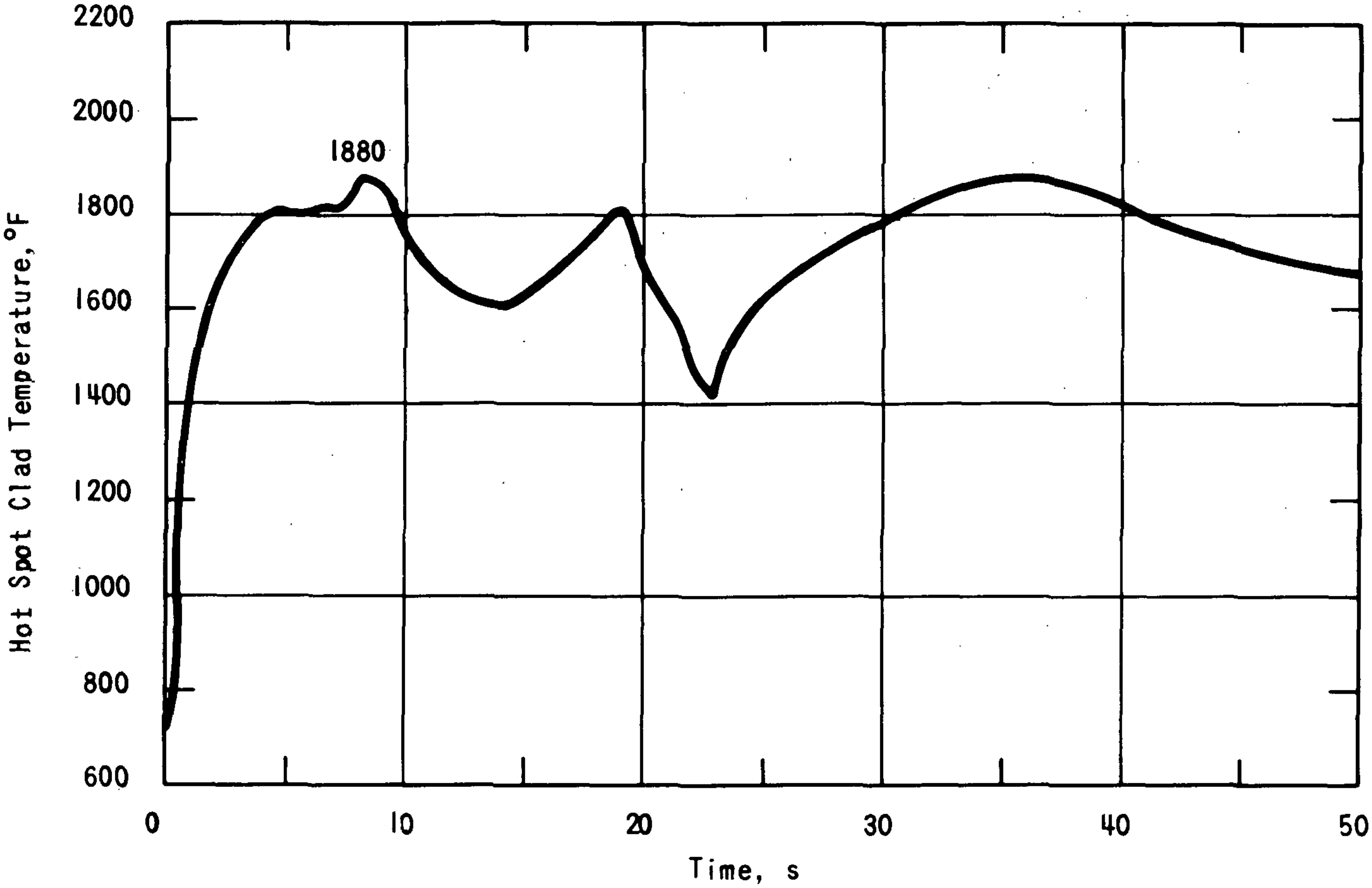
In order to show the value of the vent valves and to answer questions about entrainment, another reflood run was made with a slightly different model. In this run it was assumed that no steam generation took place until the water level in the core reached 18 inches. At that time the total core heat was transferred to the liquid to generate steam. The heat transfer coefficient in the portion of the core covered with steam was assumed to be 1/3 of the value used in the portion of the core covered with water. The heat transfer coefficient in the portion of the core covered with water was equal to or greater than the coefficient derived from FLECHT data. In addition, when the water level reached 18 inches, it was assumed that 20% of the incoming core region flow left the core as steam in addition to the steam which was generated. The reflood model has the core and the core bypass regions lumped together although the flow path through the bypass is much more restrictive. This means that the 20% core region flow could be interpreted as 31% of the core flow. The core flooding rate obtained by using this model is shown on Figure 2-4. The peak cladding temperature was 2278F. A plot showing the ratio of the steam leaving the core to the water entering the core region is shown as Curve A on Figure 2-5. The steam flow used in obtaining this ratio is that generated by core heat plus the 31% of the core inlet flow which was assumed to leave the core as steam. For comparison, Curve B in this same figure shows the ratio assuming no entrainment.

Request 3

Discuss the general applicability of the FLECHT tests to B&W reactor LOCA analyses.

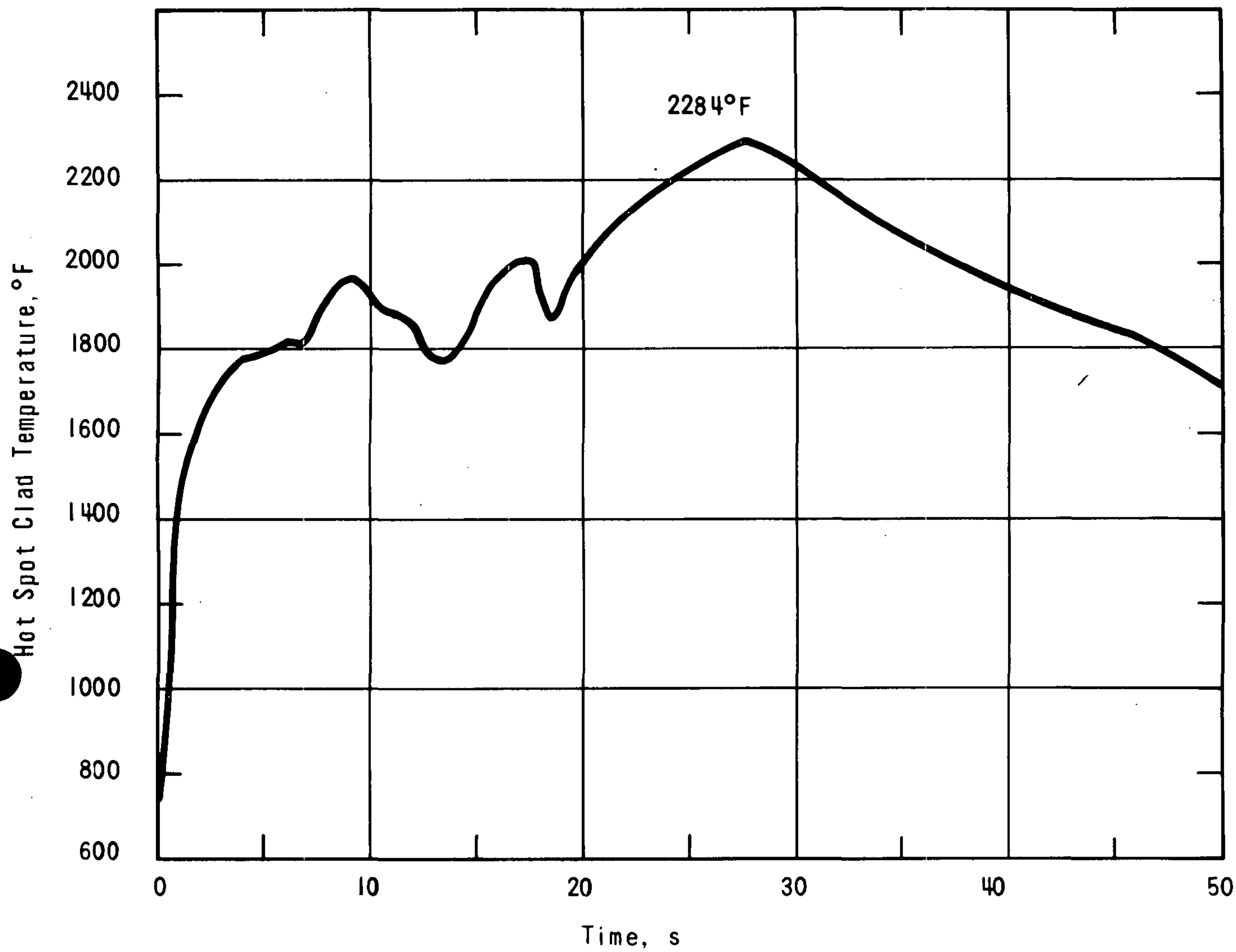
Response:

To relate this analysis to the FLECHT tests, on pages 3-123 through 3-126 of the FLECHT final report (WCAP-7665), it is stated that a 10% carryover is indicated at a flooding rate of 1 in./sec. However, Figure 3-76 indicates essentially no carryover in the first 50 seconds of reflood. The first 50 seconds of reflood is the critical time as far as cladding temperatures are concerned. Heat balances made on Run 9379 and examination of the pressure drop data indicates essentially no carryover in the first 20 seconds of reflood and perhaps none out to 60 seconds. The heat balance on Run 9379 indicates that 40 to 45% of the bundle inlet flow goes out as steam. This same heat balance shows that $\sim$10% of the incoming fluid has either been entrained by the steam or that this 10% is additional steam generated by cooling the housing surrounding the bundle. Considering that the average bundle heat rate if FLECHT is more than a factor of two greater than the average core heat rate, the methods used in calculating the flooding rate in this supplement and in BAW-10034 are apparently conservative in that 20% of the core region inlet flow was assumed to be entrained by the steam.



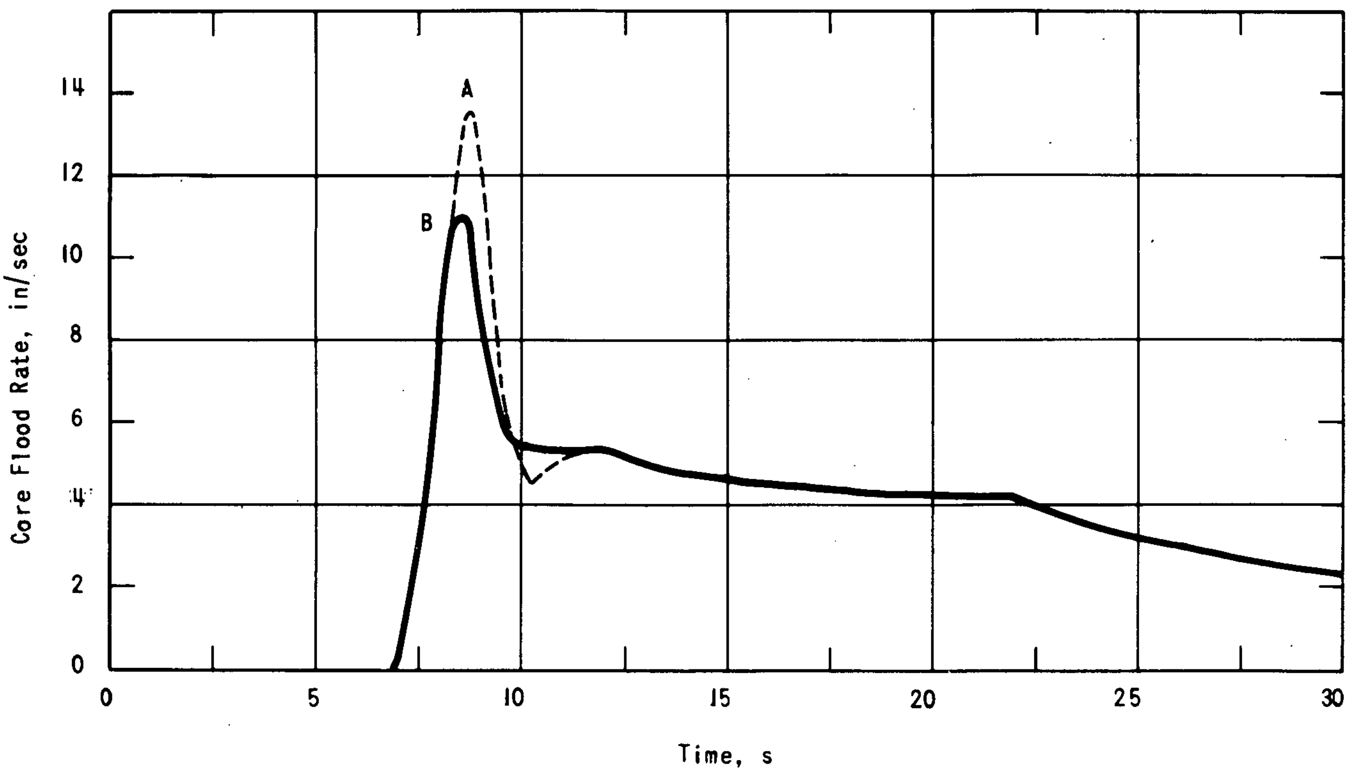
HOT SPOT CLADDING TEMPERATURE FOR
8.55 FT^2 SPLIT IN COLD LEG PIPE,
LOCKED PUMP ROTOR RESISTANCE, C_D=1.0

Figure 1-1



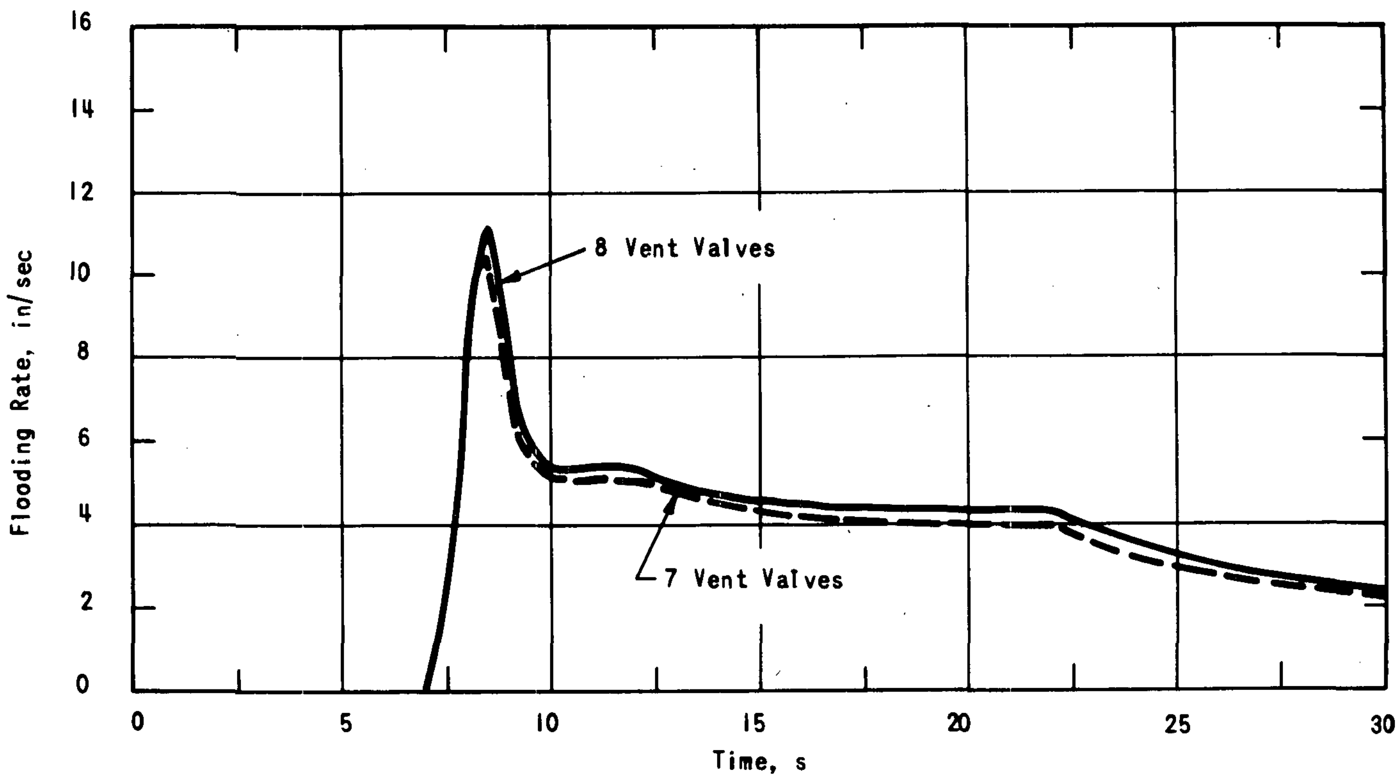
HOT SPOT CLADDING TEMPERATURE FOR $8.55\ FT^2$ SPLIT IN COLD LEG PIPE, $C_D = 1.0$

Figure 1-2



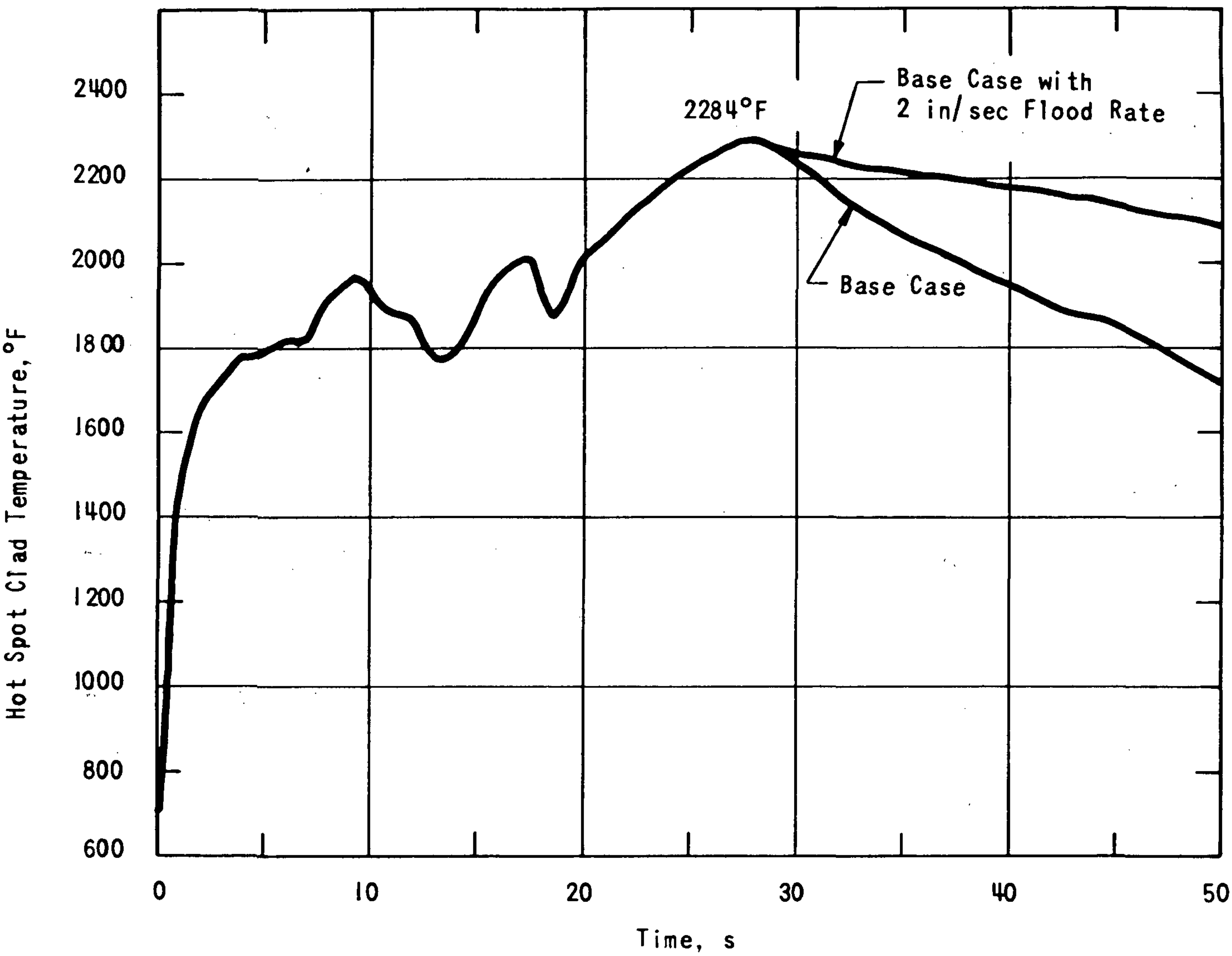
CORE FLOODING RATE FOR 8.55 FT^2
SPLIT IN COLD LEG PIPE, $C_D = 1.0$,
WITH AND WITHOUT LIQUID ENTRAINMENT
IN STEAM

Figure 2-1



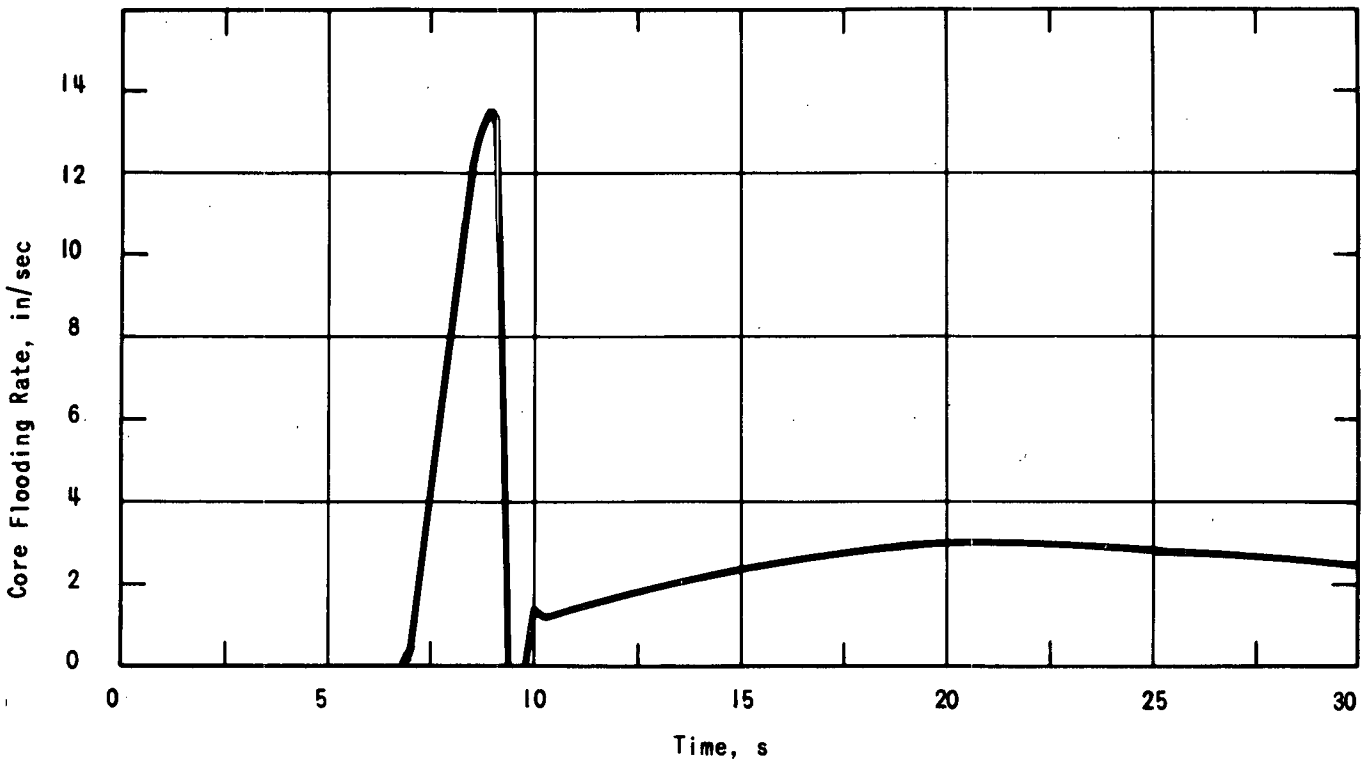
CORE FLOODING RATE FOR 8.55 FT^2
SPLIT IN COLD LEG PIPE, C_D = 1.0,
WITH 7 AND 8 VENT VALVES FUNCTIONING

Figure 2-2



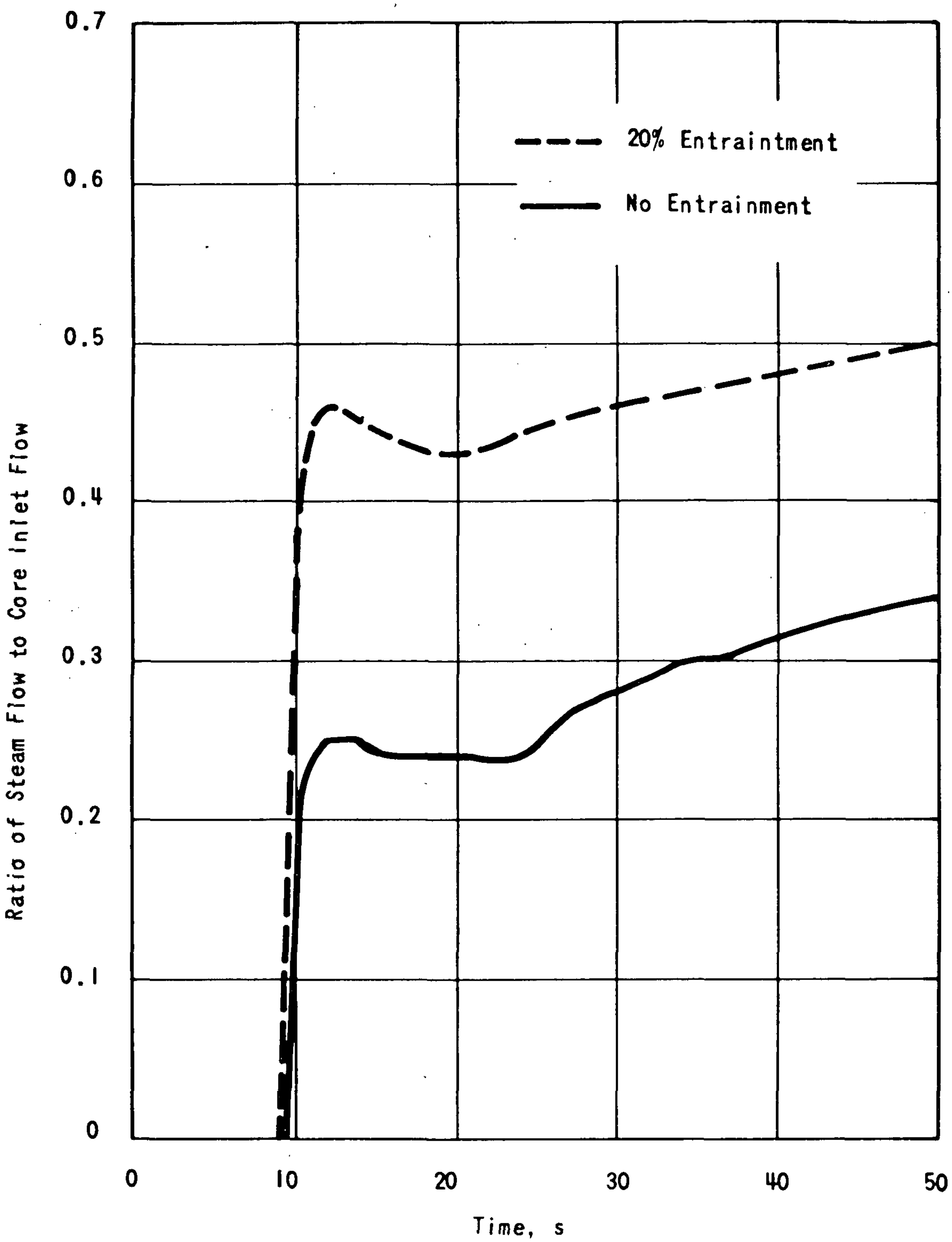
HOT SPOT CLADDING TEMPERATURE FOR 8.55 FT^2 SPLIT IN COLD LEG PIPE, C_D = 1.0, WITH ASSUMED 2 IN/SEC CORE FLOODING RATE

Figure 2-3



CORE FLOODING RATE FOR 8.55 FT^2 SPLIT IN COLD LEG PIPE, C_D = 1.0, WITH 20% OF INCOMING FLOW BEING VENTED AS STEAM IN ADDITION TO STEAM GENERATED IN THE CORE

Figure 2-4



RATIO OF STEAM LEAVING THE CORE
TO WATER ENTERING CORE REGION

Figure 2-5

DUKE POWER COMPANY

OCONEE NUCLEAR STATION

UNITS 2 AND 3

A P P L I C A T I O N   F O R   L I C E N S E S

DOCKETS 50-270 AND -287

FSAR SUPPLEMENT 11

Answers to Assistant Director for Pressurized Water Reactors
Questions of April 27, 1972

Submitted with FSAR Revision 20
May 25, 1972

LIST OF EFFECTIVE PAGES

FSAR SUPPLEMENT 11

1.0 GENERAL

Question 1.1.1

Please provide information regarding the extent to which experience with Oconee Unit 1 will affect the design and operation of Units 2 and 3.

Answer:

In the interest of standardization at the plant, the design of Units 2 and 3 will be functionally similar to Unit 1. This has obvious benefits for the operating personnel and should improve the reliability of Units 2 and 3. During the design, erection and testing phases, any problems encountered which caused a revision to equipment or systems for Unit 1 were immediately considered for Units 2 and 3. The revisions made to the later units were sometimes different in particulars but not in function due to greater design and schedule flexibility for these units.

An example would be the reactor coolant pumps where Westinghouse pumps were used on Unit 1 due to problems encountered during the testing of the Bingham pumps at the factory. Resolution of the Bingham problems was made in time to use them on Units 2 and 3. Even though the pumps are made by different manufacturers with some variation in seal and leak-off arrangements, the functional differences are slight.

Other design revisions necessary during Unit 1 startup will be made as needed on Units 2 and 3.

Experience in the startup and checkout of Unit 1 will be used in the checkout of Units 2 and 3. Areas of equipment startup problems experienced on Unit 1 startup will be closely monitored on Units 2 and 3. Procedures used and problems experienced in the Unit 1 checkout will be studied to determine appropriate improvements for Units 2 and 3.

Question 1.1.2

Please provide information regarding the extent to which Units 2 and 3 will be affected by the current Appendix A of 10 CFR 50, General Design Criteria for Nuclear Power Plants.

Answer:

The design of Units 1, 2 and 3 met the intent of the General Design Criteria (proposed rule-making published for 10CFR50 in the Federal Register of July 11, 1967) in effect during the design period for these units. FSAR Appendix 1A documents the applicant's agreement with the intent of each of these criteria. We have reviewed the criteria in the current Appendix A of 10CFR50, General Design Criteria for Nuclear Power Plants. The table 1.1.2-1 summarizes whether or not Units 2 and 3 will meet our understanding of the intent of the current General Design Criteria.

GENERAL DESIGN CRITERIA
TABLE 1.1.2-1
Sheet 1 of 6

Criterion	Title	Units 2 & 3 Meet Intent	Units 2 & 3 Do Not Meet Intent	Comments
1	Quality Standards and Records	x		
2	Design Bases for Protection Against Natural Phenomena	x		The combinations of effects of normal and accident conditions with the effects of natural phenomena which were established for Units 2 and 3 design bases are discussed in 1A.2, 1C.1, and 1C.2.
3	Fire Protection		x	Design meets intent of criterion with respect to fire detection and fighting systems, material usage, and minimizing the effect of fires. Rupture of High Pressure Service Water System piping in Auxiliary Building possibly could impair effectiveness of some equipment.
4	Environmental and Missile Design Bases		x	Limited sections of piping in Auxiliary Building for which intent of criterion is not met.
5	Sharing of Structures, Systems, & Components	x		See 1A.4 and Section 8.
10	Reactor Design	x		
11	Reactor Inherent Protection	x		
12	Suppression of Reactor Power Oscillations	x		

GENERAL DESIGN CRITERIA
TABLE 1.1.2-1
Sheet 2 of 6

Criterion	Title	Units 2 & 3 Meet Intent	Units 2 & 3 Do Not Meet Intent	Comments
13	Instrumentation and Control	x		See 1A.12 and 1A.13
14	Reactor Coolant Pressure Boundary	x		
15	Reactor Coolant System Design	x		
16	Containment Design	x		
17	Electrical Power Systems	x		See 1A.39
18	Inspection and Testing of Electrical Power Systems	x		
19	Control Room	x		See 1A.11
20	Protection System Functions	x		See 1A.14 and 1A.15
21	Protection System Reliability and Testability	x		See 1A.19
22	Protection System Independence	x		See 1A.20
23	Protection System Failure Modes	x		See 1A.26
24	Separation of Protection and Control Systems	x		See 1A.22
25	Protection System Requirements for Reactivity Control Malfunctions	x		See 1A.13, 1A.14, and 1A.31
26	Reactivity Control System Redundancy and Capability	x		See 1A.27, 1A.28, and 1A.30

GENERAL DESIGN CRITERIA
TABLE 1.1.2-1
Sheet 3 of 6

Criterion	Title	Units 2 & 3 Meet Intent	Units 2 & 3 Do Not Meet Intent	Comments
27	Combined Reactivity Control Capability	x		
28	Reactivity Limits	x		
29	Protection Against Anticipated Operational Occurrences	x		
30	Quality of Reactor Pressure Boundary	x		See 1A.16
31	Fracture Prevention of Reactor Coolant Pressure Boundary	x		
32	Inspection of Reactor Coolant Pressure Boundary		x	See Question 4.5
33	Reactor Coolant Makeup		x	Operator action of BS valves is required to meet the intent of criterion
34	Residual Heat Removal		x	Single failure not met on the single suction line from the Reactor Building emergency sump
35	Emergency Core Cooling	x		
36	Inspection of Emergency Core Cooling System		x	Cannot inspect embedded portion of the suction line from the Reactor Building sump
37	Testing of Emergency Core Cooling System	x		

GENERAL DESIGN CRITERIA
TABLE 1.1.2-1
Sheet 4 of 6

Criterion	Title	Units 2 & 3 Meet Intent	Units 2 & 3 Do Not Meet Intent	Comments
38	Containment Heat Removal	x		
39	Inspection of Containment Heat Removal System	x		
40	Testing of Containment Heat Removal System	x		
41	Containment Atmosphere Cleanup	x		Applies to Reactor Building Purge System and Hydrogen Purge System
42	Inspection of Containment Atmosphere Cleanup Systems	x		Applies to Reactor Building Purge System and Hydrogen Purge System
43	Testing of Containment Atmosphere Cleanup Systems	x		Applies to Reactor Building Purge System and Hydrogen Purge System
44	Cooling Water	x		
45	Inspection of Cooling Water System	x		
46	Testing of Cooling Water System	x		
50	Containment Design Basis	x		See Supplement 6, Question 2, page FSAR Supplement 6-22
51	Fracture Prevention of Containment Pressure Boundary	x		
52	Capability for Containment Leakage Rate Testing	x		

GENERAL DESIGN CRITERIA
TABLE 1.1.2-1
Sheet 5 of 6

Criterion	Title	Units 2 & 3 Meet Intent	Units 2 & 3 Do Not Meet Intent	Comments
53	Provisions for Containment Inspection and Testing	x		
54	Systems Penetrating Containment	x		
55	Reactor Coolant Pressure Boundary Penetrating Containment		x	One example is seal injection line. The valve outside the Reactor Building is a check valve.
56	Primary Containment Isolation		x	Examples: (a) Suction line from Reactor Building emergency sump does not have valve inside the Reactor Building. (b) Reactor Building pressure sensing instrumentation lines have manually operated, normally open isolation valves inside the Reactor Building.
57	Closed Systems Isolation Valves		x	One example is Component Cooling System supply to Reactor Building.
60	Control of Releases of Radioactive Materials to the Environment	x		
61	Fuel Storage and Handling and Radioactivity Control	x		
62	Prevention of Criticality in Fuel Storage and Handling	x		

GENERAL DESIGN CRITERIA
TABLE 1.1.2-1
Sheet 6 of 6

Criterion	Title	Units 2 & 3 Meet Intent	Units 2 & 3 Do Not Meet Intent	Comments
63	Monitoring Fuel and Waste Storage	x		See 1A.18
64	Monitoring Radioactivity Releases	x		See 1A.17 and 1A.70

Question 1.1.3

Please provide information regarding the extent to which Units 2 and 3 will be compatible with the intent of current issued Safety Guides with particular emphasis on Guides 1, 4, 7, 13, 16, 20, 21.

Answer:

Safety Guide 1, "Net Positive Suction Head for Emergency Core Cooling and Containment Heat Removal System Pumps" is addressed in the answer to question 6.1 of the Supplement. This answer applies to Oconee Units 1, 2 and 3.

The specific assumptions listed in Safety Guide 4, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss of Coolant Accident for Pressurized Water Reactors," were used in the evaluation of the environmental effects of the maximum hypothetical accident (see FSAR Section 14.2.2.4).

To address Safety Guide 7, "Control of Combustible as Concentrations in Containment Following a Loss of Coolant Accident," the following assumptions have been used in the calculation of the hydrogen generated:

(1) A maximum hypothetical accident (MHA)

For the MHA, the following assumptions were used to determine the hydrogen generated by radiolysis:

(a) 100 percent of the core inventory of noble gases is released to the Reactor Building atmosphere.

(b) 50 percent of the halogens and 1 percent of the solid fission products are ultimately mixed with the coolant water.

(c) The fraction of fission product radiation energy absorbed by the coolant is:

1. 0.0 betas from fission products in the fuel rods.

2. 1.0 for betas from fission products intimately mixed with the coolant.

3. 0.1 for gamma from fission products in the fuel rods for coolant in the core region.

4. 1.0 for gammas from fission products intimately mixed with the coolant.

(d) A hydrogen generation constant (G value) of 0.5 molecules Hz/100 eV absorbed energy.

(2) 5 percent zironcium-water reaction

(3) Chemical Hydrogen Sources

(a) Aluminum corrosion - none

(b) Zinc corrosion

1. Corrosion rate - 200 mg Zn/m^2 - day

2. 5812.5 lbs zinc in building

(4) Purging commences at 4.0 volume percent hydrogen.

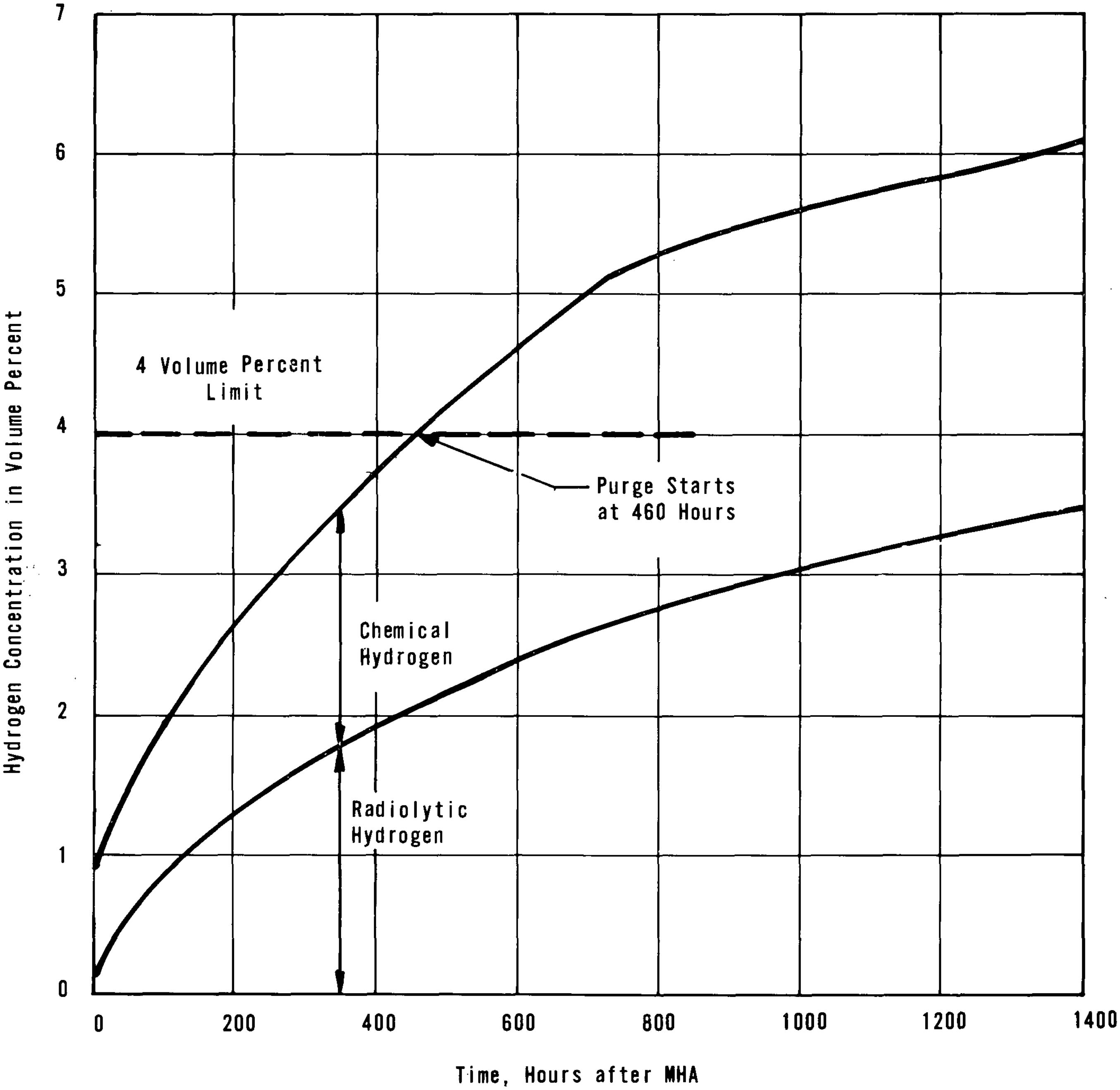
Based on the above assumptions, the hydrogen concentration in the reactor building does not reach the control limit until 460 hours after the accident as shown in Figure 1.1.3-1. The initial purge rate required to limit the hydrogen buildup is 45.3 SCFM. The doses at the boundary of the low population zone due to purging are 12.7 Rem thyroid and .05 Rem whole body.

The spent fuel pool designed for Oconee Units 1, 2 and 3 conforms to Safety Guide 13, "Fuel Storage Design Basis." The fuel pools were designed for tornado wind and missiles, turbine generator missile, and seismic conditions as listed in Table 5.5. The spent fuel pools were designed for the postulated cast drop accident as described in Section 5.7.1.2.

Technical Specification 6.6 for Unit 1 meets the intent of Safety Guide 16, "Reporting of Operation Information," except that certain items involving incidents with license material are not included in this specification, but are adequately covered by applicable AEC regulations. The technical specifications for Units 2 and 3 will reflect the requirements of Safety Guide 16.

Safety Guide 20, "Vibration Measurements on Reactor Internals," a vibration testing program which will be included in the pre-operational test program for Oconee Unit 1 is being developed, and will be discussed with the AEC/DRL staff before resumption of Hot Functional Testing on Unit 1. Unit 1 is considered the prototype and Units 2 and 3 are considered similar to the prototype design. A program is under development for Units 2 and 3 also.

Units 2 and 3 will fully comply with the intent of the current Safety Guide 21, "Measuring and Reporting of The Effluents from Nuclear Power Plants." Unit 1 will essentially be in compliance with Safety Guide 21 at the time that the facility license for Unit 1 is issued, and will be fully in compliance at the time that the facility license for Unit 2 is issued.



REACTOR BUILDING HYDROGEN CONCENTRATION
FOLLOWING AN MHA SAFETY GUIDE 7 ASSUMPTIONS

Figure 1.1.3-1

2.0 SITE AND ENVIRONMENT

Question 2.1

We understand that a minimum of one year of on-site meteorological data have been accumulated after the filling of Lake Keowee with water and that wind calibration problems with the data previously presented for the period from June 19, 1968 through June 18, 1969 have been resolved. Provide the joint frequency distribution of wind direction and wind speed by stability class for annual period(s) of record after the filling of Lake Keowee. Use vertical temperature difference measurements in currently acceptable classes to define atmospheric stability.

Answer:

Table 2.1-1 depicts joint frequencies of wind direction and speed by stability class for the period March 15, 1970 through March 14, 1972. Stability is defined in terms of vertical temperature gradient and indexed by the following schedule:

Stability Class	Vertical Temperature Gradient Class Interval
G	> + 3.3°F in 145'
F	+ 1.3 to + 3.3°F in 145'
E	- 0.5 to + 1.2°F in 145'
D	- 1.2 to - 0.6°F in 145'
B-C	- 1.5 to - 1.3°F in 145'
A	< - 1.5°F in 145'

Question 2.2

Discuss the effect of heated water discharges from the three nuclear units on the diffusion climatology of the site.

Answer:

The incremental offset in the diffusion climatology due to heated water discharge over and above effects already noted since filling of the reservoir should be in the direction of improvement, but probably will not be of a magnitude to warrant special emphasis. The effect of warmer surface waters in the vicinity of the discharge would both increase the speed change of air flow from land to water and decrease the change of wind range for such trajectories. (Reference 1. D. H. Slade, Atmospheric Dispersion Over Chesapeake Bay, Monthly Weather Rev. 90(6) 217-224 (June 1962).)

In regard to further modification of low-level stability, additional enhancement would be tempered, at least to some extent, from effects of the relatively deep reservoir. A conservative assessment, then, would assume some improvement but of minimal inpact on the total climate.

Docket 50-270 and -287
FSAR Supplement 11
May 25, 1972

OCONEE METEOROLOGICAL SURVEY TOWER DATA FOR PERIOD OF MAR. 15,1970 THRU MAR. 14,1972
SUMMARY OF PASQUILL A WIND OCCURRENCES BY SECTOR + SPEED CLASS (NO. OCCURR,PERCENT)
DATE OF REPORT 5-16-72

WIND SECTOR	ITEM	SECTOR TOTAL	WIND SPEED CLASS 1.0-3.2 .45-1.49	3.3-5.5 1.5-2.49	5.6-7.8 2.5-3.49	7.9-10.0 3.5-4.49	10.1-12.3 4.5-5.49	12.4-14.5 5.5-6.49	14.6-16.7 6.5-7.49	16.8-19.0 7.5-8.49	19.1-21.2 8.5-9.49	>21.2 MPH >=9.5 M/S
360.0	NO	132	15	68	35	8	4	0	2	0	0	0
-N-	PCT	0.92	0.10	0.47	0.24	0.05	0.03	0.00	0.01	0.00	0.00	0.00
22.5	NO	99	5	48	26	10	5	3	2	0	0	0
-NNE-	PCT	0.69	0.03	0.33	0.18	0.07	0.03	0.02	0.01	0.00	0.00	0.00
45.0	NO	172	10	56	30	16	23	18	10	9	0	0
-NE-	PCT	1.20	0.07	0.39	0.21	0.11	0.16	0.13	0.07	0.06	0.00	0.00
67.5	NO	161	8	29	31	20	32	25	13	2	1	0
-ENE-	PCT	1.12	0.05	0.20	0.22	0.14	0.22	0.17	0.09	0.01	0.01	0.00
90.0	NO	165	8	47	52	32	18	6	2	0	0	0
-E-	PCT	1.15	0.05	0.33	0.36	0.22	0.13	0.04	0.01	0.00	0.00	0.00
112.5	NO	137	18	59	35	12	11	2	0	0	0	0
-ESE-	PCT	0.96	0.13	0.41	0.24	0.08	0.08	0.01	0.00	0.00	0.00	0.00
135.0	NO	255	15	76	81	50	22	8	2	1	0	0
-SE-	PCT	1.78	0.10	0.53	0.56	0.35	0.15	0.05	0.01	0.01	0.00	0.00
157.5	NO	200	5	31	63	52	31	12	4	2	0	0
-SSE-	PCT	1.39	0.03	0.22	0.44	0.36	0.22	0.08	0.03	0.01	0.00	0.00
180.0	NO	270	11	49	64	56	45	27	14	2	2	0
-S-	PCT	1.88	0.08	0.34	0.45	0.39	0.31	0.19	0.10	0.01	0.01	0.00
202.5	NO	374	4	53	105	86	67	32	18	8	0	1
-SSW-	PCT	2.61	0.03	0.37	0.73	0.60	0.47	0.22	0.13	0.05	0.00	0.01
225.0	NO	388	5	81	113	60	44	27	33	17	5	3
-SW-	PCT	2.71	0.03	0.56	0.79	0.42	0.31	0.19	0.23	0.12	0.03	0.02
247.5	NO	204	4	50	47	17	19	16	17	14	5	15
-WSW-	PCT	1.42	0.03	0.35	0.33	0.12	0.13	0.11	0.12	0.10	0.03	0.10
270.0	NO	184	8	53	35	8	22	19	16	9	9	5
-W-	PCT	1.28	0.05	0.37	0.24	0.05	0.15	0.13	0.11	0.06	0.06	0.03
292.5	NO	113	7	31	15	10	6	8	8	8	6	14
-WNW-	PCT	0.79	0.05	0.22	0.10	0.07	0.04	0.05	0.05	0.05	0.04	0.10
315.0	NO	123	14	41	15	12	3	6	9	4	5	14
-NW-	PCT	0.86	0.10	0.29	0.10	0.08	0.02	0.04	0.06	0.03	0.03	0.10
337.5	NO	84	12	38	21	4	4	2	2	0	1	0
-NNW-	PCT	0.59	0.08	0.26	0.15	0.03	0.03	0.01	0.01	0.00	0.01	0.00
CALM	NO	0										
	PCT	0.00										
TOTAL	NO	3061	149	810	758	453	356	211	152	76	34	52
	PCT	21.36	1.04	5.65	5.36	3.16	2.48	1.47	1.06	0.53	0.24	0.36

TOTAL VALID OBSERVATIONS 14333 TOTAL OBSERVATIONS 17545

OCONEE METEOROLOGICAL SURVEY TOWER DATA FOR PERIOD OF MAR. 15,1970 THRU MAR. 14,1972
SUMMARY OF PASQUILL B+C WIND OCCURRENCES BY SECTOR + SPEED CLASS (NO. OCCURR,PERCENT)
DATE OF REPORT 5-16-72

WIND SECTOR	ITEM	SECTOR TOTAL	WIND SPEED CLASS 1.0-3.2 .45-1.49	3.3-5.5 1.5-2.49	5.6-7.8 2.5-3.49	7.9-10.0 3.5-4.49	10.1-12.3 4.5-5.49	12.4-14.5 5,5-6.49	14.6-16.7 6.5-7.49	16.8-19.0 7.5-8.49	19.1-21.2 8.5-9.49	>21.2 MPH >=9.5 M/S
360.0	NO	20	3	8	3	4	0	0	2	0	0	0
-N-	PCT	0.14	0.02	0.05	0.02	0.03	0.00	0.00	0.01	0.00	0.00	0.00
22.5	NO	34	6	8	8	2	2	5	2	1	0	0
-NNE-	PCT	0.24	0.04	0.05	0.05	0.01	0.01	0.03	0.01	0.01	0.00	0.00
45.0	NO	57	3	8	9	11	7	9	6	3	1	0
-NE-	PCT	0.40	0.02	0.05	0.06	0.08	0.05	0.06	0.04	0.02	0.01	0.00
67.5	NO	52	0	10	2	12	9	7	7	3	1	1
-ENE-	PCT	0.36	0.00	0.07	0.01	0.08	0.06	0.05	0.05	0.02	0.01	0.01
90.0	NO	37	4	11	10	5	7	0	0	0	0	0
-E-	PCT	0.26	0.03	0.08	0.07	0.03	0.05	0.00	0.00	0.00	0.00	0.00
112.5	NO	32	5	9	12	4	2	0	0	0	0	0
-ESE-	PCT	0.22	0.03	0.06	0.08	0.03	0.01	0.00	0.00	0.00	0.00	0.00
135.0	NO	51	11	16	11	9	4	0	0	0	0	0
-SE-	PCT	0.36	0.08	0.11	0.08	0.06	0.03	0.00	0.00	0.00	0.00	0.00
157.5	NO	40	1	11	12	7	6	2	1	0	0	0
-SSE-	PCT	0.28	0.01	0.08	0.08	0.05	0.04	0.01	0.01	0.00	0.00	0.00
180.0	NO	48	5	9	6	8	10	4	3	2	0	1
-S-	PCT	0.33	0.03	0.06	0.04	0.05	0.07	0.03	0.02	0.01	0.00	0.01
202.5	NO	74	2	13	12	14	11	5	10	5	2	0
-SSW-	PCT	0.52	0.01	0.09	0.08	0.10	0.08	0.03	0.07	0.03	0.01	0.00
225.0	NO	75	7	9	8	18	7	11	10	2	3	0
-SW-	PCT	0.52	0.05	0.06	0.05	0.13	0.05	0.08	0.07	0.01	0.02	0.00
247.5	NO	37	3	6	4	3	2	7	2	4	0	6
-WSW-	PCT	0.26	0.02	0.04	0.03	0.02	0.01	0.05	0.01	0.03	0.00	0.04
270.0	NO	24	3	4	3	0	4	2	2	1	0	5
-W-	PCT	0.17	0.02	0.03	0.02	0.00	0.03	0.01	0.01	0.01	0.00	0.03
292.5	NO	21	2	9	0	0	0	0	3	3	1	3
-WNW-	PCT	0.15	0.01	0.06	0.00	0.00	0.00	0.00	0.02	0.02	0.01	0.02
315.0	NO	28	4	8	2	1	3	2	0	2	1	5
-NW-	PCT	0.20	0.03	0.05	0.01	0.01	0.02	0.01	0.00	0.01	0.01	0.03
337.5	NO	26	4	8	8	3	1	0	0	0	0	2
-NNW-	PCT	0.18	0.03	0.05	0.05	0.02	0.01	0.00	0.00	0.00	0.00	0.01
CALM	NO	0										
	PCT	0.00										
TOTAL	NO	656	63	147	110	101	75	54	48	26	9	23
	PCT	4.58	0.44	1.03	0.77	0.70	0.52	0.38	0.33	0.18	0.06	0.16

TOTAL VALID OBSERVATIONS 14333 TOTAL OBSERVATIONS 17545

Table 2.1-1

Page 3 of 6

Docket 50-270 and -287
FSAR Supplement 11
May 25, 1972

OCONEE METEOROLOGICAL SURVEY TOWER DATA FOR PERIOD OF MAR. 15,1970 THRU MAR. 14,1972

SUMMARY OF PASQUILL D WIND OCCURRENCES BY SECTOR + SPEED CLASS (NO. OCCURR,PERCENT)

DATE OF REPORT 5-16-72

WIND SECTOR	ITEM	SECTOR TOTAL	WIND SPEED CLASS 1.0-3.2 .45-1.49	3.3-5.5 1.5-2.49	5.6-7.8 2.5-3.49	7.9-10.0 3.5-4.49	10.1-12.3 4.5-5.49	12.4-14.5 5.5-6.49	14.6-16.7 6.5-7.49	16.8-19.0 7.5-8.49	19.1-21.2 8.5-9.49	>21.2 MPH >=9.5 M/S
360.0	NO	30	10	10	3	4	1	1	0	1	0	0
-N-	PCT	0.21	0.07	0.07	0.02	0.03	0.01	0.01	0.00	0.01	0.00	0.00
22.5	NO	43	2	8	12	11	4	6	0	0	0	0
-NNE-	PCT	0.30	0.01	0.05	0.08	0.08	0.03	0.04	0.00	0.00	0.00	0.00
45.0	NO	95	7	10	18	9	18	19	11	2	1	0
-NE-	PCT	0.66	0.05	0.07	0.13	0.06	0.13	0.13	0.08	0.01	0.01	0.00
67.5	NO	55	4	7	10	12	13	6	0	3	0	0
-ENE-	PCT	0.38	0.03	0.05	0.07	0.08	0.09	0.04	0.00	0.02	0.00	0.00
90.0	NO	63	6	20	14	8	9	4	1	1	0	0
-E-	PCT	0.44	0.04	0.14	0.10	0.05	0.06	0.03	0.01	0.01	0.00	0.00
112.5	NO	26	4	12	7	3	0	0	0	0	0	0
-ESE-	PCT	0.18	0.03	0.08	0.05	0.02	0.00	0.00	0.00	0.00	0.00	0.00
135.0	NO	35	7	12	7	7	2	0	0	0	0	0
-SE-	PCT	0.24	0.05	0.08	0.05	0.05	0.01	0.00	0.00	0.00	0.00	0.00
157.5	NO	43	6	14	10	8	3	1	1	0	0	0
-SSE-	PCT	0.30	0.04	0.10	0.07	0.05	0.02	0.01	0.01	0.00	0.00	0.00
180.0	NO	44	4	7	7	4	7	9	3	3	0	0
-S-	PCT	0.31	0.03	0.05	0.05	0.03	0.05	0.06	0.02	0.02	0.00	0.00
202.5	NO	65	3	9	16	8	14	9	4	1	1	0
-SSW-	PCT	0.45	0.02	0.06	0.11	0.05	0.10	0.06	0.03	0.01	0.01	0.00
225.0	NO	98	2	23	25	13	9	14	11	1	0	0
-SW-	PCT	0.68	0.01	0.16	0.17	0.09	0.06	0.10	0.08	0.01	0.00	0.00
247.5	NO	38	5	10	2	2	5	8	2	1	0	3
-WSW-	PCT	0.26	0.03	0.07	0.01	0.01	0.03	0.05	0.01	0.01	0.00	0.02
270.0	NO	51	8	10	3	5	4	6	5	3	0	7
-W-	PCT	0.36	0.05	0.07	0.02	0.03	0.03	0.04	0.03	0.02	0.00	0.05
292.5	NO	24	2	6	2	1	1	2	0	3	1	6
-WNW-	PCT	0.17	0.01	0.04	0.01	0.01	0.01	0.01	0.00	0.02	0.01	0.04
315.0	NO	36	14	9	1	1	1	1	1	3	1	4
-NW-	PCT	0.25	0.10	0.06	0.01	0.01	0.01	0.01	0.01	0.02	0.01	0.03
337.5	NO	26	6	9	6	3	0	0	0	1	0	1
-NNW-	PCT	0.18	0.04	0.06	0.04	0.02	0.00	0.00	0.00	0.01	0.00	0.01
CALM	NO	0										
	PCT	0.00										
TOTAL	NO	772	90	176	143	99	91	86	39	23	4	21
	PCT	5.38	0.63	1.23	1.00	0.69	0.63	0.60	0.27	0.16	0.03	0.15

TOTAL VALID OBSERVATIONS 14333

TOTAL OBSERVATIONS 17545

Table 2.1-1

OCONEE METEOROLOGICAL SURVEY TOWER DATA — FOR PERIOD OF MAR. 15,1970 THRU MAR. 14,1972

SUMMARY OF PASQUILL E — WIND OCCURRENCES BY SECTOR + SPEED CLASS (NO. OCCURR,PERCENT)

DATE OF REPORT 5-16-72

WIND SECTOR	ITEM	SECTOR TOTAL	WIND SPEED CLASS 1.0-3.2 .45-1.49	3.3-5.5 1.5-2.49	5.6-7.8 2.5-3.49	7.9-10.0 3.5-4.49	10.1-12.3 4.5-5.49	12.4-14.5 5.5-6.49	14.6-16.7 6.5-7.49	16.8-19.0 7.5-8.49	19.1-21.2 8.5-9.49	>21.2 MPH >=9.5 M/S
360.0	NO	391	50	135	129	49	19	4	3	0	0	2
-N-	PCT	2.73	0.35	0.94	0.90	0.34	0.13	0.03	0.02	0.00	0.00	0.01
22.5	NO	392	35	92	126	64	44	21	4	6	0	0
-NNE-	PCT	2.73	0.24	0.64	0.88	0.45	0.31	0.15	0.03	0.04	0.00	0.00
45.0	NO	611	42	87	120	129	108	90	25	8	2	0
-NE-	PCT	4.26	0.29	0.61	0.84	0.90	0.75	0.63	0.17	0.05	0.01	0.00
67.5	NO	390	30	84	93	92	39	27	15	9	1	0
-ENE-	PCT	2.72	0.21	0.59	0.65	0.64	0.27	0.19	0.10	0.06	0.01	0.00
90.0	NO	313	33	92	106	46	24	8	2	0	2	0
-E-	PCT	2.18	0.23	0.64	0.74	0.32	0.17	0.05	0.01	0.00	0.01	0.00
112.5	NO	165	34	56	47	11	13	2	2	0	0	0
-ESE-	PCT	1.15	0.24	0.39	0.33	0.08	0.09	0.01	0.01	0.00	0.00	0.00
135.0	NO	182	39	57	42	21	17	3	2	0	1	0
-SE-	PCT	1.27	0.27	0.40	0.29	0.15	0.12	0.02	0.01	0.00	0.01	0.00
157.5	NO	166	21	43	44	35	20	2	1	0	0	0
-SSE-	PCT	1.16	0.15	0.30	0.31	0.24	0.14	0.01	0.01	0.00	0.00	0.00
180.0	NO	217	31	36	58	38	25	19	7	2	1	0
-S-	PCT	1.51	0.22	0.25	0.40	0.26	0.17	0.13	0.05	0.01	0.01	0.00
202.5	NO	401	18	64	75	82	73	49	28	12	0	0
-SSW-	PCT	2.80	0.13	0.45	0.52	0.57	0.51	0.34	0.20	0.08	0.00	0.00
225.0	NO	570	35	94	100	84	87	93	60	15	2	0
-SW-	PCT	3.98	0.24	0.65	0.70	0.59	0.61	0.65	0.42	0.10	0.01	0.00
247.5	NO	363	20	54	62	51	69	57	24	11	3	12
-WSW-	PCT	2.53	0.14	0.38	0.43	0.36	0.48	0.40	0.17	0.08	0.02	0.08
270.0	NO	364	39	79	37	26	33	52	32	28	16	22
-W-	PCT	2.54	0.27	0.55	0.26	0.18	0.23	0.36	0.22	0.20	0.11	0.15
292.5	NO	206	22	36	18	16	15	15	25	15	16	28
-WNW-	PCT	1.44	0.15	0.25	0.13	0.11	0.10	0.10	0.17	0.10	0.11	0.20
315.0	NO	275	36	82	50	24	15	15	8	21	5	19
-NW-	PCT	1.92	0.25	0.57	0.35	0.17	0.10	0.10	0.05	0.15	0.03	0.13
337.5	NO	233	38	89	55	19	14	8	4	0	0	6
-NNW-	PCT	1.63	0.26	0.62	0.38	0.13	0.10	0.05	0.03	0.00	0.00	0.04
CALM	NO	17										
	PCT	0.12										
TOTAL	NO	5239	523	1180	1162	787	615	465	242	127	49	89
	PCT	36.55	3.65	8.23	8.11	5.49	4.29	3.24	1.69	0.89	0.34	0.62

TOTAL VALID OBSERVATIONS 14333 — TOTAL OBSERVATIONS 17545

OCONEE METEOROLOGICAL SURVEY TOWER DATA FOR PERIOD OF MAR. 15,1970 THRU MAR. 14,1972

SUMMARY OF PASQUILL F WIND OCCURRENCES BY SECTOR + SPEED CLASS (NO. OCCURR,PERCENT)

DATE OF REPORT 5-16-72

WIND SECTOR	ITEM	SECTOR TOTAL	1.0-3.2 .45-1.49	3.3-5.5 1.5-2.49	5.6-7.8 2.5-3.49	7.9-10.0 3.5-4.49	10.1-12.3 4.5-5.49	12.4-14.5 5.5-6.49	14.6-16.7 6.5-7.49	16.8-19.0 7.5-8.49	19.1-21.2 8.5-9.49	>21.2 MPH >=9.5 M/S
360.0	NO	384	38	160	150	30	6	0	0	0	0	0
-N-	PCT	2.68	0.26	1.12	1.05	0.21	0.04	0.00	0.00	0.00	0.00	0.00
22.5	NO	213	24	93	76	16	1	2	1	0	0	0
-NNE-	PCT	1.48	0.17	0.65	0.53	0.11	0.01	0.01	0.01	0.00	0.00	0.00
45.0	NO	170	23	83	45	12	4	2	1	0	0	0
-NE-	PCT	1.19	0.16	0.58	0.31	0.08	0.03	0.01	0.01	0.00	0.00	0.00
67.5	NO	106	12	50	31	5	5	0	1	0	1	1
-ENE-	PCT	0.74	0.08	0.35	0.22	0.03	0.03	0.00	0.01	0.00	0.01	0.01
90.0	NO	88	19	30	31	5	3	0	0	0	0	0
-E-	PCT	0.61	0.13	0.21	0.22	0.03	0.02	0.00	0.00	0.00	0.00	0.00
112.5	NO	53	11	25	12	4	1	0	0	0	0	0
-ESE-	PCT	0.37	0.08	0.17	0.08	0.03	0.01	0.00	0.00	0.00	0.00	0.00
135.0	NO	84	9	33	26	13	3	0	0	0	0	0
-SE-	PCT	0.59	0.06	0.23	0.18	0.09	0.02	0.00	0.00	0.00	0.00	0.00
157.5	NO	84	10	26	26	17	5	0	0	0	0	0
-SSE-	PCT	0.59	0.07	0.18	0.18	0.12	0.03	0.00	0.00	0.00	0.00	0.00
180.0	NO	108	14	27	26	14	21	6	0	0	0	0
-S-	PCT	0.75	0.10	0.19	0.18	0.10	0.15	0.04	0.00	0.00	0.00	0.00
202.5	NO	124	8	31	35	24	12	9	3	1	1	0
-SSW-	PCT	0.86	0.05	0.22	0.24	0.17	0.08	0.06	0.02	0.01	0.01	0.00
225.0	NO	173	16	49	32	35	24	15	1	0	0	1
-SW-	PCT	1.21	0.11	0.34	0.22	0.24	0.17	0.10	0.01	0.00	0.00	0.01
247.5	NO	142	13	40	29	30	14	6	8	2	0	0
-WSW-	PCT	0.99	0.09	0.28	0.20	0.21	0.10	0.04	0.05	0.01	0.00	0.00
270.0	NO	185	34	58	29	20	15	10	11	6	2	0
-W-	PCT	1.29	0.24	0.40	0.20	0.14	0.10	0.07	0.08	0.04	0.01	0.00
292.5	NO	159	23	67	29	16	10	6	5	1	2	0
-WNW-	PCT	1.11	0.16	0.47	0.20	0.11	0.07	0.04	0.03	0.01	0.01	0.00
315.0	NO	246	39	123	50	19	6	4	1	2	1	1
-NW-	PCT	1.72	0.27	0.86	0.35	0.13	0.04	0.03	0.01	0.01	0.01	0.01
337.5	NO	337	38	155	104	30	5	4	1	0	0	0
-NNW-	PCT	2.35	0.26	1.08	0.72	0.21	0.03	0.03	0.01	0.00	0.00	0.00
CALM	NO	3										
	PCT	0.02										
TOTAL	NO	2656	331	1050	731	290	135	64	33	12	7	3
	PCT	18.53	2.31	7.33	5.10	2.02	0.94	0.45	0.23	0.08	0.05	0.02

TOTAL VALID OBSERVATIONS 14333

TOTAL OBSERVATIONS 17545

OCONEE METEOROLOGICAL SURVEY TOWER DATA FOR PERIOD OF MAR. 15,1970 THRU MAR. 14,1972

SUMMARY OF PASQUILL G WIND OCCURRENCES BY SECTOR + SPEED CLASS (NO. OCCURR,PERCENT)

DATE OF REPORT 5-16-72

WIND SECTOR	ITEM	SECTOR TOTAL	WIND SPEED CLASS 1.0-3.2 .45-1.49	3.3-5.5 1.5-2.49	5.6-7.8 2.5-3.49	7.9-10.0 3.5-4.49	10.1-12.3 4.5-5.49	12.4-14.5 5.5-6.49	14.6-16.7 6.5-7.49	16.8-19.0 7.5-8.49	19.1-21.2 8.5-9.49	>21.2 MPH >=9.5 M/S
360.0	NO	370	35	144	139	46	6	0	0	0	0	0
-N-	PCT	2.58	0.24	1.00	0.97	0.32	0.04	0.00	0.00	0.00	0.00	0.00
22.5	NO	143	28	69	38	8	0	0	0	0	0	0
-NNE-	PCT	1.00	0.20	0.48	0.26	0.05	0.00	0.00	0.00	0.00	0.00	0.00
45.0	NO	97	18	41	27	8	2	1	0	0	0	0
-NE-	PCT	0.68	0.13	0.29	0.19	0.05	0.01	0.01	0.00	0.00	0.00	0.00
67.5	NO	72	10	31	18	11	2	0	0	0	0	0
-ENE-	PCT	0.50	0.07	0.22	0.13	0.08	0.01	0.00	0.00	0.00	0.00	0.00
90.0	NO	55	7	27	13	5	1	2	0	0	0	0
-E-	PCT	0.38	0.05	0.19	0.09	0.03	0.01	0.01	0.00	0.00	0.00	0.00
112.5	NO	31	6	14	7	1	2	1	0	0	0	0
-ESE-	PCT	0.22	0.04	0.10	0.05	0.01	0.01	0.01	0.00	0.00	0.00	0.00
135.0	NO	102	11	36	39	14	2	0	0	0	0	0
-SE-	PCT	0.71	0.08	0.25	0.27	0.10	0.01	0.00	0.00	0.00	0.00	0.00
157.5	NO	65	11	22	23	8	1	0	0	0	0	0
-SSE-	PCT	0.45	0.08	0.15	0.16	0.05	0.01	0.00	0.00	0.00	0.00	0.00
180.0	NO	55	8	18	17	10	1	1	0	0	0	0
-S-	PCT	0.38	0.05	0.13	0.12	0.07	0.01	0.01	0.00	0.00	0.00	0.00
202.5	NO	64	11	23	18	10	2	0	0	0	0	0
-SSW-	PCT	0.45	0.08	0.16	0.13	0.07	0.01	0.00	0.00	0.00	0.00	0.00
225.0	NO	142	19	42	46	25	8	1	0	1	0	0
-SW-	PCT	0.99	0.13	0.29	0.32	0.17	0.05	0.01	0.00	0.01	0.00	0.00
247.5	NO	111	23	40	29	10	5	3	0	0	0	1
-WSW-	PCT	0.77	0.16	0.28	0.20	0.07	0.03	0.02	0.00	0.00	0.00	0.01
270.0	NO	99	18	37	24	10	5	2	2	1	0	0
-W-	PCT	0.69	0.13	0.26	0.17	0.07	0.03	0.01	0.01	0.01	0.00	0.00
292.5	NO	110	26	52	19	4	4	3	2	0	0	0
-WNW-	PCT	0.77	0.18	0.36	0.13	0.03	0.03	0.02	0.01	0.00	0.00	0.00
315.0	NO	168	35	80	37	8	4	3	0	1	0	0
-NW-	PCT	1.17	0.24	0.56	0.26	0.05	0.03	0.02	0.00	0.01	0.00	0.00
337.5	NO	242	33	100	77	26	4	1	0	0	0	1
-NNW-	PCT	1.69	0.23	0.70	0.54	0.18	0.03	0.01	0.00	0.00	0.00	0.01
CALM	NO	3										
	PCT	0.02										
TOTAL	NO	1926	299	776	571	204	49	18	4	3	0	2
	PCT	13.44	2.09	5.41	3.98	1.42	0.34	0.13	0.03	0.02	0.00	0.01

TOTAL VALID OBSERVATIONS 14333

TOTAL OBSERVATIONS 17545

Question 2.3

Section 2.1 of the FSAR, indicates that the population statistics are based on the 1960 census. Update this section to include the 1970 census data.

Answer:

See FSAR Section 2.

Question 2.4

Figure 2-2 of the FSAR, Plot Plan and Site Boundary, shows the one-mile radius exclusion area boundary. Provide a revised figure which clearly shows the boundary which will be used for establishing effluent release limits. [See the AEC's recently published "Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants" dated February 1972, Section 2.1.2.2] Show the nearest location siutable for dairying.

Answer:

The restricted area for release of gaseous effluents is the one mile radius exclusion area boundary with the exception of the temporary construction quarter which are considered an unrestricted area. (See Technical Specification 3.9 Bases) The restricted area for release of liquid effluents is a 154 feet wide by 216 feet long area at the Keowee dam tailrace. This area extends from the face of the powerhouse to the crest of the tailrace.

The nearest appropriate location for establishing a dairy is 5 miles due west of the plant site. This is based on the suitability of the terrain for dairying and not on plant releases.

Question 2.5

Figure 2-2 of the FSAR shows a highway passing in a northerly direction through the one-mile radius exclusion area boundary. Provide an analysis of a truck transportation incident involving toxic chemicals, explosive materials, and flammable materials on the safe operation of the Oconee Nuclear Power Station. If there are any oil or gas pipelines which pass near the reactor facility, the effect of potential accidents on the safe operation of the nuclear facility should be evaluated. If gaseous chlorine is stored onsite, describe the consequences of accidental release of chlorine on the reactor control room personnel. Describe any protective devices used to protect control room personnel during such a postulated incident.

21.

Answer:

There are no oil or gas pipelines that pass within 5 miles of the Nuclear Station.

The highways passing through the exclusion area are local roads with infrequent trucking of hazardous chemical and explosives since the general area is non-idustrial. An incident involving fire, chemicals or explosives at the closest point along the highway would be more than 1000 feet from the Reactor and Auxiliary Buildings. We believe that fire or chemical reactions at this distance would not affect plant operation. The blast pressure[1] from a truck loaded with 40,000 pounds[2] of TNT at this distance would be less than the design tornado loading on the structures.

Spilled liquid materials would follow the pattern of roadside drainage toward Lake Keowee and Keowee River. In the event flammable material should reach the cooling water intake structure and burn, the cooling water pumps and related equipment would likely not be affected, but the operation of these pumps is not required for plant safety, and the most serious consequence would be a plant shutdown due to lack of condenser cooling water.

If a highway incident should result in the release of toxic gases, the gases under most circumstances would either move in a direction away from the plant or be sufficiently dispersed by the time they reach the plant that they would not interfere with the safe operation of the plant. But if adverse environmental conditions should make it necessary, the plant could safely be operated or shutdown from the control room. The control room is an enclosed area which can be isolated from the outside environment. Portable breathing equipment is also provided to allow access to areas outside the control room.

Only small quantities of chlorine are stored on-site since chlorine is not used for condenser cleaning at Oconee. No individual container on the site contains more than 150 pounds of chlorine. The chlorine is used for drinking water purification and sanitary waste treatment, with three to five 150-pound containers typically being in use, and the maximum total number of containers on hand at any time is approximately ten. It is unlikely that leaks from these small chlorine containers could result in dangerous concentrations in the control room; but the control room can be isolated from the outside environment if necessary and portable breathing equipment, suitable for protection against chlorine, is also provided.

[1]Effects of Impact and Explosion, AD 221 586, National Defense Research Committee, Vol. 1, 1946

[2]Interstate Commerce Commission and Department of Transportation Regulations of Maximum Truck Limit

Question 2.6

Describe, if any, all industrial activities within a five mile radius of the site. If industrial facilities are located within a five mile radius of the plant, provide a description of the products manufactured, stored, or transported to indicate the maximum quantities of hazardous material likely to be processed, stored or transported.

Answer:

See FSAR Section 2.2.4.

3.0 REACTOR

Question 3.1

Describe your plans for vibration monitoring and detection of loose parts in Units 2 and 3.

Answer:

In keeping with commitments made in FSAR Supplement 4, Page 4-12c, submitted July 9, 1970, Duke Power Company has actively studied the feasibility of in-service monitoring for vibration and the detection of loose parts, and has pursued this subject with consultants and vendors of this type equipment. It has now been concluded that equipment is now commercially available which can be installed on the exterior of the primary system and remain in service during operation with reasonable availability. Reliable and consistent interpretation of the output of these instrument systems as related to the magnitude of vibration and the presence of loose parts is yet to be determined.

Neutron noise analysis techniques will be employed to monitor for vibration of the reactor core and reactor vessel internals. The permanently installed nuclear instrument system ion chambers will be used to supply the input signals for the neutron noise analysis system. It is believed through proper analysis that vertical oscillations can be separated from lateral motion and the magnitude of these oscillations determined approximately. The analysis will consider control rod worth as a function of position and the phase angle between signals detected on opposite sides of the reactor vessel.

The loose parts monitoring on Oconee Units 1, 2, and 3 within the primary system provides the plant operator with an immediate warning should abnormalities occur. The two OTSG upper heads and the reactor vessel lower head are instrumented with mechanically attached sensors (crystal type accelerometers) to transmit (through pre-amplifiers) the signals to the control room. Sensors and pre-amplifiers located within the reactor building are installed in sets of two, one of which is a spare. The
28. vessel sensors are mechanically attached to instrument guide piping just outboard of the primary shield wall penetration. The steam generator sensors are mounted on the upper vent lines and on a special wave guide attached to the handhole plug.

In the control room the signal from each installed sensor is amplified, filtered and continuously monitored by a decibel meter and an alarm circuit. The signal from any sensor can also be fed to a high fidelity speaker to permit the operator to detect any abnormal sounds. One channel is normally fed to the speaker at all times at reduced volume. Periodic checking of selected channels by the operator to detect signal abnormalities below the threshold detectable by alarm circuits is a standard practice.

To permit analysis of abnormal signals a reference signal is provided. This consists of a separate speaker and tape player on which reference tapes recorded during early plant operation may be selected to compare with signals at a later date. A tape recorder is provided to record signals from any sensor and is used to record the initial reference tapes as well as to record selected signals during operation. A reference tape of signals recorded at Oconee is also provided to acquaint operators with actual sound characteristics of loose parts in the system.

28.

Suitable connections are provided to permit the addition of on-line data analysis equipment. The on-line measurements can be made without interference to the normal operation of the monitor system. A spectrum analyzer can be used to produce permanent on-line signature analysis plots and elementary cross correlations between sensors. In addition, the connections can be used to record the signals for off-line analysis. The recorded signals can be returned to the NSS vendor for signal analysis and interpretation.

Question 3.2

Describe your procedures for preventing control rod damage due to dry trip.

Answer:

Procedures for preventing control rod damage due to dry trip are covered under 3.2.4.3.1 under "Additional Design Criteria" on Page 3-81 of FSAR.

4.0 REACTOR COOLANT SYSTEMS

Question 4.1

Describe any design modifications, operational limits and/or additional test procedures that may result from experience gained during hot functional test in Oconee Unit 1.

Answer:

The design modifications, operational limit, and/or additional test procedures which may result as a result of the Oconee Unit 1 hot functional testing incident are not yet finalized. Meetings have been scheduled with the AEC/DRL staff to discuss these points.

Question 4.2

Regarding the fracture toughness data obtained for all pressure-retaining ferritic materials of the reactor vessel, state the degree of compliance with the acceptance criteria for the recently revised ASME Code Section III fracture toughness rules (Code Case 1514). These rules require determination of the following for reactor vessel plates, forgings and the qualification welds:

a. NDT temperature obtained from drop weight (DWT) tests, and

b. Temperature, at which "weak" direction Charpy V-notch specimens exhibit at least 35 mils lateral expansion and not less than 50 ft-lbs absorbed energy.

Answer:

The pressure boundary materials used in these contracts were ordered and tested in accordance with the requirements of the 1965 Edition of Section III of the ASME Code including all addenda through Summer 1967. The 1965 Edition does not require the determination of the Nil-Ductility Temperature (NDT) as obtained by drop weight test, nor the Charpy V-notch energy levels for specimens oriented in the "weak" direction. All base materials meet the Charpy V-notch energy value requirements listed in Section III at a temperature of plus 40°F or lower. For the weld deposits the transition temperature was obtained by performing Charpy V-notch impact tests during procedure qualification on weld deposits using the same flux and filler wire combinations as the production welds. All weld deposits meet Charpy V-notch energy values required by Section III at a temperature of plus 40°F or lower.

Question 4.3

For the materials of the reactor vessel beltline (including welds) provide the initial upper shelf fracture energy levels, as determined by Charpy V-notch tests, if available, in both directions.

Answer:

In addition to the impact tests required by the ASME Code, the Nil-Ductility Temperature and Charpy V-notch energy levels at several temperatures were obtained for the two forgings that comprise the core region of the reactor vessels. The forging material is ASTM A508-64 Class 2 as modified by Code Case 1332-4. The impact tests were taken at 2 inches from surface, 1/4 and 1/2 of the forging thickness, and oriented in the circumferential direction with the length of the notch of the Charpy V-notch perpendicular to the surface of the material. The weld deposits of the core region (circumferential welds) were impact tested at plus 10°F using Charpy V-notch specimens oriented perpendicular to the direction of welding with the notch normal to the surface. No upper shelf fracture energy levels were determined for the weld deposits.

OCONEE II

DATA:

	Specimen Description	Drop Weight NDT (°F)	Cv Energy at +10°F (ft - lb)	Approximate Upper Shelf Cv Energy (ft-lb)
Top Shell Forging	C.1/5 T*	+20	86, 46, 79	129
	C.1/4 T	+10	100, 89, 72	140
	C.1/2 T	+20	62, 77, 40	141
Bottom Shell Forging	C.1/5 T*	+20	116, 93, 104	142
	C.1/4 T	+20	82, 83, 90	140
	C.1/2 T	+20	101, 89, 92	149
Top Weld Deposit	1/4T	Not Available	41, 37, 43	Not Available
Center Weld Deposit	1/4T	Not Available	38, 28, 49	Not Available
Bottom Weld Deposit	1/4T	Not Available	35, 40, 30	Not Available

21. *Circumferential, 2 inches from surface.

OCONEE III

DATA:

	Specimen Description	Drop Weight NDT (°F)	Cv Energy at +10°F (ft - lb)	Approximate Upper Shelf Cv Energy (ft-lb)
Top Shell Forging	C 1/5 T*	+40	76, 82, 46	115
	C 1/4 T	+30	85, 77, 78	139
	C 1/2 T	+30	82, 55, 91	135
Bottom Shell Forging	C 1/5 T*	+20	49, 83, 43	153
	C 1/4 T	+40	39, 50, 66	150
	C 1/2 T	+20	24, 34, 14	154
Top Weld Deposit	1/4 T	Not Available	36, 35, 26	Not Available
Outer Weld Deposit	1/4 T	Not Available	29, 35, 30	Not Available
Bottom Weld	1/4 T	Not Available	36, 43, 42	Not Available

21. *Circumferential, 2 inches from surface.

Question 4.4

Provide proposed operating limitations during startup and shutdown of the reactor coolant system using as a guide Appendix G, "Protection Against Non-Ductile Failure," of the recently revised ASME Code Section III fracture toughness rules (Code Case 1514).

Answer:

B&W proposes to use the ASME Section III Non-Mandatory Appendix G to establish the operating limitations on reactor vessel.

The general procedure to obtain such operating limitations would be as follows:

A. A computer program will be used to determine the maximum allowable pressure versus the relative temperature ($T-RT_{NDT}$) for various maximum temperature differences through the vessel wall where T is the actual temperature in °F and RT_{NDT} is the reference temperature as defined in NB-2331. The maximum temperature difference through the vessel wall will depend on the heatup and cooldown rate. The permissible operating conditions will be established so as to be to the right of these curves.

B. The effects of irradiation on the allowable pressure vs relative temperature curves will be considered as follows:

21. 1. The chemical analysis of the irradiated region materials (forgings and weld deposits) will include copper, phosphorus, and sulfur. Several curves for maximum copper content will illustrate the predicted Charpy V-notch 30 ft-lb transition temperature shift versus neutron fluence for such materials.

2. The initial reference temperature RT_{NDT} for the materials of the reactor vessel beltline will be estimated conservatively using available impact data, (i.e., from the data given in Answer 4.3, the estimated RT_{NDT} for the center weld deposit is equal to +60°F).

3. For any cooldown or heatup after the irradiated region has been irradiated to a neutron fluence (where the predicted increase in transition temperature is approximately 50°F or greater), the reference temperature, RT_{NDT} will be calculated as follows:

$$RT_{NDT} \text{ (irradiated)} = RT_{NDT} \text{ (unirradiated, estimated)} + \Delta TT$$

21. where ΔTT is the Charpy V-notch 30 ft-lb transition temperature shift based on the chemical analysis (Cu and P) and the fission spectrum neutron fluence level of the materials. The highest predicted RT_{NDT} (irradiated) of all the irradiated region materials will be used to determine the operating limitations of the reactor vessel.

C. The operating limitations for the early life of the reactor vessel
21. (early life is defined as the span where the predicted increase in transition temperature is 50°F or less) will be based on the case where the highest predicted RT_{NDT} (irradiated) material has an irradiated embrittlement shift equivalent to 50°F.

, Question 4.5

For the predicted NDT temperature shift of 250°F (FSAR, Page 4-24) at least five capsules are required by the AEC proposed "Reactor Vessel Material Surveillance Program Requirements," 50.55a, Appendix H, published in the Federal Register on July 3, 1971. Each of these surveillance capsules should include specimens from the base metal, heat affected zone and the weld metal, as recommended in ASTM E-185, Section 3.3 Section 4.4.6 of the FSAR refers to the report BAW-10006 for the description of the surveillance program consisting of six capsules, only three of which contain weld metal specimens. In effect, the proposed surveillance program consists of only three capsules containing the required number and type of impact specimens.

Describe the steps that will be taken to provide five surveillance capsules, each of which contain specimens per ASTM-E-185.

Answer:

The surveillance program for Oconee Units 2 and 3 are in accordance with BAW-10006 Revision 2, which meets the intent of ASTM-E-185-70, but not the intent of proposed Appendix H to 10 CFR 50.

The capsules for Oconee 2 have been completely assembled and are at the site per BAW-10006, Revision 2. There is not adequate material available to meet Appendix H of 10 CFR 50.

The Oconee 3 program will be modified to meet the intent of Appendix H of 10 CFR 50.

Question 4.6

Regarding preoperational mapping of the reactor vessel by ultrasonic examination, to meet the requirements of IS-232 of Section XI of the ASME Code, state the acceptance standards that were used to establish acceptability of the vessel for service.

Answer:

The acceptance standards used were those contained in N625.4 of the 1965 edition of Section III of the ASME Code, with addenda through Summer 1967.

5.0 CONTAINMENT SYSTEMS AND OTHER SPECIFIC STRUCTURES

Question 5.1

Section 5 of the FSAR contains procedures and instrumentation for the structural testing of the Unit No. 1 containment. However, the extent to which these procedures and instrumentation will be applied to the testing of Units 2 and 3 cannot be ascertained from the section. Provide this information and, in addition, indicate the extent to which the procedures and instrumentation proposed are compatible with Safety Guide No. 18.

Answer:

See FSAR Section 5.6.1.2.2. The instrumentation for Units 2 and 3 complies with Safety Guide 18 except for the building deformation measuring points. These measuring points were selected so that measurements could be directly compared to Unit 1.

Question 5.2

Reword Section 5.6.2.1 of the FSAR to make it clear (under "Integrated Leak Rate Test") that a leak rate test will be performed at the end of the ten year period also (three tests per 10 year period). Delete Item C under "Integrated Leak Rate Test."

Answer:

See FSAR Section 5.6.2.1.

6.0 ENGINEERED SAFETY FEATURES

Question 6.1

Provide analysis to justify that the ECCS and containment spray pumps will have adequate net positive suction head.

Answer:

The analysis to show that the low pressure injection pumps and the reactor building spray pumps will have adequate net positive suction head is presented in FSAR Section 6.5.2 as revised in Amendment 19 dated May 5, 1972. This analysis applies to all three Oconee units.

Question 6.2

Provide information on how failures in the engineered safety features will be detected during normal operation.

Answer:

Failures in the engineered safeguards protection system will be detected during normal plant operation by on-line periodic testing and inspection (FSAR Section 7.1.3.3.4), comparison of like readings, and by annunciators which continuously monitor critical points within the system.

Where equipment is used for emergency functions only, such as reactor building spray system, systems have been designed to permit meaningful periodic tests. See Table 6-3, Section 6 of the FSAR, for operational tests of the low and high pressure injection system and the core flooding system.

Section 6.2.4 on Page 6-18 of the FSAR gives the operational tests for the reactor building spray system.

Section 6.3.4 of the FSAR gives the operational tests for the reactor building cooling system.

The reactor building penetration room ventilation system may be actuated during normal operation for testing.

Technical specifications 4.5.1 through 4.5.4 state how the ESF systems are to be tested and the frequency of testing.

Question 6.3

Identify and field run piping used for the engineered safety systems and the manner in which such runs, if any, are checked against design predictions.

Answer:

See FSAR Sections 1C.3.5.

7.0 INSTRUMENTATION AND CONTROL

Question 7.1

Provide information on the measures being taken to prevent the type of fire which occurred in the control rod drive system transfer panel for Unit 1 on March 9, 1972.

Answer:

The localized combustion of insulation which extinguished itself in a transfer panel in Unit 1 on March 8, 1972 was determined to have been caused by a loose or high resistance connection to a rod group patch connector. To preclude a recurrence of the problem, connectors of a different design employing circular, MS type, multipin connectors are used instead of the bus clip type of connectors originally used on Unit 1. The multipin connectors are mounted on aluminum panels, whereas the original connectors on Unit 1 were clipped directly to round copper buses through an insulating panel.

Units 2 and 3 are supplied with the upgraded design.

8.0 ELECTRICAL SYSTEMS

Question 8.1

Provide information on the performance discharge tests to be performed in accordance with IEEE Standard 308 - 1970 on the station batteries.

Answer:

IEEE Standard 308 - 1971, paragraph 5.3.6 states: "Performance Discharge Test Provisions: Means shall be provided to determine the voltage time output current characteristic of the battery. The output current shall be large enough to determine if the battery has been degraded. Periodic tests are shown in Table 2."

Table 2 is titled "Illustrative Periodic Tests" and indicates a test interval as "yearly."

The tests proposed for the Instrument and Control, Keowee Station, and Switching Station 125 volt DC systems are as follows:

a. The voltage and temperature of a pilot cell in each bank shall be measured and recorded daily, five days/week.

b. The specific gravity and voltage of each cell shall be measured and recorded every month.

c. Before initial operation and at five-year intervals coincident with the refueling outages, a one-hour discharge test at the required emergency load will be made.

These tests are in agreement with IEEE-308 - 1971 except that the test interval is five years instead of yearly. The five year test interval is in accord with the proposed "IEEE Recommended Practice for Maintenance, Testing, and Replacement, Large Stationary Type Power Plant and Substation Lead Storage Batteries," which is considered adequate test intervals to detect deterioration of batteries.

11.0 RADIOACTIVE WASTE AND RADIATION PROTECTION

Question 11.1

Provide information to support the fact that the RIA-44 monitoring device to be installed in the unit vents will have the required sensitivity for measuring the anticipated levels either for a continuous or instantaneous release. Discuss iodine plate out.

Answer:

A four cubic foot per minute sample from each unit vent is pumped through a particulate filter - charcoal filter combination. A NaI scintillation detector is located within a lead shielded compartment adjacent to the two inch diameter filter assembly. A ten stage photomultiplier tube and a preamplifier located at the detector are used to amplify the signal for transmission to the control room. A single channel analyzer with a $\pm$ 5% window centered on the predominant Iodine 131 gamma peak (.36 MeV) is provided to indicate radioactive iodine accumulation on the filter. For limiting concentration in unit vents to achieve proposed 10CFR50 Appendix I concentrations at the Exclusion Area Boundary, RIA-44 will increase approximately 29 counts per minute per day over a background of approximately six counts per minute. This is considered to be sufficient sensitivity to monitor releases within Appendix I limits and above for continuous and instantaneous releases of significance.

The sensitivity of this instrumentation is based on factory calibration of the detectors using simulated Iodine 131 sources consisting of Barium 133 and Cesium 137 with traceability to Certified Radioactive Standard Reference Materials issued by the National Bureau of Standards. The initial calibration data on each channel was recorded along with data from calibration sources which were supplied with the equipment. By applying appropriate source decay correction factors, supplied on decay curves, detector sensitivities can be verified and maintained. A solenoid operated check source mounted at the detector permits the measuring circuitry to be functionally checked by depressing a pushbutton switch located on the count rate meter in the control room.

Pressure switches located at the detector provide control room annunciation of low sample flow. A rotometer and valves located in the cabinet with the detector are used to calibrate the pressure switches by measuring the flow through the detector while filter clogging and reduced sample blower efficiency are simulated by throttling a valve in series with the filter or by opening a bypass line around the sample blower. To prevent interaction between the moving filter on the air particulate monitor and the iodine monitor separate sample blowers are provided.

The sample is removed from the unit vent via a sample nozzle and a 1 1/2" diameter sch. 40 304 stainless steel sample pipe. The sample pipe contains one 90° and two 45° elbows. The sample point is located at an elevation which permits the sample to be removed greater than six diameters above the upper inlet to the vent. The minimum downward slope on the sample line is 45°. Sample line material and length should not present plateout problems for these monitors. However, losses can be empirically determined by manual air sampling in the Reactor Building and comparing this concentration with the unit vent iodine monitors. Appropriate correction factors can be applied if necessary.

Question 11.2

Verify that the charcoal to be used in the RIA-44 monitor is impregnated to assure collection of both elemental and non-elemental forms of iodine. Provide information as to the frequency at which the charcoal will be changed and tested.

Answer:

Filters for the iodine monitors will be impregnated to assure collection of elemental iodine and methyl iodide. Initial charcoal filter replacements will be made at 30-day intervals or more frequently if dust clogging is experienced.

12.0 CONDUCT OF OPERATION

Question 12.1

Expand the job description of the Shift Supervisor and the Assistant Shift Supervisor to describe their responsibilities during Units 2 and 3 operation (FSAR Section 12.1.2.5 and 12.1.2.6).

Answer:

See FSAR Sections 12.1.2.5 and 12.1.2.6.

Question 12.2

The minimum qualifications listed for the key supervisory positions do not meet the minimum requirements as specified in ANSL N.18.1 - 1971. Justify (FSAR Section 12.2).

Answer:

See FSAR Section 12.2.

Question 12.3

Describe what provisions have been made for the review of both operating and emergency procedures by the operating crew during the life of the plant, to enhance operator proficiency (FSAR Section 12.3).

Answer:

See FSAR Sections 12.3.1, 12.3.2.1 and 12.3.2.3.

Question 12.4

State whether agreements with those offsite organizations which will be called upon to perform emergency functions or services are in writing. [FSAR Sections 12.3.2.2 and 12.3.2.3(c)].

Answer:

Verbal agreements have been made with all offsite organizations that will be called upon to perform emergency functions or to provide emergency services. These verbal arrangements and agreements have been confirmed by AEC Compliance (Region II) and by the South Carolina State Board of Health, Division of Radiological Health. These verbal agreements have been reduced to writing and were forwarded by Duke Power Company to the organizations involved by letter dated May 11, 1970. Subsequent contacts to date, with these organizations by Oconee Nuclear Station personnel, for the purpose of implementing the emergency plan, have confirmed these agreements.

Question 12.5

Define what is meant by "periodic" in relation to emergency drills for the training of plant personnel. State at what frequency simulated drills involving offsite agencies will be conducted [FSAR Section 12.3.2.3(e)].

Answer:

See FSAR Section 12.3.2.3(e).

Question 12.6

The doses listed for initiation of protective measures for people in the low population zone are higher than appropriate. We suggest that you consider 2.5R for external exposure and 500 XMPC for internal exposure as possible action levels for notification of offsite support groups (FSAR Section 12.3.2.7.2).

Answer:

See FSAR Section 12.3.2.7.2(b).

Question 12.7

Define "periodically" (1) regarding the plant operations review by the Station Review Committee and (2) regarding the audit of station operations by the General Office Review Committee. (FSAR Section 12.5).

Answer:

See FSAR Section 12.5.

Question 12.8

The shift complement for the Units 2 and 3 operation is not acceptable. (Figure 12-4B FSAR). (See the letter from P. A. Morris to Duke Power Company dated February 13, 1970).

Answer:

In 1970, after receiving Dr. P. A. Morris' letter to Duke Power Company dated February 13, 1970, members of the Oconee Nuclear Station staff and General Office Steam Production staff made a presentation to AEC/DRL personnel in which several emergency procedures, placing the most severe demands on station operating manpower, were presented. This presentation established that two operators per shift could safely shut down a unit from outside the control room. Another presentation is being developed to again justify our shift commitment and will be available for AEC/DRL consideration by June 15, 1972.

Question 12.9

Define the role of the Staff Engineer (see Page 12A-6 FSAR) in the station organization (Figure 12-4B FSAR).

Answer:

The Staff Engineer who reports to the Superintendent and/or the Assistant Superintendent performs special assignments as designated by the Superintendent such as scheduling, following startup problems, and liason with the Construction Department.

Question 12.10

State how many nuclear engineers and mechanical engineers with nuclear training and experience are employed in the Mechanical Engineering Section at the present time. (FSAR Section 12A-6).

Answer:

At the present time, the Mechanical Section of the Design Engineering Department has thirty-five (35) nuclear engineers and mechanical engineers with nuclear experience.

Question 12.11

Provide resumes for all individuals selected for the positions of Shift Supervisor and Assistant Shift Supervisor.

Answer:

See FSAR Section 12A.5.

D U K E   P O W E R   C O M P A N Y

OCONEE NUCLEAR STATION

UNITS 1, 2, AND 3

APPLICATION FOR LICENSES

DOCKET 50-269, -270, -287

FSAR SUPPLEMENT 12

Submitted with FSAR Revision 21

July 26, 1972

LIST OF EFFECTIVE PAGES

FSAR SUPPLEMENT 12

The following is voluntarily submitted in response to informal questions by AEC/MEB.

Question 1

Confirm the validity of a fixed base assumption in the mathematical models for the dynamic system analyses by providing summary analytical results that indicate that the rocking and translational responses are insignificant. A brief description should be included of the method, mathematical model and damping values (rocking vertical, translation and torsion) that have been used to consider the coil-structure interaction.

Answer

The natural frequencies, as determined by the fixed base mathematical model, was validated by comparison with the natural frequencies calculated by using rocking and translational soil springs. The variation in the compared natural frequencies was insignificant.

Question 2

Describe the method employed to consider the torsional modes of vibration in the seismic analysis of the Category I building structures. If static factors are used to account for torsional accelerations in the seismic design of Category I structures, justify this procedure in lieu of a combined vertical horizontal, and torsional multimass system dynamic analysis.

Answer

Torsional modes were not considered in the seismic analysis. Insignificant torsional shear stresses were found, assuming a minimum of 10 percent eccentricity, based on "Torsion in Symmetrical Buildings," N. M. Newmark.

Question 3

Provide the dynamic methods and procedures used to determine Category I structure overturning moments. Include a description of the procedures used to account for coil reactions and vertical earthquake effects.

Answer

The safety factor against overturning due to maximum hypothetical earthquake moment is 3.6.

Question 4

With respect to Category I piping burried or otherwise located outside of the containment structure, describe the seismic design criteria employed to assure that allowable piping and structural stresses are not exceeded due to differential movement at support points, at containment penetrations, and at entry points into other structures.

Answer

See FSAR Section 1C.3.4.1

Question 5

Describe the evaluation performed to determine seismic induced effects of Category II piping systems on Category I piping.

Answer

See FSAR 1C.3.4.6

Question 6

Provide the criteria employed to determine the field location of seismic supports and restraints for Category I piping, piping system components, and equipment, including placement of snubbers and dampers. Describe the procedures followed to assure that the field location and characteristics of these supports and restraining devices are consistent with the assumptions made in the dynamic analyses of the system.

Answer

See FSAR Section 1C.3.4.5

Question 7

Discuss the seismic instrumentation provided and compare the proposed seismic instrumentation program with that described in AEC Safety Guide 12, "Instrumentation for Earthquakes." Submit the basis and justification for elements of the proposed program that differ substantially from Safety Guide 12.

Provide a description of the seismic instrumentation such as peak recording accelerographs and peak deflection recorders, that will be installed in selected Category I (Class 1 Seismic) structures and on selected Category I (Class 1 Seismic) components. Include the basis for selection of these structures and components, the basis for location of the instrumentation, and the extent to which this instrumentation will be employed to verify the seismic analyses following a seismic event.

Describe the provisions that will be used to signal the control room operator the value of the peak acceleration level experienced in the tendon access gallery of the reactor containment structure to the control room operator within a few minutes after the earthquake. Include the basis for establishing the predetermined values for activating the readout of the accelerograph to the control room operator. Provide the criteria and procedures that will be used to compare measured responses of Category I (Class 1 Seismic) structures and the selected components in the event of an earthquake with the results of the system dynamic analyses. Include consideration of different underlying soil conditions or unique structural dynamic characteristics that may produce different dynamic responses of Category I (Class 1 Seismic) structures at the site.

Answer

24. Duke will install another strong motion accelograph at a higher elevation in the reactor building and several peak recording accelerometers on selected reactor building piping and equipment. The exact location of these instruments will be designated at a future date. See FSAR Section 5.6.2.2

Question 8

Paragraph I701.5.4 of the ANSI B31.7 Nuclear Power Piping Code requires that piping shall be supported to prevent excessive vibration under startup and initial operating conditions. Submit a discussion of your vibration operational test program which will be used to verify that the piping and piping restraints within the reactor coolant pressure boundary have been designed to withstand dynamic effects due to valve closures, pump trips, etc. Provide a list of the transient conditions and the associated actions (pump trips, valve actuations, etc.) that will be used in the vibration operational test program to verify the integrity of the system. Include those transients introduced in systems other than the reactor coolant pressure boundary that will result in significant vibration response of reactor coolant pressure boundary systems and components.

Answer

See FSAR Section 1C.3.7

Question 9

Discuss the testing procedures used in the design of Category I mechanical equipment such as fans, pumps, drives, valve operators and heat exchanger tube bundles to withstand seismic, accident and operational vibratory loading conditions, including the manner in which the methods and procedures employed will consider the frequency spectra and amplitudes calculated to exist at the equipment supports. Where tests or analyses do not include evaluation of the equipment in the operating mode, describe the bases for assuring that this equipment will function when subjected to seismic and accident loadings.

Answer

See FSAR Section 1C.3.8

Question 10

Provide a brief description of the dynamic system analysis methods and procedures used to determine dynamic responses of reactor internals and associated Class I components of the reactor coolant pressure boundary (e.g., analyses and tests). The discussion should include the preoperational test program elements described in Safety Guide 20, Vibration Measurements on Reactor Internals. In the event elements of the program differ substantially from the requirements of Safety Guide 20, the basis and justification for these differences should be presented.

23. Provide a discussion of the preoperational analysis and testing results that will be used to augment the LOCA dynamic analysis methods and procedures, i.e., barrel ring and beam modes, guide tube responses, water mass and compliance effects, damping factor selection, etc.

Answer

B&W will submit a topical report in the early fall on the Oconee I internals redesign. This document will discuss the Preoperational Monitoring of the Reactor Internals in compliance with Safety Guide 20 using Oconee I as a prototype and Oconee 2 and 3 as a similar to prototype design.

Question 11

The FSAR states that faulted operating condition categories have been applied to certain reactor coolant system components. Identify any other components or systems that are not a part of the reactor coolant pressure boundary for which the design stress limits associated with faulted conditions were applied. If faulted conditions are used for such cases, then provide justification for applying such conditions, including the bases for the loading conditions and combinations, and associated design stress limits which were applied.

In addition, for all components and systems comparable to ASME Code Class 2 and 3, provide the design condition categories (normal, upset or emergency), the associated design loading combinations and the design stress limits which will be applied for each loading combination. This information may be submitted in tabular form as suggested below:

System and/or Component	Design Loading Combinations	Design Condition Categories (Normal, Upset, or Emergency)	Design Stress Limits

If any design stress limits allow inelastic deformation (or are comparable to the faulted condition limits defined in ASME Section III for Class I components) then provide the bases for the use of inelastic design limits by demonstrating that the component will maintain its functional or structural integrity under the specified design loading combination. Include a brief description of the methods and design procedures that were used in such cases.

Answer

Faulted operating conditions were not applied to any components that were not a part of the reactor coolant pressure boundary.

The design stress limits for components comparable to the ASME Code Class 2 and 3 did not allow inelastic deformation.

Question 12

Describe the design and installation criteria applicable to the mounting of the pressure-relieving devices (safety valves and relief valves) for the overpressure protection of systems with Class 2 components. In particular, specify the design criteria used to take into account full discharge loads (i.e., thrust, bending, torsion) imposed on valves and on connected piping in the event all the valves are required to discharge. Indicate the provisions made to accommodate these loads.

Answer

See FSAR Section 1C.3.6

Question 13

Categorize all transients or combinations of transients listed in TAble 4-8 of the FSAR with respect to the conditions identified as "normal", "upset", "emergency", or "faulted" as defined in the ASME Section III Nuclear Component Code. In addition, provide the design loading combinations and the associated stress or deformation criteria.

Answer

Reference Revised Table 4-8 in the FSAR

Question 14

To facilitate review of the bases for the pressure relieving capacity of the reactor coolant pressure boundary, submit (as an appendix to the FSAR) the "Report on Overpressure Protection" that has been prepared in accordance with the requirements of the ASME Section III Nuclear Power Plant Components Code or, if the report is not available, indicate the approximate date for submission. In the event the report is not expected to be available until either the Operating License review or late in the construction schedule for the plant, provide in the FSAR the bases and analytical approach (e.g., preliminary analyses) being utilized to establish the overpressure relieving capacity required for the reactor coolant pressure boundary.

Answer

Reference BAW-10043

LIST OF EFFECTIVE PAGES

FSAR Supplement 13

Page	Revision	Page	Revision
LOEP 1 of 2	Rev. 37	13-23	Original
LOEP 2 of 2	Rev. 37	13-24	Original
Cover Sheet Supplement 13	Rev. 26	13-25	Original
13-1	Original	13-26	Original
13-2	Original	13-27	Original
13-3	Original	13-28	Original
13-4	Rev. 28	13-29	Original
13-5	Rev. 28	13-30	Original
13-6	Original	13-31	Original
13-7	Original	13-32	Original
13-8	Original	13-33	Original
13-9	Original	13-34	Original
13-10	Original	13-35	Original
13-11	Original	13-36	Original
13-12	Original	13-37	Original
13-13	Original	13-38	Original
13-14	Original	13-39	Original
13-15	Original	13-40	Original
13-16	Original	13-41	Original
13-17	Original	13-42	Original
13-18	Original	13-43	Original
13-19	Original	13-44	Original
13-20	Original	13-45	Original
13-21	Original	13-46	Original
13-22	Original	13-47	Original

LIST OF EFFECTIVE PAGES

FSAR SUPPLEMENT 13 (CONT'D)

DUKE POWER COMPANY

OCONEE NUCLEAR STATION

FSAR SUPPLEMENT 13

Submitted with FSAR Revision 26
January 29, 1973

In response to questions contained in Mr. R. C. DeYoung's letter of September 26, 1972, the following information is submitted:

Question - Review Oconee Nuclear Station Units 1, 2 and 3 to determine whether the failure of any non-category I (seismic) equipment, particularly in the circulating water system and fire protection, could result in a condition, such as flooding or the release of chemicals, that might potentially adversely affect the performance of safety-related equipment required for safe shutdown of the facilities or to limit the consequences of an accident.

Answer - There exists a remote possibility of flooding in the turbine building at the basement level due to failure of expansion joints in the condenser cooling water system near the condenser water box inlet or outlet nozzles.

Condenser cooling water intake and discharge pipes are embedded in the turbine building substructure mat to points immediately below the inlet or outlet connections on the condenser inlet and outlet water boxes. At each waterbox connection a 78" steel pipe is turned up and projected above the basement level and connected to a butterfly valve. A rubber expansion joint is located between each valve and waterbox connection. The rubber joint spans across a 4 1/4" physical gap in the 78" intake pipe and across a 2" physical gap in the 78" discharge. At maximum flow conditions through any condenser a complete rupture of the 4 1/4" intake pipe joint (all rubber removed) would result in a 235 cfs leak into the turbine building basement area. This is the worst case leak condition due to the higher head and wider possible gap situation that exists on the intake side of the condenser.

Each foot of depth in this 202 feet wide by 790 feet long structure contains a volume of 160,000 cubic feet. Therefore, a joint rupture would fill the turbine building at the rate of .088 feet per minute until the water surface reaches the height of the rupture and a reduced rate thereafter due to reduced differential head conditions, provided all flood water could be contained in the turbine building.

Curbs 1.5 feet high will be provided prior to Oconee 1 criticality around doorway entrances to the auxiliary building from the turbine building to contain flood water in the turbine building until action is taken to control flooding. (This will provide 17 minutes storage in the turbine building basement.) Turbine building sump level alarms will alert the control room operators.

Whenever alarms in the control room indicate a possibility of flood conditions in the turbine building basement an operator will immediately investigate the situation by visual inspection and initiate the appropriate valve operation to control the flooding if a joint rupture has occurred. Each half of each condenser shell of each unit can be isolated from the remainder of the cooling water system without unit shutdown in the event of joint failure.

The auxiliary building could be subject to flooding from two sources: the fire protection system and the ventilation cooling water system. The fire protection system does not constitute a threat due to the fact that the headers inside the auxiliary building will be empty and dry except when manually energized to fight a fire. The possibility of flooding from the ventilation cooling water system is reduced by flow limiting valves installed in all non-category 1 supply lines entering the auxiliary building larger than 3" in diameter. The maximum flow which can flood the building from a single rupture is 1140 gpm. Without taking credit for auxiliary builidng sump pumps, over 10 minutes is available for corrective action before safety-related equipment would be affected. Flooding by this source will be detected by high level alarm sensors in the auxiliary building sumps and necessary action taken by the operator to isolate the line rupture.

In response to questions contained in Mr. A. Schwencer's letter of October 25, 1972, the following information is submitted.

Question - 14.1.1 Provide analyses of the containment pressure-time response for various (i.e., break sizes and break location) postulated design basis loss-of-coolant accidents. Include analysis of a double-ended break at the largest reactor outlet pipe, the reactor coolant pump suction pipe and at the reactor coolant pump discharge pipe as well as a spectrum of smaller pipe breaks at the same locations. The analyses should be extended, as a minimum, through the blowdown, reflood and post-reflood phases of the accidents (i.e., about an hour following the accident).

All assumptions used in these analyses should be explained. Assumptions should be selected on a conservative basis in the calculations of containment pressure response.

14.1.2 The reflood model that is used following blowdown should be described in detail. The description should include the assumptions used to develop the model, e. g., hydraulic modeling of the primary coolant system, resistances of components (primary coolant pump, steam generator, piping and reactor core), method to compute steam generation in core and energy sources (core stored energy, decay heat, thick and thin metal stored energy, and steam generator stored energy).

14.1.3 If the blowdown model, used for these analyses, differs from that described in the SAR for containment sizing, the difference should be discussed in detail.

14.1.4 For the cold leg break of the size and location resulting in the highest calculated containment pressure (analyzed in 14.1.1 above), provide tables of mass release (pounds/sec) and the enthalpy of the mass (BTU/pound) released from the core, and the mass and enthalpy released to the containment throughout the blowdown and reflood phases of the accident. Provide a graph showing core inlet velocity as a function of time for the reflood phase of the accident.

Answer - See Next Page

Answers to 14.1.1, 14.1.2, 14.1.3, 14.1.4 -- SAFETY ANALYSIS

The reactor building pressure-time response is shown in the FSAR for a spectrum of breaks in the hot leg piping after it was determined that breaks in the hot leg resulted in the highest building pressures. The assumptions used for that analysis are also given the FSAR. Mass and energy release rates from the reactor coolant system were determined by the use of the 3-region FLASH code.

The revised analysis presented here utilized mass and energy release rates determined by the use of a multiregion CRAFT code. The nodal arrangement of the CRAFT model is shown on Figure 14-1. The secondary side of the steam generators and the associated feedwater piping are represented by control volumes 6, 20, 7, and 21. Main feedwater was added directly into the secondary side of the steam generators over a period of 17 seconds as the feedwater control valve closed. At 15 seconds, the paths connecting the feedwater piping to the steam generators (which were assumed to be closed in the CRAFT model until this time) were opened so that the mass and energy trapped in the piping could enter the steam generator. Auxiliary feedwater started at approximately 35 seconds. Sensible heat stored in the steam generator tubes was modeled by slabs of metal in the secondary control volumes. Stored heat in the primary metal was simulated by slabs of metal in the control volumes which discharge their stored energy as the reactor coolant system temperature decreases.

To ensure a conservative calculation, the CRAFT code was run at 102% core power (2619 MWt) and the mode of heat transfer in the core was assumed to be nucleate boiling until the quality of the coolant was approximately 1.0. Decay heat was calculated by using the ANS Standard times 1.2.

The entire transient (blowdown and reflood periods) was simulated using the CRAFT code. In order to present a conservative analysis for the reactor building pressure response, it was assumed that the single failure was a power failure which resulted in minimum reactor building cooling (2 coolers and 1 spray). The coolers were assumed to be operative in 25 seconds and sprays started at 75 seconds. This same failure allows operation of only 1 HPI and 1 LPI pump. To see the effect of this assumption on the containment pressure, runs were made using 2 LPI pumps. A negligible difference in pressure was seen.

As in the FSAR analysis, the highest building pressure occurred for a hot leg break. The highest pressure (53.5 psig) was obtained for the largest break (14.1 ft^2). These hot leg breaks resulted in removing heat from the steam generators because of the backflow of emergency coolant through the steam generators. Figure 14-2 through 14-5 show the pressure time response for 4 hot leg breaks.

Additional analyses of the 14.1 ft. hot leg break show that for reduced reactor building cooling capability, pressure-time responses follow the trends shown
28. in FSAR Section 14, Figure 14.63h. Without reactor building spray and only two coolers operable, a maximum building pressure of 53.8 psig was obtained. For no spray and one cooler, the maximum pressure was 54.2 psig and for no spray and no coolers the maximum pressure was 54.6 psig.

Breaks in the cold leg piping at the pump suction and the pump discharge were analyzed. It was determined that breaks at the pump suction resulted in the higher pressures than those at the pump discharge. Four break sizes at the pump discharge (0.5 ft^2, 3 ft^2, 5.13 ft^2, and 8.55 ft^2) were analyzed as well as six break sizes at the pump suction (0.5 ft^2, 2 ft^2, 3 ft^2 5.13 ft^2, 7 ft^2, and 8.55 ft^2). Guillotine and split type breaks were studied, but in all cases, splits yielded the higher pressures. The 7.0 ft^2 break resulted in the highest pressure of 53.4 psig which is approximately the same pressure as the worst hot leg break. The pressure time responses for the six suction breaks are shown on Figures 14-6 through 14-11 and the 4 discharge breaks are shown on Figures 14-12 through 14-15. Although the primary path for flow from the core is through the vent valves, the secondary side energy is removed. The effect is obvious in the pressure-time response curves. Table 14-1 shows the peak pressure and the time when the peak occurs for each of the breaks in the hot and cold legs.

The core inlet velocity during the reflood stage is oscillatory in nature but, by using the integral of the core inlet flow path, the core average velocity was determined and is shown on Figure 14-16 for the 7.0 ft^2 suction break. The CRAFT code conservatively calculates an average carryout rate fraction of approximately 0.9 of the core inlet flow. Table 14-2 shows the leak flow rate and enthalpy for the 14.1 ft^2 hot leg break and Table 14-3 shows the same quantities for the 7.0 ft^2 cold leg break at the pump suction. As can be seen from Table 14-3, the leak flow was zero from 36 to 40 seconds and again from 44 to 48 seconds which is caused by the building pressure, as calculated by CRAFT, coming into equilibrium with RCS pressure. To show that the input is still conservative, Figure 14-17 shows the comparison between the building pressure calculated by CRAFT and CONTEMPT. As can be seen from this figure, the CONTEMPT pressure is higher during the first 200 seconds than the CRAFT pressure which implies that zero leak flow would have occurred earlier.

In order to provide a better understanding of where the energy came from, Table 14-4 shows an energy balance at t = 0 sec., and at the time of peak pressure at t = 120 sec. for the 7 ft^2 cold leg break at the pump suction. Energy added by the core, steam generators ECCS, and building cooling systems is also shown. The reference temperature for these calculations was 32^{o}F with the exception of the reactor building structures where the initial building temperature of 100^{o}F was used. As can be seen from this table and Figure 14-7, all of the available energy sources have contributed substantially to the reactor building pressure response and considerable margin still remains between the peak building pressure and building design pressure.

LIST OF TABLES
APPLYING TO QUESTIONS
14.1.1, 14.1.2, 14.1.3 and 14.1.4

LIST OF FIGURES
APPLYING TO QUESTIONS 14.1.1,14.1.2, 14.1.3, 14.1.4

TABLE 14-1

PEAK REACTOR BUILDING PRESSURE VERSUS BREAK AREA AND LOCATION

Area, ft^2	Location	Peak Pressure, psig	Time of Peak, sec
14.1	Hot Leg	53.5	94
11.0	Hot Leg	53.3	100
8.55	Hot Leg	52.9	100
5	Hot Leg	52.7	140
8.55	Cold Leg (Pump Suction)	52.3	120
7	Cold Leg (Pump Suction)	53.4	120
5.13	Cold Leg (Pump Suction)	52.6	140
3	Cold Leg (Pump Suction)	50.0	160
2	Cold Leg (Pump Suction)	49.1 (1st Peak) 48.7 (2nd Peak)	76 157
.5	Cold Leg (Pump Suction)	41.1	240
8.55	Cold Leg (Pump Discharge)	49.8	21
5.13	Cold Leg (Pump Discharge)	48.5	26
3	Cold Leg (Pump Discharge)	47.8	35
.5	Cold Leg (Pump Discharge)	41.1	239

TABLE 14-2

MASS RATE AND ENTHALPY TO THE REACTOR BUILDING FOR A 14.1-FT^2 HOT LEG BREAK

Time Interval (sec)	Average Mass Flow Rate (lb/sec)	Average Enthalpy (Btu/lb)
0-2	80,249.5	599.2
2-4	60,009.	595.1
4-6	46,675.5	601.9
6-8	30,404.5	636.1
8-10	14,993.	736.3
10-12	4,807.	1031.8
12-14	2376.	1181.3
14-16	1636.	1065.7
16-18	1326.5	611.3
18-21	259.33	780.2
21-26	0.	0.
26-30	321.5	269.8
30-34	953.	274.1
34-38	54.	263.8
38-42	546.75	304.5
42-44	168.	857.1
44-46	1540.5	296.9
46-48	2936.5	325.3
48-50	2084.	510.5
50-52	2337.5	459.6
52-54	2419.5	504.2
54-56	2963.	403.4
56-58	3391.5	369.4
58-60	3268.5	395.5
60-70	2938.1	404.8
70-80	2185.1	472.7
80-90	1778.3	510.4
90-100	1387.	420.0
100-120	493.85	455.6
120-140	403.65	376.3
140-160	366.2	366.0
160-180	363.75	350.1
180-200	368.	337.0
200-220	370.85	326.2
220-240	365.8	315.3
240-260	397.25	303.0
260-280	482.7	293.3
280-300	453.8	290.6

TABLE 14-3

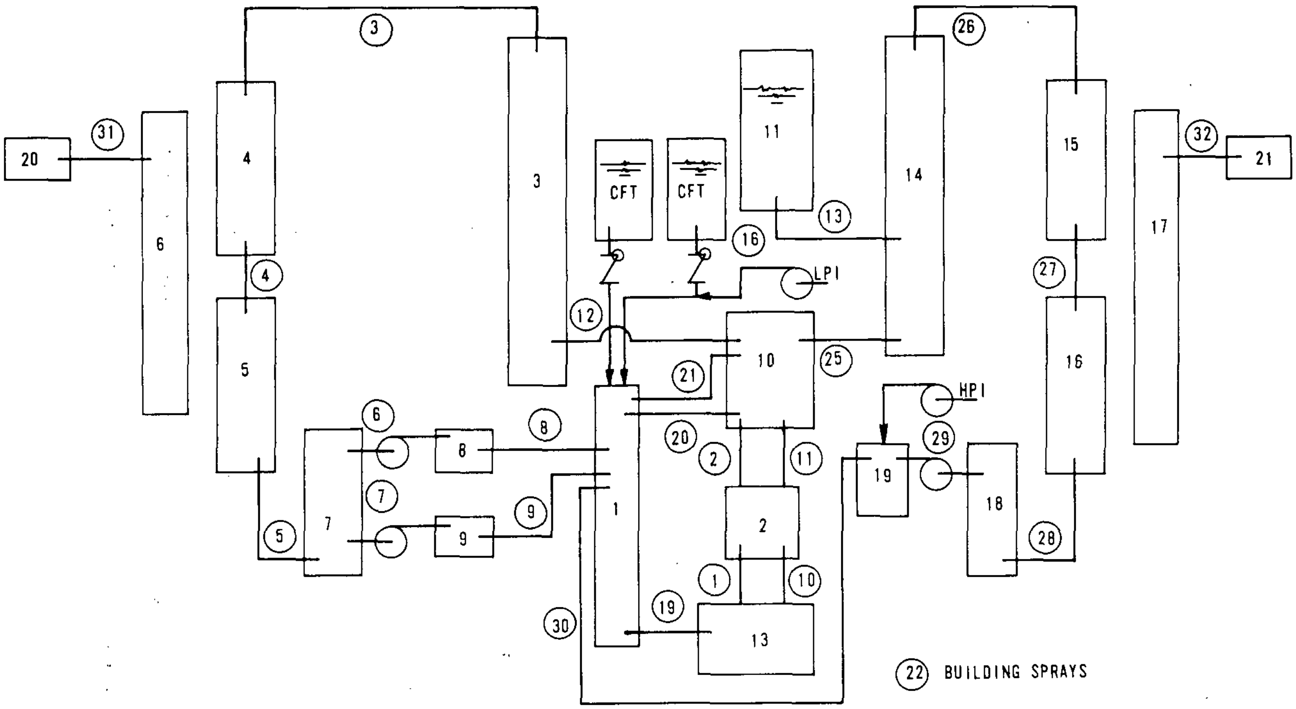
MASS RATE AND ENTHALPY TO THE REACTOR BUILDING
FOR A 7-FT^2 SPLIT AT THE PUMP SUCTION

Time Interval (sec)	Average Mass Flow Rate (lb/sec)	Average Enthalpy (Btu/lb)
0-2	57300	558.333
2-4	53350	566.382
4-6	47035	583.019
6-8	34195	617.780
8-10	22187	689.503
10-12	12904	765.916
12-14	4768	1086.838
14-16	4630	735.501
16-18	5605	520.250
18-20	5416	457.903
20-24	2893	419.165
24-28	889	412.658
28-32	14	298.246
32-36	38	337.748
36-40	0.0	0.0
40-44	48	333.333
44-48	0.0	0.0
48-56	79	265.263
56-62	1427	416.472
62-68	656	1073.895
68-74	1477	604.153
74-80	2688	461.519
80-90	2479	477.207
90-100	1959	538.497
100-110	1131	692.002
110-120	561	888.632
120-140	157	1133.524
140-160	54	1180.801
160-180	50	1182.093
180-200	48	1148.691
200-240	323	431.353
240-280	673	474.770
280-320	556	438.506
320-360	340	344.001
360-400	413	322.518
400-440	540	316.633
440-480	391	300.288
480-520	383	280.439
520-560	359	275.384
560-600	345	273.663

TABLE 14-4

ENERGY DISTRIBUTION FOR THE 7 FT^2 BREAK (SPLIT) AT REACTOR COOLANT PUMP SUCTION

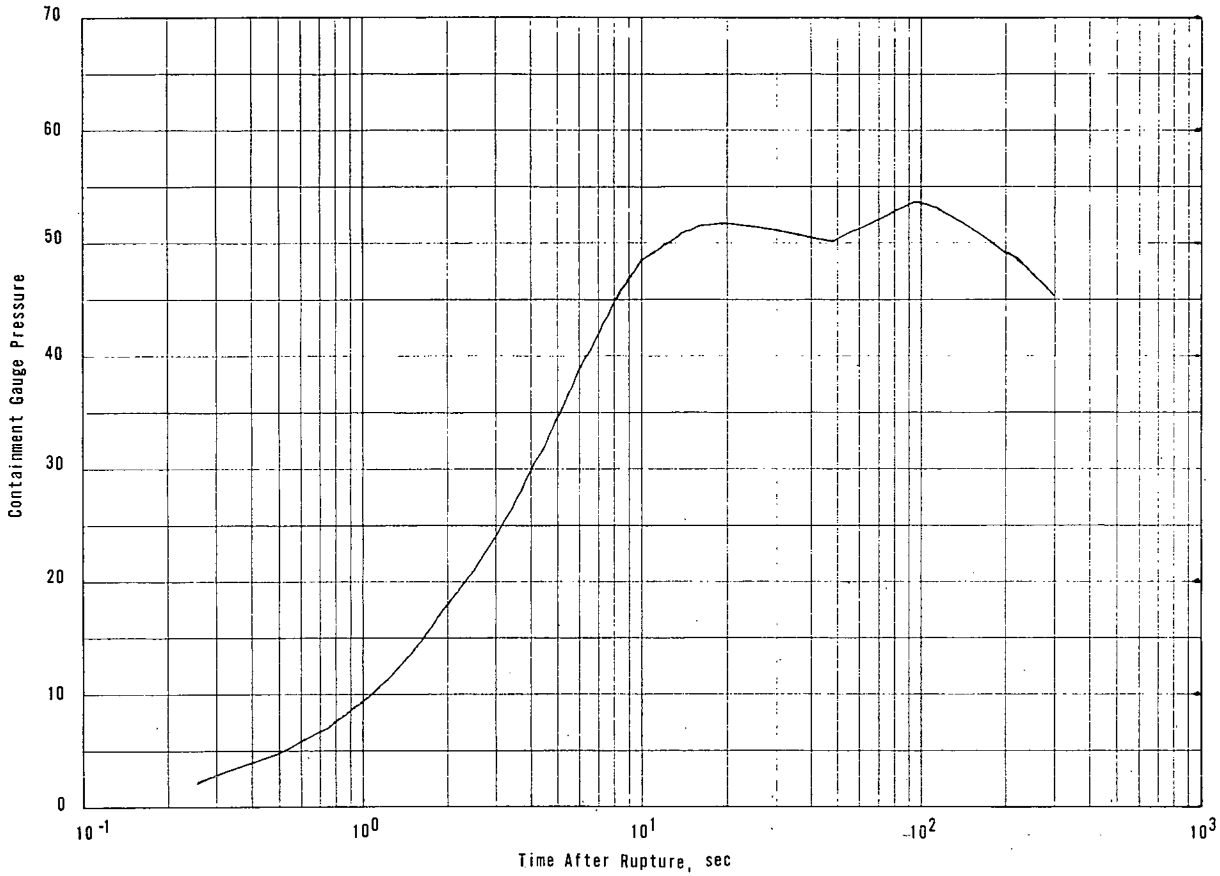
Description	Energy Before Accident, Btu x 10^{-6}	Energy Added Between 0 and 120 s	Energy at Time of Peak Pressure (120 sec), Btu x 10^{-6}
1. Reactor Coolant System, Coolant	298.17		24.28
2. Reactor Coolant System, Structures			
a. Fuel & Cladding	22.95		5.89
b. Vessel, piping, pressurizer and primary side of steam generators	157.22		147.31
3. Core Heat Generation		17.32	
4. ECCS Coolant		13.04	
5. From Secondary System Including Tubes		31.03	
6. Reactor Building Atmosphere	1.78		245.72
7. Reactor Building Sump	0.0		82.0
8. Reactor Building Structures	0.0		32.89
9. Reactor Building Coolers		- 3.96	
10. Reactor Building Sprays		.54	
	480.12	57.97	538.09



REACTOR BUILDING IS NODE 12

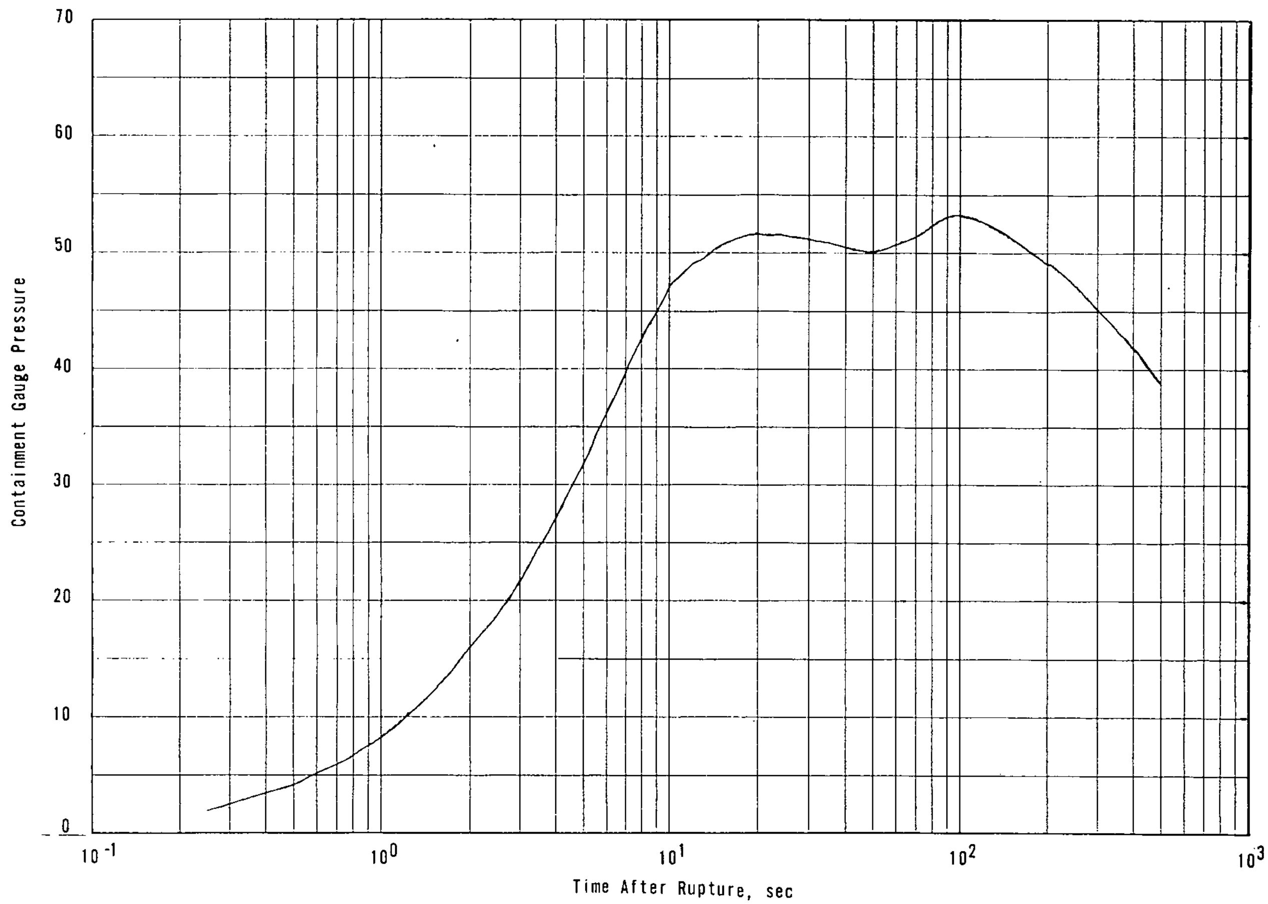
MULTINODE REPRESENTATION OF
NUCLEAR STEAM SUPPLY SYSTEM

Figure 14-1



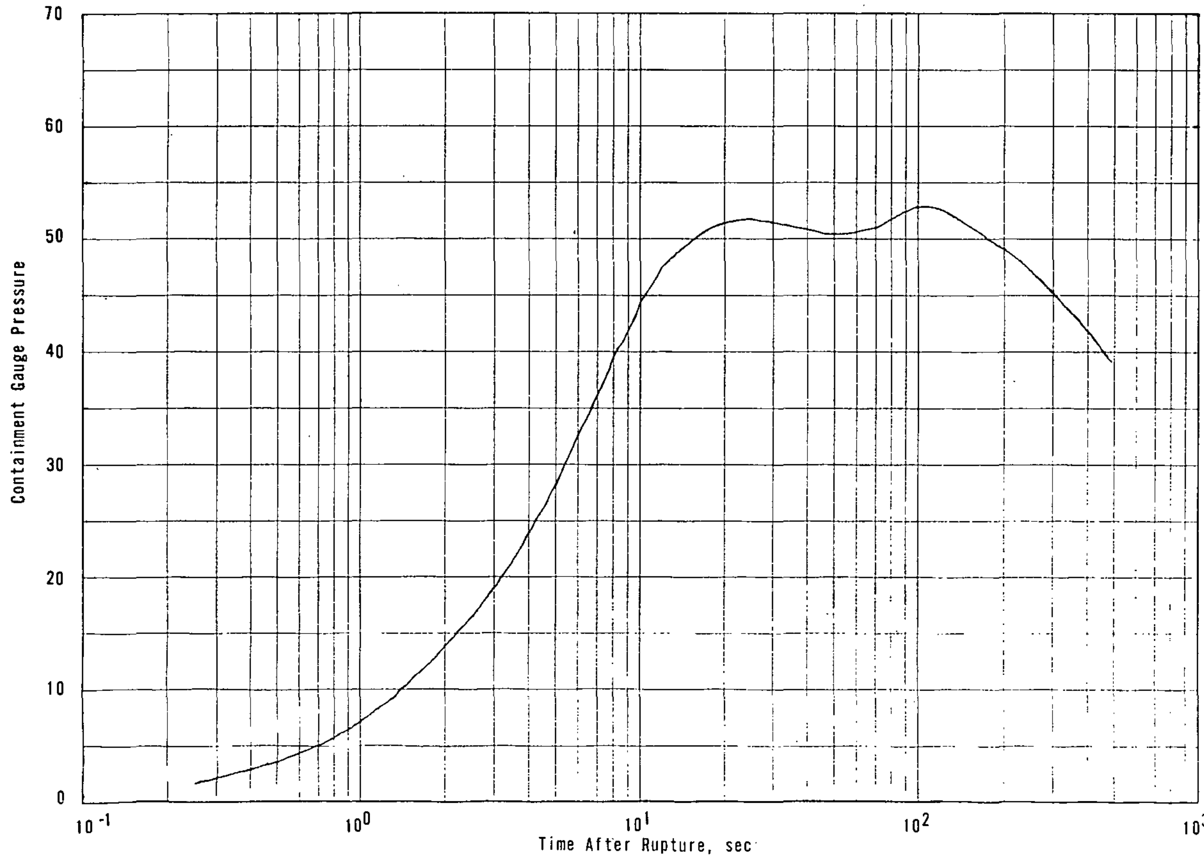
REACTOR BUILDING PRESSURE VERSUS
TIME FOR A 14.1 FT^2 HOT LEG BREAK

Figure 14-2



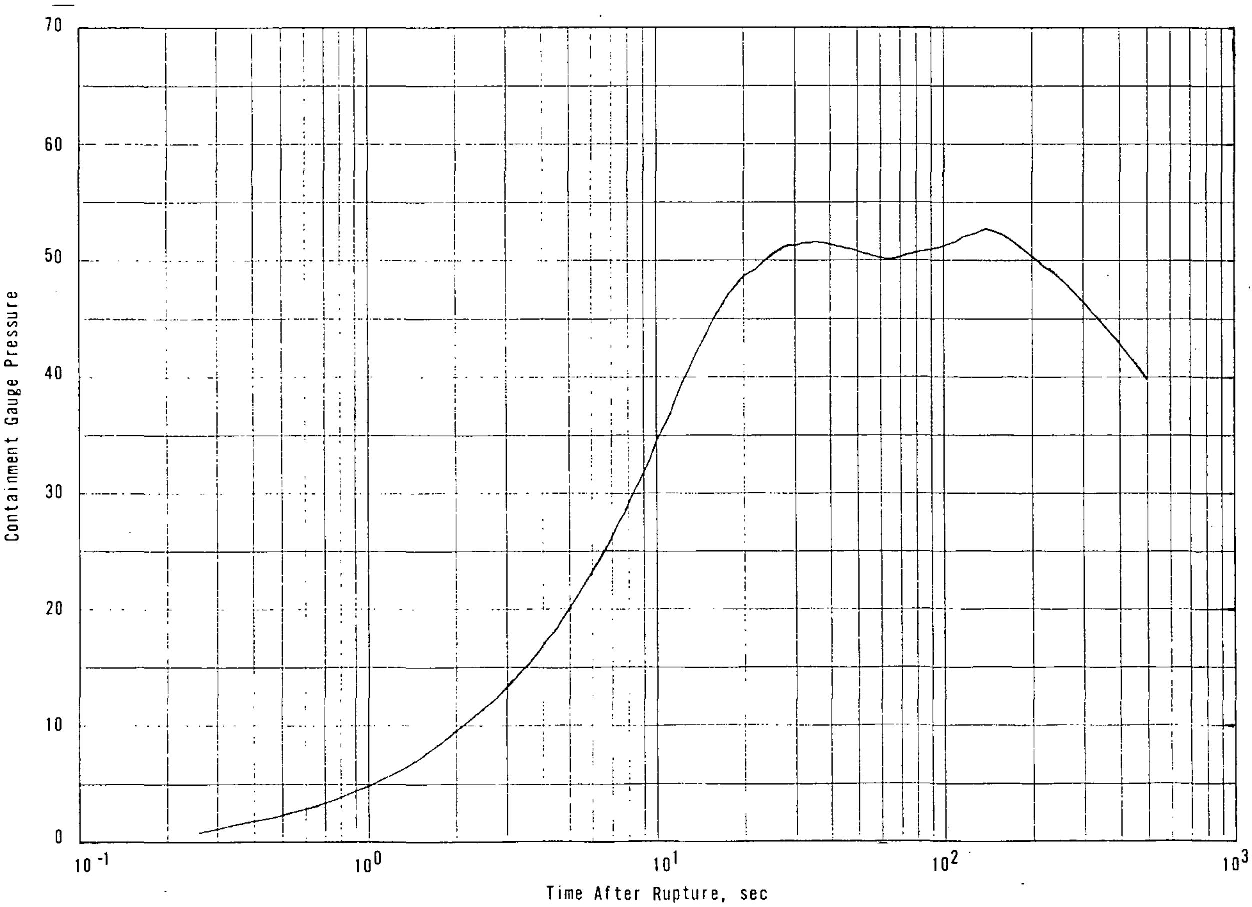
REACTOR BUILDING PRESSURE VERSUS TIME FOR A 11.0 FT^2 HOT LEG BREAK

Figure 14-3



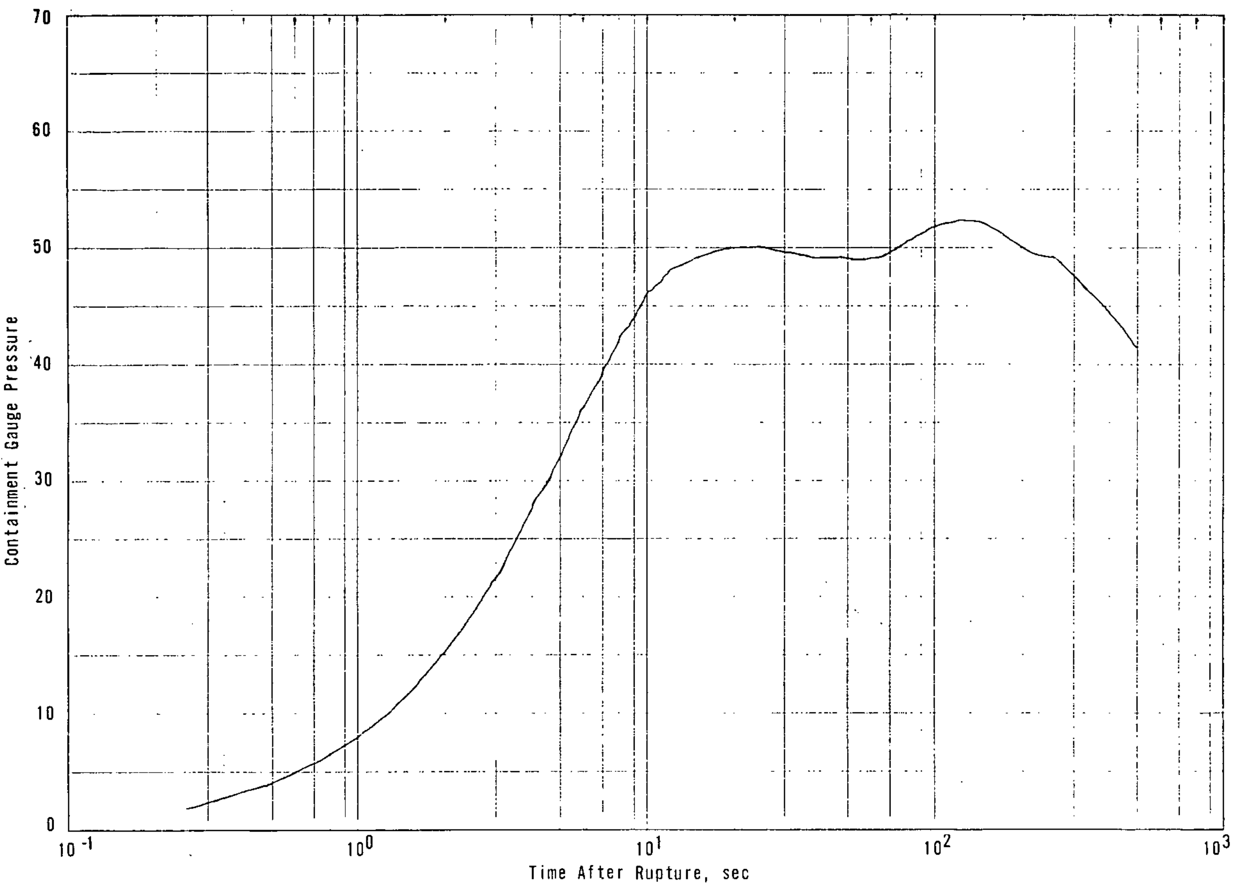
REACTOR BUILDING PRESSURE VERSUS TIME FOR A 0.56 FT^2 HOT LEG BREAK

Figure 14-4



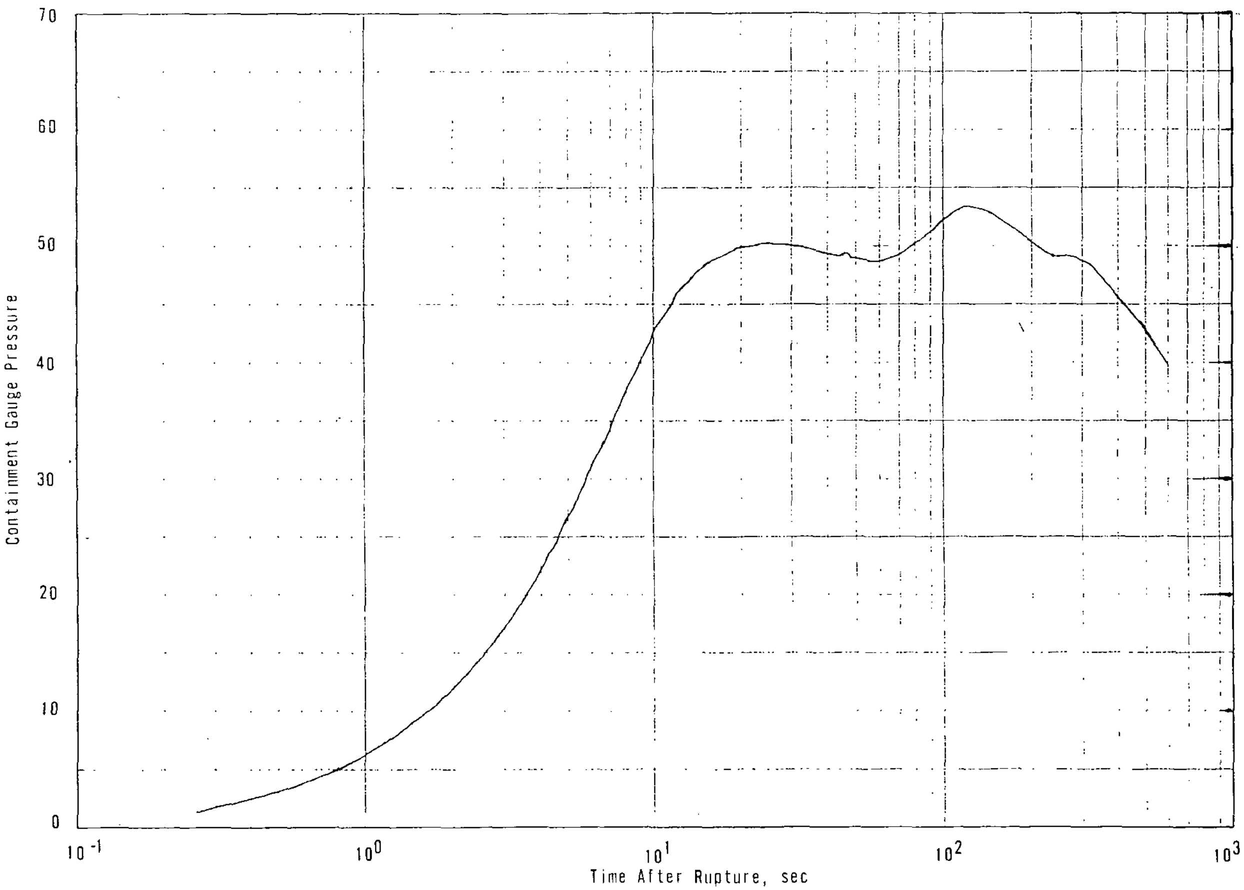
REACTOR BUILDING PRESSURE VERSUS TIME FOR A 5.0 FT^2 HOT LEG BREAK

Figure 14-5



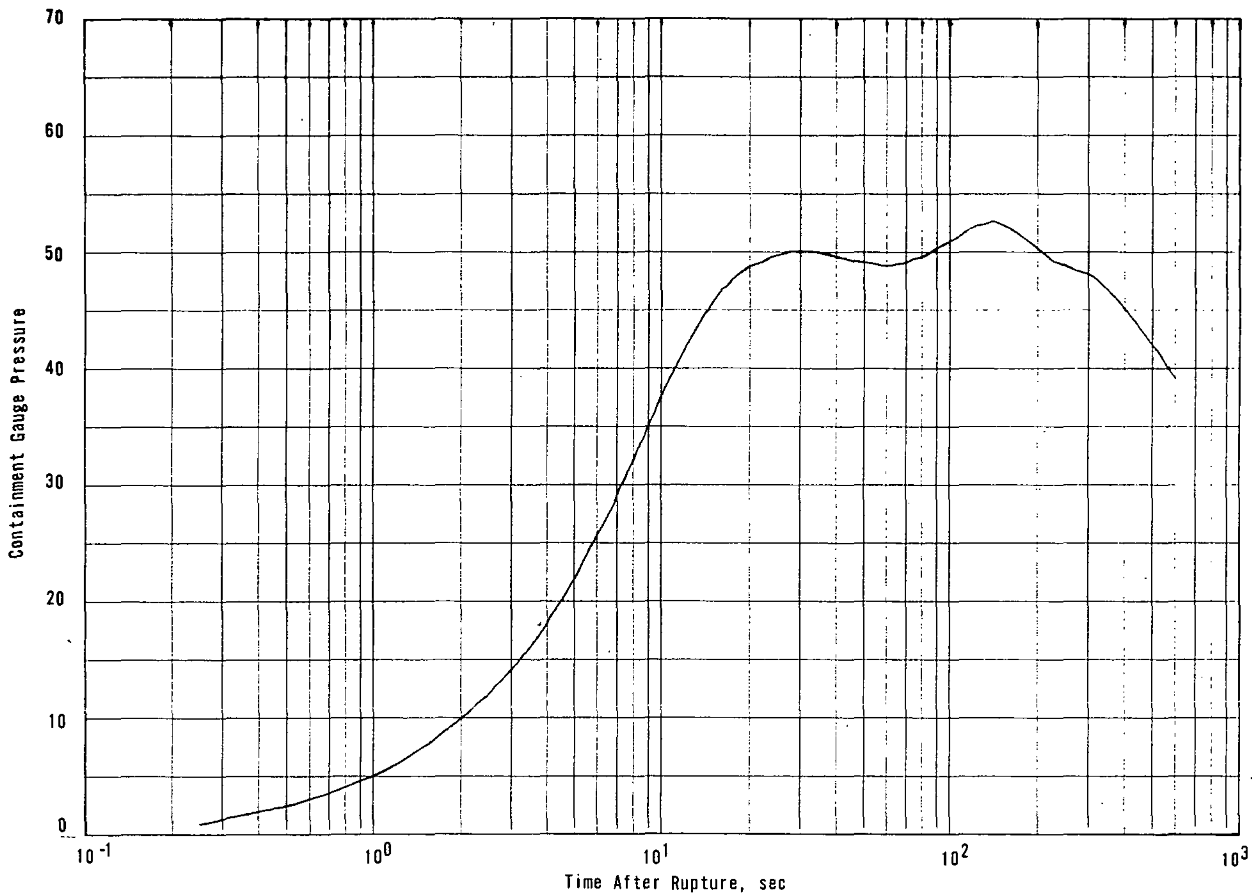
REACTOR BUILDING PRESSURE VERSUS TIME FOR A 8.55 FT^2 COLD LEG BREAK (PUMP SUCTION)

Figure 14-6



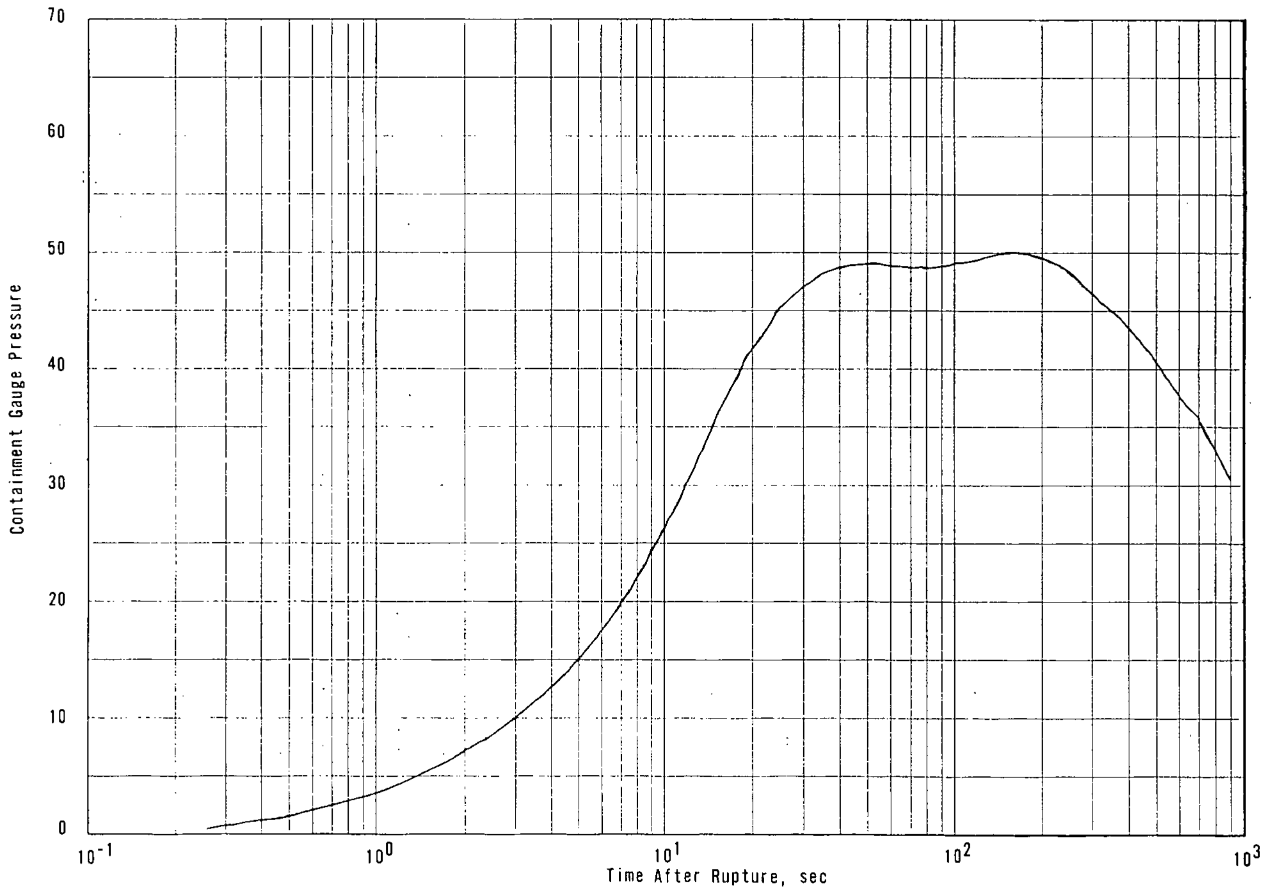
REACTOR BUILDING PRESSURE VERSUS TIME FOR A 7.0 FT^2 COLD LEG BREAK (PUMP SUCTION)

Figure 14-7



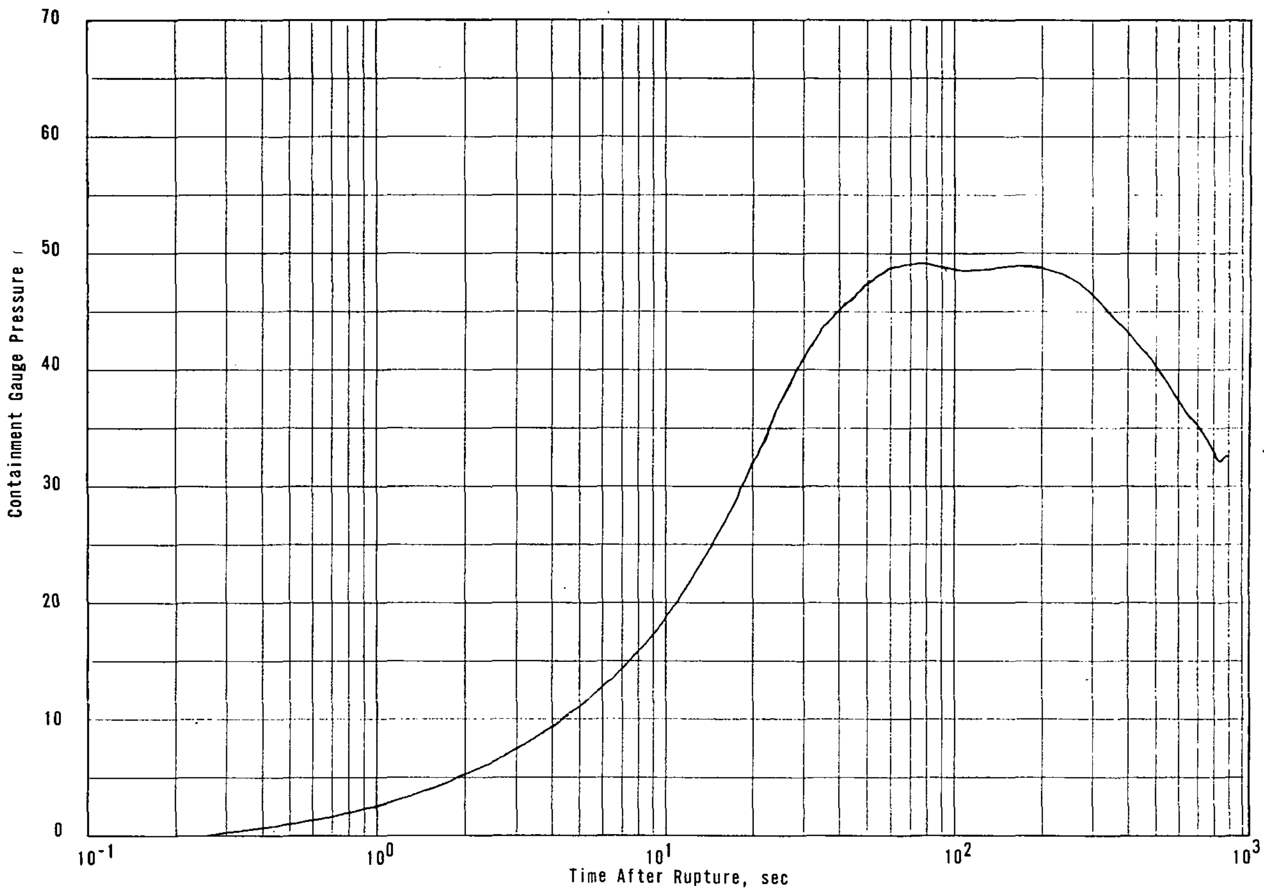
REACTOR BUILDING PRESSURE VERSUS TIME FOR A 5.13 FT^2 COLD LEG BREAK (PUMP SUCTION)

Figure 14-8



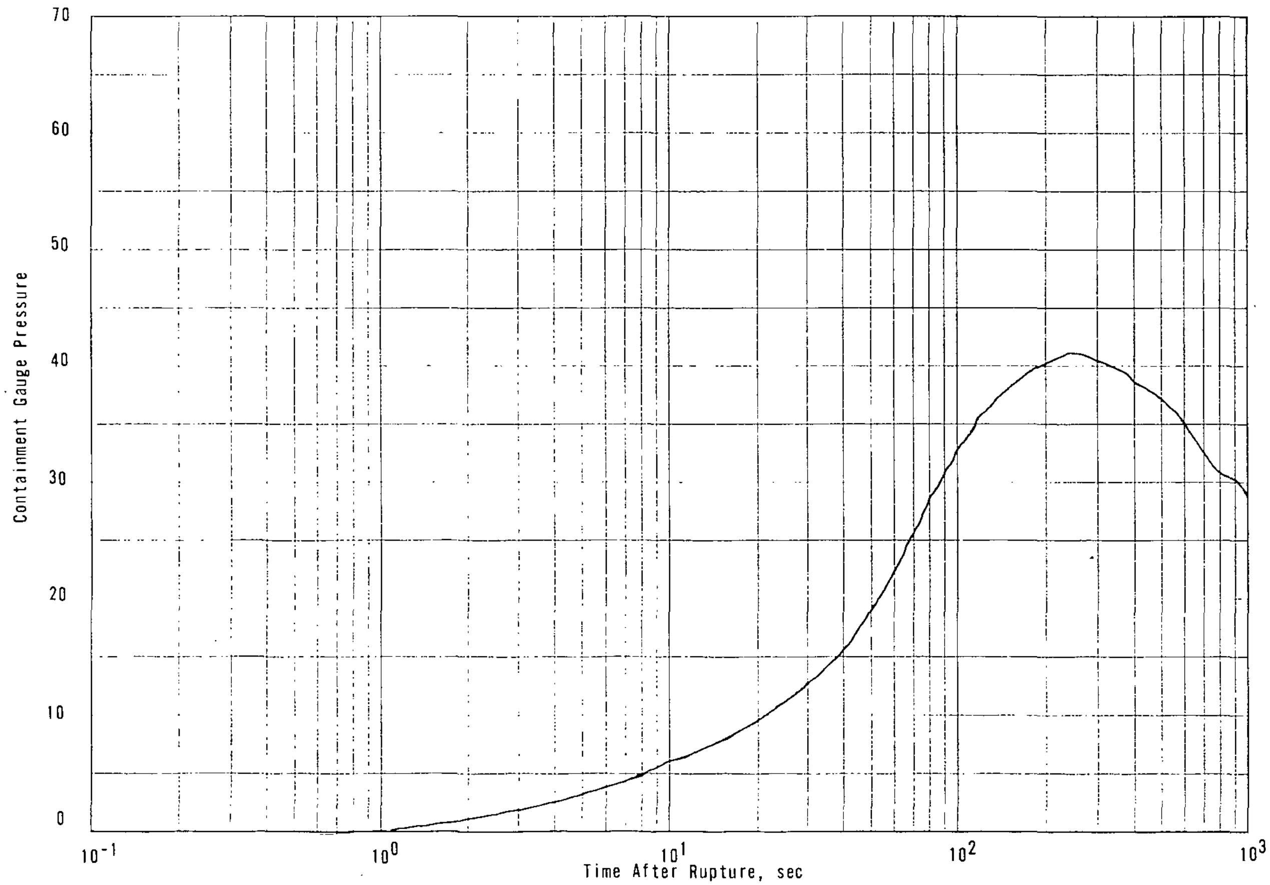
REACTOR BUILDING PRESSURE VERSUS TIME FOR A 3.0 FT^2 COLD LEG BREAK (PUMP SUCTION)

Figure 14-9



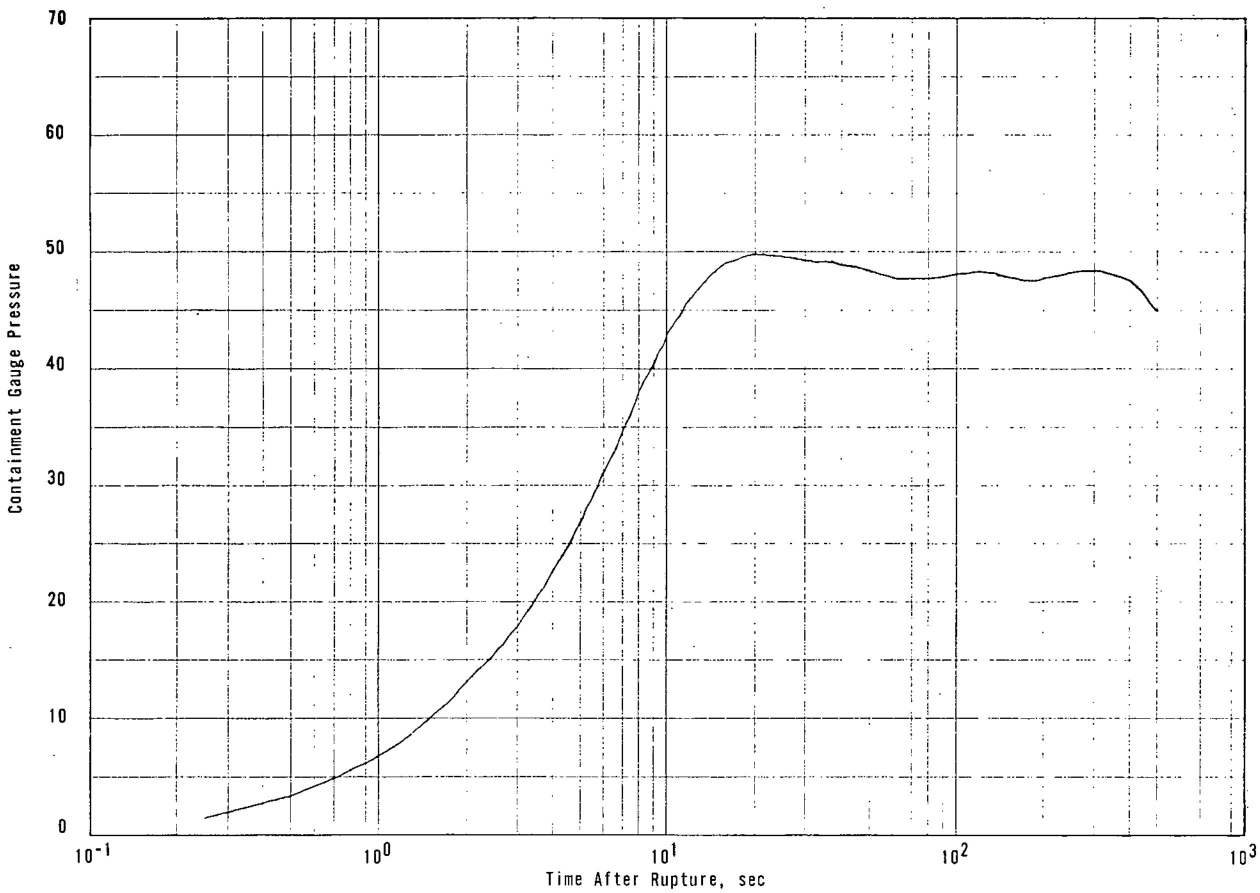
REACTOR BUILDING PRESSURE VERSUS TIME FOR A 2.0 FT^2 COLD LEG BREAK (PUMP SUCTION)

Figure 14-10



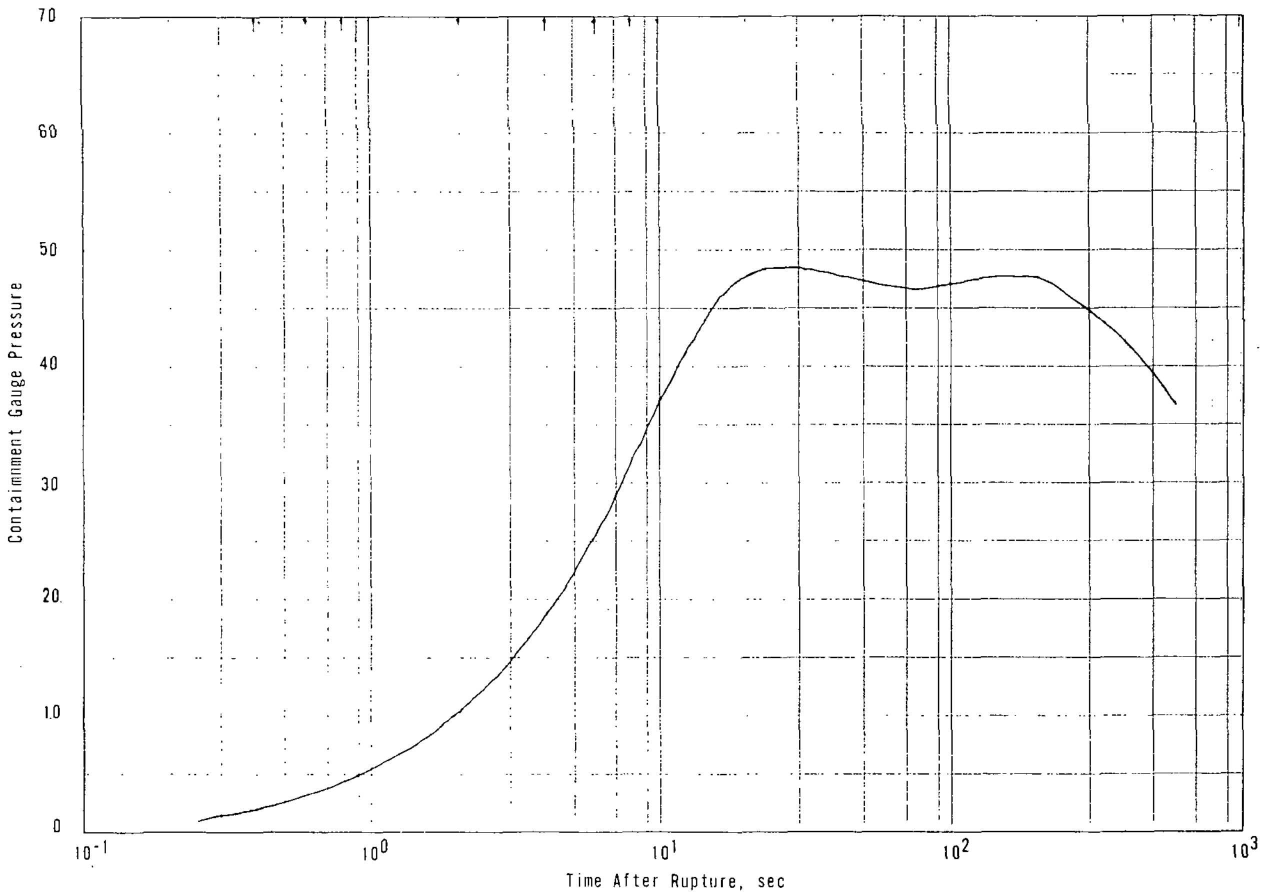
REACTOR BUILDING PRESSURE VERSUS TIME FOR A 0.5 FT^2 COLD LEG BREAK (PUMP SUCTION)

Figure 14-11



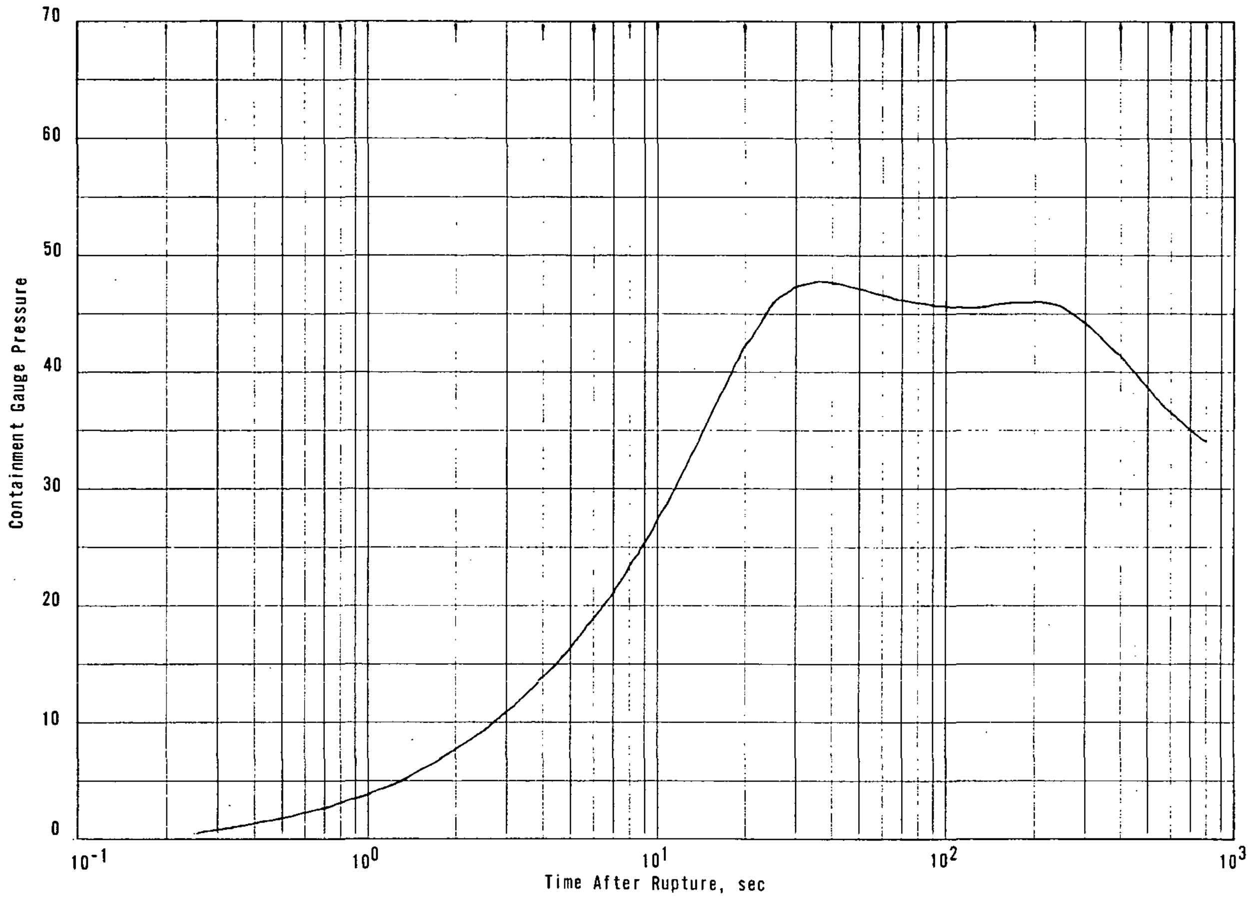
REACTOR BUILDING PRESSURE VERSUS TIME FOR A 8.55 FT^2 COLD LEG BREAK (PUMP DISCHARGE)

Figure 14-12



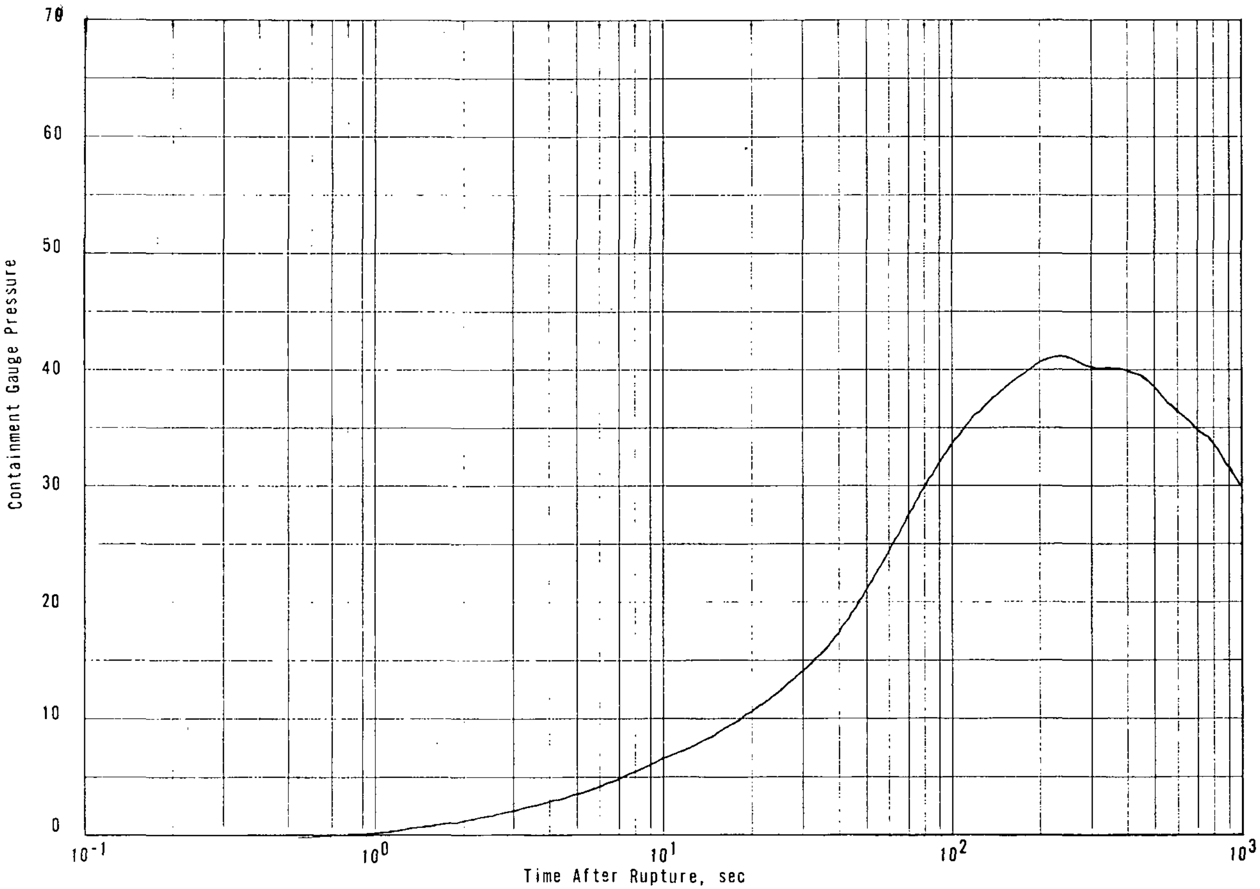
REACTOR BUILDING PRESSURE VERSUS TIME FOR A 5.13 FT^2 COLD LEG BREAK (PUMP DISCHARGE)

Figure 14-13



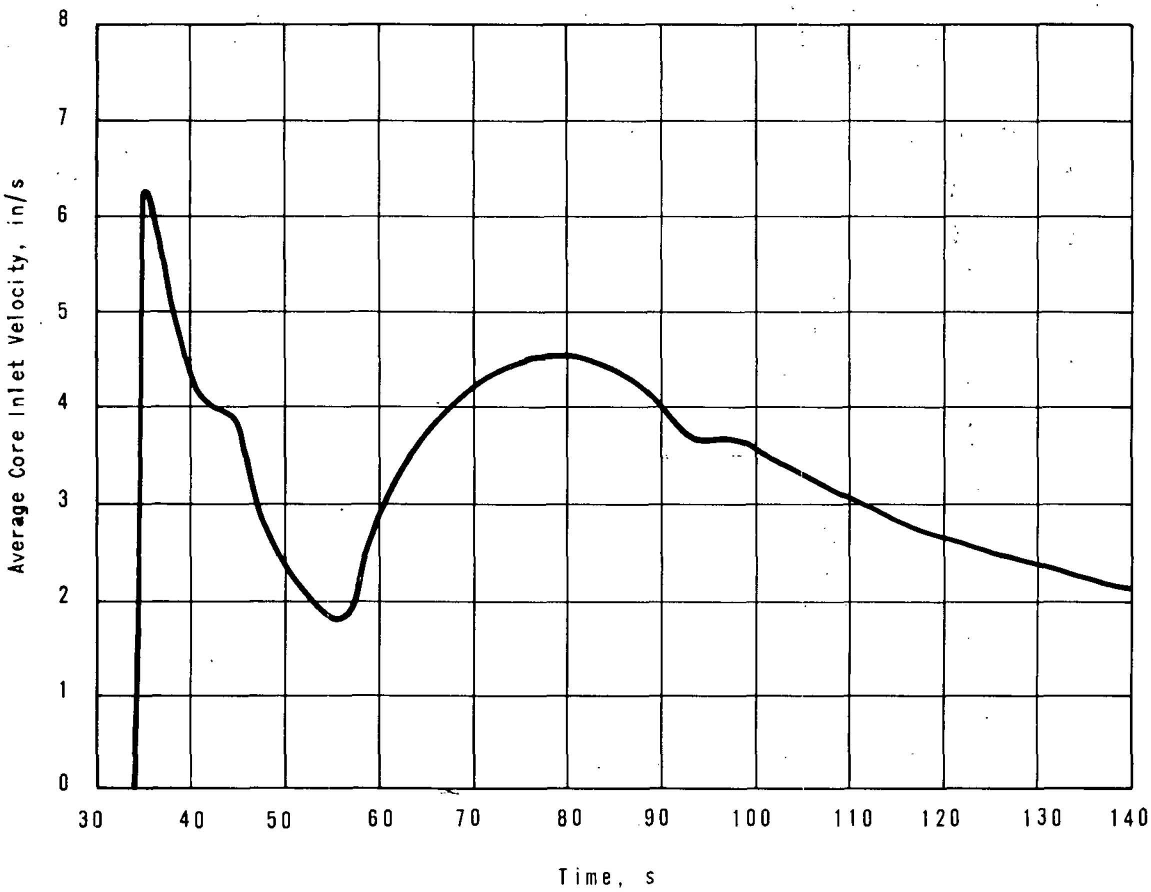
REACTOR BUILDING PRESSURE VERSUS TIME FOR A 3.0 FT^2 COLD LEG BREAK (PUMP DISCHARGE)

Figure 14-14



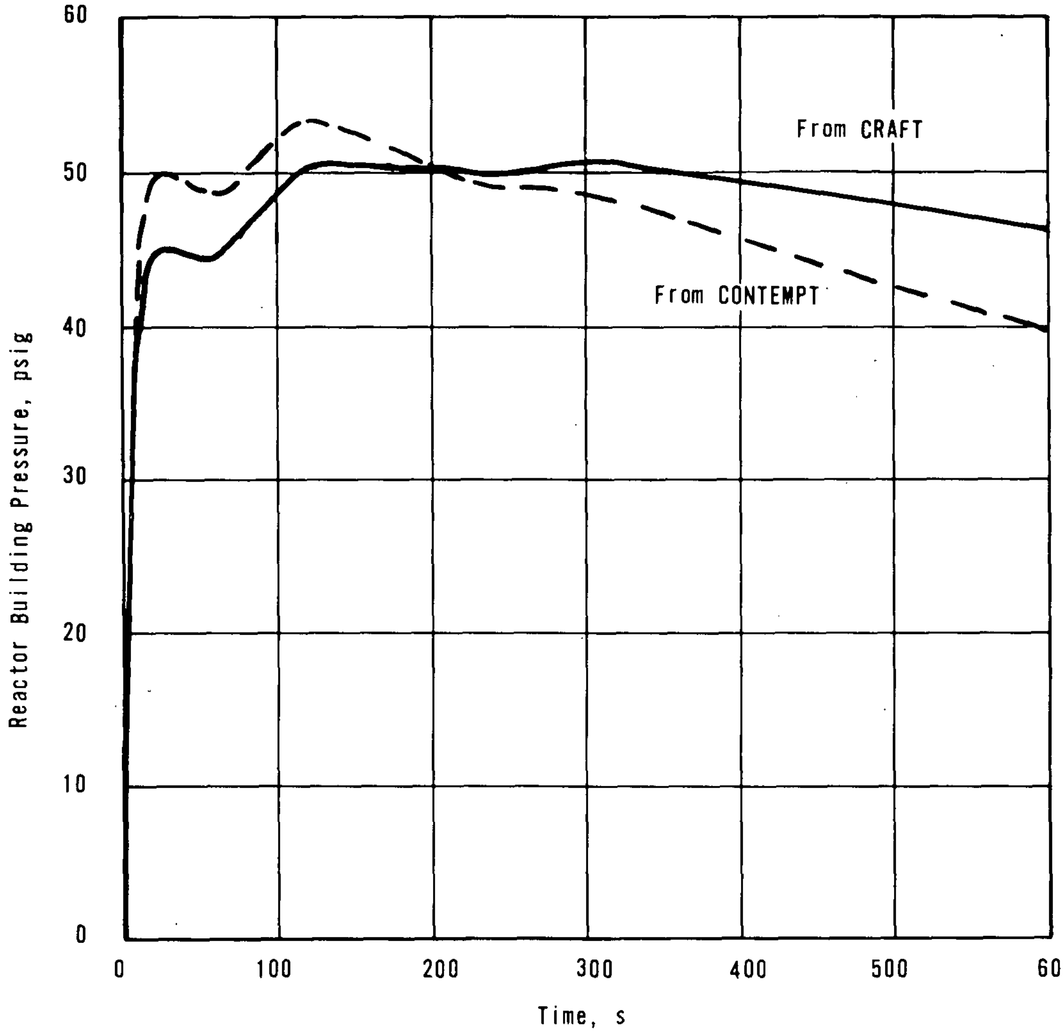
REACTOR BUILDING PRESSURE VERSUS TIME FOR A 0.5 FT^2 COLD LEG BREAK (PUMP DISCHARGE)

Figure 14-15



AVERAGE CORE INLET VELOCITY VERSUS TIME FOR A 7.0 FT^2 COLD LEG BREAK (PUMP SUCTION)

Figure 14-16



COMPARISON OF CRAFT AND CONTEMPT
REACTOR BUILDING PRESSURES

Figure 14-17

In response to questions contained in Mr. R. C. DeYoung's letter of September 26, 1972, the following information is submitted:

Question- 14.2 For spectrum of breaks, recompute peak cladding temperature using the CRF model in the REFLOOD code. Include, in addition to usual information (T_{clad}, metal-water reaction, etc.), the procedures for calculating h versus time, after beginning of core recovery. Vary location of axial q max.

Answer-- The evaluation of Babcock and Wilcox's reactor design that uses internal vent valves and have power levels up to 2568 MWt during a loss-of-coolant accident is reported in BAW-10034, Rev. 3[1]. For that analysis, the REFLOOD code[2], used by B&W for flooding rate computation, combined FLECHT heat transfer coefficients with a conservative entrainment assumption (20%) to determine effluent from the core during reflooding. As an alternate, however, B&W has developed a correlation for the carryout rate fraction using data from the PWR-FLECHT program.

This correlation has now been used in the REFLOOD program for the spectrum of breaks reported in BAW-10034, Rev. 3, to determine the fraction of core inlet flow that is entrained and/or converted into steam during each loss-of-coolant accident. This change in the reflood model results in slightly lower flooding rates and minor increases in the calculated maximum, hot spot cladding temperatures. Table 14.2-1 summarizes pertinent parameters and results for the spectrum of breaks considered, and Figures 14.2-1 through 14.2-30 show the core flooding rate, post-blowdown heat transfer coefficient, and the cladding temperature response for each break in the order presented in Table 14.2-1. For the 8.55 ft^2 split case with a symmetrical power shape (peak at midplane), the heat transfer coefficient and coolant temperature during the blowdown period are also shown. All of the pertinent parameters related to core cooling for this break and power shape are shown on Figures 14.2-25 through 14.2-30.

The REFLOOD code uses a carryout rate fraction that assumes that entrainment of water by the steam does not start until the quench front or water level has reached an elevation of 18 inches in the core resulting in an initial high flooding rate. This carryout rate fraction is described in detail in the redirect and rebuttal testimony filed by B&W on October 26, 1972 at the ECCS public hearing. A further justification for using 18 inches as the starting point for entrainment can be obtained by examining the axial pressure drop data from the FLECHT tests. These data show that the pressure drop can be related to the static head of the incoming water until the water level reaches ≈24 inches in the bundle. This can be seen for flooding rates ranging from 1 to 6 inches/sec.

Using this correlation for the worst cold leg break, 8.55 ft^2 split at the pump discharge, the maximum cladding temperature increased from 2177F to 2186F. The largest increase in peak clad temperature occurred for the 3.0 ft^2 split in the cold leg pipe at the pump discharge. An increase of 76F in the peak cladding temperature was calculated when the carryout rate fraction correlation based on FLECHT data was used. For this relatively small break, the reflood portion of the transient is restricted somewhat due to the lack of flow available from the core flooding tanks after the quench front reaches 18 inches into the core. Thus, throwing away the CFT water which entered during the blowdown period plus using the FLECHT carryout rate fraction correlation causes a reduction in flooding rate and heat transfer coefficient which in turn results in a higher cladding temperature. In addition, the reflood analysis for the 0.5 ft^2 split in a cold leg pipe predicts a low flooding rate, shown in Figure 14.2-19, shortly after the water reaches the bottom of the core. To conservatively analyze the cladding response during this time period (81 to 85s), the adiabatic heatup of the fuel was extended until 85 seconds after the end of blowdown. At that time FLECHT heat transfer coefficients based on a 1 in/s flooding rate were used.

To insure that the use of the 1.7 design, axial power shape, which peaks at the 3 foot elevation, is still conservative when the carryout rate fraction correlation is used, the worst cold leg break was analyzed using a 1.67 symmetrical power shape. Calculations show that the maximum cladding temperature is 2135F. Since the same break produces a cladding temperature of 2186F, 51F higher, when Qmax is located at the 3.0 foot elevation; the B&W evaluation model is justifiably conservative in its selection of an inlet, axial power shape.

Although the carryout rate fraction correlation has only a slight effect on the peak cladding temperatures, its use does decrease the rate at which the cladding temperature falls after the peak has been reached. This phenomena produces an increase in local and core metal-water reaction. In fact, the values shown in Table 14.2-1 are approximately a factor of two greater than those reported in BAW-10034, Rev. 3. This increase is quite significant, but still well within the limitation set forth in the AEC Interim Acceptance Criteria.

The heat transfer coefficients used in the reflood portion of the THETA 1-B calculations are computed using the FLECHT correlations for various flooding rates. To end some confusion concerning methods employed by B&W in converting the flooding rates predicted by the REFLOOD code to FLECHT heat transfer coefficients, a detailed explanation of the calculations involved for the worst cold leg break is presented.

The 8.55 ft^2 split at the pump discharge has been shown to result in the worst hot spot cladding temperature. For this particular break, the end of blowdown is calculated to be 18.75, and the hot pin is assumed to undergo adiabatic heatup until the water in the vessel reaches the bottom of the core (6.93 seconds later). The instantaneous flooding rate that follows is shown in Figure 14.2-7. The REFLOOD code continually integrates the entering core flow, and the resulting integrated mass is used to determine the uniform (square wave) flooding rates.

For Figure 14.2-7, two (square wave) flooding rates are used to calculate the FLECHT heat transfer coefficients. The first is used for the time interval of 6.93s to 10.4s; the flooding rate is fairly high initially because entrainment of water is assumed to be zero until the water reaches the 18 inch elevation in the core. The end of the first interval (10.4s) is selected at the point where the flooding rate drops to its lowest value following the initiation of entrainment. At 10.4s the total mass injected into the core is calculated to be 6895.82 pounds. To determine the (square wave) flooding rate the total mass injected into the core from 6.93s to 10.4s is converted to an equivalent flooding rate as follows:

$$\text{Flooding Rate} = \frac{(\text{Total Mass Injected})\ (\text{Specific Volume of Water})}{(\text{Core Flooding Area})\qquad (\text{Time Interval})}$$

$$= \frac{(6895.82\ \text{lb})\ (.01726\ \text{ft}^3/\text{lb})\ \ (12\ \text{in/ft})}{(65.7\ \text{ft}^2)\qquad (10.4 - 6.93\text{s})}$$

$= 6.265$ in/s   Where the specific volume of water is taken at saturated pressure conditions (49 psia) and the core and the core bypass are conservatively assumed to flood at the same rate.

For the actual heat transfer calculations, a flooding rate of 6.25 in/s is used. For the remainder of the reflooding period, 10.4s to 70s, an additional 35422 pounds of water are injected into the core. In a like manner, an equivalent flooding rate of 1.8 in/s is computed.

Having established the square wave approximations to the reflood curve, the heat transfer coefficients for each flooding rate are calculated using the FLECHT correlation (WCAP-7665)[3]. Input to the FLECHT correlation is based on the following initial conditions which exist at the end of the adiabatic heat up period:

Initial Clad Temperature	= 2100F
Pressure	= 49 psia
Power	= 1.7 kw/ft
Percent Blockage	= 0.0
Inlet Subcooling	= 90F

The linear heat rate (1.7 kw/ft) represents the peak power for a symmetrical power shaped curve which matches the axial energy generation profile for the 1.7 inlet peak distribution up to the three foot elevation. Heat transfer coefficients as a function of time, for input into THETA 1-B are calculated in a manner similar to that suggested in the PWR-FLECHT Final Report.[3] The FLECHT heat transfer correlation is applied for flooding rates of 6.25 in/s and 1.8 in/s. The former flooding rate is used to determine the heat transfer coefficient during the first interval (3.47s). The 6.25 in/s flooding rate for 3.47 sec is equivalent to 1.8 in/s for 12 seconds. Therefore, the run made to determine h is entered in at 12 seconds for the 1.8 in/s flooding rate. However, the heat transfer coefficient is always adjusted so that the heat transfer coefficient is always equal to or smaller than that given by a simple excess mass approach. The resulting heat transfer coefficient is shown in Figure 14.2-8.

REFERENCES

1. C. E. Parks, et. al., Multi-Node Analysis of B&W's 2568 MWt Nuclear Plants During a Loss-of-Coolant Accident, BAW-10034, Rev. 3, Babcock & Wilcox Lynchburg, Va., May, 1972.

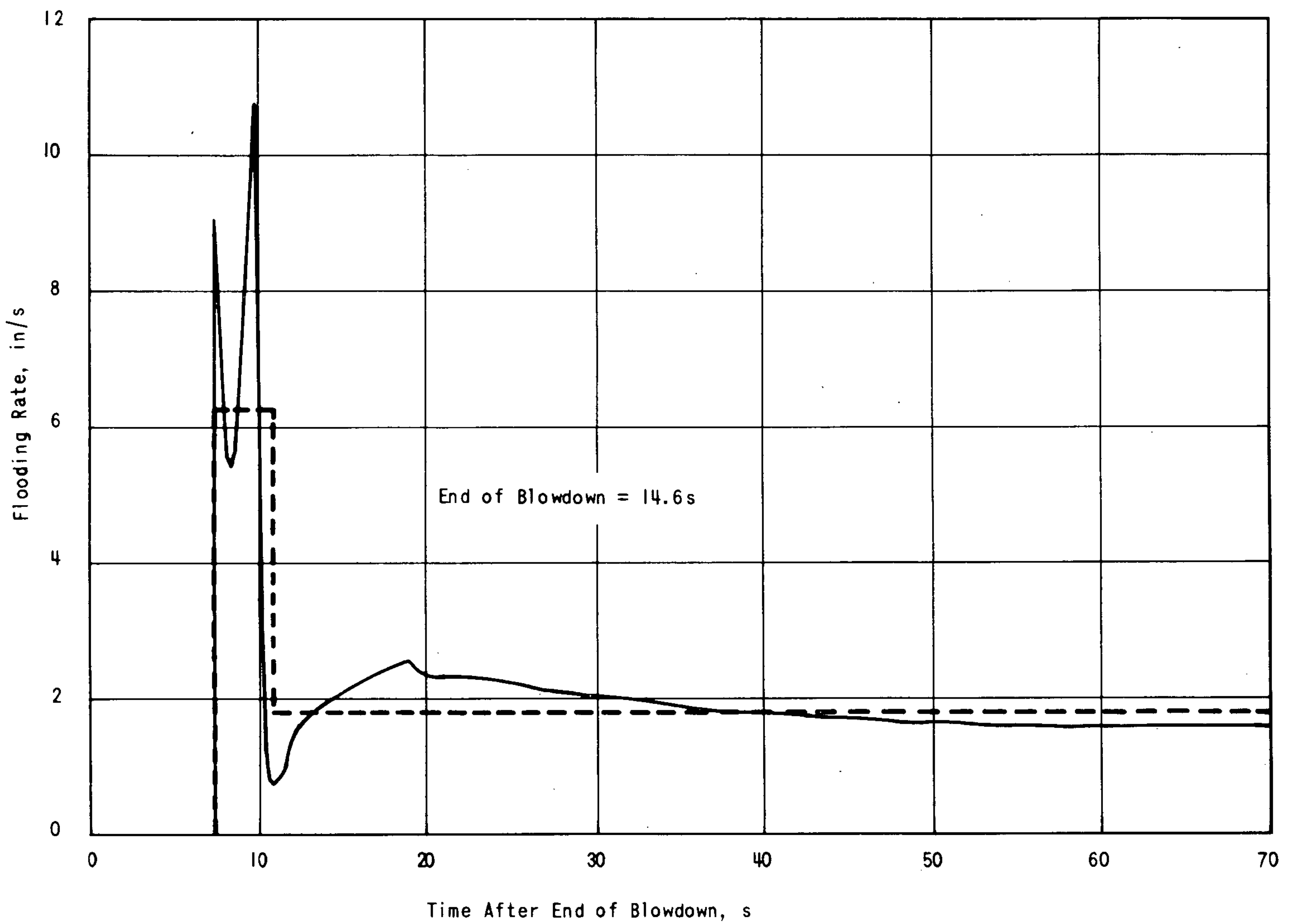
2. C. E. Parks and K. C. Shieh, REFLOOD - Description of Model for Multinode Core Reflood Analysis, BAW-10031, Supplement 1, Babcock & Wilcox, Lynchburg, Va., April, 1972.

3. F. F. Cader, et. al., PWR FLECHT (Full-Length Emergency Cooling Heat Transfer) Final Report, WCAP-7665, April, 1971.

TABLE 14.2-1
SUMMARY OF BREAK RESULTS

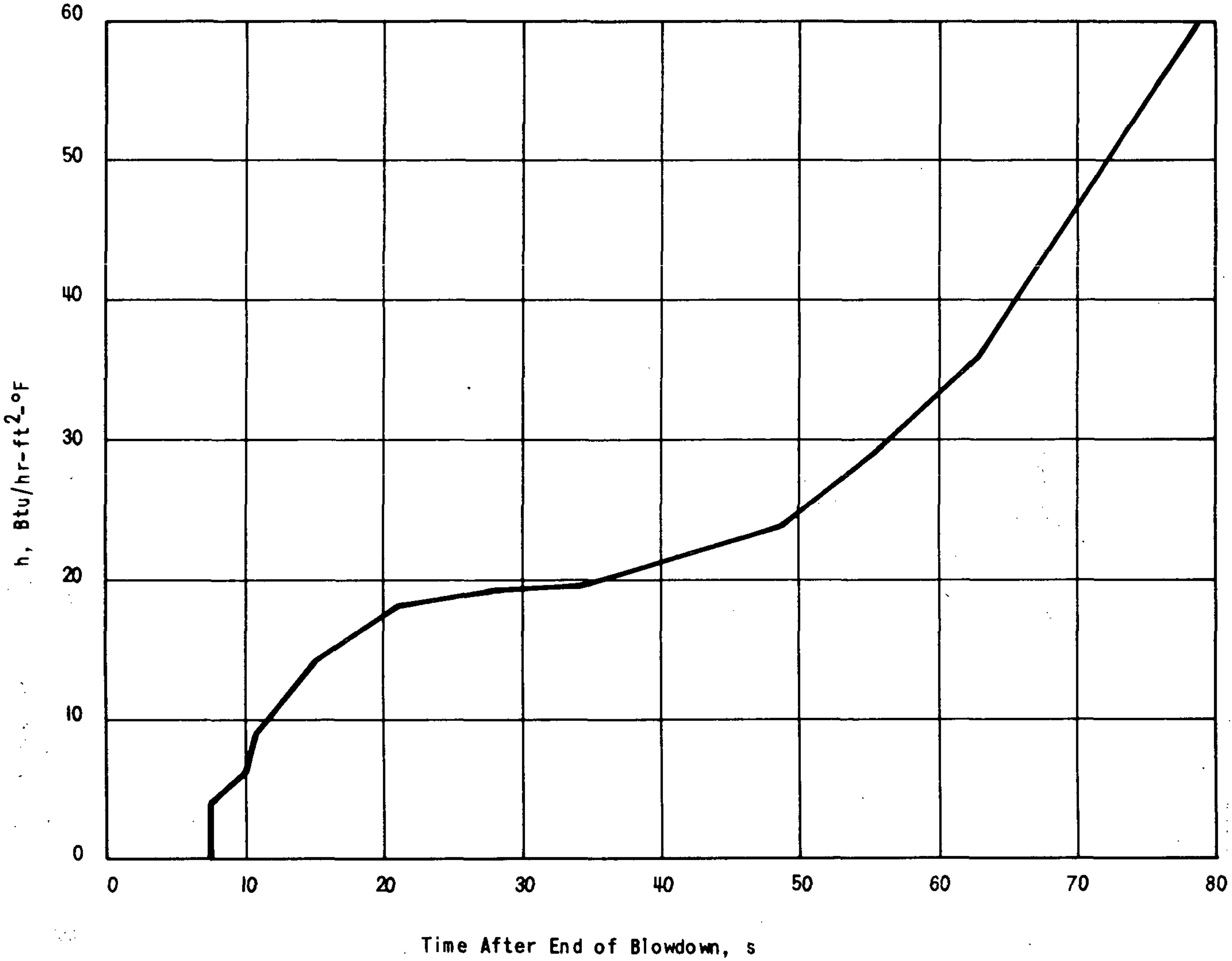
Break Size, ft²/Descrip.	Break Location	Start of CF Tank Injection,s	End of Blowdown,s	End of CF Tank Injection,s	Peak Cladding Temp., F	Metal-Water Reaction, % Local	Metal-Water Reaction, % Core
8.55/cold leg (Guillotine)	Pump Discharge	9.3	14.6	39.4	2082	2.11	.073
5.13/cold leg (Guillotine)	Pump Discharge	12.9	21.5	43.9	2029	1.8	.058
8.55/cold leg (Split)	Pump Discharge	11.1	18.7	41.9	2186	2.98	.09
5.13/cold leg (Split)	Pump Discharge	13.8	24.0	44.4	1994	1.8	.056
8.55/cold leg (Split)	Pump Suction	10.5	21.0	41.8	1899	1.15	.042
3.0/cold leg (Split)	Pump Discharge	20.3	31.8	51.6	1728	.046	.011
0.5/cold leg (Split)	Pump Discharge	119	192.5	197.5	1660	0.22	0.00
14.1/hot leg (Split)	Reactor Vessel Outlet	7.0	16.0	37.8	1670	.14	.003
8.55/cold leg (Split-Cosine Peak)	Pump Discharge	10.8	19.6	41.4	2135	4.2	.24

LIST OF FIGURES APPLYING TO QUESTION 14.2



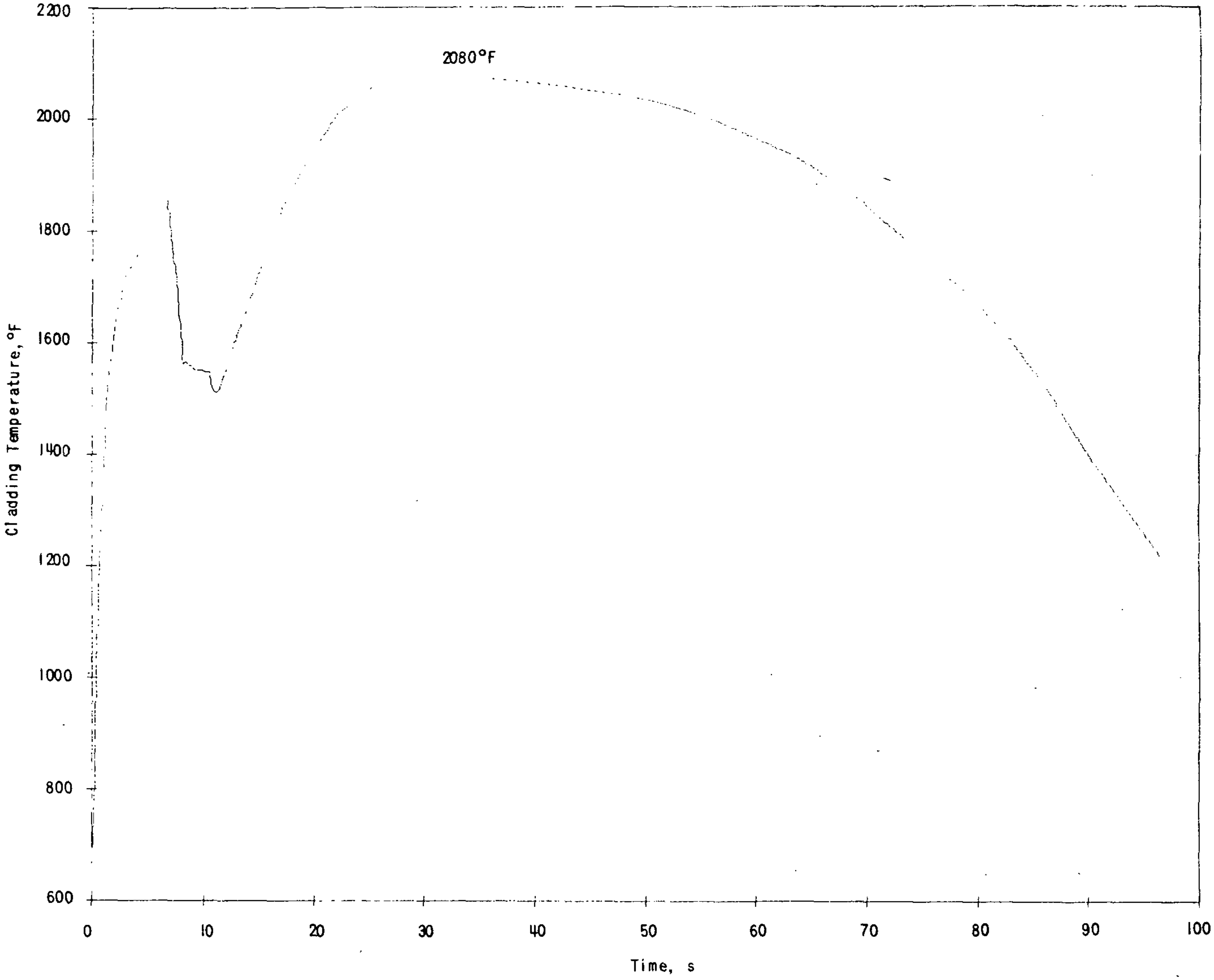
CORE FLOODING RATE FOR 8.55 FT^2 GUILLOTINE
COLD LEG BREAK AT PUMP DISCHARGE

Figure 14.2-1



POST BLOWDOWN HOT SPOT HEAT TRANSFER COEFFICIENT FOR 8.55 FT^2 GUILLOTINE COLD LEG BREAK AT PUMP DISCHARGE

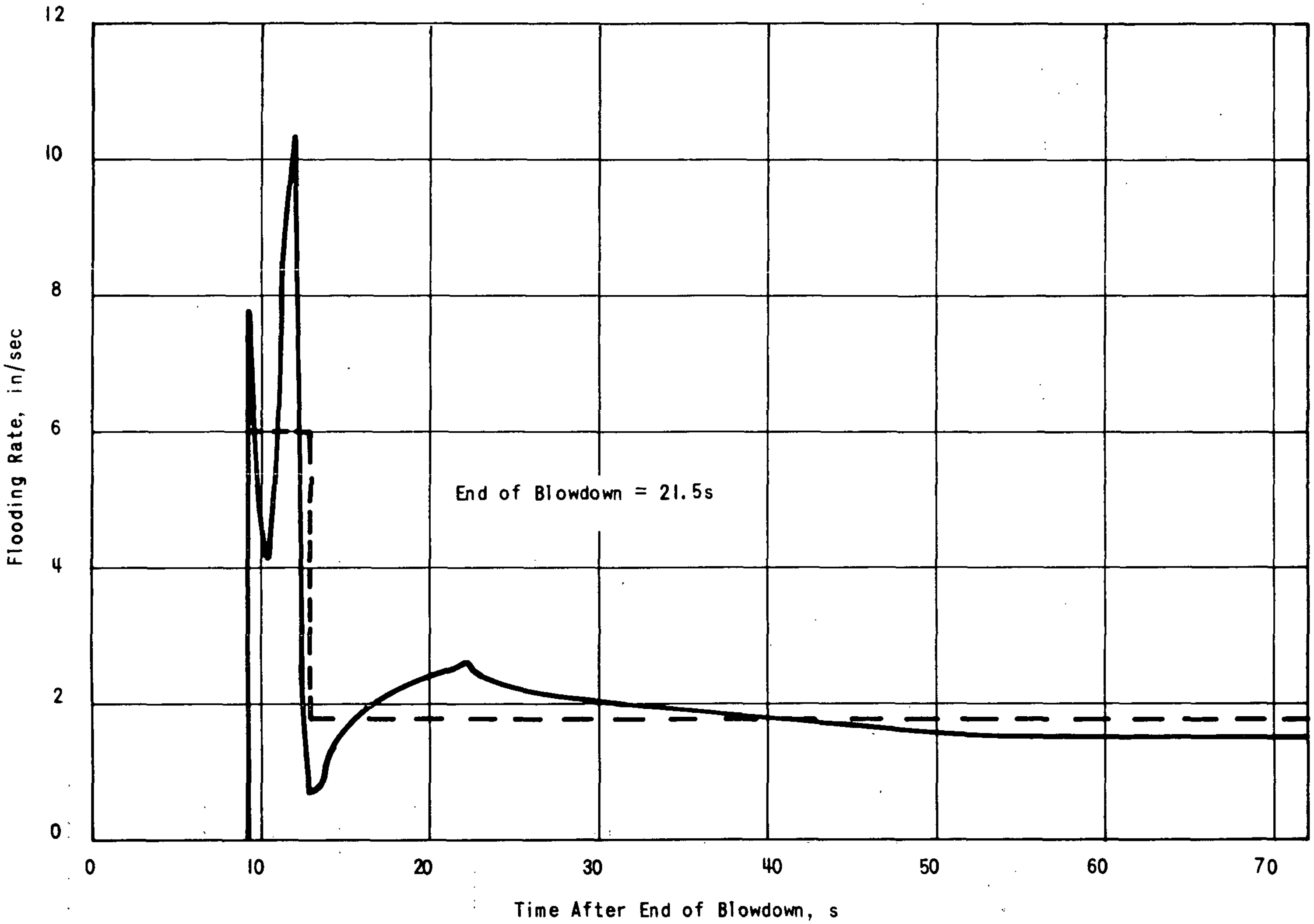
Figure 14.2-2



HOT SPOT CLADDING TEMPERATURE FOR 8.55 FT^2
GUILLOTINE COLD LEG BREAK AT PUMP DISCHARGE

Figure 14.2-3

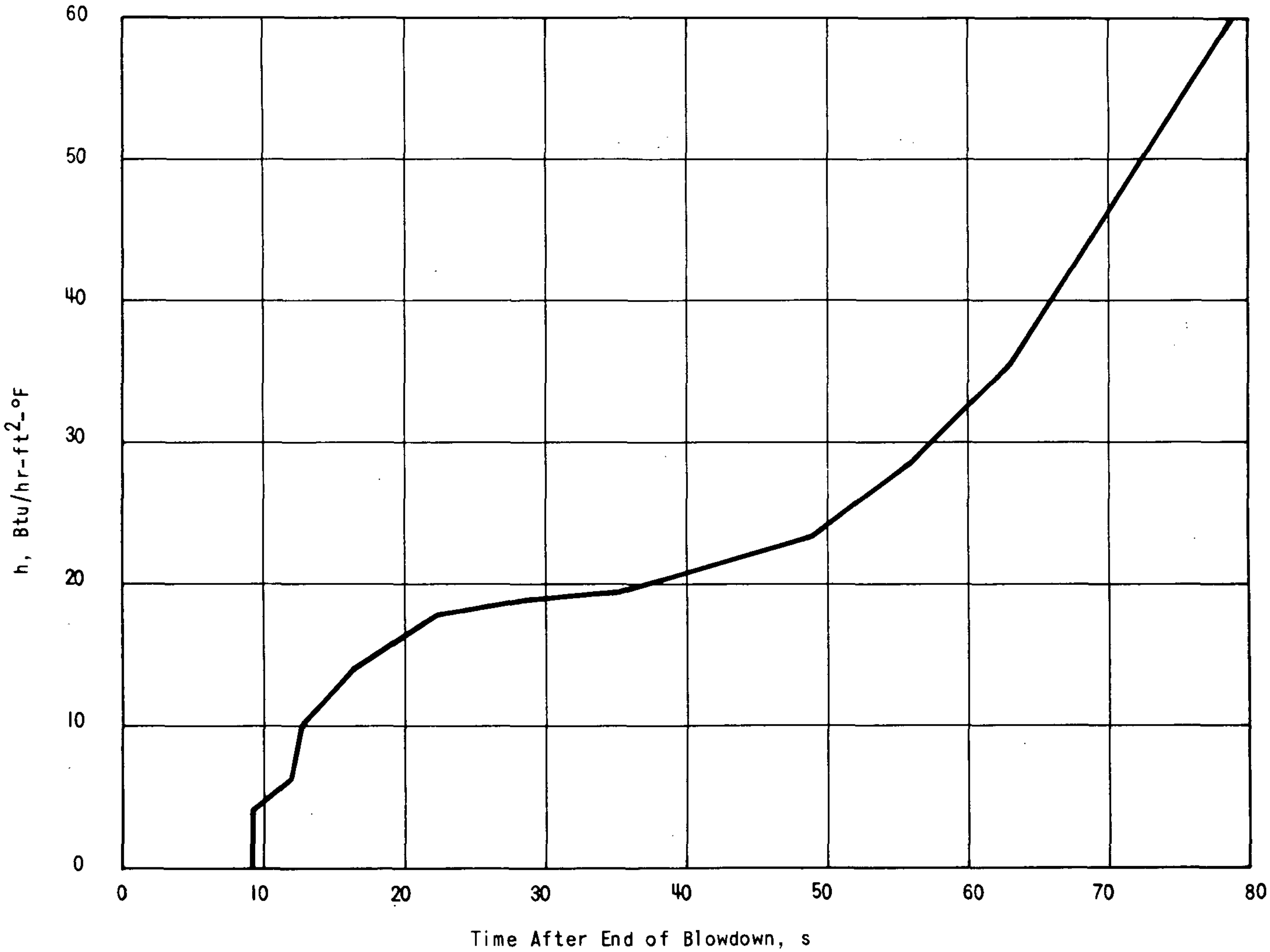
Docket 50-270 and -287
FSAR Supplement 13
January 29, 1973



CORE FLOODING RATE FOR 5.13 FT^2 GUILLOTINE
COLD LEG BREAK AT PUMP DISCHARGE

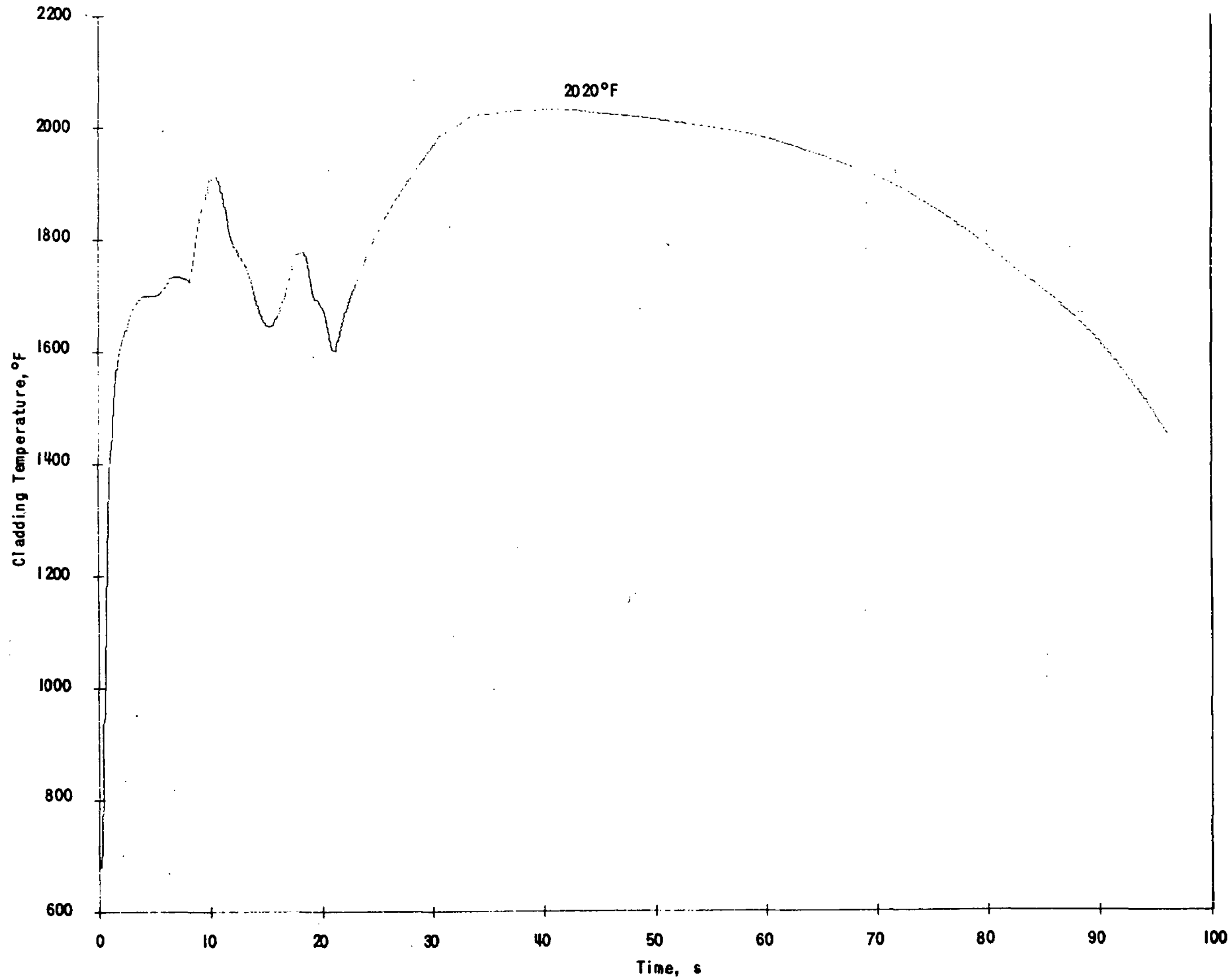
Figure 14.2-4

Docket 50-270 and -287
FSAR Supplement 13
January 29, 1973



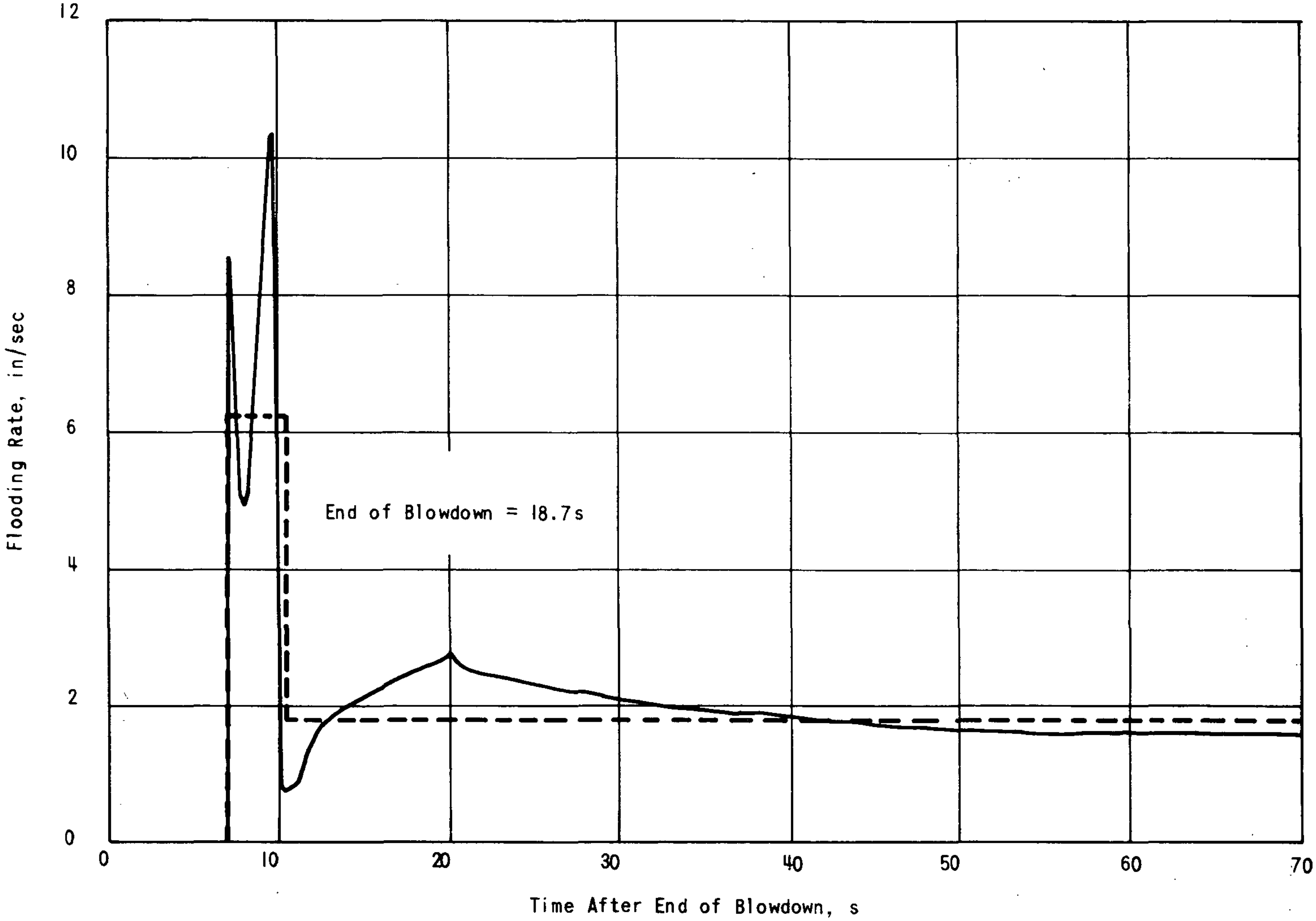
POST BLOWDOWN HOT SPOT HEAT TRANSFER
COEFFICIENT FOR 5.13 FT^2 GUILLOTINE
COLD LEG BREAK AT PUMP DISCHARGE

Figure 14.2-5



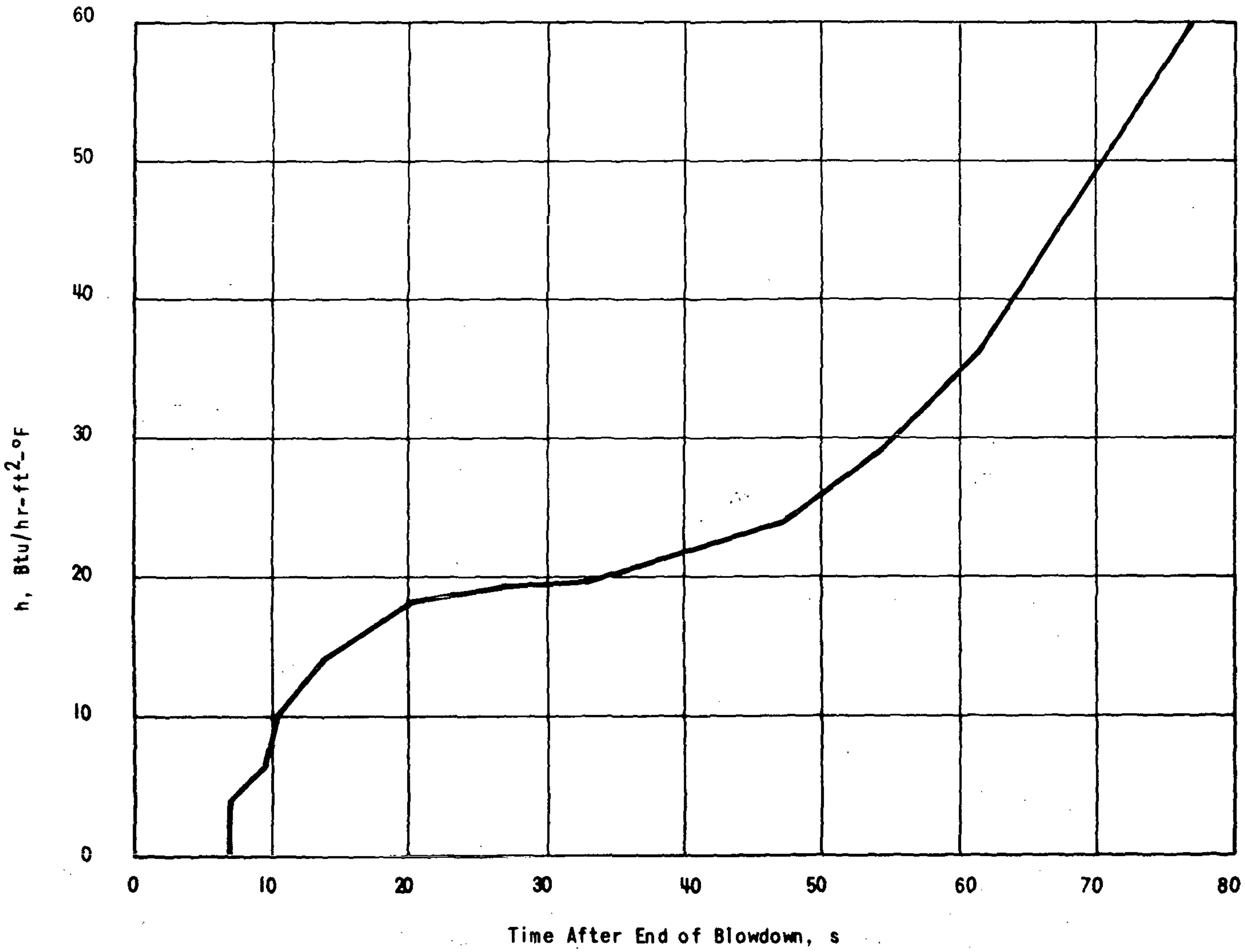
HOT SPOT CLADDING TEMPERATURE FOR 5.13 FT^2
GUILLOTINE COLD LEG BREAK AT PUMP DISCHARGE

Figure 14.2-6



CORE FLOODING RATE FOR 8.55 FT^2 SPLIT
IN COLD LEG PIPE AT PUMP DISCHARGE

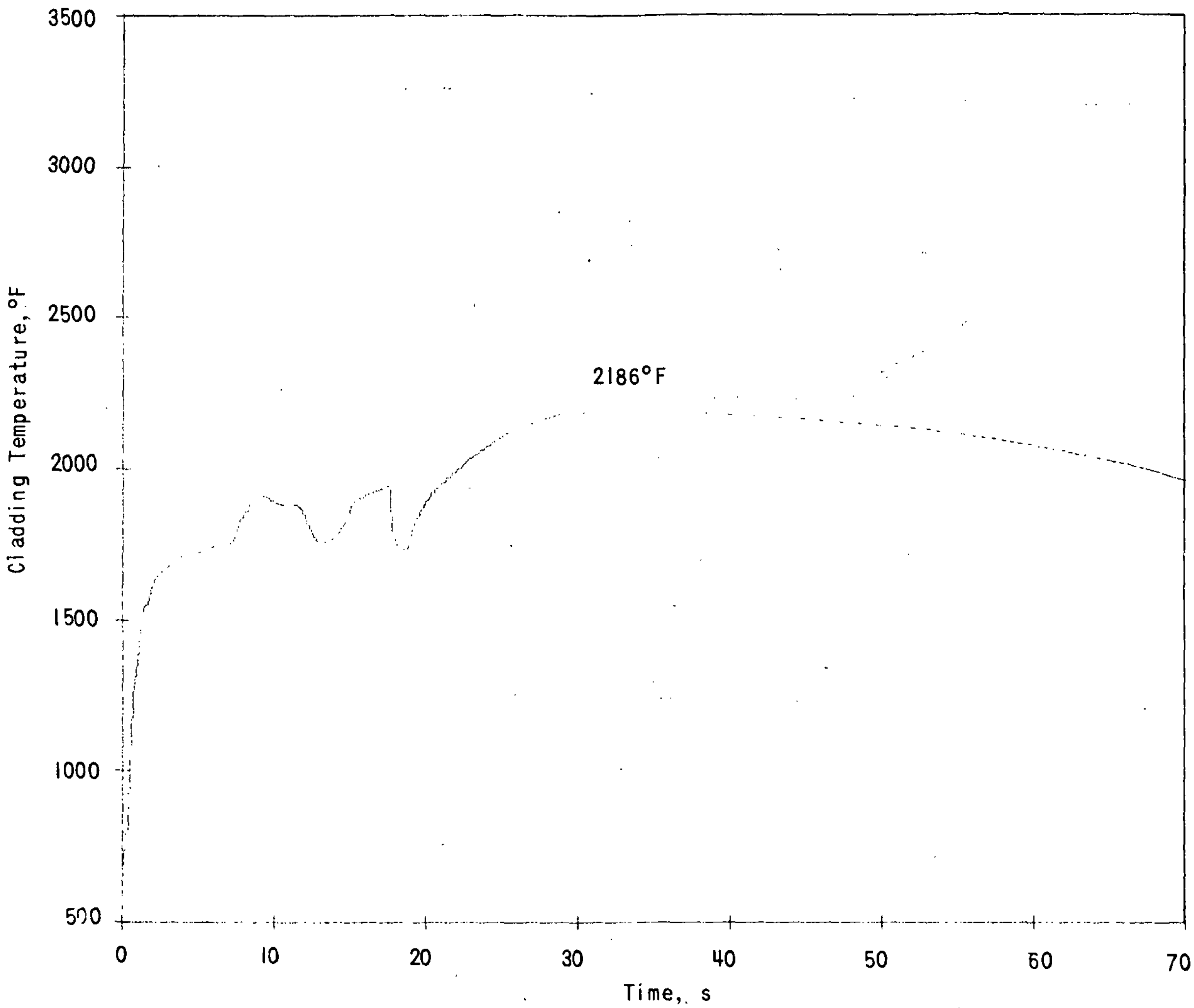
Figure 14.2-7



POST BLOWDOWN HOT SPOT HEAT TRANSFER COEFICIENT FOR 8.55 FT^2 SPLIT IN COLD LEG PIPE AT PUMP DISCHARGE

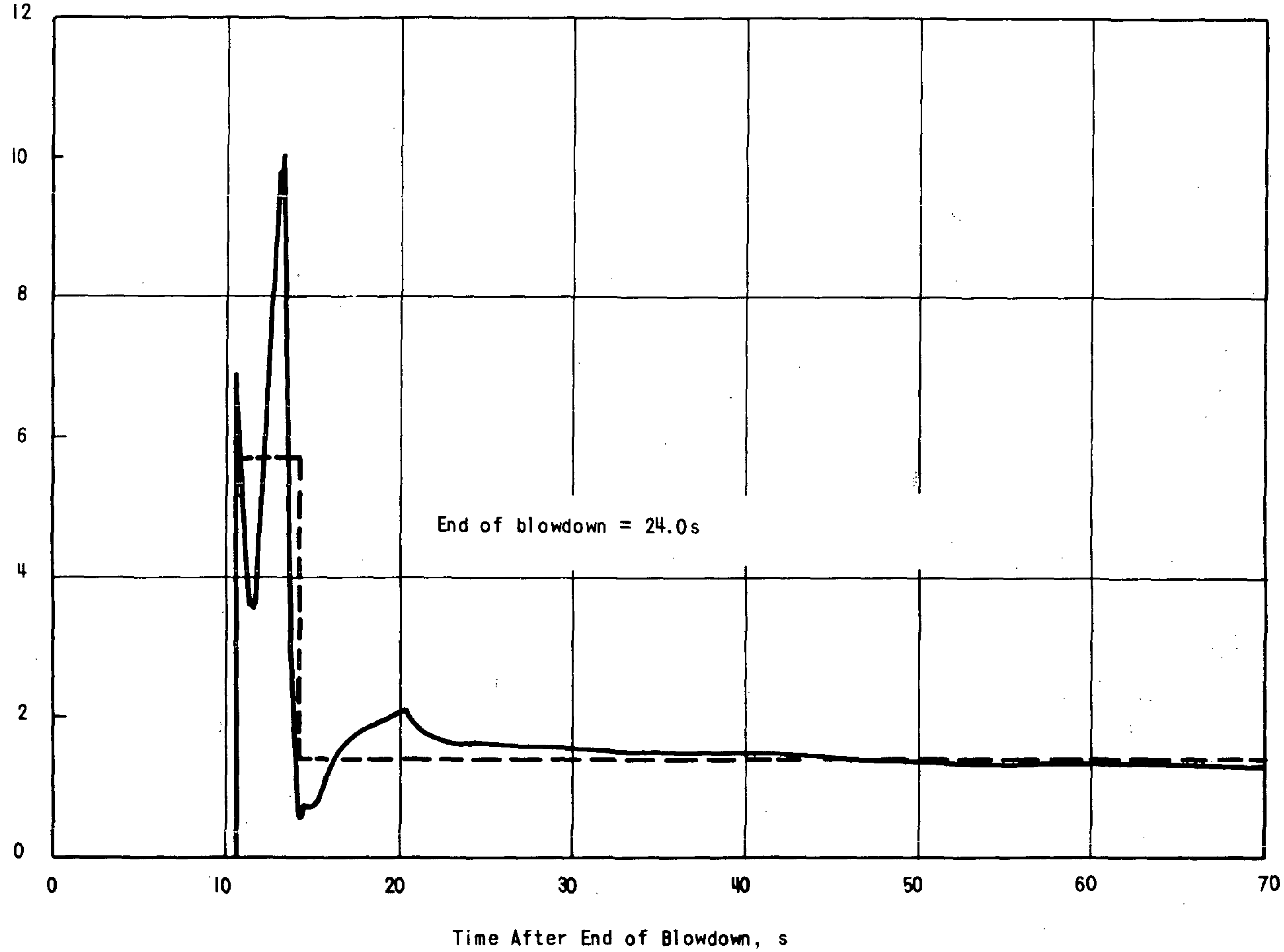
Figure 14.2-8

Docket 50-270 and -287
FSAR Supplement 13
January 29, 1973



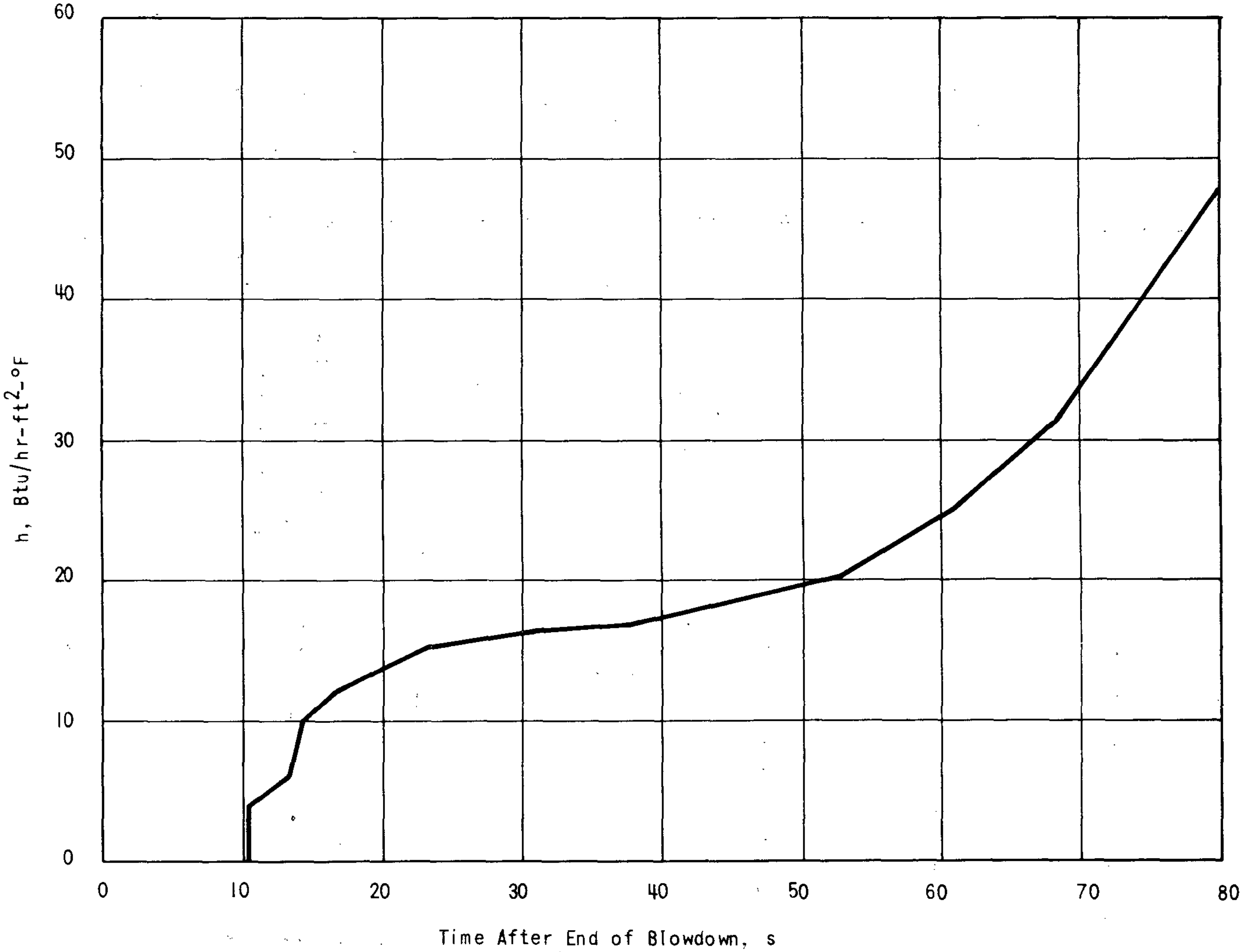
HOT SPOT CLADDING TEMPERATURE FOR 8.55 FT^2
SPLIT IN COLD LEG PIPE AT PUMP DISCHARGE

Figure 14.2-9



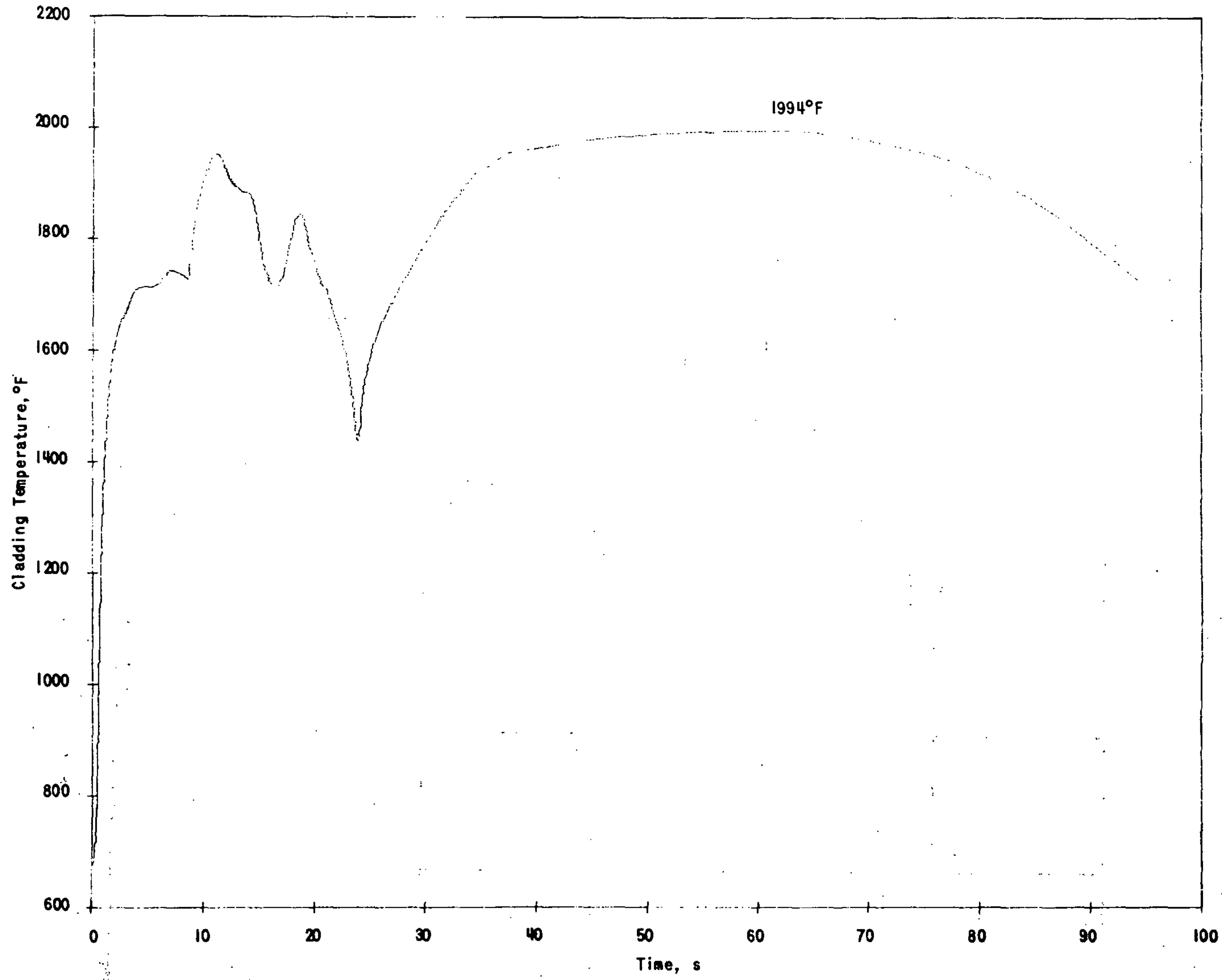
CORE FLOODING RATE FOR 5.13 FT^2 SPLIT
IN COLD LEG PIPE AT PUMP DISCHARGE

Figure 14.2-10



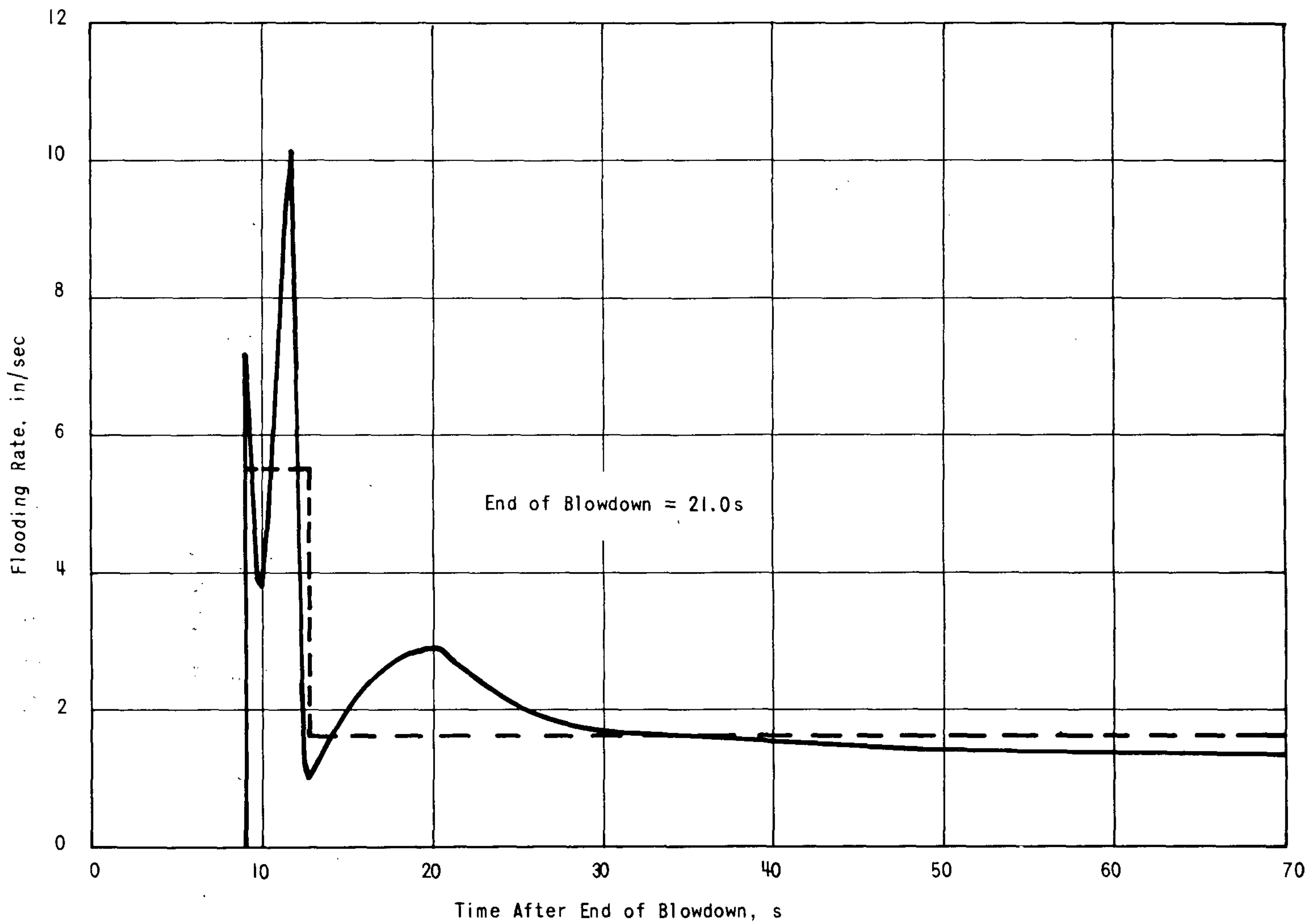
POST BLOWDOWN HOT SPOT HEAT TRANSFER COEFFICIENT FOR 5.13 FT^2 SPLIT IN COLD LEG PIPE AT PUMP DISCHARGE

Figure 14.2-11



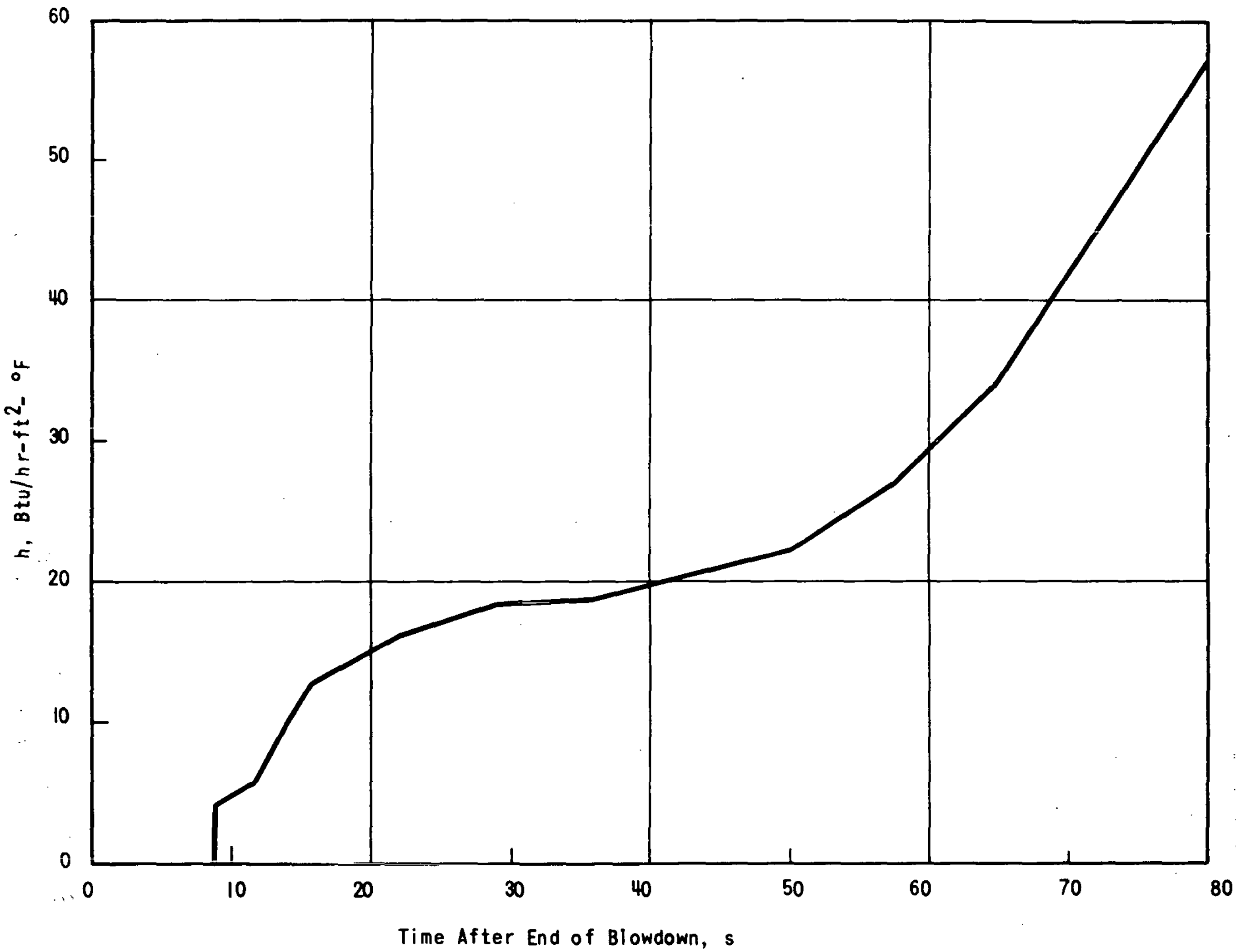
HOT SPOT CLADDING TEMPERATURE FOR 5.13 FT^2
SPLIT IN COLD LEG PIPE AT PUMP DISCHARGE

Figure 14.2-12



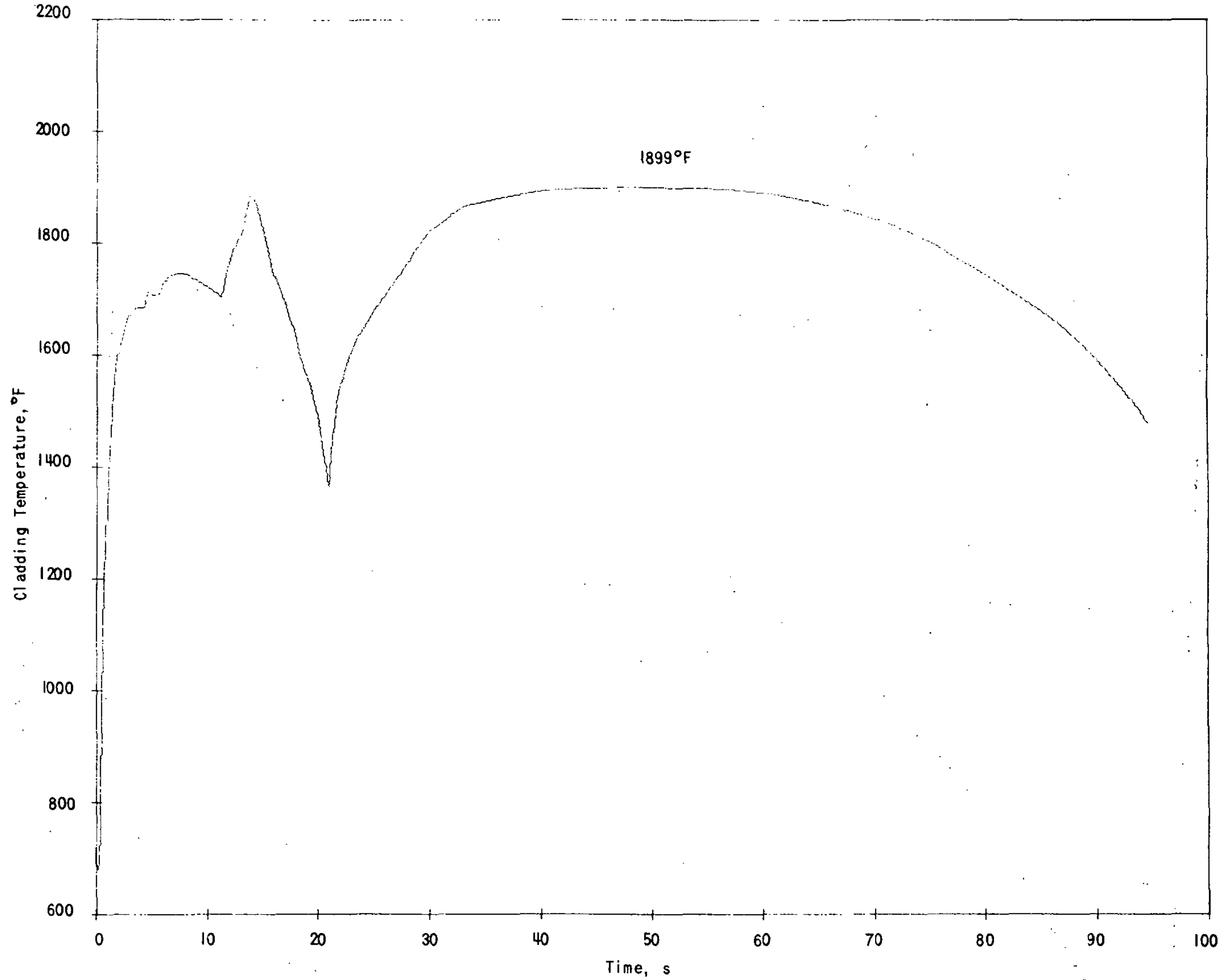
CORE FLOODING RATE FOR 8.55 FT^2 SPLIT
IN COLD LEG PIPE AT PUMP SUCTION

Figure 14.2-13



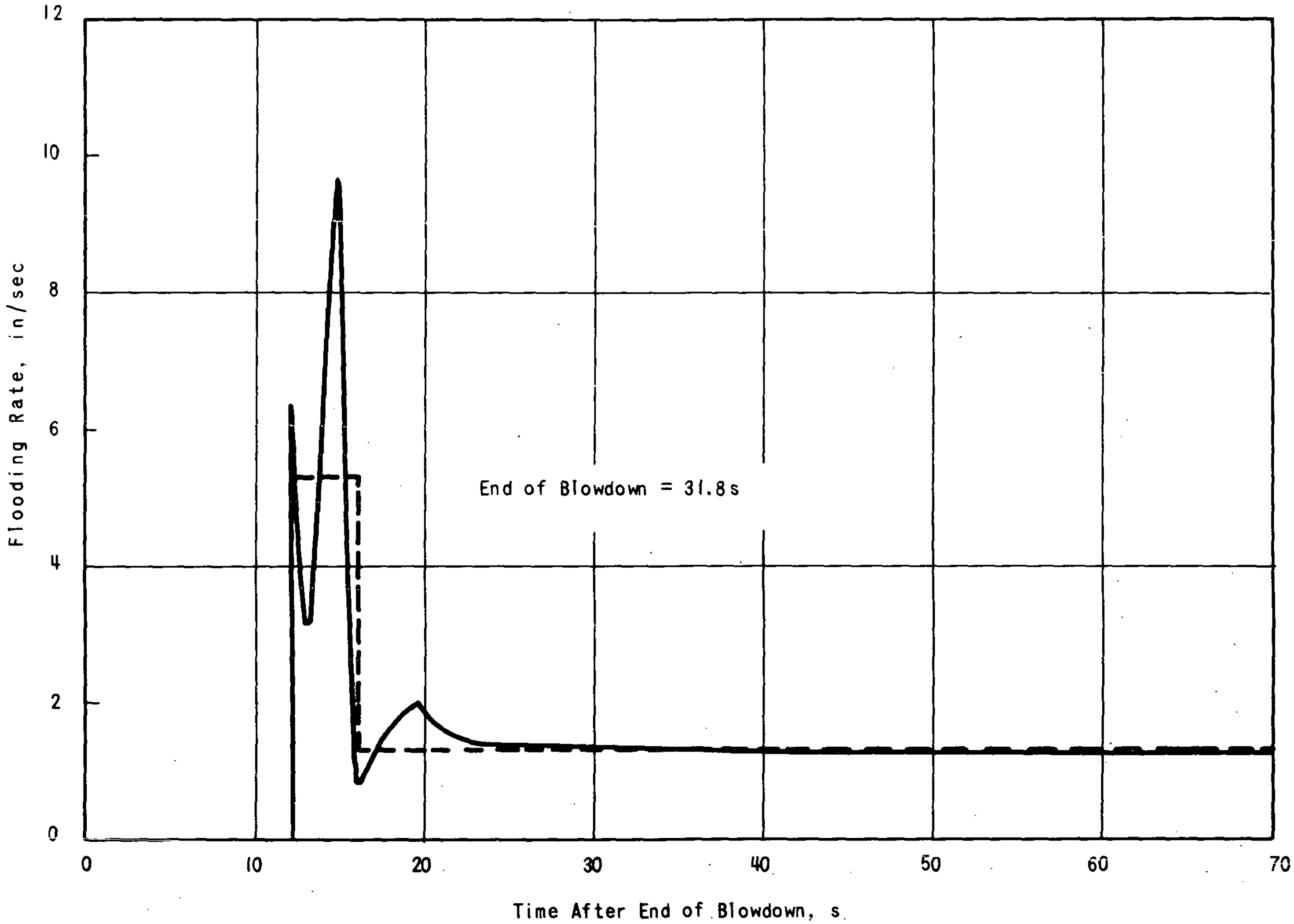
POST BLOWDOWN HOT SPOT HEAT TRANSFER
COEFFICIENT FOR 8.55 FT^2 SPLIT IN
COLD LEG PIPE AT PUMP SUCTION

Figure 14.2-14



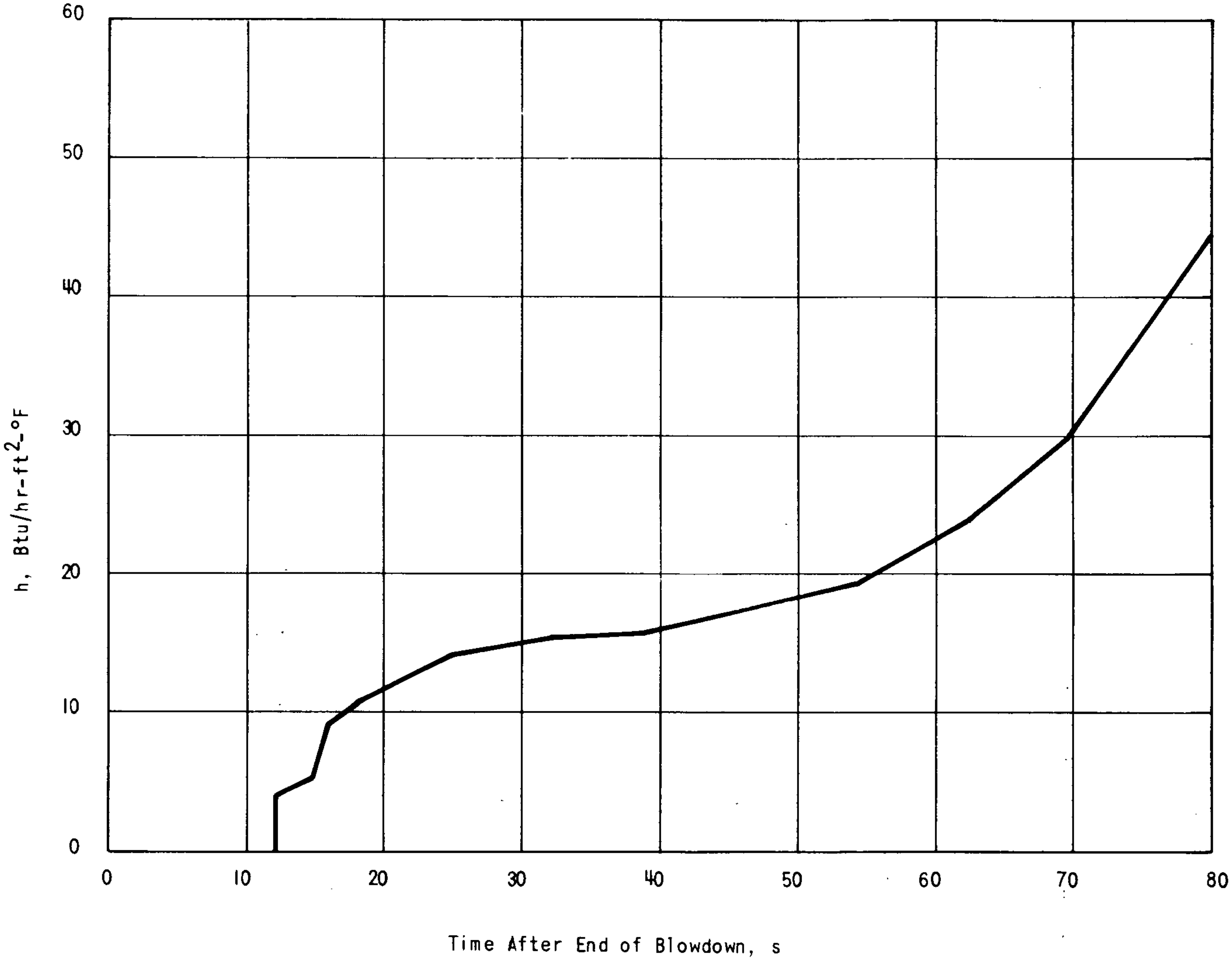
HOT SPOT CLADDING TEMPERATURE FOR 8.55 FT^2
SPLIT IN COLD LEG PIPE AT PUMP SUCTION

Figure 14.2-15



CORE FLOODING RATE FOR 3.0 FT^2 SPLIT
IN COLD LEG PIPE AT PUMP DISCHARGE

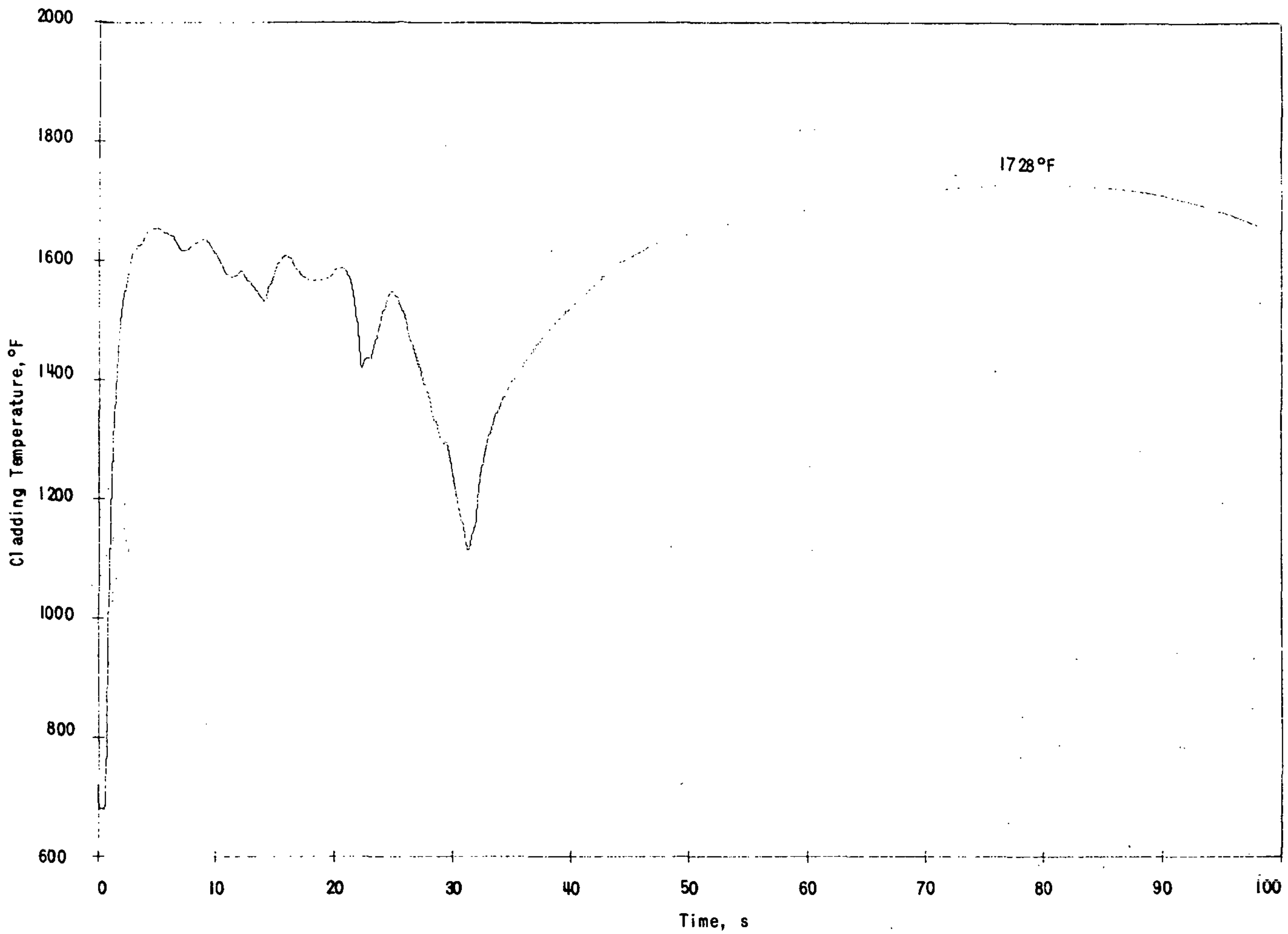
Figure 14.2-16



POST BLOWDOWN HOT SPOT HEAT TRANSFER COEFFICIENT FOR 3.0 FT^2 SPLIT IN COLD LEG PIPE AT PUMP DISCHARGE

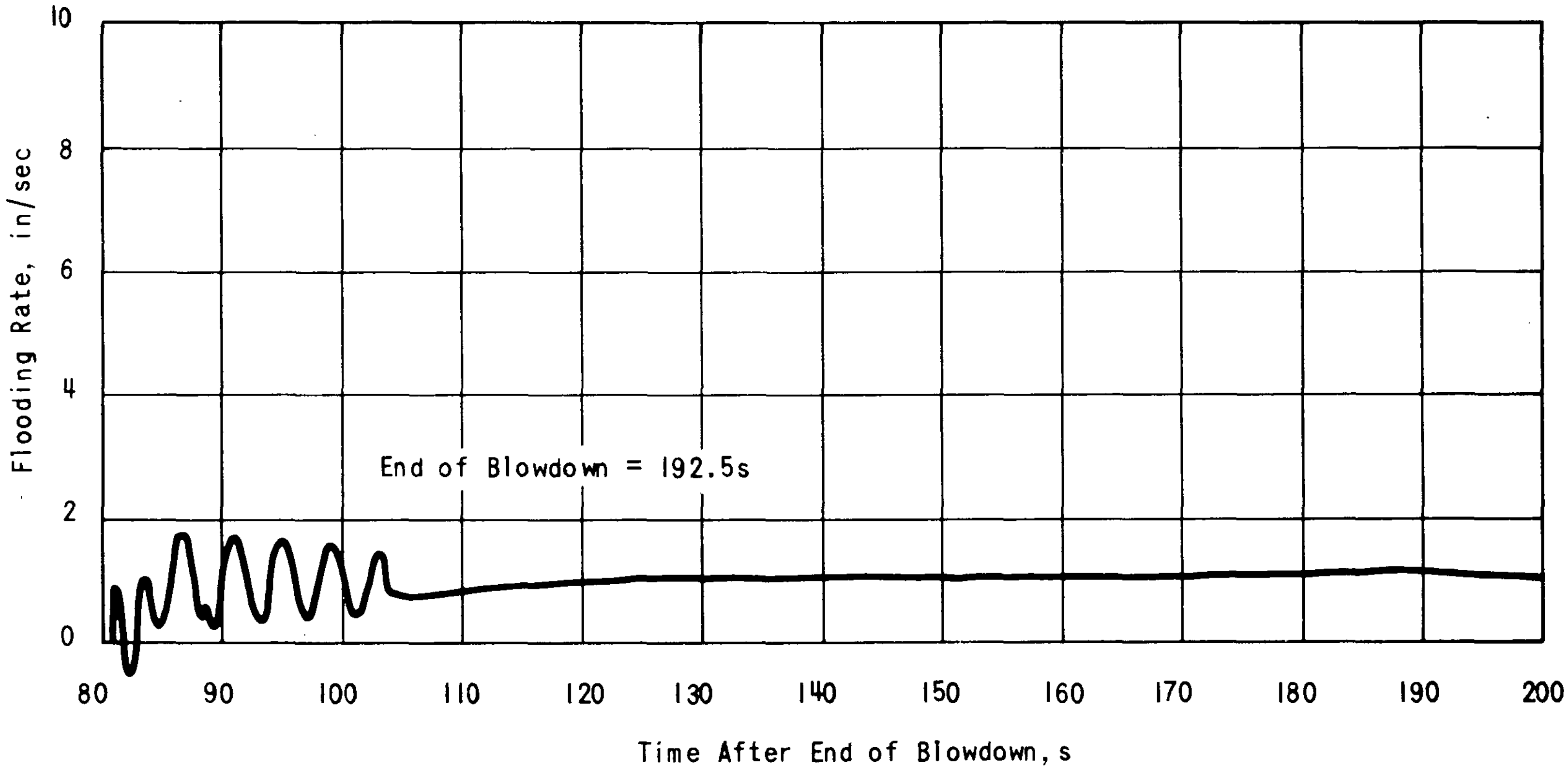
Figure 14.2-17

Docket 50-270 and -287
FSAR Supplement 13
January 29, 1973



HOT SPOT CLADDING TEMPERATURE FOR 3.0 FT^2
SPLIT IN COLD LEG PIPE AT PUMP DISCHARGE

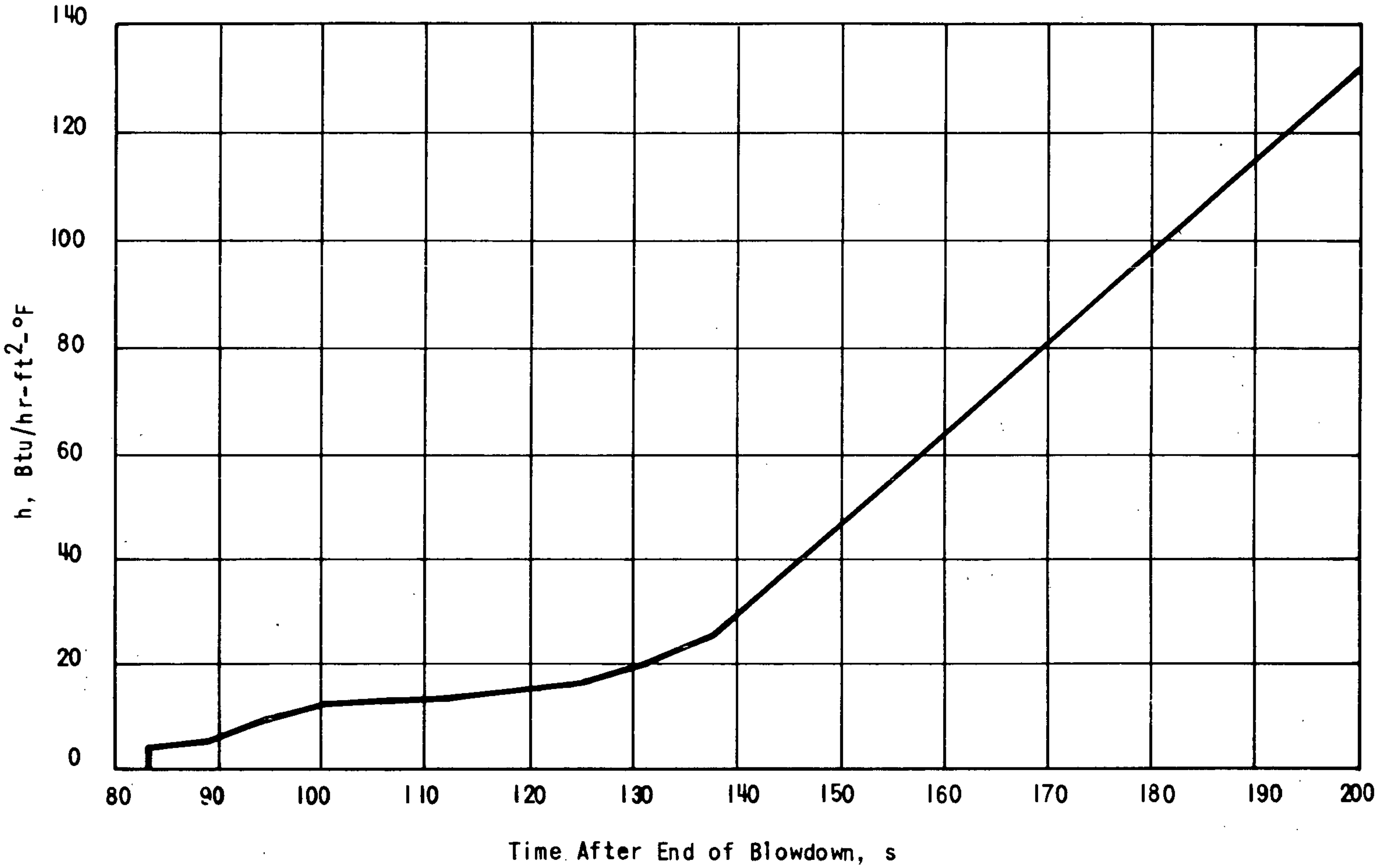
Figure 14.2-18



CORE FLOODING RATE FOR 0.5 FT^2 SPLIT
IN COLD LEG PIPE AT PUMP DISCHARGE

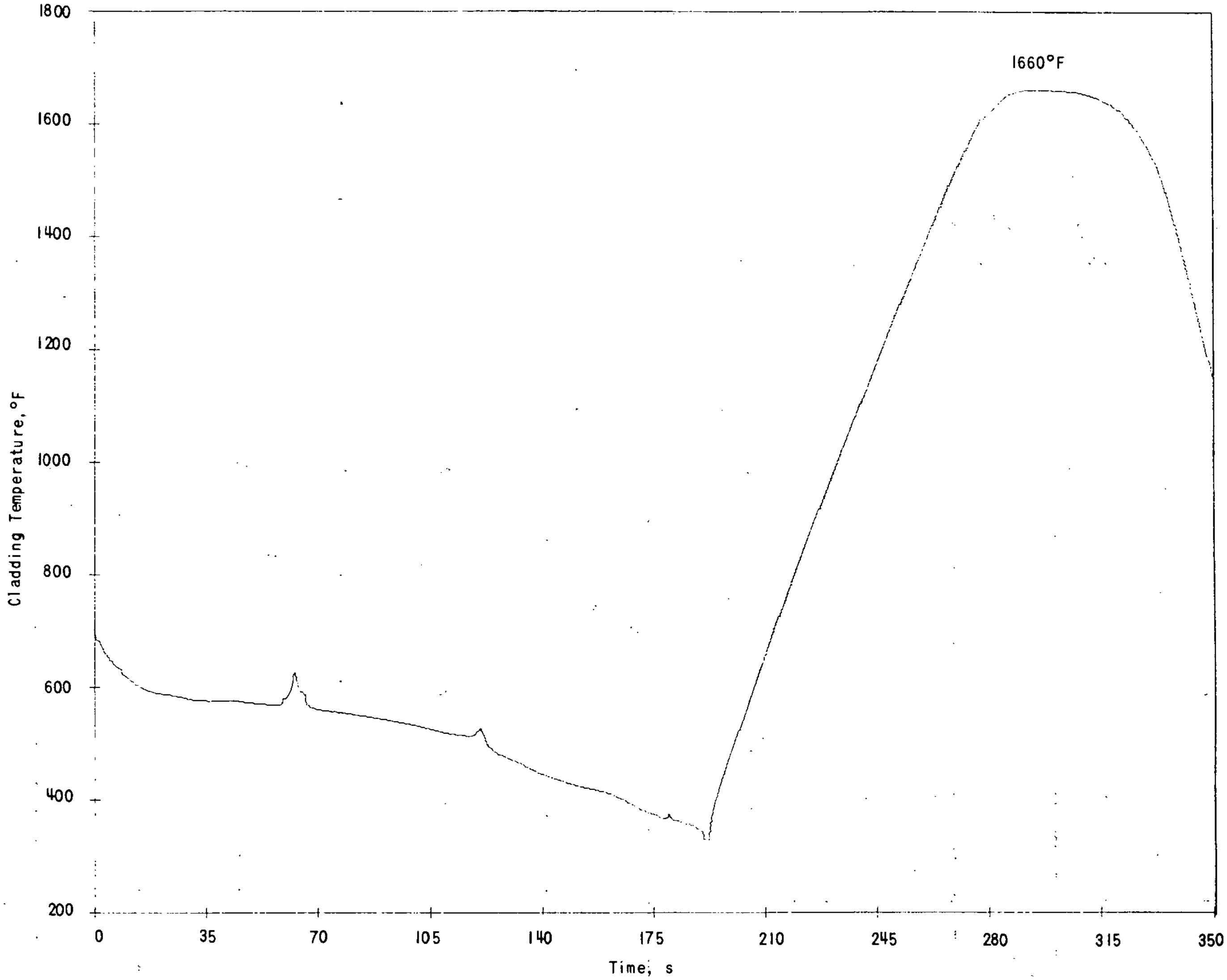
Figure 14.2-19

Docket 50-270 and -287
FSAR Supplement 13
January 29, 1973



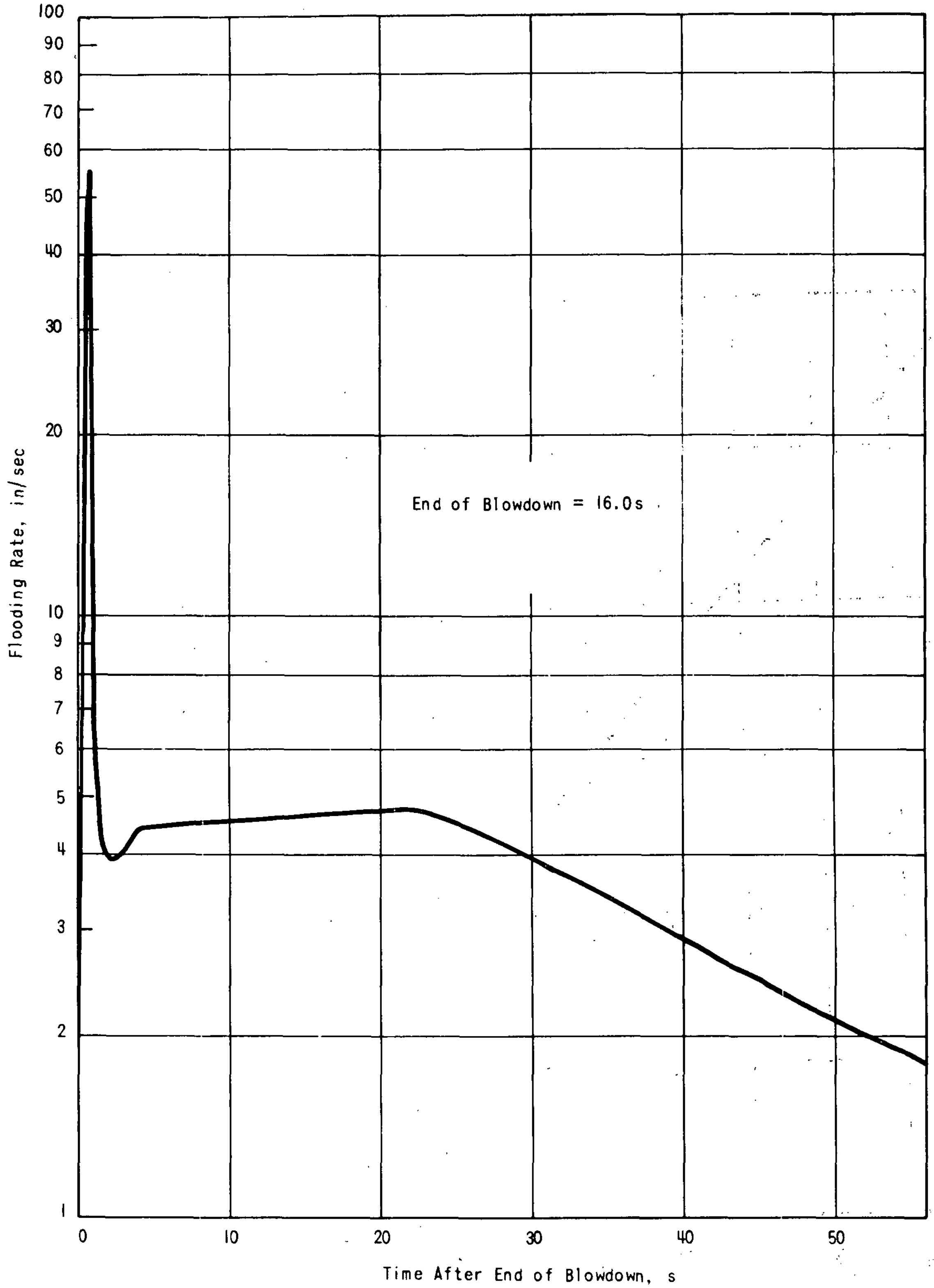
POST BLOWDOWN HOT SPOT HEAT TRANSFER COEFFICIENT FOR 0.5 FT^2 SPLIT IN COLD LEG PIPE AT PUMP DISCHARGE

Figure 14.2-20



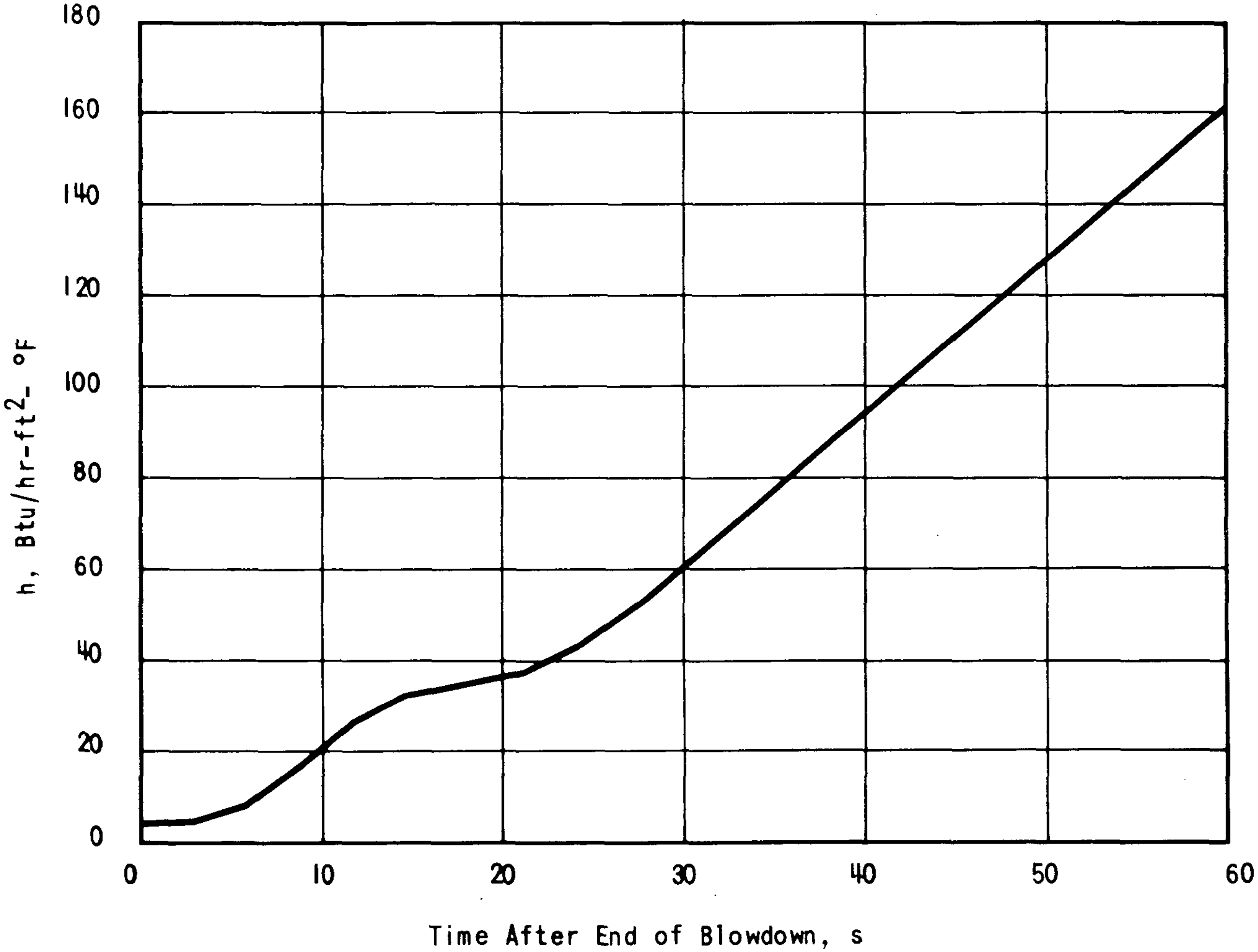
HOT SPOT CLADDING TEMPERATURE FOR 0.5 FT^2
SPLIT IN COLD LEG PIPE AT PUMP DISCHARGE

Figure 14.2-21



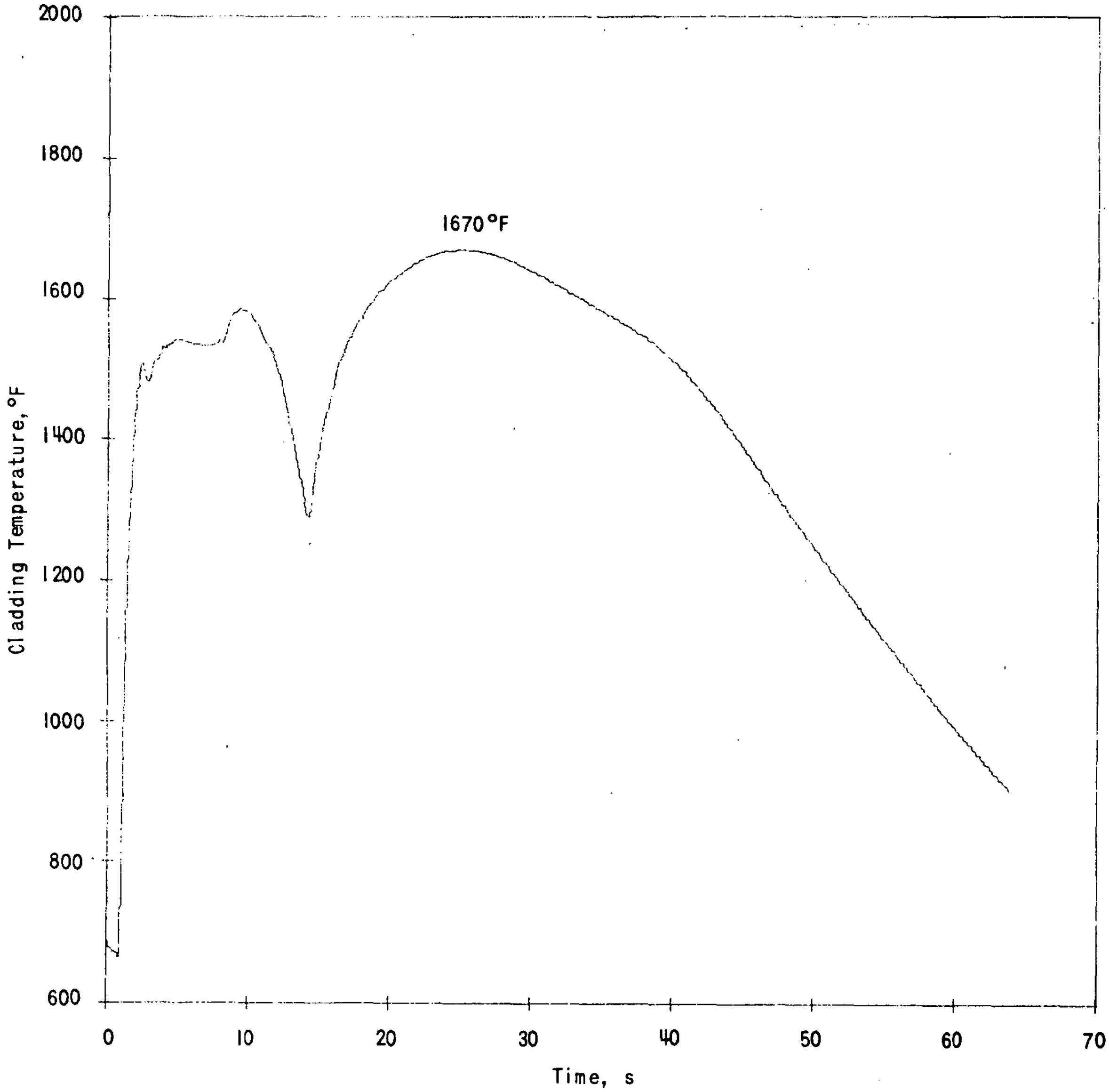
CORE FLOODING RATE FOR 14.1 FT
HOT LEG BREAK

Figure 14.2-22



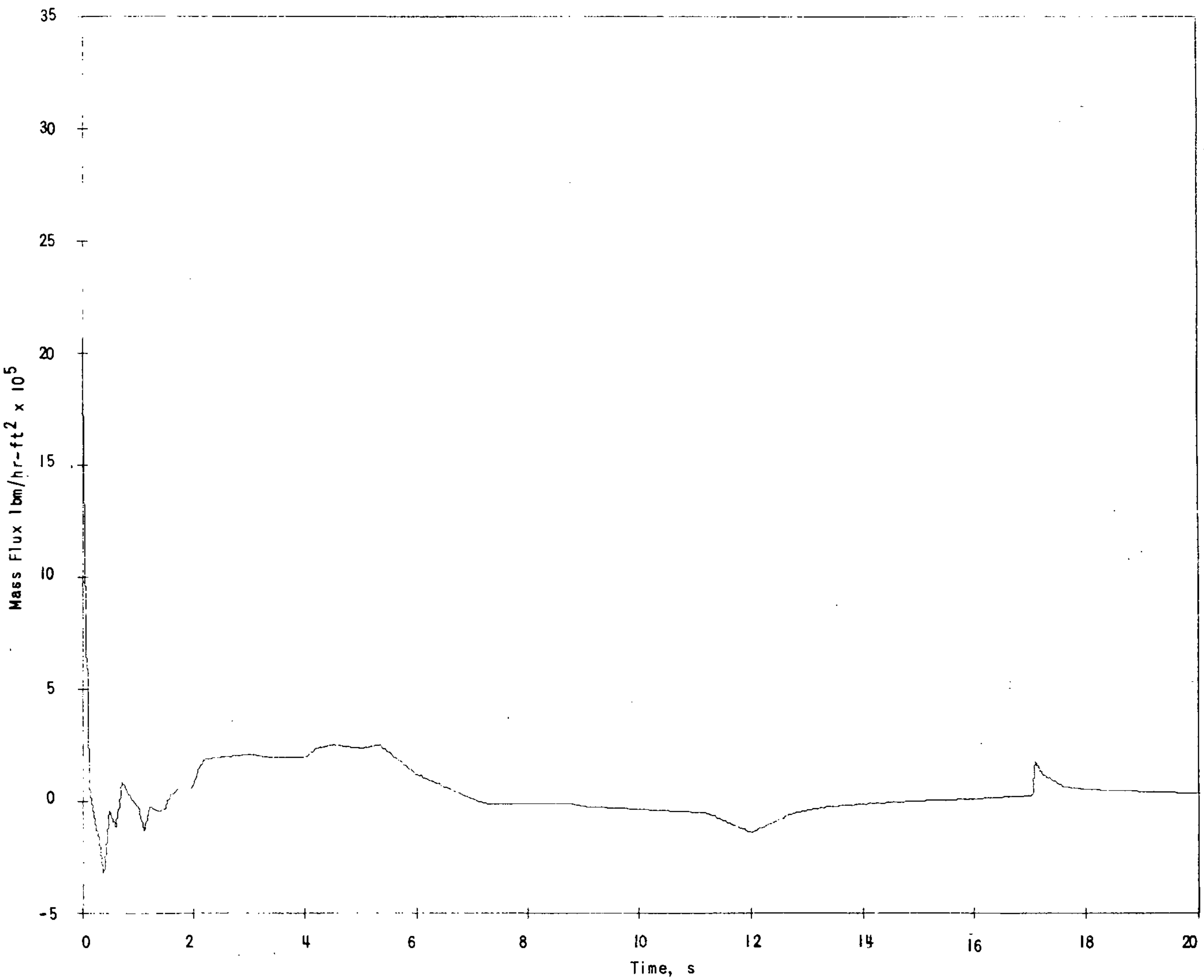
POST BLOWDOWN HOT SPOT HEAT TRANSFER COEFFICIENT FOR 14.1 FT^2 HOT LEG BREAK

Figure 14.2-23



HOT SPOT CLADDING TEMPERATURE FOR 14.1 FT^2 HOT LEG BREAK

Figure 14.2-24



SMOOTHED HOT SPOT MASS FLUX FOR 8.55 FT^2
SPLIT IN COLD LEG PIPE AT PUMP DISCHARGE
WITH A SYMMETRICAL POWER SHAPE

Figure 14.2-25

LIST OF EFFECTIVE PAGES

FSAR SUPPLEMENT 14

DUKE POWER COMPANY

OCONEE NUCLEAR STATION

UNIT 1

APPLICATION FOR LICENSES

Docket 50-269

FSAR SUPPLEMENT 14

LOSS OF COOLANT ACCIDENT
CORE FLOOD LINE BREAK

Submitted with FSAR Revision 26

January 29, 1973

INTRODUCTION

Supplement 14 to the Final Safety Analysis Report contains responses to several questions received in reference to the core flooding tank line break and, in particular, the questions received by telecon and discussed at the meeting held in Bethesda, Maryland on January 4, 1973. The information presented in this supplement is the same as that presented in a letter to Mr. A. Giambusso, Deputy Director for Reactor Projects, on January 15, 1973 except that the results of the stress analysis are now complete and are included.

Duke Power Company has installed a flow restrictor in the core flooding nozzle to restrict the magnitude of the blowdown and retain more water in the reactor vessel during the accident.

Part 1 provides an analysis of the core flooding line break with the flow restrictor in place.

Part 2 shows the effect of the device on the operation of the emergency core cooling system for both large and small breaks.

Part 3 provides a description and mechanical design of the device.

PART 1

ANALYSIS OF A CORE FLOODING LINE BREAK LOSS OF COOLANT ACCIDENT FOR THE OCONEE 1 REACTOR WITH INSERT IN CFT NOZZLE

LIST OF FIGURES

LIST OF FIGURES

(Continued)

1. INTRODUCTION

The scope of this analysis is a guillotine break of the core flood tank line between the reactor vessel nozzle and the first check valve. Flow out of the reactor vessel is limited to an effective area of 0.44 ft^2 due to a flow limiting insert.

Since the leak area is less than 0.5 ft^2, the B&W small leak evaluation model, BAW-10052[1], "Multinode Analysis of Small Breaks for B&W's 2568-MWt Nuclear Plants", is used. Consideration is given to three different axial power distributions. The following assumptions are made for conditions and system responses during the accident:

1. The reactor is operating at 102% of the steady-state power level of 2568-MWt.

2. A single failure is assumed in addition to the CFL break. The worst single failure results in an injection flow from only one high pressure injection pump and the second core flooding tank.

3. The leak occurs instantaneously, and a discharge coefficient of 1.0 is used for the entire analysis.

4. The reactor trips on low pressure at 2050 psig.

2. SUMMARY AND CONCLUSIONS

The maximum cladding temperature for this analysis is a function of the axial power shape. For the three axial power shapes analyzed, the maximum cladding temperature is 1172 F. A peak temperature of this magnitude or less presents no potential for either metal-water reaction or cladding swelling; therefore, the core geometry remains unchanged and amenable to cooling. This analysis indicates that adequate core cooling is maintained. The operator may initiate additional injection to refill the vessel in about one-half hour. Even without this additional long term cooling established however, all conditions of the AEC Interim Acceptance Criteria are met.

The peak temperatures for the three different axial power shapes analyzed in this report are as follows:

Elevation of Power Peak from Bottom of Core ft	Elevation of Peak Cladding Temperature from Bottom of Core ft	Peak Cladding Temperature F
5.5	5.5	663
7.8	11.4	666
10.6	11.4	1172

3. METHOD OF ANALYSIS

The method of analysis used to determine the cladding temperature response is the same as described in BAW-10052. This is consistent with the interim policy statement because, although the CFT line cross-sectional area is 0.72 ft^2, the flow from the reactor vessel is limited in the vessel nozzle by an insert which has a cross-sectional area of 0.44 ft^2.

The basic assumptions of the CRAFT[2] model, used for core hydrodynamics, are the same as those applied to the small breaks except that the noding scheme is somewhat different due to the nature of the break. The noding scheme and legend are shown in Figure 3-1.

A THETA-1B[3] model is used to determine the heat transfer in the flow controlled region. The 10 axial region model is necessary to describe in sufficient detail the three axial power shapes that are analyzed. The Quench model, as described in BAW-10052, is used when the heat transfer regime is not flow controlled.

When the flow in the core has decayed sufficiently, the core is in a relatively quiescent condition in which the lower portion is covered by a two-phase mixture, which is cooled by pool boiling, and the upper portion is steam cooled at a flow rate consistent with the boil-off rate in the lower portion This steam cooling results in reduced heat transfer coefficients and significant temperature transients. Therefore, it is very important to properly and conservatively determine the height of the two-phase mixture. Steam production as calculated by CRAFT and the Redfield-Murphy bubble rise model[4] is used to determine the two-phase mixture height. Conservatism is applied by limiting the void fraction in the inner vessel mixture to a maximum value of 0.5 for thermal analysis. The void fraction as calculated by CRAFT results in higher mixture levels in the inner vessel. By limiting the void fraction to 0.5, more of the core is uncovered, which results in less steam flow and more superheating in the upper region.

The CRAFT model was examined to determine its sensitivity to various noding schemes. Figure 3-2 shows the results of three noding approaches in terms of inner vessel liquid volume and confirms our conservatism in using a two node inner vessel model because it eliminates the "pancaking"effect.

4. RESULTS OF ANALYSIS

The CRAFT run was made with a single core flow path and is applicable to any of the three axial power shapes examined. Figures 4-1 and 4-2 show the core pressure and power history for this accident. Figure 4-3 depicts the inner vessel fluid volumes as used for the heat-up calculations. The vent valves are located above the top of the active region; therefore, the vent valve flow shown in Figure 4-4 will appear to be inconsistent with the mixture height shown in Figure 4-3. In section 3, it was explained that a conservative limit was placed on the core void fraction. This effect is visible here because CRAFT shows a mixture height in the area of the vent valves while the mixture height which is used in the thermal analysis is much lower. Mixture flowing out of the vent valves is conservative because it removes liquid water from the core region.

The leak flow as a function of time is shown in Figure 4-5. The core and downcomer water heights are shown in Figure 4-6, while Figure 4-7 shows that calculated fluid velocities between the core and downcomer are not of sufficient magnitude to consider entrainment of the CFT water.

The 10 axial region THETA model was used until 300 seconds for the three axial profiles shown in Figures 4-8, 4-18, and 4-28. The Quench code was then used to analyze the cladding thermal response for the remainder of the transient.

The center-peaking power shape used in analyzing Case 1 is shown in Figure 4-8. The upper portion of the core is uncovered during the transient; but the low power obtained in this region does not result in significant cladding temperature increases. Therefore, the center region of the core, axial level 5, remains the hottest portion of the core with a peak cladding temperature of 663 F. Figures 4-9 through 4-11 show data related to axial level 5 and Figures 4-12 through 4-14 and 4-15 through 4-17 are relevant to axial levels 9 and 10 respectively.

Outlet power peaks are important to this analysis because higher power regions will be uncovered and the amount of superheat in the cooling steam will be higher. Case 2, Figure 4-18, is typical of an outlet peak experienced in the core. The maximum power peak occurs in axial level 7. It is never uncovered by the two-phase mixture, and the cladding temperature remains low. Figures 4-19 through 4-21 provide information on this level. Axial level 9 is uncovered for only a short period of time between 740 and 770 seconds and does not undergo a large cladding temperature rise. Figures 4-22 through 4-24 apply to this level. The highest peak cladding temperature is achieved at level 10 which is uncovered from approximately 1300 seconds to 2000 seconds and reaches a peak cladding temperature of 666 F. Information for this level is shown in Figures 4-25 through 4-27. The cladding temperature is declining slightly at the end of the analysis.

To see the effect of an outlet peak, Case 3 which is shown on Figure 4-28 was chosen for examination. This power shape represents one of the most adverse tilts toward the exit of the core. This shape is used for design purposes and while it is not an expected or normal operating condition, this shape is allowable for operation for the last few days of core life. In the heatup calculation, the power peak occurs at axial level 9. The temperature at the location of peak power, level 9, does rise as it is uncovered, but the recovering of the level keeps this from being the worst location. Figures 4-29 through 4-31 provide information at this location. The peak cladding temperature, 1172 F, occurs at level 10 because of the extended steam cooling from 1300 seconds on. The cladding temperature, Figure 4-34, is high because of the low heat transfer coefficient, Figure 4-33, and the degree of superheat, Figure 4-32. Both effects are caused by the shift of power to the upper regions of the core.

5. HIGH PRESSURE INJECTION FLOW

The analysis of the core flooding line break shows that the core can be cooled using only one core flooding tank and one high pressure injection (HPI) pump. The HPI flow used in the analysis was based on test results from Oconee 1. A least-squares regression analysis of the data was performed which showed a flow of 352 gpm and 353 gpm in each of two strings at a RCS pressure of 1500 psig. Considering an instrumentation error of ± 1% and the relative error in the regression fit, the flow at 1500 psig was reduced to 340 gpm for use in the analysis. Similarly at 600 psia, the least-squares fit resulted in a flow of 457 gpm which was reduced to 440 gpm for use in the analysis. Using these points, together with the tested pump head capacity curve, the high pressure injection over the full pressure range was established.

6. LONG-TERM ECCS OPERATION

The preceding analyses were based on ECCS capability from one HPI pump and one core flooding tank. This condition assumed that the active LPI pump was lined up to pump to the core flood line that had the break and the other LPI pump was inoperative by the criteria of a single active failure.

Increased long-term safety margin can be obtained by operator action to initiate low pressure injection through the unbroken core flood line. This
27. action can easily be taken within 15 minutes after the CFT line break. The operator will open control room operated cross connect valves at the LPI pump discharge and check flow indicators in each of LPI lines to determine that some flow is going through each line. Equalization of flow in the lines can be accomplished from the control room by positioning of control valves in each LPI line. When flow is equalized through each line, the LPI flow into the reactor vessel will be at least 1500 gpm with one pump operating and 3000 gpm if both pumps are operating.

The Oconee station operating procedures will be changed as follows:

1. Prior to switching suction on the ECCS and RB spray pumps from the BWST to the RB sump or before shutting off all HPI pumps, check LPI flow indication and LPI pump operation to assure flow into the reactor vessel. This requires flow indication in each line since it is not known which line has the break.

2. If only one pump is running, the operator should take the following actions.

   a. Attempt to start idle LPI pump. Failure to start may be ES actuation failure. Operator can operate valves and start pumps by remote manual control from control room.

   b. If pump (LP-P1C) is available, place in operation on the LPI string where pump is not running by opening valves in suction and discharge crossover lines and starting pump LP-P1C from the control room. Observe flow indication in the LPI line. This action produces 3000 gpm through each LPI line.

   c. If operator cannot start either of the two LPI pumps (steps a and b), perform the following steps to achieve flow into the reactor vessel from the one active pump:

      Open discharge crossover valves to get LPI flow into each of the LPI lines.

      Monitor LPI flow indication to assure flow through each line.

Adjust the throttle valve in each LPI line until a flow balance is achieved. This will give approximately 1500 gpm through each line.

28. All valves also have local handwheels that can be manually actuated. Procedures will provide for manual operation of these valves in the event they cannot be operated from the control room. These valves will also be manually cycled during refueling periods to give assurance that long-term emergency core cooling can be established in a timely manner. Shift supervisors will ensure that access to the valve handwheels is not impaired by other plant activities.

7. COMPARISON OF EVALUATION MODEL WITH APPENDIX A, PART 4 OF THE AEC INTERIM ACCEPTANCE CRITERIA FOR EMERGENCY CORE COOLING SYSTEMS

Although Appendix A, Part 4 is strictly appropriate only for breaks larger than 0.5 ft^2, a check list comparison of that evaluation model to the one used in this report may be of convenience to the reader. Evaluation and explanations are provided on a point by point basis with a subdivision consistent with Appendix A, Part 4.

The first paragraph of Appendix A, Part 4 lists several reports written by B&W which document techniques to be applied in the large break evaluation model. These reports are appropriate as follows:

1. CRAFT - This report and the code described are used for the CFT line break.
2. REFLOOD - This code is used for the purpose of calculating the refilling of a vessel once that vessel has reached end of blowdown. As that situation does not occur for the CFT line break, the code and its report do not apply and are not used.
3. THETA 1-B - This report and the code described are used for the CFT line break.
4. BAW-10034 - This report is written for large breaks.

Appendix A, Part 4 goes on to list specific instructions for the large break evaluation model.

1.1 Core and System Noding

1.1.1 Only one core node has been used in the CFT line break analysis.

1.1.2 The Theta model used during the flow controlled heat transfer regime had 6 fuel nodes, 2 clad nodes, and 10 axial levels. After the flow controlled heat transfer regime, after 300 seconds, the Quench code is used. This code has 1 fuel and 1 clad node and must be applied individually at separate axial levels.

1.2 Pump Model

This model is the same used in BAW-10052 and is discussed in that report. It is different from the model used in large break analysis though both models are consistent with the Appendix A, Part 4 guideline.

1.3 Break Characteristics

This statement does not apply to a specific break like the CFT line break.

1.4 Discharge Coefficient

As suggested by Appendix A, Part 4, a discharge coefficient of 1.0 has been used.

1.5 Decay Heat

The decay heat curve suggested in Appendix A, Part 4 was used in the CFT line break analysis.

1.6 Time to Departure from Nucleate Boiling (DNB)

This was done as suggested in the large break evaluation model.

1.7 Film Boiling Heat Transfer

This was done as suggested by the large break evaluation model except that for pool film boiling used in the QUENCH code, the Morgan correlation was used.

1.8 Metal-Water Reaction Rate

This was done as suggested by the large break evaluation model. Temperatures for this accident, however, prohibit any significant metal-water reaction.

1.9 Core Flow Rate

This was done as suggested by the large break evaluation model while flow was controlling the heat transfer.

1.10 Enthalpy and Pressure

This was done as suggested by the large break evaluation model.

1.11 Core Flooding Tank Bypass

As downcomer steam flows were insufficient to cause entrainment of core flooding tank water, this bypass assumption was not imposed on the CFT line break analysis.

Appendix A, Part 4 then proceeds to describe the evaluation model for the reflood portion of the large break. As there is no classic reflood portion for the CFT line break, this section does not apply to the analysis.

LIST OF REFERENCES

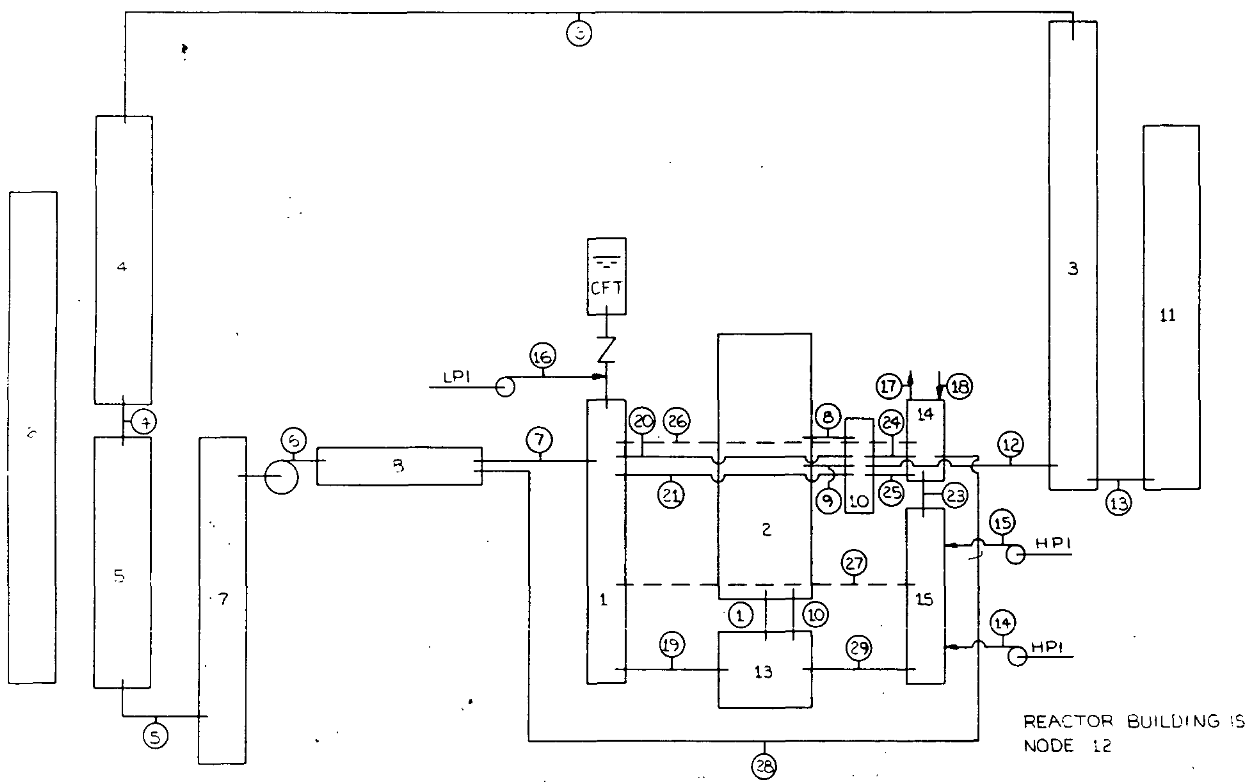
1. C.E. Parks, B.M. Dunn, and R.C. Jones, Multinode Analysis of Small Breaks for B&W's 2568-MWt Nuclear Plants, BAW-10052, Babcock and Wilcox, Lynchburg, Virginia, September, 1972.

2. CRAFT - Description of Model for Equilibrium LOCA Analysis Program, BAW-10030, Babcock and Wilcox, Lynchburg, Virginia, October, 1971.

3. C.J. Hocever and T.W. Wineinger, THETA 1-B - A Computer Code for Nuclear Reactor Core Thermal Analysis, IN - 1445, Idaho Nuclear Corp., February, 1971.

4. J.A. Redfield and J.H. Murphy, Void Fraction and Residual Water Predictions During Loss of Coolant, WAPD-T-2155, Westinghouse, BAPL, September 1968.

FIGURE 3-1 CRAFT EVALUATION MODEL



LEGEND

NODE NO.	IDENTIFICATION	PATH NO.	IDENTIFICATION
1	DOWNCOMER	1	CORE
2	CORE & UPPER HEAD	3	HOT LEG PIPING
3	HOT LEG PIPING	4	SG TUBES
4	SG	5	COLDLEG PIPING
5	SG	6	PUMP
6	SG(SECONDARY SIDE)	7	COLD LEG PIPING
7	COLD LEG PIPING	8	PLENUM TO OUTLET NOZZLE
8	COLD LEG PIPING	9	PLENUM TO OUTLET NOZZLE
10	OUTLET NOZZLE	10	CORE BYPASS
11	PRESSURIZER	12	HOT LEG PIPING
12	CONTAINMENT	13	PRESSURIZER SURGE LINE
13	LOWER HEAD	14	HIGH PRESSURE INJECTION
14	DOWNCOMER	15	HIGH PRESSURE INJECTION
15	DOWNCOMER	16	LOW PRESSURE INJECTION
		17	LEAK PATH
		18	RETURN PATH
		19	DOWNCOMER
		20	VENT VALVES
		21	LEAKAGE
		23	DOWNCOMER
		24	VENT VALVES
		25	LEAKAGE
		26	CROSS FLOW-DOWNCOMER
		27	CROSS FLOW-DOWNCOMER
		28	COLD LEG PIPING
		29	DOWNCOMER

FIGURE 3-2 SENSITIVITY OF INNER VESSEL LIQUID VOLUME TO NODING

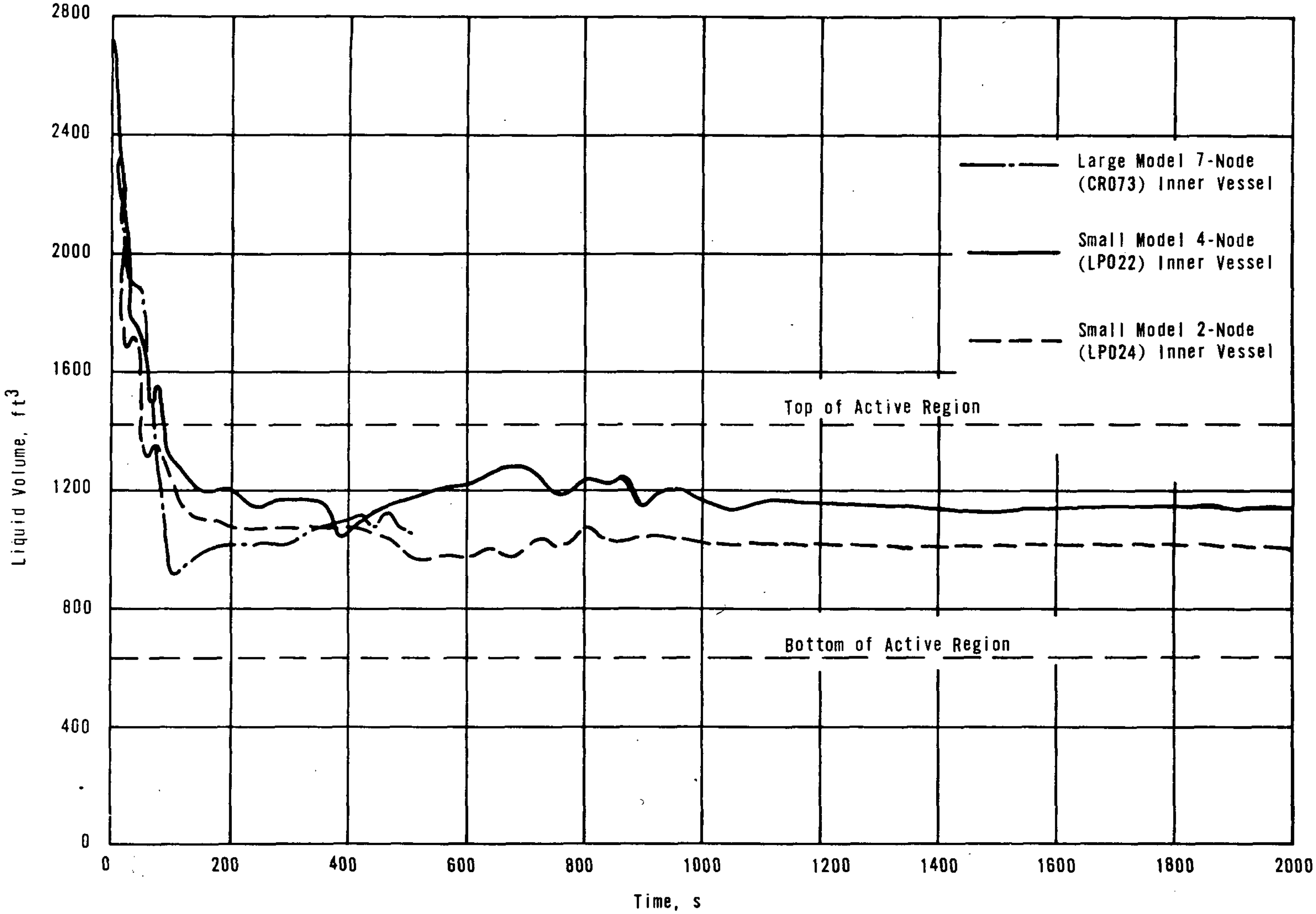


FIGURE 4-1 CORE PRESSURE VERSUS TIME

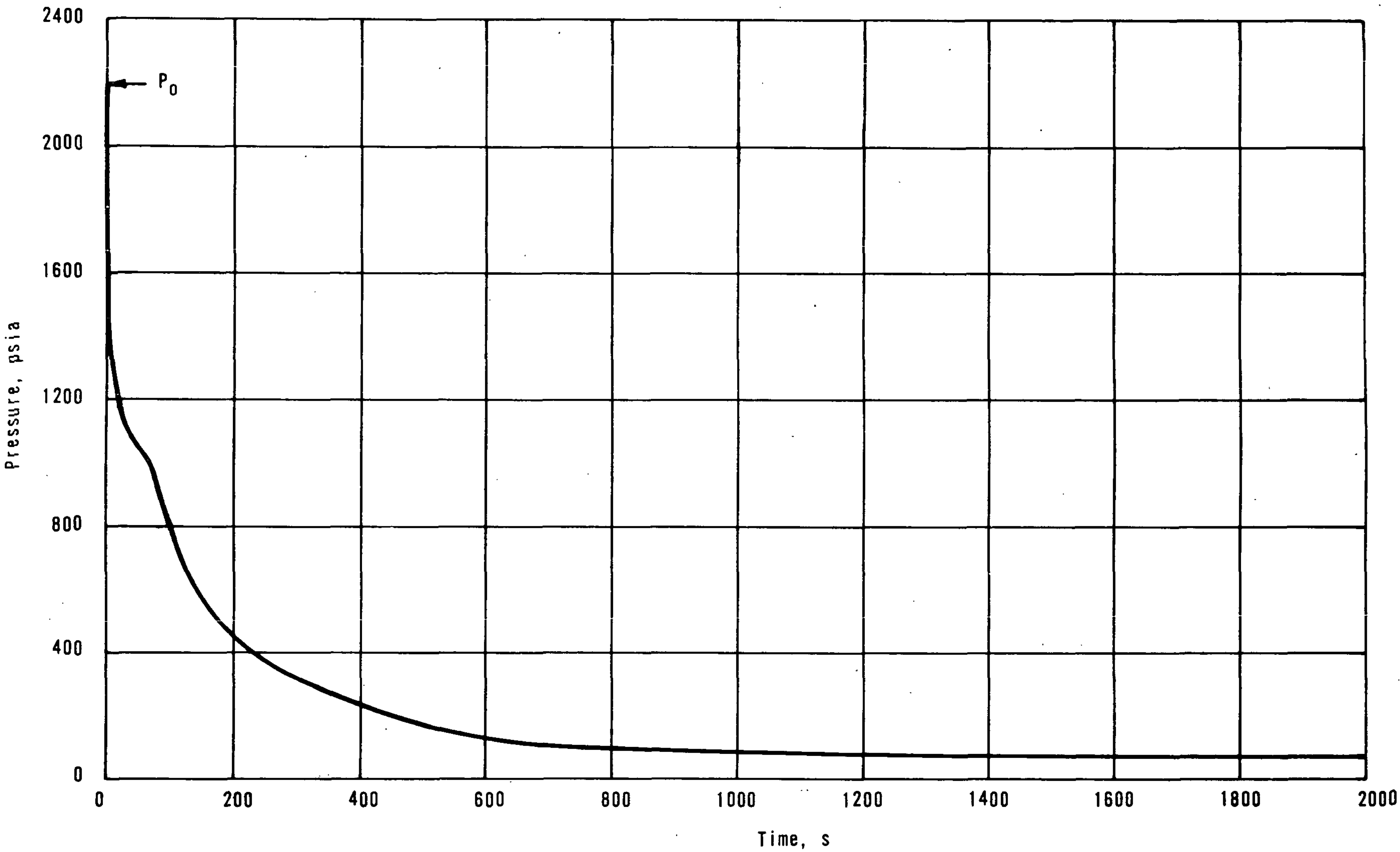
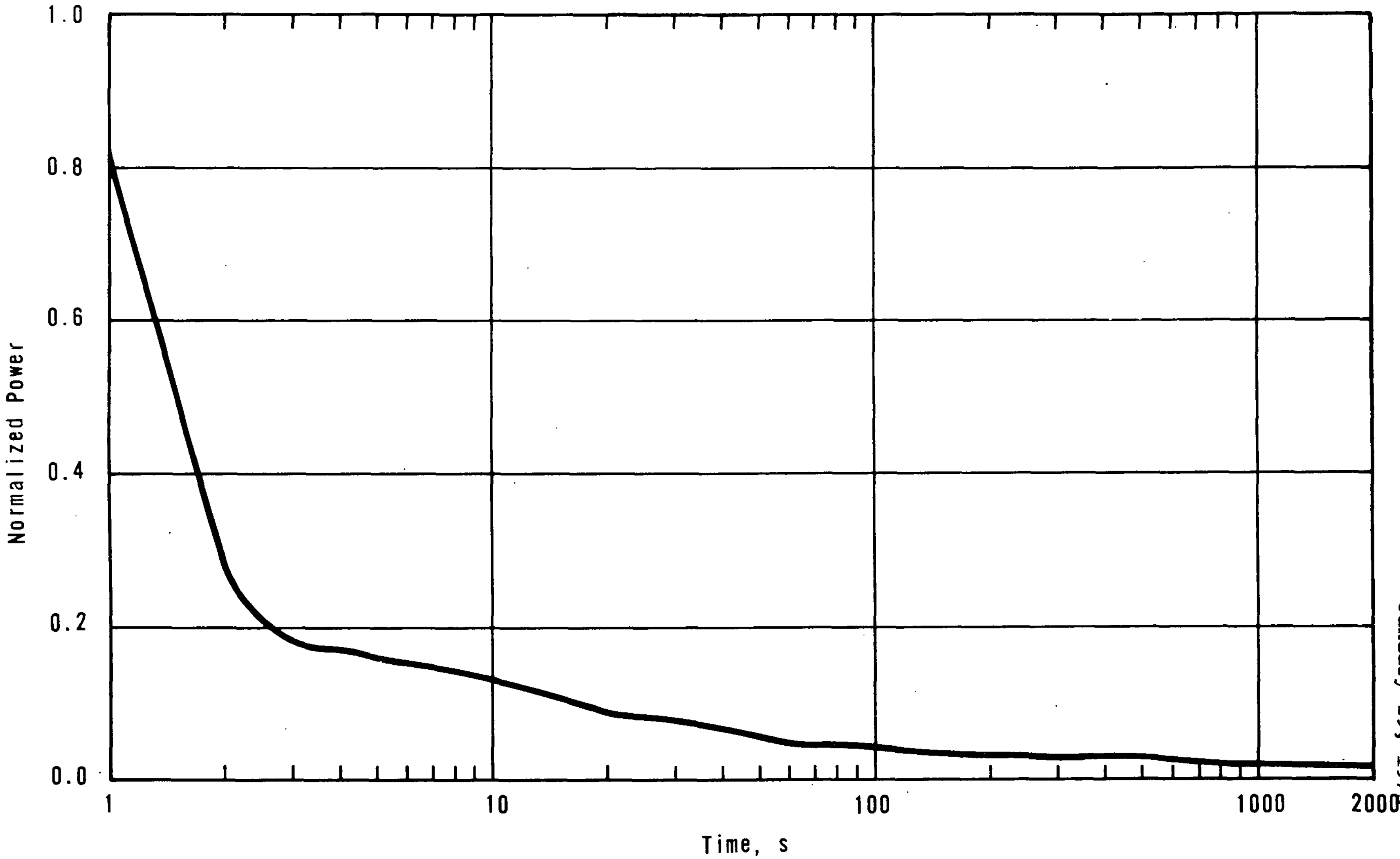


FIGURE 4-2 CORE POWER VERSUS TIME



Docket 50-269
FSAR Supplement 14
January 29, 1973

FIGURE 4-3 INNER VESSEL FLUID VOLUMES VERSUS TIME

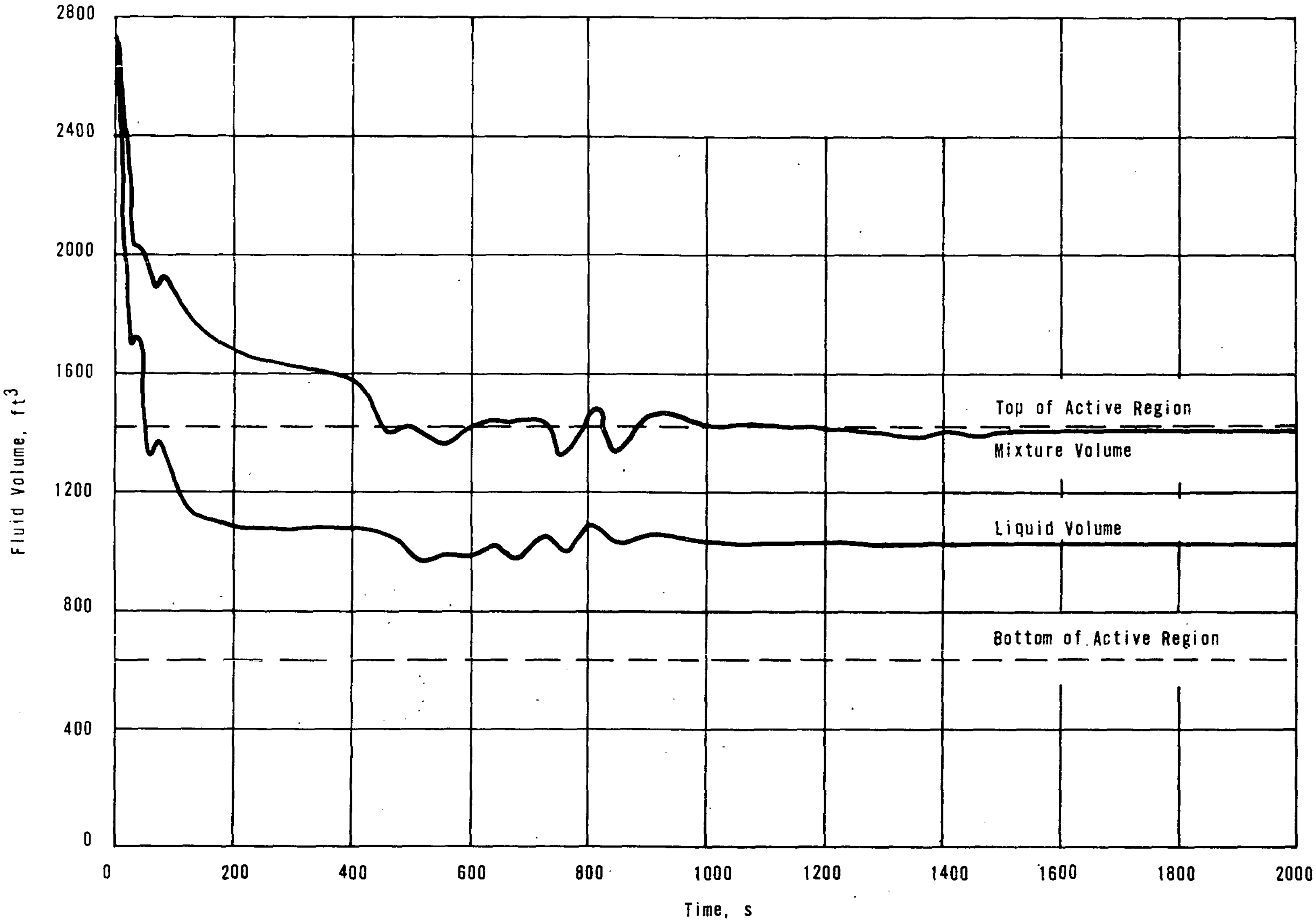


FIGURE 4-4 VENT VALVE FLOW VERSUS TIME

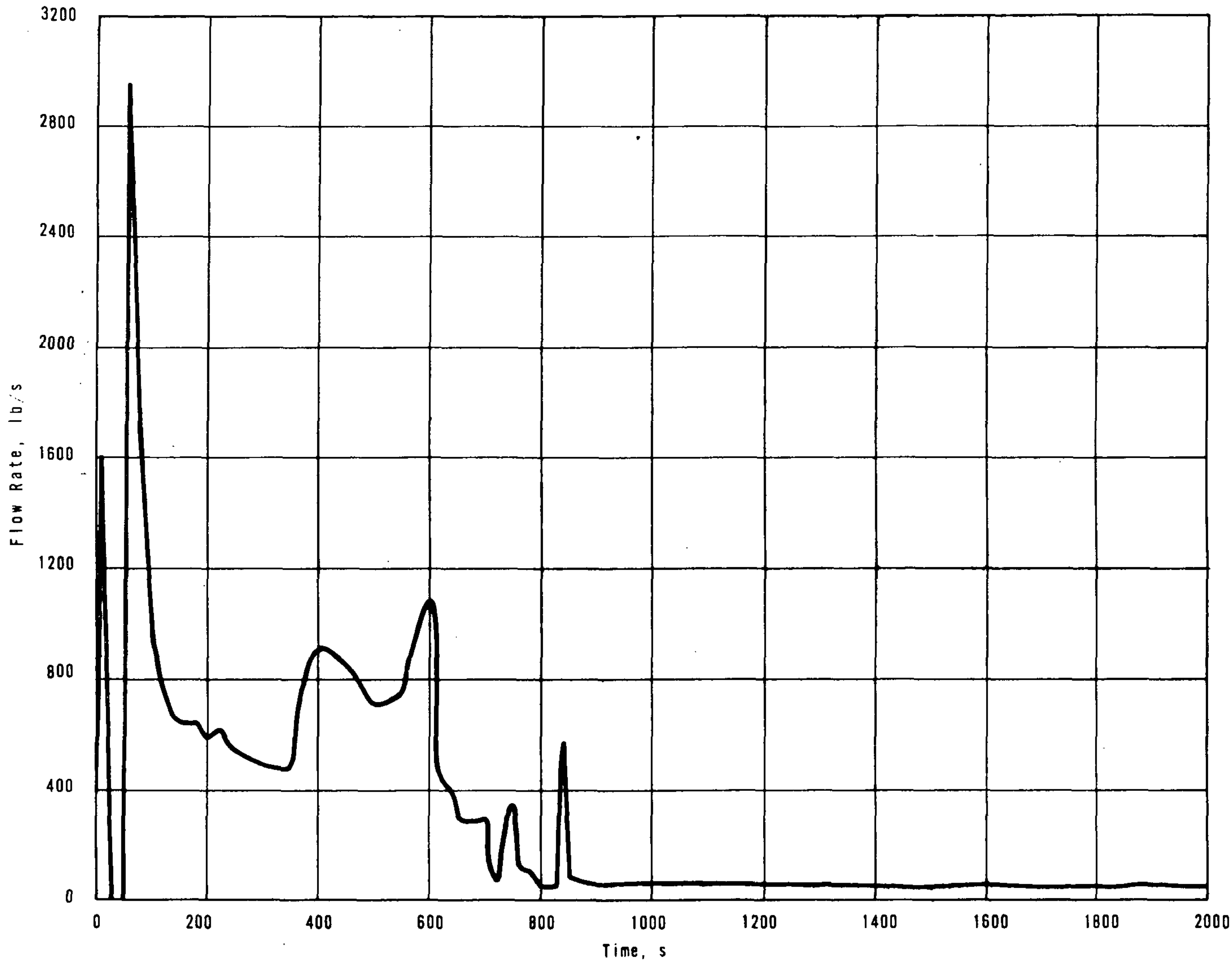


FIGURE 4-5 LEAK FLOW VERSUS TIME

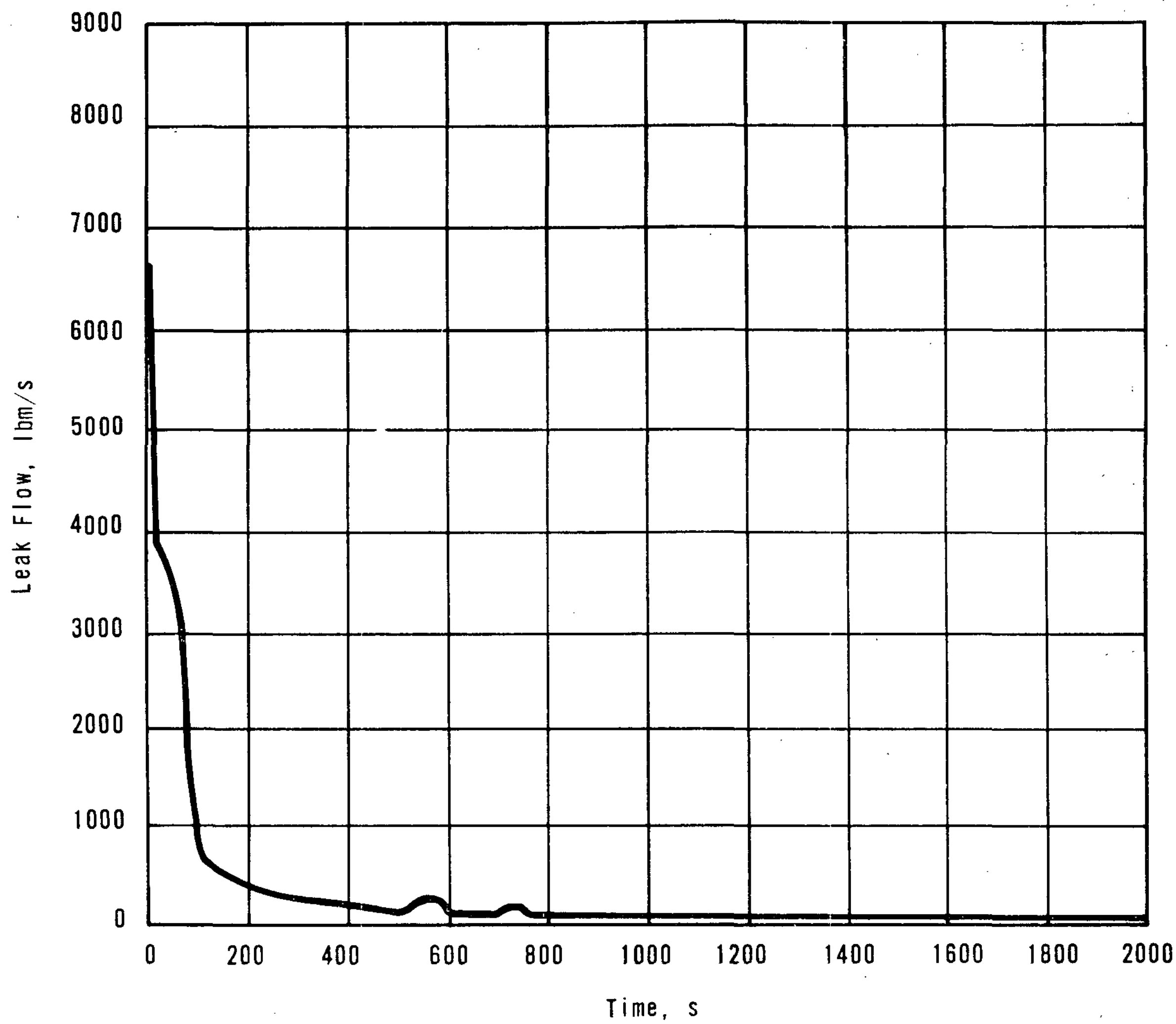


FIGURE 4-6 WATER HEIGHT IN CORE AND DOWNCOMER VERSUS TIME

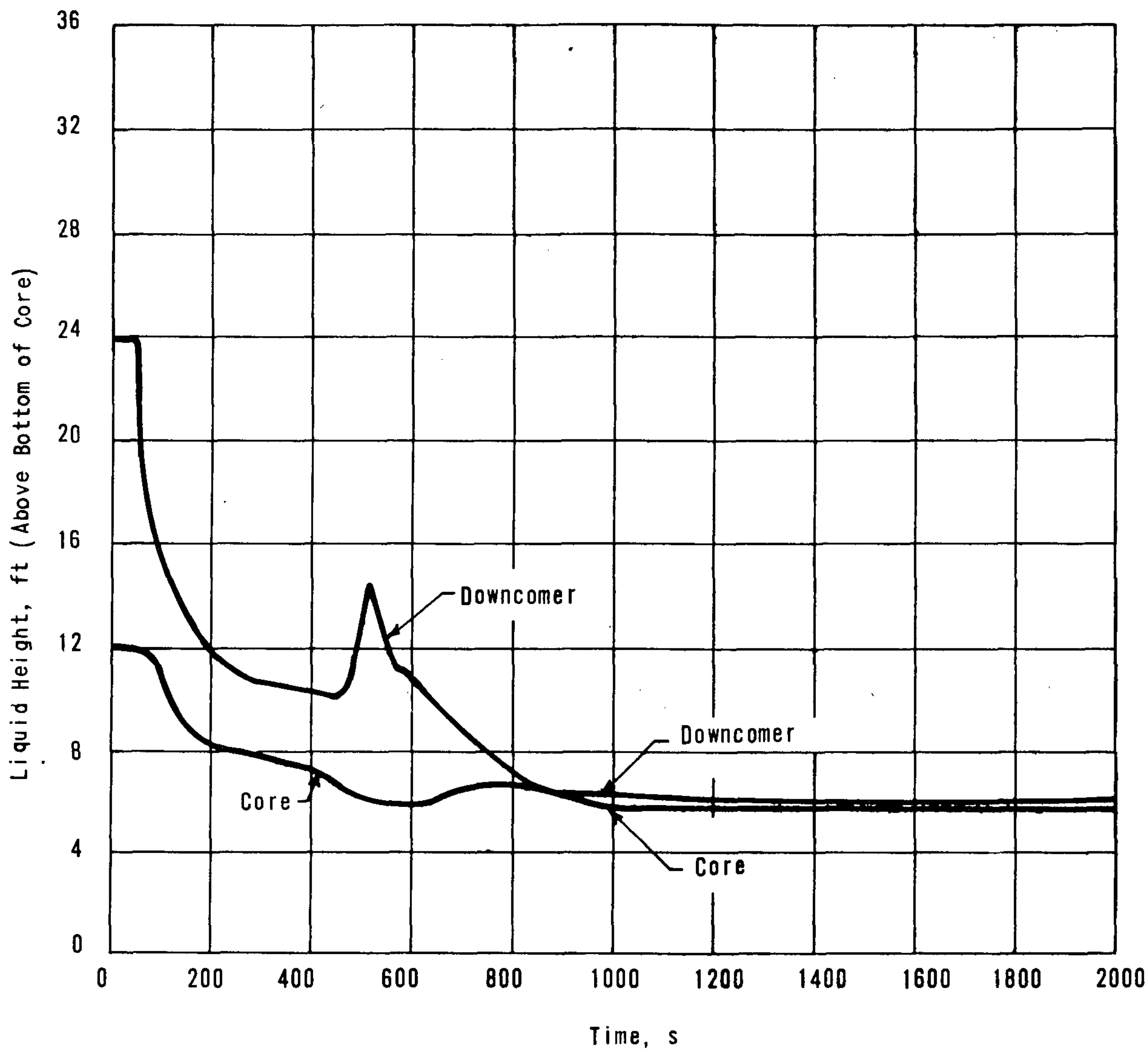
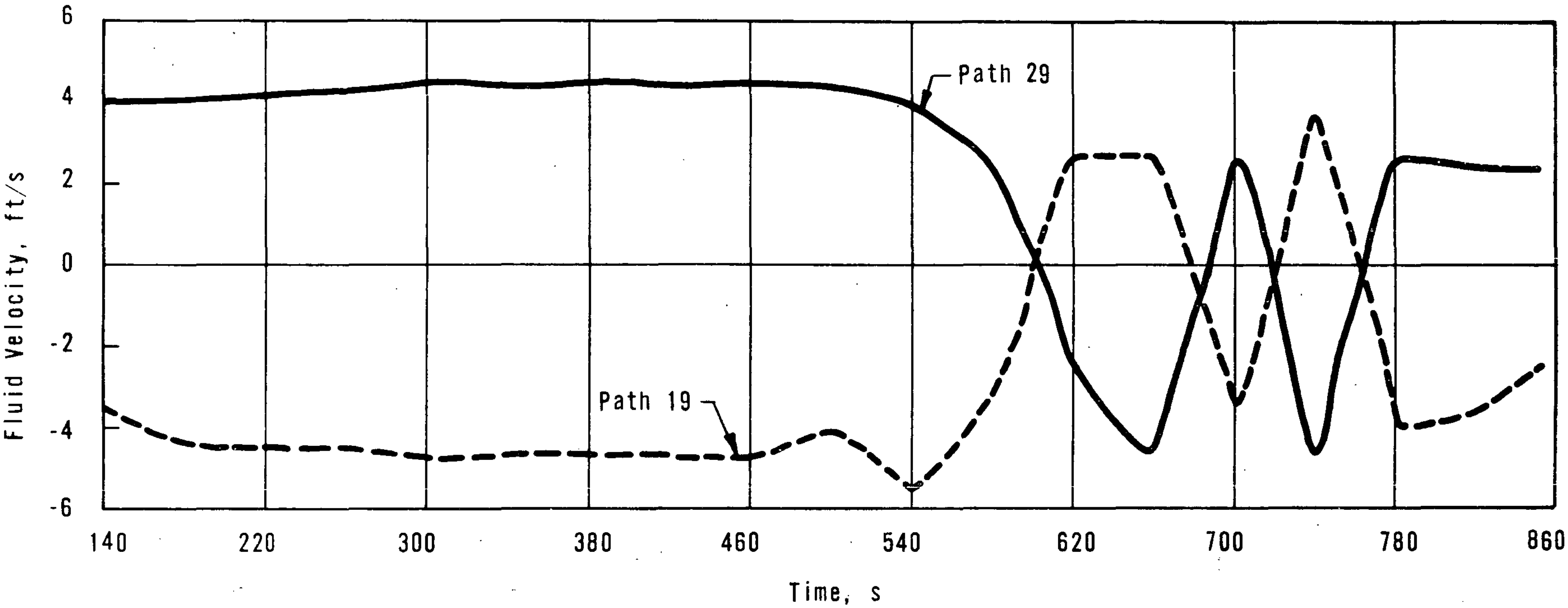
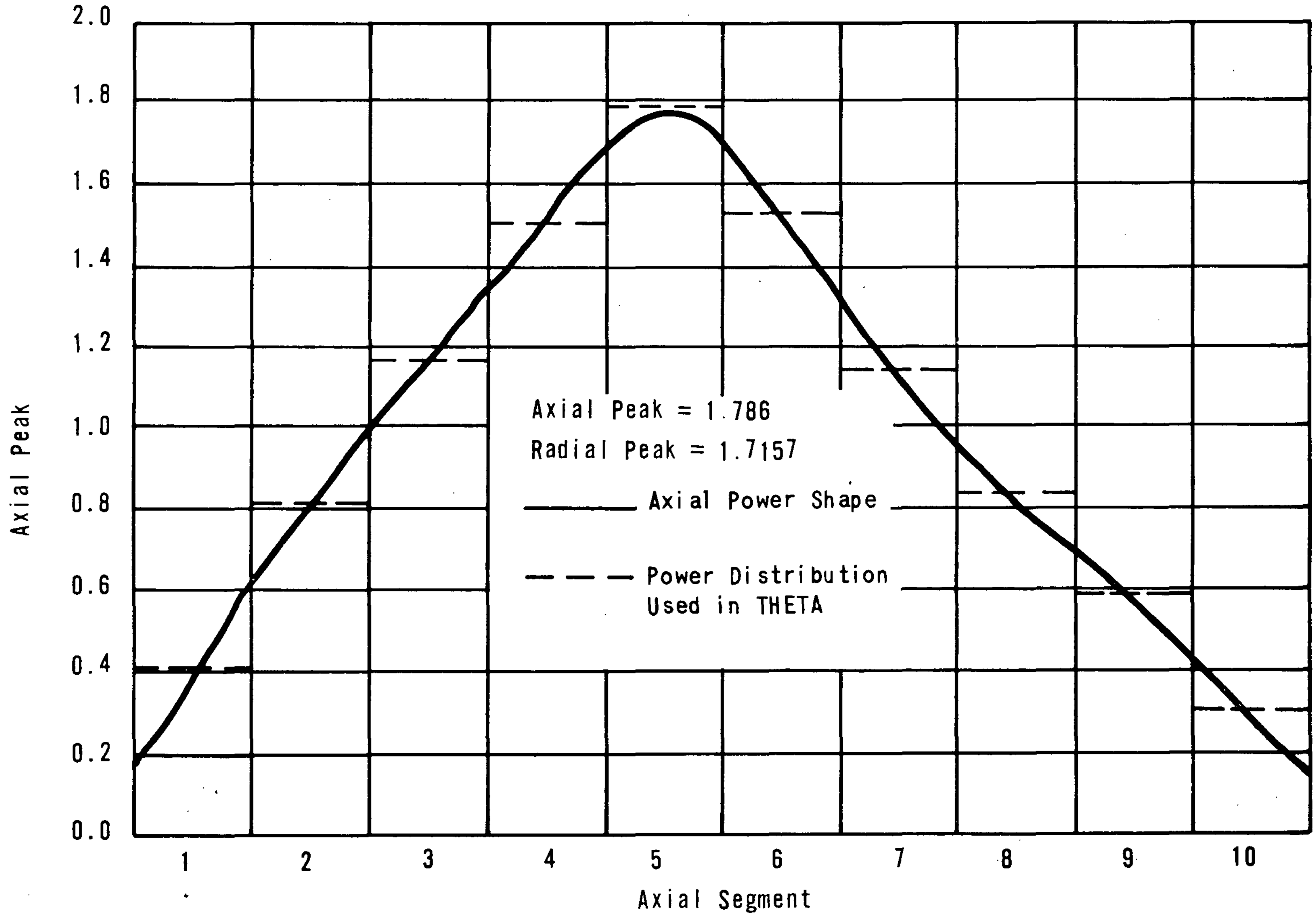


FIGURE 4-7 FLUID VELOCITIES FROM THE DOWNCOMER TO THE LOWER HEAD DURING CFT INJECTION



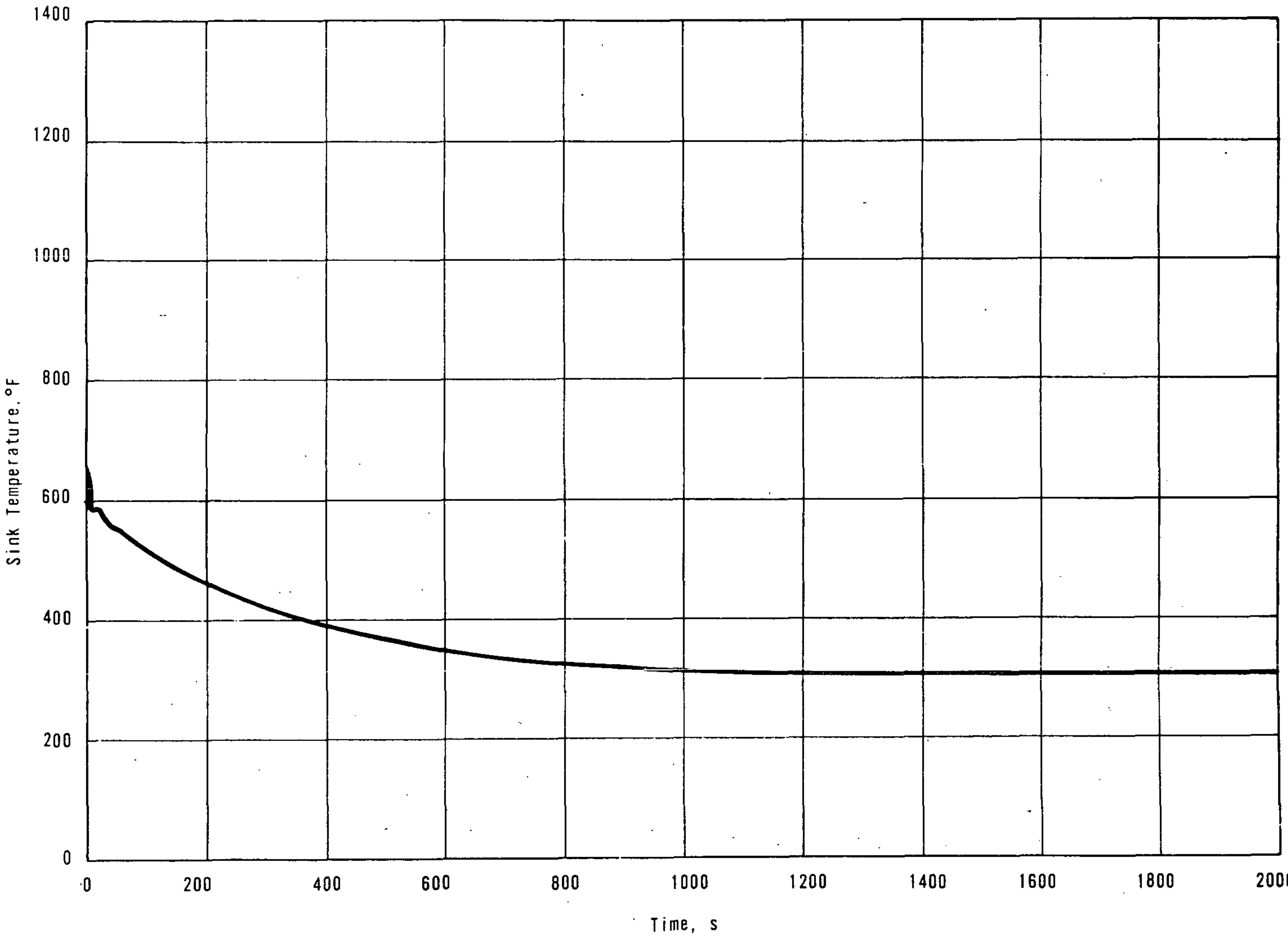
Docket 50-269
FSAR Supplement 14
January 29, 1973

FIGURE 4-8 AXIAL POWER SHAPE FOR CASE 1



Docket 50-269
FSAR Supplement 14
January 29, 1973

FIGURE 4-9 CASE 1, LEVEL 5 - SINK TEMPERATURE VERSUS TIME



Docket 50-269
FSAR Supplement 14
January 29, 1973

FIGURE 4-10 CASE 1, LEVEL 5 - HEAT TRANSFER COEFFICIENT VERSUS TIME

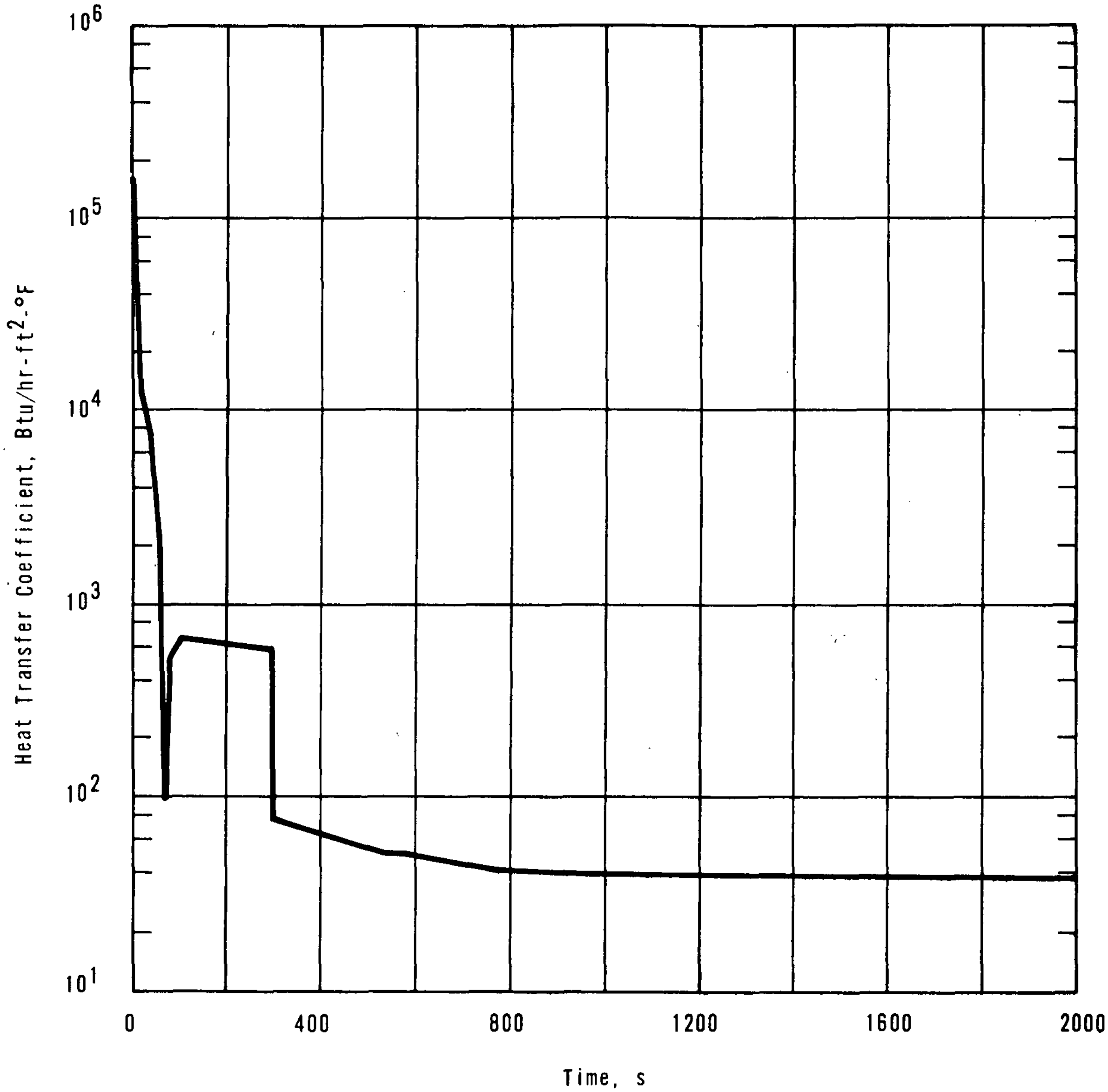


FIGURE 4-11 CASE 1, LEVEL 5 - CLADDING TEMPERATURE VERSUS TIME

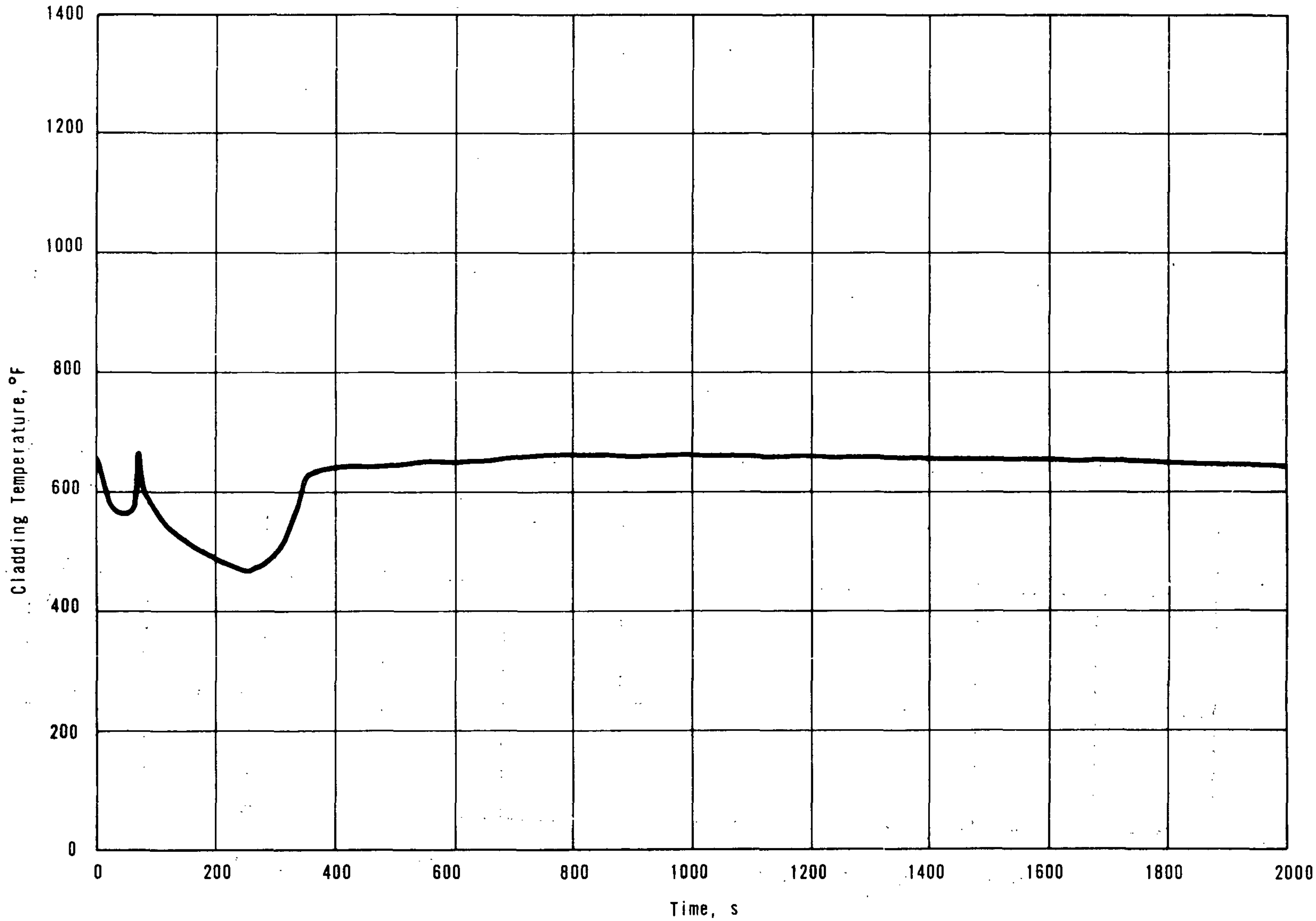


FIGURE 4-12 CASE 1, LEVEL 9 - SINK TEMPERATURE VERSUS TIME

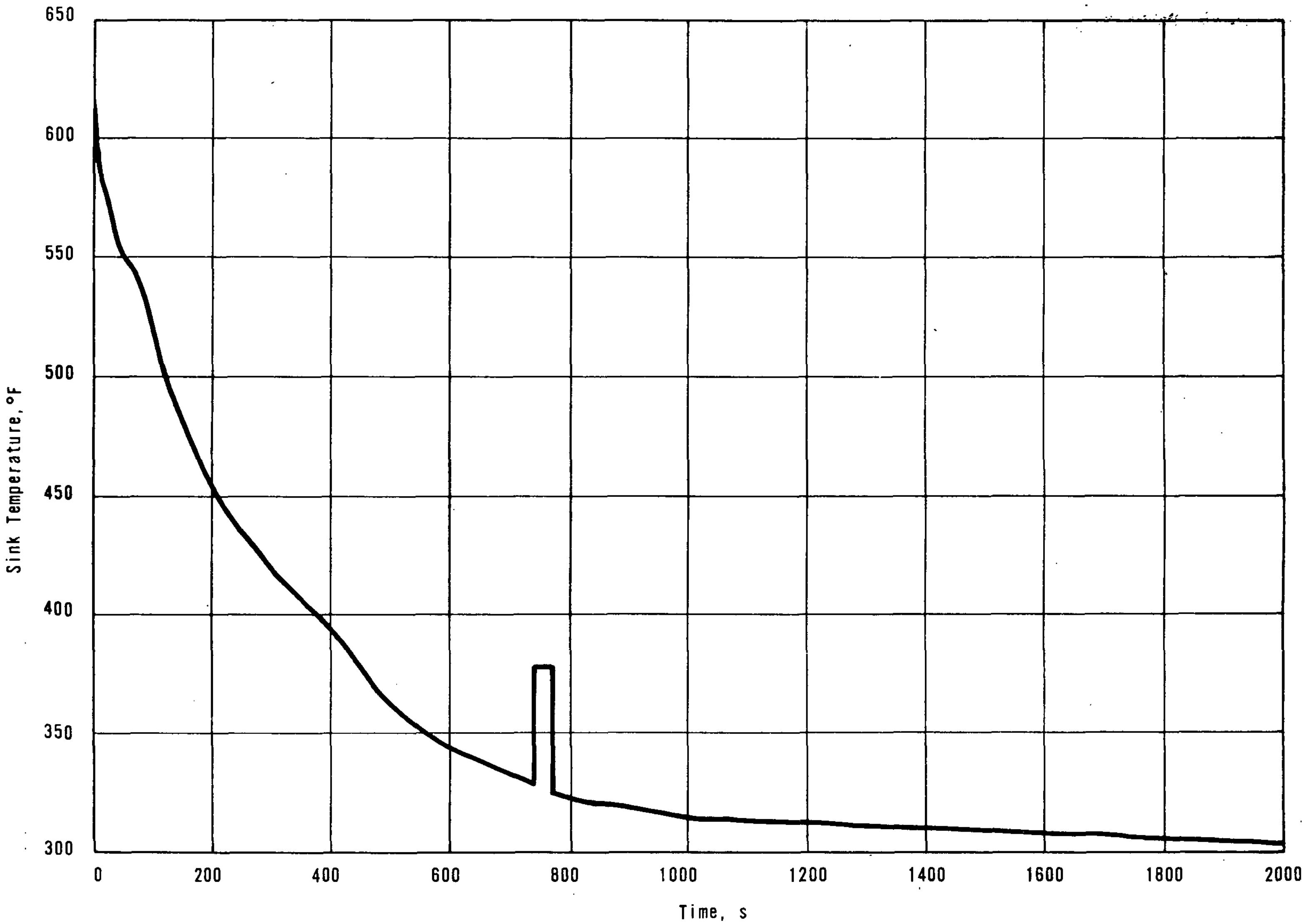


FIGURE 4-13 CASE 1, LEVEL 9 - HEAT TRANSFER COEFFICIENT VERSUS TIME

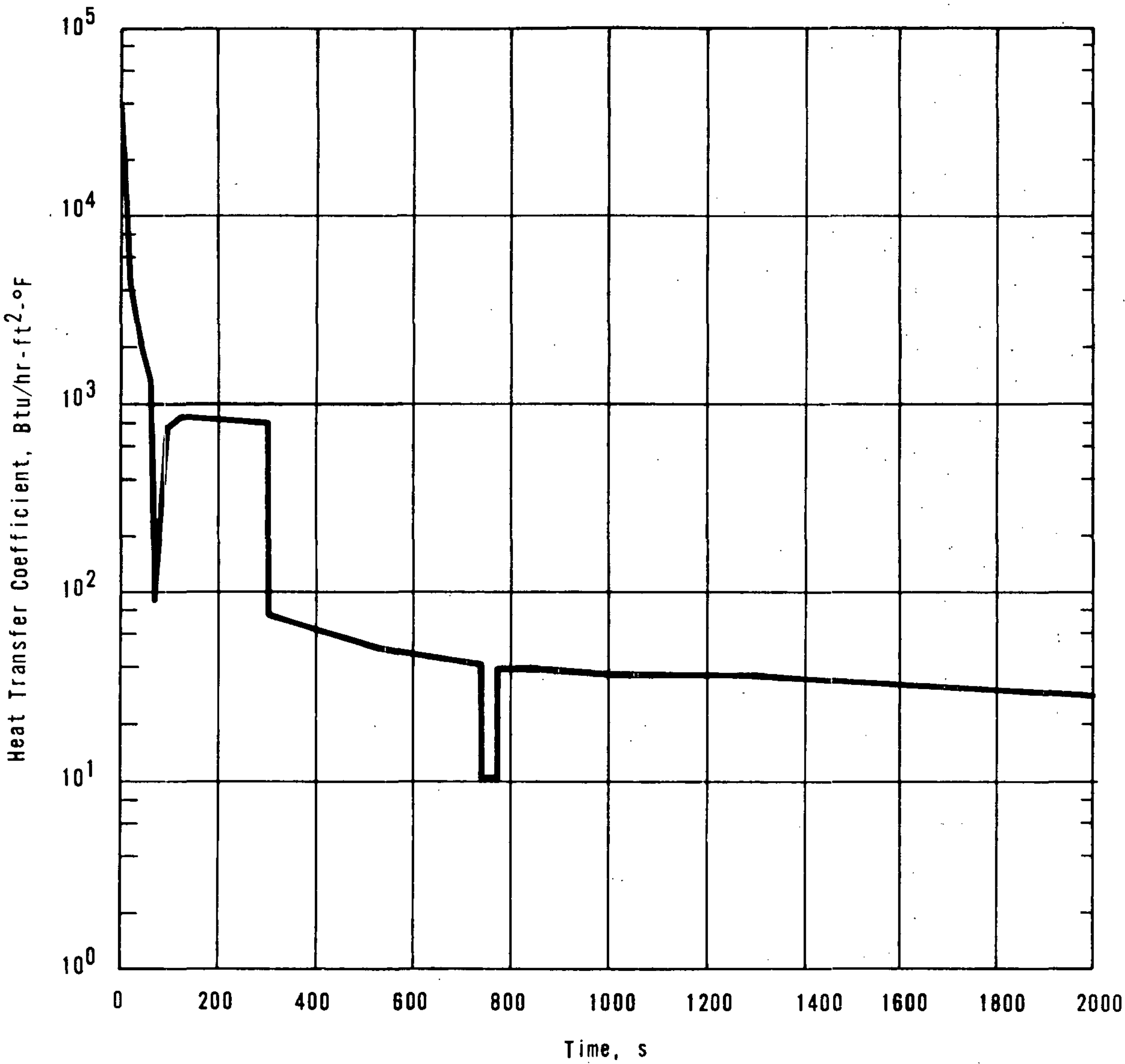


FIGURE 4-14 CASE 1, LEVEL 9 - CLADDING TEMPERATURE VERSUS TIME

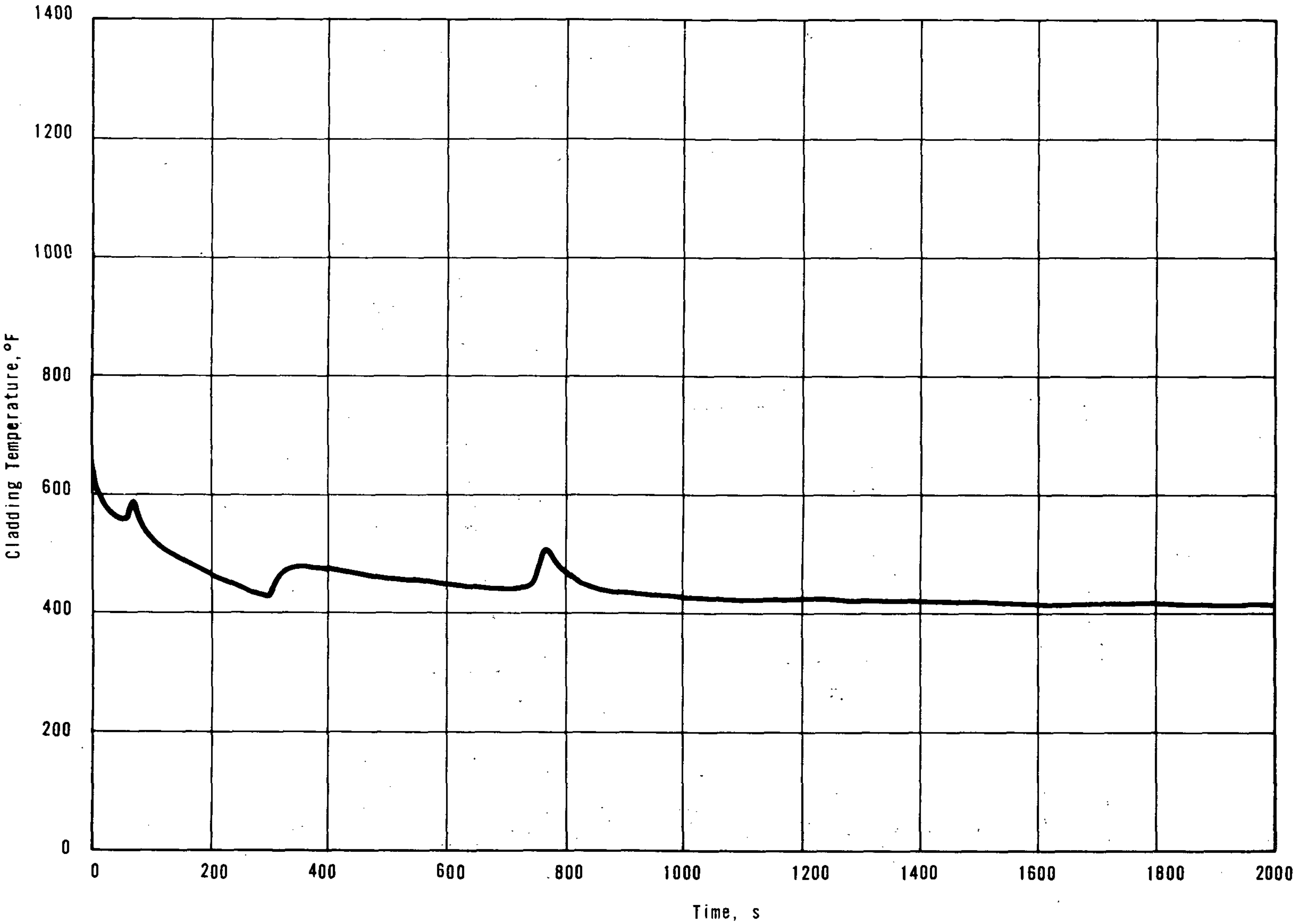


FIGURE 4-15 CASE 1, LEVEL 10 - SINK TEMPERATURE VERSUS TIME

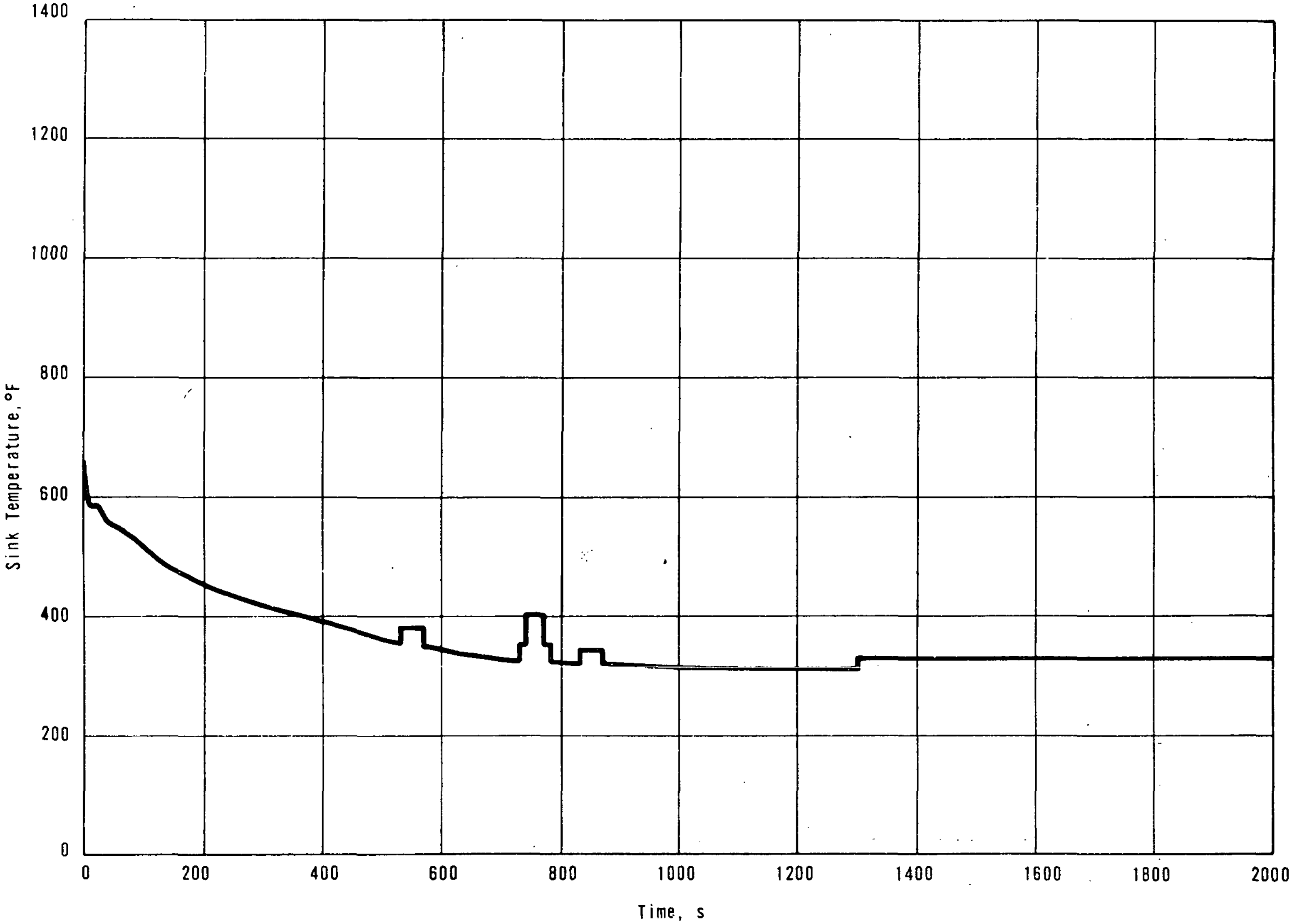


FIGURE 4-16 CASE 1, LEVEL 10 - HEAT TRANSFER COEFFICIENT VERSUS TIME

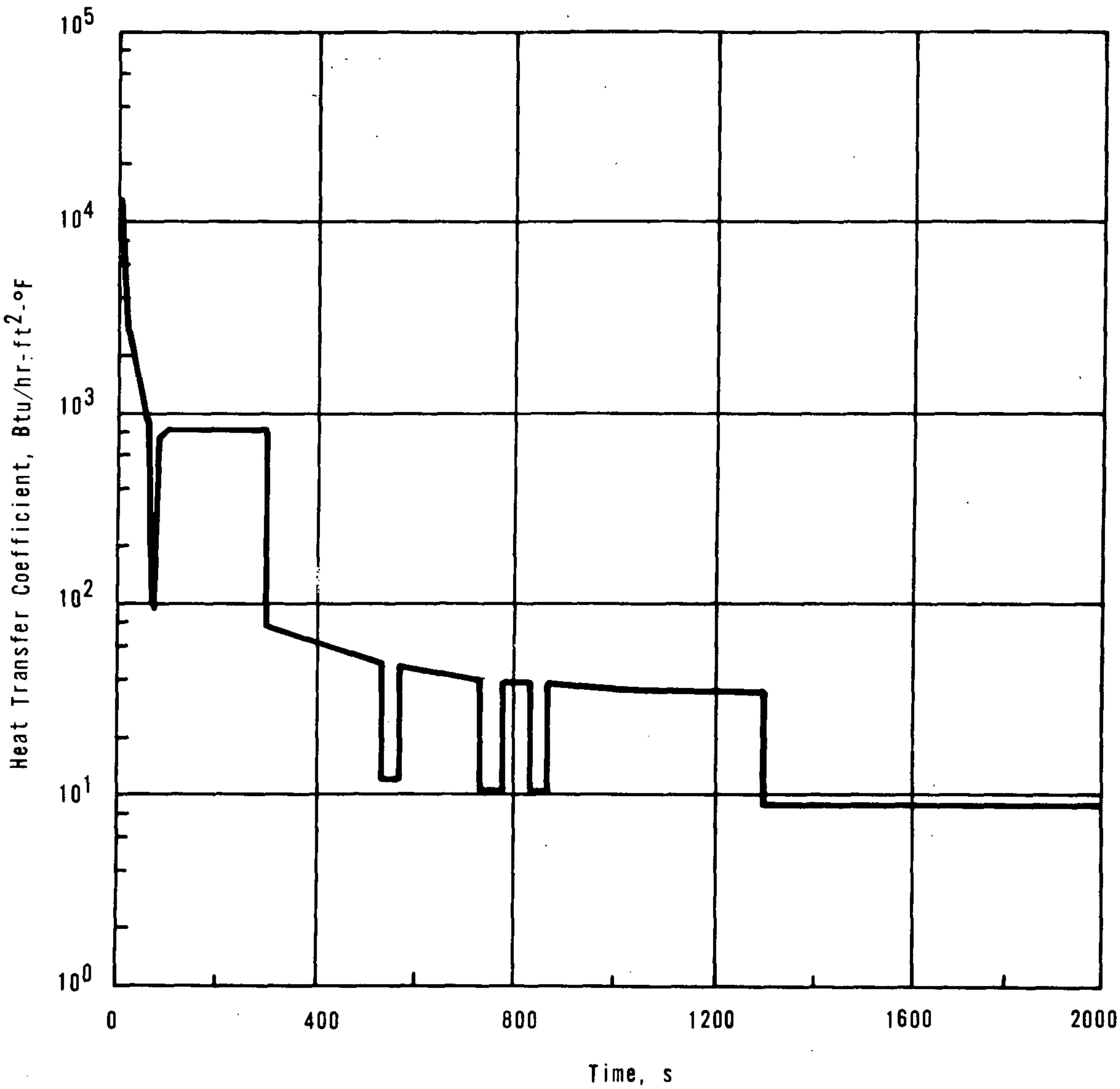
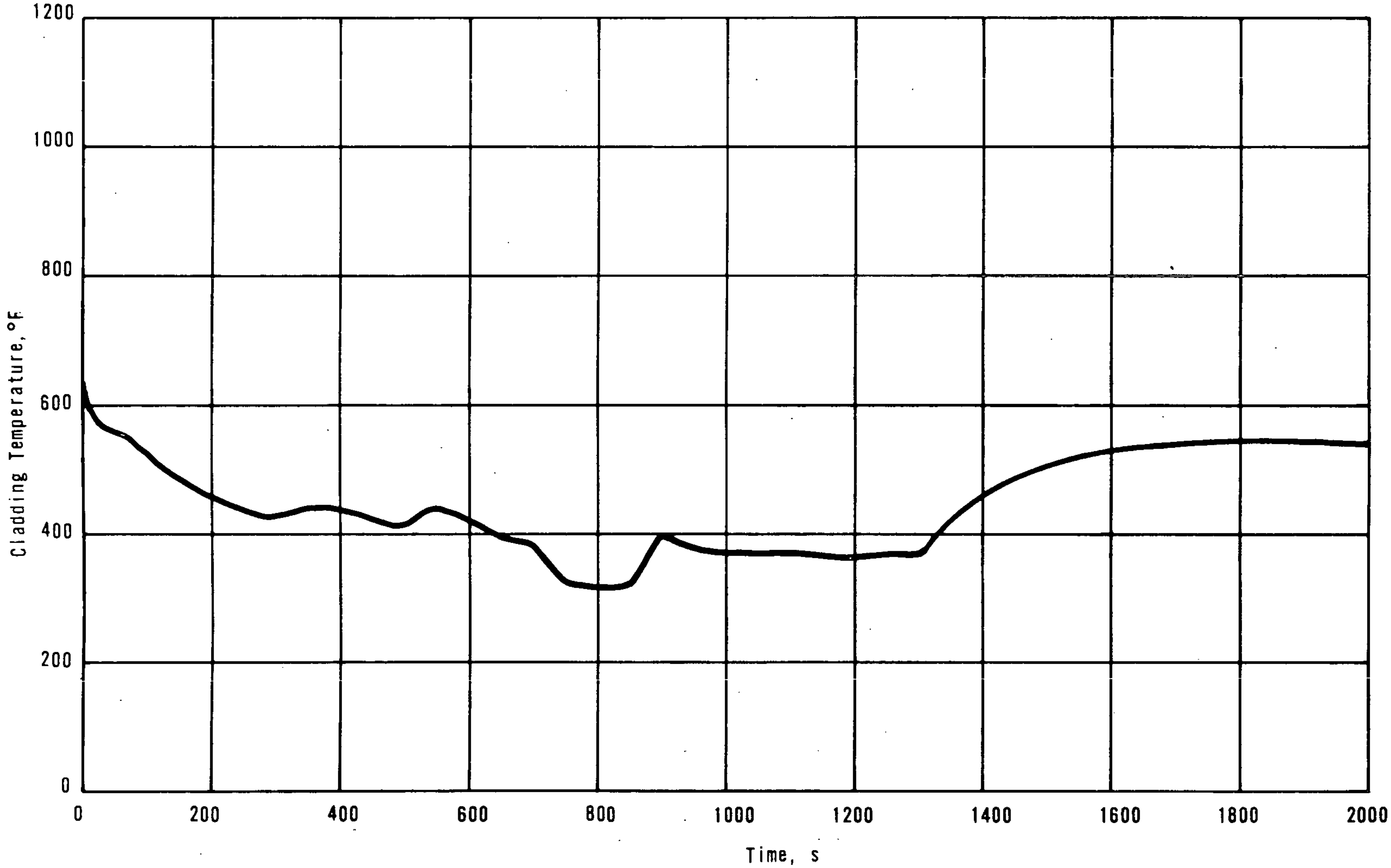


FIGURE 4-17 CASE 1, LEVEL 10 - CLADDING TEMPERATURE VERSUS TIME



Docket 50-269
FSAR Supplement 14
January 29, 1973

FIGURE 4-18 AXIAL POWER SHAPE FOR CASE 2

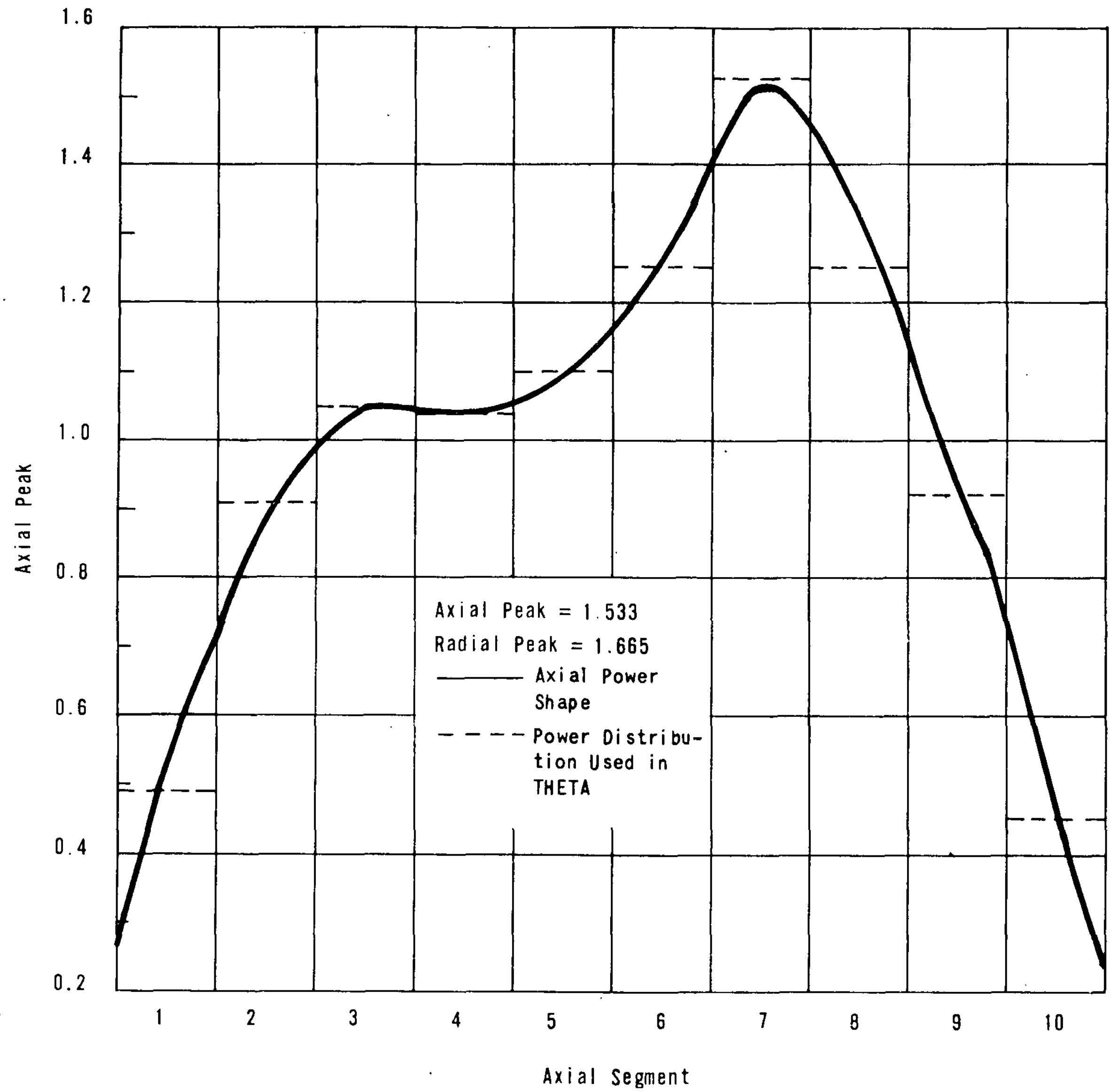


FIGURE 4-19 CASE 2, LEVEL 7 - SINK TEMPERATURE VERSUS TIME

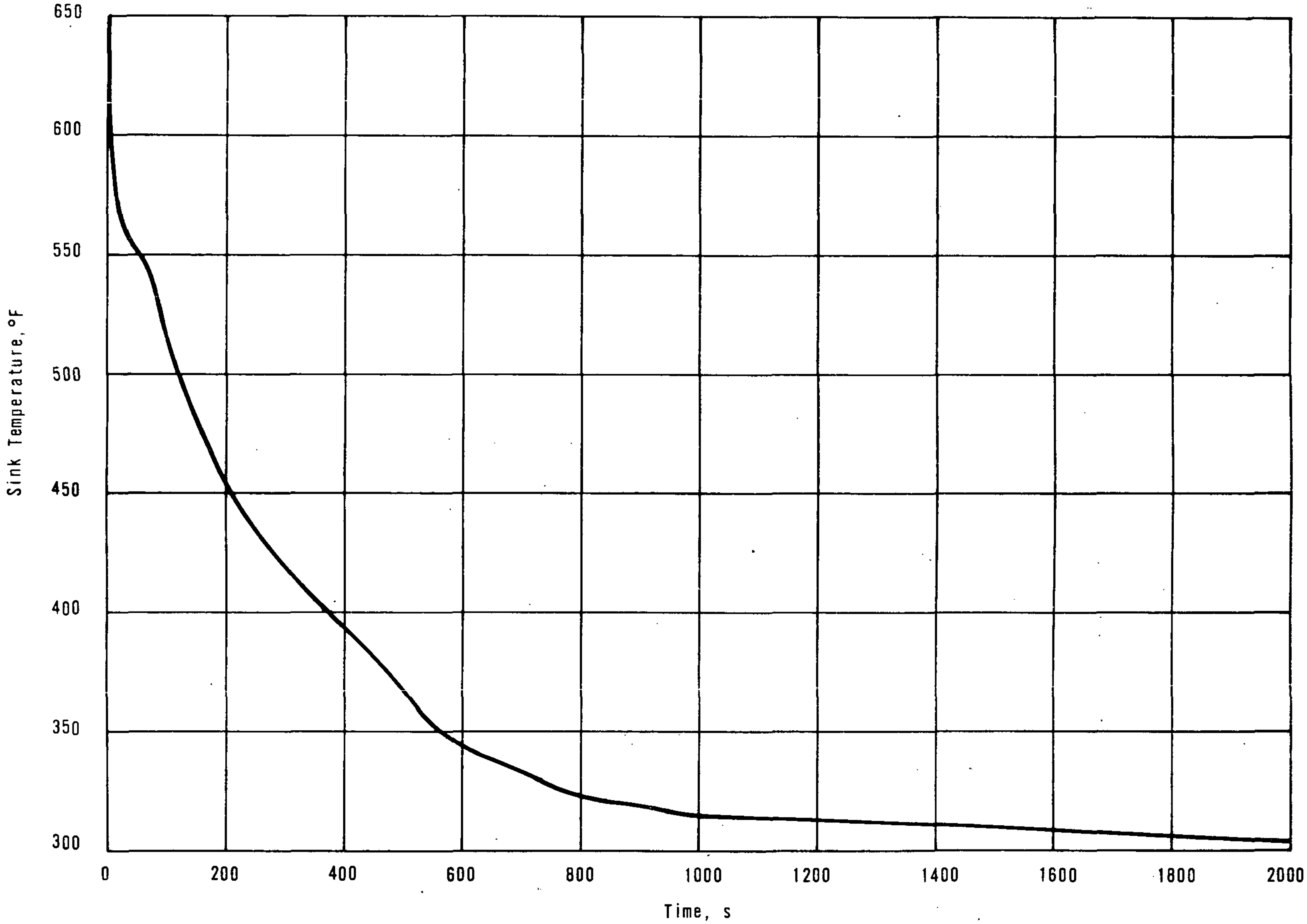


FIGURE 4-20 CASE 2, LEVEL 7 - HEAT TRANSFER COEFFICIENT VERSUS TIME

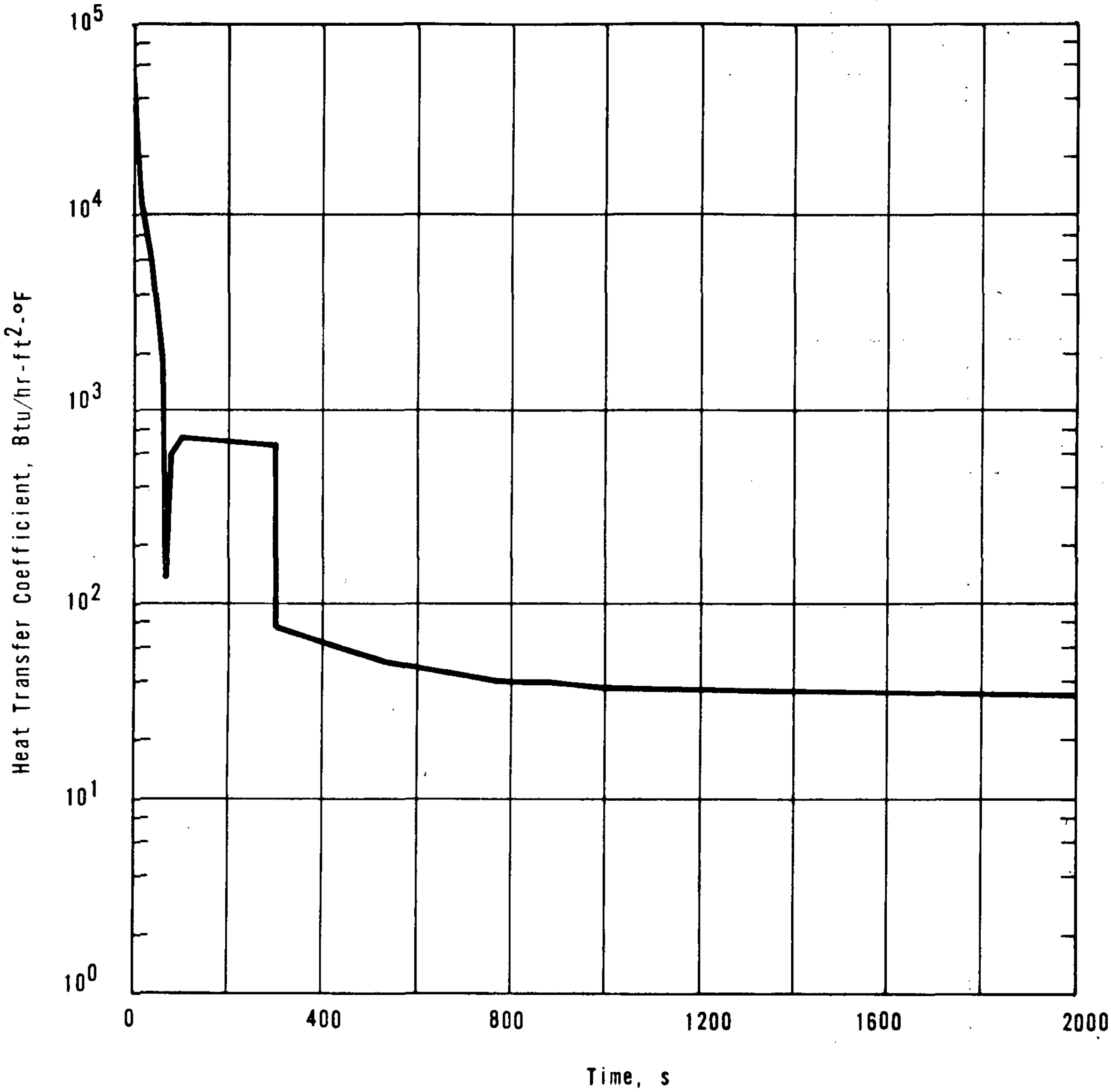


FIGURE 4-21 CASE 2, LEVEL 7 - CLADDING TEMPERATURE VERSUS TIME

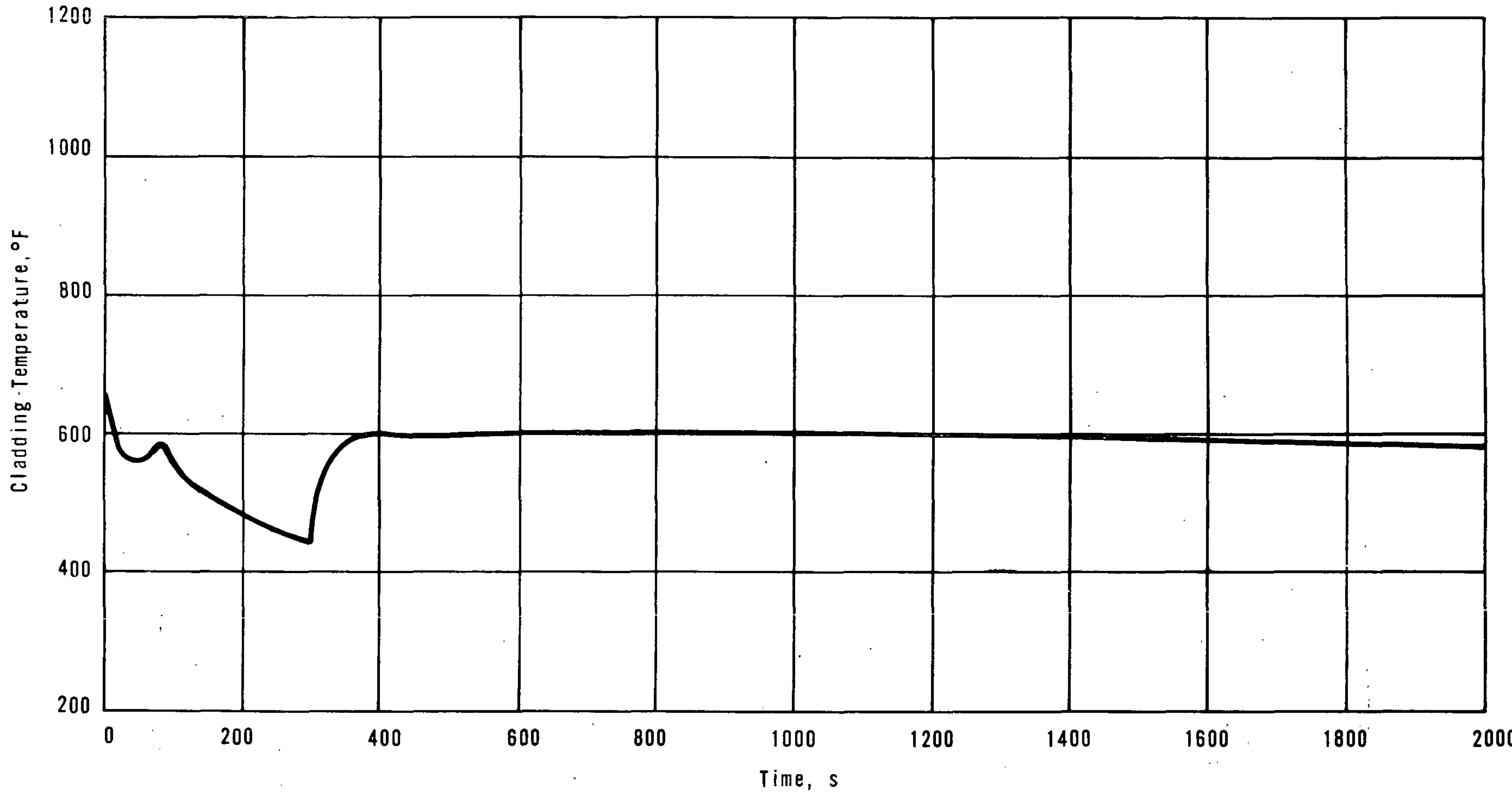


FIGURE 4-22 CASE 2, LEVEL 9 - SINK TEMPERATURE VERSUS TIME

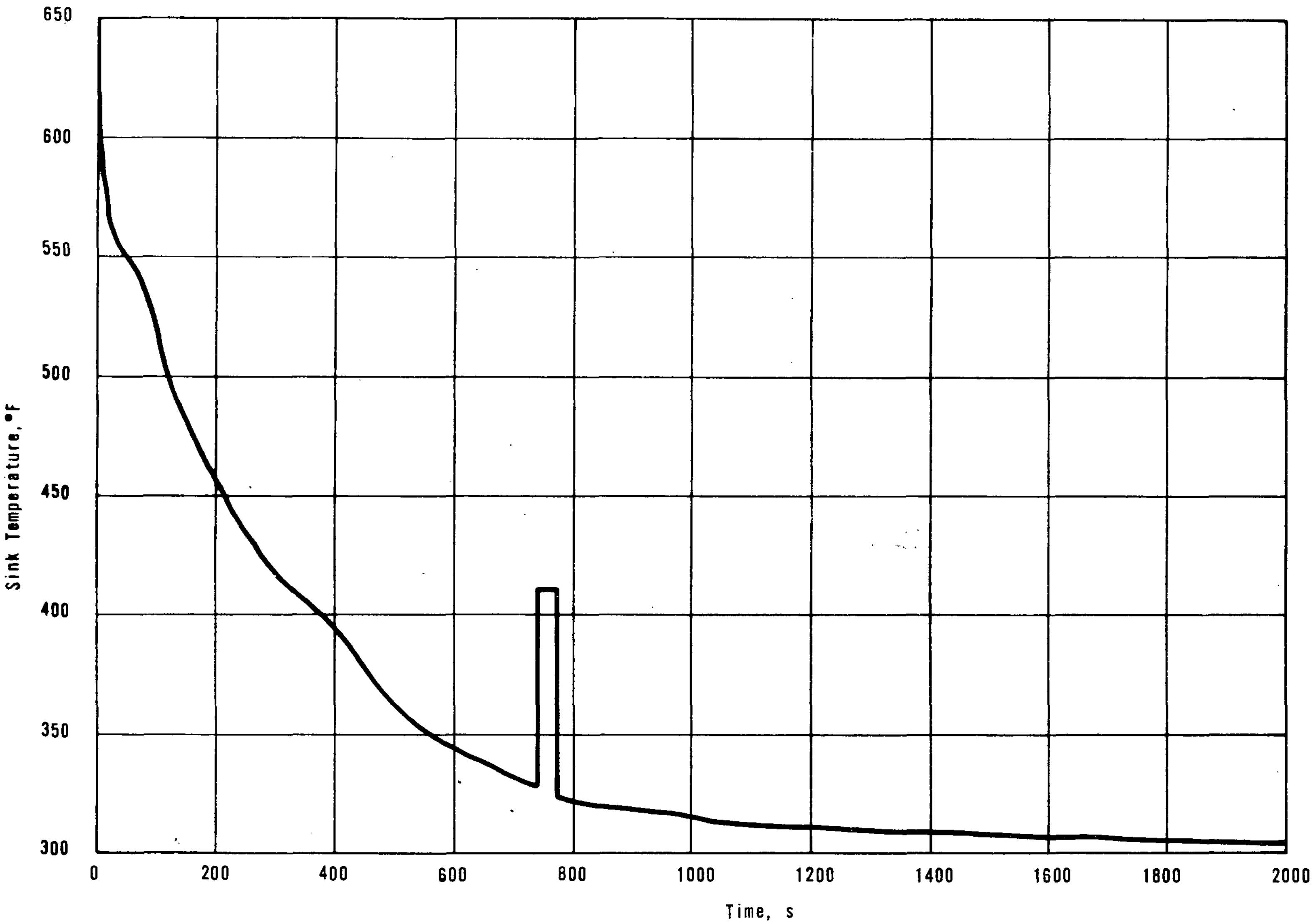


FIGURE 4-23 CASE 2, LEVEL 9 - HEAT TRANSFER COEFFICIENT VERSUS TIME

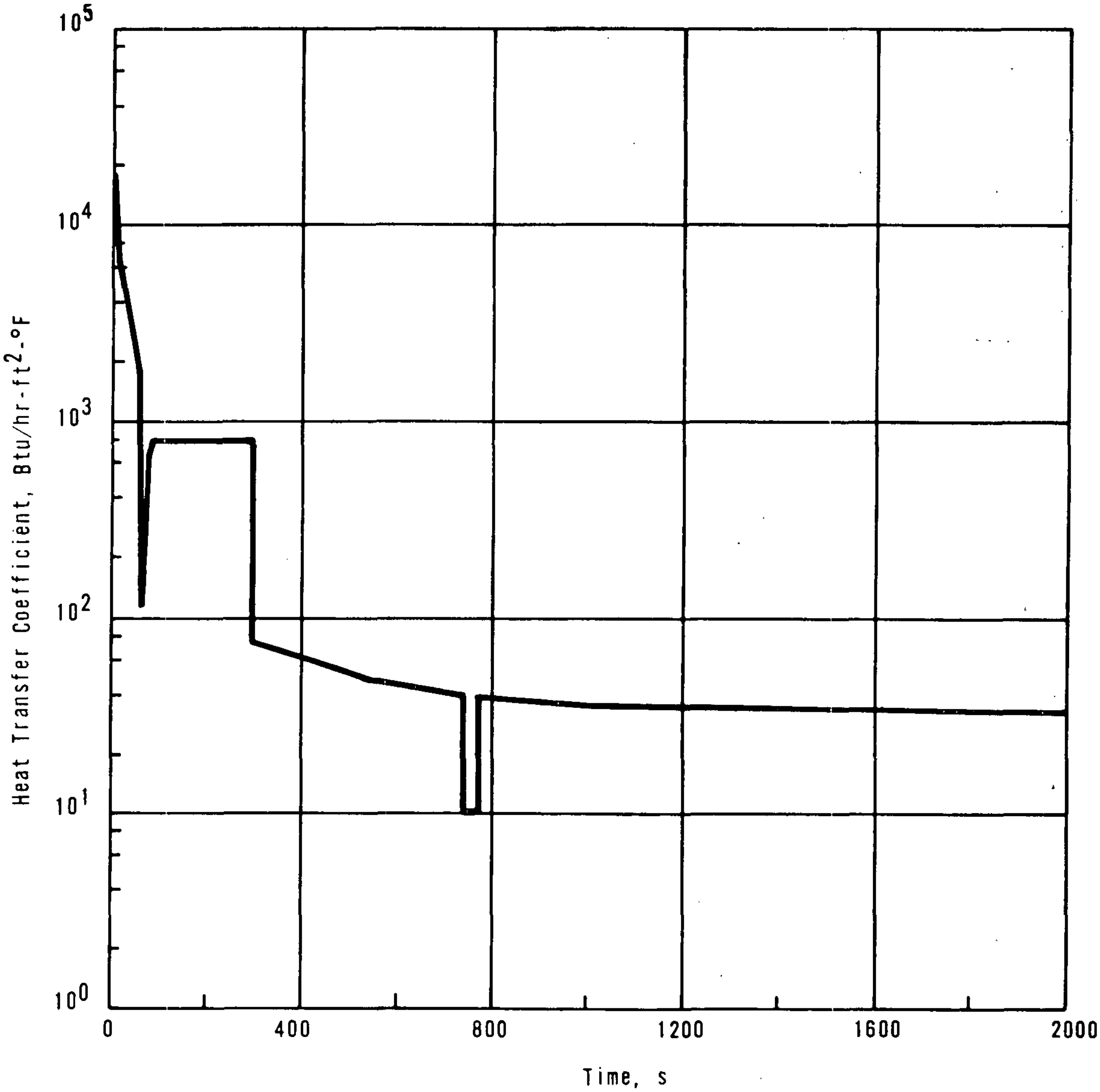
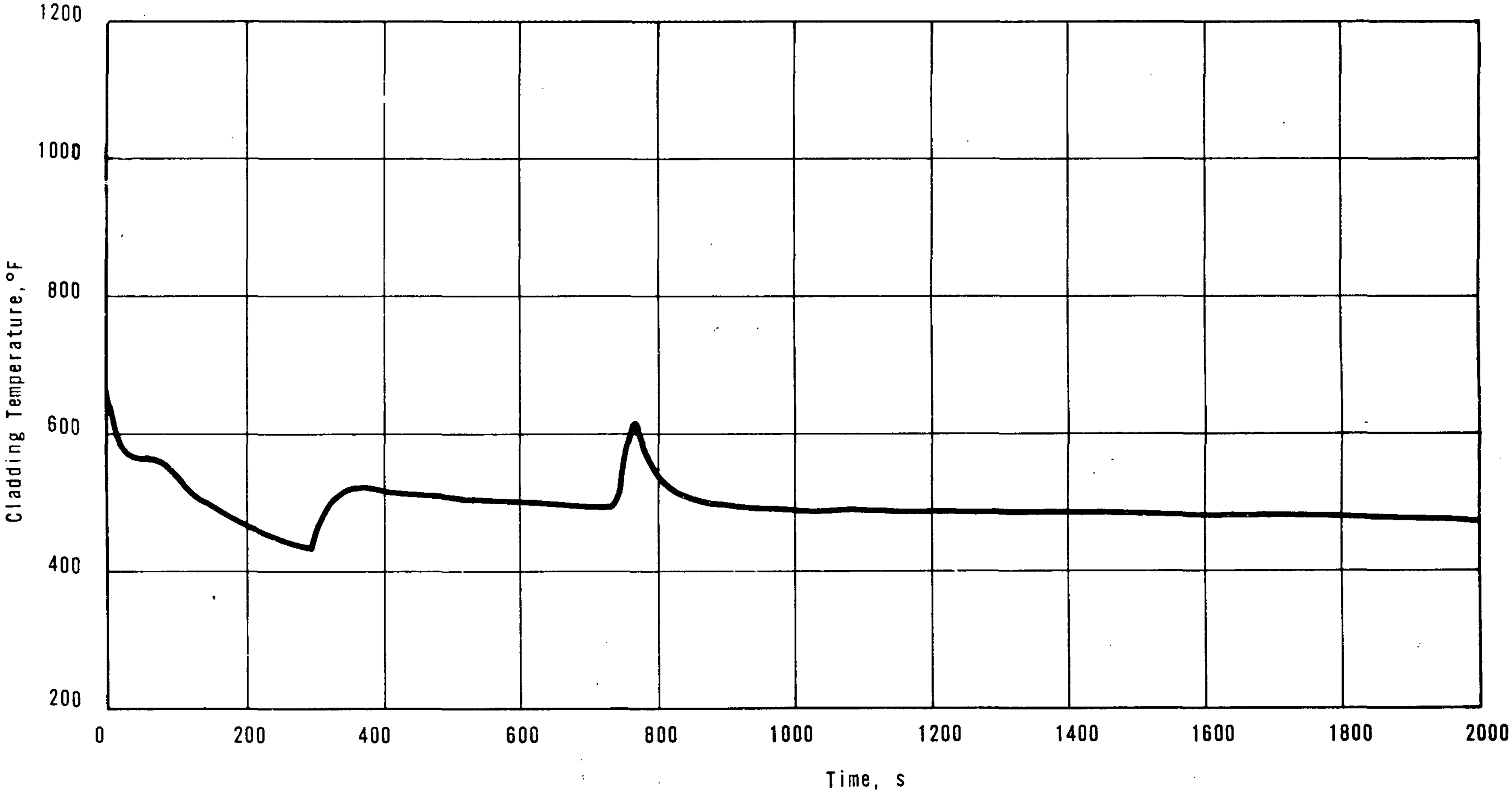
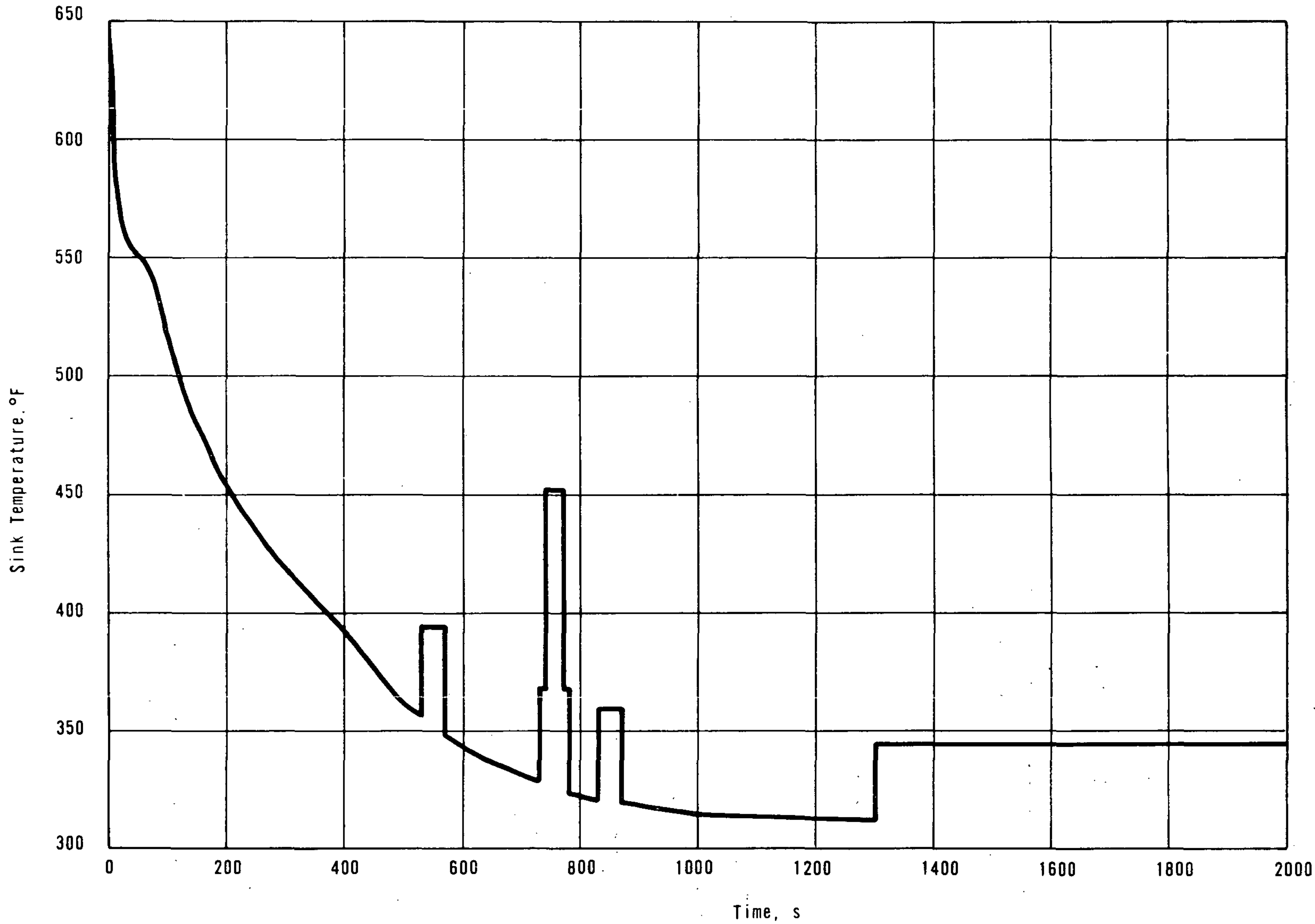


FIGURE 4-24 CASE 2, LEVEL 9 - CLADDING TEMPERATURE VERSUS TIME



Docket 50-269
FSAR Supplement 14
January 29, 1973

FIGURE 4-25 CASE 2, LEVEL 10 - SINK TEMPERATURE VERSUS TIME



Docket 50-269
FSAR Supplement 14
January 29, 1973

FIGURE 4-26 CASE 2, LEVEL 10 - HEAT TRANSFER COEFFICIENT VERSUS TIME

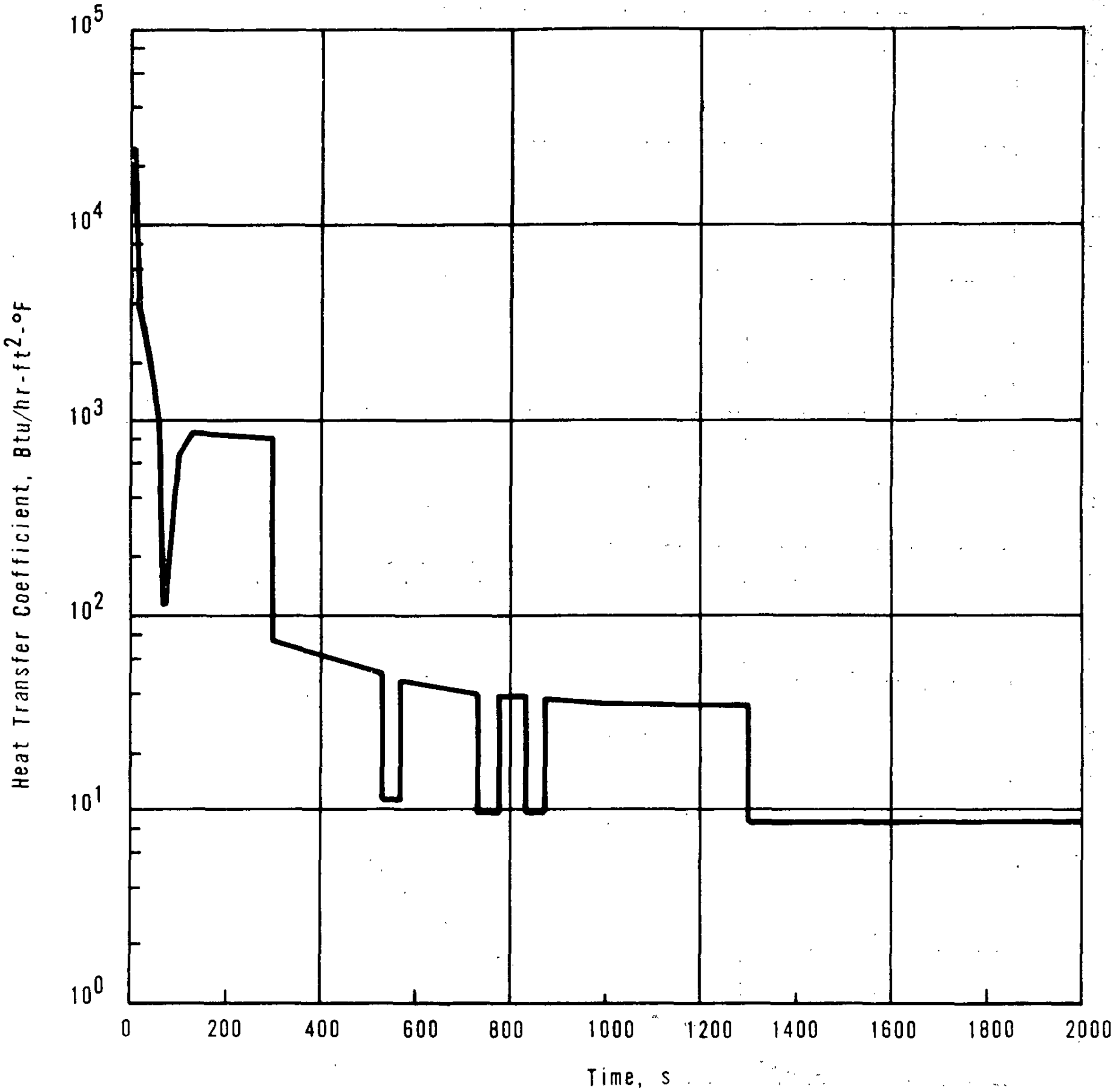
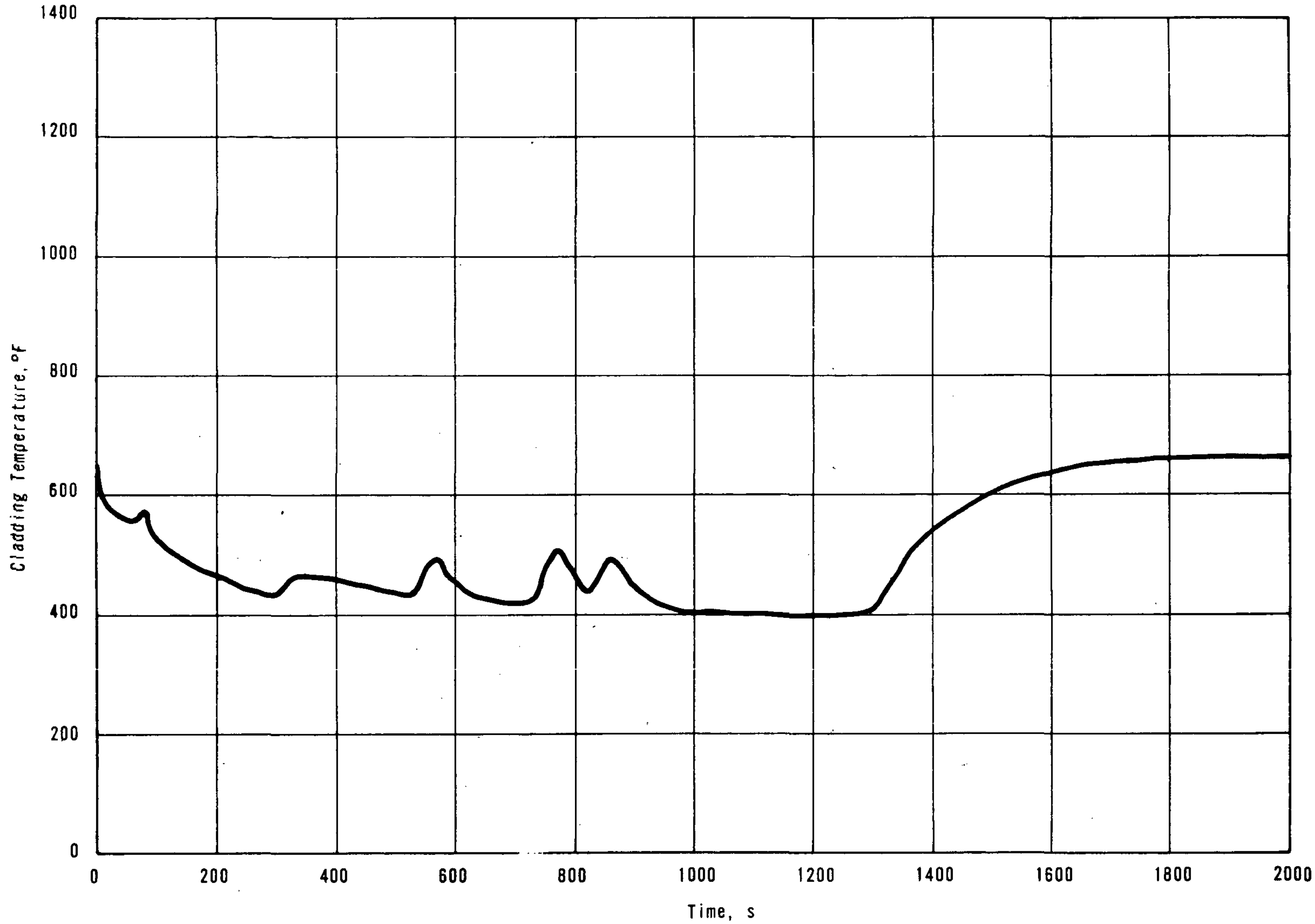


FIGURE 4-27 CASE 2, LEVEL 10 - CLADDING TEMPERATURE VERSUS TIME



Docket 50-269
FSAR Supplement 14
January 29, 1973

FIGURE 4-28 AXIAL POWER SHAPE FOR CASE 3

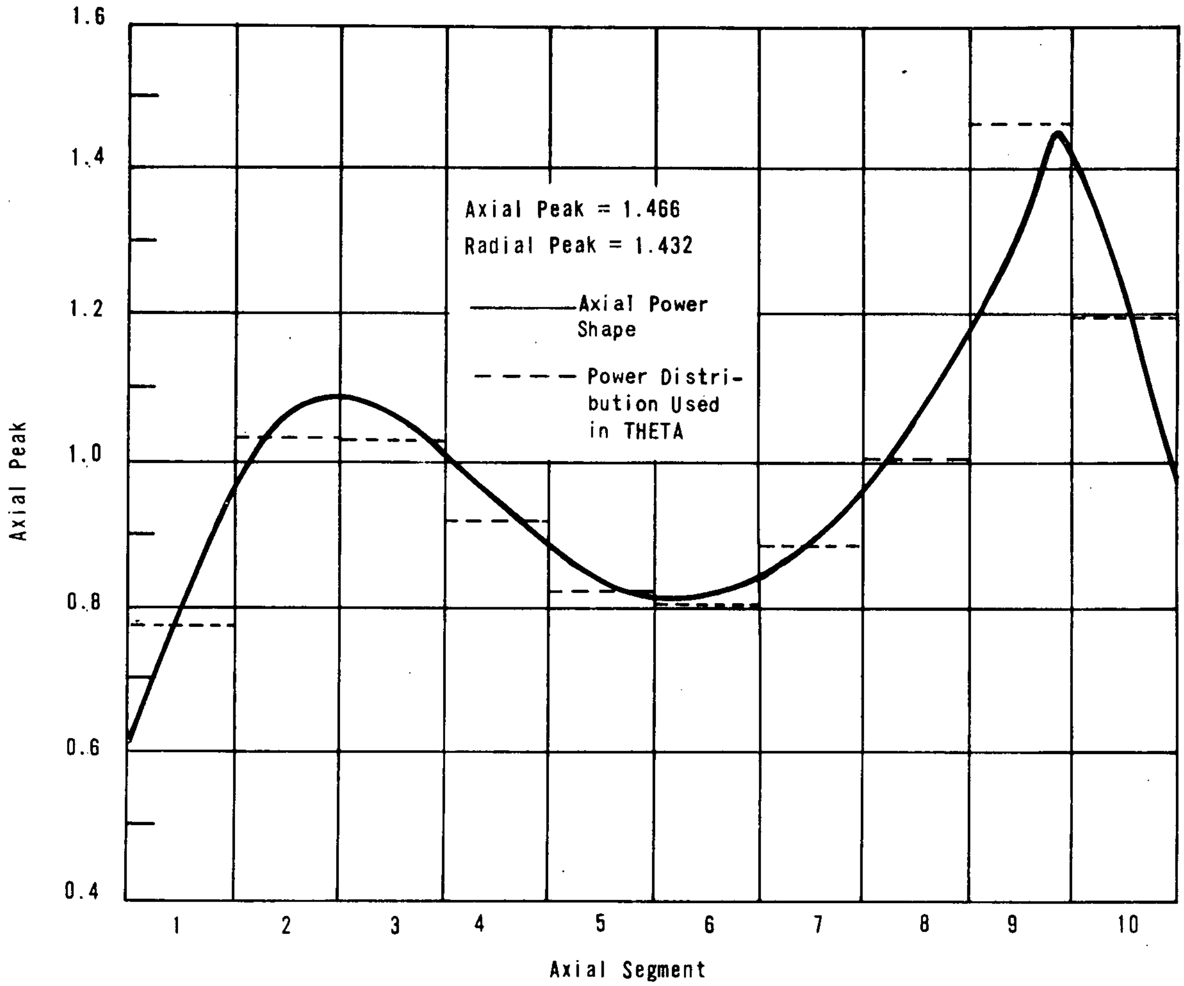


FIGURE 4-29 CASE 3, LEVEL 9 - SINK TEMPERATURE VERSUS TIME

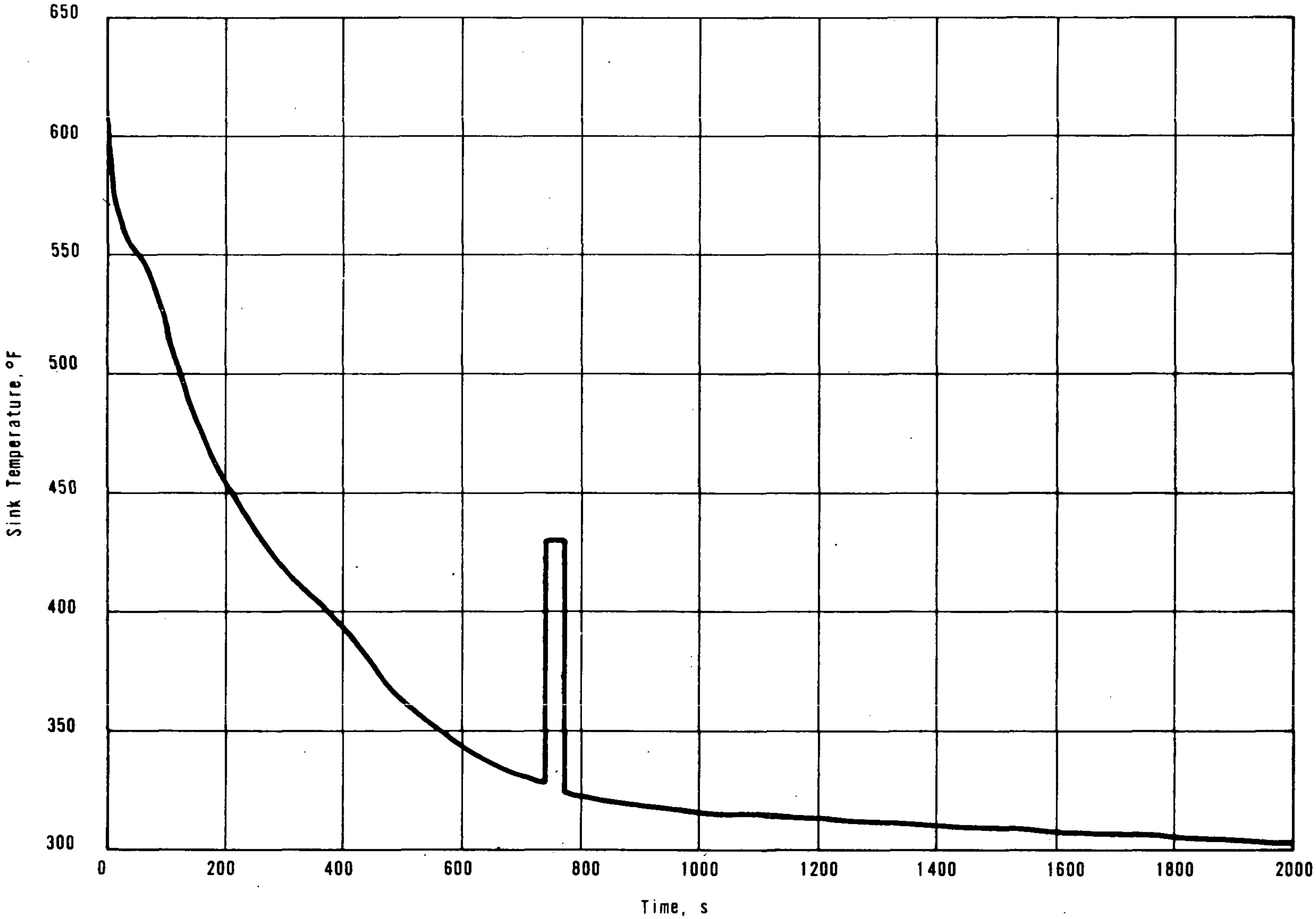


FIGURE 4-30 CASE 3, LEVEL 9 - HEAT TRANSFER COEFFICIENT VERSUS TIME

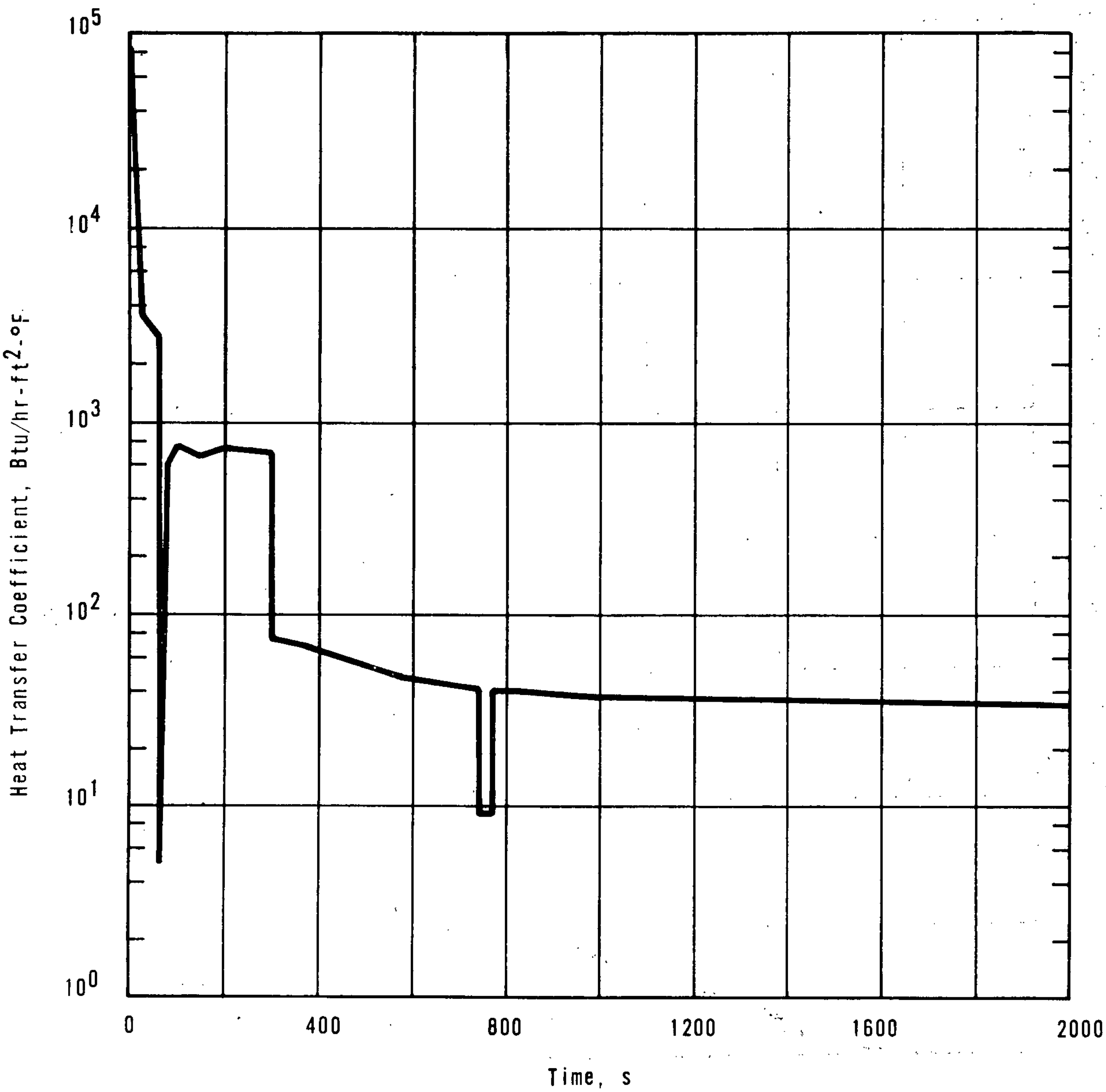
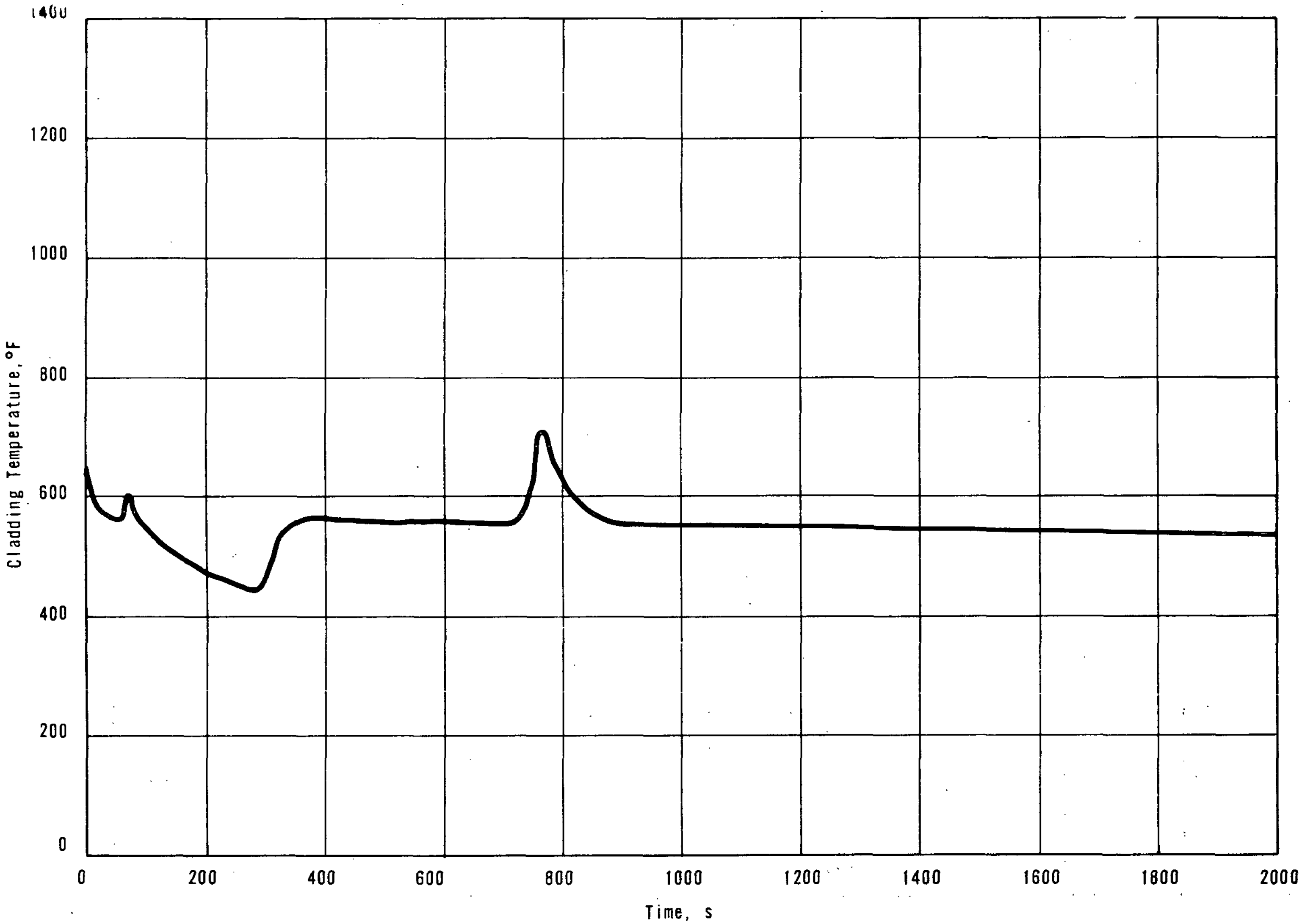


FIGURE 4-31 CASE 3, LEVEL 9 - CLADDING TEMPERATURE VERSUS TIME



Docket 50-269
FSAR Supplement 14
January 29, 1973

FIGURE 4-32 CASE 3, LEVEL 10 - SINK TEMPERATURE VERSUS TIME

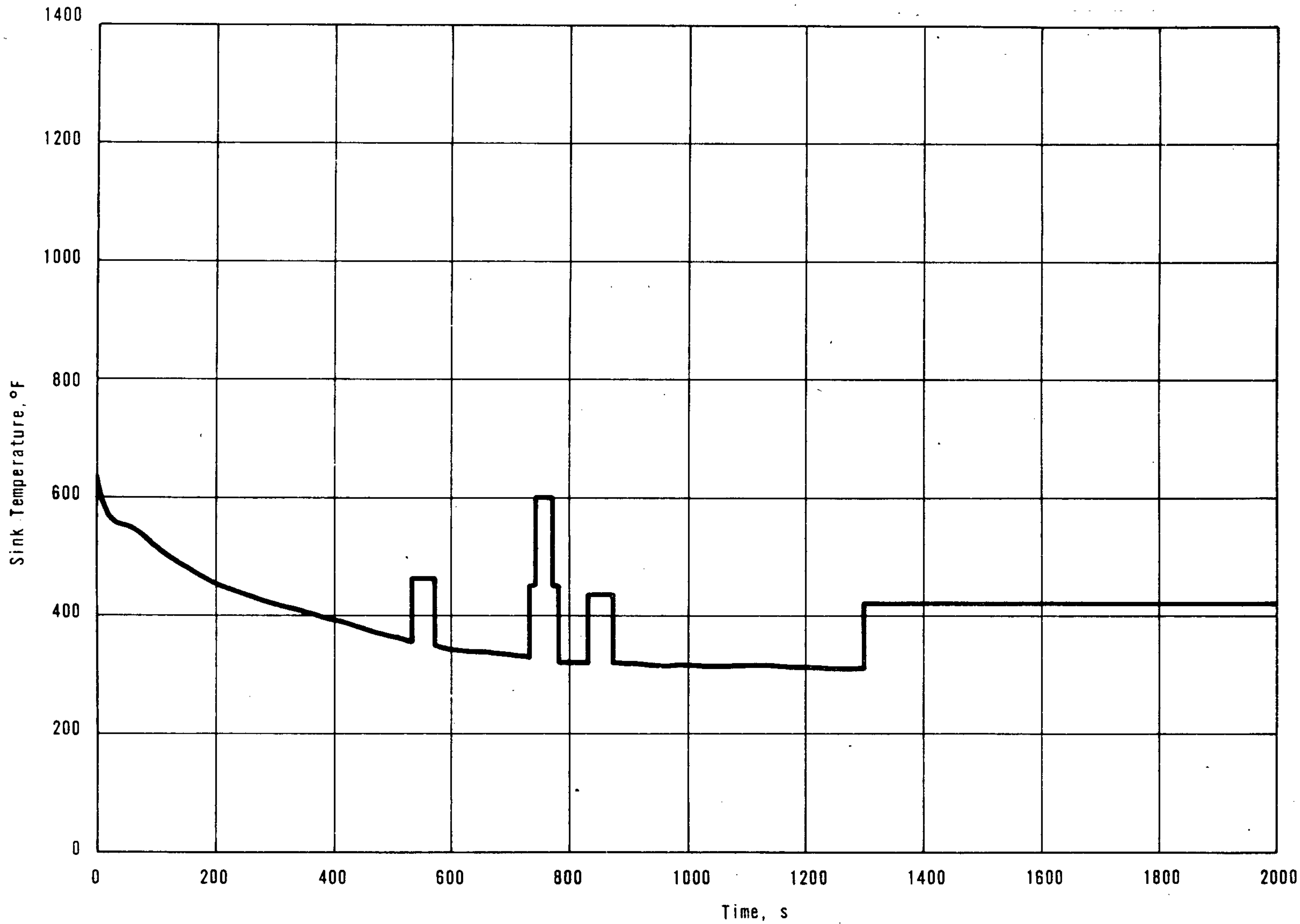


FIGURE 4-33 CASE 3, LEVEL 10 - HEAT TRANSFER COEFFICIENT VERSUS TIME

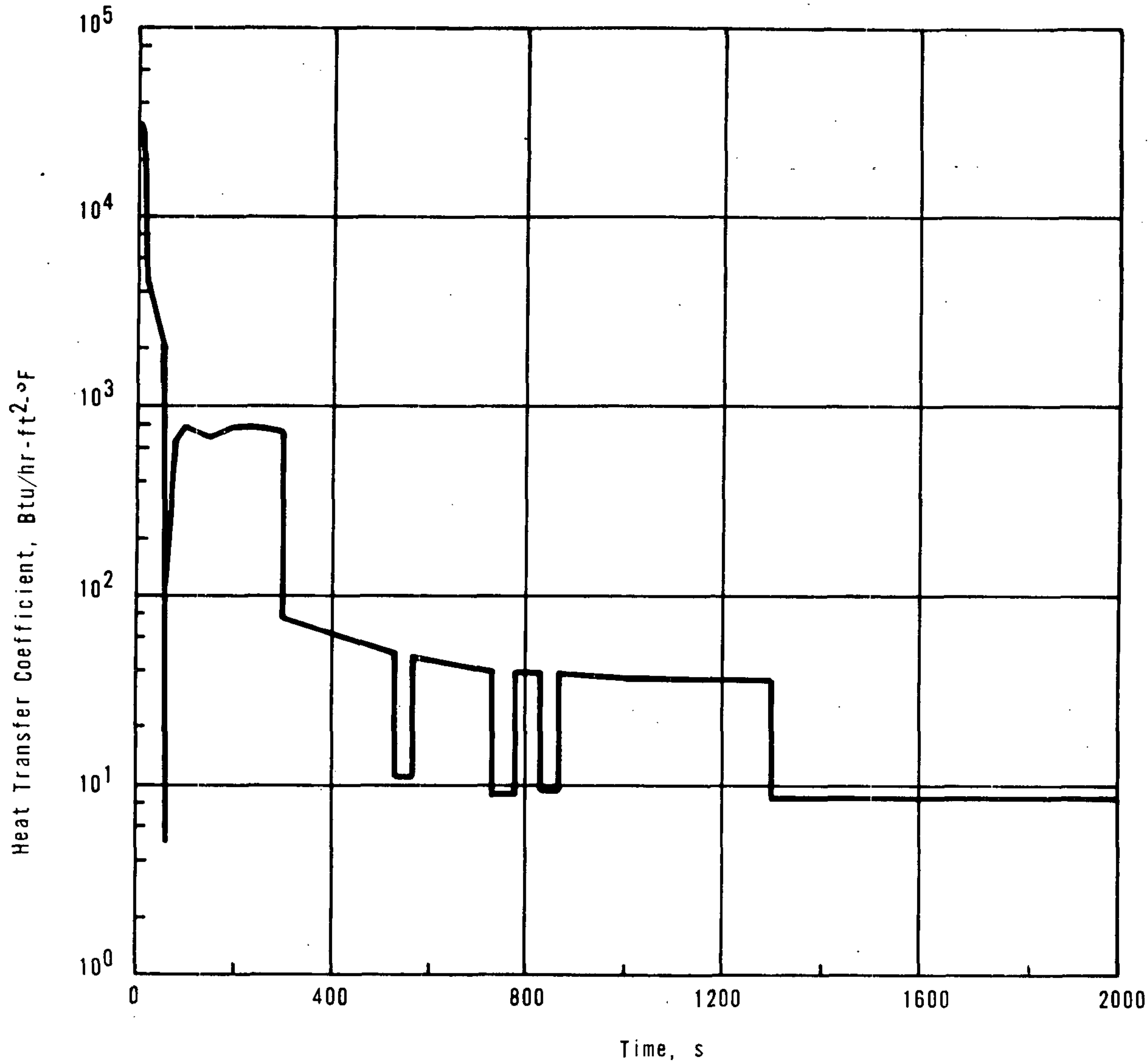
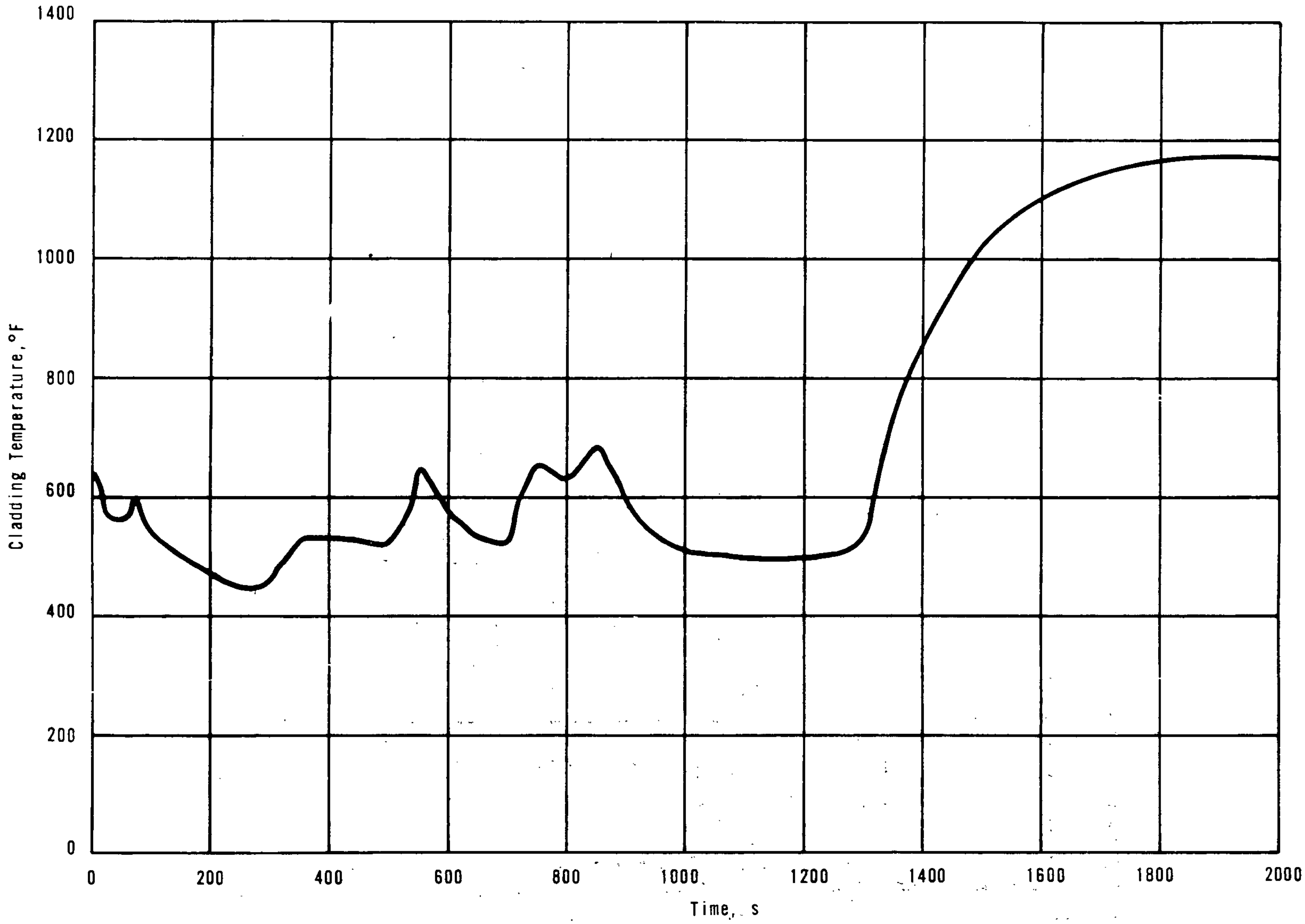


FIGURE 4-34 CASE 3, LEVEL 10 - CLADDING TEMPERATURE VERSUS TIME



Docket 50-269
FSAR Supplement 14
January 29, 1973

PART 2

EFFECT OF CORE FLOOD LINE RESTRICTOR ON ECCS FOR LARGE AND SMALL BREAKS

Previous analyses have shown that a marginal amount of water is maintained in the core following a core flooding tank line break. This is due to a partial degrading of the ECC System as a natural consequence of the accident. In order to improve the safety of the design, a flow limiting insert is to be placed in the core flooding tank nozzle to control the accident. By limiting the violence of the blowdown, less reactor coolant system water will be ejected from the primary system. Thus, more water will be maintained in the core during and after blowdown.

The acceptability of the insert is dependent on three points. First, it must be shown that it can be built and installed to function as designed. This has been addressed in Part 1. Second, it is necessary to show that the insert as designed will produce the desired results. This has been addressed in Part 2. Finally, it is necessary to show that the insert will not jeopardize the performance of the ECC System during other accidents. This is addressed in this Part 3.

By standard methods of analysis, a k-factor has been determined for the insert. The value of this factor is 0.2 based upon an area of 0.7213 ft^2 and is used to solve for the pressure loss Δp in the following equation:

$$\Delta p = \frac{k|W|W}{288\ \rho g\ A^2}$$

where

W = flow rate, Lbm/sec

ρ = density, Lbm/ft^3

g = gravitional constant, $\frac{32\ \text{Lbm ft}}{\text{Lbf sec}^2}$

A = Area, ft^2

The k-factor for the core flooding line resistance used by B&W in the evaluation of LOCA's presented in BAW-10034 was 6.3. This value is typical of all plants of this type. The proposed insert would increase the resistance by only 3%. To show that such an increase is acceptable, an analysis of the worst case large break, an 8.5 ft^2 cold leg split, has been carried out for two different k-factors. The first value, k = 8.3, has an increase (6.3 + 2.0) an order of magnitude higher than the proposed insert. This was chosen before the insert had been designed in order to bound the result. The second value, k = 5.5, is based on an experimental measurement of the line k-factor without the insert, k = 4.8, plus a conservative evaluation of the insert effect, k insert = 0.7. With a more concrete design and a better evaluation of the effect of the insert, we expect the actual line resistance to be k = 4.8, plus k insert = 0.2 or k = 5.0.

The results of the two different k-factors are shown in Figures 1 and 2. Figure 1 shows the core flooding tank injection rate for two tanks. Figure 2 shows the resulting peak cladding temperatures. Both temperatures are acceptable and are within the AEC Interim Acceptance Criteria. These results show that there is no adverse effect of the insert for large breaks.

For small breaks, the core flooding tanks provide water at a very slow rate. Thus, the important parameter in the core flooding tank system is its pressure volume relationship and not line resistance. An increase of only 3% in CFT line resistance would have no effect on small breaks.

FIGURE 1 CORE FLOODING TANK INJECTION RATE DURING AN 8.5 FT^2 COLD LEG LOCA FOR VARIOUS FLOW RESISTANCES

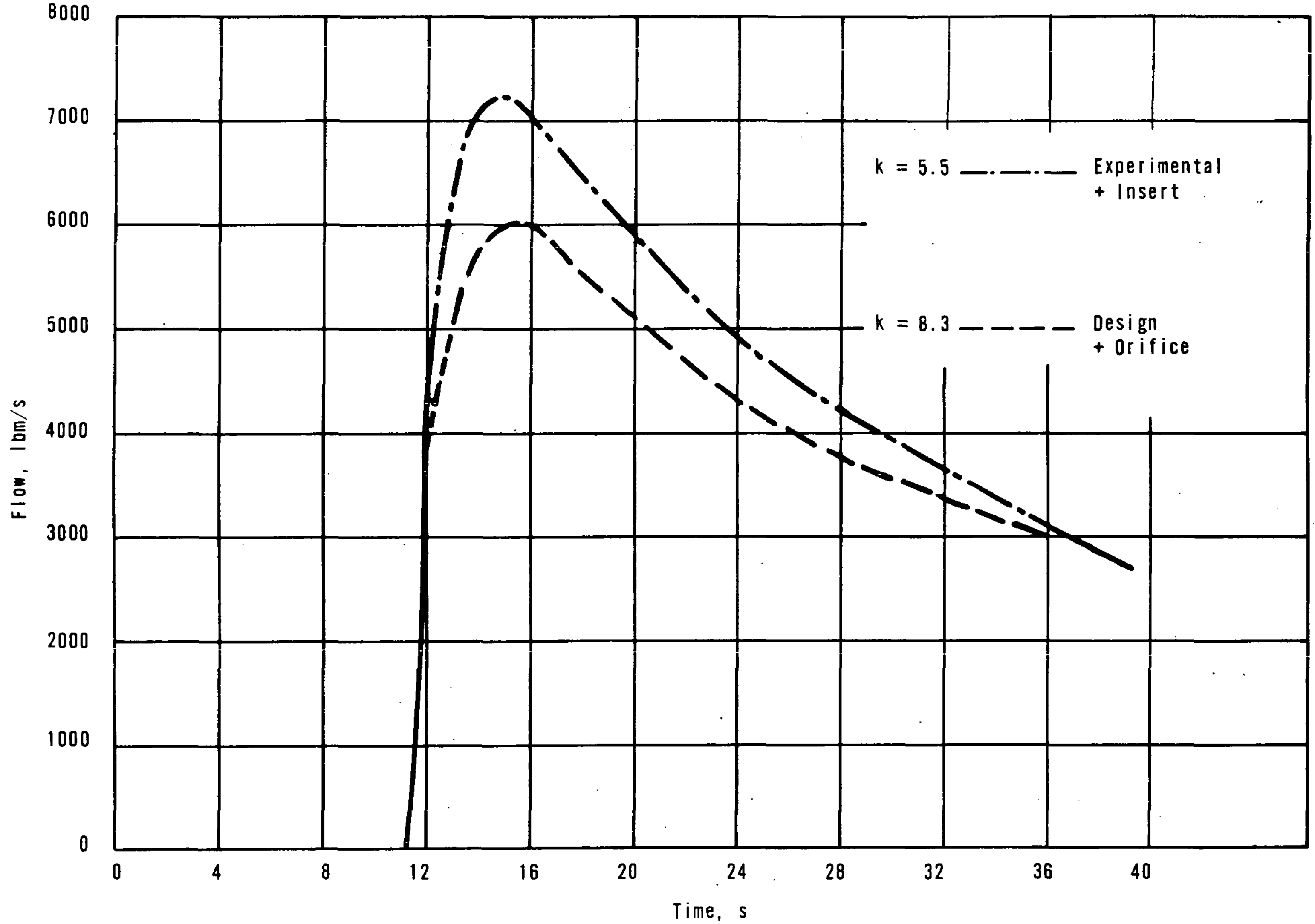
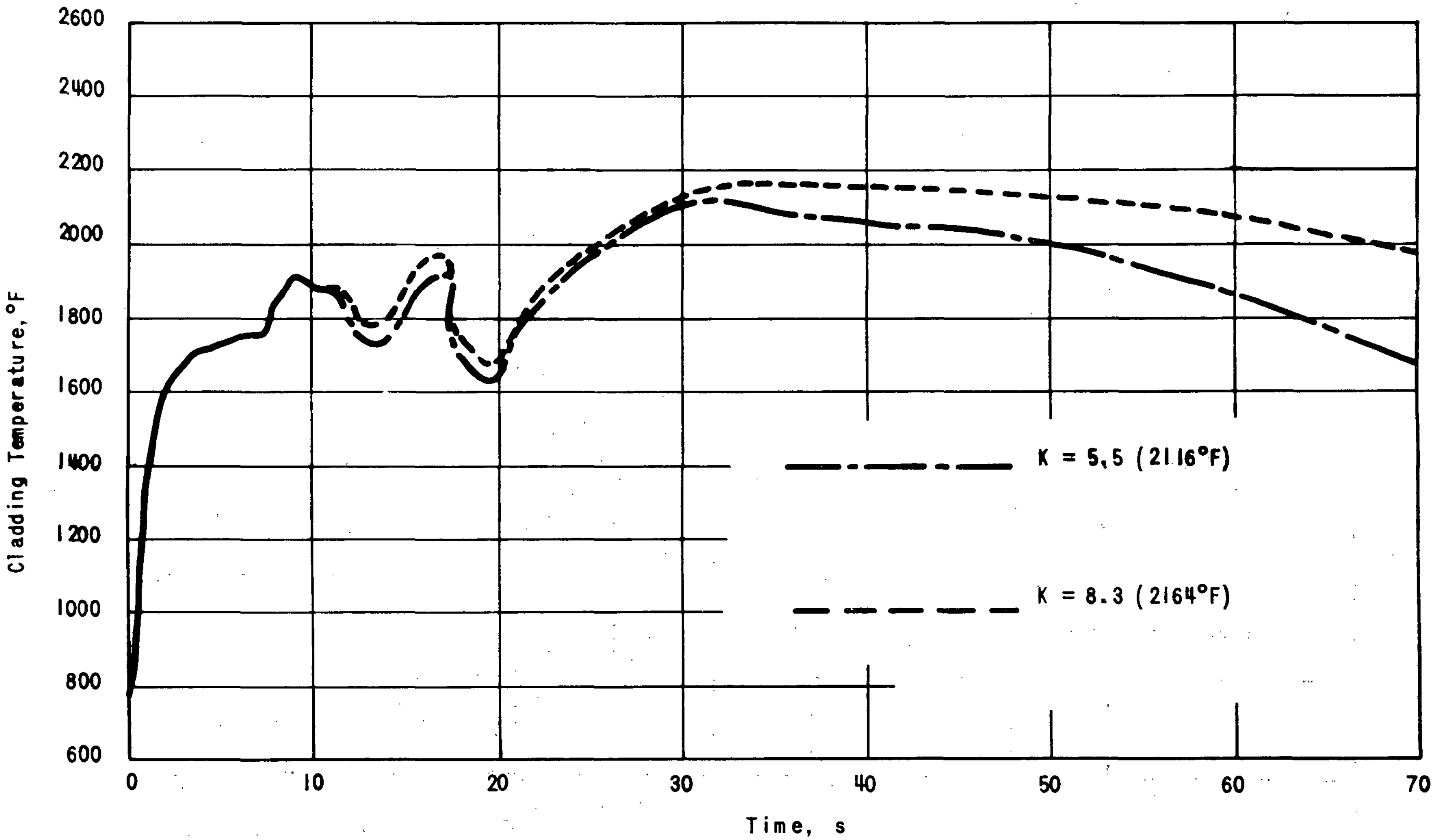


FIGURE 2 HOT SPOT CLADDING TEMPERATURE DURING AN 8.5 FT^2 COLD LEG LOCA FOR VARIOUS FLOW RESISTANCES IN THE CORE FLOOD TANK LINE



PART 3

DESCRIPTION AND MECHANICAL DESIGN OF THE RESTRICTOR

The core flooding nozzle modification for the Oconee 1 nuclear plant is basically a variable diameter thermal sleeve which is slotted into 45^o segments at its thinner end.

The existing core flooding nozzle and thermal sleeve are shown in Figure 1. The existing sleeve has been removed to enable installation of the modified sleeve.

Fabrication

The restrictor was fabricated in two stages as shown on Figure 2. Because of schedule restraints related to installing this modification at Oconee, it was necessary to use available material. Thus, Type 304 stainless steel pipe was used and weld overlay was deposited to meet the required nine inch I.D. The pipe and weld overlay were U.T. examined before machining with P.T. examination after final machining. These examinations were performed in accordance with ASME Section III.

Field Installation

Removal of the existing sleeve was accomplished by grinding the weld buttons which hold it in place and performing a P.T. examination on the ground areas (Figure 3). Installing the restrictor is then accomplished by welding and machining the rings as shown in Figure 4. The restrictor is inserted and a full penetration weld with permanent backing ring is made in accordance with ASME Section III (Figure 5). A progressive P.T. is performed to insure a quality weld. The weld ring centers the restrictor and controls vibration.
27. The slots are machined into the restrictor to allow additional flexibility where the sleeve is attached to the I.D. of the nozzle. Also, the slot on the bottom vertical axis will allow a small flow of water behind the restrictor to prevent crud buildup.

Design and Stress Analysis

The analysis of the sleeve is concerned with two major criteria: (1) Acceptability of the design from a fatigue standpoint, and (2) acceptability of the design for pressure thrust loads resulting from either a primary pipe LOCA or a core flood line LOCA.

The critical thrust load occurs for a core flood line LOCA where an instantaneous pressure differential of 2250 psi is assumed to occur. This results in a load of $(2)\ (2250)\ (\pi/4)\ (12.25^2 - 9^2) = 244.1$ kips. The factor of 2 is a dynamic factor for instantaneous loading. The critical shear section is at the weld, which is $\pi(.75)\ (11.73) = 27.6$ in.^{2}. Thus, the average shear across the section is 8.84 ksi. Using the normal shear criteria of ASME Section III of .6 S_m, the allowable stress is 9.18 ksi based on the S_m of TP304 stainless steel at 650^o of 15.3 ksi; however, it should be noted that this is actually a faulted condition. The critical tensile area just above the weld in the sleeve is $(\pi/4)[(11.73)^2 - (10.875)^2] - 8(3/8) = 12.26$ in.^{2}. The direct tensile stress is 21.7 ksi. For a faulted condition, the Code allowable for direct tension is 1.5 S_m - 22.9 ksi.

During the core flooding transient, the maximum Δp across the nozzle is expected to be approximately 200 psi. This is a factor of greater than 20 less than the design loading assumptions. Therefore, it is not considered credible that the restrictor retaining weld would fail during core flooding tank discharge.

During operation of the decay heat system, the Δp loads on the restrictor are insignificant.

The analysis of the restrictor and attachment weld for fatigue considerations is a lengthy analysis which will be incorporated into the stress report. The analysis procedure and results only will be given in this report.

The temperature distribution was determined by a two dimensional finite difference program that solves time and space dependent heat balance equations. The nozzle was originally analyzed ignoring the original thermal sleeve. A comparison of nozzle temperature distribution including the restrictor and original distribution shows only minor differences. Therefore, a reanalysis of the resulting stresses in the nozzle was not necessary and the effort was concentrated on the analysis of the restrictor and weld attachment of the restrictor.

A temperature distribution was obtained for heatup, a composite temperature excursion based on various power level changes, and cooldown including the core flooding test transient and decay heat removal initiation. The highest thermal stresses resulted from the core flooding test transient. Thermal and pressure motions for the cylindrical portion of the restrictor were obtained. The thermal motions for the opening in the shell were obtained by a computer program which is a flat plate theory nodel finite difference program. The pressure motions for the opening were obtained by calculating the nominal hoop stress in the vessel, multiplying by 2.5 as a correction for maximum hoop stress around a hole in a two dimensional stress field, and then using the cylindrical strain equation for a uniaxial hoop stress field; $\sigma_H = E\Delta/R$ or $\Delta = R\sigma_H/E$. The pressure rotation was assumed to be 0. The beam thermal motions were hand calculated based on a linear radial distribution equivalent to the actual distribution. An interaction analysis was then performed between the restrictor cylinder, beam slot section, and the weld attachment assuming a rigid weld attachment.

The redundants obtained were then used in hand calculation to calculate stresses in the weld attachment and slotted beam sections. The loads were also input to a computer program to obtain stress results on the cylindrical sleeve portion.

LIST OF FIGURES

FIGURE 1 EXISTING CORE FLOODING NOZZLE SLEEVE

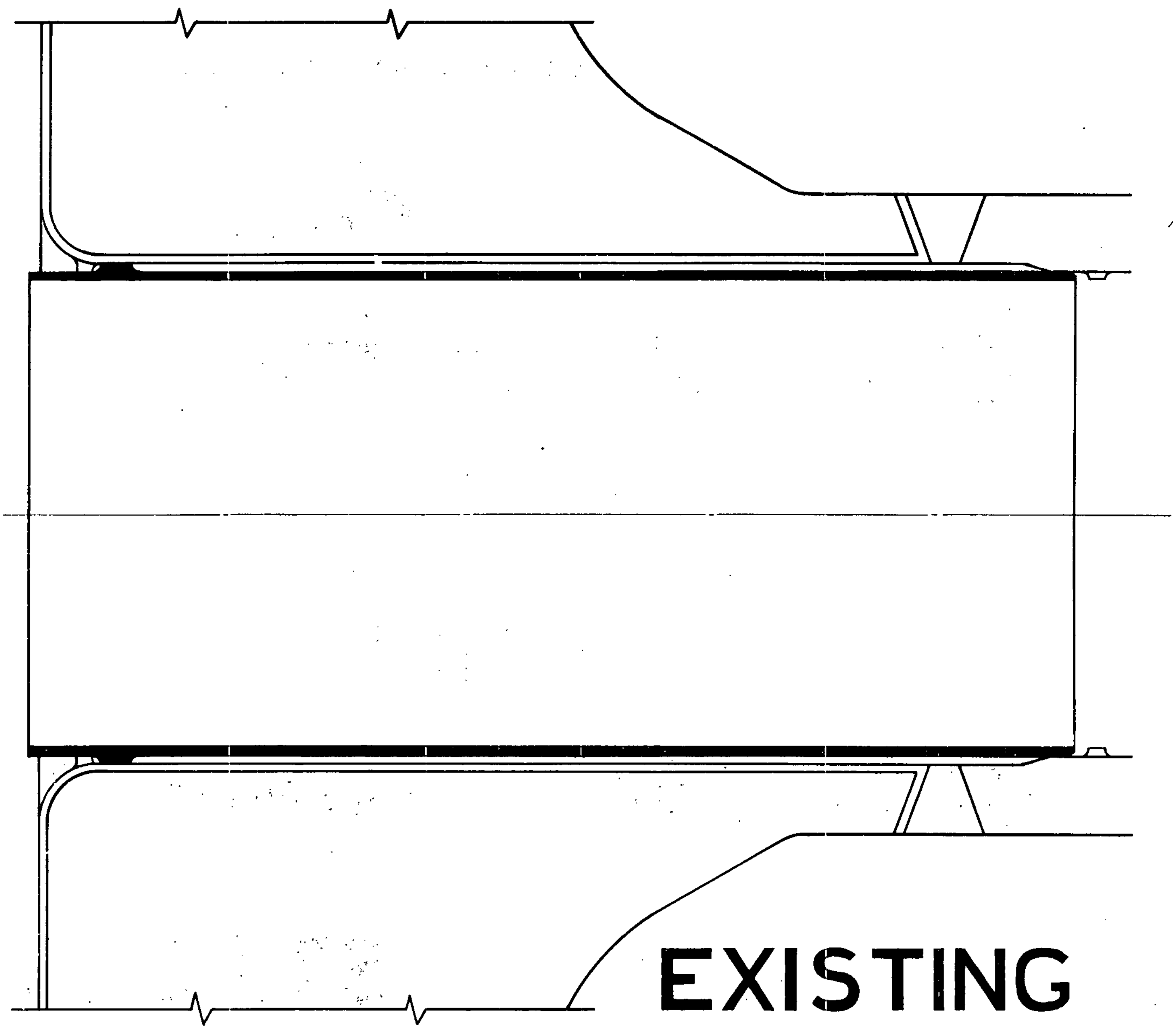


FIGURE 2 OCONEE I CORE FLOODING NOZZLE INSERT

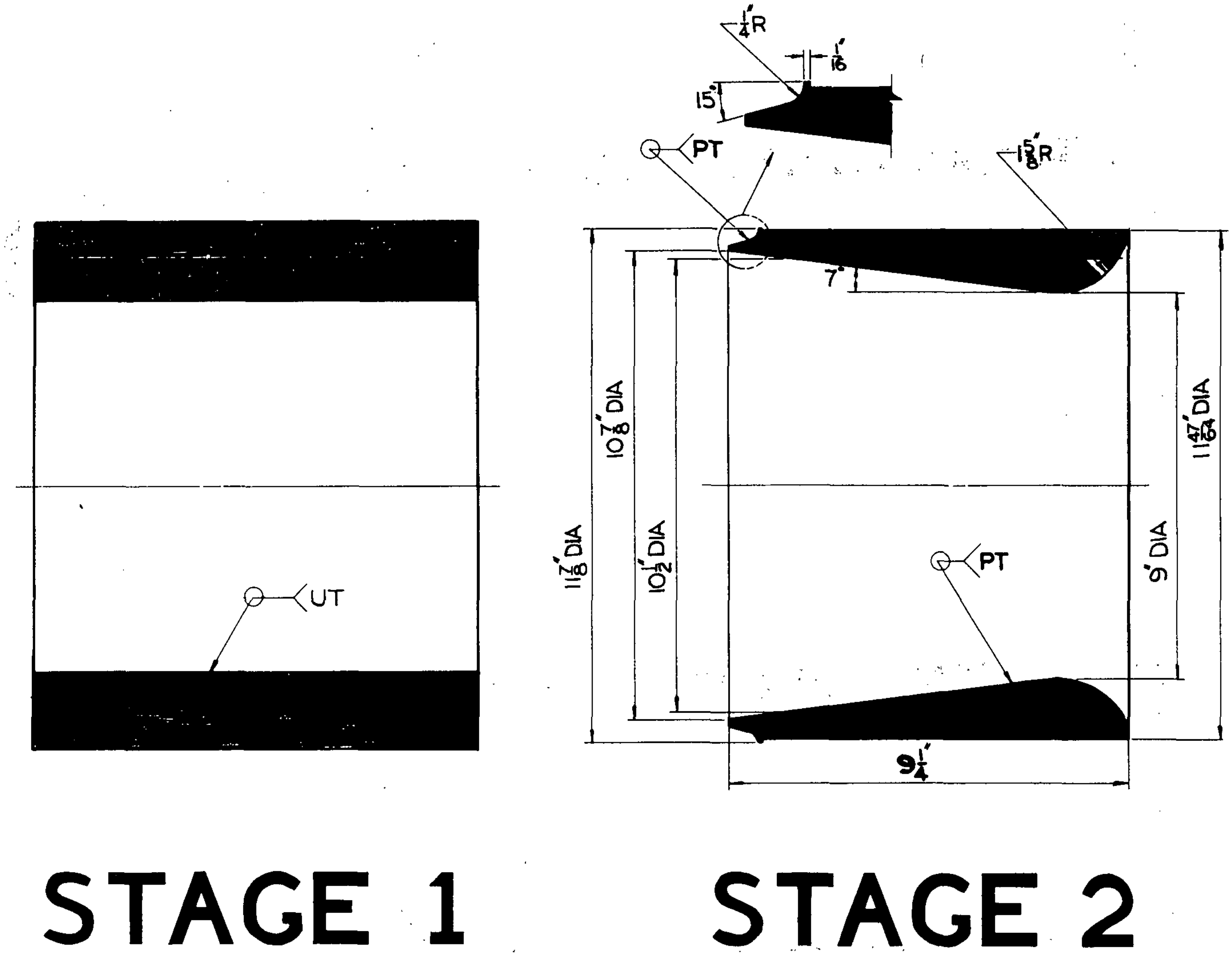


FIGURE 3 CORE FLOODING NOZZLE AFTER REMOVING EXISTING SLEEVE

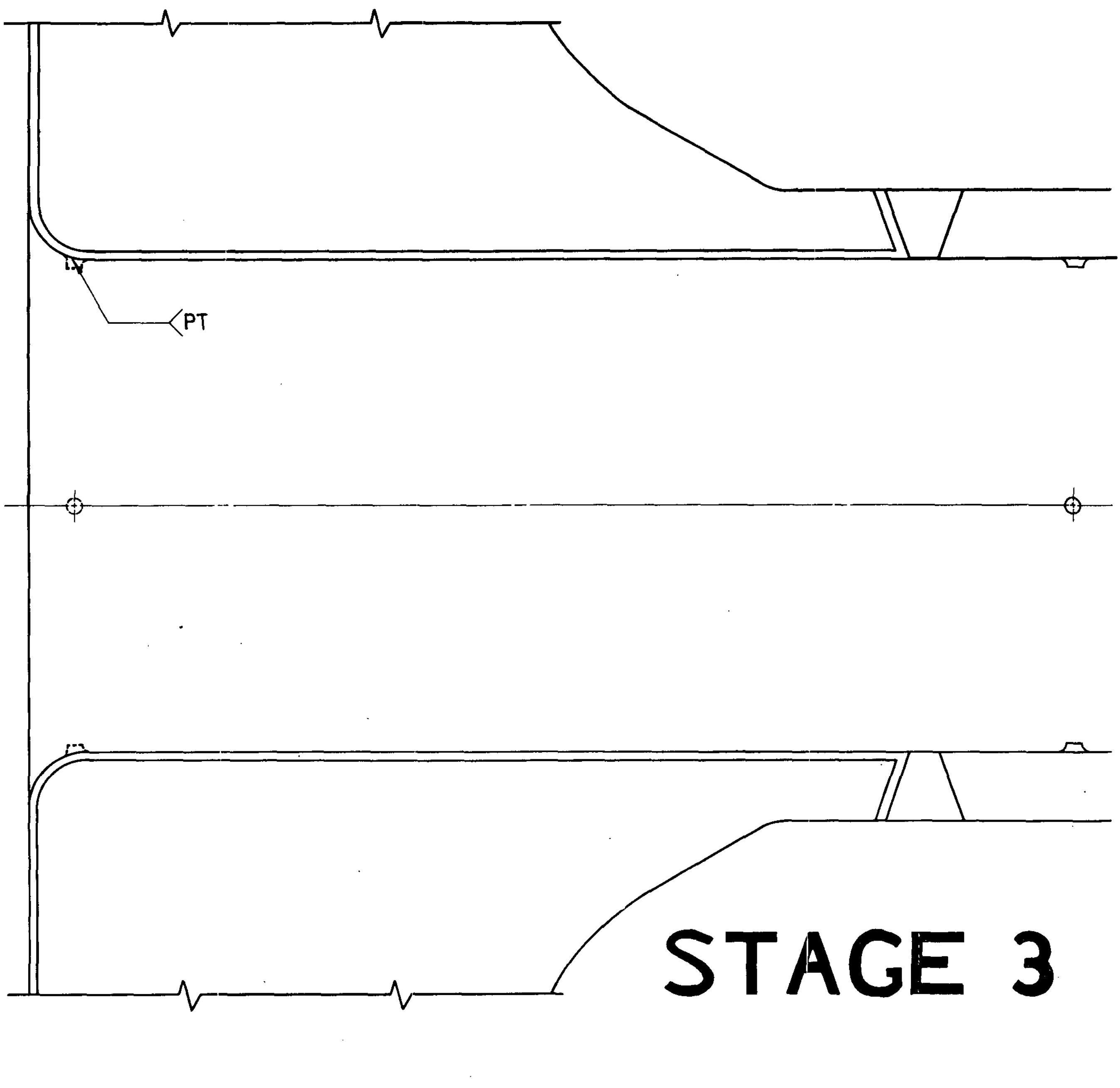


FIGURE 4 CORE FLOODING NOZZLE WELD PREPARATION PRIOR TO INSERTING MODIFIED INSERT

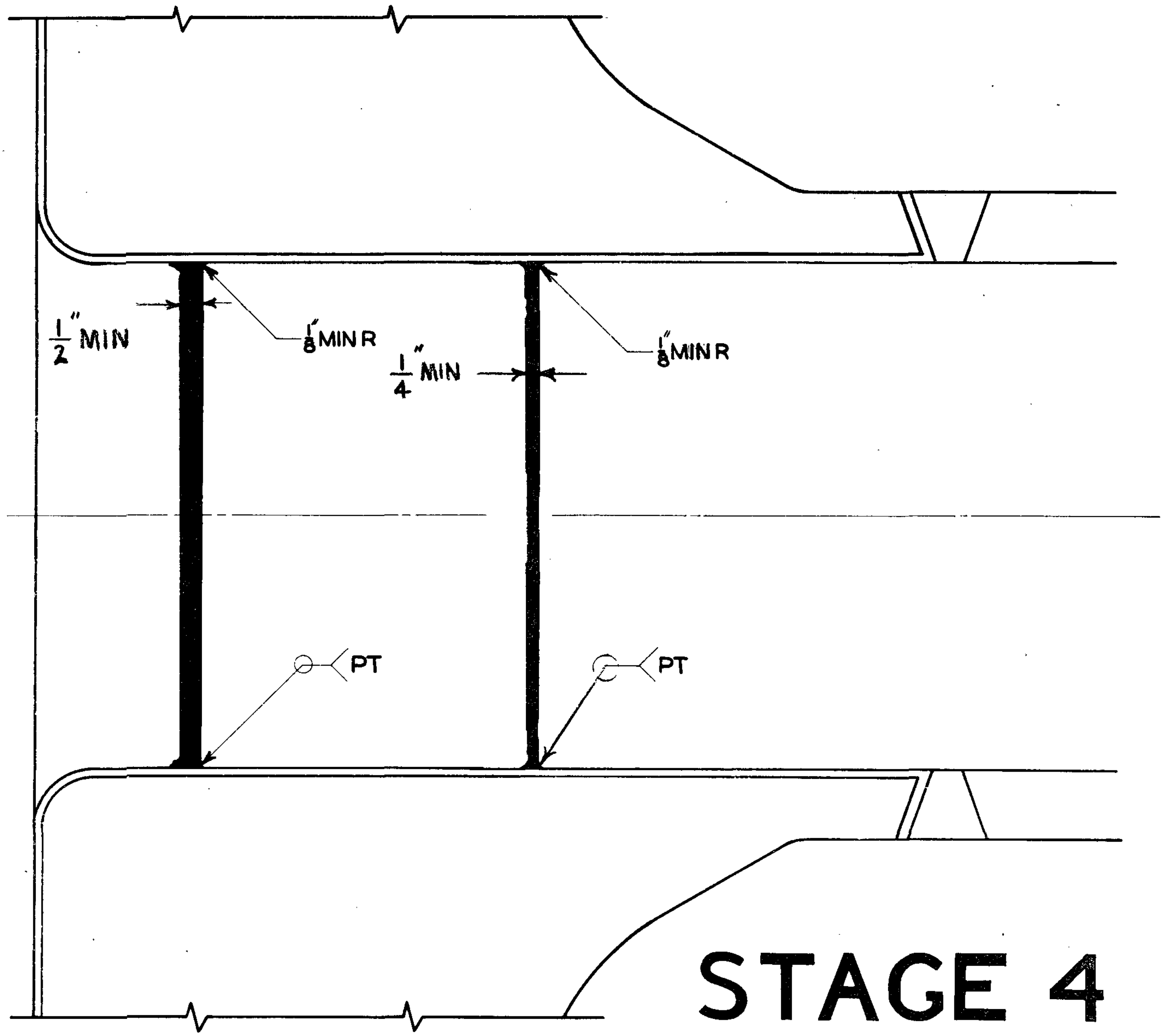
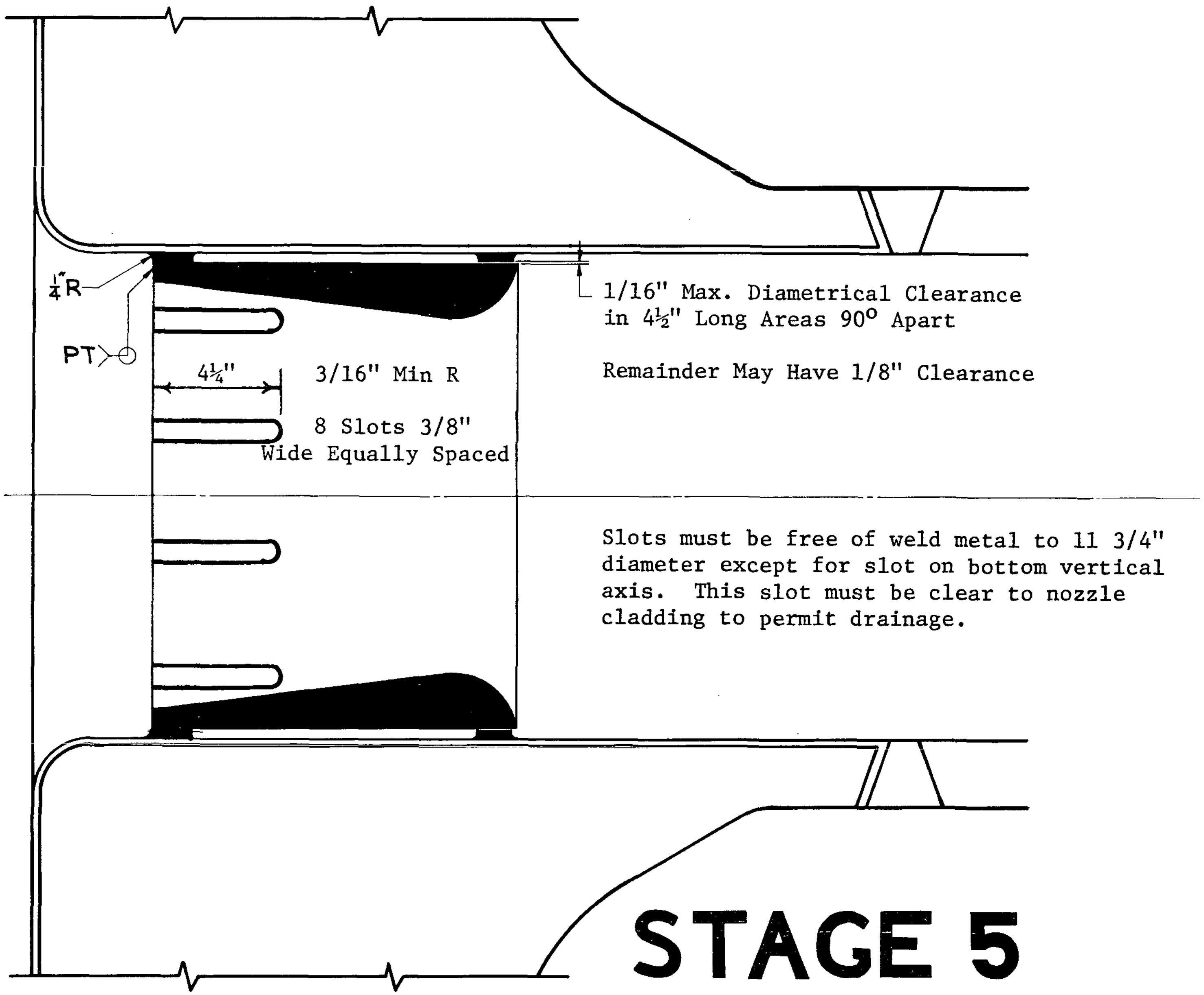


FIGURE 5   CORE FLOODING NOZZLE INSERT INSTALLED IN CORE FLOODING NOZZLE



DUKE POWER COMPANY

OCONEE NUCLEAR STATION

UNITS 2 AND 3

APPLICATION FOR LICENSES

Dockets 50-270 and -287

FSAR SUPPLEMENT 15

OCONEE UNITS 2 AND 3

ACTIVE VALVE OPERATILITY

Submitted with FSAR Revision 29

June 29, 1973

LIST OF EFFECTIVE PAGES
FSAR SUPPLEMENT 15

INTRODUCTION

Supplement 15 to the Final Safety Analysis Report contains the response to Mr. R. C. DeYoung's letter of January 2, 1973 which requested additional information which confirms the operability of active valves. The information presented in this supplement is the same as that presented in a letter to Mr. DeYoung on May 1, 1973.

OCONEE UNITS 2 AND 3
ACTIVE VALVE OPERABILITY

In each of Oconee Units 2 and 3, there are seven valves that meet the definition of being active and also part of the reactor coolant pressure boundary in accordance with 10CFR50. These valves are required to actuate upon an engineered safeguards signal and to either isolate the reactor building or to open an engineered safeguard system flow path. These valves and their design conditions are listed in Table I. Actual system operating conditions are significantly less severe than design conditions, as shown in Table I. A summary of these valves follows:

Mark Number	Service	Size	Qty.	System	Actuator
HP-V2A&B	Letdown Cooler Isolation	2½"	2	High Pressure Injection	Limitorque EMO
HP-V3	Letdown Isolation	2½"	1	High Pressure Injection	Sheffer Pneumatic
HP-V24A & B	High Pressure Injection Isolation	4"	2	High Pressure Injection	Limitorque EMO
LP-V4A&B	Low Pressure Injection Isolation	10"	2	Low Pressure Injection	Limitorque EMO

All of these valves receive extensive preoperational testing prior to initial fuel loading. The electric motor operators are all of the Limitorque SMB series and have a history of qualification testing to verify their reliability and operability. Extensive testing has been carried out by Limitorque and by an independent institute. The testing has been done over a number of years, and the most recent testing in the summer of 1972 bears out the operability confirmed initially.

It is also noted that the insulation on HP-V24A&B and LP-V4A&B EMO motors is Class B rather than the Class H insulation which is used on EMO's inside the reactor building and on the EMO's on which the tests were run.

One valve with a pneumatic cylinder operator is noted above. Analyses have
31. been performed assuming a 5g horizontal and 5g vertical seismic loading with the resultant acting in a direction to maximize deflection. It was shown that under these loading conditions, binding between the operator piston and cylinder tube or between the piston rod and the cylinder busing will not occur. The natural frequency of the cylinder structure has been analyzed and found acceptable.

1. ELECTRIC MOTOR OPERATOR QUALIFICATION TESTING

A. Shock and Vibration Testing

In August, 1970, a Limitorque SMB series operator was mounted on a test stand having a threaded valve stem driven by the operator simulating opening and closing a valve. The operator was electrically connected to stop at the full close position by means of a torque switch and stop at the full open position by means of a geared limit switch. The operator had a four-train geared limit switch installed and all contacts not being used for motor control were wired to electric indicating lights at a remote panel.

The unit successfully completed a 5.3g shock level at 32 Hz with no discrepancies noted. An exploratory scan of 5 Hz to 35 Hz was made and no critical resonant frequencies were noted on the operator. The unit was shocked and vibrated in each of three different axes a total of two minutes on, one minute off, three times per axis. The unit was operated electrically to both the full open and full close position and all torque switches and limit switches functioned properly. None of the auxiliary limit switches wired to indicating lights ever flickered or indicated they were opening or flickering. All electrical and mechanical devices on the operator performed successfully.

B. Heat Testing

In January, 1969, a completely assembled and operational SMB series operator was placed in an oven where the temperature was maintained at approximately 325°F for a duration of 12 hours. The unit was electrically operated every thirty minutes for a period of approximately two minutes per cycle and the geared limit switches were used to stop the actuator at the full open and full closed position of travel. Indicating light circuits were also wired to the geared limit switches.

The test was successful in every respect. There were no malfunctions of the operator and upon inspection of the component parts used, there was no noticeable deterioration or wear.

C. Live Steam Testing

In January, 1969, a complete SMB series operator was set up for electrical operation and live steam was piped into the conduit taps on the top of the limit switch compartment. One of the bottom conduit taps was left open to drain off any condensate. The operator was set on a timer basis for operation every thirty minutes for two minutes per cycle over a period of approximately nine hours. During this test, the live steam in the switch compartment had no effect on the function of the limit switches in their control of the operator at the full open and full closed position of travel. In addition, the limit switches were wired to indicating lights which operated satisfactorily.

The test was successful and there was no noticeable effect on the function of any of the parts in the limit switch comparment.

D. Life Cycle Testing

In January 1969, the operator was mounted on a stand inside a test chamber and a 150 cycle load test was made on the unit. This test cycle consisted of stroking a 2-3/8" diameter valve stem at a speed of 6 inches per minutes for a total of approximately 12 inches in two minutes. The valve stem in the full closed position produced a thrust of 16,500 pounds on a rigid plate securely bolted to the test chamber. The unit was wired so that the open position geared limit switch stopped the unit in the full open position.

After the life cycle testing was completed, the unit was inspected and found to be in excellent condition. There was no noticeable wear on any of the parts.

E. Simulated Accident Environment Testing

In November, 1968, an electric motor operator was tested under conditions which simulated the temperature, humidity and chemical environments that could be expected in the containment following some postulated accident such as the rupture of a major reactor coolant pipe.

The operator was placed in an Autoclave type chamber and subjected to 90 psig saturated steam. At specified intervals, the operator was cycled to assure proper operation. Forty minutes after the introduction of steam, a 1.5% boric acid solution was sprayed on the operator assembly. The operator continued to operate satisfactorily. Later, the steam pressure was periodically reduced to simulate post accident conditions. The boric acid spray was allowed to continue for four hours. The steam pressure was eventually reduced to 15 psig. The test continued for seven days.

During this time, the operation of the operator became erratic. The corrosive effects of the steam and boric acid spray caused electrical contact malfunctions which were bypassed by the use of an appropriate jumper. The valve continued to cycle during the seven day period.

A design change was made to the limit switch in order to correct the erratic operation, and it was tested under similar accident conditions and found to operate satisfactorily. This design change has been incorporated into all subsequent applicable models of this operator.

2. RECENT TESTING

More recent tests on Limitorque SMB series operator were conducted during the summer of 1972 by the Franklin Institute Research Laboratories.* In these tests an operator was exposed to gamma radiation (200 megarads), a steam/chemical environment (for twelve days), a steam environment at temperatures as high as 340^{o}F during the first day (test consisted of a 30 day exposure) and a seismic test similar to those conducted in August, 1970. During all of these tests, the operator was periodically cycled and was found to operate satisfactorily.

3. VALVE PURCHASE SPECIFICATION

In addition to a proven record as verified by the previous testing, the valve vendor must also comply with the purchase specification requirements. The purchase specifications for these seven valves require that they be hydrostatically tested, leak tested and cycled between the extremes of fully opened or closed. The hydrostatic test is in accordance with the Standard for Steel Pipe Flanges and Flapped Fittings (USAS B16.5). The leak test requires that with the disc closed tight, hydrostatic pressure shall be applied alternately on each side of the closed disc with the side opposite the pressure open for inspection. Acceptance criteria require that valves not show a leakage greater than 10 cubic centimeters per hour per inch of seat diameter, or permanent deformation when the valves are subjected to two times design pressure, except that the stress developed at test pressure shall not exceed 90% of the specified minimum yield strength based on the minimum specified wall thickness.

Valve vendors have submitted generic calculations to B&W which show that when similar valve assemblies are subjected to a 3g horizontal force and to a 2g vertical force, the stresses incurred are within the code allowable stresses. These calculations also verified that the first natural frequency is above 20 Hz for these valves.

4. PREOPERATIONAL TESTING

The testing procedures for valves that require operation to meet engineered safeguards requirements are quite extensive during the preoperational testing program. These tests demonstrate proper installation, strength and functional performance of valves. Subsequent to satisfactory preoperational testing, surveillance testing requirements have been

*[1]Qualification test of Limitorque valve operator, motor brake, and other units in a simulated reactor containment post-accident environment, Final Report F-C3327, July, 1972.

[2]Qualification test of Limitorque valve operators in a simulated reactor containment post-accident steam environment, Final Report F-C3441, September 1972.

established to assure continued satisfactory operation of these valves. Furthermore, if maintenance or repair of these valves is required, appropriate functional testing will be accomplished to assure proper operation subsequent to the maintenance or repair.

A. System Electrical Test

1. Electric Motor Operated Valves

The purpose of these tests is to verify electrical characteristics of valve operators in performing their function. Preliminary checkout of the operator valve assembly requires that the valve be free to move and that if the motor operated valve travels in the wrong direction from its mid-travel position, its breaker must be tripped immediately as there would be no torque limit protection. The valve can be operated manually with a handwheel to ascertain its freedom of movement.

The phase rotation of the operator is checked. During valve operation, verification that the valve travel and motor are stopped is done by closing the torque limit switch. Similarly, the opening of the valve is terminated by the opening of the limit switch.

2. Pneumatic Cylinder Operated Valves

The purpose of these tests is to verify proper operation of the piston operated valve and the solenoid controlling the air supply to the valve. Valves with handwheels are checked for freedom of movement prior to applying air to the pistons. Valve position limit switches are set during this check.

The valves are then operated using air supplied through the solenoids. Proper valve travel, solenoid, and limit switch operation is verified.

3. ES Test (Both EMO and Piston Valves)

In checking the valve for ES actuation, the valve is placed in the position opposite to its ES position and then an ES signal is simulated. The valve moves to its ES position. Then the control room switch is turned to the position opposite of ES operation and the valve is verified as remaining in its ES position. Similarly, turning the circuit breaker panel switch to the position opposite of ES operation has no effect on the valve.

Acceptance criteria for these electrical tests are:

(1) Valves must open, close, and travel in the proper direction in response to control and engineered safeguards signals.

(2) The valve open and closed indicating lights must indicate correctly.

(3) Valve electric motor operator resistance-to-ground readings must be within specification.

(4) The specified valve travel time is within specification requirements.

B. System Engineered Safeguards Test

The purpose of these tests is to demonstrate actual valve performance for its intended engineered safeguard use. Initially, all valves are placed in their non-ES position prior to simulating an ES signal. Upon initiation of an ES signal, the tests for the subject valve demonstrate containment isolation and also emergency injection flow capability to the reactor coolant system from the low pressure injection system and the high pressure injection system.

C. System Functional Testing

The purpose of this testing is to verify that the valves perform as intended for normal operation. Cycling the valves under conditions of specified differential pressure and/or flow that may be encountered during plant operation will verify that the valve operator does not exceed maximum cycle time.

D. Integrated ES Actuation Test

The purpose of this test, in which these valves are used, is to demonstrate the full operational sequence that would bring the emergency core cooling systems and the containment pressure reducing systems into action, including the transfer to alternate power sources.

General acceptance criteria for this test are:

1. The ES systems operate as described in the FSAR.

2. Upon actuation of an ES signal, high pressure and low pressure injection to the reactor coolant system are supplied in accordance with FSAR requirements.

3. Upon loss of normal station power, the ES systems continue to perform their designed functions without interruption.

Following completion of the preoperational test program and issuance of an operating license for the facility, these valves are functionally tested as required by the FSAR.

5. SYSTEM HYDROSTATIC TESTS

Fluid systems hydrostatic tests are performed on the various systems to assure leak tight installation of the valve in the piping system.

6. TECHNICAL SPECIFICATIONS

The technical specification testing requires that these valves be operated quarterly to assure their continued availability. During the life of the facility, these valves will be appropriately operated subsequent to any required maintenance, repair or replacement.

7. ACTUAL SEISMIC CONDITIONS

In summary, we want to emphasize the results of the dynamic seismic analysis for the specific piping systems in which these valves are located (Table II). The maximum acceleration of 1.05g indicated is considerably less than the maximum g force in either the horizontal or vertical direction that the seven valves are required to withstand. The entire scope of testing verifies valve operability from conditions of extreme duress to normal operation, and the results of the earliest environmental, vibratory, and load testing have been verified in later independent testing.

TABLE I

ACTIVE - REACTOR COOLANT PRESSURE BOUNDARY VALVES

Mark Number	System	Service	Size	Purchased By	System Valve Class	System Design Rating	System Cond During Opr	Type	Motor Operator Type	Valve Mfg.	Valve Movement
2HP-V2A 3HP-V2A	High Pressure Injection	Letdown Cooler Outlet	2½"	B&W	B	2500 psig 650°F	2170 psig 135°F 40-100 gpm	Globe	Limitorque SMB-00-15	Rockwell	Full Open to Full Close
2HP-V2B 3HP-V2B	High Pressure Injection	Letdown Cooler Outlet	2½"	B&W	B	2500 psig 650°F	2170 psig 135°F 40-140 gpm	Globe	Limitorque SMB-00-15	Rockwell	Full Open to Full Close
2HP-V3 3HP-V3	High Pressure Injection	Letdown Line RB Isolation	2½"	B&W	C	2500 psig 200°F	2170 psig 135°F 40-140 gpm	Globe	Sheffer Piston	Rockwell	Full Open to Full Close
2HP-V24A 3HP-V24A	High Pressure Injection	HP Inj RB Isolation	4"	B&W	B	3050 psig 200°F	2200-2950 psig 120-245°F 450 gpm	Globe	Limitorque SMB-1-25	Rockwell	Full Close to Full Open
2HP-V24B 3HP-V24B	High Pressure Injection	HP Inj RB Isolation	4"	B&W	B	3050 psig 200°F	2200-2950 psig 450 gpm 120-245°F	Globe	Limitorque SMB-1-25	Rockwell	Full Close to Full Open
2LP-V4A 2LP-V4B	Low Pressure Injection	LP Inj RB Isolation	10"	B&W	B	2500 psig 300°F	255 psig 280°F 3000 gpm	Gate	Limitorque SMB-4-150	Walworth	Full Close to Full Open
3LP-V4A 3LP-V4B	Low Pressure Injection	LP Inj RB Isolation	10"	B&W	B	2500 psig 300°F	255 psig 280°F 3000 gpm	Gate	Limitorque SMB-4-100	Velan	Full Close to Full Open

TABLE II
SEISMIC INFORMATION FOR R. B. ISOLATION VALVES
OCONEE NUCLEAR STATION
UNITS 2-3

Valve Body

Valve No.	Displacements (Inches) X + Y EQ. Dx	Dy	Dz	Displacements (Inches) Y + Z EQ. Dx	Dy	Dz	Accelerations (G) X + Y EQ. Ax	Ay	Az	Accelerations (G) Y + Z EQ. Ax	Ay	Az
2HP-V2A 3HP-V2A	Rigidly Mounted To Building											
2HP-V2B 3HP-V2B	Rigidly Mounted To Building											
2HP-V3 3HP-V3	Rigidly Mounted To Building											
2HP-V24A 3HP-V24A	.010	.024	.010	.006	.018	.006	.040	.340	.066	.028	.116	.052
2HP-V24B 3HP-V24B	.120	.001	.042	.040	.002	.014	1.050	.012	.376	.354	.034	.122
2LP-V4A 3LP-V4A	.094	.001	.050	.046	.001	.024	.288	.007	.154	.170	.012	.088
2LP-V4B 3LP-V4B	.058	.002	.036	.006	.002	.038	.272	.009	.160	.400	.010	.238

Operator

Valve No.	Displacements (Inches) X + Y EQ. Dx	Dy	Dz	Displacements (Inches) Y + Z EQ. Dx	Dy	Dz	Accelerations (G) X + Y EQ. Ax	Ay	Az	Accelerations (G) Y + Z EQ. Ax	Ay	Az
2HP-V2A 3HP-V2A												
2HP-V2B 3HP-V2B												
2HP-V3 3HP-V3												
2HP-V24A 3HP-V24A	.024	.042	.022	.014	.034	.014	.146	.600	.178	.114	.220	.146
2HP-V24B 3HP-V24B	.080	.006	.046	.026	.002	.014	.700	.114	.390	.236	.052	.134
2LP-V4A 3LP-V4A	.100	.001	.055	.050	.001	.028	.326	.008	.176	.224	.012	.124
2LP-V4B 3LP-V4B	.082	.024	.003	.088	.024	.002	.386	.198	.005	.560	.216	.006

Notes:
1. Dx = Seismic Displacement (x-Direction)
2. Ax = Seismic Acceleration (x-Direction)
3. X-Direction = North-South
4. Z-Direction = East-West
5. Values are Design Basis Earthquake

LIST OF EFFECTIVE PAGES
FSAR SUPPLEMENT 16

DUKE POWER COMPANY

OCONEE NUCLEAR STATION

UNIT 2

APPLICATION FOR LICENSES

Docket 50-270

FSAR SUPPLEMENT 16

FUEL ROD AND CLADDING STUDY IN OCONEE 2

Submitted with FSAR Revision 30

September 4, 1973

FUEL ROD AND CLADDING STUDY IN OCONEE 2

Introduction

The Babcock & Wilcox Company proposes to irradiate 48 fuel rods and four surveillance orifice rod assemblies (ORA) in Oconee II, Core 1. The fuel rods would be loaded into 24 peripheral fuel rod locations in two batch 2 fuel assemblies. The surveillance ORA's would be initially inserted into four batch 3 fuel assemblies (FA). This work is being conducted to: (1) provide additional information on the effect of pellet and cladding design variables on pellet-to-cladding mechanical interaction, (2) obtain data relevant to the fuel densification phenomenon, and (3) obtain additional data on Zr-4 cladding creep and irradiation growth rates. All fueled and unfueled rods (orifice rods) will be initially characterized before irradiation. They will then be subjected to nondestructive interim and post-irradiation examination in the Oconee spent fuel storage pool as well as destructive post-irradiation examination at the B&W hot cells in Lynchburg, Virginia.

Description

A. Fueled Rods

1. Twelve of the 24 fueled rods in each identical fuel assembly contain standard Oconee II batch 2 fuel pellets. These pellets are clad with ZR-4 tubing identical to the Oconee II cladding except for mechanical properties. Equal numbers of three different types of tubing are used. The minimum yield strength and ductility (elongation percentage) of these three cladding types are:

	Yield Strength (psi)	Ductility (percent)	Test Temperature (^{o}F)
Type 1	19,000	25	650
Type 2	23,000	14	750*
Type 3	38,000	10	750*
Standard Cladding	45,000	18	650

*These tubes were purchased based on acceptable mechanical property values at 750^{o}F. However, these properties will be determined and evaluated by B&W at 650^{o}F prior to fuel rod fabrication.

2. The other twelve fueled rods in each fuel assembly contain fuel pellets designed and fabricated to minimize pellet shrinkage (thermally stable) and fuel-cladding mechanical interaction. These pellets are manufactured using standard B&W QC and QA procedures. These fuel pellets are 2.66% U-235/U enriched while standard batch 2 fuel is 2.75% U-235/U. The effect of 12 rods of lower enrichment (2.66% w/o U-235/U versus 2.75 w/o) in each of the two FA's on changing the power peaking has been computed to be less than .04%. The fuel diameter, fuel-cladding gap, fuel rod fill gas volume, and internal pressurization are the same as in the standard fuel rods in the core.

   The four types of cladding used with these thermally stable pellets are identified as Type 1, 2, and 3 given above as well as the standard Oconee II fuel cladding.

   The presence of these 48 fuel rods ($\sim$.1% of the total number of rods in-core) represents a minute total effect on the reactor and its safe operation and will not alter the transient, steady-state, and accident modes of reactor operation.

B. Unfueled Rods - Surveillance Orifice Rod Assemblies

The four surveillance ORA's are identical to each other and each contains 16 pieces of cladding. Twelve of these pieces are prepressurized to produce compressive loads of 1300, 1600, and 1900 lbs. on the cladding circumference at reactor operating conditions. These pressurized rods contain solid Zr-4 internal supports to prevent cladding collapse; a cladding diametral creep to 0.030 inch is permitted. After the first fuel cycle at least one of the four surveillance ORA's would be moved from the core and at least one of the remaining three in-core assemblies would be moved into a higher flux region. At least one ORA will remain in-core through three complete fuel cycles.

The presence of the four surveillance orifice rod assemblies will not alter the core nuclear, thermal and hydraulic characteristics and they pose no safety threat during transient, steady-state or accident conditions.

LIST OF EFFECTIVE PAGES
FSAR SUPPLEMENT 17

DUKE POWER COMPANY

OCONEE NUCLEAR STATION

APPLICATION FOR LICENSES

Docket Nos. 50-269, -270, and -287

FSAR SUPPLEMENT 17

FLUX/FLOW TRIP ANALYSIS

Submitted with FSAR Revision 31
February 15, 1974

FLUX/FLOW TRIP ANALYSIS

BACKGROUND

Reactor coolant flow measurements are obtained by pressure drop taps installed in the hot leg of all 177 fuel assembly plants. Oconee 1 test data have shown that the average ΔP measurement produces an accurate indication of the reactor coolant flow. However, the "As-Built" sensing string has a measured time constant of 1.4 seconds versus the 0.65 seconds assumed in the FSAR. It is the purpose of this report to demonstrate that the calculated value of the flux/flow trip ratio is still conservative and adequate.

THERMAL-HYDRAULIC METHODS

To determine the flux/flow trip setpoint that is necessary to meet the hot-channel DNB ratio criteria, several calculational steps are required. These steps involve such things as the determination of steady-state operating conditions, fuel densification effects and transient calculations.

A. Thermal-Hydraulic Conditions During Normal Operation

The hot channel thermal hydraulic conditions are calculated for design conditions at 108 percent of rated power. The power level of 108 percent includes operation at 102 percent of rated power plus a maximum power level measurement error of 6 percent (4 percent neutron flux error and 2 percent heat balance error). This serves as the benchmark calculation from which the densification penalty and the transient effects can be determined. The steady-state analysis is performed using the TEMP computer code (BAW-10021)(1) with the appropriate hot channel factors, coolant inlet temperature and system pressure errors, and a 5 percent hot assembly flow maldistribution factor applied. These conservatisms are consistent with the calculational techniques employed in the FSAR analyses. The design flow rate of 131.32×10^6 #/hr. was used. The hot assembly power distribution consisted of a 1.78 radial-local nuclear factor ($F\,\Delta H$) with a 1.5 cosine axial flux shape. The primary output of the calculation is the minimum hot channel DNB ratio as calculated by either the W-3 or the BAW-2 (BAW-10000)(2) correlations.

B. Densification Effects

The fuel densification penalty applied to the hot channel was determined by the methods discussed in the Oconee 2 Fuel Densification Report, BAW-1395, June 1973, page A-5. A conservative slumped and spiked 1.83 outlet peaked axial power shape was used in conjunction with a 1.49 radial-local factor to determine the maximum fuel densification effect on DNB ratio. This reduced hot channel DNB ratio is the basis for establishing the initial conditions for the transient calculations.

C. Transient Hot Channel Conditions During a Loss of Flow

The flux/flow trip setpoint is derived to protect the core during a two pump coastdown. A two pump coastdown is analyzed because it is assumed that one pump monitor is inoperative (single failure assumption) and would not trip the reactor on the loss of two pumps (the pump monitors are set such that the reactor will not trip on a pump monitor signal as the result of the loss of a single pump).

The thermal-hydraulic response of the hot channel is calculated by RADAR computer code (BAW-10069)[3]. The initial hot channel DNB ratio is set equal to the steady-state value with densification effects included. The time zero DNB ratio is then further reduced to account for the possibility of a vent valve being stuck open. This conservatism reduces the effective core flow by 4.6 percent. The RADAR output in the form of Hot Channel DNB ratio versus time is the basis for establishing the flux/flow ratio trip setpoint.

PROCEDURE FOR DETERMINING FLUX/FLOW SETPOINT

The determination of the flux/flow setpoint is accomplished in four basic steps. The result of these steps is designed to yield a value of the flux/flow ratio that will prevent the minimum Hot Channel DNBR from going below 1.3 for the coastdown for which protection is required. These steps are as follows:

A. Total Time Determination

From a plot of minimum DNBR versus time find the time that yields a DNBR of 1.3 for the maximum power level (108 percent) for the maximum number of pumps lost for which the flux/flow trip must provide protection (two pumps for Oconee).

B. Coasting Time Determination

The total time to reach a DNBR of 1.3 minus a conservative value of the total trip delay time gives the maximum allowable coasting time prior to trip initiation.

C. Minimum Flow Determination

From a plot of flow versus time for the coastdown of interest, the percent flow for the maximum allowable coasting time is found. This yields the flow at which trip must be initiated.

D. Flux/Flow Ratio Calculation

The maximum allowable flux/flow ratio is the maximum real power level of interest (108 percent) minus the power level measurement error (6 percent) divided by the minimum flow.

CALCULATIONAL RESULTS

Figure 1 shows the flow versus time that is the design basis for the determination of the flux/flow ratio. Figure 2 shows the results of the calculation of DNBR versus time for the undensified Oconee fuel using the W-3

CHF correlation. From Figure 2 it is seen that a DNBR of 1.3 occurs at about 2.30 seconds. Using the design trip delay time as 0.65 seconds, the maximum coasting time is 1.65 seconds. From Figure 1, the flow at 1.65 seconds is slightly less than 93 percent. The flux/flow ratio as determined in the original Oconee Technical Specifications is thus (1.08-0.06)/0.93 or 1.10.

Figure 3 shows the calculational DNBR versus time with the effects of densification included. From this figure it is seen that a DNBR of 1.3 is reached at about 2.15 seconds using the technique explained previously; this yields a flux/flow ratio of 1.08. This is the value presented in the FSAR Technical Specifications for densified fuel for Oconee 1. Figures 4, 5, and 6 show DNBR versus time for an Oconee two-pump coastdown using B&W's standard analysis methods which includes the use of the BAW-2[2] correlation. From Figure 4 using the previous method and a trip delay time of 1.40 seconds rather than 0.65 seconds it is seen that a flux/flow ratio of 1.10 is obtained. Similarly a flux/flow ratio of 1.09 is obtained for Oconee II using the data shown in Figure 5. For Oconee 3, due to a slightly modified control rod drive system, the appropriate trip delay time is 1.50 seconds. Using this value of delay time and the data presented in Figure 6, the calculated flux/flow ratio is 1.11.

CONCLUSIONS

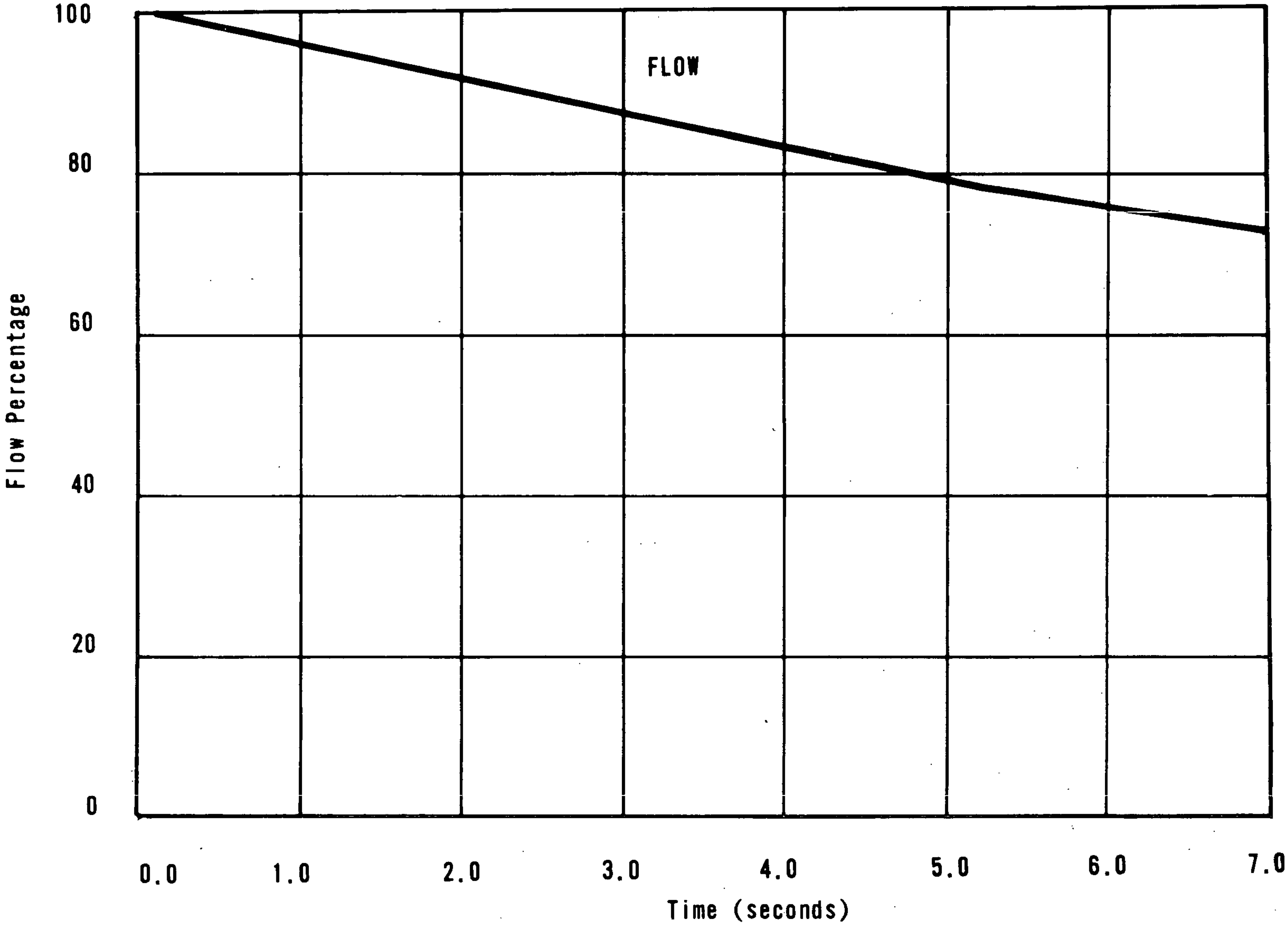
Due to differences in the assumed and "As-Built" sensing strings, the trip delay time for Oconee and subsequent plants has been changed from 0.65 seconds to approximately 1.4 seconds. The analysis presented in this report has demonstrated that when using B&W's standard techniques including the BAW-2 correlation, a technical specification setpoint of the flux/flow ratio of 1.08 is conservative even with the delay time increased from 0.65 seconds to 1.40 seconds. Since the technical specification is conservative, it does not need to be changed. The analyses in the FSAR and Fuel Densification Report where the flux/flow ratio is used, do not need to be revised since the results are conservative.

REFERENCES

(1) B&W Topical Report BAW-10021, "TEMP-Thermal Enthalpy Mixing Program," April 1970.

(2) B&W Topical Report BAW-10000, "Correlation of Critical Heat Flux in a Bundle Cooled by Pressurized Water," March 1970.

(3) B&W Topical Report BAW-10069, "RADAR-Reactor Thermal and Hydraulic Analysis During Reactor Flow Coastdown," July 1973.

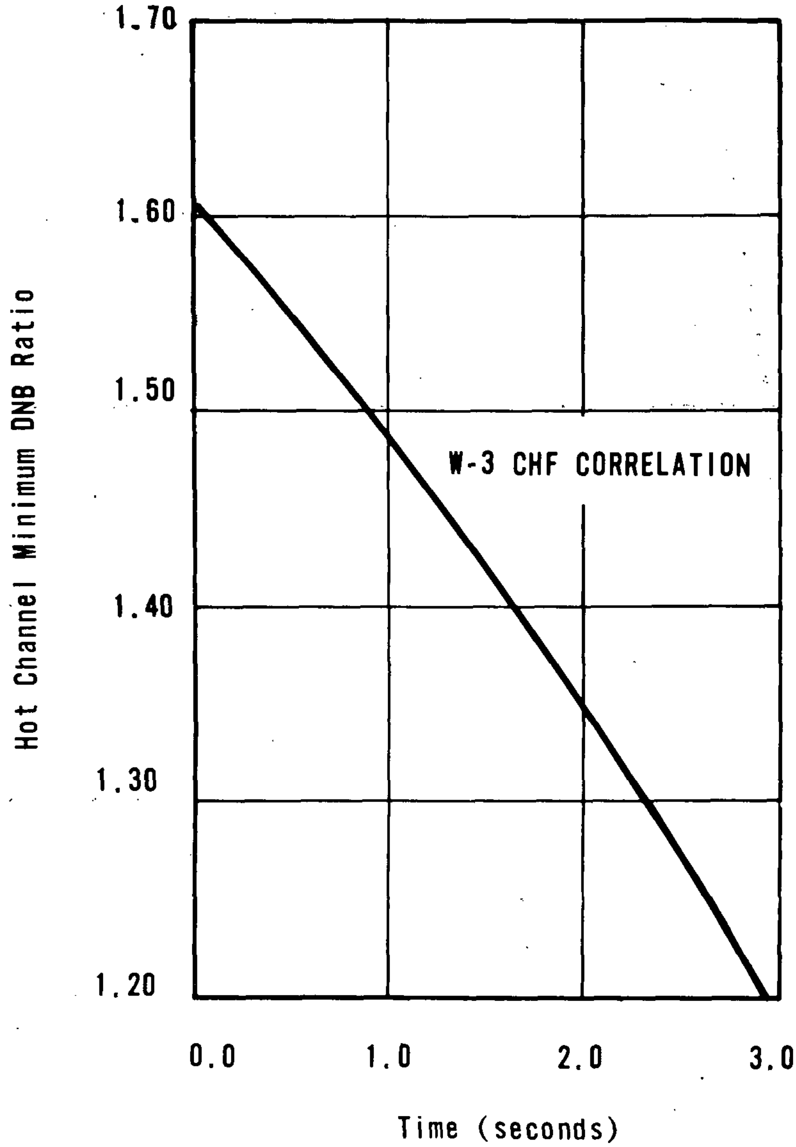


OCONEE 1,2,3

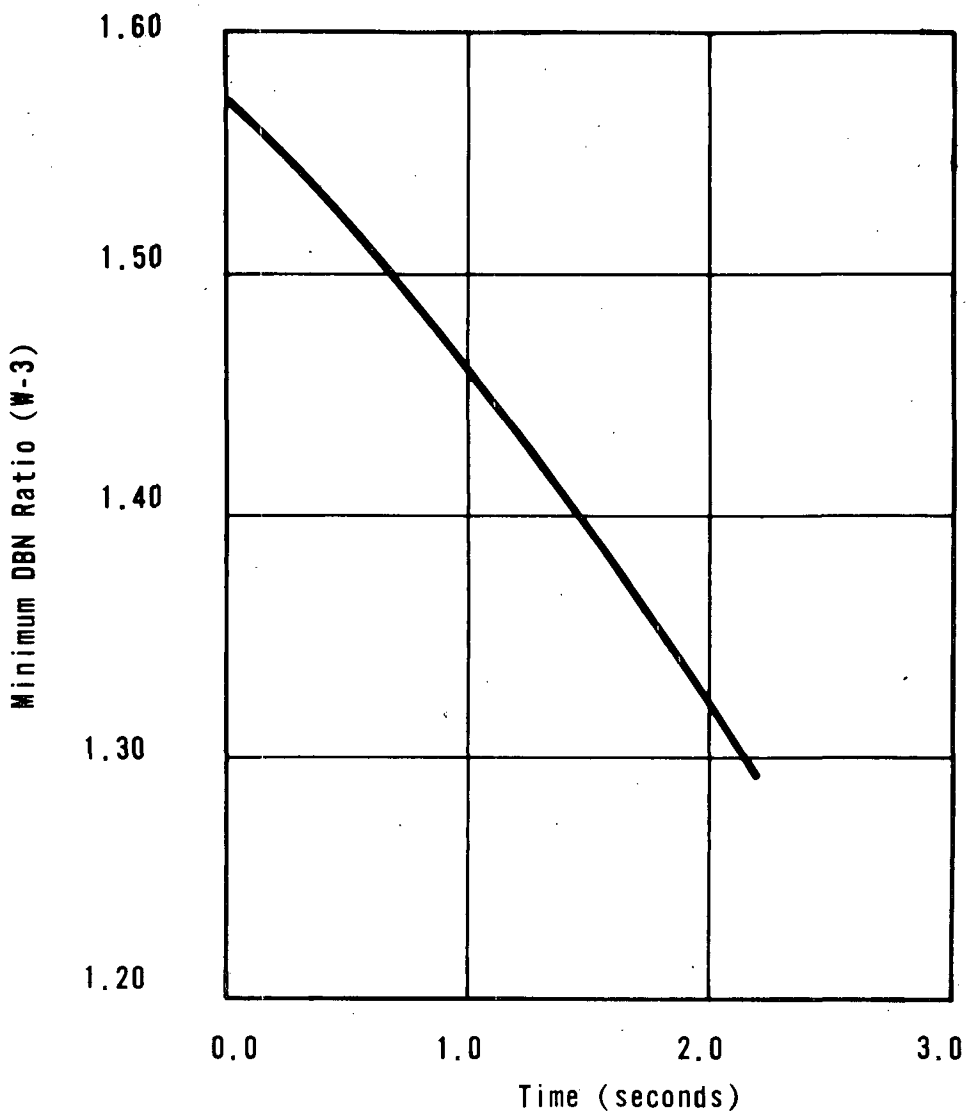
FLOW PERCENTAGE DURING 2 PUMP COASTDOWN FROM 4 PUMP OPERATION

Figure 1

Docket 50-269, -270, and -287
FSAR Supplement 17
February 15, 1974



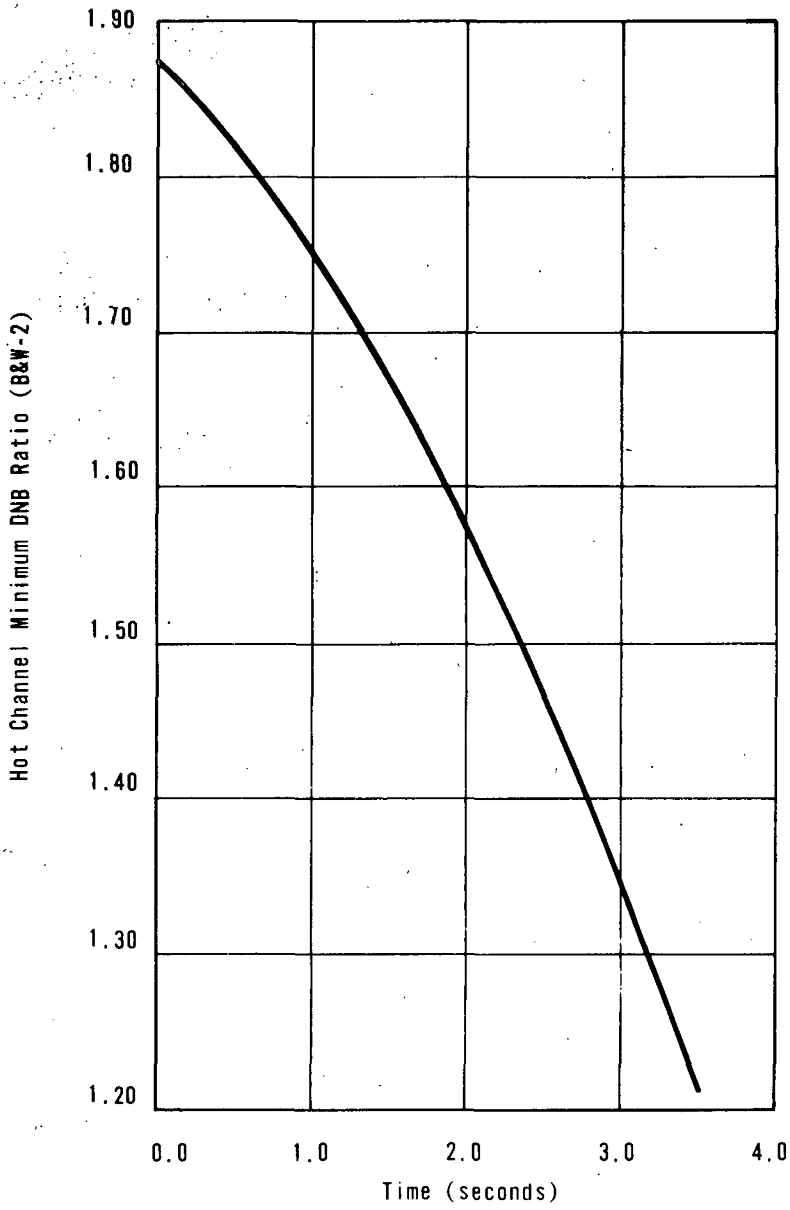
DNBR VERSUS TIME FOR OCONEE—
UNDENSIFIED TWO PUMP COASTDOWN
Figure 2



OCONEE 1

DENSIFIED FUEL HOT CHANNEL DNB RATIO VS. TIME DURING THE TWO PUMP COASTDOWN-OPEN VENT VALVE

Figure 3



OCONEE 1

HOT CHANNEL MINIMUM DNBR (B&W-2) VERSUS TIME DURING TRANSIENT FOR 2 PUMP COASTDOWN FROM 4 PUMP OPERATION ANALYSIS INCLUDES FUEL DENSIFICATION

Figure 4