ATTACHMENT 1

SAFETY EVALUATION

FOR THE

VIRGIL C. SUMMER NUCLEAR STATION

TRANSITION TO WESTINGHOUSE 17x17 VANTAGE 5 FUEL

SOUTH CAROLINA ELECTRIC & GAS COMPANY

VIRGIL C. SUMMER NUCLEAR STATION

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1.0 INTRODUCTION

The V. C. Summer Power Plant is currently operating in Cycle 4 with a Westinghouse 17x17 low-parasitic (LOPAR) fueled core. For subsequent cycles, it is planned to refuel and operate the V. C. Summer Plant with the Westinghouse VANTAGE 5 improved fuel design defined in Reference 1, except for replacing the VANTAGE 5 Bottom Nozzle with a Debris Filter Bottom Nozzle (DFBN). As a result, future core loadings would range from approximately 50%-60% LOPAR and 40%-50% VANTAGE 5 transition core (Cycle 5) to eventually an all VANTAGE 5 fueled core. The VANTAGE 5 fuel assembly is designed as a modification to the current 17x17 LOPAR (standard fuel) and the optimized fuel assembly (OFA) designs, Reference 2.

The VANTAGE 5 design features were conceptually packaged to be licensed as a single entity. This was accomplished via the NRC review and approval of the "VANTAGE 5 Fuel Assembly Reference Core Report," WCAP-10444-P-A, Reference 1. The initial irradiation of a fuel region containing all the VANTAGE 5 design features occurred in the Callaway Plant during the last quarter of 1987. The Callaway VANTAGE 5 licensing submittal was made to the NRC in March 31, 1987 (ULNRC-1470, Docket No. 50-483). Several of the VANTAGE 5 design features, such as axial blankets, reconstitutable top nozzles, extended burnup modified fuel assemblies and Integral Fuel Burnable Absorbers have been successfully licensed as individual design features and are currently in operating Westinghouse plants. Also, four VANTAGE 5 demonstration assemblies are currently in a 3rd cycle of irradiation in the V. C. Summer core.

Descriptions and evaluations of the DFBN and VANTAGE 5 design features are presented in Section 3.0 of this evaluation report. A brief summary of the VANTAGE 5 design features and its major advantages compared to the LOPAR fuel design are given below.

Integral Fuel Burnable Absorber (IFBA) - The IFBA features a zirconium diboride coating on the fuel pellet surface on the central portion of the enriched UO_2 fuel stack. In a typical reload core, approximately one fourth of the fuel rods in the feed region are expected to include IFBAs. IFBAs provide power peaking and moderator temperature coefficient control.

Intermediate Flow Mixer (IFM) - Three IFM grids located between the four upper most Zircaloy grids (Figure 3.1) provide increased DNB margin. Increased margin permits an increase in the design basis $F_{\Delta H}$ and F_Q.

Reconstitutable Top Nozzle - A mechanical disconnect feature facilitates the top nozzle removal. Changes in the design of both the top and bottom nozzles increase burnup margins by providing additional plenum space and room for fuel rod growth.

Extended Burnup - The VANTAGE 5 fuel design will be capable of achieving extended burnups. The basis for designing to extended burnup is contained in the approved Westinghouse topical WCAP-10125-P-A, Reference 3.

Blankets - The axial blanket consists of a nominal six inches of natural UO_2 pellets at each end of the fuel stack to reduce neutron leakage and to improve uranium utilization.

This report is to serve as a reference safety evaluation/analysis report for the region-by-region reload transition from the present V. C. Summer LOPAR fueled core to an all VANTAGE 5 fueled core. This report examines the differences between the VANTAGE 5 and LOPAR fuel assembly designs and evaluates the effect of these differences on the cores during the transition to an all VANTAGE 5 core. The transition and VANTAGE 5 core evaluation/analyses were performed at a core thermal power level of 2775 MWt with the following conservative assumptions made in the safety evaluations: a full power $F_{\Delta H}$ of 1.62 for a reload transition core and 1.68 for the VANTAGE 5 fueled core (except for the Large Break LOCA analysis which used 1.62 for both transition and full VANTAGE 5 cores), an increase in the maximum F_Q to 2.45, 15% plant uniform steam generator tube plugging*, and a positive moderator temperature coefficient (PMTC) of +7 pcm/°F from 0 to 70% power and then decreasing linearly to 0 pcm/°F between 70 to 100% power.

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* Assumes 15% of steam generator tubes in each generator are plugged and corresponds to the worst plugging level of any steam generator.

The axial offset strategy will be the licensed RAOC with Base Loaded Technical Specifications as an optional strategy during operation at or near steady-state equilibrium conditions. An F_Q Surveillance Technical Specification will be implemented. RAOC and F_Q surveillance were approved by the NRC, as shown in WCAP-10217-A (Reference 4).

This report utilizes the standard reload design methods described in Reference 5 and will be used as a basic reference document in support of future V. C. Summer Reload Safety Evaluations (RSE) for VANTAGE 5 fuel reloads. Sections 3.0 through 6.0 of the report summarize the Mechanical, Nuclear, Thermal and Hydraulic, and Accident Evaluations, respectively. Section 7.0 gives a summary of the Technical Specifications changes needed. Attachments 2 and 3 contain the Technical Specification change pages and non-LOCA safety analyses results, respectively. Attachment 4 contains the large and small break LOCA safety analyses. Attachment 5 contains the Thimble Plug Removal Evaluation. Attachment 6 contains the No Significant Hazards Consideration Justification, and Attachment 7 shows no adverse radiological consequence when using VANTAGE 5 fuel.

Consistent with the Westinghouse standard reload methodology, Reference 5, parameters are chosen to maximize the applicability of the safety evaluations for future cycles. The objective of subsequent cycle specific RSEs will be to verify that applicable safety limits are satisfied based on the reference evaluation/analyses established in this report.

In order to demonstrate early performance of the VANTAGE 5 design product features in a commercial reactor, four VANTAGE 5 demonstration assemblies (17x17) were loaded into the V. C. Summer Unit 1 Cycle 2 core and began power production in December of 1984. These assemblies completed one cycle of irradiation in October of 1985 with an average burnup of 11,357 MWD/MTU. Post-irradiation examinations showed all 4 demonstration assemblies were of good mechanical integrity. No mechanical damage or wear was evident on any of the VANTAGE 5 components. Likewise, the IFM grids on the VANTAGE 5 demonstration assemblies had no effect on the adjacent fuel assemblies. All four demonstration assemblies were reinserted into the V. C. Summer core for a second cycle of irradiation. This cycle was completed in March of 1987, at which time the demonstration assemblies achieved an average burnup of about 30,000 MWD/MTU. The observed behavior of the four demonstration assemblies at the end of 2 cycles of irradiation was as good as that observed at the end of the first cycle of irradiation. The four assemblies were reinserted for a third cycle of irradiation.

In addition to V. C. Summer, individual VANTAGE 5 product features have been demonstrated at other nuclear plants. IFBA demonstration fuel rods have been irradiated in Turkey Point Units 3 and 4 for two reactor cycles. Unit 4 contains 112 fuel rods equally distributed in four demonstration assemblies. The IFBA coating performed well with no loss of coating integrity or adherence. The IFM grid feature has been demonstrated at McGuire Unit 1. The demonstration assembly at McGuire has been irradiated for two reactor cycles and is of good mechanical integrity.

2.0 SUMMARY AND CONCLUSIONS

Consistent with the Westinghouse standard reload methodology for analyzing cycle specific reloads, Reference 5, parameters were selected to conservatively bound the values for each subsequent reload cycle and to facilitate determination of the applicability of 10CFR50.59. The objective of subsequent cycle specific reload safety evaluations will be to verify that applicable safety limits are satisfied based on the reference evaluation/analyses established in this report. The mechanical, thermal and hydraulic, nuclear, and accident evaluations considered the transition core effects described for a VANTAGE 5 mixed core in Reference 1. The summary of these evaluations for the V. C. Summer core transitions to an all VANTAGE 5 core are given in the following sections of this submittal.

The transition design and safety evaluations consider the following conditions: 2775 MWt core thermal power, 552.3°F core inlet temperature, 2250 psia system pressure and 277,800 gpm RCS thermal design flow. These conditions are used in core design and safety evaluations to justify safe operation with the conservative assumptions noted in Section 1.0. The conditions summarized in the SER for the VANTAGE 5 reference core report, WCAP-10444, have been considered in the V. C. Summer plant-specific safety evaluations.

The results of evaluation/analysis described herein lead to the following conclusions:

1. The Westinghouse VANTAGE 5 reload fuel assemblies for the V. C. Summer Nuclear Plant are mechanically compatible with the current LOPAR fuel assemblies, control rods, secondary source rods and reactor internals interfaces. The VANTAGE 5/LOPAR fuel assemblies satisfy the current design bases for the V. C. Summer reactor.

2. Evaluations/analyses have shown that all or any combination of thimble plugs may be removed from the Cycle 5 core and subsequent reload cores.

3. Changes in the nuclear characteristics due to the transition from LOPAR to VANTAGE 5 fuel will be within the range normally seen from cycle to cycle due to fuel management effects.

4. The reload VANTAGE 5 fuel assemblies are hydraulically compatible with previously irradiated LOPAR fuel assemblies.

5. The core design and safety analyses results documented in this report show the core's capability for operating safely for the rated V. C. Summer Plant design thermal power with an $F_{\Delta H}$ of 1.62 for mixed fuel cores (eg. Cycle 5) and for all VANTAGE 5 fueled cores,* F_Q = 2.45, uniform steam generator tube plugging levels up to 15%, and a positive MTC of +7 pcm/°F from 0 to 70% power and then decreasing linearly to 0 pcm/°F at 100% power.

6. Plant operating limitations given in the Technical Specifications will be satisfied with the proposed changes noted in Section 7.0 of this report. A reference is established upon which to base Westinghouse reload safety evaluations for future reloads with VANTAGE 5 fuel.

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* Allows for a number of low power LOPAR assemblies in essentially all VANTAGE 5 cores. The number and maximum power of the LOPAR assemblies will be defined in the cycle specific Reload Safety Evaluation (RSE).

3.0 MECHANICAL EVALUATION

This section evaluates the mechanical design and the compatibility of the 17x17 VANTAGE 5 fuel assembly with the current LOPAR fuel assemblies during the transition through mixed-fuel cores to an all VANTAGE 5 core. The VANTAGE 5 fuel assembly has been designed to be compatible with the LOPAR fuel assemblies, reactor internals interfaces, the fuel handling equipment, and the refueling equipment. The VANTAGE 5 design is intended to replace and be compatible with fuel cores containing fuel of the LOPAR designs. The VANTAGE 5 design dimensions as shown on Figure 3.1 are essentially equivalent to these designs from an exterior assembly envelope and reactor internals interface standpoint. The design basis and design limits are essentially the same as those for the LOPAR designs. As such, compliance with the "Acceptance Criteria" of the Standard Review Plan (SRP, NUREG 0800) Section 4.2 Fuel System Design was fully demonstrated.

The significant new mechanical features of the VANTAGE 5 design relative to the current LOPAR fuel design in operation include the following:

- Integral Fuel Burnable Absorber (IFBA)
- Intermediate Flow Mixer (IFM) Grids
- Reconstitutable Top Nozzle
- Extended Burnup Capability
- Axial Blankets
- Replacement of six intermediate Inconel grids with Zircaloy grids
- Reduction in fuel rod, guide thimble and instrumentation tube diameter

Table 3.1 provides a comparison of the VANTAGE 5 and LOPAR fuel assembly design parameters.

Another new mechanical design feature is the Debris Filter Bottom Nozzle (DFBN) which is used instead of the bottom nozzle described in the Reference 1 VANTAGE 5 report.

Fuel Rod Performance

Fuel rod performance for all fuel rod designs is shown to satisfy the SRP fuel rod design bases on a region by region basis. These same bases are applicable to all fuel rod designs, including the Westinghouse LOPAR and VANTAGE 5 fuel designs, with the only difference being that the VANTAGE 5 fuel is designed to achieve a higher burnup consistent with WCAP-10125-P-A, Reference 3, and VANTAGE 5 fuel is designed to operate with a higher $F_{\Delta H}$ limit. The design bases for Westinghouse VANTAGE 5 fuel are discussed in Reference 1.

There is no effect from a fuel rod design standpoint due to having fuel with more than one type of geometry simultaneously residing in the core during the transition cycles. The mechanical fuel rod design evaluation for each region incorporates all appropriate design features of the region, including any changes to the fuel rod or pellet geometry from that of previous fuel regions.

The IFBA coated fuel pellets are identical to the enriched uranium dioxide pellets except for the addition of a thin coating on the pellet cylindrical surface. Coated pellets occupy the central portion of the fuel column. The number and pattern of IFBA rods within an assembly may vary depending on specific application. The ends of the enriched coated pellets and enriched uncoated pellets are dished to allow for axial expansion at the pellet centerline and void volume for fission gas release. Analysis of IFBA rods includes any geometry changes necessary to model the presence of burnable absorber, and conservatively models the gas release from the coating. An evaluation and test program for the IFBA design features are given in Section 2.5 in Reference 1.

Fuel performance evaluations are completed for each fuel region to demonstrate that the design criteria will be satisfied for all fuel rod types in the core under the planned operating conditions. Any changes from the plant operating conditions originally considered in the mechanical design of a fuel region (for example, a power uprating or an increase in the peaking factors) are addressed for all affected fuel regions.

Fuel rod design evaluations are currently performed using the NRC approved models in References 6, 7, and 8 to demonstrate that the SRP fuel rod design criteria (including the rod internal pressure design basis in Reference 9) will be satisfied.

Grid Assemblies

The top and bottom Inconel (non-mixing vane) grids of the VANTAGE 5 fuel assemblies are nearly identical in design to the Inconel grids of the LOPAR fuel assemblies. The only differences are: 1) the grid spring and dimple heights have been modified to accommodate the reduced diameter fuel rod, and 2) the grid spring force has been reduced in the top grid. The six intermediate (mixing vane) structural grids are made of Zircaloy material rather than the Inconel used in the LOPAR design, the straps are thicker and the grid height is greater compared to the LOPAR design.

The Intermediate Flow Mixer (IFM) grids shown in Figure 3.1 are located in the three uppermost spans between the Zircaloy mixing vane structural grids and incorporate a similar mixing vane array. Their prime function is mid-span flow mixing in the hottest fuel assembly spans. Each IFM grid cell contains four dimples which are designed to prevent mid-span channel closure in the spans containing IFMs and fuel rod contact with the mixing vanes. This simplified cell arrangement allows short grid cells so that the IFM grid can accomplish its flow mixing objective with minimal pressure drop.

The IFM grids are not intended to be structural members. The outer strap configuration was designed similar to current fuel designs to preclude grid hang-up and damage during fuel handling. Additionally, the grid envelope is smaller which further minimizes the potential for damage and reduces calculated forces during seismic/LOCA events. A coolable geometry is, therefore, assured at the IFM grid elevation, as well as at the structural grid elevation.

Reconstitutable Top Nozzle

The reconstitutable top nozzle for the VANTAGE 5 fuel assembly differs from the LOPAR design in two ways: a groove is provided in each thimble thru-hole in the nozzle plate to facilitate attachment and removal; and the nozzle plate thickness is reduced to provide additional axial space for fuel rod growth.

To remove the top nozzle, a tool is first inserted through a lock tube and expanded radially to engage the bottom edge of the tube. An axial force is then exerted on the tool which overrides local lock tube deformations and withdraws the lock tube from the insert. After the lock tubes have been withdrawn, the nozzle is removed by raising it off the upper slotted ends of the nozzle inserts which deflect inwardly under the axial lift load.

With the top nozzle removed, direct access is provided for fuel rod examinations or replacement. Reconstitution is completed by the remounting of the nozzle and the insertion of lock tubes. Additional details of this design feature, the design bases and evaluation of the reconstitutable top nozzle are given in Section 2.3.2 in Reference 1.

Debris Filter Bottom Nozzle

It is planned to introduce the debris filter bottom nozzle (DFBN) into the V. C. Summer Region 7 fuel assemblies to reduce the possibility of fuel rod damage due to debris-induced fretting. The relatively large flow holes in a conventional bottom nozzle are replaced with a new pattern of smaller flow holes for the DFBN. The holes are sized to minimize passage of debris particles large enough to cause damage while providing sufficient flow area, comparable pressure drop, and continued structural integrity of the nozzle. Tests to measure pressure drop and demonstrate structural integrity have been performed to verify that the debris filter bottom nozzle is totally compatible with the current design.

The 304 stainless steel DFBN is similar to the LOPAR design used for the V. C. Summer Region 6 fuel assemblies. Significant changes compared to the LOPAR design involve 1) a modified flow hole size and pattern as described above, and 2) a decreased nozzle height and thinner top plate (identical to the existing VANTAGE 5 bottom nozzle) to accommodate the high burnup fuel rods. The DFBN retains the design reconstitution feature which facilitates easy removal of the nozzle from the fuel assembly in the same manner as the bottom nozzle used for the Region 6 LOPAR fuel assemblies.

Axial Blankets

Although noted as a new mechanical feature of the VANTAGE 5 design and licensed in Reference 1, axial blankets have been and are currently operating in Westinghouse plants. A description and design application of this feature are contained in Reference 1, Section 3.0.

Mechanical Compatibility of Fuel Assemblies

Based on the evaluation of the VANTAGE 5/LOPAR design differences and hydraulic test results (Reference 1) and the evaluation of the DFBN, it is concluded that the two designs are mechanically compatible with each other. The VANTAGE 5 fuel rod mechanical design bases remain unchanged from that used for the LOPAR Region 6 fuel assemblies in the V. C. Summer Cycle 4 core.

Rod Bow

It is predicted that the 17x17 Vantage 5 rod bow magnitudes, like those of the Westinghouse OFA fuel, will be within the bounds of existing 17x17 LOPAR assembly rod bow data. The current NRC approved methodology for comparing rod bow for two different fuel assembly designs is given in Reference 10.

Rod bow in fuel rods containing IFBAs is not expected to differ in magnitude or frequency from that currently observed in Westinghouse LOPAR fuel rods under similar operating conditions. No indications of abnormal rod bow have been observed on visual or dimensional inspections performed on the test IFBA rods. Rod growth measurements were also within predicted bounds.

Fuel Rod Wear

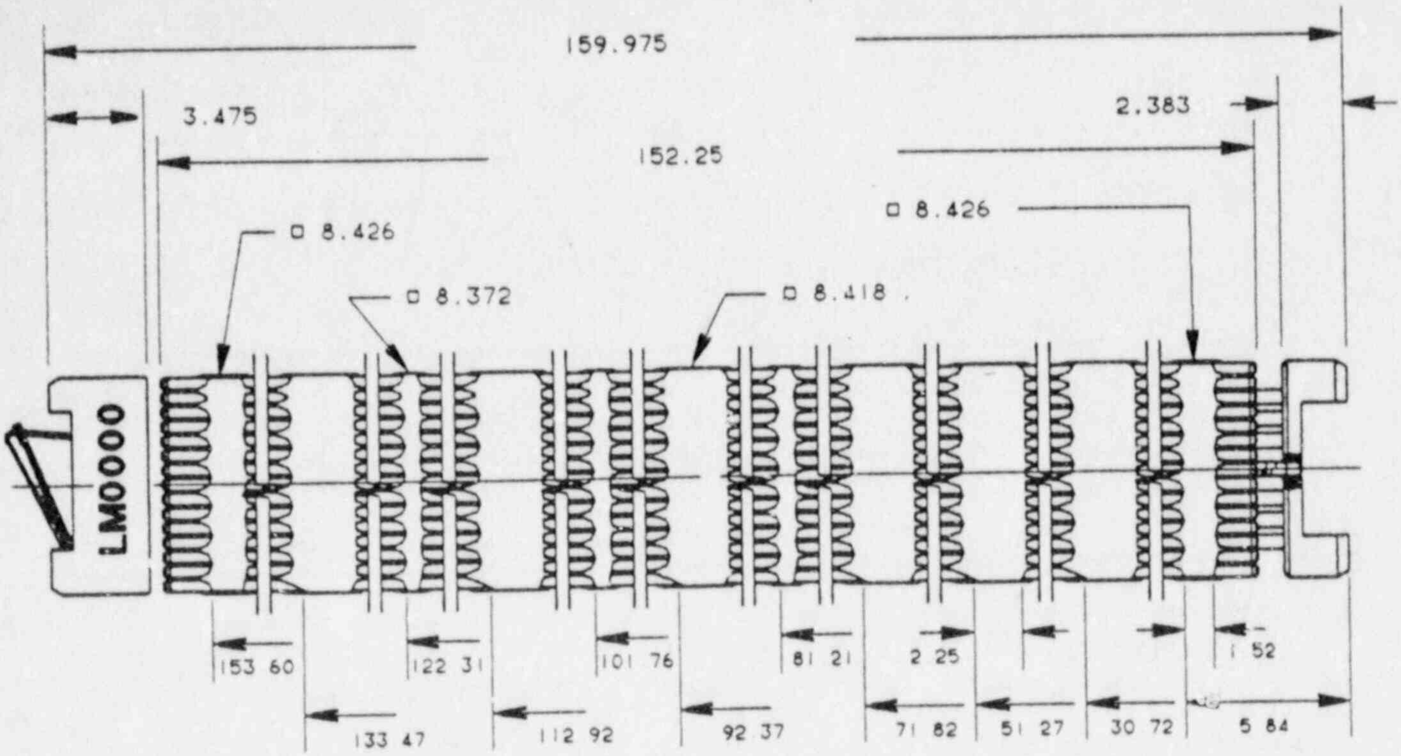
Fuel rod wear is dependent on both the support conditions and the flow environment to which the fuel rod is subjected. Due to the LOPAR and VANTAGE 5 fuel assembly designs employing different grids, there is an unequal axial pressure distribution between the assemblies. Crossflow resulting from this unequal pressure distribution was evaluated to determine the induced rod vibration and subsequent wear. Hydraulic tests (Reference 1, Appendix A.1.4) were performed to verify that the crossflows were negligible and also to check hydraulic compatibility of the LOPAR and VANTAGE 5 designs. The VANTAGE 5 fuel assembly was flow tested adjacent to a 17x17 OFA, since vibration test results indicated that the crossflow effects produced by this fuel assembly combination would have the most detrimental effect on fuel rod wear.

Results of the wear inspection and analysis discussed in Reference 1, Appendix A.1.4, revealed that the VANTAGE 5 fuel assembly wear characteristic was similar to that of the 17x17 OFA when both sets of data were normalized to the test duration time. It was concluded that the VANTAGE 5 fuel rod wear would be less than the maximum wear depth established, Reference 11, for the 17x17 OFA at EOL.

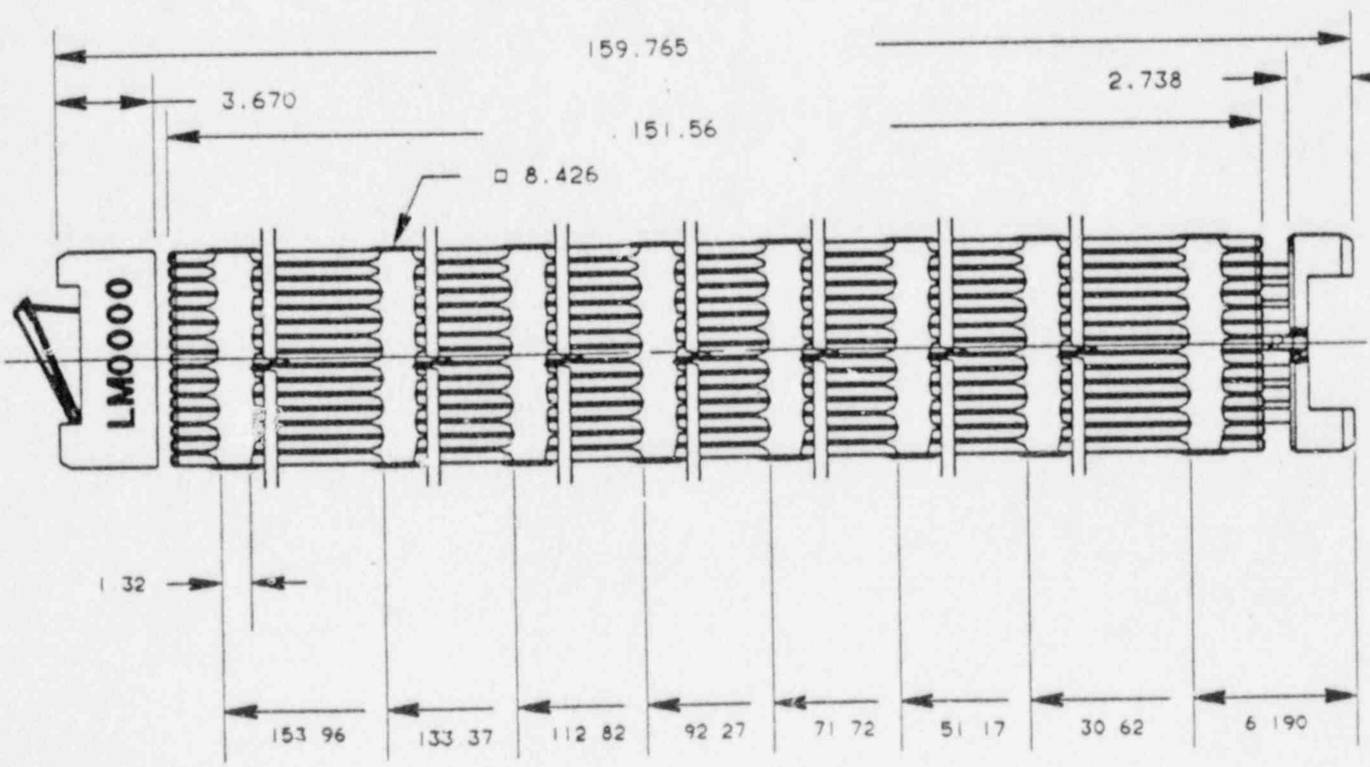
TABLE 3.1

COMPARISON OF 17x17 LOPAR
and
17x17 VANTAGE 5 FUEL ASSEMBLY DESIGN PARAMETERS

PARAMETER	17x17 LOPAR DESIGN	17x17 VANTAGE 5 DESIGN
Fuel Assy Length, in	159.765	159.975
Fuel Rod Length, in	151.56	152.255
Assembly Envelope, in	8.426	8.426
Compatible with Core Internals	Yes	Yes
Fuel Rod Pitch, in	.496	.496
Number of Fuel Rods/Assy	264	264
Number/Guide Thimble Tubes/Assy	24	24
Number/Instrumentation Tube/Assy	1	1
Fuel Tube Material	Zircaloy 4	Zircaloy 4
Fuel Rod Clad OD, in	0.374	0.360
Fuel Rod Clad Thickness, in	.0225	.0225
Fuel/Clad Gap, mil	6.5	6.2
Fuel Pellet Diameter, in	.3225	.3088
Fuel Pellet Length, in	.530	.507
Guide Thimble Material	Zircaloy 4	Zircaloy 4
Guide Thimble OD, in.	.482	.474
Instrumentation Tube Material	Zircaloy 4	Zircaloy 4
Instrumentation Tube OD, in.	.482	.474



17X17 VANTAGE 5 FUEL ASSEMBLY



17X17 RECONSTITUTABLE LOPAR FUEL ASSEMBLY

V.C. SUMMER UNIT 1

FIGURE 3.1

17X17 VANTAGE 5 / LOPAR FUEL ASSEMBLY COMPARISON

4.0 NUCLEAR DESIGN

The evaluation of the transition and equilibrium cycle VANTAGE 5 cores presented in Reference 1, as well as the V. C. Summer specific transition and equilibrium core evaluations, demonstrate that the impact of implementing VANTAGE 5 does not cause a significant change to the physics characteristics of the V. C. Summer cores beyond the normal range of variations seen from cycle to cycle.

The methods and core models used in the V. C. Summer reload transition core analysis are described in References 1, 5, 12, and 13. These licensed methods and models have been used for V. C. Summer and other previous Westinghouse reload designs using the LOPAR and VANTAGE 5 fuel. No changes to the nuclear design philosophy, methods, or models are necessary because of the transition to VANTAGE 5 fuel. Increased emphasis will be placed on the use of three-dimensional nuclear models because of the axially heterogeneous nature of the VANTAGE 5 fuel design when axial blankets and reduced length WABA/IFBAs are used.

From the nuclear design area, the following V. C. Summer Technical Specification changes are proposed:

1) Increased $F_{\Delta H}$ limits. These higher limits will allow loading pattern designs with lower leakage which in turn will allow longer cycles.

2) Increased F_Q limit. The increased F_Q limit will provide greater flexibility with regard to accommodating the axially heterogeneous cores (blankets and short burnable absorbers).

3) F_Q Surveillance. This revision to surveillance requirements on the heat flux hot channel factor, $F_Q(z)$, has been proposed to increase plant operating flexibility while more directly monitoring F_Q. Rather than performing surveillance on Fxy(z), the radial component of the total peaking factor, surveillance is performed directly on $F_Q(z)$. The steady-state $F_Q(z)$ is measured and increased by

applicable uncertainties. This quantity is further increased by an analytical factor called W(z) which accounts for possible increases in the steady-state $F_Q(z)$ resulting from operation within the allowed axial flux difference limits. The resulting $F_Q(z)$ is compared to the $F_Q(z)$ limit to demonstrate operation below the heat flux hot channel factor limit.

4) RAOC/Base Load implementation. The RAOC strategy allows greater operator flexibility with regard to core operation. The margin created by the increased F_Q limit is being partly converted into operational flexibility.

During operation at or near steady state equilibrium conditions, core peaking factors are significantly reduced. Through the use of a Base Load Tech Spec, this reduction in core peaking factors are recognized.

As illustrated in Attachment A to this section, the AFD operating spaces may be presented as a function of cycle burnup to further enhance operational flexibility during portions of the cycle. This is accomplished by performing an analysis consistent with the methodology described in Reference 4, which takes credit for "burndown" characteristics of both radial and axial power shapes.

It is proposed that the resulting AFD operating spaces for RAOC and Base Load operations be deleted from the Tech Spec and instead, be incorporated in the Peaking Factor Limit Report (PFLR). This eliminates the potential necessity of Tech Spec amendments or AFD limits for future reload cycles, while providing adequate assurance that the correct AFD operating spaces will be followed. A sample PFLR containing AFD operating limits and W(Z) values can be found in the Attachment A.

5) Positive MTC. Due to increased boron concentration associated with a higher enriched fuel and longer fuel residence times, it is necessary to increase the MTC limit. The increased limit value is +7.0 pcm/°F from HZP to 70% rated power, then a negative linear ramp from 70% power to 0.0 pcm/°F at HFP.

Power distributions and peaking factors show slight changes as a result of the incorporation of axial blankets, reduced length WABA/IFBAs, and increased peaking factor limits, in addition to the normal variations experienced with different loading patterns. The usual methods of enrichment variation and burnable absorber usage can be employed in the transition and full VANTAGE 5 cores to ensure compliance with the peaking factor Technical Specifications.

The key safety parameters evaluated for V. C. Summer reactor as it transitions to an all VANTAGE 5 core show little change relative to the range of parameters experienced for the all LOPAR core. The changes in values of the key safety parameters are typical of the normal cycle-to-cycle variations experienced as loading patterns change. As is current practice, each reload core design will be evaluated to assure that design and safety limits are satisfied according to the reload methodology. The design and safety limits will be documented in each cycle specific reload safety evaluation (RSE) report which serves as a basis for any significant changes which may require a future NRC review.

Attachment A to Section 4.0

SAMPLE PEAKING FACTOR REPORT

FOR

V. C. SUMMER PLANT

PEAKING FACTOR LIMIT REPORT FOR V. C. SUMMER CYCLE 5
RAOC AND BASE LOAD OPERATION

This Peaking Factor Limit Report is provided in accordance with Paragraph 6.9.1.11 of the V. C. Summer Technical Specifications.

The V. C. Summer Cycle 5 allowed Axial Flux Difference (AFD) operating space for RAOC operation for beginning-of-life (BOL), middle-of-life (MOL), and end-of-life (EOL) portions are shown in Figures 1 through 3, respectively. The cycle burnup ranges applicable to each specified operating space are indicated in each of the figures. The specified allowed AFD operational spaces for RAOC operation were determined using the method described in Reference 1.

The V. C. Summer Cycle 5 elevation dependent W(z) values for RAOC operation at BOL, MOL, and near EOL are shown in Figures 4 through 6, respectively. This information is sufficient to determine W(z) versus core height for Cycle 5 burnups in the range of 0 MWD/MTU to EOL burnup through the use of three point interpolation.

The V. C. Summer Cycle 5 allowed AFD target bands during base load operation for BOL, MOL, and EOL were determined to be as follows:

BOL (0 - xxxx MWD/MTU)	: + or - x % about a measured target value
MOL (xxxx - xxxx MWD/MTU)	: + or - x % about a measured target value
EOL (xxxx - xxxxx MWD/MTU)	: + or - x % about a measured target value

The V. C. Summer Cycle 5 elevation dependent W(z) values for base load operation between x% and 100% of rated thermal power with the above specified target band about a measured target value at BOL, MOL, and near EOL are shown in Figures 7 through 9, respectively. This information is sufficient to determine W(z) versus core height for Cycle 5 burnups in the range of 0 MWD/MTu to EOL burnup through the use of three point interpolation.

W(z) values for RAOC and base load operation were calculated using the method described in Part B of Reference 1.

The minimum allowable power level for base load operation, APL^{ND}, for V. C. Summer Cycle 5 is xx percent of rated thermal power.

The appropriate W(z) function is used to confirm that the heat flux hot channel factor, Fq(z), will be limited to the Technical Specification values

$Fq(z) \leq 2.45 / P (k(z))$ for $P > 0.50$ and

$Fq(z) \leq 4.90 (k(z))$ for $P \leq 0.50$

The appropriate elevation dependent W(z) values, when applied to a power distribution measured under equilibrium conditons, demonstrate that the initial conditions assumed in the LOCA are met, along with the ECCS acceptance criteria of 10CFR50.46.

(1) WCAP-10215-P-A, Relaxation of Constant Axial Control - Fq Surveillance Technical Specification

SAMPLE

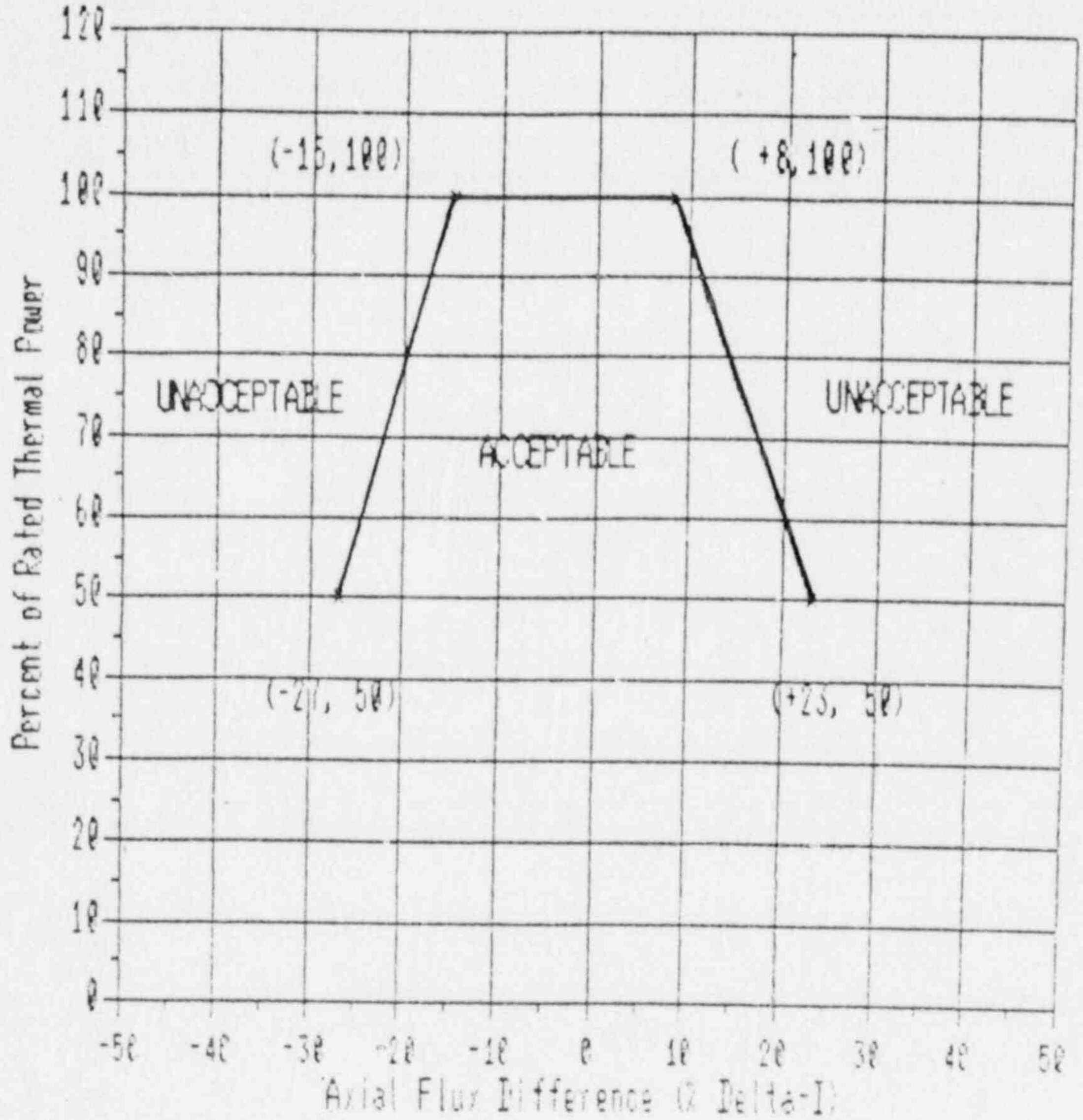


FIGURE 1

AXIAL FLUX DIFFERENCE LIMITS AS A FUNCTION OF
RATED THERMAL POWER FOR CYCLE BURNUP BOL - 4000 MWD/MTU

SAMPLE

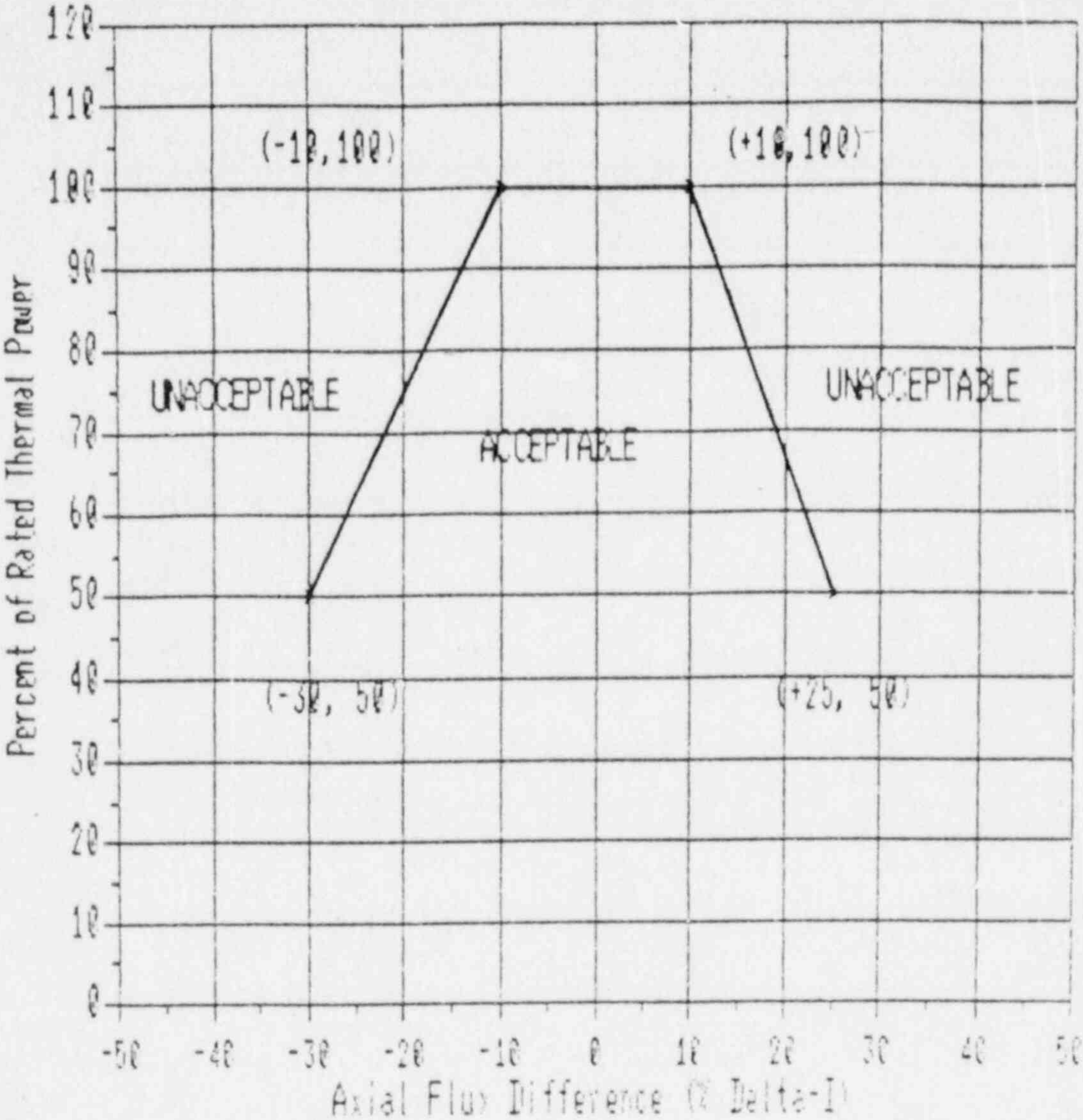


FIGURE 2

AXIAL FLUX DIFFERENCE LIMITS AS A FUNCTION OF
RATED THERMAL POWER FOR CYCLE BURNUP 4000 -10000 MWD/MTU

SAMPLE

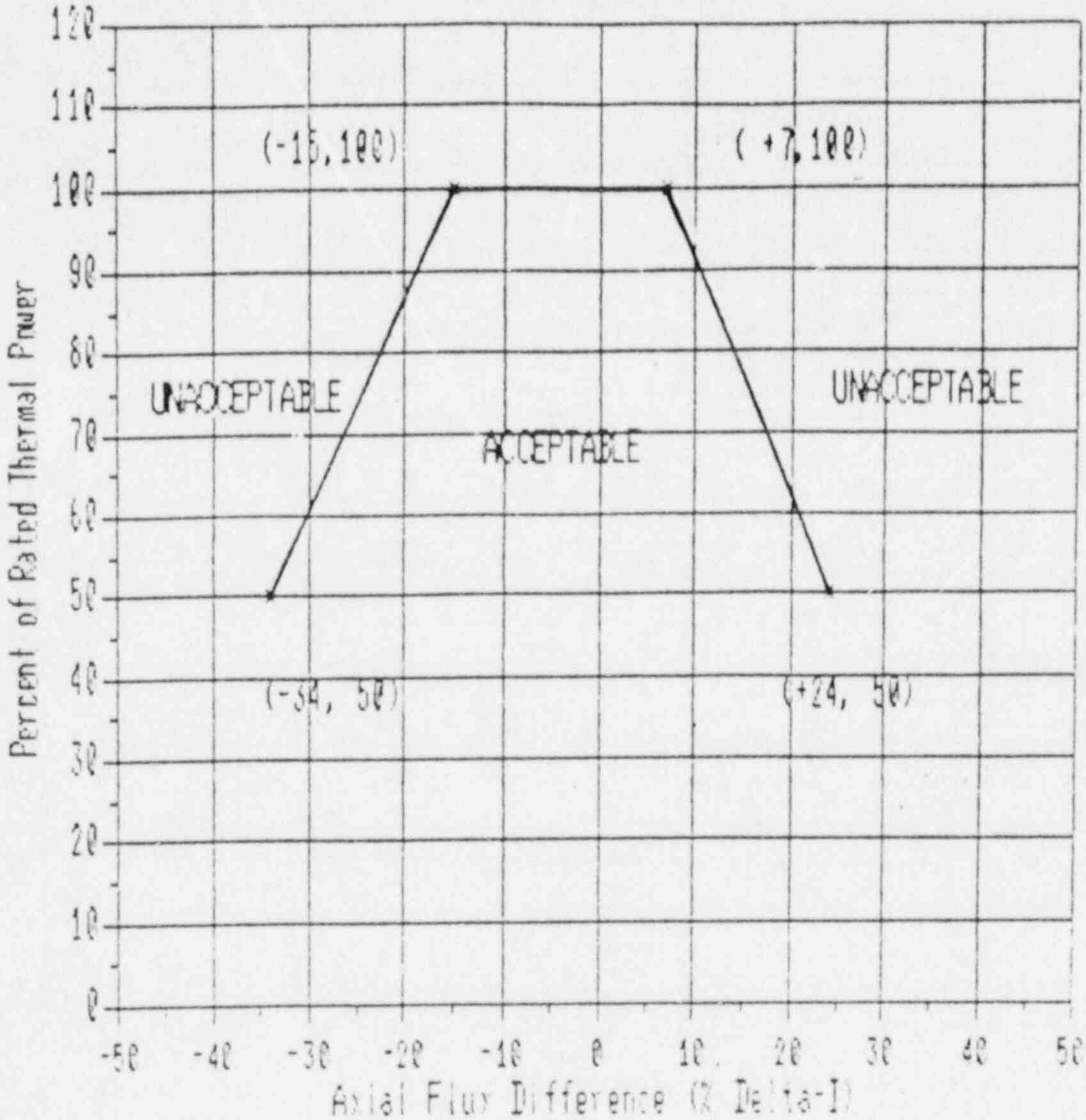
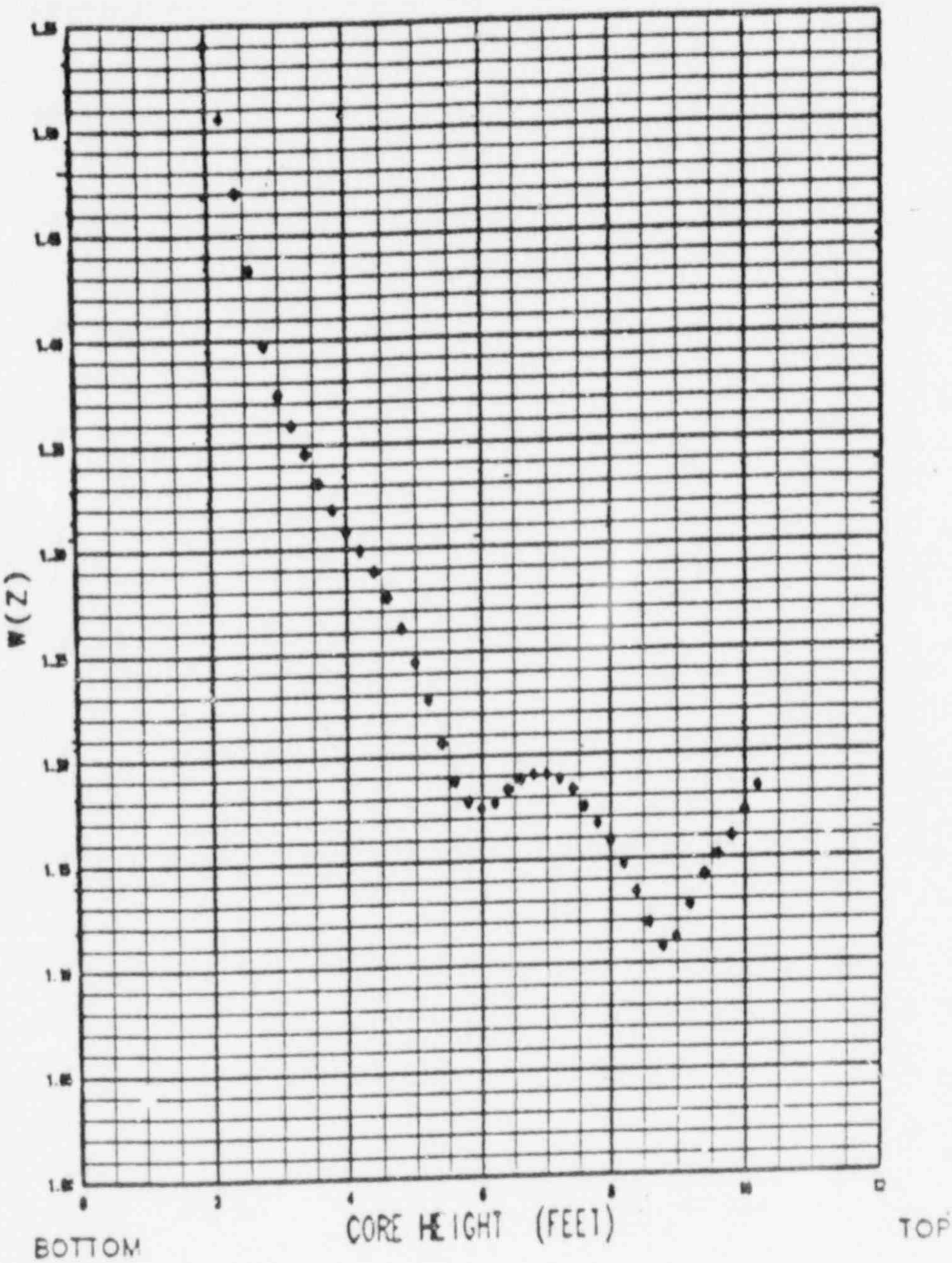


FIGURE 3

AXIAL FLUX DIFFERENCE LIMITS AS A FUNCTION OF
RATED THERMAL POWER FOR CYCLE BURNUP 10000 MWD/MTU - EOL

SAMPLE



	HEIGHT (FT.)	BOL W(Z)
•	0.0	1.0000
•	0.2	1.0000
•	0.4	1.0000
•	0.6	1.0000
•	0.8	1.0000
•	1.0	1.0000
•	1.2	1.0000
•	1.4	1.0000
•	1.6	1.0000
•	1.8	1.0000
	2.0	1.5401
	2.2	1.5065
	2.4	1.4703
	2.6	1.4335
	2.8	1.3975
	3.0	1.3737
	3.2	1.3591
	3.4	1.3455
	3.6	1.3315
	3.8	1.3137
	4.0	1.3033
	4.2	1.2992
	4.4	1.2890
	4.6	1.2764
	4.8	1.2618
	5.0	1.2453
	5.2	1.2274
	5.4	1.2065
	5.6	1.1880
	5.8	1.1782
	6.0	1.1750
	6.2	1.1773
	6.4	1.1836
	6.6	1.1884
	6.8	1.1905
	7.0	1.1906
	7.2	1.1885
	7.4	1.1834
	7.6	1.1752
	7.8	1.1670
	8.0	1.1583
	8.2	1.1473
	8.4	1.1340
	8.6	1.1192
	8.8	1.1078
	9.0	1.1121
	9.2	1.1277
	9.4	1.1417
	9.6	1.1512
	9.8	1.1599
	10.0	1.1716
	10.2	1.1833
•	10.4	1.0000
•	10.6	1.0000
•	10.8	1.0000
•	11.0	1.0000
•	11.2	1.0000
•	11.4	1.0000
•	11.6	1.0000
•	11.8	1.0000
•	12.0	1.0000

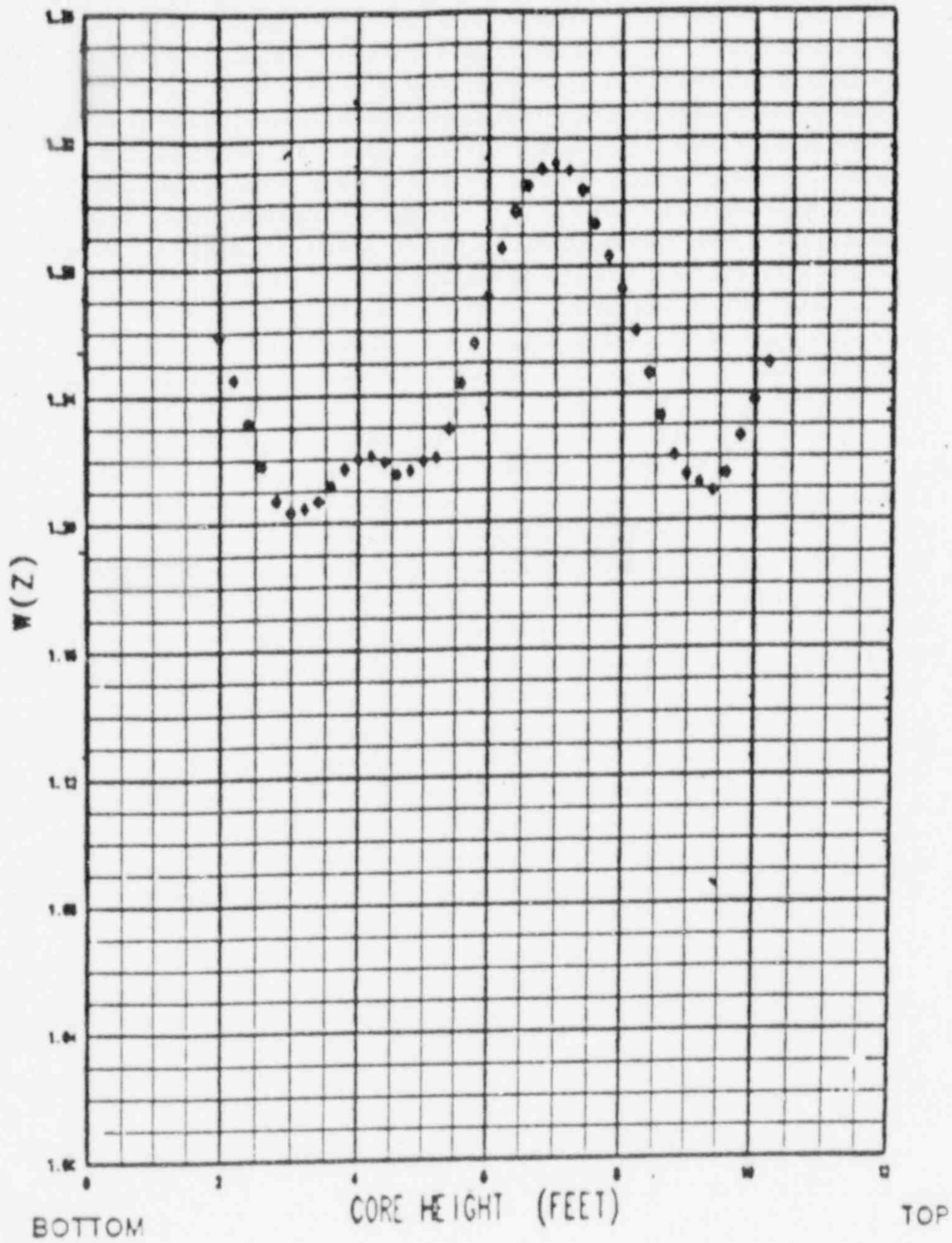
FIGURE 4

V. C. SUMMER RAOC W(Z) AT BOL

(150 MWD/MTU)

TOP AND BOTTOM 15% EXCLUDED AS PER TECH SPEC 4.2.2.2.G

SAMPLE



	HEIGHT (FT.)	MOL W(Z)
•	0.0	1.0000
•	0.2	1.0000
•	0.4	1.0000
•	0.6	1.0000
•	0.8	1.0000
•	1.0	1.0000
•	1.2	1.0000
•	1.4	1.0000
•	1.6	1.0000
•	1.8	1.0000
	2.0	1.2581
	2.2	1.2464
	2.4	1.2315
	2.6	1.2180
	2.8	1.2072
	3.0	1.2034
	3.2	1.2044
	3.4	1.2067
	3.6	1.2116
	3.8	1.2170
	4.0	1.2199
	4.2	1.2210
	4.4	1.2193
	4.6	1.2153
	4.8	1.2184
	5.0	1.2194
	5.2	1.2204
	5.4	1.2294
	5.6	1.2436
	5.8	1.2580
	6.0	1.2707
	6.2	1.2858
	6.4	1.2971
	6.6	1.3053
	6.8	1.3105
	7.0	1.3121
	7.2	1.3101
	7.4	1.3038
	7.6	1.2932
	7.8	1.2830
	8.0	1.2727
	8.2	1.2597
	8.4	1.2463
	8.6	1.2329
	8.8	1.2205
	9.0	1.2144
	9.2	1.2120
	9.4	1.2093
	9.6	1.2146
	9.8	1.2263
	10.0	1.2376
	10.2	1.2493
•	10.4	1.0000
•	10.6	1.0000
•	10.8	1.0000
•	11.0	1.0000
•	11.2	1.0000
•	11.4	1.0000
•	11.6	1.0000
•	11.8	1.0000
•	12.0	1.0000

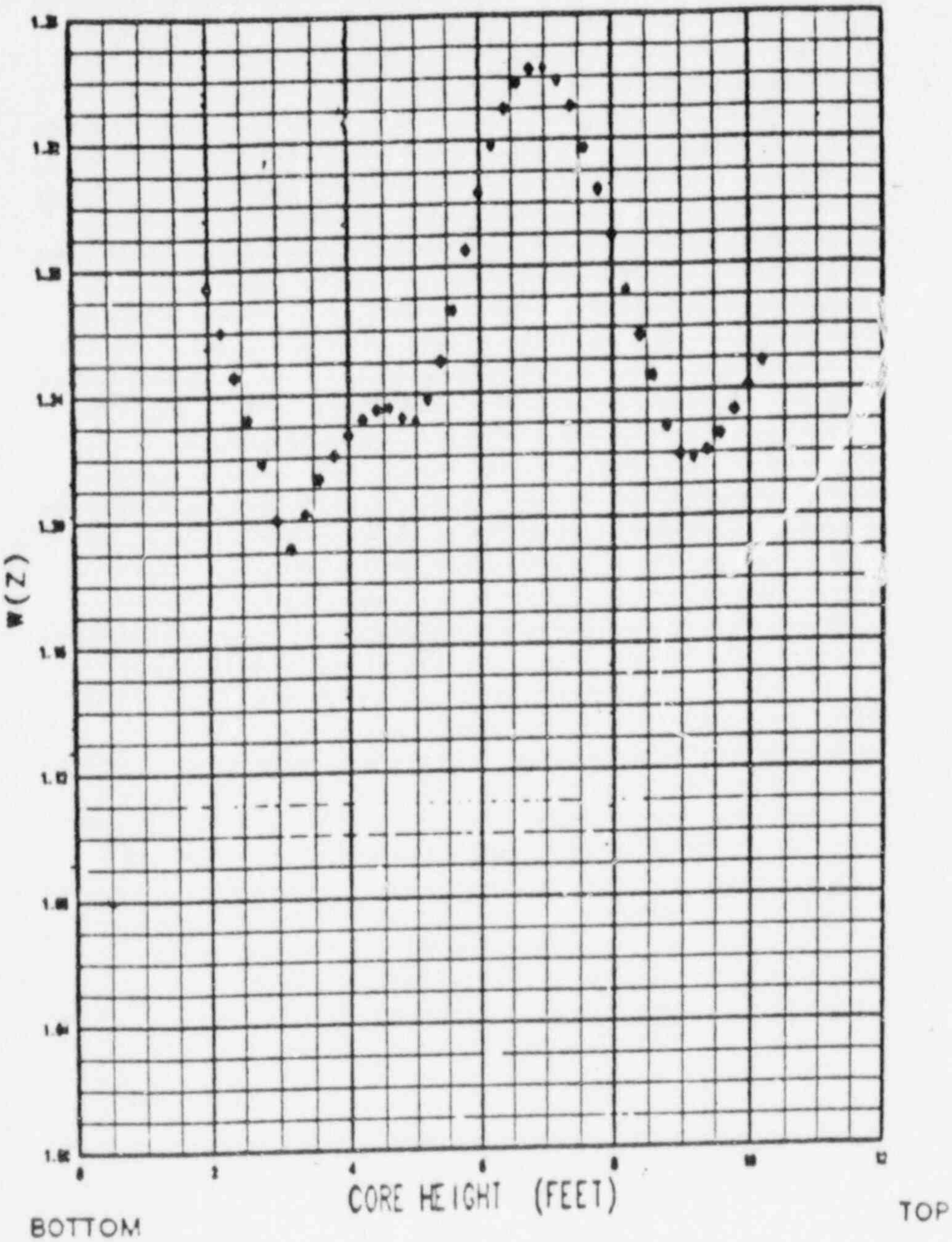
FIGURE 5

V. C. SUMMER RAOC W(Z) AT MOL

(8000 MWD/MTU)

TOP AND BOTTOM 15% EXCLUDED AS PER TECH SPEC 4.2.2.2.G

SAMPLE



	HEIGHT (FT.)	EOL W(Z)
*	0.0	1.0000
*	0.2	1.0000
*	0.4	1.0000
*	0.6	1.0000
*	0.8	1.0000
*	1.0	1.0000
*	1.2	1.0000
*	1.4	1.0000
*	1.6	1.0000
*	1.8	1.0000
	2.0	1.2744
	2.2	1.2601
	2.4	1.2454
	2.6	1.2321
	2.8	1.2183
	3.0	1.2002
	3.2	1.1911
	3.4	1.2017
	3.6	1.2130
	3.8	1.2204
	4.0	1.2270
	4.2	1.2318
	4.4	1.2347
	4.6	1.2355
	4.8	1.2325
	5.0	1.2312
	5.2	1.2381
	5.4	1.2601
	5.6	1.2861
	5.8	1.2850
	6.0	1.3031
	6.2	1.3183
	6.4	1.3300
	6.6	1.3382
	6.8	1.3425
	7.0	1.3427
	7.2	1.3389
	7.4	1.3304
	7.6	1.3171
	7.8	1.3038
	8.0	1.2893
	8.2	1.2716
	8.4	1.2575
	8.6	1.2445
	8.8	1.2286
	9.0	1.2195
	9.2	1.2157
	9.4	1.2207
	9.6	1.2259
	9.8	1.2332
	10.0	1.2410
	10.2	1.2485
*	10.4	1.0000
*	10.6	1.0000
*	10.8	1.0000
*	11.0	1.0000
*	11.2	1.0000
*	11.4	1.0000
*	11.6	1.0000
*	11.8	1.0000
*	12.0	1.0000

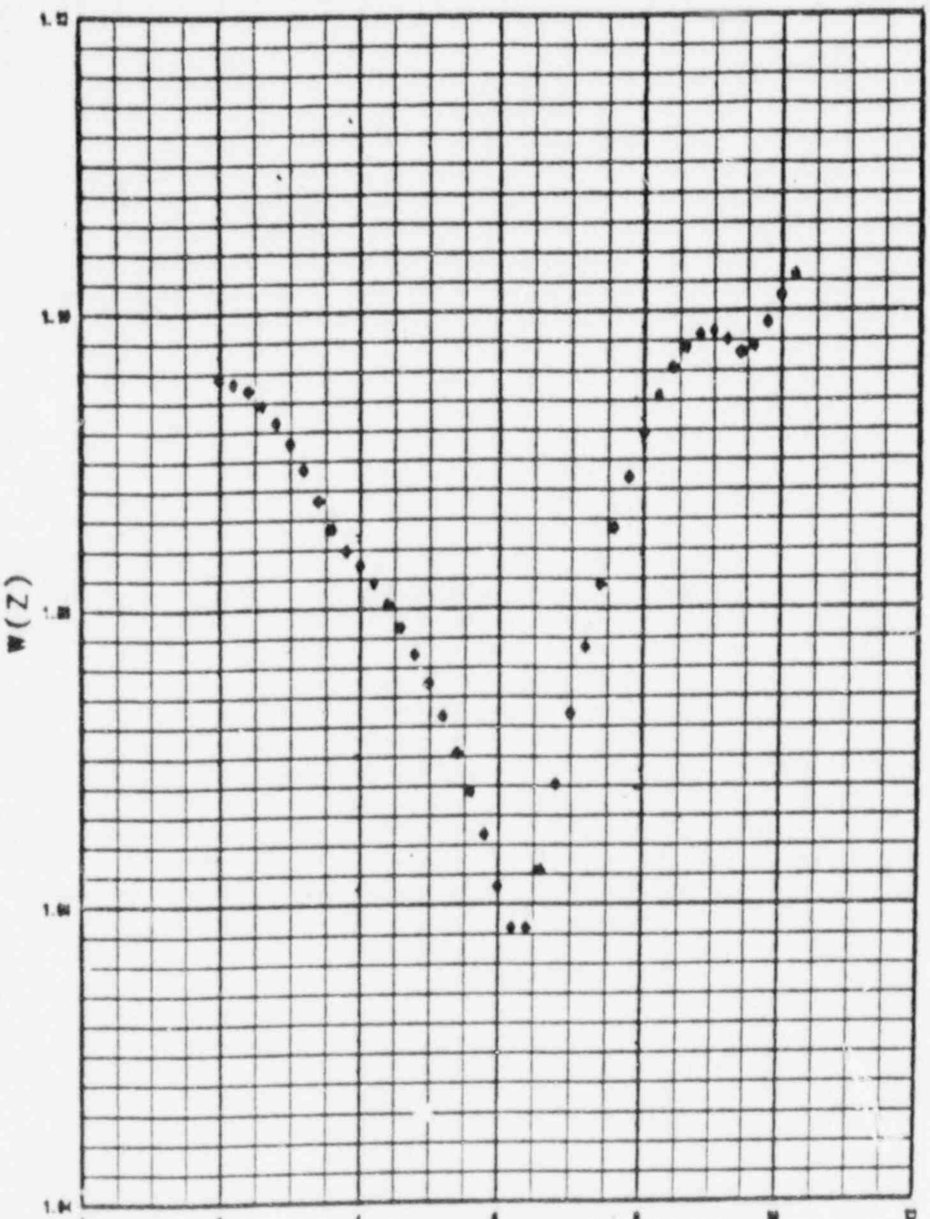
FIGURE 6

V. C. SUMMER RAOC W(Z) NEAR EOL

(14000 MWD/MTU)

TOP AND BOTTOM 15% EXCLUDED AS PER TECH SPEC 4.2.2.2.G

SAMPLE



CORE HEIGHT (FEET)

BOTTOM TOP

	HEIGHT (FT.)	BOL W(Z)
*	0.0	1.0000
*	0.2	1.0000
*	0.4	1.0000
*	0.6	1.0000
*	0.8	1.0000
*	1.0	1.0000
*	1.2	1.0000
*	1.4	1.0000
*	1.6	1.0000
*	1.8	1.0000
	2.0	1.0956
	2.2	1.0953
	2.4	1.0944
	2.6	1.0934
	2.8	1.0926
	3.0	1.0912
	3.2	1.0894
	3.4	1.0873
	3.6	1.0854
	3.8	1.0839
	4.0	1.0829
	4.2	1.0817
	4.4	1.0803
	4.6	1.0787
	4.8	1.0769
	5.0	1.0749
	5.2	1.0727
	5.4	1.0702
	5.6	1.0676
	5.8	1.0647
	6.0	1.0612
	6.2	1.0584
	6.4	1.0584
	6.6	1.0623
	6.8	1.0681
	7.0	1.0729
	7.2	1.0774
	7.4	1.0816
	7.6	1.0854
	7.8	1.0888
	8.0	1.0918
	8.2	1.0943
	8.4	1.0962
	8.6	1.0976
	8.8	1.0984
	9.0	1.0986
	9.2	1.0981
	9.4	1.0972
	9.6	1.0976
	9.8	1.0992
	10.0	1.1010
	10.2	1.1024
*	10.4	1.0000
*	10.6	1.0000
*	10.8	1.0000
*	11.0	1.0000
*	11.2	1.0000
*	11.4	1.0000
*	11.6	1.0000
*	11.8	1.0000
*	12.0	1.0000

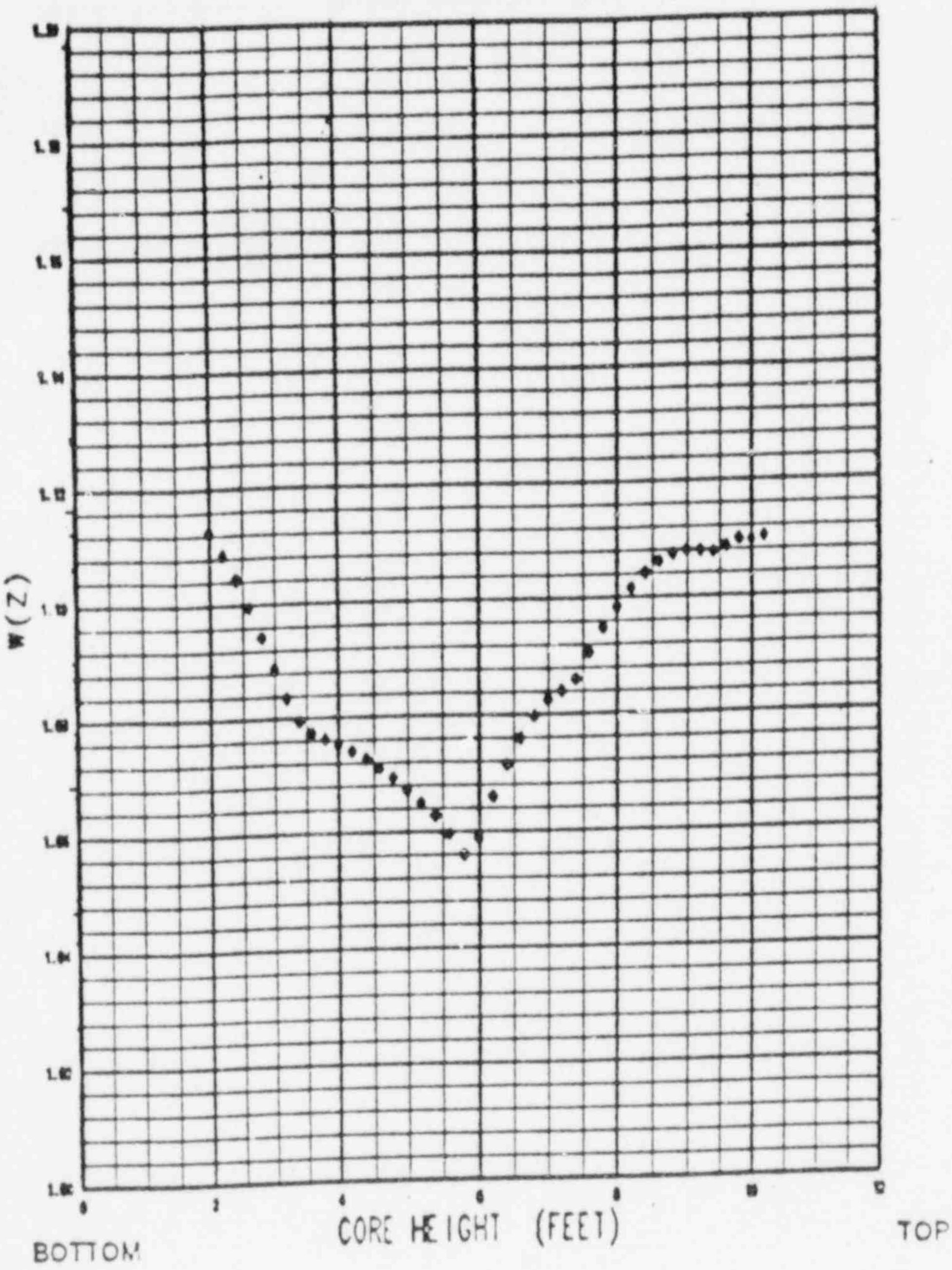
FIGURE 7

V. C. SUMMER BASELOAD W(Z) FOR POWERS BETWEEN 80% AND 100% OF RATED THERMAL POWER WITHIN X % AFD OF THE MEASURED TARGET AT BOL

(150 MWD/MTU)

TOP AND BOTTOM 15% EXCLUDED AS PER TECH SPEC 4.2.2.4G

SAMPLE



	HEIGHT (FT.)	MOL W(Z)
•	0.0	1.0000
•	0.2	1.0000
•	0.4	1.0000
•	0.6	1.0000
•	0.8	1.0000
•	1.0	1.0000
•	1.2	1.0000
•	1.4	1.0000
•	1.6	1.0000
•	1.8	1.0000
	2.0	1.1125
	2.2	1.1086
	2.4	1.1043
	2.6	1.0995
	2.8	1.0943
	3.0	1.0888
	3.2	1.0836
	3.4	1.0785
	3.6	1.0774
	3.8	1.0764
	4.0	1.0754
	4.2	1.0742
	4.4	1.0728
	4.6	1.0712
	4.8	1.0694
	5.0	1.0673
	5.2	1.0650
	5.4	1.0628
	5.6	1.0596
	5.8	1.0560
	6.0	1.0549
	6.2	1.0660
	6.4	1.0715
	6.6	1.0760
	6.8	1.0788
	7.0	1.0826
	7.2	1.0840
	7.4	1.0860
	7.6	1.0906
	7.8	1.0949
	8.0	1.0984
	8.2	1.1015
	8.4	1.1041
	8.6	1.1060
	8.8	1.1072
	9.0	1.1078
	9.2	1.1078
	9.4	1.1076
	9.6	1.1085
	9.8	1.1097
	10.0	1.1095
	10.2	1.1101
•	10.4	1.0000
•	10.6	1.0000
•	10.8	1.0000
•	11.0	1.0000
•	11.2	1.0000
•	11.4	1.0000
•	11.6	1.0000
•	11.8	1.0000
•	12.0	1.0000

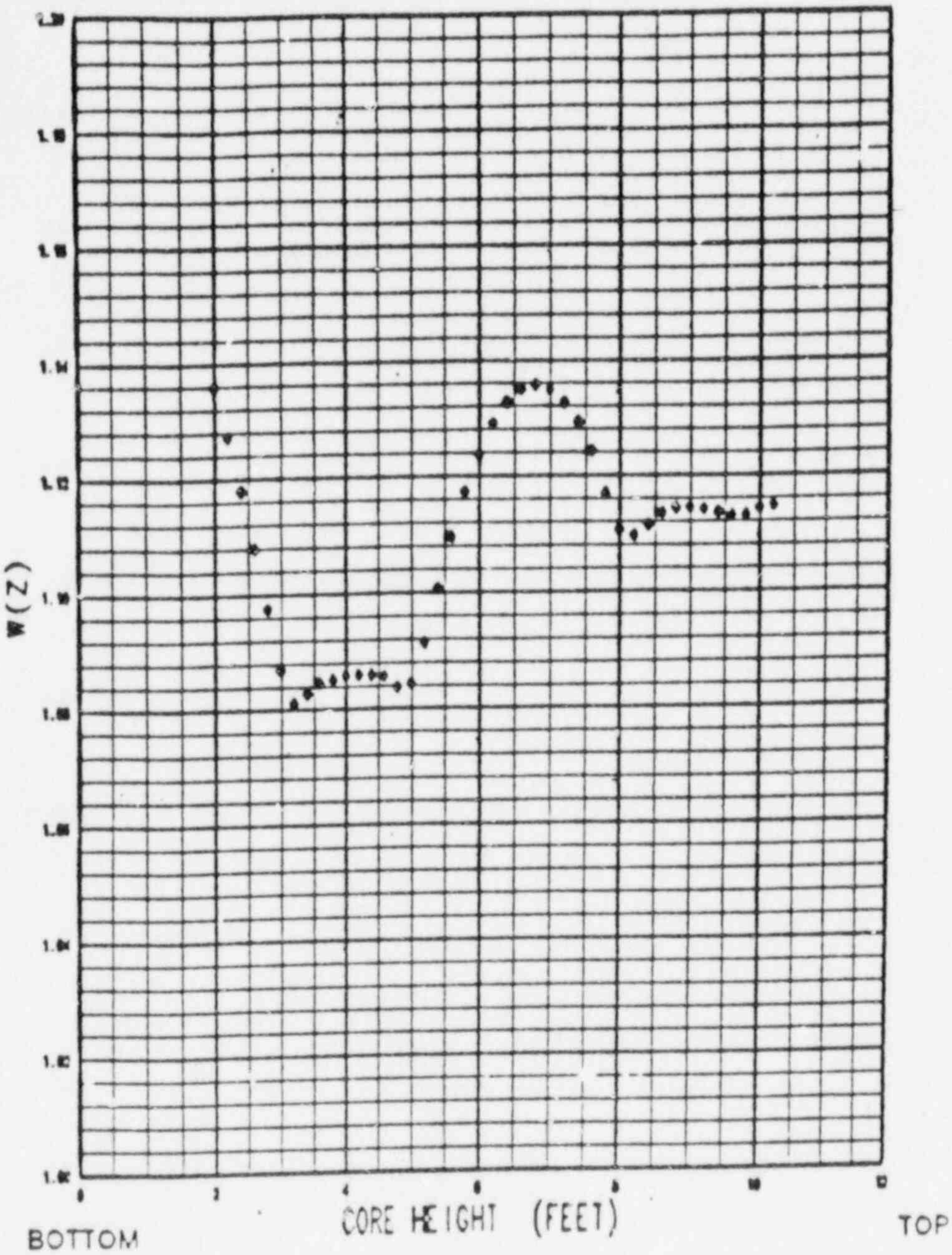
FIGURE 8

V. C. SUMMER BASELOAD W(Z) FOR POWERS BETWEEN 80% AND 100% OF RATED THERMAL POWER WITHIN X % AFD OF THE MEASURED TARGET AT MOL

(8000 MWD/MTU)

TOP AND BOTTOM 15% EXCLUDED AS PER TECH SPEC 4.2.2.4.G

SAMPLE



	HEIGHT (FT.)	EOL W(Z)
*	0.0	1.0000
*	0.2	1.0000
*	0.4	1.0000
*	0.6	1.0000
*	0.8	1.0000
*	1.0	1.0000
*	1.2	1.0000
*	1.4	1.0000
*	1.6	1.0000
*	1.8	1.0000
	2.0	1.1283
	2.2	1.1273
	2.4	1.1178
	2.6	1.1080
	2.8	1.0975
	3.0	1.0870
	3.2	1.0811
	3.4	1.0828
	3.6	1.0846
	3.8	1.0851
	4.0	1.0857
	4.2	1.0859
	4.4	1.0859
	4.6	1.0857
	4.8	1.0838
	5.0	1.0843
	5.2	1.0913
	5.4	1.1006
	5.6	1.1086
	5.8	1.1172
	6.0	1.1237
	6.2	1.1290
	6.4	1.1327
	6.6	1.1350
	6.8	1.1357
	7.0	1.1350
	7.2	1.1327
	7.4	1.1292
	7.6	1.1243
	7.8	1.1165
	8.0	1.1108
	8.2	1.1095
	8.4	1.1114
	8.6	1.1134
	8.8	1.1143
	9.0	1.1145
	9.2	1.1142
	9.4	1.1136
	9.6	1.1130
	9.8	1.1130
	10.0	1.1141
	10.2	1.1147
*	10.4	1.0000
*	10.6	1.0000
*	10.8	1.0000
*	11.0	1.0000
*	11.2	1.0000
*	11.4	1.0000
*	11.6	1.0000
*	11.8	1.0000
*	12.0	1.0000

FIGURE 9

V. C. SUMMER BASELOAD W(Z) FOR POWERS BETWEEN 80% AND 100% OF RATED THERMAL POWER WITHIN X % AFD OF THE MEASURED TARGET NEAR EOL

(14000 MWD/MTU)

TOP AND BOTTOM 15% EXCLUDED AS PER TECH SPEC 4.2.2.4.G

5.0 THERMAL AND HYDRAULIC DESIGN

The analysis of the LOPAR and VANTAGE 5 fuel is based on the Improved Thermal Design Procedure (ITDP) described in Reference 14. The LOPAR fuel analysis uses the WRB-1 DNB correlation in Reference 15 while the VANTAGE 5 fuel utilizes the WRB-2 DNB correlation in Reference 1. These DNB correlations take credit for the significant improvement in the accuracy of the critical heat flux predictions over previous DNB correlations. The WRB-2 DNB correlation also takes credit for the VANTAGE 5 fuel assembly mixing vane design. A DNBR limit of 1.17 is applicable for both the WRB-1 and WRB-2 correlations. In addition, The W-3 DNBR correlation is used where appropriate (e.g., accidents analyzed in Sections 15.2.1 and 15.4.2 of Attachment 3). Table 5.1 summarizes the pertinent thermal and hydraulic design parameters.

The design method employed to meet the DNB design basis is the Improved Thermal Design Procedures which has been approved by the NRC, Reference 16. Uncertainties in plant operating parameters, nuclear, and thermal parameters, and fuel fabrication parameters are considered statistically such that there is at least 95 percent probability at a 95 percent confidence level that the minimum DNBR will be greater than or equal to 1.17 for the limiting power rod. Plant parameter uncertainties are used to determine the plant DNBR uncertainties. These DNBR uncertainties, combined with the DNBR limit, establish a DNBR value which must be met in plant safety analyses. Since the parameter uncertainties are considered in determining the design DNBR value, the plant safety analyses are performed using values of input parameters without uncertainties. For this application, the minimum required DNBR values for the LOPAR fuel analysis are a 1.35 for thimble cold wall cells (three fuel rods and a thimble tube) and 1.36 for typical cell (four fuel rods). The design DNBR values for the VANTAGE 5 fuel are a 1.32 and a 1.33 for thimble and typical cells, respectively.

In addition to the above considerations, a plant-specific DNBR margin has been considered in the analyses. In particular, safety analysis DNBR limits of 1.44 for thimble and 1.48 for typical cells for LOPAR fuel, and 1.60 and 1.68 for thimble and typical cells respectively for the VANTAGE 5 fuel, were employed in the safety analyses. The DNBR margin between the DNBRs used in the safety analyses and the design DNBR values is broken down as follows. A fraction of the margin is utilized to accommodate the transition core penalty (12.5% for VANTAGE 5 fuel and none for LOPAR fuel) and the appropriate fuel rod bow DNBR penalty, Reference 10, which is less than 1.3%. The existing 6.3% margin in the LOPAR fuel and 17.5% margin in the VANTAGE 5 fuel between the design and safety analysis DNBR limits also includes a greater than 5% DNBR margin in the LOPAR fuel and a greater than 3.7% DNBR margin in the VANTAGE 5 fuel reserved for flexibility in the design.

The LOPAR and VANTAGE 5 designs have been shown to be hydraulically compatible in Reference 1.

The major impact of thimble plug removal on the thermal-hydraulic analysis is the increase in bypass flow which is reflected in Table 5.1.

The phenomena of fuel rod bowing, as described in Reference 10, must be accounted for in the DNBR safety analysis of Condition I and Condition II events for each plant application. Internal to the fuel rod, the IFBA and fuel pellet designs are not expected to increase the propensity for fuel rods to bow. External to the VANTAGE 5 fuel rod, the Inconel non-mixing vane and Zircaloy mixing vane grids provide fuel rod support. Additional restraint is provided with the Intermediate Flow Mixer (IFM) grids. Applicable generic credits for margin resulting from retained conservatism in the evaluation of DNBR are used to offset the effect of rod bow. The safety analysis for the V. C. Summer Plants maintain sufficient margin between the safety analysis limit DNBRs and the design limit DNBRs to accommodate full-flow and low-flow DNBR penalties.

The Westinghouse transition core DNB methodology is given in References 2 and 17 and has been approved by the NRC via Reference 18. Using this methodology, transition cores are analyzed as if they were full cores of one assembly type (full LOPAR or full VANTAGE 5), applying the applicable transition core penalties. This penalty is included in the safety analysis limit DNBRs such that sufficient margin over the design limit DNBR exists to accommodate the transition core penalty and the appropriate rod bow DNBR penalty.

The fuel temperatures for use in safety analysis calculations for the VANTAGE 5 fuel are evaluated using the same methods as those used to evaluate the LOPAR fuel. Westinghouse uses the PAD performance code described in Reference 6 to perform both design and licensing calculations. When the code is used to calculate fuel temperatures to be used as initial conditions in safety analyses, a conservative thermal safety model, Reference 7, is used.

TABLE 5.1

V. C. SUMMER THERMAL AND HYDRAULIC DESIGN PARAMETERS

Thermal and Hydraulic Design Parameters		Design Parameters
(Using ITDP)		
Reactor Core Heat Output, MWt		2775
Reactor Core Heat Output, 10^6 BTU/Hr		9469
Heat Generated in Fuel, %		97.4
Core Pressure, Nominal, psia		2280
Radial Power Distribution	(LOPAR)	1.56 [1+0.3(1-P)]*
	(V-5)	1.62 [1+0.3(1-P)]*
Limit DNBR for Design Transients		
Typical Flow Channel	(LOPAR)	1.48
	(V-5)	1.68
Thimble (Cold Wall) Flow Channel	(LOPAR)	1.44
	(V-5)	1.60
DNB Correlation	(LOPAR)	WRB-1
	(V-5)	WRB-2

* The 4% radial power uncertainty has been removed for statistical combination with other uncertainties in the ITDP analysis.

TABLE 5.1 (Continued)

V. C. SUMMER THERMAL AND HYDRAULIC DESIGN PARAMETERS

HFP Nominal Coolant Conditions		Design Parameters
Vessel Minimum Measured Flow+		
Rate (including Bypass), 10^6 lbm/hr		106.2
GPM		283,500
Vessel Thermal Design Flow+		
Rate (including Bypass), 10^6 lbm/hr		104.1
GPM		277,800
Core Flow Rate		
(excluding Bypass, based on TDF)		
10^6 lbm/hr		94.8
GPM		253,080
Fuel Assembly Flow Area++		
for Heat Transfer, ft^2	(LOPAR)	41.55
	(V-5)	44.04
Core Inlet Mass Velocity,		
10^6 lbm/hr-ft (Based on TDF)	(LOPAR)	2.28
	(V-5)	2.15

+ Includes 15% steam generator tube plugging

++ Assumes all LOPAR or VANTAGE 5 core

TABLE 5.1 (Continued)

V. C. SUMMER THERMAL AND HYDRAULIC DESIGN PARAMETERS

Thermal and Hydraulic Design Parameters		Design Parameters
(Based on Thermal Design Flow)		
Nominal Vessel/Core Inlet Temperature, °F		552.3
Vessel Average Temperature, °F		585.5
Core Average Temperature, °F		590.5
Vessel Outlet Temperature, °F		618.7
Average Temperature Rise in Vessel, °F		66.4
Average Temperature Rise in Core, °F		72.0
Heat Transfer		
Active Heat Transfer Surface Area,++ ft^2	(LOPAR)	43,598
	(V-5)	46,779
Average Heat Flux, $BTU/hr\text{-}ft^2$	(LOPAR)	189,820
	(V-5)	197,200
Average Linear Power, kw/ft		5.45
Peak Linear Power for Normal Operation,+++ kw/ft		13.30

++ Assumes all LOPAR or VANTAGE 5 core

+++ Based on 2.45 F_Q peaking factor

6.0 ACCIDENT EVALUATION

6.1 Non-LOCA Accidents

This section addresses the impact of the VANTAGE 5 design features and modified safety analysis assumptions for the V. C. Summer Plant non-LOCA accident analyses.

6.1.1 VANTAGE 5 Design Features

The design features of VANTAGE 5 fuel, considered in the non-LOCA analysis are:

- Fuel Rod Dimensions
- Axial Blankets
- Integral Fuel Burnable Absorbers (IFBAs) and Wet Annular Burnable Absorbers (WABA)
- Intermediate Flow Mixer Grids (IFMs)
- Reconstitutable Top Nozzle
- Fuel Enrichment
- Extended Burnup Fuel Assembly Design

A brief description of each of these and its consideration in the safety analyses follows.

Fuel Rod Dimensions

The VANTAGE 5 fuel rod dimensions which determine the safety analysis temperature versus linear power density relationship include rod diameter, pellet diameter, initial pellet-to-clad gap size, and stack height. The non-LOCA safety analysis fuel temperature and rod geometry assumptions consider this geometry change and bound both LOPAR (Standard) and VANTAGE 5 fuel.

Axial Blankets and IFBAs

Axial blankets reduce power at the ends of the rod which increases axial peaking at the interior of the rod. Used alone, axial blankets reduce DNB margin, but the effect may be offset by the presence of part length Integral Fuel Burnable Absorbers (IFBAs) which flatten the power distribution. The net effect on the axial shape is a function of the number and configuration of IFBAs in the core and time in life. The effects of axial blankets and IFBAs on the reload safety analysis parameters are taken into account in the reload design process. The axial power distribution assumption in the safety analyses kinetics calculations have been determined to be applicable for evaluating the introduction of axial blankets in the V. C. Summer plant.

IFM Grids and Reconstitutable Top Nozzle

The IFM grid feature of the VANTAGE 5 fuel design increases DNB margin. The fuel safety analysis limit DNBR values contain significant DNB margin (see Section 4.0). This DNB margin was set to ensure that the core thermal safety limits for the VANTAGE 5 fuel with an F_{WH} of 1.68 are acceptable. However, for the transition cycles the LOPAR fuel core limits with an F_{WH} of 1.62 are more restrictive than the VANTAGE 5 fuel core limits. Thus the most restrictive core limits correspond to the LOPAR fuel design. Any transition core penalty is accounted for with the available DNBR margin.

The IFM grid feature of the VANTAGE 5 fuel design increases the core pressure drop. The control rod scram time to the dashpot is increased from 2.3 to 2.7 seconds. The increased drop time primarily affects the fast reactivity transients. These accidents were reanalyzed for this report. The revised safety analysis assumption was incorporated in all the reanalyzed events requiring this parameter and the remaining transients have been evaluated.

Core flow areas and loss coefficients were preserved in the design of the reconstitutable top nozzle. As such, no parameters important to non-LOCA safety analyses are impacted.

Fuel Enrichment

The VANTAGE 5 fuel design increased fuel enrichment is conservatively bounded by the maximum safety analysis assumptions.

Extended Burnup Fuel Assembly Design

WCAP-10125-P-A, "Extended Burnup Evaluation of Westinghouse Fuel," evaluates the impact of extended burnup on the design and operation of Westinghouse fuel. The major effect of the extended burnup rod design is on power shaping between fresh and burned assemblies.

6.1.2 Modified Safety Analysis Assumptions

Listed below are the analysis assumptions which represent a departure from that currently used for V. C. Summer.

- Changes in Moderator Temperature Coefficient (Most Positive, Most Negative)
- Increased Design Enthalpy Rise Hot Channel Factor (F_{WH}) and F_Q
- Increase in Allowable Steam Generator Tube Plugging Level
- Reactor Coolant System Flow Reduction
- Thimble Plug Deletion
- Debris Filter Bottom Nozzle
- Increased Overpower/Overtemperature WT Reactor Trip Response Time
- Improved Thermal Design Procedure

A brief description of each of these assumptions follows.

Changes in Moderator Temperature Coefficient (Most Positive, Most Negative)

A positive moderator temperature coefficient (PMTC) of +7 pcm/degree F from 0% to 70% power and decreasing linearly to 0 pcm/degree F at 100% power was incorporated into the safety analyses performed for this report.

In general, the analyses presented are based on a +7 pcm/degree F moderator temperature coefficient, which is assumed to remain constant for variations in temperature. Exceptions are rod ejection and rod withdrawal from subcritical which are based on a MTC of +7 pcm/degree F at zero power nominal average temperature and which, due to moderator temperature feedback modeled in the TWINKLE diffusion-theory code, becomes less positive for higher temperatures.

Incorporation of the described level of PMTC into the safety analyses is, in all cases, a conservative assumption for this report.

In order to accommodate longer fuel cycles and extended fuel burnup, a negative moderator temperature coefficient of -50 pcm/degree F corresponding to end of life, full power conditions was conservatively incorporated into the safety analyses performed for this report.

Increased Design Enthalpy Rise Peaking Factor ($F_{\Delta H}$) and F_Q

The $F_{\Delta H}$ for the LOPAR and Vantage 5 fuel during the transition cycles is 1.62. The non-LOCA calculations applicable for the VANTAGE 5 core have assumed a full power $F_{\Delta H}$ of 1.68. This is a conservative safety analysis assumption for this report.

The design core limits for this report incorporate the increased $F\Delta H$ for both the LOPAR and VANTAGE 5 fuel.

The inc[illegible]e in the Technical Specification maximum LOCA F_Q from 2.25 to 2.45 for both LOPAR and VANTAGE 5 fuel is conservatively accounted for in the non-LOCA transients.

Increased Steam Generator Tube Plugging

All non-LOCA safety analyses reanalyzed for this report have incorporated up to a maximum of 15% plant total steam generator tube plugging. It is assumed that no one steam generator exceeds 15% tube plugging.

Reactor Coolant System Flow Reduction

All non-LOCA safety analyses reanalyzed for this report have incorporated a reduction in the reactor coolant system flow. The reduced flow corresponds to a thermal design flow of 92600 gpm/loop, and a minimum measured flow of 94500 gpm/loop.

Thimble Plug Deletion

The non-LOCA analyses performed incorporated the impact of thimble plug deletion. Thimble plug deletion affects core pressure drops and bypass flow. These effects have been conservatively incorporated into the non-LOCA safety analyses.

Debris Filter Bottom Nozzle

The VANTAGE 5 fuel design will also include the Debris Filter Bottom Nozzle (DFBN). In the DFBN, the relatively large flow holes in the conventional bottom nozzle are replaced with a new pattern of smaller flow holes. These holes are sized to minimize the passage of debris particles large enough to cause damage while still providing sufficient flow area, comparable pressure drop, and continued structural integrity of the nozzle. As such, no parameters important to the non-LOCA safety analyses are impacted.

Increased Overpower/Overtemperature ΔT Reactor Trip Response Time

The total time delay of the overtemperature ΔT and overpower ΔT trips (including RTD time response, trip circuitry and channel electronics delay) assumed in the non-LOCA analyses is 8.5 seconds. The 8.5 second delay includes a 7 second first order lag incorporated into the determination of the time at which the overtemperature ΔT and overpower ΔT trip setpoints are reached. The remaining 1.5 seconds is the delay from the time at which the trip signal is initiated until the rod cluster control assemblies are free to drop into the core.

Improved Thermal Design Procedure

The calculational method utilized to meet the DNB design basis is the ITDP, discussed in Reference 14. Uncertainties in plant operating parameters are statistically treated such that there is at least a 95 percent probability at a 95 percent confidence level that the minimum DNBR will be greater than 1.17. Since the parameter uncertainties are considered in determining the design DNBR value, the plant safety analyses are performed using nominal input parameters without uncertainties.

The LOPAR fuel DNB analyses use the WRB-1 correlation, while the VANTAGE 5 fuel analyses use the WRB-2 correlation.

6.1.3 Non-LOCA Safety Evaluation Methodology

The non-LOCA safety evaluation process is described in References 1 and 2. The process determines if a core configuration is bounded by existing safety analyses in order to confirm that applicable safety criteria are satisfied. The methodology systematically identifies parameter changes on a cycle-by-cycle basis which may invalidate existing safety analysis assumptions and identifies the transients which require reevaluation. This methodology is applicable to the evaluation of VANTAGE 5 transition and full cores.

Any required reevaluation identified by the reload methodology is one of two types. If the identified parameter is only slightly out of bounds, or the transient is relatively insensitive to that parameter, a simple evaluation may be made which conservatively evaluates the magnitude of the effect and explains why the actual analysis of the event does not have to be repeated. Alternatively, should the deviation be large and/or expected to have a significantly or not easily quantifiable effect on the transients, reanalyses are required. The reanalysis approach will typically utilize the analytical methods which have been used in previous submittals to the NRC. These methods are those which have been presented in FSARs, subsequent submittals to the NRC for a specific plant, reference SARs, or report submittals for NRC approval.

The key safety parameters are documented in Reference 5. Values of these safety parameters which bound both fuel types (LOPAR and VANTAGE 5) were assumed in the safety analyses. For subsequent fuel reloads, the key safety parameters will be evaluated to determine if violations of these bounding values exist. Reevaluation of the affected transients would take place and would be documented for the cycle-specific reload design, as per Reference 5.

6.1.4 Conclusions

Descriptions of the transients reanalyzed for this report, method of analysis, results, and conclusions are contained in Attachment 3. The analytical procedures and computer codes used are identified in Section 15.1. Attachment 3 has been prepared conforming to the format of the V. C. Summer FSAR.

For each of the accidents reanalyzed, it was found that the appropriate safety criteria are met. In addition, evaluations have been performed regarding the impact of VANTAGE 5 fuel on the steam line break mass and energy release analyses, both inside and outside containment. The results of this evaluation verify that the mass and energy releases previously calculated, are not adversely impacted by the transition to VANTAGE 5 fuel.

6.2 LOCA Accidents

6.2.1 Large Break LOCA

6.2.1.1 Description of Analysis/Assumptions for 17x17 VANTAGE 5 Fuel

Consistent with the methodology developed in the VANTAGE 5 Reference Core Report (Reference 1), a large break loss-of-coolant accident (LOCA) analysis based on a full VANTAGE 5 core has been performed to define peaking factor limits for use during and subsequent to the transition to VANTAGE 5 fuel at V. C. Summer. The Westinghouse 1981 Evaluation Model with BASH (Reference 19) was utilized for a spectrum of cold leg breaks. Key assumptions include:

- Core thermal power of 2775 MWt.
- 15% uniform steam generator tube plugging.
- A $F_{\Delta H}$ of 1.62.
- Fuel data based on the Revised Thermal Model (Reference 7).
- A limiting chopped cosine power shape (Reference 26).

During the transition period, a PCT penalty is applied to the full VANTAGE 5 core for the purpose of demonstrating conformance with the 10CFR50.46 PCT limit of 2200°F. (See Section 5.2.3)

Reference 25 states three restrictions related to the use of the 1981 Evaluation Model + BASH calculational model. The application of these restrictions to the plant specific large break LOCA analysis was addressed with the following conclusions:

V. C. Summer is neither an Upper Head Injection (UHI) nor an Upper Plenum Injection (UPI) plant, so Restriction 1 does not apply.

V. C. Summer Plant specific LOCA analysis analyzed both minimum and maximum ECCS cases to address Restriction 2. The C_d = 0.4 Double Ended Cold Leg Guillotine (DECLG) with minimum ECCS flows was found to result in most limiting consequences.

Generic sensitivity studies have been performed by Westinghouse which justify the continued use of the chopped cosine power shape as limiting for 3-loop plants which addresses restriction 3.

6.2.1.2 Method of Analysis

The methods used in analyzing the V. C. Summer Power Plant for VANTAGE 5 fuel, including computer codes used and assumptions, are described in detail in Attachment 4, Section 15.4.1.1.2.

6.2.1.3 Results

The Double Ended Cold Leg Guillotine (DECLG, CD=0.4) with minimum ECCS was found to result in the most limiting consequences. The peak clad temperature was 2141°F at a total peaking factor of 2.45. The maximum local metal-water reaction was 10.13%, and the total core wide metal-water reaction was less than 0.3% for all cases analyzed. The clad temperature transients turned around at a time when the core geometry was still amenable to cooling.

The impact of the transition core cycles was conservatively assumed to be a 50°F increase in calculated peak cladding temperature which would yield a transition core PCT of 2191.0°F. The transition core penalty can be accommodated by the margin to the 10CFR50.46 limit of 2200°F.

The results of this analysis, including tabular and plotted results of the break spectrum analyzed, are provided in Appendix C which has been prepared using the NRC Standard Format and Content Guide, Regulatory Guide 1.70, Revision 1 for accidents applicable to the V. C. Summer plant.

6.2.1.4 Conclusions

The large break LOCA analysis performed for the V. C. Summer Power Plant has demonstrated that for breaks up to a double-ended severance of the reactor coolant piping, the Emergency Core Cooling System (ECCS) will meet the acceptance criteria of Title 10 CFR Part 50 Section 46, that is:

1. The calculated peak cladding temperature will remain below the required 2200°F.

2. The [illegible] fuel cladding that reacts chemically with the water or steam does [illegible] ed one percent of the total fuel rod cladding.

3. The localized cladding oxidation limit of 17 percent is not exceeded during or after quenching.

4. The core remains amenable to cooling during and after the LOCA.

5. The core temperature is reduced and decay heat is removed for an extended period of time. This is required to remove the heat produced by the long-lived radioactivity remaining in the core.

Thus, the ECCS analysis for the V. C. Summer Power Plant is in compliance with the requirements of 10CFR50.46 including consideration for transition core configurations.

6.2.2 Small Break LOCA

6.2.2.1 Description of Analysis/Assumptions for 17x17 VANTAGE 5

Consistent with the methodology developed in the VANTAGE 5 Reference Core Report (Reference 1), a small break loss-of-coolant accident (LOCA) analysis was performed assuming a full core of VANTAGE 5 fuel to determine the peak clad temperature. The currently approved Small Break ECCS Evaluation Model, using NOTRUMP, Reference 21, was utilized for a spectrum of cold leg breaks. Attachment 4, Section 15.3.1, includes a full description of the analysis and assumptions utilized. Key assumptions include an $F_{\Delta H}$ of 1.68, a total peaking factor corresponding to 2.5 at the core mid-plane, 15% uniform steam generator tube plugging, and a core thermal power level of 2775 MWt.

Sensitivity studies performed using the NOTRUMP small break evaluation model have demonstrated that VANTAGE 5 fuel is more limiting than OFA fuel in calculated ECCS performance. Similar studies using the WFLASH evaluation model, Reference 22, have previously shown that OFA fuel is more limiting than LOPAR fuel. For the small break LOCA, the effect of the fuel difference is more pronounced during core uncovery periods and, therefore, shows up predominantly in the LOCTA-IV calculation in the evaluation model analysis. Consequently, the previous conclusion drawn from the WFLASH studies, regarding the fuel difference, may be extended to this NOTRUMP analysis. Thus, only VANTAGE 5 fuel was analyzed, since it is the more limiting of the two types of fuel residing in the core.

6.2.2.2 Method of Analysis

The methods of analysis, including codes used and assumptions, are described in detail in Attachment 4, Section 15.3.1.

6.2.2.3 Results

The small break VANTAGE 5 LOCA analysis for the V. C. Summer Power Plant, utilizing the currently approved NOTRUMP Evaluation Model, resulted in a peak clad temperature of 2095°F for the 3.0 inch diameter cold leg break. The analysis assumed the limiting small break power shape consistent with a LOCA F_Q envelope of 2.50 at core midplane elevation and 2.26 at the top of the core. The maximum local metal-water reaction is 5.69 percent, and the total core metal-water reaction is less than 0.3 percent for all cases analyzed. The clad temperature transients turn around at a time when the core geometry is still amendable to cooling. These results are applicable for V. C. Summer with a full core of VANTAGE 5 fuel and with transition cores.

6.2.2.4 Conclusions

Analyses presented in Attachment 4, Section 15.3 show that one centrifugal charging pump, together with the accumulators, provide sufficient core flooding to keep the calculated peak clad temperature well below the required limits of 10CFR50.46. It can also be seen that the ECCS analysis remains in compliance with all other requirements of 10CFR50.46. Adequate protection is therefore afforded by the ECCS in the event of a small break LOCA.

6.2.3 Transition Core Effects on LOCA

The V. C. Summer large and small break analysis have been performed in accordance with the transition core LOCA methodology defined in Section 5.2.3 of VANTAGE 5 Reference Core Report. These analyses are based on a full core of VANTAGE 5. To cover a transition core, the maximum PCT penalty of 50°F has been applied to the full VANTAGE 5 core results for large breaks to ensure conformance with the 10CFR50.46 PCT limit of 2200°F. Application of this maximum penalty, conservatively accounts for the potential increases in PCT due to the effects of mixed core hydraulic resistance mismatch as described in Reference 2. No PCT penalty has been applied to the small break results since mixed core hydraulic resistance mismatch is not a significant factor for the analysis. Following the transition period, the large break LOCA analysis will apply without the 50°F PCT penalty.

6.2.4 Containment Integrity Mass and Energy Releases

The extent to which fuel changes can impact containment mass and energy releases, used to determine containment peak pressure, is dependent upon changes to:

1) Core fluid volume.
2) Core stored energy.
3) Core hydraulic resistance.

The VANTAGE 5 fuel design utilizes a fuel rod of smaller diameter than the 17x17 LOPAR (standard) fuel presently installed in the V. C. Summer Power Plant. This smaller fuel rod diameter leads to a reduction in core stored energy which is beneficial in reducing the mass and energy releases calculated for a hypothetical LOCA. The small VANTAGE 5 fuel rod will also result in a slight increase in core fluid volume; and, the use of Intermediate Flow Mixing grids will increase hydraulic resistance. These changes are offset by the reduction in core stored energy. Based on this offset, a reanalysis of containment integrity mass and energy releases is not necessary for the implementation of VANTAGE 5 fuel at the V. C. Summer Power Plant. Thus, the implementation of VANTAGE 5 fuel at the V. C. Summer Power Plant will not result in an increase in the containment peak pressure reported in the V. C. Summer FSAR or increase the offsite radiological consequences associated with high containment pressures resulting from a hypothetical LOCA.

6.2.5 Steam Generator Tube Rupture

The consequences of a Steam Generator Tube Rupture (SGTR), as analyzed in the V. C. Summer Power Plant FSAR, are dependent upon the initial reactor and steam generator conditions of power, pressure, and temperature. Changes in initial operating conditions as a result of implementation of VANTAGE 5 fuel at V. C. Summer Power Plant have been evaluated and concluded that the consequences of a SGTR will not be increased by the implementation of VANTAGE 5 fuel. Thus, a reanalysis of the FSAR Steam Generator Tube Rupture was determined to be unnecessary for the implementation of VANTAGE 5 fuel and the current FSAR SGTR analysis remains applicable.

6.2.6 Blowdown Reactor Vessel and Loop Forces

The forces created by a hypothetical break in the RCS piping are principally caused by the motion of the decompression wave through the RCS. The strength of the decompression wave is primarily a result of the assumed break opening time, break area, and RCS operating conditions of power, temperature, and pressure. These parameters will not be affected by a change in fuel at the V. C. Summer Power Plant from 17x17 Standard to VANTAGE 5. The forces in the vicinity of the core are affected by the core flow area/volume. An increase in core flow area/volume will tend to more effectively dissipate the decompression wave resulting in a reduction of the forces acting on the reactor vessel internals. VANTAGE 5 fuel, having a smaller rod diameter than 17x17 standard fuel, increases the core flow area and volume which is beneficial in reducing forces associated with a hypothesized LOCA. Forces acting on the RCS loop piping as a result of a hypothesized LOCA are not influenced by changes in fuel assembly design. Thus, the implementation of VANTAGE 5 fuel at the V. C. Summer Power Plant will not result in an increase of the calculated consequences of a hypothesized LOCA on the reactor vessel internals or RCS loop piping. The current FSAR analysis for forces on the reactor internals and RCS piping resulting from a hypothesized LOCA remains applicable to the application of VANTAGE 5 fuel at the V. C. Summer Power Plant.

6.2.7 Post-LOCA Long-Term Core Cooling (ECCS Flows, Core Subcriticality, and Switchover of the ECCS to Hot Leg Recirculation)

The implementation of VANTAGE 5 fuel at the V. C. Summer Power Plant does not impact the assumptions for decay heat, core reactivity, or boron concentration for sources of water residing in the containment sump Post-LOCA. Thus, these licensing requirements associated with LOCA are not significantly affected by the implementation of VANTAGE 5 fuel. Additionally, Westinghouse performs an independent check on core subcriticality for each fuel cycle operated at the V. C. Summer Plant.

7.0 SUMMARY OF TECHNICAL SPECIFICATION CHANGES

The proposed changes to the Virgil C. Summer Nuclear Station (VCSNS) Technical Specifications are summarized in Table 7.1. These changes reflect the impact of the design, analytical methodology, and safety analysis assumptions outlined in the SCE&G amendment request and are given in the proposed Technical Specification page changes (see Attachment 2 of this report). A brief overview of the significant changes follows.

7.1 Core Safety Limits

Core safety limits and associated bases for 3-loop operation during modes 1 and 2 are revised to reflect the impact of the transition to VANTAGE 5 with:

1. The use of ITDP and the WRB-1 and WRB-2 DNB Correlation.

2. An $F_{\Delta H}$ of 1.62 (see Section 7.11).

3. Reduced RCS flow to accommodate the increased resistance of the VANTAGE 5 fuel assembly and 2% flow margin over and above that required to support SG tube plugging up to 15% (see Section 7.2).

The proposed limits corresponds to those for the LOPAR fuel which are limiting during the transition period. Less limiting values will be possible with a full core of VANTAGE 5.

7.2 Thermal Design Flow

The VCSNS thermal design flow is being decreased from 96,200 gpm per loop to 92,600 gpm per loop. This flow reduction accommodates:

a. The increased resistance of the VANTAGE 5 fuel assembly.

b. Up to 15% SG tube plugging in all three SG's.

c. 2% additional flow margin.

The revision proposed for Table 2.2-1 corresponds to the minimum measured flow value used as input to the ITDP DNBR analyses for the loss of flow event.

The reduced thermal design flow has also been factored into the limiting conditions of operation defined by RCS flow and the Nuclear Enthalpy Rise Hot Channel Factor (see Section 7.11) within Technical Specification 3.2.3. Indicated RCS flow is derived from the reduced thermal design flow based on VCSNS's currently approved flow measurement uncertainty of 2.1%.

7.3 <u>OPΔT/OTΔT Setpoints</u>

Revisions to the limiting safety-system settings for the thermal overpower ΔT and overtemperature ΔT trip functions are proposed to maintain consistency with the non-LOCA Accident Analyses provided in the Transition Safety Evaluation. These trip functions provide primary protection against departure from nucleate boiling and fuel centerline melting (excessive kw/ft) during postulated transients. The proposed settings have been derived consistent with WCAP-8745, "Design Bases for the Thermal Overpower ΔT and Thermal Overtemperature ΔT Trip Functions", based on the core safety limits (see Section 7.1) for the VANTAGE 5 transition and with instrument uncertainties accounted for.

7.4 <u>Shutdown Margin for Modes 3, 4, and 5</u>

Figure 3.1-3 of the VCSNS Technical Specifications defines shutdown margin requirements as a function of average RCS boron concentration during Modes 3, 4, and 5. The proposed revisions are based on the reanalyses of the boron dilution event with VANTAGE 5 fuel (see the Transition Safety Evaluation) and are required to maintain the current bases of the Technical Specification.

No new technology was employed in the VANTAGE 5 Boron Dilution analysis. The revised limits reflect solely the impact of VANTAGE 5; the more negative boron worths and higher initial boron concentrations (both initial and critical) are the primary factors leading to the modified shutdown margin requirements.

7.5 Moderator Temperature Coefficient

Revisions to the VCSNS Technical Specifications for Moderator Temperature coefficient are proposed.

The BOL limits are increased from 0 $\Delta K/K/°F$ for the all rods withdrawn (BOL, HZP) to +7.0 pcm/°F from HZP to 70% rated power with a negative linear ramp from +7.0 pcm/°F at 70% power to 0.0 pcm/°F at HFP. This change is required due to the increased RCS boron concentrations for VANTAGE 5 and the positive shift in moderator coefficient caused by the larger H/U ratio and small MTU loading for the smaller (compared to LOPAR) VANTAGE 5 fuel rod.

The EOL limits are also increased (more negative) to accommodate longer fuel cycles and extended fuel burnup.

7.6 Borated Water Sources

The technical specification requirements for borated water sources during all operating modes were evaluated to determine if current limits remain applicable for the transition to VANTAGE 5 fuel. The bases for the technical specification were preserved by current limits except for the minimum contained borated water volume in the boric acid storage system during Modes 1-4. An increase in the minimum water volume of 100 gallons (from 13200 gallons to 13300 gallons) is requested to assure minimum shutdown margin from full power equilibrium xenon conditions.

In conjunction with the above, the examples of maximum expected boration capabilities in the bases section are deleted. These examples were based on operation with LOPAR fuel and are not applicable for the transition to VANTAGE 5 fuel. Margin to the technical specification limits will be confirmed in the future on a cycle specific bases.

7.7 Rod Drop Time

The VANTAGE 5 guide thimbles are identical to those in the LOPAR design except for a reduction in the guide thimble diameter and length above the dashpot.

The reduction to the guide tube diameters is required due to the thicker zircaloy grid straps and reduced cell size; whereas; the VANTAGE 5 thimble tube is shorter due to the reconstitutable top nozzle feature. To accommodate these changes the scram time to the dashpot for accident analyses is increased from 2.3 seconds to 2.7 seconds for the transition to VANTAGE 5. The increased rod drop time has been used in the safety analyses provided in the Transition Safety Evaluation.

7.8 Axial Flux Difference

Axial power distribution control at VCSNS is currently achieved by following the Constant Axial Offset Control (CAOC) procedure. In conjunction with the transition to VANTAGE 5, SCE&G proposes to replace CAOC with a modified version of the NRC approved Relaxed Axial Offset Control (RAOC) procedure described in WCAP-10216-PA. The proposed implementation at VCSNS includes:

a. Operation within the standard RAOC AFD limits as a function of power.

b. Option to operate in a base load mode above a minimum allowable power level with AFD maintained within a specified target band about a target flux difference.

c. Use of burnup dependent AFD limits and baseload target band widths on a cycle specific basis. These valves would be supplied in the peaking factor limit report.

The Modified RAOC Technical Specification is a logical extension of the Westinghouse Standard RAOC Technical Specification. Basically, this concept involves the removal of the "Axial Flux Difference Limits as a function of Rated Thermal Power" figure from the Technical Specifications. The AFD Technical Specification is then modified to reference a cycle specific Peaking Factor Limit Report (PFLR) as the reference document for the AFD limit figure(s).

The base load option provides the additional capability to make use of the reduction in core peaking factors that result due to the limited amount of xenon skewing that occurs during operation at or near equilibrium conditions.

7.9 Heat Flux Hot Channel Factor - F_Q (z)

It is proposed to increase the VCSNS F_Q limit from 2.25 to 2.45 for greater flexibility and to accommodate the axially heterogeneous aspects (blankets and short burnable absorbers) of the VANTAGE 5 core. Furthermore, the K(z) curve, which defines the axial dependancy of F_Q, is modified to remove the third line segment applicable to the top of the core.

The full power F_Q limit value of 2.45 was selected to support steam generator tube plugging level of up to 15% while still limiting large break LOCA peak clad temperature values to less than 2200°F, with transition core penalties included.

The axial power profile used to perform the VCSNS small-break LOCA analysis was derived using the recently improved Westinghouse power shape methodology. Among the most notable aspects of this methodology are the use of a comprehensive data base and the elimination of the third line segment from the K(z) Technical Specification curve.

7.10 F_Q Surveillance

Another requested Technical Specification change revises the surveillance technique for the heat flux hot channel factor from F_{xy} (z) to the NRC approved F_Q (z) Surveillance described in WCAP-10216-PA. This revised surveillance procedure for the total peaking factor, F_Q, will increase plant operating flexibility while more directly monitoring the parameter of interest.

It is further proposed to supply the height dependent analytical factor W (z) on a burnup dependent, cycle specific bases within the Peaking Factor Limit Report to take advantage of changes in the axial flux and xenon concentration distributions which occur with burnup and from cycle to cycle.

7.11 Nuclear Enthalpy Rise Hot Channel Factor

The following $F_{\Delta H}$ values (includes uncertainties) are proposed for the VANTAGE 5 transition.

$$F_{\Delta H} = 1.56 [1 + 0.3 (1-P)]$$

where P is the fraction of full power. These higher values allow increased fuel cycle design flexibility and lower leakage core loading patterns.

7.12 DNB Parameters

The proposed limits on DNB related parameters (T_{avg} and Pressurizer Pressure) assure that each are maintained within the normal steady state envelope of operation assumed in the transient and accident analyses. The proposed revisions are consistent with new accident analyses supplied in the Transient Safety Evaluation which utilizes the ITDP (see Section 5.0) for DNB evaluations.

The T_{avg} reflects the nominal baseline T_{avg} of 585.5°F assumed in the VANTAGE 5 analysis in order to support full power operation with:

1. 15% uniform SG tube plugging.

2. A thermal design flow conservatively reduced 2% below that required to support 15% SG tube plugging and the use of VANTAGE 5 fuel.

3. No increase in the V. C. Summer original T_{HOT} of 618.7°F.

The analyses provided herein either reflect the use of or are conservative relative to the proposed T_{avg} of 585.5°F.

7.13 OTΔT,/OPΔT Trip Response Times

To reflect the assumptions used in the VCSNS analyses performed to support the transition to VANTAGE 5 fuel, a response time of 8.5 seconds for the overtemperature ΔT and overpower ΔT is requested. It corresponds to the total time delay (including RTD and thermowell time response, trip circuit and channel electronic delay) from the time the temperature difference in the loop exceeds the trip setpoint until the rods are free to fall and was assumed within the non-LOCA analyses. Its use is intended to allow the potential future removal of the RTD bypass manifold without the need to perform additional safety analyses.

7.14 RC Pump Underfrequency Trip Response Time

A decrease in the RC Pump Underfrequency trip response time for 0.9 to 0.6 seconds is requested. This change reflects the new analysis value for the Complete Loss of Forced Reactor Coolant Flow accident in the Transition Safety Evaluation.

7.15 ECCS Accumulators

A change to the contained borated water volume of the ECCS accumulators is requested. The proposed change would require maintenance of accumulator water volume between 7489 and 7685 gallons during Modes 1, 2, and 3 (pressurized pressure above 1000 psig). The net benefit is a total gain of 2 inches of usable inventory which improves core reflooding during a large break LOCA. This impact is included in the large break LOCA analyses provided in Attachment 4.

An evaluation of the potential effects of this increased accumulator water volume in areas of safety other than the Large Break LOCA has been performed. Areas considered include small break LOCA, non-LOCA analyses, LOCA forces, mass and energy releases, and the steam generator tube rupture event. In all cases the increased accumulator volume had either a beneficial or inconsequential impact on the analysis results.

7.16 Charging Pump Surveillance

Minimum SI flow for the VANTAGE 5 transition LOCA and non-LOCA analyses assumed that the charging pump mini-flow path (recirculation) remained open for the duration of the accident. Consequently, from a safety Analysis standpoint, isolation of mini-flow to maximize SI is an optional action (i.e., not required for accident mitigation) during the injection phase.

To maintain consistency with the new analysis assumptions, it is proposed that the charging pump flow balance test limits be changed to reflect the mini-flow path open.

TABLE 7.1
VIRGIL SUMMER
TECHNICAL SPECIFICATION CHANGES FOR CYCLE 5 RELOAD

PAGE	SECTION	DESCRIPTION OF CHANGE	JUSTIFICATION
2-1	2.1.1	Delete reference to two loop operation	Two loop operation is not currently licensed
2-2	Figure 2.1-1	Core limits are revised	Core limits are revised for Vantage 5 fuel due to ITDP, new peaking factors and reduced RCS flow
2-5 2-8 2-9 2-10	Table 2.2-1	Setpoints and Thermal Design Flow (TDF) are changed	Setpoints and TDF are consistent with the new safety limits, instrument uncertainty and reduced flow
B 2-1	Bases 2.1.1	Discussion of Thermal and Hydraulic analysis	The analysis is based on ITDP methodology and the WRB-1 and -2 correlations
B 2-4	Bases 2.2.1	DNBR limit replaced by "safety analysis DNBR limit"	Future changes in analyses will not require a change in Bases
3/4 1-3a	Figure 3.1-3	Changed shutdown margin for Modes 3, 4 and 5	This is based on reanalysis of boron dilution with Vantage 5 fuel
3/4 1-4	3.1.1.3	Changed the MTC limits	This is based on Vantage 5 fuel
3/4 1-5	4.1.1.3	Changed the MTC Limits	This is based on Vantage 5 fuel
3/4 1-12	3.1.2.6.a	Changed minimum borated water volume for boric acid system	This is based on Vantage 5 fuel and extended fuel cycles
3/4 1-19	3.1.3.4	Changed rod drop time	This is based on Vantage 5 fuel
3/4 2-1, 3/4 2-2, 3/4 2-3	3.2.1	Changed AFD requirements	This is based on Vantage 5 fuel, RAOC and base load operation

TABLE 7.1
VIRGIL SUMMER
TECHNICAL SPECIFICATION CHANGES FOR CYCLE 5 RELOAD

PAGE	SECTION	DESCRIPTION OF CHANGE	JUSTIFICATION
3/4 2-4	3.2.2	Changed F(Q)	This is based on the optimized selection of parameters
3/4 2-5, 3/4 2-6	4.2.2.1	Changed F(Q) Surveillance Requirements	This is based on Vantage 5 fuel, RAOC and base load operation
3/4 2-7	Figure 3.2-2	Changed K(z) curve and figure number to 3.2-1	The K(z) curve reflects the limits in accordance with LOCA analysis.
3/4 2-8 3/4 2-9	3.2.3	Changed R to reflect F-Delta-H of 1.62 and changed figure number 3.2-3	This is based on the optimized selection of parameters
3/4 2-10	Figure 3.2-3	Changed total RCS flow rate requirement and figure number to 3.2-2	This is based on the analyzed RCS flow limit
3/4 2-16	Table 3.2-1	Changed DNB parameters	Changes reflect new analysis values
3/4 3-9	Table 3.3-2	Changed OTDT, OPDT response times	This is based on the optimized selection of parameters
3/4 3-10	Table 3.3-2	Changed underfrequency response time	The change reflects new analysis value
3/4 5-1	3.5.1	Changed accumulator water volume	This is based on the revised LOCA and non-LOCA analysis

TABLE 7.1
VIRGIL SUMMER
TECHNICAL SPECIFICATION CHANGES FOR CYCLE 5 RELOAD

PAGE	SECTION	DESCRIPTION OF CHANGE	JUSTIFICATION
3/4 5-6	4.5.2	Changed charging pump flow balance limit to reflect testing with recirculation	This is based on the revised safety analyses, where the charging pump recirculation was not isolated during the accidents
3/4 10-2	4.10.2.2	Changed surveillance section numbers to reflect use of FQ as opposed to Fxy surveillance	Reflects correct specification numbers for FQ surveillance
B 3/4 1-2	3/4 1.1.3	Changed MTC limits	This is based on Vantage 5 fuel
B 3/4 1-3	3/4.1.2	Changed boration volumes	This removes the cycle specific values.
B 3/4 2-1 B 3/4 2-2 B 3/4 2-3 B 3/4 2-4 B 3/4 2-5	3/4.2, 3/4.2.1, 3/4.2.2 and 3/4.2.3	Changed F(Q), DNB limit, deleted Fxy, and revised discussion of AFD	This is based on optimized selection of parameters, Vantage 5 fuel, ITDP, RAOC and baseload operation
B 3/4 5-1	3/4.5.1	Added comment for borated water and secondary pipe ruptures	The accumulators are actuated in the steam line break analysis
6-18	6.9.1.11	Change discussion of peaking factor report	This is based on Vantage 5 fuel, RAOC and base load operation. The approved methodology is referenced

8.0 REFERENCES

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12. Camden, T. M., et al., "PALADON-Westinghouse Nodal Computer Code," WCAP-9485-P-A, December 1979 and Supplement 1, September 1981.

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ATTACHMENT 2

TECHNICAL SPECIFICATIONS CHANGE PAGES

FOR THE

V. C. SUMMER PLANT

TRANSITION TO 17x17 VANTAGE 5 FUEL

2.0 SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

2.1 SAFETY LIMITS

REACTOR CORE

2.1.1 The combination of THERMAL POWER, pressurizer pressure, and the highest operating loop coolant temperature (T_{avg}) shall not exceed the limits shown in Figures 2.1-1 ~~and 2.1-2~~ for 3 ~~and 2~~ loop operation. ~~respectively.~~

APPLICABILITY: MODES 1 and 2.

ACTION:

Whenever the point defined by the combination of the highest operating loop average temperature and THERMAL POWER has exceeded the appropriate pressurizer pressure line, be in HOT STANDBY within 1 hour, and comply with the requirements of Specification 6.7.1.

REACTOR COOLANT SYSTEM PRESSURE

2.1.2 The Reactor Coolant System pressure shall not exceed 2735 psig.

APPLICABILITY: MODES 1, 2, 3, 4 and 5.

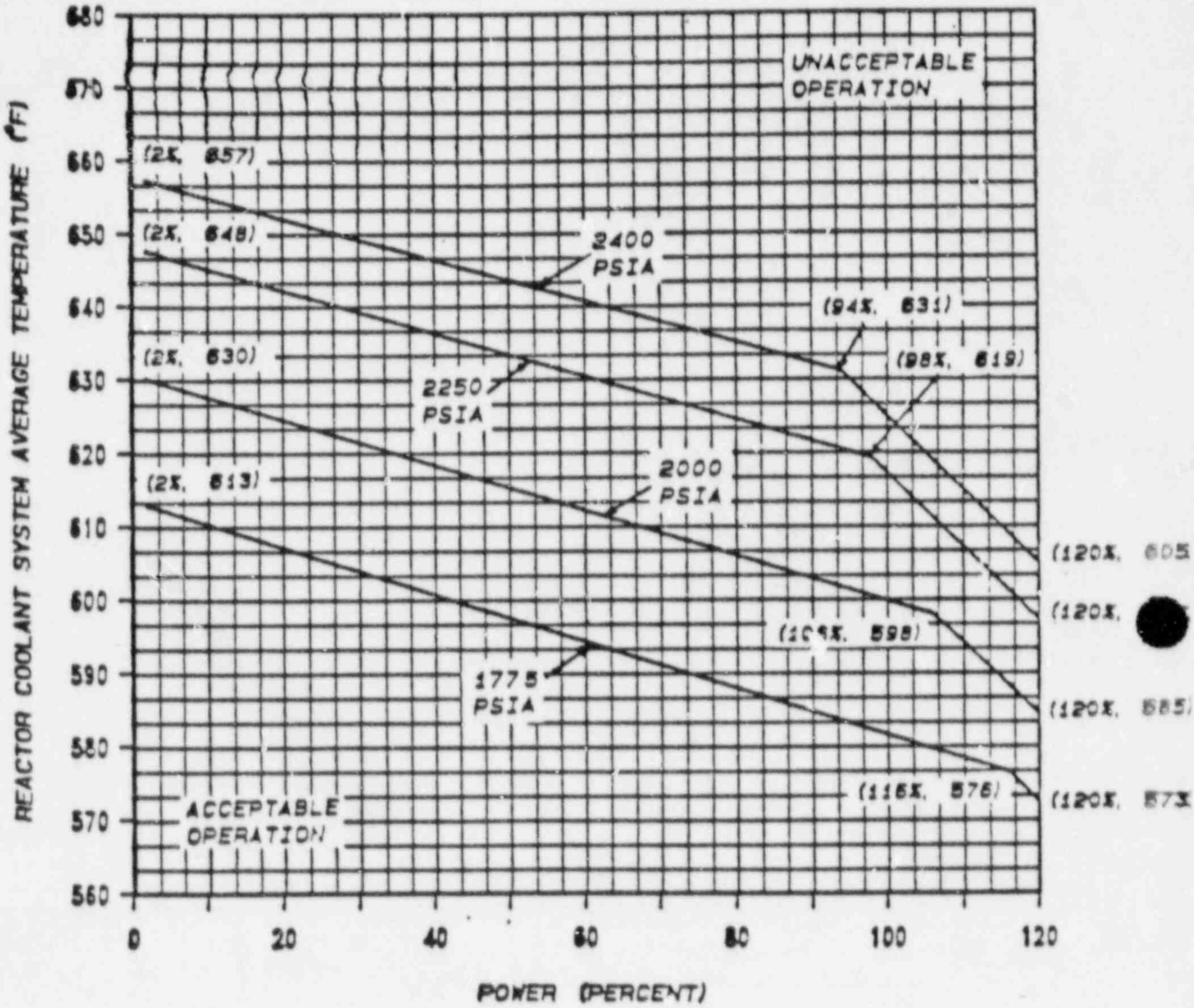
ACTION:

MODES 1 and 2

Whenever the Reactor Coolant System pressure has exceeded 2735 psig, be in HOT STANDBY with the Reactor Coolant System pressure within its limit within 1 hour, and comply with the requirements of Specification 6.7.1.

MODES 3, 4 and 5

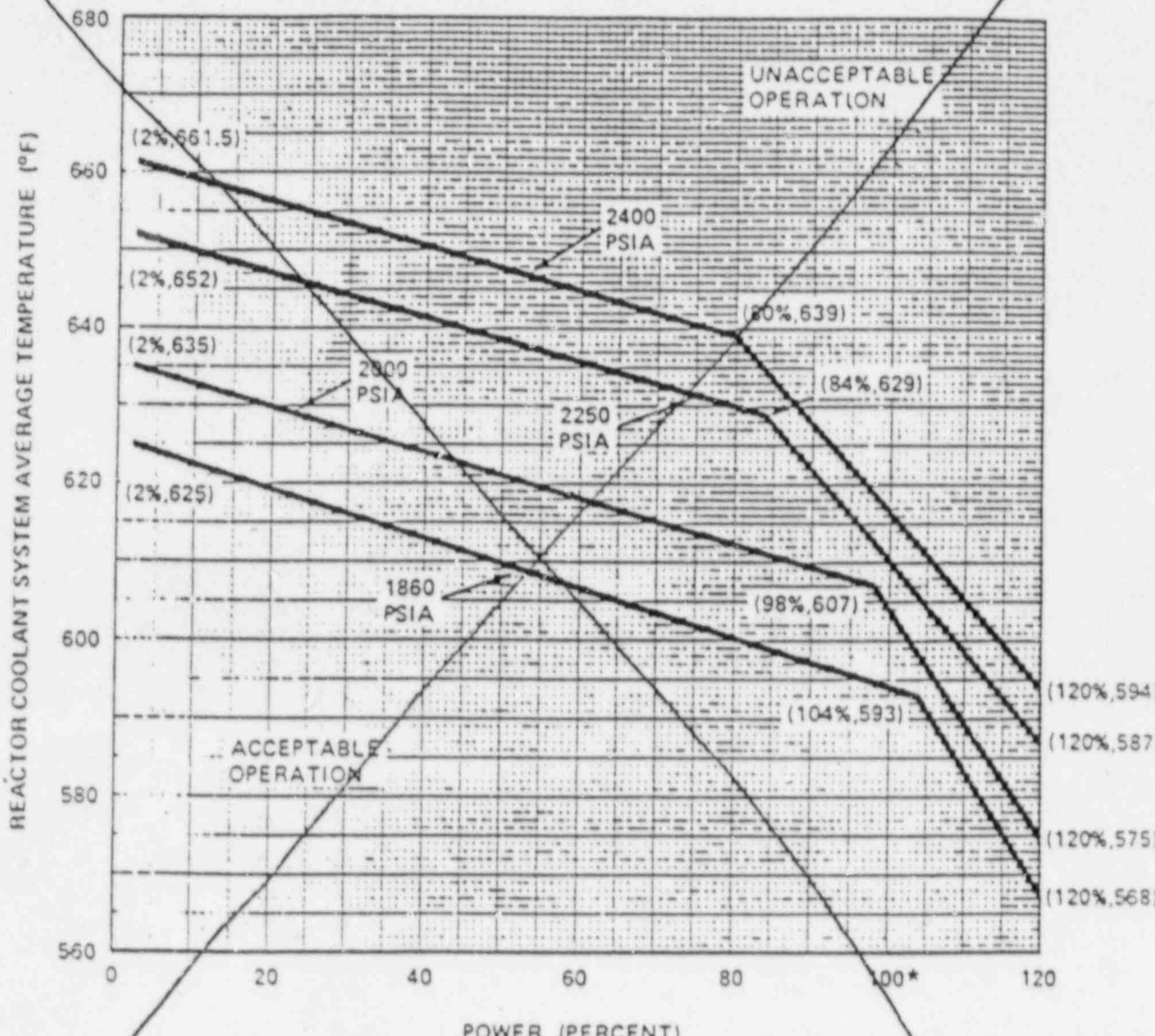
Whenever the Reactor Coolant System pressure has exceeded 2735 psig, reduce the Reactor Coolant System pressure to within its limit within 5 minutes, and comply with the requirements of Specification 6.7.1.



*When operating in the reduced RTP region of Technical Specification 3.2.3 (Figure 3.2-3), the restricted power level must be considered 100% RTP for this figure.

Figure 2.1-1 Reactor Core Safety Limit - Three Loops in Operation

REPLACE WITH NEW
FIGURE 2.2-1
ON OPPOSITE PAGE



*When operating in the reduced RTP region of Technical Specification 3.2.3 (Figure 3.2-3), the restricted power level must be considered 100% RTP for this figure.

Figure 2.1-1 Reactor Core Safety Limit - Three Loops in Operation

TABLE 2.2-1

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

Functional Unit	Total Allowance (TA)	Z	S	Trip Setpoint	Allowable Value
1. Manual Reactor Trip	Not Applicable	NA	NA	NA	NA
2. Power Range, Neutron Flux High Setpoint	7.5	4.56	0	$\leq$109% of RTP	$\leq$111.2% of RTP
Low Setpoint	8.3	4.56	0	$\leq$25% of RTP	$\leq$27.2% of RTP
3. Power Range, Neutron Flux High Positive Rate	1.6	0.5	0	$\leq$5% of RTP with a time constant $\geq$2 seconds	$\leq$6.3% of RTP with a time constant $\geq$2 seconds
4. Power Range, Neutron Flux High Negative Rate	1.6	0.5	0	$\leq$5% of RTP with a time constant $\geq$2 seconds	$\leq$6.3% of RTP with a time constant $\geq$2 seconds
5. Intermediate Range, Neutron Flux	17.0	8.4	0	$\leq$25% of RTP	$\leq$31% of RTP
6. Source Range, Neutron Flux	17.0	10.0	0	$\leq 10^5$ cps	$\leq 1.4 \times 10^5$ cps
7. Overtemperature ΔT	9.8 ~~7.1~~	7.29 ~~2.94~~	1.9 & 1.2** ~~1.8~~	See note 1	See note 2
8. Overpower ΔT	5.2 ~~4.5~~	2.26 ~~1.4~~	1.9 ~~1.2~~	See note 3	See note 4
9. Pressurizer Pressure-Low	3.1	0.71	1.5	$\geq$1870 psig	$\geq$1859 psig
10. Pressurizer Pressure-High	3.1	0.71	1.5	$\leq$2380 psig	$\leq$2391 psig
11. Pressurizer Water Level-High	5.0	2.18	1.5	$\leq$92% of instrument span	$\leq$93.8% of instrument span
12. Loss of Flow	2.5	1.0	1.5	$\geq$90% of ~~loop design flow*~~ MINIMUM MEASURED FLOW*	$\geq$89.2% of ~~loop design flow*~~ MINIMUM MEASURED FLOW*

~~*loop design flow = 96,200 gpm~~

RTP = RATED THERMAL POWER

* MINIMUM MEASURED FLOW = 94500 gpm/loop

** 1.9% SPAN FOR DELTA-T (RTDS) AND 1.2% FOR PRESSURIZER PRESSURE

TABLE 2.2-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

NOTATION

NOTE 1: OVERTEMPERATURE ΔT

$$\Delta T \leq \Delta T_o \{K_1 - K_2 \frac{(1 + \tau_1 S)}{(1 + \tau_2 S)} [T - T'] + K_3(P - P') - f_1(\Delta I)\}$$

Where: ΔT = Measured ΔT by RTD Manifold Instrumentation

ΔT_o $\leq$ ~~=~~ Indicated ΔT at RATED THERMAL POWER

K_1 $\leq$ ~~=~~ ~~1.090~~ 1.203

K_2 $\geq$ ~~=~~ ~~0.01450~~ 0.03006

$\frac{1 + \tau_1 S}{1 + \tau_2 S}$ = The function generated by the lead-lag controller for T_{avg} dynamic compensation

τ_1, & τ_2 = Time constants utilized in the lead-lag controller for T_{avg}, τ_1 $\geq$ ~~=~~ 28 secs., τ_2 $\leq$ ~~=~~ 4 secs.

T = Average temperature °F

T' $\leq$ ~~587.4~~ 585.5°F Reference T_{avg} at RATED THERMAL POWER

K_3 $\geq$ ~~=~~ ~~.0006728~~ 0.00147

P = Pressurizer pressure, psig

P' $\geq$ ~~=~~ 2235 psig, Nominal RCS operating pressure

S = Laplace transform operator, sec^{-1}.

TABLE 2.2-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

NOTATION (Continued)

NOTE 1: (Continued)

and $f_1(\Delta I)$ is a function of the indicated difference between top and bottom detectors of the power-range nuclear ion chambers; with gains to be selected based on measured instrument response during plant startup tests such that:

(i) for $q_t - q_b$ between -~~34~~ -24 percent and + ~~8~~ 4 percent $f_1(\Delta I) = 0$ where q_t and q_b are percent RATED THERMAL POWER in the top and bottom halves of the core respectively, and $q_t + q_b$ is total THERMAL POWER in percent of RATED THERMAL POWER.

(ii) for each percent that the magnitude of $q_t - q_b$ exceeds -~~34~~ -24 percent, the ΔT trip setpoint shall be automatically reduced by ~~1.62~~ 2.27 percent of its value at RATED THERMAL POWER.

(iii) for each percent that the magnitude of $q_t - q_b$ exceeds +~~8~~ 4 percent, the ΔT trip setpoint shall be automatically reduced by ~~1.11~~ 2.14 percent of its value at RATED THERMAL POWER.

NOTE 2: The channel's maximum trip setpoint shall not exceed its computed trip point by more than 2.0 ~~3.4~~ percent ΔT span.

NOTE 3: OVERPOWER ΔT

$$\Delta T \leq \Delta T_o \{K_4 - K_5 \left(\frac{\tau_3 S}{1 + \tau_3 S}\right) T - K_6 [T - T''] \}$$

Where:

ΔT = as defined in Note 1

ΔT_o = as defined in Note 1

K_4 $\leq$ ~~=~~ ~~1.09~~ 1.0875

K_5 $\geq$ ~~=~~ 0.02/°F for increasing average temperature and 0 for decreasing average temperature

$\frac{\tau_3 S}{1 + \tau_3 S}$ = The function generated by the rate-lag controller for T_{avg} dynamic compensation

TABLE 2.2-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

NOTATION (Continued)

NOTE 3: (Continued)

τ_3	=	Time constant utilized in the rate-lag controller for T_{avg}, τ_3 ~~<~~ ≥ 10 secs.
K_6	~~=~~ ≥	0.~~001190~~ 0.00156/°F for $T > T''$ and $K_6 = 0$ for $T \leq T''$
T	=	as defined in Note 1
T''	≤	~~581.4~~ 585.5°F Reference T_{avg} at RATED THERMAL POWER
S	=	as defined in Note 1

NOTE 4: The channel's maximum trip setpoint shall not exceed its computed trip point by more than ~~2.7~~ percent ΔT span.

2.0

2.1 SAFETY LIMITS

BASES

2.1.1 REACTOR CORE

REPLACE WITH INSERT 1

~~The restrictions of this Safety Limit prevent overheating of the fuel and possible cladding perforation which would result in the release of fission products to the reactor coolant. Overheating of the fuel cladding is prevented by restricting fuel operation to within the nucleate boiling regime where the heat transfer coefficient is large and the cladding surface temperature is slightly above the coolant saturation temperature.~~

~~Operation above the upper boundary of the nucleate boiling regime could result in excessive cladding temperatures because of the onset of departure from nucleate boiling (DNB) and the resultant sharp reduction in heat transfer coefficient. DNB is not a directly measurable parameter during operation and therefore THERMAL POWER and Reactor Coolant Temperature and Pressure have been related to DNB through the W-3 correlation. The W-3 DNB correlation has been developed to predict the DNB flux and the location of DNB for axially uniform and non-uniform heat flux distributions. The local DNB heat flux ratio, DNBR, defined as the ratio of the heat flux that would cause DNB at a particular core location to the local heat flux, is indicative of the margin to DNB.~~

~~The minimum value of the DNBR during steady state operation, normal operational transients, and anticipated transients is limited to 1.30. This value corresponds to a 95 percent probability at a 95 percent confidence level that DNB will not occur and is chosen as an appropriate margin to DNB for all operating conditions.~~

~~The curves of Figures 2.1-1 and 2.1-2 show the loci of points of THERMAL POWER, Reactor Coolant System pressure and average temperature for which the minimum DNBR is no less than 1.30, or the average enthalpy at the vessel exit is equal to the enthalpy of saturated liquid.~~

These curves are based on an enthalpy hot channel factor, $F^N_{\Delta H}$, of ~~1.55~~ 1.56 (includes measurement uncertainty) and a reference cosine with a peak of 1.55 for axial power shape. An allowance is included for an increase in $F^N_{\Delta H}$ at reduced power based on the expression:

$$F^N_{\Delta H} = \cancel{1.55}\ 1.56\ [1+ \cancel{0.2}\ 0.3\ (1-P)]$$

where P is the fraction of RATED THERMAL POWER

These limiting heat flux conditions are higher than those calculated for the range of all control rods fully withdrawn to the maximum allowable control rod insertion assuming the axial power imbalance is within the limits of the f_1 (delta I) function of the Overtemperature trip. When the axial power imbalance is not within the tolerance, the axial power imbalance effect on the Overtemperature delta T trips will reduce the setpoints to provide protection consistent with core safety limits.

INSERT 1

The restrictions of this safety limit prevent overheating of the fuel and possible cladding perforation which would result in the release of fission products to the reactor coolant. Overheating of the fuel cladding is prevented by restricting fuel operation to within the nucleate boiling regime where the heat transfer coefficient is large and the cladding surface temperature is slightly above the coolant saturation temperature.

Operation above the upper boundary of the nucleate boiling regime could result in excessive cladding temperatures because of the onset of departure from nucleate boiling (DNB) and the resultant sharp reduction in heat transfer coefficient. DNB is not a directly measurable parameter during operation and therefore THERMAL POWER and Reactor Coolant Temperature and Pressure have been related to DNB. This relation has been developed to predict the DNB flux and the location of DNB for axially uniform and non-uniform heat flux distributions. The local DNB heat flux ratio (DNBR) defined as the ratio of the heat flux that would cause DNB at a particular core location to the local heat flux, is indicative of the margin to DNB.

The DNB design basis is as follows: there must be at least a 95 percent probability that the minimum DNBR of the limiting rod during Condition I and II events is greater than or equal to the DNBR limit of the DNB correlation being used (the WRB-1 or WRB-2 correlation in this application). The correlation DNBR limit is established based on the entire applicable experimental data set such that there is a 95 percent probability with 95 percent confidence that DNB will not occur when the minimum DNBR is at the DNBR limit (1.17 for the WRB-1 or WRB-2 Correlation).

In meeting this design basis, uncertainties in plant operating parameters, nuclear and thermal parameters, and fuel fabrication parameters are considered statistically such that there is at least a 95% probability with 95% confidence level that the minimum DNBR for the limiting rod is greater than or equal to the DNBR limit. The uncertainties in the above plant parameters are used to determine the plant DNBR uncertainty. This DNBR uncertainty, combined with the correlation DNBR limit, establishes a design DNBR value which must be met in plant safety analyses using values of input parameters without uncertainties. In addition, margin has been maintained in the design by meeting safety analysis DNBR limits in performing safety analyses.

The curves of Figure 2.1-1 show the loci of points of THERMAL POWER, Reactor Coolant System pressure and average temperature below which the calculated DNBR is no less than the design DNBR value or the average enthalpy at the vessel exit is less than the enthalpy of saturated liquid.

REACTOR TRIP SYSTEM INSTRUMENTATION SETPOINTS (Continued)

The various reactor trip circuits automatically open the reactor trip breakers whenever a condition monitored by the Reactor Protection System reaches a preset or calculated level. In addition to redundant channels and trains, the design approach provides a Reactor Protection System which monitors numerous system variables, therefore, providing protection system functional diversity. The Reactor Protection System initiates a turbine trip signal whenever reactor trip is initiated. This prevents the reactivity insertion that would otherwise result from excessive reactor system cooldown and thus avoids unnecessary actuation of the Engineered Safety Features Actuation System.

Manual Reactor Trip

The Reactor Protection System includes manual reactor trip capability.

Power Range, Neutron Flux

In each of the Power Range Neutron Flux channels there are two independent bistables, each with its own trip setting used for a high and low range trip setting. The low setpoint trip provides protection during subcritical and low power operations to mitigate the consequences of a power excursion beginning from low power, and the high setpoint trip provides protection during power operations to mitigate the consequences of a reactivity excursion from all power levels.

The low setpoint trip may be manually blocked above P-10 (a power level of approximately 10 percent of RATED THERMAL POWER) and is automatically reinstated below the P-10 setpoint.

Power Range, Neutron Flux, High Rates

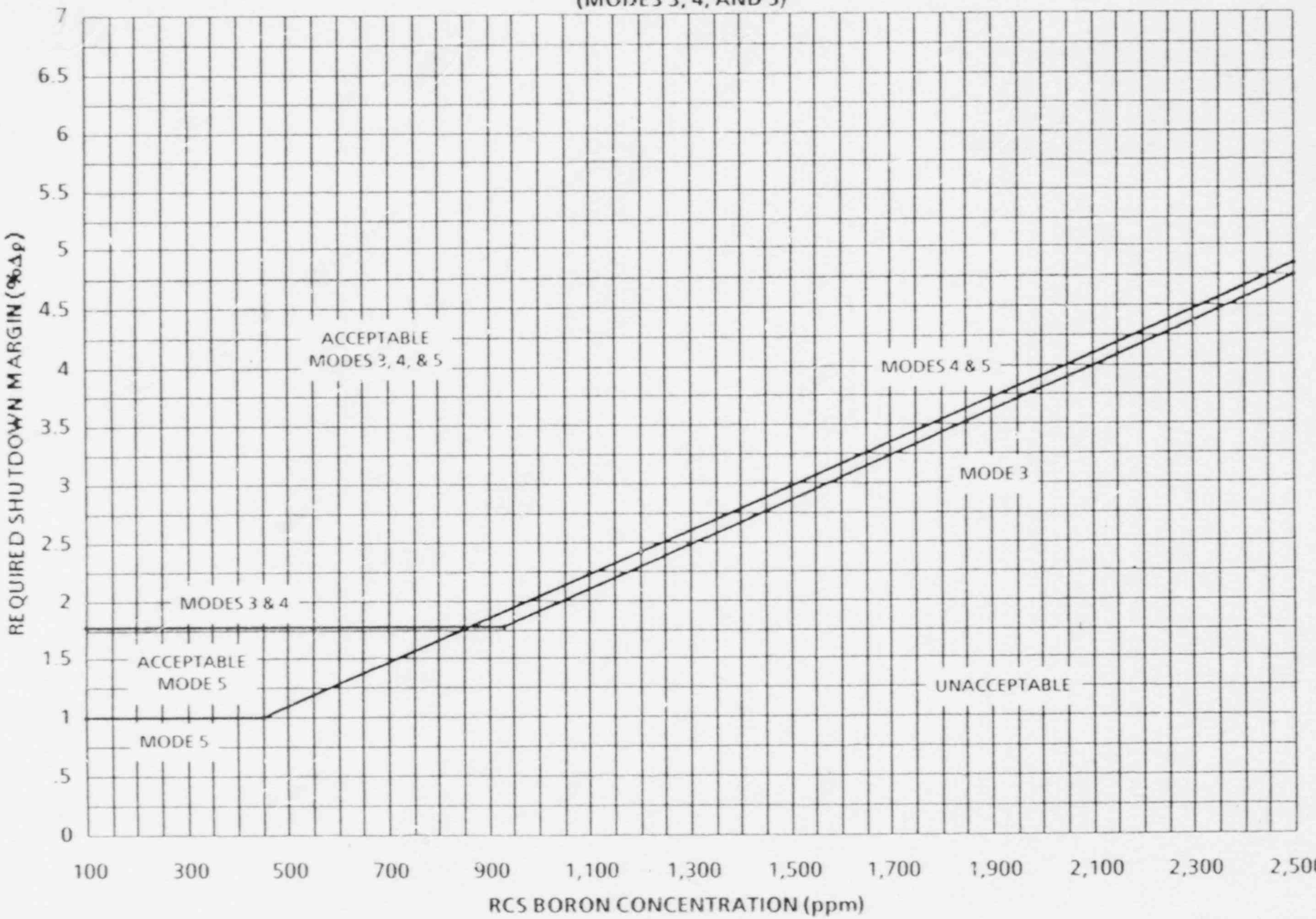
The Power Range Positive Rate trip provides protection against rapid flux increases which are characteristic of a rupture of a control rod drive housing. Specifically, this trip complements the Power Range Neutron Flux High and Low trips to ensure that the criteria are met for rod ejection from mid-power.

The Power Range Negative Rate trip provides protection for control rod drop accidents. At high power, a rod drop accident of a single or multiple rods could cause local flux peaking which could cause an unconservative local DNBR to exist. The Power Range Negative Rate trip will prevent this from occurring by tripping the reactor. No credit is taken for operation of the Power Range Negative Rate trip for those control rod drop accidents for which DNBR's will be greater than ~~1.30~~ the limit value.

Intermediate and Source Range, Nuclear Flux

The Intermediate and Source Range, Nuclear Flux trips provide reactor core protection during reactor startup to mitigate the consequences of an

FIGURE 3.1-3
REQUIRED SHUTDOWN MARGIN
(MODES 3, 4, AND 5)



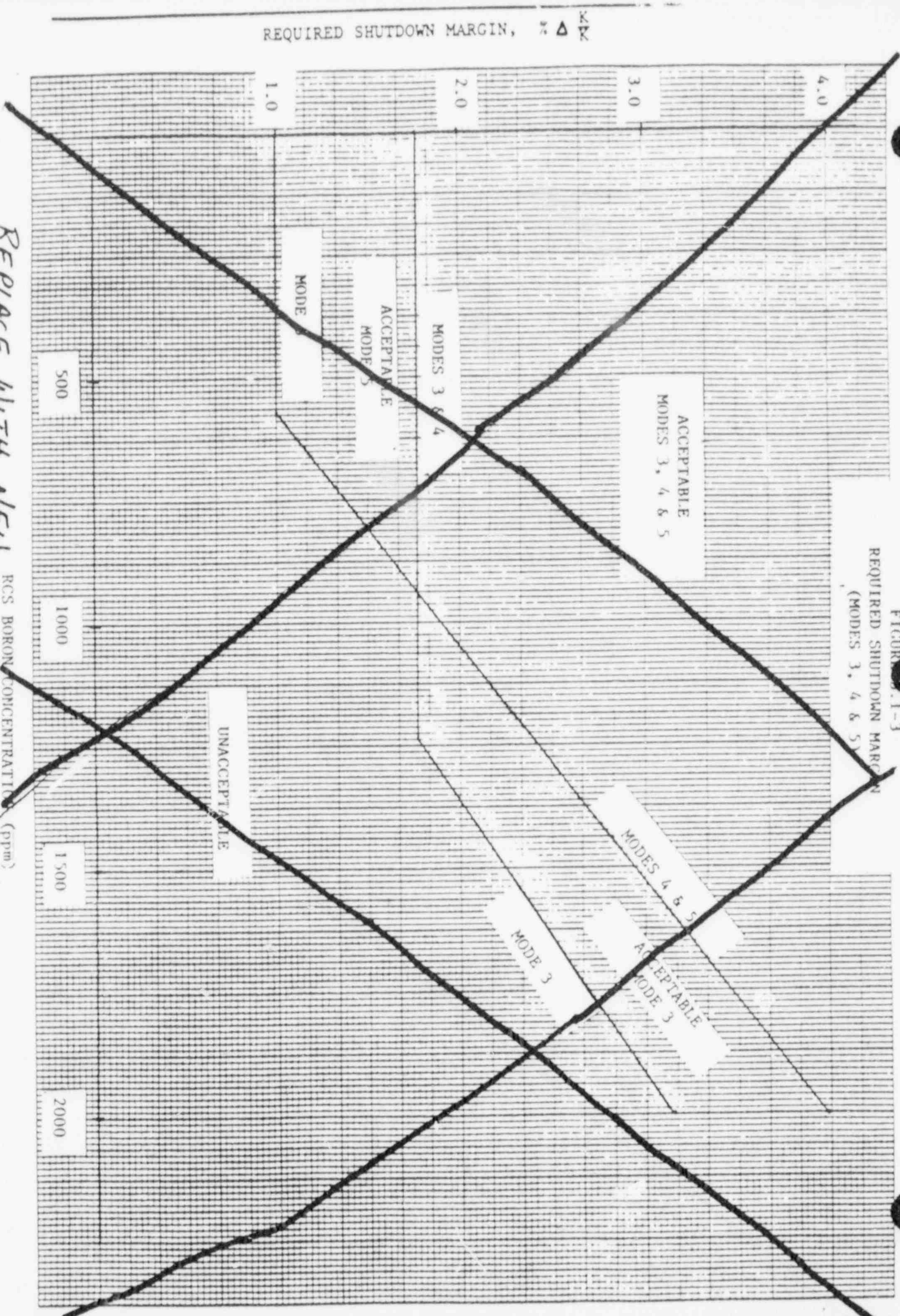


FIGURE 3.1-3
REQUIRED SHUTDOWN MARGIN
(MODES 3, 4 & 5)

REPLACE WITH NEW FIGURE 3.1-3 ON OPPOSITE PAGE.

MODERATOR TEMPERATURE COEFFICIENT

LIMITING CONDITION FOR OPERATION

3.1.1.3 The moderator temperature coefficient (MTC) shall be:

a. Less positive than the limits shown in Figure 3.1-0, and ~~0 delta k/k/°F for the all rods withdrawn, beginning of cycle life (BOL), hot zero THERMAL POWER condition.~~

b. Less negative than ~~-4.2~~ -5.0 x 10^{-4} delta k/k/°F for the all rods withdrawn, end of cycle life (EOL), RATED THERMAL POWER condition.

APPLICABILITY: Specification 3.1.1.3.a - MODES 1 and 2* only#
Specification 3.1.1.3.b - MODES 1, 2 and 3 only#

ACTION:

a. With the MTC more positive than the limit of 3.1.1.3.a above operation in MODES 1 and 2 may proceed provided:

1. Control rod withdrawal limits are established and maintained sufficient to restore the MTC to less positive than ~~0 delta k/k/°F~~ the limits shown in Figure 3.1-0 within 24 hours or be in HOT STANDBY within the next 6 hours. These withdrawal limits shall be in addition to the insertion limits of Specification 3.1.3.6.

2. The control rods are maintained within the withdrawal limits established above until a subsequent calculation verifies that the MTC has been restored to within its limit for the all rods withdrawn condition.

3. In lieu of any other report required by Specification 6.9.1, a Special Report is prepared and submitted to the Commission pursuant to Specification 6.9.2 within 10 days, describing the value of the measured MTC, the interim control rod withdrawal limits and the predicted average core burnup necessary for restoring the positive MTC to within its limit for the all rods withdrawn condition.

b. With the MTC more negative than the limit of 3.1.1.3.b above, be in HOT SHUTDOWN within 12 hours.

*With K_{eff} greater than or equal to 1.0

#See Special Test Exception 3.10.3

SURVEILLANCE REQUIREMENTS

4.1.1.3 The MTC shall be determined to be within its limits during each fuel cycle as follows:

a. The MTC shall be measured and compared to the BOL limit of Specification 3.1.1.3.a, above, prior to initial operation above 5% of RATED THERMAL POWER, after each fuel loading.

b. The MTC shall be measured at any THERMAL POWER and compared to -4.1 ~~-3.3~~ x 10^{-4} delta k/k/°F (all rods withdrawn, RATED THERMAL POWER condition) within 7 EFPD after reaching an equilibrium boron concentration of 300 ppm. In the event this comparison indicates the MTC is more negative than ~~-3.3~~ -4.1 x 10^{-4} delta k/k/°F, the MTC shall be remeasured, and compared to the EOL MTC limit of specification 3.1.1.3.b, at least once per 14 EFPD during the remainder of the fuel cycle.

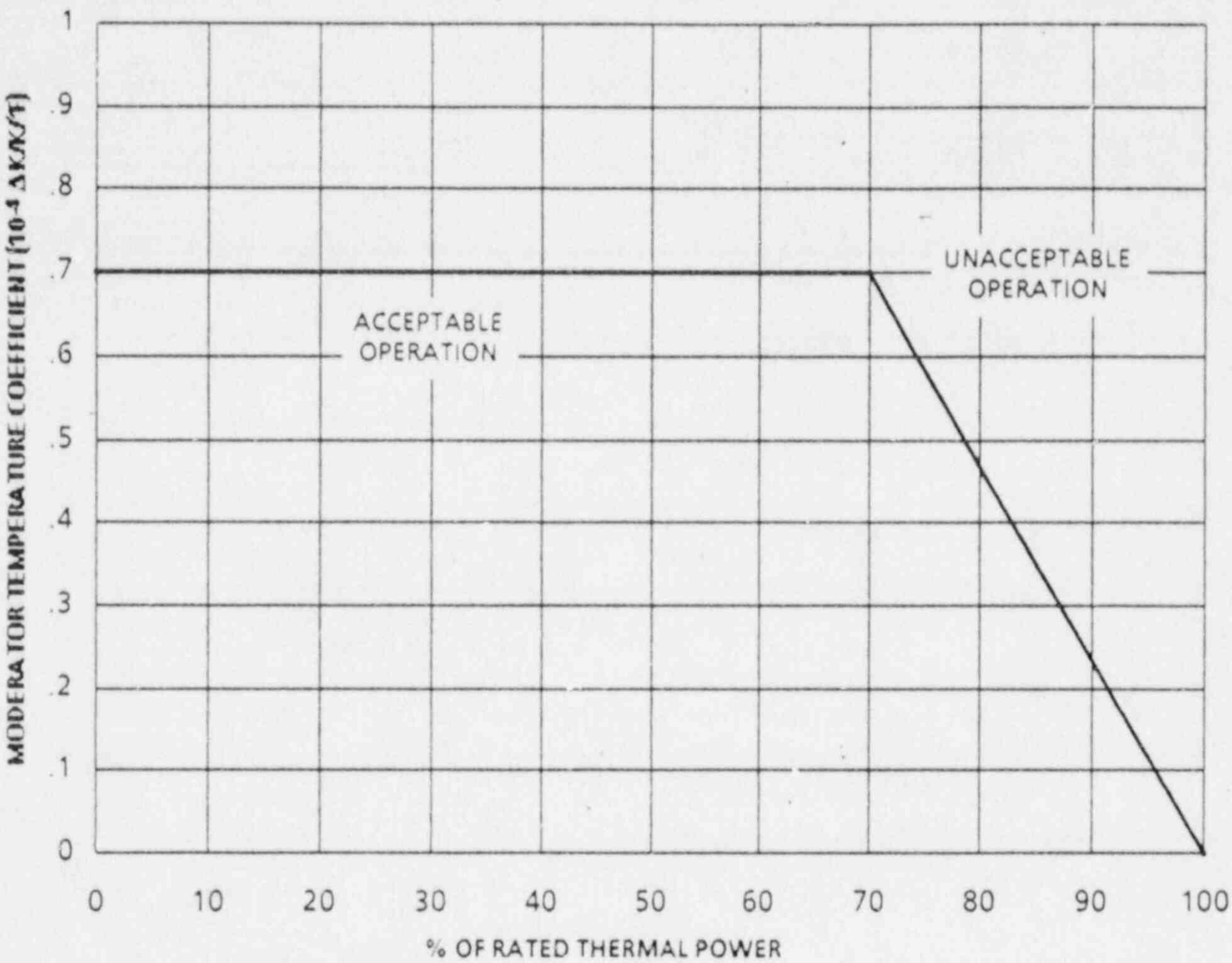


FIGURE 3.1-0
MODERATOR TEMPERATURE COEFFICIENT VS POWER LEVEL

REACTIVITY CONTROL SYSTEMS

BORATED WATER SOURCES - OPERATING

LIMITING CONDITION FOR OPERATION

3.1.2.6 As a minimum, the following borated water source(s) shall be OPERABLE as required by Specification 3.1.2.2:

a. A boric acid storage system with:

13,300

1. A minimum contained borated water volume of ~~13,200~~ gallons,

2. Between 7000 and 7700 ppm of boron, and

3. A minimum solution temperature of 65°F.

b. The refueling water storage tank with:

1. A minimum contained borated water volume of 453,800 gallons,

2. A minimum boron concentration of 2300 ppm, and

3. A minimum solution temperature of 40°F.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

a. With the boric acid storage system inoperable and being used as one of the above required borated water sources, restore the storage system to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and borated to a SHUTDOWN MARGIN equivalent to at least 2 percent delta k/k at 200°F; restore the boric acid storage system to OPERABLE status within the next 7 days or be in COLD SHUTDOWN within the next 30 hours.

b. With the refueling water storage tank inoperable, restore the tank to OPERABLE status within one hour or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

ROD DROP TIME

LIMITING CONDITION FOR OPERATION

3.1.3.4 The individual full length (shutdown and control) rod drop time from the fully withdrawn position shall be less than or equal to ~~[illegible]~~ 2.7 seconds from beginning of decay of stationary gripper coil voltage to dashpot entry with:

a. T_{avg} greater than or equal to 551°F, and

b. All reactor coolant pumps operating.

APPLICABILITY: MODES 1 and 2.

ACTION:

With the drop time of any full length rod determined to exceed the above limit, restore the rod drop time to within the above limit prior to proceeding to MODE 1 or 2.

SURVEILLANCE REQUIREMENTS

4.1.3.4 The rod drop time of full length rods shall be demonstrated through measurement prior to reactor criticality:

a. For all rods following each removal of the reactor vessel head,

b. For specifically affected individual rods following any maintenance on or modification to the control rod drive system which could affect the drop time of those specific rods, and

c. At least once per 18 months.

REPLACE TEXT ON PAGE 3/4 2-1 & 3/4 2-2 WITH INSERT 2.

3/4.2 POWER DISTRIBUTION LIMITS

3/4.2.1 AXIAL FLUX DIFFERENCE (AFD)

LIMITING CONDITION FOR OPERATION

3.2.1 The indicated AXIAL FLUX DIFFERENCE (AFD) shall be maintained within a ±5% target band (flux difference units) about the target flux difference.

APPLICABILITY: MODE 1 above 50% of RATED THERMAL POWER*

ACTION:

a. With the indicated AXIAL FLUX DIFFERENCE outside of the ±5% target band about the target flux difference and with THERMAL POWER.

   1. Above 90% of RATED THERMAL POWER, within 15 minutes either:

      a) Restore the indicated AFD to within the target band limits, or

      b) Reduce THERMAL POWER to less than 90% of RATED THERMAL POWER.

   2. Between 50% and 90% of RATED THERMAL POWER:

      a) POWER OPERATION may continue provided:

         1) The indicated AFD has not been outside of the ±5% target band for more than 1 hour penalty deviation cumulative during the previous 24 hours, and

         2) The indicated AFD is within the limits shown on Figure 3.2-1. Otherwise, reduce THERMAL POWER to less than 50% of RATED THERMAL POWER within 30 minutes and reduce the Power Range Neutron Flux-High Trip Setpoints to less than or equal to 55% of RATED THERMAL POWER within the next 4 hours.

      b) Surveillance testing of the Power Range Neutron Flux Channels may be performed pursuant to Specification 4.3.1.1 provided the indicated AFD is maintained within the limits of Figure 3.2-1. A total of 16 hours operation may be accumulated with the AFD outside of the target band during this testing without penalty deviation.

b. THERMAL POWER shall not be increased above 90% of RATED THERMAL POWER unless the indicated AFD is within the ±5% target band and ACTION a.2.a) 1), above has been satisfied.

*See Special Test Exception 3.10.2

REPLACE WITH INSERT 2

REPLACE WITH INSERT 2

ACTION (Continued)

c. THERMAL POWER shall not be increased above 50% of RATED THERMAL POWER unless the indicated AFD has not been outside of the ±5% target band for more than 1 hour penalty deviation cumulative during the previous 24 hours. Power increases above 50% of RATED THERMAL POWER do not require being within the target band provided the accumulative penalty deviation is not violated.

SURVEILLANCE REQUIREMENTS

4.2.1.1 The indicated AXIAL FLUX DIFFERENCE shall be determined to be within its limits during POWER OPERATION above 15% of RATED THERMAL POWER by:

a. Monitoring the indicated AFD for each OPERABLE excore channel:

1. At least once per 7 days when the AFD Monitor Alarm is OPERABLE, and

2. At least once per hour for the first 24 hours after restoring the AFD Monitor Alarm to OPERABLE status.

b. Monitoring and logging the indicated AXIAL FLUX DIFFERENCE for each OPERABLE excore channel at least once per hour for the first 24 hours and at least once per 30 minutes thereafter, when the AXIAL FLUX DIFFERENCE Monitor Alarm is inoperable. The logged values of the indicated AXIAL FLUX DIFFERENCE shall be assumed to exist during the interval preceding each logging.

4.2.1.2 The indicated AFD shall be considered outside of its ±5% target band when 2 or more OPERABLE excore channels are indicating the AFD to be outside the target band. Penalty deviation outside of the ±5% target band shall be accumulated on a time basis of:

a. One minute penalty deviation for each one minute of POWER OPERATION outside of the target band at THERMAL POWER levels equal to or above 50% of RATED THERMAL POWER, and

b. One-half minute penalty deviation for each one minute of POWER OPERATION outside of the target band at THERMAL POWER levels between 15% and 50% of RATED THERMAL POWER.

4.2.1.3 The target flux difference of each OPERABLE excore channel shall be determined by measurement at least once per 92 Effective Full Power Days. The provisions of Specification 4.0.4 are not applicable.

4.2.1.4 The target flux difference shall be updated at least once per 31 Effective Full Power Days by either determining the target flux difference pursuant to 4.2.1.3 above or by linear interpolation between the most recently measured value and 0 percent at the end of the cycle life. The provisions of Specification 4.0.4 are not applicable.

INSERT 2

3.2.1 The indicated AXIAL FLUX DIFFERENCE (AFD) shall be maintained within:

a. the allowed operational space defined in the Peaking Factor Limit Report (PFLR) for Relaxed Axial Offset Control (RAOC) operation,
   or

b. within the target band specified in the PFLR about the target flux difference during base load operation.

APPLICABILITY: MODE 1 above 50% of RATED THERMAL POWER*.

ACTION:

a. For RAOC operation with the indicated AFD outside of the applicable limits specified in the PFLR,

   1. Either restore the indicated AFD to within the PFLR specified limits within 15 minutes, or

   2. Reduce THERMAL POWER to less than 50% of RATED THERMAL POWER within 30 minutes and reduce the Power Range Neutron Flux - High Trip setpoints to less than or equal to 55% of RATED THERMAL POWER within the next 4 hours.

b. For Base Load operation above APL^{ND}** with the indicated AFD outside of the applicable target band about the target flux difference:

   1. Either restore the indicated AFD to within the PFLR specified target band within 15 minutes, or

   2. Reduce THERMAL POWER to less than APL^{ND} of RATED THERMAL POWER and discontinue Base Load operation within 30 minutes.

c. THERMAL POWER shall not be increased above 50% of RATED THERMAL POWER unless the indicated AFD is within the applicabl RAOC limits.

---

* See Special Text Exception 3.10.2.

**APL^{ND} is the minimum allowable power level for base load operation and will be provided in the Peaking Factor Limit Report per Specification 6.9.1.11.

POWER DISTRIBUTION LIMITS

SURVEILLANCE REQUIREMENTS

4.2.1.1 The indicated AFD shall be determined to be within its limits during POWER OPERATION above 50% of RATED THERMAL POWER by:

a. Monitoring the indicated AFD for each OPERABLE excore channel at least once per 7 days when the AFD Monitor Alarm is OPERABLE.

b. Monitoring and logging the indicated AFD for each OPERABLE excore channel at least once per hour for the first 24 hours and at least once per 30 minutes thereafter, when the AFD Monitor Alarm is inoperable. The logged values of the indicated AFD shall be assumed to exist during the interval preceding each logging.

4.2.1.2 The indicated AFD shall be considered outside of its limits when two or more OPERABLE excore channels are indicating the AFD to be outside the limits.

4.2.1.3 When in Base Load operation, the target axial flux difference of each OPERABLE excore channel shall be determined by measurement at least once per 92 Effective Full Power Days. The provisions of Specification 4.0.4 are not applicable.

4.2.1.4 When in Base Load operation, the target flux difference shall be updated at least once per 31 Effective Full Power Days by either determining the target flux difference in conjunction with the surveillance requirements of Specification 4.2.1.3 above or by linear interpolation between the most recently measured value and the calculated value at the end of cycle life. The provisions of Specification 4.0.4 are not applicable.

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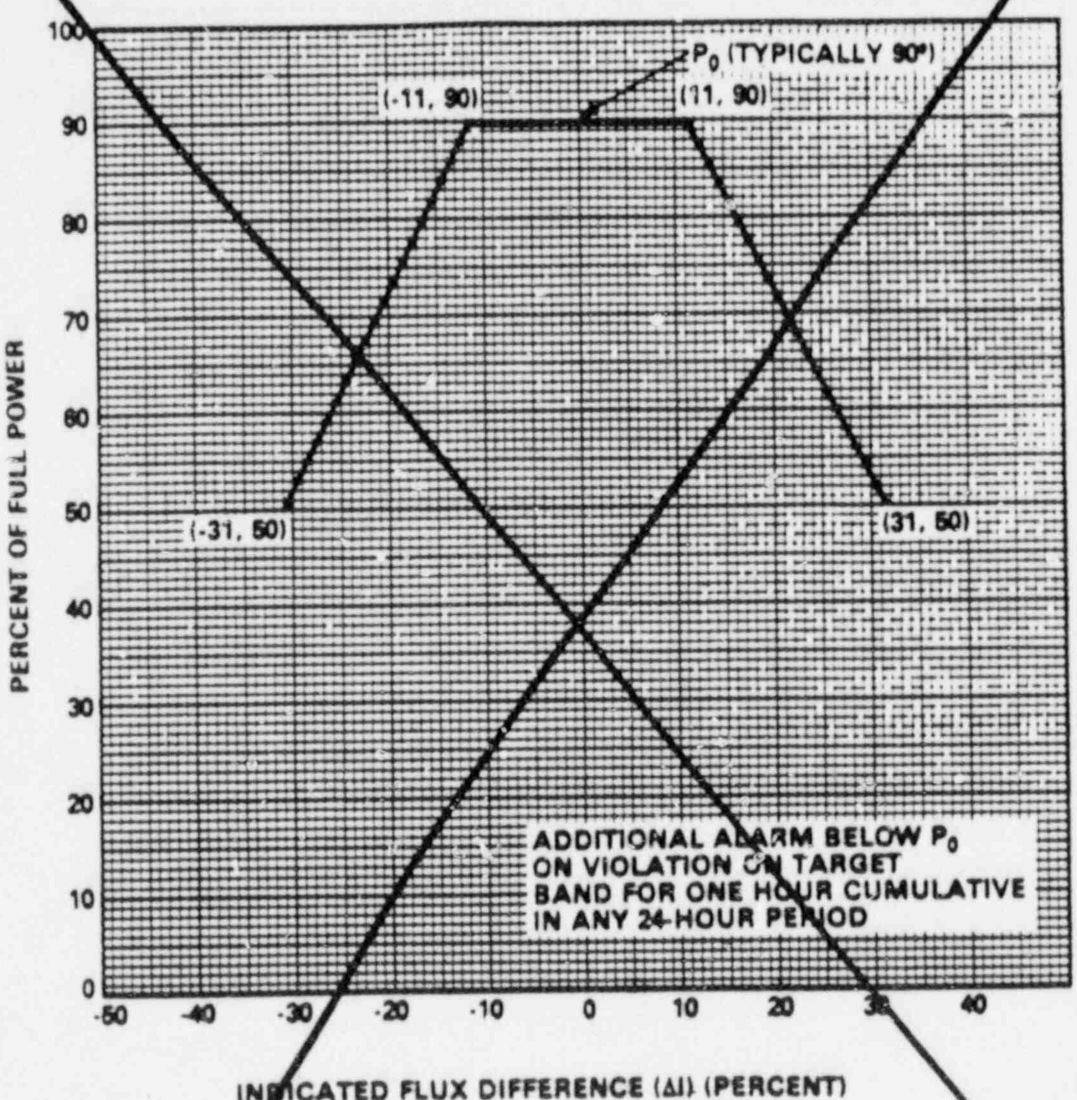

Figure 3.2-1 Axial Flux Difference Limits as a Function of Rated Thermal Power

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POWER DISTRIBUTION LIMITS

3/4.2.2 HEAT FLUX HOT CHANNEL FACTOR - $F_Q(Z)$

LIMITING CONDITION FOR OPERATION

3.2.2 $F_Q(Z)$ shall be limited by the following relationships:

$$F_Q(Z) \le \left[\frac{\cancel{2.25}\ 2.45}{P}\right] [K(Z)] \text{ for } P > 0.5$$

$$F_Q(Z) \le [\cancel{4.5}\ 4.9] [K(Z)] \text{ for } P \le 0.5$$

where $P = \frac{\text{THERMAL POWER}}{\text{RATED THERMAL POWER}}$

and K(Z) is the function obtained from Figure ~~3.2-2~~ 3.2-1 for a given core height location.

APPLICABILITY: MODE 1.

ACTION.

With $F_Q(Z)$ exceeding its limit:

a. Reduce THERMAL POWER at least 1% for each 1% $F_Q(Z)$ exceeds the limit within 15 minutes and similarly reduce the Power Range Neutron Flux-High Trip Setpoints within the next 4 hours; POWER OPERATION may proceed for up to a total of 72 hours; subsequent POWER OPERATION may proceed provided the Overpower delta T Trip Setpoints have been reduced at least 1% for each 1% $F_Q(Z)$ exceeds the limit.

b. Identify and correct the cause of the out of limit condition prior to increasing THERMAL POWER above the reduced limit required by a, above; THERMAL POWER may then be increased provided $F_Q(Z)$ is demonstrated through incore mapping to be within its limit.

Replace pp {3/4 2-5, 3/4 2-6} with following 5 pages

SURVEILLANCE REQUIREMENTS

4.2.2.1 The provisions of Specification 4.0.4 are not applicable.

4.2.2.2 F_{xy} shall be evaluated to determine if $F_Q(Z)$ is within its limit by:

a. Using the movable incore detectors to obtain a power distribution map at any THERMAL POWER greater than 5% of RATED THERMAL POWER.

b. Increasing the measured F_{xy} component of the power distribution map by 3% to account for manufacturing tolerances and further increasing the value by 5% to account for measurement uncertainties.

c. Comparing the F_{xy} computed (F_{xy}^C) obtained in b, above to:

1. The F_{xy} limits for RATED THERMAL POWER (F_{xy}^{RTP}) for the appropriate measured core planes given in e. and f. below, and

2. The relationship:

$$F_{xy}^L = F_{xy}^{RTP}\,[1+0.2(1-P)]$$

where F_{xy}^L is the limit for fractional THERMAL POWER operation expressed as a function of F_{xy}^{RTP} and P is the fraction of RATED THERMAL POWER at which F_{xy} was measured.

d. Remeasuring F_{xy} according to the following schedule:

1. When F_{xy}^C is greater than the F_{xy}^{RTP} limit for the appropriate measured core plane but less than the F_{xy}^L relationship, additional power distribution maps shall be taken and F_{xy}^C compared to F_{xy}^{RTP} and F_{xy}^L:

a) Either within 24 hours after exceeding by 20% of RATED THERMAL POWER or greater, the THERMAL POWER at which F_{xy}^C was last determined, or

b) At least once per 31 EFPD, whichever occurs first.

SURVEILLANCE REQUIREMENTS (Continued)

2. When the F_{xy}^{C} is less than or equal to the F_{xy}^{RTP} limit for the appropriate measured core plane, additional power distribution maps shall be taken and F_{xy}^{C} compared to F_{xy}^{RTP} and F_{xy}^{L} at least once per 31 EFPD.

e. The F_{xy} limits for RATED THERMAL POWER (F_{xy}^{RTP}) shall be provided for all core planes containing bank "D" control rods and all unrodded core planes in a Radial Peaking Factor Limit Report per Specification 6.9.1.11.

f. The F_{xy} limits of e., above, are not applicable in the following core planes regions as measured in percent of core height from the bottom of the fuel:

1. Lower core region from 0 to 15%, inclusive.
2. Upper core region from 85 to 100%, inclusive.
3. Grid plane regions at 17.8 ± 2%, 32.1 ± 2%, 46.4 ± 2%, 60.6 ± 2% and 74.9 ± 2%, inclusive. (17 x 17 fuel elements).
4. Core plane regions within ± 2% of core height (± 2.88 inches) about the bank demand position of the bank "D" control rods.

g. With F_{xy}^{C} exceeding F_{xy}^{L} the effects of F_{xy} on $F_{Q}(Z)$ shall be evaluated to determine if $F_{Q}(Z)$ is within its limits.

4.2.2.3 When $F_{Q}(Z)$ is measured for other than F_{xy} determinations, an overall measured $F_{Q}(Z)$ shall be obtained from a power distribution map and increased by 3% to account for manufacturing tolerances and further increased by 5% to account for measurement uncertainty.

P. 1 of 5

POWER DISTRIBUTION LIMITS

SURVEILLANCE REQUIREMENTS

4.2.2.1 The provisions of Specification 4.0.4 are not applicable.

4.2.2.2 For RAOC operation, $F_Q(z)$ shall be evaluated to determine if $F_Q(z)$ is within its limit by:

a. Using the movable incore detectors to obtain a power distribution map at any THERMAL POWER greater than 5% of RATED THERMAL POWER.

b. Increasing the measured $F_Q(z)$ component of the power distribution map by 3% to account for manufacturing tolerances and further increasing the value by 5% to account for measurement uncertainties. Verify the requirements of Specification 3.2.2 are satisfied.

c. Satisfying the following relationship:

$$F_Q^M(z) \leq \frac{2.45 \times K(z)}{P \times W(z)} \text{ for } P > 0.5$$

$$F_Q^M(z) \leq \frac{2.45 \times K(z)}{W(z) \times 0.5} \text{ for } P \leq 0.5$$

where $F_Q^M(z)$ is the measured $F_Q(z)$ increased by the allowances for manufacturing tolerances and measurement uncertainty, 2.45 is the F_Q limit, K(z) is given in Figure 3.2-1, P is the relative THERMAL POWER, and W(z) is the cycle dependent function that accounts for power distribution transients encountered during normal operation. This function is given in the Peaking Factor Limit Report as per Specification 6.9.1.11.

d. Measuring $F_Q^M(z)$ according to the following schedule:

1. Upon achieving equilibrium conditions after exceeding by 10% or more of RATED THERMAL POWER, the THERMAL POWER at which $F_Q(z)$ was last determined,* or

2. At least once per 31 Effective Full Power Days, whichever occurs first.

*During power escalation at the beginning of each cycle, power level may be increased until a power level for extended operation has been achieved and a power distribution map obtained.

P. 2 of 5

POWER DISTRIBUTION LIMITS

SURVEILLANCE REQUIREMENTS (Continued)

e. ~~With measurements indicating~~ With the maximum value of ~~maximum over z~~

$$\frac{F_Q^M(z)}{K(z)}$$

over the core height (z) increasing ~~has increased~~ since the previous determination of $F_Q^M(z)$ either of the following actions shall be taken:

1) $F_Q^M(z)$ shall be increased by 2% over that specified in Specification 4.2.2.2c. or

2) $F_Q^M(z)$ shall be measured at least once per 7 Effective Full Power Days until two successive maps indicate that the maximum value of ~~maximum over z~~ $\frac{F_Q^M(z)}{K(z)}$ over the core height (z) is not increasing.

f. With the relationships specified in Specification 4.2.2.2c. above not being satisfied:

1) Calculate the maximum percent over the core height (z) that $F_Q(z)$ exceeds its limit by the following expression:

$$\left\{\left[\frac{F_Q^M(z) \times W(z)}{\frac{\cancel{2.32}\ 2.45}{P} \times K(z)}\right] - 1\right\} \times 100 \quad \text{for } P \geq 0.5$$

$$\left\{\left[\frac{F_Q^M(z) \times W(z)}{\frac{\cancel{2.32}\ 2.45}{0.5} \times K(z)}\right] - 1\right\} \times 100 \quad \text{for } P < 0.5$$

2) One of the following actions shall be taken:

a) Within 15 minutes, control the AFD to within new AFD limits which are determined by reducing the acceptable AFD limits ~~of ± 5%~~ by 1% AFD for each percent $F_Q(z)$ exceeds its limits as determined in Specification 4.2.2.2f.1). Within 8 hours, reset the AFD alarm setpoints to these modified limits, or

b) Comply with the requirements of Specification 3.2.2 for $F_Q(z)$ exceeding its limit by the percent calculated above, or

c) Verify that the requirements of Specification 4.2.2.3 for Base Load operation are satisfied and enter Base Load operation.

P. 3 of 5

POWER DISTRIBUTION LIMITS

SURVEILLANCE REQUIREMENTS (Continued)

g. The limits specified in Specifications 4.2.2.2c, 4.2.2.2e., and 4.2.2.2f. above are not applicable in the following core plane regions:

1. Lower core region from 0 to 15%, inclusive.
2. Upper core region from 85 to 100%, inclusive.

4.2.2.3 Base Load operation is permitted at powers above APL^{ND} if the following conditions are satisfied:

a. Prior to entering Base Load operation, maintain THERMAL POWER above APL^{ND} and less than or equal to that allowed by Specification 4.2.2.2 for at least the previous 24 hours. Maintain Base Load operation surveillance (AFD within applicable target band about the ~~±5% of~~ target flux difference) during this time period. Base Load operation is then permitted providing THERMAL POWER is maintained between APL^{ND} and APL^{BL} or between APL^{ND} and 100% (whichever is most limiting) and FQ surveillance is maintained pursuant to Specification 4.2.2.4. APL^{BL} is defined as the minimum value of:

$$APL^{BL} = \cancel{\genfrac{}{}{0pt}{}{\text{minimum}}{\text{over } Z}} \left[\frac{2.45 \times K(Z)}{F_Q^M(Z) \times W(Z)_{BL}} \right] \times 100\%$$

over the core height (z) where: $F_Q^M(z)$ is the measured $F_Q(z)$ increased by the allowances for manufacturing tolerances and measurement uncertainty. The F_Q limit is ~~2.32~~ 2.45. K(z) is given in Figure ~~3.2-2~~ 3.2-1. $W(z)_{BL}$ is the cycle dependent function that accounts for limited power distribution transients encountered during base load operation. The function is given in the Peaking Factor Limit Report as per Specification ~~6.9.1.~~ 6.9.1.11.

b. During Base Load operation, if the THERMAL POWER is decreased below APL^{ND} then the conditions of 4.2.2.3.a shall be satisfied before re-entering Base Load operation.

4.2.2.4 During Base Load Operation $F_Q(Z)$ shall be evaluated to determine if $F_Q(Z)$ is within its limit by:

a. Using the movable incore detectors to obtain a power distribution map at any THERMAL POWER above APL^{ND}.

b. Increasing the measured $F_Q(Z)$ component of the power distribution map by 3% to account for manufacturing tolerances and further increasing the value by 5% to account for measurement uncertainties. Verify the requirements of Specification 3.2.2 are satisfied.

P. 4 of 5

POWER DISTRIBUTION LIMITS

SURVEILLANCE REQUIREMENTS (Continued)

c. Satisfying the following relationship:

$$F_Q^M(Z) \leq \frac{\cancel{2.32}\ 2.45 \times K(Z)}{P \times W(Z)_{BL}} \text{ for } P > APL^{ND}$$

where: $F_Q^M(Z)$ is the measured $F_Q(Z)$. The F_Q limit is ~~2.32~~ 2.45.

K(Z) is given in Figure ~~3.2-2~~ B.2-1. P is the relative THERMAL POWER. $W(Z)_{BL}$ is the cycle dependent function that accounts for limited power distribution transients encountered during normal operation. This function is given in the Peaking Factor Limit Report as per Specification 6.9.1.11.

d. Measuring $F_Q^M(Z)$ in conjunction with target flux difference determination according to the following schedule:

1. Prior to entering BASE LOAD operation after satisfying Section 4.2.2.3 unless a full core flux map has been taken in the previous 31 EFPD with the relative thermal power having been maintained above APL^{ND} for the 24 hours prior to mapping, and
2. At least once per 31 effective full power days.

e. ~~With measurements indicating~~ with the maximum value of

~~maximum over Z~~ $\left\{ \frac{F_Q^M(Z)}{K(Z)} \right\}$ over the core height (z) increasing

~~has increased~~ since the previous determination of $F_Q^M(Z)$ either of the following actions shall be taken:

1. $F_Q^M(Z)$ shall be increased by 2 percent over that specified in 4.2.2.4.c, or
2. $F_Q^M(Z)$ shall be measured at least once per 7 EFPD until 2 successive maps indicate that the maximum value of

~~maximum over Z~~ $\left\{ \frac{F_Q^M(Z)}{K(Z)} \right\}$ over the core height (z) is not increasing.

f. With the relationship specified in 4.2.2.4.c above not being satisfied, either of the following actions shall be taken:

1. Place the core in an equilibrium condition where the limit in 4.2.2.2.c is satisfied, and remeasure $F_Q^M(Z)$, or

P. 5 of 5

POWER DISTRIBUTION LIMITS

SURVEILLANCE REQUIREMENTS (Continued)

2. Comply with the requirements of Specification 3.2.2 for $F_Q(Z)$ exceeding its limit by the maximum percent calculated over the core height (z) with the following expression:

$$\left[\left\{ \cancel{\text{max. over z of}} \left[\frac{F_Q^M(Z) \times W(Z)_{BL}}{\cancel{\frac{\ldots}{P}}\ 2.45 \times K(Z)} \right] \right\} - 1 \right] \times 100 \text{ for } P \geq APL^{ND}$$

g. The limits specified in 4.2.2.4.c, 4.2.2.4.e, and 4.2.2.4.f above are not applicable in the following core plan regions:

1. Lower core region 0 to 15 percent, inclusive.
2. Upper core region 85 to 100 percent, inclusive.

4.2.2.5 When $F_Q(Z)$ is measured for reasons other than meeting the requirements of specification 4.2.2.2 an overall measured $F_Q(z)$ shall be obtained from a power distribution map and increased by 3% to account for manufacturing tolerances and further increased by 5% to account for measurement uncertainty.

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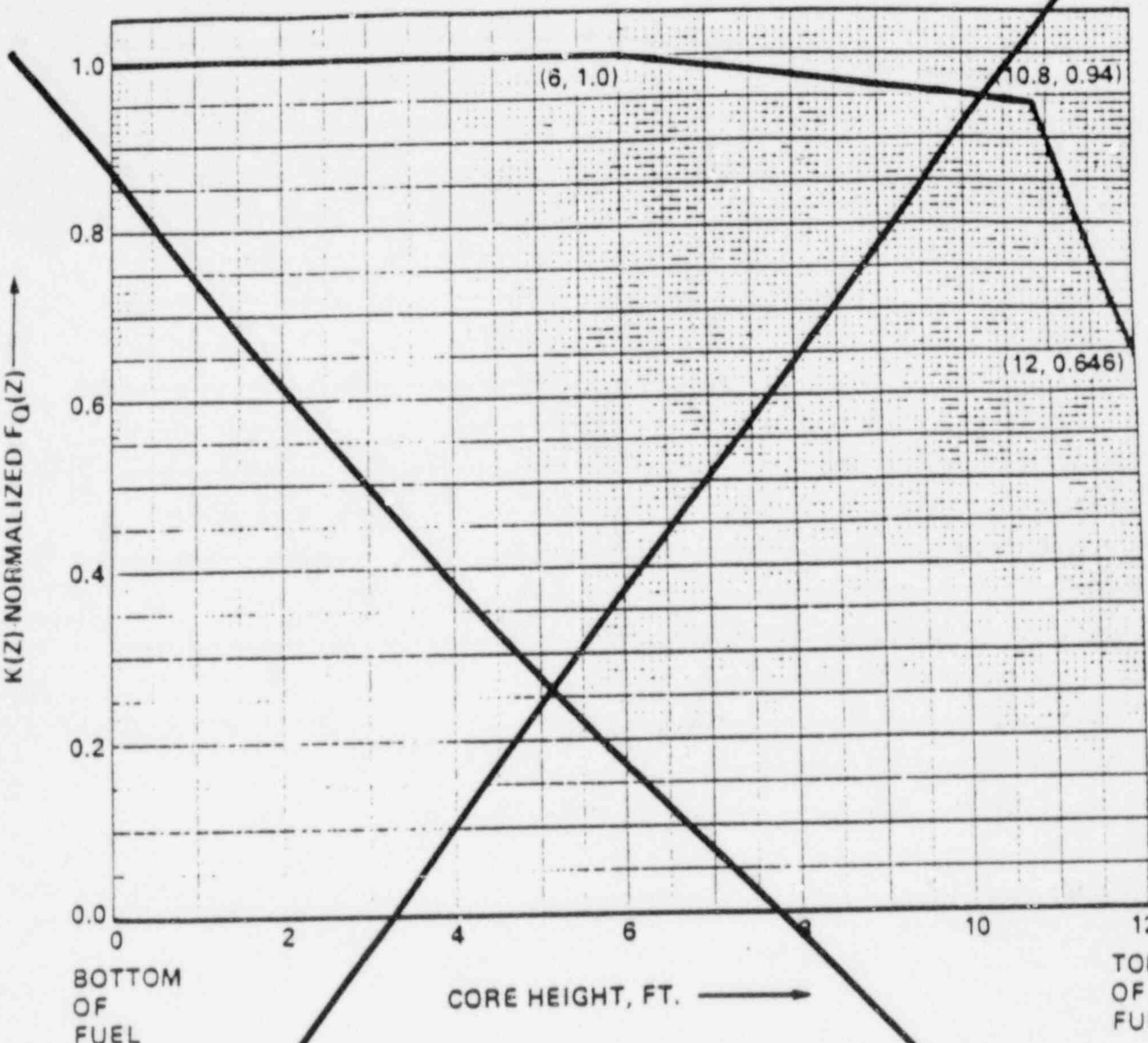


FIGURE 3.2-2

K(Z)-NORMALIZED $F_Q(Z)$ AS A FUNCTION OF CORE HEIGHT

REPLACE WITH NEW FIGURE 3.2-1

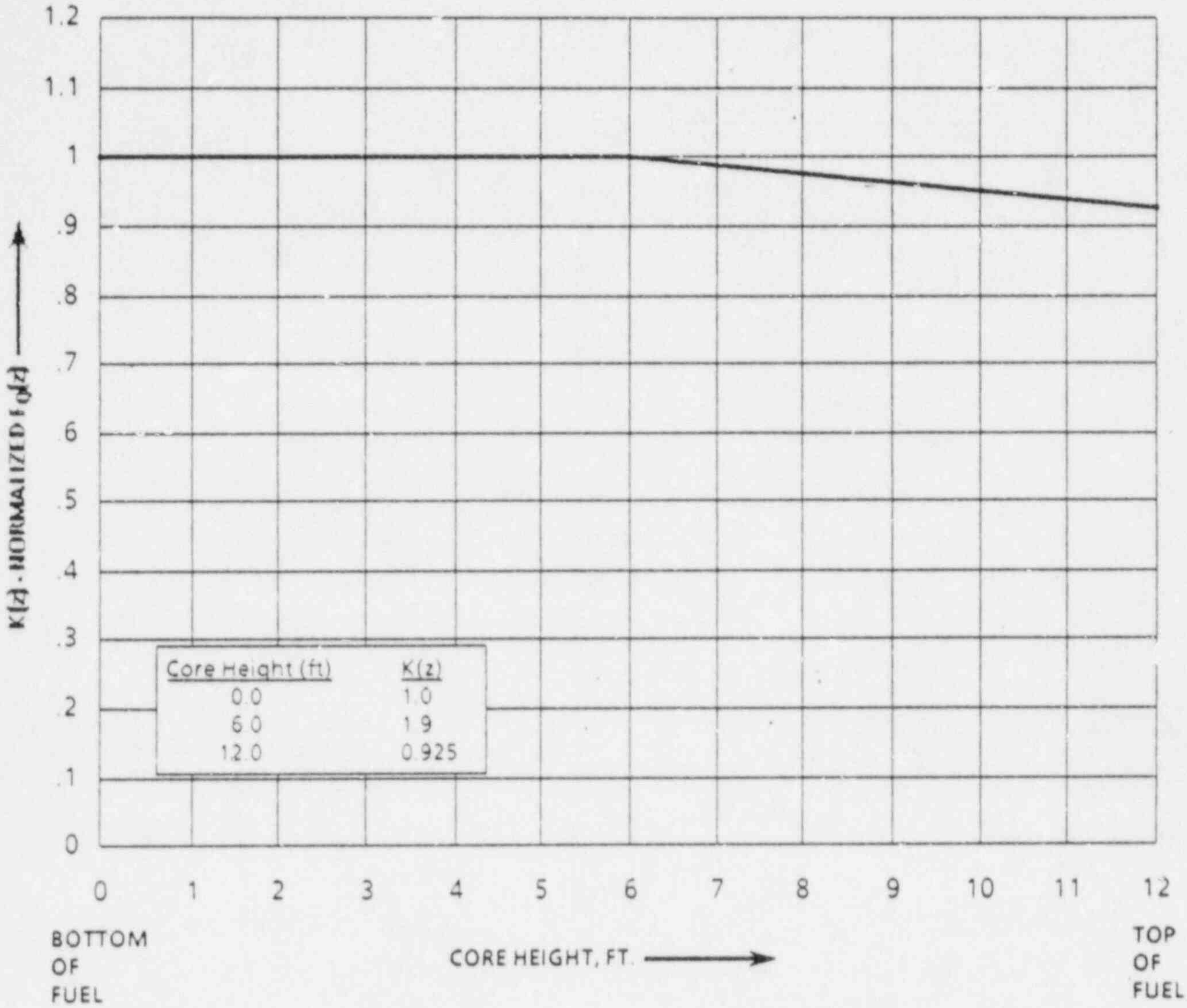


FIGURE 3.2-1

K(z) - NORMALIZED $F_Q(z)$ AS A FUNCTION OF CORE HEIGHT

POWER DISTRIBUTION LIMITS

3/4.2.3 RCS FLOW RATE AND NUCLEAR ENTHALPY RISE HOT CHANNEL FACTOR

LIMITING CONDITION FOR OPERATION

3.2.3 The combination of indicated Reactor Coolant System (RCS) total flow rate and R shall be maintained within the region of allowable operation shown on Figure ~~3.2-3~~ 3.2-2 for 3 loop operation.

Where:

a. $$R = \frac{F^N_{\Delta H}}{\cancel{1.49}\ 1.56\ [1.0 + \cancel{0.2}\ 0.3\ (1.0 - P)]},$$

b. $$P = \frac{\text{THERMAL POWER}}{\text{RATED THERMAL POWER}},$$

c. $F^N_{\Delta H}$ = Measured values of $F^N_{\Delta H}$ obtained by using the movable incore detectors to obtain a power distribution map. The measured values of $F^N_{\Delta H}$ shall be used to calculate R since Figure ~~3.2-3~~ 3.2-2 includes measurement uncertainties of 2.1% for flow and 4% for incore measurement of $F^N_{\Delta H}$, and

APPLICABILITY: MODE 1.

ACTION:

With the combination of RCS total flow rate and R outside the region of acceptable operation shown on Figure ~~3.2-3~~ 3.2-2:

a. Within 2 hours either:

1. Restore the combination of RCS total flow rate and R to within the above limits, or
2. Reduce THERMAL POWER to less than 50% of RATED THERMAL POWER and reduce the Power Range Neutron Flux - High trip setpoint to less than or equal to 55% of RATED THERMAL POWER within the next 4 hours.

b. Within 24 hours of initially being outside the above limits, verify through incore flux mapping and RCS total flow rate comparison that the combination of R and RCS total flow rate are restored to within the above limits, or reduce THERMAL POWER to less than 5% of RATED THERMAL POWER within the next 2 hours.

c. Identify and correct the cause of the out-of-limit condition prior to increasing THERMAL POWER above the reduced THERMAL POWER limit required by ACTION items a.2. and/or b. above; subsequent POWER OPERATION may proceed provided that the combination of R and indicated RCS total flow rate are demonstrated, through incore flux mapping and RCS total flow rate comparison, to be within the region of acceptable operation shown on Figure ~~3.2-3~~ 3.2-2 prior to exceeding the following THERMAL POWER levels:

ACTION: (Continued)

1. A nominal 50% of RATED THERMAL POWER,
2. A nominal 75% of RATED THERMAL POWER, and
3. Within 24 hours of attaining greater than or equal to 95% of RATED THERMAL POWER.

SURVEILLANCE REQUIREMENTS

4.2.3.1 The provisions of Specification 4.0.4 are not applicable.

4.2.3.2 The combination of indicated RCS total flow rate and R shall be determined to be within the region of acceptable operation of Figure ~~3.2-3~~: 3.2-2

a. Prior to operation above 75% of RATED THERMAL POWER after each fuel loading, and

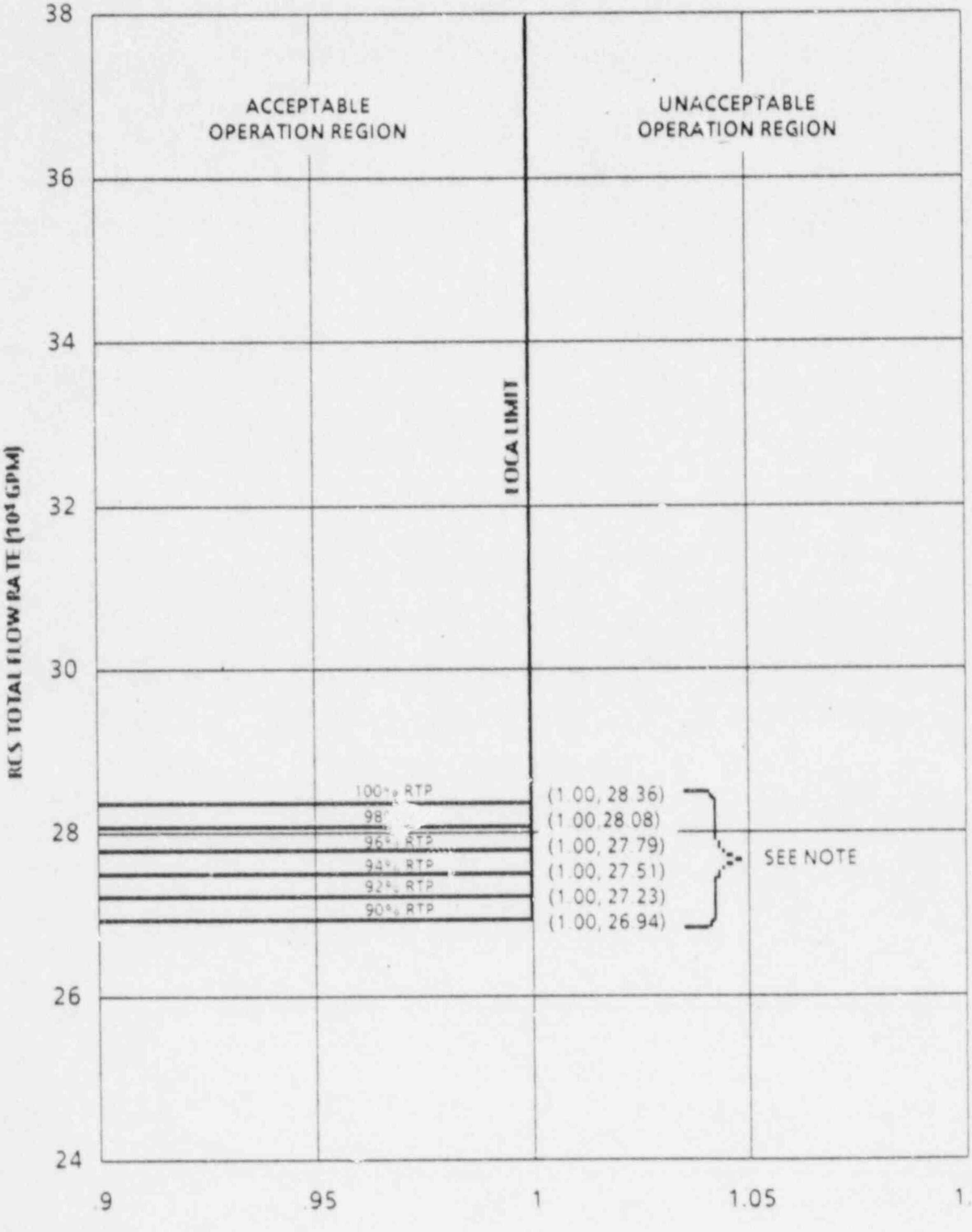
b. At least once per 31 Effective Full Power Days.

4.2.3.3 The indicated RCS total flow rate shall be verified to be within the region of acceptable operation of Figure ~~3.2-3~~ 3.2-3 at least once per 12 hours when the most recently obtained value of R obtained per Specification 4.2.3.2, is assumed to exist.

4.2.3.4 The RCS total flow rate indicators shall be subjected to a CHANNEL CALIBRATION at least once per 18 months.

4.2.3.5 The RCS total flow rate shall be determined by measurement at least once per 18 months.

MEASUREMENT UNCERTAINTIES OF 2.1% FOR FLOW AND 4.0% FOR INCORE MEASUREMENT OF $F^N_{\Delta H}$ ARE INCLUDED IN THIS FIGURE



$R = F^N_{\Delta H} / 1.56 \, [1.0 + 0.3(1.0 - P)]$

FIGURE 3.2-2 RCS TOTAL FLOW RATE VS. R THREE LOOP OPERATION

NOTE: When operationg in this region, the restricted power levels shall be cosidered to be 100% of rated thermal power (RTP) for Figure 2.1-1

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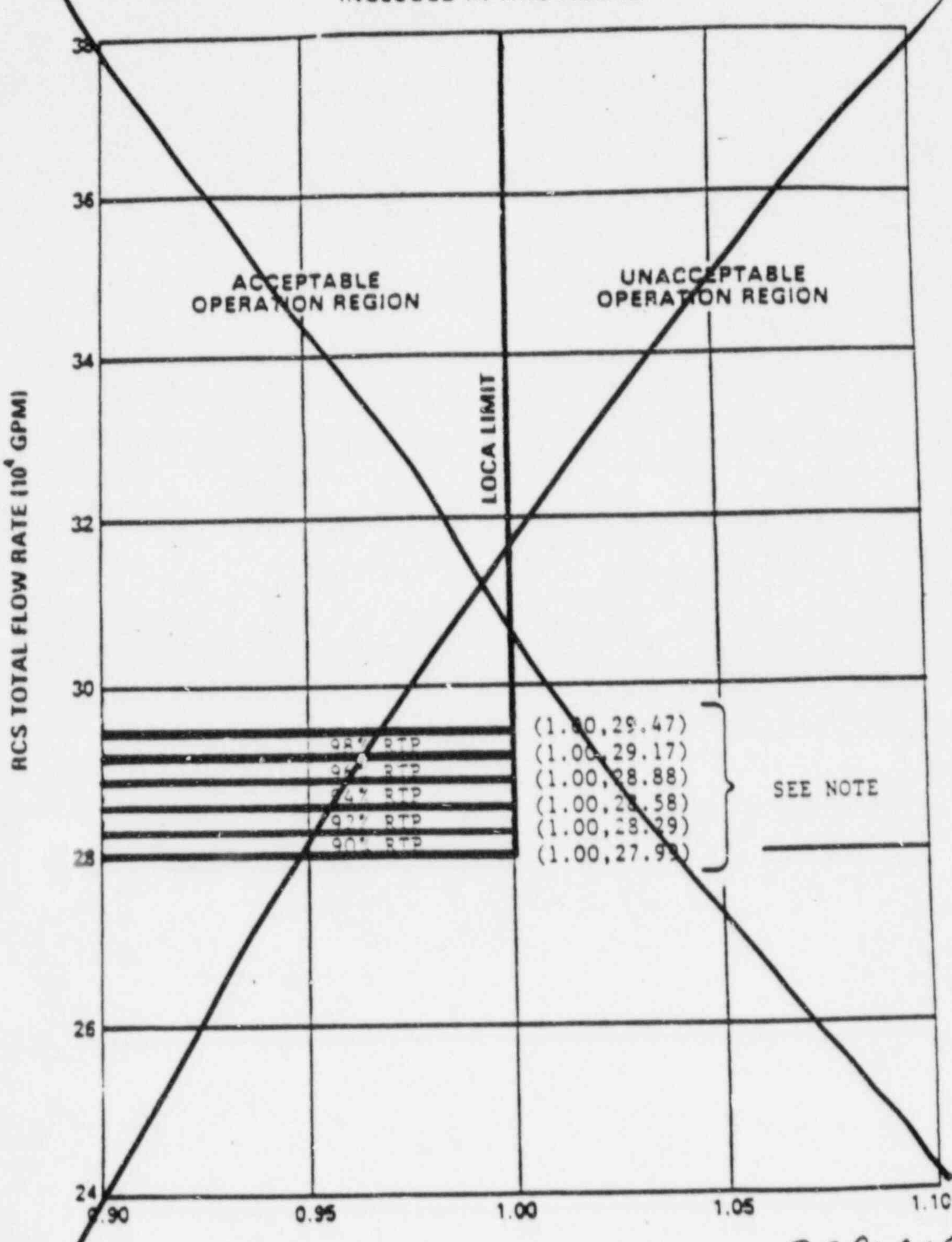
MEASUREMENT UNCERTAINTIES OF 2.1% FOR FLOW AND 4.0% FOR INCORE MEASUREMENT OF $F^N_{\Delta H}$ ARE INCLUDED IN THIS FIGURE



$$R = F^N_{\Delta H}/1.49\left[1.0 + 0.2(1.0 - P)\right]$$

REPLACE WITH New Fig. 3.2-2 on OPPOSITE PAGE.

FIGURE 3.2-3 RCS TOTAL FLOW RATE VS. R THREE LOOP OPERATION

NOTE: When operating in this region, the restricted power levels shall be considered to be 100% of rated thermal power (RTP) for Figure 2.1-1.

TABLE 3.2-1

DNB PARAMETERS

PARAMETER	LIMITS: 3 Loops In Operation	LIMITS: 2 Loops in Operation
Reactor Coolant System T_{avg}	$\leq$ ~~592°F~~ 589.8°F	**
Pressurizer Pressure	$\geq$ ~~2220 psia*~~ 2206 psia*	**

*Limit not applicable during either a THERMAL POWER ramp in excess of 5% of RATED THERMAL POWER per minute or a THERMAL POWER step in excess of 10% of RATED THERMAL POWER.

**These values left blank pending NRC approval of two-loop operation.

TABLE 3.3-2

REACTOR TRIP SYSTEM INSTRUMENTATION RESPONSE TIMES

FUNCTIONAL UNIT	RESPONSE TIME
1. Manual Reactor Trip	Not Applicable
2. Power Range, Neutron Flux	$\leq$ 0.5 seconds*
3. Power Range, Neutron Flux, High Positive Rate	Not Applicable
4. Power Range, Neutron Flux, High Negative Rate	$\leq$ 0.5 seconds*
5. Intermediate Range, Neutron Flux	Not Applicable
6. Source Range, Neutron Flux	Not Applicable
7. Overtemperature ΔT	$\leq$ ~~[illegible]~~ 8.5 seconds*
8. Overpower ΔT	~~Not Applicable~~ $\leq$ 8.5 seconds*
9. Pressurizer Pressure--Low	$\leq$ 2.0 seconds
10. Pressurizer Pressure--High	$\leq$ 2.0 seconds
11. Pressurizer Water Level--High	Not Applicable

*Neutron detectors are exempt from response time testing. Response time of the neutron flux signal portion of the channel shall be measured from detector output or input of first electronic component in channel.

TABLE 3.3-2 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION RESPONSE TIMES

FUNCTIONAL UNIT	RESPONSE TIME
12. A. Loss of Flow - Single Loop (Above P-8)	≤ 1.0 seconds
B. Loss of Flow - Two Loops (Above P-7 and below P-8)	≤ 1.0 seconds
13. Steam Generator Water Level--Low-Low	≤ 2.0 seconds
14. Steam/Feedwater Flow Mismatch and Low Steam Generator Water Level	Not Applicable
15. Undervoltage-Reactor Coolant Pumps	≤ 1.5 seconds
16. Underfrequency-Reactor Coolant Pumps	≤ ~~0.5~~ 0.6 seconds
17. Turbine Trip	
A. Low Fluid Oil Pressure	Not Applicable
B. Turbine Stop Valve Closure	Not Applicable
18. Safety Injection Input from ESF	Not Applicable
19. Reactor Trip System Interlocks	Not Applicable
20. Reactor Trip Breakers	Not Applicable
21. Automatic Trip Logic	Not Applicable

3/4.5.1 ACCUMULATORS

LIMITING CONDITION FOR OPERATION

3.5.1 Each reactor coolant system accumulator shall be OPERABLE with:

a. The isolation valve open,

7489 7685

b. A contained borated water volume of between ~~7368~~ and ~~7582~~ gallons,

c. A boron concentration of between 2200 and 2500 ppm, and

d. A nitrogen cover-pressure of between 600 and 656 psig.

APPLICABILITY: MODES 1, 2 and 3.*

ACTION:

a. With one accumulator inoperable, except as a result of a closed isolation valve, restore the inoperable accumulator to OPERABLE status within one hour or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

b. With one accumulator inoperable due to the isolation valve being closed, either immediately open the isolation valve or be in at least HOT STANDBY within one hour and in HOT SHUTDOWN within the following 12 hours.

SURVEILLANCE REQUIREMENTS

4.5.1.1 Each accumulator shall be demonstrated OPERABLE:

a. At least once per 12 hours by:

1. Verifying the contained borated water volume and nitrogen cover-pressure in the tanks, and

2. Verifying that each accumulator isolation valve is open.

*Pressurizer pressure above 1000 psig.

SPECIAL TEST EXCEPTIONS

3/4.10.2 GROUP HEIGHT, INSERTION AND POWER DISTRIBUTION LIMITS

LIMITING CONDITION FOR OPERATION

3.10.2 The group height, insertion and power distribution limits of Specifications 3.1.3.1, 3.1.3.5, 3.1.3.6, 3.2.1 and 3.2.4 may be suspended during the performance of PHYSICS TESTS provided:

a. The THERMAL POWER is maintained less than or equal to 85% of RATED THERMAL POWER, and

b. The limits of Specifications 3.2.2 and 3.2.3 are maintained and determined at the frequencies specified in Specification 4.10.2.2 below.

APPLICABILITY: MODE 1

ACTION:

With any of the limits of Specifications 3.2.2 or 3.2.3 being exceeded while the requirements of Specifications 3.1.3.1, 3.1.3.5, 3.1.3.6, 3.2.1 and 3.2.4 are suspended, either:

a. Reduce THERMAL POWER sufficient to satisfy the ACTION requirements of Specifications 3.2.2 and 3.2.3, or

b. Be in HOT STANDBY within 6 hours.

SURVEILLANCE REQUIREMENTS

4.10.2.1 The THERMAL POWER shall be determined to be less than or equal to 85% of RATED THERMAL POWER at least once per hour during PHYSICS TESTS.

4.10.2.2 The Surveillance Requirements of the below listed Specifications (a. and b.) shall be performed at least once per 12 hours during PHYSICS TESTS:

a. Either Specifications 4.2.2.2 ~~and 4.2.2.3~~ OR 4.2.2.4 and Specification 4.2.2.5.

b. Specification 4.2.3.2.

SURVEILLANCE REQUIREMENTS (Continued)

h. By performing a flow balance test, during shutdown, following completion of modifications to the ECCS subsystems that alter the subsystem flow characteristics and verifying that:

1) For centrifugal charging pump lines, with a single pump running and with recirculation flow:

a) The sum of the injection line flow rates, excluding the highest flow rate, is greater than or equal to ~~374~~ 338 gpm, and

b) The total pump flow rate is less than or equal to 680 gpm.

i. By performing a flow test, during shutdown, following completion of modifications to the ECCS subsystems that alter the subsystem flow characteristics and verifying that:

1) For residual heat removal pump lines, with a single pump running the sum of the injection line flow rates is greater than or equal to 3663 gpm.

MODERATOR TEMPERATURE COEFFICIENT (Continued)

involved subtracting the incremental change in the MDC associated with a core condition of all rods inserted (most positive MDC) to an all rods withdrawn condition and, a conversion for the rate of change of moderator density with temperature at RATED THERMAL POWER conditions. This value of the MDC was then transformed into the limiting MTC value $\cancel{-4.2}$ -5.0 $\times 10^{-4}$ delta k/k/°F. The MTC value of $\cancel{3.5}$ -4.1 $\times 10^{-4}$ delta k/k/°F represents a conservative value (with corrections for burnup and soluble boron) at a core condition of 300 ppm equilibrium boron concentration and is obtained by making these corrections to the limiting MTC value of $\cancel{-4.2}$ -5.0 $\times 10^{-4}$ k/k/°F.

The surveillance requirements for measurement of the MTC at the beginning and near the end of the fuel cycle are adequate to confirm that the MTC remains within its limits since this coefficient changes slowly due principally to the reduction in RCS boron concentration associated with fuel burnup.

3/4.1.1.4 MINIMUM TEMPERATURE FOR CRITICALITY

This specification ensures that the reactor will not be made critical with the Reactor Coolant System average temperature less than 551°F. This limitation is required to ensure 1) the moderator temperature coefficient is within its analyzed temperature range, 2) the protective instrumentation is within its normal operating range, 3) the pressurizer is capable of being in an OPERABLE status with a steam bubble, and 4) the reactor pressure vessel is above its minimum RT_{NDT} temperature.

3/4.1.2 BORATION SYSTEMS

The boron injection system ensures that negative reactivity control is available during each mode of facility operation. The components required to perform this function include 1) borated water sources, 2) charging pumps, 3) separate flow paths, 4) boric acid transfer pumps, and 5) an emergency power supply from OPERABLE diesel generators.

With the RCS average temperature above 200°F, a minimum of two boron injection flow paths are required to ensure single functional capability in the event an assumed failure renders one of the flow paths inoperable. The boration capability of either flow path is sufficient to provide the required SHUTDOWN

BORATION SYSTEMS (Continued) bounded 13,300

MARGIN from expected operating conditions of 1.77% delta k/k or as required by Figure 3.1-3 after xenon decay and cooldown to 200°F. The maximum expected boration capability requirement occurs from full power equilibrium xenon conditions and is ~~satisfied~~ by ~~12475~~ gallons of 7000 ppm borated water from the boric acid storage tanks or 54,040 gallons of 2300 ppm borated water from the refueling water storage tank. 453,800

With the RCS temperature below 200°F, one injection system is acceptable without single failure consideration on the basis of the stable reactivity condition of the reactor and the additional restrictions prohibiting CORE ALTERATIONS and positive reactivity changes in the event the single injection system becomes inoperable.

The limitation for a maximum of one centrifugal charging pump to be OPERABLE and the Surveillance Requirement to verify all charging pumps except the required OPERABLE pump to be inoperable below 275°F provides assurance that a mass addition pressure transient can be relieved by the operation of a single PORV.

bounded

The boron capability required below 200°F is sufficient to provide the required SHUTDOWN MARGIN of 1 percent delta k/k or as required by Figure 3.1-3 after xenon decay and cooldown from 200°F to 140°F. This condition is ~~satisfied~~ by either 2080 gallons of 7000 ppm borated water from the boric acid storage tanks or ~~9580~~ gallons of 2300 ppm borated water from the refueling water storage tank.

2,700 37,900

The contained water volume limits include allowance for water not available because of discharge line location and other physical characteristics.

The OPERABILITY of one boron injection system during REFUELING ensures that this system is available for reactivity control while in MODE 6.

3/4.1.3 MOVABLE CONTROL ASSEMBLIES

The specifications of this section ensure that (1) acceptable power distribution limits are maintained, (2) the minimum SHUTDOWN MARGIN is maintained, and (3) limit the potential effects of rod misalignment on associated accident analyses. OPERABILITY of the control rod position indicators is required to determine control rod positions and thereby ensure compliance with the control rod alignment and insertion limits.

3/4.2 POWER DISTRIBUTION LIMITS

BASES

REPLACE WITH INSERT 3

The specifications of this section provide assurance of fuel integrity during Condition I (Normal Operation) and II (Incidents of Moderate Frequency) events by: (a) maintaining the minimum DNBR in the core greater than or equal to 1.30 during normal operation and in short term transients, and (b) limiting the fission gas release, fuel pellet temperature and cladding mechanical properties to within assumed design criteria. In addition, limiting the peak linear power density during Condition I events provides assurance that the initial conditions assumed for the LOCA analyses are met and the ECCS acceptance criteria limit of 2200°F is not exceeded.

The definitions of certain hot channel and peaking factors as used in these specifications are as follows:

$F_Q(Z)$ Heat Flux Hot Channel Factor, is defined as the maximum local heat flux on the surface of a fuel rod at core elevation Z divided by the average fuel rod heat flux, allowing for manufacturing tolerances on fuel pellets and rods.

$F^N_{\Delta H}$ Nuclear Enthalpy Rise Hot Channel Factor, is defined as the ratio of the integral of linear power along the rod with the highest integrated power to the average rod power.

$F_{xy}(Z)$ Radial Peaking Factor, is defined as the ratio of peak power density to average power density in the horizontal plane at core elevation Z.

3/4.2.1 AXIAL FLUX DIFFERENCE

REPLACE WITH INSERT 4

The limits on AXIAL FLUX DIFFERENCE (AFD) assure that the $F_Q(Z)$ upper bound envelope of 2.25 times the normalized axial peaking factor is not exceeded during either normal operation or in the event of xenon redistribution following power changes.

Target flux difference is determined at equilibrium xenon conditions. The full length rods may be positioned within the core in accordance with their respective insertion limits and should be inserted near their normal position for steady state operation at high power levels. The value of the target flux difference obtained under these conditions divided by the fraction of RATED THERMAL POWER is the target flux difference at RATED THERMAL POWER for the associated core burnup conditions. Target flux differences for other THERMAL POWER levels are obtained by multiplying the RATED THERMAL POWER value by the appropriate fractional THERMAL POWER level. The periodic updating of the target flux difference value is necessary to reflect core burnup considerations.

CONTINUED ON NEXT PAGE

AXIAL FLUX DIFFERENCE (Continued)

REPLACE WITH INSERT 4

Although it is intended that the plant will be operated with the AFD within the + 5% target band about the target flux difference, during rapid plant THERMAL POWER reductions, control rod motion will cause the AFD to deviate outside of the target band at reduced THERMAL POWER levels. This deviation will not affect the xenon redistribution sufficiently to change the envelope of peaking factors which may be reached on a subsequent return to RATED THERMAL POWER (with the AFD within the target band) provided the time duration of the deviation is limited. Accordingly, a 1 hour penalty deviation limit cumulative during the previous 24 hours is provided for operation outside of the target band but within the limits of Figure 3.2-1 while at THERMAL POWER levels between 50% and 90% of RATED THERMAL POWER. For THERMAL POWER levels between 15% and 50% of RATED THERMAL POWER, deviations of the AFD outside of the target band are less significant. The penalty of 2 hours actual time reflects this reduced significance.

Provisions for monitoring the AFD on an automatic basis are derived from the plant process computer through the AFD Monitor Alarm. The computer determines the one minute average of each of the OPERABLE excore detector outputs and provides an alarm message immediately if the AFD for at least 2 or more OPERABLE excore channels are outside the target band and the THERMAL POWER is greater than 90% of RATED THERMAL POWER. During operation at THERMAL POWER levels between 50% and 90% and between 15% and 50% RATED THERMAL POWER, the computer outputs an alarm message when the penalty deviation accumulates beyond the limits of 1 hour and 2 hours, respectively.

Figure B 3/4 2-1 shows a typical monthly target band.

3/4.2.2 and 3/4.2.3 HEAT FLUX HOT CHANNEL FACTOR and RCS FLOWRATE AND NUCLEAR ENTHALPY RISE HOT CHANNEL FACTOR

The limits on heat flux hot channel factor, RCS flowrate, and nuclear enthalpy rise hot channel factor ensure that 1) the design limits on peak local power density and minimum DNBR are not exceeded and 2) in the event of a LOCA the peak fuel clad temperature will not exceed the 2200°F ECCS acceptance criteria limit.

Each of these is measurable but will normally only be determined periodically as specified in Specifications 4.2.2 and 4.2.3. This periodic surveillance is sufficient to insure that the limits are maintained provided:

a. Control rods in a single group move together with no individual rod insertion differing by more than ± 13 steps, indicated, from the group demand position.

b. Control rod groups are sequenced with overlapping groups as described in Specification 3.1.3.6.

INSERT 3

The specifications of this section provide assurance of fuel integrity during Condition I (Normal Operation) and II (Incidents of Moderate Frequency) events by: (1) maintaining the calculated DNBR in the core at or above the design limit during normal operation and in short-term transients, and (2) limiting the fission gas release, fuel pellet temperature, and cladding mechanical properties to within assumed design criteria. In addition, limiting the peak linear power density during Condition I events provides assurance that the initial conditions assumed for the LOCA analyses are met and the ECCS acceptance criteria limit of 2200°F is not exceeded.

The definitions of certain hot channel and peaking factors as used in these specifications are as follows:

$F_Q(Z)$ Heat Flux Hot Channel Factor, is defined as the maximum local heat flux on the surface of a fuel rod at core elevation Z divided by the average fuel rod heat flux, allowing for manufacturing tolerances on fuel pellets and rods;

$F^N_{\Delta H}$ Nuclear Enthalpy Rise Hot Channel Factor, is defined as the ratio of the integral of linear power along the rod with the highest integrated power to the average rod power.

INSERT 4

The limits on AXIAL FLUX DIFFERENCE (AFD) assure that the $F_Q(Z)$ upper bound envelope of 2.45 times the normalized axial peaking factor is not exceeded during either normal operation or in the event of xenon redistribution following power changes.

The limits on AFD will be provided in the Peaking Factor Limit Report (PFLR) per Technical Specification 6.9.1.11.

Target flux difference is determined at equilibrium xenon conditions. The full-length rods may be positioned within the core in accordance with their respective insertion limits and should be inserted near their normal position for steady-state operation at high power levels. The value of the target flux difference obtained under these conditions divided by the fraction of RATED THERMAL POWER is the target flux difference at RATED THERMAL POWER for the associated core burnup conditions. Target flux differences for other THERMAL POWER levels are obtained by multiplying the RATED THERMAL POWER value by the appropriate fractional THERMAL POWER level. The periodic updating of the target flux difference value is necessary to reflect core burnup considerations.

At power levels below APL^{ND}, the limits on AFD are defined in the PFLR consistent with the Relaxed Axial Offset Control (RAOC) operating procedure and limits. These limits were calculated in a manner such that expected operational transients, e.g. load follow operations, would not result in the AFD deviating outside of those limits. However, in the event such a deviation occurs, the short period of time allowed outside of the limits at reduced power levels will not result in significant xenon redistribution such that the envelope of peaking factors would change sufficiently to prevent operation in the vicinity of the APL^{ND} power level.

At power levels greater than APL^{ND}, two modes of operation are permissible; 1) RAOC, the AFD limit of which are defined in the PFLR and 2) Base Load operation, which is defined as the maintenance of the AFD within PFLR specifications band about a target value. The RAOC operating procedure above APL^{ND} is the same as that defined for operation below APL^{ND}. However, it is possible when following extended load following maneuvers that the AFD limits may result in restrictions in the maximum allowed power or AFD in order to guarantee operation with $F_Q(z)$ less than its limiting value. To allow operation at the maximum permissible power level the Base Load operating procedure restricts the indicated AFD to relatively small target band (as specified in the PFLR) and power swings($APL^{ND} \leq$ power $\leq APL^{BL}$ or 100% Rated Thermal Power, whichever is lower). For Base Load operation, it is expected that the plant will operate within the target band. Operation outside of the target band for the short time period allowed will not result in significant xenon redistribution such that the envelope of peaking factors would change sufficiently to prohibit continued operation in the power region defined above. To assure there is no residual xenon redistribution impact from past operation on the Base Load operation, a 24-hour waiting period at a power level above APL^{ND} and allowed by RAOC is necessary. During this time period load changes and rod motion are restricted to that allowed by the Base Load procedure. After the waiting period extended Base Load operation is permissible.

The computer determines the one minute average of each of the OPERABLE excore detector outputs and provides an alarm message immediately if the AFD for at least 2 of 4 or 2 of 3 OPERABLE excore channels are: 1) outside the allowed delta-I power operating space (for RAOC operation), or 2) outside the allowed delta-I target band (for Base Load operation). These alarms are active when power is greater than: 1) 50% of RATED THERMAL POWER (for RAOC operation), or 2) APL^{ND} (for Base Load operation). Penalty deviation minutes for Base Load operation are not accumulated based on the short period of time during which operation outside of the target band is allowed.

REPLACE WITH New PAGE B 3/4 2-3
(This change deletes this FIGURE FROM Tech Specs)

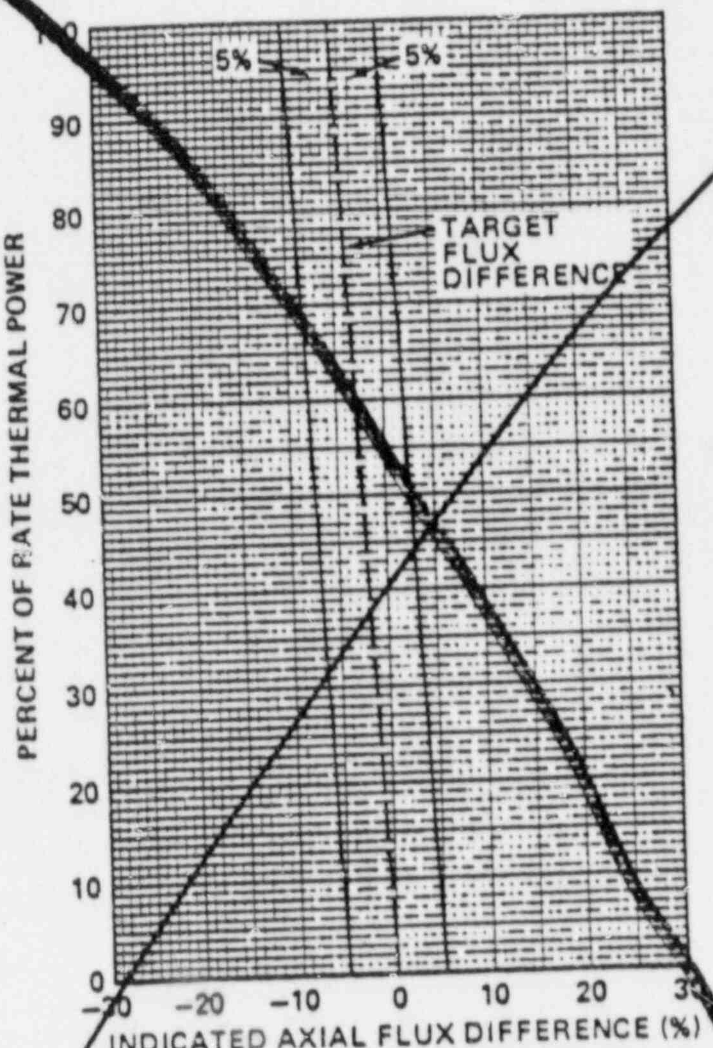


FIGURE B 3/4 2-1
INDICATED AXIAL FLUX DIFFERENCE
VERSUS THERMAL POWER

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HEAT FLUX HOT CHANNEL FACTOR and RCS FLOWRATE and NUCLEAR ENTHALPY RISE HOT CHANNEL FACTOR (Continued)

c. The control rod insertion limits of Specifications 3.1.3.5 and 3.1.3.6 are maintained.

d. The axial power distribution, expressed in terms of AXIAL FLUX DIFFERENCE, is maintained within the limits.

$F^N_{\Delta H}$ will be maintained within its limits provided conditions a. through d. above are maintained. As noted on Figure ~~3.2-3~~ 3.2-2, RCS flow rate and $F^N_{\Delta H}$ may be "traded off" against one another (i.e., a low measured RCS flow rate is acceptable if the measured $F^N_{\Delta H}$ is also low) to ensure that the calculated DNBR will not be below the design DNBR value. The relaxation of $F^N_{\Delta H}$ as a function of THERMAL POWER allows changes in the radial power shape for all permissible rod insertion limits.

R, as calculated in 3.2.3 and used in Figure ~~3.2.3~~ 3.2-2, accounts for $F^N_{\Delta H}$ less than or equal to ~~1.49~~ 1.56. This value is used in the various accident analyses where $F^N_{\Delta H}$ influences parameters other than DNBR, e.g., peak clad temperature and thus is the maximum "as measured" value allowed.

Replace With Insert 5 from following page

~~Fuel rod bowing reduces the value of DNB ratio. Credit is available to offset this reduction in the generic margin. The generic design margins, totaling 9.1% DNBR, completely offset any rod bow penalties.* This margin includes the following:~~

~~1) Design limit DNBR of 1.30 vs. 1.28~~
~~2) Grid Spacing (K_s) of 0.046 vs. 0.059~~
~~3) Thermal Diffusion Coefficient of 0.038 vs. 0.059~~
~~4) DNBR Multiplier of 0.86 vs. 0.88~~
~~5) Pitch reduction~~

~~The applicable value of rod bow penalties is referenced in the FSAR.~~

When an F_Q measurement is taken, an allowance for both experimental error and manufacturing tolerance must be made. An allowance of 5% is appropriate for a full core map taken with the incore detector flux mapping system and a 3% allowance is appropriate for manufacturing tolerance.

Replace With Insert 6 next page

~~The radial peaking factor $F_{xy}(Z)$ is measured periodically to provide assurance that the hot channel factor, $F_Q(Z)$, remains within its limit. The~~ → next page

Delete {

~~*The generic margins also offset the penalty associated with the thermal design flow reduction included in Amendment 45 to the Technical Specifications.~~

HEAT FLUX HOT CHANNEL FACTOR and RCS FLOWRATE and NUCLEAR ENTHALPY RISE HOT CHANNEL FACTOR (Continued)

Replace with Insert 6 - continued from previous page.

~~F_{xy} limit for Rated Thermal Power (F_{xy}^{RTP}) as provided in the Radial Peaking Factor Limit Report per specification 6.9.1.11 was determined from expected power control maneuvers over the full range of burnup conditions in the core.~~

When RCS flow rate and $F_{\Delta H}^{N}$ are measured, no additional allowances are necessary prior to comparison with the limits of Figures 3.2-3. Measurement errors of 2.1% for RCS total flow rate and 4% for $F_{\Delta H}^{N}$ have been allowed for in determining the limits of Figure 3.2-3.

The 12 hour periodic surveillance of indicated RCS flow is sufficient to detect only flow degradation which could lead to operation outside the acceptable region of operation shown on Figure 3.2-3.

3/4.2.4 QUADRANT POWER TILT RATIO

The quadrant power tilt ratio limit assures that the radial power distribution satisfies the design values used in the power capability analysis. Radial power distribution measurements are made during startup testing and periodically during power operation.

The limit of 1.02, at which corrective action is required, provides DNB and linear heat generation rate protection with x-y plane power tilts. A limiting tilt of 1.025 can be tolerated before the margin for uncertainty in F_Q is depleted. The limit of 1.02 was selected to provide an allowance for the uncertainty associated with the indicated power tilt.

The two hour time allowance for operation with a tilt condition greater than 1.02 but less than 1.09 is provided to allow identification and correction of a dropped or misaligned control rod. In the event such action does not correct the tilt, the margin for uncertainty on F_Q is reinstated by reducing the maximum allowed power by 3 percent for each percent of tilt in excess of 1.0.

For purposes of monitoring QUADRANT POWER TILT RATIO when one excore detector is inoperable the movable incore detectors are used to confirm that the normalized symmetric power distribution is consistent with the QUADRANT POWER TILT RATIO. The incore detector monitoring is done with a full incore flux map or two sets of 4 symmetric thimbles. These locations are C-8, E-5, E-11, H-3, H-13, L-5, L-11, N-8.

3/4.2.5 DNB PARAMETERS

The limits on the DNB related parameters assure that each of the parameters are maintained within the normal steady state envelope of operation assumed in the transient and accident analyses. The limits are consistent

INSERT 5

Margin is maintained between the safety analysis limit DNBR and the design limit DNBR. This margin is more than sufficient to offset any rod bow penalty and transition core penalty. The remaining margin is available for plant design flexibility.

INSERT 6

The hot channel factor $F^M_Q(z)$ is measured periodically and increased by a cycle and height dependent power factor appropriate to either RAOC or Base Load operation, $W(z)$ or $W(z)_{BL}$, to provide assurance that the limit on the hot channel factor, $F_Q(z)$ is met. $W(z)$ accounts for the effects of normal operation transients and was determined from expected power control maneuvers over the full range of burnup conditions in the core. $W(z)_{BL}$ accounts for the more restrictive operating limits allowed by Base Load operation which result in less severe transient values. The $W(z)$ and $W(z)_{BL}$ functions described above for normal operation are provided in the Peaking Factor Limit Report per Specification 6.9.1.11.

3/4.5 EMERGENCY CORE COOLING SYSTEMS

BASES

3/4.5.1 ACCUMULATORS

The OPERABILITY of each Reactor Coolant System (RCS) accumulator ensures that a sufficient volume of borated water will be immediately forced into the reactor core through each of the cold legs in the event the RCS pressure falls below the pressure of the accumulators. This initial surge of water into the core provides the initial cooling mechanism during large RCS pipe ruptures. (Insert 7) ADD

The limits on accumulator volume, boron concentration and pressure ensure that the assumptions used for accumulator injection in the safety analysis are met.

The accumulator power operated isolation valves are considered to be "operating bypasses" in the context of IEEE Std. 279-1971, which requires that bypasses of a protective function be removed automatically whenever permissive conditions are not met. In addition, as these accumulator isolation valves fail to meet single failure criteria, removal of power to the valves is required.

The limits for operation with an accumulator inoperable for any reason except an isolation valve closed minimizes the time exposure of the plant to a LOCA event occurring concurrent with failure of an additional accumulator which may result in unacceptable peak cladding temperatures. If a closed isolation valve cannot be immediately opened, the full capability of one accumulator is not available and prompt action is required to place the reactor in a mode where this capability is not required.

3/4.5.2 and 3/4.5.3 EMERGENCY CORE COOLING SYSTEM (ECCS) SUBSYSTEMS

The OPERABILITY of two independent ECCS subsystems ensures that sufficient emergency core cooling capability will be available in the event of a LOCA assuming the loss of one subsystem through any single failure consideration. Either subsystem operating in conjunction with the accumulators is capable of supplying sufficient core cooling to limit the peak cladding temperatures within acceptable limits for all postulated break sizes ranging from the double ended break of the largest RCS cold leg pipe downward. In addition, each ECCS subsystem provides long term core cooling capability in the recirculation mode during the accident recovery period.

With the RCS temperature below 350°F, one OPERABLE ECCS subsystem is acceptable without single failure consideration on the basis of the stable reactivity condition of the reactor and the limited core cooling requirements.

INSERT 7

In addition, the borated water servos to limit the maximum power which may be reached during large secondary pipe ruptures.

e. Type of container (e.g., LSA, Type A, Type B, Large Quantity), and

f. Solidification agent (e.g., cement, urea formaldehyde).

The radioactive effluent release reports shall include unplanned releases from site to unrestricted areas of radioactive materials in gaseous and liquid effluents on a quarterly basis.

The radioactive effluent release reports shall include any changes to the Process Control Program (PCP) made during the reporting period.

MONTHLY OPERATING REPORT

6.9.1.10 Routine reports of operating statistics and shutdown experience, including documentation of all challenges to the PORV's or safety valves, shall be submitted on a monthly basis to the Director, Office of Resource Management, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555, with a copy to the Regional Office of Inspection and Enforcement, no later than the 15th of each month following the calendar month covered by the report.

Any changes to the OFFSITE DOSE CALCULATION MANUAL shall be submitted with the Monthly Operating Report within 90 days in which the change(s) was made effective. In addition, a report of any major changes to the radioactive waste treatment systems shall be submitted with the Monthly Operating Report for the period in which the evaluation was reviewed and accepted as set forth in 6.5 above.

REPLACE WITH INSERT 8

~~RADIAL PEAKING FACTOR LIMIT REPORT~~

~~6.9.1.11 The F_{xy} limit for Rated Thermal Power (F_{xy}^{RTP}) shall be provided to the Regional Administrator of the Regional Office of Inspection and Enforcement, with a copy to the Director, Nuclear Reactor Regulation, Attention Chief of the Core Performance Branch, U. S. Nuclear Regulatory Commission, Washington, D.C. 20555 for all core planes containing bank "D" control rods and all unrodded core planes at least 60 days prior to cycle initial criticality. In the event that the limit would be submitted at some other time during core life, it shall be submitted 60 days prior to the date the limit would become effective unless otherwise exempted by the Commission.~~

~~Any information needed to support F_{xy}^{RTP} will be by request from the NRC and need not be included in this report.~~

SPECIAL REPORTS

6.9.2 Special reports shall be submitted to the Regional Administrator of the Office of Inspection and Enforcement Regional Office within the time period specified for each report.

6.10 RECORD RETENTION

In addition to the applicable record retention requirements of Title 10, Code of Federal Regulations, the following records shall be retained for at least the minimum period indicated.

INSERT 8

6.9.1.11 PEAKING FACTOR LIMIT REPORT

The AFD limits, the W(z) Functions for RAOC and Base Load operation and the value for APL^{ND} (as required) shall be established for each reload core and implemented prior to use.

The methodology used to generate the W(z) functions for RAOC and Base Load Operation and the value for APL^{ND} shall be those previously reviewed and approved by the NRC*. If changes to these methods are deemed necessary they will be evaluated in accordance with 10CFR50.59 and submitted to the NRC for review and approval prior to their use if the change is determined to involve an unreviewed safety question or if such a change would require amendment of previously submitted documentation.

A report containing the AFD limits, the W(z) functions for RAOC and Base Load operation and the value for APL^{ND} (as required) shall be provided to the NRC document control desk with copies to the regional administrator and the resident inspector within 30 days of their implementation.

Any information needed to support W(z), $W(z)_{BL}$ and APL^{ND} will be by request from the NRC and need not be included in this report.

*WCAP-10216 P-A "Relaxation of Constant Axial Offset Control-F_Q Surveillance Technical Specification".

ATTACHMENT 3

NON-LOCA ACCIDENT ANALYSIS

FOR THE

V. C. SUMMER PLANT

TRANSITION TO 17x17 VANTAGE 5 FUEL

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FIGURES

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FIGURES

Chapter 15

ACCIDENT ANALYSES

Since 1970, the ANS classification of plant conditions has been used to divide plant conditions into four categories in accordance with anticipated frequency of occurrence and potential radiological consequences to the public. The four categories are as follows:

(1) Condition I: Normal Operation and Operational Transients

(2) Condition II: Faults of Moderate Frequency

(3) Condition III: Infrequent Faults

(4) Condition IV: Limiting Faults.

The basic principle applied in relating design requirements to each of the conditions is that the most frequent occurrences must yield little or no radiological risk to the public, and those extreme situations having the potential for the greatest risk to the public shall be those least likely to occur. Where applicable, reactor trip system and engineered safety features functioning is assumed, to the extent allowed by considerations such as the single failure criterion, in fulfilling this principle.

In the evaluation of the radiological consequences associated with initiation of a spectrum of accident conditions, numerous assumptions must be postulated. In many instances these assumptions are a product of extremely conservative judgments. This is due to the fact that many physical phenomena, in particular fission product transport under accident conditions, are not understood to the extent that accurate predictions can be made. Therefore, the set of assumptions postulated would predominantly determine the accident classification.

The specific accident sequences analyzed in this chapter include those required by Revision 1 of Regulatory Guide 1.70, Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants, and others considered significant for V. C. Summer. Because the V. C. Summer design differs from other plants, some of the accidents identified in Table 15-1 of Regulatory Guide 1.70, Revision 1, are not applicable to this plant; some comments on these items are as follows:

(Item 10) - There are no pressure regulators or regulating instruments in the Westinghouse pressurized water reactor (PWR) design whose failure could cause heat removal greater than heat generation.

(Item 11) - Reactor coolant flow controller is not a feature of the Westinghouse PWR design. Treatment of the performance of the reactivity controller in a number of accident conditions is offered in this chapter.

(Item 12) - The analysis of specific effects of internal and external events such as major and minor fires, floods, storms, or earthquakes are generally discussed in Chapter 3. Refer to Section 3.1.2.1 for guidance on which FSAR sections specifically address GDCs 2, 3 and 4.

(Item 22) - No instrument lines from the RCS boundary in the V. C. Summer design penetrate the containment[a].

(a) For definition of the RCS boundary, refer to the 1972 issue of ANS N18.2, Nuclear Safety Criteria for the Design of Stationary PWR Plants.

(Item 26) - Habitability of the control room following accident conditions is discussed. In addition, Chapter 7 contains an analysis showing that the plant can be brought to, and maintained in, the hot shutdown condition from outside the control room.

(Item 27) - Overpressurization of the residual heat removal system (RHRS) is considered extremely unlikely due to the isolation valve interlocks described in Section 7.6.

(Item 28) - This event is covered by the analyses of Section 15.2.7, Loss of External Electrical Load and/or Turbine Trip.

(Item 29) - Same as Item 28 above.

(Item 30) - Loss of the service water system is discussed in Section 9.2.

(Item 31) - Loss of one DC system is discussed in Chapter 8.

(Item 33) - The effects of turbine trip on the RCS are presented in Section 15.2.7. Turbine trip with failure of the generator breakers to open is discussed in Chapter 10.

(Item 34) - Malfunctions of this system are discussed in Chapter 9.

(Item 35) - The radiological effects of this event are not significant for PWR plants.

TABLE 15.0-1

EQUIPMENT AVAILABLE FOR TRANSIENT AND ACCIDENT CONDITIONS

Incident	Reactor Trip Functions	ESF Actuation Functions	Other Equipment	ESF Equipment
1. Uncontrolled RCAA Bank Withdrawal from	Power range high flux (low s.p.), manual	-	-	-
2. Uncontrolled RCCA Bank Withdrawal at Power	Power range high flux OTΔT, hi pressurizer pressure, manual	-	Pressurizer safety valves, steam generator safety valves	-
3. RCCA Misalignment	Power range negative flux rate, manual	-	-	-
4. Uncontrolled Boron Dilution	Source range high flux, power range high flux, OTΔT, manual	Low insertion limit annunciators for boration	-	-
5. Startup of an Inactive Reactor Coolant Loop	Power range high flux, manual	-	-	-
6. Loss of External Electrical Load and/or Turbine Trip	High pressurizer pressure OTΔT, manual	-	Pressurizer safety valves, steam generator safety valves	-
7. Loss of Normal Feedwater	Steam generator lo-lo level, manual	Steam generator lo-lo level	-	One motor driven emergency feedwater pump
8. Loss of Offsite Power to the Station Auxiliaries	Same as 7	Same as 7	Same as 7	Same as 7
9. Excess Heat Removal due to Feedwater System Malfunctions	Power range high flux, high steam generator level, manual	High steam generator level produced feedwater isolation and turbine trip	Feedwater isolation valves	-
10. Excessive Load Increase Incident	Power range high flux, OTΔT, OPΔT, manual	-	Pressurizer self-actuated safety valves, steam generator safety valves	-
11. Accidental Depressurization of the RCS	Pressuri er low pressur OTΔT, manual	-	-	-
12. Major Rupture of Main Steam Line	SIS, manual	Low pressurizer pressure, low compensated steam line pressure, hi-1 containment pressure, manual	Feed line isolation valves, steam line isolation valves	Emergency feedwater system, SI equipment minus either one SI charging pump, or one diesel generator.

TABLE 15.0-1 (Cont'd)

EQUIPMENT AVAILABLE FOR TRANSIENT AND ACCIDENT CONDITIONS

Incident	Reactor Trip Functions	ESF Actuation Functions	Other Equipment	ESF Equipment
13. Complete Loss of Forced Reactor Coolant Flow	Low flow, undervoltage underfrequency, manual	-	-	-
14. Rupture of a Control Rod Drive Mechanism Housing	Power range high flux, manual	-	-	-
15. Single RCCA Withdrawal at Full Power	OTΔT, manual	-	-	-
16. Major Rupture of a Main Feedwater Line	Lo steam generator level plus steam/feed mismatch, SIS, manual	High containment pressure, high pressurizer pressure, steam generator low-low water level, low compensated steam line pressure	Steam line isolation valves, feed line isolation pressurizer self-actuated safety valves, steam generator safety valves	Emergency feedwater pumps
17. Large Break LOCA	Reactor trip system	Engineered safety features actuation system	Service water system, component cooling water system	Emergency core cooling system, containment heat removal system, emergency power system
18. Small Break LOCA	Reactor trip system	Engineered safety features actuation system	Service water system, component cooling water system, generator safety and/or relief valves	Emergency core cooling system, emergency feedwater system containment heat removal system emergency power system
19. Steam Generator Tube Rupture	Reactor trip system	Engineered safety features actuation system	Service water system, component cooling water system, steam generator shell side fluid operating system, steam generator safety and/or relief valve, steam line isolation valves	Emergency core cooling system, emergency feedwater system, emergency power systems

15.1 CONDITION I - NORMAL OPERATION AND OPERATIONAL TRANSIENTS

Condition I occurrences are those that are expected frequently or regularly in the course of power operation, refueling, maintenance, or maneuvering of the plant. As such, Condition I occurrences are accommodated with margin between any plant parameter and the value of that parameter which would require either automatic or manual protective action. Inasmuch as Condition I occurrences occur frequently or regularly, they must be considered from the point of view of affecting the consequences of fault conditions (Conditions II, III and IV). In this regard, analysis of each fault condition is generally based on a conservative set of initial conditions corresponding to the most adverse set of conditions that can occur during Condition I operation.

A typical list of Condition I events is shown below:

(1) Steady state and shutdown operations

Mode 1 - Power operation (> 5% of rated thermal power)

Mode 2 - Startup ($K_{eff} \geq 0.99$, $\leq$ 5% of rated thermal power)

Mode 3 - Hot standby ($K_{eff} < 0.99$, $T_{avg} \geq 350°F$)

Mode 4 - Hot shutdown (subcritical, residual heat removal system in operation, $K_{eff} < 0.99$, $200°F < T_{avg} < 350°F$)

Mode 5 - Cold shutdown (subcritical, residual heat removal system in operation, $K_{eff} < 0.99$, $T_{avg} \leq 200°F$)

Mode 6 - Refueling ($K_{eff} \leq 0.95$, $T_{avg} \leq 140°F$)

(2) Operation with permissible deviations

Various deviations that may occur during continued operation as permitted by the plant Technical Specifications[1] must be considered in conjunction with other operational modes. These include:

(a) Operation with components or systems out of service

(b) Leakage from fuel with cladding defects

(c) Activity in the reactor coolant

1. Fission products

2. Corrosion products

3. Tritium

(d) Operation with steam generator leaks up to the maximum allowed by the Technical Specifications

(e) Testing .s allowed by the Technical Specifications

(3) Operational transients

(a) Plant heatup and cooldown (up to 100°F/hour for the reactor coolant system (RCS); 200°F/hour for the pressurizer)

(b) Step load changes (up to ±10%)

(c) Ramp load changes (up to 5% per minute)

(d) Load rejection up to and including design load rejection transient

15.1.1 Optimization of Control Systems

A setpoint study[2] has been performed in order to simulate performance of the reactor control and protection systems. Emphasis was placed on the development of a control system that will automatically maintain prescribed conditions in the plant even under the most conservative set of reactivity parameters with respect to both system stability and transient performance.

For each mode of plant operation, a group of optimum controller setpoints is determined. In areas where the resultant setpoints are different, compromises based on the optimum overall performance are made and verified. A consistent set of control system parameters is derived satisfying plant operational requirements throughout the core life and for power levels between 15 and 100%. The study comprises an analysis of the following control systems: rod cluster assembly control, steam dump, steam generator level, pressurizer pressure, and pressurizer level.

15.1.2 Initial Power Conditions Assumed in Accident Analyses

Reactor power-related initial conditions assumed in the accident analyses presented in this chapter are described in this section.

15.1.2.1 Power Rating

Table 15.1-1 lists the principal power rating values that are assumed in analyses performed in this section. Two ratings are given:

(1) The guaranteed nuclear steam supply system (NSSS) thermal power output. This power output includes the thermal power generated by the reactor coolant pumps.

(2) The engineered safety features (ESF) design rating. The Westinghouse-supplied ESFs are designed for a thermal power higher than the guaranteed value in order not to preclude realization of future potential power capabilty. This higher thermal power value is designated as the ESF design rating. This power output includes the thermal power generated by the reactor coolant pumps.

Where initial power operating conditions are assumed in accident analyses, the guaranteed NSSS thermal power output (plus allowance for errors in steady state power determination for some accidents) is assumed. Where demonstration of the adequacy of the containment and ESF is concerned, the ESF design rating plus allowance for error is assumed. The thermal power values for each transient analyzed are given in Table 15.1-4.

15.1.2.2 Initial Conditions

For most accidents which are DNB limited, nominal values of initial conditions are assumed. The allowances on power, temperature, and pressure are determined on a statistical basis and are included in the limit DNBR, as described in Reference 3. This procedure is known as the "Improved Thermal Design Procedure" (ITDP) and these accidents utilize the WRB-1 and WRB-2 DNB correlations (References 4 and 5). ITDP allowances may be more restrictive than non-ITDP allowances. The initial conditions for other key parameters are selected in such a manner to maximize the impact on DNBR. Minimum measured flow is used in all ITDP transients.

For accident evaluations that are not DNB-limited, or for which the Improved Thermal Design Procedure is not employed, the initial conditions are obtained by adding maximum steady state errors to rated values. The following steady state errors are considered:

(1) Core power	+2.0%/-2.1% allowance calorimetric error
(2) Average RCS temperature	+4.0°F/-4.3°F allowance for deadband and measurement error
(3) Pressurizer pressure	+33 psi/-45 psi allowance for steady state fluctuations and measurement error.

15.1.2.3 Power Distribution

The transient response of the reactor system is dependent on the initial power distribution. The nuclear design of the reactor core minimizes adverse power distribution through the placement of fuel assemblies, control rods, and by operation instructions. The power distribution may be characterized by the radial peaking factor $F_{\Delta H}$ and the total peaking factor F_Q. The peaking factor limits are given in Technical Specification 3/4.2.

For transients that may be DNB-limited, the radial peaking factor is of importance. The radial peaking factor increases with decreasing power level due to rod insertion. This increase in $F_{\Delta H}$ is included in the core limits illustrated on Figure 15.1-1. All transients that may be DNB-limited are assumed to begin with an $F_{\Delta H}$ consistent with the initial power level defined in the Technical Specifications.

The axial power shape used in the DNB calculation is discussed in Section 4.4.3.

For transients that may be overpower-limited, the total peaking factor F_Q is of importance. The value of F_Q may increase with decreasing power level so that the full power hot spot heat flux is not exceeded, i.e., F_Q x Power = design hot spot heat flux. All transients that may be overpower-limited are assumed to begin with a value of F_Q consistent with the initial power level as defined in the Technical Specifications.

The value of peak kW/ft can be directly related to fuel temperature as illustrated on Figures 4.4-1 and 4.4-2. For transients that are slow with respect to the fuel rod thermal time constant (approximately 5 seconds), the fuel temperatures are illustrated on Figures 4.4-1 and 4.4-2. For transients that are fast with respect to the fuel rod thermal time constant (for example, rod ejection), a detailed heat transfer calculation is made.

15.1.3 Trip Points and Time Delays to Trip Assumed in Accident Analyses

A reactor trip signal acts to open two trip breakers connected in series feeding power to the control rod drive mechanisms. The loss of power to the mechanism coils causes the mechanism to release the rod cluster control assemblies (RCCAs) which then fall by gravity into the core. There are various instrumentation delays associated with each trip function, including delays in signal actuation, in opening the trip breakers, and in the release of the rods by the mechanisms. The total delay to trip is defined as the time delay from the time that trip conditions are reached to the time the rods are free and begin to fall. Limiting trip setpoints assumed in accident analyses and the time delay assumed for each trip function are given in Table 15.1-2. Reference is made in that table to the overtemperature and overpower ΔT trip shown on Figure 15.1-1. This figure presents the allowable reactor coolant loop average temperature and ΔT for the design flow and the NSSS Design Thermal Power distribution as a function of primary coolant pressure. The boundaries of operation defined by the Overpower ΔT trip and the Overtemperature ΔT trip are represented as "protection lines" on this diagram. The protection lines are drawn to include all adverse instrumentation and setpoint errors so that under nominal conditions trip would occur well within the area bounded by these lines. The utility of this diagram is in the fact that the limit imposed by any given DNBR can be represented as a line. The DNB lines represent the locus of conditions for which the DNBR equals the safety analysis limit values (1.44 and 1.48 for Standard thimble cell and typical cells, respectively; 1.60 and 1.68 for V-5 thimble cell and typical cells, respectively) for ITDP accidents. All points below and to the left of a DNB line for a given pressure have a DNBR greater than the limit values. The diagram shows that DNB is prevented for all cases if the area enclosed with the maximum protection lines is not traversed by the applicable DNBR line at any point.

The area of permissible operation (power, pressure and temperature) is bounded by the combination of reactor trips: high neutron flux (fixed setpoint); high pressurizer pressure (fixed setpoint); low pressurizer pressure (fixed setpoint); Overpower and Overtemperature ΔT (variable setpoints).

The limit values, which were used as the DNBR limits for all accidents analyzed with the Improved Thermal Design Procedure are conservative compared to the actual design DNBR values required to meet the DNB design basis.

The difference between the limiting trip point assumed for the analysis and the normal trip point represents an allowance for instrumentation channel error and setpoint error. During startup tests, it is demonstrated that actual instrument errors and time delays are equal to or less than the assumed values.

15.1.4 Rod Cluster Control Assembly Insertion Characteristic

The negative reactivity insertion following a reactor trip is a function of the acceleration of the RCCA and the variation in rod worth as a function of rod position.

With respect to accident analyses, the critical parameter is the time of insertion up to the dashpot entry or approximately 85% of the rod cluster travel. For accident analyses, the insertion time to dashpot entry is conservatively taken as 2.7 seconds. The RCCA position versus time assumed in accident analyses is shown on Figure 15.1-2.

Figure 15.1-3 shows the fraction of total negative reactivity insertion for a core where the axial distribution is skewed to the lower region of the core. This curve is used as input to all point kinetics core models used in transient analyses.

There is inherent conservatism in the use of this curve in that it is based on a skewed axial power distribution that would exist relatively infrequently. For cases other than those associated with xenon oscillations, significant negative reactivity would have been inserted due to the more favorable axial power distribution existing prior to trip.

The normalized RCCA negative reactivity insertion versus time is shown on Figure 15.1-4. The curve shown in this figure was obtained from Figures 15.1-2 and 15.1-3. A total negative reactivity insertion following a trip of 4.8% Δk is assumed in the transient analyses except where specifically noted otherwise. This assumption is conservative with respect to the calculated trip reactivity worth available as shown in Tables 4.3-2 and 4.3-3.

The normalized RCCA negative reactivity insertion versus time curve for an axial power distribution skewed to the bottom (Figure 15.1-4) is used in transient analyses.

Where special analyses require the use of three-dimensional or axial one-dimensional core models, the negative reactivity insertion resulting from reactor trip is calculated directly by the reactor kinetic code and is not separable from other reactivity feedback effects. In this case, the RCCA position versus time of Figure 15.1-2 is used as a code input.

15.1.5 Reactivity Coefficients

The transient response of the reactor coolant system is dependent on reactivity feedback effects, in particular the moderator temperature coefficient and the Doppler power coefficient. These reactivity coefficients and their values are discussed in detail in Chapter 4.

In the analysis of certain events, conservatism requires the use of large reactivity coefficient values, whereas in the analysis of other events, conservatism requires the use of small reactivity coefficient values. Some analyses, such as loss of reactor coolant from cracks or ruptures in the RCS, do not depend on reactivity feedback effects. The values used are given in Table 15.1-4; reference is made in that table to Figure 15.1-5 that shows the upper and lower Doppler power coefficients, as a function of power, used in the transient analysis. The justification for use of conservatively large versus small reactivity coefficient values is treated on an event-by-event basis.

15.1.6 Fission Product Inventories

The fission product inventories existing in the core and fuel rod gaps are described in Section 15.1.7 of the FSAR. The description of the models used for calculating fuel gap activities is included in Section 15.1.7.2 of the FSAR.

15.1.7 Residual Decay Heat

Residual heat in a subcritical core consists of:

(1) Fission product decay energy

(2) Decay of neutron capture products

(3) Residual fissions due to the effect of delayed neutrons.

These constituents are discussed separately in the following paragraphs.

15.1.7.1 Fission Product Decay

For short times ($<10^3$ seconds) after shutdown, data on yields of short-half-life isotopes is sparse. Very little experimental data is available for the gamma ray contributions and even less for the beta ray contribution. Several authors have compiled the available data into a conservative estimate of fission product decay energy for short times after shutdown, notably Shure(6), Dudziak(7), and Teage(8). Of these three selections, Shure's curve is the highest and is based on the data of Stehn and Clancy(9) and Obenshain and Foderaro(10). The fission product contribution to decay heat that has been assumed in the LOCA accident analyses is the curve of Shure increased by 20% for conservatism. This curve with the 20% factor included is shown on Figure 15.1-6. For the non-LOCA analyses the 1979 ANS decay heat curve is used(11). Figure 15.1-7 presents this curve as a function of time after shutdown.

15.1.7.2 Decay of U-238 Capture Products

Betas and gammas from the decay of U-239 (23.5-minute half-life) and Np-239 (2.35-day half-life) contribute significantly to the heat generation after shutdown. The cross sections for production of these isotopes and their decay schemes are relatively well known. For long irradiation times their contribution can be written as:

$$P_1/P_o = \frac{(E_{\gamma 1} + E_{\beta 1})}{200\ \text{MeV}} c(1+\alpha)\ e^{-\lambda_1 t}\ \text{watts/watt} \qquad (15.1\text{-}1)$$

$$P_2/P_o = \frac{(E_{\gamma 2} + E_{\beta 2})}{200\ \text{MeV}} c(1+\alpha) \left[\frac{\lambda_2}{\lambda_1 - \lambda_2}\left(e^{-\lambda_2 t} - e^{-\lambda_1 t}\right) + e^{-\lambda_2 t}\right]\ \text{watts/watt} \qquad (15.1\text{-}2)$$

where:

P_1/P_o is the energy from U-239 decay

P_2/P_o is the energy from Np-239 decay

t is the time after shutdown (seconds)

$c(1+\alpha)$ is the ratio of U-238 captures to total fissions = 0.6(1 + 0.2)

λ_1 = the decay constant of U-239 = 4.91×10^{-4} seconds^{-1}

λ_2 = the decay constant of Np-239 = 3.41×10^{-6} seconds^{-1}

$E_{\gamma 1}$ = total γ ray energy from U-239 decay = 0.06 MeV

$E_{\gamma 2}$ = total γ ray energy from Np-239 decay = 0.30 MeV

$E_{\beta 1}$ = total β ray energy from U-239 decay = 1/3[a] x 1.18 MeV

$E_{\beta 2}$ = total β ray energy from Np-239 decay = 1/3[a] x .43 MeV

This expression with a margin of 10% is shown on Figure 15.1-6 as it is used in the LOCA analysis. The 10% margin, compared to 20% for fission product decay, is justified by the availability of the basic data required for this analysis. The decay of other isotopes, produced by neutron reactions other than fission, is neglected. For the non-LOCA analysis, the decay of U-238 capture products is included as an integral part of the 1979 decay heat curve presented as Figure 15.1-7.

15.1.7.3 Residual Fissions

The time dependence of residual fission power after shutdown depends on core properties throughout a transient under consideration. Core average conditions are more conservative for the calculation of reactivity and power level than actual local conditions as they would exist in hot areas of the core. Thus, unless otherwise stated in the text, static power shapes have been assumed in the analysis and these are factored by the time behavior of core average fission power calculated by a point kinetics model calculation with six delayed neutron groups.

For the purpose of illustration, only one delayed neutron group calculation, with a constant shutdown reactivity of -4% Δk is shown on Figure 15.1-6.

(a) Two-thirds of the potential β-energy is assumed to escape by the accompanying neutrinos.

15.1.7.4 Distribution of Decay Heat Following Loss of Coolant Accident

During a loss-of-coolant accident (LOCA), the core is rapidly shut down by void formation or RCCA insertion, or both, and long-term shutdown is assured by the borated ECCS water. A large fraction of the heat generation to be considered comes from fission product decay gamma rays. This heat is not distributed in the same manner as steady state fission power. Local peaking effects that are important for the neutron dependent part of the heat generation do not apply to the gamma ray source contribution. The steady state factor of 97.4% that represents the fraction of heat generated within the cladding and pellet drops to 95% for the hot rod in a LOCA.

For example, consider the transient resulting from the postulated double-ended break of the largest RCS pipe; 1/2 second after the rupture, about 30% of the heat generated in the fuel rods is from gamma ray absorption. The gamma power shape is less peaked than the steady state fission power shape, reducing the energy deposited in the hot rod at the expense of adjacent colder rods. A conservative estimate of this effect is a reduction of 10% of the gamma ray contribution or 3% of the total. Since the water density is considerably reduced at this time, an average of 98% of the available heat is deposited in the fuel rods, the remaining 2% being absorbed by water, thimbles, sleeves, and grids. The net effect is a factor of 0.95, rather than 0.974, to be applied to the heat production in the hot rod.

15.1.8 Computer Codes Utilized

Summaries of some of the principal computer codes used in transient analyses are given below. Other codes, in particular, very specialized codes in which the modeling has been developed to simulate one given accident, such as the SATAN-VI code used in the analysis of the RCS pipe rupture (Section 15.4), and which consequently have a direct bearing on the analysis of the accident itself, are summarized in their respective accident analyses sections. The codes used in the analyses of each transient are listed in Table 15.1-4.

15.1.8.1 FACTRAN

FACTRAN calculates the transient temperature distribution in a cross section of a metal clad UO_2 fuel rod (see Figure 15.1-8) and the transient heat flux at the surface of the clad using as input the nuclear power and the time-dependent coolant parameters (pressure, flow, temperature and density). The code uses a fuel model that exhibits the following features simultaneously:

(1) A sufficiently large number of finite difference radial space increments to handle fast transients such as rod ejection accidents

(2) Material properties that are functions of temperature and a sophisticated fuel-to-clad gap heat transfer calculation

(3) The necessary calculations to handle post-DNB transients: film boiling heat transfer correlations, zircaloy-water reaction and partial melting of the materials.

The gap heat transfer coefficient is calculated according to an elastic pellet model. The thermal expansion of the pellet is calculated as the sum of the radial (one-dimensional) expansions of the rings. Each ring is assumed to expand freely. The clad diameter is calculated based on thermal expansion and internal and external pressures.

If the outside radius of the expanded pellet is smaller than the inside radius of the expanded clad, there is no fuel-clad contact and the gap conductance is calculated on the basis of the thermal conductivity of the gas contained in the gap. If the pellet outside radius so calculated is larger than the clad inside radius (negative gap), the pellet and the clad are pictured as exerting upon each other a pressure sufficient to reduce the gap to zero by elastic deformation of both. This contact pressure determines the heat transfer coefficient.

FACTRAN is further discussed in Reference 12.

15.1.8.2 LOFTRAN

The LOFTRAN program is used for studies of transient response of a PWR system to specified perturbations in process parameters. LOFTRAN simulates a multiloop system by modeling the reactor core and vessel, hot and cold leg piping, steam generator (tube and shell-sides), reactor coolant pumps and the pressurizer with up to four reactor coolant loops. The pressurizer heaters, spray, relief and safety valves are also considered in the program. Point model neutron kinetics and reactivity effects of the moderator, fuel, boron, and rods are included. The secondary side of the steam generator utilizes a homogeneous, saturated mixture for the thermal transients and a water level correlation for indication and control. The reactor protection system is simulated to include reactor trips on neutron flux, overpower and overtemperature reactor coolant ΔT, high and low pressure, low flow, and high pressurizer level. Control systems are also simulated including rod control, steam dump, feedwater control, and pressurizer pressure control. The safety injection system (SIS), including the accumulators, is also modeled.

LOFTRAN is a versatile program that is suited to both accident evaluation and control studies as well as parameter sizing. LOFTRAN also has the capability of calculating the transient value of DNB based on the input from the core limits illustrated on Figure 15.1-1. The core limits represent the minimum value of DNBR as calculated for a typical or thimble cell.

LOFTRAN is further discussed in Reference 13.

15.1.8.3 LEOPARD

The LEOPARD computer program determines fast and thermal spectra using only basic geometry and temperature data. The code optionally computes fuel depletion effects for a dimensionless reactor and recomputes the spectra before each discrete burnup step.

LEOPARD is further discussed in Reference 14.

15.1.8.4 TURTLE

TURTLE is a two-group, two-dimensional neutron diffusion code featuring direct treatment of the nonlinear effects of xenon, enthalpy, and Doppler feedback. Fuel depletion is allowed.

TURTLE was written for the study of azimuthal xenon oscillations, but the code is useful for general analysis. The input is simple, fuel management is handled directly, and a boron criticality search is allowed.

TURTLE is further described in Reference 15.

15.1.8.5 TWINKLE

The TWINKLE program is a multidimensional spatial neutron kinetics code which was patterned after steady state codes presently used for reactor core design. The code uses an implicit finite-difference method to solve the two-group transient neutron diffusion equations in one, two, and three dimensions. The code uses six delayed neutron groups and contains a detailed multiregion fuel-clad-coolant heat transfer model for calculating pointwise Doppler and moderator feedback effects. The code handles up to 2000 spatial points and performs its own steady state initialization. Aside from basic cross section data and thermal-hydraulic parameters, the code accepts as input basic driving functions such as inlet temperature, pressure, flow, boron concentration, control rod motion, and others. Various edits provide channelwise power, axial offset, enthalpy, volumetric surge, pointwise power, fuel temperatures, and so on.

The TWINKLE code is used to predict the kinetic behavior of a reactor for transients that cause a major perturbation in the spatial neutron flux distribution.

TWINKLE is further described in Reference 16.

15.1.8.6 THINC

The THINC code is described in Section 4.4.3 of the FSAR.

15.1.9 REFERENCES

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TABLE 15.1-1

NUCLEAR STEAM SUPPLY SYSTEM POWER RATING

Core thermal power (license level)	2775
Thermal power generated by the reactor coolant pumps	12
Nuclear steam supply system thermal power output	2787
Engineered safety features design rating (maximum calculated turbine rating)[a]	2912

(a) The unit will not be operated at this rating because it exceeds the license rating.

TABLE 15.1-2

TRIP POINTS AND TIME DELAYS TO TRIP ASSUMED IN ACCIDENT ANALYSES

Trip Function	Limiting Trip Point Assumed In Analyses	Time Delay, sec
Power range high neutron flux, high setting	118%	0.5
Power range high neutron flux, low setting	35%	0.5
Overtemperature ΔT	Variable, see Figure 15.1-1	8.5[a]
Overpower ΔT	Variable, see Figure 15.1-1	8.5[a]
High pressurizer pressure	2440 psig	2
Low pressurizer pressure	1760 psig	2
Low reactor coolant flow (from loop flow detectors)	87% loop flow	1
Undervoltage trip	(b)	1.5

TABLE 15.1-2

Trip Function	Limiting Trip Point Assumed In Analyses	Time Delay, sec
Turbine trip	Not applicable	2
Low-low steam generator level	0% of narrow range level span	2
High-high steam generator level trip of the feedwater pumps and turbine; closure of feedwater system valves*	96% of narrow range level span	2 13* (for feedwater isolation)

(a) Total time delay (including RTD and thermowell time response, trip circuit and channel electronics delay) from the time the temperature difference in the coolant loops exceeds the trip setpoint until the rods are free to fall.

(b) A specific undervoltage setpoint was not assumed in the safety analysis.

TABLE 15.1-4

SUMMARY OF INITIAL CONDITIONS AND COMPUTER CODES USED

Faults	Computer Codes Utilized	Assumed Reactivity Coefficients: Moderator Temp.[a], pcm/°F[d]	Assumed Reactivity Coefficients: Moderator Density[a], Δk/gm/cc	Doppler[b]	Initial NSSS Thermal Power Output Assumed[c], MWt
CONDITION II					
Uncontrolled RCCA bank withdrawal from a subcritical condition	TWINKLE, FACTRAN, THINC	+7	-	Consistent with lower limit on Fig. 15.1-5	0
Uncontrolled RCCA bank withdrawal at power	LOFTRAN	+7	0.50	Lower and Upper	2790
RCCA misoperation	THINC, TURTLE LOFTRAN	-	-	-	2787
Uncontrolled boron dilution					0 and 2787
Partial loss of forced reactor coolant flow	LOFTRAN FACTRAN, THINC	0	-	Upper	2787
Startup of an inactive reactor coolant loop	LOFTRAN, FACTRAN, THINC	-	0.50	Lower	1672
Loss of external electrical load and/or turbine trip	LOFTRAN	+7	0.50	Lower and Upper	2787
Loss of normal feedwater	LOFTRAN	+7	-	Upper	2790
Loss of offsite power to the plant auxiliaries (plant blackout)	LOFTRAN	+7	-	Upper	2790

TABLE 15.1-4

Faults	Computer Codes Utilized	Assumed Reactivity Coefficients: Moderator Temp.[a], pcm/°F[d]	Moderator Density[a], Δk/gm/cc	Doppler[b]	Initial NSSS Thermal Power Output Assumed[c], MWt
CONDITION II (Cont'd)					
Excessive heat removal due to feedwater system malfunctions	LOFTRAN	-	0.50	Lower	0 and 2787
Excessive load increase	LOFTRAN	-	0 and 0.50	Lower and Upper	2787
Accidental depressurization of the reactor coolant system	LOFTRAN	+7	-	Lower	2787
Accidental depressurization of the main steam system	LOFTRAN	-	Function of the moderator density. See Sec. 15.2.13 (Figure 15.2.13-1)	See Figure 15.4.2-1	0 (Subcritical)
Spurious operation of the SIS at power	LOFTRAN	+7	0.50	Lower and Upper	2787
CONDITION III					
Loss of reactor coolant from small ruptured pipes or from cracks in large pipe which actuate emergency core cooling	NOTRUMP SBLOCTA	-	-	-	2775[e]

TABLE 15.1-

Faults	Computer Codes Utilized	Assumed Reactivity Coefficients: Moderator Temp.[a] pcm/°F[d],	Assumed Reactivity Coefficients: Moderator Density[a], Δk/gm/cc	Doppler [b]	Initial NSSS Thermal Power Output Assumed[c], MWt
CONDITION III (Cont'd)					
Inadvertent loading of a fuel assembly into an improper position	LEOPARD, TURTLE	-	-	-	2775[e]
Complete loss of force reactor coolant flow	LOFTRAN, FACTRAN, THINC	0	-	Upper	2787
Single RCCA withdrawal at full power	TURTLE, THINC, LEOPARD	-	-	-	2787
CONDITION IV					
Major rupture of pipes containing reactor coolant up to and including double-ended rupture of the largest pipe in the reactor coolant system (loss-of-coolant accident)	SATAN-VI COCO BASH WREFLOOD LOCBART	Function of moderator density. See Sec. 15.4.1	-	Function of fuel temp. See Sec. 15.4.1	2775[e]
Major secondary system pipe rupture up to and including double-ended rupture (rupture of a steam pipe)	LOFTRAN	-	Function of the Moderator Density see Section 15.2.13 (Figure 15.2.13-1)	See Figure 15.4.2-1	0 (Subcritical)

TABLE 15.1-4

Faults	Computer Codes Utilized	Assumed Reactivity Coefficients: Moderator Temp.[a], pcm/°F[d]	Moderator Density[a], Δk/gm/cc	Doppler[b]	Initial NSSS Thermal Power Output Assumed[c], MWt
CONDITION IV (Cont'd)					
Major secondary system pipe rupture up to and including double-ended rupture (rupture of a feedline)	LOFTRAN	-	0.50	Upper	2912
Single reactor coolant pump locked rotor	LOFTRAN FACTRAN, THINC	0	-	Upper	2787
Rupture of a control rod mechanism housing (RCCA ejection)	TWINKLE, FACTRAN, LEOPARD	+7.1 BOL -23. EOL	-	Consistent with lower limit on Fig. 15.1-5	0 and 2775[e]

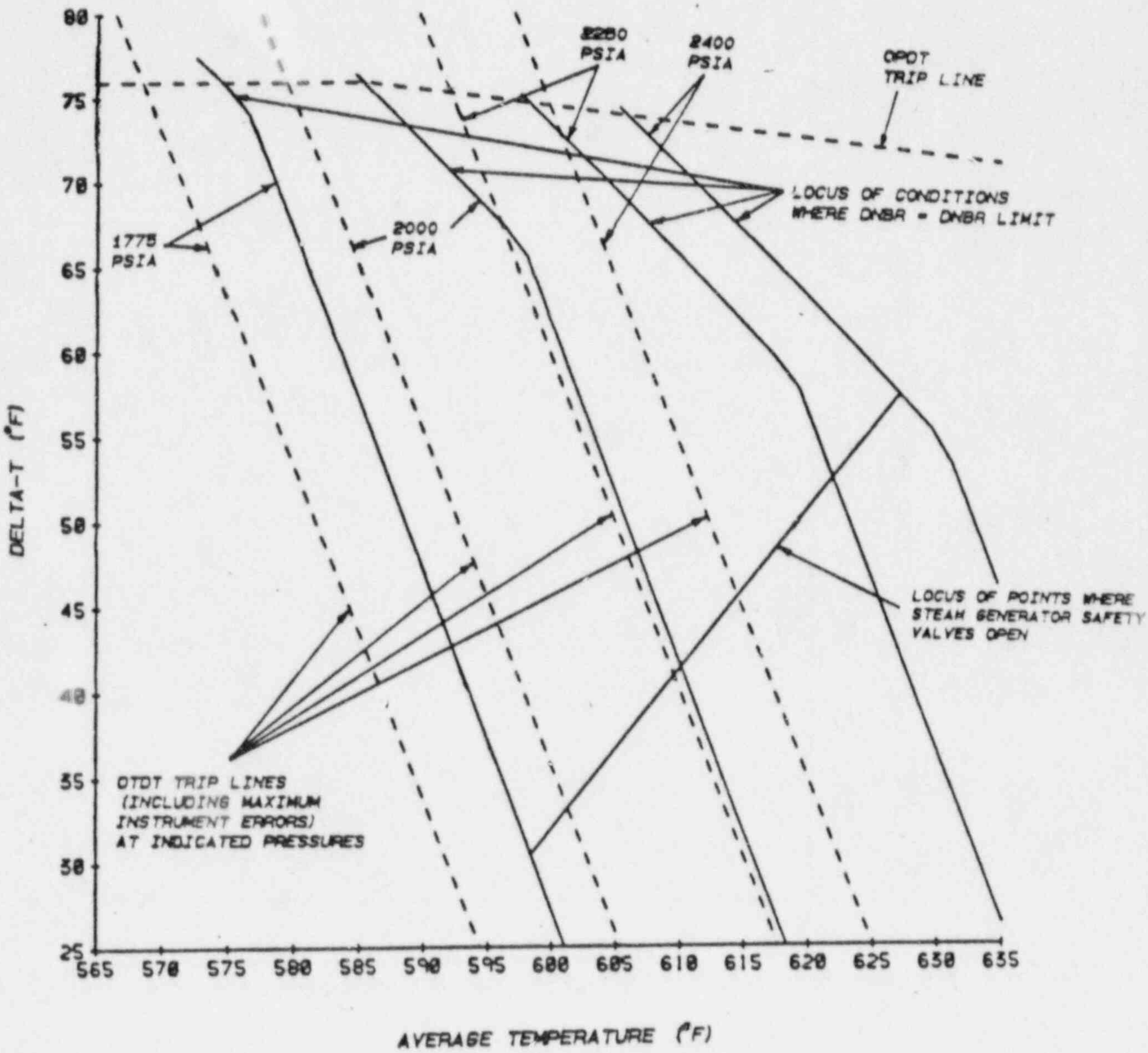
(a) Only one is used in analysis, i.e., either moderator temperature or moderator density coefficient.

(b) Reference Figure 15.1-5.

(c) Appropriate calorimetric error considered where applicable.

(d) Pcm means percent mille. See footnote Table 4.3-1.

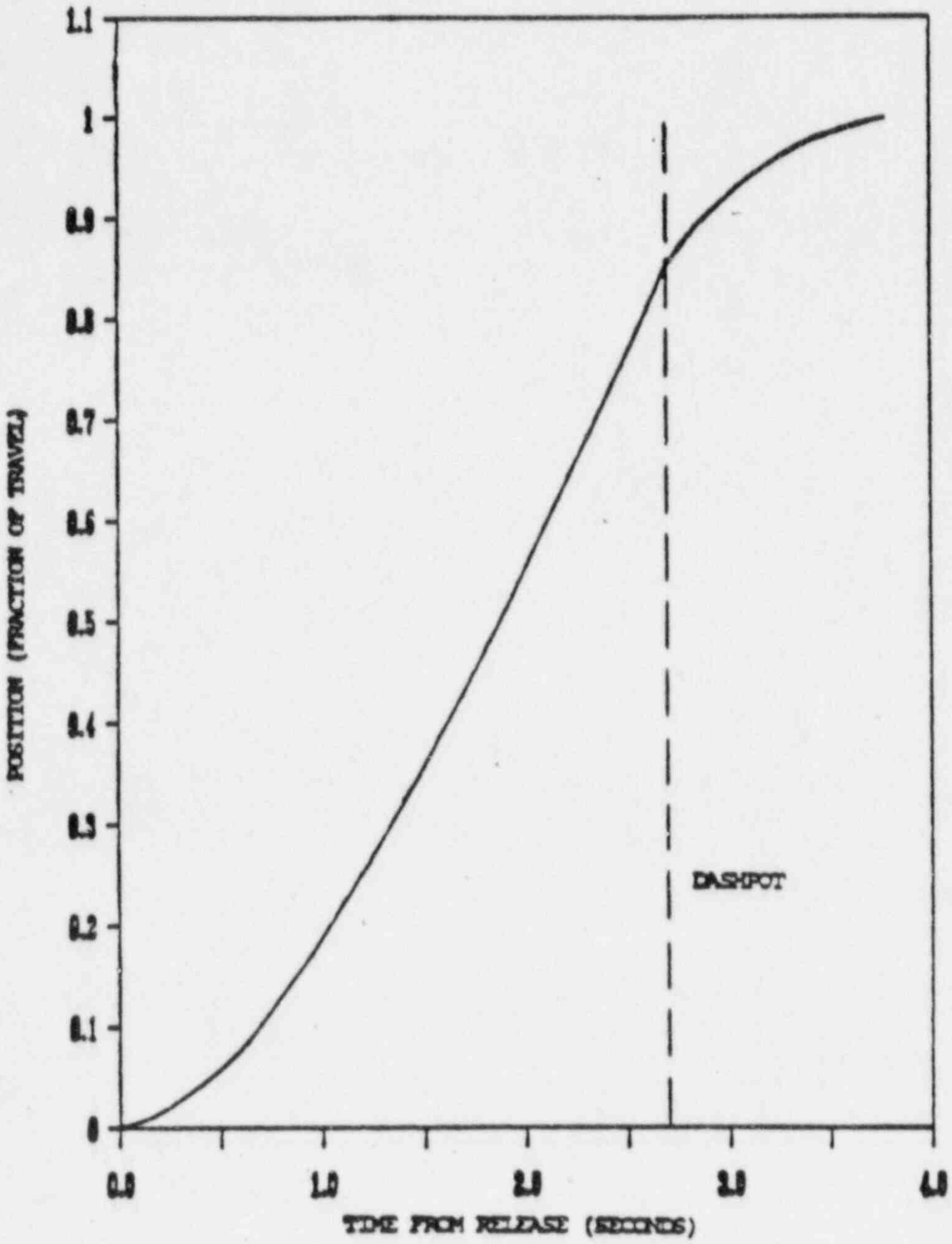
(e) Core power.



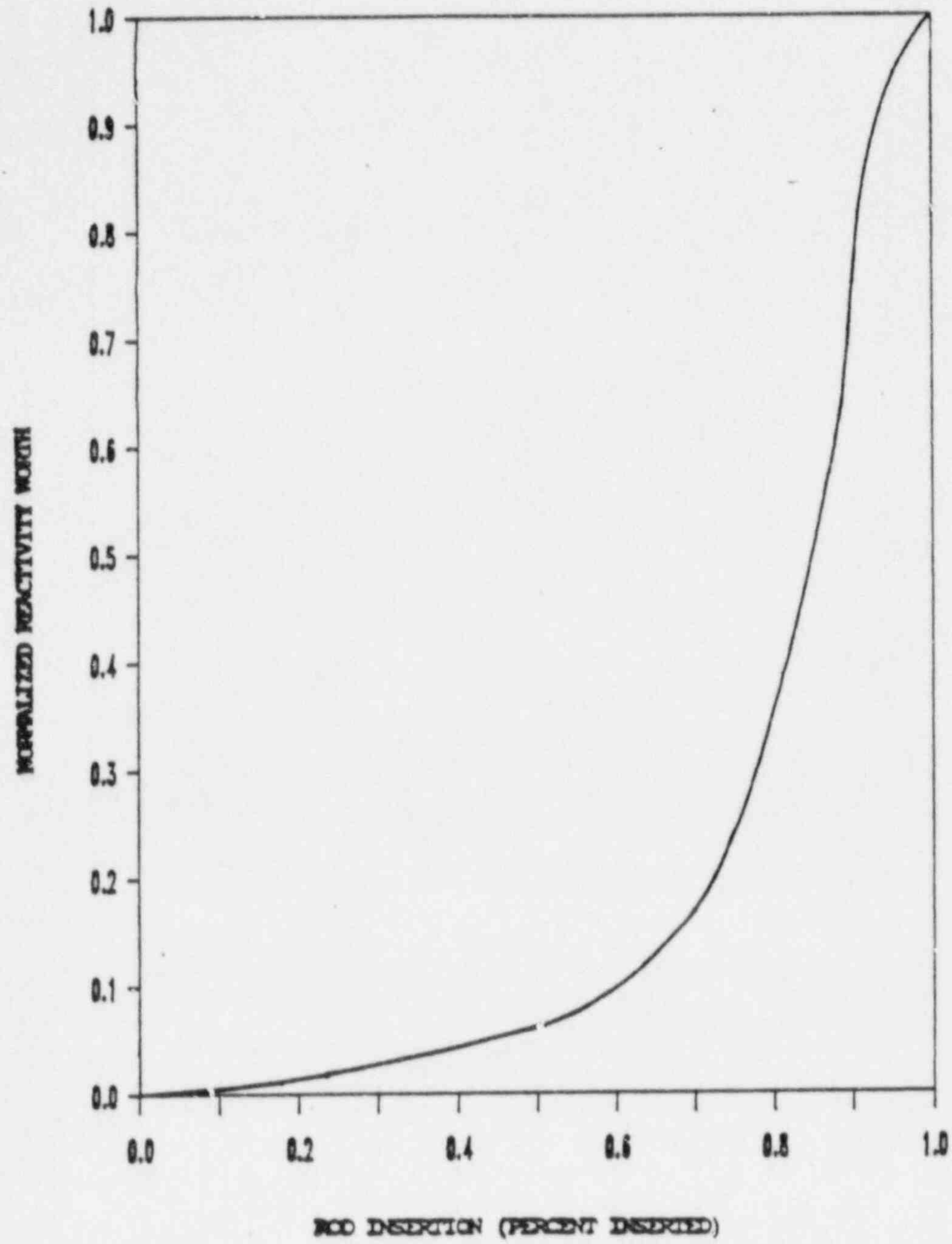
V. C. Summer

Figure 15.1-1

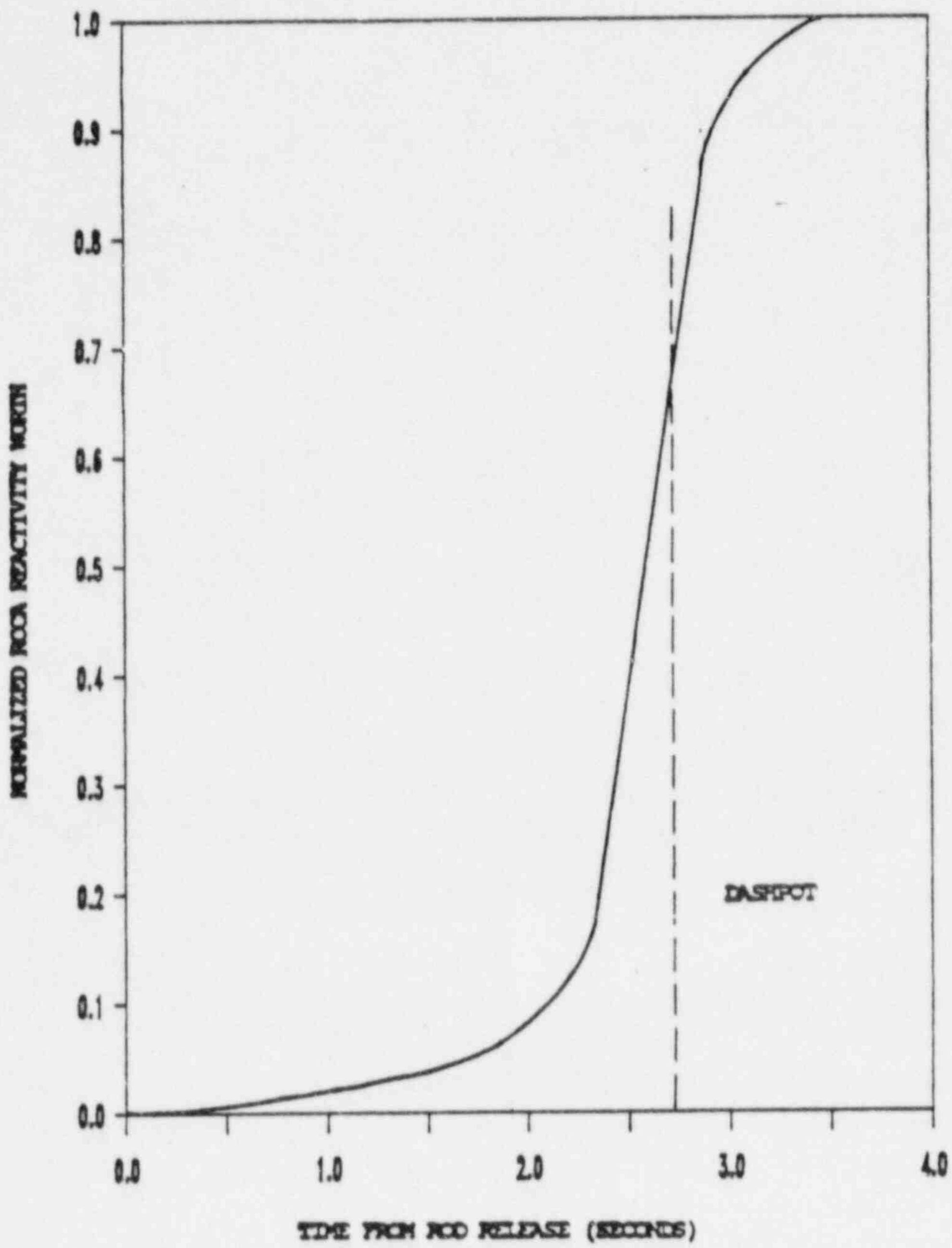
Overtemperature and Overpower Delta-T Protection



V. C. Summer
Figure 15.1-2
Rod Position vs. Time On Reactor Trip



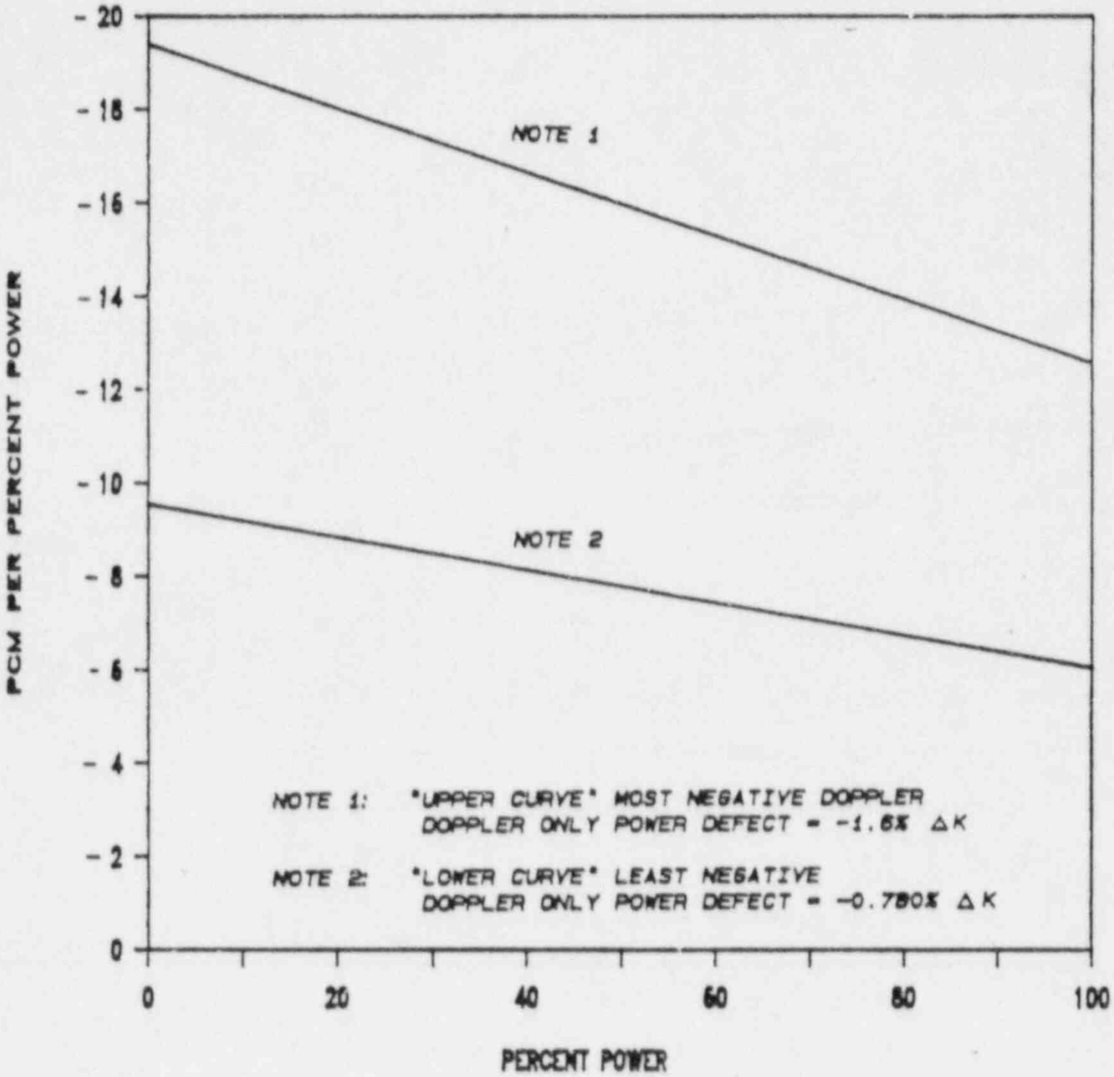
V. C. Summer
Figure 15.1-3
Normalized RCCA Reactivity Worth vs. Percent Insertion



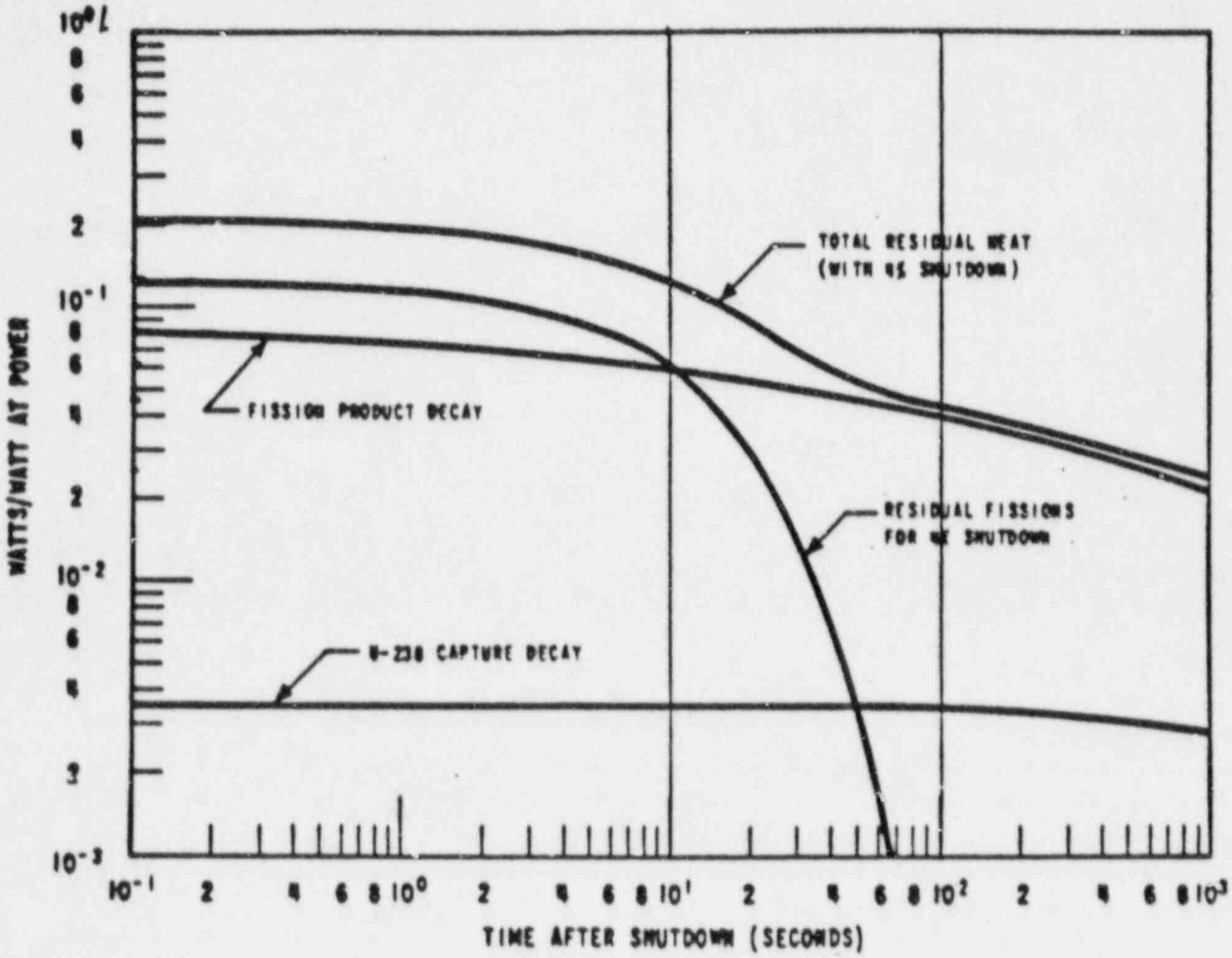
V. C. Summer

Figure 15.1-4

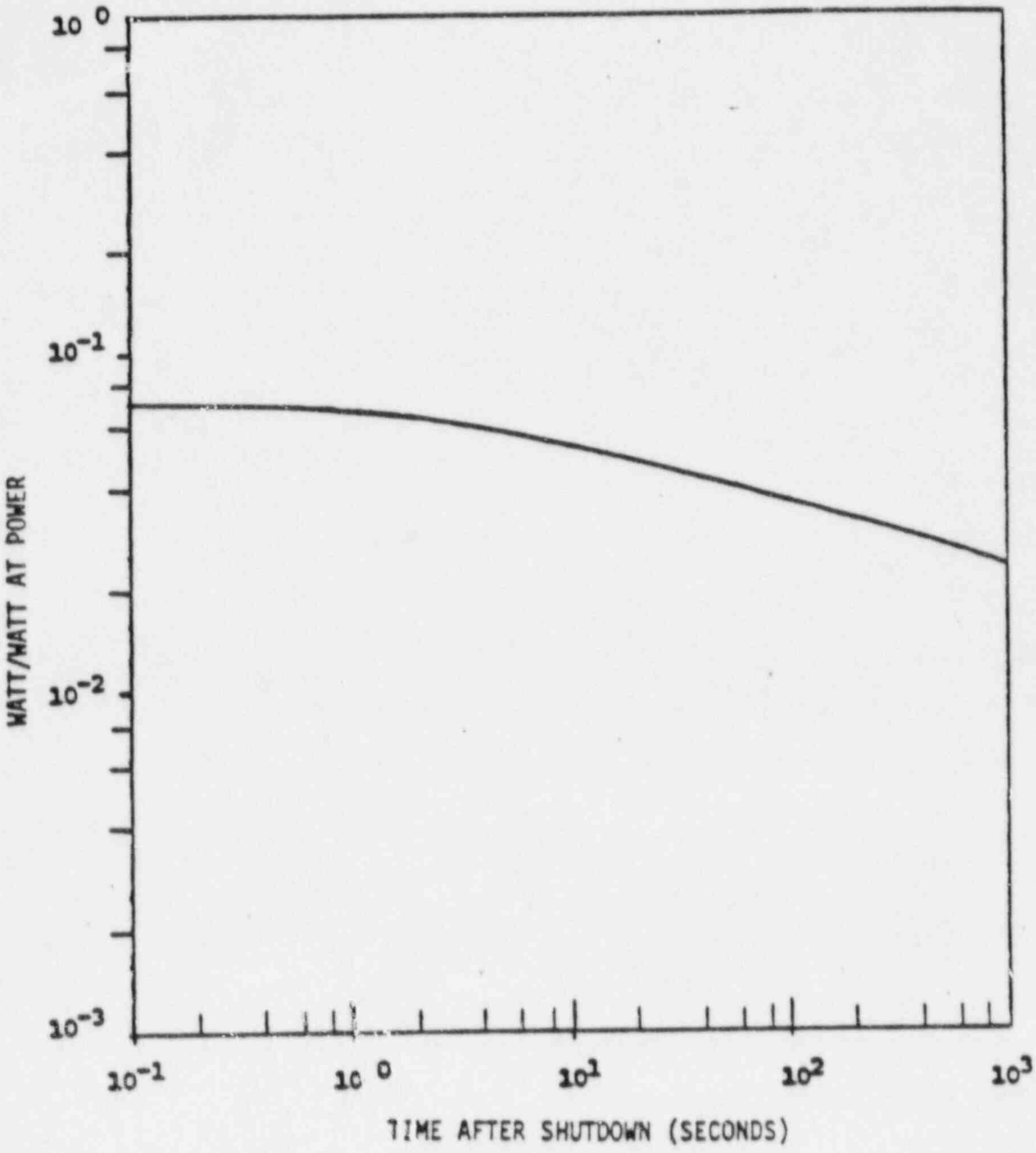
Normalized RCCA Bank Worth vs. Time After Trip



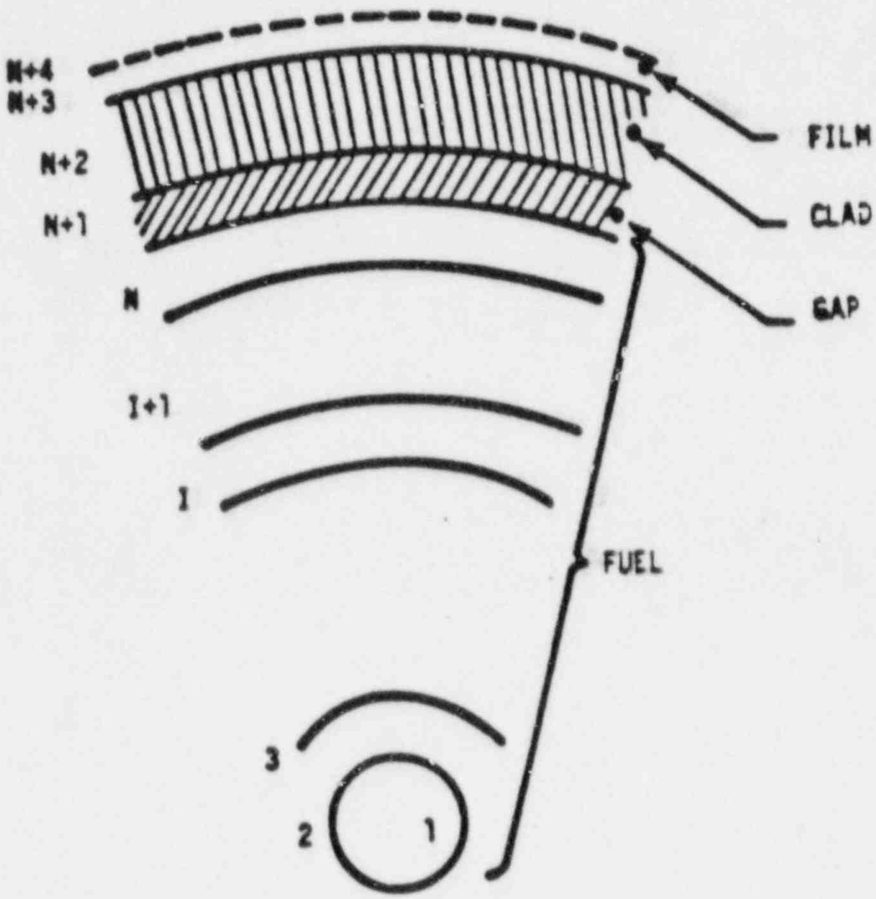
V. C. Summer
Figure 15.1-5
Doppler Power Coefficient Used In Accident Analyses



V. C. Summer
Figure 15.1-6
Residual Decay Heat



V. C. Summer
Figure 15.1-7
1979 ANS Decay Heat



V. C. Summer
Figure 15.1-8
Fuel Rod Cross Section

15.2 CONDITION II - FAULTS OF MODERATE FREQUENCY

These faults result at worst in reactor shutdown with the plant being capable of returning to operation. By definition, these faults (or events) do not propagate to cause a more serious fault, i.e., a Condition III or IV fault. In addition, Condition II events are not expected to result in fuel rod failures or reactor coolant system (RCS) overpressurization. For the purposes of this report the following faults have been grouped into these categories:

(1) Uncontrolled rod cluster control assembly (RCCA) bank withdrawal from a subcritical condition

(2) Uncontrolled RCCA bank withdrawal at power

(3) RCCA misoperation

(4) Uncontrolled boron dilution

(5) Partial loss of forced reactor coolant flow

(6) Startup of an inactive reactor coolant loop

(7) Loss of external electrical load and/or turbine trip

(8) Loss of normal feedwater

(9) Loss of offsite power to the station auxiliaries (station blackout)

(10) Excessive heat removal due to feedwater system malfunctions

(11) Excessive load increase

(12) Accidental RCS depressurization

(13) Accidental main steam system depressurization

(14) Spurious operation of safety injection system (SIS) at power.

Each of these faults of moderate frequency are analyzed in this section. In general, each analysis includes an identification of causes and description of the accident, an analysis of effects and consequences, a presentation of results, and relevant conclusions.

An evaluation of the reliability of the reactor protection system actuation following initiation of Condition II events has been completed and is presented in Reference 1 for the relay protection logic. Standard reliability engineering techniques were used to assess the likelihood of the trip failure due to random component failures. Common-mode failures were also qualitatively investigated. It was concluded from the evaluation that the likelihood of no trip following initiation of Condition II events is extremely small (2×10^{-7} derived for random component failures). The reliability of the solid-state protection system has also been evaluated using the same methods. The calculated reliability is of the same order of magnitude as that obtained for the relay protection logic.

Hence, because of the high reliability of the protection system, no special provision is included in the design to cope with the consequences of Condition II events without trip.

The time sequence of events for the Condition II faults are shown in Table 15.2-1.

15.2.1 Uncontrolled Rod Cluster Control Assembly Bank Withdrawal from a Subcritical Condition

15.2.1.1 Identification of Causes and Accident Description

An RCCA withdrawal accident is defined as an uncontrolled increase in reactivity in the reactor core caused by withdrawal of RCCAs resulting in a power excursion. Such a transient could be caused by a malfunction of the reactor control or control rod drive systems. This could occur with the reactor at either subcritical, hot zero power, or at power. The at-power case is discussed in Section 15.2.2.

Although the reactor is normally brought to power from a subcritical condition by means of RCCA withdrawal, initial startup procedures with a clean core call for boron dilution. The maximum rate of reactivity increase in the case of boron dilution is less than that assumed in this analysis (see Section 15.2.4).

The RCCA drive mechanisms are wired into preselected bank configurations that are not altered during core reactor life. These circuits prevent the assemblies from being withdrawn in other than their respective banks. Power supplied to the banks is controlled so that no more than two banks can be withdrawn at the same time. The RCCA drive mechanisms are of the magnetic latch type and coil actuation is sequenced to provide variable speed travel. The maximum reactivity insertion rate analyzed in the detailed plant analysis is that occurring with the simultaneous withdrawal of the two control banks having the maximum combined worth at maximum speed.

The neutron flux response to a continuous reactivity insertion is characterized by a very fast rise terminated by the reactivity feedback effect of the negative Doppler coefficient. This self-limitation of the power burst is of primary importance since it limits the power to a tolerable level during the delay time for protection action. Should a continuous RCCA withdrawal accident occur, the transient will be terminated by the following automatic features of the reactor protection system:

15.2.1.1.1 Source Range High Neutron Flux Reactor Trip

The source range high neutron flux reactor trip is actuated when either of two independent source range channels indicates a neutron flux level above a preselected manually adjustable setpoint. This trip function may be manually bypassed when either intermediate range flux channel indicates a flux level above a specified level. It is automatically reinstated when both intermediate range channels indicate a flux level below a specified level.

15.2.1.1.2 Intermediate Range High Neutron Flux Reactor Trip

The intermediate range high neutron flux reactor trip is actuated when either of two independent intermediate range channels indicates a flux level above a preselected manually adjustable setpoint. This trip function may be manually bypassed when two of the four power range channels give readings above approximately 10% of full power and is automatically reinstated when three of the four channels indicate a power below this value.

15.2.1.1.3 Power Range High Neutron Flux Reactor Trip (Low Setting)

The power range high neutron flux trip (low setting) is actuated when two-out-of-four power range channels indicate a power level above approximately 25% of full power. This trip function may be manually bypassed when two of the four power range channels indicate a power level above approximately 10% of full power and is automatically reinstated when three of the four channels indicate a power level below this value.

15.2.1.1.4 Power Range High Neutron Flux Reactor Trip (High Setting)

The power range high neutron flux reactor trip (high setting) is actuated when two-out-of-four power range channels indicate a power level above a preset setpoint. This trip function is always active. In addition, control rod stops on high intermediate range flux level (one-of-two) and high power range flux level (one-out-of-four) serve to discontinue rod withdrawal and prevent the need to actuate the intermediate range flux level trip and the power range flux level trip, respectively.

15.2.1.1.5 High Neutron Flux Rate Trip

The high neutron flux rate trip is actuated when the rate of change in power exceeds the positive or negative setpoint in two-out-of-four power range channels. This function is always active.

15.2.1.2 Analysis of Effects and Consequences

The analysis of the uncontrolled rod withdrawal from subcritical accident is performed in three stages: first a core average nuclear power transient calculation is performed, followed by an average core heat transfer calculation, and finally a DNBR calculation. The core average nuclear power transient calculation is performed using a spatial neutron kinetics code, TWINKLE[2], to determine the average power generation with time including the various total core feedback effects, i.e., Doppler and moderator reactivity. The average heat flux and temperature transients are determined by performing a fuel rod transient heat transfer calculation in FACTRAN[3]. The average heat flux is next used in THINC[7] for the transient DNBR calculation.

The core axial power distribution is severely peaked to the bottom of the core for the limiting transient. The W-3 DNB correlation is used to evaluate DNBR in the span between the lower non-mixing vane grid. The WRB-1 correlation (LOPAR fuel) and the WRB-2 correlation (VANTAGE 5 fuel) remain applicable for the rest of the fuel assembly.

In order to give conservative results for a startup accident, the following assumptions are made concerning the initial reactor conditions:

(1) Since the magnitude of the power peak reached during the initial part of the transient for any given rate of reactivity insertion is strongly dependent on the Doppler coefficient, conservative values (low absolute magnitude) as a function of power are used. See Section 15.1.5 and Table 15.1-4.

(2) Contribution of the moderator reactivity coefficient is negligible during the initial part of the transient because the heat transfer time between the fuel and the moderator is much longer than the

neutron flux response time. However, after the initial neutron flux peak, the succeeding rate of power increase is affected by the moderator reactivity coefficient. A conservative value, given in Table 15.1-4, is used in the analysis to yield the maximum peak heat flux.

(3) The reactor is assumed to be at hot zero power. This assumption is more conservative than that of a lower initial system temperature. The higher initial system temperature yields a larger fuel-water heat transfer coefficient, larger specific heats, and a less negative (smaller absolute magnitude) Doppler coefficient, all of which tend to reduce the Doppler feedback effect thereby increasing the neutron flux peak. The initial effective multiplication factor is assumed to be 1 since this results in maximum neutron flux peaking.

(4) Reactor trip is assumed to be initiated by power range high neutron flux (low setting). The most adverse combination of instrument and setpoint errors, as well as delays for trip signal actuation and RCCA release, is taken into account. A 10% increase is assumed for the power range flux trip setpoint, raising it from the nominal value of 25 to 35%. Previous results, however, show that the rise in neutron flux is so rapid that the effect of errors in the trip setpoint on the actual time at which the rods are released is negligible. In addition, the reactor trip insertion characteristic is based on the assumption that the highest worth RCCA is stuck in its fully withdrawn position. See Section 15.1.4 for RCCA insertion characteristics.

(5) The maximum positive reactivity insertion rate assumed is greater than that for the simultaneous withdrawal of the combination of the two control banks having the greatest combined worth at maximum speed (45 inches/minute). Control rod drive mechanism design is discussed in Section 4.2.3 of the FSAR.

(6) The initial power level was assumed to be below the power level expected for any shutdown condition. The combination of highest reactivity insertion rate and lowest initial power produces the highest peak heat flux.

(7) Two reactor coolant pumps are assumed to be operating.

15.2.1.3 Results

The calculated sequence of events for this accident is shown on Table 15.2-1.

Figures 15.2.1-1 and 15.2.1-2 show the transient behavior for the indicated reactivity insertion rate with the accident terminated by reactor trip at 35% nominal power. This insertion rate is greater than that for the two highest worth control banks, both assumed to be in their highest incremental worth region.

Figure 15.2.1-1 shows the nuclear power transient. The nuclear power overshoots the full power nominal value but this occurs for only a very short time period. Hence, the energy release and the fuel temperature increase are relatively small. The thermal flux response, of interest for departure from nucleate boiling (DNB) considerations, is shown on Figure 15.2.1-1. The beneficial effect on the inherent thermal lag in the fuel is evidenced by a peak heat flux less than the full power nominal value.

Figure 15.2.1-2 shows the response of the hot spot fuel average and clad temperatures. The hot spot fuel average temperature increases to a value lower than the nominal full power value.

15.2.1.4 Conclusions

In the event of an RCCA withdrawal accident from the subcritical condition, the core and the RCS are not adversely affected since the combination of thermal power and the coolant temperature result in a departure from nucleate boiling ratio (DNBR) greater than the design limit value. Thus, no fuel or clad damage is predicted as a result of DNB.

15.2.2 Uncontrolled Rod Cluster Control Assembly Bank Withdrawal at Power

15.2.2.1 Identification of Causes and Accident Description

Uncontrolled RCCA bank withdrawal at power results in an increase in the core heat flux. Since the heat extraction from the steam generator lags behind the core power generation until the steam generator pressure reaches the relief or safety valve setpoint, there is a net increase in the reactor coolant temperature. Unless terminated by manual or automatic action, the power mismatch and resultant coolant temperature rise would eventually result in DNB. Therefore, in order to avert damage to the cladding, the reactor protection system is designed to terminate any such transient before the DNBR falls below the safety analysis limit values.

The automatic features of the reactor protection system that prevent core damage following the postulated accident include the following:

(1) The power range neutron flux instrumentation actuates a reactor trip if two-out-of-four channels exceed a high flux setpoint;

(2) The reactor trip is actuated if any two-out-of-three ΔT channels exceed an overtemperature ΔT setpoint. This setpoint is automatically varied with axial power imbalance, coolant temperature, and pressure to protect against DNB;

(3) The reactor trip is actuated if any two-out-of-three ΔT channels exceed an overpower ΔT setpoint to ensure that the allowable heat generation rate (kw/ft) is not exceeded;

(4) A high pressurizer pressure reactor trip actuated from any two-out-of-three pressure channels that are set at a fixed point. This set pressure is less than the set pressure for the pressurizer safety valves;

(5) A high pressurizer water level reactor trip actuated from any two-out-of-three level channels that are set at a fixed point.

In addition to the above listed reactor trips, there are the following RCCA withdrawal blocks:

(1) High neutron flux (one-out-of-four);

(2) Overpower ΔT (two-out-of-three);

(3) Overtemperature ΔT (two-out-of-three).

The manner in which the combination of overpower and overtemperature ΔT trips provide protection over the full range of RCS conditions is described in Chapter 7. Figure 15.1-1 presents allowable reactor coolant loop average temperature and ΔT for the design power distribution and flow as a function of primary coolant pressure. The boundaries of operation defined by the overpower ΔT trip and the overtemperature ΔT are represented as protection lines on this diagram. The protection lines are drawn to include all adverse instrumentation and setpoint errors so that under nominal conditions a trip would occur well within the area bounded by these lines. The utility of this diagram is in the fact that the limit imposed by a given DNBR can be represented as a line. The DNB lines represent the locus of conditions for which the DNBR equals the safety analysis limit value. All points below and to the left of a DNB line for a given pressure have a DNBR greater than the limit. The diagram shows that DNB is prevented for all cases if the area enclosed with the maximum protection lines is not traversed by the applicable DNBR line at any point.

The area of permissible operation (power, pressure, and temperature) is bounded by the combination of reactor trips: high neutron flux (fixed setpoint); high-pressure (fixed setpoint); low-pressure (fixed setpoint); overpower and overtemperature ΔT (variable setpoints).

15.2.2.2 Analysis of Effects and Consequences

The uncontrolled RCCA bank withdrawal at power transient is analyzed by the LOFTRAN code[4]. This code simulates the neutron kinetics, RCS, pressurizer, pressurizer relief and safety valves, pressurizer spray, steam generator, and steam generator safety valves. The code computes pertinent plant variables including temperatures, pressures, and power level. The core limits as illustrated on Figure 15.1-1 are used as input to LOFTRAN to determine the minimum DNBR during the transient.

This accident is analyzed with the Improved Thermal Design Procedure as described in Reference 5. In order to obtai onservative results, the following assumptions are made:

(1) Initial conditions of nominal core power and reactor coolant average temperatures and nominal reactor coolant pressure are assumed. Uncertainties in initial conditions are included in the limit DNBR as described in Reference 5;

(2) Reactivity Coefficients - two cases are analyzed:

(a) Minimum reactivity feedback. A positive moderator coefficient of reactivity of +7 pcm/°F is assumed. A variable Doppler power coefficient with core power is used in the analysis. A conservatively small (in absolute magnitude) value is assumed;

(b) Maximum reactivity feedback. A conservatively large positive moderator density coefficient and a large (in absolute magnitude) negative Doppler power coefficie t are assumed;

(3) The reactor trip on high neutron flux is assumed to be actuated at a conservative value of 118% of nominal full power. The ΔT trips include all adverse instrumentation and setpoint errors, while the delays for the trip signal actuation are assumed at their maximum values;

(4) The RCCA trip insertion characteristic is based on the assumption that the highest worth assembly is stuck in its fully withdrawn position;

(5) The maximum positive reactivity insertion rate is greater than that which would be obtained from the simultaneous withdrawal of the two control rod banks having the maximum combined worth at maximum speed.

The effect of RCCA movement on the axial core power distribution is accounted for by causing a decrease in overtemperature and overpower ΔT trip setpoints proportional to a decrease in margin to DNB.

15.2.2.3 Results

Figures 15.2.2-1 and 15.2.2-2 show the response of nuclear power, pressure, average coolant temperature, and DNBR to a rapid RCCA withdrawal starting from full power. Reactor trip on high neutron flux occurs shortly after the start of the accident. Since this is rapid with respect to the thermal time constants of the plant, small changes in T_{avg} and pressure result and a large margin to DNB is maintained.

The response of nuclear power, pressure, average coolant temperature, and DNBR for a slow control rod assembly withdrawal from full power is shown on Figures 15.2.2-3 and 15.2.2-4. Reactor trip on overtemperature ΔT occurs after a longer period and the rise in temperature and pressure is consequently larger than for rapid RCCA withdrawal. Again, the minimum DNBR is never less than the safety analysis limit values.

Figure 15.2.2-5 shows the minimum DNBR as a function of reactivity insertion rate from initial full power operation for the minimum and for the maximum reactivity feedbacks. It can be seen that two reactor trip channels provide protection over the whole range of reactivity insertion rates. These are the high neutron flux and overtemperature ΔT trip channels. The minimum DNBR is never less than the safety analysis limit values.

Figures 15.2.2-6 and 15.2.2-7 show the minimum DNBR as a function of reactivity insertion rate for RCCA withdrawal incidents starting at 60% and 10% power, respectively. The results are similar to the 100% power case, except that as the initial power is decreased, the range over which the overtemperature ΔT trip is effective is increased. In neither case does the DNBR fall below the safety analysis limit values.

The shape of the curves of minimum DNB ratio versus reactivity insertion rate in the reference figures is due both to reactor core and coolant system transient response and to protection system action in initiating a reactor trip.

Referring to Figure 15.2.2-7, for example, it is noted that:

1. For reactivity insertion rates ~ above 20 pcm/sec reactor trip is initiated by the high neutron flux trip for the minimum reactivity feedback cases. The neutron flux level in the core rises rapidly for these insertion rates while core heat flux and coolant system temperature lag behind due to the thermal capacity of the fuel and coolant system fluid. Thus, the reactor is tripped prior to a significant increase in heat flux or water temperature with resultant high minimum DNB ratios during the transient. As the reactivity insertion rate decreases, core heat flux and coolant temperatures can remain more nearly in equilibrium with the neutron flux. Minimum DNBR during the transient thus decreases with decreasing insertion rate.

2. The overtemperature ΔT reactor trip circuit initiates a reactor trip when measured coolant loop ΔT exceeds a setpoint based on measured Reactor Coolant System average temperature and pressure. It is important to note that the average temperature contribution to the circuit is lead-lag compensated in order to decrease the effect of the thermal capacity of the Reactor Coolant System in response to power increases.

3. For reactivity insertion rate below ~ 20 pcm/sec the overtemperature ΔT trip terminates the transient.

For reactivity insertion rates between ~ 20 pcm/sec and ~ 5 pcm/sec the effectiveness of the overtemperature ΔT trip increases (in terms of increased minimum DNBR) due to the fact that with lower insertion rates the power increase rate is slower, the rate of rise of average coolant temperature is slower and the system lags and delays become less significant.

4. For reactivity insertion rates less than ~ 5 pcm/sec, the rise in the reactor coolant temperature is sufficiently high so that the steam generator safety valve setpoint is reached prior to trip. Opening of these valves, which act as an additional heat load on the Reactor Coolant System, sharply decreases the rate of increase of Reactor Coolant System average temperature. This decrease in rate of increase of the average coolant system temperature during the transient is accentuated by the lead-lag compensation causing the overtemperature ΔT trip setpoint to be reached later with a resulting lower minimum DNBR.

For transients initiated from higher power levels (for example, see Figure 15.2.2-5) the effect described in item 4 above, which results in the sharp peak in minimum DNBR at approximately 5 pcm/sec, does not occur since the steam generator safety valves are not actuated prior to trip.

Figures 15.2.2-5, 15.2.2-6, and 15.2.2-7 illustrate minimum DNBRs calculated for minimum and maximum reactivity feedback.

Since the RCCA withdrawal at power incident is an overpower transient, the fuel temperatures rise during the transient until after reactor trip occurs. For high reactivity insertion rates, the overpower transient is fast with respect to the fuel rod thermal time constant, and the core heat flux lags behind the neutron flux response. Due to this lag, the peak core heat flux does not exceed 118 percent of its nominal value (i.e., the high neutron flux trip setpoint assumed in the analysis). Taking into account the effect of the RCCA withdrawal on the axial core power distribution, the peak fuel centerline temperature will still remain below the fuel melting temperature.

For slow reactivity insertion rates, the core heat flux remains more nearly in equilibrium with the neutron flux. The overpower transient is terminated by the overtemperature ΔT reactor trip before a DNB condition is reached. The peak heat flux again is maintained below 118 percent of its nominal value. Taking into account the effect of the RCCA withdrawal on the axial core power distribution, the peak fuel centerline temperature will remain below the fuel melting temperature.

Since DNB does not occur at any time during the RCCA withdrawal at power transient, the ability of the primary coolant to remove heat from the fuel rod is not reduced. Thus, the fuel cladding temperature does not rise significantly above its initial value during the transient.

The calculated sequence of events for this accident is shown on Table 15.2-1. With the reactor tripped, the plant eventually returns to a stable condition. The plant may subsequently be cooled down further by following normal plant shutdown procedures.

15.2.2.4 Conclusions

The high neutron flux and overtemperature ΔT trip channels provide adequate protection over the entire range of possible reactivity insertion rates; i.e., the minimum value of DNBR is always larger than the safety analysis limit values.

15.2.3 Rod Cluster Control Assembly Misoperation

This section discusses RCCA misoperation that can result either from system malfunction or operator error.

15.2.3.1 Identification of Causes and Accident Description

RCCA misalignment accidents include:

(1) One or more dropped RCCAs within the same group;

(2) A dropped RCCA bank;

(3) Statically misaligned RCCA.

Each RCCA has a position indicator channel that displays the position of the assembly. The displays of assembly positions are grouped for the operator's convenience. Fully inserted assemblies are further indicated by a rod at bottom signal, which actuates a local alarm and a control room annunciator. Group demand position is also indicated.

RCCAs are always moved in preselected banks, and the banks are always moved in the same preselected sequence. Each bank of RCCAs is divided into two groups. The rods comprising a group operate in parallel through multiplexing thyristors. The two groups in a bank move sequentially such that the first group is always within one step of the second group in the bank. A definite schedule of actuation (or deactuation of the stationary gripper, movable gripper, and lift coils of a mechanism) is required to withdraw the RCCA attached to the mechanism. Since the stationary gripper, movable gripper, and lift coils associated with the four RCCAs of a rod group are driven in parallel, any single failure that would cause rod withdrawal would affect a minimum of one group. Mechanical failures are in the direction of insertion, or immobility.

A dropped RCCA, or RCCA bank, is detected by:

(1) A sudden drop in the core power level as seen by the nuclear instrumentation system;

(2) Asymmetric power distribution as seen on out-of-core neutron detectors or core-exit thermocouples;

(3) Rod at bottom signal;

(4) Rod deviation alarm;

(5) Rod position indication;

(6) Negative neutron flux rate trip circuitry.

Misaligned RCCAs are detected by:

(1) Asymmetric power distribution as seen on out-of-core neutron detectors or core-exit thermocouples;

(2) Rod deviation alarm;

(3) Rod position indicators.

The deviation alarm alerts the operator whenever an individual rod position signal deviates from the other rods in the bank by a preset limit. If the rod deviation alarm is not operable, the operator is required to take action as required by the Technical Specifications[6].

If one or more rod position indicator channels should be out of service, detailed operating instructions are followed to ensure the alignment of the nonindicated RCCAs. The operator is also required to take action as required by the Technical Specifications.

15.2.3.2 Analysis of Effects and Consequences

Method of Analysis

(1) One or More Dropped RCCAs from the Same Group

For evaluation of the dropped RCCA event, the transient system response is calculated using the LOFTRAN code. The code simulates the neutron kinetics, RCS, pressurizer, pressurizer relief and safety valves, pressurizer spray, steam generator, and steam generator safety valves. The code computes pertinent plant variables including temperatures, pressures, and power level.

Statepoints are calculated and nuclear models are used to obtain a hot channel factor consistent with the primary system conditions and reactor power. By incorporating the primary conditions from the transient and the hot channel factor from the nuclear analysis, the DNB design basis is shown to be met using the THINC code[7]. The transient response, nuclear peaking factor analysis, and DNB design basis confirmation are performed in accordance with the methodology described in Reference 10.

(2) Dropped RCCA Bank

Analysis is not required since the dropped RCCA bank results in a trip.

(3) Statically Misaligned RCCA

Steady state power distributions are analyzed using the computer codes as described in Table 4.1-2 of the FSAR. The peaking factors are then used as input to the THINC code to calculate the DNBR.

15.2.3.3 Results

(1) One or More Dropped RCCAs

Single or multiple dropped RCCAs within the same group result in a negative reactivity insertion that may be detected by the power range negative neutron flux rate trip circuitry. If detected, the reactor is tripped within approximately 2.7 seconds following the drop of the RCCAs. The core is not adversely affected during this period since power is decreasing rapidly. Following reactor trip, normal shutdown procedures are followed. The operator may manually retrieve the RCCA by following approved operating procedures.

For those dropped RCCAs that do not result in a reactor trip, power may be reestablished either by reactivity feedback or control bank withdrawal. Following a dropped rod event in manual rod control, the plant will establish a new equilibrium condition. The equilibrium process without control system interaction is monotonic, thus removing power overshoot as a concern and establishing the automatic rod control mode of operation as the limiting case.

For a dropped RCCA event in the automatic rod control mode, the rod control system detects the drop in power and initiates control bank withdrawal. Power overshoot may occur due to this action by the automatic rod controller after which the control system will insert the control bank to restore nominal power. Figures 15.2.3-1 and 15.2.3-2 show a typical transient response to a dropped RCCA (or RCCAs) in automatic control. In all cases, the minimum DNBR remains above the safety analysis limit value.

(2) Dropped RCCA Bank

A dropped RCCA bank typically results in a reactivity insertion of greater than 500 pcm which will be detected by the power range negative neutron flux rate trip circuitry. The reactor is tripped within approximately 2.7 seconds following the drop of a RCCA bank. The core is not adversely affected during this period since power is decreasing rapidly. Following the reactor trip, normal shutdown procedures are followed to further cool down the plant. Any action required of the operator to maintain the plant in a stabilized condition will be in a time frame in excess of 10 minutes following the incident.

(3) Statically Misaligned RCCA

The most severe misalignment situations with respect to DNBR at significant power levels arise from cases in which one RCCA is fully inserted, or where Bank D is fully inserted with one RCCA fully withdrawn. Multiple independent alarms, including a bank insertion limit alarm, alert the operator well before the postulated conditions are approached. The bank can be inserted to its insertion limit with any one assembly fully withdrawn without the DNBR falling below the safety analysis limit value.

The insertion limits in the Technical Specifications may vary from time to time depending on a number of limiting criteria. It is preferable, therefore, to analyze the misaligned RCCA case at full power for a position of the control bank as deeply inserted as the criteria on minimum DNBR and power peaking factor will allow. The full power insertion limits on control Bank D must then be chosen to be above that position and will usually be dictated by other criteria. Detailed results will vary from cycle to cycle depending on fuel arrangements.

For this RCCA misalignment, with Bank D inserted to its full power insertion limit and one RCCA fully withdrawn, DNBR does not fall below the safety analysis limit value. This case is analyzed assuming the initial reactor power, pressure, and RCS temperatures are at their nominal values but with the increased radial peaking factor associated with the misaligned RCCA.

DNB calculations have not been performed specifically for RCCAs missing from other banks; however, power shape calculations have been done as required for the fully withdrawn analysis. Inspection of the power shapes shows that the DNB and peak kW/ft situation is less severe than the Bank D case discussed above assuming insertion limits on the other banks equivalent to a Bank D full-in insertion limit.

For RCCA misalignments with one RCCA fully inserted, the DNBR does not fall below the limit value. This case is analyzed assuming the initial reactor power, pressure, and RCS temperatures are at their nominal values, but with the increased radial peaking factor associated with the misaligned RCCA.

DNB does not occur for the RCCA misalignment incident and thus the ability of the primary coolant to remove heat from the fuel rod is not reduced. The peak fuel temperature corresponds to a linear heat generation rate based on the radial peaking factor penalty associated with the misaligned RCCA and the design axial power distribution. The resulting linear heat generation is well below that which would cause fuel melting.

Following the identification of an RCCA group misalignment condition by the operator, the operator is required to take action as required by the plant Technical Specifications and operating instructions.

15.2.3.4 Conclusions

For all cases of dropped RCCAs or dropped banks, for which the reactor is tripped by the power range negative neutron flux rate trip, there is no reduction in the margin to core thermal limits and, consequently, the DNB design basis is met. It is shown for all cases which do not result in reactor trip that the DNBR remains greater than the safety analysis limit value and, therefore, the DNB design basis is met.

For all cases of any RCCA inserted, or Bank D inserted to its rod insertion limits with any single RCCA in that bank fully withdrawn (static misalignment), the DNBR remains greater than the safety analysis limit value.

15.2.4 Uncontrolled Boron Dilution

15.2.4.1 Identification of Causes and Accident Description

Reactivity can be added to the core by feeding primary grade water into the reactor coolant system (RCS) via the reactor makeup portion of the chemical and volume control system (CVCS). Boron dilution is a manual operation under strict administrative controls with procedures calling for a limit on the rate and duration of dilution. A boric acid blend system is provided to permit the operator to match the boron concentration of reactor coolant makeup water during normal charging to that in the RCS. The CVCS is designed to limit, even under various postulated failure modes, the potential rate of dilution to a value which, after indication through alarms and instrumentation, provides the operator sufficient time to correct the situation in a safe and orderly manner.

The opening of the primary water makeup control valve provides makeup to the RCS which can dilute the reactor coolant. Inadvertent dilution from this source can be readily terminated by closing the control valve. In order for makeup water to be added to the RCS at pressure, at least one charging pump must be running in addition to a primary makeup water pump.

The rate of addition of unborated makeup water to the RCS is limited by a flow limiting orifice between the reactor makeup water pumps and the boric acid blender. As demonstrated by tests at the plant, flow is within the bounds of unborated water used in analyses in this section.

The boric acid from the boric acid tank is blended with primary grade water in the blender and the composition is determined by the preset flow rates of boric acid and primary grade water on the control board. In order to dilute two separate operations are required:

1. The operator must switch from the automatic makeup mode to the dilute mode;

2. The start/stop switch is in the start position.

Omitting either step would prevent dilution.

The status of the RCS makeup is continuously available to the operator by:

1. Indication of the boric acid and blended flow rates,

2. CVCS and RMWS pump status lights,

3. Deviation alarms if the boric acid or blended flow rates deviate by more than 10% from the preset values,

Indication of a dilution event is available to the operator by:

1. Source Range Neutron Flux - when reactor is subcritical;

   a. High flux at shutdown alarm. A separate alarm will be provided for each channel,

   b. Indicated source range neutron flux count rates, and

   c. Audible source range neutron flux count rate.

2. With the reactor critical;

   a. Axial flux difference alarm (reactor power $\geq$ 50% RTP),

   b. Control rod insertion limit low and low-low alarms,

   c. Overtemperature ΔT alarm (at power),

   d. Overtemperature ΔT turbine runback (at power),

   e. Overtemperature ΔT reactor trip, and

   f. Power range neutron flux - high, both high and low setpoint Reactor Trips.

15.2.4.2 Analysis of Effects and Consequences

To cover all phases of plant operation, boron dilution during refueling, cold shutdown, hot standby, startup, and power operation are considered in this analysis. The hot shutdown case is bounded by the analysis for cold shutdown and hot standby. Table 15.2-1 contains the time sequence of events for this accident.

1. Dilution During Refueling

An uncontrolled boron dilution accident based on a failure in the primary water makeup system cannot occur during refueling. This accident is prevented by administrative controls which isolate the RCS from the potential source of unborated water.

Valves 8454, 8441, 8430, and 8439 will be locked closed during refueling operations. These valves will block the flow paths which could allow unborated makeup water to reach the RCS. Any makeup which is required during refueling will be added to the Reactor Coolant System by unlocking these valves as appropriate and initiating the required blended makeup water flow. After the required volume of blended makeup flow has been added, these valves will again be locked closed. An alternate source of borated water that could be used is from the Refueling Water Storage Tank to the Charging Pump suction.

The most limiting alternate source of unborated water is from the boron thermal regeneration system (BTRS). For this case, highly borated RCS water is depleted of boron as it passes through the BTRS and is returned via the volume control tank. The following conditions are assumed for an uncontrolled boron dilution during refueling.

Technical Specifications require the reactor to be borated to at least 2,000 ppm or shutdown by at least 5.0 percent $\Delta k/k$ at refueling.

If an inadvertent dilution from the BTRS occurs during refueling with the reactor vessel head off and the refueling cavity filled with borated water (i.e., in a condition to move fuel), the maximum dilution capability of the BTRS is insufficient to cause a return to criticality.

The maximum dilution capability of the BTRS at these conditions is conservatively estimated tc e 250 ppm. However, the minimum change in boron concentration necessary to bring the reactor critical at these conditions is conservatively estimated to be 800 ppm. An initial boron concentration of 2500 ppm is assumed.

Therefore, a dilution to criticality from the BTRS at these refueling conditions cannot occur.

The most limiting conditions for an inadvertent boron dilution from the BTRS during refueling occur when the reactor head is unbolted but in place and the reactor coolant level is at the vessel/head junction. The dilution capability of the BTRS at these conditions is sufficient to cause a return to criticality. The minimum volume in the reactor coolant system corresponding to this condition is conservatively estimated to be 3300 ft^3. The critical boron concentration is conservatively estimated to be 1700 ppm.

2. Dilution During Cold Shutdown

Technical Specifications specify the required shutdown margin as a function of RCS boron concentration during cold shutdown. The specified shutdown margin ensures sufficient time for the operator to terminate the dilution. For a boron concentration of 1000 ppm, the required shutdown margin is 2.0% Δk. If the reactor is in cold shutdown and on the residual heat removal system with RCS piping filled and vented, the following conditions are assumed for an uncontrolled boron dilution. Dilution flow is assumed to be a maximum of 150 gpm, which is the capability of one primary water makeup pump to deliver unborated water to the RCS. Mixing of the reactor coolant is accomplished by the operation of one residual heat removal pump.

A volume of 4816.2 ft^3 in the reactor coolant system is used. This corresponds to the active volume of the reactor coolant system minus the pressurizer volume, while on the residual heat removal system.

If the reactor is in cold shutdown and the RCS water level is drained down from a filled and vented condition while on RHR, an inadvertent dilution is prevented by administrative controls which isolate the RCS from the potential source of unborated water. Valves 8454, 8441, 8430, and 8439 will be locked closed during operations in these conditions. These valves block all flow paths that could allow unborated makeup water to reach the RCS. Any makeup which is required will be added to the Reactor Coolant System by unlocking these valves as appropriate and initiating the required blended makeup water flow. After the required volume of blended makeup flow has been added, these valves will again be locked closed. An alternate source of borated water which may be used is from the Refueling Water Storage Tank to the Charging Pump suction.

3. Dilution during Hot Standby

Technical Specifications specify the required shutdown margin as a function of RCS boron concentration. For a boron concentration of 1500 ppm, the required shutdown margin is conservatively estimated to be 2.85% Δk.

The following conditions are assumed for a continuous boron dilution during hot standby:

Dilution flow is assumed to be a maximum of 150 gpm, which is the capability of one primary water makeup pump to deliver unborated water to the RCS.

A minimum RCS water volume of 5050 ft^3 is used. This is a conservative estimate of the active RCS volume with one reactor coolant pump operating.

4. Dilution During Startup

Prior to startup, the RCS is filled with borated water at a boron concentration of 2200 ppm. This is a conservative estimate with the reactor at a 1.77% Δk/k shutdown margin at 557°F.

Dilution flow is assumed to be a maximum of 150 gpm, which is the capability of one primary water makeup pump to deliver unborated water to the RCS. A minimum volume of 7682 ft^3 in the reactor coolant system is used. This is a conservative estimate of the active volume of the RCS excluding the pressurizer.

5. Dilution During Full Power Operation

During power operation, the plant may be operated two ways, under manual operator control or under automatic Tavg/rod control. The Technical Specifications require three reactor coolant pumps operating and a shutdown margin of at least 1.77% Δk/k. The RCS is conservatively assumed to be filled with borated water at a boron concentration of 2200 ppm.

While the plant is in manual control, the dilution flow is assumed to be a maximum of 150 gpm, which is the capacity of one reactor makeup water pump to deliver unborated water to the RCS. When in automatic control, the dilution flow is limited by the maximum letdown flow (approximately 125 gpm).

A minimum RCS water volume of 7682 ft^3 is used. This is a conservative estimate of the active volume of the RCS excluding the pressurizer.

15.2.4.3 Conclusion

Dilution During Refueling

During refueling, an inadvertent dilution from the reactor makeup water system is prevented by administrative controls which isolate the RCS from the potential source of unborated makeup water.

The most limiting conditions for an inadvertent dilution from the BTRS occur when the reactor vessel head is unbolted and the vessel water level is at the vessel/head junction. The high flux at shutdown alarm, set at twice the background flux level measured by the source range nuclear instrumentation, is available at these conditions to alert the operator that a dilution event is in progress.

For this case, the operator has 48 minutes from the high flux at shutdown alarm to recognize and terminate the dilution before shutdown margin is lost and the reactor becomes critical.

Dilution During Cold Shutdown

While in cold shutdown, the high flux at shutdown alarm set at twice the background flux level measured by the source range nuclear instrumentation, is available to alert the operator that a dilution event is in progress.

During the cold shutdown mode while operating on the residual heat removal system (RHRS) with the RCS piping filled and vented, the shutdown margin requirement ensures that the operator has at least 13.6 minutes from the high flux at shutdown alarm to recognize and terminate the uncontrolled reactivity insertion before shutdown margin is lost.

During the cold shutdown mode while operating on the RHRS with the RCS drained down from a filled and vented condition, an inadvertent dilution is precluded by administrative controls which isolate the RCS from the potential source of unborated water.

Dilution During Hot Shutdown

Analysis for a dilution during hot shutdown is bounded by the analysis for a dilution during cold shutdown and hot standby.

Dilution During Hot Standby

While in hot standby, the high flux at shutdown alarm, set at twice the background flux level measured by the source range nuclear instrumentation, is available to alert the operator that a dilution event is in progress.

During hot standby, the shutdown margin requirement ensures that the operator has at least 13.4 minutes from the high flux at shutdown alarm to recognize and terminate the uncontrolled reactivity insertion before shutdown margin is lost.

Dilution During Startup

In the event of an unplanned approach to criticality or dilution during power escalation while in the startup mode, the operator is alerted to an uncontrolled reactivity insertion by a reactor trip at the Power Range Neutron Flux-High, low setpoint (nominally 25% RTP). After reactor trip there is at least 20.6 minutes for operator action prior to return to criticality.

Dilution at Power

During the at power mode with manual control, the operator is alerted to an uncontrolled reactivity insertion by an overtemperature ΔT trip. 19.0 minutes are available from the trip for the operator to recognize and terminate the uncontrolled dilution. The sensitivity and alarm thresholds are already assumed to be degraded to the maximum extent allowable for the overtemperature ΔT trip function (see Section 15.2.2).

During the at power mode with automatic control, the operator is alerted to an uncontrolled reactivity insertion by the rod insertion limit alarms. Two insertion limit alarms are available: the first occurs when the rods are 10 steps above the insertion limit (LO Insertion Limit Alarm) and the second occurs at the insertion limit (Lo-Lo Insertion Limit Alarm). The analysis assumed that the operator is alerted to the need for action by the Lo-Lo Alarm although action would be taken when the first alarm occurs. Thus the analysis already assumes a 10 step allowance for rod position indicator inaccuracies. Even with this conservatism, there are still 23.0 minutes available from the time of alarm until all shutdown margin is lost. In addition to the above, other indications are available. The main indication would be a violation of the axial offset control band which could result in a reactor trip (reduction in overtemperature ΔT setpoint).

15.2.5 Partial Loss of Forced Reactor Coolant Flow

15.2.5.1 Identification of Causes and Accident Description

A partial loss of coolant flow accident can result from a mechanical or electrical failure in a reactor coolant pump, or from a fault in the power supply to the pump. If the reactor is at power at the time of the accident, the immediate effect of a loss of forced reactor coolant flow is a rapid increase in the coolant temperature. This increase could result in DNB with subsequent fuel damage if the reactor is not tripped promptly.

The necessary protection against a partial loss of coolant flow accident is provided by the low primary coolant flow reactor trip that is actuated by two-out-of-three low flow signals in any reactor coolant loop. Above approximately 38% power (Permissive 8), low flow in any loop will actuate a reactor trip. Between approximately 10% power (Permissive 7) and the power level corresponding to Permissive 8 low flow in any two loops will actuate a reactor trip. Reactor trip on low flow is blocked below Permissive 7.

A reactor trip signal from the pump breaker position is also provided. When operating above Permissive 7, a breaker open signal from any two pumps will actuate a reactor trip. This serves as a backup to the low flow trip. Reactor trip on reactor coolant pump breakers open is blocked below Permissive 7.

Normal power for each pump is supplied through individual buses connected to the isolated phase bus duct between the generator circuit breaker and the main transformer. Faults in the substation may cause a trip of the main transformer high side circuit breaker leaving the generator to supply power to the reactor coolant pumps. When a generator circuit breaker trip occurs because of electrical faults, the pumps are automatically transferred to an alternate power supply and the pumps will continue to supply coolant flow to the core. Following any turbine trip where there are no electrical faults, the generator circuit breaker is tripped and the reactor coolant pumps remain connected to the network through the transformer high side breaker. Continuity of power to the pump buses is achieved without motoring the generator since means are provided to isolate the generator without isolating the pump buses from the external power lines (e.g., a generator output breaker is provided as well as a station output breaker).

15.2.5.2 Analysis of Effects and Consequences

15.2.5.2.1 Method of Analysis

The following case has been analyzed:

All loops operating, one loop coasting down

This transient is analyzed by three digital computer codes. First the LOFTRAN code is used to calculate the loop and core flow during the transient. The LOFTRAN code is also used to calculate the time of reactor trip, based on the calculated flows and the nuclear power transient following reactor trip. The FACTRAN code is then used to calculate the heat flux transient based on the nuclear power and flow from LOFTRAN. Finally, the THINC code[7] is used to calculate the minimum DNBR during the transient based on the heat flux from FACTRAN and the flow from LOFTRAN. The DNBR transient presented represents the minimum of the typical and thimble cells for Standard and VANTAGE 5 fuel.

15.2.5.2.2 Initial Conditions

The assumed initial operating conditions are the most adverse with respect to the margin to DNB, i.e., nominal steady state power level, nominal steady state pressure, and nominal steady state coolant average temperature. See Section 15.1.2 for an explanation of initial conditions. The accident is analyzed using the Improved Thermal Design Procedure as described in Reference 5.

15.2.5.2.3 Reactivity Coefficients

A conservatively large absolute value of the Doppler-only power coefficient is used (see Table 15.1-4). The total integrated Doppler reactivity from 0 to 100% power is assumed to be -0.016 Δk.

The least negative moderator temperature coefficient at full power (0 pcm/°F) is assumed since this results in the maximum hot spot heat flux during the initial part of the transient when the minimum DNBR is reached.

15.2.5.2.4 Flow Coastdown

The flow coastdown analysis is based on a momentum balance around each reactor coolant loop and across the reactor core. This momentum balance is combined with the continuity equation, a pump momentum balance, and the pump characteristics and is based on high estimates of system pressure losses to calculate the flow coastdown.

15.2.5.3 Results

The calculated sequence of events is shown in Table 15.2-1. Figures 15.2.5-1 and 15.2.5-2 show the vessel flow coastdown, the faulted loop flow coastdown, the nuclear power and heat flux transient. The minimum DNBR is not less than the safety analysis limit value. A plot of DNBR vs. time is given in Figure 15.2.5-3 for the most limiting typical or thimble cell for Standard and VANTAGE 5 fuel.

15.2.5.4 Conclusions

The analysis shows that the DNBR will not decrease below the safety analysis limit values at any time during the transient. Thus, no core safety limit is violated.

15.2.6 Startup of an Inactive Reactor Coolant Loop

In accordance with Technical Specification 3/4.4.1, V. C. Summer operation during startup and power operation with less than three loops operating is not permitted. This analysis is presented for completeness.

15.2.6.1 Identification of Causes and Accident Description

If a plant is operating with one pump out of service, there is reverse flow through the loop due to the pressure difference across the reactor vessel. The cold leg temperature in an inactive loop is identical to the cold leg temperature of the active loops (the reactor core inlet temperature). If the reactor is operated at power, and assuming the secondary side of the steam generator in the inactive loop is not isolated, there is a temperature drop across the steam generator in the inactive loop and, with the reverse flow, the hot leg temperature of the inactive loop is lower than the reactor core inlet temperature.

Administrative procedures require that the unit be brought to a load of less than 25% of full power prior to starting a pump in an inactive loop in order to bring the inactive loop hot leg temperature closer to the core inlet temperature. Starting of an idle reactor coolant pump without bringing the inactive loop hot leg temperature close to the core inlet temperature would result in the injection of cold water into the core which causes a rapid reactivity insertion and subsequent power increase.

This event is classified as an ANS Condition II incident (an incident of moderate frequency) as defined in Section 15.0.

Should the startup of an inactive reactor coolant pump at an incorrect temperature occur, the transient will be terminated automatically by a reactor trip on low coolant loop flow when the power range neutron flux (two out of four channels) exceeds the P-8 setpoint, which has been previously reset for two loop operation.

15.2.6.2 Analysis of Effects and Consequences

This transient is analyzed by three digital computer codes. The LOFTRAN Code[4] is used to calculate the loop and core flow, nuclear power and core pressure and temperature transients following the startup of an idle pump. FACTRAN[3] is used to calculate the core heat flux transient based on core flow and nuclear power from LOFTRAN. The THINC Code[7] is then used to calculate the DNBR during the transient based on system conditions (pressure, temperature, and flow) calculated by LOFTRAN and heat flux as calculated by FACTRAN.

In order to obtain conservative results for the startup of an inactive pump accident, the following assumptions are made:

(1) Initial conditions of maximum core power and reactor coolant average temperatures and minimum reactor coolant pressure resulting in minimum initial margin to DNB. These values are to be consistent with maximum steady state power level allowed with all but one loop in operation including appropriate allowances for calibration and instrument errors. The high initial power gives the greatest temperature difference between the core inlet temperature and the inactive loop hot leg temperature.

(2) Following the start of the idle pump, the inactive loop flow reverses and accelerates to its nominal full flow value.

(3) A conservatively large (absolute value) negative moderator temperature coefficient associated with end of life conditions.

(4) A conservatively low (absolute value) negative Doppler power coefficient is used.

(5) The initial reactor coolant loop flows are at the appropriate values for one pump out of service.

(6) The reactor trip is assumed to occur on low coolant flow when the power range neutron flux exceeds the P-8 setpoint. The P-8 setpoint is conservatively assumed to be 74 percent of rated power.

15.2.6.3 Results

The results following the startup of an idle pump with the above listed assumptions are shown in Figures 15.2.6-1 through 15.2.6-4. As shown in these curves, during the first part of the transient, the increase in core flow with cooler water results in an increase in nuclear power and a decrease in core average temperature. The minimum DNBR during the transient is considerably greater than the safety analysis limit values.

Reactivity addition for the inactive loop startup accident is due to the decrease in core water temperature. During the transient, this decrease is due both to (1) the increase in reactor coolant flow and, (2) as the inactive loop flow reverses, to the colder water entering the core from the hot leg side (colder temperature side prior to the start of the transient) of the steam generator in the inactive loop. Thus, the reactivity insertion rate for this transient changes with time. The resultant core nuclear power transient, computed with consideration of both moderator and Doppler reactivity feedback effects, is shown on Figure 15.2.6-1.

The calculated sequence of events for this accident is shown in Table 15.2-1. The transient results illustrated in Figures 15.2.6-1 through 15.2.6-4 indicate that a stabilized plant condition, with the reactor tripped, is approached rapidly. Plant cooldown may subsequently be achieved by following normal shutdown procedures.

15.2.6.4 Conclusions

The transient results show that the core is not adversely affected. There is considerable margin to the safety analysis DNBR limit values; thus, no fuel or clad damage is predicted.

15.2.7 Loss of External Electrical Load and/or Turbine Trip

15.2.7.1 Identification of Causes and Accident Description

A major load loss on the plant can result from either a loss of external electrical load or from a turbine trip. For either case, offsite power is available for the continued operation of plant components such as the reactor coolant pumps. The case of loss of all ac power (station blackout) is analyzed in Section 15.2.9.

For a turbine trip, the reactor would be tripped directly (unless it is below approximately 50% power) from a signal derived from the turbine autostop oil pressure and turbine stop valves. The automatic steam dump system accommodates the excess steam generation. Reactor coolant temperatures and pressure do not significantly increase if the steam dump system and pressurizer pressure control system are functioning properly. If the turbine condenser were not available, the excess steam generation would be dumped to the atmosphere. Additionally, main feedwater flow would be lost if the turbine condenser were not available. For this situation, steam generator level would be maintained by the emergency feedwater system.

For a loss of external electrical load without subsequent turbine trip, no direct reactor trip signal would be generated. With full load rejection capability the plant would be expected to continue operating without a reactor trip. A continued steam load of approximately 5% would exist after total loss of external electrical load because of the electrical demand of plant auxiliaries.

In the event the steam dump valves fail to open following a large loss of load, the steam generator safety valves may lift and the reactor may be tripped by the high pressurizer pressure signal, the high pressurizer water level signal, or the overtemperature ΔT signal. The steam generator shell-side pressure and reactor coolant temperatures will increase rapidly. The pressurizer safety valves and steam generator safety valves are, however, sized to protect the RCS and steam generator against overpressure for all load losses without assuming the operation of the steam dump system, pressurizer spray, pressurizer power-operated relief valves, automatic RCCA control, or direct reactor trip on turbine trip.

The steam generator safety valve capacity is sized to remove the steam flow at the engineered safeguards design rating (104.5% of steam flow at rated power) from the steam generator without exceeding 110% of the steam system design pressure. The pressurizer safety valve capacity is sized based on a complete loss of heat sink with the plant initially operating at the maximum calculated turbine load along with operation of the steam generator safety valves. The pressurizer safety valves are then able to maintain the RCS pressure within 110% of the RCS design pressure without direct or immediate reactor trip action.

A more complete discussion of overpressure protection can be found in Reference 8.

15.2.7.2 Analysis of Effects and Consequences

In this analysis, the behavior of the unit is evaluated for a complete loss of steam load from full power without a direct reactor trip. This is done to show the adequacy of the pressure-relieving devices and to demonstrate core protection margins. The reactor is not tripped until conditions in the RCS result in a trip. The turbine is assumed to trip without actuating all the turbine stop valve limit switches. This assumption delays reactor trip until conditions in the RCS result in a trip due to other signals. Thus, the analysis assumes a worst case transient. In addition, no credit is taken for steam dump. Main feedwater flow is terminated at the time of turbine trip, with no credit taken for emergency feedwater (except for long-term recovery) to mitigate the consequences of the transient.

The total loss of load transients are analyzed with the LOFTRAN computer program (see Section 15.1). The program simulates the neutron kinetics, RCS, pressurizer, pressurizer relief and safety valves, pressurizer spray, steam generator, and steam generator safety valves. The program computes pertinent plant variables including temperatures, pressures, and power level.

Major assumptions are summarized below:

(1) Initial Operating Conditions

The initial reactor power and RCS temperatures are assumed at their maximum values consistent with the steady-state full power operation including allowances for calibration and instrument errors. The initial RCS pressure is assumed at a minimum value consistent with the steady-state full power operation including allowances for calibration and instrument errors. This results in the maximum power difference for the load loss, and the minimum margin to core protection limits at the initiation of the accident.

(2) Moderator and Doppler Coefficients of Reactivity

The turbine trip is analyzed with both maximum and minimum reactivity feedback. The maximum feedback (EOL) cases assume a large negative moderator temperature coefficient and the most negative Doppler power coefficient. The minimum feedback (BOL) cases assume a minimum moderator temperature coefficient and the least negative Doppler coefficient.

(3) Reactor Control

From the standpoint of the maximum pressures attained, it is conservative to assume that the reactor is in manual control. If the reactor were in automatic control, the control rod banks would move prior to trip and reduce the severity of the transient.

(4) Steam Release

No credit is taken for the operation of the steam dump system or steam generator power-operated relief valves. The steam generator pressure rises to the safety valve setpoint where steam release through safety valves limits secondary steam pressure at the setpoint value.

(5) Pressurizer Spray and Power-operated Relief Valves

Two cases for both the BOL and EOL are analyzed:

(a) Full credit is taken for the effect of pressurizer spray and power-operated relief valves in reducing or limiting the coolant pressure. Safety valves are also available.

(b) No credit is taken for the effect of pressurizer spray and power-operated relief valves in reducing or limiting the coolant pressure. Safety valves are operable.

(6) Feedwater Flow

Main feedwater flow to the steam generators is assumed to be lost at the time of turbine trip. No credit is taken for emergency feedwater flow since a stabilized plant condition will be reached before emergency feedwater initiation is normally assumed to occur; however, the emergency feedwater pumps would be expected to start on a trip of the main feedwater pumps. The emergency feedwater flow would remove core decay heat following plant stabilization.

Reactor trip is actuated by the first reactor protection system trip setpoint reached with no credit taken for the direct reactor trip on the turbine trip.

15.2.7.3 Results

The transient responses for a total loss of load from full power operation are shown for four cases; two cases for the BOL and two cases for the EOL on Figures 15.2.7-1 through 15.2.7-12.

Figures 15.2.7-1, 15.2.7-2 and 15.2.7-3 show the transient responses for the total loss of steam load at BOL assuming full credit for the pressurizer spray and pressurizer power-operated relief valves. No credit is taken for the steam dump. The reactor is tripped by the overtemperature ΔT trip channel. The minimum DNBR is well above the limit value. The pressurizer safety valves are actuated for this case and maintain system pressure below 110 percent of the design value. The steam generator safety valves open and limit the secondary steam pressure increase.

Figures 15.2.7-4, 15.2.7-5 and 15.2.7-6 show the responses for the total loss of load at EOL assuming a large (absolute value) negative moderator temperature coefficient. All other plant parameters are the same as in the above case. The reactor is tripped by the overtemperature ΔT trip channel. The DNBR increases throughout the transient and never drops below its initial value.

Total loss of load was also studied assuming the plant to be initially operating at full power with no credit taken for the pressurizer spray, pressurizer power-operated relief valves, or steam dump. The reactor is tripped on the high pressurizer pressure signal. Figures 15.2.7-7, 15.2.7-8 and 15.2.7-9 show the BOL transients. The nuclear power remains at or above full power until the reactor is tripped. The DNBR generally increases throughout the transient. In this case the pressurizer safety valves are actuated and maintain the system pressure below 110 percent of the design value.

Figures 15.2.7-10, 15.2.7-11 and 15.2.7-12 show the transient at EOL with the other assumptions being the same as on Figures 15.2.7-7 through 15.2.7-9. Again, the DNBR increases throughout the transient and the pressurizer safety valves are actuated to limit the primary pressure.

Reference 8 presents additional results for a complete loss of heat sink including loss of main feedwater. This report shows the overpressure protection that is afforded by the pressurizer and steam generator safety valves.

15.2.7.4 Conclusions

Results of the analyses, including those in Reference 8, show that the plant design is such that a total loss of external electrical load without a direct or immediate reactor trip presents no hazard to the integrity of the RCS or the main steam system. Pressure-relieving devices incorporated in the two systems are adequate to limit the maximum pressures to within the design limits.

The integrity of the core is maintained by operation of the reactor protection system; i.e., the DNBR will be maintained above the safety analysis limit values. Thus, no core safety limit will be violated.

15.2.8 Loss of Normal Feedwater

15.2.8.1 Identification of Causes and Accident Description

A loss of normal feedwater (from pump failures, valve malfunctions, or loss of offsite ac power) results in a reduction in capability of the secondary system to remove the heat generated in the reactor core. If the reactor were not tripped during this accident, core damage would possibly occur from a sudden loss of heat sink. If an alternative supply of feedwater were not supplied to the plant, residual heat following reactor trip would heat the primary system water to the point where water relief from the pressurizer would occur. Significant loss of water from the RCS could conceivably lead to core damage. Since the plant is tripped well before the steam generator heat transfer capability is reduced, the primary system variables never approach a DNB condition.

The following provide the necessary protection against a loss of normal feedwater:

(1) Reactor trip on low-low water level in any steam generator;

(2) Reactor trip on steam flow-feedwater flow mismatch in coincidence with low steam generator water level;

(3) Two motor-driven emergency feedwater (EFW) pumps that are started on:

- (a) Low-low level in any steam generator,
- (b) Trip of all main feedwater pumps,
- (c) Any safety injection signal,
- (d) Loss of offsite power (automatic transfer to diesel generators),
- (e) Manual actuation.

(4) One turbine-driven emergency feedwater pump that is started on:

(a) Low-low level in any two steam generators,

(b) Loss of offsite power,

(c) Manual actuation.

The motor-driven EFW pumps are connected to vital buses and are supplied by the diesels if a loss of offsite power occurs. The turbine-driven pump utilizes steam from the secondary system and exhausts it to the atmosphere. The controls are designed to start both types of pumps within 1 minute even if a loss of all ac power occurs simultaneously with loss of normal feedwater. The EFW pumps take suction from the condensate storage tank for delivery to the steam generators.

The analysis shows that following a loss of normal feedwater, the EFW system is capable of removing the stored and residual heat thus preventing either overpressurization of the RCS or loss of water from the reactor core.

15.2.8.2 Analysis of Effects and Consequences

A detailed analysis using the LOFTRAN code(4) is performed in order to determine the plant transient following a loss of normal feedwater. The code describes the plant thermal kinetics, RCS including natural circulation, pressurizer, steam generators, and feedwater system, and computes pertinent variables, including the pressurizer pressure, pressurizer water level, and reactor coolant average temperature.

Major assumptions are:

(1) Reactor trip occurs on steam generator low-low level at 23.2% of narrow range span.

(2) The plant is initially operating at 102% of the NSSS design rating.

(3) Conservative core residual heat generation based on long-term operation at the initial power level preceding the trip is assumed. The 1979 decay heat ANSI 5.1 + 2 SIGMA was used for calculation of residual decay heat levels.

(4) The emergency feedwater system is actuated by the low-low steam generator water level signal.

(5) The worst single failure in the emergency feedwater system occurs (turbine-driven pump) and one motor-driven pump is assumed to be unavailable. The emergency feedwater system is assumed to supply a total of 380 gpm to two steam generators from the available motor-driven pump.

(6) The pressurizer sprays and PORVs are assumed operable. This maximizes the peak transient pressurizer water volume.

(7) Secondary system steam relief is achieved through the self-actuated safety valves. Note that steam relief will, in fact, be through the power-operated relief valves or condenser dump valves for most cases of loss of normal feedwater. However, for the sake of analysis these have been assumed unavailable.

(8) The initial reactor coolant average temperature is 4.0°F higher than the nominal value to allow for uncertainty on nominal temperature. The initial pressurizer pressure uncertainty is 33 psi.

15.2.8.3 Results

Figures 15.2.8-1 and 15.2.8-2 show plant parameters following a loss of normal feedwater.

Following the reactor and turbine trip from full load, the water level in the steam generators will fall due to the reduction of steam generator void fraction and because steam flow through the safety valves continues to dissipate the stored and generated heat. One minute following the initiation of the low-low level trip, the motor-driven EFW pump is automatically started, reducing the rate of water level decrease.

The capacity of the motor-driven EFW pump is such that the water level in the steam generator being fed does not recede below the lowest level at which sufficient heat transfer area is available to dissipate core residual heat without water relief from the RCS relief or safety valves.

From Figure 15.2.8-2 it can be seen that at no time is there water relief from the pressurizer. If the emergency feed delivered is greater than that of one motor-driven pump, the initial reactor power is less than 102% of the NSSS design rating, or the steam generator water level in one or more steam generators is above the low-low level trip point at the time of trip, then the results for this transient will be less limiting.

The calculated sequence of events for this accident is listed in Table 15.2-1. As shown in Figures 15.2.8-1 and 15.2.8-2, the plant approaches a stabilized condition following reactor trip and emergency feedwater initiation. Plant procedures may be followed to further cool down the plant.

15.2.8.4 Conclusions

Results of the analysis show that a loss of normal feedwater does not adversely affect the core, the RCS, or the steam system since the EFW capacity is such that the reactor coolant water is not relieved from the pressurizer relief or safety valves.

15.2.9 Loss of OffSite Power to the Station Auxiliaries (Station Blackout)

15.2.9.1 Identification of Causes and Accident Description

During a complete loss of offsite power and a turbine trip there will be loss of power to the plant auxiliaries, i.e., the reactor coolant pumps, condensate pumps, etc.

The events following a loss of ac power with turbine and reactor trip are described in the sequence listed below:

(1) Plant vital instruments are supplied by emergency power sources.

(2) As the steam system pressure rises following the trip, the steam system power-operated relief valves are automatically opened to the atmosphere. Steam dump to the condenser is assumed not to be available. If the power-operated relief valves are not available, the steam generator self-actuated safety valves may lift to dissipate the sensible heat of the fuel and coolant plus the residual heat produced in the reactor.

(3) As the no-load temperature is approached, the steam system power-operated relief valves (or the self-actuated safety valves, if the power-operated relief valves are not available) are used to dissipate the residual heat and to maintain the plant at the hot standby condition.

(4) The emergency diesel generators started on loss of voltage on the plant emergency buses begin to supply plant vital loads.

The EFW system is started automatically as discussed in the loss of normal feedwater analysis. The steam-driven emergency feedwater pump utilizes steam from the secondary system and exhausts to the atmosphere. The two motor-driven EFW pumps are supplied by power from the diesel generators. The pumps take suction directly from the condensate storage tank for delivery to the steam generators.

Upon the loss of power to the reactor coolant pumps, coolant flow necessary for core cooling and the removal of residual heat is maintained by natural circulation in the reactor coolant loops.

15.2.9.2 Analysis of Effects and Consequences

A detailed analysis using the LOFTRAN code[4] is performed in order to determine the plant transient following a station blackout. The code describes the plant thermal kinetics, RCS including natural circulation, pressurizer, steam generators, and feedwater system, and computes pertinent variables, including the pressurizer pressure, pressurizer water level, and reactor coolant average temperature.

Major assumptions differing from those in a loss of normal feedwater are:

(1) No credit is taken for immediate response of control rod drive mechanisms caused by a loss of offsite power.

(2) A heat transfer coefficient in the steam generator associated with RCS natural circulation is assumed following the reactor coolant pump coastdown.

The time sequence of events for the accident is given in Table 15.2-1. The first few seconds after the loss of power to the reactor coolant pumps will closely resemble a simulation of the complete loss of flow incident (see Section 15.3.4); i.e., core damage due to rapidly increasing core temperatures is prevented by promptly tripping the reactor. After the reactor trip, stored and residual heat must be removed to prevent damage to either the RCS or the core. The LOFTRAN code results show that the natural circulation flow available is sufficient to provide adequate core decay heat removal following reactor trip and RCP coastdown.

15.2.9.3 Conclusions

Results of the analysis show that, for the loss of offsite power to the station auxiliaries event, all safety criteria are met. Since the DNBR remains above the safety analysis limit, the core is not adversely affected. EFW capacity is sufficient to prevent water relief through the pressurizer relief and safety valves; this assures that the RCS is not overpressurized.

Analysis of the natural circulation capability of the RCS demonstrates that sufficient long-term heat removal capability exists following reactor coolant pump coastdown to prevent fuel or clad damage.

15.2.10 Excessive Heat Removal Due to Feedwater System Malfunctions

15.2.10.1 Identification of Causes and Accident Description

Excessive feedwater additions are a means of increasing core power above full power. Such transients are attenuated by the thermal capacity of the secondary plant and of the RCS. The overpower and overtemperature protection (high neutron flux, overtemperature ΔT, and overpower ΔT trips) prevent any power increase that could lead to a DNBR that is less than the DNBR limit.

An example of excessive feedwater flow would be a full opening of a feedwater control valve due to a feedwater control system malfunction or an operator error. At power, this excess flow causes a greater load demand on the RCS due to increased subcooling in the steam generator. With the plant at no-load conditions, the addition of cold feedwater may cause a decrease in RCS temperature and thus a reactivity insertion due to the effects of the negative moderator coefficient of reactivity. Continuous excessive feedwater addition is prevented by the steam generator high-high level trip, which closes the feedwater valves.

15.2.10.2 Analysis of Effects and Consequences

The excessive heat removal due to a feedwater system malfunction transient is analyzed with the LOFTRAN code. This code simulates a multiloop system, neutron kinetics, the pressurizer, pressurizer relief and safety valves, pressurizer spray, steam generator, and steam generator safety valves. The code computes pertinent plant variables including temperatures, pressures, and power level.

The system is analyzed to evaluate plant behavior in the event of a feedwater system malfunction.

Excessive feedwater addition due to a control system malfunction or operator error that allows a feedwater control valve to open fully is considered. Two cases are analyzed as follows:

(1) Accidental opening of one feedwater control valve with the reactor just critical at zero load conditions assuming a conservatively large moderator density coefficient characteristic of end-of-life conditions.

(2) Accidental opening of one feedwater control valve with the reactor in manual control at full power.

The reactivity insertion rate following a feedwater system malfunction is calculated with the following assumptions:

(1) For the feedwater control valve accident at full power, one feedwater control valve is assumed to malfunction, resulting in a step increase of 250% of nominal feedwater flow to one steam generator.

(2) For the feedwater control valve accident at zero load conditions, a feedwater valve malfunction occurs that results in a step increase in flow to one steam generator from zero to the nominal full load value for one steam generator.

(3) For the zero load condition, feedwater temperature is at a conservatively low value of 70°F.

(4) No credit is taken for the heat capacity of the RCS and steam generator thick metal in attenuating the resulting plant cooldown.

(5) No credit is taken for the heat capacity of the steam and water in the unaffected steam generators.

(6) The feedwater flow resulting from a fully open control valve is terminated by the steam generator high-high level signal that closes all feedwater control valves, closes all feedwater bypass valves, trips the main feedwater pumps, and shuts the feedwater isolation valves. The steam generator high-high level signal also produces a signal to trip the turbine.

15.2.10.3 Results

In the case of an accidental full opening of one feedwater control valve with the reactor at zero power and the above mentioned assumptions, the maximum reactivity insertion rate is less than the maximum reactivity insertion rate analyzed in Section 15.2.1, Uncontrolled RCCA Bank Withdrawal from a Subcritical Condition, and its analysis is, therefore, covered by that of the latter. It should be noted that if the incident occurs with the unit just critical at no-load, the reactor may be tripped by the power range high neutron flux trip (low setting) set at approximately 25% of nominal full power.

The full power case (end-of-life, without control) gives the largest reactivity feedback and results in the greatest power increase. Assuming the reactor to be in the automatic control mode results in a slightly less severe transient. The rod control system is not required to function for an excessive feedwater flow event. A turbine trip is actuated when the steam generator level reaches the high-high level setpoint. For convenience, reactor trip is assumed to be initiated upon turbine trip. However, this function is not necessary. Should turbine trip not initiate a reactor trip signal, reactor trip will occur on power range high neutron flux.

For all cases of excessive feedwater, continuous addition of cold feedwater is prevented by closure of all feedwater control valves, closure of all feedwater bypass valves, a trip of the feedwater pumps, and closures of the feedwater isolation valves on steam generator high-high level.

Transient results (see Figures 15.2.10-1 and 15.2.10-2) show the core heat flux, pressurizer pressure, T_{avg}, and DNBR, as well as the increase in nuclear power and loop ΔT associated with the increased thermal load on the reactor. Steam generator level rises until the feedwater is terminated as a result of the high-high steam generator level trip. The DNBR does not drop below the limit safety analysis DNBR value.

15.2.10.4 Conclusions

The reactivity insertion rate that occurs at no-load following excessive feedwater addition is less than the maximum value considered in the analysis of the rod withdrawal from a subcritical condition. Also, the DNBRs encountered for excessive feedwater addition at power are well above the safety analysis limit DNBR value.

15.2.11 Excessive Load Increase Incident

15.2.11.1 Identification of Cause and Accident Description

An excessive load increase incident is defined as a rapid increase in the steam flow that causes a power mismatch between the reactor core power and the steam generator load demand. The reactor control system is designed to accommodate a 10% step-load increase or a 5% per minute ramp load increase in the range of 15 to 100% of full power. Any loading rate in excess of these values may cause a reactor trip actuated by the reactor protection system.

This accident could result from either an administrative violation such as excessive loading by the operator or an equipment malfunction in the steam dump control or turbine speed control.

During power operation, steam dump to the condenser is controlled by reactor coolant condition signals; i.e., high reactor coolant temperature indicates a need for steam dump. A single controller malfunction does not cause steam dump; an interlock is provided that blocks the opening of the valves unless a large turbine load decrease or a turbine trip has occurred.

Protection against an excessive load increase accident is provided by the following reactor protection system signals:

(1) Overpower ΔT,

(2) Overtemperature ΔT,

(3) Power range high neutron flux.

15.2.11.2 Analysis of Effects and Consequences

This accident is analyzed using the LOFTRAN code[4]. The code simulates the neutron kinetics, RCS, pressurizer, pressurizer relief and safety valves, pressurizer spray, feedwater system, steam generator, and steam generator safety valves. The code computes pertinent plant variables including temperatures, pressures, and power level.

Four cases are analyzed to demonstrate the plant behavior following a 10% step load increase from rated load. These cases are as follows:

(1) Reactor control in manual with BOL minimum moderator reactivity feedback,

(2) Reactor control in manual with EOL maximum moderator reactivity feedback,

(3) Reactor control in automatic with BOL minimum moderator reactivity feedback,

(4) Reactor control in automatic with EOL maximum moderator reactivity feedback.

For the BOL minimum moderator feedback cases, the core has the least negative moderator temperature coefficient of reactivity and the least negative Doppler only power coefficient curve; therefore the least inherent transient response capability. For the EOL maximum moderator feedback cases, the moderator temperature coefficient of reactivity has its highest absolute value and the most negative Doppler only power coefficient curve. This results in the largest amount of reactivity feedback due to changes in coolant temperature.

A conservative limit on the turbine valve opening is assumed, and all cases are studied without credit being taken for pressurizer heaters.

This accident is analyzed with the Improved Thermal Design Procedure as described in Reference 5. Initial reactor power, RCS pressure and temperature are assumed to be at their nominal values. Uncertainties in initial conditions are included in the limit DNBR as described in Reference 5.

Plant characteristics and initial conditions are further discussed in Section 15.1.

Normal reactor control systems and engineered safety systems are not required to function. The reactor protection system is assumed to be operable; however, reactor trip is not encountered for most cases due to the error allowances assumed in the setpoints. No single active failure will prevent the reactor protection system from performing its intended function.

The cases which assume automatic rod control are analyzed to ensure that the worst case is presented. The automatic function is not required.

15.2.11.3 Results

The calculated sequence of events for the excessive load increase incident are shown on Table 15.2-1.

Figures 15.2.11-1 through 15.2.11-4 illustrate the transient with the reactor in the manual control mode. As expected, for the BOL minimum moderator feedback case, there is a slight power increase, and the average core temperature shows a large decrease. This results in a DNBR which increases above its initial value. For the EOL maximum moderator feedback manually controlled case, there is a much larger increase in reactor power due to the moderator feedback. A reduction in DNBR is experienced but DNBR remains above the limit value.

Figures 15.2.11-5 through 15.2.11-8 illustrate the transient assuming the reactor is in the automatic control mode. Both the BOL minimum and EOL maximum moderator feedback cases show that core power increases, thereby reducing the rate of decrease in coolant average temperature and pressurizer pressure. For both of these cases, the minimum DNBR remains above the limit value.

For all cases, the plant rapidly reaches a stabilized condition at the higher power level. Normal plant operating procedures would then be followed to reduce power.

The excessive load increase incident is an overpower transient for which the fuel temperatures will rise. Reactor trip does not occur for any of the cases analyzed, and the plant reaches a new equilibrium condition at a higher power level corresponding to the increase in steam flow.

Since DNB does not occur at any time during the excessive load increase transients, the ability of the primary coolant to remove heat from the fuel rod is not reduced. Thus, the fuel cladding temperature does not rise significantly above its initial value during the transient.

15.2.11.4 Conclusions

The analysis presented above shows that for a 10% step load increase, the DNBR remains above the safety analysis limit value, thereby precluding fuel or clad damage. The plant reaches a stabilized condition rapidly, following the load increase.

15.2.12 Accidental Depressurization of the Reactor Coolant System

15.2.12.1 Identification of Causes and Accident Description

An accidental depressurization of the Reactor Coolant System could occur as a result of an inadvertent opening of a pressurizer relief or safety valve. Since a safety valve is sized to relieve approximately twice the steam flowrate of a relief valve, and will therefore allow a much more rapid depressurization upon opening, the most severe core conditions resulting from an accidental depressurization of the RCS are associated with an inadvertent opening of a pressurizer safety valve. Initially, the event results in a rapidly decreasing RCS pressure until this pressure reaches a value corresponding to the hot leg saturation pressure. At that time, the pressure decrease is slowed considerably. The pressure continues to decrease, however, throughout the transient. The effect of the pressure decrease would be to decrease the neutron flux via the moderator density feedback, but the reactor control system (if in the automatic mode) functions to maintain the power and average coolant temperature essentially constant throughout the initial stage of the transient. Pressurizer level increases initially due to expansion caused by depressurization and then decreases following reactor trip.

The reactor will be tripped by the following reactor protection system signals:

(1) Pressurizer low pressure,

(2) Overtemperature ΔT.

15.2.12.2 Analysis of Effects and Consequences

The accidental depressurization transient is analyzed with the LOFTRAN code[4]. The code simulates the neutron kinetics, RCS, pressurizer, pressurizer relief and safety valves, pressurizer spray, steam generator, and steam generator safety valves. The code computes pertinent plant variables including temperatures, pressures, and power level.

This accident is analyzed with the Improved Thermal Design Procedure as described in Reference 5.

In calculating the DNBR the following conservative assumptions are made:

(1) Plant characteristics and initial conditions are discussed in Section 15.1. Uncertainties and initial conditions are included in the limit DNBR as described in Reference 5.

(2) A positive moderator temperature coefficient of reactivity for BOL operation in order to provide a conservatively high amount of positive reactivity feedback due to changes in moderator temperature. The spatial effect of voids due to local or subcooled boiling is not considered in the analysis with respect to reactivity feedback or core power shape. These voids would tend to flatten the core power distribution.

(3) A low (absolute value) Doppler coefficient of reactivity such that the resultant amount of negative feedback is conservatively low in order to maximize any power increase due to moderator reactivity feedback.

15.2.12.3 Results

Figure 15.2.12-1 illustrates the nuclear power transient following the RCS depressurization accident. The flux increases until the time reactor trip occurs on overtemperature ΔT, thus resulting in a rapid decrease in the nuclear flux. The time of reactor trip is shown in Table 15.2-1. The pressure decay transient following the accident is given on Figure 15.2.12-2. The resulting DNBR never goes below the safety analysis limit value as shown on Figure 15.2.12-3.

15.2.12.4 Conclusions

The pressurizer low pressure and the overtemperature ΔT reactor protection system signals provide adequate protection against this accident, and the minimum DNBR remains in excess of the safety analysis limit value.

15.2.13 Accidental Depressurization of the Main Steam System

15.2.13.1 Identification of Causes and Accident Description

The most severe core conditions resulting from an accidental depressurization of the main steam system are associated with an inadvertent opening of a single steam dump, relief, or safety valve. The analyses, assuming a rupture of a main steam pipe, are discussed in Section 15.4.

The steam released as a consequence of this accident results in an initial increase in steam flow that decreases during the accident as the steam pressure falls. The energy removal from the RCS causes a reduction of coolant temperature and pressure. In the presence of a negative moderator temperature coefficient, the cooldown results in a reduction of core shutdown margin.

The analysis is performed to demonstrate that the following criterion is satisfied: assuming a stuck RCCA and a single failure in the engineered safety features (ESF) the limit DNBR value will be met after reactor trip for a steam release equivalent to the spurious opening, with failure to close, of the largest of any single steam dump, relief, or safety valve.

The following systems provide the necessary mitigation of an accidental depressurization of the main steam system.

(1) Safety injection system (SIS) actuation from any of the following:

   (a) Two-out-of-three low pressurizer pressure signals,

   (b) High differential pressure signals between steam lines,

   (c) Two-out-of-three high-1 containment pressure signals.

(2) The overpower reactor trips (neutron flux and ΔT) and the reactor trip occurring in conjunction with receipt of the safety injection signal.

(3) Redundant isolation of the main feedwater lines: sustained high feedwater flow would cause additional cooldown. Therefore, a safety injection signal will rapidly close all feedwater control valves, trip the main feedwater pumps, and close the feedwater isolation valves.

15.2.13.2 Analysis of Effects and Consequences

The following analyses of a secondary system steam release are performed:

(1) A full plant digital simulation using LOFTRAN[4] to determine RCS temperature and pressure during cooldown.

(2) An analysis to ascertain that the reactor does not exceed the limit DNBR value.

The following conditions are assumed to exist at the time of a secondary system break accident.

(1) EOL shutdown margin at no-load, equilibrium xenon conditions, and with the most reactive assembly stuck in its fully withdrawn position. Operation of RCCA banks during core burnup is restricted in such a way that addition of positive reactivity in a secondary system break accident will not lead to a more adverse condition than the case analyzed.

(2) A negative moderator coefficient corresponding to the EOL rodded core with the most reactive RCCA in the fully withdrawn position. The variation of the coefficient with temperature and pressure is included. The k_{eff} versus temperature curve at 1150 psia corresponding to the negative moderator temperature coefficient plus the Doppler temperature effect used is shown on Figure 15.2.13-1.

(3) Minimum capability for injection of high concentration boric acid solution corresponding to the most restrictive single failure in the safety injection system. The injection curve is shown on Figure 15.2.13-2. This corresponds to the flow delivered by one charging pump delivering its full contents to the cold leg header. No credit has been taken for the low concentration boric acid that must be swept from the safety injection lines downstream of the refueling water storage tank (RWST) isolation valves prior to the delivery of high concentration boric acid (2300 ppm) to the reactor coolant loops.

(4) The case studied is an initial total steam flow of 255 lb/sec at 1100 psia from one steam generator with offsite power available. This is the maximum capacity of any single steam dump or safety valve. Initial hot shutdown conditions at time zero are assumed since this represents the most pessimistic initial condition.

Should the reactor be just critical or operating at power at the time of a steam release, the reactor will be tripped by the normal overpower protection when the power level reaches a trip point. Following a trip at power the RCS contains more stored energy than at no-load, the average coolant temperature is higher than at no-load, and there is appreciable energy stored in the fuel.

Thus, the additional energy stored is removed via the cooldown caused by the steam line break before the no-load conditions of RCS temperature and shutdown margin assumed in the analyses are reached. After the additional stored energy has been removed, the cooldown and reactivity insertions proceed then in the same manner as in the analysis which assumes no-load condition at time zero. However, since the initial steam generator water inventory is greatest at no-load, the magnitude and duration of the RCS cooldown are less for steam line breaks occurring at power.

(5) In computing the steam flow, the Moody Curve for $fL/D = 0$ is used.

(6) Perfect moisture separation in the steam generator is assumed.

15.2.13.3 Results

The results presented are a conservative indication of the events that would occur assuming a secondary system steam release since it is postulated that all of the conditions described above occur simultaneously.

Figures 15.2.13-3 and 15.2.13-4 show the transient arising as the result of a steam release having an initial steam flow of 255 lb/sec at 1100 psia with steam release from one safety valve. The assumed steam release is the maximum capacity of any single steam dump or safety valve. In this case, safety injection is initiated automatically by low pressurizer pressure. Operation of one centrifugal charging pump is considered. Boron solution at 2300 ppm enters the RCS providing sufficient negative reactivity to prevent core damage. The reactivity transient for the case shown on Figure 15.2.13-4 is more severe than that of a failed steam generator safety or relief valve that is terminated by steam line differential pressure, or a failed condenser dump valve that is terminated by low pressurizer pressure and level. The transient is quite conservative with respect to cooldown since no credit is taken for the energy stored in the system metal other than that of the fuel elements or the energy stored in the other steam generators. Since the transient occurs over a period of about 5 minutes, the neglected stored energy is likely to have a significant effect in slowing the cooldown.

15.2.13.4 Conclusions

The analysis has shown that the criteria stated earlier in this section are satisfied. For an accidental depressurization of the main steam system, the DNB design basis is met. This case is less limiting than the rupture of a main steam pipe case presented in Section 15.4.

15.2.14 Spurious Operation of the Safety Injection System at Power

15.2.14.1 Identification of Causes and Accident Description

Spurious SIS operation at power could be caused by operator error or a false electrical actuating signal. A spurious signal in any of the following channels could cause this accident.

(1) High containment pressure,

(2) Low pressurizer pressure,

(3) High steam line differential pressure,

(4) Low steam line pressure,

(5) Manual actuation.

Following the actuation signal, the suction of the coolant charging pumps is diverted from the volume control tank to the refueling water storage tank (RWST). The charging pumps then force highly concentrated (2300 ppm) boric acid solution from the RWST through the header and injection line and into the cold legs of each loop. The safety injection pumps also start automatically but provide no flow when the reactor coolant system (RCS) is at normal pressure. The passive injection system and the low-head system also provide no flow at normal RCS pressure.

A safety injection system (SIS) signal normally results in a reactor trip followed by a turbine trip. However, it cannot be assumed that any single fault that actuates the SIS will also produce a reactor trip. Therefore, two different courses of events are considered.

Case A: Trip occurs at the same time spurious injection starts.

Case B. The reactor protection system produces a trip later in the transient.

For Case A, the operator should determine if the spurious signal was transient or steady state in nature, i.e., an occasional occurrence or a definite fault. The operator will determine this by following approved procedures. In the transient case, the operator would stop the safety injection and bring the plant to the hot shutdown condition. If the SIS must be disabled for repair, boration should continue and the plant brought to cold shutdown.

For Case B, the reactor protection system does not produce an immediate trip and the reactor experiences a negative reactivity excursion due to the injected boron causing a decrease in the reactor power. At beginning of life, the power mismatch causes a drop in T_{avg} and consequent coolant shrinkage, and pressurizer pressure and level drop. Load will decrease due to the effect of reduced steam pressure on load when the turbine throttle valve is fully open. If automatic rod control is used, these effects will be lessened until the rods have moved out of the core. The transient is eventually terminated by the reactor protection system low-pressure trip or by manual trip.

Results at end of life are similar except that moderator feedback effects result in a slower transient. The pressurizer pressure and level increase slowly and the coolant T_{avg} decreases slowly. The transient is eventually terminated by the reactor protection system high pressurizer pressure or high pressurizer level trip or by manual trip.

The time to trip is affected by initial operating conditions including core burnup history that affects initial boron concentration, rate of change of boron concentration, and Doppler and moderator coefficients.

Recovery from this incident for Case B is in the same manner as for Case A. The only difference is the lower T_{avg} and pressure associated with the power imbalance during this transient. The time at which reactor trip occurs is of no concern for this occurrence. At lower loads coolant contraction will be slower resulting in a longer time to trip.

15.2.14.2 Analysis of Effects and Consequences

The spurious operation of the SIS system is analyzed with the LOFTRAN code[4]. The code simulates the neutron kinetics, RCS, pressurizer, pressurizer relief and safety valves, pressurizer spray, steam generator, steam generator safety valves, and the effect of the SIS. The program computes pertinent plant variables including temperatures, pressures, and power level.

Because of the power and temperature reduction during the transient, operating conditions do not approach the core limits. Analyses of several cases show that the results are relatively independent of time to trip.

A typical transient is considered representing conditions at BOL.

This accident is analyzed with the Improved Thermal Design Procedure as described in Reference 5. The assumptions made in the analysis are:

(1) Initial Operating Conditions

The initial reactor power, pressure and RCS temperatures are assumed to be at their nominal values. Uncertainties in initial conditions are included in the limit DNBR as described in Reference 5.

(2) Moderator and Doppler Coefficients of Reactivity

A positive BOL moderator temperature coefficient was used. A low absolute value Doppler power coefficient was assumed.

(3) Reactor Control

The reactor was assumed to be in manual control.

(4) Pressurizer Heaters

Pressurizer heaters were assumed to be inoperative in order to increase the rate of pressure drop.

(5) Boron Injection

At time zero, two charging pumps inject 2300 ppm borated water into the cold legs of each loop.

(6) Turbine Load

Turbine load was assumed constant until the governor drives the throttle valve wide open. Then turbine load drops as steam pressure drops.

(7) Reactor Trip

Reactor trip was initiated by low pressurizer pressure. The trip was conservatively assumed to be delayed until the pressure reached 1775 psia.

15.2.14.3 Results

The transient response for the minimum feedback case is shown on Figures 15.2.14-1 through 15.2.14-2. Nuclear power starts decreasing immediately due to boron injection, but steam flow does not decrease until 25 seconds into the transient when the turbine throttle valve goes wide open. The mismatch between load and nuclear power causes T_{avg}, pressurizer water level, and pressurizer pressure to drop. The low-pressure trip setpoint is reached at 54 seconds and rods start moving into the core at 56 seconds.

15.2.14.4 Conclusions

Results of the analysis show that spurious safety injection with or without immediate reactor trip presents no hazard to the integrity of the RCS.

DNBR is never less than the initial value. Thus, there will be no cladding damage and no release of fission products to the reactor coolant system.

If the reactor does not trip immediately, the low-pressure reactor trip will be actuated. This trips the turbine and prevents excess cooldown thereby expediting recovery from the incident.

15.2.15 References

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2. Risher, D. H. Jr. and Barry, R. F., "TWINKLE-A Multi-Dimensional Neutron Kinetics Computer Code," WCAP-7979-P-A (Proprietary) and WCAP-8028-A (Non-proprietary), January 1975.

3. H. G. Hargrove, FACTRAN - A Fortran IV Code for Thermal Transients in A UO_2 Fuel Rod, WCAP-7908, June 1972.

4. T. W. T. Burnett, et al., LOFTRAN Code Description, WCAP-7907-P-A (Proprietary), WCAP-7907-A (Non-proprietary), April 1984.

5. Chelemer, H., et al., "Improved Thermal Design Procedure," WCAP-8567 (Proprietary) and WCAP-8568 (Non-proprietary), July 1975.

6. Technical Specifications, V. C. Summer Nuclear Station Appendix A to License No. NPF-12, as amended through Amendment Number 66.

7. Chelemer, H. et al, Subchannel Thermal Analysis of Rod Bundle Cores, WCAP-7015, Rev. 1, January 1969.

8. M. A. Mangan, Overpressure Protection for Westinghouse Pressurized Water Reactor, WCAP-7769, October 1971.

9. J. S. Shefcheck, Application of the THINC Program to PWR Design, WCAP-7359-L, August 1969 (Proprietary) and WCAP-7838, January 1972.

10. Morita, T., et. al., "Dropped Rod Methodology for Negative Flux Rate Trip Plant," WCAP-10297-P-A (Proprietary) and WCAP-10298-A (Non-proprietary), June 1983.

TABLE 15.2-1

TIME SEQUENCE OF EVENTS FOR CONDITION II EVENTS

Accident	Event	Time, sec
Uncontrolled RCCA Withdrawal from a Subcritical Condition	Initiation of uncontrolled rod withdrawal 9.0×10^{-4} Δk/sec reactivity insertion rate from 10^{-9} of nominal power	0.0
	Power range high neutron flux low setpoint reached	8.8
	Peak nuclear power occurs	8.9
	Rods begin to fall into core	9.3
	Peak heat flux occurs	11.8
	Peak hot spot average clad temperature occurs	11.8
	Peak hot spot average fuel temperature occurs	12.1

Accident	Event	Time, sec
Uncontrolled RCCA Withdrawal at Power		
1. Case A	Initiation of uncontrolled RCCA withdrawal at a high reactivity insertion rate (7.5×10^{-4} Δk/sec)	0.0
	Power range high neutron flux high trip setpoint reached	1.5
	Rods begin to fall into core	2.0
	Minimum DNBR occurs	2.8
2. Case B	Initiation of uncontrolled RCCA withdrawl at a small reactivity insertion rate (5.0×10^{-5} Δk/sec)	
	Overtemperature ΔT reactor trip signal initiated	19.6
	Rods begin to fall into core	20.1
	Minimum DNBR occurs	20.7

Accident	Event	Time, sec
Uncontrolled Boron Dilution		
1. Dilution during refueling	Dilution begins	0
	Operator receives high flux at shutdown alarm, set at twice background	1791
	Operator isolates source of dilution; minimum margin to criticality occurs	4680
2. Dilution during cold shutdown	Dilution begins	0
	Operator receives high flux at shutdown alarm set at twice background	1358
	Operator isolates source of dilution; shutdown margin is lost	2181
3. Dilution during hot standby	Dilution begins	0
	Operator receives high flux at shutdown alarm, set at twice background	1405

Accident	Event	Time, sec
	Operator isolates source of dilution; shutdown margin is lost	2273
4. Dilution during startup	Power Range-low setpoint Reactor trip due to dilution	0
	Shutdown margin lost (if dilution continues after trip)	1236
5. Dilution during full power operation		
a. Automatic reactor control	Operator receives lo-lo rod insertion limit alarm due to dilution	0
	Shutdown margin lost	1380
b. Manual reactor control	Overtemperature ΔT reactor trip due to dilution	0
	Shutdown margin lost (if dilution continues after trip)	1140

Accident	Event	Time, sec
Partial Loss of Forced Reactor Coolant Flow		
All loops operating, one pump coasting down		
	Coastdown begins	0.0
	Low-flow reactor trip	1.49
	Rods begin to drop	2.49
	Minimum DNBR occurs	3.4

Accident	Event	Time, sec
Startup of an Inactive Reactor Coolant Loop	Initiation of pump startup	0.0
	Power reaches high nuclear flux trip	3.1
	Rods begin to drop	3.6
	Minimum DNBR occurs	4.2

Accident	Event	Time, sec
Loss of External Electrical Load		
1. With pressurizer control (BOL)	Loss of electrical load	0.0
	Overtemperature ΔT	6.6
	Rods begin to drop	8.1
	Minimum DNBR occurs	9.0
	Initiation of steam release from steam generator safety valves	10.0
	Peak pressurizer pressure occurs	10.6
2. With pressurizer control (EOL)	Loss of electrical load	0.0
	Overtemperature ΔT trip setpoint reached	6.9
	Rods begin to drop	8.4
	Peak pressurizer pressure occurs	8.2

Accident	Event	Time, sec
	Initiation of steam release from steam generator safety valves	9.8
	Minimum DNBR occurs	(a)
3. Without pressurizer control (BOL)	Loss of electrical load	0.0
	High pressurizer pressure reactor trip setpoint reached	5.2
	Rods begin to drop	7.2
	Peak pressurizer pressure occurs	8.8
	Initiation of steam release from steam generator safety valves	10.2
	Minimum DNBR occurs	(a)

Accident	Event	Time, sec
4. Without pressurizer control (EOL)	Loss of electrical load	0.0
	High pressurizer pressure reactor trip setpoint reached	5.4
	Rods begin to drop	7.4
	Peak pressurizer pressure occurs	8.2
	Initiation of steam release from steam generator safety valves	10.0
	Minimum DNBR occurs	(a)

Accident	Event	Time, sec W/Power	Time, sec W/O Power
Loss of Normal Feedwater and Loss of Offsite Power to the Station Auxiliaries (Station Blackout)	Main feedwater flow stops	10	10
	Low-low steam generator water level reactor trip	56.3	56.3
	Rods begin to drop	58.3	58.3
	Reactor coolant pumps begin to coast down	-	60.3
	Peak water level in pressurizer occurs	62	62
	Two steam generators begin to receive emergency feedwater from one motor driven emergency feedwater pump	116.3	116.3
	Cold emergency feedwater is delivered to the steam generators	128	128
	Core decay heat plus pump heat decreases to emergency feedwater heat removal capacity	~3600	~1200

Accident	Event	Time, sec
Excessive Feedwater Flow at Full Load	One main feedwater control valve fails fully open	0
	High-high steam generator level signal generated	27.6
	Turbine trip occurs due to high-high steam generator level	29.6
	Minimum DNBR occurs	30.0
	Reactor trip due to turbine trip[b]	31.6
	Feedwater isolation valves fully closed	40.6

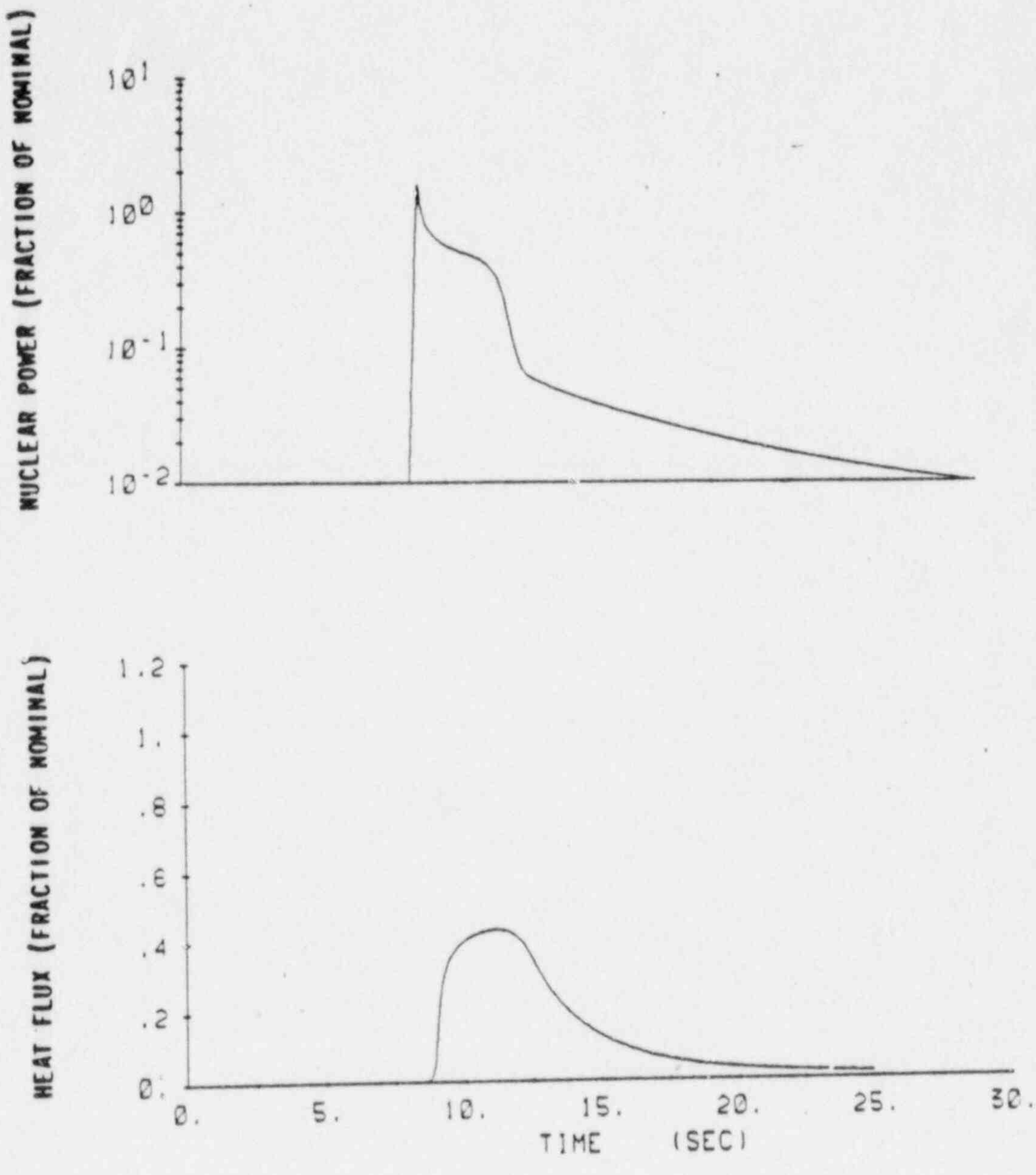
Accident	Event	Time, sec
Excessive Load Increase		
1. Manual reactor control (BOL minimum moderator feedback)	10% step load increase	0.0
	Equilibrium conditions reached (approximate times only)	300
2. Manual reactor control (EOL maximum moderator feedback)	10% step load increase	0.0
	Equilibrium conditions reached (approximate times only)	100
3. Automatic reactor control (BOL minimum moderator feedback)	10% step load increase	0.0
	Equilibrium conditions reached (approximate times only)	200

Accident	Event	Time, sec
4. Automatic reactor control (EOL maximum moderator feedback)	10% step load increase	0.0
	Equilibrium conditions reached (approximate times only)	100
Accidental Depressurization of the Reactor Coolant System	Inadvertent opening of one RCS safety valve	0.0
	Overtemperature ΔT Trip Setpoint Reached	22.8
	Rods begin to drop	24.3
	Minimum DNBR occurs	24.8
Accidental Depressurization of the Main Steam System	Inadvertent opening of one main steam safety or relief valve	0.0
	Pressurizer empties	193
	Boron from the RWST reaches RCS loops	262

Accident	Event	Time, sec
Inadvertent Operation of ECCS During Power Operation	Charging pumps begin injecting borated water	0.0
	Low-pressure trip setpoint reached	54
	Rods begin to drop	56

(a) DNBR does not decrease below its initial value.

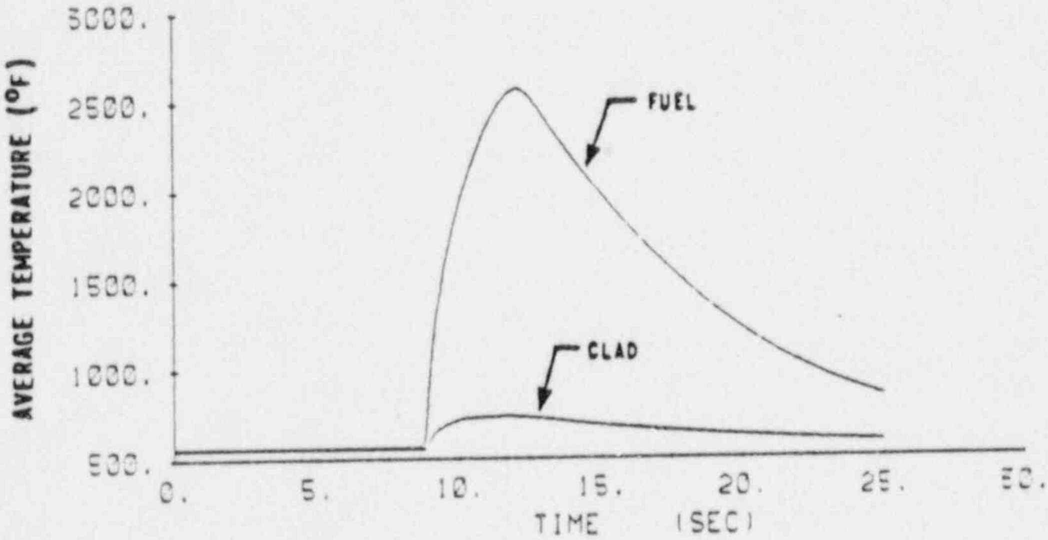
(b) Not a required safety function.



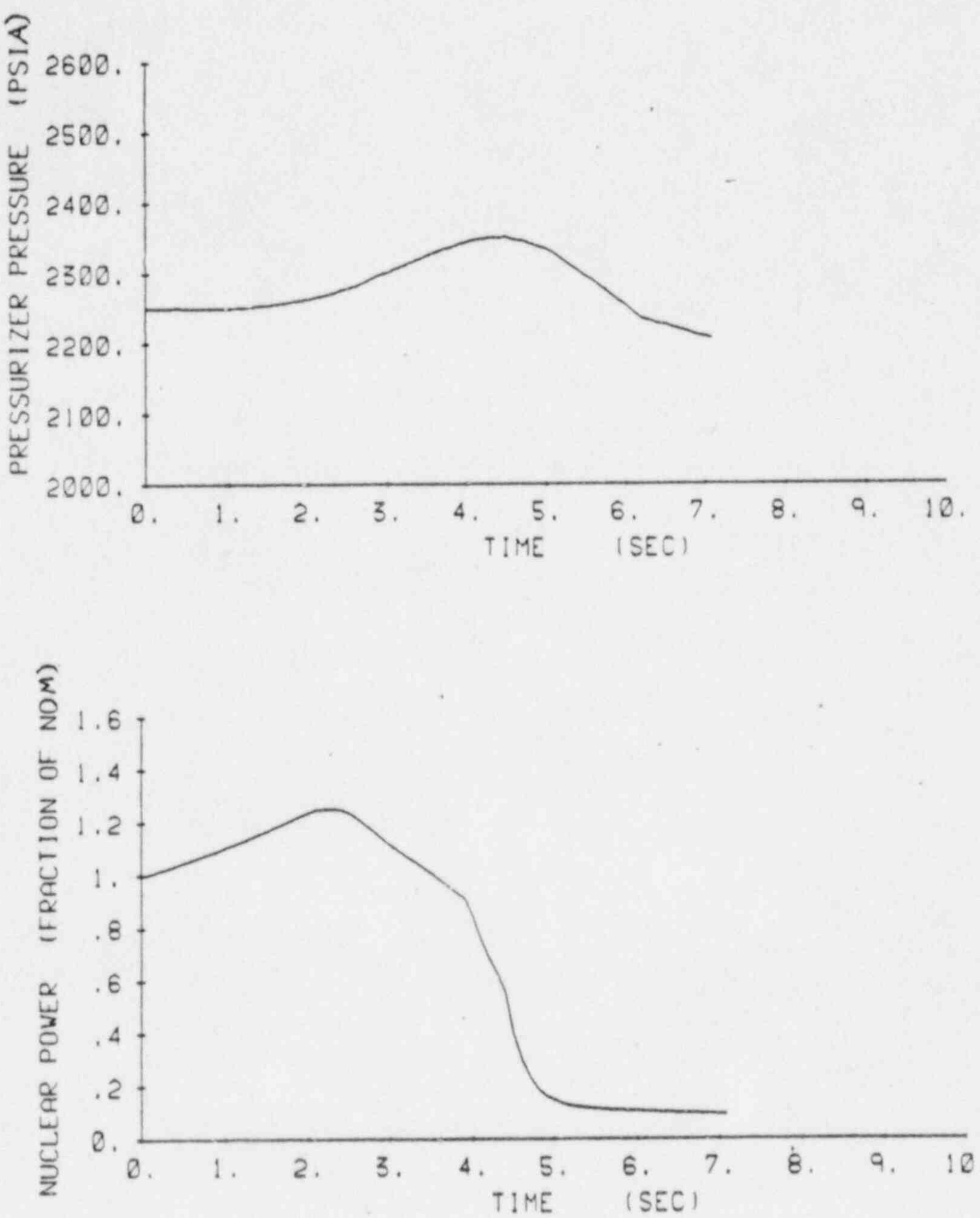
V. C. Summer

Figure 15.2.1-1

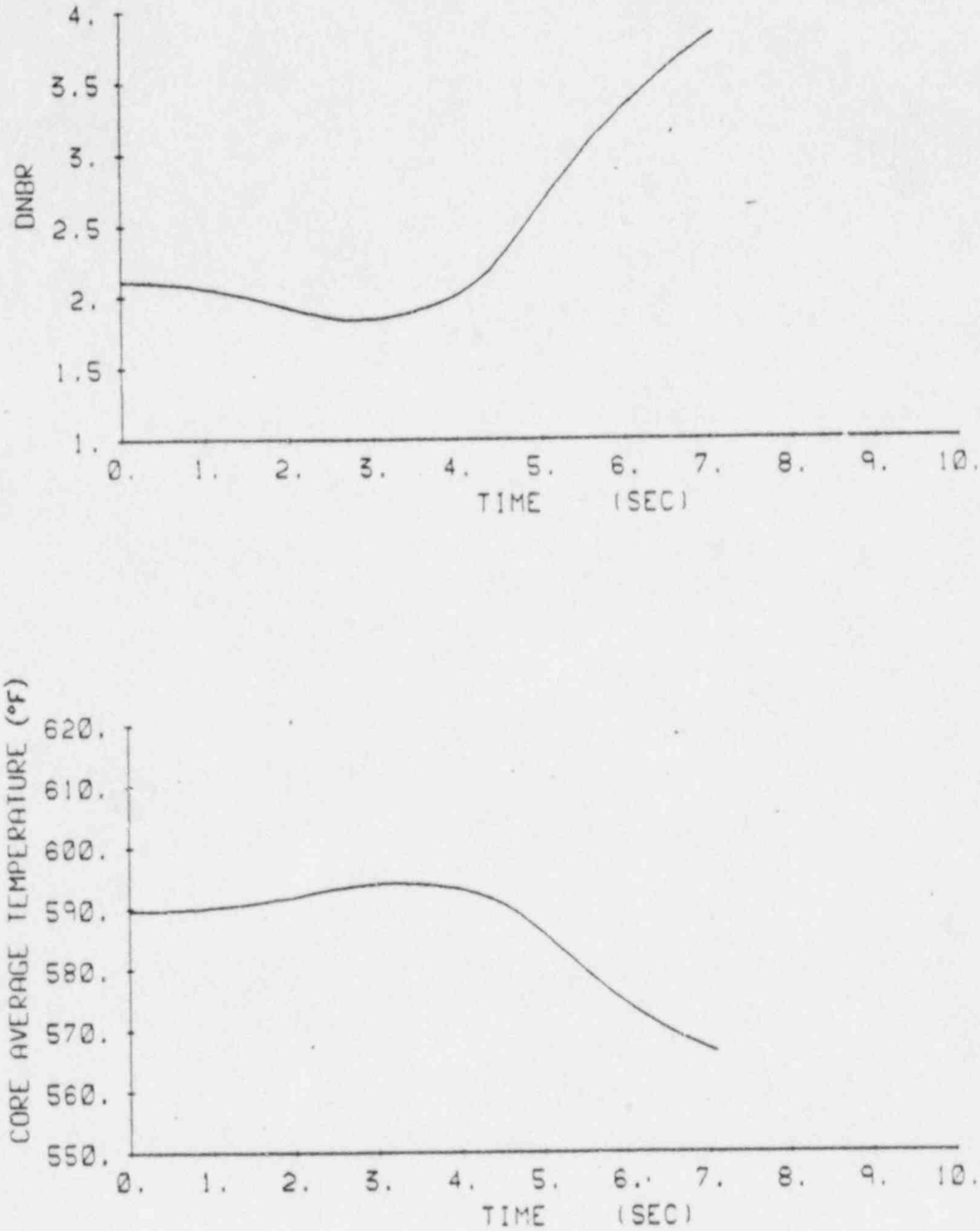
Uncontrolled Rod Withdrawal From A Subcritical Condition Nuclear Power and Core Heat Flux vs. Time



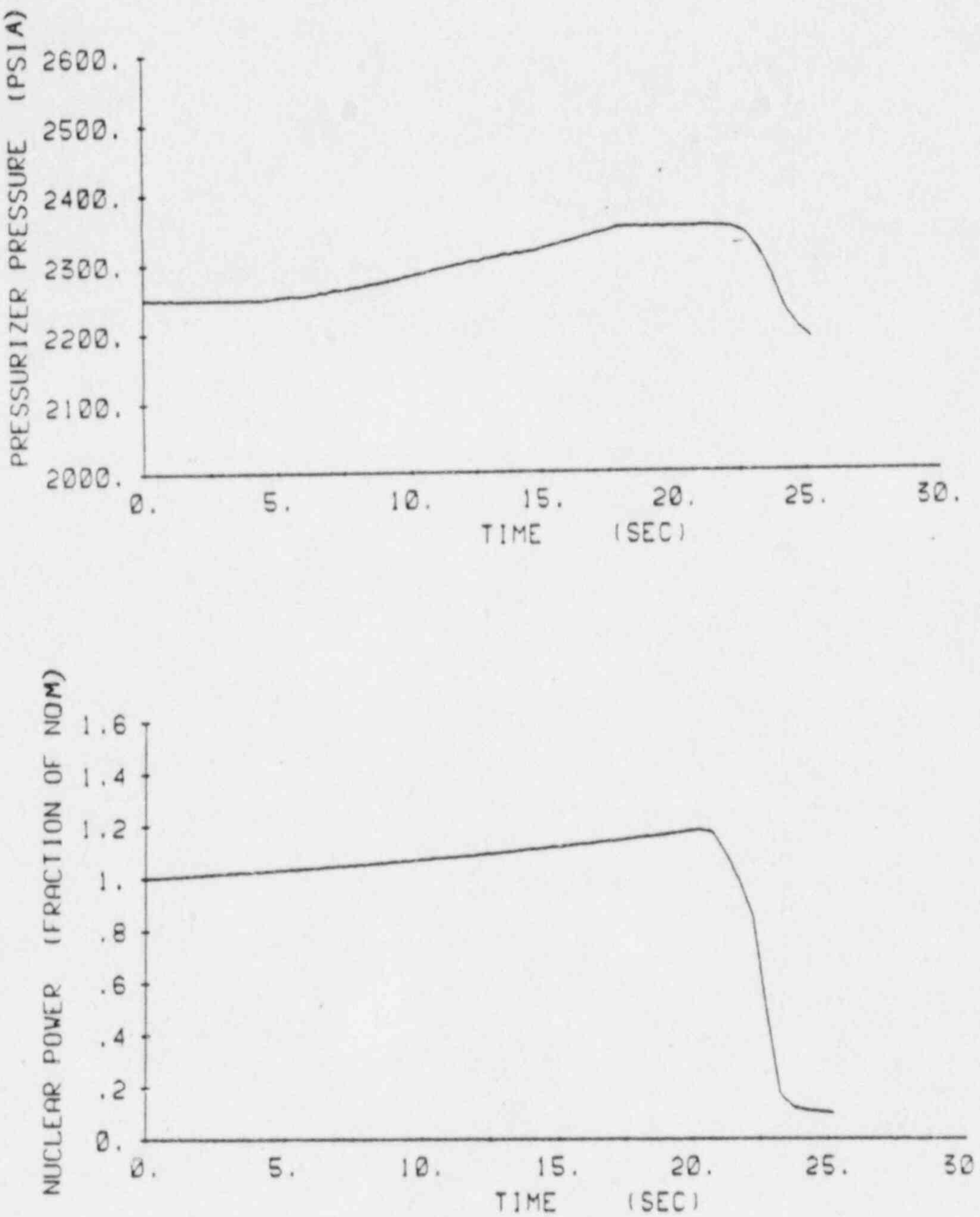
V. C. Summer
Figure 15.2.1-2
Uncontrolled Rod Withdrawal From A Subcritical Condition Hot Spot Fuel Average and Clad Temperature vs. Time



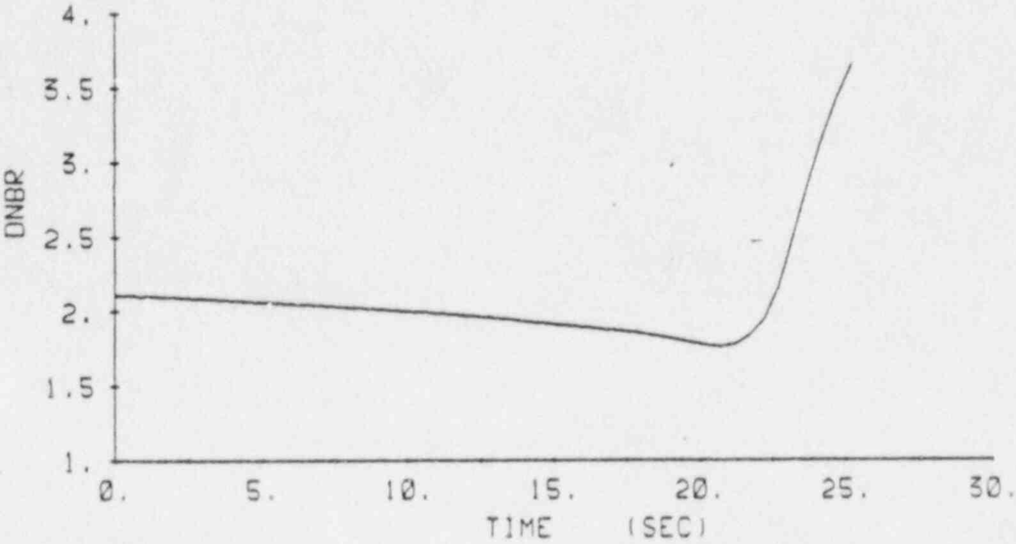
V. C. Summer
Figure 15.2.2-1
Uncontrolled Rod Withdrawal From 100% Power Terminated by High Neutron Flux Trip Pressurizer Pressure and Nuclear Power vs. Time

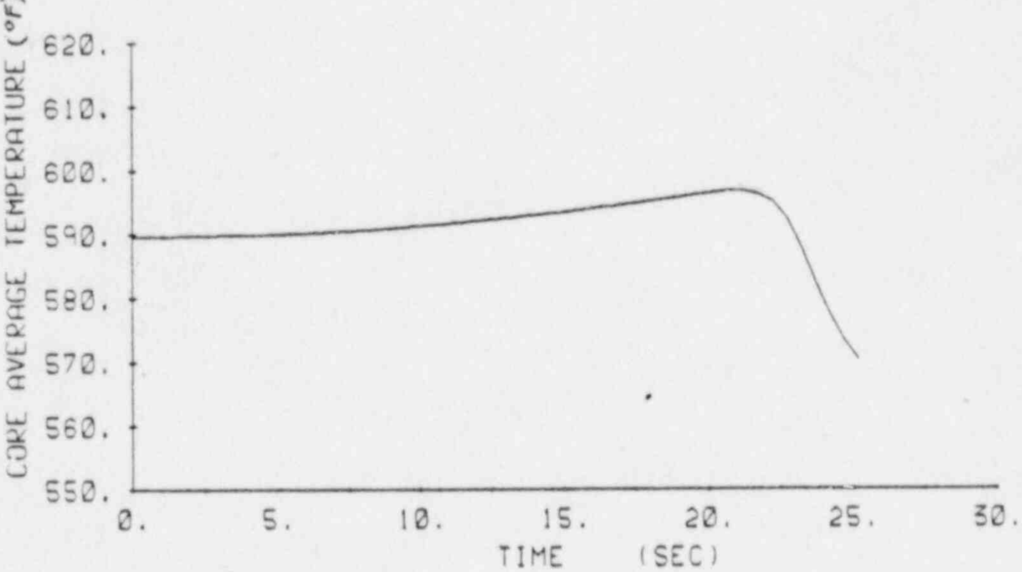


V. C. Summer
Figure 15.2.2-2
Uncontrolled Rod Withdrawal From 100% Power Terminated by High Neutron Flux Trip DNBR and Tavg vs. Time

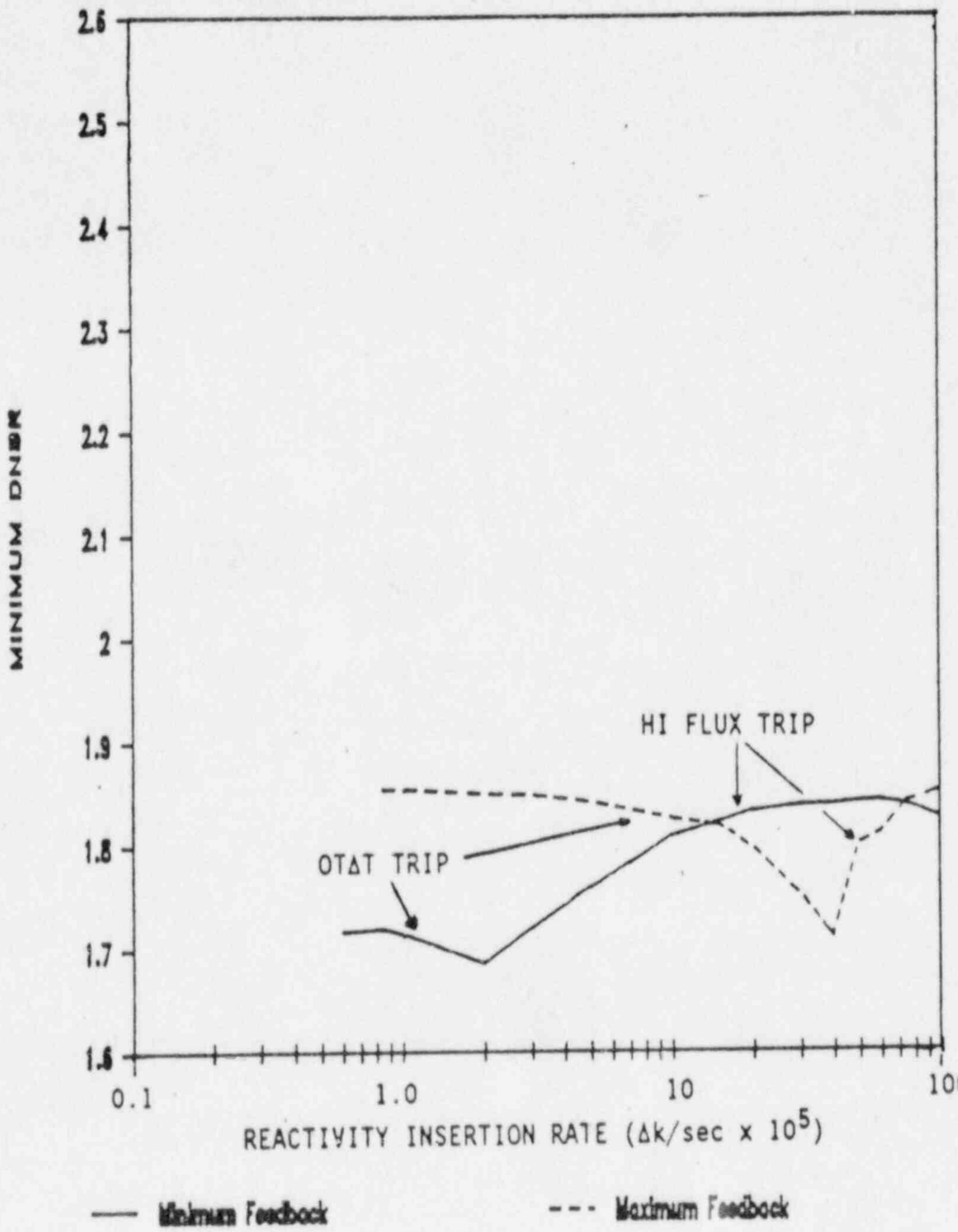


V. C. Summer
Figure 15.2.2-3
Uncontrolled Rod Withdrawal From 100% Power Terminated by Overtemperature Delta-T Pressurizer Pressure and Nuclear Power vs. Time

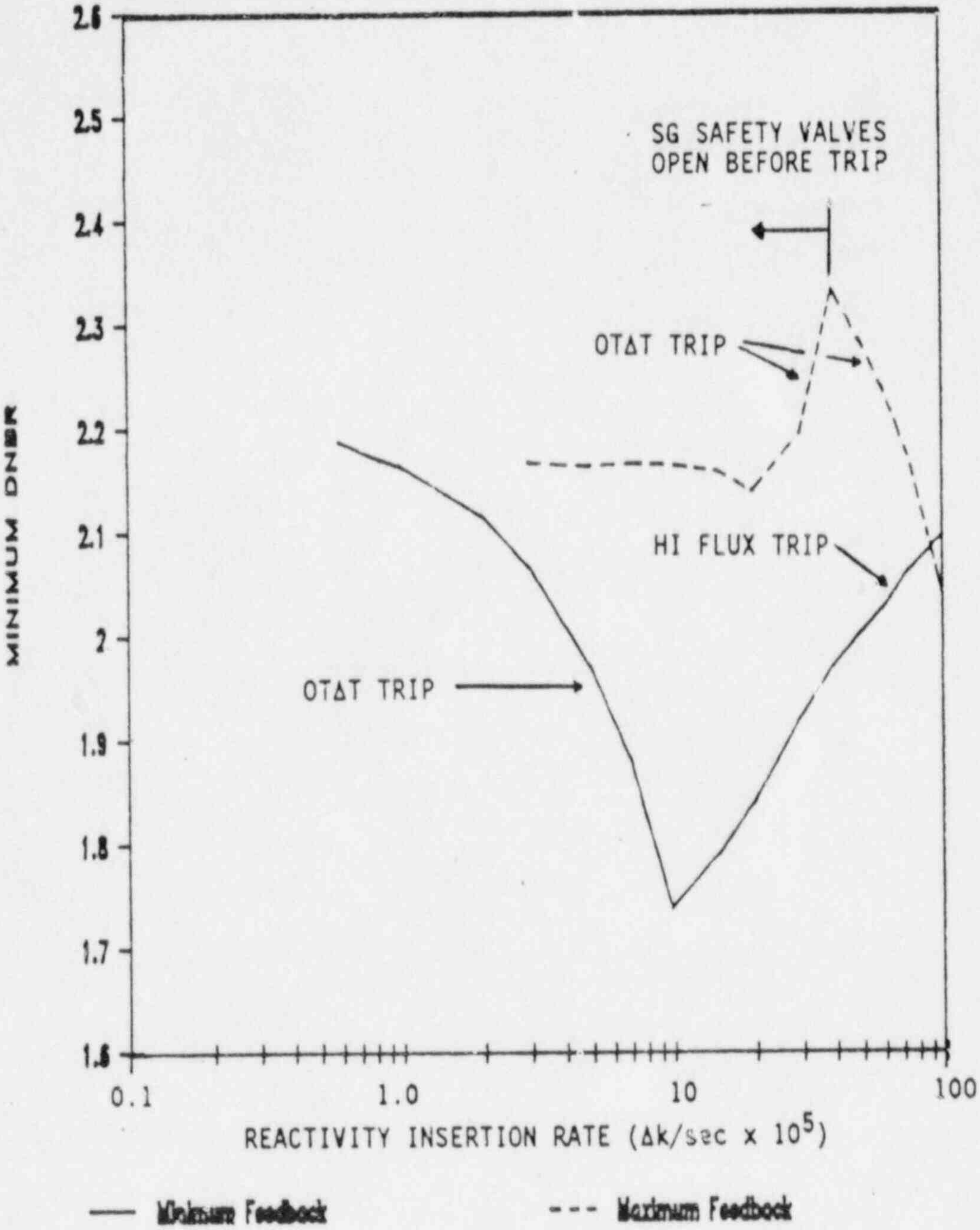




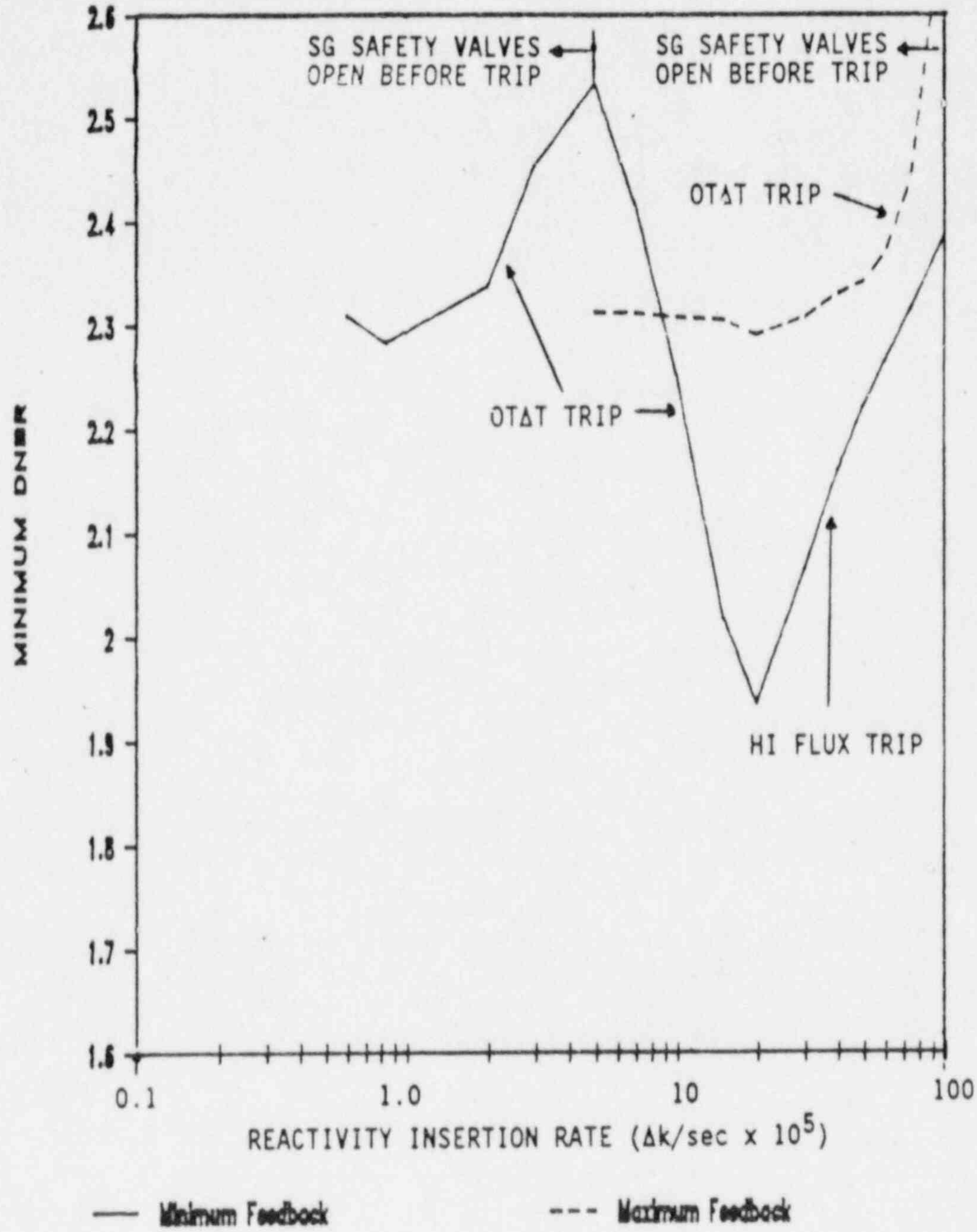
V. C. Summer
Figure 15.2.2-4
Uncontrolled Rod Withdrawal From 100% Power Terminated by Overtemperature Delta-T DNBR and Tavg vs. Time



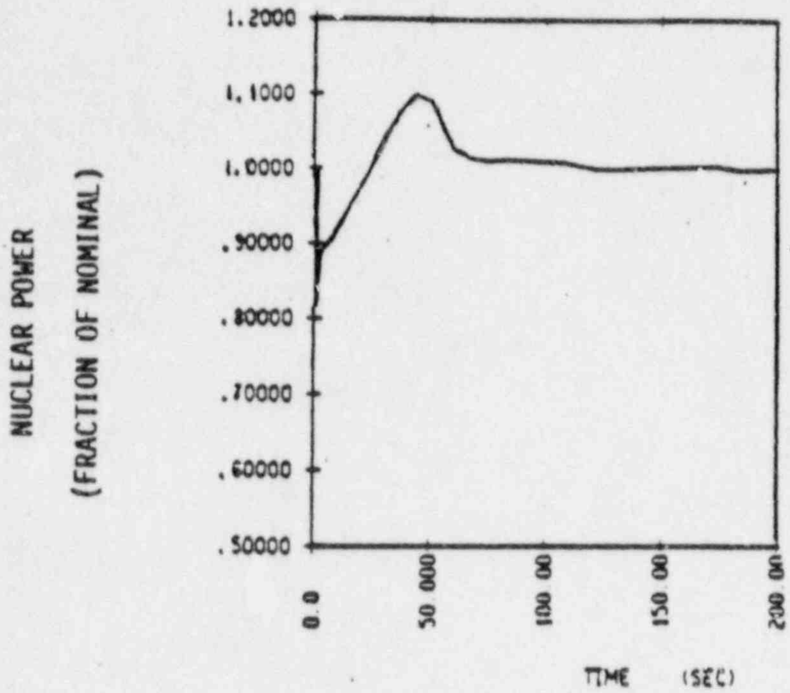
V. C. Summer
Figure 15.2.2-5
Effect of Reactivity Insertion Rate on Minimum DNBR For a Rod Withdrawal Accident at 100% Power

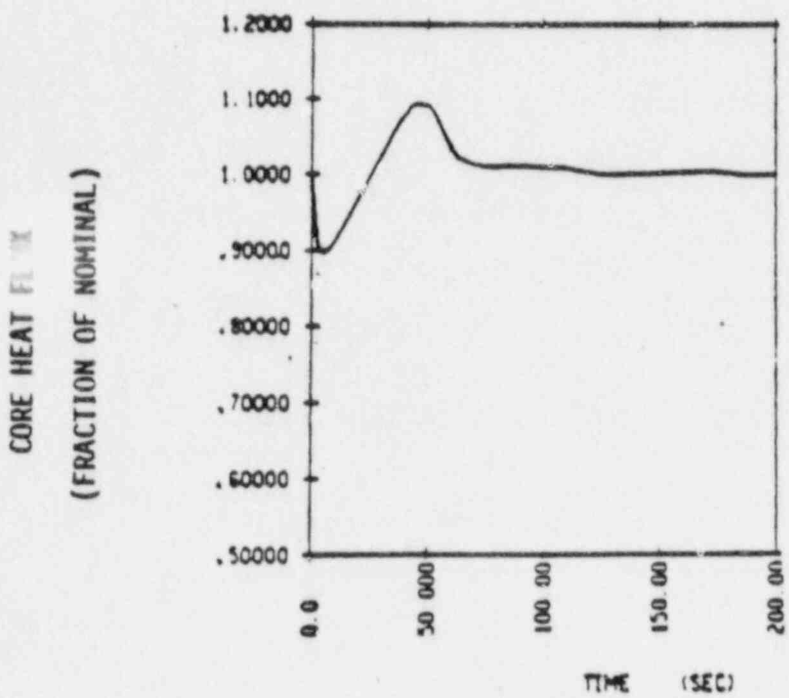


V. C. Summer
Figure 15.2.2-6
Effect of Reactivity Insertion Rate on Minimum DNBR For a Rod Withdrawal Accident at 60% Power

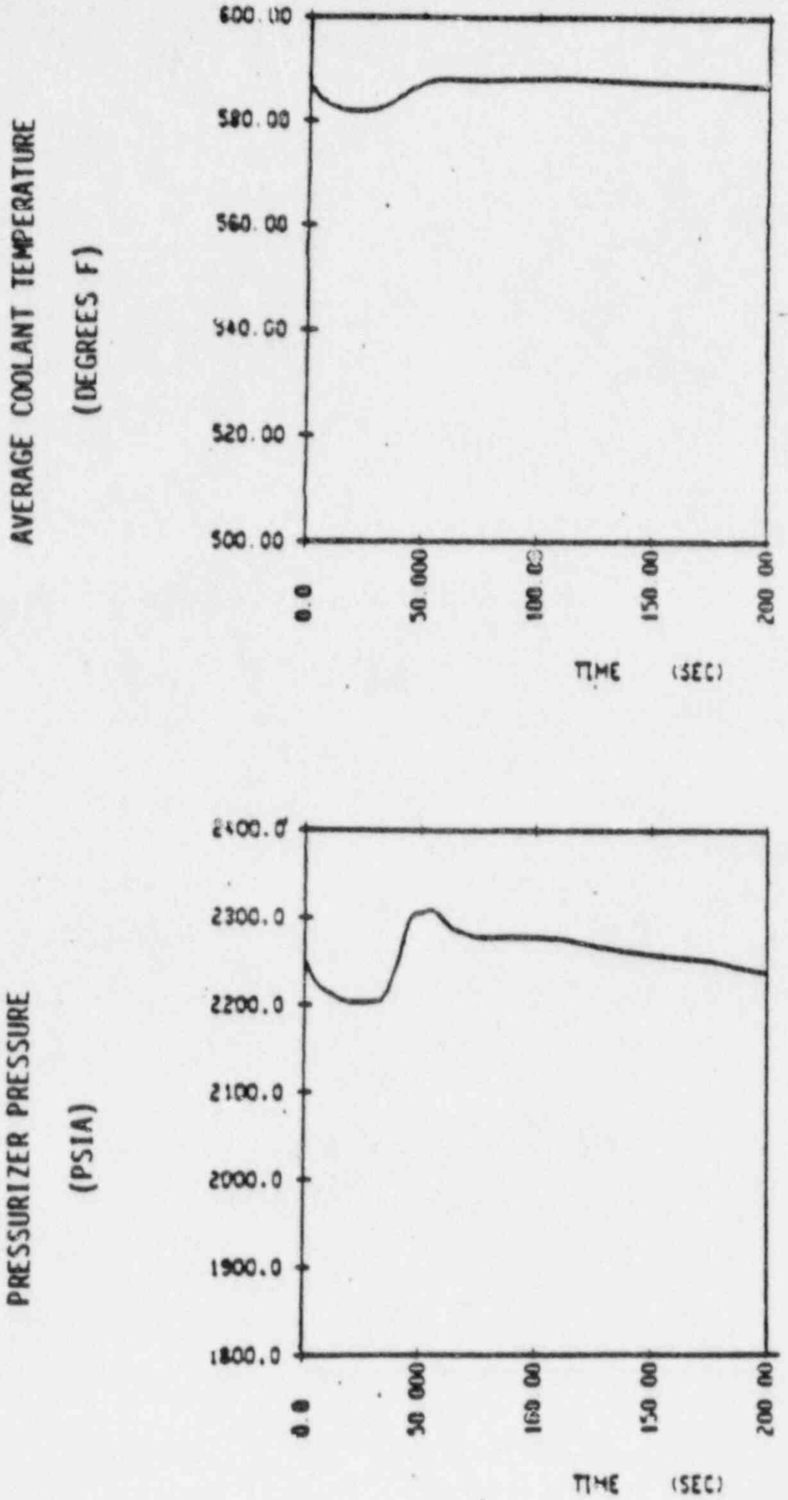


V. C. Summer
Figure 15.2.2-7
Effect of Reactivity Insertion Rate on Minimum DNBR For a Rod Withdrawal Accident at 10% Power

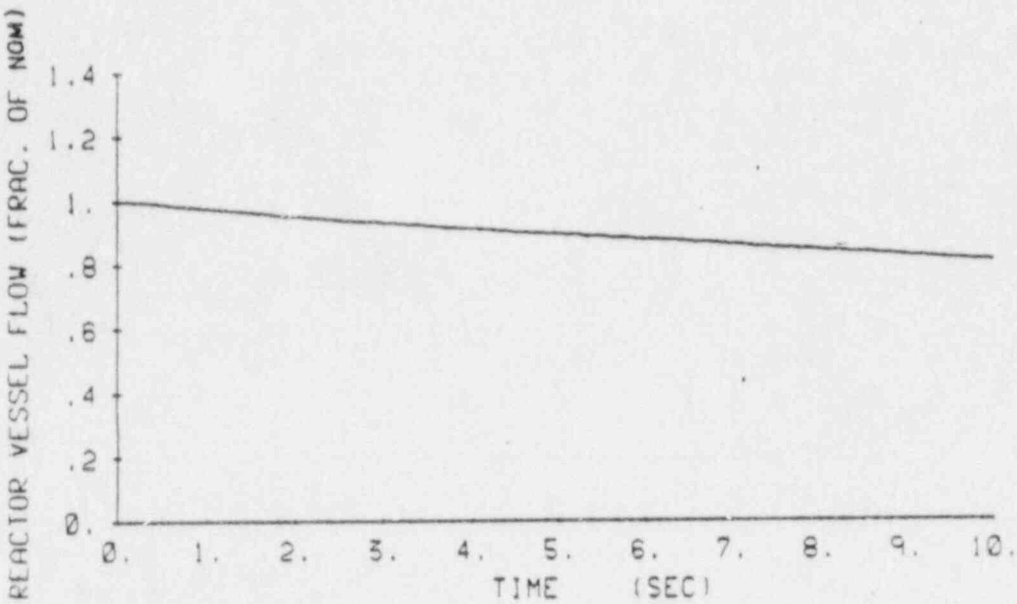


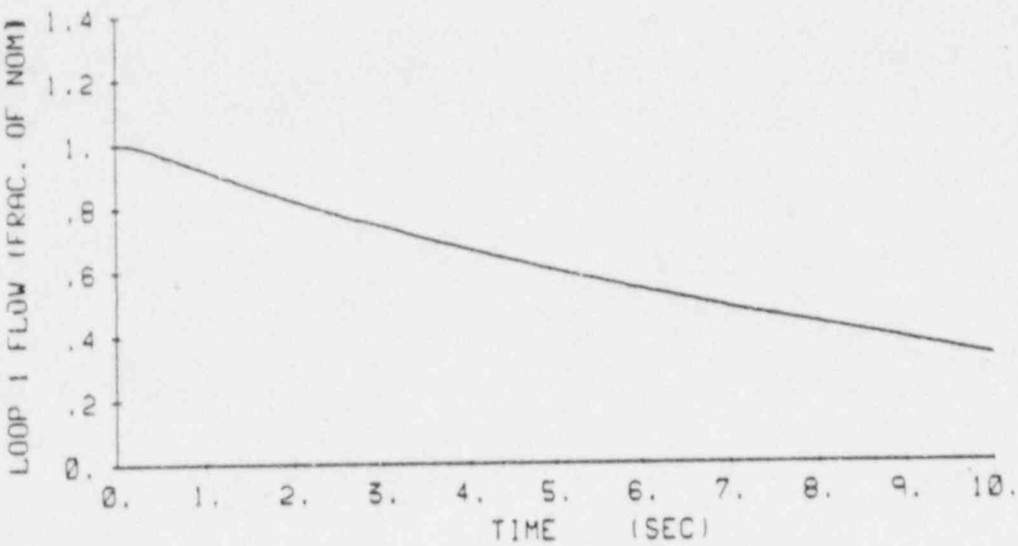


V. C. Summer
Figure 15.2.3-1
Transient Response To A Dropped RCCA Nuclear Power and Heat Flux vs. Time



V. C. Summer
Figure 15.2.3-2
Transient Response To A Dropped RCCA Tavg and Pressurizer Press. vs. Time

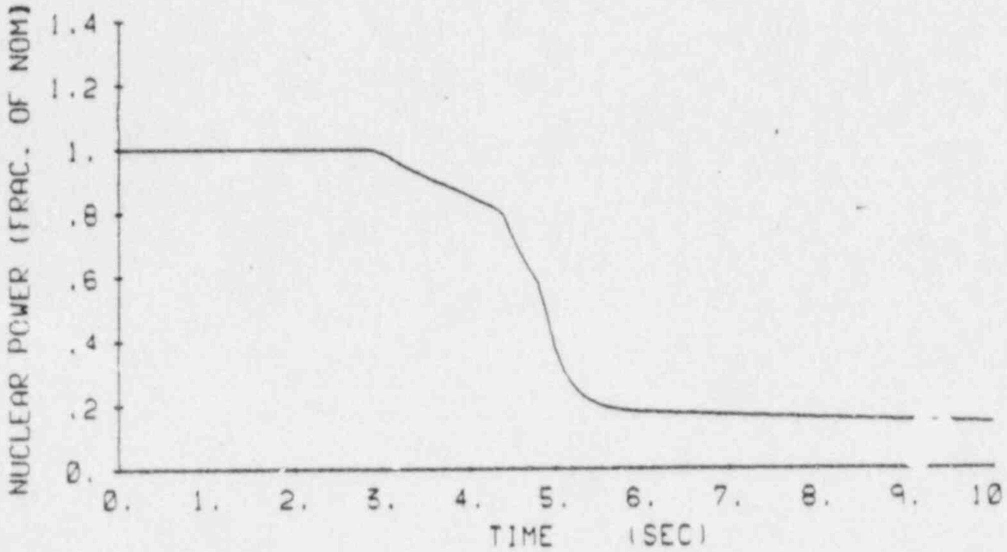


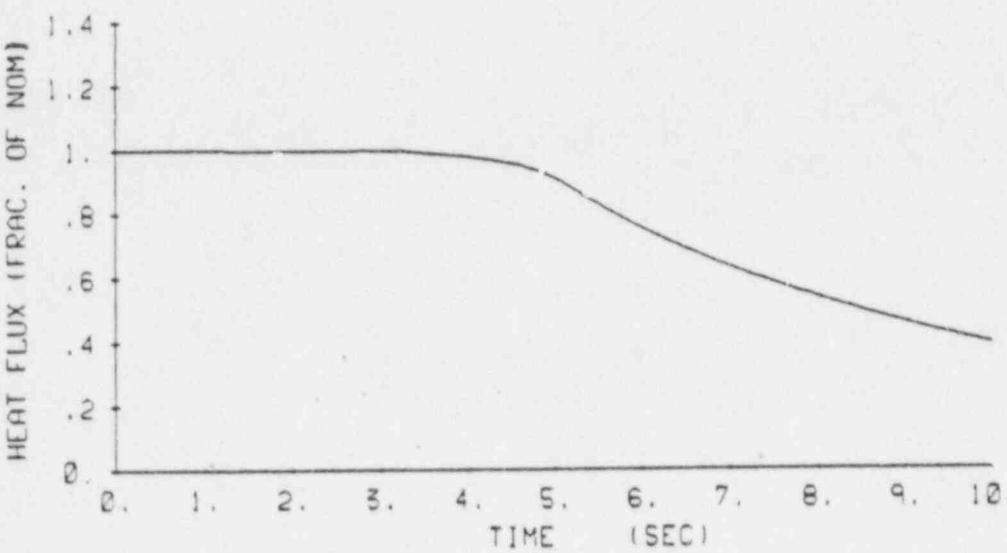


V. C. Summer

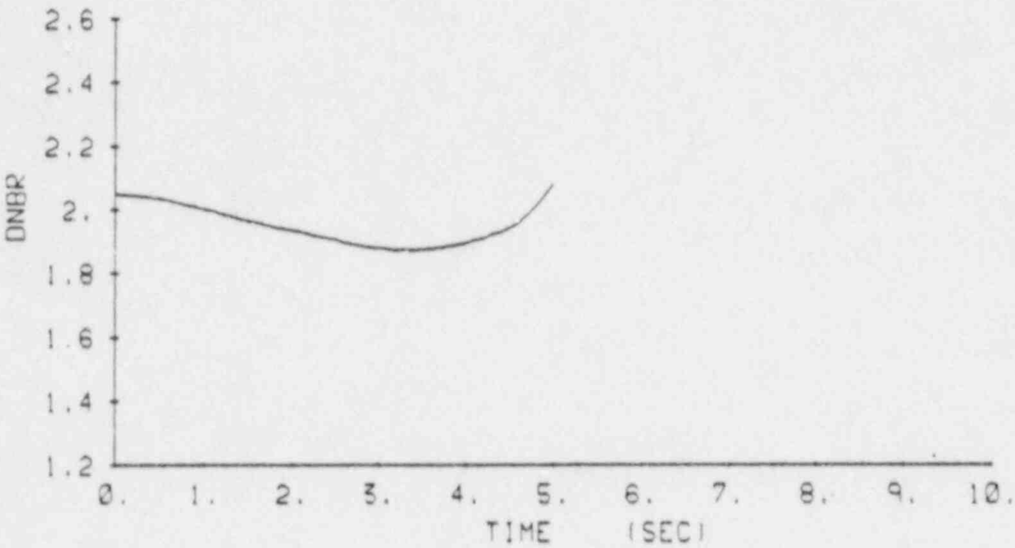
Figure 15.2.5-1

All Loops Operating
One Loop Coasting Down
Vessel Flow and Faulted
Loop Flow vs. Time

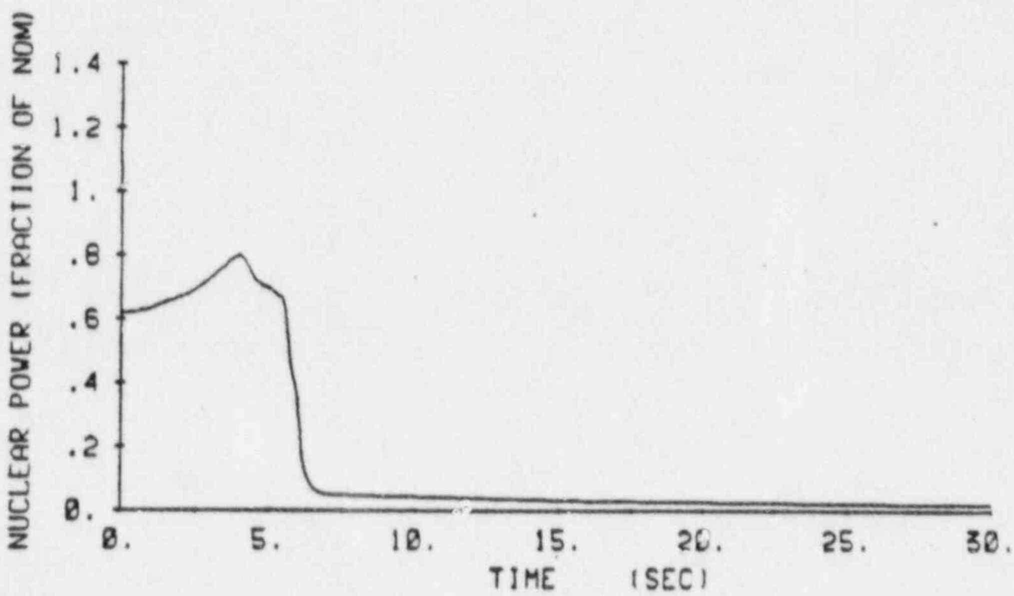




V. C. Summer
Figure 15.2.5-2
All Loops Operating One Loop Coasting Down Nuclear Power and Heat Flux vs. Time



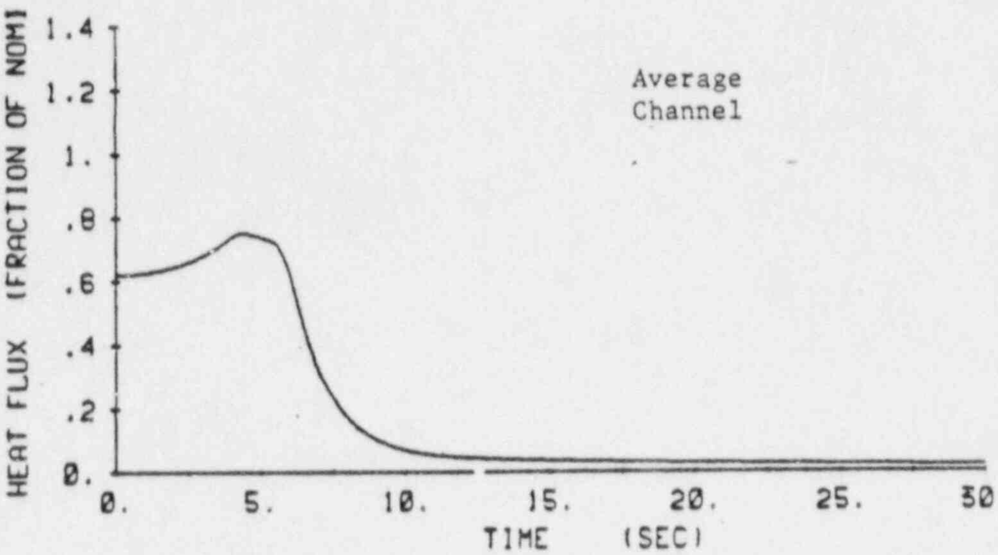
V. C. Summer
Figure 15.2.5-3
All Loops Operating One Loop Coasting Down DNBR vs. Time

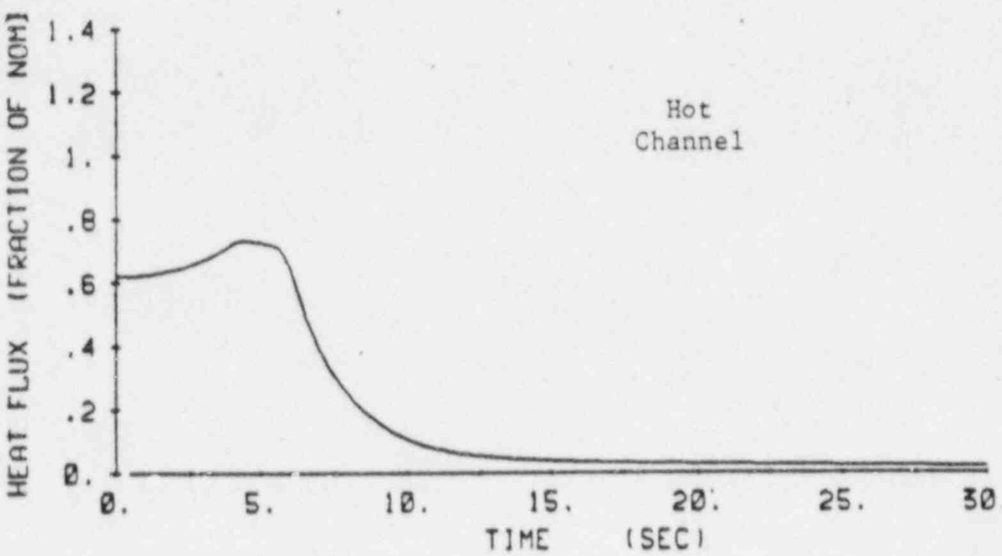


V. C. Summer

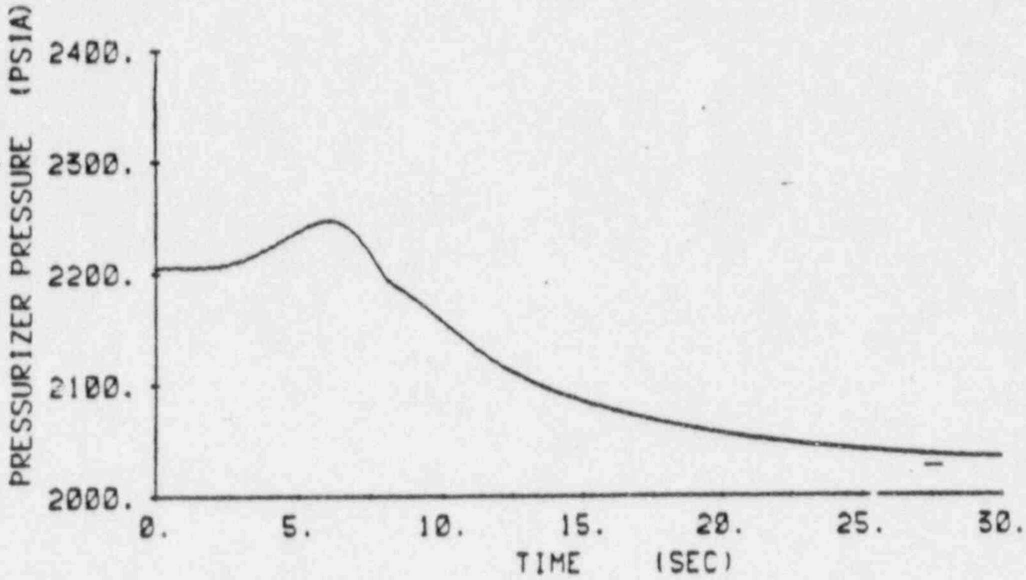
Figure 15.2.6-1

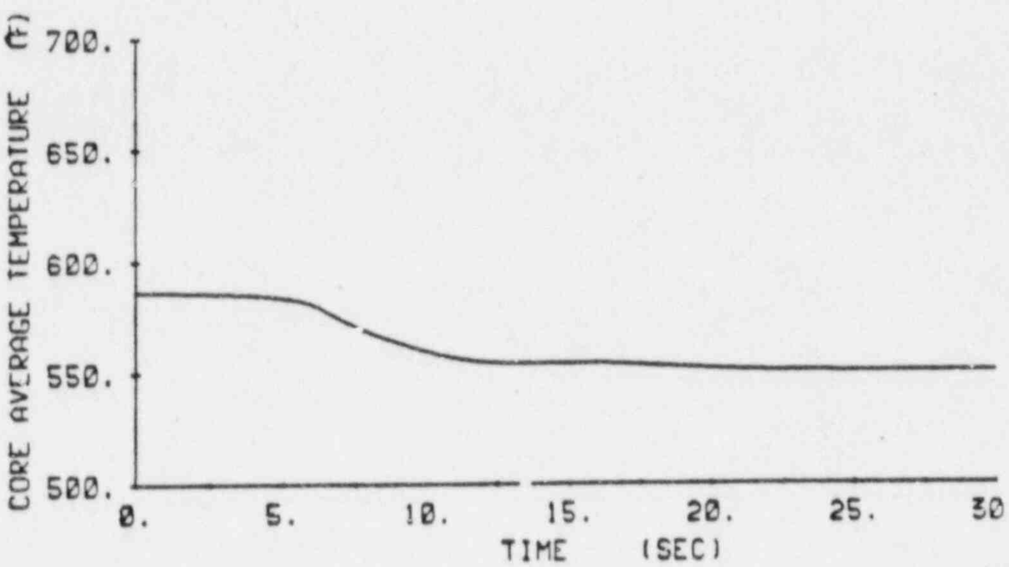
Startup of an Inactive Loop
Nuclear Power vs.Time



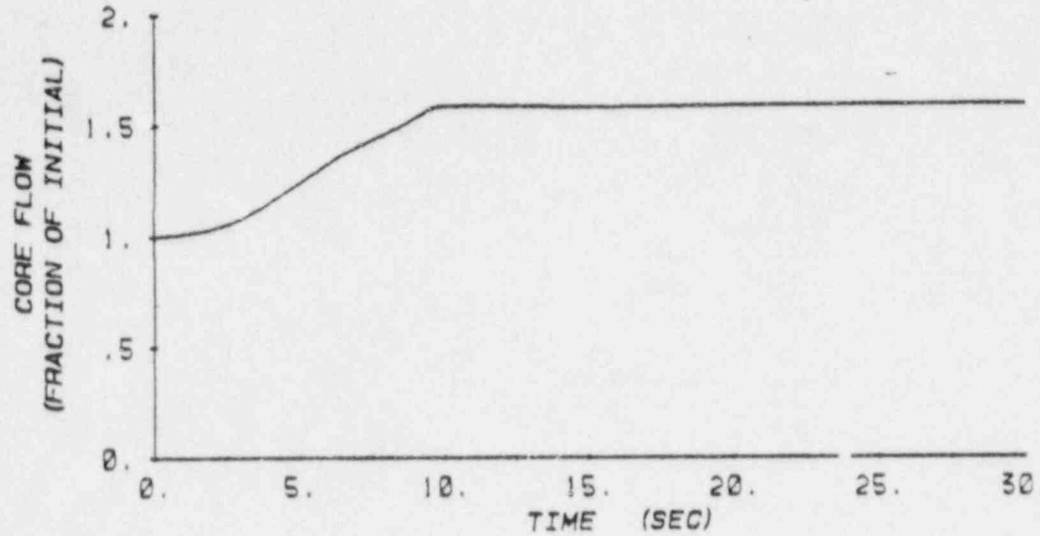


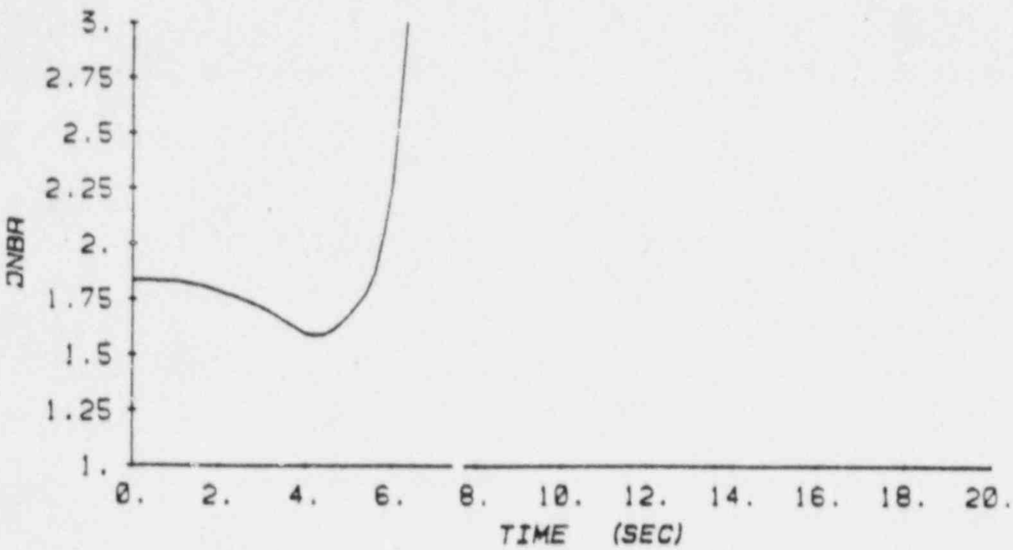
V. C. Summer
Figure 15.2.6-2
Startup of an Inactive Loop Average and Hot Channel Heat Flux vs. Time





V. C. Summer
Figure 15.2.6-3
Startup of an Inactive Loop Pressurizer Press. and Core Tavg vs.Time

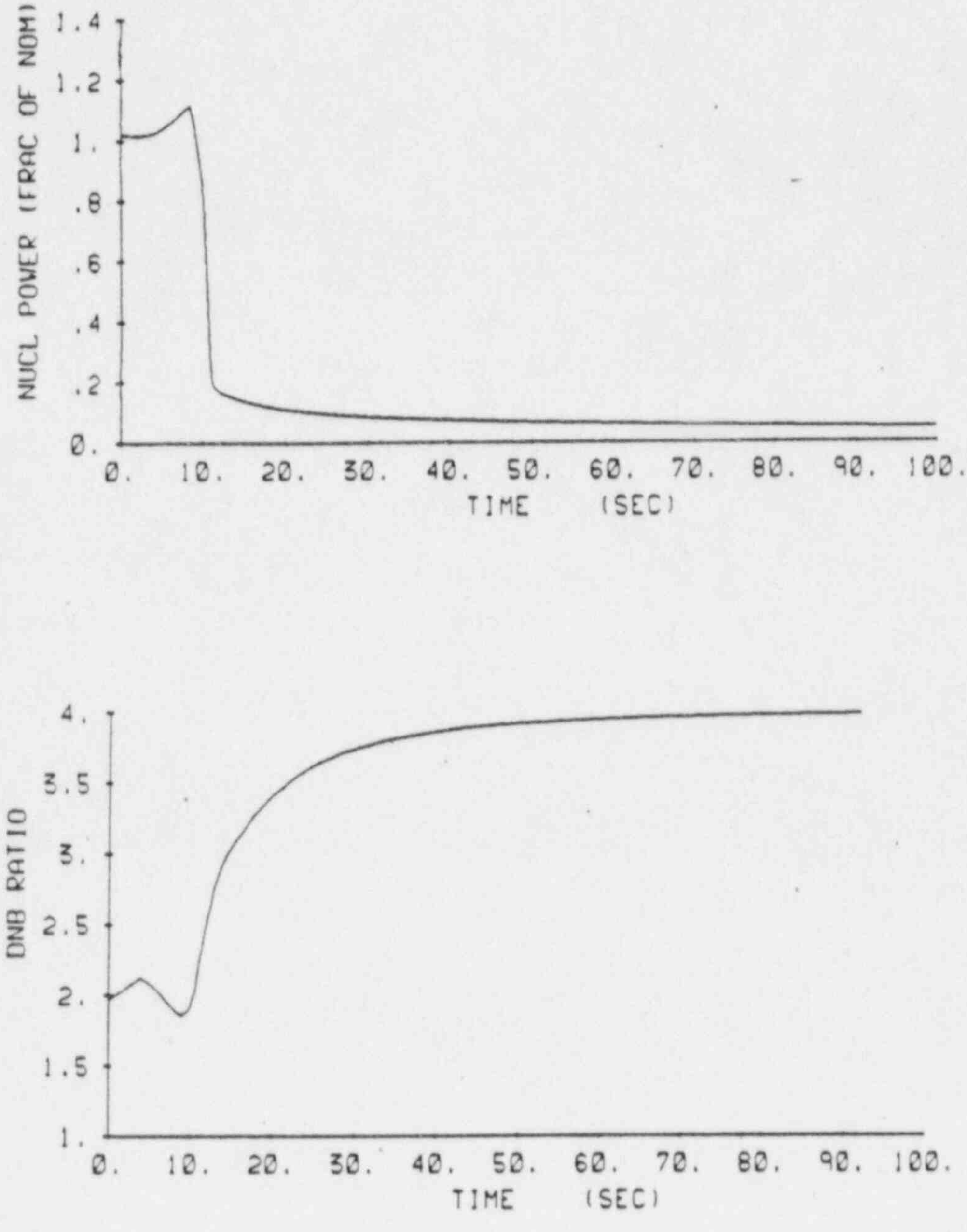




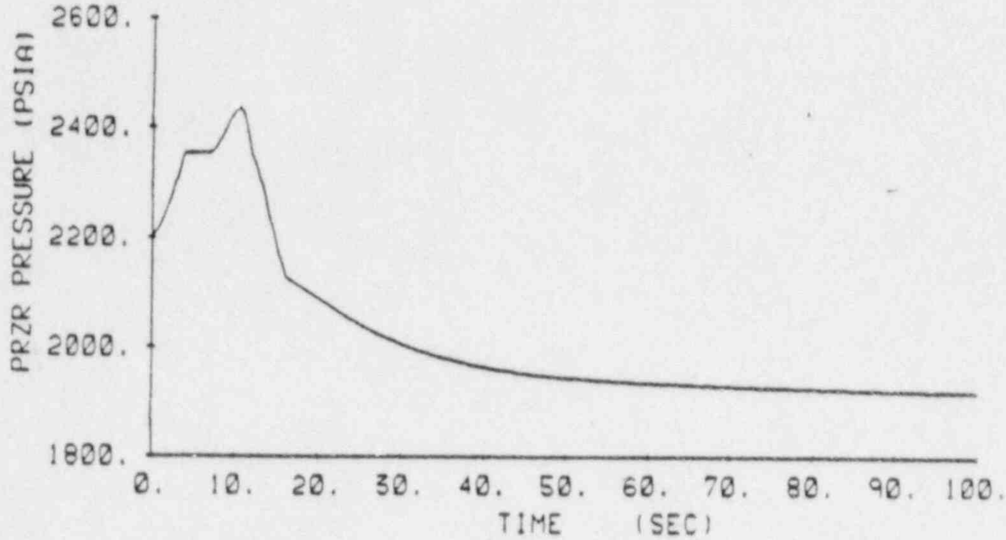
V. C. Summer

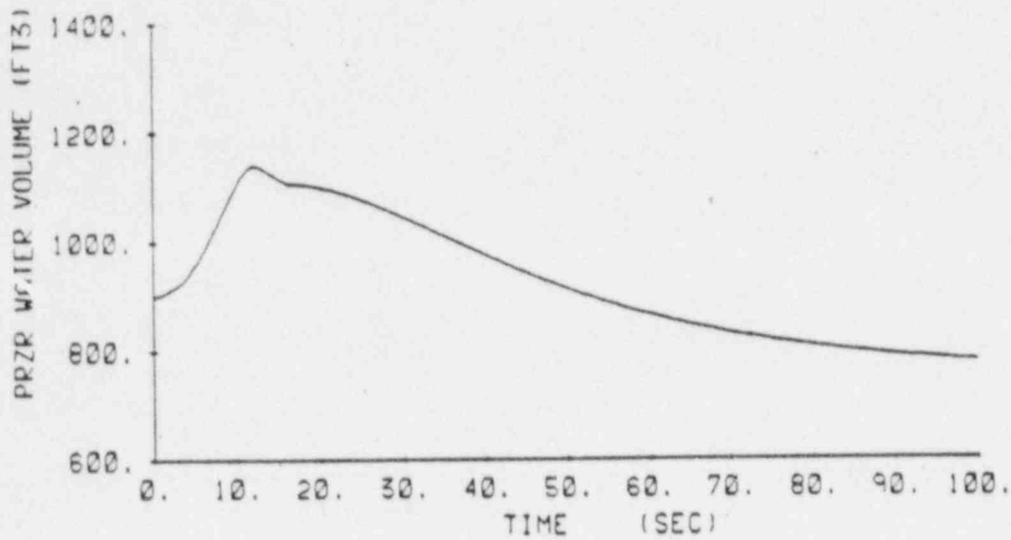
Figure 15.2.6-4

Startup of an Inactive Loop Core Flow and DNBR vs.Time

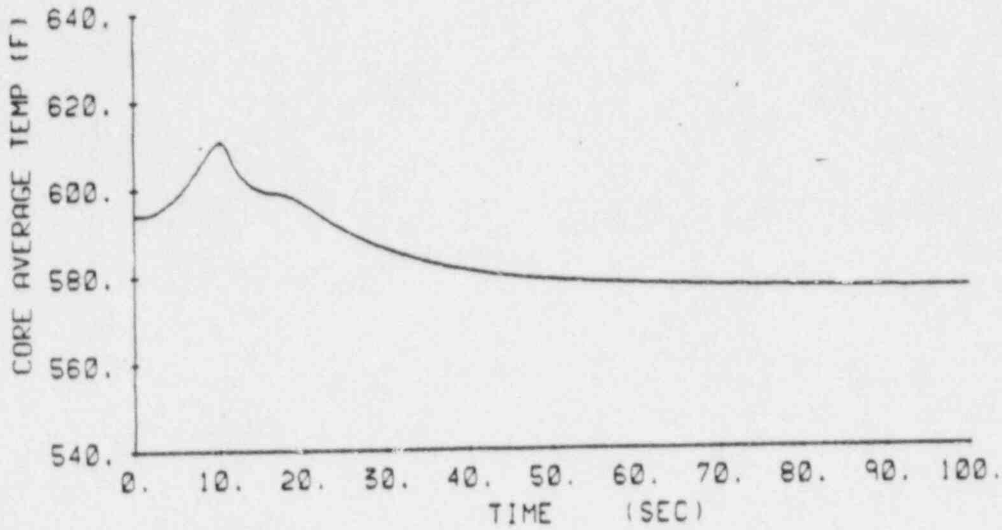


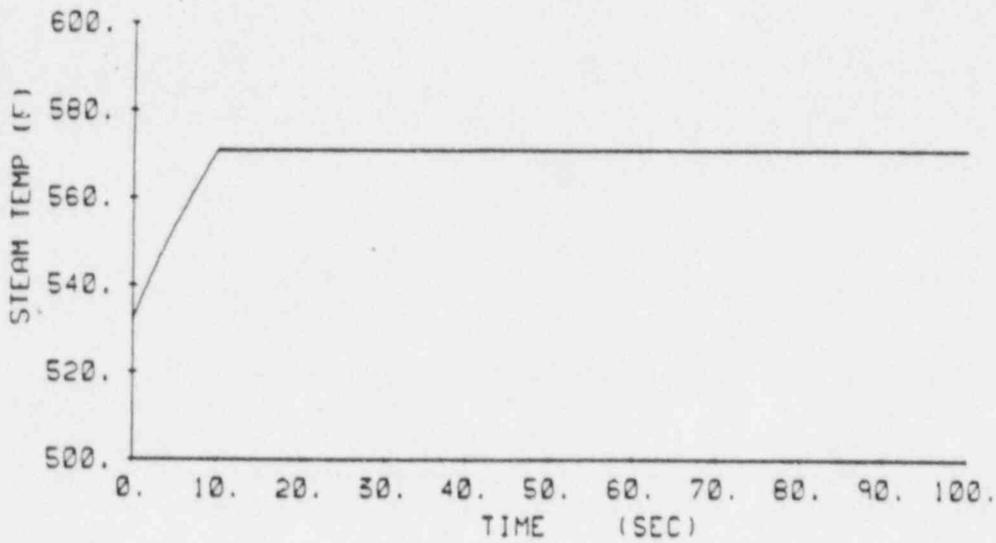
V. C. Summer
Figure 15.2.7-1
Loss of Load w/ Pressurizer Spray and PORVs at BOL Nuclear Power and DNBR vs. Time



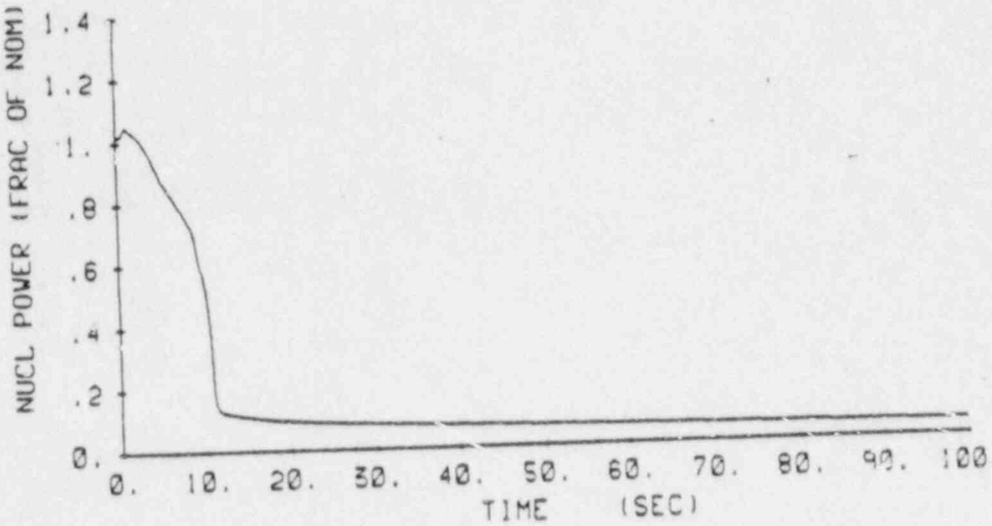


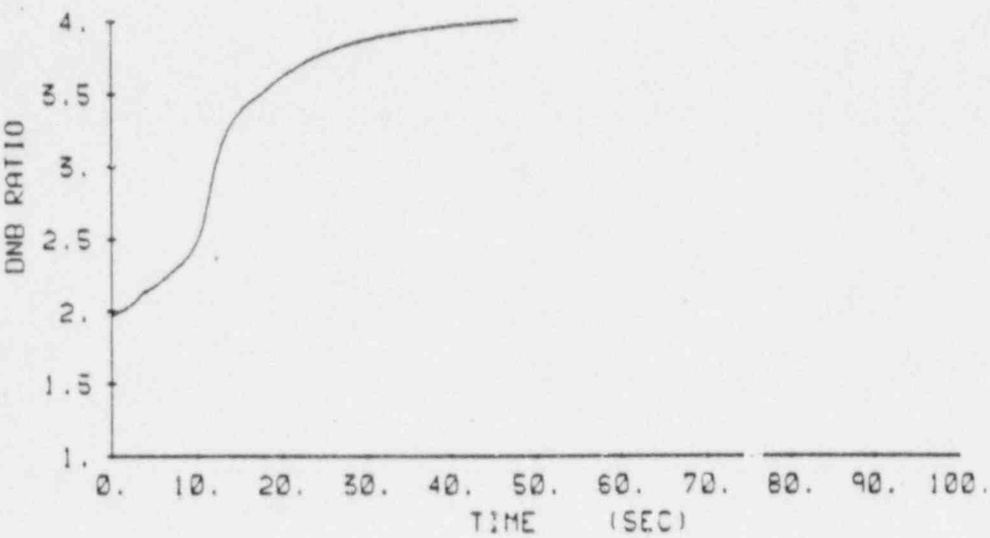
V. C. Summer
Figure 15.2.7-2
Loss of Load w/ Pressurizer Spray and PORVs at BOL Pressurizer Pressure and Water Volume vs. Time



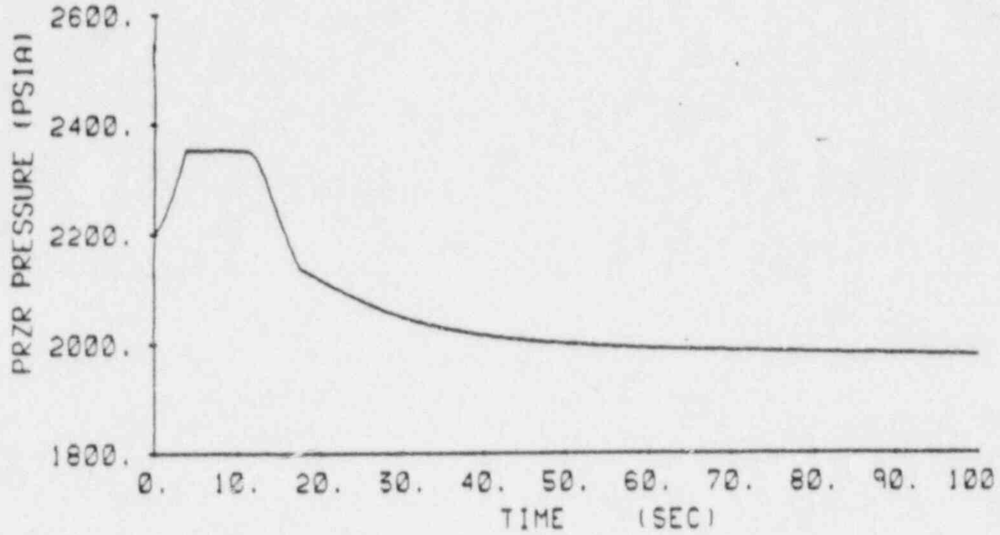


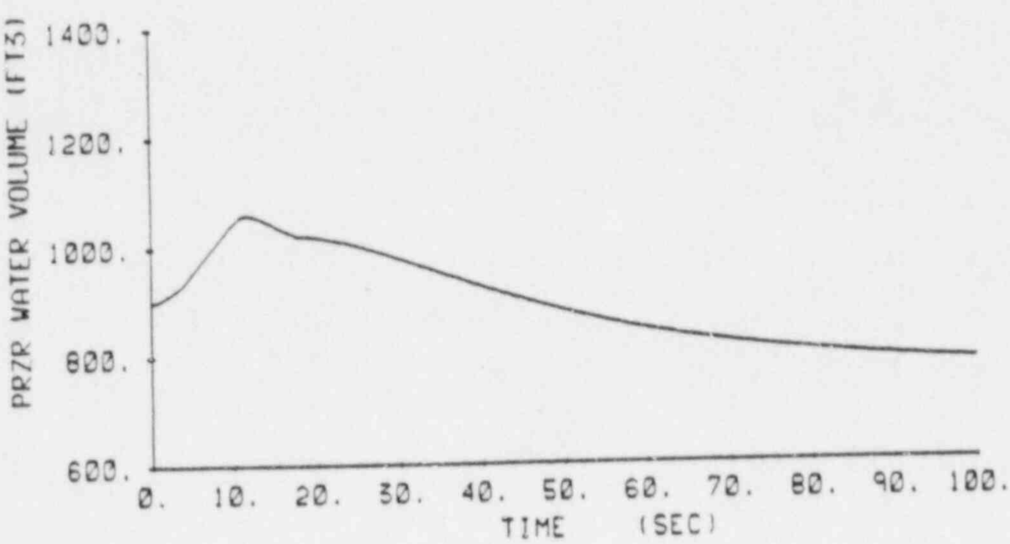
V. C. Summer
Figure 15.2.7-3
Loss of Load w/ Pressurizer Spray and PORVs at BOL Core Tavg and Steam Temperature vs. Time



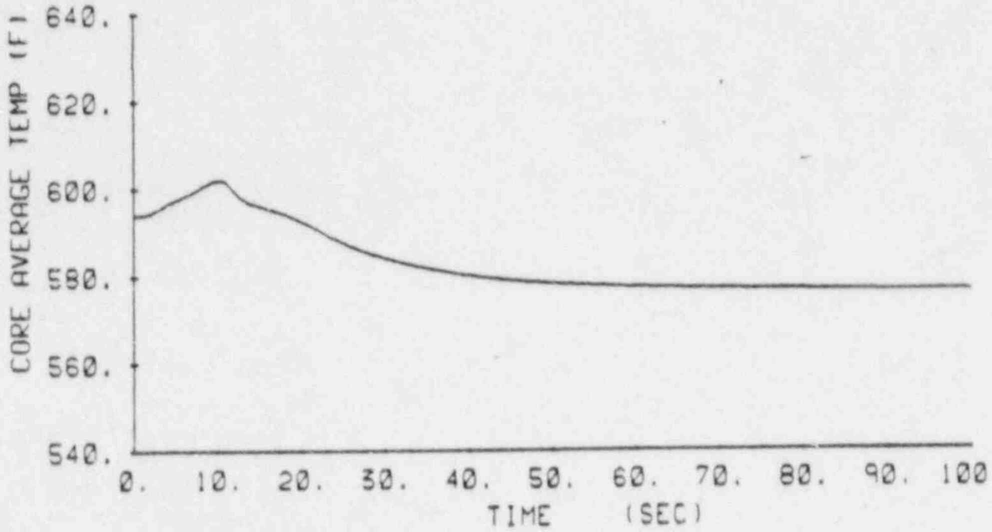


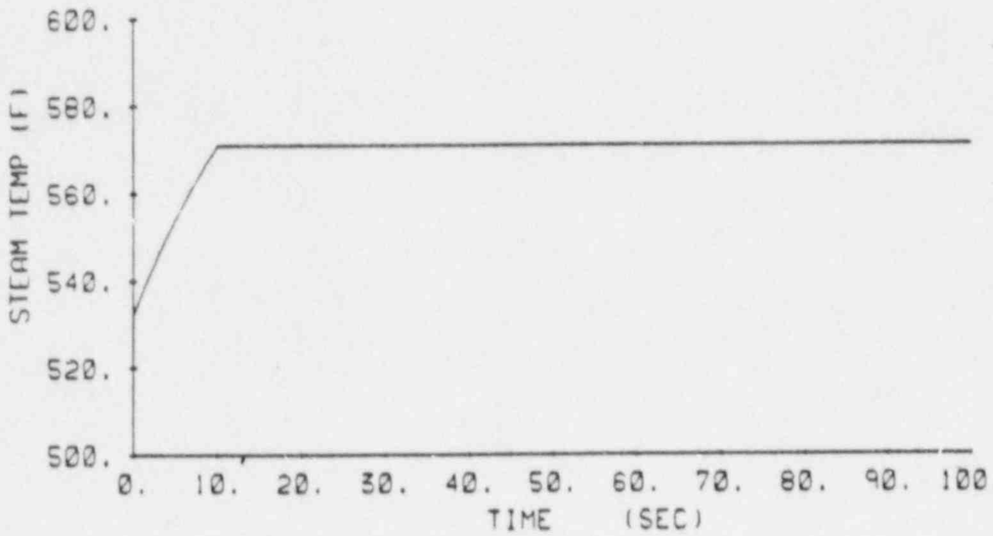
V. C. Summer
Figure 15.2.7-4
Loss of Load w/ Pressurizer Spray and PORVs at EOL Nuclear Power and DNBR vs. Time





V. C. Summer
Figure 15.2.7-5
Loss of Load w/ Pressurizer Spray and PORVs at EOL Pressurizer Pressure and Water Volume vs. Time

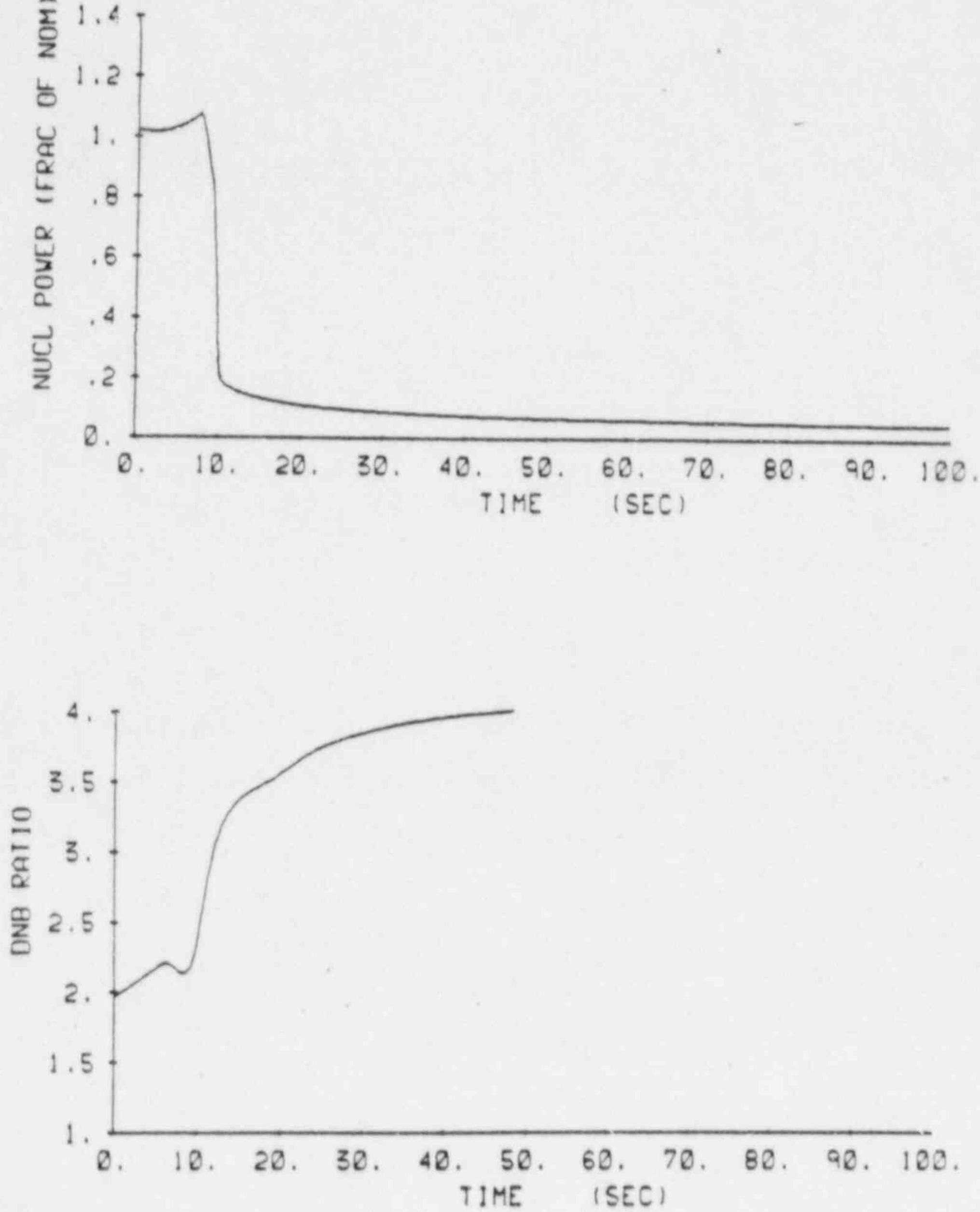




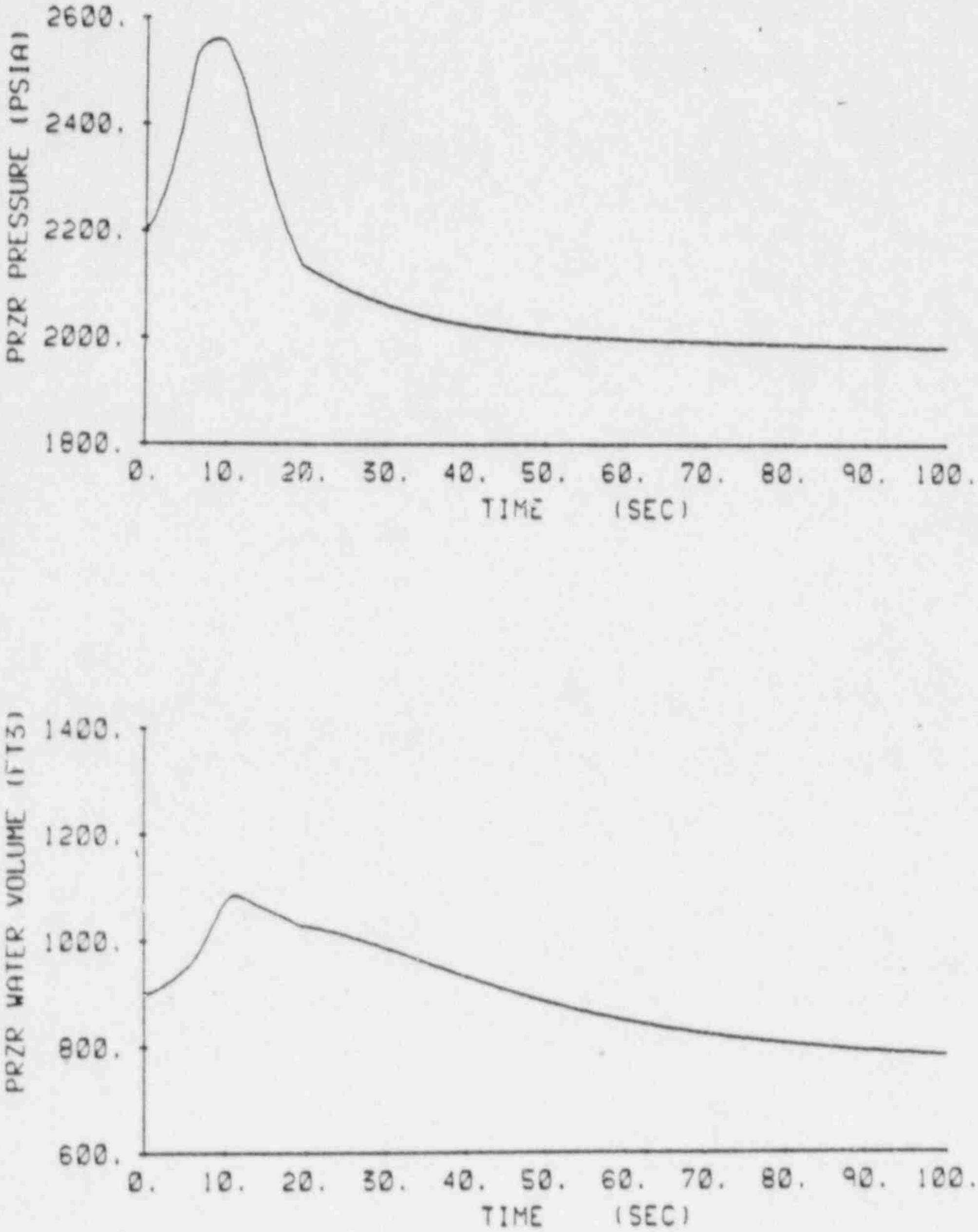
V. C. Summer

Figure 15.2.7-6

Loss of Load w/ Pressurizer Spray and PORVs at EOL Core Tavg and Steam Temperature vs. Time



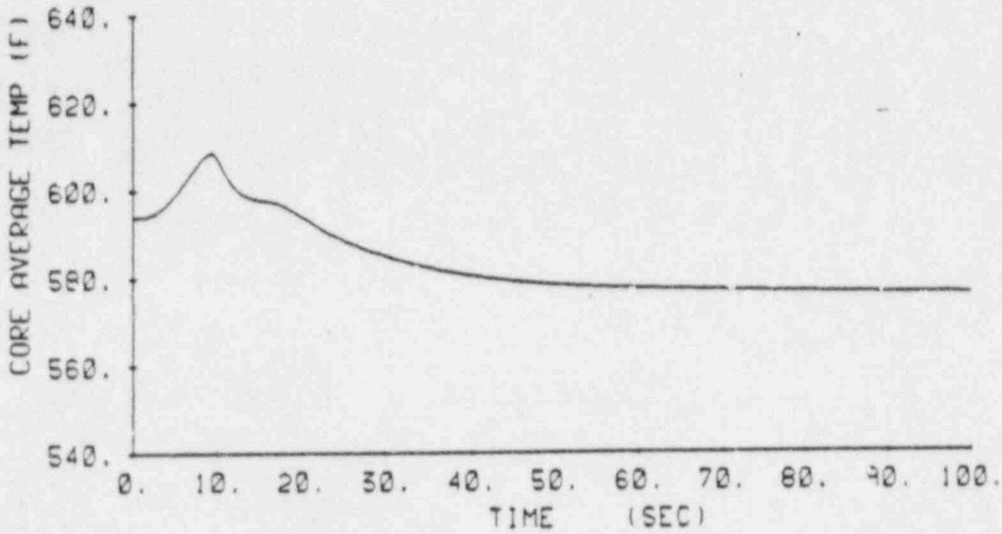
V. C. Summer
Figure 15.2.7-7
Loss of Load w/o Pressurizer Spray and PORVs at BOL Nuclear Power and DNBR vs. Time

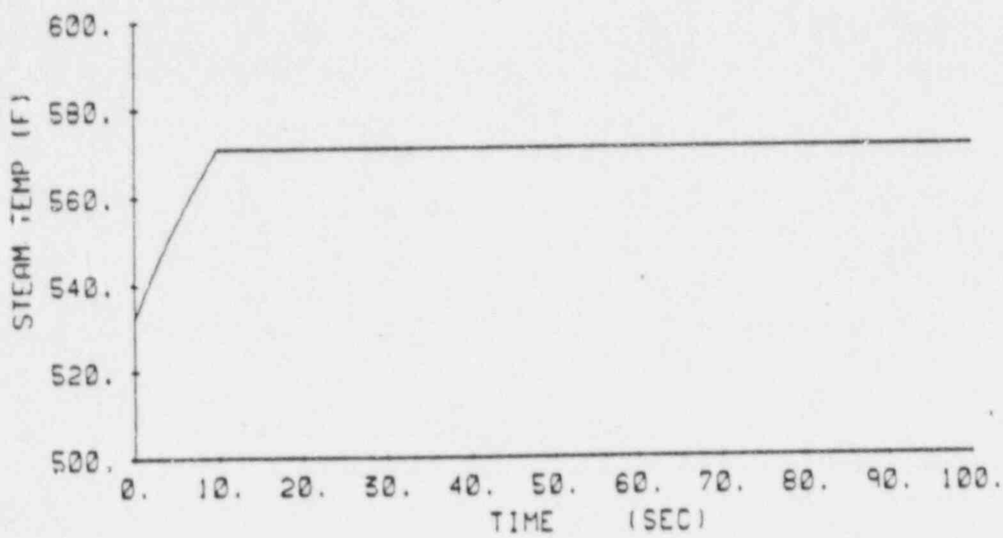


V. C. Summer

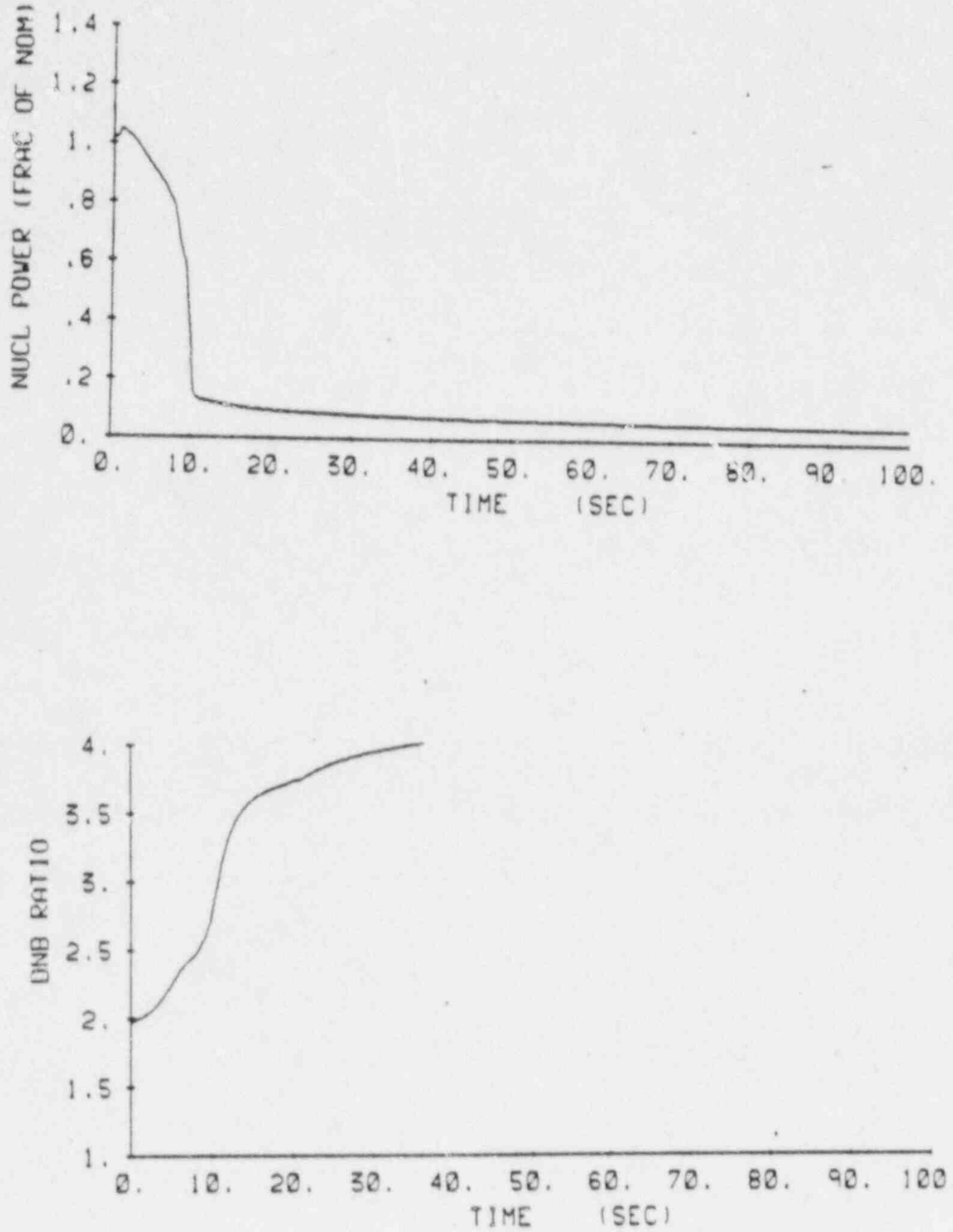
Figure 15.2.7-8

Loss of Load w/o Pressurizer Spray and PORVs at BOL Pressurizer Pressure and Water Volume vs. Time

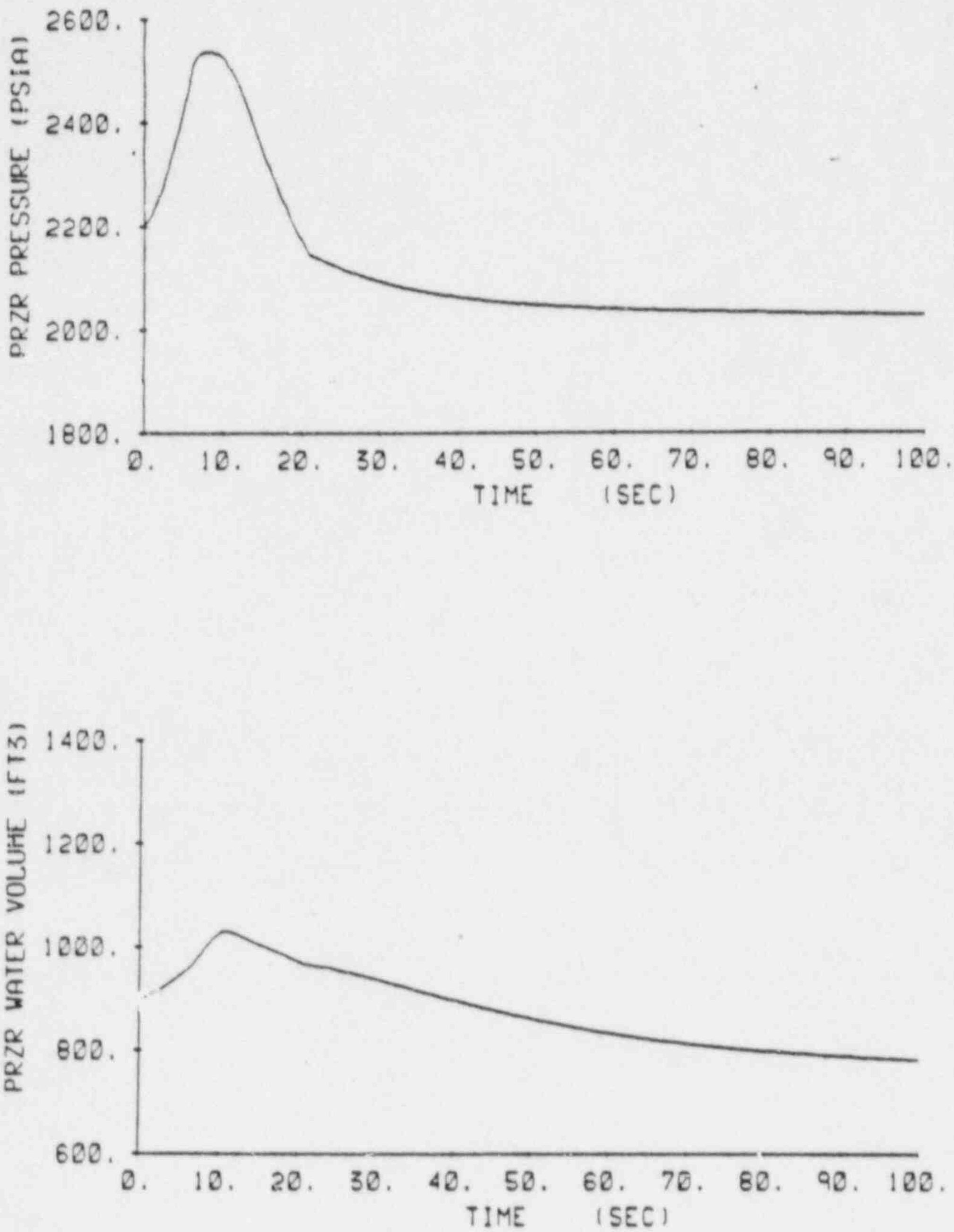




V. C. Summer
Figure 15.2.7-9
Loss of Load w/o Pressurizer Spray and PORVs at BOL Core Tavg and Steam Temperature vs. Time



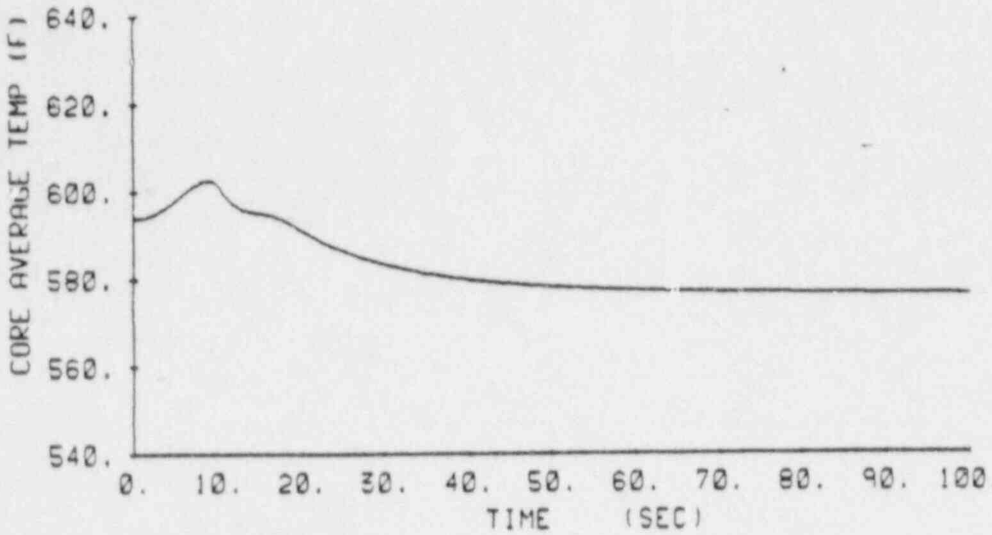
V. C. Summer
Figure 15.2.7-10
Loss of Load w/o Pressurizer Spray and PORVs at EOL Nuclear Power and DNBR vs. Time

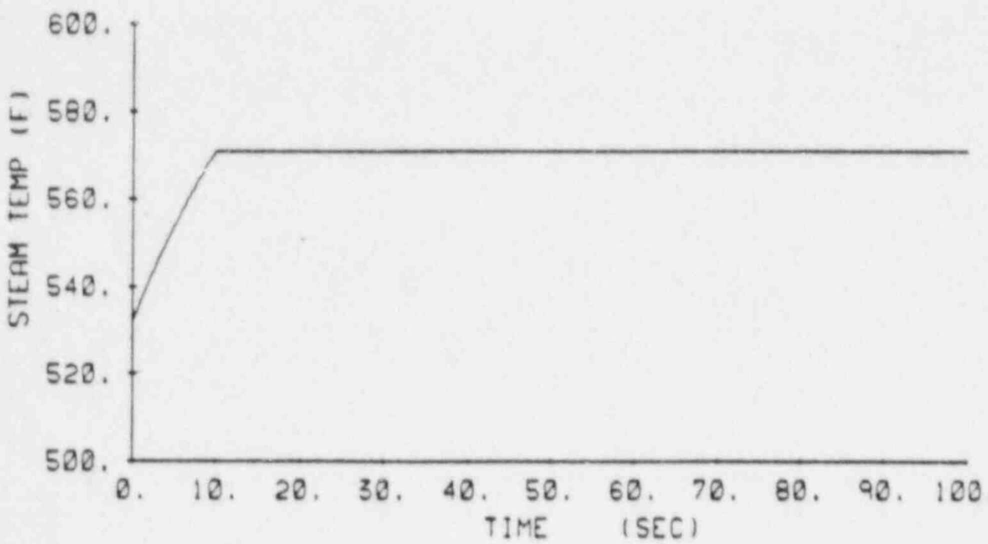


V. C. Summer

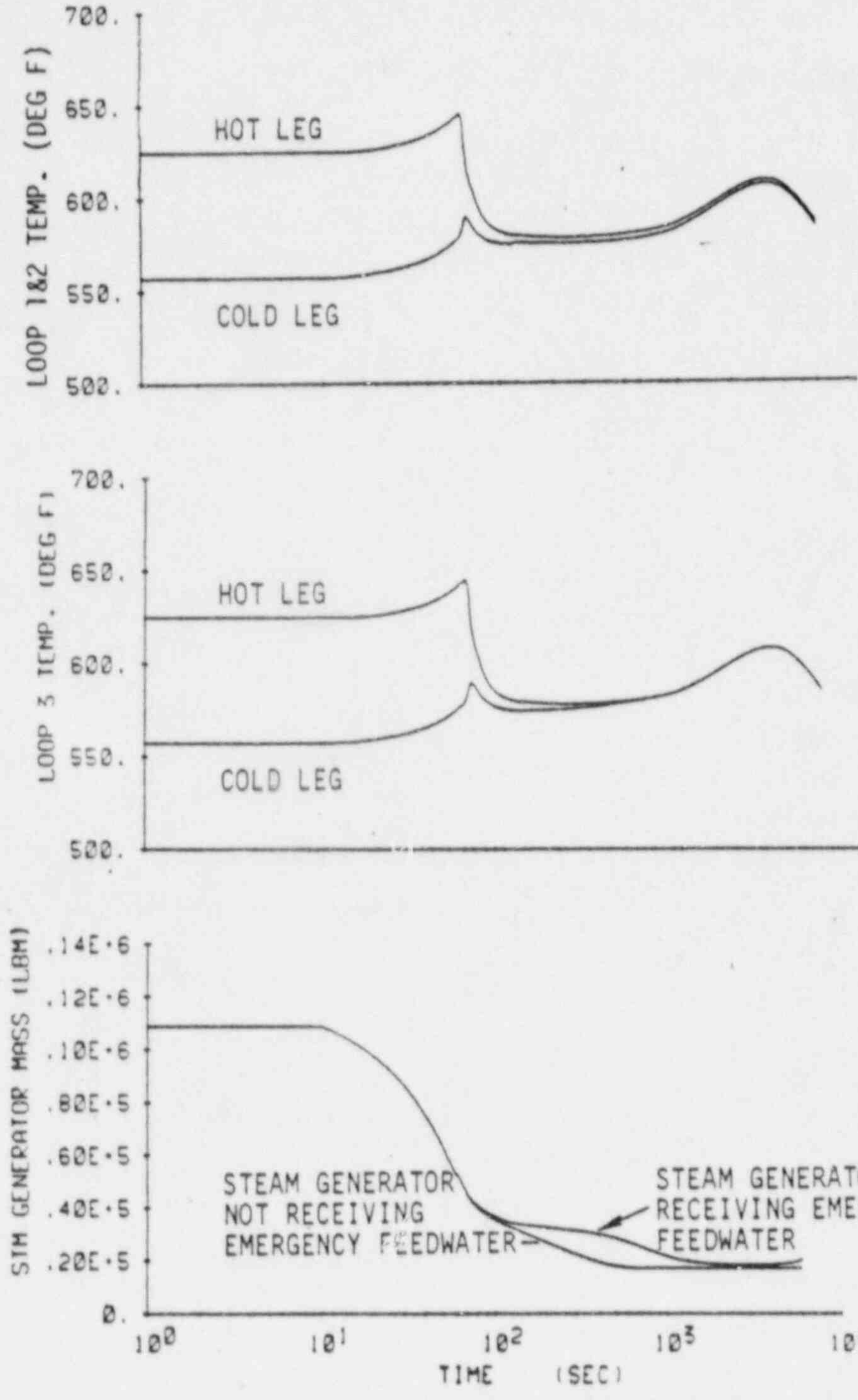
Figure 15.2.7-11

Loss of Load w/o Pressurizer Spray and PORVs at EOL Pressurizer Pressure and Water Volume vs. Time

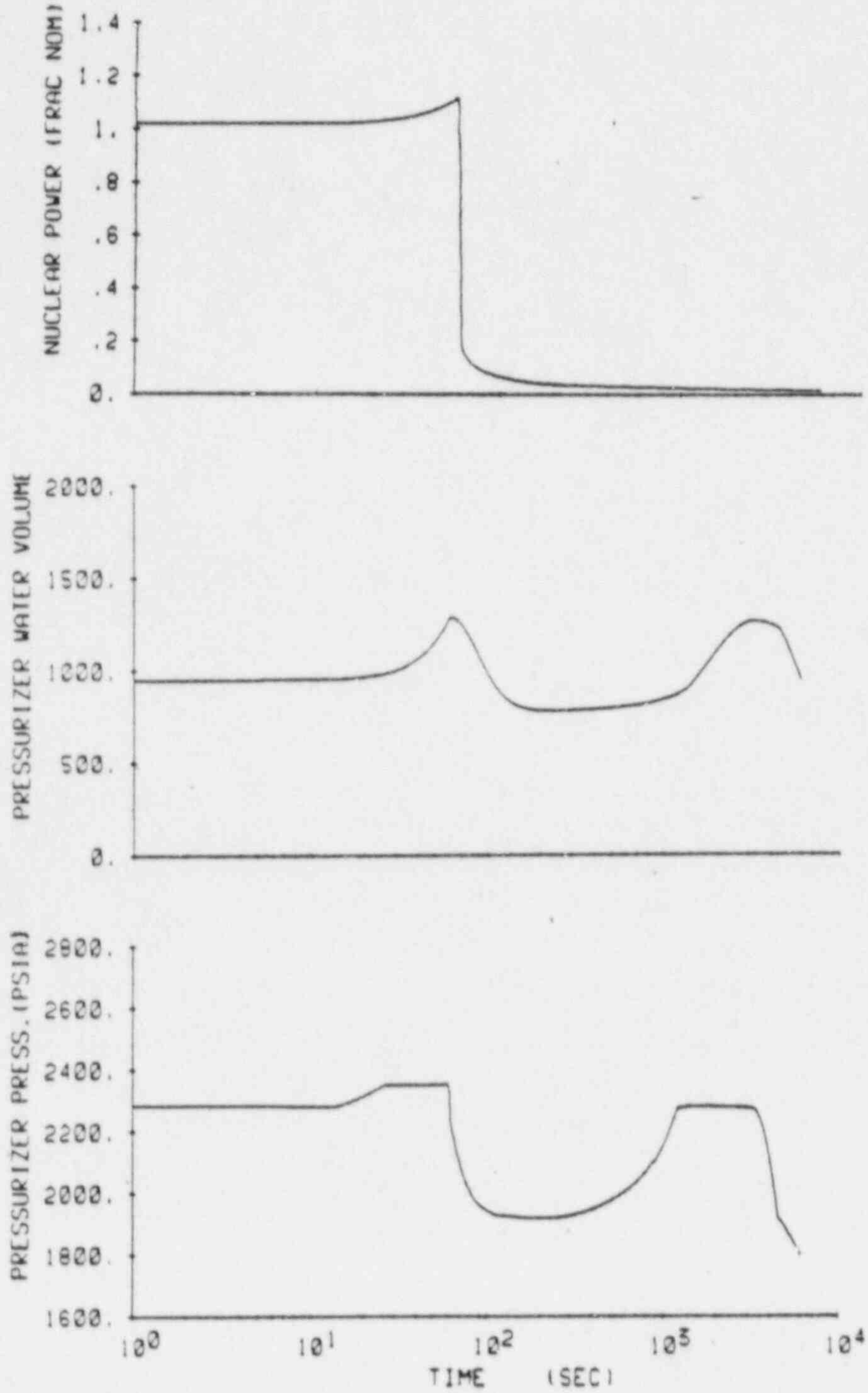




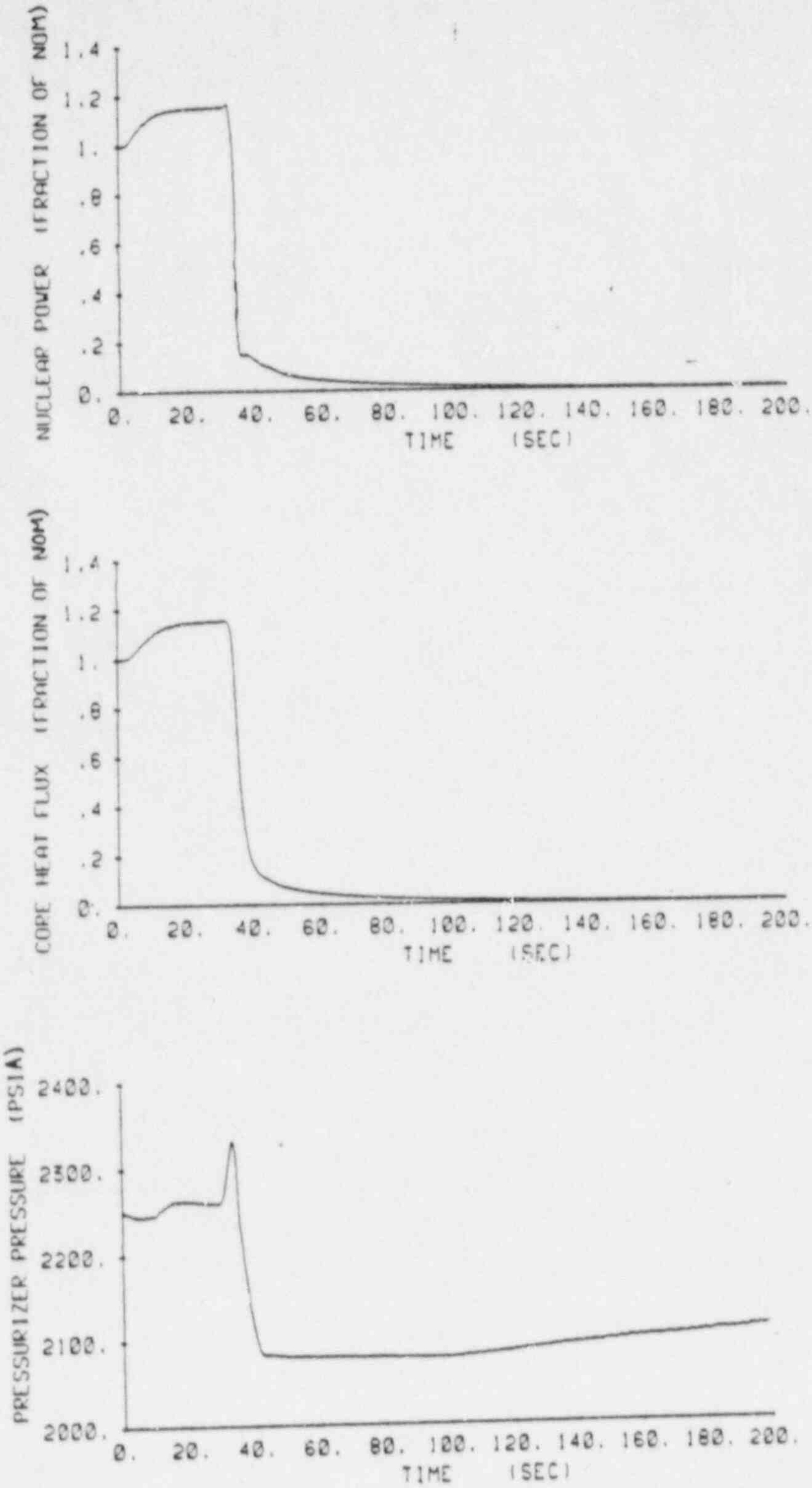
V. C. Summer
Figure 15.2.7-12
Loss of Load w/o Pressurizer Spray and PORVs at EOL Core Tavg and Steam Temperature vs. Time



V. C. Summer
Figure 15.2.8-1
Loss of Normal Feedwater Loops 1 and 2 Temperatures, Loop 3 Temperatures, Steam Generator Water Mass vs. Time



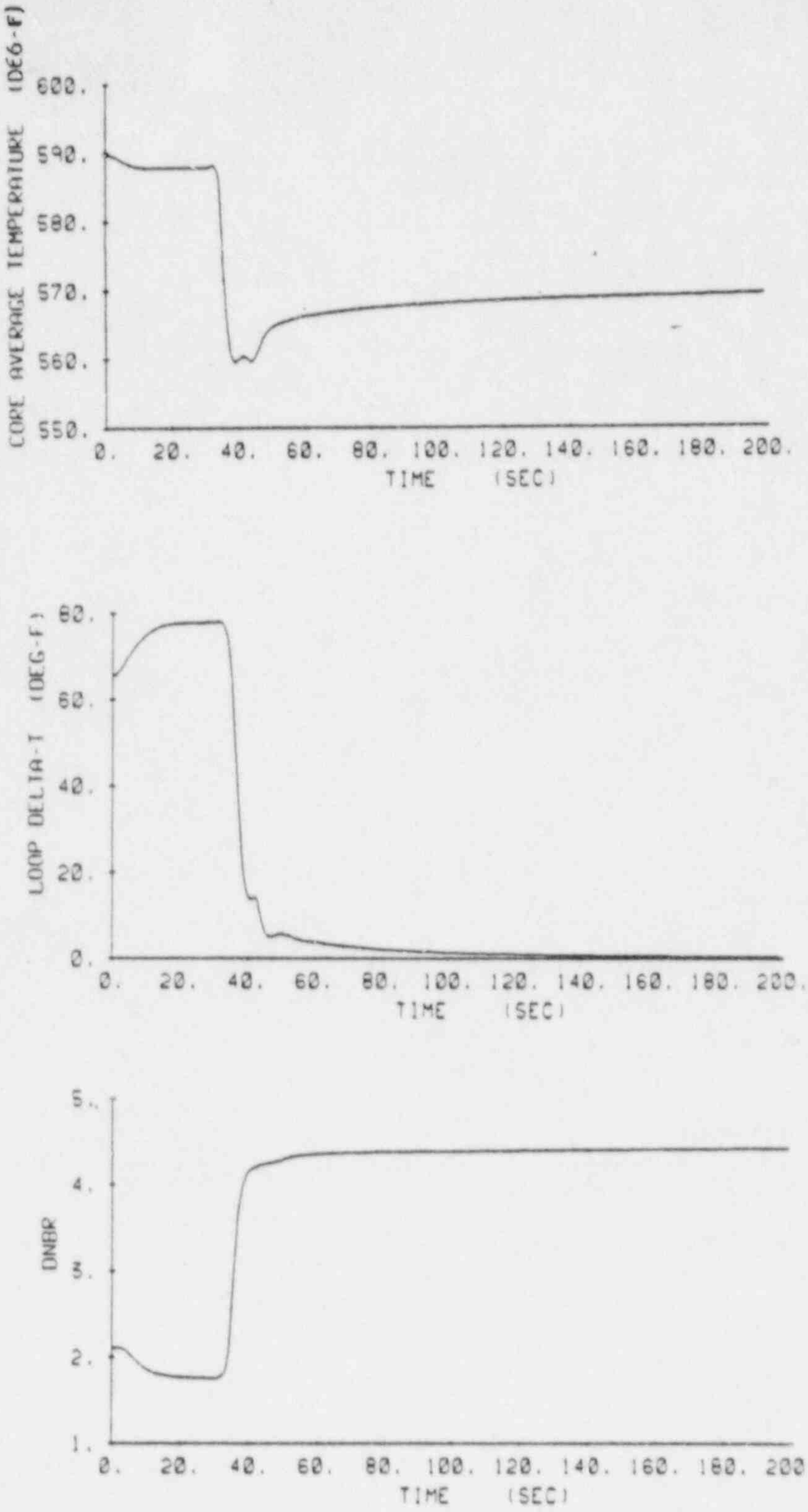
V. C. Summer
Figure 15.2.8-2
Loss of Normal Feedwater Nuclear Power, Pressurizer Water Volume and Pressurizer Pressure vs. Time



V. C. Summer

Figure 15.2.10-1

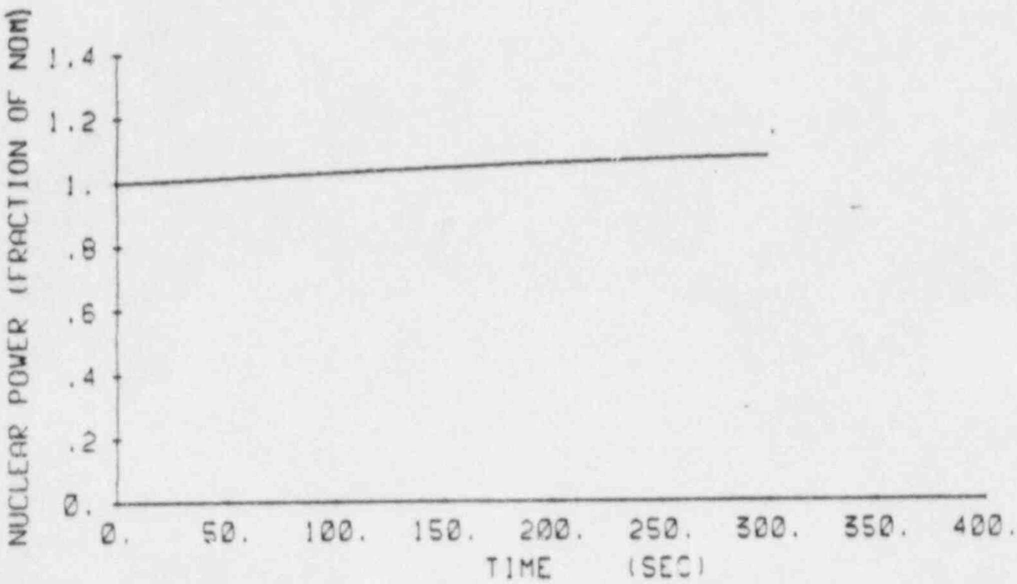
Feedwater System Malfunction
Nuclear Power, Core Heat
Flux and Pressurizer
Pressure vs. Time

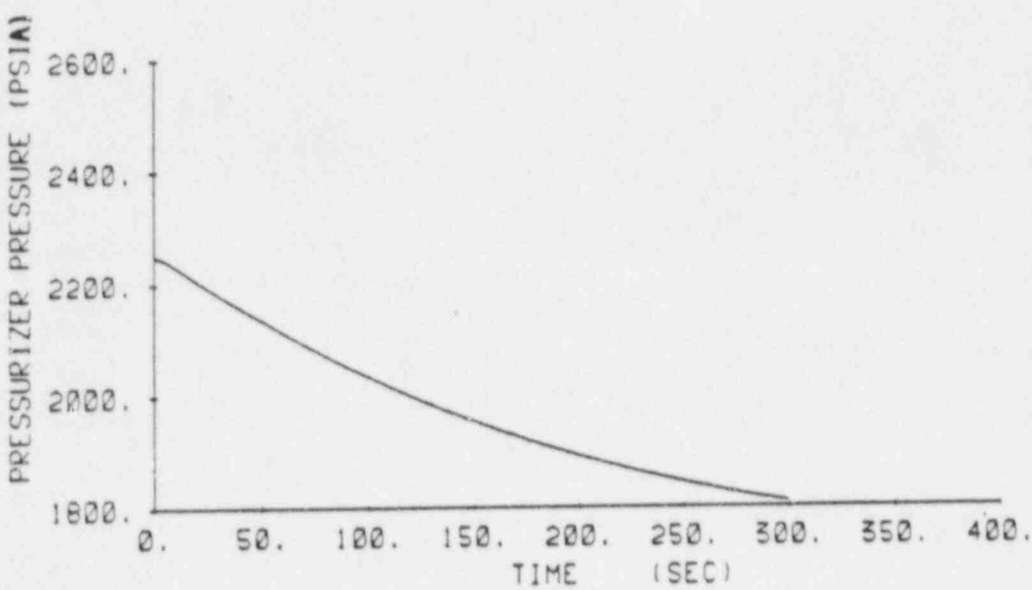


V. C. Summer

Figure 15.2.10-2

Feedwater System Malfunction
Loop Delta-T, Core Tavg
and DNBR vs. Time





V. C. Summer
Figure 15.2.11-1
Excessive Load Increase w/o Control, Minimum Feedback Nuclear Power and Pressurizer Pressure vs.Time

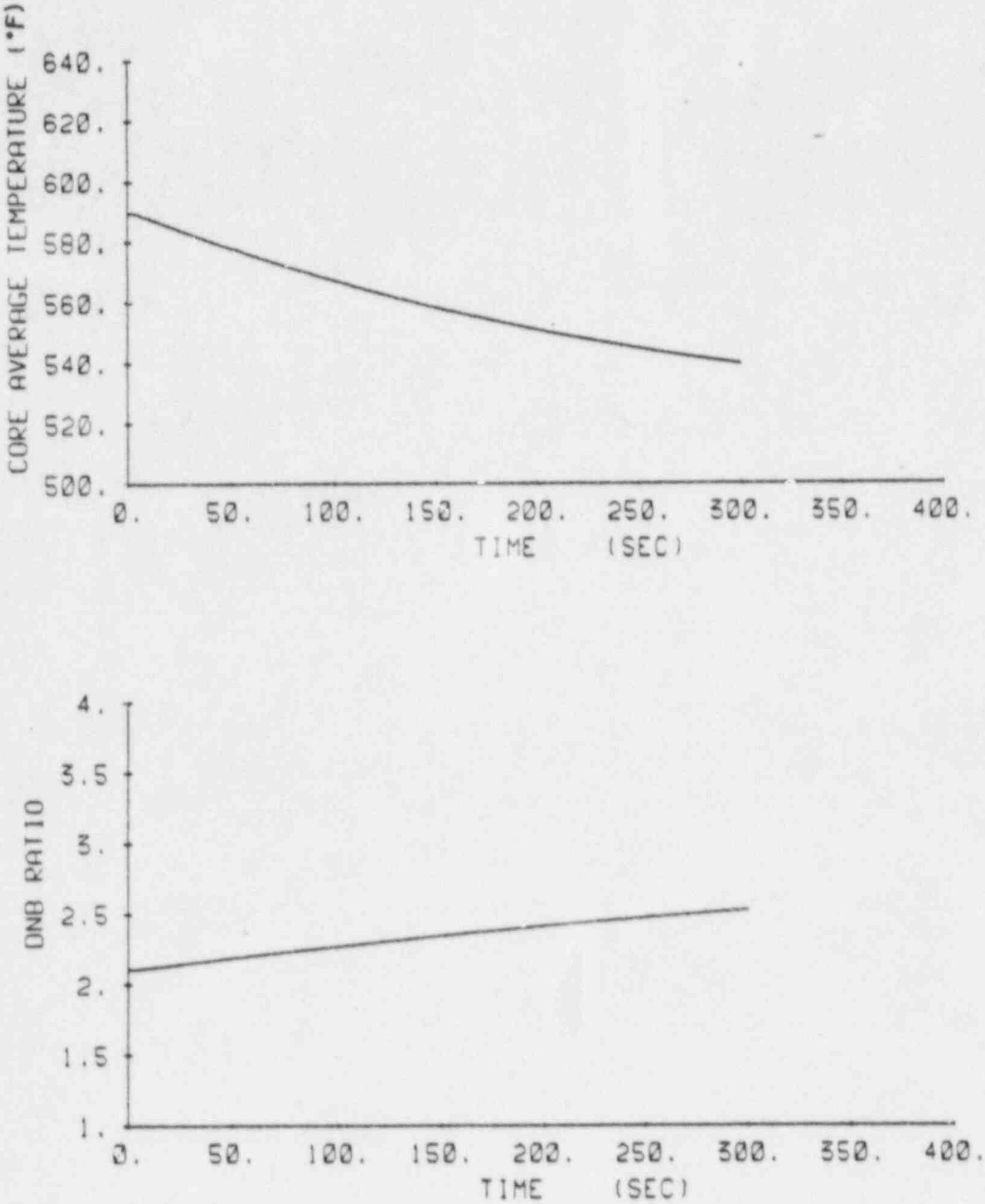


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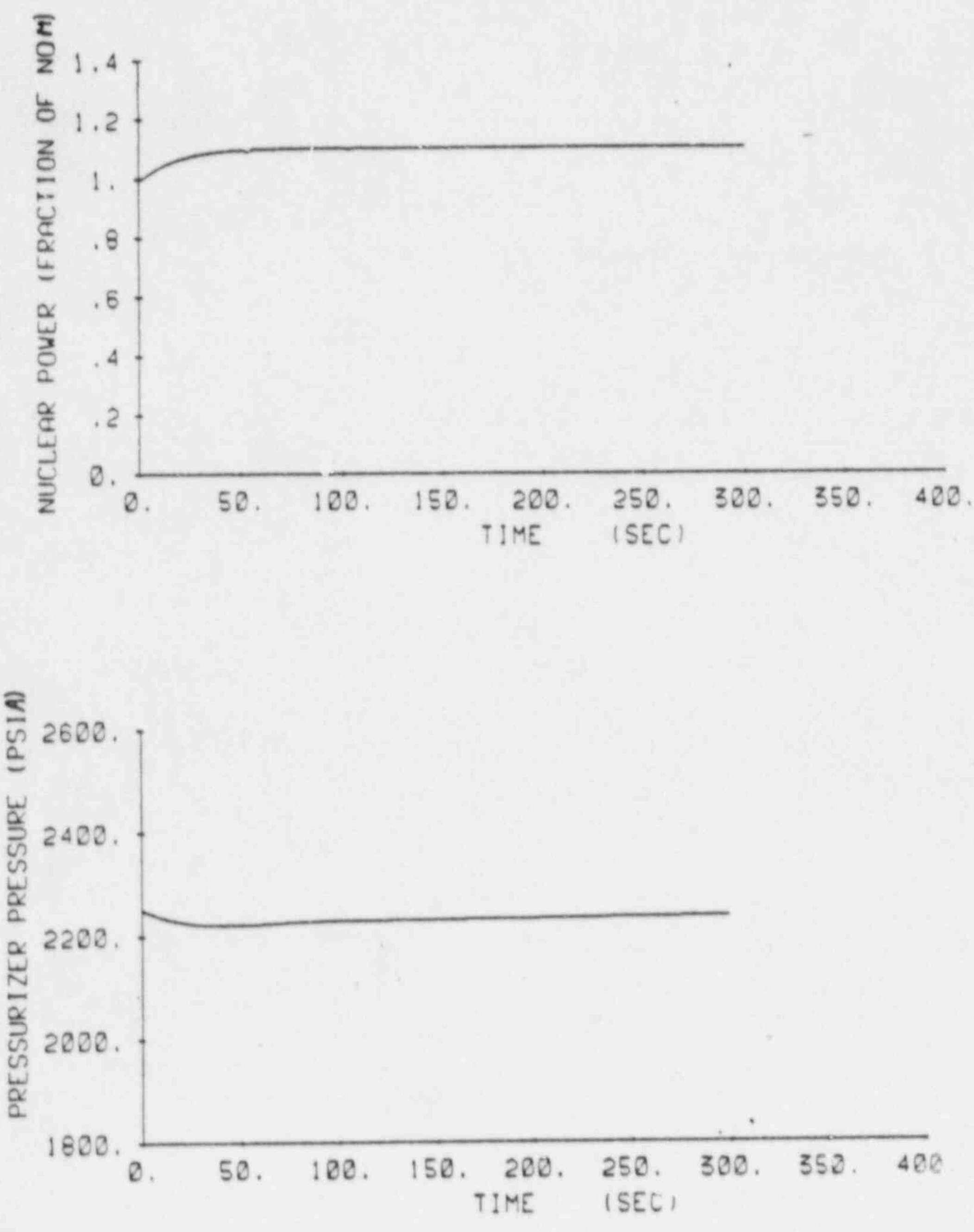
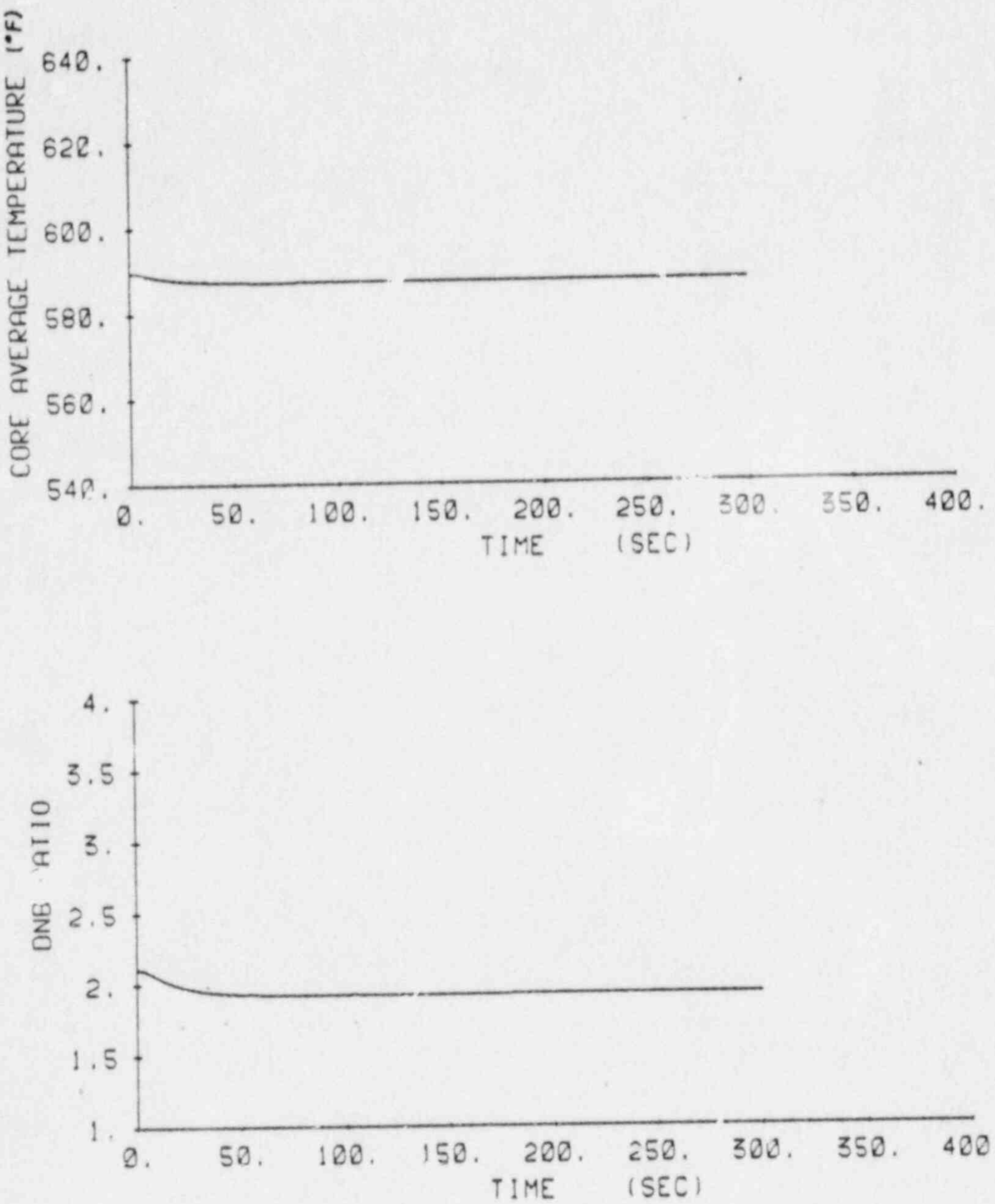
Figure 15.2.11-2

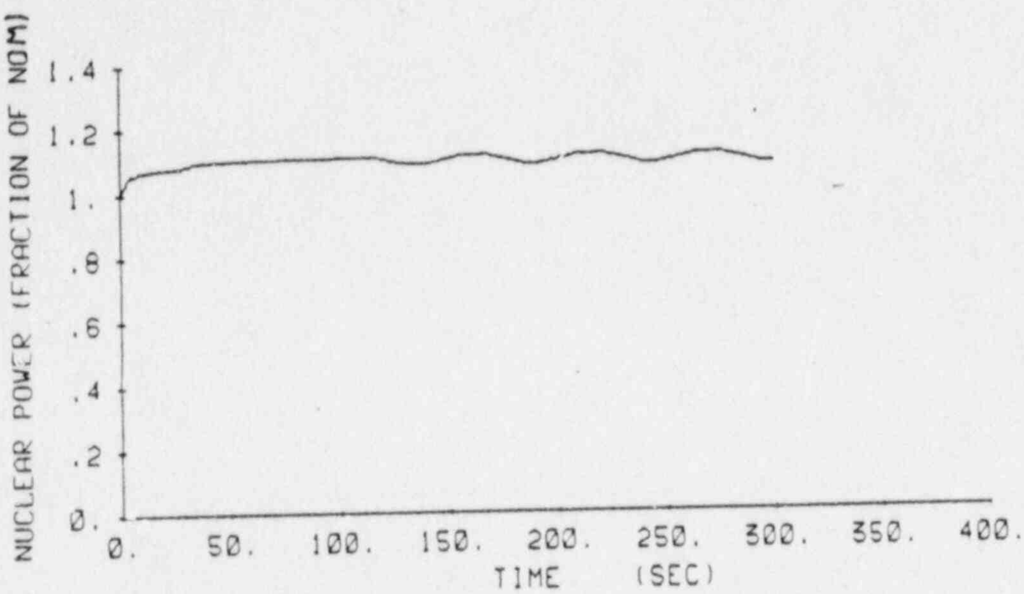
Excessive Load Increase w/o Control, Minimum Feedback Tavg and DNBR vs. Time

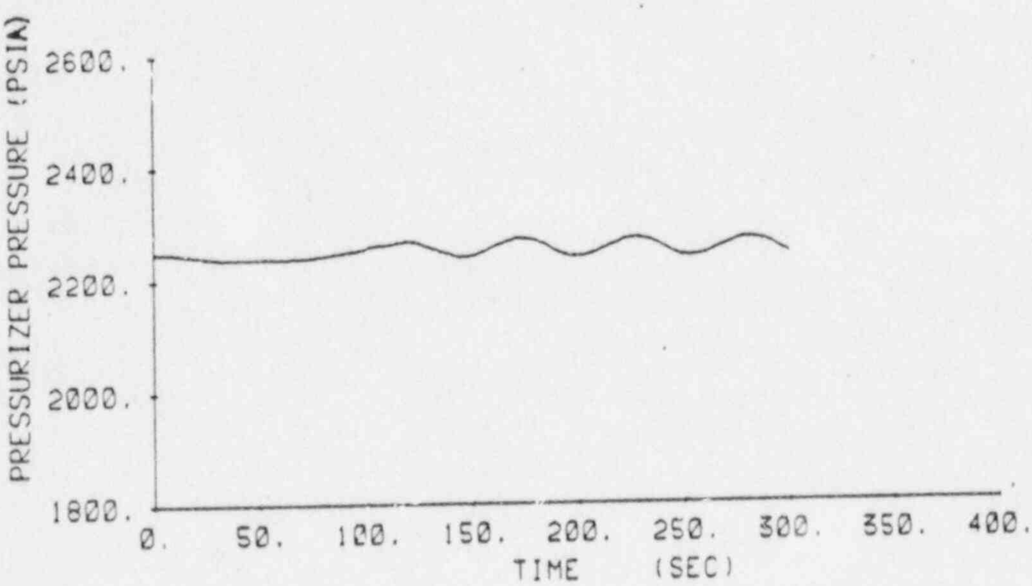


V. C. Summer
Figure 15.2.11-3
Excessive Load Increase w/o Control, Maximum Feedback Nuclear Power and Pressurizer Pressure vs. Time

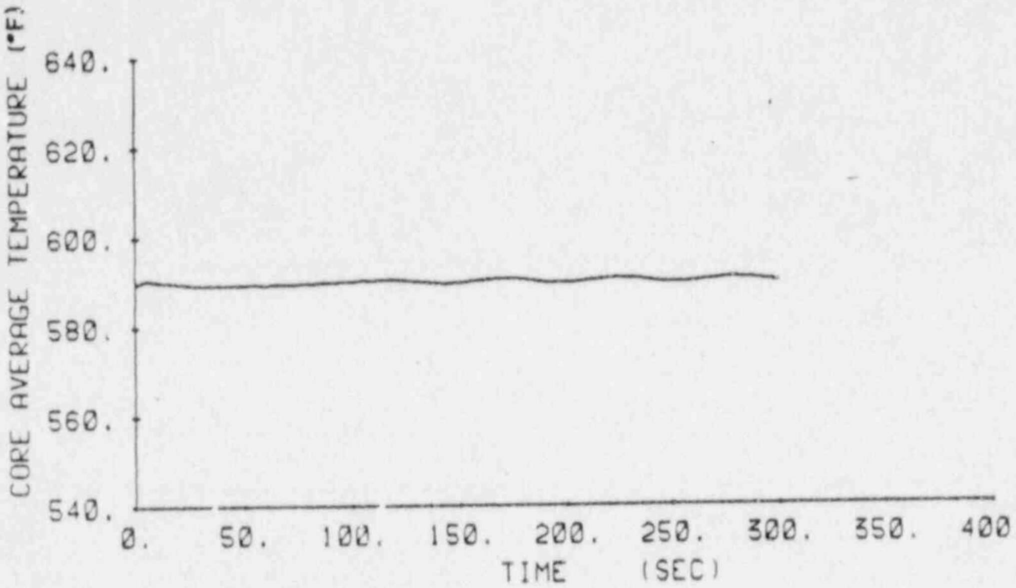


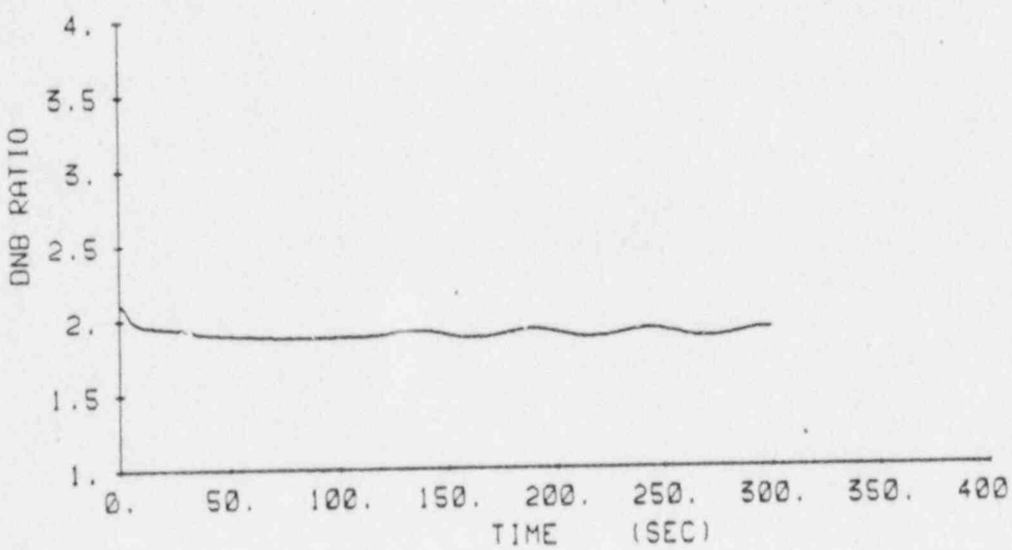
V. C. Summer
Figure 15.2.11-4
Excessive Load Increase w/o Control, Maximum Feedback Tavg and DNBR vs. Time



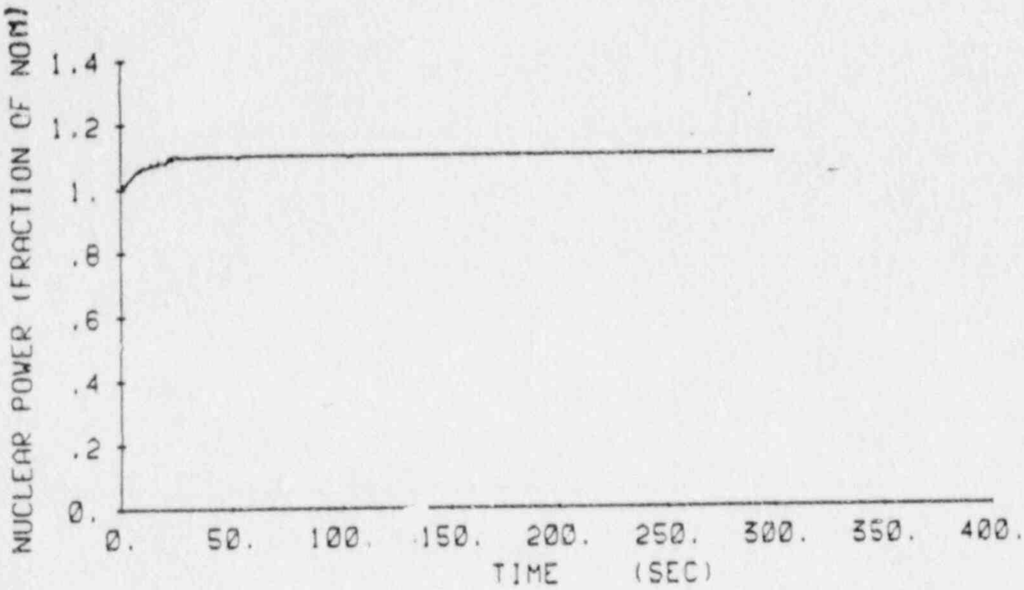


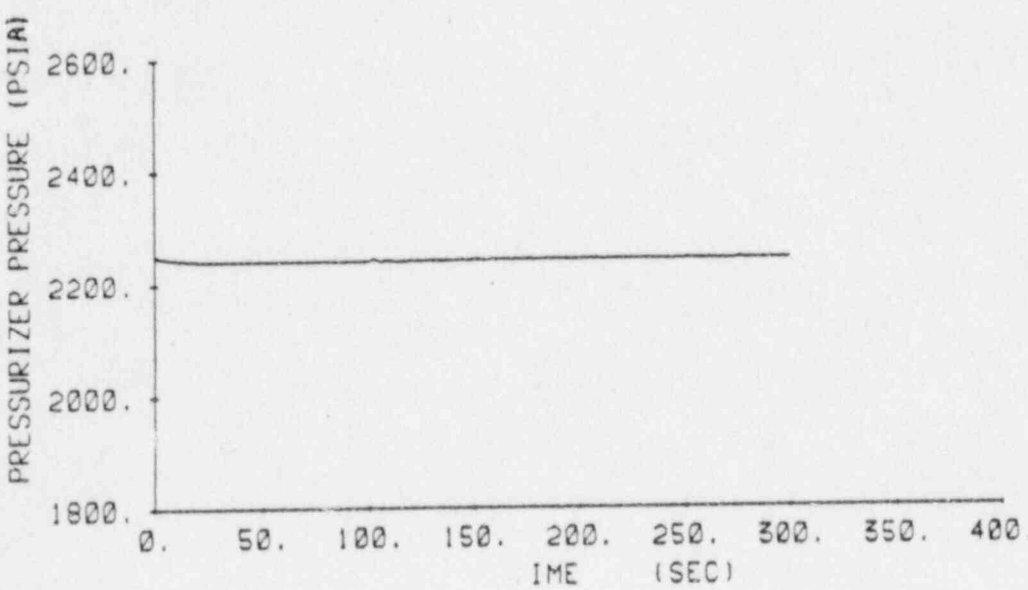
V. C. Summer
Figure 15.2.11-5
Excessive Load Increase w/ Control, Minimum Feedback Nuclear Power and Pressurizer Pressure vs. Time



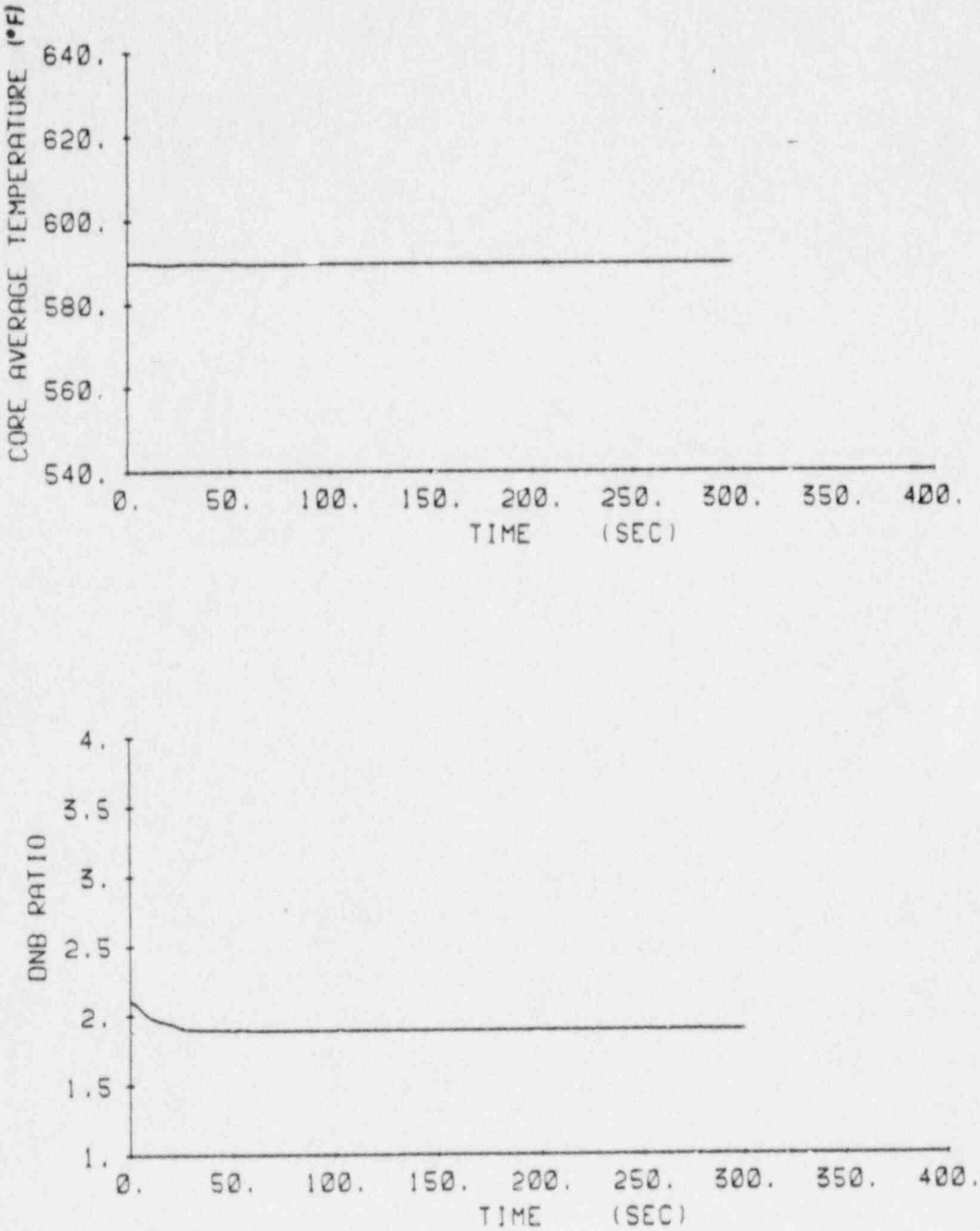


V. C. Summer
Figure 15.2.11-6
Excessive Load Increase w/ Control, Minimum Feedback Tavg and DNBR vs. Time





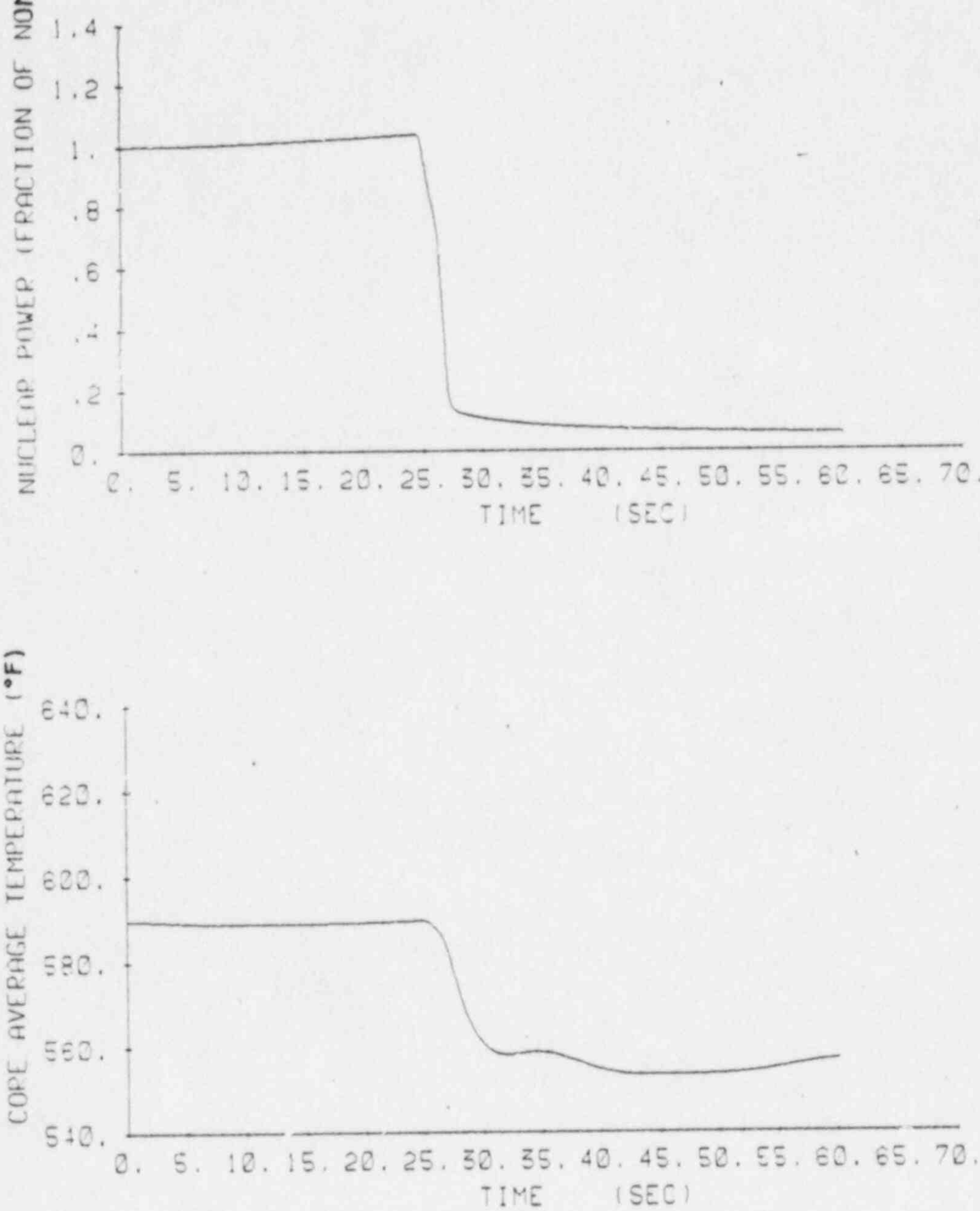
V. C. Summer
Figure 15.2.11-7
Excessive Load Increase w/ Control, Maximum Feedback Nuclear Power and Pressurizer Pressure vs. Time



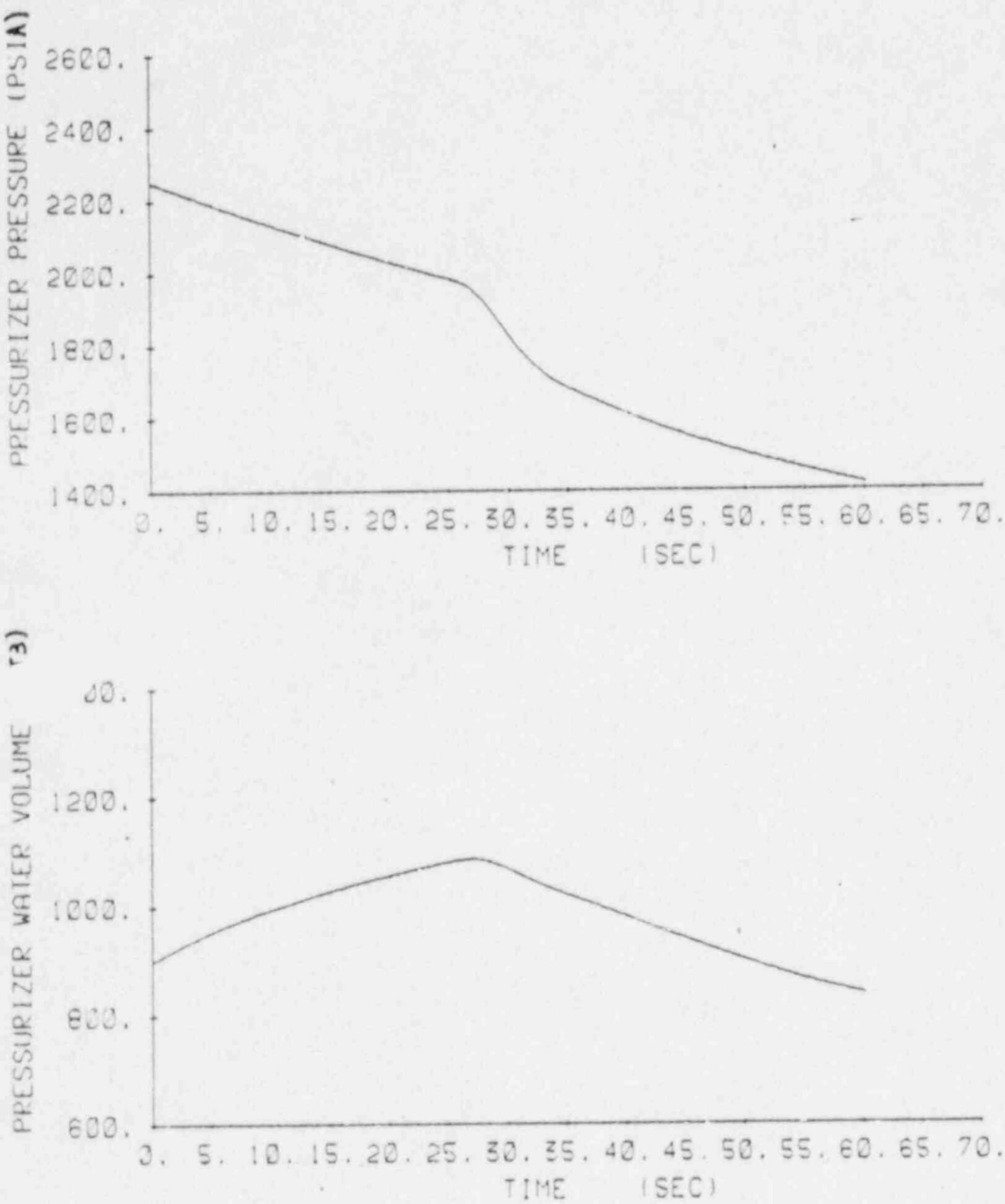
V. C. Summer

Figure 15.2.11-8

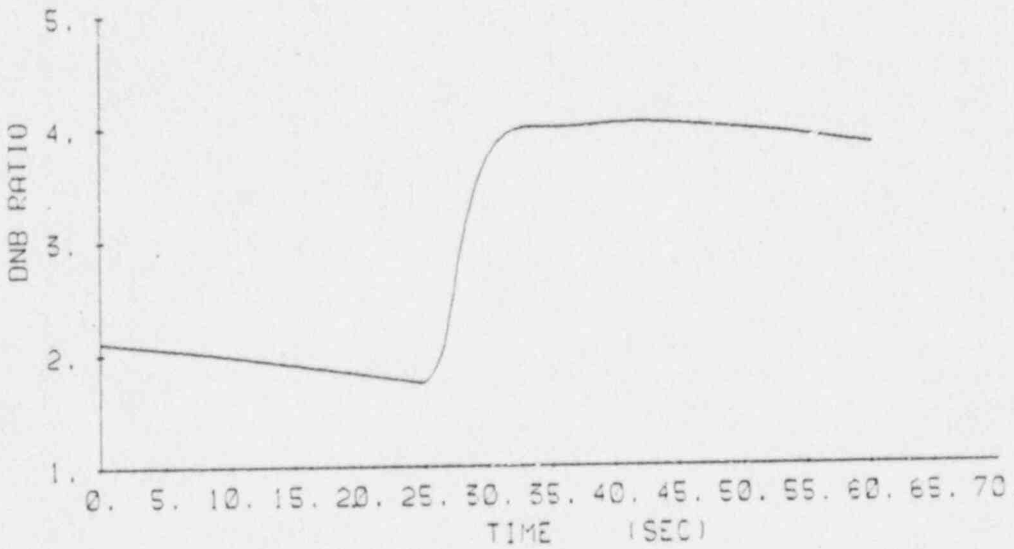
Excessive Load Increase w/ Control, Maximum Feedback Tavg and DNBR vs. Time



V. C. Summer
Figure 15.2.12-1
Accidental Depressurization of the Reactor Coolant System Nuclear Power and Core Tavg vs. Time



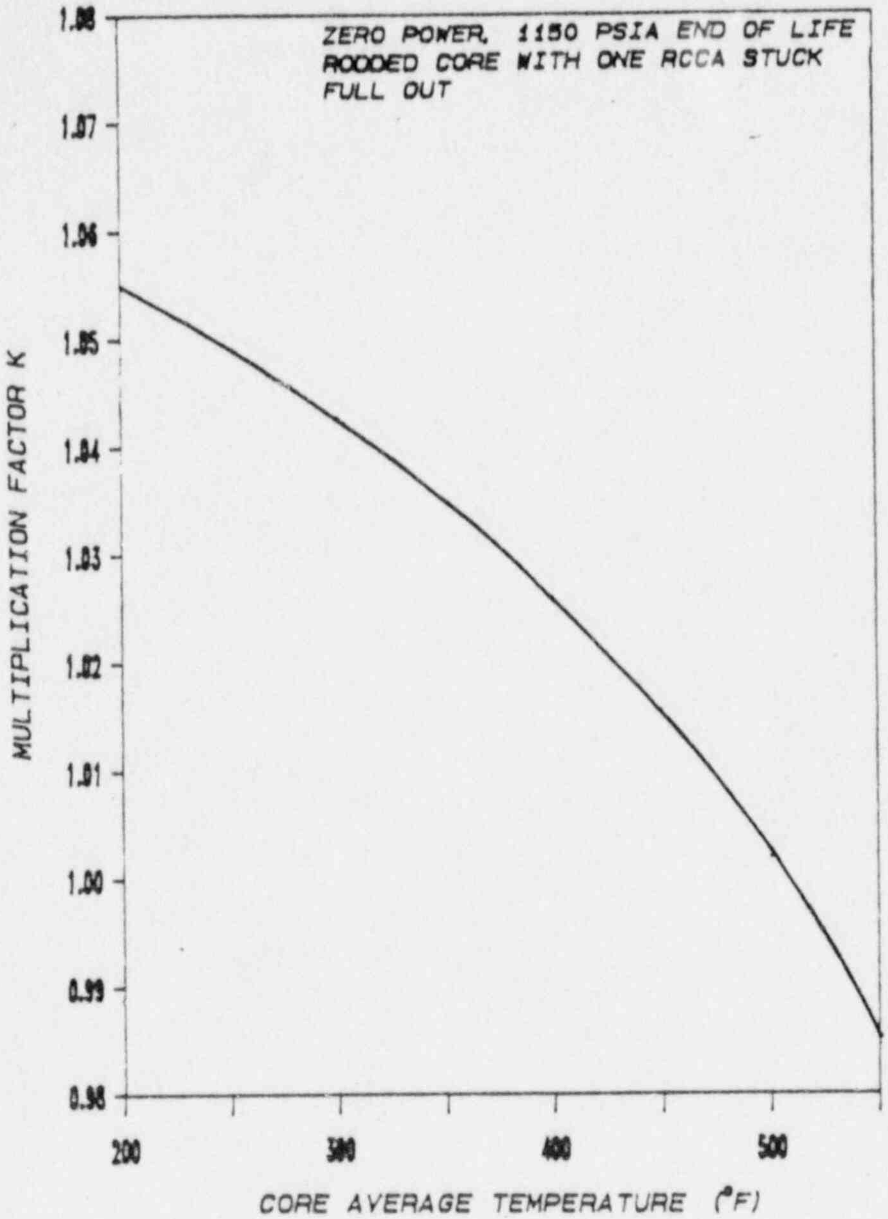
V. C. Summer
Figure 15.2.12-2
Accidental Depressurization of the Reactor Coolant System Pressurizer Pressure and Water Volume vs. Time



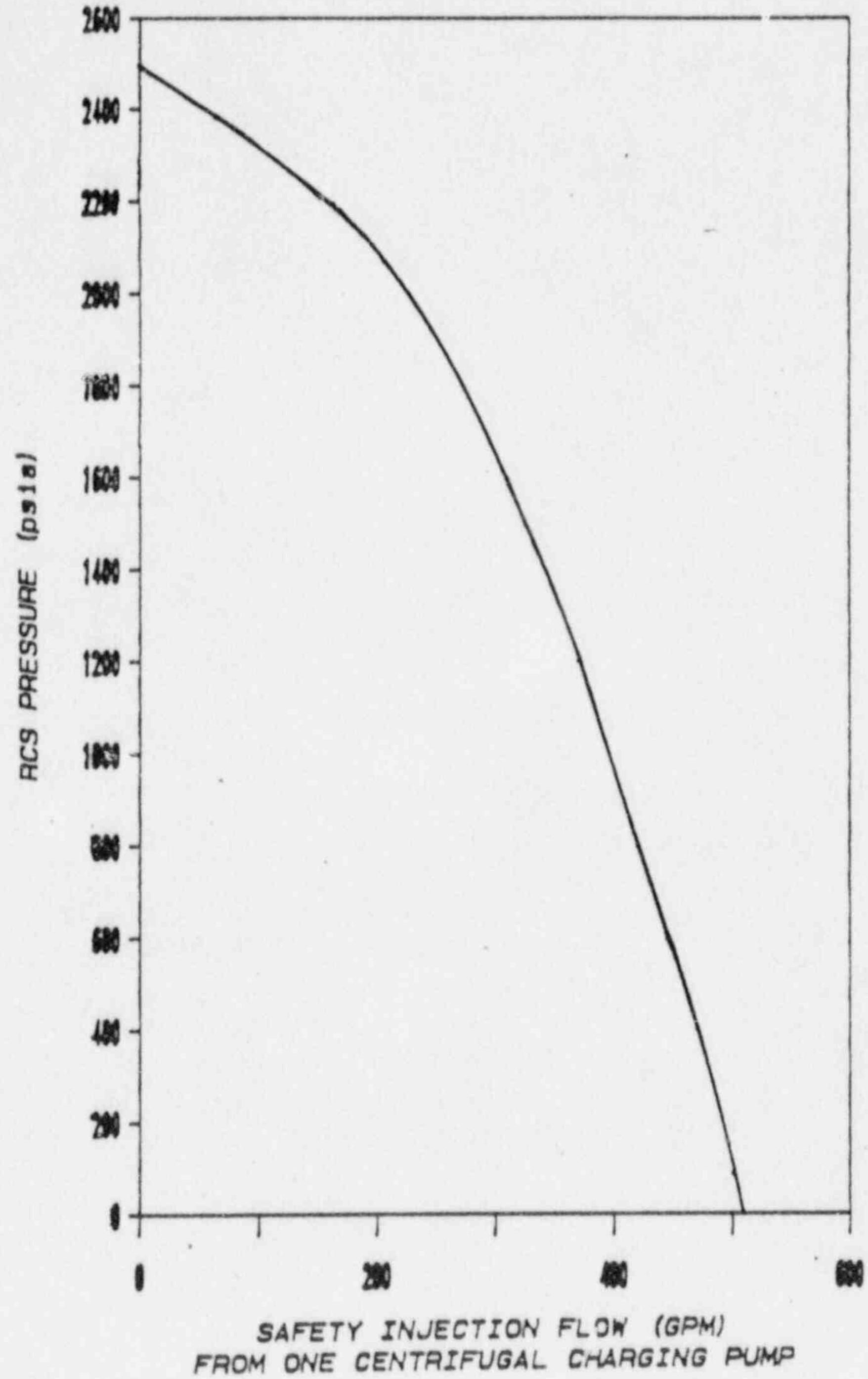
V. C. Summer

Figure 15.2.12-3

Accidental Depressurization of the Reactor Coolant System DNBR vs. Time



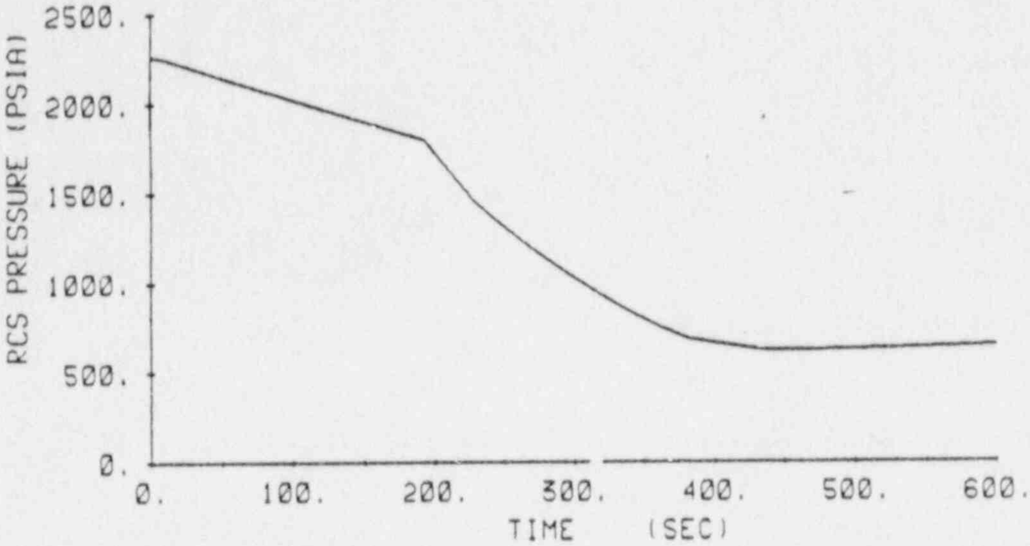
V. C. Summer
Figure 15.2.13-1
Main Steam Depressurization Variation of Keff with Core Temperature

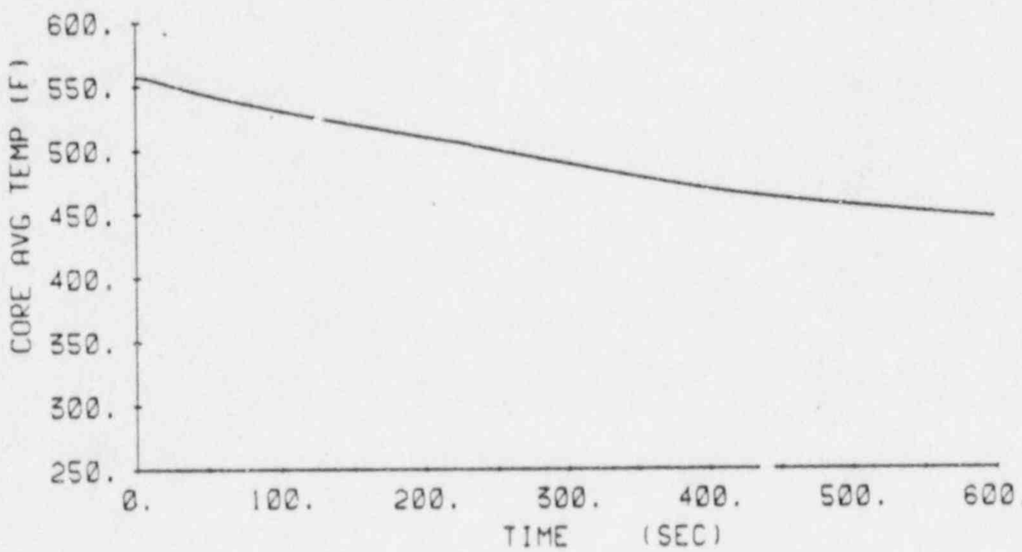


V. C. Summer

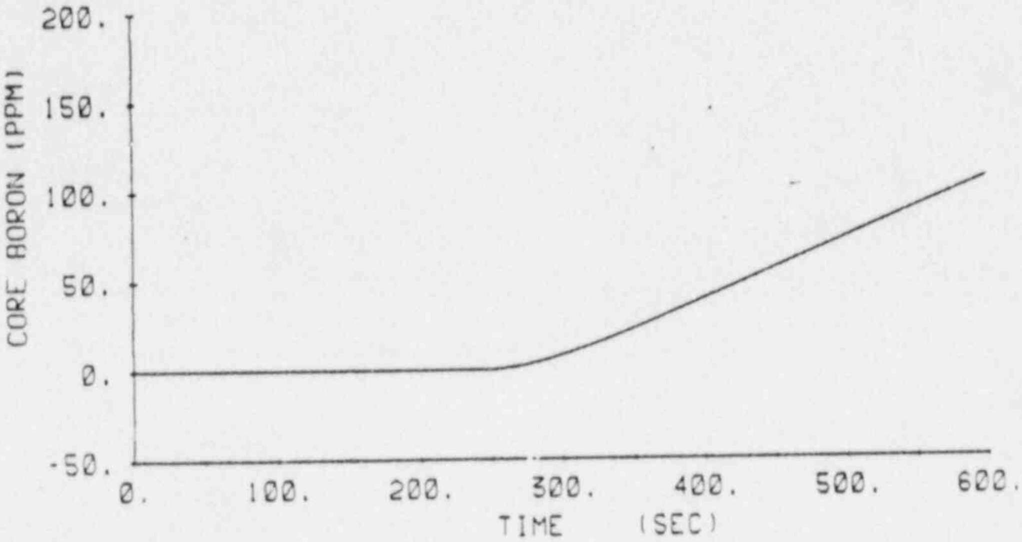
Figure 15.2.13-2

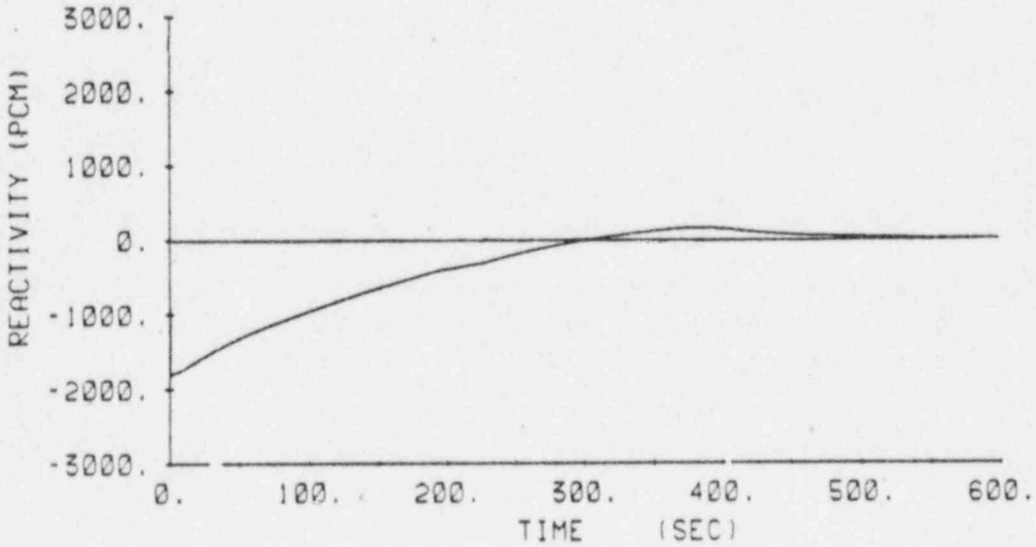
Main Steam Depressurization
Safety Injection Flowrate



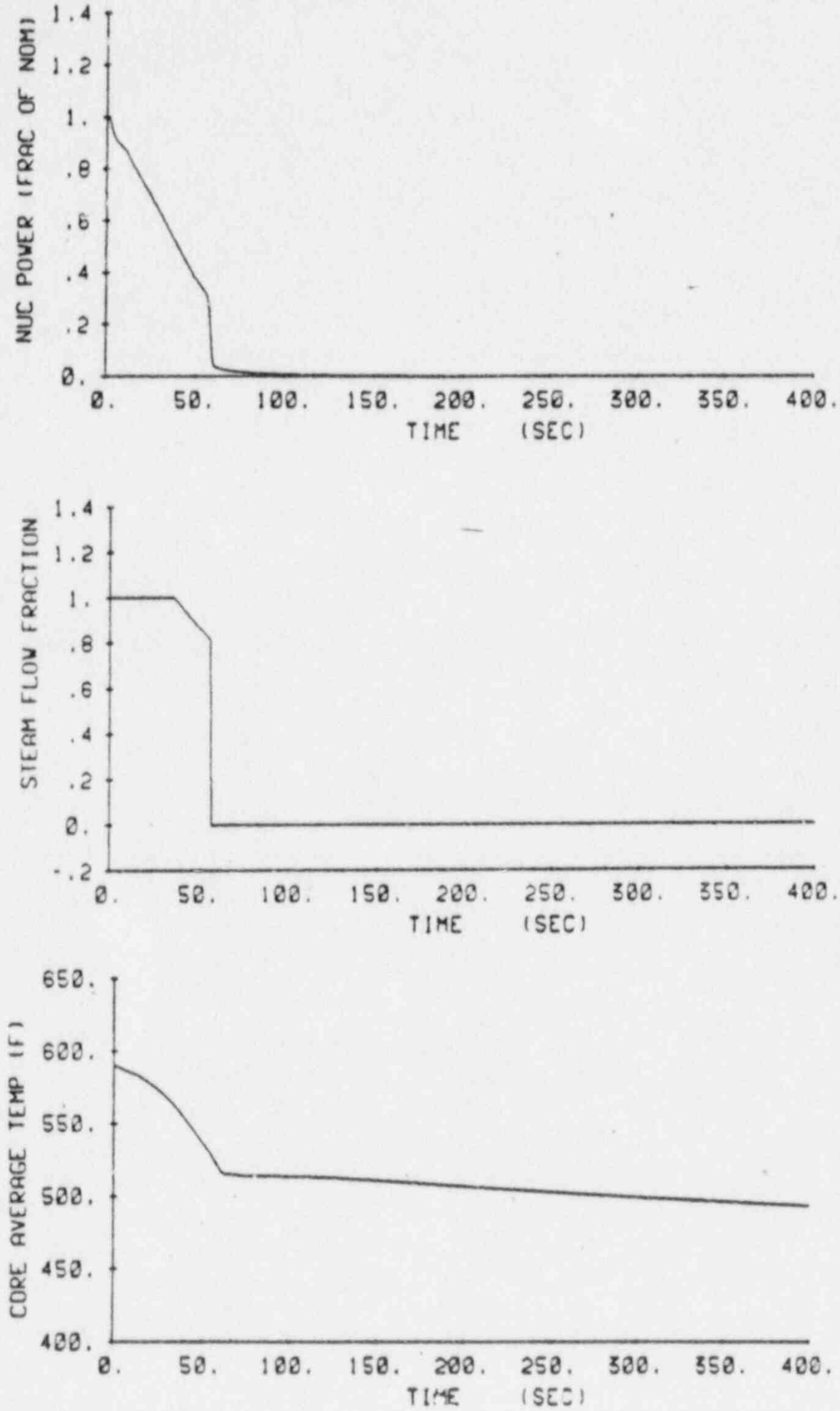


V. C. Summer
Figure 15.2.13-3
Transient Response For A Steam Line Break Equivalent To 255 lb/sec at 1100 psia w/ Offsite Power Available

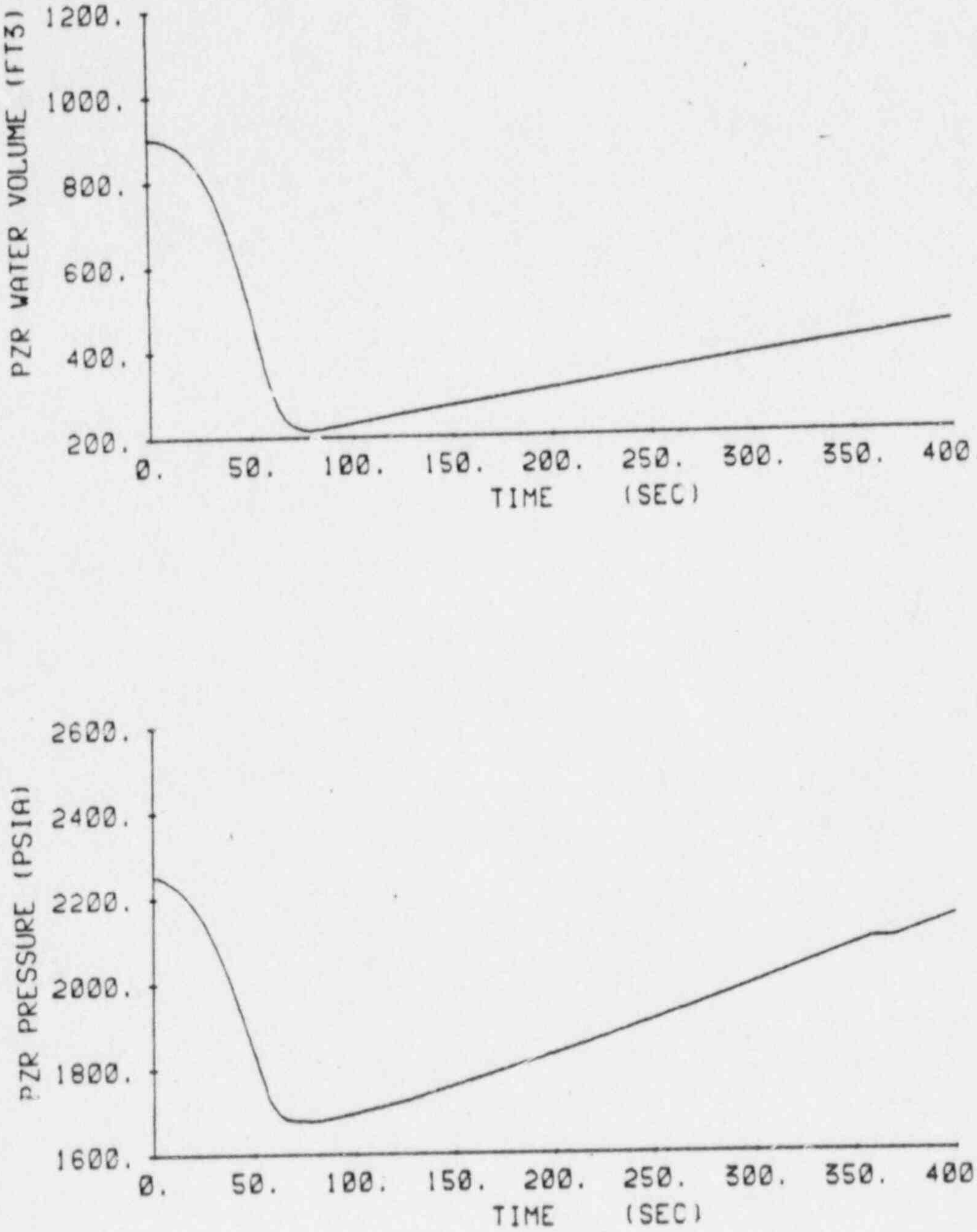




V. C. Summer
Figure 15.2.13-4
Transient Response For A Steam Line Break Equivalent To 255 lb/sec at 1100 psia w/ Offsite Power Available



V. C. Summer
Figure 15.2.14-1
Spurious Actuation Of The Safety Injection System Nuclear Power, Steam Flow and Core Tavg vs. Time



V. C. Summer

Figure 15.2.14-2

Spurious Actuation Of The Safety Injection System Pressurizer Water Volume and Pressurizer Pressure vs. Time

15.3 CONDITION III - INFREQUENT FAULTS

By definition, Condition III occurrences are faults which may occur very infrequently during the life of the plant. They will be accommodated with the failure of only a small fraction of the fuel rods although sufficient fuel damage might occur to preclude resumption of the operation for a considerable outage time. The release of radioactivity will not be sufficient to interrupt or restrict public use of those areas beyond the exclusion radius. A Condition III fault will not, by itself, generate a Condition IV fault or result in a consequential loss of function of the reactor coolant system or containment barriers. For the purpose of this report the following faults have been grouped into this category:

1. Loss of reactor coolant, from small ruptured pipes or from cracks in large pipes, which actuates the emergency core cooling system.

2. Minor secondary system pipe breaks.

3. Inadvertent loading of fuel assembly into an improper position.

4. Complete loss of forced reactor coolant flow.

5. Single rod cluster control assembly withdrawal at full power.

Each of these infrequent faults are analyzed in this section. In general, each analysis includes an identification of causes and description of the accident, an analysis of effects and consequences, a presentation of results, and relevant conclusions.

The time sequence of events during applicable Condition III faults 1 and 4 above is shown in Table 15.3-1.

15.3.2 Minor Secondary System Pipe Breaks

15.3.2.1 Identification of Causes and Accident Description

Included in this grouping are ruptures of secondary system lines which would result in steam release rates equivalent to a 6-inch-diameter break or smaller.

15.3.2.2 Analysis of Effects and Consequences

Minor secondary system pipe breaks must be accommodated with the failure of only a small fraction of the fuel elements in the reactor. Since the results of analysis presented in Section 15.4.2 for a major secondary system pipe rupture also meet these criteria, separate analyses for minor secondary system pipe breaks is not required.

The analyses of the more probable accidental opening of a secondary system steam dump, relief, or safety valve is presented in Section 15.2.13. These analyses are illustrative of a pipe break equivalent in size to a single valve opening.

15.3.2.3 Conclusions

The analysis presented in Section 15.4.2 demonstrates that the consequences of a minor secondary system pipe break are acceptable since a departure from nucleate boiling ratio (DNBR) of less than the design basis values does not occur even for a more critical major secondary system pipe break.

15.3.3 Inadvertent Loading of a Fuel Assembly Into an Improper Position

15.3.3.1 Identification of Causes and Accident Description

Fuel and core loading errors such as can arise from the inadvertent loading of one or more fuel assemblies into improper positions, loading a fuel rod during manufacture with one or more pellets of the wrong enrichment will lead to increased heat fluxes if the error results in placing fuel in core positions calling for fuel of lesser enrichment. Also included among possible core loading errors is the inadvertent loading of one or more fuel assemblies requiring burnable poison rods into a new core without burnable poison rods.

Any error in enrichment, beyond the normal manufacturing tolerances, can cause power shapes which are more peaked than those calculated with the correct enrichments. There is a 5 percent uncertainty margin included in the design value of power peaking factor assumed in the analysis of Condition I and Condition II transients. The incore system of movable flux detectors which is used to verify power shapes at the start of life is capable of revealing any assembly enrichment error or loading error which causes power shapes to be peaked in excess of the design value.

To reduce the probability of core loading errors, each fuel assembly is marked with an identification number and loaded in accordance with a core loading diagram. During core loading, the identification number will be checked before each assembly is moved into the core. Serial numbers read during fuel movement are subsequently recorded on the loading diagram as a further check on proper placing after the loading is completed.

The power distortion due to any combination of misplaced fuel assemblies would significantly raise peaking factors and would be readily observable with incore flux monitors. In addition to the flux monitors, thermocouples are located at the outlet of about one third of the fuel assemblies in the core. there is a high probability that these thermocouples would also indicate any abnormally high coolant enthalpy rise. Incore flux measurements are taken during the startup subsequent to every refueling operation.

15.3.3.2 Analysis of Effects and Consequences

Steady-state power distribution in the x-y plane of the core are calculated using the TURTLE[4] Code based on macroscopic cross section calculated by the LEOPARD[5] Code. A discrete representation is used wherein each individual fuel rod is described by a mesh interval. The power distribution in the x-y plane for a correctly loaded core assembly are also given in Chapter 4 of the FSAR based on enrichments given in that section.

For each core loading error case analyzed, the percent deviations from detector readings for a normally loaded core are shown at all incore detector locations (see Figures 15.3.3-1 to 15.3.3-5 inclusive).

15.3.3.3 Results

The following core loading error cases have been analyzed:

1. Case A

   Case in which a Region 1 assembly is interchanged with a Region 3 assembly. The particular case considered was the interchange of two adjacent assemblies near the periphery of the core (see Figure 15.3.3-1).

2. Case B

   Case in which a Region 1 assembly is interchanged with a neighboring Region 2 fuel assembly. Two analyses have been performed for this case (see Figures 15.3.3-2 and 15.3.3-3).

   In Case B-1, the interchange is assumed to take place with the burnable poison rods transferred with the Region 2 assembly mistakenly loaded into Region 1.

   In Case B-2, the interchange is assumed to take place closer to core center and with burnable poison rods located in the correct Region 2 position but in a Region 1 assembly mistakenly loaded into the Region 2 position.

3. Case C

   Enrichment error: Case in which a Region 2 fuel assembly is loaded in the core central position (see Figure 15.3.3-4).

4. Case D

   Case in which a Region 2 fuel assembly instead of a Region 1 assembly is loaded near the core periphery (see Figure 15.3.3-5).

15.3.3.4 Conclusions

Fuel assembly enrichment errors would be prevented by administrative procedures implemented in fabrication.

In the event that a single pin or pellet has a higher enrichment than the nominal value, the consequences in terms of reduced DNBR and increased fuel and clad temperatures will be limited to the incorrectly loaded pin or pins.

Fuel assembly loading errors are prevented by administrative procedures implemented during core loading. In the unlikely event that a loading error occurs, analyses in this section confirm that resulting power distribution effects will either be readily detected by the incore movable detector system or will cause a sufficiently small perturbation to be acceptable within the uncertainties allowed between nominal and design power shapes.

15.3.4 Complete Loss of Forced Reactor Coolant Flow

15.3.4.1 Identification of Causes and Accident Description

A complete loss of forced reactor coolant flow may result from a simultaneous loss of electrical supplies to all reactor coolant pumps. If the reactor is at power at the time of the accident, the immediate effect of a loss of forced coolant flow is a rapid increase in the coolant temperature. This increase could result in departure from nucleate boiling (DNB) with subsequent fuel damage if the reactor were not tripped promptly. The following provide necessary protection against a loss of coolant flow accident:

(1) Undervoltage or underfrequency on reactor coolant pump power supply buses;

(2) Low reactor coolant loop flow.

The reactor trip on reactor coolant pump bus undervoltage is provided to protect against conditions that can cause a loss of voltage to all reactor coolant pumps, i.e., station blackout. The reactor trip on reactor coolant pump underfrequency is provided to open the reactor coolant pump breakers and trip the reactor for an underfrequency condition, resulting from frequency disturbances on the major power grid. The trip disengages the reactor coolant pumps from the power grid so that the pumps' flywheel kinetic energy is available for full coastdown. Both trips are blocked below approximately 10% power (Permissive 7).

The reactor trip on low primary coolant loop flow is provided to protect against loss-of-flow conditions that affect only one reactor coolant loop. It also serves as a backup to the undervoltage and underfrequency trips. This function is generated by two-out-of-three low-flow signals per reactor coolant loop. Above approximately 38% power (Permissive 8), low flow in any loop will actuate a reactor trip. Between approximately 10 and 38% power (Permissive 7 and Permissive 8), low-flow in any two loops will actuate a reactor trip.

Normal power for each pump is supplied through individual busses connected to the isolated phase bus duct between the generator circuit breaker and the main transformer. Faults in the substation may cause a trip of the main transformer high side circuit breaker leaving the generator to supply power to the reactor coolant pumps. When a generator circuit breaker trip occurs because of electrical faults, the pumps are automatically transferred to an alternate power supply and the pumps will continue to supply coolant flow to the core. Following any turbine trip where there are no electrical faults, the generator circuit breaker is tripped and the reactor coolant pumps remain connected to the network through the transformer high side breaker. Continuity of power to the pump buses is achieved without motoring the generator since means are provided to isolate the generator without isolating the pump buses from the external power lines (e.g., a generator output breaker is provided as well as a station output breaker).

15.3.4.2 Analysis of Effects and Consequences

This transient is analyzed by three digital computer codes. First, the LOFTRAN code[1] is used to calculate the loop and core flow during the transient. The LOFTRAN code is also used to calculate the time of reactor trip based on the calculated flows and the nuclear power transient following reactor trip. The FACTRAN code[2] is then used to calculate the heat flux transient based on the nuclear power and flow from LOFTRAN. Finally, the THINC code[3] is used to calculate the minimum DNBR during the transient based on the heat flux from FACTRAN and flow from LOFTRAN. The transients presented represent the minimum of the typical and thimble cells for Standard and VANTAGE 5 fuel.

The following case has been analyzed:

All loops operating, all loops coasting down.

The method of analysis and the assumptions made regarding initial operating conditions and reactivity coefficients are identical to those discussed in Section 15.2, except that following the loss of supply to all pumps at power, a reactor trip is actuated by either bus undervoltage or bus underfrequency.

15.3.4.3 Results

The calculated sequence of events is shown in Table 15.3-3. Figures 15.3.4-1 and 15.3.4-2 show the flow coastdown, nuclear power and heat flux transients and minimum DNBR for the limiting complete loss of flow event. The reactor is assumed to trip on the undervoltage signal. The DNBR versus time plot represents the limiting cell for the three-loop coastdown.

15.3.4.4 Conclusions

The analysis performed has demonstrated that for the complete loss of forced reactor coolant flow, the DNBR does not decrease below the safety analysis limit values during the transient, and thus, no core safety limit is violated.

15.3.6 Single Rod Cluster Control Assembly Withdrawal at Full Power

15.3.6.1 Identification of Causes and Accident Description

No single electrical or mechanical failure in the rod control system could cause the accidental withdrawal of a single RCCA from the inserted bank at full power operation. The operator could deliberately withdraw a single RCCA in the control bank; this feature is necessary in order to retrieve an assembly should one be accidentally dropped. In the extremely unlikely event of simultaneous electrical failures that could result in single RCCA withdrawal, rod deviation and rod control urgent failure would both be displayed on the plant annunciator, and the rod position indicators would indicate the relative positions of the assemblies in the bank. The urgent failure alarm also inhibits automatic rod motion in the group in which it occurs. Withdrawal of a single RCCA by operator action, whether deliberate or by a combination of errors, would result in activation of the same alarm and the same visual indications.

Each bank of RCCAs in the system is divided into two groups of four mechanisms each. The rods comprising a group operate in parallel through multiplexing thyristors. The two groups in a bank move sequentially such that the first group is always within one step of the second group in the bank. A definite schedule of actuation and deactuation of the stationary gripper, movable gripper, and lift coils of a mechanism is required to withdraw the RCCA attached to the mechanism. Since the four stationary grippers, movable grippers, and lift coils associated with the four RCCAs of a rod group are driven in parallel, any single failure that would cause rod withdrawal would affect a minimum of one group, or four RCCAs. Mechanical failures are either in the direction of insertion or immobility.

In the unlikely event of multiple failures that result in continuous withdrawal of a single RCCA, it is not possible, in all cases, to provide assurance of automatic reactor trip so that core safety limits are not violated. Withdrawal of a single RCCA results in both positive reactivity insertion tending to increase core power, and an increase in local power density in the core area covered by the RCCA.

15.3.6.2 Analysis of Effects and Consequences

Power distributions within the core are calculated by the TURTLE code based on a macroscopic cross section generated by LEOPARD. The peaking factors calculated by TURTLE are then used by THINC to calculate the minimum DNB for the event. The plant was analyzed for the case of the worst rod withdrawn from Bank C inserted at the insertion limit, with the reactor initially at full power.

15.3.6.3 Results

Two cases have been considered as follows:

(1) If the reactor is in the automatic control mode, withdrawal of a single RCCA will result in the immobility of the other RCCAs in the controlling bank. The transient will then proceed in the same manner as Case 2 described below. For such cases as above, a trip will ultimately ensue, although not sufficiently fast in all cases to prevent a minimum DNBR in the core of less than the safety limit.

(2) If the reactor is in the manual control mode, continuous withdrawal of a single RCCA results in both an increase in core power and coolant temperature, and an increase in the local hot channel factor in the area of the failed RCCA. In terms of the overall system response, this case is similar to those presented in Section 15.2; however, the increased local power peaking in the area of the withdrawn RCCA results in lower minimum DNBR than for the withdrawn bank cases. Depending on initial bank insertion and location of the withdrawn RCCA, automatic reactor trip may not occur sufficiently fast to prevent the minimum core DNBR from falling below the safety limit value. Evaluation of this case at the power and coolant condition at which overtemperature ΔT trip would be expected to trip the plant shows that an upper limit for the number of rods with a DNBR less than the safety limit value is 5%.

15.3.6.4 Conclusions

For the case of one RCCA fully withdrawn, with the reactor in either the automatic or manual control mode and initially operating at full power with Bank D at the insertion limit, an upper bound of the number of fuel rods experiencing DNBR less than the design limit is 5% or less of the total fuel rods in the core.

For both cases discussed, the indicators and alarms mentioned would function to alert the operator to the malfunction before DNB could occur. For Case 2 discussed above, the insertion limit alarms (low and low-low alarms) would also serve in this regard.

15.3.7 References

1. T. W. T. Burnett, et. al., LOFTRAN Code Description, WCAP-7907-P-A (Proprietary), WCAP-7907-A (Non-Proprietary), April 1984.

2. H. G. Hargrove, FACTRAN-A Fortran-IV Code for Thermal Transients in a UO_2 Fuel Rod, WCAP-7908, June 1972.

3. J. S. Shefcheck, Application of the THINC Program to PWR Design, WCAP-7359-L, August 1969 (proprietary), WCAP-7838, January 1972 (Non-Proprietary).

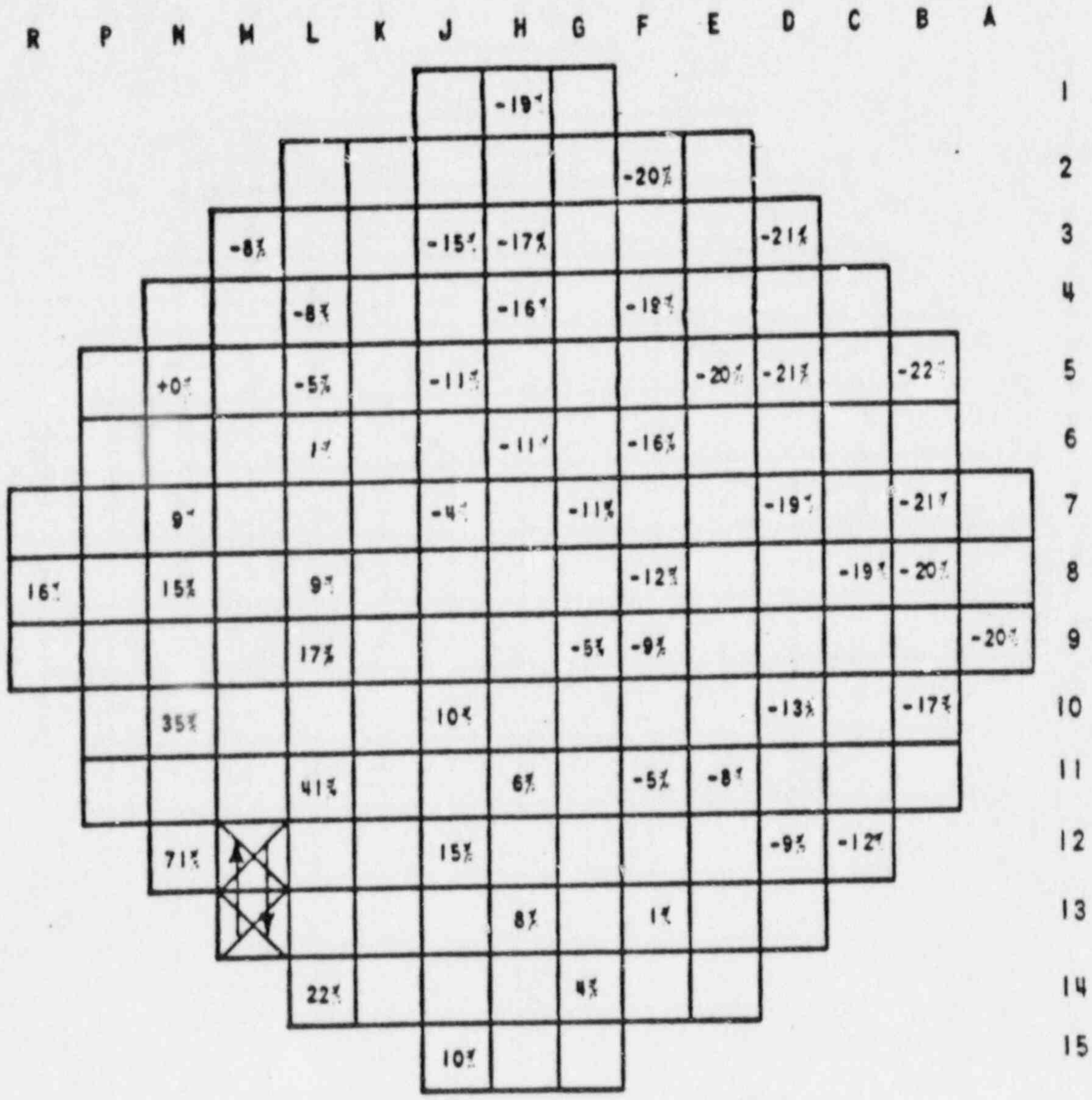
4. R. F. Barry and S. Altomare, The TURTLE 24.0 Diffusion Depletion Code, WCAP-7213-P-A (Proprietary), WCAP-7758-A (Non-Proprietary), January 1975.

5. R. F. Barry, LEOPARD-A Spectrum Dependent Non-Spatial Depletion Code for the IBM-7904, WCAP-3269-26, September 1963.

TABLE 15.3-3

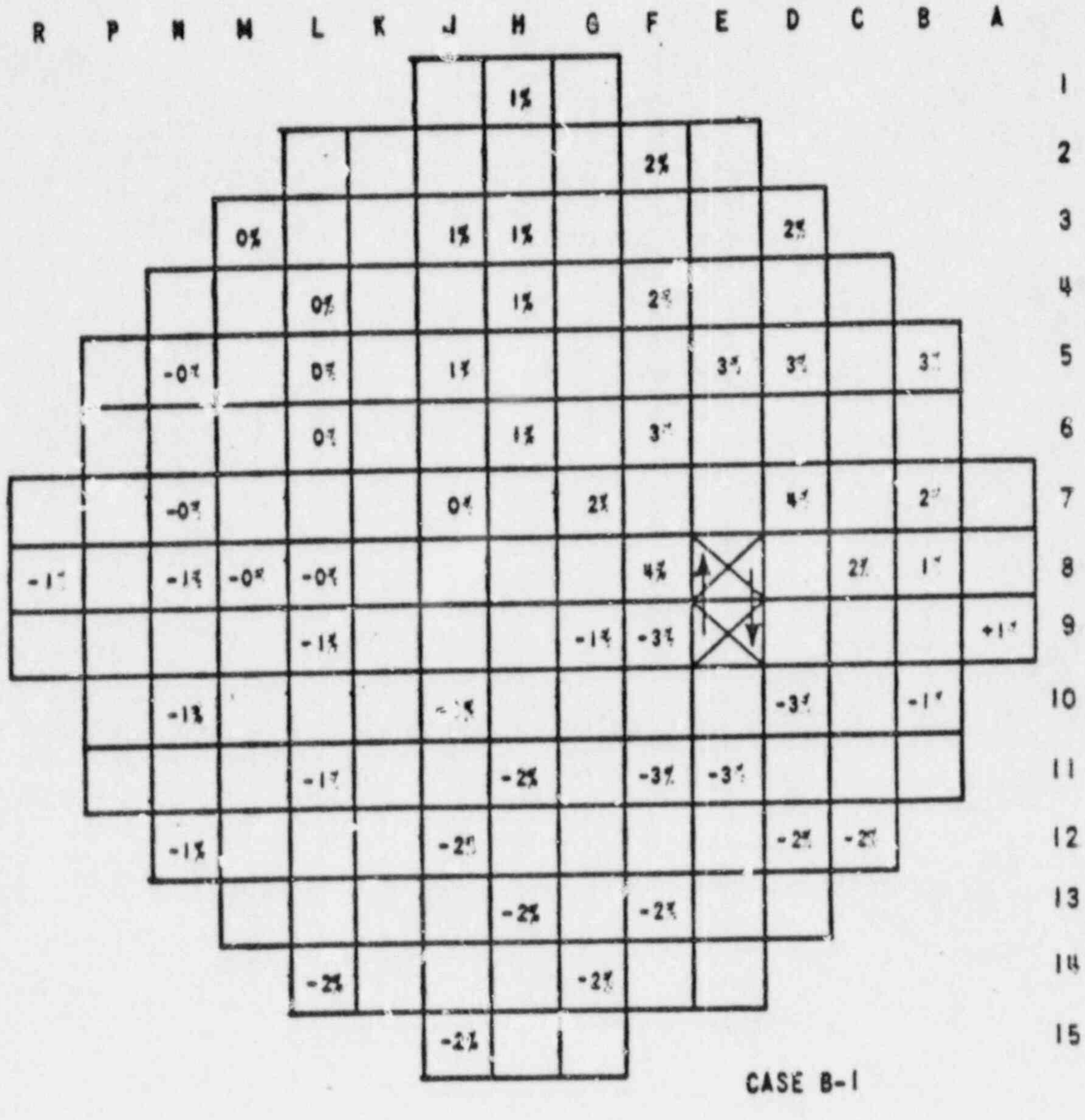
TIME SEQUENCE OF EVENTS FOR CONDITION III EVENTS

Accident	Event	[illegible] sec
Complete Loss of Forced Reactor Coolant Flow		
All loops operating, all pumps coasting down	Coastdown begins	0.0
	Rod motion begins	1.5
	Minimum DNBR occurs	3.4



CASE A

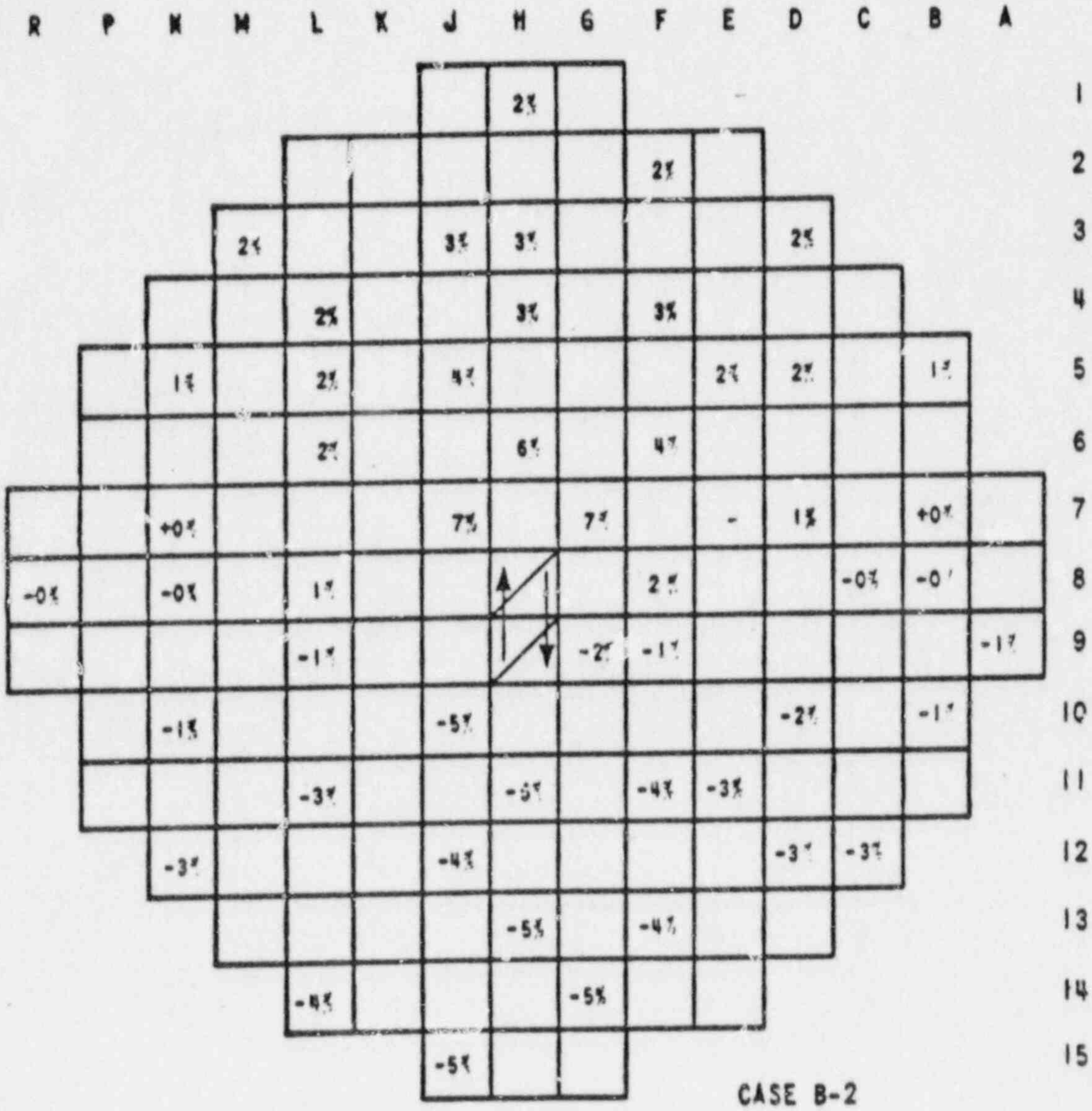
V. C. Summer
Figure 15.3.3-1
Inadvertent Fuel Misloading Interchange of Region 1 and Region 3 Assembly



V. C. Summer

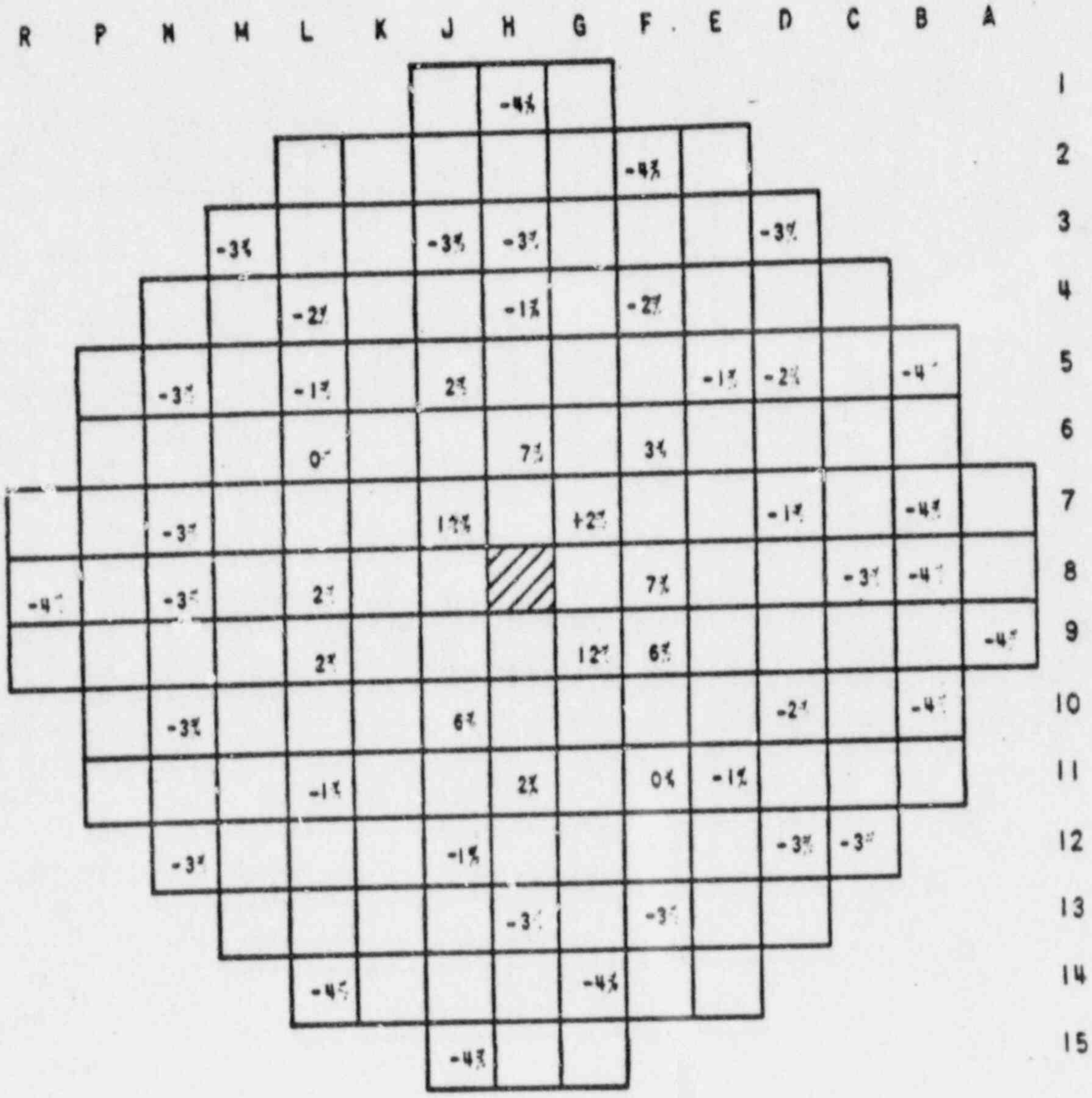
Figure 15.3.3-2

Inadvertent Fuel Misloading Interchange of Region 1 and Region 2 Assembly, Poison Rods Retained In Region 2 Assembly



CASE B-2

V. C. Summer
Figure 15.3.3-3
Inadvertent Fuel Misloading Interchange of Region 1 and Region 2 Assembly, Poison Rods Transferred To Region 1 Assembly

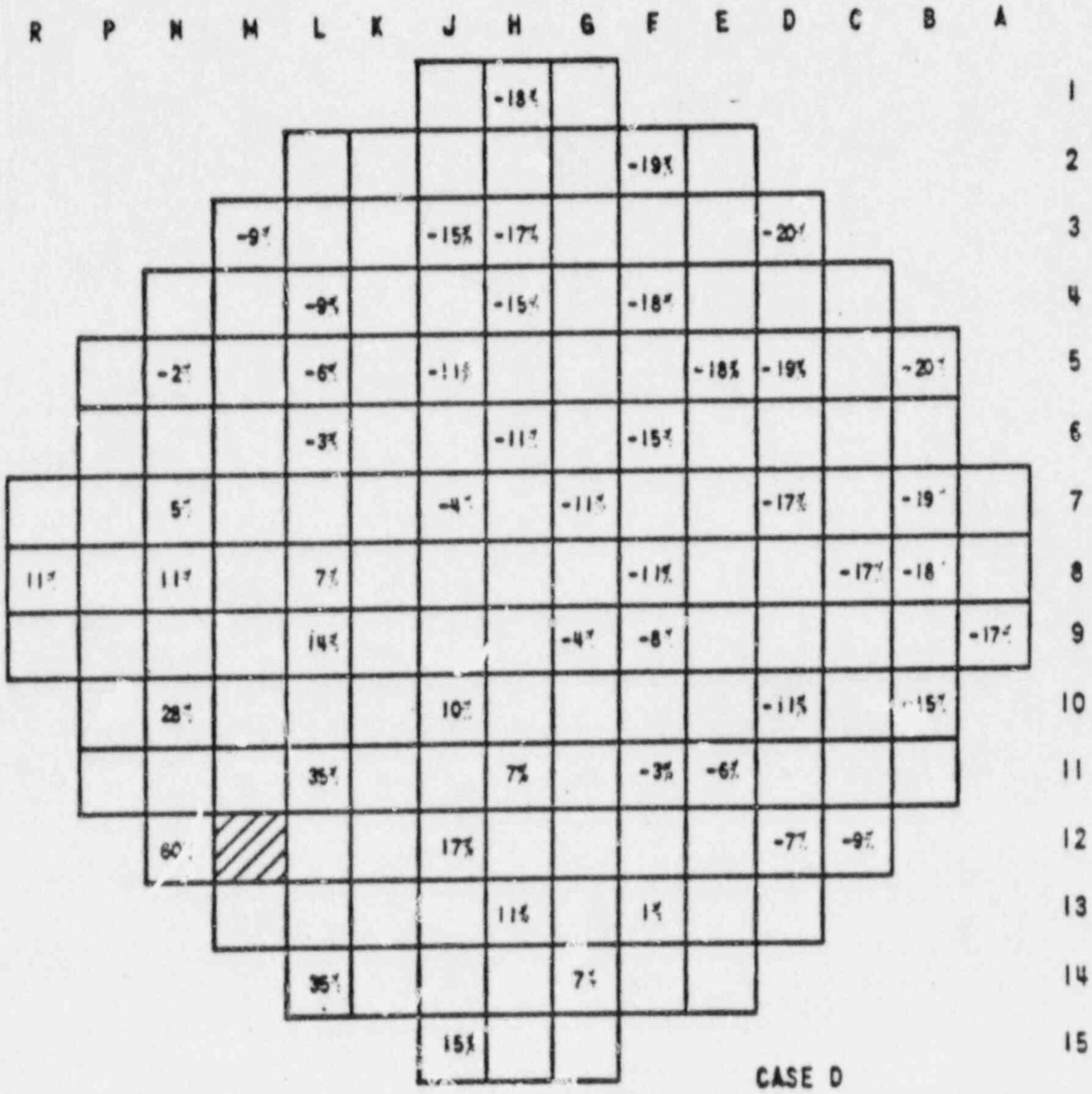


CASE C

V. C. Summer

Figure 15.3.3-4

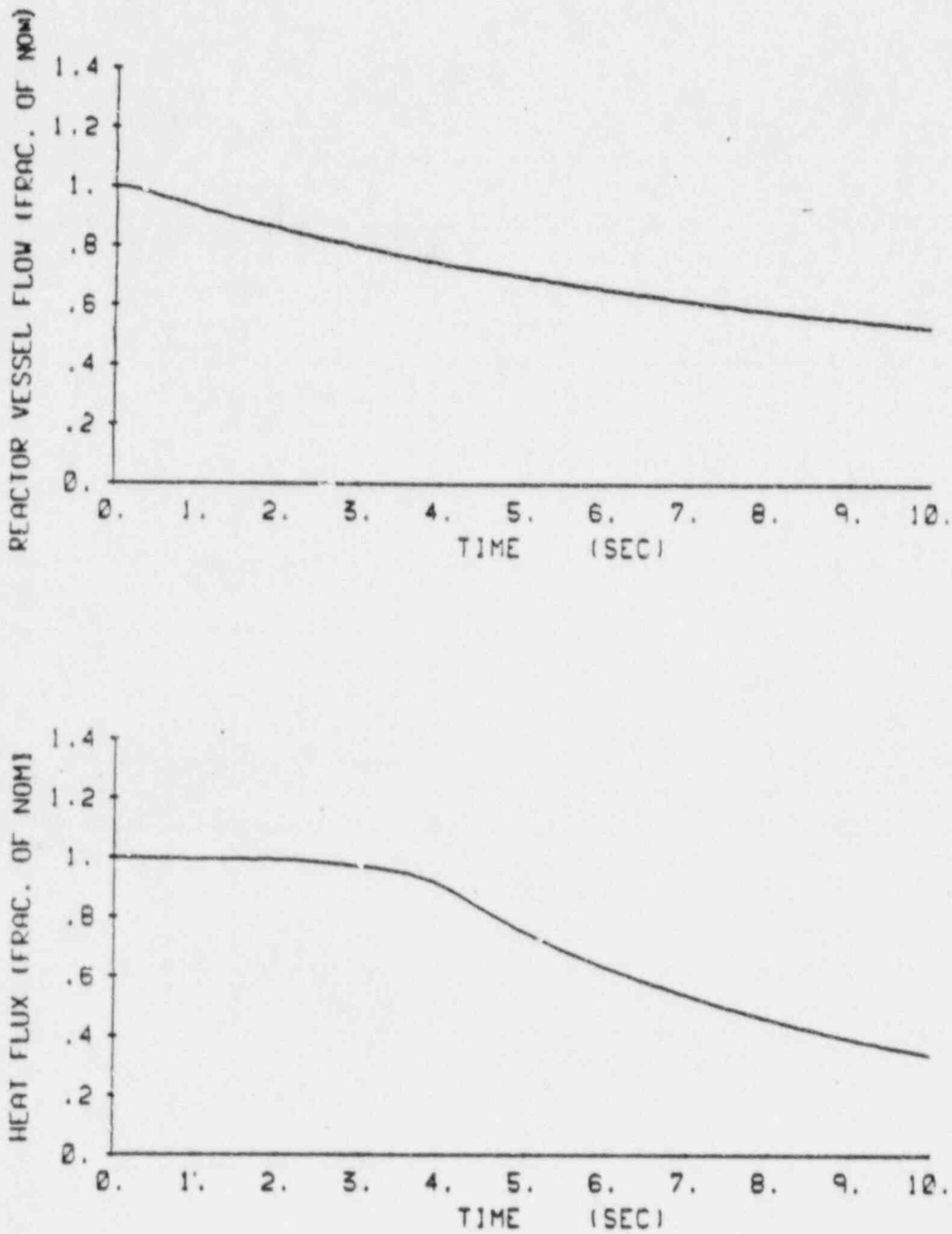
Inadvertent Fuel Misloading
Enrichment Error,
A Region 2 Assembly,
Loaded Into The Core Center



V. C. Summer

Figure 15.3.3-5

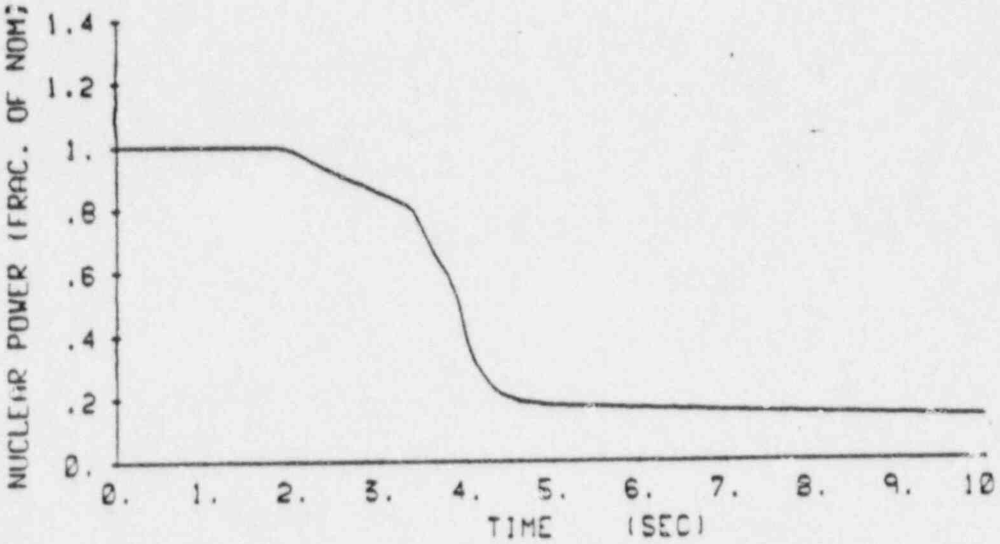
Inadvertent Fuel Misloading
A Region 2 Assembly Loaded
Into A Region 1 Position
Near Core Periphery

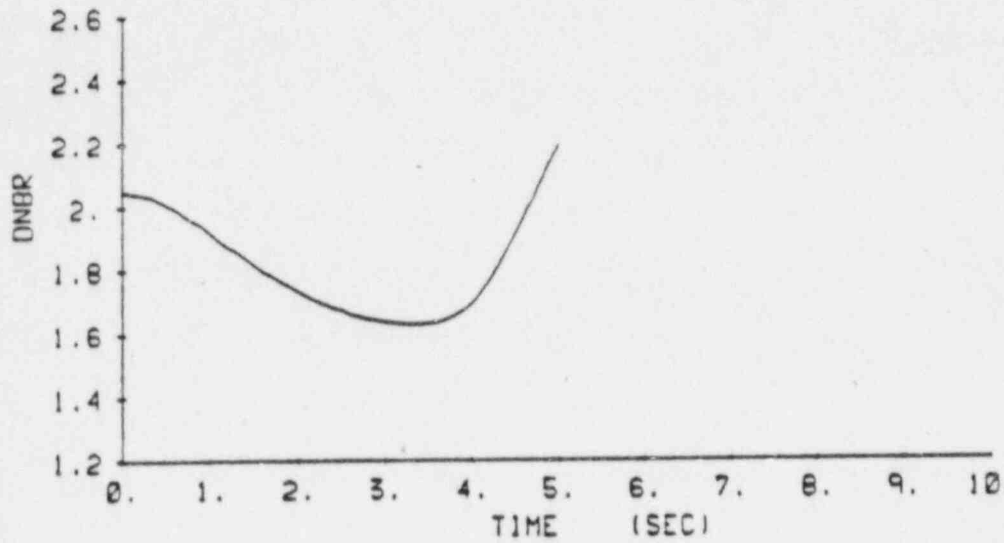


V. C. Summer

Figure 15.3.4-1

All Loops Operating
All Loops Coasting Down
Vessel Flow and Heat
Flux vs. Time





V. C. Summer
Figure 15.3.4-2
All Loops Operating All Loops Coasting Down Nuclear Power and DNBR vs. Time

15.4 CONDITION IV - LIMITING FAULTS

Condition IV occurrences are faults that are not expected to take place, but are postulated because their consequences would include the potential for the release of significant amounts of radioactive material. These are the most drastic occurrences that must be designed against and represent limiting design cases. Condition IV faults shall not cause a fission product release to the environment resulting in an undue risk to public health and safety in excess of guideline values of 10 CFR 100. A single Condition IV fault shall not cause a consequential loss of required functions of systems needed to cope with the fault including those of the emergency core cooling system (ECCS) and the containment. For the purposes of this report the following faults have been classified in this category:

(1) Major rupture of pipes containing reactor coolant up to and including double-ended rupture of the largest pipe in the reactor coolant system (RC^), i.e., loss-of-coolant accident (LOCA);

(2) Major secondary system pipe ruptures;

(3) Steam generator tube rupture;

(4) Single reactor coolant pump (RCP) locked rotor;

(5) Rupture of a control rod mechanism housing (rod cluster control assembly [RCCA] ejection).

Each of these five limiting faults is analyzed in Section 15.4. In general, each analysis includes an identification of causes and description of the accident, an analysis of effects and consequences, a presentation of results, and relevant conclusions.

15.4.2 Major Secondary System Pipe Rupture

Two major secondary system pipe ruptures are analyzed in this section: rupture of a main steam line and rupture of a main feedwater pipe. The time sequence of events for each of these events is provided in Table 15.4-8.

15.4.2.1 Rupture of a Main Steam Line

15.4.2.1.1 Identification of Causes and Accident Description

The steam release arising from a rupture of a main steam pipe would result in an initial increase in steam flow that decreases during the accident as the steam pressure falls. The energy removal from the RCS causes a reduction of coolant temperature and pressure. In the presence of a negative moderator temperature coefficient, the cooldown results in a reduction of core shutdown margin. If the most reactive RCCA is assumed stuck in its fully withdrawn position after reactor trip, there is an increased possibility that the core will become critical and return to power. A return to power following a steam pipe rupture is a potential problem mainly because of the high power peaking factors that exist assuming the most reactive RCCA to be stuck in its fully withdrawn position. The core is ultimately shut down by the boric acid injection delivered by the SIS.

The analysis of a main steam pipe rupture is performed to demonstrate that the following criteria are satisfied:

(1) Assuming a stuck RCCA, with or without offsite power, and assuming a single failure in the engineered safety features (ESF) there is no consequential damage to the primary system and the core remains in place and intact;

(2) Energy release to containment from the worst steam pipe break does not cause failure of the containment structure.

Although DNB and possible cladding perforation following a steam pipe rupture are not necessarily unacceptable, the following analysis, in fact, shows that no DNB occurs for any rupture assuming the most reactive assembly stuck in its fully withdrawn position.

The following functions provide the necessary protection against a steam pipe rupture:

(1) SIS actuation from any of the following:

(a) Two-out-of-three low pressurizer pressure signals;

(b) High differential pressure signals between steam lines;

(c) Two-out-of-three low steam line pressure signals;

(d) Two-out-of-three high-1 containment pressure signals.

(2) The overpower reactor trips (neutron flux and ΔT) and the reactor trip occurring in conjunction with receipt of the safety injection signal.

(3) Redundant isolation of the main feedwater lines: sustained high feedwater flow would cause additional cooldown. Therefore, a safety injection signal will rapidly close all feedwater control valves, trip the main feedwater pumps, and close the feedwater isolation valves that backup the control valves.

(4) Trip of the main steam line isolation valves on: (See Technical Specifications[1] Table 3.3-5)

(a) High steam flow in two-out-of-three main steam lines in coincidence with two-out-of-three low-low T_{avg} signals;

(b) High-2 containment pressure signal;

(c) Two-out-of-three low steam line pressure signals.

For breaks downstream of the isolation valves, closure of all valves would completely terminate the blowdown. For any break, in any location, no more than one steam generator would blow down even if one of the isolation valves fails to close. A description of steam line isolation is included in Chapter 10 of the FSAR.

Steam flow is measured by monitoring dynamic head inside the steam pipes. Nozzles that are of considerably smaller diameter than the main steam pipe are located in the steam generators and serve to limit the maximum steam flow for any break at any location.

15.4.2.1.2 Analysis of Effects and Consequences

The analysis of the steam pipe rupture has been performed to determine:

(1) The core heat flux and RCS temperature and pressure resulting from the cooldown following the steam line break. The LOFTRAN code[2] has been used.

(2) The thermal and hydraulic behavior of the core following a steam line break. A detailed thermal and hydraulic digital-computer code, THINC[3], has been used to determine if DNB occurs for the core conditions computed in (1) above.

The following conditions were assumed to exist at the time of a main steam line break accident.

(1) End of life (EOL) shutdown margin at no-load, equilibrium xenon conditions, and the most reactive assembly stuck in its fully withdrawn position. Operation of the control rod banks during core burnup is restricted in such a way that addition of positive reactivity in a steam line break accident will not lead to a more adverse condition than the case analyzed.

(2) The negative moderator coefficient corresponding to the EOL rodded core with the most reactive rod in the fully withdrawn position. The variation of the coefficient with temperature and pressure has been included. The k_{eff} versus temperature at 1150 psia corresponding to the negative moderator temperature coefficient plus the Doppler temperature effect used is shown on Figure 15.2.13-1. The effect of power generation in the core on overall reactivity is shown on Figure 15.4.2-1.

The core properties associated with the sector nearest the affected steam generator and those associated with the remaining sector were conservatively combined to obtain average core properties for reactivity feedback calculations. Further, it was conservatively assumed that the core power distribution was uniform. These two conditions cause underprediction of the reactivity feedback in the high-power region near the stuck rod. To verify the conservatism of this method, the reactivity as well as the power distribution was checked. These core analyses considered the Doppler reactivity from the high fuel temperature near the stuck RCCA, moderator feedback from the high water enthalpy near the stuck RCCA, power redistribution and nonuniform core inlet temperature effects. For cases in which steam generation occurs in the high flux regions of the core, the effect of void formation was also included. It was determined that the reactivity employed in the kinetics analysis was always larger than the true reactivity verifying conservatism; i.e., underprediction of negative reactivity feedback from power generation.

(3) Minimum capability for injection of high concentration boric acid (2300 ppm) solution corresponding to the most restrictive single failure in the SIS. The characteristics of the injection unit used are shown on Figure 15.2.13-2. This corresponds to the flow delivered by one charging pump delivering its full flow to the cold leg header. No credit has been taken for the low concentration of boric acid that

must be swept from the safety injection lines downstream of the refueling water storage tank (RWST) isolation valves prior to the delivery of highly concentrated boric acid to the reactor coolant loops. This effect has been allowed for in the analysis. The modeling of the SIS in LOFTRAN is described in Reference 2.

For the case where offsite power is assumed, the sequence of events in the SIS is the following: After the generation of the safety injection signal (appropriate delays for instrumentation, logic, and signal transport included), the appropriate valves begin to operate and the high-head injection pump starts. In 27 seconds, the valves are assumed to be in their final position and the pump is assumed to be at full speed. The volume containing the low concentration borated water is swept before the 2300 ppm boron reaches the core. This delay is inherently included in the modeling. In cases where offsite power is not available, an additional 10-second delay is assumed to be required to start the diesels and to load the necessary safety injection equipment onto them. That is, after a total of 37 seconds following an SIS signal, the SIS is assumed to be capable of delivering flow to the RCS.

(4) Two cases have been considered in determining the core power and RCS transients:

   (a) Complete severance of a pipe with the plant initially at no-load conditions, full reactor coolant flow with offsite power available,

   (b) Complete severance of a pipe with the plant initially at no-load conditions with offsite power unavailable.

(5) Power peaking factors corresponding to one stuck RCCA and nonuniform core inlet coolant temperatures are determined at EOL. The coldest core inlet temperatures are assumed to occur in the sector with the stuck rod. The power peaking factors account for the effect of the local void in the region of the stuck control assembly during the

return to power phase following the steam line break. This void in conjunction with the large negative moderator coefficient partially offsets the effect of the stuck assembly. The power peaking factors depend on the core power, operating history, temperature, pressure, and flow, and thus are different for each case studied.

Both cases assume initial hot shutdown conditions at time zero since this represents the most pessimistic initial condition. Should the reactor be just critical or operating at power at the time of a steam line break, the reactor will be tripped by the normal overpower protection system when power level reaches a trip point. Following a trip at power the RCS contains more stored energy than at no-load, the average coolant temperature is higher than at no-load, and there is appreciable energy stored in the fuel. Thus, the additional stored energy is removed via the cooldown caused by the steam line break before the no-load conditions of RCS temperature and shutdown margin assumed in the analyses are reached. After the additional stored energy has been removed, the cooldown and reactivity insertions proceed in the same manner as in the analysis which assumes no-load condition at time zero.

However, since the initial steam generator water inventory is greatest at no-load, the magnitude and duration of the RCS cooldown are less for steam line breaks occurring at power.

(6) In computing the steam flow during a steam line break, the Moody Curve[4] for fL/D = 0 is used. The Moody Multiplier is 1 with a discharge at dry saturated steam conditions.

(7) Perfect moisture separation in the steam generator is assumed. The assumption leads to conservative results since, in fact, considerable water would be discharged. Water carryover would reduce the magnitude of the temperature decrease in the core and the pressure increase in the containment.

15.4.2.1.3 Results

The results presented are a conservative indication of the events that would occur assuming a steam line rupture since it is postulated that all of the conditions described above occur simultaneously.

Figures 15.4.2-2 and 15.4.2-3 show the response of pertinent system parameters following a main steam pipe rupture. Offsite power is assumed to be available such that full reactor coolant flow exists. The transient shown assumes an uncontrolled steam release from only one steam generator.

As can be seen, the core attains criticality with RCCAs inserted (with the design shutdown assuming one stuck RCCA) before boric acid solution at 2300 ppm enters the RCS from the SIS which is drawing from the RWST. The delay time consists of the time to receive and actuate the safety injection signal and the time to completely open valve trains in the safety injection lines. The safety injection pumps are then ready to deliver flow. At this stage, a further delay is incurred before 2300 ppm boron solution can be injected to the RCS due to the low concentration solution being swept from the safety injection lines. Should a partial loss of offsite power occur such that power is lost to the ESF functions while the reactor coolant pumps remain in operation, an additional safety injection delay of 10 seconds would occur while the diesel generators startup and the necessary safety injection equipment is loaded onto them. A peak core power well below the nominal full power value is attained.

The calculation assumes the boric acid is mixed with and diluted by the water flowing in the RCS prior to entering the reactor core. The concentration after mixing depends on the relative flowrates in the RCS and the SIS. The variation of mass flowrate in the RCS due to water density changes is included in the calculation as is the variation of flowrate from the SIS and the accumulator due to changes in the RCS pressure. The SIS flow calculation includes the line losses in the system as well as the pump head curve. The accumulators provide an additional source of borated water after the RCS pressure has decreased to below 600 psia.

Should the core be critical at near zero power when the rupture occurs, the initiation of safety injection by high differential pressure between any steam line and the remaining steam lines or low steam line pressure will trip the reactor. Steam release from more than one steam generator will be prevented by automatic trip of the isolation valves in the steam lines by low steam line pressure or the high steam flow signal in coincidence with low-low RCS temperature. The steam line isolation valves are designed to be fully closed in less than 5 seconds after receipt of closure signal.

Figures 15.4.2-4 and 15.4.2-5 show the responses of the salient parameters for the case discussed above with a total loss of offsite power at the time of the rupture. This results in a coastdown of the reactor coolant pumps. In this case, the core power increases at a slower rate and reaches a lower peak value than in the cases in which offsite power is available to the reactor coolant pumps. The ability of the emptying steam generator to extract heat from the RCS is reduced by the decreased flow in the RCS.

It should be noted that following a steam line break, only one steam generator blows down completely. Thus, the remaining steam generators are still available for dissipation of decay heat after the initial transient is over. In case of a loss of offsite power, this heat is removed to the atmosphere via the steam line safety valves.

15.4.2.1.4 Conclusion

A DNB analysis was performed for the above cases. It was found that the DNB design basis[16] is met.

15.4.2.2 Major Rupture of a Main Feedwater Pipe

15.4.2.2.1 Identification of Causes and Accident Description

A major feedwater line rupture is defined as a break in a feedwater pipe large enough to prevent the addition of sufficient feedwater to the steam generators to maintain shell-side fluid inventory in the steam generators. If the break is postulated in a feedline between the check valve, the forward flush valve, or the reverse flush valve and the steam generator, fluid from the steam generator may also be discharged through the break. (A break upstream of the feedline check valve, or downstream of the forward or reverse flush valves would affect the nuclear steam supply system [NSSS] only as a loss of feedwater. This case is covered by the evaluation in Section 15.2.8.)

Depending on the size of the break and the plant operating conditions at the time of the break, the break could cause either an RCS cooldown (by excessive energy discharge through the break), or an RCS heatup. The potential RCS cooldown resulting from a secondary pipe rupture is evaluated in Section 15.4.2.1, Rupture of a Main Steam Pipe. Therefore, only the RCS heatup effects are evaluated for a feedline rupture.

A feedline rupture reduces the ability to remove heat generated by the core from the RCS for the following reasons:

(1) Feedwater to the steam generators is reduced. Since feedwater is subcooled, its loss may cause reactor coolant temperatures to increase prior to reactor trip;

(2) Liquid in the steam generator may be discharged through the break, and would then not be available for decay heat removal after trip;

(3) The break may be large enough to prevent the addition of any main feedwater after trip.

An emergency feedwater system is provided to assure that adequate feedwater will be available such that:

(1) No substantial overpressurization of the reactor coolant system shall occur; and

(2) Liquid in the reactor coolant system shall be sufficient to cover the reactor core at all times.

The following provide the necessary protection against a main feedwater line rupture:

(1) A reactor trip on any of the following conditions:

(a) High pressurizer pressure;

(b) Overtemperature ΔT;

(c) Low-low steam generator water level in any steam generator;

(d) Low steam generator level plus steam/feedwater flow mismatch in any steam generator;

(e) Safety injection signals from any of the following:

1. Low steam line pressure,

2. High containment pressure (Hi-1),

3. High steamline differential pressure.

(Refer to Chapter 7 for a description of the actuation system.)

(2) An emergency feedwater system to provide an assured source of feedwater to the steam generators for decay heat removal. (Refer to FSAR Section 10.4.9 for a description of the emergency feedwater system.)

15.4.2.2.2 Analysis of Effects and Consequences

A detailed analysis using the LOFTRAN[2] code is performed in order to determine the plant transient following a feedline rupture. The code describes the plant thermal kinetics, RCS including natural circulation, pressurizer, steam generators, and feedwater system, and computes pertinent variables, including the pressurizer pressure, pressurizer water level, and reactor coolant average temperature.

Major assumptions are:

(1) The plant is initially operating at 102% of the ESF design rating.

(2) Initial reactor coolant average temperature is 4.0°F above the nominal value, and the initial pressurizer pressure is 33 psi above its nominal value.

(3) A conservatively high initial pressurizer level is assumed; initial steam generator water level is at the nominal value plus 5% in the faulted steam generator, and at the nominal value minus 5% in the intact steam generators.

(4) No credit is taken for the pressurizer power-operated relief valves or pressurizer spray.

(5) No credit is taken for the high pressurizer pressure reactor trip.

Note: This assumption is made for calculational convenience. Pressurizer power-operated relief valves and spray could act to delay the high pressure trip. Assumptions 3 and 4 permit evaluation of one hypothetical, limiting case rather than two possible cases: one with a high pressure trip and no pressure control; and one with pressure control but no high pressure trip.

(6) Main feed to all steam generators is assumed to stop at the time the break occurs. (All main feedwater spills out through the break.)

(7) A conservative feedline break discharge quality is assumed prior to the time the reactor trip occurs, thereby maximizing the time the trip setpoint is reached. After the trip occurs, a saturated liquid discharge is assumed until all water inventory is discharged from the affected steam generator. This minimize the heat removal capability of the affected steam generator.

(8) Reactor trip is assumed to be initiated when the low-low level trip setpoint in the ruptured steam generator is reached. A low-low level setpoint of 0% narrow range span is assumed.

(9) The worst possible break area which minimizes the steam generator fluid inventory at the time of trip and is assumed maximizes the blowdown discharge rate following the time of trip, and thereby maximizes the resultant heatup of the reactor coolant.

(10) No credit is taken for heat energy deposited in RCS metal during the RCS heatup.

(11) No credit is taken for charging or letdown.

(12) Steam generator heat transfer area is assumed to decrease as the shell-side liquid inventory decreases.

(13) Conservative core residual heat generation based on long-term operation at the initial power level preceding the trip is assumed. The 1979 ANS 5.1[5] decay heat standard plus uncertainty was used for calculation of residual decay heat levels.

(14) The emergency feedwater system is actuated by the low-low steam generator water level signal. The emergency feedwater system is assumed to supply a total 380 gpm to the unaffected steam generators. A 60 second delay following reactor trip is assumed to allow time for startup of the emergency diesel generators and the emergency feed pumps. Before the relatively cold (120°F) emergency feedwater enters the unaffected steam generators, additional time is modeled to allow for the purging of 5 cubic feet of hot water contained in the emergency feedwater system lines.

15.4.2.2.3 Results

Results for two feedline break cases are presented. Results for a case in which offsite power is assumed to be available are presented in Section 15.4.2.2.3.1. Results for a case in which offsite power is assumed to be lost following reactor trip are presented in Section 15.4.2.2.3.2. The calculated sequence of events for both cases is listed in Table 15.4-8.

15.4.2.2.3.1 Feedline Rupture with Offsite Power Available

The system response following a feedwater line rupture, assuming offsite power is available, is presented in Figures 15.4.2-6 through 15.4.2-9. Results presented in Figures 15.4.2-7 and 15.4.2-9 show that pressures in the RCS and main steam system remain below 110% of the respective design pressures. Pressurizer pressure decreases after reactor trip on low-low steam generator water level due to the reduction of heat input. Following this initial decrease, pressurizer pressure increases to the pressurizer safety valve setpoint. This increase in pressure is the result of coolant expansion caused by the reduction in heat transfer capability in the steam generators. Figure 15.4.2-7 shows that the water volume in the pressurizer increases in response to the heatup, pressurizer water relief begins at 1512 seconds. At approximately 3300 seconds, decay heat generation decreases to a level such that the total RCS heat generation (decay heat plus pump heat) is less than emergency feedwater heat removal capability, and RCS pressure and temperature begin to decrease.

The results show that the core remains covered at all times and that no boiling occurs in the reactor coolant loops.

15.4.2.2.3.2 Feedline Rupture with Offsite Power Unavailable

The system response following a feedwater line rupture without offsite power available is similar to the case with offsite power available. However, as a result of the loss of offsite power (assumed to occur at reactor trip), the reactor coolant pumps coast down. This results in a reduction in total RCS heat generation by the amount produced by pump operation.

The reduction in total RCS heat generation produces a milder transient than in the case where offsite power is available. Results presented in Figures 15.4.2-11 and 15.4.2-13 show that pressure in the RCS and main steam system remain below 110% of the respective design pressures. Pressurizer pressure decreases after reactor trip on low-low steam generator water level due to the reduction of heat input. Following this initial decrease, pressurizer pressure increases to a peak pressure of 2502 psia at 746 seconds. This increase in pressure is the result of coolant expansion caused by the reduction in heat transfer capability in the steam generators. Figure 15.4.2-11 shows that the water volume in the pressurizer increases in response to the heatup but does not fill the pressurizer. At approximately 1200 seconds, decay heat generation decreases to a level less than the emergency feedwater heat removal capability, and RCS temperatures begin to decrease. The results show that the core remains covered at all times since the pressurizer does not empty.

15.4.2.2.4 Conclusion

Results of the analysis show that for the postulated feedline rupture, the assumed emergency feedwater system capacity is adequate to remove decay heat, to prevent overpressurizing the RCS, and to prevent uncovering the reactor core.

15.4.4 Single Reactor Coolant Pump Locked Rotor

15.4.4.1 Identification of Causes and Accident Description

The accident postulated is an instantaneous seizure of an RCP rotor. Flow through the affected reactor coolant loop is rapidly reduced, leading to an initiation of a reactor trip on a low flow signal.

Following initiation of the reactor trip, heat stored in the fuel rods continues to be transferred to the coolant causing the coolant to expand. At the same time, heat transfer to the shell-side of the steam generators is reduced, first because the reduced flow results in a decreased tube-side film coefficient and then because the reactor coolant in the tubes cools down while the shell-side temperature increases (turbine steam flow is reduced to zero upon plant trip). The rapid expansion of the coolant in the reactor core, combined with reduced heat transfer in the steam generators causes an insurge into the pressurizer and a pressure increase throughout the RCS. The insurge into the pressurizer compresses the steam volume, actuates the automatic spray system, opens the power-operated relief valves, and opens the pressurizer safety valves in that sequence. The three power-operated relief valves are designed for reliable operation and would be expected to function properly during the accident. However, for conservatism, their pressure-reducing effect as well as the pressure-reducing effect of the spray is not included in the analysis.

15.4.4.2 Analysis of Effects and Consequences

Two digital computer codes are used to analyze this transient. The LOFTRAN[2] code is used to calculate the resulting loop and core coolant flow following the pump seizure. The LOFTRAN code is also used to calculate the time of reactor trip, based on the calculated flow, the nuclear power following reactor trip, and to determine the peak pressure. The thermal behavior of the fuel located at the core hot spot is investigated using the FACTRAN[6] code, using the core flow and the nuclear power calculated by LOFTRAN. The FACTRAN code includes the use of a film boiling heat transfer coefficient.

The following case is analyzed:

All loops operating, one locked rotor.

At the beginning of the postulated locked rotor accident, i.e., at the time the shaft in one of the RCPs is assumed to seize, the plant is assumed to be operating under the most adverse steady state operating conditions, i.e., maximum steady state power level, maximum steady state pressure, and maximum steady state coolant average temperature.

When the peak pressure is evaluated, the initial pressure is conservatively estimated as 33 psi above nominal pressure, 2250 psia, to allow for errors in the pressurizer pressure measurement and control channels. This is done to obtain the highest possible rise in the coolant pressure during the transient. To obtain the maximum pressure in the primary side, conservatively high loop pressure drops are added to the calculated pressurizer pressure. The pressure response is shown on Figure 15.4.4-1.

15.4.4.2.1 Evaluation of the Pressure Transient

After pump seizure and reactor trip, the neutron flux is rapidly reduced by control rod insertion effect. Rod motion is assumed to begin 1 second after the flow in the affected loop reaches 87% of nominal flow. No credit is taken for the pressure-reducing effect of the pressurizer relief valves (PORVs), pressurizer spray, steam dump, or controlled feedwater flow after plant trip.

Although these operations are expected to occur and would result in a lower peak pressure, an additional degree of conservatism is provided by ignoring their effect.

The pressurizer safety valves are assumed to initially open at 2500 psia and achieve rated flow at 2575 psia (3% accumulation).

15.4.4.2.2 Evaluation of the Effects of DNB in the Core During the Accident

For this accident, DNB is assumed to occur in the core and, therefore, an evaluation of the consequences with respect to fuel rod thermal transients is performed. Results obtained from analysis of this hot spot condition represent the upper limit with respect to cladding temperature and zirconium-water reaction.

In the evaluation, the rod power at the hot spot is conservatively assumed to be approximately 2.6 times the average rod power at the initial core power level.

15.4.4.2.3 Film Boiling Coefficient

The film boiling coefficient is calculated in the FACTRAN code using the Bishop-Sandberg-Tong film boiling correlation The fluid properties are evaluated at film temperature (average between wall and bulk temperatures). The program calculates the film coefficient at every time step based on the actual heat transfer conditions at the time. The neutron flux, system pressure, bulk density, and mass flowrate as a function of time are used as program input.

For this analysis, the initial values of the pressure and the bulk density are used throughout the transient since they are the most conservative with respect to cladding temperature response. For conservatism, DNB was assumed to start at the beginning of the accident.

15.4.4.2.4 Fuel Cladding Gap Coefficient

The magnitude and time dependence of the heat transfer coefficient between fuel and cladding (gap coefficient) has a pronounced influence on the thermal results. The larger the value of the gap coefficient, the more heat is transferred between pellet and cladding. Based on investigations on the effect of the gap coefficient upon the maximum cladding temperature during the transient, the gap coefficient was assumed to increase from a steady state value consistent with the initial fuel temperature to 10,000 BTU/hr-ft^2-°F at the initiation of the transient. This assumption causes energy stored in the fuel to be released to the cladding at the initiation of the transient and maximizes the cladding temperature during the transient.

15.4.4.2.5 Zirconium-steam Reaction

The zirconium-steam reaction can become significant above 1800°F (cladding temperature). The Baker-Just parabolic rate equation shown below is used to define the rate of the zirconium-steam reaction.

$$\frac{d(w^2)}{dt} = 33.3 \times 10^6 \exp - [\frac{45,500}{1.986T}] \qquad (15.4\text{-}1)$$

where:

w = amount reacted, mg/cm^2

t = time, sec

T = temperature, °K

The reaction heat is 1510 cal/gm.

15.4.4.3 Results

Transient values of RCS pressure, RCS flow, faulted loop flow, nuclear power hot channel heat flux, and clad temperature are shown in Figure 15.4.4-1 and Figure 15.4.4-2.

Maximum RCS pressure, maximum cladding temperature, and amount of zirconium-water reaction are contained in Table 15.4-9.

15.4.4.4 Conclusions

(1) Since the peak RCS pressure reached during any of the transients is less than that which would cause stresses to exceed the faulted condition stress limits, the integrity of the primary coolant system is not endangered.

(2) Since the peak cladding surface temperature calculated for the hot spot during the worst transient remains considerably less than 2700°F and the amount of zirconium-water reaction is small, the core will remain in place and intact with no consequential loss of core cooling capability.

(3) The results of the transient analysis show that less than 15.0% of the fuel rods will have DNBRs below the safety analysis limit values.

15.4.6 Rupture of a Control Rod Drive Mechanism Housing (Rod Cluster Control Assembly Ejection)

15.4.6.1 Identification of Causes and Accident Description

This accident is defined as the mechanical failure of a control rod mechanism pressure housing resulting in the ejection of a rod cluster control assembly (RCCA) and drive shaft. The consequence of this mechanical failure is a rapid positive reactivity insertion and system depressurization together with an adverse core power distribution, possibly leading to localized fuel rod damage.

15.4.6.1.1 Design Precautions and Protection

Certain features of the VCSPP are intended to preclude the possibility of a rod ejection accident, or to limit the consequences if the accident were to occur. These include a sound, conservative mechanical design of the rod housings, together with a thorough quality control (testing) program during assembly, and a nuclear design that lessens the potential ejection worth of RCCAs, and minimizes the number of assemblies inserted at high power levels.

15.4.6.1.2 Mechanical Design

The mechanical design is discussed in FSAR Section 4.2. Mechanical design and quality control procedures intended to preclude the possibility of an RCCA drive mechanism housing failure are listed below:

(1) Each full length control rod drive mechanism housing is completely assembled and shop tested at 4100 psi.

(2) The mechanism housings are individually hydrotested after they are attached to the head adapters in the reactor vessel head, and checked during the hydrotest of the completed reactor coolant system.

(3) Stress levels in the mechanism are not affected by anticipated system transients at power, or by the thermal movement of the coolant loops. Moments induced by the design-basis earthquake can be accepted within the allowable primary working stress range specified by the ASME Code, Section III, for Class I components.

(4) The latch mechanism housing and rod travel housing are each a single length of forged Type-304 stainless steel. This material exhibits excellent notch toughness at all temperatures that will be encountered.

A significant margin of strength in the elastic range together with the large energy absorption capability in the plastic range gives additional assurance that gross failure of the housing will not occur. The joints between the latch mechanism housing and head adapter, and between the latch mechanism housing and rod travel housing, are threaded joints reinforced by canopy-type rod welds which are subject to periodic inspections.

15.4.6.1.3 Nuclear Design

Even if a rupture of an RCCA drive mechanism housing is postulated, the operation of a plant utilizing chemical shim is such that the severity of an ejected RCCA is inherently limited. In general, the reactor is operated with the RCCAs inserted only far enough to permit load follow. Reactivity changes caused by core depletion and xenon transients are compensated by boron changes. Further, the location and grouping of control RCCA banks are selected during the nuclear design to lessen the severity of an RCCA ejection accident. Therefore, should a RCCA be ejected from its normal position during full-power operation, only a minor reactivity excursion, at worst, could be expected to occur.

However, it may be occasionally desirable to operate with larger than normal insertions. For this reason, a rod insertion limit is defined as a function of power level. Operation with the RCCAs above this limit guarantees adequate shutdown capability and acceptable power distribution. The position of all RCCAs is continuously indicated in the control room. An alarm will occur if a bank of RCCAs approaches its insertion limit or if one RCCA deviates from its bank. There are low and low-low level insertion monitors with visual and audio signals. Operating instructions require boration at low-level alarm and emergency boration at the low-low alarm.

15.4.6.1.4 Reactor Protection

The reactor protection in the event of a rod ejection accident has been described in Reference 7. The protection for this accident is provided by the power range high neutron flux trip (high and low setting) and high rate of neutron flux increase trip. These protection functions are described in detail in FSAR Section 7.2.

15.4.6.1.5 Effects on Adjacent Housings

Disregarding the remote possibility of the occurrence of an RCCA mechanism housing failure, investigations have shown that failure of a housing due to either longitudinal or circumferential cracking is not expected to cause damage to adjacent housings leading to increased severity of the initial accident.

15.4.6.1.6 Limiting Criteria

Due to the extremely low probability of an RCCA ejection accident, limited fuel damage is considered an acceptable consequence.

Comprehensive studies of the threshold of fuel failure and of the threshold of significant conversion of the fuel thermal energy to mechanical energy have been carried out as part of the SPERT project by the Idaho Nuclear Corporation[8]. Extensive tests of zirconium-clad UO_2 fuel rods representative of those in PWR-type cores have demonstrated failure thresholds in the range of 240 to 257 cal/gm. However, other rods of a slightly different design have exhibited failures as low as 225 cal/gm. These results differ significantly from the TREAT[9] results, which indicated a failure threshold of 280 cal/gm. Limited results have indicated that this threshold decreases by about 10% with fuel burnup. The cladding failure mechanism appears to be melting for zero burnup rods and brittle fracture for irradiated rods. Also important is the conversion ratio of thermal to mechanical energy. This ratio becomes marginally detectable above 300 cal/gm for unirradiated rods and 200 cal/gm for irradiated rods; catastrophic failure, (large fuel dispersal, large pressure rise) even for irradiated rods, did not occur below 300 cal/gm.

In view of the above experimental results, conservative criteria are applied to ensure that there is little or no possibility of fuel dispersal in the coolant, gross lattice distortion, or severe shock waves. These criteria are:

(1) Average fuel pellet enthalpy at the hot spot below 225 cal/gm for unirradiated fuel and 200 cal/gm for irradiated fuel;

(2) Average cladding temperature at the hot spot below the temperature at which cladding embrittlement may be expected (2700°F);

(3) Peak reactor coolant pressure less than that which would cause stresses to exceed the faulted condition stress limits;

(4) Fuel melting will be limited to less than 10% of the fuel volume at the hot spot even if the average fuel pellet enthalpy is below the limits of Criterion (1) above.

15.4.6.2 Analysis of Effects and Consequences

The analysis of the RCCA ejection accident is performed in two stages: (a) an average core nuclear power transient calculation and (b) a hot spot heat transfer calculation. The average core calculation is performed using spatial neutron kinetics methods to determine the average power generation with time including the various total core feedback effects, i.e., Doppler reactivity and moderator reactivity. Enthalpy and temperature transients in the hot spot are then determined by multiplying the average core energy generation by the hot channel factor and performing a fuel rod transient heat transfer calculation. The power distribution calculated without feedback is pessimistically assumed to persist throughout the transient.

A detailed discussion of the method on analysis can be found in Reference 10.

15.4.6.2.1 Average Core Analysis

The spatial kinetics computer code, TWINKLE[11], is used for the average core transient analysis. This code solves the two group neutron diffusion theory kinetic equations in one, two, or three spatial dimensions (rectangular coordinates) for six delayed neutron groups and up to 2000 spatial points. The computer code includes a detailed multiregion, transient fuel-clad-coolant heat transfer model for calculating pointwise Doppler, and moderator feedback effects.

In this analysis, the code is used as a one-dimensional axial kinetics code since it allows a more realistic representation of the spatial effects of axial moderator feedback and RCCA movement and the elimination of axial feedback weighting factors. However, since the radial dimension is missing, it is still necessary to employ very conservative methods (described below) of calculating the ejected rod worth and hot channel factor. A further description of TWINKLE appears in Section 15.1.8.

15.4.6.2.2 Hot Spot Analysis

The average core energy addition, calculated as described above, is multiplied by the appropriate hot channel factors, and the hot spot analysis is performed using the detailed fuel and cladding transient heat transfer computer code, FACTRAN(6). This computer code calculates the transient temperature distribution in a cross section of a metal clad UO_2 fuel rod, and the heat flux at the surface of the rod, using as input the nuclear power versus time and the local coolant conditions. The zirconium-water reaction is explicitly represented, and all material properties are represented as functions of temperature. A parabolic radial power generation is used within the fuel rod.

FACTRAN uses the Dittus-Boelter(12) or Jens-Lottes(13) correlation to determine the film heat transfer before DNB, and the Bishop-Sandberg-Tong correlation(14) to determine the film boiling coefficient after DNB. The DNB heat flux is not calculated; instead the code is forced into DNB by specifying a conservative DNB heat flux. The gap heat transfer coefficient can be calculated by the code; however, it is adjusted in order to force the full power steady state pellet temperature distribution to agree with that predicted by design fuel heat transfer codes.

For full power cases, the design initial hot channel factor (F_Q) is input to the code. The hot channel factor during the transient is assumed to increase from the steady state design value to the maximum transient value in 0.1 seconds, and remain at the maximum for the duration of the transient. This is conservative, since detailed spatial kinetics models show that the hot channel factor decreases shortly after the nuclear power peak due to power flattening caused by preferential feedback in the hot channel. Further description of FACTRAN appears in Section 15.1.8.

15.4.6.2.3 System Overpressure Analysis

Because safety limits for fuel damage specified earlier are not exceeded, there is little likelihood of fuel dispersal into the coolant. The pressure surge may therefore be calculated on the basis of conventional heat transfer from the fuel and prompt heat generation in the coolant.

The pressure surge is calculated by first performing the fuel heat transfer calculation to determine the average and hot spot heat flux versus time. Using this heat flux data, a THINC calculation is conducted to determine the volume surge. Finally, the volume surge is simulated in a plant transient computer code. This code calculates the pressure transient taking into account fluid transport in the system, heat transfer to the steam generators, and the action of the pressurizer spray and pressure relief valves. No credit is taken for the possible pressure reduction caused by the assumed failure of the control rod pressure housing[15].

15.4.6.2.4 Calculation of Basic Parameters

Input parameters for the analysis are conservatively selected on the basis of calculated values for this type of core. The more important parameters are discussed below. Table 15.4-10 presents the parameters used in this analysis.

15.4.6.2.5 Ejected Rod Worths and Hot Channel Factors

The values for ejected rod worths and hot channel factors are calculated using three dimensional calculations. Standard nuclear design codes are used in the analysis. No credit is taken for the flux-flattening effects of reactivity feedback. The calculation is performed for the maximum allowed bank insertion at a given power level as determined by the rod insertion limits. Adverse xenon distributions are considered in the calculations.

Appropriate margins are added to the results to allow for calculational uncertainties, including an allowance for nuclear power peaking due to densification.

15.4.6.2.6 Reactivity Feedback Weighting Factors

The largest temperature rises, and hence the largest reactivity feedbacks, occur in channels where the power is higher than average. Since the weight of regions is dependent on flux, these regions have high weights. This means that the reactivity feedback is larger than that indicated by a simple single channel analysis. Physics calculations were carried out for temperature changes with a flat temperature distribution, and with a large number of axial and radial temperature distributions. Reactivity changes were compared and effective weighting factors determined. These weighting factors take the form of multipliers that, when applied to single channel feedbacks, correct them to effective whole core feedbacks for the appropriate flux shape. In this analysis, since a one-dimensional (axial) spatial kinetics method is employed, axial weighting is not used. In addition, no weighting is applied to the moderator feedback. A conservative radial weighting factor is applied to the transient fuel temperature to obtain an effective fuel temperature as a function of time accounting for the missing spatial dimension. These weighting factors were shown to be conservative compared to three-dimensional analysis.

15.4.6.2.7 Moderator and Doppler Coefficient

The critical boron concentrations at the beginning-of-life (BOL) and end-of-life (EOL) are adjusted in the nuclear code in order to obtain moderator density coefficient curves which are conservative compared to actual design conditions for the plant. As discussed above, no weighting factor is applied to these results.

The Doppler reactivity defect is determined as a function of power level using the one-dimensional steady state computer code with a Doppler weighting factor of 1. The resulting curve is conservative compared to design predictions for this plant. The Doppler weighting factor should be larger than 1 (approximately 1.2) just to make the present calculation agree with design predictions before ejection. This weighting factor will increase under accident conditions, as discussed above. The Doppler defect used as an initial condition is 900 pcm at BOL and 840 pcm at EOL.

15.4.6.2.8 Delayed Neutron Fraction

Calculations of the effective delayed neutron fraction (β_{eff}) typically yield values of 0.70% at BOL and 0.50% at EOL for the first cycle. The accident is sensitive to β if the ejected rod worth is nearly equal to or greater than β as in zero power transients. In order to allow for future fuel cycles, pessimistic estimates of β of 0.54% at beginning of cycle and 0.44% at end of cycle were used in the analysis.

15.4.6.2.9 Trip Reactivity Insertion

The trip reactivity insertion assumed is given in Table 15.4-10 and includes the effect of one stuck rod. These values are reduced by the ejected rod reactivity. The shutdown reactivity was simulated by dropping a rod of the required worth into the core. The start of rod motion occurred 0.5 seconds after the high neutron flux trip point was reached. This delay is assumed to consist of 0.2 seconds for the instrument channel to produce a signal, 0.15 seconds for the trip breaker to open, and 0.15 seconds for the coil to release the rods. The analyses presented are applicable for a rod insertion time of 2.7 seconds from coil release to entrance to the dashpot. The choice of such a conservative insertion rate means that there is over 1 second after the trip point is reached before significant shutdown reactivity is inserted into the core. This is a particularly important conservatism for hot full power accidents.

The rod insertion versus time is described in Section 15.1.4.

15.4.6.3 Results

The values of the parameters used in the analysis, as well as the results of the analysis, are presented in Table 15.4-10 and discussed below.

15.4.6.3.1 Beginning of Cycle, Full Power

Control Bank D was assumed to be inserted to its insertion limit. The worst ejected rod worth and hot channel factor were conservatively assumed to be 0.20% ΔK and 6.5, respectively. The peak hot spot clad average temperature was 2524°F. The peak hot spot fuel center temperature exceeded the BOL melting temperature of 4900°F. However, melting was restricted to less than 10% of the pellet.

15.4.6.3.2 Beginning of Cycle, Zero Power

For this condition, control Bank D was assumed to be fully inserted and C was at its insertion limit. The worst ejected rod is located in control Bank D and was conservatively assumed to have a worth of 0.855% ΔK and a hot channel factor of 13. The peak hot spot clad average temperature reached 2476°F. The peak hot spot fuel center temperature reached 4697°F.

15.4.6.3.3 End-of-Cycle, Full Power

Control Bank D was assumed to be inserted to its insertion limit. The ejected rod worth and hot channel factors were conservatively assumed to be 0.21% ΔK and 7.0, respectively. This resulted in a peak hot spot clad average temperature of 2414°F. The peak hot spot fuel center temperature exceeded the EOL melting temperature of 4800°F. However, melting was restricted to less than 10% of the pellet.

15.4.6.3.4 End of Cycle, Zero Power

The ejected rod worth and hot channel factor for this case were obtained assuming control Bank D to be fully inserted and Bank C at its insertion limit. The results were 0.90% ΔK and 22.5, respectively. The peak clad average and fuel center temperatures were 2344 and 4104°F, respectively.

A summary of the cases presented is given in Table 15.4-10. The nuclear power and hot spot fuel clad temperature transients for the worst cases (BOL full power and zero power) are presented on Figures 15.4.6-1 through 15.4.6-2.

15.4.6.3.5 Fission Product Release

It is assumed that fission products are released from the gaps of all rods entering DNB. In all cases considered, less than 10% of the rods entered DNB based on a detailed three-dimensional THINC analysis. Although limited fuel melting at the hot spot was predicted for the full power cases, in practice melting is not expected since the analysis conservatively assumed that the hot spots before and after ejection were coincident.

15.4.6.3.6 Pressure Surge

A detailed calculation of the pressure surge for an ejection worth of one dollar at BOL, hot full power, indicates that the peak pressure does not exceed that which would cause stress to exceed the faulted condition stress limits. Since the severity of the present analysis does not exceed this worst case analysis, the accident for this plant will not result in an excessive pressure rise or further damage to the RCS.

15.4.6.3.7 Lattice Deformations

A large temperature gradient will exist in the region of the hot spot. Since the fuel rods are free to move in the vertical direction, differential expansion between separate rods cannot produce distortion. However, the temperature gradients across individual rods may produce a force tending to bow the midpoint of the rods toward the hot spot. Physics calculations indicate that the net result of this would be a negative reactivity insertion. In practice, no significant bowing is anticipated, since the structural rigidity of the core is more than sufficient to withstand the forces produced. Boiling in the hot spot region would produce a net flow away from that region. However, the heat from fuel is released to the water relatively slowly, and it is considered inconceivable that cross flow will be sufficient to produce significant lattice forces. Even if massive and rapid boiling, sufficient to distort the lattice, is hypothetically postulated, the large void fraction in the hot spot region would produce a reduction in the total core moderator to fuel ratio, and a large reduction in this ratio at the hot spot. The net effect would therefore be a negative feedback. It can be concluded that no conceivable mechanism exists for a net positive feedback resulting from lattice deformation. In fact, a small negative feedback may result. The effect is conservatively ignored in the analyses.

15.4.6.4 Conclusions

Even on a conservative basis, the analyses indicate that the described fuel and cladding limits are not exceeded. It is concluded that there is no danger of sudden fuel dispersal into the coolant. Since the peak pressure does not exceed that which would cause stresses to exceed the faulted condition stress limits, it is concluded that there is no danger of further consequential damage to the reactor coolant system. The analyses have demonstrated that the upper limit in fission product release as a result of a number of fuel rods entering DNB amounts to 10%[15].

15.4.7 References

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16. Westinghouse letter dated March 25, 1986, NS-NRC-86-3116, "Westinghouse Response to Additional Request on WCAP-9226-P/WCAP-9227-N-P, Reactor Core Response to Excessive Secondary Steam Release," (Non-Proprietary).

TABLE 15.4-8

TIME SEQUENCE OF EVENTS FOR
MAJOR SECONDARY SYSTEM PIPE RUPTURES

Accident	Event	Time (sec)
Major Steam Line Rupture		
A. Offsite power available	Steam line ruptures	0
	Criticality attained	19
	Boron from RWST reaches core	65
	Accumulators actuate	76
	Peak heat flux attained	76
	Core becomes subcritical	~340
B. Without offsite power	Steam line ruptures	0
	Criticality attained	22
	Boron from RWST reaches core	68
	Peak heat flux attained	~270
	Core becomes subcritical	~370

Accident	Event	Time, sec
Rupture of Main Feedwater Pipe (Offsite Power Available)	Feedline rupture occurs	10
	Low-low steam generator level reactor trip setpoint reached in affected steam generator	32.4
	Rod begins to drop	34.4
	Emergency Feedwater is started	92.4
	Feedwater lines are purged and emergency feedwater is delivered to two of three intact steam generators	104
	Low steamline pressure setpoint reached	225
	Steamline and feedline isolation occurs	235
	Steam generator safety valves lift in intact loops	612
	Pressurizer water relief begins	1512
	Total RCS heat generation (decay heat + pump heat) decreases to emergency feedwater heat removal capability	~3300

TABLE 15.4-8

Accident	Event	Time, sec
Rupture of Main Feedwater Pipe (Offsite Power Unavailable)	Feedline rupture occurs	10
	Low-low steam generator level reactor trip setpoint reached in affected steam generator	32.4
	Rod begins to drop	34.4
	Reactor coolant pump coastdown	36.4
	Emergency feedwater is started	92.4
	Feedwater lines are purged and emergency feedwater is delivered to two of three intact steam generators	104
	Low steamline pressure setpoint reached	254
	Steamline and feedline isolation occurs	264
	Steam generator safety valves lift in intact loops	768
	Total RCS heat generation decreases to emergency feedwater heat removal capability	~1200
	Peak pressurizer water level reached	~1470

TABLE 15.4-9

SUMMARY OF RESULTS FOR LOCKED ROTOR TRANSIENT

	3 Loops Operating Initially 1 Locked Rotor
Maximum RCS pressure, psia	2606
Maximum clad temperature, °F core hot spot	1983
Amount of $Zr-H_2O$ at core hot spot, % by weight	0.6%

TABLE 15.4-10

PARAMETERS USED IN THE ANALYSIS OF THE ROD CLUSTER CONTROL ASSEMBLY EJECTION ACCIDENT

Time in Life	Beginning	Beginning	End	End
Power level, %	102	0.0	102	0.0
Ejected rod worth, %Δk	0.20	0.855	0.21	0.90
Delayed neutron fraction, %	0.54	0.54	0.44	0.44
Feedback reactivity weighting	1.30	2.07	1.30	3.55
Trip reactivity, %Δk	4	2	4	2
F_Q before rod ejection	2.61	-	2.61	-
F_Q after rod ejection	6.5	13	7.0	22.5
Number of operating pumps	3	3	3	3
Maximum fuel pellet average temperature, °F	4219	3888	4050	3463
Maximum fuel center temperature, °F	*	4697	*	4104
Maximum clad average temperature, °F	2524	2476	2414	2344
Maximum fuel stored energy, cal/gm	186	169	177	147

*Less than 10% fuel melt

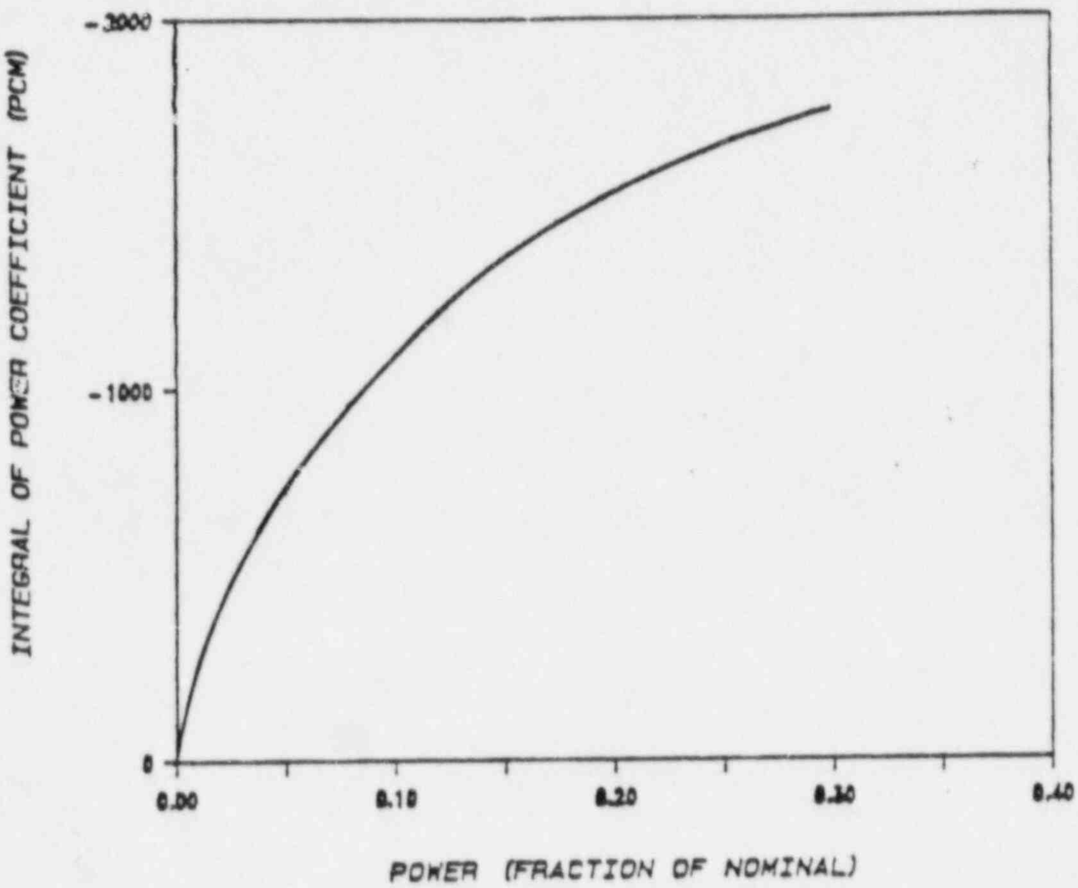

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Figure 15.4.2-1
Variation Of Reactivity With Power At Constant Core Average Temperature

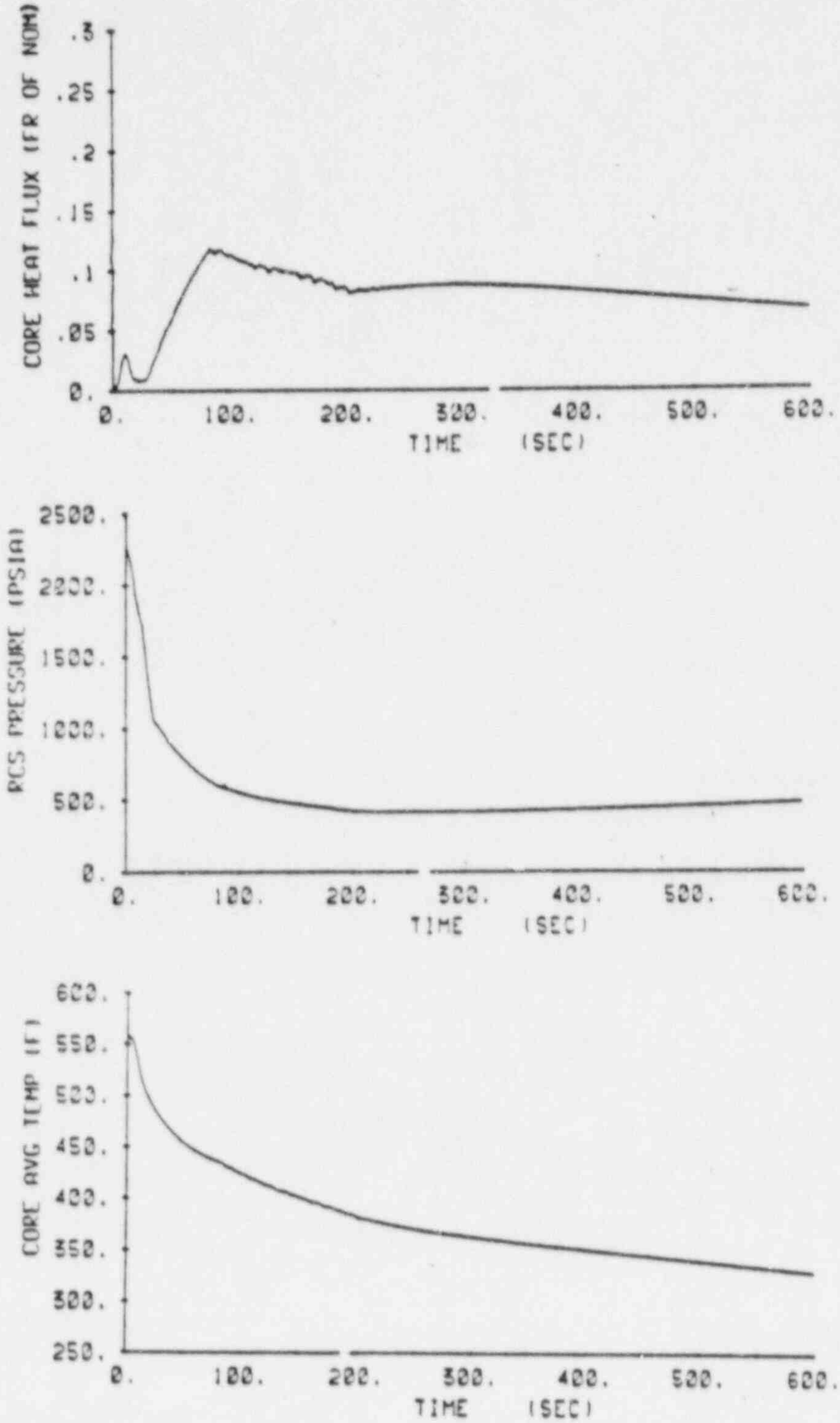

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Figure 15.4.2-2

Transient Response To A Steam Line Break Double Ended Rupture With Offsite Power Available (Case A)

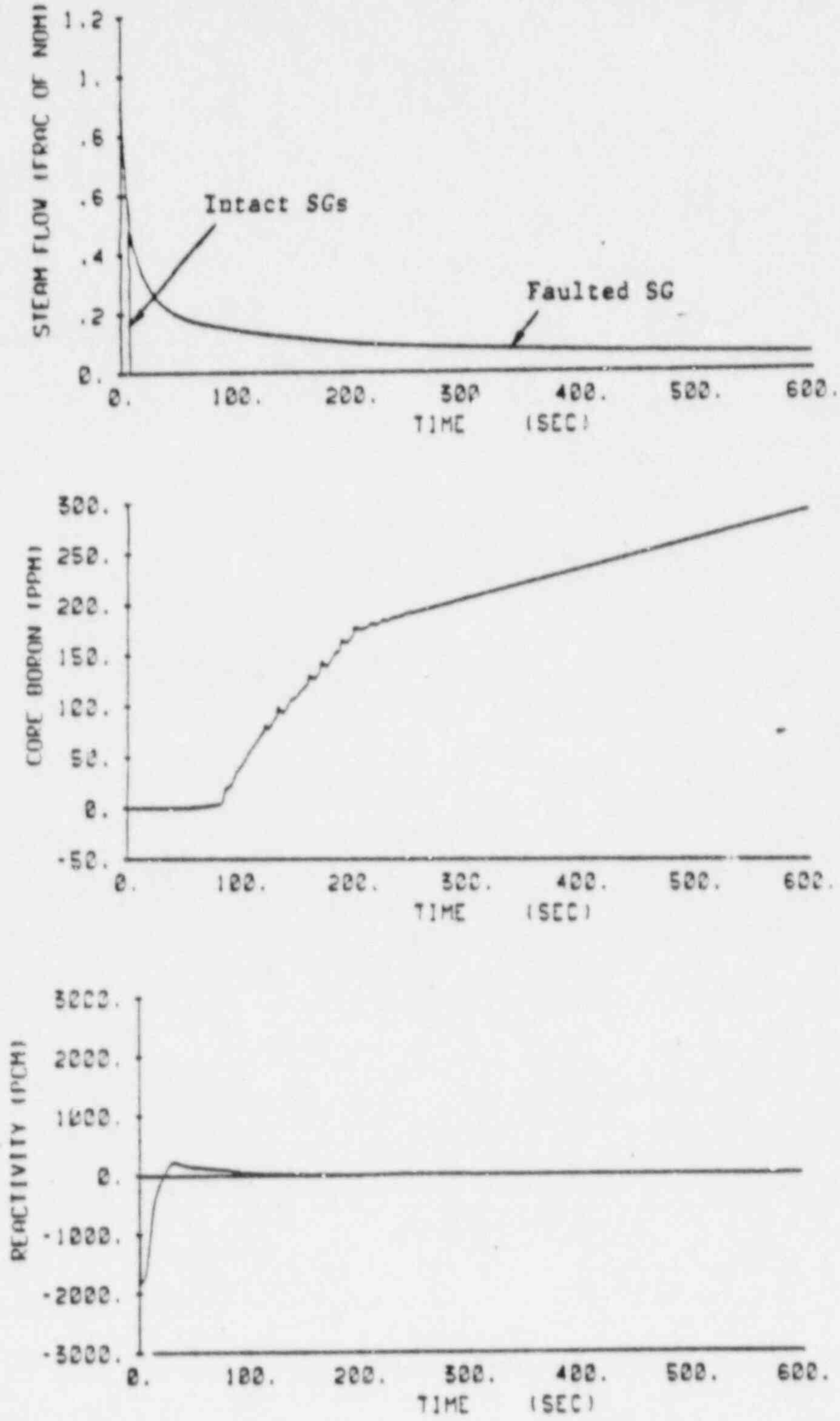

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Figure 15.4.2-3
Transient Response To A Steam Line Break Double Ended Rupture With Offsite Power Available (Case A)

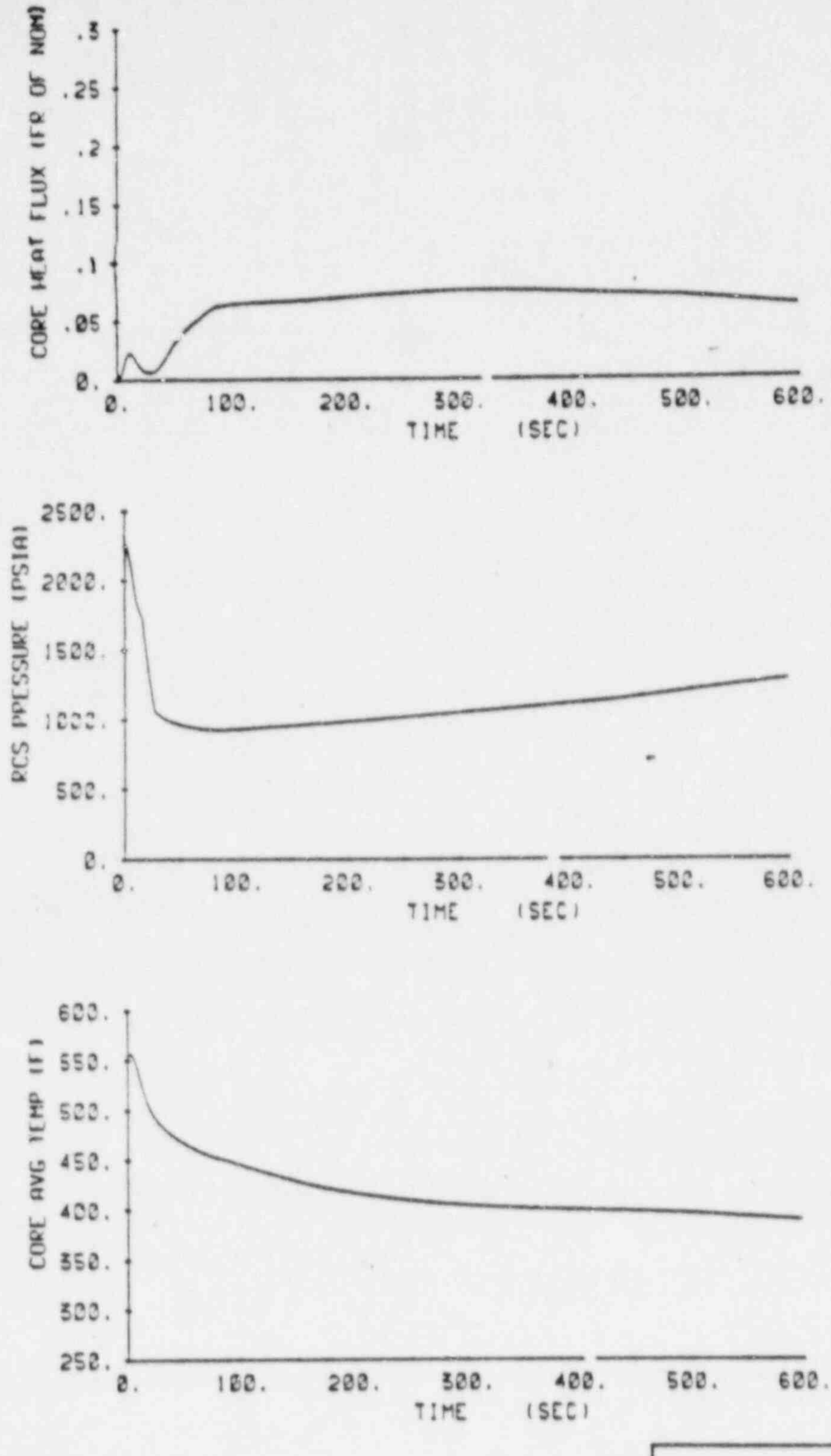

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Figure 15.4.2-4

Transient Response To A Steam Line Break Double Ended Rupture With No Offsite Power Available (Case B)

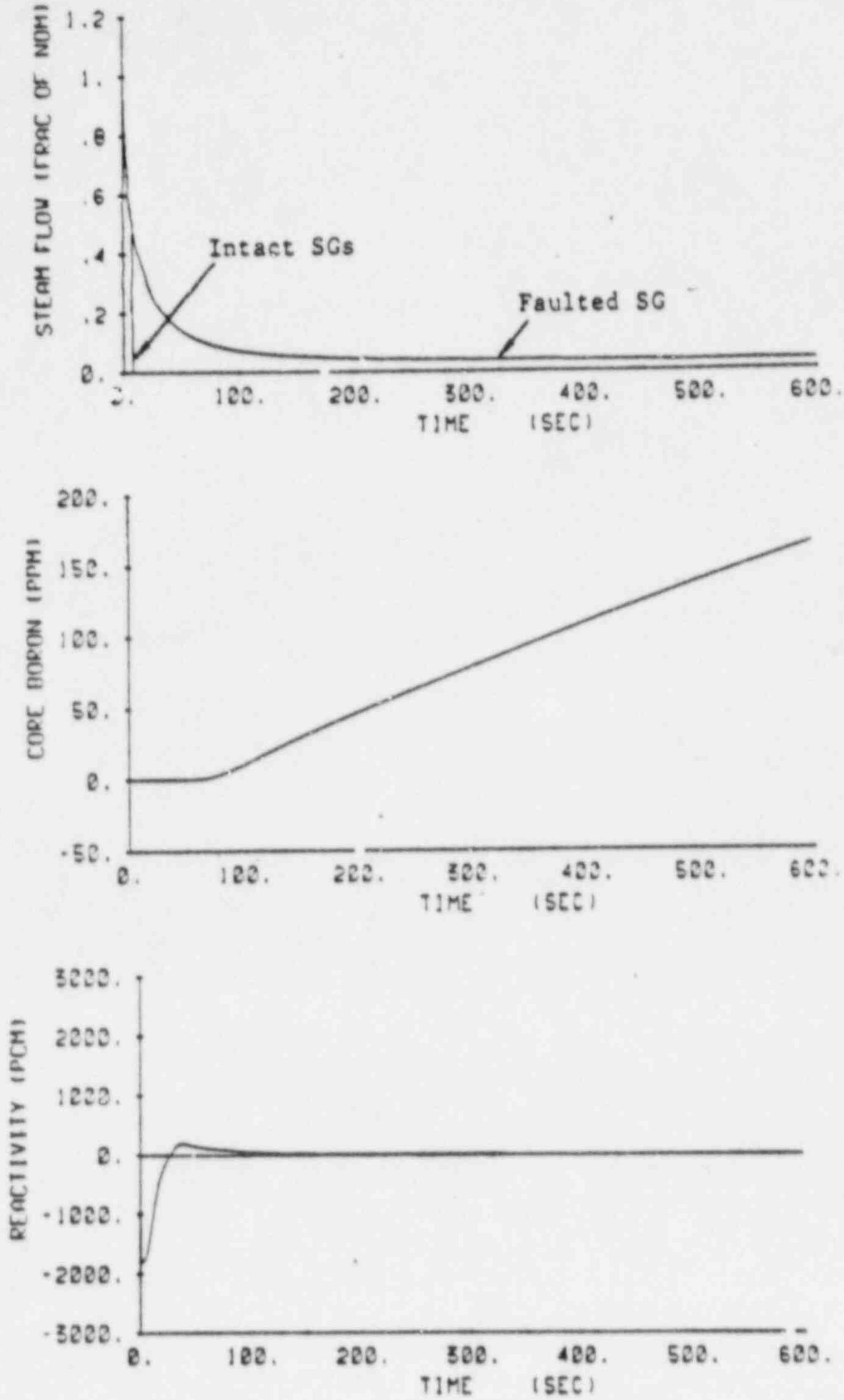

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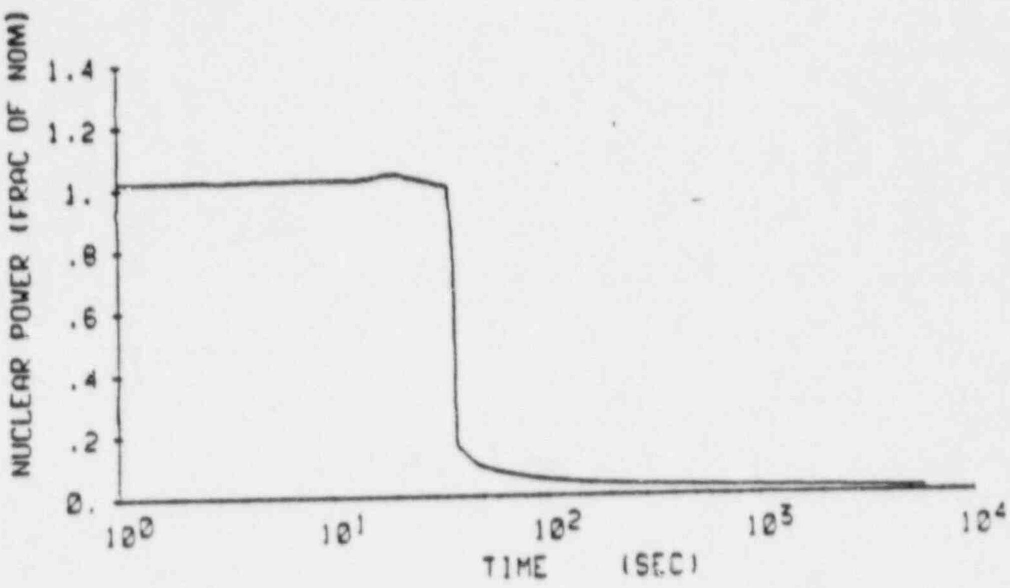
Figure 15.4.2-5
Transient Response To A Steam Line Break Double Ended Rupture With No Offsite Power Available (Case B)



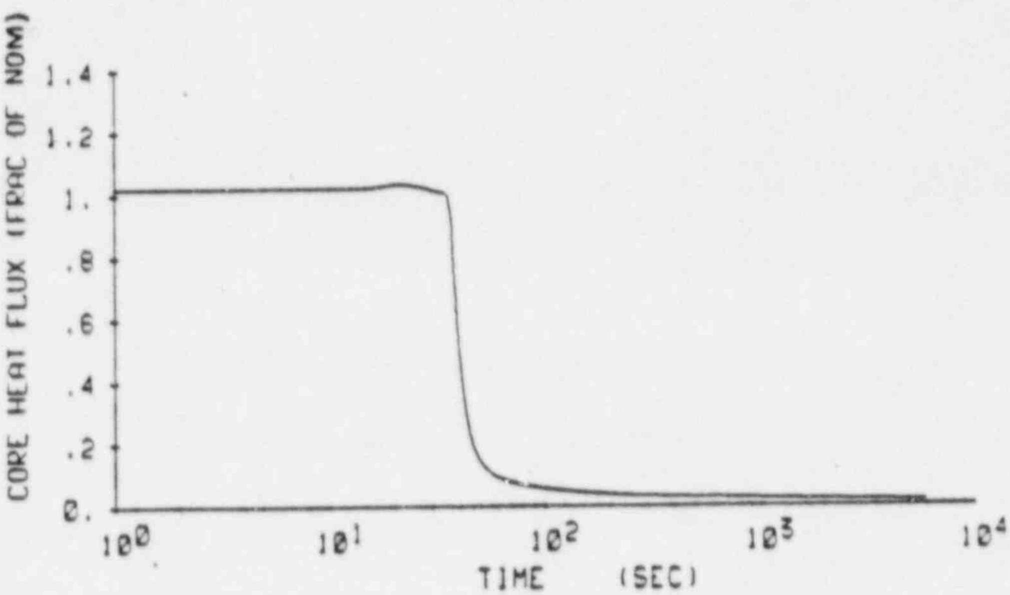


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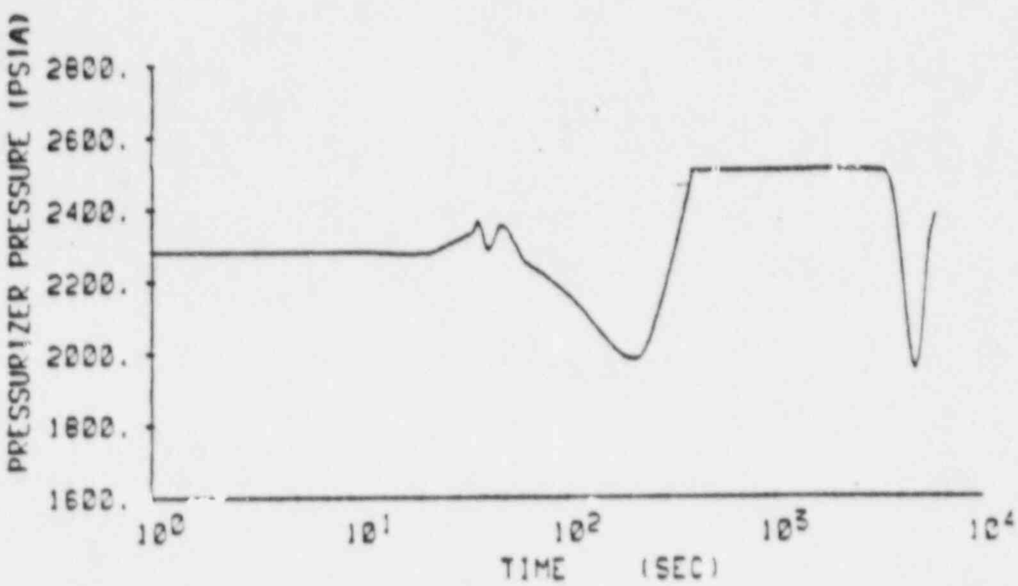
Figure 15.4.2-6
Main Feedline Rupture With Offsite Power Nuclear Power and Core Heat Flux vs. Time



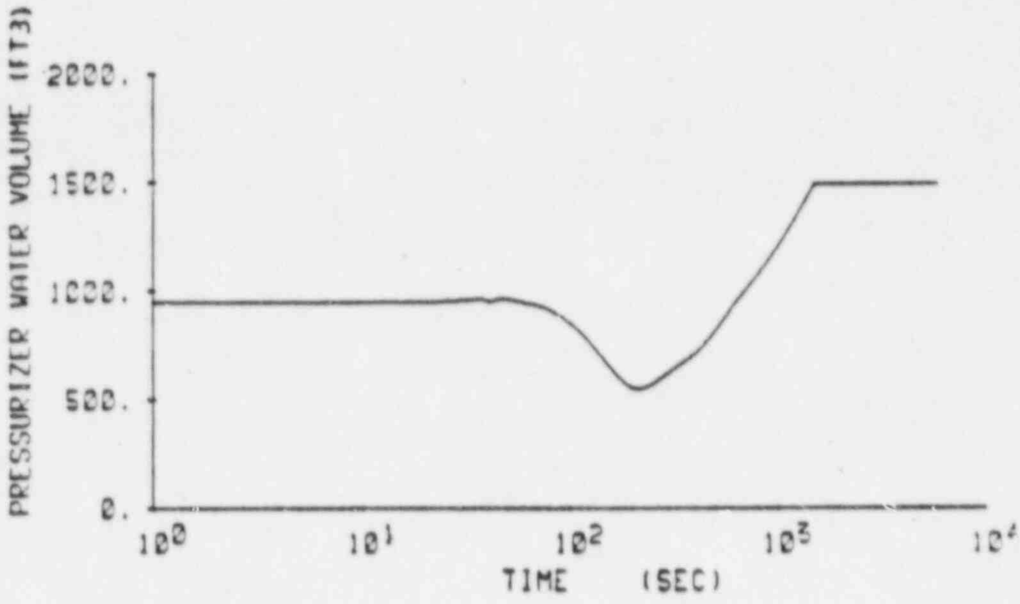


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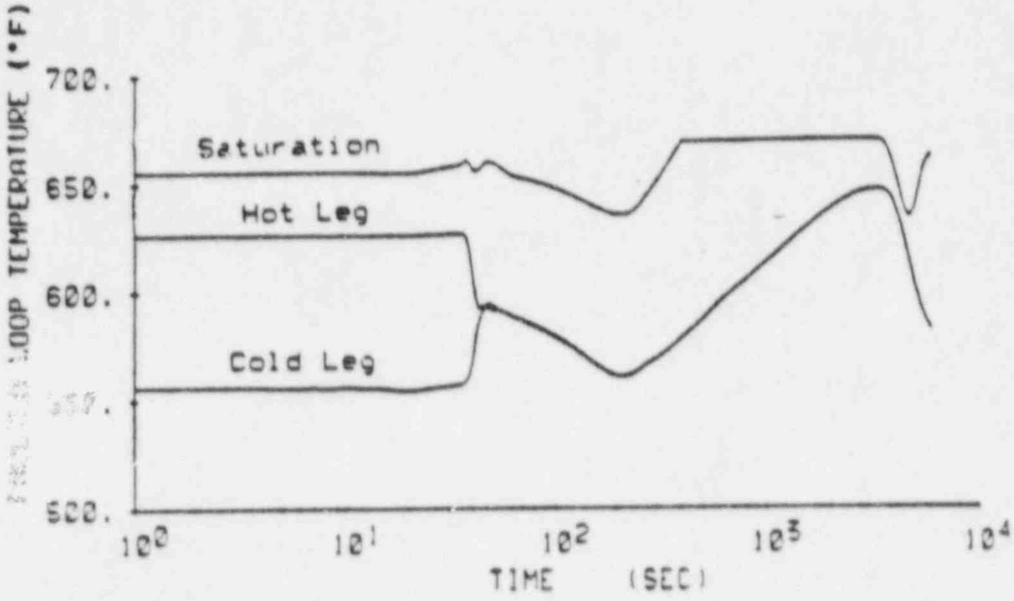
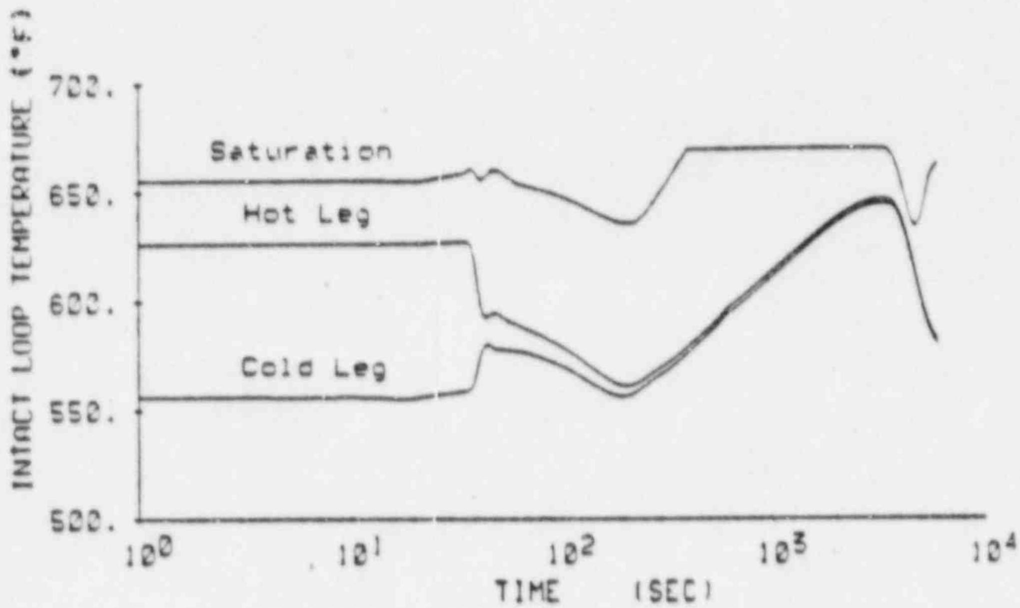
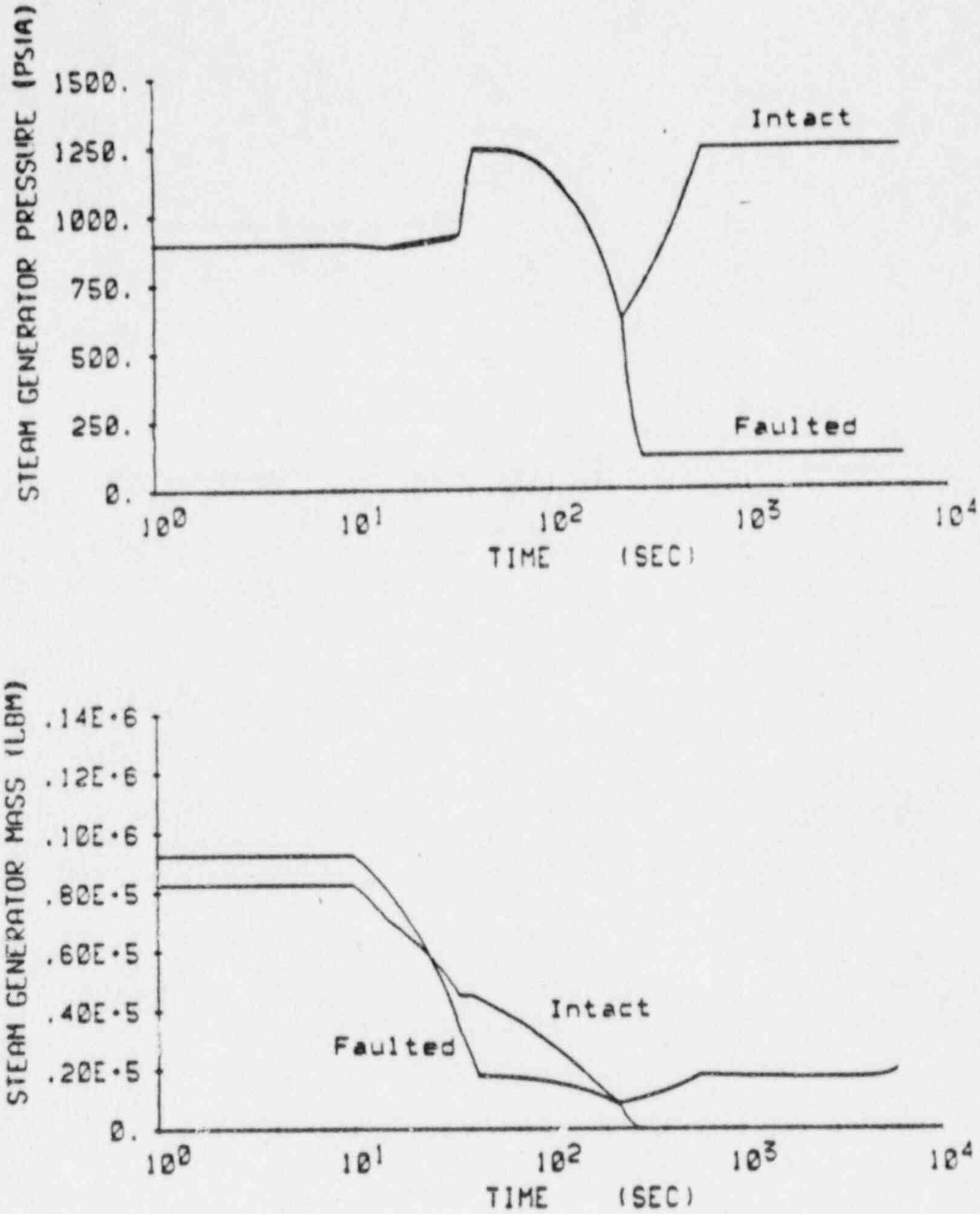
Figure 15.4.2-7

Main Feedline Rupture With Offsite Power Pressurizer Pressure and Water Volume vs. Time





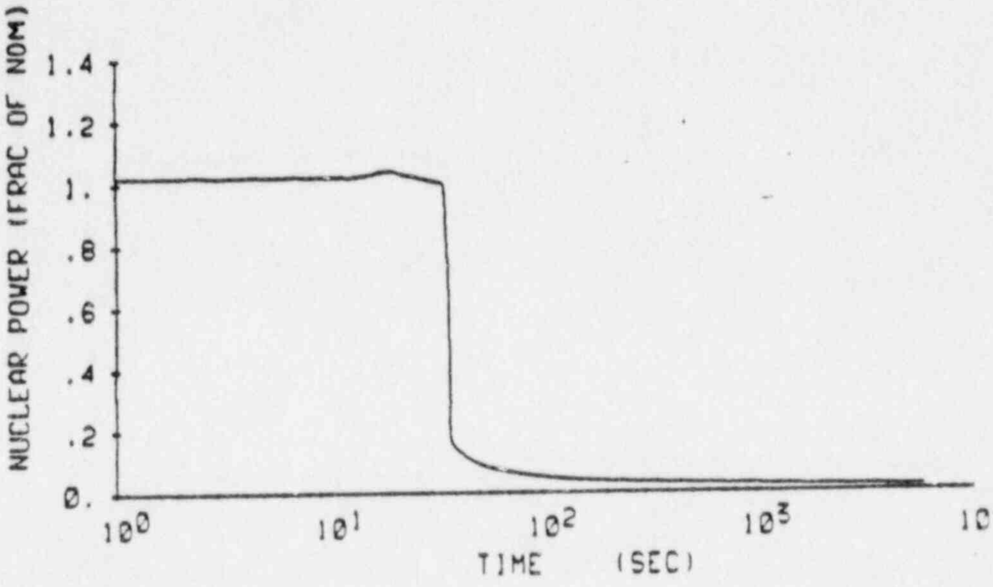
V. C. Summer
Figure 15.4.2-8
Main Feedline Rupture With Offsite Power Faulted and Intact Loop Coolant Temperatures vs. Time

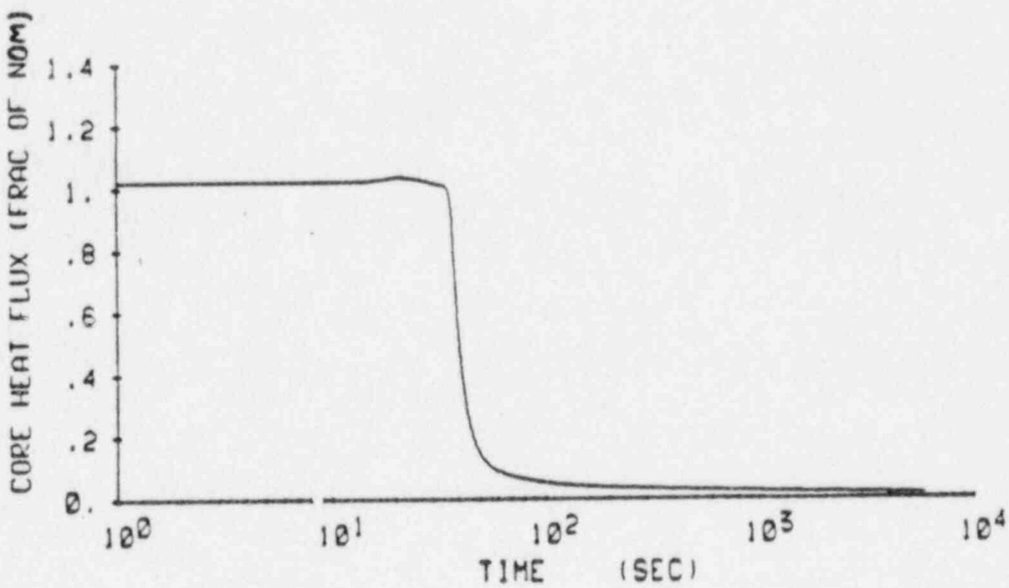


V. C. Summer

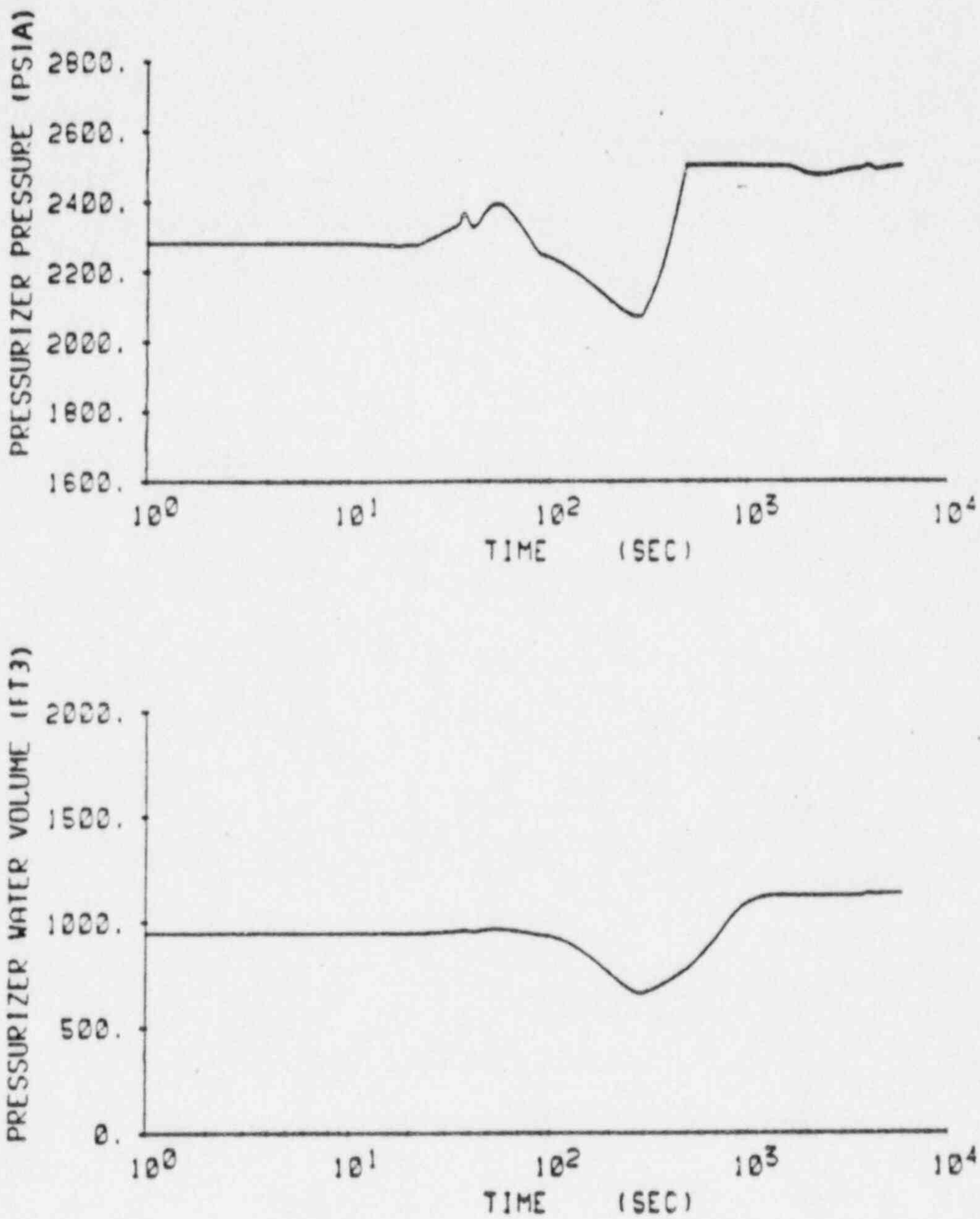
Figure 15.4.2-9

Main Feedline Rupture
With Offsite Power
Steam Generator Pressure and
Water Mass vs. Time

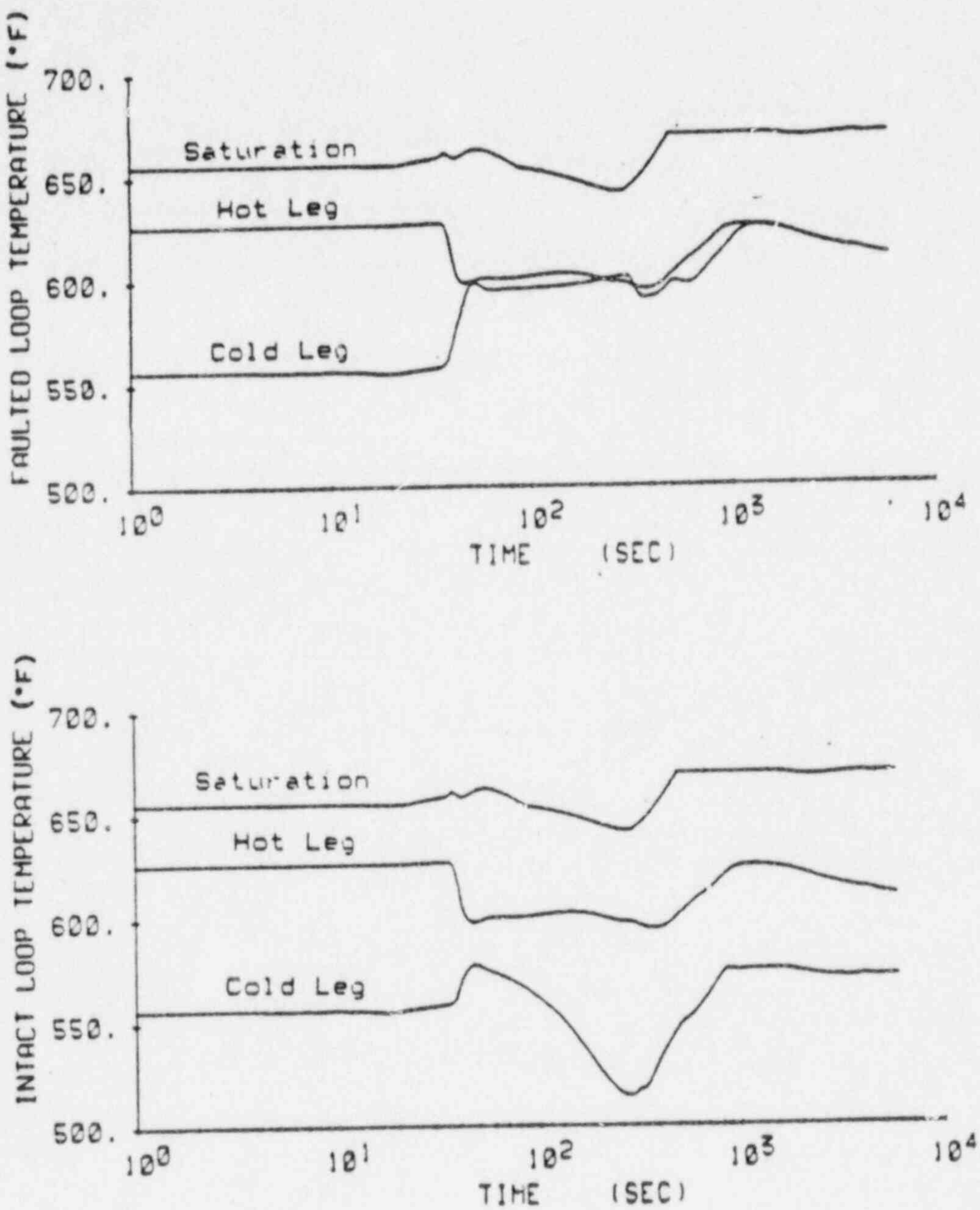




V. C. Summer
Figure 15.4.2-10
Main Feedline Rupture Without Offsite Power Nuclear Power and Core Heat Flux vs. Time



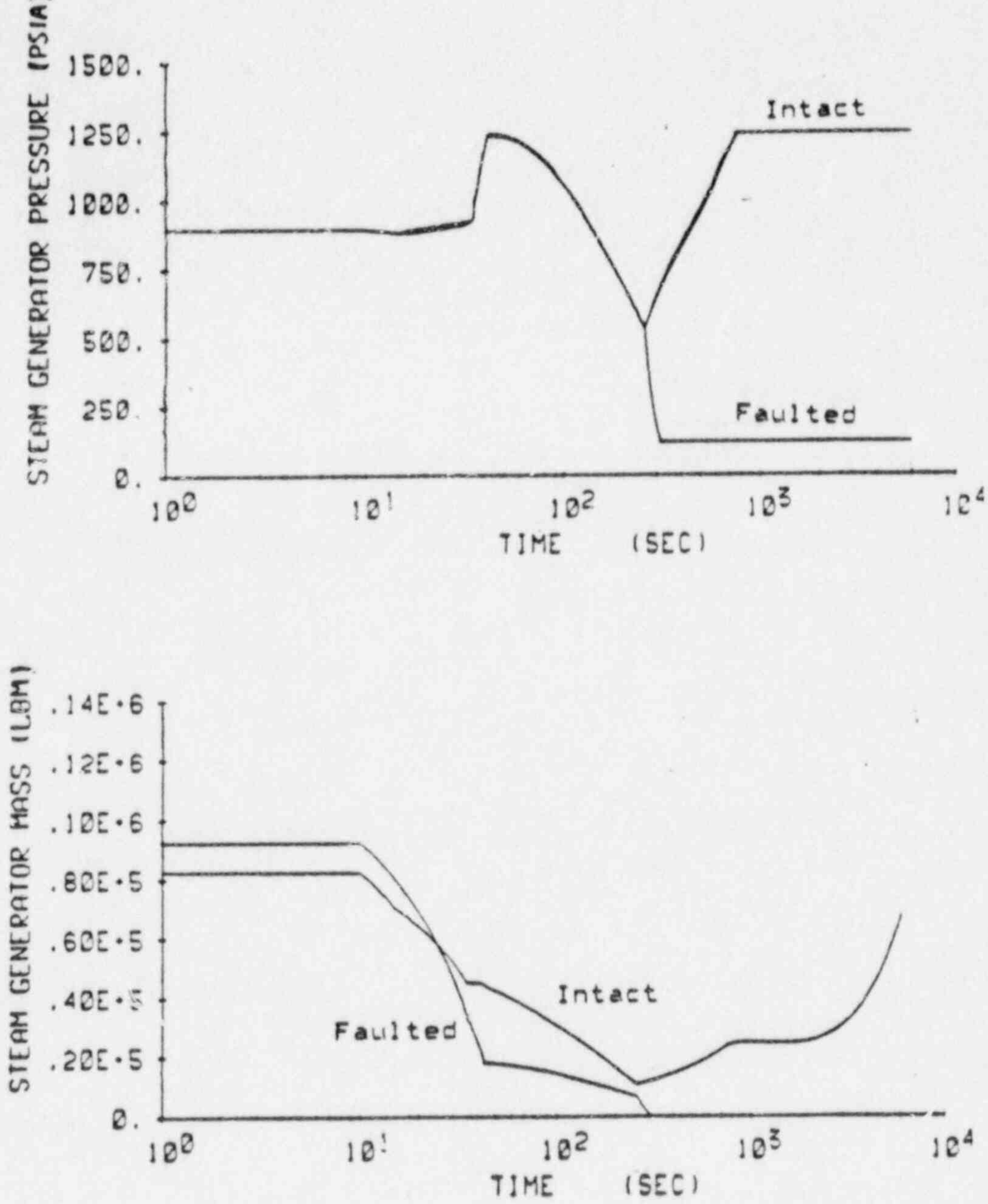
V. C. Summer
Figure 15.4.2-11
Main Feedline Rupture Without Offsite Power Pressurizer Pressure and Water Volume vs. Time



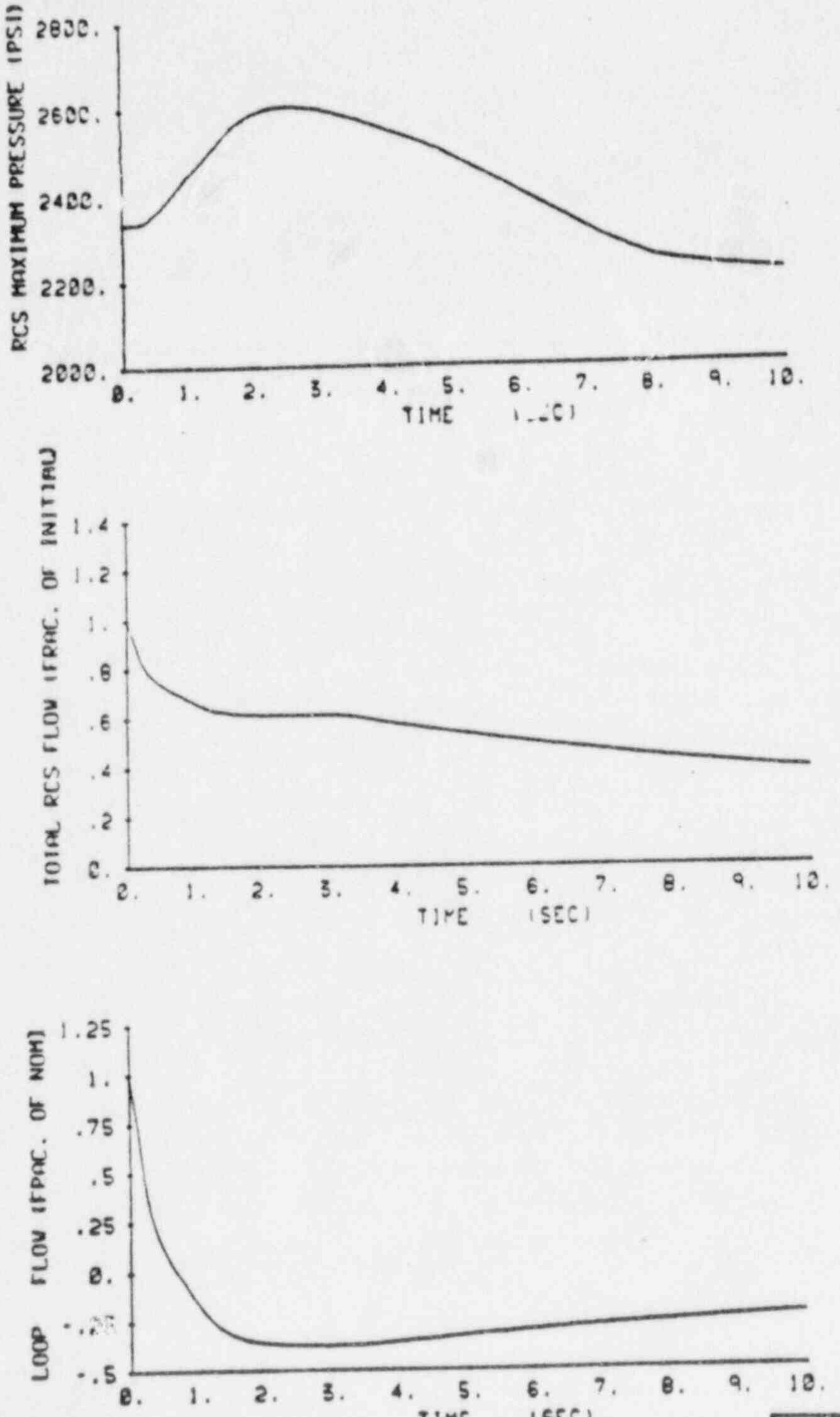
V. C. Summer

Figure 15.4.2-12

Main Feedline Rupture
Without Offsite Power
Faulted and Intact Loop
Temperatures vs. Time



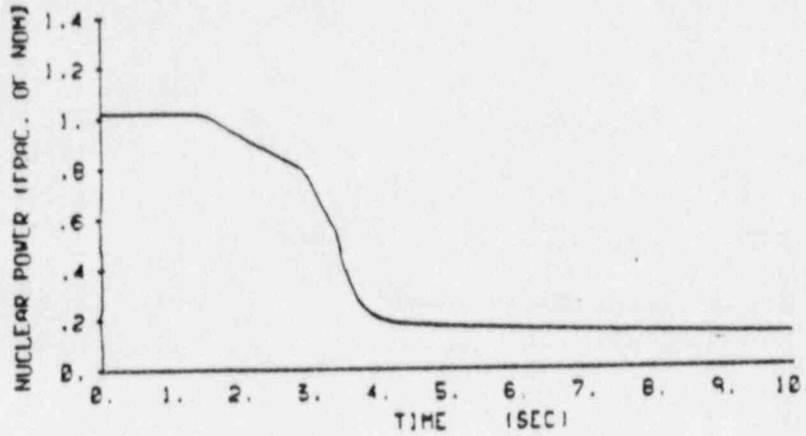
V. C. Summer
Figure 15.4.2-13
Main Feedline Rupture Without Offsite Power Steam Generator Pressure and Water Mass vs. Time

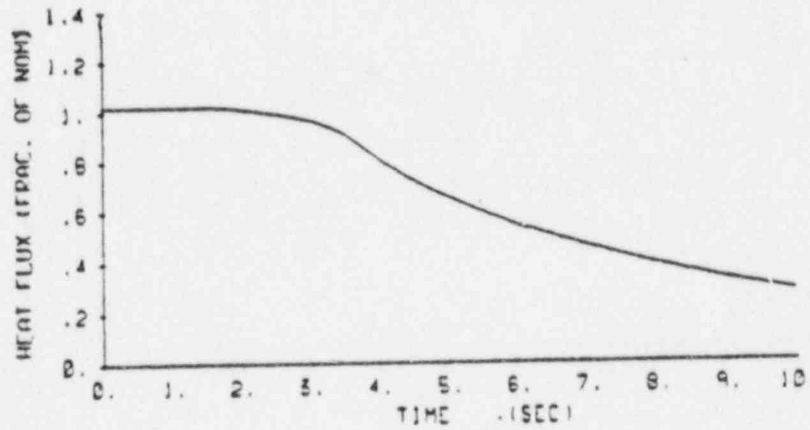


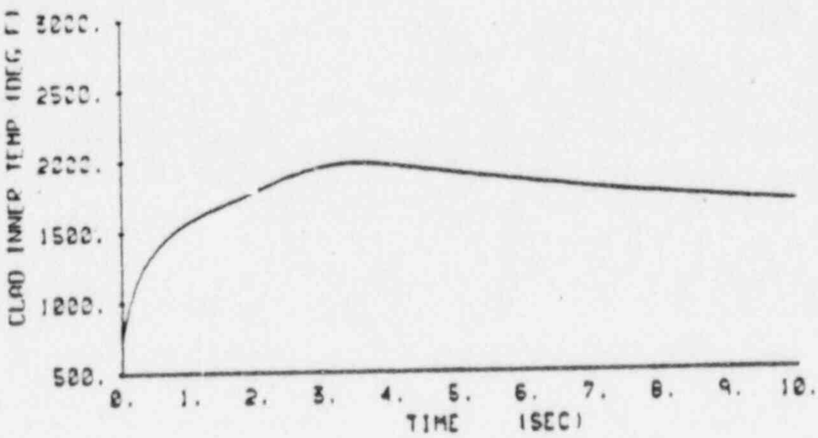
V. C. Summer

Figure 15.4.4-1

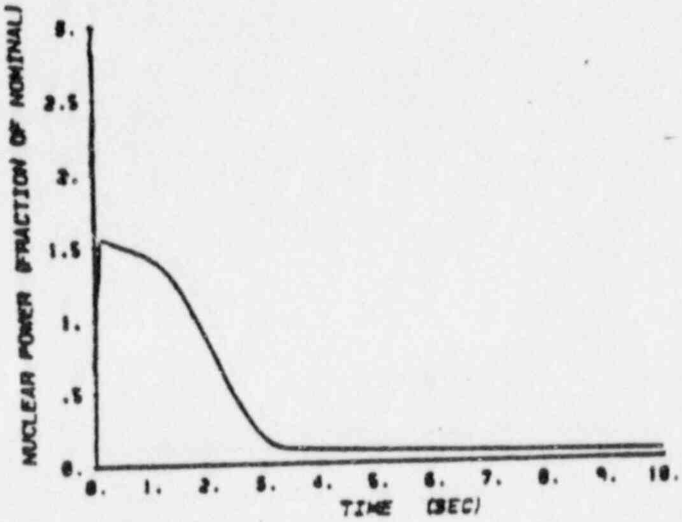
All Loops Operating
One Locked Rotor
RCS Pressure, RCS Flow and
Faulted Loop Flow vs. Time

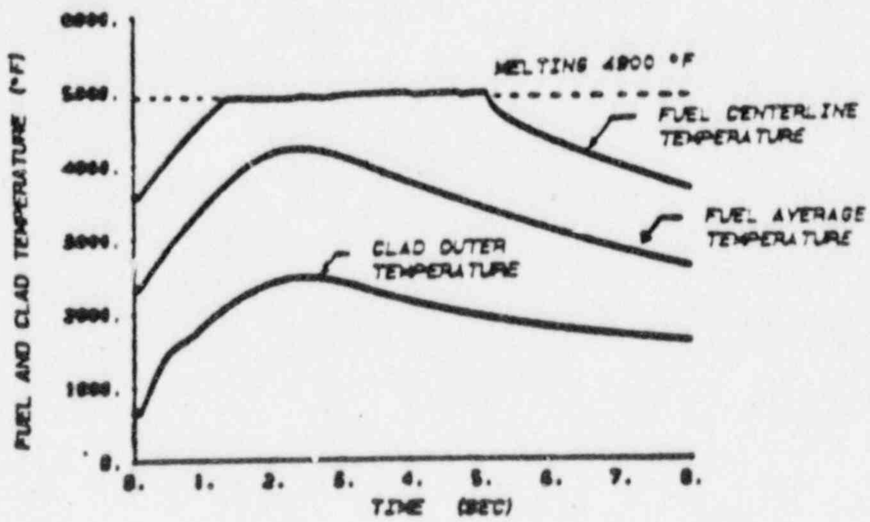




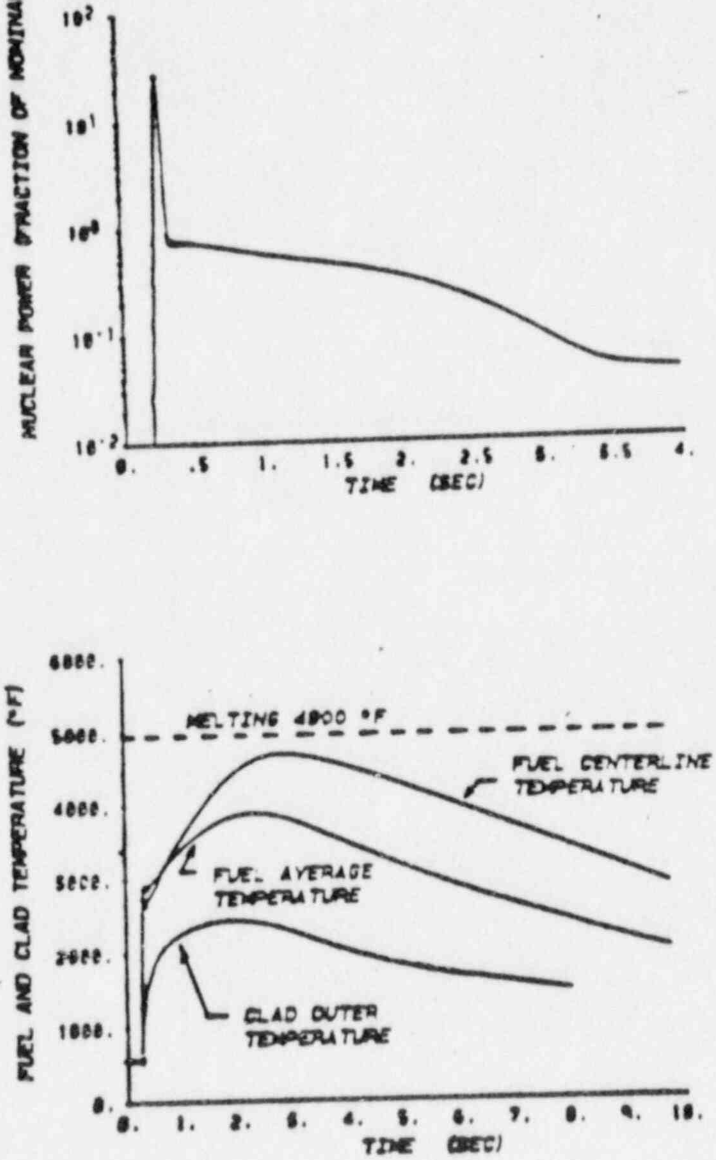


V. C. Summer
Figure 15.4.4-2
All Loops Operating One Locked Rotor Nuclear Power, Heat Flux and Clad Temperature vs. Time





V. C. Summer
Figure 15.4.8-1
Rod Ejection Accident BOL HFP Nuclear Power, Hot Spot Fuel and Clad Temperature vs. Time



V. C. Summer
Figure 15.4.6-2
Rod Ejection Accident BOL HZP Nuclear Power, Hot Spot Fuel and Clad Temperature vs. Time

ATTACHMENT 4

LOCA ACCIDENT ANALYSES

FOR THE

V. C. SUMMER PLANT

TRANSITION TO 17x17 VANTAGE 5 FUEL

15.3 CONDITION III - INFREQUENT FAULTS

By definition, Condition III occurrences are faults which may occur very infrequently during the life of the plant. They will be accommodated with the failure of only a small fraction of the fuel rods although sufficient fuel damage might occur to preclude resumption of the operation for a considerable outage time. The release of radioactivity will not be sufficient to interrupt or restrict public use of those areas beyond the exclusion radius. A Condition III fault will not, by itself, generate a Condition IV fault or result in a consequential loss of function of the reactor coolant system or containment barriers. For the purpose of this report the following faults have been grouped into this category:

1. Loss of reactor coolant, from small ruptured pipes or from cracks in large pipes, which actuates the emergency core cooling system.

2. Minor secondary system pipe breaks.

3. Inadvertent loading of fuel assembly into an improper position.

4. Complete loss of forced reactor coolant flow.

5. Single rod cluster control assembly withdrawal at full power.

Each of these infrequent faults are analyzed in this section. In general, each analysis includes an identification of causes and description of the accident, an analysis of effects and consequences, a presentation of results, and relevant conclusions.

The time sequence of events during applicable Condition III faults 1 and 4 above is shown in Table 15.3-1.

15.3.1 LOSS OF REACTOR COOLANT FROM SMALL RUPTURED PIPES OR FROM CRACKS IN LARGE PIPES WHICH ACTUATES THE EMERGENCY CORE COOLING SYSTEM

15.3.1.1 Identification of Causes and Accident Description

A loss of coolant accident is defined as a rupture of the reactor coolant system piping or of any line connected to the system. See Section 5.2 for a more detailed description of the loss of reactor coolant accident boundary limits. Ruptures of small cross section will cause expulsion of the coolant at a rate which can be accommodated by the charging pumps which would maintain an operational water level in the pressurizer, permitting the operator to execute an orderly shutdown. The coolant which would be released to the containment contains the fission products existing in it.

The maximum break size for which the normal makeup system can maintain the pressurizer level is obtained by comparing the calculated flow from the reactor coolant system through the postulated break against the charging pump makeup flow at normal reactor coolant system pressure, i.e., 2250 psia. A makeup flow rate from one centrifugal charging pump is adequate to sustain pressurizer level at 2250 psia for a break through a 3/8 inch diameter hole. This break results in a loss of approximately 17.5 lb/sec.

Should a larger break occur, depressurization of the reactor coolant system causes fluid to flow to the reactor coolant system from the pressurizer, resulting in a pressure and level decrease in the pressurizer. Reactor trip occurs when the pressurizer low pressure trip setpoint is reached. The safety injection system is actuated when the appropriate setpoint is reached. The consequence of the accident are limited in two ways:

1. Reactor trip and borated water injection complement void formation in causing rapid reduction of nuclear power to a residual level corresponding to the delayed fission and fission product decay.

2. Injection of borated water ensures sufficient flooding of the core to prevent excessive clad temperatures.

Before the break occurs, the plant is an equilibrium condition, i.e., the heat generated in the core is being removed via the secondary system. During blowdown, heat from decay, hot internals and the vessel continues to be transferred to the reactor coolant system. The heat transfer between the reactor coolant system and the secondary system may be in either direction, depending on the relative temperatures. In the case of continued heat addition to the secondary, system pressure increases and steam dump may occur. Makeup to the secondary side is automatically provided by the emergency feedwater pumps. The safety injection signal stops normal feedwater flow by closing the main feedwater line isolation valves and initiates emergency feedwater flow by starting the emergency feedwater pumps. The secondary flow aids in the reduction of reactor coolant system pressure. When the reactor coolant system depressurizes to 600 psia, the accumulators begin to inject water into the reactor coolant loops. The reactor coolant pumps are assumed to be tripped at the initialization of the accident, and effects of pump coastdown are included in the blowdown analyses.

15.3.1.2 <u>Analysis of Effects and Consequences</u>

15.3.1.2.1 Method of Analysis

For loss-of-coolant accidents due to small breaks less than 1 square foot, the NOTRUMP computer code[13,14] is used to calculate the transient depressurization of the RCS as well as to describe the mass and enthalpy of flow through the break. The NOTRUMP computer code is a state-of-the-art one-dimensional general network code incorporating a number of advanced features. Among these are calculation of thermal non-equilibrium in all fluid volumes, flow regime-dependent drift flux calculations with counter-current flooding limitations, mixture level tracking logic in multiple-stacked fluid nodes and regime-dependent heat transfer correlations. The NOTRUMP small-break LOCA emergency core cooling (ECCS) evaluation model was developed to determine the RCS response to design basis small break LOCAs, and to address NRC concerns expressed in NUREG-0611, "Generic Evaluation of Feedwater Transients and Small Break Loss-of-Coolant Accidents in Westinghouse-Designed Operating Plants".

The reactor coolant system model is nodalized into volumes interconnected by flowpaths. The broken loop is modelled explicitly, while the two intact loops are lumped into a second loop. Transient behavior of the system is determined from the governing conservation equations of mass, energy, and momentum. The multinode capability of the program enables explicit, detailed spatial representation of various system components; which, among other capabilities, enables a proper calculation of the behavior of the loop seal during a loss-of-coolant accident. The reactor core is represented as heated control volumes with associated phase separation models to permit transient mixture height calculations. Detailed descriptions of the NOTRUMP code and the evaluation model are provided in References 13 and 14.

Safety injection systems consist of gas pressurized accumulator tanks and pumped injection systems. Minimum emergency core cooling system availability is assumed for the analysis. Assumed pumped safety injection characteristics as a function of RCS pressure used as boundary conditions in the analysis are shown in Figure 15.3-1. The injection rate is based upon the pump performance curves, but degraded for conservatism and to account for possible reduced injection rates due to pump cooling recirculation miniflow operation. Injection is delayed after the occurence of the injection signal as indicated in Table 15.3-1 to account for diesel generator startup and emergency power bus loading in case of a loss of offsite power coincident with an accident.

Peak clad temperature calculations are performed with the LOCTA-IV[1] code using the NOTRUMP calculated core pressure, fuel rod power history, uncovered core steam flow and mixture height as boundary conditions. Figure 15.3-2 depicts the hot rod axial power shape used to perform the small break analysis. This shape was chosen because it represents a distribution with power concentrated in the upper regions of the core. Such a distribution is limiting for small-break LOCAs because it minimizes coolant level swell, while maximizing vapor superheating and fuel rod heat generation in the uncovered elevations. Figure 15.3-3 presents the normalized core power curve as a function of time after reactor trip. The scram delay times denoted in Table 15.3-1 reflect the assumption that the core is assumed to continue to operate at full rated power until the control rods are completely inserted.

15.3.1.2.2 Results

This section presents results of the limiting break size analysis as determined by the highest peak fuel rod clad temperature for a range of break sizes. The limiting break size was found to be a 3-inch diameter cold leg break. The maximum temperature attained during the transient was 2095°F. Important parameters are summarized in Table 15.3-2, while the key transient event times are listed in Table 15.3-1. Figures 15.3-4 through 9 show for the three-inch break transient, respectively:

- RCS pressure,
- Core mixture level,
- Peak clad temperature,
- Core outlet steam flow,
- Hot spot rod surface heat transfer coefficient, and
- Hot spot fluid temperature.

During the initial period of the small-break transient, the effect of the break flow is not strong enough to overcome the flow maintained by the reactor recirculation cooling pumps as they coast down. Normal upward flow is maintained through the core and core heat is adequately removed. At the low heat generation rates following shutdown the fuel rods continue to be well cooled as long as the core is covered by a two-phase mixture level. From the clad temperature transient for the 3-inch break calculation shown in Figure 15.3-6, it is seen that the peak clad temperature occurs near the time at which the core is most deeply uncovered when the top of the core is steam cooled. This time is also accompanied by the highest vapor superheating above the mixture level.

15.3.1.2.3 Additional Break Sizes

Studies documented in references 9 and 10 determined that the limiting small-break size occurred for breaks less than 10 inches in diameter. To insure that the 3-inch diameter break was limiting, calculations were run with breaks of 2 inches and 4 inches. The results of these calculations are shown in the Sequence of Events Table 15.3-1, and the Results Table 15.3-2. Plots of the following parameters are shown in Figures 15.3-10 through 15 for the 2-inch break, and Figures 15.3-16 through 21 for the 4-inch break.

- RCS pressure,
- Core mixture level,
- Peak clad temperature,
- Core outlet steam flow,
- Hot spot rod surface heat transfer coefficient, and
- Hot spot fluid temperature.

As seen in Table 15.3-2 the maximum clad temperatures were calculated to be less than that for the 3-inch break.

15.3.1.2.4 Additional Analysis

NUREG-0737[11], Section II.K.3.31, required plant-specific small break LOCA analysis using an Evaluation Model revised per Section II.K.3.30. In accordance with NRC Generic Letter 83-65[12], generic analyses using NOTRUMP[17,18] were performed and are presented in WCAP-11145[15]. Those results demonstrate that in a comparison of cold leg, hot leg and pump suction leg break locations, the cold leg break location is limiting.

Analyses of a LOCA in the pressurizer vapor space such as that caused by opening a pressurizer relief valve or a safety valve were provided in WCAP-9600[10]. The conclusion presented in WCAP-9600 is that these breaks are not limiting since little or no core uncovery will take place. WCAP-9600 states that the analyses reported therein apply to all Westinghouse designed plants.

Calculations were also performed for the Virgil Summer plant with the NOTRUMP[13,14] and LOCTA-IV[1] codes to examine the influence of initial loop fluid operating temperatures on small break LOCA peak clad temperature. The results showed that peak clad temperature decreased as loop operating temperature decreased.

15.3.1.3 Conclusions

Analyses presented in this section show that the high head portion of the emergency core cooling system, together with accumulators, provide sufficient core flooding to keep the calculated peak clad temperatures below required limits of 10 CFR 50.46. Hence, adequate protection is afforded by the emergency core cooling system in the event of a small break loss of coolant accident.

15.3.2 MINOR SECONDARY SYSTEM PIPE BREAKS

15.3.2.1 Identification of Causes and Accident Description

Included in this grouping are ruptures of secondary system lines which would result in steam release rates equivalent to a 6 inch diameter break or smaller.

15.3.2.2 Analysis of Effects and Consequences

Minor secondary system pipe breaks must be accommodated with the failure of only a small fraction of the fuel elements in the reactor. Since the results of analysis presented in Section 15.4.2 for a major secondary system pipe rupture also meet this criteria, separate analysis for minor secondary system pipe breaks is not required.

The analysis of the more probable accidental opening of a secondary system steam dump, relief or safety valve is presented in Section 15.2.13. These analyses are illustrative of a pipe break equivalent in size to a single valve opening.

3. The iodine partition factor for activity released from the break is 0.1.

4. The concentration of radioactive nuclides in the reactor coolant is listed in Table 11.1-2 for the conservative case and in Table 11.1-5 for the realistic case.

Using the previously listed assumptions, isotopic releases to the environment are determined to be those listed in Tables 15.3-6 and 15.3-7 for the realistic and conservative cases, respectively.

Gamma, beta and thyroid doses at the site boundary for the realistic case are 7.30×10^{-6} Rem, 6.81×10^{-6} Rem and 5.90×10^{-4} Rem, respectively. Corresponding doses at the low population zone are 8.20×10^{-7} Rem, 7.65×10^{-7} Rem and 6.63×10^{-5} Rem, respectively.

Gamma, beta and thyroid doses at the site boundary for the conservative case are 3.18×10^{-2} Rem, 3.66×10^{-2} Rem and 6.72×10^{-1} Rem, respectively. Corresponding doses at the low population zone are 1.85×10^{-3} Rem, 2.13×10^{-3} Rem and 3.90×10^{-2} Rem, respectively.

Doses resulting from this accident are well within the limits defined in 10 CFR 20 (25 Rem whole body and 300 Rem thyroid).

15.3.8 REFERENCES

1. Bordelon, F. M., et al., "LOCTA-IV Program: Loss of Coolant Transient Analysis," WCAP-8301 (Proprietary) and WCAP-8305 (Non-Proprietary), June, 1974.

2. Hellman, J. M., "Fuel Densification Experimental Results and Model for Reactor Application," WCAP-8218-P-A (Proprietary) and WCAP-8219-A (Non-Proprietary), March, 1975.

3. Altamore, S. and Barry, R. F., "The TURTLE 24.0 Diffusion Depletion Code," WCAP-7213-P-A (Proprietary) and WCAP-7758-A (Non-Proprietary), January, 1975.

4. Barry, R. F., "LEOPARD - A Spectrum Dependent Non-Spatial Depletion Code for the IBM-7094," WCAP-3269-26, September, 1963.

5. Baldwin, M. S., Merrian, M. M., Schenkel, H. S. and Van De Walle, D. J., "An Evaluation of Loss of Flow Accidents Caused by Power System Frequency Transients in Westinghouse PWR's," WCAP-8424, Revision 1, June, 1975.

6. Bordelon, F. M., "Calculation of Flow Coastdown After Loss of Reactor Coolant Pump (PHOENIX Code)," WCAP-7973, September, 1972.

7. Burnett, T. W. T., et al., "LOFTRAN Code Description," WCAP-7907, June, 1972.

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9. Salvatori, R., "Westinghouse ECCS Plant Sensitivity Studies," WCAP-8340 (Proprietary) and WCAP-8356 (Non-Proprietary), July, 1974.

10. "Report on Small Break Accidents for Westinghouse NSSS System, "WCAP-9600 (Proprietary) and WCAP-9601 (Non-Proprietary), June, 1979.

11. "Clarification of TMI Action Plan Requirements", NUREG-0737, November, 1980.

12. NRC Generic Letter 83-85 from D. G. Eisenhut, "Clarification of TMI Action Plan Item II.K.3.31", November 2, 1983.

13. Meyer, P. E., "NOTRUMP - A Nodal Transient Small Break and General Network Code", WCAP-10079-P-A, August 1985.

14. Lee, N. et al., "Westinghouse Small Break ECCS Evaluation Model Using the NOTRUMP Code", WCAP-10054-P-A, August 1985.

15. Rupprecht, S. D., et al., "Westinghouse Small Break LOCA ECCS Evaluation Model Generic Study With the NOTRUMP Code"; WCAP-11145.

TABLE 15.3-1

TIME SEQUENCE OF EVENTS FOR CONDITION III EVENTS

Small-break Loss of Coolant Accident

Event	Time (s) Break Size: 2-Inch	3-Inch	4-Inch
Break occurs	0	0	0
Reactor trip signal	125.7	34.15	21 07
Core power shutdown	131.4	39.85	26.77
Safety injection signal	143.3	49.84	33.04
Safety injection begins	175.3	81.84	65.04
Top of core uncovered	1605	726	450
Accumulator injection begins	N/A	1183	660
Peak clad temperature occurs	2550	1374	752
Core recovered	5575	3700	1875

TABLE 15.3-1 (Continued)

TIME SEQUENCE OF EVENTS FOR
CONDITION III EVENTS

Accident	Event	Time (Seconds)
Complete Loss of Forced Reactor Coolant Flow		
1. Three pumps in operation, three pumps coasting down	All operating pumps loos power and begin coasting down	0
	Reactor coolant pump under-voltage trip pointed reached	0
	Rods begin to drop	1.5
	Minimum DNBR occurs	3.0
Complete Loss of Forced Reactor Coolant Flow		
2. Two pumps in operation, two pumps coasting down	All operating pumps lose power and begin coasting down	0
	Reactor coolant pump under-voltage trip point reached	0
	Rods begin to drop	1.5
	Minimum DNBR occurs	2.7

TABLE 15.3-2

SMALL-BREAK LOSS OF COOLANT ACCIDENT CALCULATION

RESULTS

PARAMETER	VALUE		
Break Size:	2-Inch	3-Inch	4-Inch
Peak clad temperature (°F)	1284	2095	1314
Elevation (ft)	11.75	12.00	11.50
Zr/H_2O cumulative reaction			
Maximum local (%)	0.28	9.44	0.11
Elevation (ft)	11.75	12.00	11.50
Total core (%)	< 0.3	< 0.3	< 0.3
Rod Burst	None	None	None

SIGNIFICANT INPUT PARAMETERS

Licensed core power	2775 MW
Peak linear heat generation rate	13.303 kW/ft
Accumulator	
Tank water volume	1014 ft^3
Pressure	600 psi

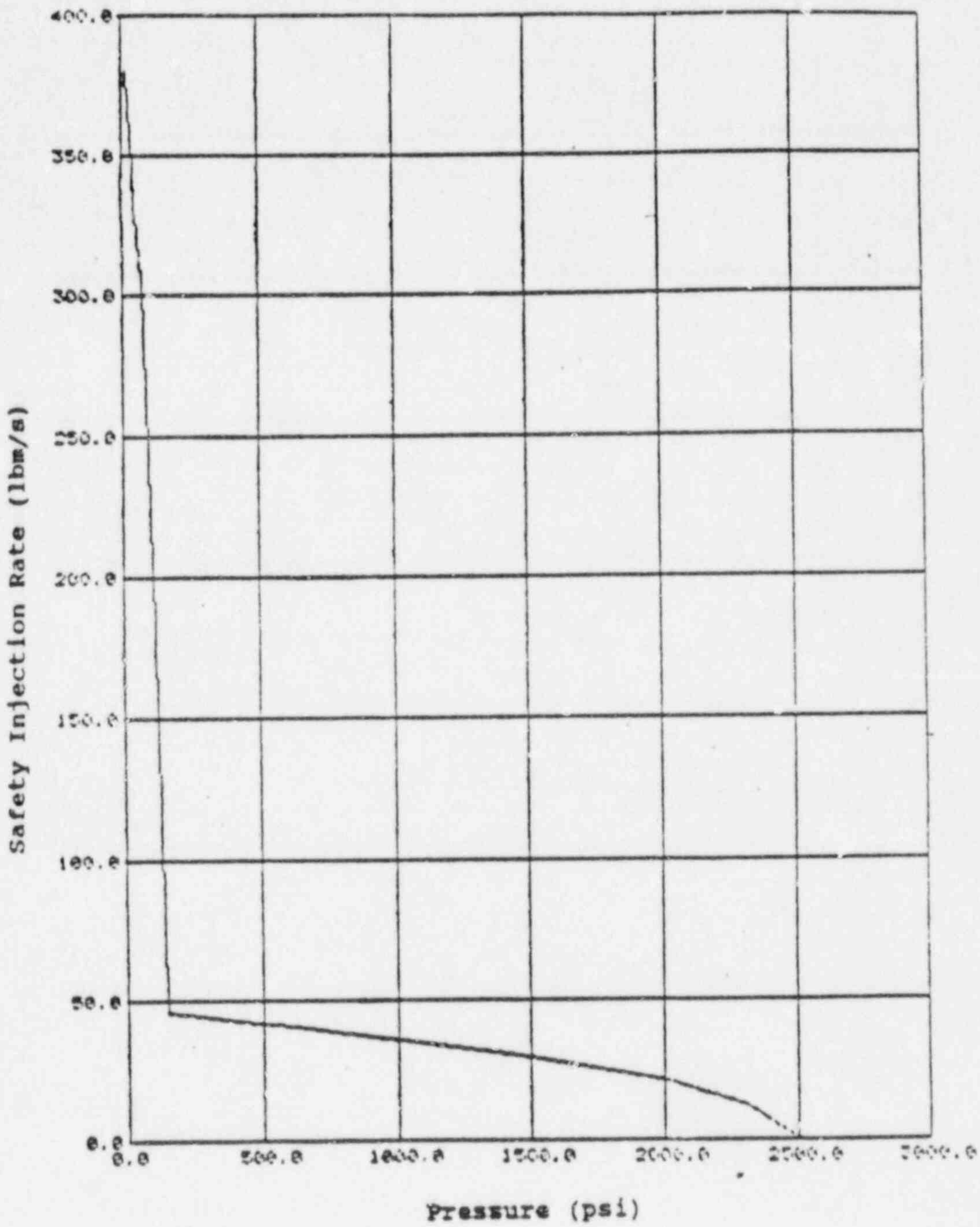

Figure 15.3-1. Pumped safety injection rate as a function of reactor coolant system pressure.

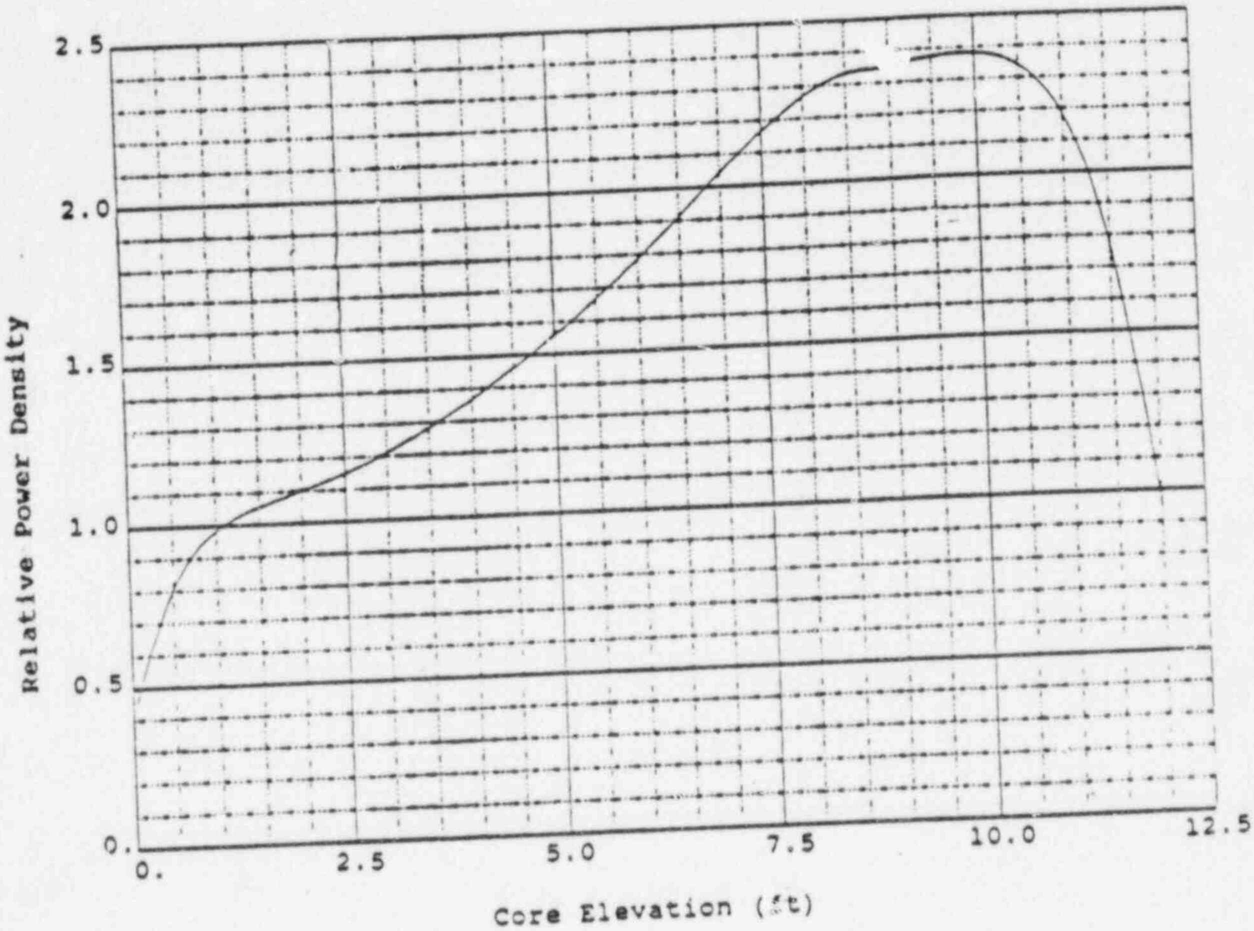

Figure 15.3-2. Peak rod axial power shape as a function of core average linear heat generation rate.

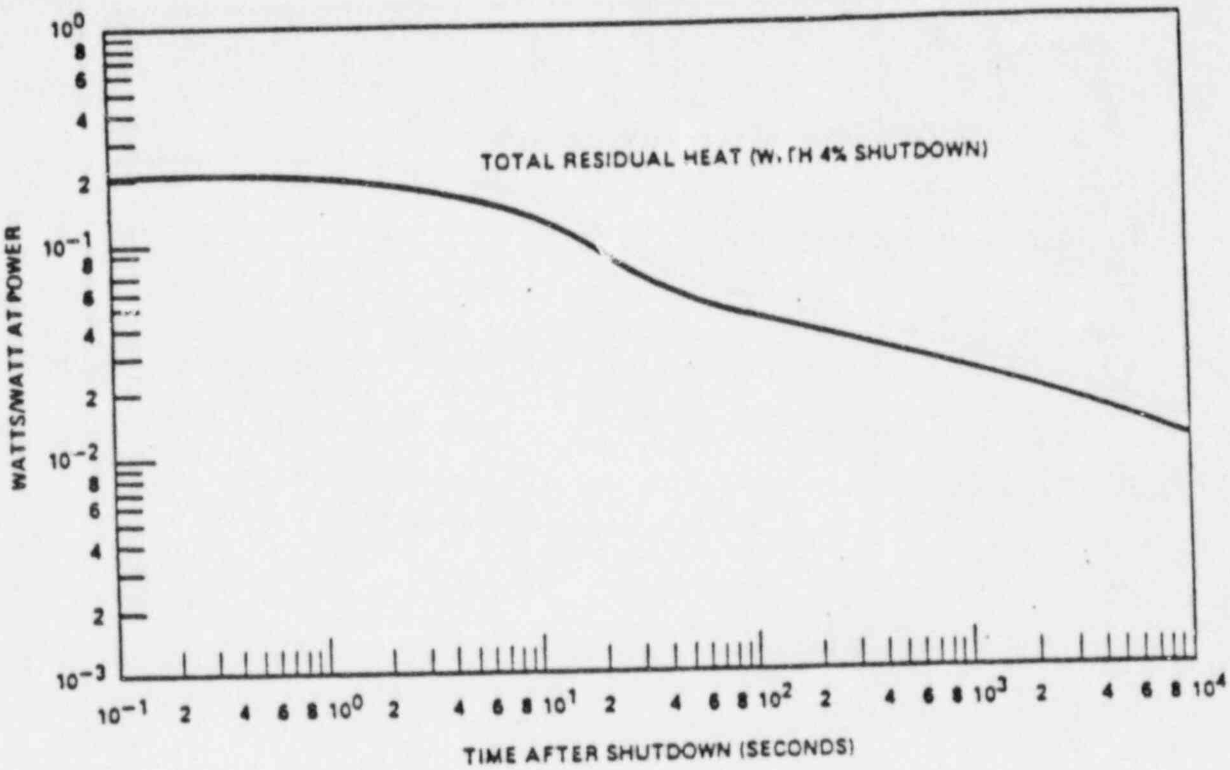

Figure 15.3-3. Normalized core heat generation rate following shutdown (full rod insertion).

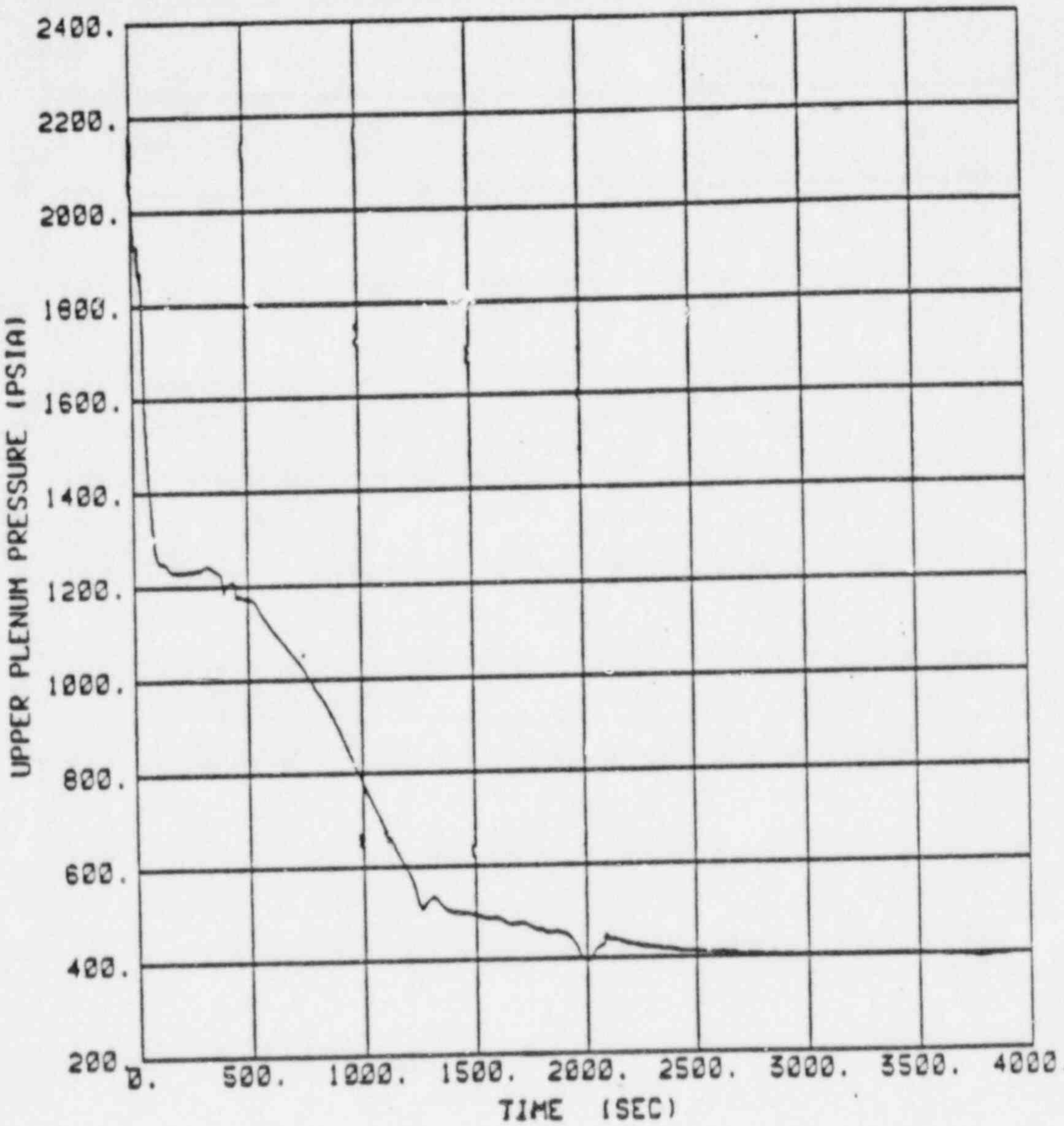

Figure 15.3-4. Reactor coolant system depressurization transient (3-inch break).

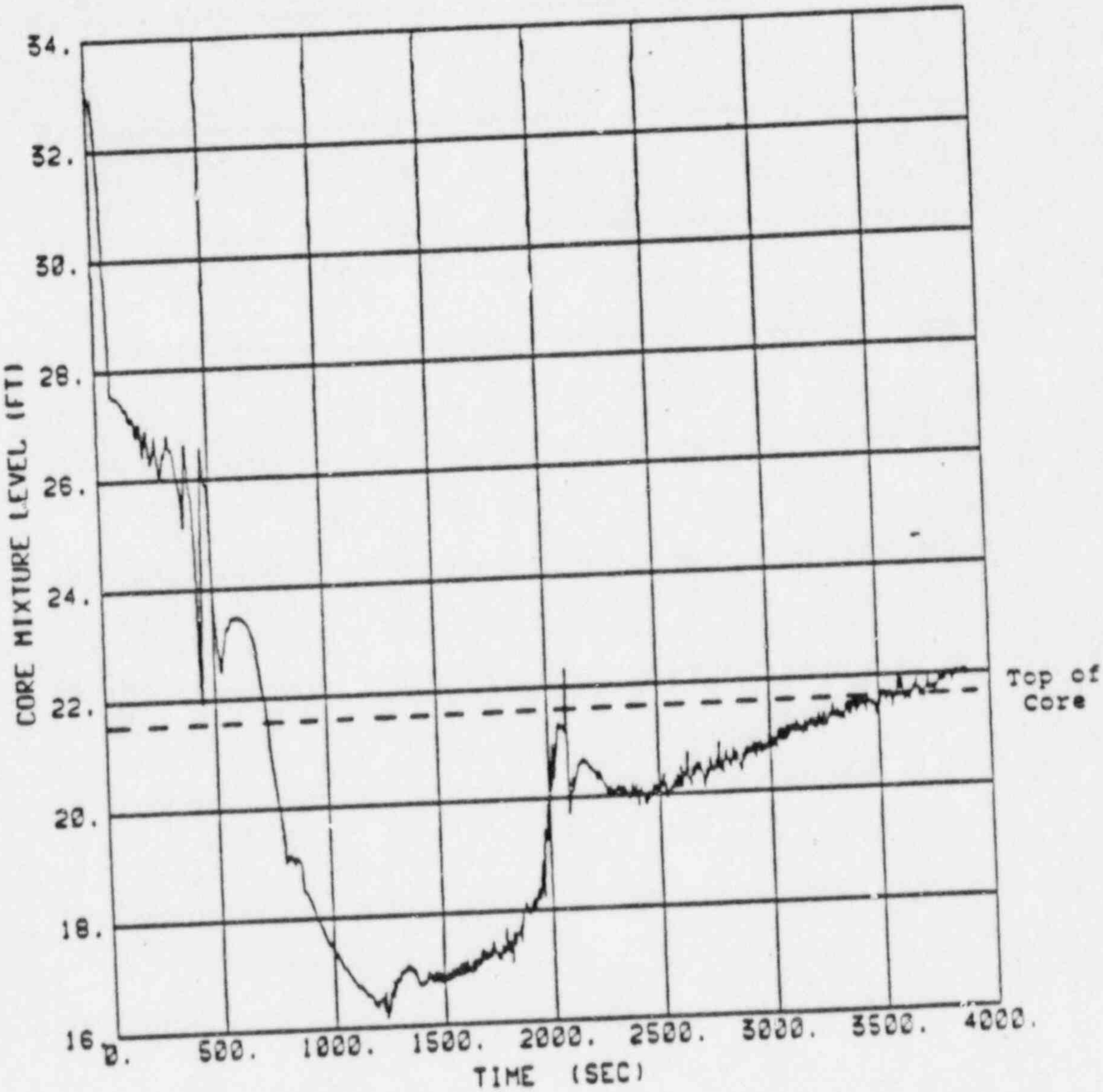

Figure 15.3-5. Core mixture height (3-inch break).

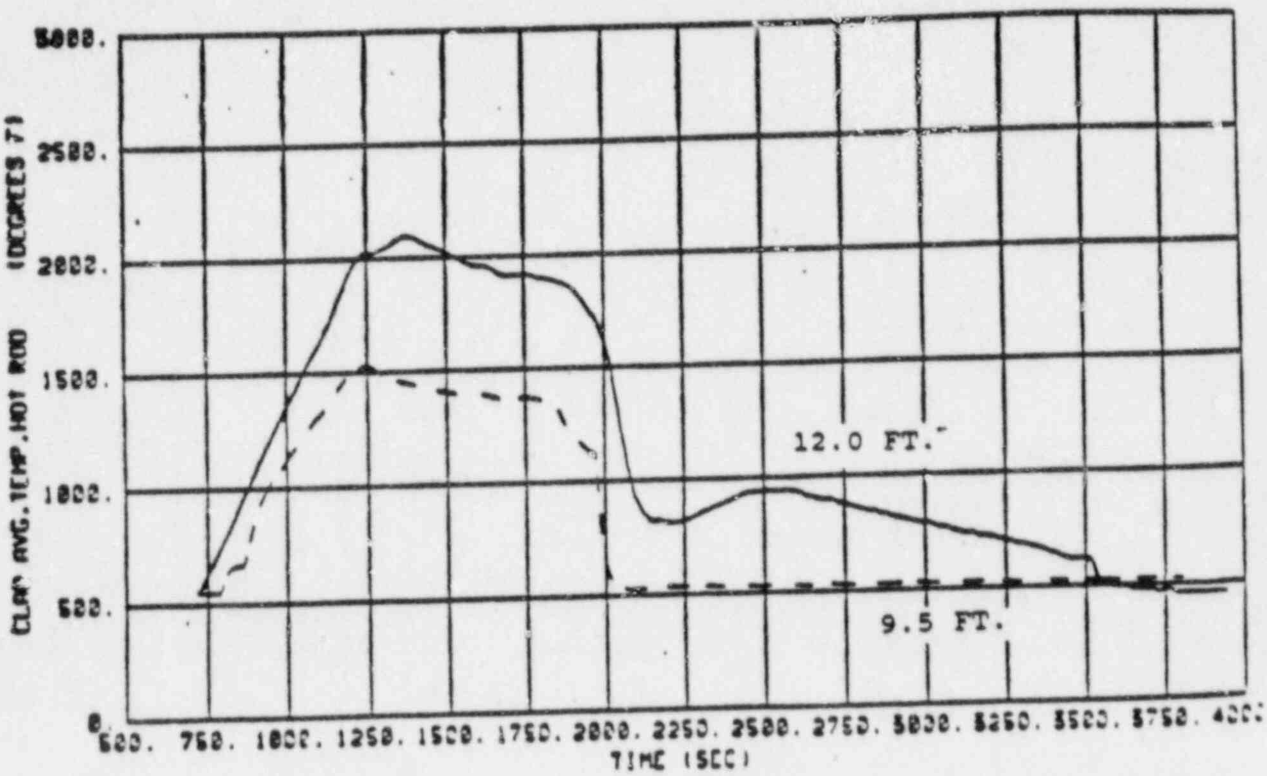

Figure 15.3-6. Clad temperature transient at peak temperature elevation and a selected lower elevation (3-inch break).

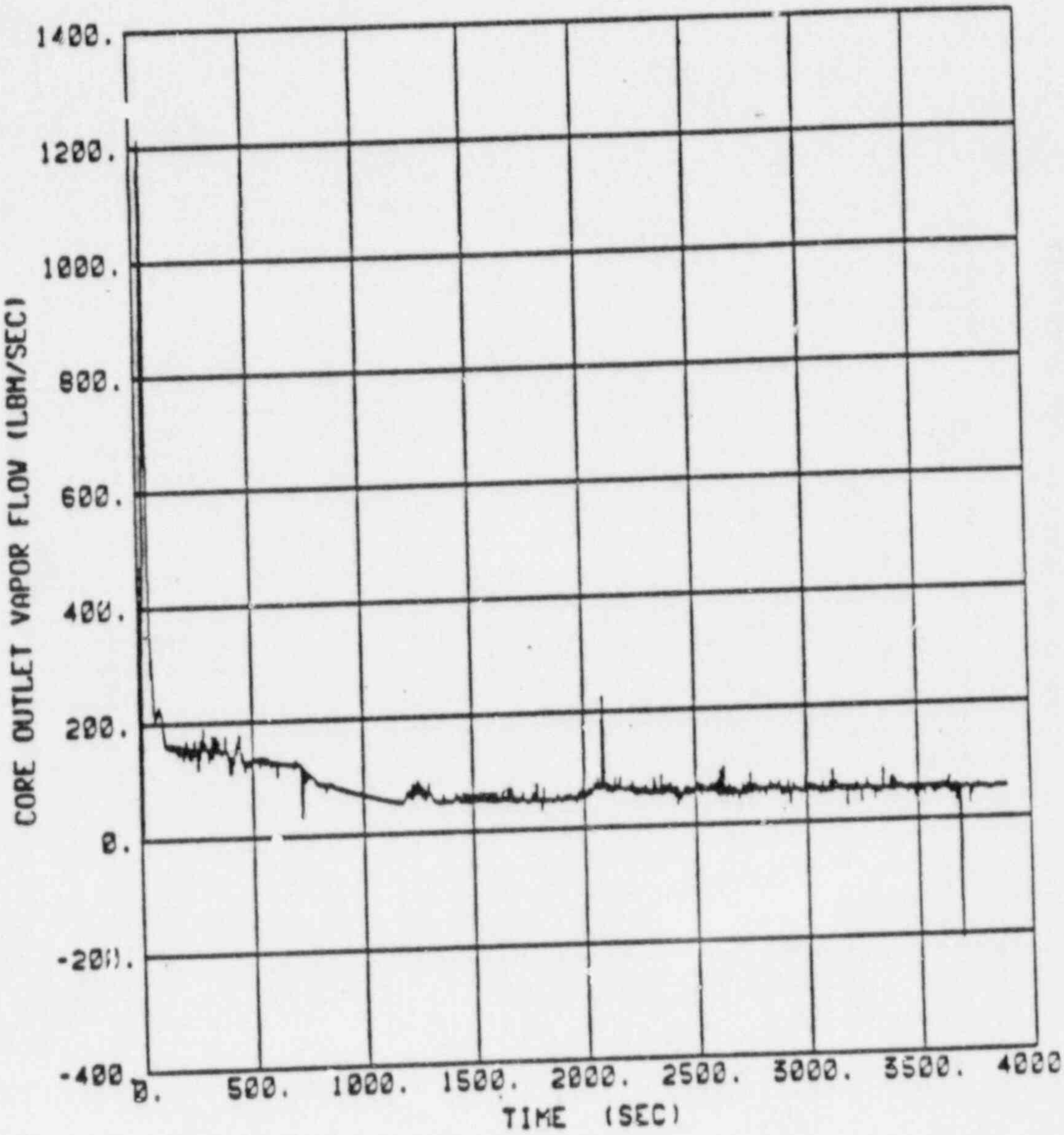

Figure 15.3-7. Steam mass flowrate out the top of the core (3-inch break).

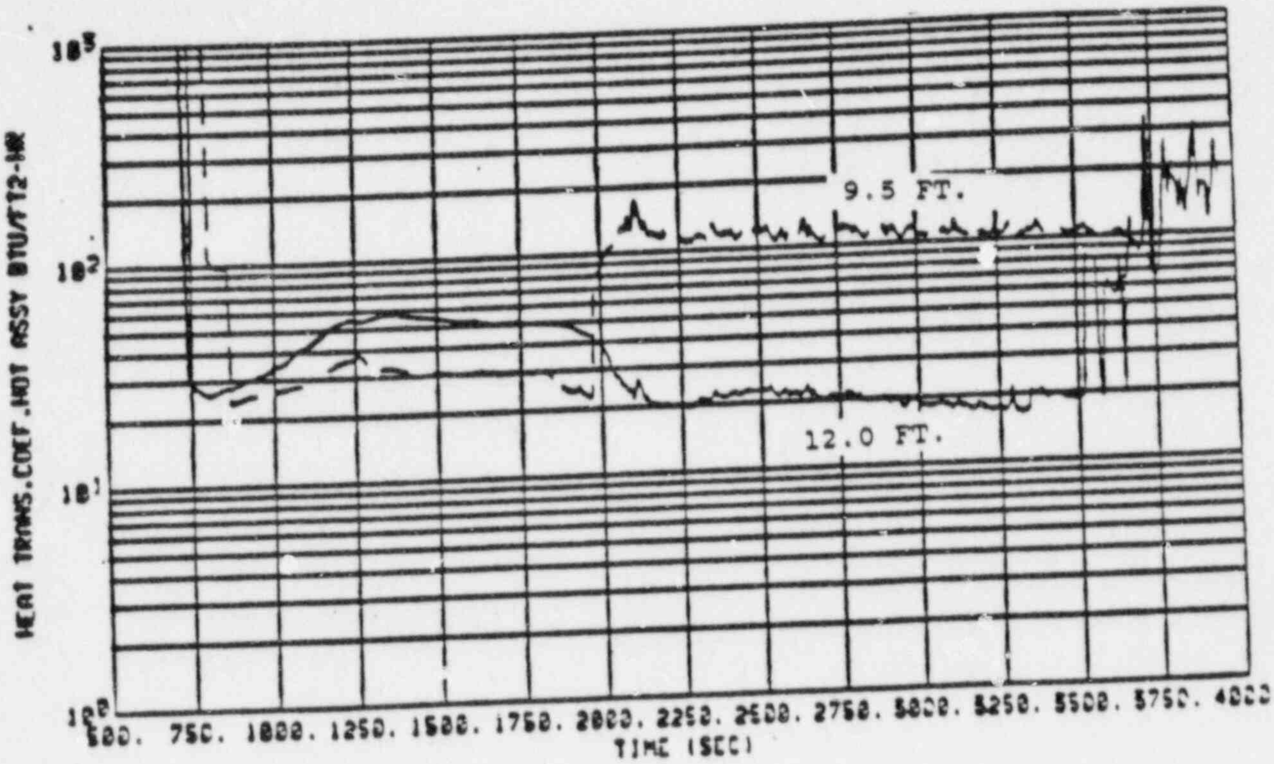

Figure 15.3-8. Clad surface heat transfer coefficients on the peak power rod (3-inch break).

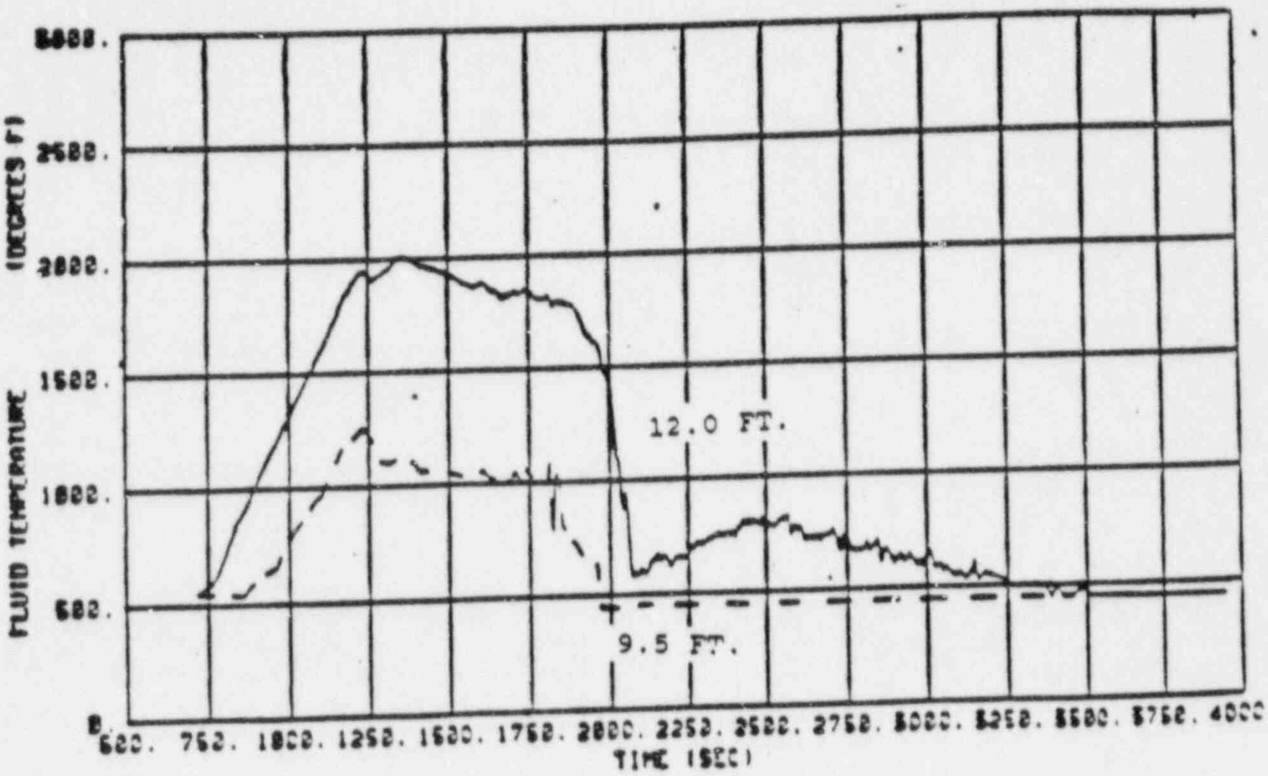

Figure 15.3-9. Fluid temperature at the peak temperature elevation and a selected lower elevation (3-inch break)

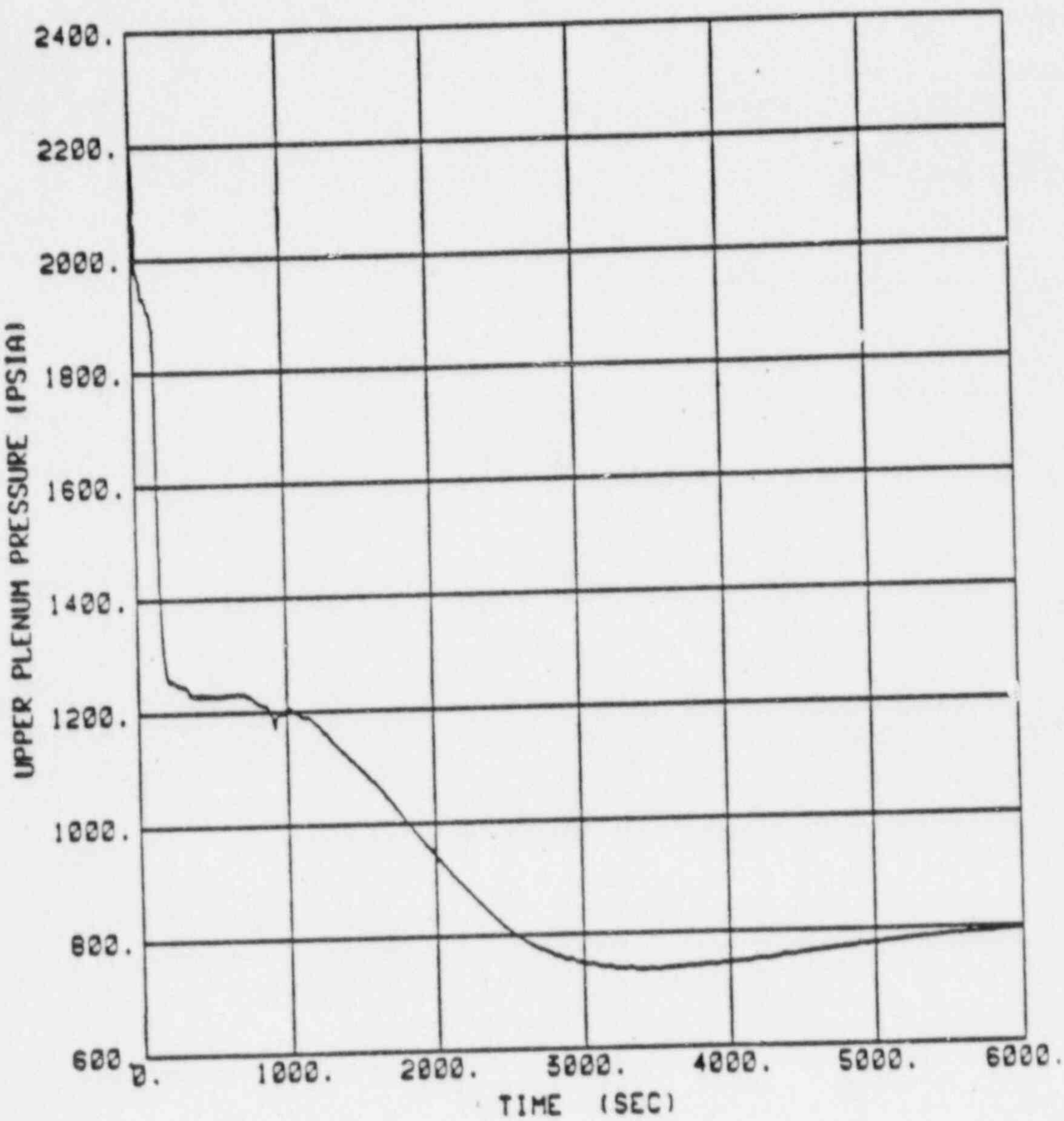

Figure 15.3-10. Reactor coolant system depressurization transient (2-inch break).

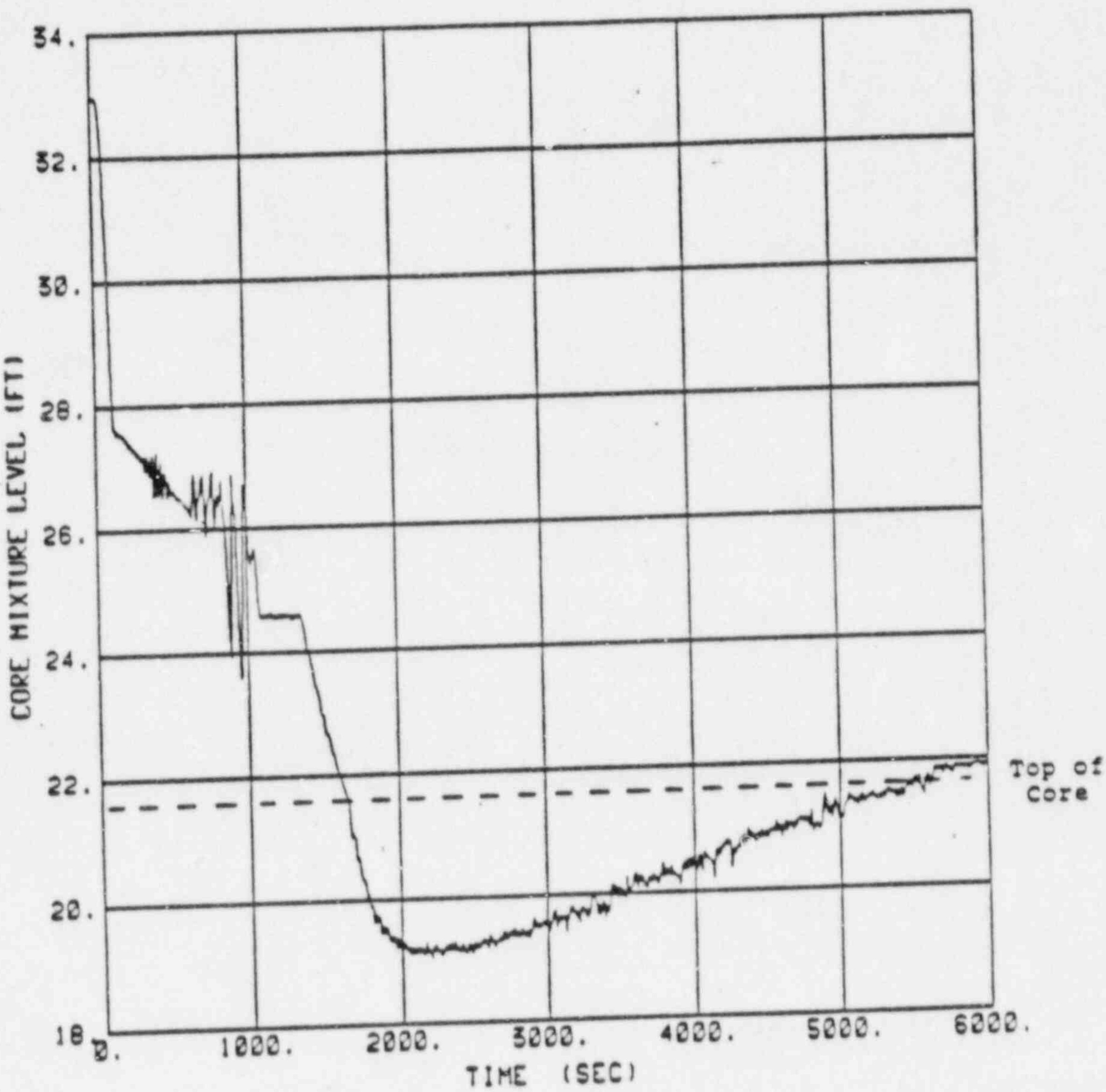

Figure 15.3-11. Core mixture height (2-inch break).

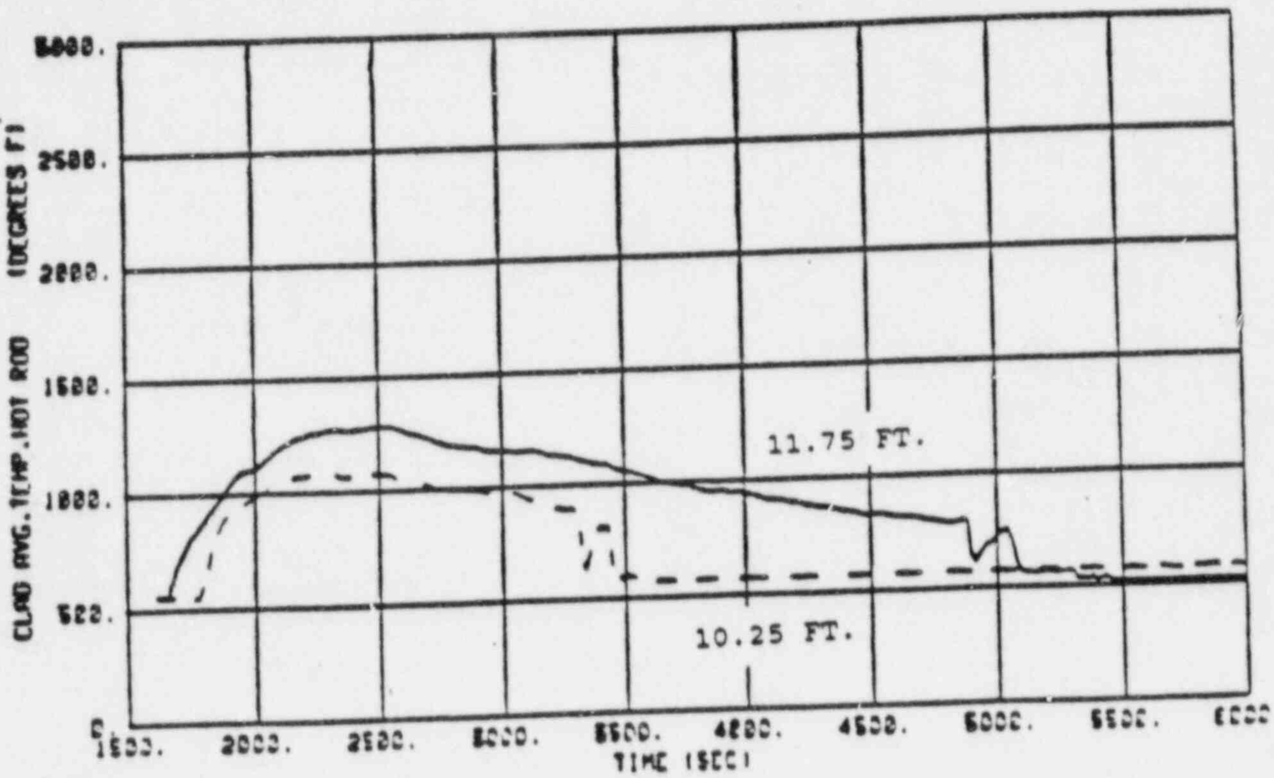

Figure 15.3-12. Clad temperature transient at peak temperature elevation and a selected lower elevation (2-inch break).

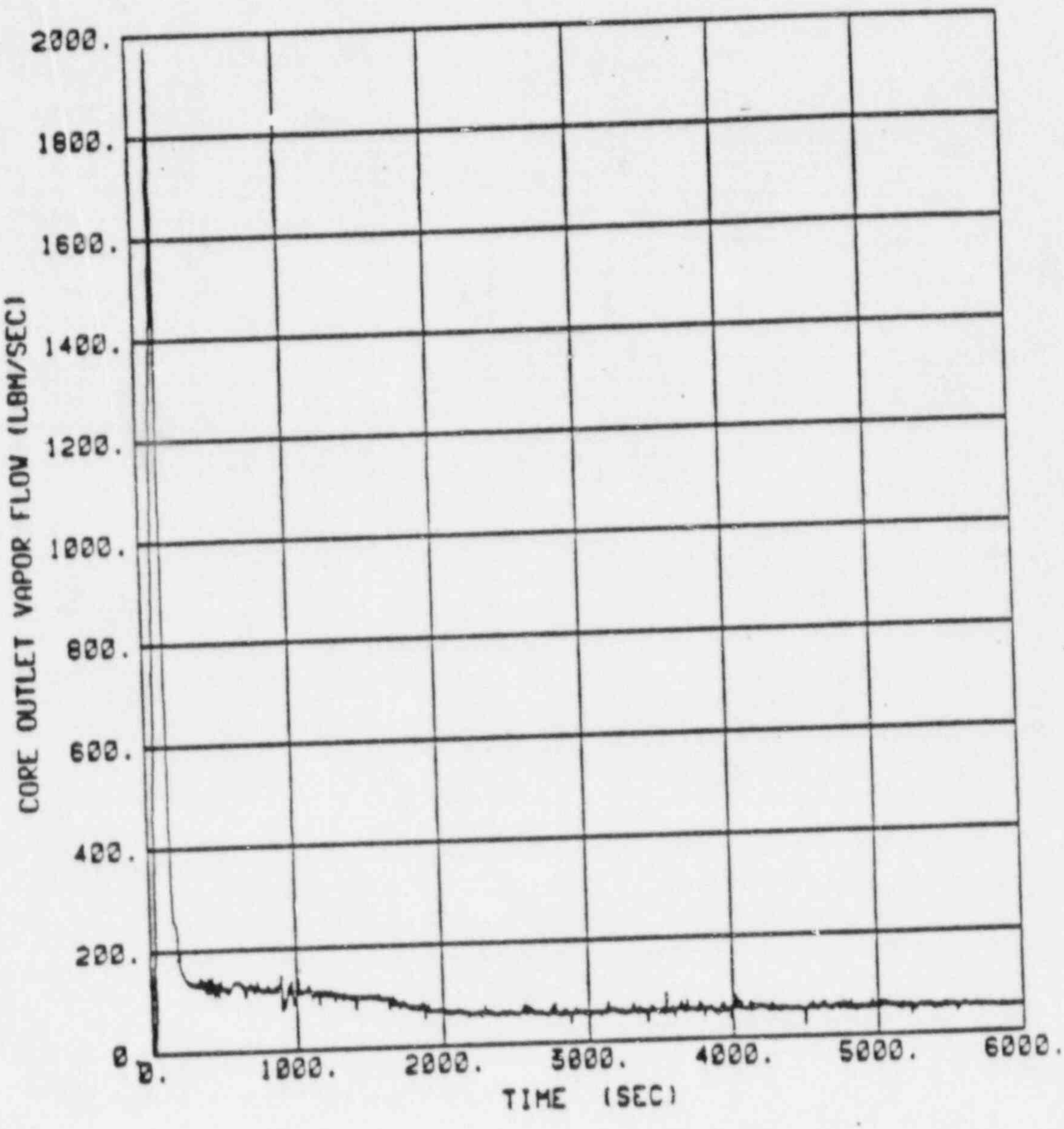

Figure 15.3-13. Steam mass flowrate out the top of the core (2-inch break).

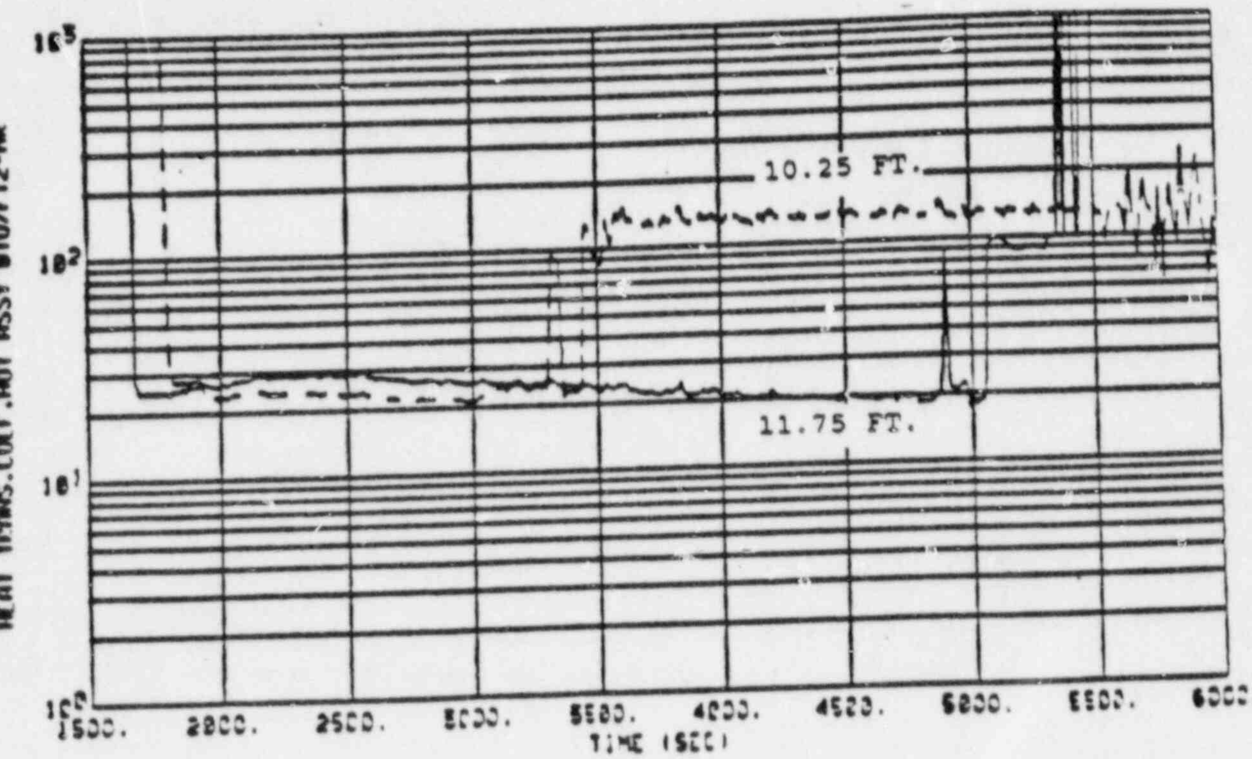

Figure 15.3-14. Clad surface heat transfer coefficients on the peak power rod (2-inch break).

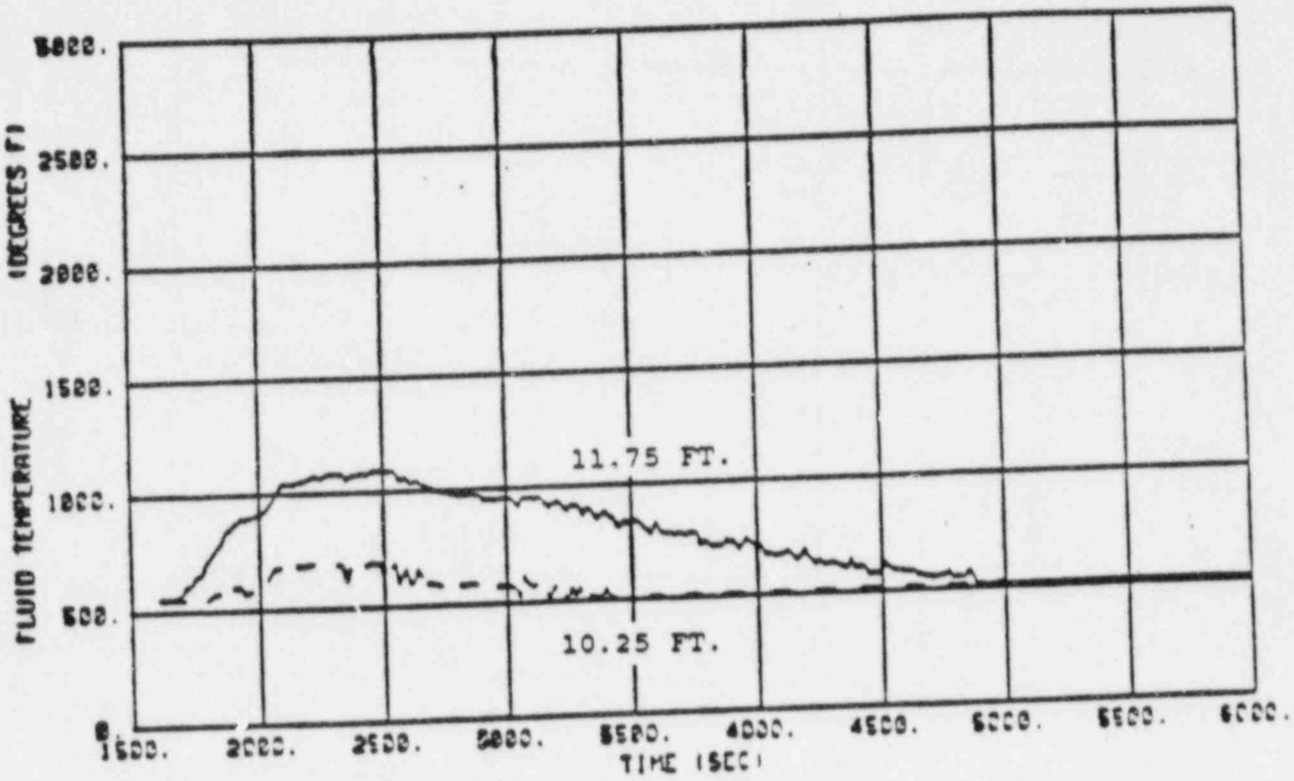

Figure 15.3-15. Fluid temperature at the peak temperature elevation and a selected lower elevation (2-inch break)

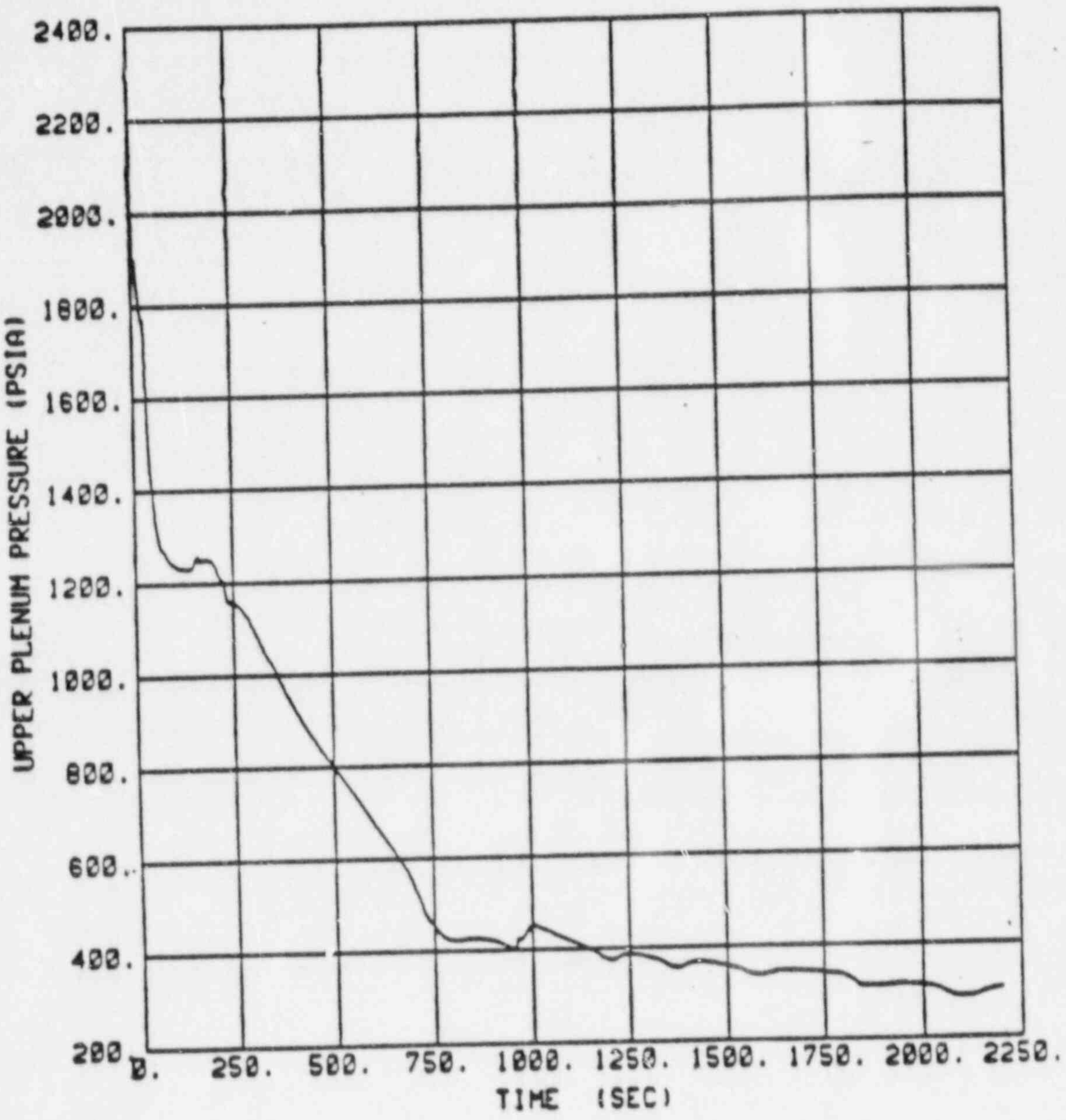

Figure 15.3-16. Reactor coolant system depressurization transient (4-inch break)

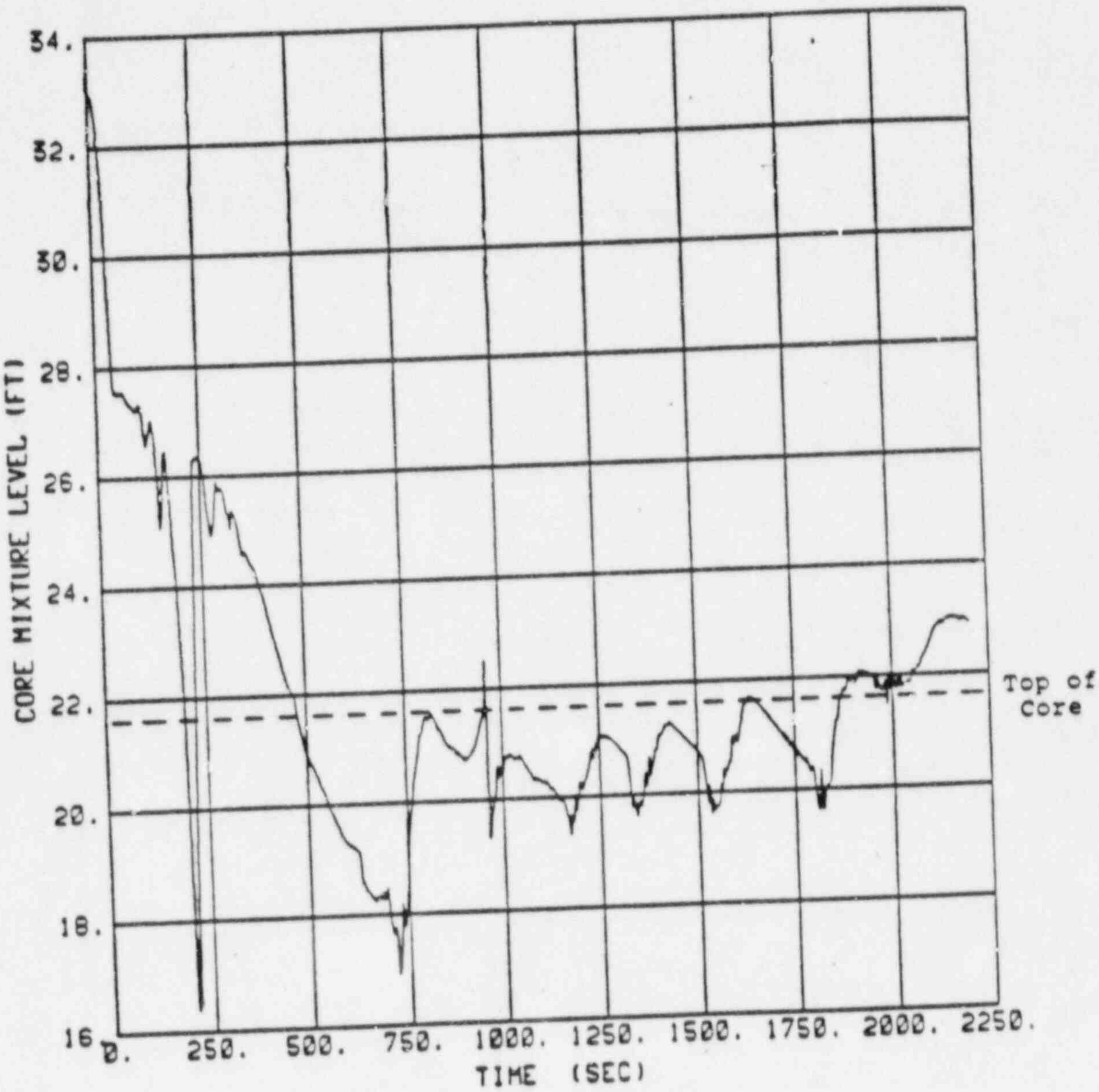

Figure 15.3-17. Core mixture height (4-inch break)

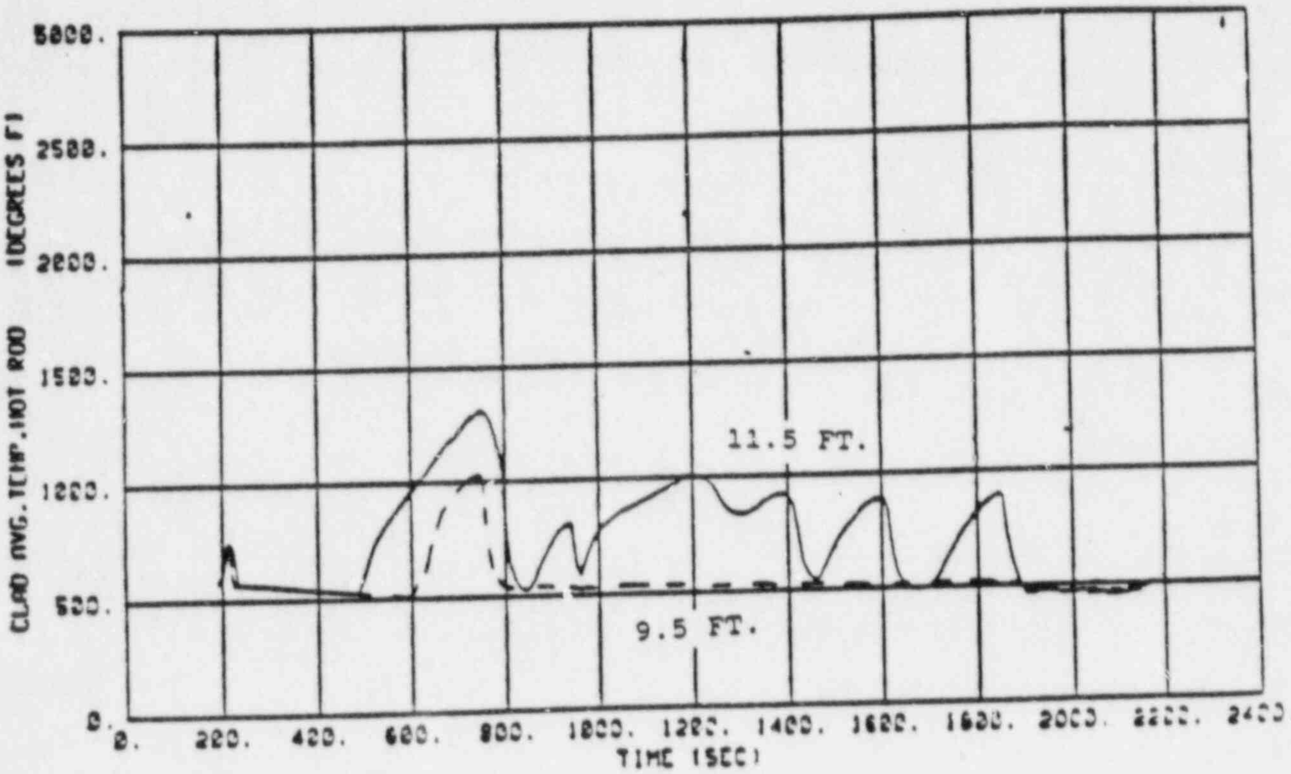

Figure 15.3-18. Clad temperature transient at peak temperature elevation and a selected lower elevation (4-inch break)

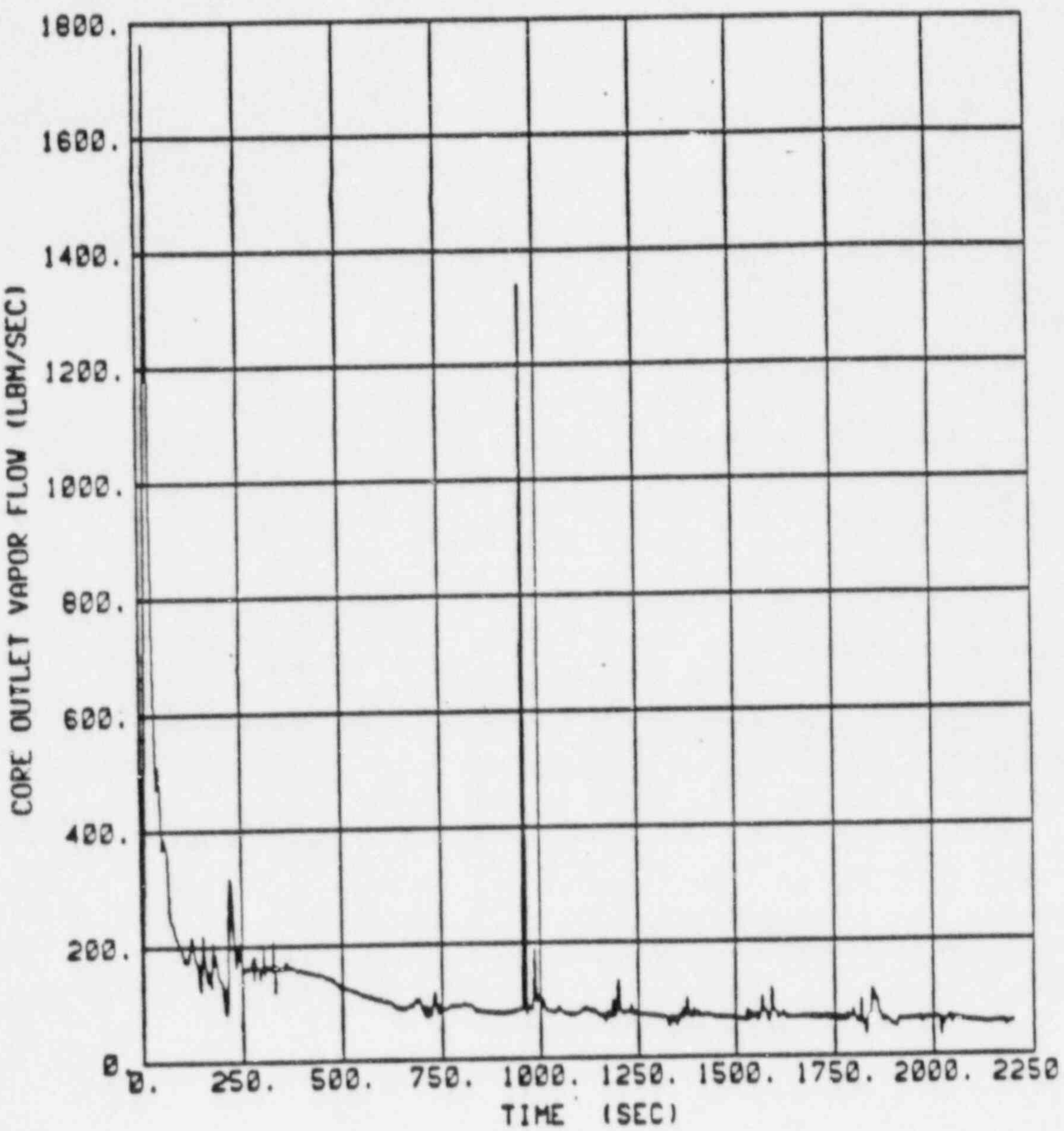

Figure 15.3-19. Steam mass flowrate out the top of the core (4-inch break)

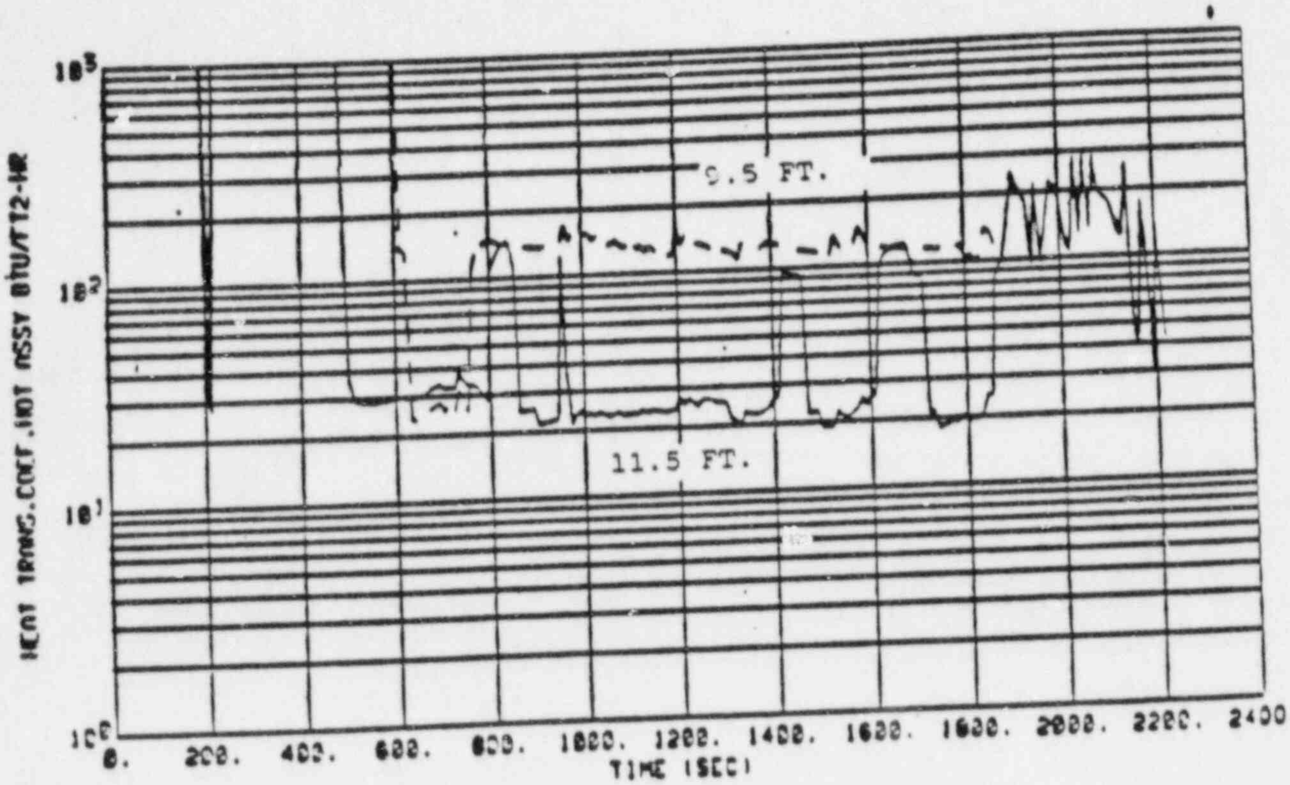

Figure 15.3-20. Clad surface heat transfer coefficients on the peak power rod (4-inch break)

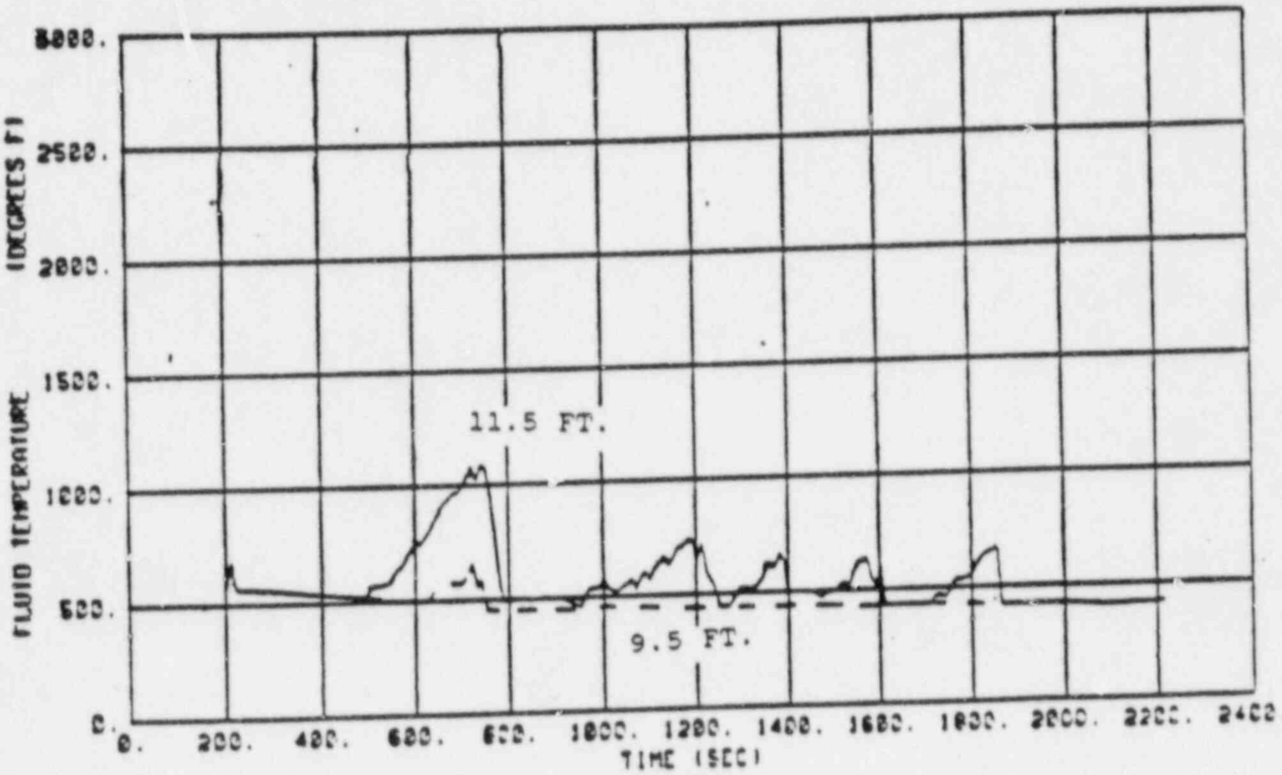

Figure 15.3-21. Fluid temperature at the peak temperature elevation and a selected lower elevation (4-inch break)

15.4 CONDITION IV - LIMITING FAULTS

Condition IV occurrences are faults that are not expected to take place, but are postulated because their consequences would include the potential for the release of significant amounts of radioactive material. These are the most drastic occurrences that must be designed against and represent limiting design cases. Condition IV faults shall not cause a fission product release to the environment resulting in an undue risk to public health and safety in excess of guideline values of 10 CFR 100. A single Condition IV fault shall not cause a consequential loss of required functions of systems needed to cope with the fault including those of the emergency core cooling system (ECCS) and the containment. For the purposes of this report the following faults have been classified in this category:

(1) Major rupture of pipes containing reactor coolant up to and including double-ended rupture of the largest pipe in the reactor coolant system (RCS), i.e., loss-of-coolant accident (LOCA);

(2) Major secondary system pipe ruptures;

(3) Steam generator tube rupture;

(4) Single reactor coolant pump (RCP) locked rotor;

(5) Rupture of a control rod mechanism housing (rod cluster control assembly [RCCA] ejection).

Each of these five limiting faults is analyzed in Section 15.4. In general, each analysis includes an identification of causes and description of the accident, an analysis of effects and consequences, a presentation of results, and relevant conclusions.

15.4.1 MAJOR REACTOR COOLANT SYSTEM PIPE RUPTURES (LOSS OF COOLANT ACCIDENT)

15.4.1.1 Identification of Causes and Frequency Classification

A loss-of-coolant accident (LOCA) is the result of a pipe rupture of the RCS pressure boundary. For the analyses reported here, a major pipe break (large break) is defined as a rupture with a total cross-sectional area equal to or greater than 1.0 ft^2. This event is considered an ANS Condition IV event, a limiting fault, in that it is not expected to occur during the lifetime of Virgil C. Summer, but is postulated as a conservative design basis. The results for the small break loss of coolant accident are presented in Section 15.3.1. The boundary considered for loss of coolant accidents are related to connecting pipe is defined in Section 3.6.

The Acceptance Criteria for the LOCA are described in 10 CFR 50.46 (10 CFR 50.46 and Appendix K of 10 CFR 50 1974)[1] as follows:

1. The calculated peak fuel element clad temperature is below the requirement of 2,200°F.

2. The amount of fuel element cladding that reacts chemically with water or steam does not exceed 1 percent of the total amount of Zircaloy in the reactor.

3. The clad temperature transient is terminated at a time when the core geometry is still amenable to cooling. The localized cladding oxidation limit of 17 percent is not exceeded during or after quenching.

4. The core remains amenable to cooling during and after the break.

5. The core temperature is reduced and decay heat is removed for an extended period of time, as required by the long-lived radioactivity remaining in the core.

These criteria were established to provide significant margin in emergency core cooling system (ECCS) performance following a LOCA. WASH-1400 (USNRC 1975)(10) presents a recent study in regards to the probability of occurrence of RCS pipe ruptures.

15.4.1.2 Sequence of Events and Systems Operations

Should a major break occur, depressurization of the RCS results in a pressure decrease in the pressurizer. The reactor trip signal subsequently occurs when the pressurizer low pressure trip setpoint is reached. A safety injection signal is generated when the appropriate setpoint is reached. These countermeasures will limit the consequences of the accident in two ways:

1. Reactor trip and borated water injection supplement void formation in causing rapid reduction of power to a residual level corresponding to fission product decay heat. However, no credit is taken in the LOCA analysis for the boron content of the injection water. In addition, the insertion of control rods to shut down the reactor is neglected in the large break analysis.

2. Injection of borated water provides for heat transfer from the core and prevents excessive clad temperatures.

15.4.1.3 Description of Large Break Loss-of-Coolant Accident Transient

The sequence of events following a large break LOCA is presented in Table 15.4.1-1.

Before the break occurs, the unit is in an equilibrium condition; that is, the heat generated in the core is being removed via the secondary system. During blowdown, heat from fission product decay, hot internals and the vessel continues to be transferred to the reactor coolant. At the beginning of the blowdown phase, the entire RCS contains subcooled liquid which transfers heat from the core by forced convection with some fully developed nucleate boiling. After the break develops, the time to departure from nucleate boiling is calculated, consistent with Appendix K of 10 CFR 50.[1] Thereafter the core heat transfer is unstable, with both nucleate boiling and film boiling occurring. As the core becomes uncovered, both turbulent and laminar forced convection and radiation are considered as core heat transfer mechanisms.

The heat transfer between the RCS and the secondary system may be in either direction, depending on the relative temperatures. In the case of continued heat addition to the secondary system, the secondary system pressure increases and the main steam safety valves may actuate to limit the pressure. Makeup water to the secondary side is automatically provided by the emergency feedwater system. The safety injection signal actuates a feedwater isolation signal, which isolates normal feedwater flow by closing the main feedwater isolation valves, and also initiates emergency feedwater flow by starting the emergency feedwater pumps. The secondary flow aids in the reduction of RCS pressure.

When the RCS depressurizes to 600 psia, the accumulators begin to inject borated water into the reactor coolant loops. The conservative assumption is made that accumulator water injected bypasses the core and goes out through the break until the termination of bypass. This conservatism is again consistent with Appendix K of 10CFR50. Since loss of offsite power (LOOP) is assumed, the RCPs are assumed to trip at the inception of the accident. The effects of pump coastdown are included in the blowdown analysis.

The blowdown phase of the transient ends when the RCS pressure (initially assumed at 2280 psia) falls to a value approaching that of the containment atmosphere. Prior to or at the end of the blowdown, the mechanisms that are responsible for the emergency core cooling water injected into the RCS bypassing the core are calculated not to be effective. At this time (called end-of-bypass) refill of the reactor vessel lower plenum begins. Refill is completed when emergency core cooling water has filled the lower plenum of the reactor vessel, which is bounded by the bottom of the fuel rods (called bottom of core recovery time).

The reflood phase of the transient is defined as the time period lasting from the end-of-refill until the reactor vessel has been filled with water to the extent that the core temperature rise has been terminated. From the latter stage of blowdown and then the beginning-of-reflood, the safety injection accumulator tanks rapidly discharge borated cooling water into the RCS, contributing to the filling of the reactor vessel downcomer. The downcomer water elevation head provides the driving force required for the reflooding of the reactor core. The low head and high head safety injection pumps aid in the filling of the downcomer and subsequently supply water to maintain a full downcomer and complete the reflooding process.

Continued operation of the ECCS pumps supplies water during longterm cooling. Core temperatures have been reduced to longterm steady state levels associated with dissipation of residual heat generation. After the water level of the residual water storage tank (RWST) reaches a minimum allowable value, coolant for long-term cooling of the core is obtained by switching to the cold recirculation phase of operation, in which spilled borated water is drawn from the engineered safety features (ESF) containment sumps by the low head safety injection (residual heat removal) pumps and returned to the RCS cold legs. The containment spray system continues to operate to further reduce containment pressure.

Approximately 11 hours after initiation of the LOCA, the ECCS is realigned to supply water to the RCS hot legs in order to control the boric acid concentration in the reactor vessel.

15.4.1.4 Core and System Performance

15.4.1.4.1 Mathematical Model

The requirements of an acceptable ECCS evaluation model are presented in Appendix K of 10 CFR 50 (Federal Register 1974).[1]

15.4.1.4.2 Large Break LOCA Evaluation Model

The analysis of a large break LOCA transient is divided into three phases: (1) blowdown, (2) refill, and (3) reflood. There are three distinct transients analyzed in each phase, including the thermal-hydraulic transient in the RCS, the pressure and temperature transient within the containment, and the fuel and clad temperature transient of the hottest fuel rod in the core. Based on these considerations, a system of interrelated computer codes has been developed for the analysis of the LOCA.

A description of the various aspects of the LOCA analysis methodology is given by Bordelon, Massie, and Zordon (1974).(6) This document describes the major phenomena modeled, the interfaces among the computer codes, and the features of the codes which ensure compliance with the Acceptance Criteria. The SATAN-VI, WREFLOOD, BASH, LOCBART, and COCO codes, which are used in the LOCA analysis, are described in detail by Bordelon et at. (1974)(5); Kelly et al. (1974)(9); Young et al. (1987)(4); Bordelon and Murphy (1974)(3). Code modifications are specified in References 2, 7 and 14. These codes assess the core heat transfer geometry and determine if the core remains amenable to cooling throughout and subsequent to the blowdown, refill, and reflood phases of the LOCA. The SATAN-VI computer code analyses the thermal-hydraulic transient in the RCS during blowdown and the WREFLOOD computer code calculates this transient during the refill phase of the accident. The BASH code is used to determine the system response during the reflood phase of the transient. The COCO code is used for the complete containment pressure history for dry containments. The LOCBART computer code calculates the thermal transient of the hottest fuel rod during the three phases. The Revised Pad Fuel Thermal Safety Model, described in Reference 14, generates the initial fuel rod conditions input to LOCBART.

SATAN-VI calculates the RCS pressure, enthalpy, density, and the mass and energy flow rates in the RCS, as well as steam generator energy transfer between the primary and secondary systems as a function of time during the blowdown phase of the LOCA. SATAN-VI also calculates the accumulator water mass and internal pressure and the pipe break mass and energy flow rates that are assumed to be vented to the containment during blowdown. At the end of the blowdown, information on the state of the system is transferred to the WREFLOOD code which performs the calculation of the refill period to bottom of core (BOC) recovery time. Once the vessel has refilled to the bottom of the core, the reflood portion of the transient begins. The BASH code is used to calculate the thermal-hydraulic simulation of the RCS for the reflood phase.

Information concerning the core boundary conditions is taken from all of the above codes and input to the LOCBART code for the purpose of calculating the core fuel rod thermal response for the entire transient. From the boundary conditions, LOCBART computes the fluid conditions and heat transfer coefficient for the full length of the fuel rod by employing mechanistic models appropriate to the actual flow and heat transfer regimes. Conservative assumptions ensure that the fuel rods modeled in the calculation represent the hottest rods in the entire core.

The containment pressure analysis is performed with the COCO code [3], which is interactive with the WREFLOOD code. The transient pressure computed by the COCO code is then input to the BASH code for the purpose of supplying a backpressure at the break plane while computing the reflood transient.

The large break analysis was performed with the December 1981 version of the Evaluation Model modified to incorporate the BASH(4) computer code.

15.4.1.4.3 Input Parameters and Initial Conditions

The analysis presented in this section was performed with a reactor vessel upper head temperature equal to the RCS cold leg temperature.

The bases used to select the numerical values that are input parameters to the analysis have been conservatively determined from extensive sensitivity studies (Westinghouse 1974(12); Salvatori 1974(11)). In addition, the requirements of Appendix K regarding specific model features were met by selecting models which provide a significant overall conservatism in the analysis. The assumptions which were made pertain to the conditions of the reactor and associated safety system equipment at the time that the LOCA occurs, and include such items as the core peaking factors, the containment pressure, and the performance of the ECCS. Decay heat generated throughout the transient is also conservatively calculated.

15.4.1.4.4 Results

Based on the results of the LOCA sensitivity studies (Westinghouse 1974[12]; Salvatori 1974[11]) the limiting large break was found to be the double ended cold leg guillotine (DECLG). Therefore, only the DECLG break is considered in the large break ECCS performance analysis. Calculations were performed for a range of Moody break discharge coefficients. The results of these calculations are summarized in Tables 15.4.1-1 and 15.4.1-2. The hot spot is defined to be the location of the maximum peak clad temperature. This location is given in Table 15.4.1-2 for each break size analyzed.

Containment data used to calculated ECCS back pressure is presented in Table 15.4.1-3.

Figures 15.4-1 through 15.4-67 show transient plots of important parameters from the ECCS Evaluation Model calculations. Plots are grouped by break size as follows:

Figures 15.4-1 through 15.4-16	C_d=0.4 MIN SI
Figures 15.4-17 through 15.4-32	C_d=0.6 MIN SI
Figures 15.4-33 through 15.4-48	C_d=0.8 MIN SI
Figures 15.4-49 through 15.4-64	C_d=0.4 MAX SI

For each break size, a series of plots is presented, showing the transients of the following parameters.

I. For the blowdown portion of the transient:

A. RCS Pressure
B. Core inlet and outlet flow rates
C. Cold leg accumulator delivery rate
D. Core pressure drop

E. Break mass flow rate
F. Break energy discharge rate
G. Normalized core power

II. For the reflood portion of the transient

A. Core and Downcomer liquid levels
B. The core inlet fluid velocity, as input to the rod thermal analysis code
C. The accumulator flow rates
D. Pumped safety injection flow rates

III. From the fuel rod thermal analysis, at the peak temperature location:

A. Fluid mass flux
B. Rod heat transfer coefficient
C. Clad peak temperature transient
D. Temperature transient at the hot rod burst elevation
E. Fluid tempeature

For the most limiting break size, the containment pressure transient and the containment wall condensing heat transfer coefficient are presented in Figures 15.4-65 and 66, and the safety injection flow rate is presented in Figures 15.4-67.

In addition to the above, Tables 15.4.1-4 and 15.4.1-5 present the reflood mass and energy release to the containment and the broken loop accumulator mass and energy flowrate to the containment, respectively.

The maximum clad temperature calculated for a large break is 2141°F which is less than the Acceptance Criteria limit of 2200°F. The maximum local metal-water reaction is 10.128 percent, which is well below the embrittlement limit of 17 percent as required by 10 CFR 50.46. The total core metal-water reaction is less than 0.3 percent for all breaks, as compared with the 1 percent criterion of 10 CFR 50.46. The clad temperature transient is terminated at a time when the core geometry is still amenable to cooling. As a result, the core temperature will continue to drop and the ability to remove decay heat generated in the fuel for an extended period of time will be provided.

REFERENCES FOR SECTION 15.4.1

1. "Acceptance Criteria for Emergency Core Cooling System for Light Water Cooled Nuclear Power Reactors," 10 CFR 50.46 and Appendix K of 10 CFR 50, Federal Register 1974, Volume 39, Number 3.

2. Rahe, E. P. (Westinghouse), letter to J. R. Miller (USNRC), Letter No. NS-EPRS-2679, November 1982.

3. Bordelon, F. M., and Murphy, E. T., "Containment Pressure Analysis Code (COCO)," WCAP-8327 (Proprietary), WCAP-8326 (Non-Proprietary), June, 1974.

4. Young, M. Y. et al, "The 1981 Version of the Westinghouse ECCS Evaluation Model Using the BASH Code," WCAP-10266-P-A Rev. 2 (Proprietary), 1987.

5. Bordelon, F. M. et al., "SATAN-VI Program: Comprehensive Space, Time Dependent Analysis of Loss-of-Coolant," WCAP-8302 (Proprietary) and WCAP-8306 (Non-Proprietary), 1974.

6. Bordelon, F. M.; Massie, H. W.; and Zordon, T. A., "Westinghouse ECCS Evaluation Model - Summary," WCAP-8339, 1974.

7. Rahe, E. P., "Westinghouse ECCS Evaluation Model, 1981 Version," WCAP-9220-P-A (Proprietary Version), WCAP-9221-P-A (Non-Proprietary Version), Revision 1, 1981.

9. Kelly, R. D. et al., "Calculational Model for Core Reflooding After a Loss-of-Coolant Accident (WREFLOOD Code," WCAP-8170 (Proprietary) and WCAP-8171 (Non-Proprietary), 1974.

10. U. S. Nuclear Regulatory Commission 1975, "Reactor Safety Study - An Assessment of Accident Risks in U. S. Commercial Nuclear Power Plant," WASH-1400, NUREG-75/014.

11. Salvatori, R., "Westinghouse ECCS - Plant Sensitivity Studies," WCAP-8340 (Proprietary) and WCAP-8356 (Non-Proprietary), 1974.

12. "Westinghouse ECCS Evaluation Model Sensitivity Studies," WCAP-8341 (Proprietary) and WCAP-8342 (Non-Proprietary), 1974.

13. "Bordelon, F. M., et al., "Westinghouse ECCS Evaluation Model - Supplementary Information," WCAP-8471 (Proprietary) and WCAP-8472 (Non-Proprietary), 1975.

14. Letter from J. F. Stoltz (NRC) to T. M. Anderson (Westinghouse), Subject: Review of WCAP-8720, Improved Analytical Models used in Westinghouse Fuel Rod Design Computations.

TABLE 15.4.1-1

LARGE BREAK

TIME SEQUENCE OF EVENTS

	DECLG (C_D=0.8) (Sec)	DECLG (C_D=0.6) (Sec)	DECLG (C_D=0.4) (Sec)
START	0.0	0.0	0.0
Reactor Trip Signal	0.433	0.440	0.452
S. I. Signal	0.62	0.70	0.85
Acc. Injection	8.96	11.25	15.10
End of Blowdown/Bypass	19.95	23.09	30.41
Bottom of Core Recovery	32.31	35.41	43.13
Pump Injection	32.62	32.70	32.85
Acc. Empty	42.269	45.234	50.527

TABLE 15.4.1-2

LARGE BREAK

	DECLG (C_D=0.8)	DECLG (C_D=0.6)	DECLG (C_D=0.4)
Results			
Peak Clad Temp. °F	1767.	1877.	2141.
Peak Clad Location Ft.	7.0	8.0	7.0
Local Zr/H_2O Reaction (max) %	2.08	2.63	10.13
Local Zr/H_2O Location Ft.	6.25	8.0	6.0
Total Zr/H_2O Reaction %	<0.3	<0.3	<0.3
Hot Rod Burst Time sec	43.37	41.46	40.2
Hot Rod Burst Location Ft.	6.25	6.0	6.0

Calculation	
NSSS Power Mwt 102% of	2775
Peak Linear Power kw/ft 102% of	13.340
Peaking Factor (At License Rating)	2.45
Accumulator Water Volume (ft^3) (minimum plus line volume)	1039

Fuel region + cycle analyzed	Cycle	Region
UNIT 1	5 and Beyond	ALL

15% Steam Generator Tube Plugging in each steam generator is assumed.

TABLE 15.4.1-3

LARGE BREAK
CONTAINMENT DATA
(DRY CONTAINMENT)

NET FREE VOLUME	1.9×10^6 ft^3
INITIAL CONDITIONS	
Pressure	14.7 psia
Temperature	90°F
RWST Temperature	40°F
Service Water Temperature	NA
Outside Temperature	19°F
SPRAY SYSTEM	
Total Flow Rate	6000 gpm
Actuation Time	52 secs
SAFEGUARDS FAN COOLERS	
Number of Fan Coolers Operating	2
Fastests Post Accicent Initiation of Fan Coolers	43 secs

TABLE 15.4.1-3 (Continued)

LARGE BREAK
CONTAINMENT DATA
(DRY CONTAINMENT)

STRUCTURAL HEAT SINKS

Thickness (In)		Area (Ft^2)
0.348 Carbon Steel	48.0 Concrete	57,397
0.264 Carbon Steel	36.0 Concrete	20,241
0.125 Carbon Steel	24.0 Concrete	11,694
18.0 Concrete		315
22.56 Concrete		43,537
12.0 Concrete		10,811
48.0 Concrete		19.020
1.52 Stainless Steel		409
1.13 Stainless Steel		551
0.6 Stainless Steel		1939
0.336 Stainless Steel		2194
0.06 Stainless Steel		88481
6.672 Carbon Steel		3300
3.504 Carbon Steel		130
2.376 Carbon Steel		2324
1.7568 Carbon Steel		4323
0.87 Carbon Steel		8787
0.744 Carbon Steel		17734
0.324 Carbon Steel		16929
0.06 Carbon Steel		654508

TABLE 15.4.1-4

REFLOOD MASS/ENERGY RELEASES* (C_D = 0.4)

TIME (SEC)	TOTAL MASS FLOWRATE (LBM/SEC)	TOTAL ENERGY FLOWRATE 10^5 BTU/SEC
43.00	0.0	0.0
49.00	38.65	0.503
59.00	87.75	1.093
79.0	179.26	1.392
109.0	265.1	1.537
129.0	271.31	1.491
169.0	306.53	1.468

*Accumulator nitrogen was released between 50.0 and 70.0 seconds at a mass flow rate of 198.67 lbm/sec.

TABLE 15.4.1-5

BROKEN LOOP ACCUMULATOR FLOWRATE TO CONTAINMENT FOR LIMITING CASE - DECLG (C_D = 0.4)

TIME (SEC)	MASS FLOWRATE* (LBM/SEC)
0.0	4763.3
3.01	3596.8
4.01	3369.0
5.01	3179.8
8.01	2752.3
11.01	2449.3
16.01	2093.6
21.01	1847.4
24.01	1734.0
25.55	0.0

*Enthalpy of accumulator water is 58 BTU/LBM

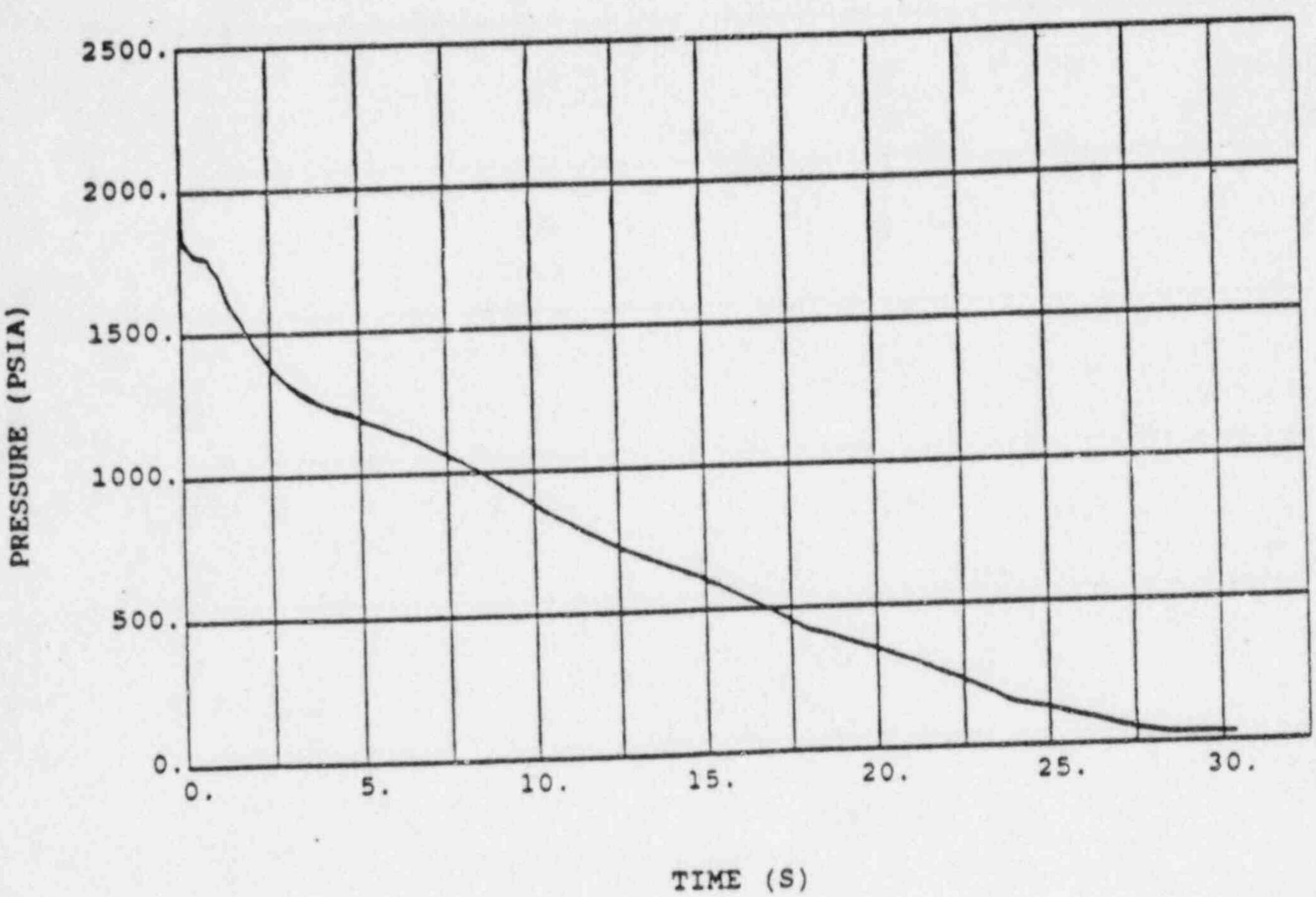

Figure 15.4-1
REACTOR COOLANT SYSTEM PRESSURE -
DECLG (C_D = 0.4, MIN SI)
Virgil C. Summer Nuclear Station
Final Safety Analysis Report

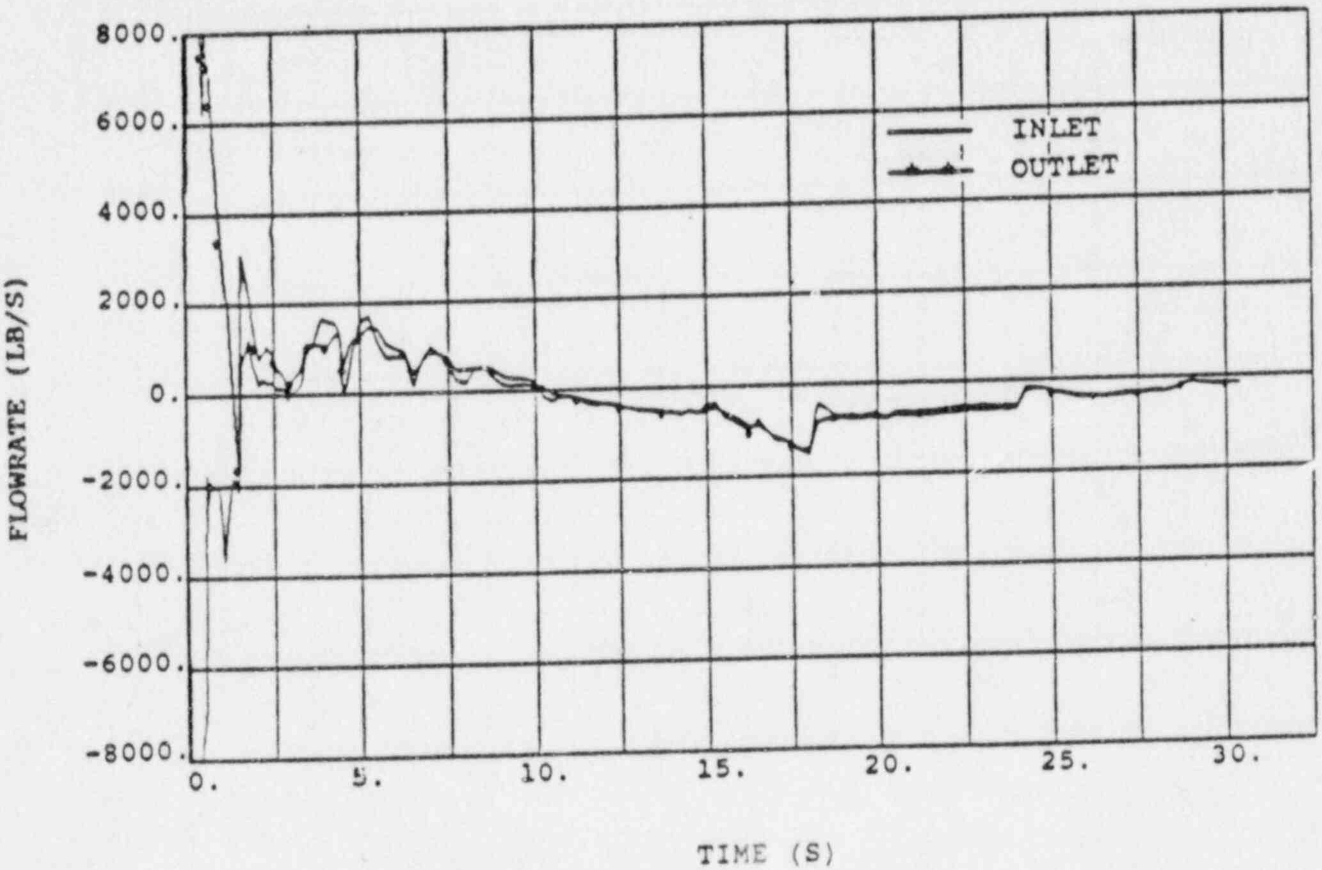

Figure 15.4-2
CORE FLOWRATE -
DECLG (C_D = 0.4, MIN SI)
Virgil C. Summer Nuclear Station
Final Safety Analysis Report

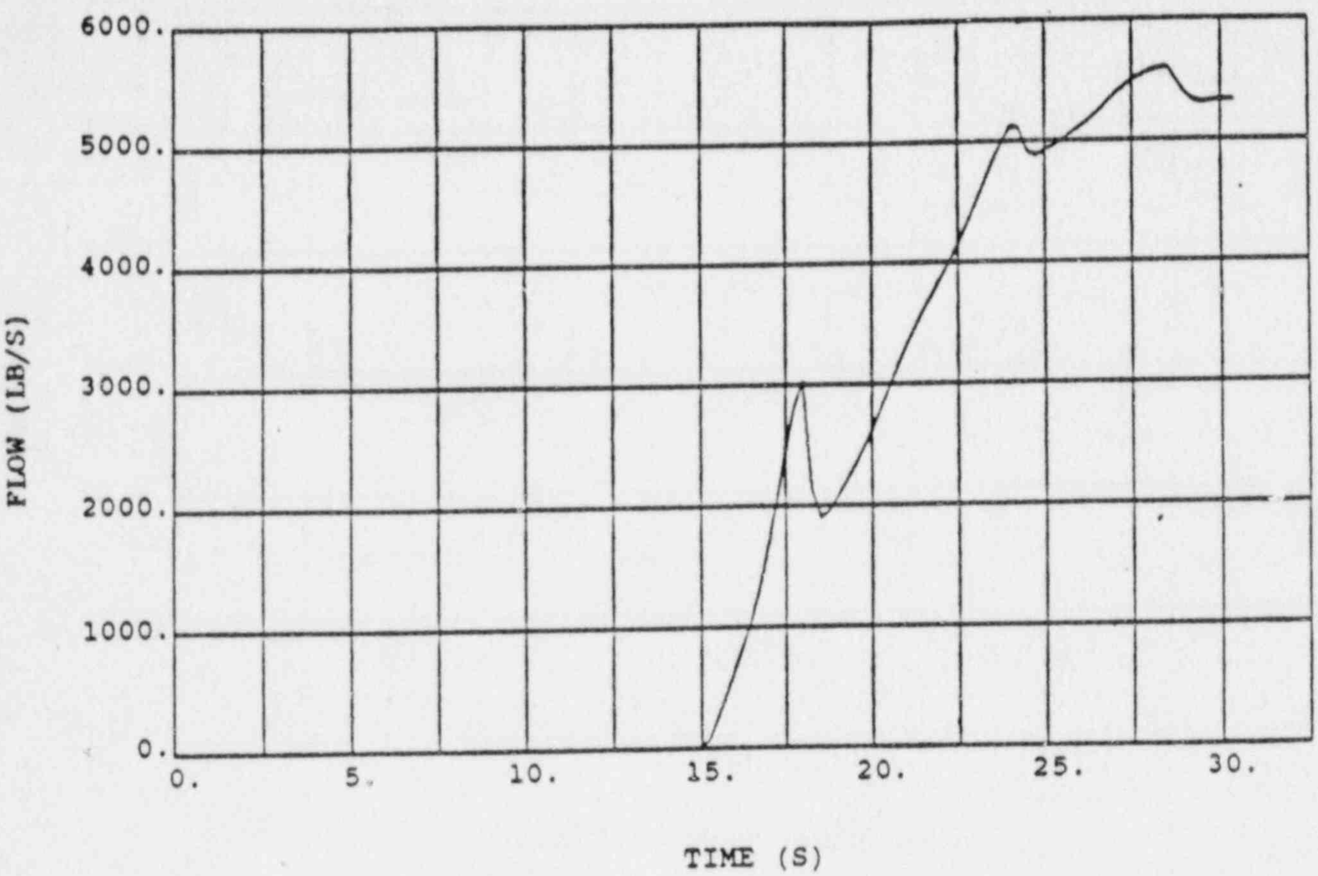

Figure 15.4-3
ACCUMULATOR FLOW DURING BLOWDOWN -
DECLG (C_D = 0.4, MIN SI)
Virgil C. Summer Nuclear Station
Final Safety Analysis Report

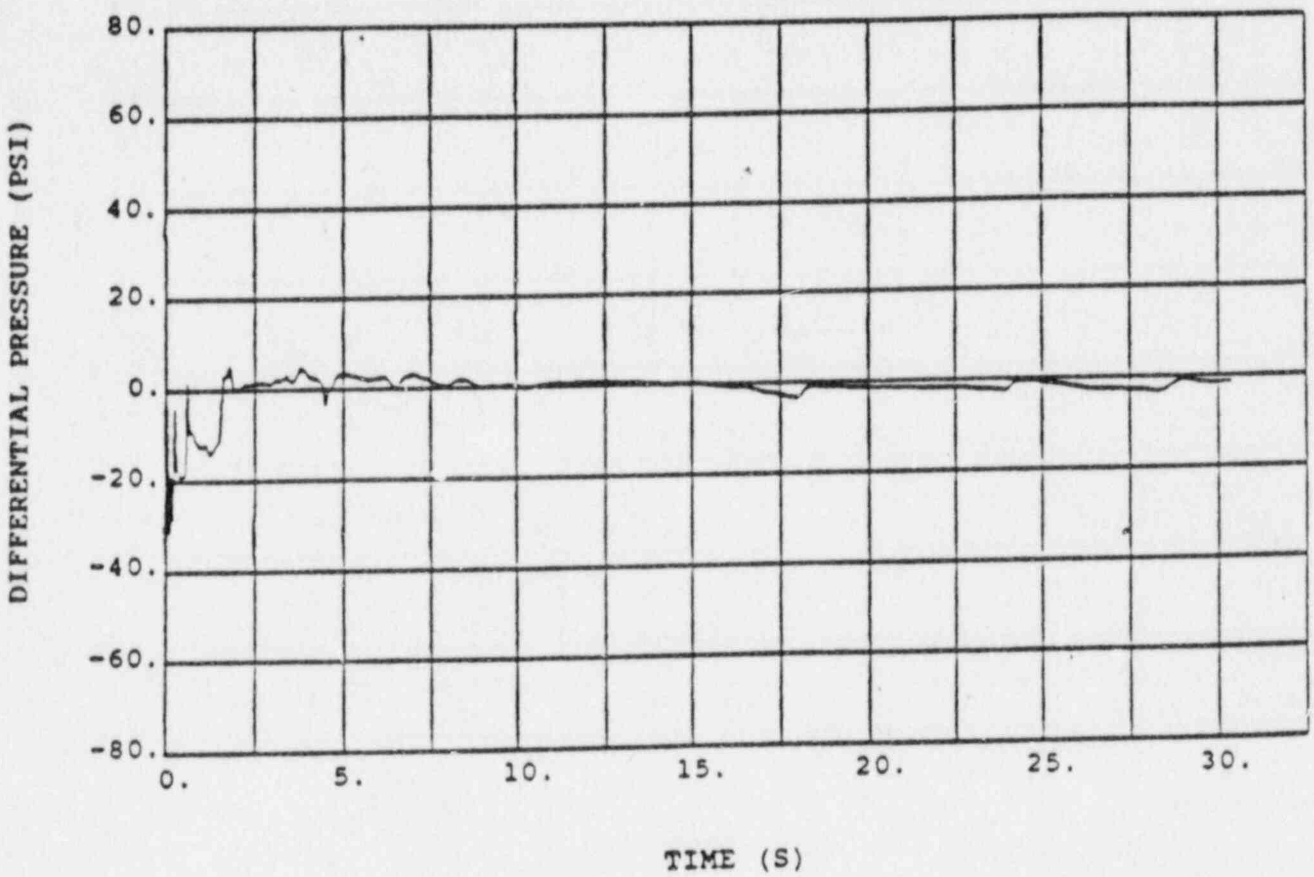

Figure 15.4-4
CORE PRESSURE DROP -
DECLG (C_D = 0.4, MIN SI)
Virgil C. Summer Nuclear Station
Final Safety Analysis Report

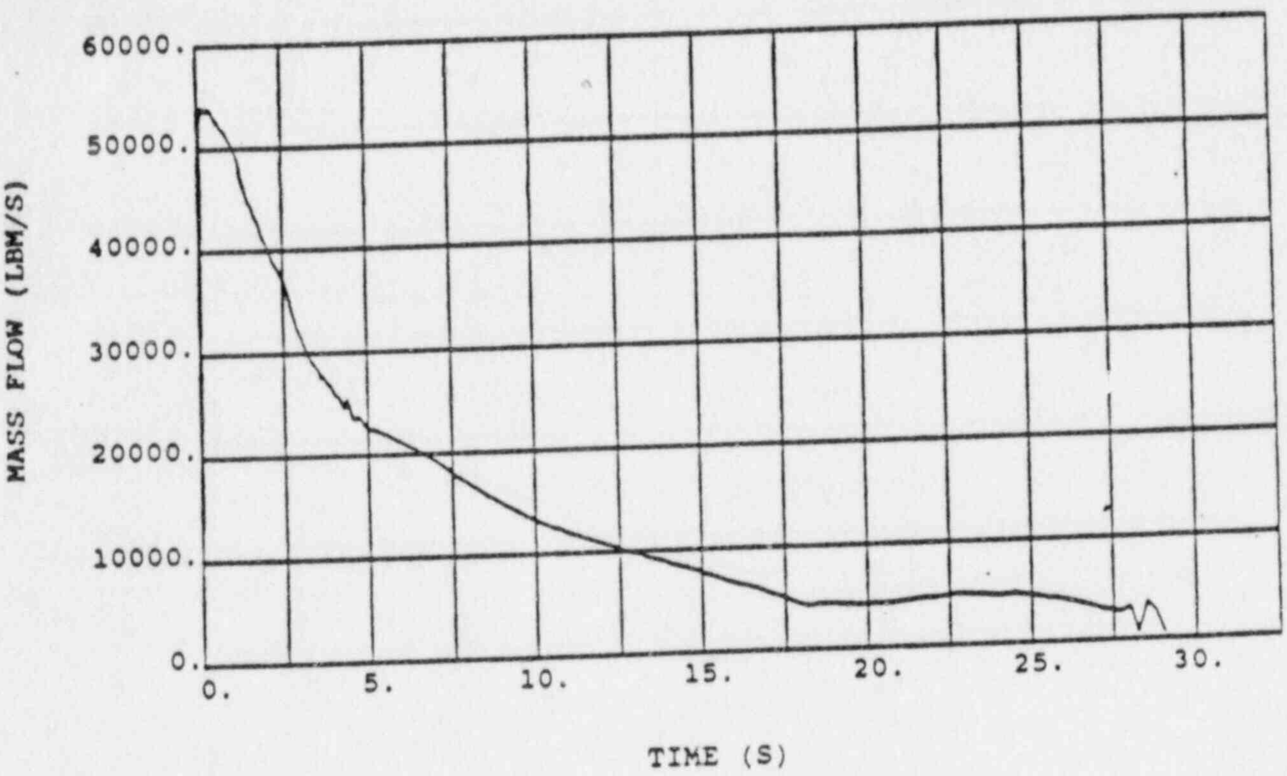

Figure 15.4-5
BREAK FLOW DURING BLOWDOWN -
DECLG (C_D = 0.4, MIN SI)
Virgil C. Summer Nuclear Station
Final Safety Analysis Report

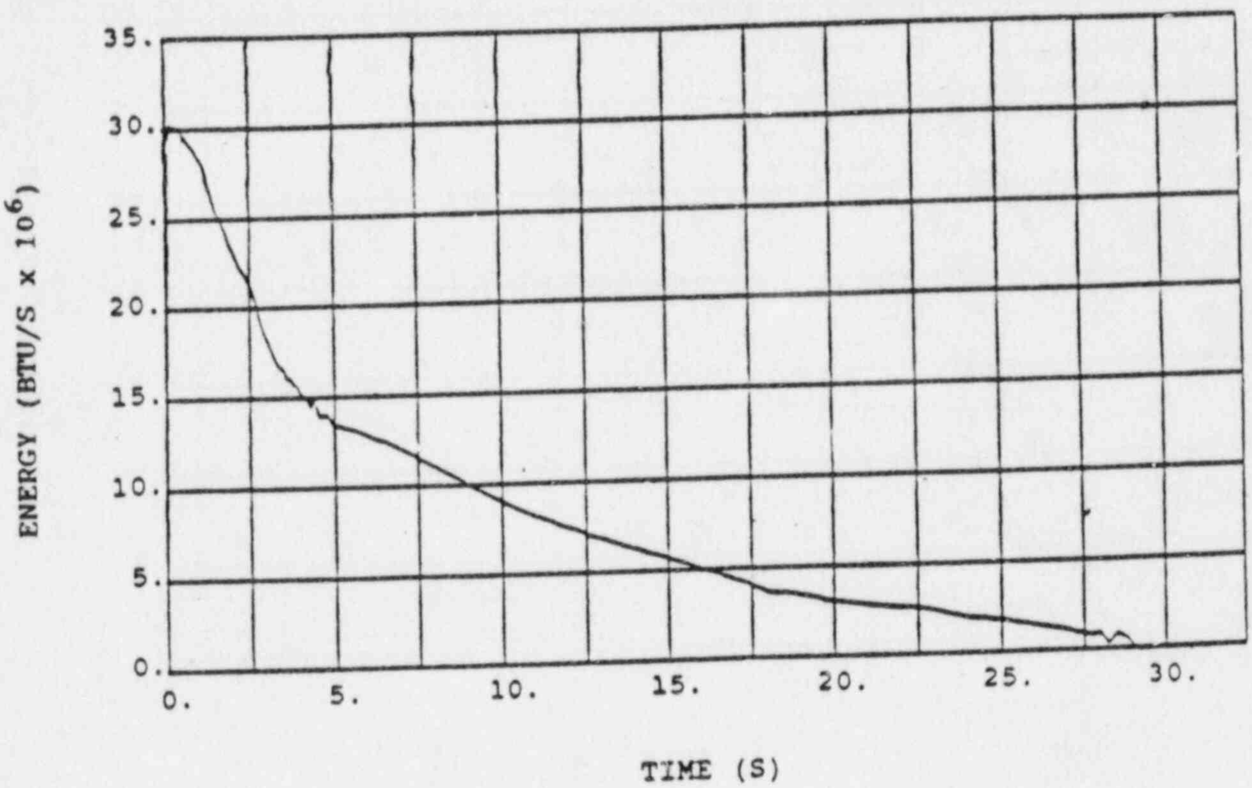

Figure 15.4-6
BREAK ENERGY DURING BLOWDOWN - DECLG (C_D = 0.4, MIN SI)
Virgil C. Summer Nuclear Station
Final Safety Analysis Report

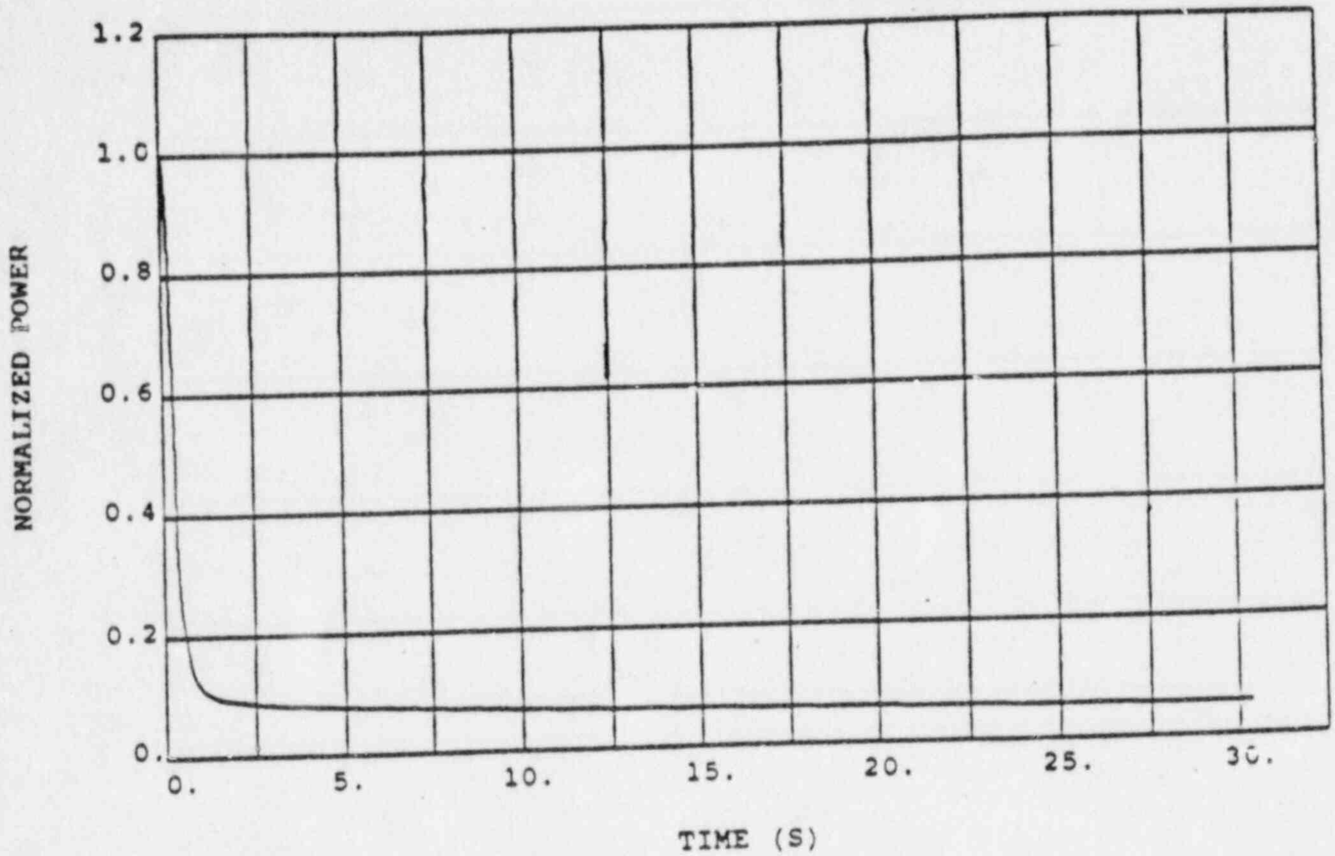

Figure 15.4-7
CORE POWER -
DECLG (C_D = 0.4, MIN SI)
Virgil C. Summer Nuclear Station
Final Safety Analysis Report

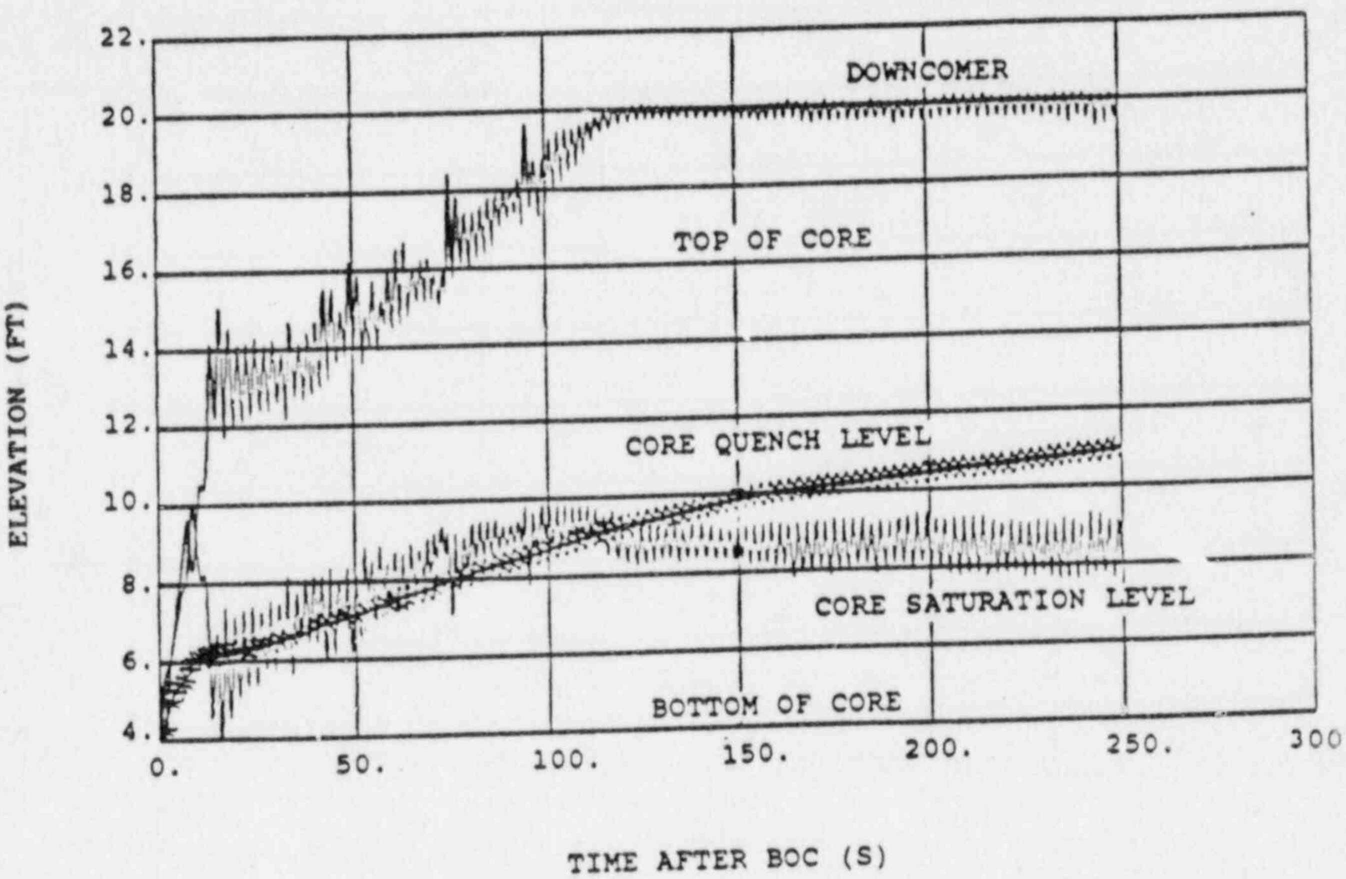

Figure 15.4-8
CORE AND DOWNCOMER LIQUID LEVELS
DURING REFLOOD
DECLG (C_D = 0.4, MIN SI)
Virgil C. Summer Nuclear Station
Final Safety Analysis Report

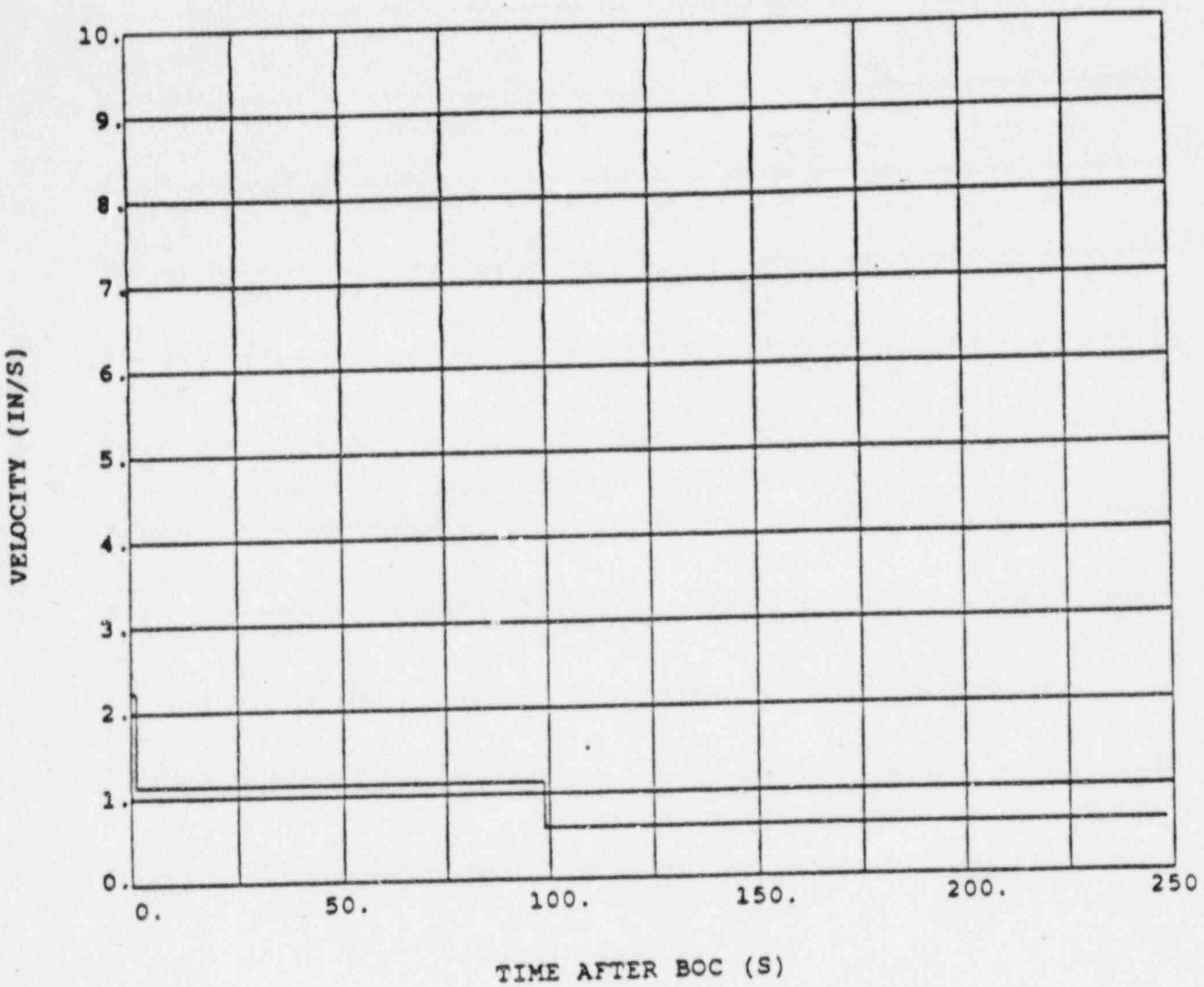

Figure 15.4-9
CORE INLET FLUID VELOCITY FOR ROD THERMAL ANALYSIS
DECLG (C_D = 0.4, MIN SI)
Virgil C. Summer Nuclear Station
Final Safety Analysis Report

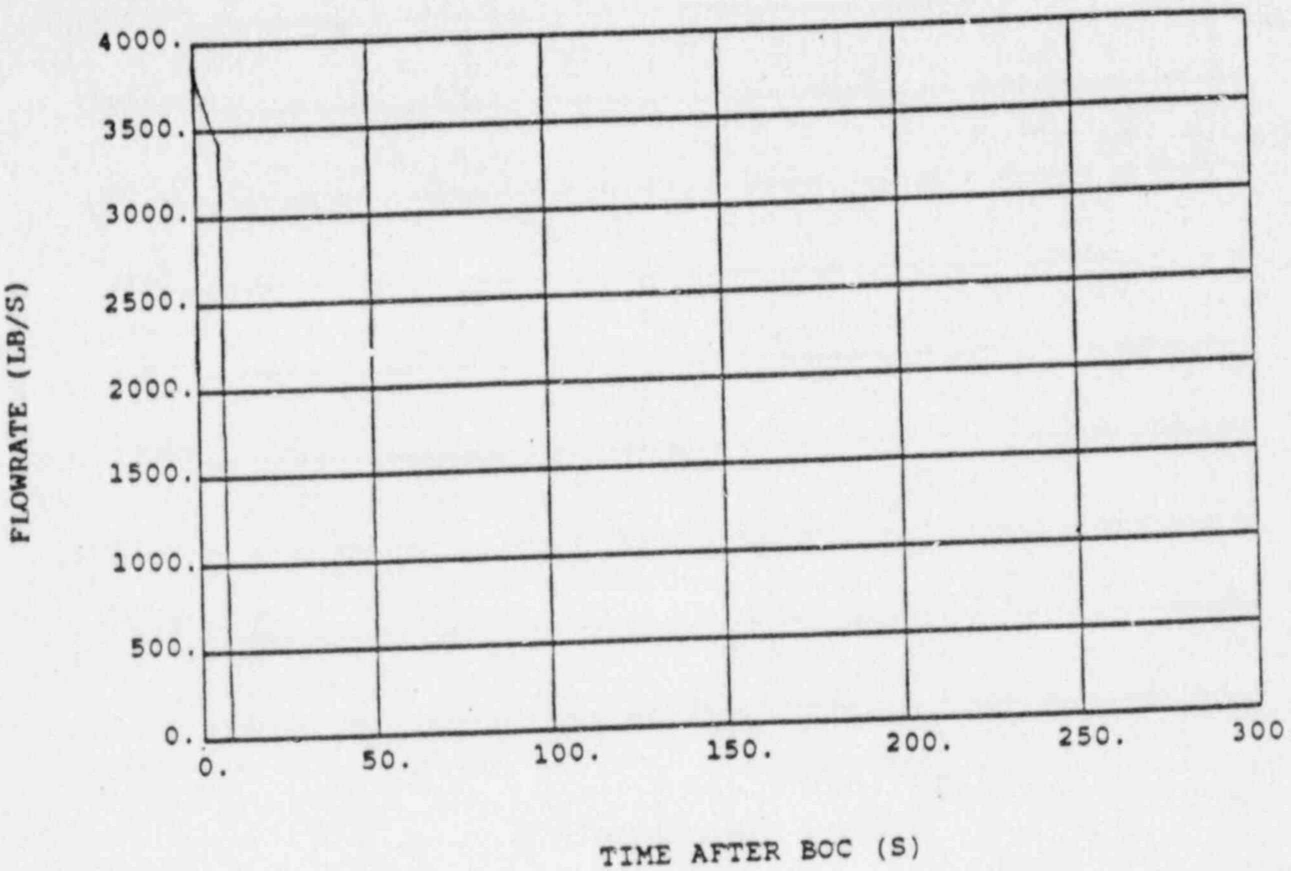

Figure 15.4-10
ACCUMULATOR FLOW DURING REFLOOD
DECLG (C_D = 0.4, MIN SI)
Virgil C. Summer Nuclear Station
Final Safety Analysis Report

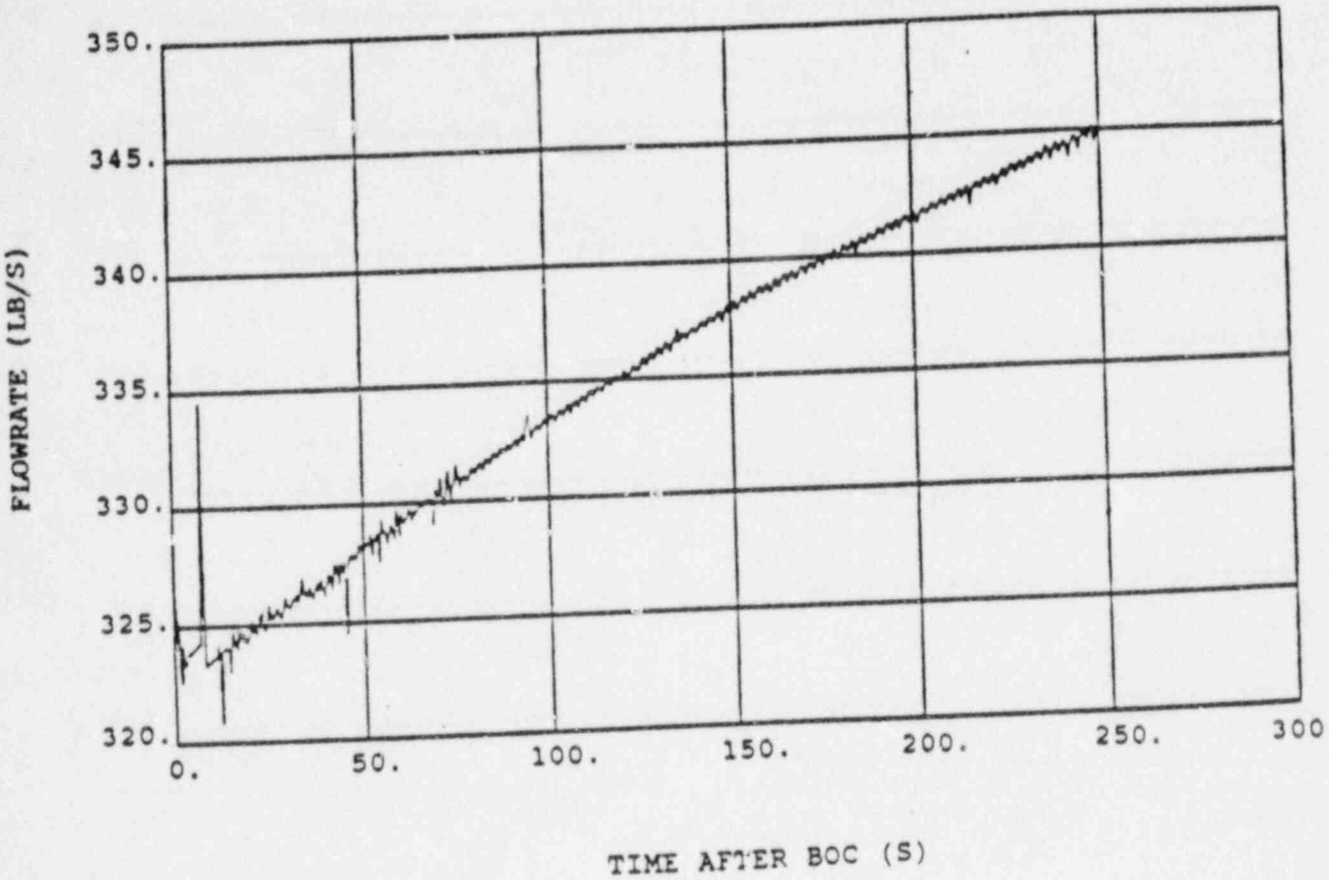

Figure 15.4-11
SI FLOW DURING REFLOOD
DECLG (C_D = 0.4, MIN SI)
Virgil C. Summer Nuclear Station

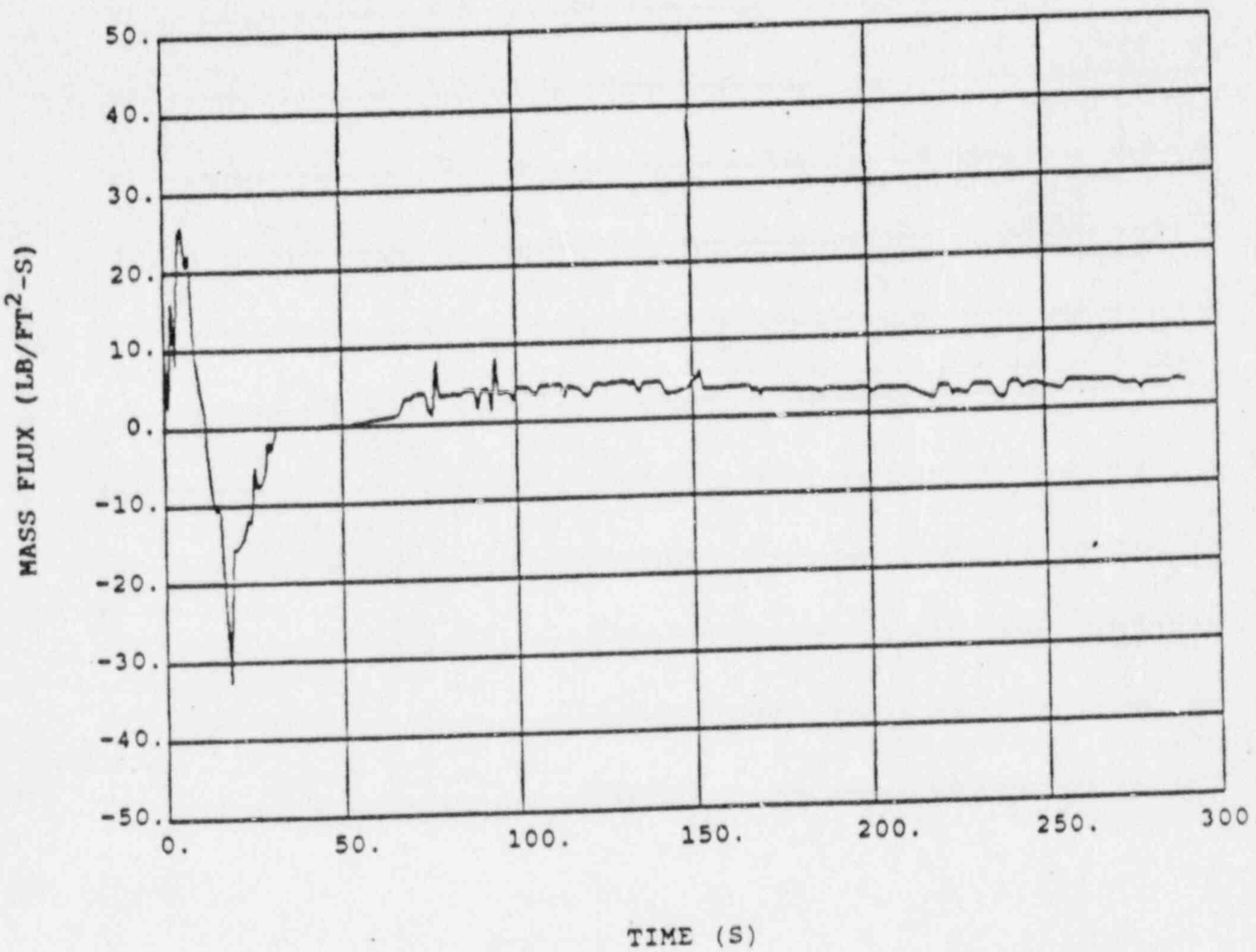

Figure 15.4-12
MASS FLUX AT THE PEAK ROD TEMPERATURE ELEVATION
DECLG (C_D = 0.4, MIN SI)
Virgil C. Summer Nuclear Station
Final Safety Analysis Report

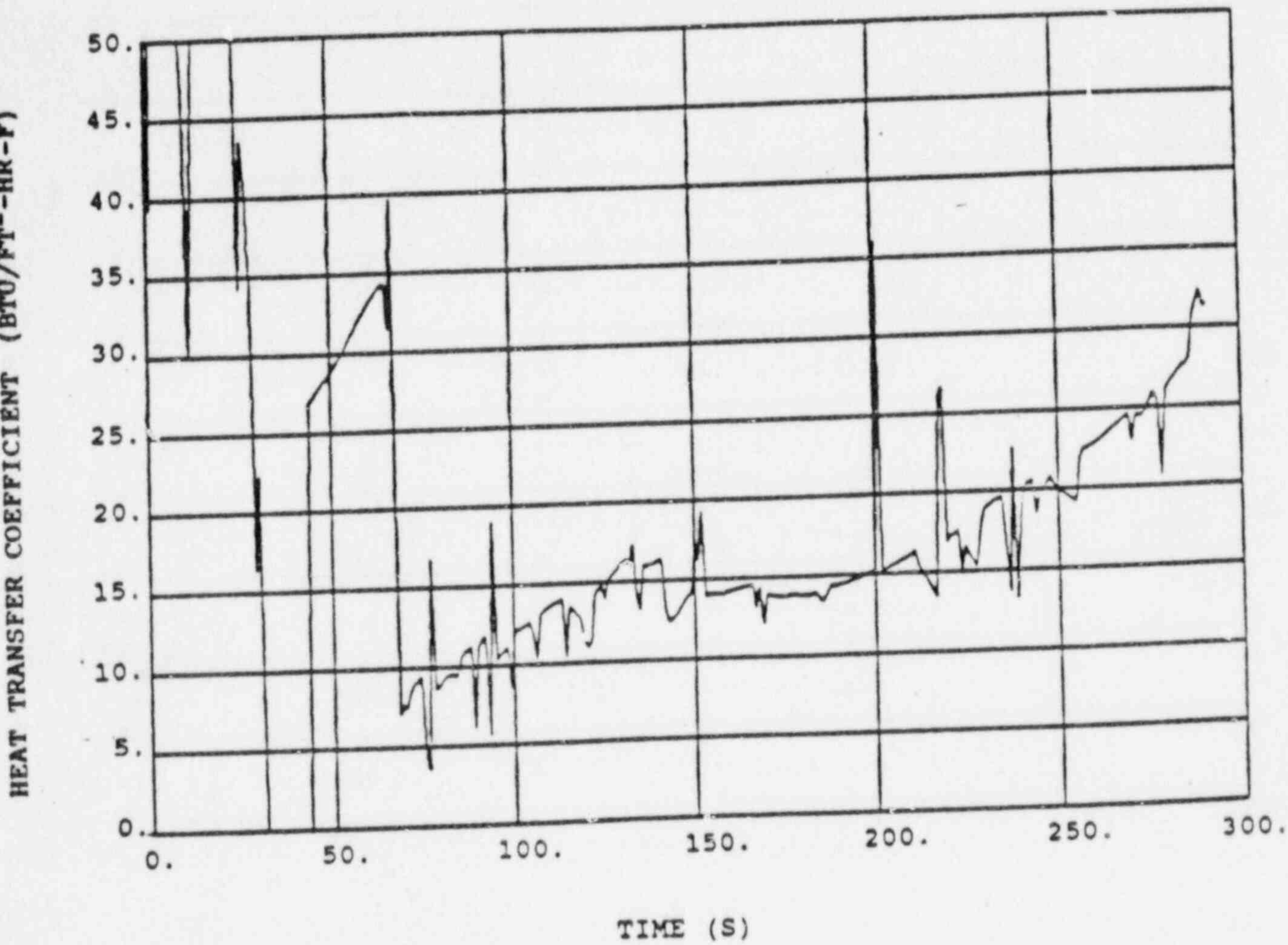

Figure 15.4-13
ROD HEAT TRANSFER COEFFICIENT AT
THE PEAK TEMPERATURE LOCATION
DECLG (C_D = 0.4, MIN SI)
Virgil C. Summer Nuclear Station
Final Safety Analysis Report

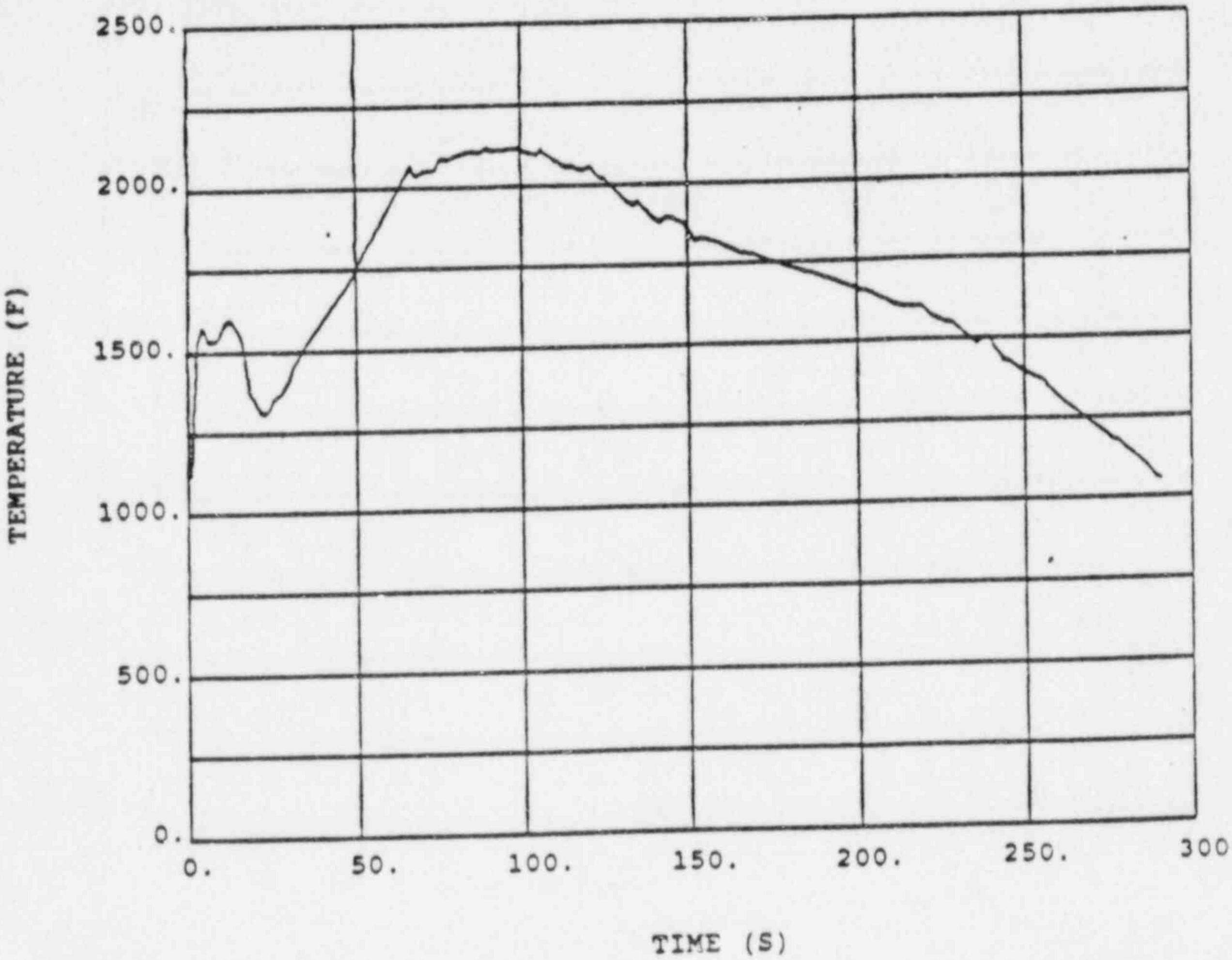

Figure 15.4-14
FUEL ROD PEAK CLAD TEMPERATURE
DECLG (C_D = 0.4, MIN SI)
Virgil C. Summer Nuclear Station
Final Safety Analysis Report

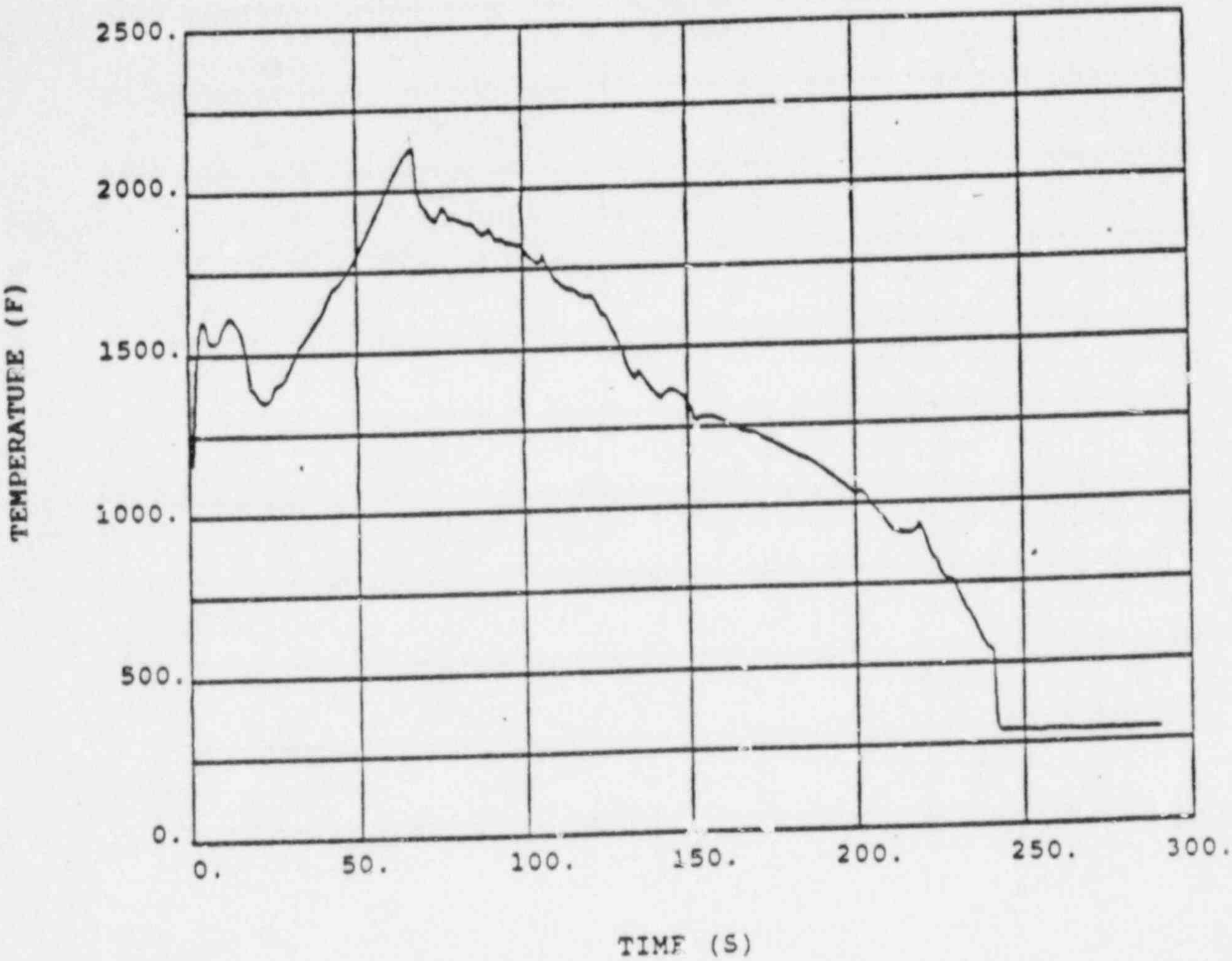

Figure 15.4-15
CLAD TEMPERATURE AT THE BURST NODE
DECLG (C_D = 0.4, MIN SI)
Virgil C. Summer Nuclear Station
Final Safety Analysis Report

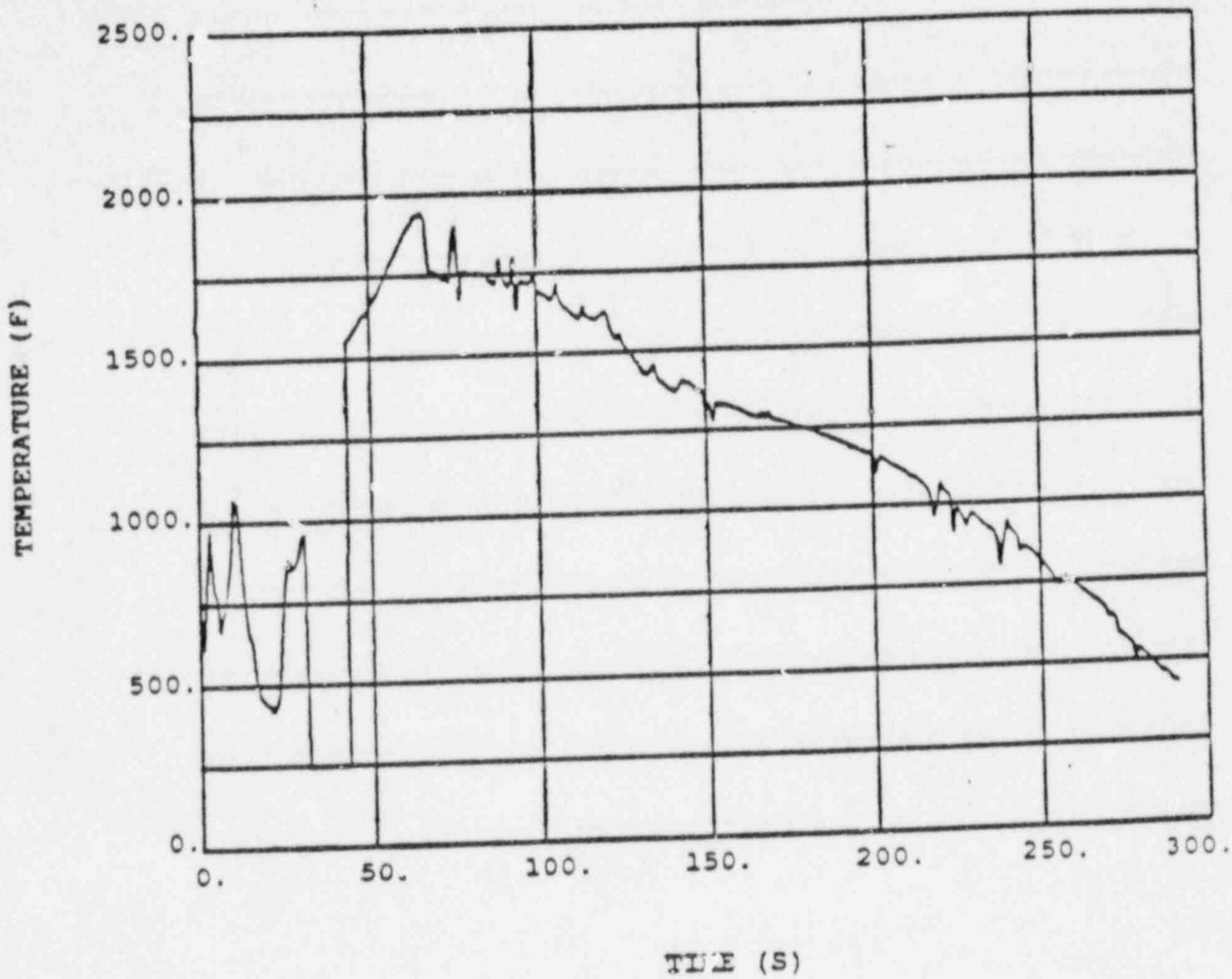

Figure 15.4-16
FLUID TEMPERATURE
DECLG (C_D = 0.4, MIN SI)
Virgil C. Summer Nuclear Station
Final Safety Analysis Report

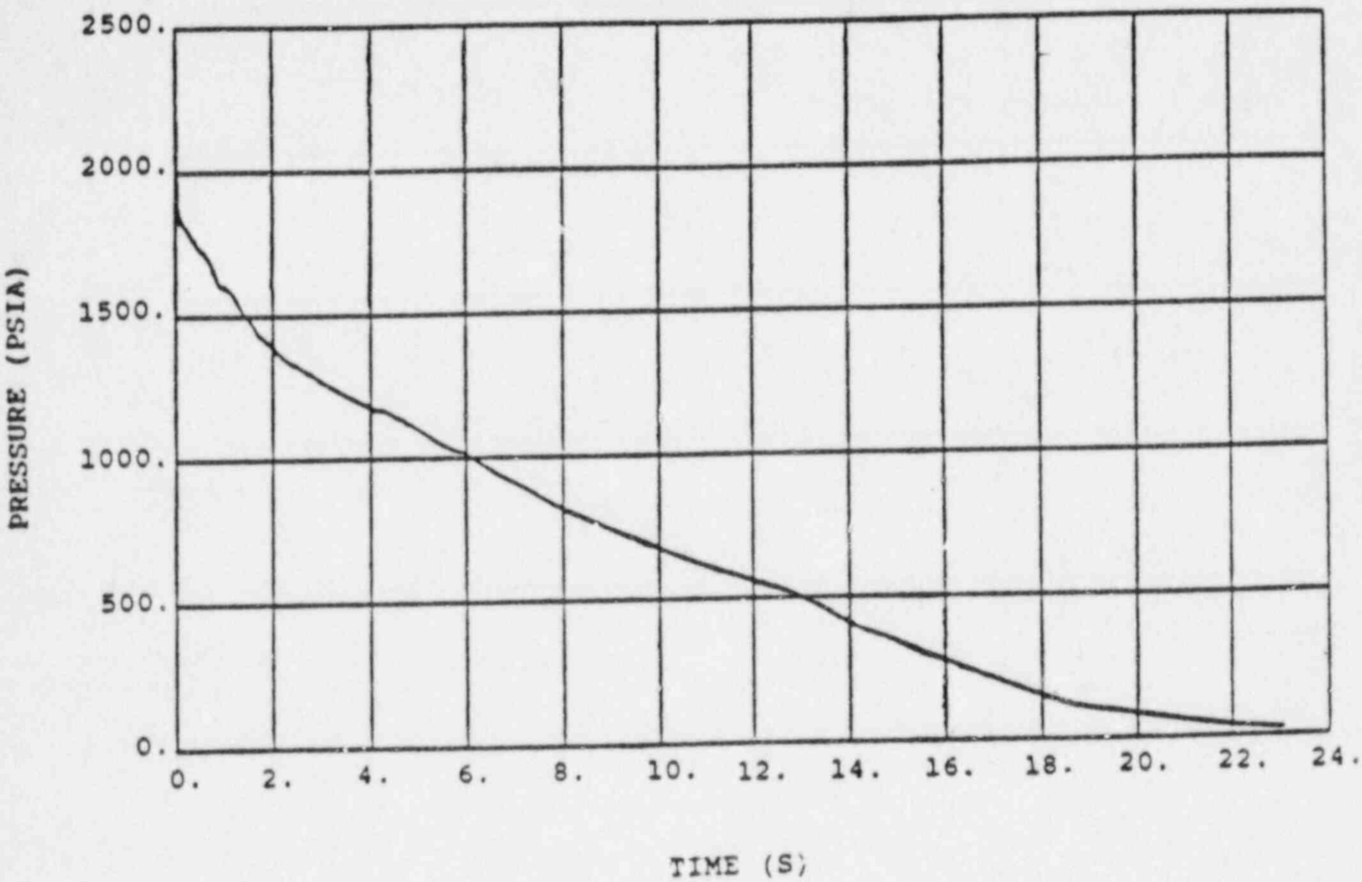

Figure 15.4-17
REACTOR COOLANT SYSTEM PRESSURE - DECLG (C_D = 0.6, MIN SI)
Virgil C. Summer Nuclear Station
Final Safety Analysis Report

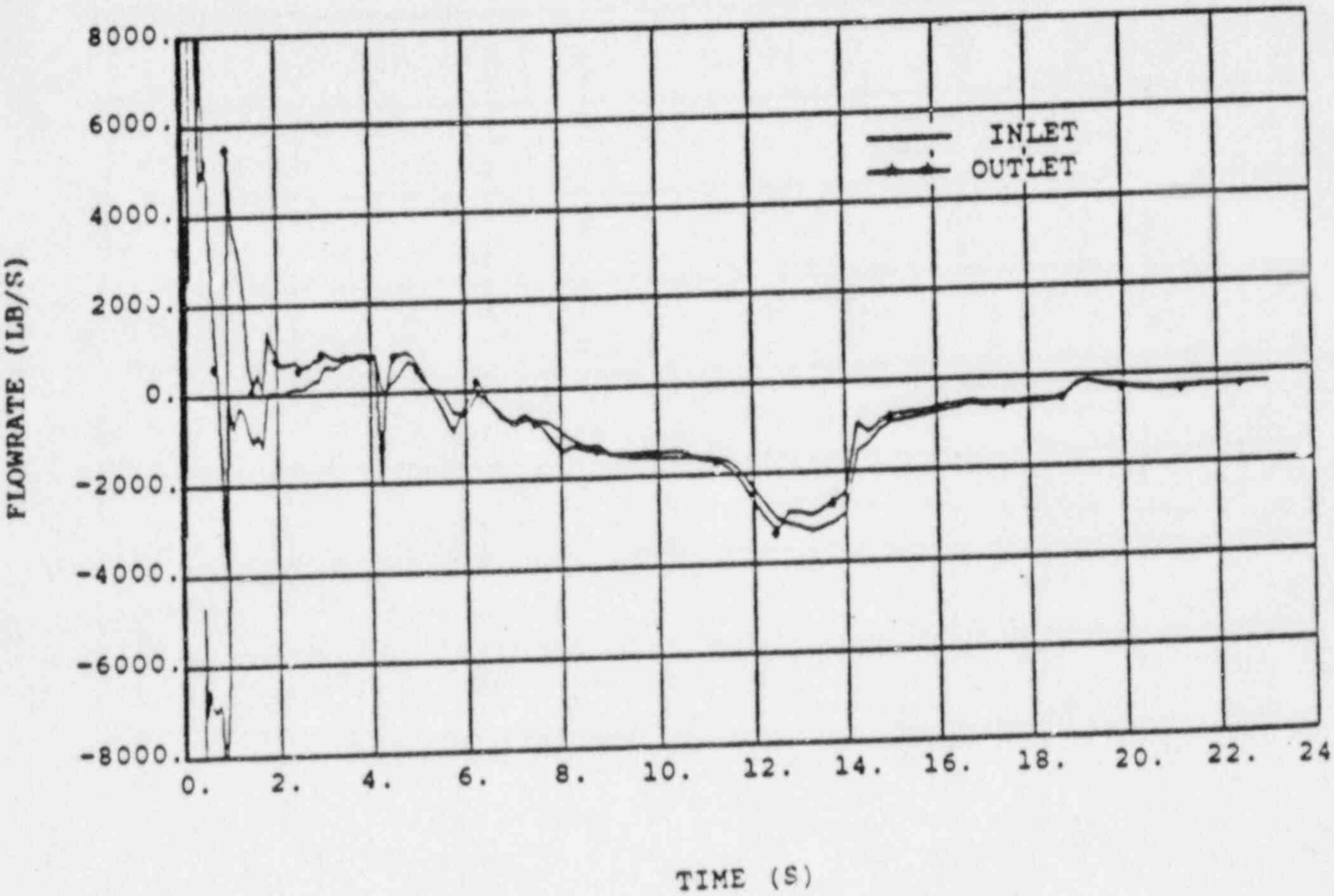

Figure 15.4-18
CORE FLOWRATE -
DECLG (C_D = 0.6, MIN SI)
Virgil C. Summer Nuclear Station
Final Safety Analysis Report

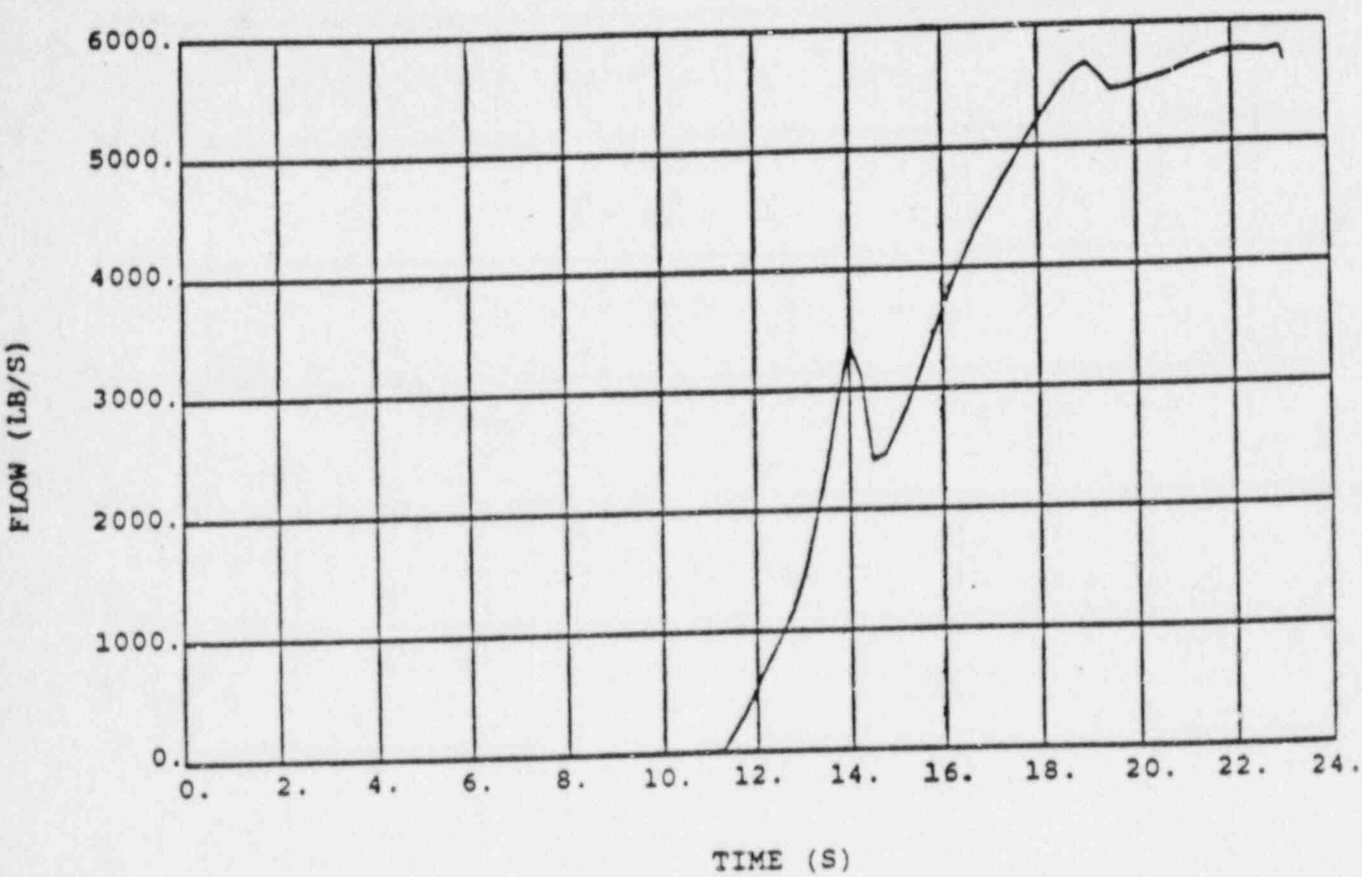

Figure 15.4-19
ACCUMULATOR FLOW DURING BLOWDOWN - DECLG (C_D = 0.6, MIN SI)
Virgil C. Summer Nuclear Station
Final Safety Analysis Report

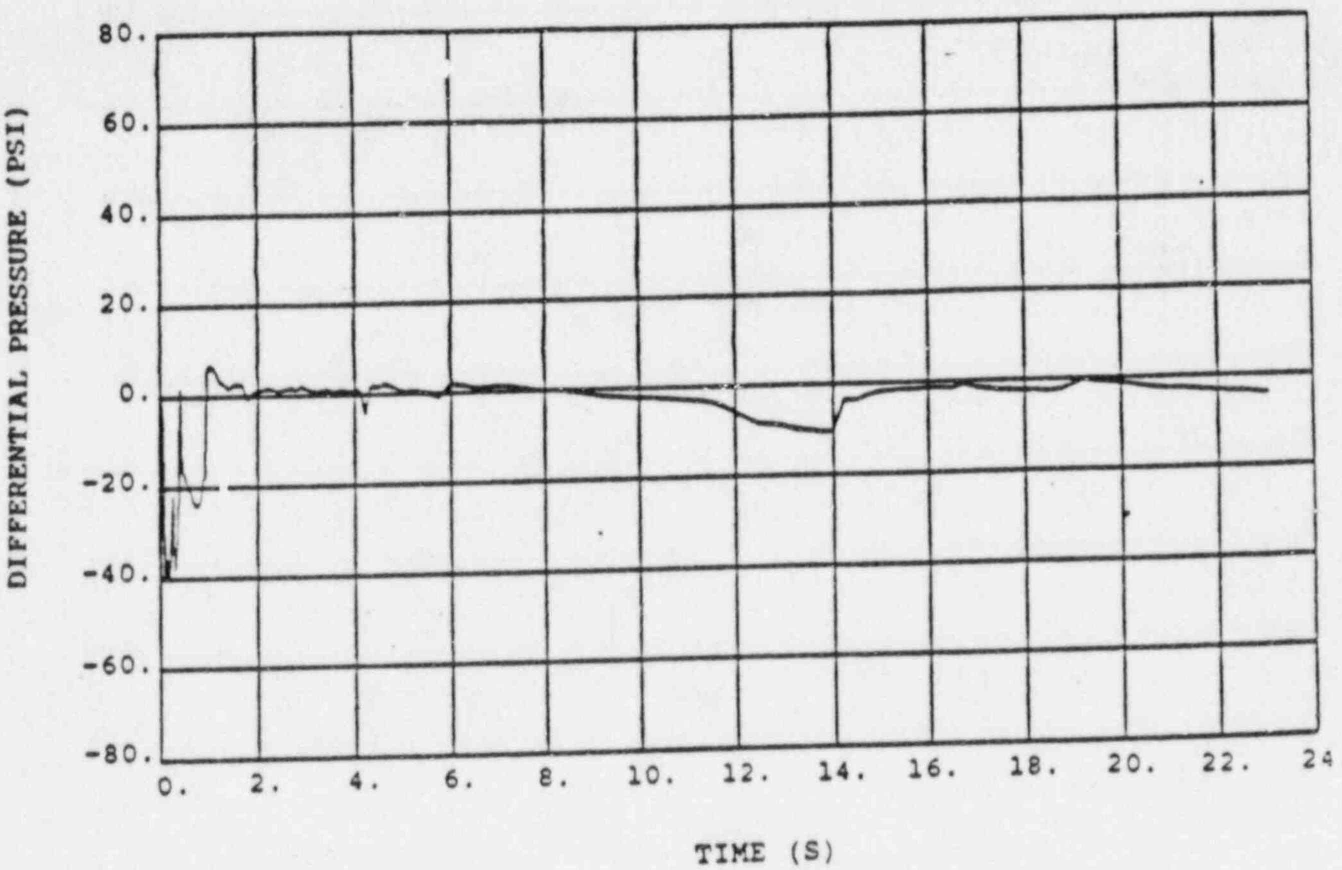

Figure 15.4-20
CORE PRESSURE DROP -
DECLG (C_D = 0.6, MIN SI)
Virgil C. Summer Nuclear Station
Final Safety Analysis Report

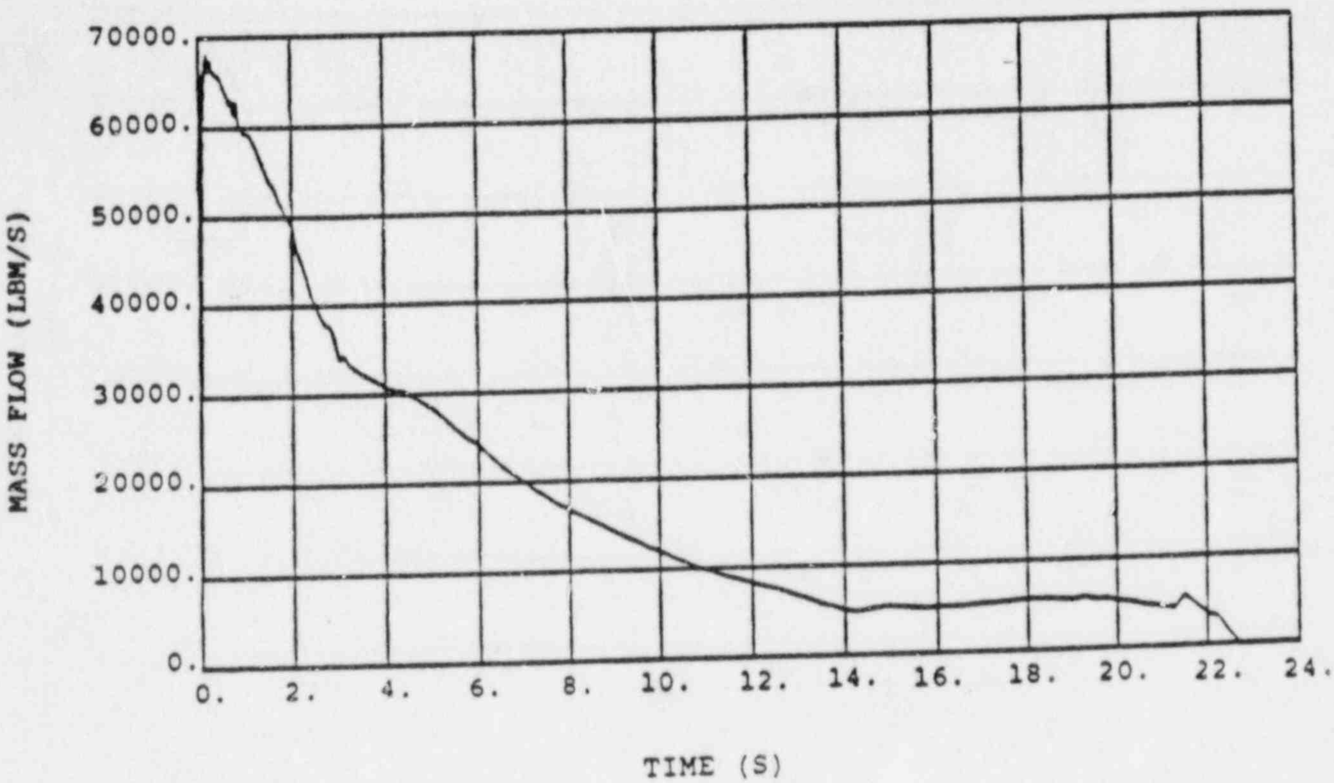

Figure 15.4-21
BREAK FLOW DURING BLOWDOWN -
DECLG (C_D = 0.6, MIN SI)
Virgil C. Summer Nuclear Station
Final Safety Analysis Report

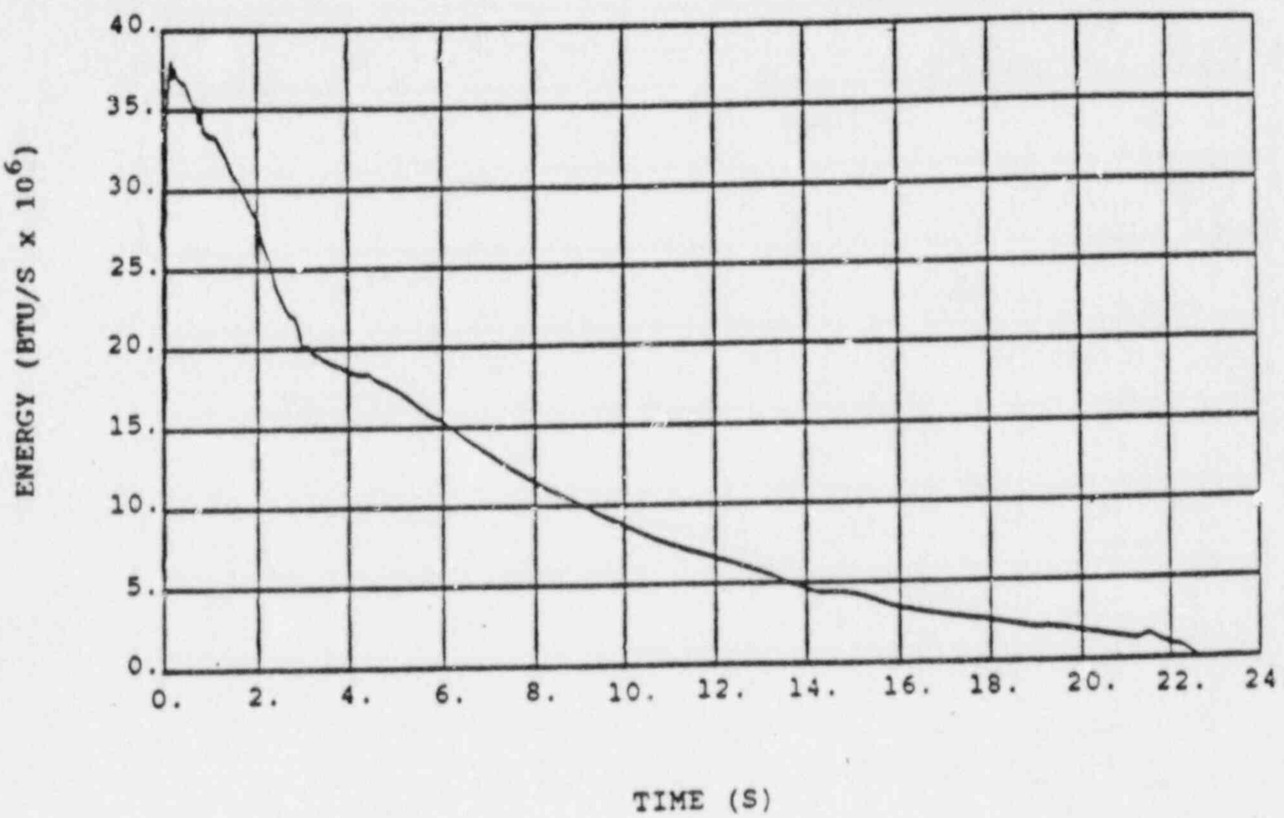

Figure 15.4-22
BREAK ENERGY DURING BLOWDOWN -
DECLG (C_D = 0.6, MIN SI)
Virgil C. Summer Nuclear Station
Final Safety Analysis Report

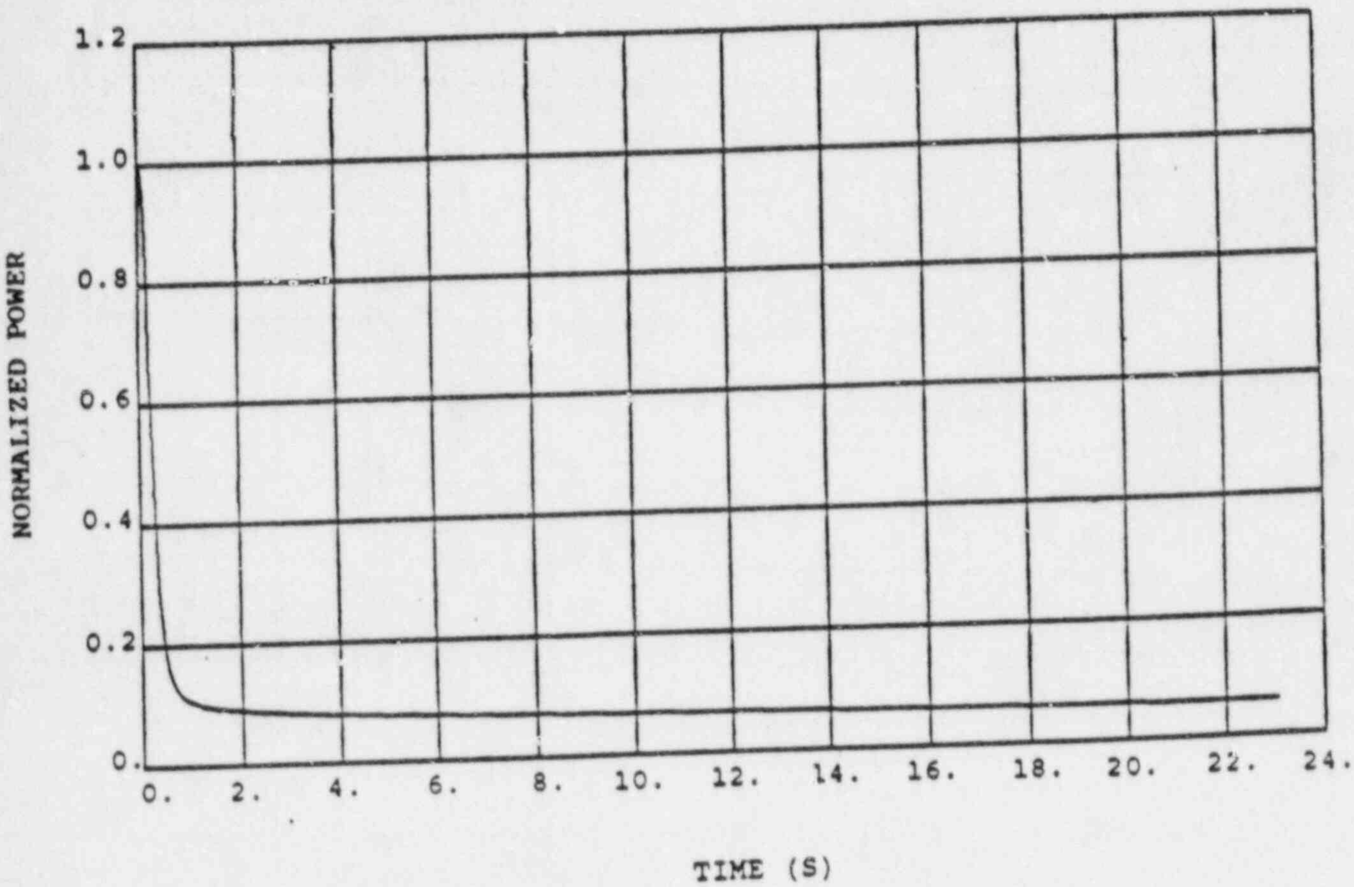

Figure 15.4-23
CORE POWER -
DECLG (C_D = 0.6, MIN SI)
Virgil C. Summer Nuclear Station
Final Safety Analysis Report

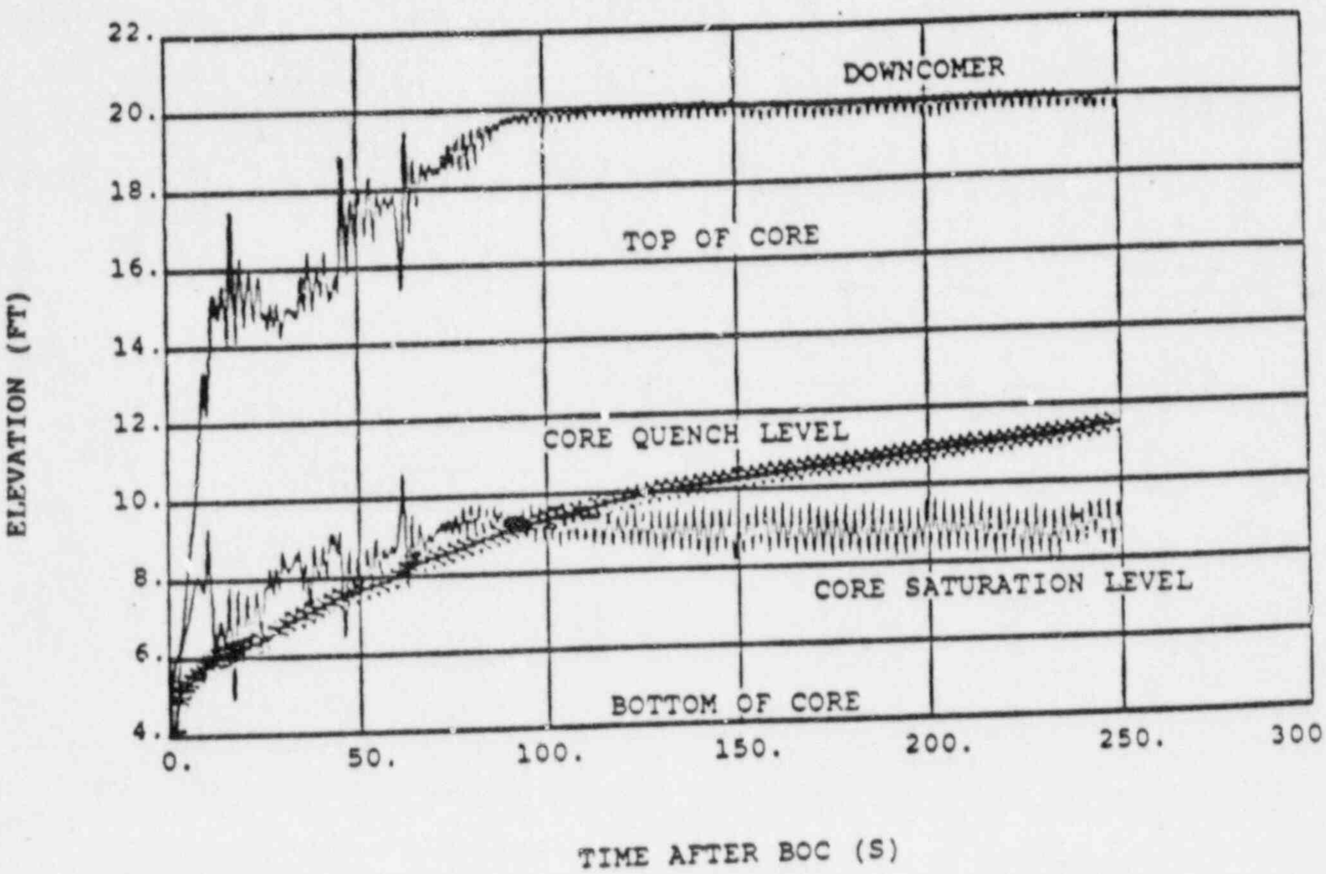

Figure 15.4-24
CORE AND DOWNCOMER LIQUID LEVELS
DURING REFLOOD
DECLG (C_D = 0.6, MIN SI)
Virgil C. Summer Nuclear Station
Final Safety Analysis Report

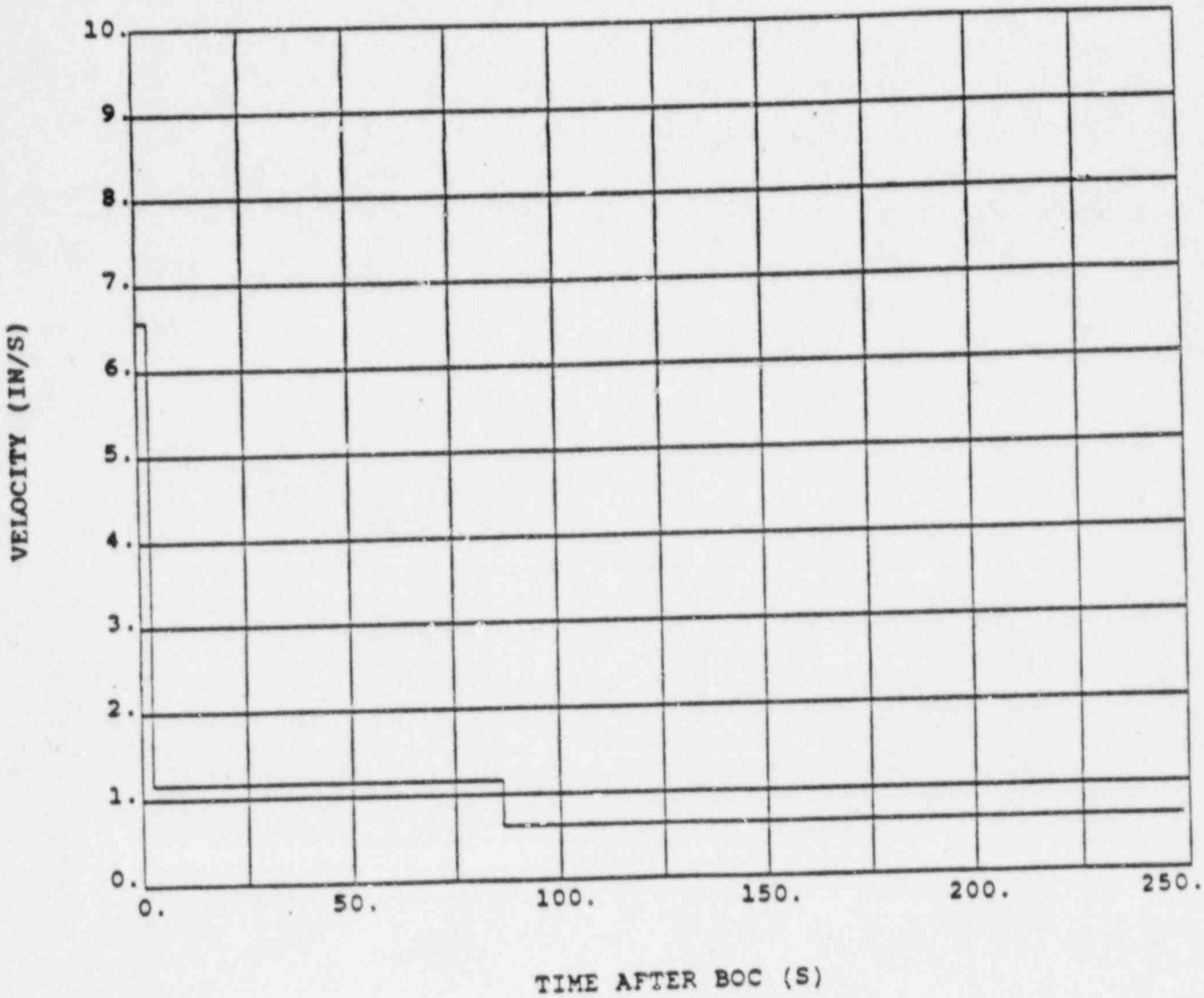

Figure 15.4-25
CORE INLET FLUID VELOCITY FOR ROD THERMAL ANALYSIS
DECLG (C_D = 0.6, MIN SI)
Virgil C. Summer Nuclear Station
Final Safety Analysis Report

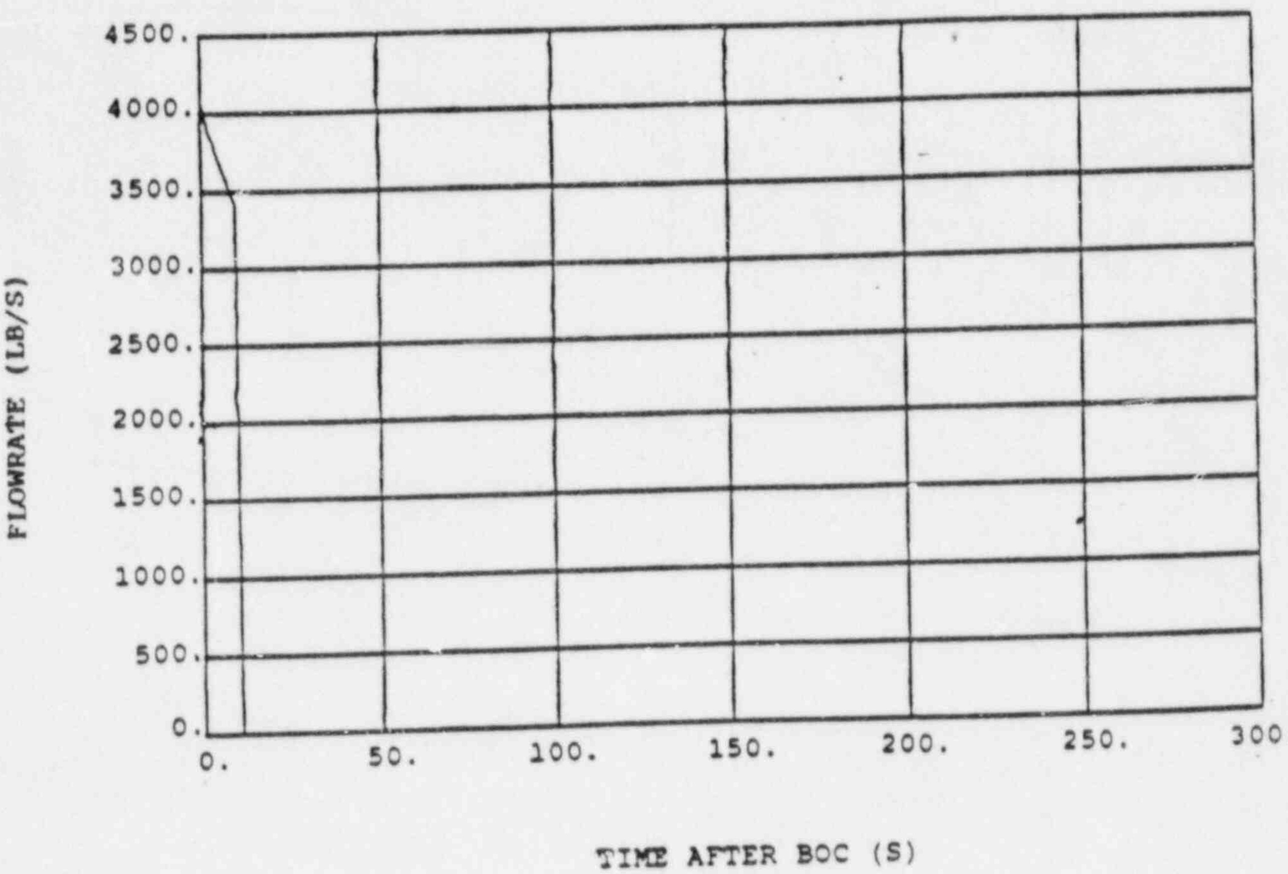

Figure 15.4-26
ACCUMULATOR FLOW DURING REFLOOD
DECLG (C_D = 0.6, MIN SI)
Virgil C. Summer Nuclear Station
Final Safety Analysis Report

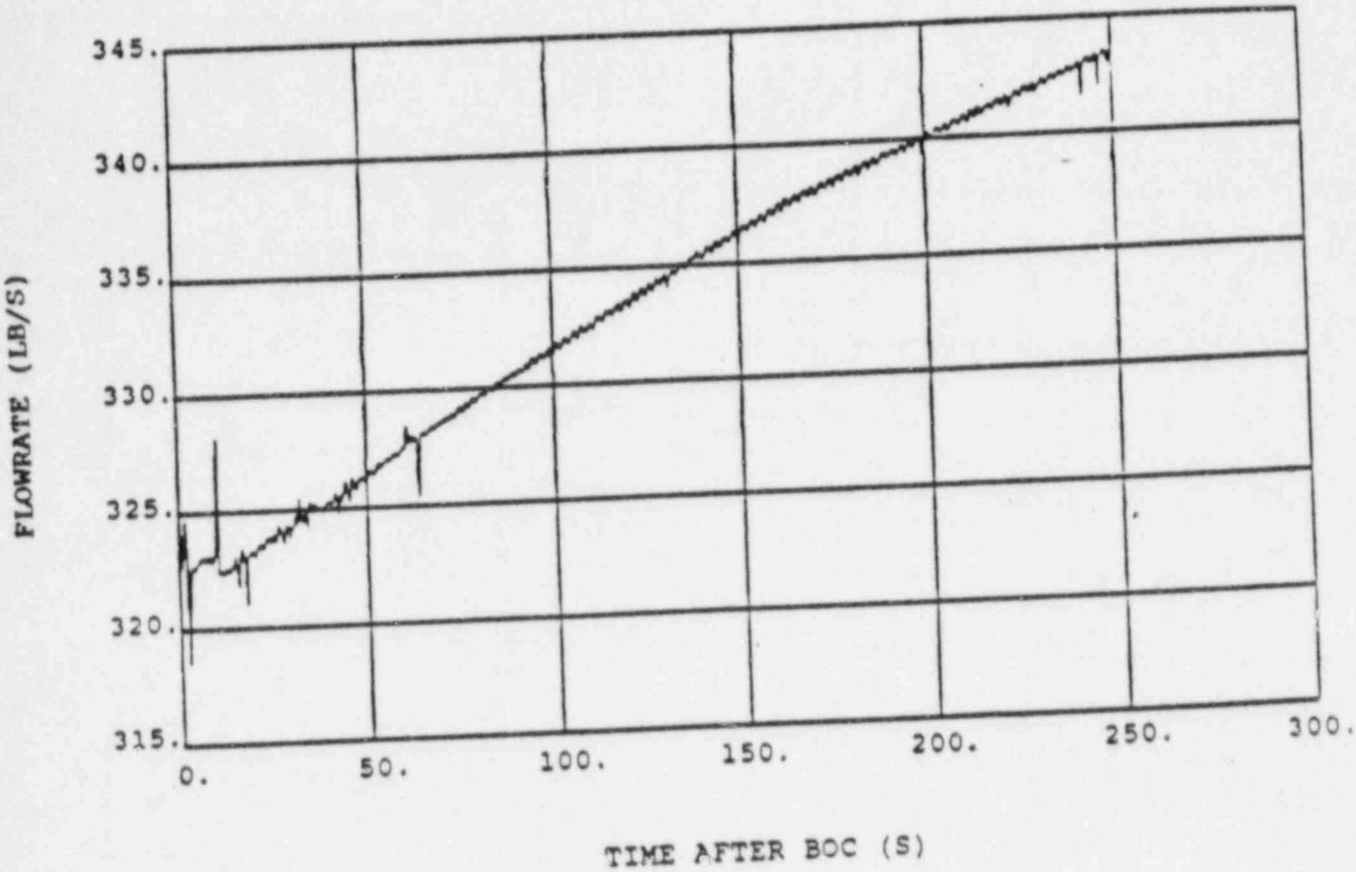

Figure 15.4-27
SI FLOW DURING REFLOOD
DECLG (C_D = 0.6, MIN SI)
Virgil C. Summer Nuclear Station
Final Safety Analysis Report

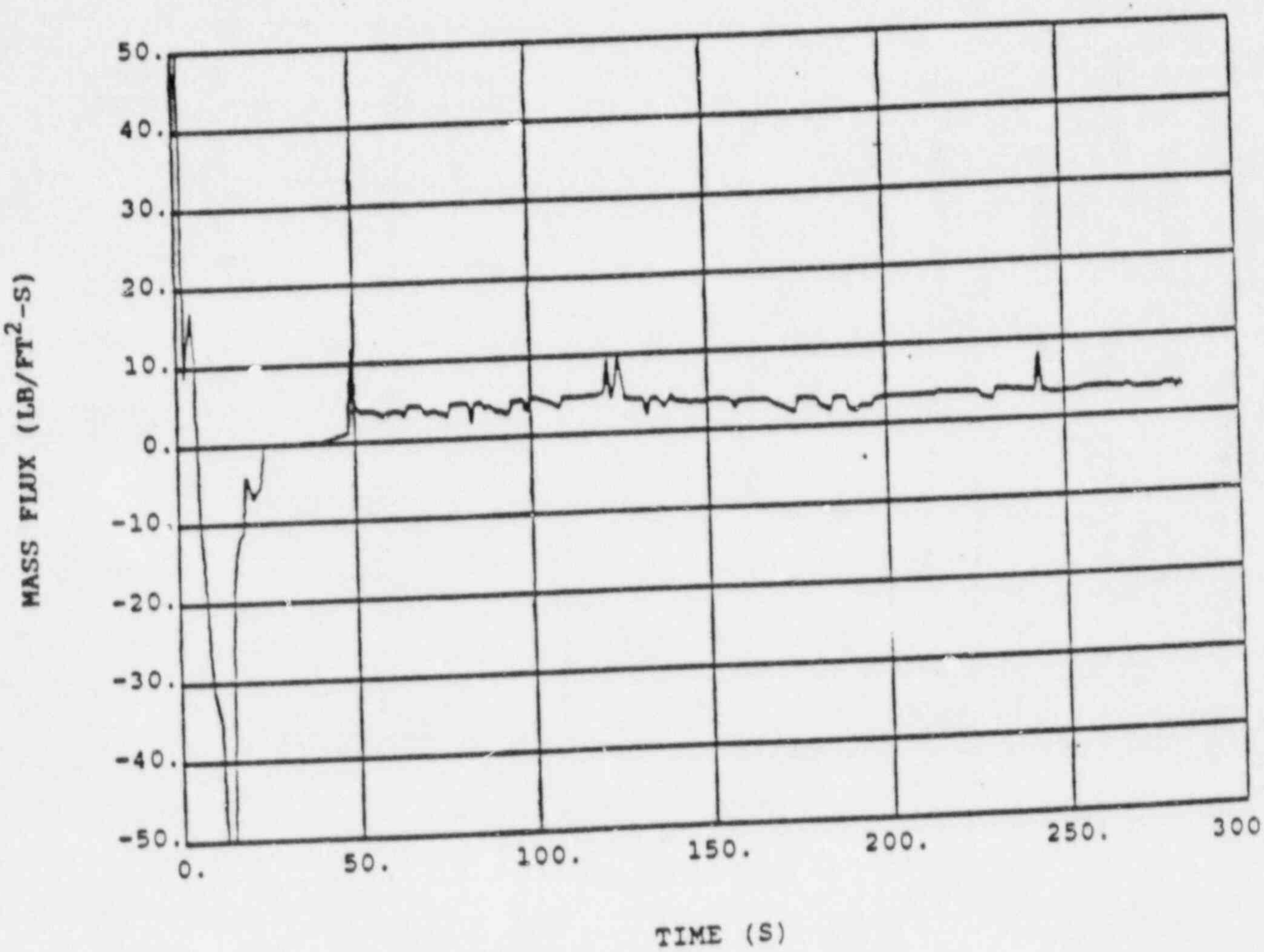

Figure 15.4-28
MASS FLUX AT THE PEAK ROD
TEMPERATURE ELEVATION
DECLG (C_D = 0.6, MIN SI)
Virgil C. Summer Nuclear Station
Final Safety Analysis Report

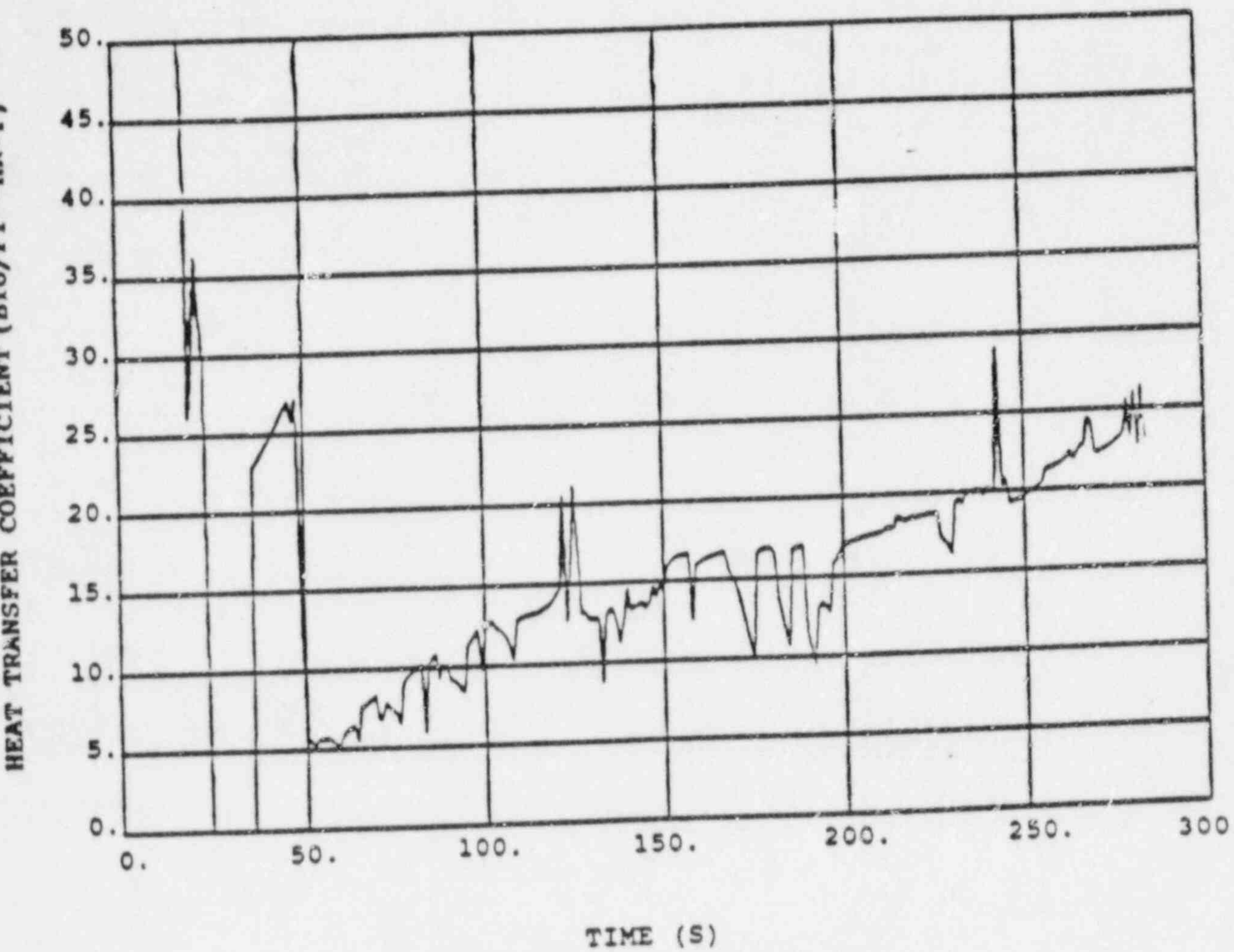

Figure 15.4-29
ROD HEAT TRANSFER COEFFICIENT AT
THE PEAK TEMPERATURE LOCATION
DECLG (C_D = 0.6, MIN SI)
Virgil C. Summer Nuclear Station
Final Safety Analysis Report

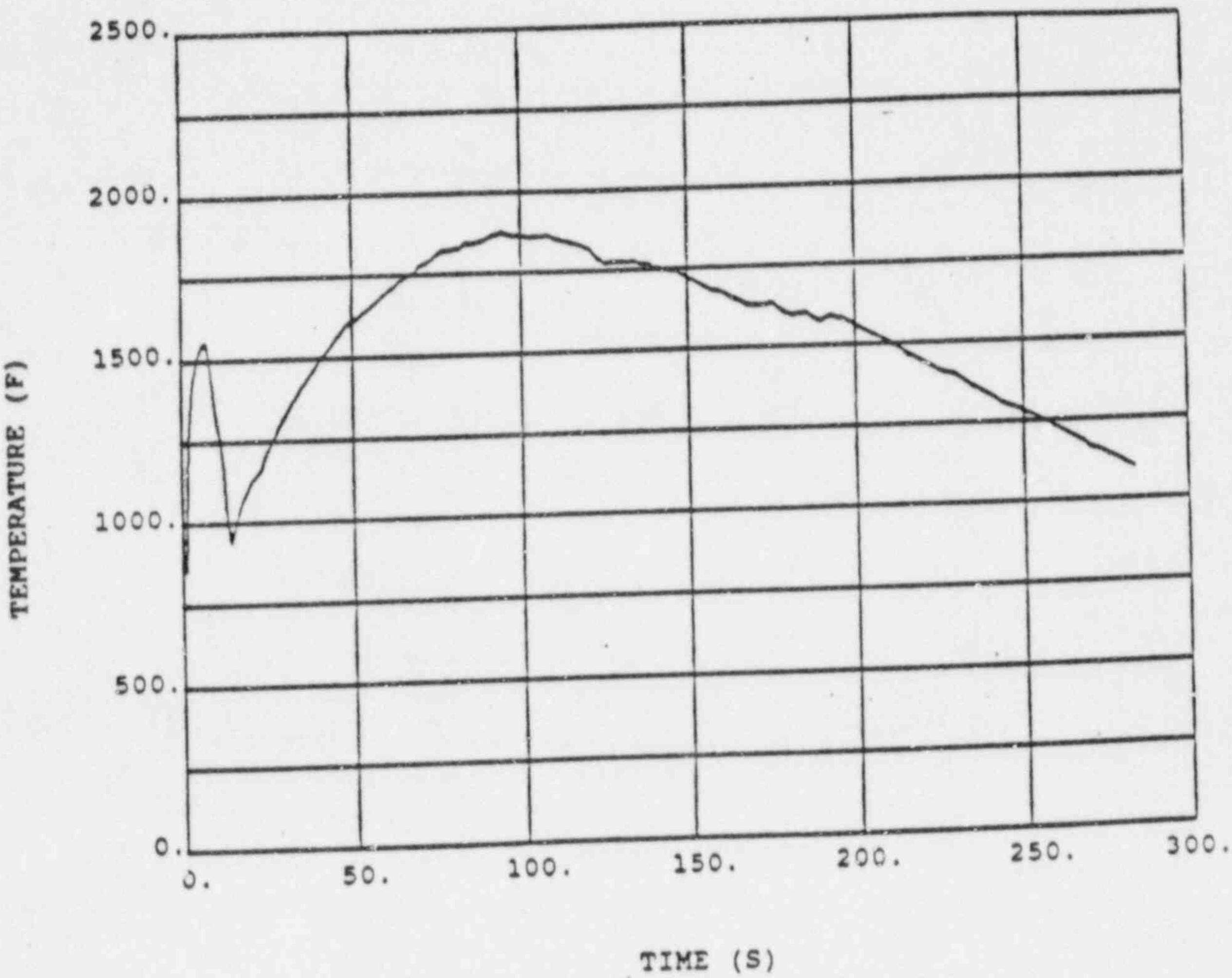

Figure 15.4-30
FUEL ROD PEAK CLAD TEMPERATURE
DECLG (C_D = 0.6, MIN SI)
Virgil C. Summer Nuclear Station
Final Safety Analysis Report

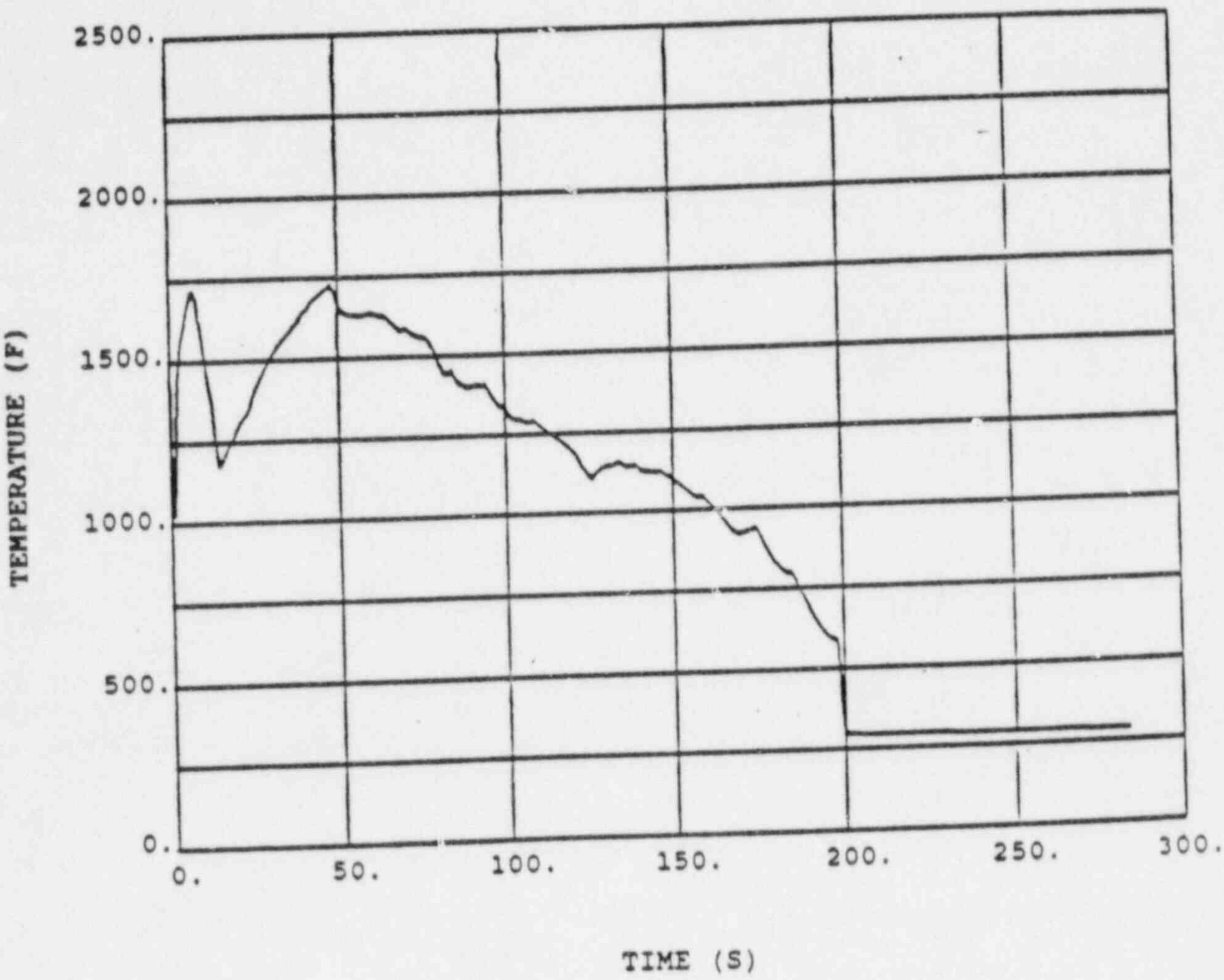

Figure 15.4-31
CLAD TEMPERATURE AT THE BURST NODE
DECLG (C_D = 0.6, MIN SI)
Virgil C. Summer Nuclear Station
Final Safety Analysis Report

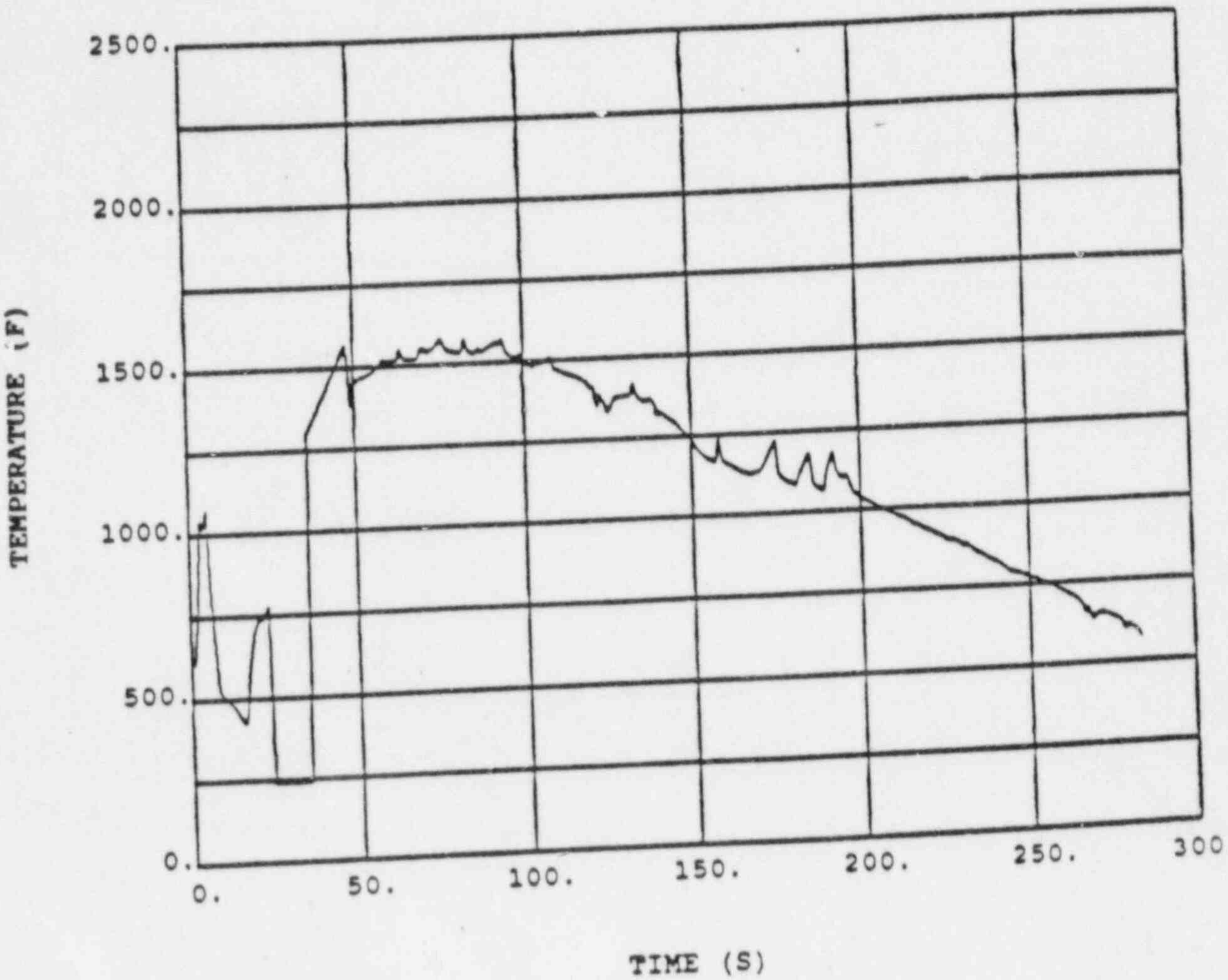

Figure 15.4-32
FLUID TEMPERATURE
DECLG (C_D = 0.6, MIN SI)
Virgil C. Summer Nuclear Station
Final Safety Analysis Report

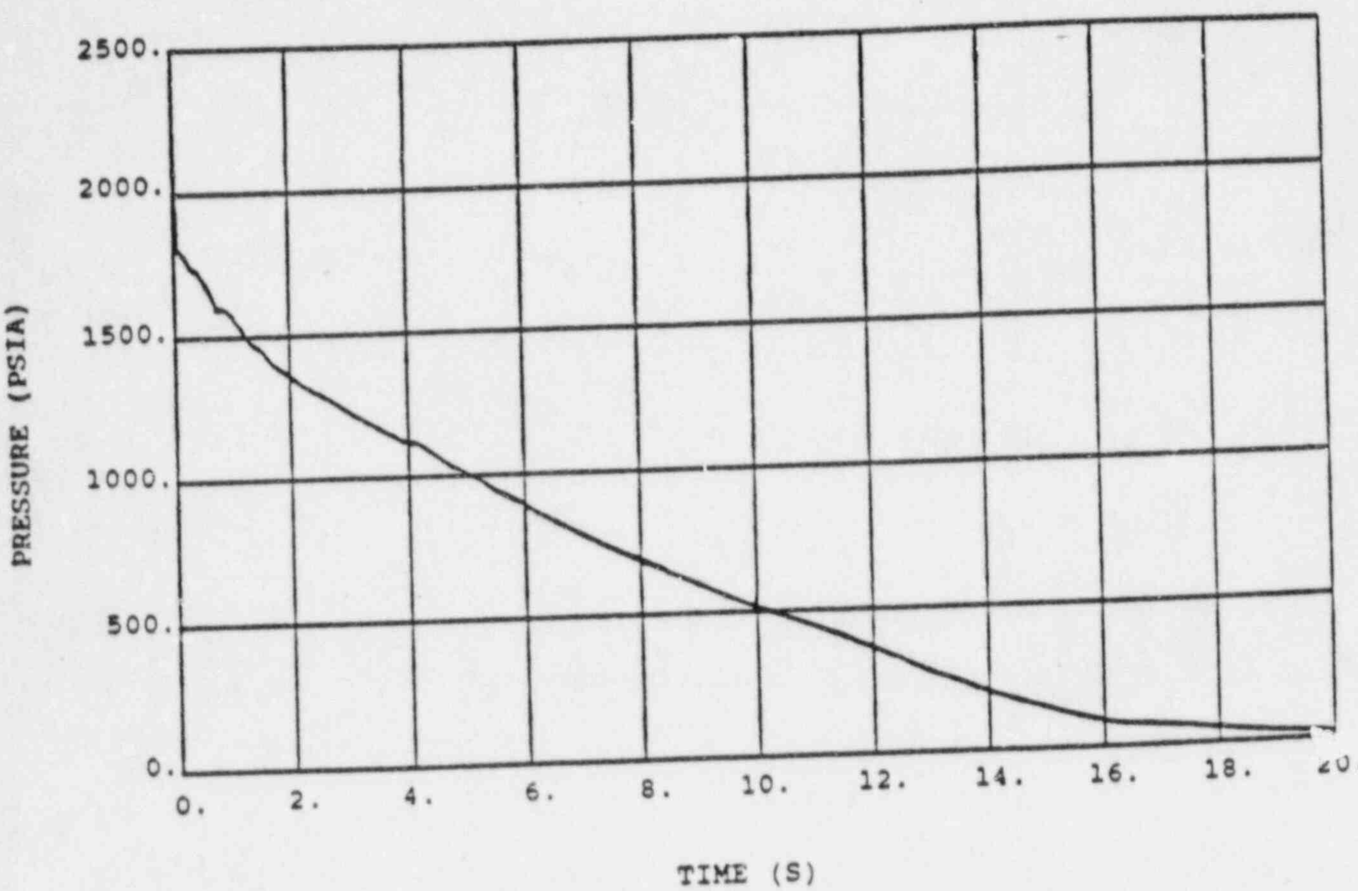

Figure 15.4-33
REACTOR COOLANT SYSTEM PRESSURE -
DECLG (C_D = 0.8, MIN SI)
Virgil C. Summer Nuclear Station
Final Safety Analysis Report

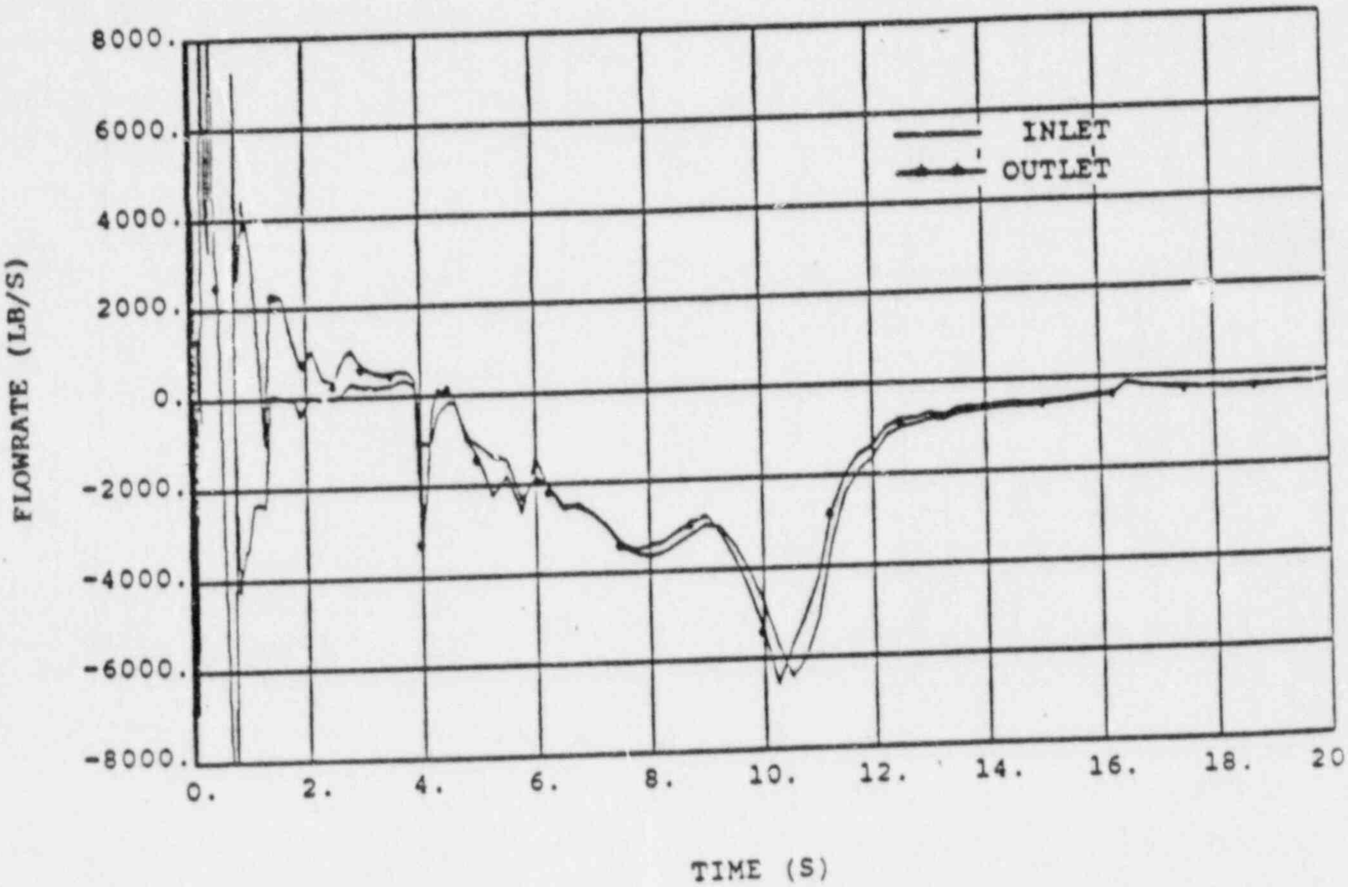

Figure 15.4-34
CORE FLOWRATE -
DECLG (C_D = 0.8, MIN SI)
Virgil C. Summer Nuclear Station
Final Safety Analysis Report

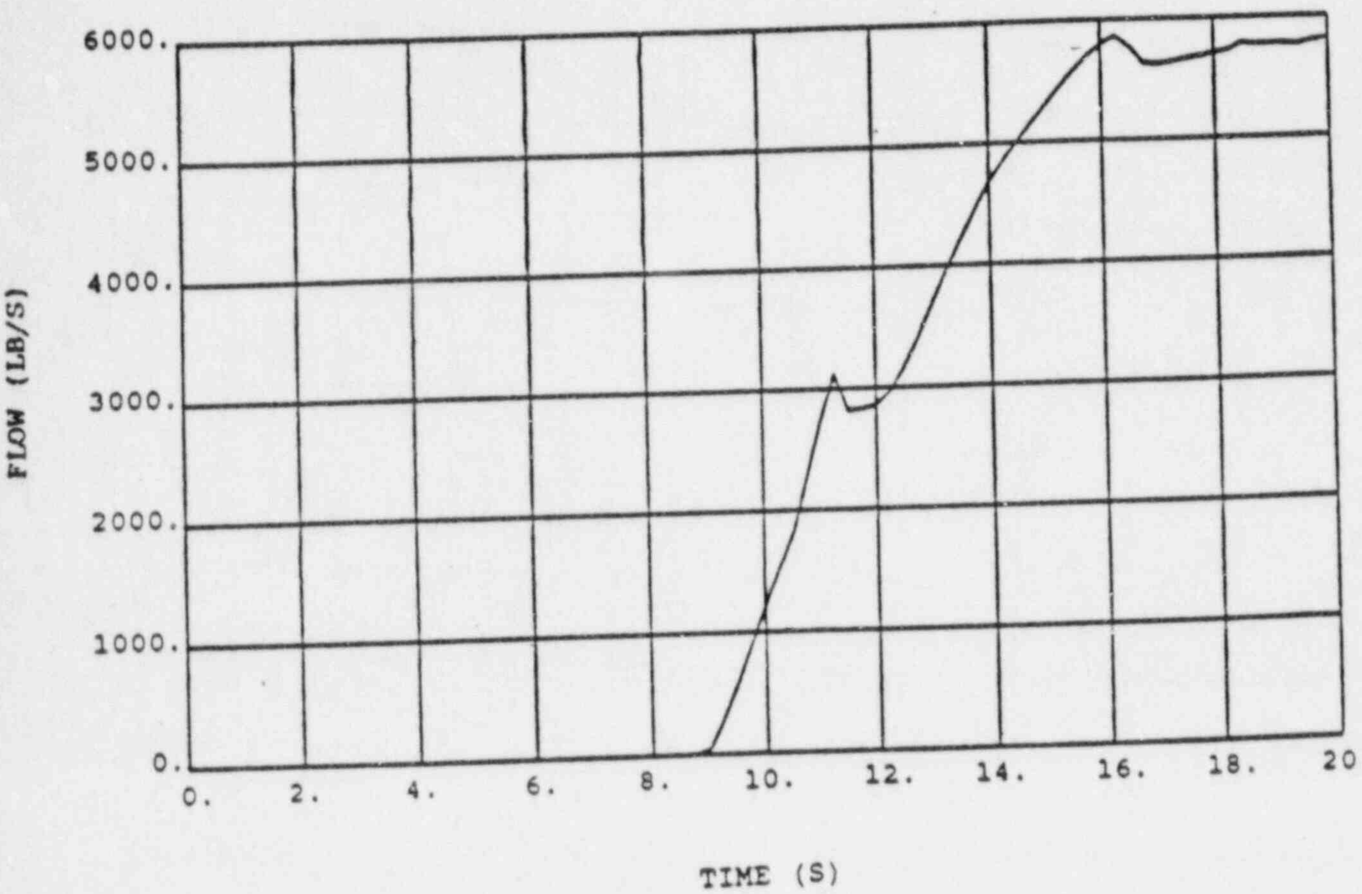

Figure 15.4-35
ACCUMULATOR FLOW DURING BLOWDOWN - DECLG (C_D = 0.8, MIN SI)
Virgil C. Summer Nuclear Station
Final Safety Analysis Report

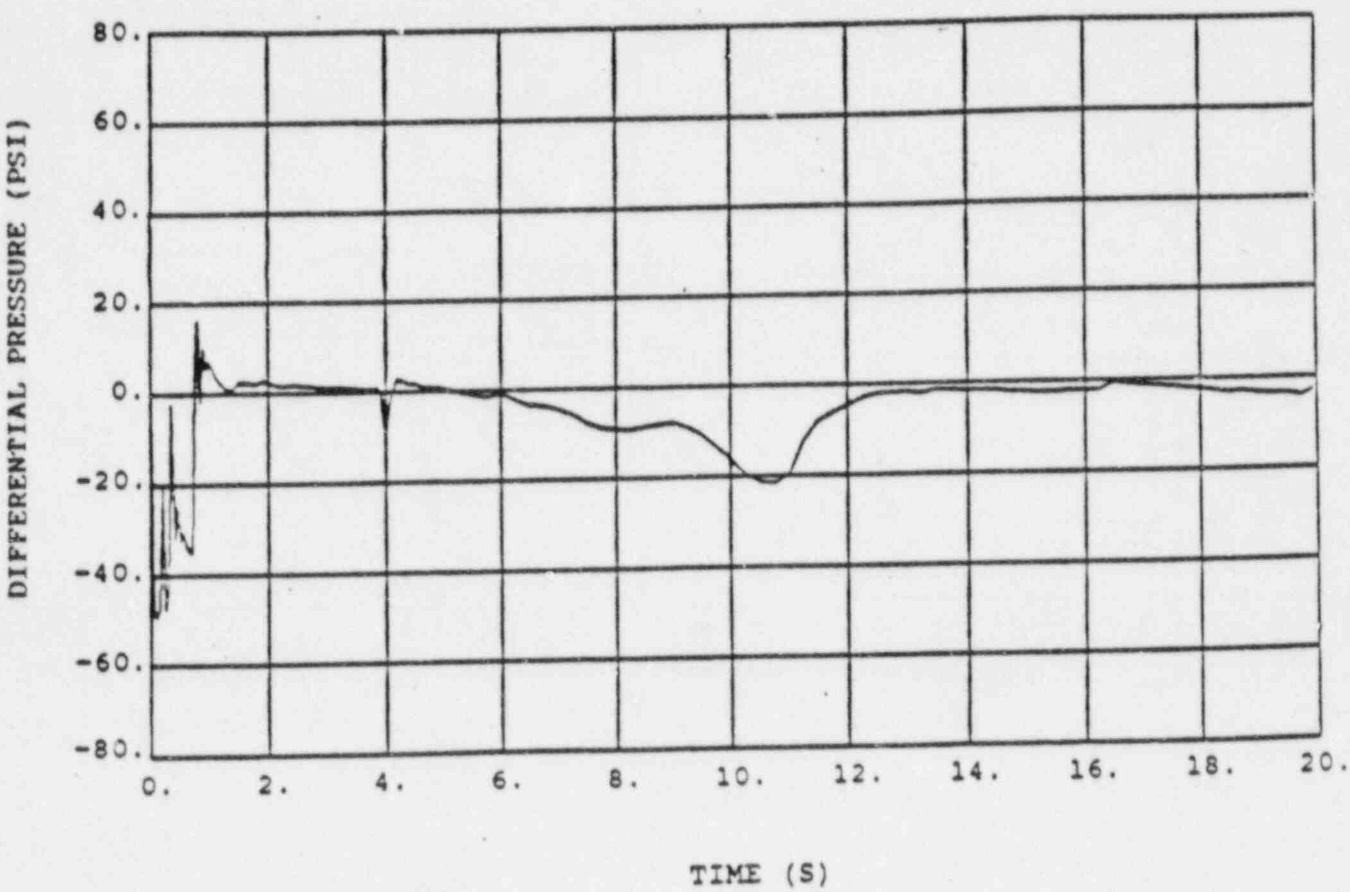

Figure 15.4-36
CORE PRESSURE DROP -
DECLG (C_D = 0.8, MIN SI)
Virgil C. Summer Nuclear Station
Final Safety Analysis Report

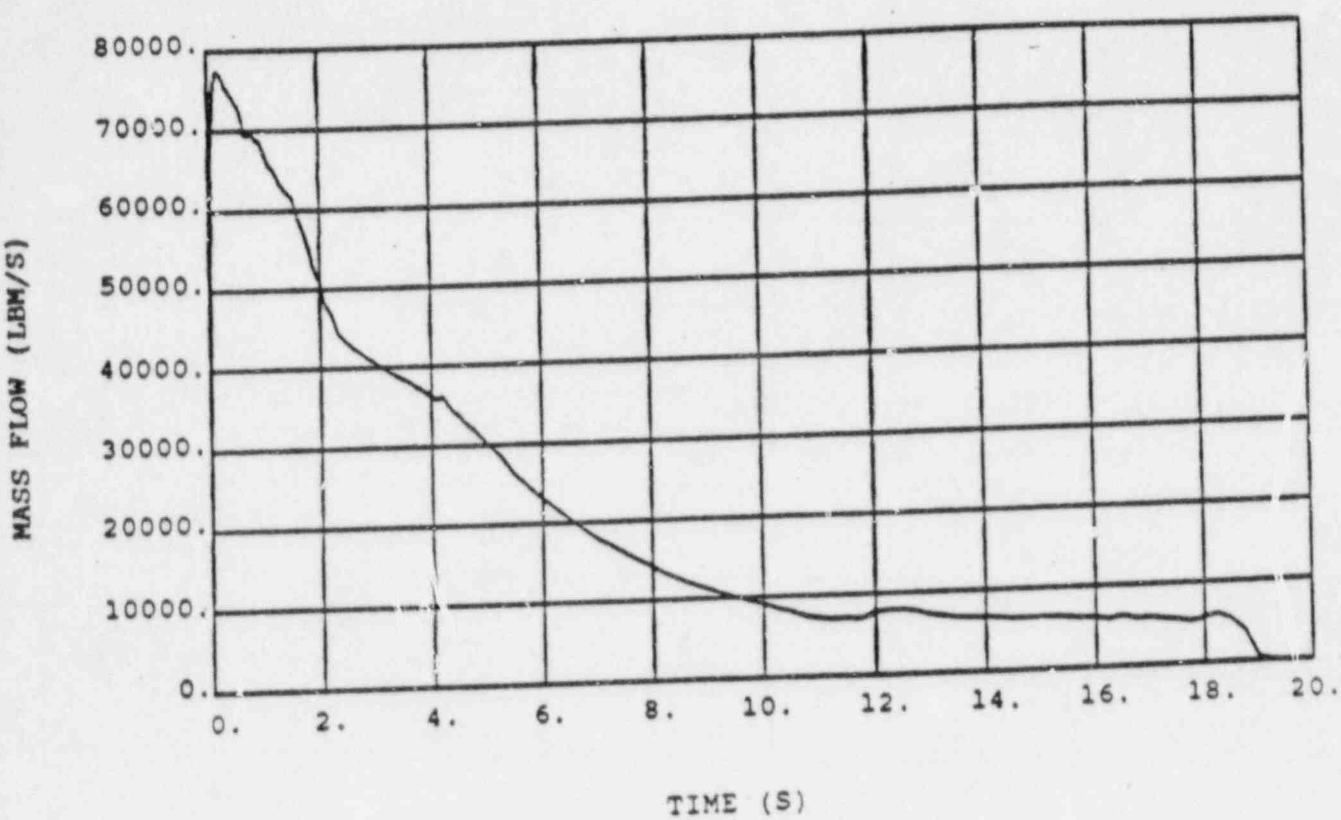

Figure 15.4-37
BREAK FLOW DURING BLOWDOWN -
DECLG (C_D = 0.8, MIN SI)
Virgil C. Summer Nuclear Station
Final Safety Analysis Report

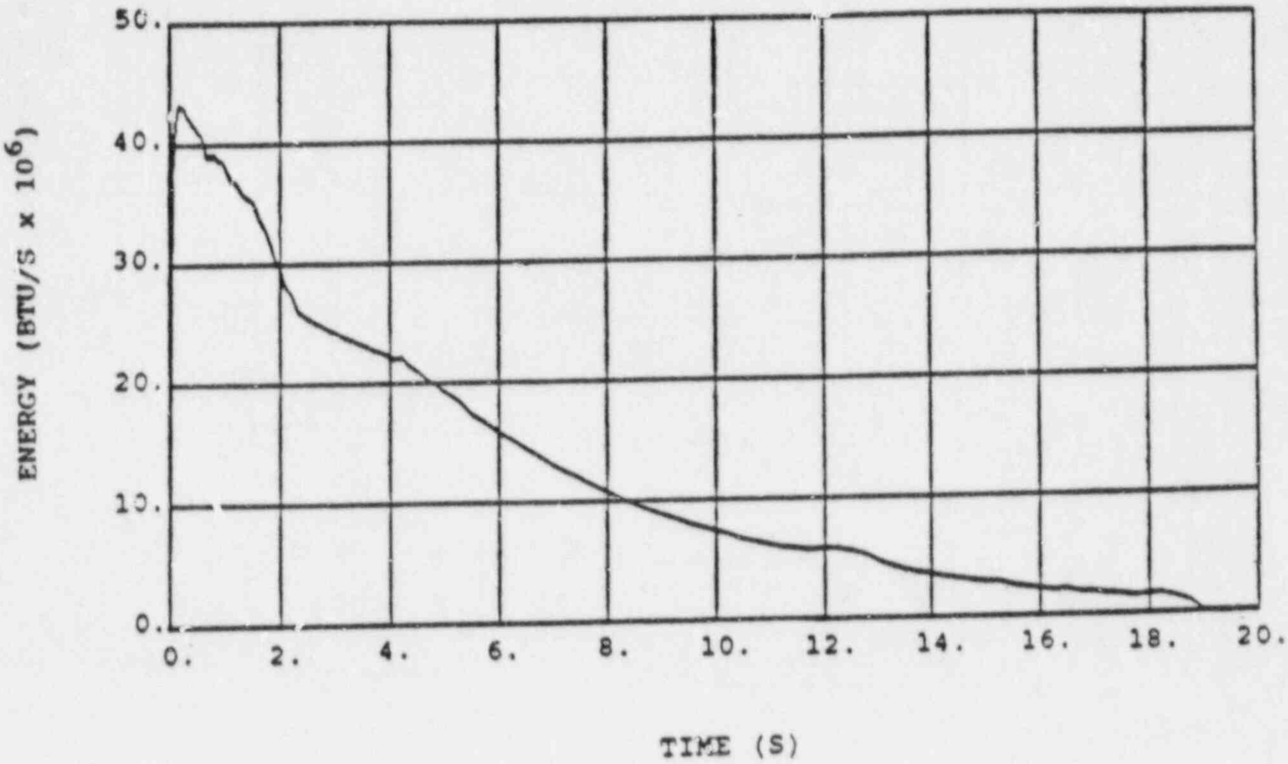

Figure 15.4-38
BREAK ENERGY DURING BLOWDOWN -
DECLG (C_D = 0.8, MIN SI)
Virgil C. Summer Nuclear Station
Final Safety Analysis Report

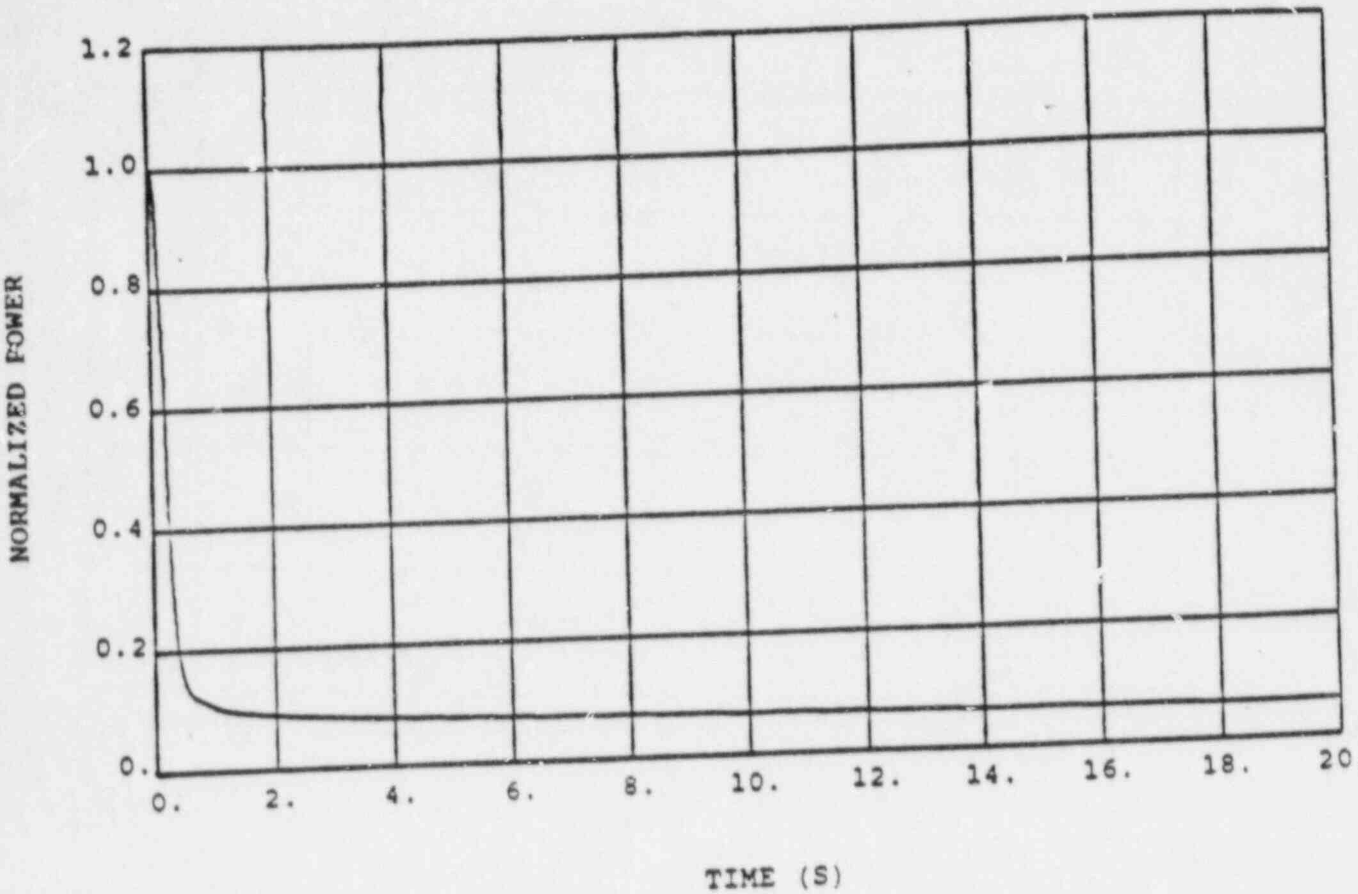

Figure 15.4-39
CORE POWER -
DECLG (C_D = 0.8, MIN SI)
Virgil C. Summer Nuclear Station
Final Safety Analysis Report

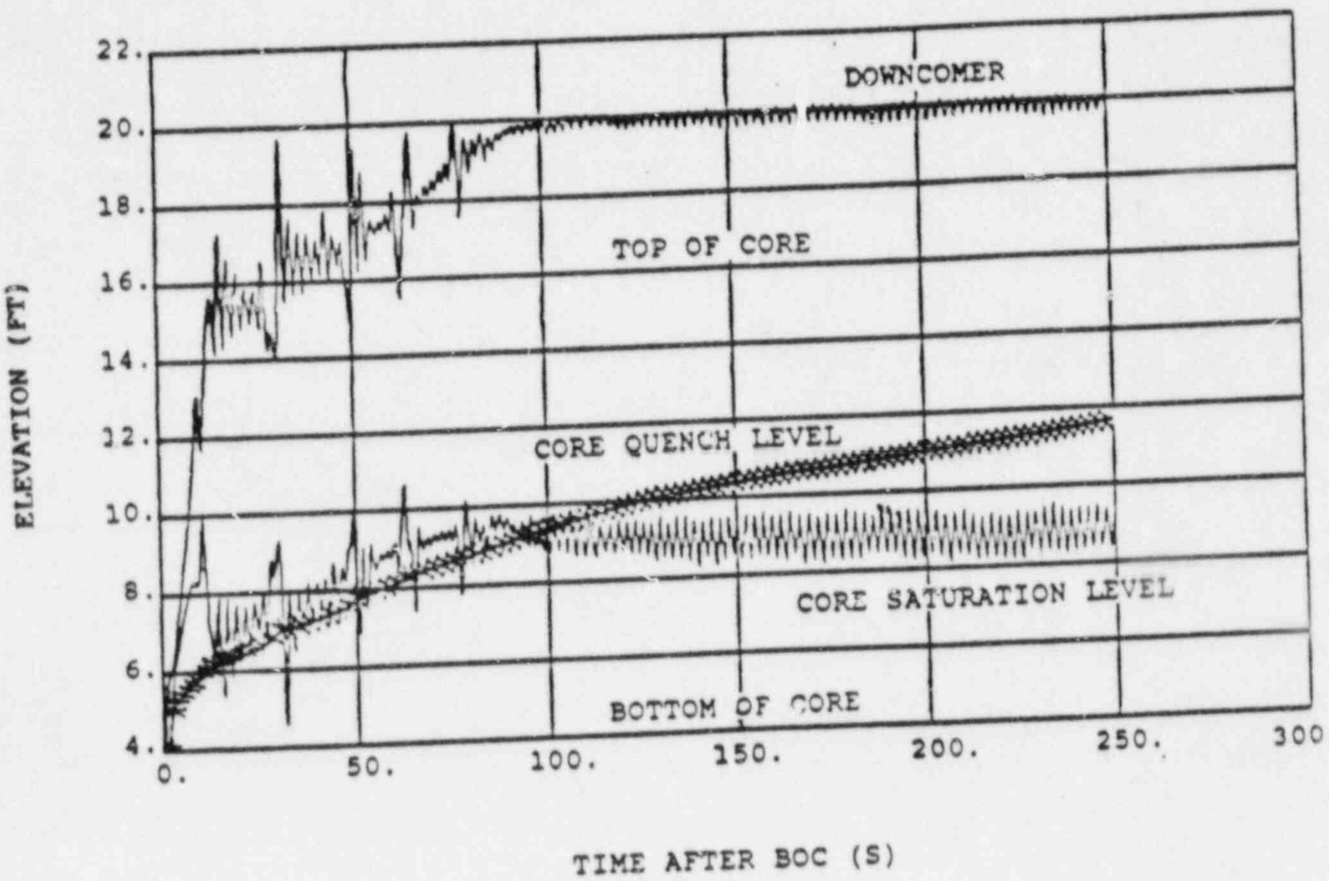

Figure 15.4-40
CORE AND DOWNCOMER LIQUID LEVELS
DURING REFLOOD
DECLG (C_D = 0.8, MIN SI)
Virgil C. Summer Nuclear Station
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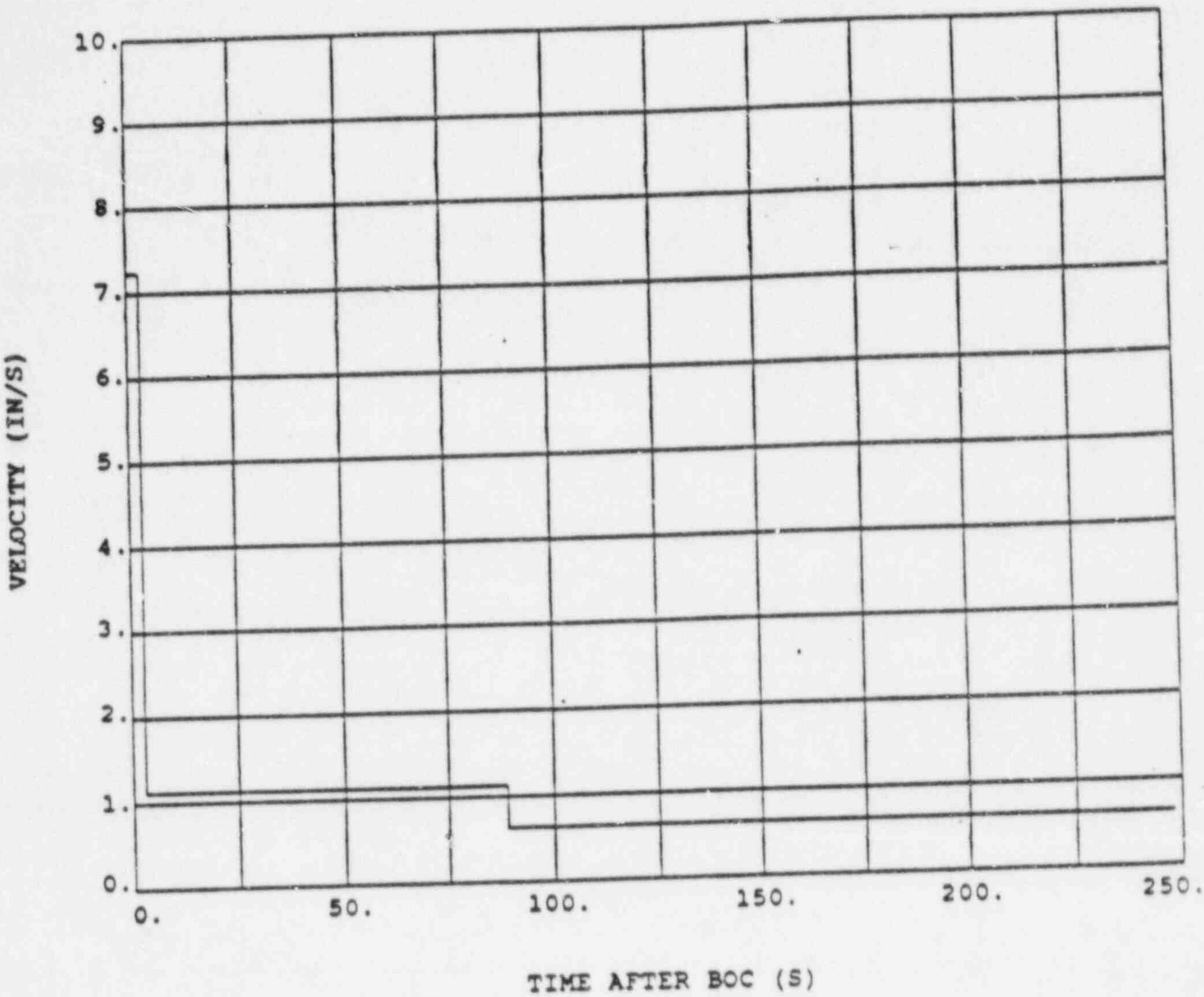

Figure 15.4-41
CORE INLET FLUID VELOCITY FOR ROD THERMAL ANALYSIS
DECLG (C_D = 0.8, MIN SI)
Virgil C. Summer Nuclear Station
Final Safety Analysis Report

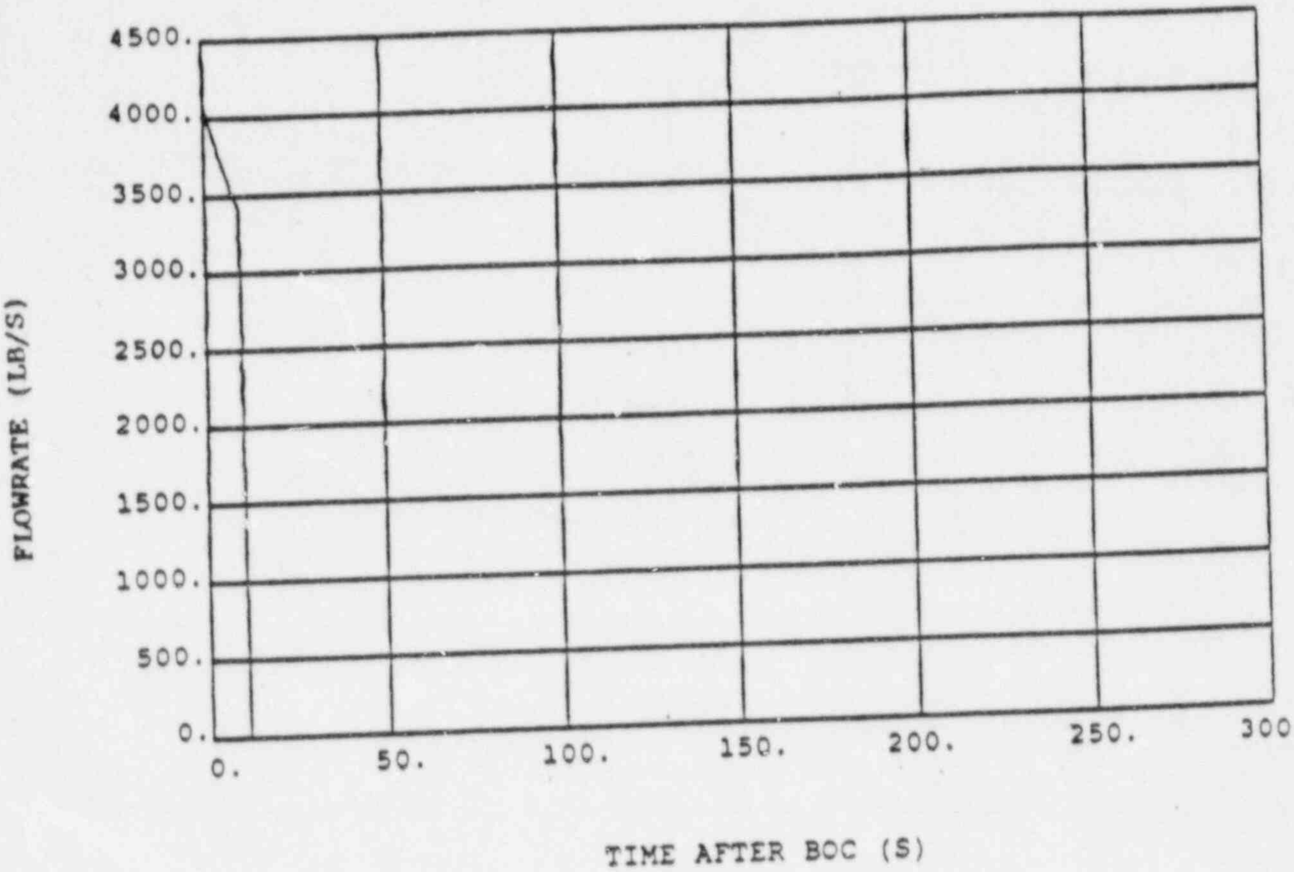

Figure 15.4-42
ACCUMULATOR FLOW DURING REFLOOD
DECLG (C_D = 0.8, MIN SI)
Virgil C. Summer Nuclear Station
Final Safety Analysis Report

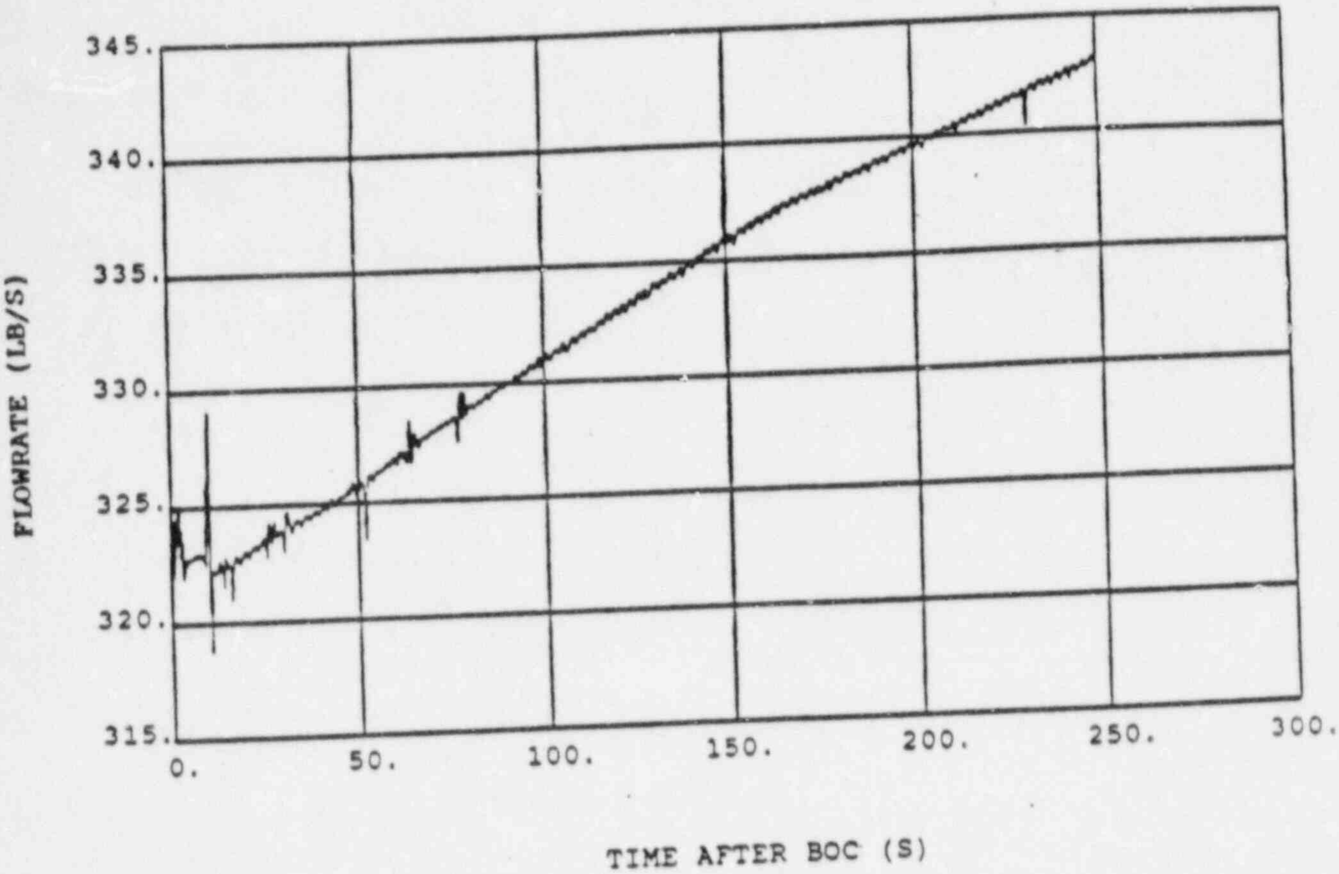

Figure 15.4-43
SI FLOW DURING REFLOOD
DECLG (C_D = 0.8, MIN SI)
Virgil C. Summer Nuclear Station
Final Safety Analysis Report

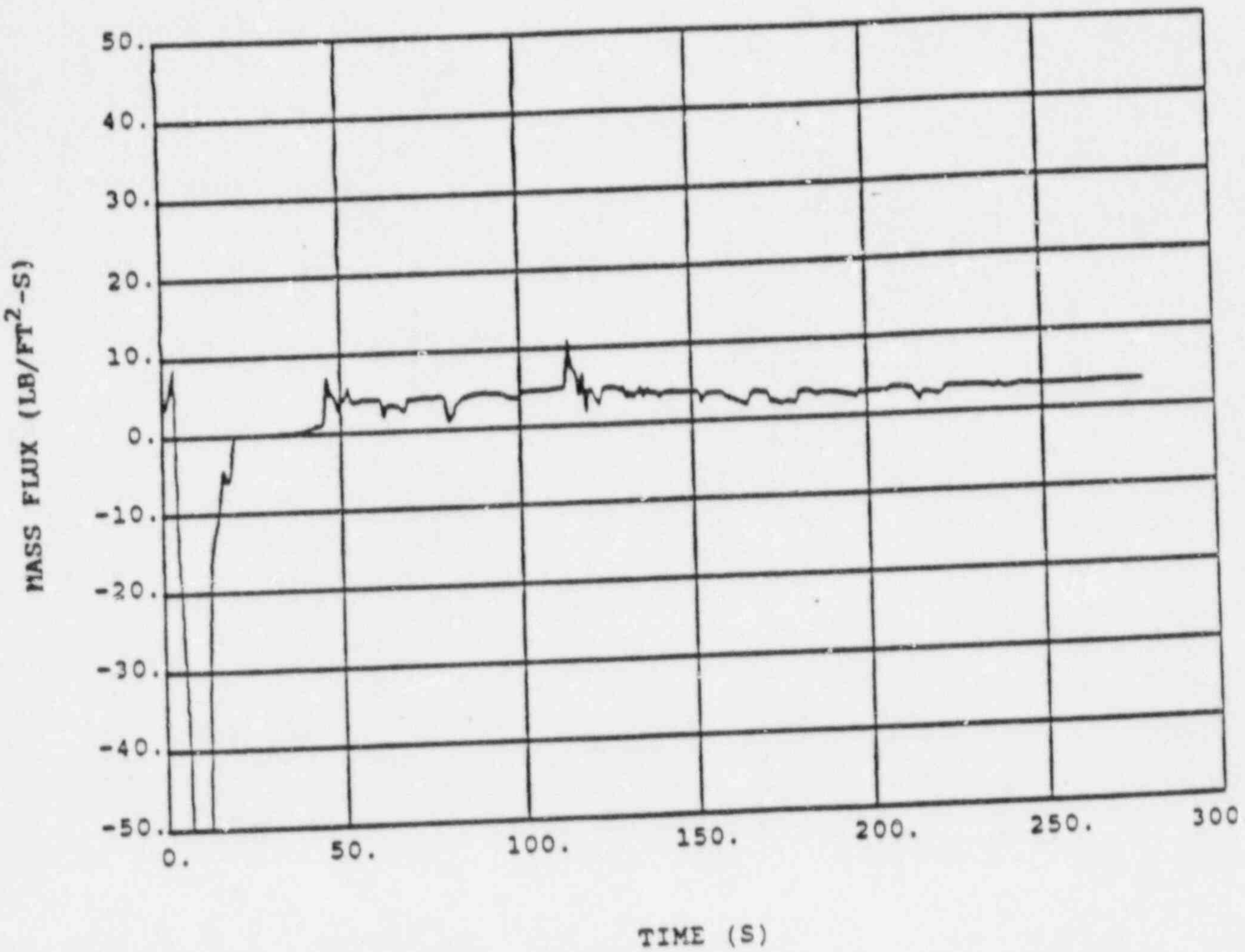

Figure 15.4-44
MASS FLUX AT THE PEAK ROD
TEMPERATURE ELEVATION
DECLG (C_D = 0.8, MIN SI)
Virgil C. Summer Nuclear Station
Final Safety Analysis Report

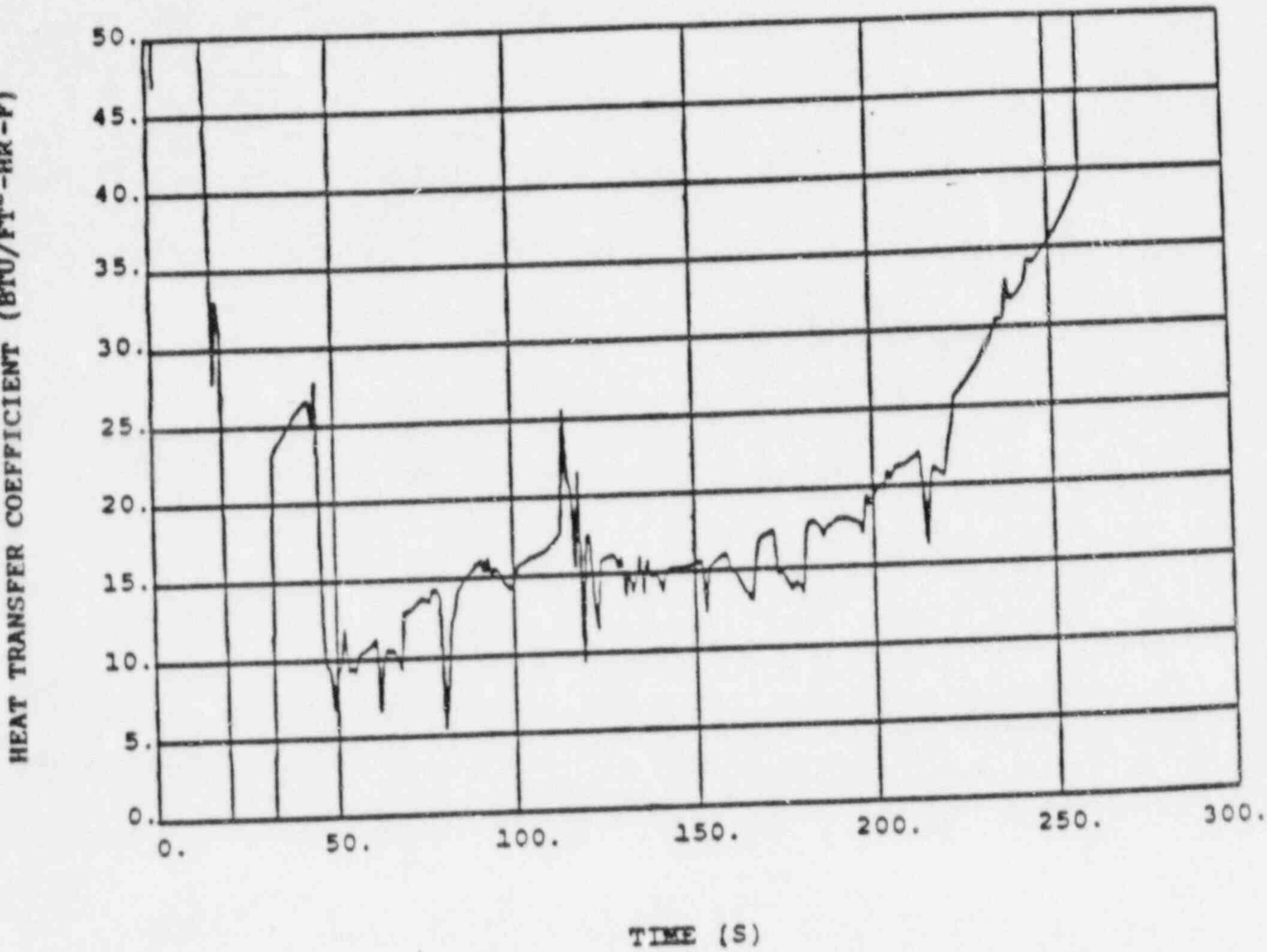

Figure 15.4-45
ROD HEAT TRANSFER COEFFICIENT AT THE PEAK TEMPERATURE LOCATION
DECLG (C_D = 0.8, MIN SI)
Virgil C. Summer Nuclear Station
Final Safety Analysis Report

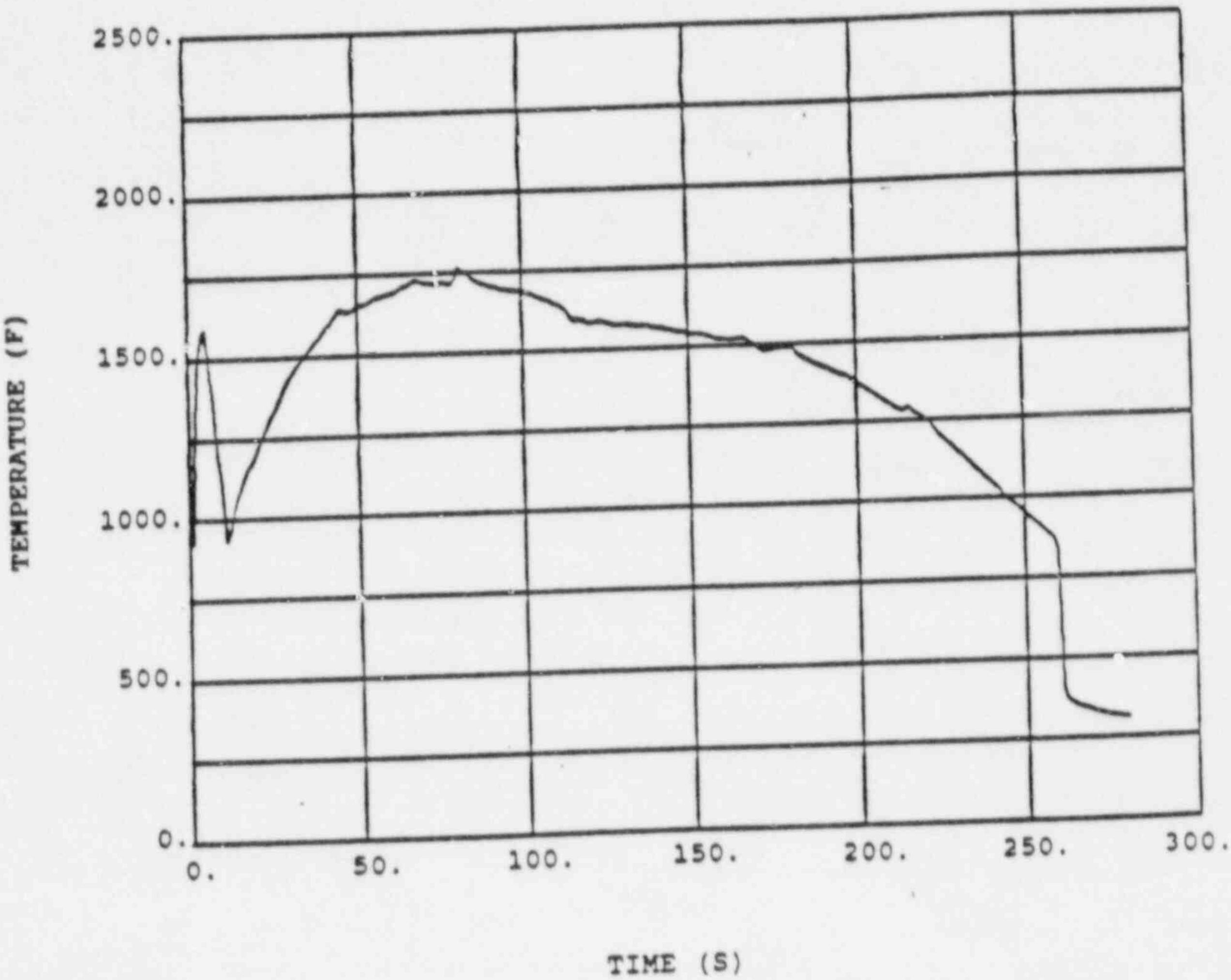

Figure 15.4-46
FUEL ROD PEAK CLAD TEMPERATURE
DECLG (C_D = 0.8, MIN SI)
Virgil C. Summer Nuclear Station
Final Safety Analysis Report

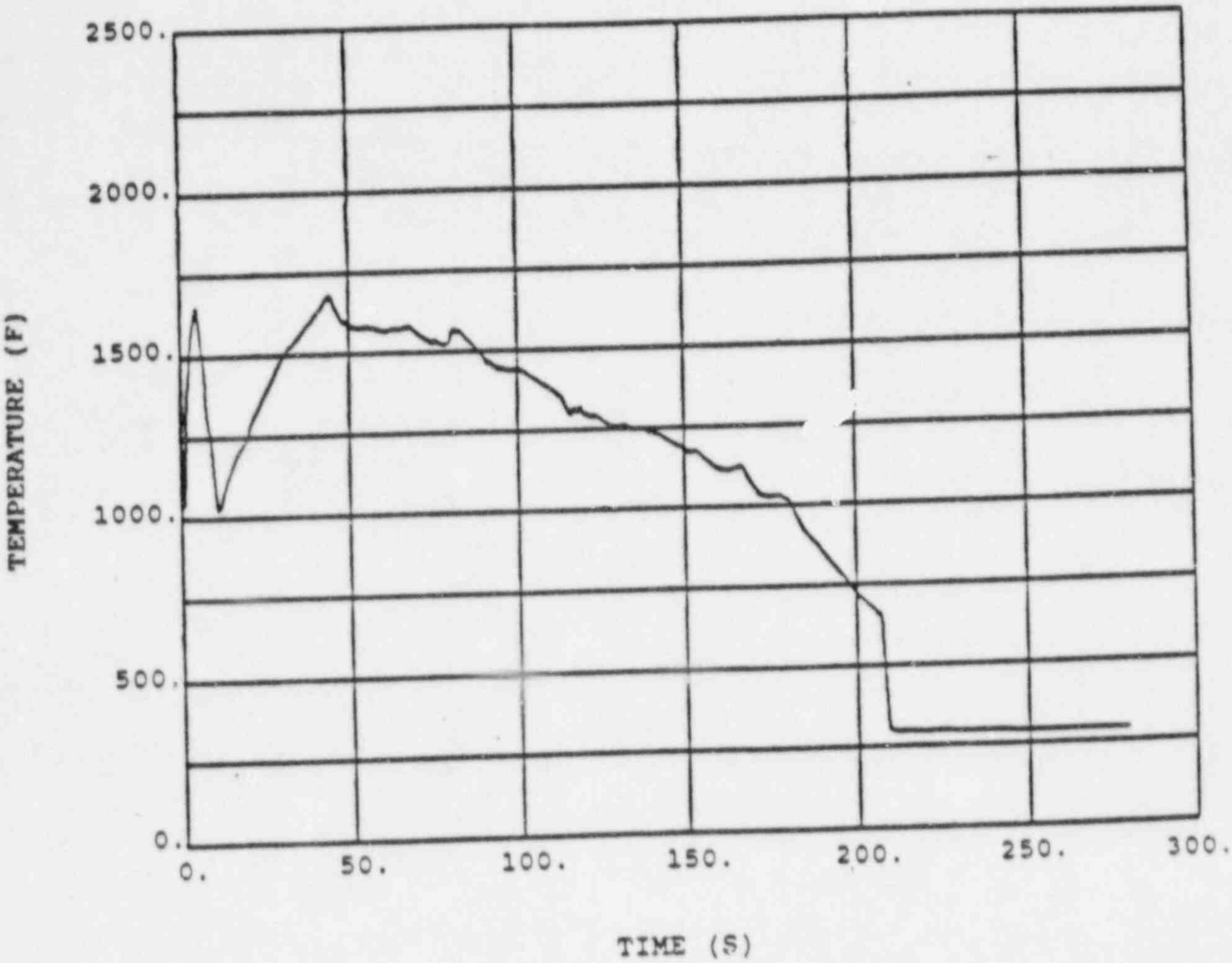

Figure 15.4-47
CLAD TEMPERATURE AT THE BURST NODE
DECLG (C_D = 0.8, MIN SI)
Virgil C. Summer Nuclear Station
Final Safety Analysis Report

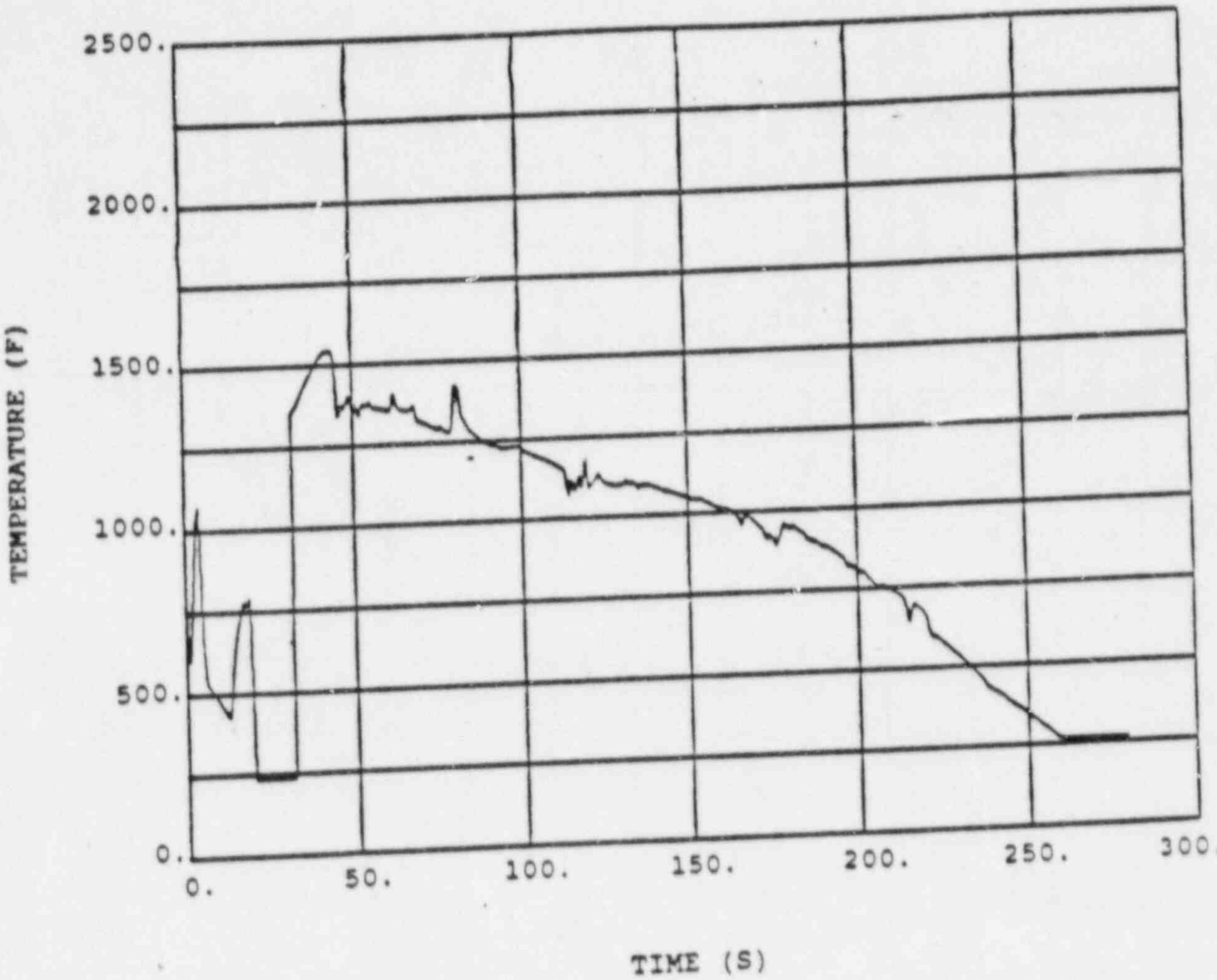

Figure 15.4-48
FLUID TEMPERATURE
DECLG (C_D = 0.8, MIN SI)
Virgil C. Summer Nuclear Station
Final Safety Analysis Report

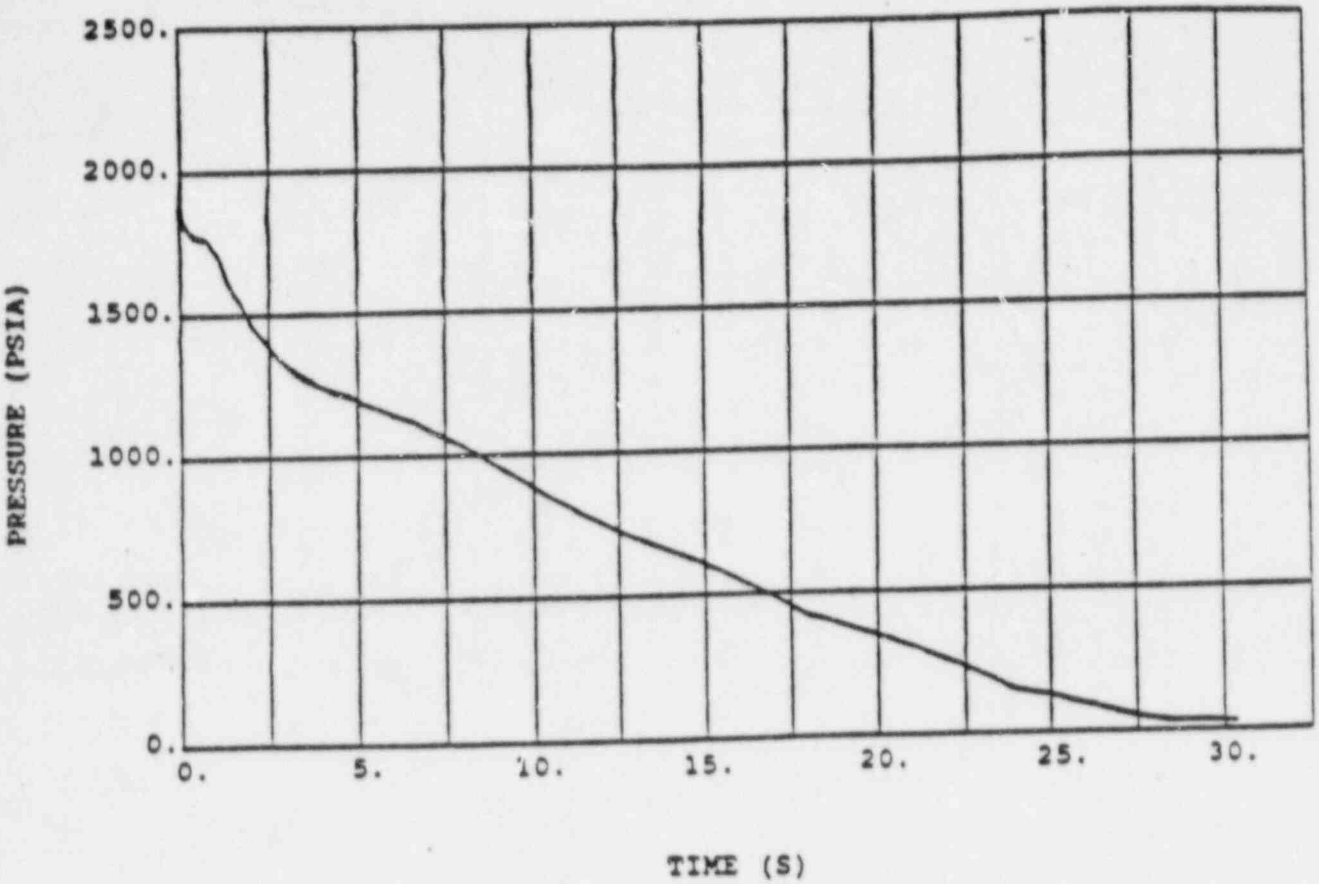

Figure 15.4-49
REACTOR COOLANT SYSTEM PRESSURE -
DECLG (C_D = 0.4, MAX SI)
Virgil C. Summer Nuclear Station
Final Safety Analysis Report

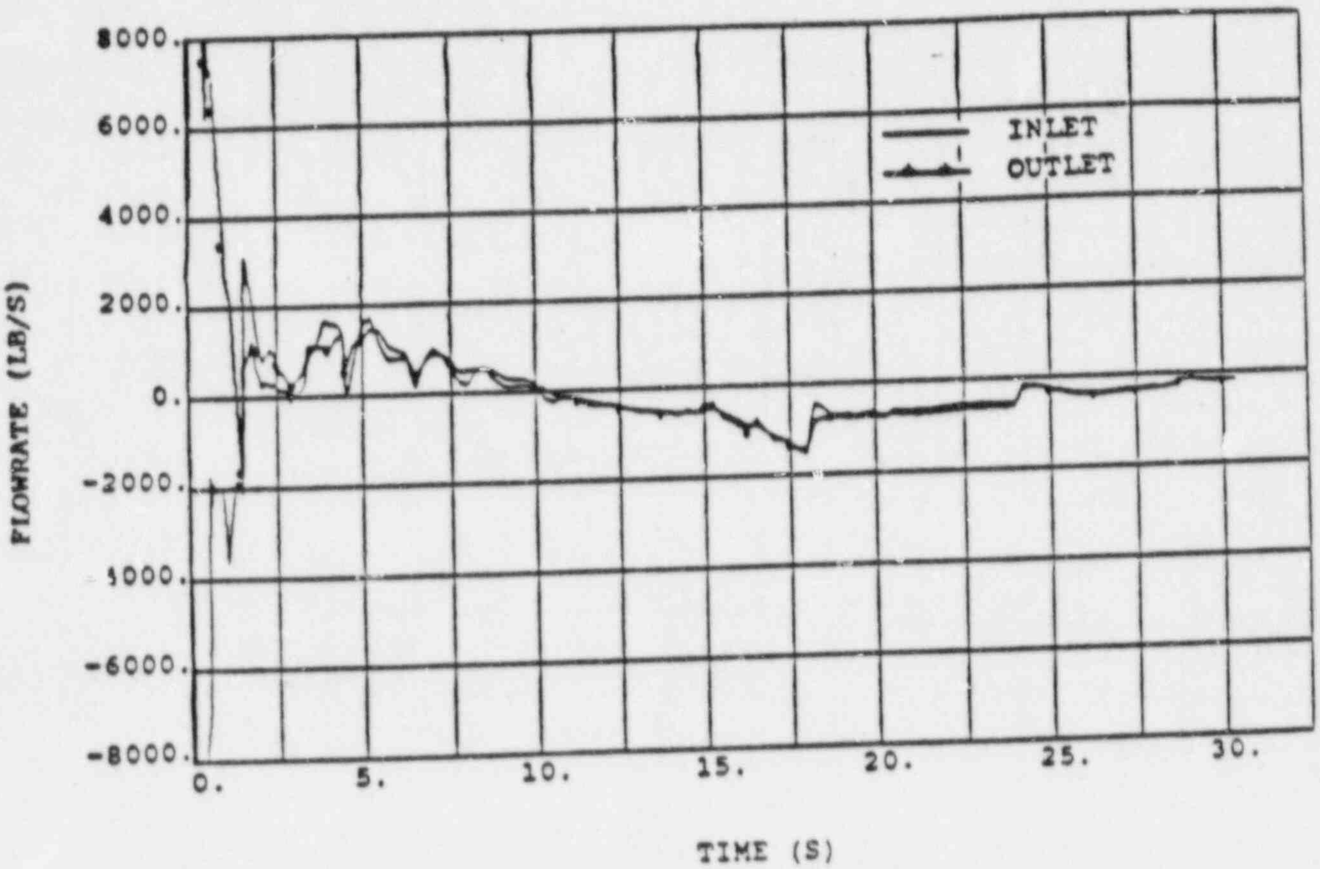

Figure 15.4-50
CORE FLOWRATE -
DECLG (C_D = 0.4, MAX SI)
Virgil C. Summer Nuclear Station
Final Safety Analysis Report

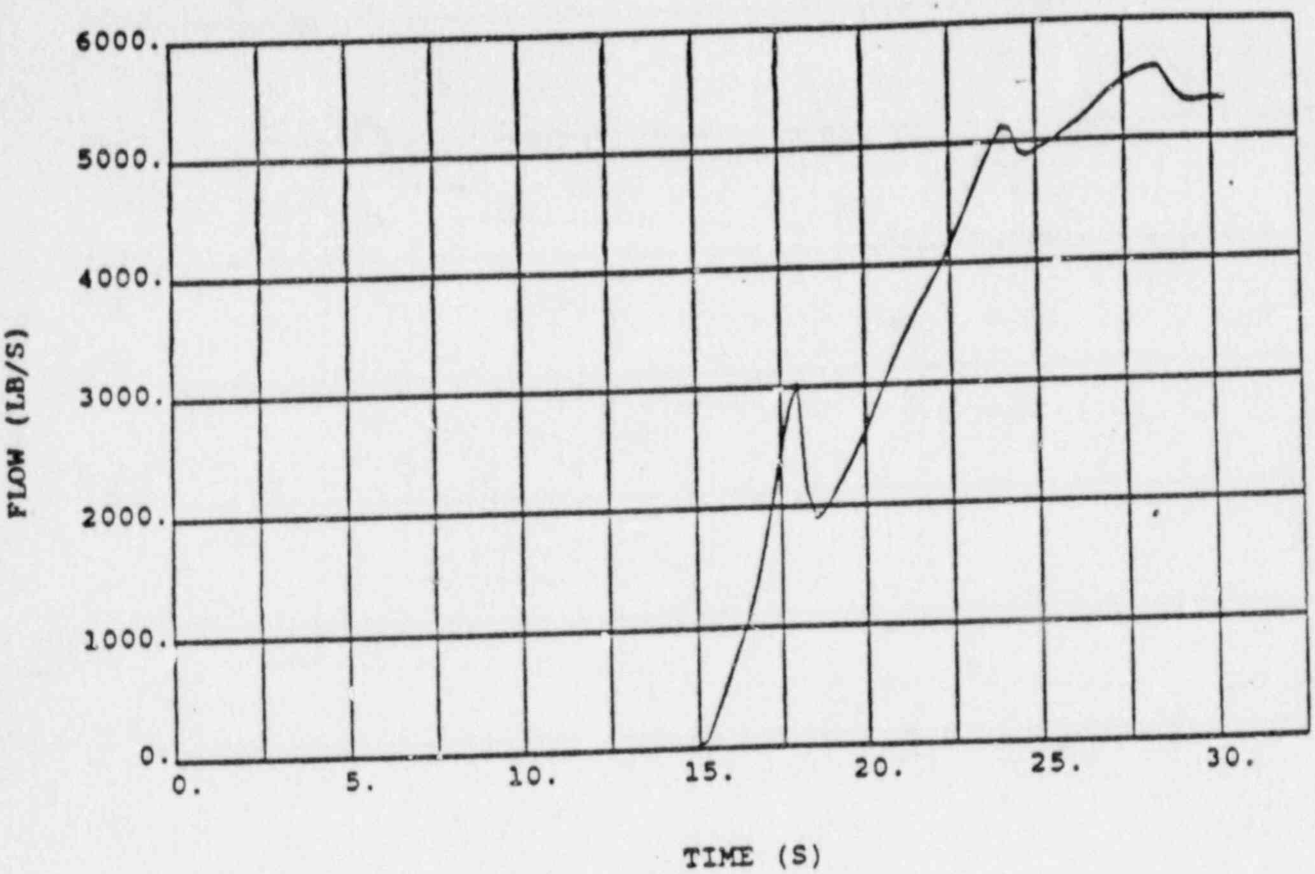

Figure 15.4-51
ACCUMULATOR FLOW DURING BLOWDOWN -
DECLG (C_D = 0.4, MAX SI)
Virgil C. Summer Nuclear Station
Final Safety Analysis Report

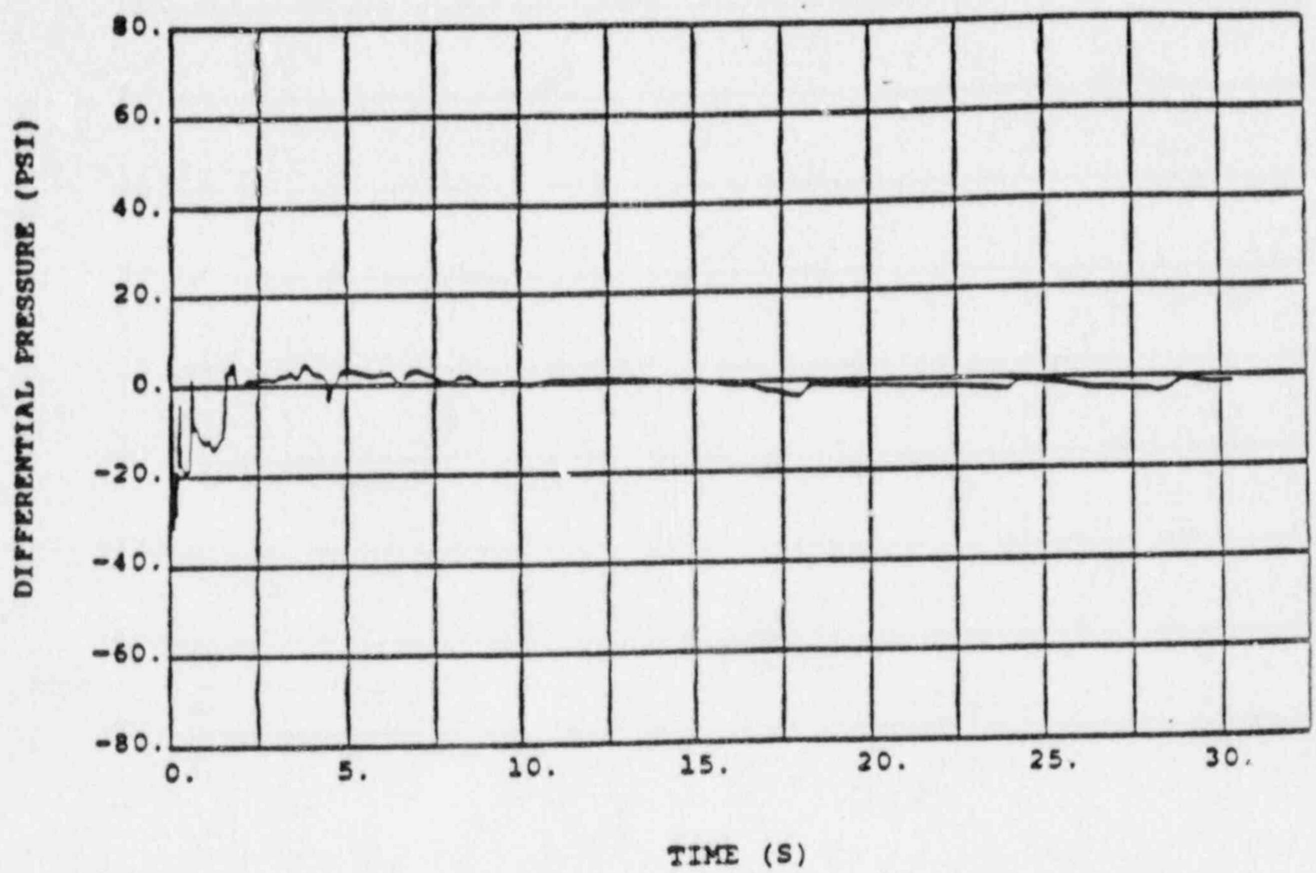

Figure 15.4-52
CORE PRESSURE DROP -
DECLG (C_D = 0.4, MAX SI)
Virgil C. Summer Nuclear Station
Final Safety Analysis Report

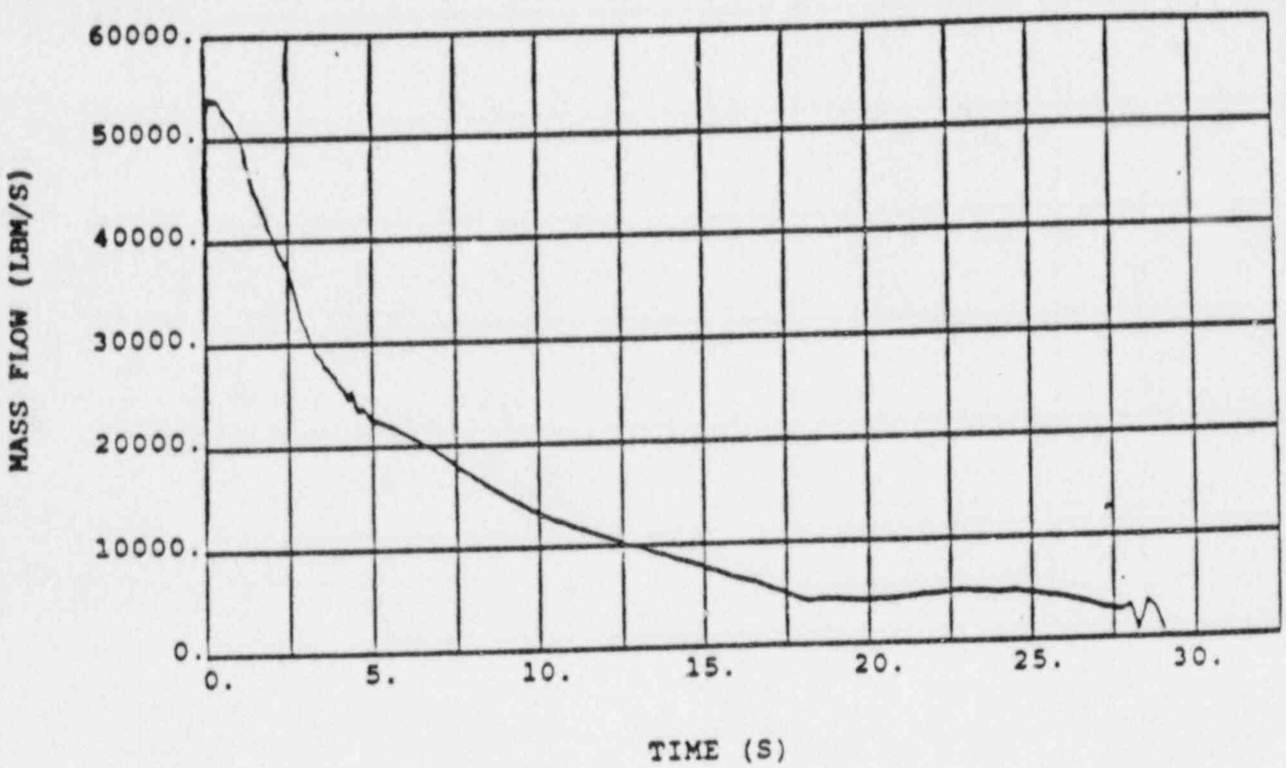

Figure 15.4-53
BREAK FLOW DURING BLOWDOWN -
DECLG (C_D = 0.4, MAX SI)
Virgil C. Summer Nuclear Station
Final Safety Analysis Report

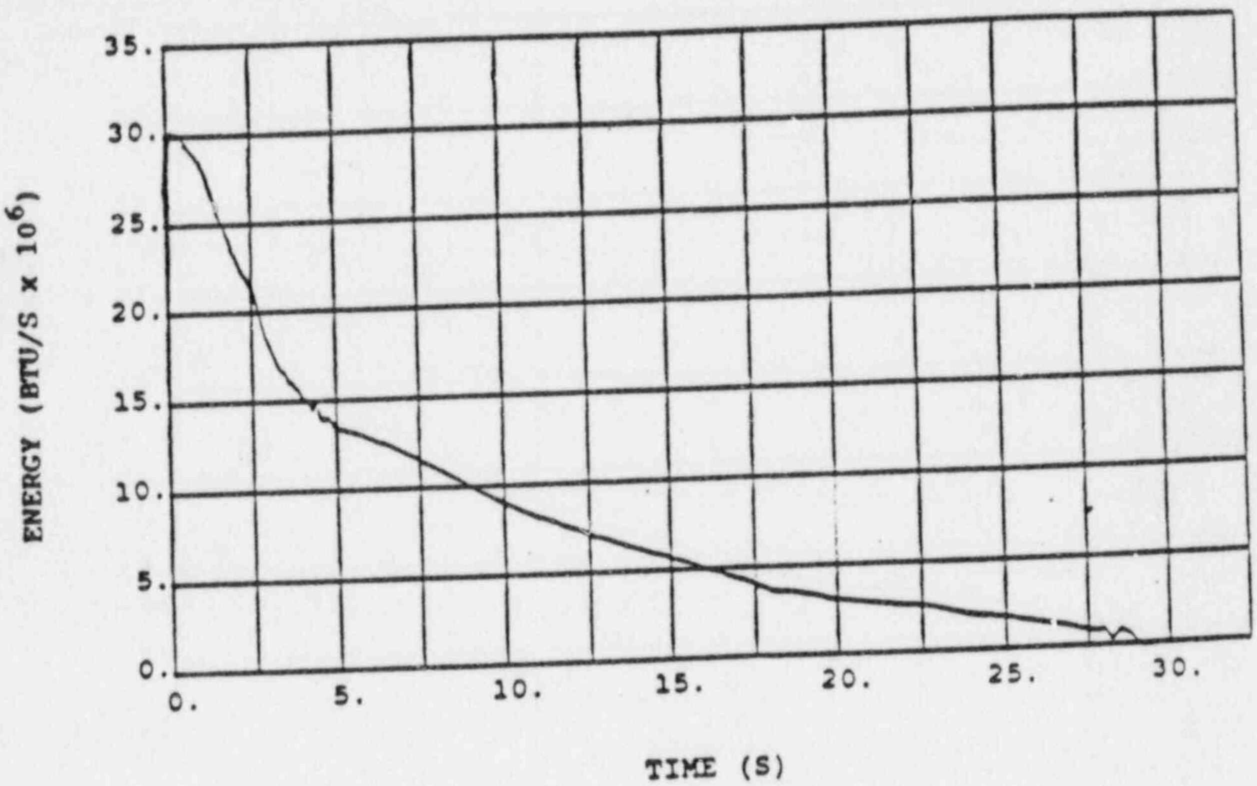

Figure 15.4-54
BREAK ENERGY DURING BLOWDOWN - DECLG (C_D = 0.4, MAX SI)
Virgil C. Summer Nuclear Station
Final Safety Analysis Report

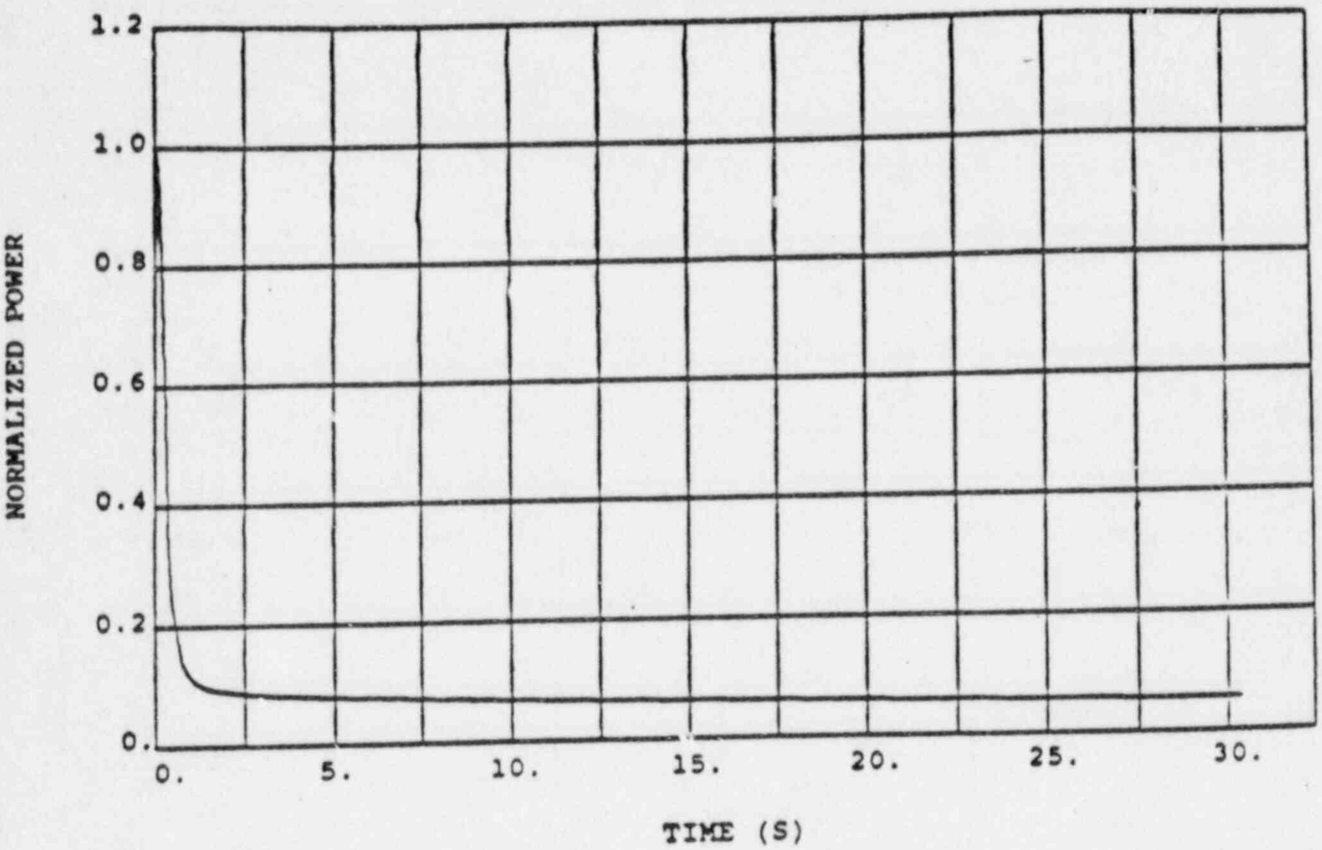

Figure 15.4-55
CORE POWER -
DECLG (C_D = 0.4, MAX SI)
Virgil C. Summer Nuclear Station
Final Safety Analysis Report

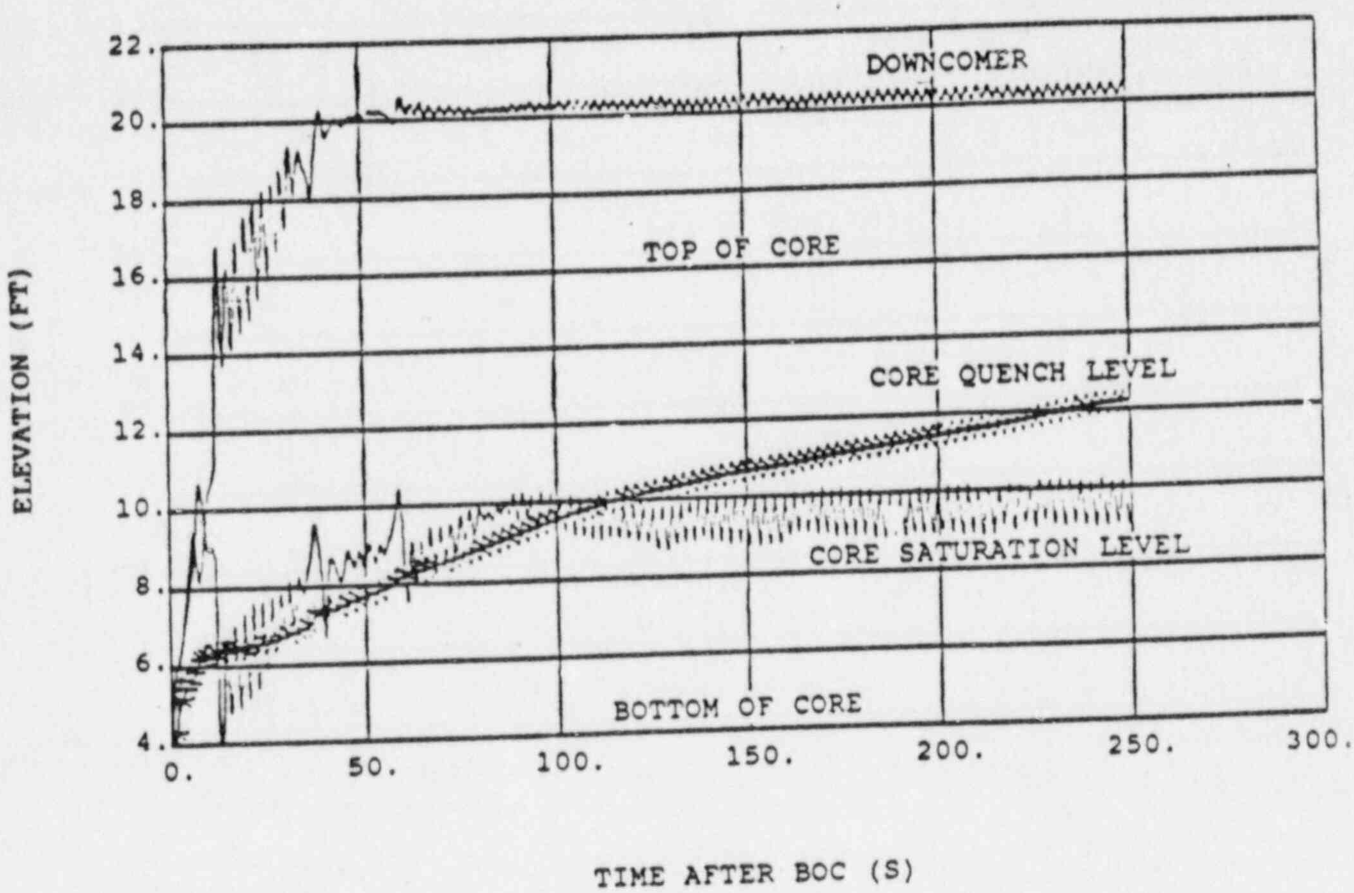

Figure 15.4-56
CORE AND DOWNCOMER LIQUID LEVELS
DURING REFLOOD
DECLG (C_D = 0.4, MAX SI)
Virgil C. Summer Nuclear Station
Final Safety Analysis Report

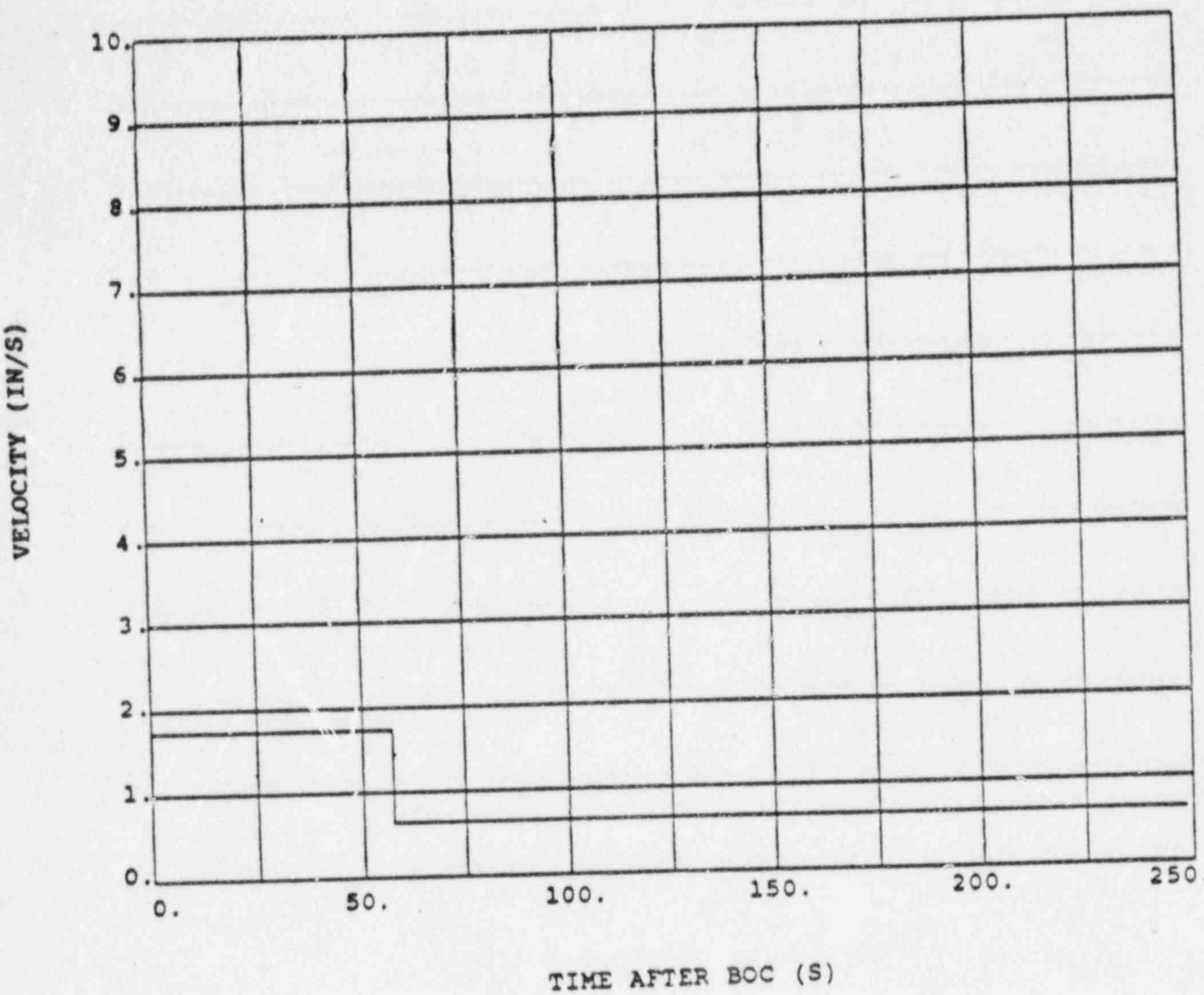

Figure 15.4-57
CORE INLET FLUID VELOCITY FOR ROD THERMAL ANALYSIS
DECLG (C_D = 0.4, MAX SI)
Virgil C. Summer Nuclear Station
Final Safety Analysis Report

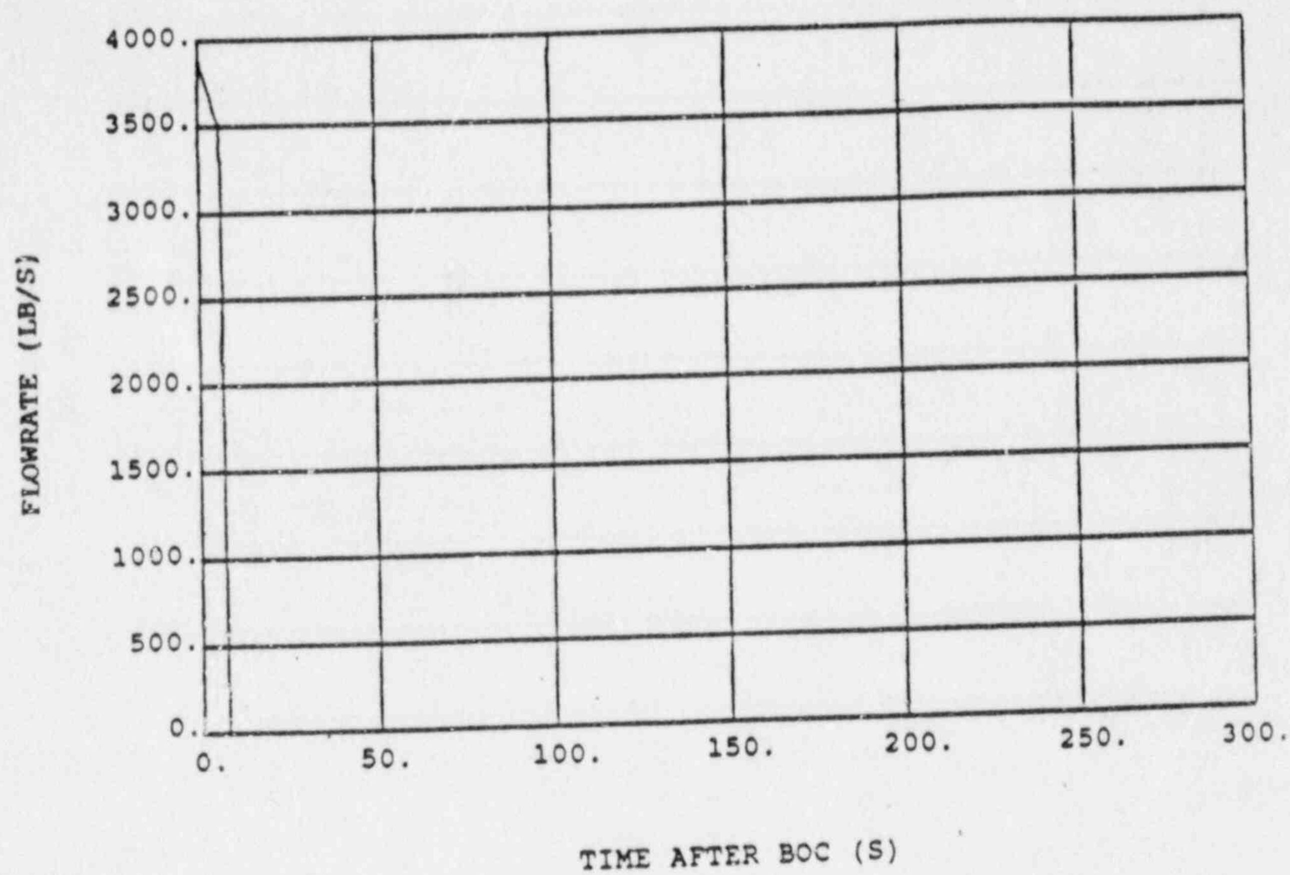

Figure 15.4-58
ACCUMULATOR FLOW DURING REFLOOD
DECLG (C_D = 0.4, MAX SI)
Virgil C. Summer Nuclear Station
Final Safety Analysis Report

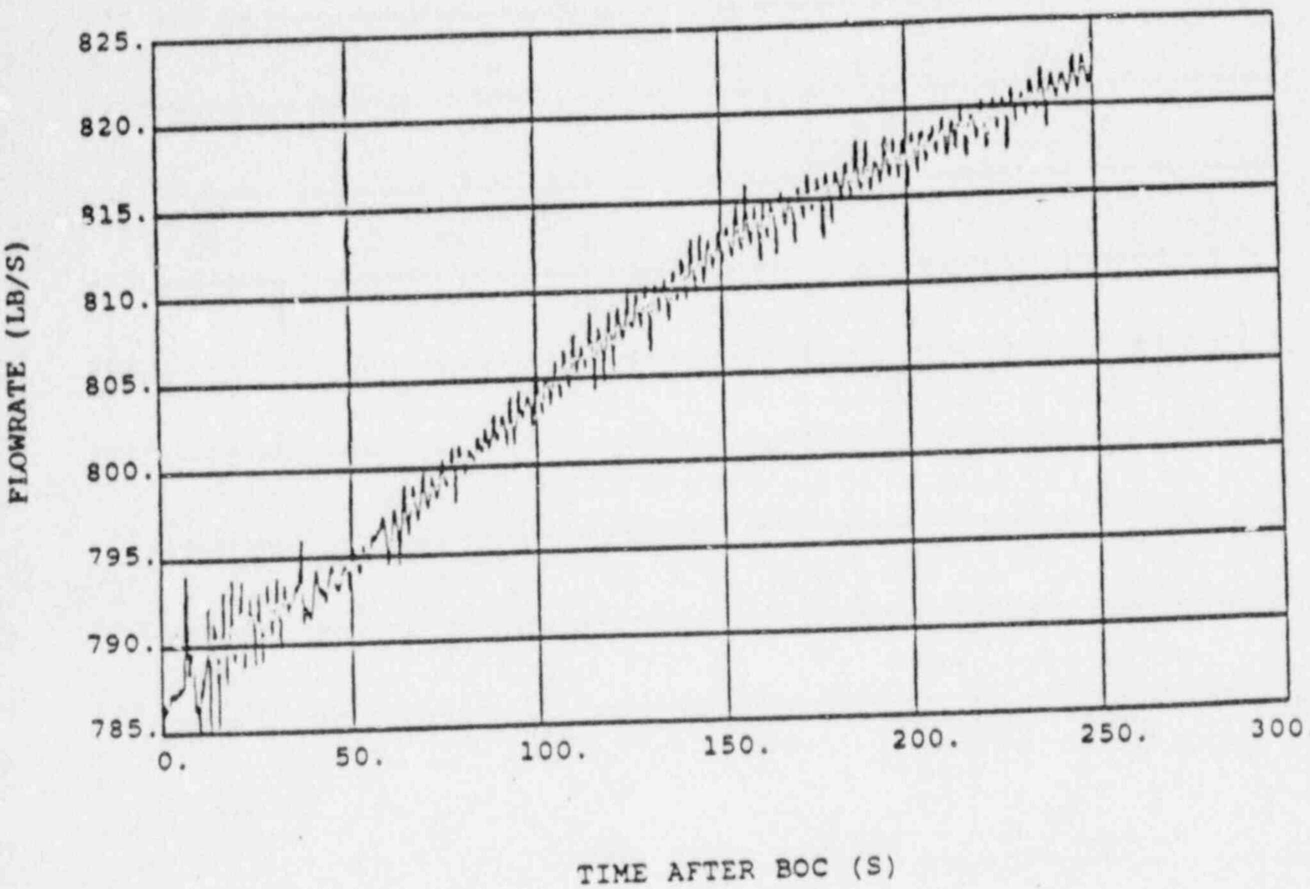

Figure 15.4-59
SI FLOW DURING REFLOOD
DECLG (C_D = 0.4, MAX SI)
Virgil C. Summer Nuclear Station
Final Safety Analysis Report

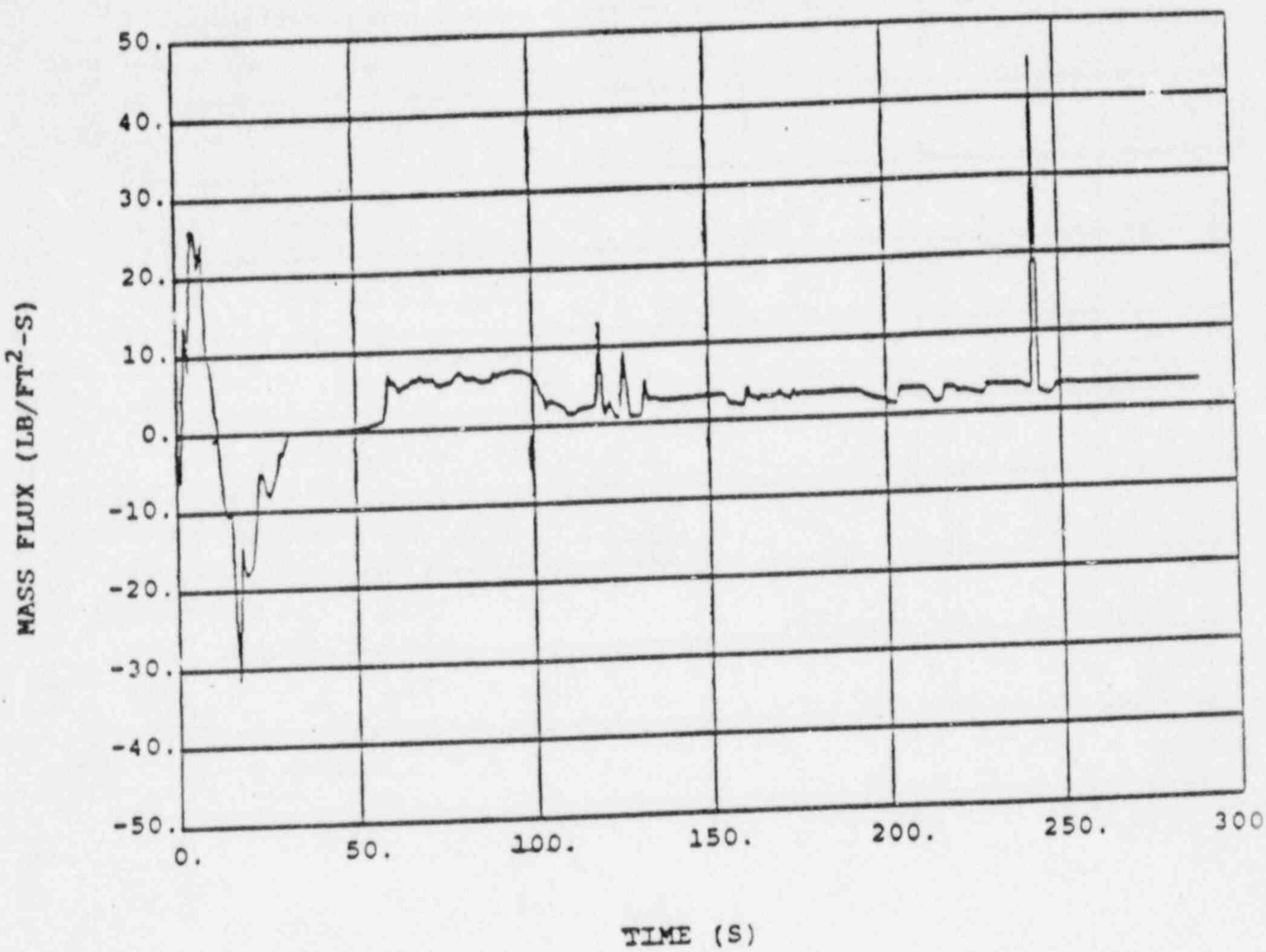

Figure 15.4-60
MASS FLUX AT THE PEAK ROD
TEMPERATURE ELEVATION
DECLG (C_D = 0.4, MAX SI)
Virgil C. Summer Nuclear Station
Final Safety Analysis Report

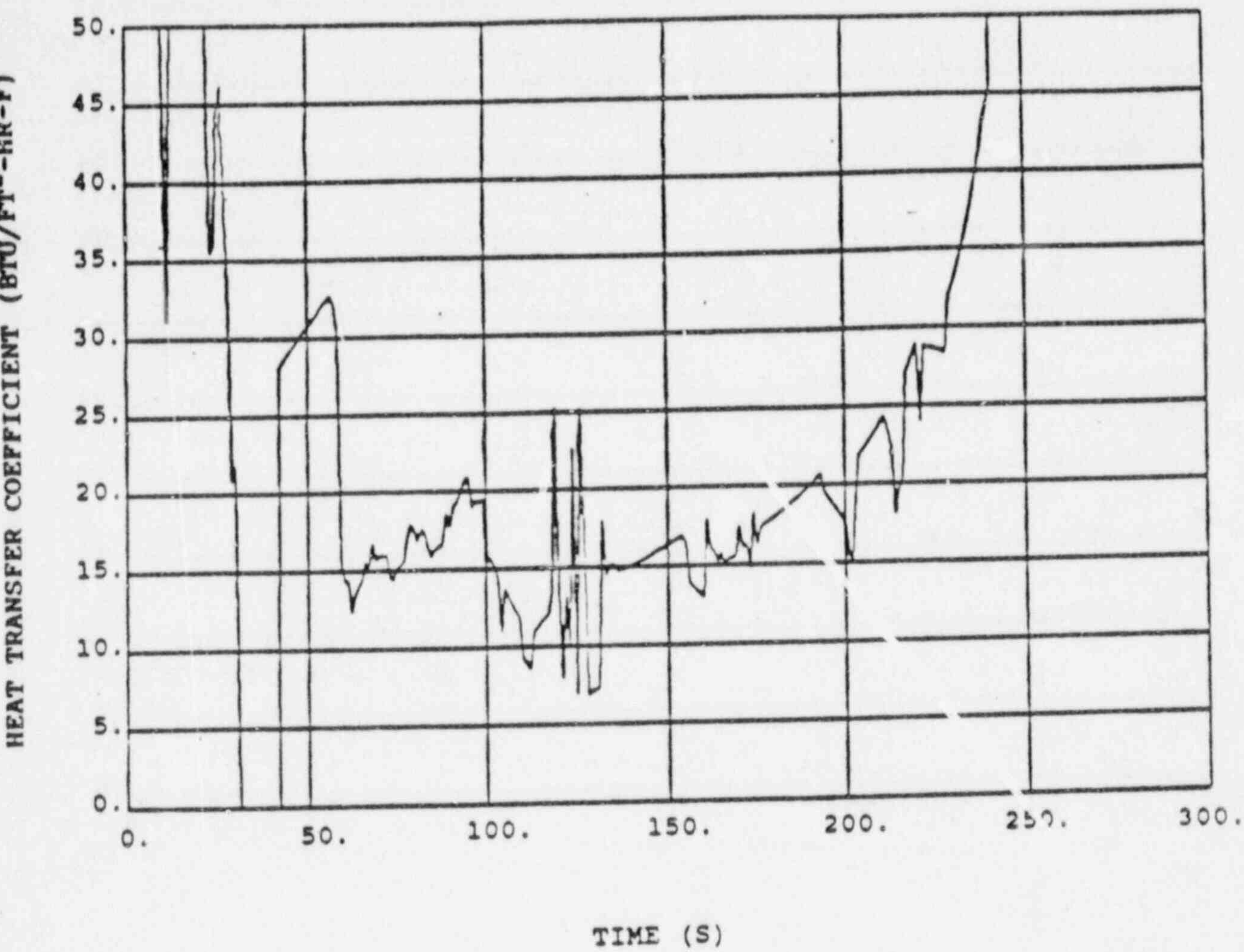

Figure 15.4-61
ROD HEAT TRANSFER COEFFICIENT AT THE PEAK TEMPERATURE LOCATION
DECLG (C_D = 0.4, MAX SI)
Virgil C. Summer Nuclear Station
Final Safety Analysis Report

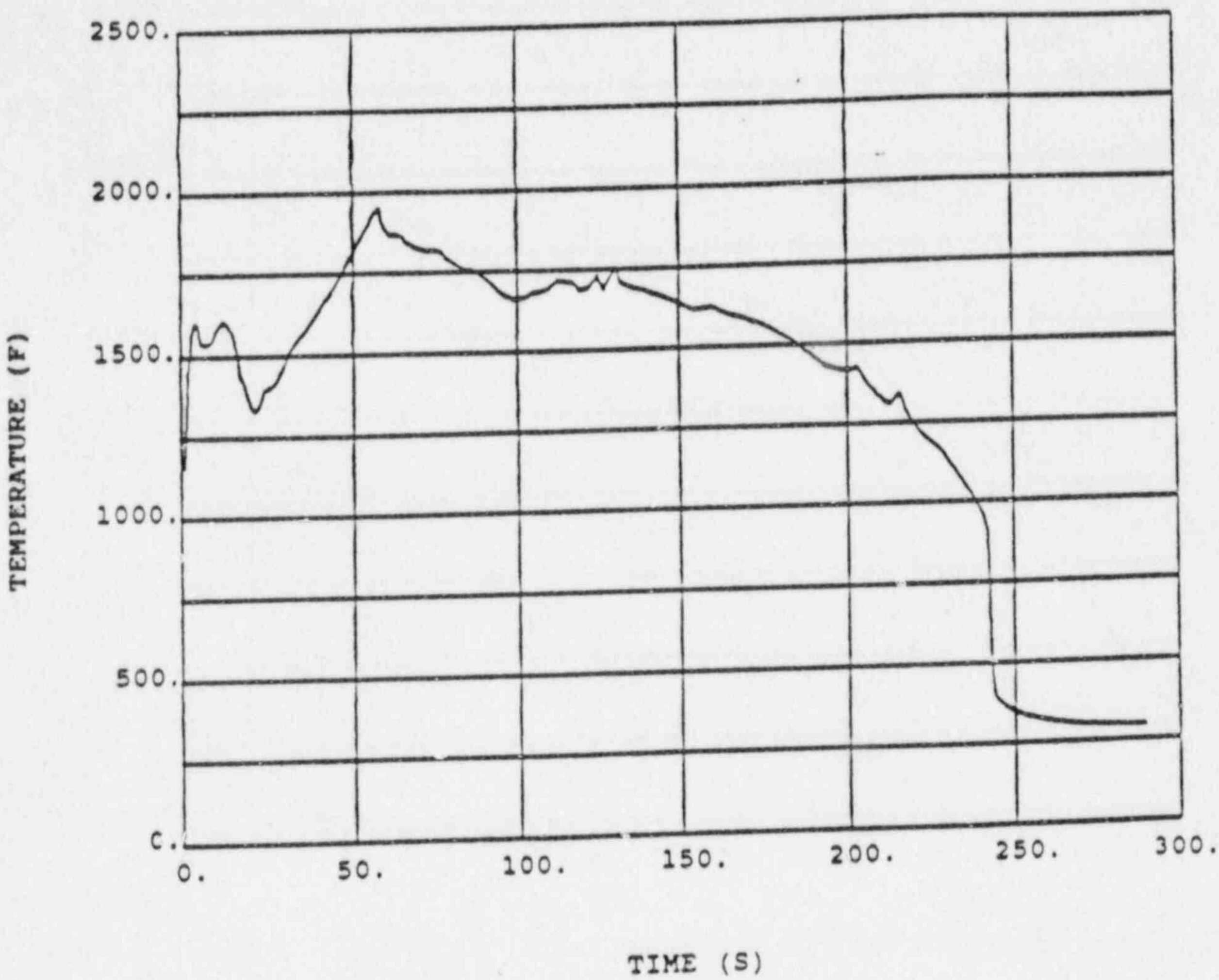

Figure 15.4-62
FUEL ROD PEAK CLAD TEMPERATURE
DECLG (C_D = 0.4, MAX SI)
Virgil C. Summer Nuclear Station
Final Safety Analysis Report

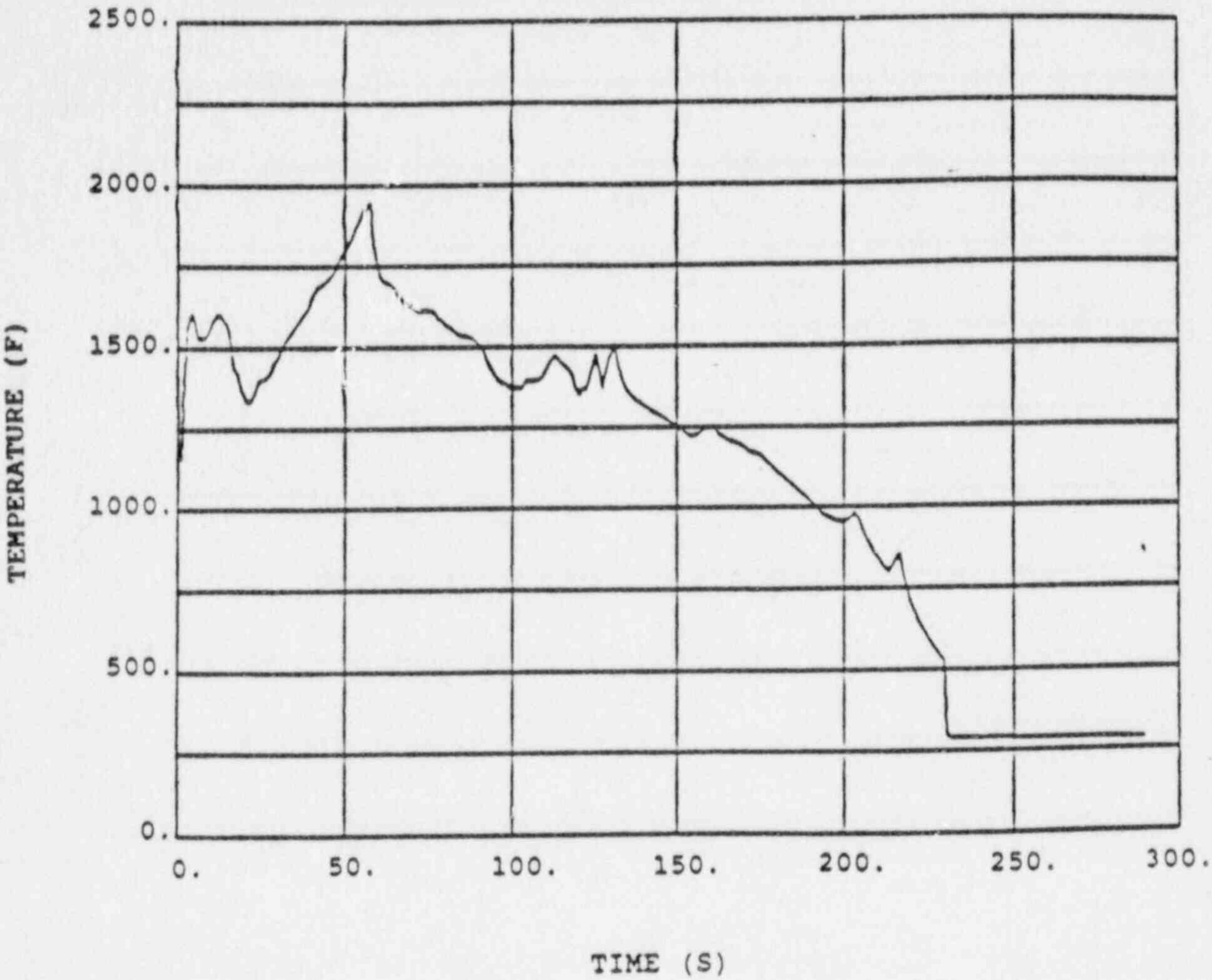

Figure 15.4-63
CLAD TEMPERATURE AT THE BURST NODE
DECLG (C_D = 0.4, MAX SI)
Virgil C. Summer Nuclear Station
Final Safety Analysis Report

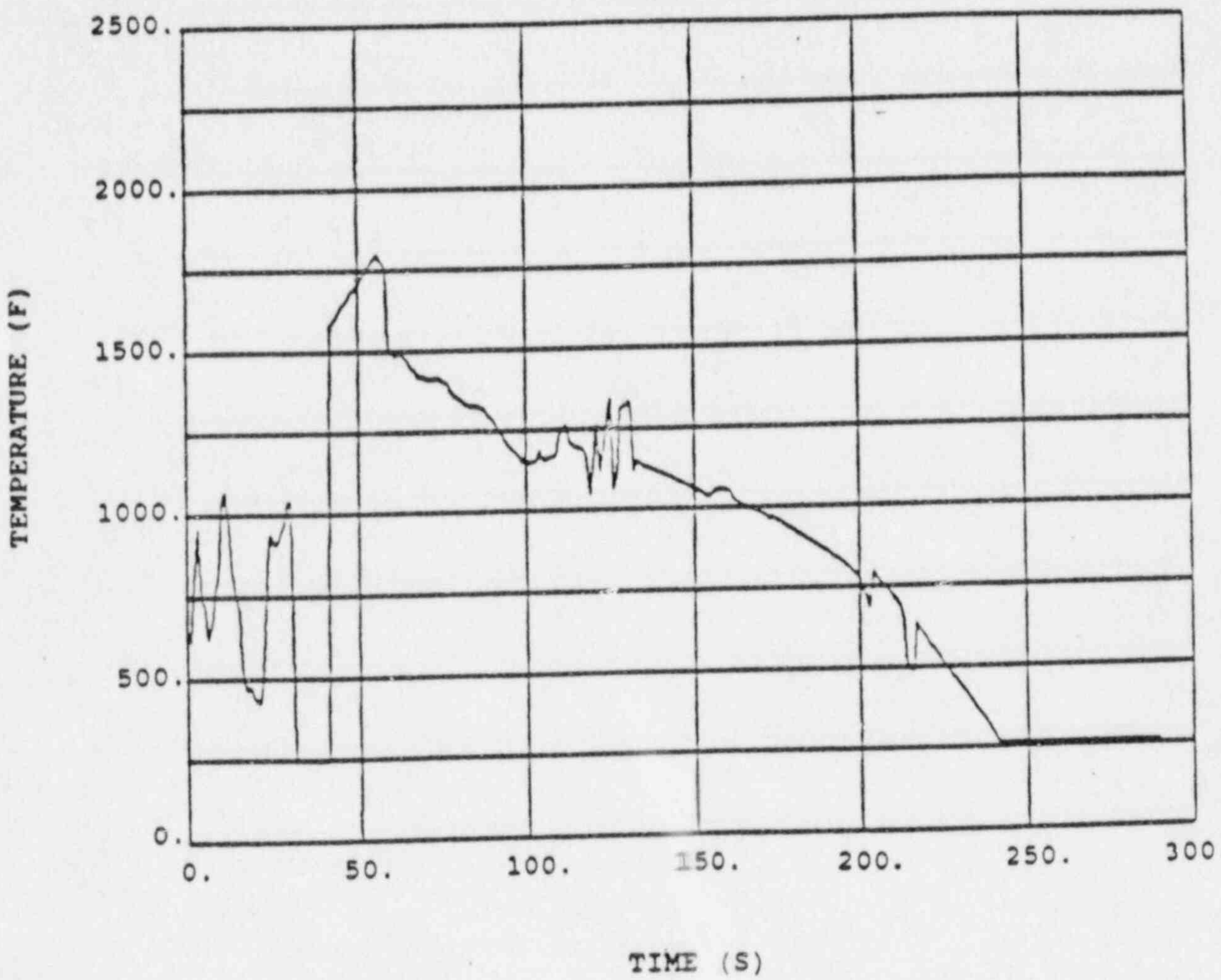

Figure 15.4-64
FLUID TEMPERATURE
DECLG (C_D = 0.4, MAX SI)
Virgil C. Summer Nuclear Station
Final Safety Analysis Report

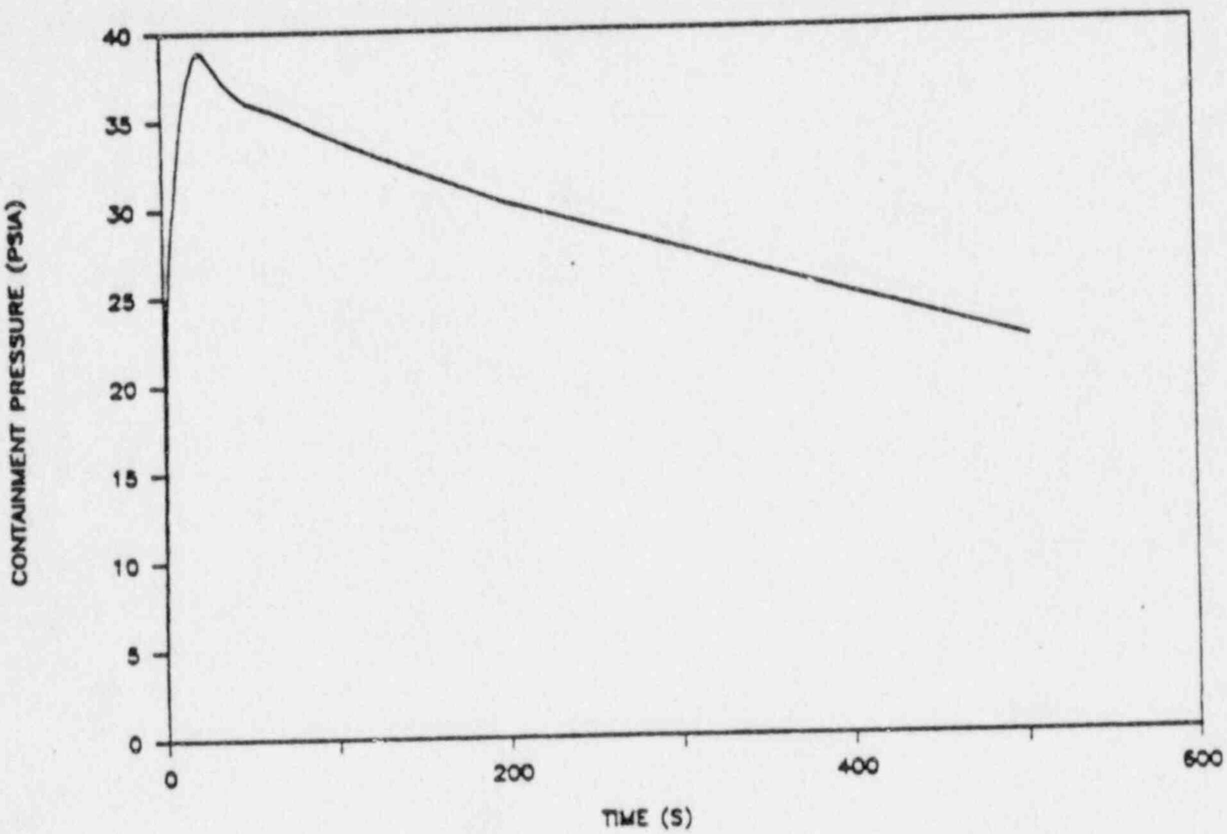

Figure 15.4-65
CONTAINMENT PRESSURE -
DECLG (C_D = 0.4, MIN SI)
Virgil C. Summer Nuclear Station
Final Safety Analysis Report

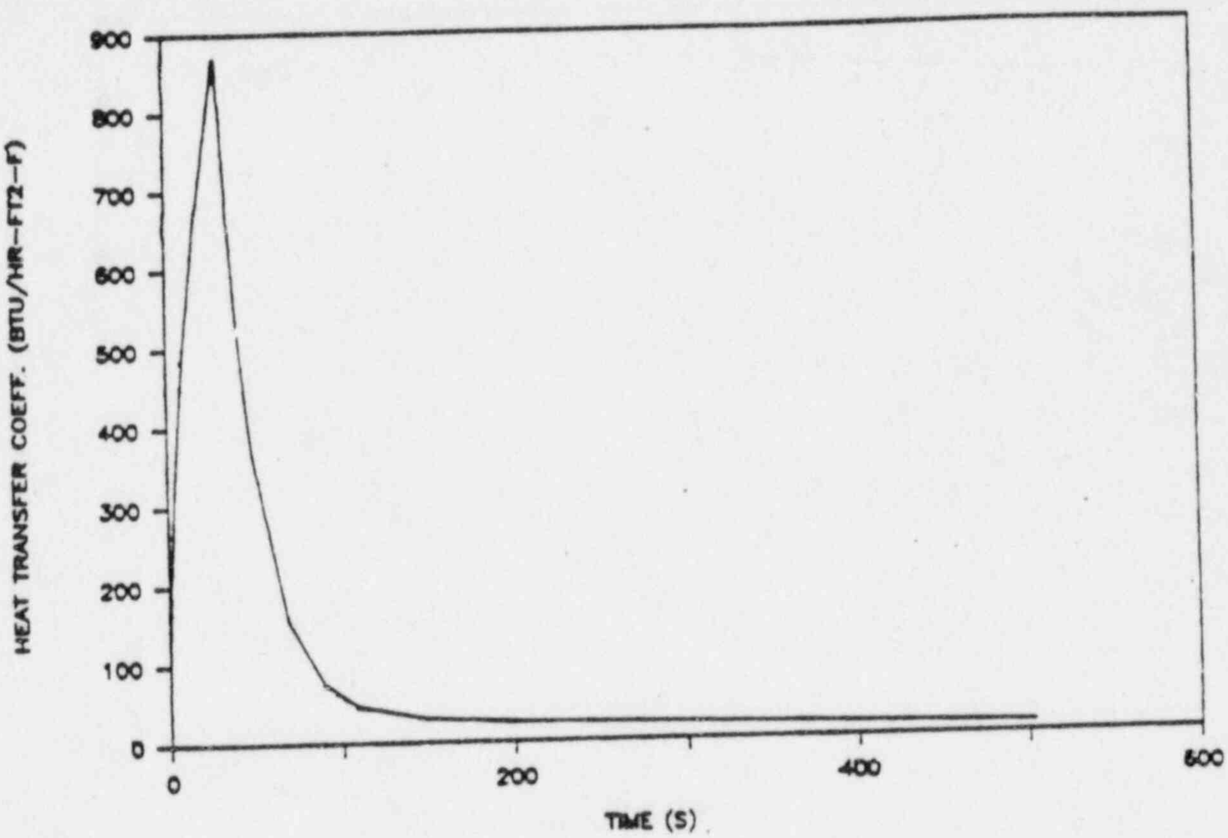

Figure 15.4-66
CONTAINMENT WALL CONDENSING HEAT
TRANSFER COEFFICIENT -
DECLG (C_D = 0.4, MIN SI)
Virgil C. Summer Nuclear Station
Final Safety Analysis Report

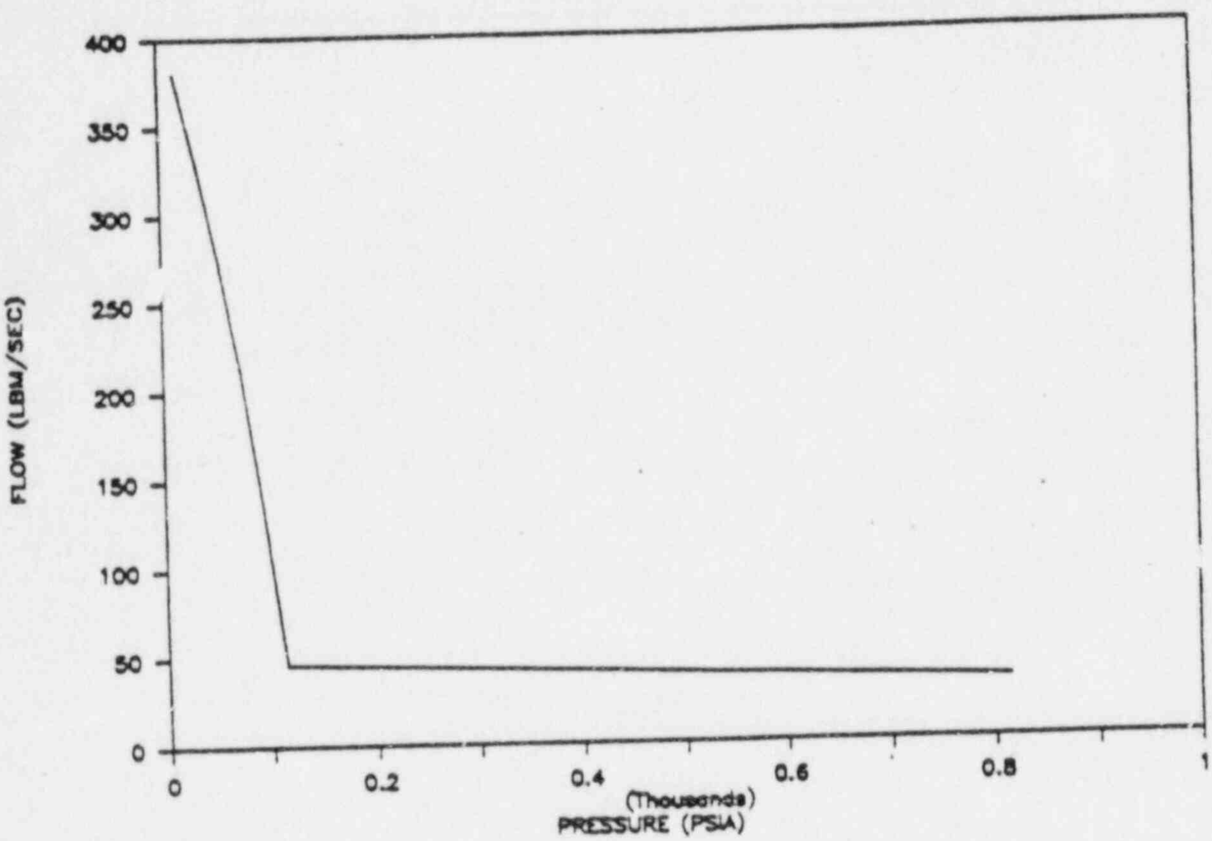

Figure 15.4-67
SAFETY INJECTION FLOW RATE, MIN SI
Virgil C. Summer Nuclear Station
Final Safety Analysis Report

ATTACHMENT 5

THIMBLE PLUG REMOVAL

EVALUATION

FOR THE

V. C. SUMMER PLANT

TRANSITION TO 17x17 VANTAGE 5 FUEL

THIMBLE PLUG REMOVAL EVALUATION

1.0 INTRODUCTION AND SUMMARY

Coincident with the fuel transition to the first region of VANTAGE 5, it is planned to remove thimble plugging devices from the core. This includes the removal of thimble plugs from the VANTAGE 5 assemblies, LOPAR assemblies, and all new core component clusters (burnable absorbers and sources).

Thimble plugging devices are currently utilized in the V. C. Summer reactors to limit the core bypass flow. All guide thimble tubes that are not under RCC locations or are not equipped with sources and burnable absorbers currently have thimble plugs inserted in them. A net gain of around 2% in DNBR margin is realized due to their presence.

Westinghouse has evaluated the effect of thimble plug removal and has concluded that it is feasible to remove all or any combination of these devices from the V. C. Summer core. This evaluation is described in the following sections and addresses the effect of thimble plug removal only unless explicitly stated otherwise.

2.0 THERMAL HYDRAULIC DESIGN EVALUATION

2.1 Bypass Flow

The main impact of thimble plug removal is the increase in core bypass flow. Calculations performed by Westinghouse have shown that the design value of core bypass flow needs to increase from 6.4% to 8.9% (non-ITDP). This increase is due to the combined effect of thimble plug removal and the slightly higher pressure drop of the VANTAGE 5 fuel assembly. The safety analysis has been performed assuming that thimble plugs have been removed from the core.

2.2 Primary System Flow Rate

Thimble plugs removal also results in a reduction to the fuel assembly hydraulic loss coefficient. Westinghouse has performed tests to quantify the magnitude of this effect. Based on these tests, it is estimated that there will be a slight increase in primary system flow rate due to thimble plug removal from the V. C. Summer core. No mechanical design criteria are impacted by this slight increase in flow rate.

2.3 Fuel Assembly Hydraulic Lift

The hydraulic lift force on the fuel assembly can be represented by the following function:

$$F_{LIFT} \alpha K_{FA} \times (\text{core flow})^2$$

Westinghouse has performed hydraulic tests to quantify the magnitude of the effect of thimble plug removal on fuel assembly hydraulic loss coefficient (K_{FA}). The results show that there is a net reduction in F_{LIFT} due to a reduced fuel assembly loss coefficient (caused by thimble plug removal) which more than compensates for the slight increase in vessel flow rate. Thimble plug removal is therefore acceptable from a fuel assembly lift standpoint.

2.4 Effect of Outlet Hydraulic Mismatch on DNB

Current DNB analyses are performed assuming the presence of an uniform static pressure distribution at the core outlet, even though pressure gradients and core outlet loss coefficient mismatches are known to exist. This is acceptable because these mismatch effects do not propagate upstream into the DNB zone. Westinghouse has performed numerous sensitivity studies to demonstrate the insensitivity of as calculated DNBR's to non-uniform outlet pressure distributions and to variations in outlet loss coefficients.

The effect of thimble plug removal on the corewide distribution of outlet loss coefficients for the V. C. Summer cores has been evaluated. It was demonstrated that the variations in outlet loss coefficient due to thimble plug removal are within the bounds of the sensitivity studies that had been performed. Therefore, it is concluded that thimble plug removal will not result in the reduction of DNBR margin due to mismatches in core outlet pressure gradients and loss coefficients.

3.0 MECHANICAL DESIGN EVALUATION

3.1 Fuel Rod Fretting Wear

The removal of thimble plugging devices changes the distribution of core outlet loss coefficients. The core outlet loss coefficient (PFO) distribution shows an increase in PFO mismatch after thimble plug removal. Therefore, the issue of crossflow induced fuel rod vibration and wear due to this increased PFO mismatch is addressed.

The maximum PFO mismatch that exists in the V. C. Summer core after removal of all or any combination of thimble plugs is less than 1.5. Westinghouse, however, has recently performed fuel rod vibration tests with a PFO mismatch of approximately 17 between two 17x17 fuel assemblies. The results showed that there was no significant difference in fuel rod response between the tests performed with and without this large PFO mismatch. Therefore, it is concluded that thimble plug removal will not have a detrimental effect on fuel rod vibration and wear.

3.2 Control Rod Wear

Westinghouse studies on control rod wear have shown that most of the wear tends to be in the upper internals region. When thimble plugs are removed the hydraulic resistance at the outlet for these assemblies is reduced. This in turn causes the flow through the RCCA guide tubes to be reduced, because more flow is now going through the outlet of the assemblies which were previously fitted with thimble plugs. This reduction of flow through the RCCA guide tubes is in the direction that would tend to reduce control rod wear.

However, since the core PFO distribution changes when thimble plugs are removed, the effect of potential control rod vibration due to inter assembly crossflows in the region of the control rod/fuel assembly guide thimble interface needs to be addressed. The control rods can be directly affected in the core region only by inter assembly crossflows through the gap (~0.75") between the top nozzle and upper core plate. For the V. C. Summer reactor upper internals configuration, it was concluded that the maximum PFO mismatch between an RCC location and an adjacent assembly does not increase with thimble plug removal. Therefore, the magnitude of the crossflow seen by the control rods and the vibration of the rods caused by this crossflow will not be increased.

Based on the above evaluation, thimble plug removal will not have an adverse impact on control rod wear for the V. C. Summer reactor.

3.3 Seismic/LOCA Transient Loading

The thimble plugging device (approximately 11 lbs.) contributes a very small percentage of the total fuel assembly weight. Therefore, the removal of these devices will have a negligible effect on the fuel structural responses to seismic or LOCA transient loading.

3.4 Reactor Internals Structural Adequacy

There is a negligible impact of thimble plug removal on the internals time history analysis. Also, the effect of thimble plug removal on increased RCS flow will have a negligible effect on the structural adequacy of the internals. Thimble plug removal is therefore acceptable from a Reactor Internals standpoint.

4.0 SAFETY ANALYSIS

The non-LOCA analyses (Attachment 3) and LOCA analysis (Attachment 4) have conservatively assumed that all thimble plugs were removed from the core to produce the maximum core bypass flow and core consequences.

5.0 CONCLUSION

Detailed evaluations have shown that the main effect of thimble plug removal is the increase in core bypass flow. This increase has been incorporated into the non-LOCA and LOCA safety analyses that have been performed in support of the VANTAGE 5/LOPAR fuel transition cores

Based on the assessment of the impact of the thimble plug removal on system and component structural adequacy and core plant safety, it is concluded that it is acceptable to remove all or any combination of these devices from the V. C. Summer core(s). The evaluation also bounds the use of any combination of dually compatible thimble plugs, WABAs and source rods.

ATTACHMENT 6

SIGNIFICANT HAZARDS EVALUATION

FOR THE V. C. SUMMER PLANT

TRANSITION TO WESTINGHOUSE 17x17 VANTAGE 5 FUEL ASSEMBLIES

SIGNIFICANT HAZARDS EVALUATION FOR
VIRGIL C. SUMMER NUCLEAR STATION TRANSITION TO
WESTINGHOUSE VANTAGE 5 FUEL

Description of amendment request:

South Carolina Electric & Gas Company (SCE&G) requests an amendment to the Virgil C. Summer Nuclear Station (VCSNS) Technical Specifications to support (a) the Cycle 5 core reload to permit operation with Westinghouse VANTAGE 5 (V-5) fuel assemblies in addition to the Westinghouse Low Parasitic (LOPAR) assemblies remaining in the core from Cycle 4 and (b) subsequent operating cycles with up to a full core of V-5. Design features of the V-5 fuel include assemblies with up to approximately 4.25 weight percent U-235, axial blankets, integral fuel burnable absorbers, intermediate flow mixers, reconstitutable top nozzles, and extended burnup capability. A debris filter bottom nozzle (DFBN) will be introduced to replace the standard V-5 bottom nozzle to reduce the possibility of fuel rod damage due to debris-induced fretting. This requires changes to the Technical Specifications due to the use of the V-5 fuel and use of the following analytical methods and assumptions.

- o The Improved Thermal Design Procedure (ITDP)
- o The WRB-1 and WRB-2 departure from nucleate boiling (DNB) correlations
- o The BASH large break loss-of-coolant accident (LOCA) model
- o The NOTRUMP small break LOCA model
- o The ANSI/ANS 5.1-1979 decay heat model for non-LOCA accidents and transients
- o Relaxed Axial Offset Control (RAOC)
- o $F_Q(z)$ Surveillance
- o Analysis baseline changes as outlined in Table 1

As a result of the above, changes to the following Technical Specifications and corresponding bases, as appropriate, are proposed:

- o Core Safety Limits
- o Reactor Coolant Flow Allowable Values

- Overtemperature delta T Reactor Trip Setpoints
- Overpower delta T Reactor Trip Setpoints
- Shutdown Margin for Modes 3, 4, and 5
- Moderator Temperature Coefficient
- Rod Drop Time
- Axial Flux Difference
- Heat Flux Hot Channel Factor - $F_Q(z)$
- Nuclear Enthalpy Rise Hot Channel Factor - $F_{\Delta H}$
- DNB Parameters
- Reactor Trip(s) Response Time
- ECCS Accumulator water volume range
- Borated Water Sources for Modes 1-4
- Reactor Trips and Emergency Safety Features Actuation System Drift Allowances for Determination of Operability
- Charging Pump Flow Balance Surveillance

Basis for proposed no significant hazards consideration determination:

SCE&G has evaluated the proposed changes associated with the transition to V-5 fuel against the Significant Hazards Criteria of 10CFR50.92 and against the Commission guidance concerning application of this standard. VCSNS's proposed license amendment is closely related to an example (51 FR 7751) of action not likely to involve a significant hazard. Specifically, example (iii) of the guidance states:

"For a nuclear power reactor, a change resulting from a nuclear reactor core reloading, if no fuel assemblies significantly different from those found previously acceptable to the NRC for a previous core at the facility in question are involved. This assumes that no significant changes are made to the acceptance criteria for the Technical Specifications, that the analytical methods used to demonstrate conformance with Technical Specifications and regulations are not significantly changed, and the NRC has previously found such methods acceptable."

The VCSNS proposed licensing amendment is directly related to the above example in that the core reload uses V-5 fuel which is not significantly different from previous cores at VCSNS, the changes to the Technical Specifications are as a result of the core reload and not because of any significant change made to the acceptance criteria for Technical Specifications, and the analytical methods used in the required reload analysis have been previously found acceptable by the NRC. Therefore, based on the above, SCE&G concludes that the proposed Technical Specifications changes do not involve a significant hazard consideration.

SCE&G has evaluated the proposed changes in design, analytical methodologies and Technical Specifications associated with the transition to V-5 fuel against the Significant Hazards Criteria of 10CFR50.92. The results of SCE&G's evaluations demonstrate that the changes do not involve any significant hazard as described below.

a. The probability or consequences of an accident previously evaluated is not significantly increased.

   The V-5 reload fuel assemblies are mechanically and hydraulically compatible with the current LOPAR fuel assemblies, control rods and reactor internals interfaces. Also, implementation of V-5 does not cause a significant change in the physics characteristics of the VCSNS cores beyond the normal range of variation seen from cycle to cycle. Thus, both fuel types satisfy the design basis for VCSNS as proposed for this amendment.

   Thimble plug removal has a negligible impact on the system and component structural adequacy but does cause core bypass flow to increase. The revised core thermal-hydraulic design and safety analysis, however, show that the DNB penalty due to removal of the thimble plugs is more than offset by the increase in DNB margin resulting from the use of the ITDP and V-5 fuel.

The proposed changes have been assessed from a core design and safety analysis standpoint. No increase in the probability of occurrence of any accident was identified but an extensive reanalysis, as described in the Transition Safety Evaluation, was required to demonstrate compliance to the revised VCSNS Technical Specifications as proposed herein. These reanalyses applied methods which have been previously found acceptable by the NRC. The results, which include transition core effects, show changes in consequences of accidents previously analyzed. However, the results are all clearly within pertinent acceptance criteria and demonstrate the plants capability to operate safely at 100% power. Thus, it is concluded that there is not significant increase in the consequences of an accident previously evaluated.

b. The possibility for an accident or malfunction of a different type than any evaluated previously in the safety analysis reports is not created.

These proposed changes do not significantly effect the overall method and manner of VCSNS operation and can be accommodated without compromising the performance or qualification of safety-related equipment. Thus, the creation of a new or different kind of accident from any previously evaluated accident is not considered a possibility.

c. The margins of safety as defined in the bases of the Technical Specifications is not significantly reduced.

The evaluations and analyses described herein show some changes in the consequences of previously analyzed accidents. In some cases, an increase in event consequences occurs and may reduce margin.
However, in all cases, the results of the changes are clearly within all pertinent design and safety acceptance criteria. Thus, there is no significant reduction in the margin of safety as a result of the proposed changes.

Table 1

VCSNS ANALYSIS BASELINE

Parameter	Current Value	Proposed Value For Vantage 5 Transition
NSS Power, MWt	2785	2787
Core Power, MWt	2775	2775
System Pressure, psia	2250	2250
Thermal Design Flow, gpm/loop	[illegible]0	92600*
Core Bypass Flow, %	[illegible].4	8.9**
T_{AVE}, °F	587.4	585.5
T_{HOT}, °F	618.7	618.7
$F_{\Delta H}$	1.55	1.62
$F_{\Delta H}$ Multipier	0.2	0.3
LOCA F_Q	2.25	2.45
SG Tube Plugging, %	16	15
AFD Control	CAOC	RAOC
Peaking Surveillance	F_{xy} (z)	F_Q (z)
High Head Safety Injection	Recirculation Isolated	Recirculation Not Isolated
Thimble Plugs	Yes	Optional

* Includes 2% additional flow margin for conservatism

** Non-ITDP

ATTACHMENT 7

RADIOLOGICAL IMPACT ASSESSMENT

FOR THE V. C. SUMMER PLANT

TRANSITION TO WESTINGHOUSE 17x17 VANTAGE 5 FUEL ASSEMBLIES

Radiological Impact Assessment

The use of VANTAGE 5 fuel in the Virgil C. Summer Nuclear Station will result in a higher discharge region average burnup. While fission product inventories are roughly proportional to operating power level, the level of fuel burnup has little impact except for isotopes with long half-lives. This is supported by the Westinghouse topical report, WCAP-10125-P-A (Proprietary), titled "Extended Burnup Evaluation of Westinghouse Fuel," which demonstrates that extension of fuel burnup to the higher discharge region average burnups evaluated in the topical report, which are bounding for V. C. Summer, would have only a small impact on the core fission product inventories. This change in fission product inventories would not significantly affect the radiological consequences of postulated accidents.

For the fuel handling accident, extending the discharge region average burnup to the maximum value evaluated in WCAP-10125-P-A would result in an increase of approximately four percent in the thyroid dose. This increase is based on continued use of the fuel handling accident analysis assumption, defined in Regulatory Guide 1.25, that ten percent of the core inventory of short-lived isotopes and thirty percent of core inventory of long-lived isotopoes are in the fuel rod gap. The short-lived isotopes are of greatest concern in regard to radiological consequences of the accident, and analysis shows that the fraction of short-lived isotopes in the fuel rod gap, when at it's maximum, would be about one percent or less of the core inventory; not the value of ten percent assumed in Regulatory Guide 1.25. Extending the fuel burnup actually will result in a reduction in the gap inventories of short-lived isotopes due to operation at lower power levels for the latter part of the residence time in the core which results in a concomitant reduced production rate for fission products. Also, with operating at a reduced power level, the fuel pellet temperature will be reduced resulting in a lower rate of diffusion of fission products into the rod gap.

The radiological consequences of the fuel handling accident are also impacted by the fact that the proposed fuel design has a F delta H (radial peaking actor) of 1.68 specified. The fuel handling accident for V. C. Summer

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utilizes the guidance of Regulatory Guide 1.25 which specifies that a minimum radial peaking factor of 1.65 be used in determining the maximum fuel assembly fission product inventory. With a radial peaking factor of 1.69, the doses reported in the FSAR would be increased by approximately two percent. The increase in calculated dose due to the combination of extended burnup and increased radial peaking factor is not significant and the doses are still well within the acceptance criteria defined in the Standard Review Plan.

The radiological consequences of accidents other than the fuel handling accident are also impacted to a slight degree. As discussed in WCAP-10125-P-A, the impact of extended fuel burnup on the consequences of a rod ejection accident would be to slightly increase the thyroid dose (about two percent) and to decrease the whole body dose. This same effect on radiological consequences would also be seen in other accidents involving release of reactor coolant activity whether or not there is any fuel damage as a result of the accident. These increases in radiological consequences are insignificant, being within the uncertainty of the calculational assumptions. Thus, there is no need to recalculate the radiological consequences of the accidents due to extending the fuel burnup within the limits of the study reported in WCAP-10125-P-A. In addition, it is noted that the radiological consequences of accidents reported in the FSAR are well within the limits of 10 CFR 100; thus, if increases such as those discussed above were applied to the FSAR doses, there would be no impact on their acceptability.

Other than the increased level of burnup and the increase in radial peaking factor discussed above, no features of the VANTAGE 5 fuel design have an impact on the radiological consequences of normal operation or of the postulated accidents.