Enclosure 1

Vogtle Electric Generating Plant Units 1 and 2

Request for Technical Specifications Changes

VANTAGE-5 Fuel Design

Basis for Proposed Changes

ENCLOSURE 1

VOGTLE ELECTRIC GENERATING PLANT
REQUEST FOR TECHNICAL SPECIFICATIONS CHANGES
VANTAGE-5 FUEL DESIGN

BASIS FOR PROPOSED CHANGES

Proposed Changes

The proposed Technical Specifications changes are listed in the attached table. The proposed changes are based on the operational and core design benefits provided through use of the VANTAGE-5 fuel design in conjunction with improved computer code methodologies.

Basis

In order to implement a long-term fuel management strategy planned by Georgia Power Company (GPC) for the Vogtle Electric Generating Plant (VEGP) Units 1 and 2, it has been decided to use reload fuel assemblies of the Westinghouse VANTAGE-5 design. This will require a transition from the current LOPAR fueled core to a full VANTAGE-5 fueled core. The transition is expected to be completed by the third loading of VANTAGE-5 fuel for each VEGP unit. This long-term strategy includes the implementation of high energy 18-month fuel cycles with high capacity factors, low leakage loading patterns, and extended fuel burnup. In addition, GPC plans to power rerate VEGP from 3411 to 3565 MWt; however, the Technical Specifications changes for power rerate are not being pursued in this license amendment request.

The Technical Specifications changes provided in Enclosure 3 are based on the use of VANTAGE-5 fuel specific design features and use of improved computer code methodologies. The fuel design features include the smaller diameter (OFA) fuel rods, mid-span zircaloy grids, Intermediate Flow Mixer (IFM) grids, natural uranium oxide (UO_2) axial blankets, Integral Fuel Burnable Absorbers (IFBA), extended fuel burnup, and reconstitutable top nozzles. The new computer code methodologies relative to the current VEGP safety analyses include the BART/BASH (large-break LOCA), NOTRUMP (small-break LOCA), and improved THINC-IV (thermal-hydraulics) computer codes, as well as the Revised Thermal Design Procedure (RTDP), WRB-2 DNB correlation, and Relaxed Axial Offset Control (RAOC) strategy. Each Technical Specifications change associated with either the change to VANTAGE-5 fuel or change in methodology is discussed below.

Enclosure 3 provides instructions for incorporating the proposed VANTAGE-5 Fuel Design Technical Specifications changes. Since VEGP uses combined Units 1 and 2 Technical Specifications, the instructions for incorporating the proposed changes for each unit will be done in two phases. The first phase is the proposed Technical Specifications changes involving the first fuel loading of VANTAGE-5 fuel in Unit 1 in late September 1991. At this point in time, the proposed Technical Specifications changes discussed below only apply to VEGP Unit 1, while VEGP Unit 2 will continue to operate with the existing Technical Specifications until its initial loading of

VANTAGE-5 fuel scheduled for February 1992. Therefore, the second phase fully implements the proposed VANTAGE-5 Fuel Design Technical Specifications changes discussed below for both VEGP Units 1 and 2. Enclosure 3 provides the Technical Specifications changes which implement this two-phase process.

The Technical Specifications changes for the core safety limits and over-temperature delta-T and over-power delta-T setpoints (Technical Specifications 2.1 and 2.2) were changed as a result of using DNB margin gained through the use of the VANTAGE-5 IFM grid feature, improved THINC-IV code, WRB-1 and WRB-2 DNB correlation, and RTDP. BART/BASH, NOTRUMP, and RAOC implementation were also factored into these Technical Specifications changes. The Technical Specifications core limits and setpoints changes accommodate the use of higher peaking factors for F-delta-H and F_Q, deletion of thimble plugs, axial blankets, IFBAs, the VANTAGE-5 rod and lattice geometry, higher steam generator tube plugging, mixed fuel type core DNB and LOCA peak clad temperature transition penalties, and future power rerate changes. The higher peaking factors will be revised through the use of the Core Operating Limits Report (COLR).

The Technical Specifications changes to describe the WRB-1, WRB-2, and RTDP correlation/methods are discussed in the BASES changes to the Technical Specifications (2.1.1 BASES, 3/4.2 BASES, and 3/4.4 BASES). The use of the VANTAGE-5 fuel design requires the use of the WRB-2 correlation. The WRB-1 correlation is used with the RTDP methodology for the LOPAR fuel design. The RTDP methodology was used to statistically combine the uncertainties in the DNB correlations with the uncertainties in the plant instrumentation, to provide operational margin to increase the maximum reactor coolant system average temperature (T_{AVG}) limit, reduce the pressurizer pressure limit, and reduce the reactor coolant system flow by the reduction in the flow measurement uncertainty (Technical Specifications changes to Sections 3/4.2.5 and 3/4.2.5 BASES). These DNB parameter Technical Specifications changes provide for improved plant operating flexibility.

The Technical Specifications change to Section 3/4.1.3.4 is proposed to increase the control rod drop time from 2.2 to 2.7 seconds. The VANTAGE-5 IFM grid feature slightly increases the core pressure drop. The VANTAGE-5 guide thimble inside diameter is slightly reduced compared to VEGP's current LOPAR fuel design. Both of these VANTAGE-5 design changes result in an increased control rod drop time. Therefore, the safety analyses performed for VEGP's VANTAGE-5 fuel design incorporated an increased Technical Specifications control rod drop time of 2.7 seconds.

The Technical Specifications changes to 3/4.2.1, 3/4.2.2, 3/4.2 BASES, and 6.8.1.6 are proposed to implement the NRC-approved RAOC methodology. The

RAOC methodology provides for increased plant operating flexibility in axial power shape control. The RAOC methodology, WCAP-10216-P-A (June 1983), forms the basis for changes to the axial flux difference and F_Q surveillance Technical Specifications changes. Since RAOC is being implemented with the use of VANTAGE-5 fuel, the F_Q surveillance Technical Specifications changes (includes use of a W(z) function) will replace the current F_{xy} surveillance Technical Specification (Section 3/4.2.2). The proposed changes to the Core Operating Limits Report (COLR) references listed in Technical Specification 6.8.1.6 include the switch from the current VEGP constant axial offset control methodology to RAOC. The new axial flux difference operating limit envelope associated with RAOC implementation will be revised through the use of the COLR. The F_{xy} limits currently given in the COLR will be removed and replaced with cycle and burnup-dependent W(z) functions to be implemented with the F_Q surveillance Technical Specifications proposed changes.

Technical Specifications changes to 3/4.1.2.5, 3/4.1.2.6, and 3/4.5.4 are proposed to the RWST Minimum Water Temperature to provide additional plant operating flexibility. The changes reduce the minimum RWST water temperature limiting condition for operation and surveillance limit from 54°F to 44°F and from 50°F to 40°F, respectively. These proposed changes were only affected by those safety analyses being performed to support the use of VANTAGE-5 fuel; therefore, these proposed changes were included in the VANTAGE-5 safety analyses.

Proposed Technical Specifications changes to the BASES Section 3/4.5.4 will provide additional operational and core design flexibility. The change uses the existing RWST boron concentration limits and applies the Leak Before Break (LBB) methodology to the Large Break Loss of Coolant Accident (LBLOCA) to allow for control rod insertion following the LBLOCA. Allowance for control rod insertion upon a reactor trip during a LBLOCA (all rods inserted less two control rods---one ejected rod and one stuck rod) eliminates the need for higher RWST boron concentration limits to maintain subcriticality at cold conditions during long-term core cooling, post-LOCA conditions. Without LBB methodology application to allow for control rod insertion during post-LOCA conditions, the RWST boron concentration limits in the Technical Specifications would have had to be increased which would place undesirable duty on plant equipment. Enclosure 4 supports the use of the LBB methodology to allow for control rod insertion during the LOCA.

The Technical Specifications changes to 3/4.5.1 provide plant operational flexibility by widening the accumulator water level limits. It is desirable to widen the accumulator water level range to accommodate potential changes in water levels that may be experienced over an 18-month operating cycle. These proposed Technical Specifications changes only affect those safety analyses being performed to support the use of VANTAGE-5 fuel; therefore, these proposed changes were included in the VANTAGE-5 safety analyses.

The Technical Specifications change to 3/4.3.2, Table 3.3-3 is proposed to raise the P-11 setpoint from 1970 psig to 2000 psig. The change to the P-11 setpoint does not affect any safety analysis results in the VEGP FSAR. The change to this setpoint increases the band between the safety injection (SI) block and the SI setpoint. The wider band provides improved plant operational flexibility.

A significant hazards evaluation (Enclosure 2) has been performed to support GPC's conclusion that these proposed Technical Specifications changes do not involve significant hazards considerations. Safety evaluations and analyses (Enclosure 4 and Appendices A, B, and C) also have been performed by Westinghouse to support these proposed Technical Specifications changes.

SUMMARY AND JUSTIFICATION FOR THE VOGTLE UNITS 1 AND 2 TECHNICAL SPECIFICATIONS CHANGES FOR VANTAGE 5 FUEL

Page	Section	Description	Justification
2-2 2-4, 2-8 2-9, 2-10 2-11	2.1 2.2	Change to core limits, to the OTDT and OPDT Setpoints.	This change is a result of changes associated with the VANTAGE 5 fuel and to provide operational flexibility.
B 2-1 B 2-2 B 3/4 2-1 B 3/4 2-2 B 3/4 2-4 B 3/4 4-1	2.1.1 Basis 3/4.2 Basis 3/4.4 Basis	Addition of the WRB-1 and WRB-2 correlation	This change reflects the DNB correlation used in analyses.
3/4 1-11 3/4 1-12 3/4 1-13 3/4 5-10	3/4.1.2.5 3/4.1.2.6 3/4.5.4	RWST minimum solution temperature	This change is to allow operational flexibility.
B 3/4 5-2	3/4.5.4 Basis	RWST Bases	This change is to allow operational flexibility.
3/4 1-19	3/4.1.3.4	Revised rod drop time to less than or equal to 2.7 seconds	This change is a result of changes associated with the VANTAGE 5 fuel. The effect of this increase on the safety analysis has been considered.
3/4 2-1 3/4 2-2 3/4 2-4 3/4 2-6 3/4 2-7 B 3/4 2-1 B 3/4 2-2 B 3/4 2-4	3/4.2.1 3/4.2.2 B 3/4.2 Basis	Axial Flux Difference and $F_Q(Z)$ changes	This change is made to give the plant operating flexibility and also as a result of changes associated with the VANTAGE 5 fuel.
3/4 2-13 B 3/4 2-5	3/4.2.5 3/4.2.5 Basis	DNB parameter changes	This change is made to give the plant operating flexibility and also as a result of changes associated with the VANTAGE 5 fuel.
3/4 3-33	3/4.3.2	Pressurizer pressure trip setpoint	This change is to provide operational flexibility.
3/4 5-1	3/4.5.1	Accumulator Water Level Range	This change is to provide operational flexibility.
6-21 6-21a	6.8.1.6	Core Operating Limits Report	This change is to provide operational flexibility.

Enclosure 2

Vogtle Electric Generating Plant Units 1 and 2

Request for Technical Specifications Changes

VANTAGE-5 Fuel Design

10 CFR 50.92 Evaluation

Enclosure 2

VOGTLE ELECTRIC GENERATING PLANT
REQUEST FOR TECHNICAL SPECIFICATIONS CHANGES
VANTAGE 5 FUEL DESIGN

10 CFR 50.92 EVALUATION

Pursuant to 10 CFR 50.92, each application for amendment to an operating license must be reviewed to determine if the proposed change involves a significant hazards consideration. The amendment, as defined below, describing the Technical Specifications changes associated with implementation of VANTAGE 5 fuel assemblies, has been reviewed and deemed not to involve significant hazards considerations. The basis for this determination follows.

Background

In order to implement the long term fuel management strategy planned by Georgia Power for the VEGP Units 1 and 2, it has been decided to use reload fuel assemblies of the Westinghouse VANTAGE 5 design. This strategy includes the implementation of high energy eighteen month fuel cycles with high capacity factors, low leakage loading patterns, and extended fuel burnup. Westinghouse VANTAGE 5 fuel has been designed to accommodate these operating characteristics by inclusion of specific design features and by use of improved methodologies previously approved by the NRC, or under review. These design features include Intermediate Flow Mixer (IFM) grids, natural uranium oxide (UO_2) axial blankets, Integral Fuel Burnable Absorbers (IFBA), extended fuel burnup, and Reconstitutable Top Nozzles (RTN). Each feature and methodology change is discussed below.

Intermediate Flow Mixer (IFM) Grids - The IFM grids promote flow mixing within the assembly and provide increased DNB margin. This additional margin can then be applied to accommodate higher design peaking factor values of F-delta-H and F_Q while still maintaining operational margin to new core safety limits and trip setpoints (Technical Specifications Sections 2.1 and 2.2 and associated BASES). The IFM grids also offer additional structural support for the fuel rods, reduced rod bow and improved seismic stability of the assembly. The IFM grid feature also increases the core pressure drop; thus the control rod drop time (Technical Specifications Section 3.1.3.4) was increased as a result of the increase in core pressure drop.

Axial Blankets - Axial blankets consist of six inches of natural UO_2 pellets (instead of enriched uranium) within the fuel rods at each end of the fuel stack, which reduces neutron leakage and improves uranium utilization.

Integral Fuel Burnable Absorber (IFBA) - The advantages provided by the IFBA include improved neutron utilization, since the neutron absorber material is a thin boride coating on the fuel pellets themselves. This reduces the need for individual absorber rods which displaces water molecules. Also, burnable absorber rods continue to absorb neutrons later in core life when this function is no longer necessary, creating a residual penalty which inhibits economic fuel use. In contrast, the IFBA's capacity to absorb neutrons is limited and precisely matched to the reactivity depletion of the fuel. The resulting more efficient use of neutrons enables fuel to last longer, reducing fuel cost. Because control of reactivity is not required at the blanketed ends, the IFBA is only added to the fuel in the central portion of the rods.

Extended Burnup - Longer fuel cycles result in increased rod growth and production of fission product gas which increases rod internal pressure. By increasing the fuel rod plenum and providing additional space between the fuel rod and nozzle, the effects of increased internal rod pressure and rod growth resulting from longer cycles and extended fuel burnups to a lead rod average of 60,000 MWD/MTU can be accommodated.

Reconstitutable Top Nozzle (RTN) - The assembly reconstitutable top nozzle provides the capability to replace damaged fuel rods. This feature avoids the discharging of an entire assembly for minor fuel damage. The advantage is increased fuel reliability and reduced fuel costs.

Each of these features supports the safe, efficient fuel management scheme planned for the VEGP Units 1 and 2. Margin provided by the VANTAGE 5 fuel design and by recently NRC approved (or under review) methodologies will be used to revise core design parameters that will provide increased operational flexibility (Technical Specifications Sections 3/4.1.2.5, 3/4.1.2.6, 3/4.3.2, 3/4.5.1, 3/4.5.4 and associated BASES). These methodologies include the use of BART/BASH, NOTRUMP, and the improved THINC-IV computer codes, as well as the Revised Thermal Design Procedure (RTDP), the WRB-2 DNB correlation, and the Relaxed Axial Offset Cortrol (RAOC) Strategy.

BART/BASH - These codes are utilized in the analysis of the large break LOCA. BART employs rigorous mechanistic models to generate heat transfer coefficients appropriate to the actual flow and heat transfer regimes experienced by the fuel rods. The BART code has been coupled with a loop model to form the BASH code. BART provides the entrainment rate for a given flooding rate. The coupling of the BART code with a loop code produces a dynamic flooding transient which reflects the close coupling between core thermal-hydraulics and loop behavior. The BASH code provides a realistic thermal-hydraulic simulation of the reactor core and RCS during the reflood phase of a large break LOCA by utilizing a sophisticated treatment of steam/water flow phenomena in the RCS. The use of this methodology provides additional margin in peak clad temperature to support an increase in F_Q peaking factor.

NOTRUMP - This code is utilized in the analysis of the small break LOCA. Features of this code include 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. In NOTRUMP, the RCS is nodalized into volumes interconnected by flow paths. The broken loop is modeled explicitly with the intact loops lumped into a second loop. The transient behavior of the system is determined from the governing conservation equations of mass, energy and momentum applied throughout the system. The use of this methodology also provides additional margin in peak clad temperature to support an increase in F_Q peaking factor.

Improved THINC-IV - This code is used to perform thermal-hydraulic calculations for the non-LOCA transients. It calculates coolant density, mass velocity, enthalpy, void fractions, static pressure and DNBR distributions along the flow channels within a reactor core under all expected accident conditions. The improved THINC-IV design modeling methodology currently under NRC review (WCAP-12330-P, August 1989) replaces the present THINC-IV model.

Revised Thermal Design Procedure (RTDP) - With this methodology, DNBR uncertainties associated with uncertainties in plant operating parameters, nuclear and thermal parameters and computer codes are statistically combined with the DNBR correlation uncertainties. Since the parameter uncertainties are considered in determining the design limit DNBR values, the plant safety analyses are performed using values of input parameters without uncertainties. The RTDP methodology was used to provide additional operating margin for the proposed Technical Specifications Section 3/4.2.5 and associated BASES changes 2.1.1 and 3/4.2.

WRB-2 - This is a DNB correlation that is used for fuel assemblies utilizing IFM grids. It takes credit for significant improvement in the accuracy of the critical heat flux predictions over previous DNB correlations as well as for the reduced grid-to-grid spacing of the VANTAGE 5 fuel assembly mixing vanes. The WRB-2 DNB correlation is incorporated into the Technical Specifications BASES Sections 2.1.1, 3/4.2, and 3/4.4.

Relaxed Axial Offset Control (RAOC) - This strategy was developed to provide wider control band widths and more operator freedom than with Constant Axial Offset Control (CAOC). RAOC provides wider control bands particularly at reduced power by utilizing core margin effectively. The wider operating space increases plant availability by allowing quicker plant startups and increased maneuvering flexibility without reactor trip or reportable occurrences. The RAOC strategy forms the basis for Technical Specifications changes 3/4.2.1, 3/4.2.2 and the associated BASES, and Section 6.8.1.6.

Analysis

The proposed Technical Specifications changes to reflect the operational and core design benefits provided through use of the VANTAGE 5 design in conjunction with NRC approved (or under review) methodologies are summarized in Enclosure 1 and the changes are provided in Enclosure 3. The effect of these changes on plant safety is discussed in detail in the Safety Evaluation portion of this submittal (Enclosure 4). The implementation of VANTAGE 5 fuel includes conservatism for power rerating to 3565 MWt, additional increased peaking factors, steam generator tube plugging up to ten percent, a reduction in thermal design flow and various other plant operational margins. Therefore, the NSSS design parameters to which the safety analyses have been performed in support of VANTAGE 5 fuel have been generated to account for these future changes. However, these future changes are not being requested in the licensing amendment.

Results

Based on the information presented above and the Safety Evaluation in Enclosure 4, the following conclusions can be reached with respect to 10 CFR 50.92.

1. The VANTAGE 5 fuel related Technical Specifications changes do not involve a significant increase in the probability or consequences of an accident previously evaluated in the VEGP FSAR. The mechanical design changes associated with VANTAGE 5 fuel result in the capability for relaxation of analytical input parameters such that increased margin can be generated without violation of any acceptance criteria. This margin can then be applied towards relaxation of operational limits such as reduced safety injection flow or increased steam generator tube plugging. In each case however, the appropriate design and acceptance criteria are met. No new performance requirements are being imposed on any system or component in order to support the revised analysis assumptions. Subsequently, overall plant integrity is not reduced. Furthermore, the parameter changes are associated with features used as limits or mitigators to assumed accident scenarios and are not accident initiators. Therefore, the probability of an accident has not increased.

The consequences of an accident previously evaluated in the VEGP FSAR are not increased due to the VANTAGE 5 fuel related Technical Specifications changes. Evaluation of the radiological consequences of extended fuel burnup to lead rod average burnups of 60,000 MWD/MTU has been performed for the transition to VANTAGE 5 fuel. These evaluations have confirmed that the doses remain within previously approved acceptable limits as well as those defined by 10 CFR 100. Therefore, the consequences to the public resulting from any accident previously evaluated in the VEGP FSAR has not significantly increased.

2. The VANTAGE 5 fuel related Technical Specifications changes do not create the possibility of a new or different kind of accident from any accident previously evaluated in the VEGP FSAR. Mechanical evaluations have been performed on the fuel assemblies, fuel rods and control rod drop times to confirm that their function and reliability are consistent with the originally supplied equipment. No new accident scenarios, failure mechanisms or limiting single failures are introduced as a result of the fuel transition. The presence of VANTAGE 5 fuel assemblies in the core or the revised analytical assumptions have no adverse effect and do not challenge the performance of any other safety related system. Therefore, the possibility of a new or different kind of accident is not created.

3. The VANTAGE 5 Technical Specifications changes do not involve a significant reduction in the margin of safety. The margin of safety for fuel related parameters associated with the VANTAGE 5 transition are defined in the BASES to those Technical Specifications identified in Enclosure 1. These BASES and the supporting Technical Specifications values are defined by the accident analyses which are performed to conservatively bound the operating conditions. These operating conditions are defined by the Technical Specifications such that the regulatory acceptance limits will not be exceeded. Performance of analyses and evaluations for the VANTAGE 5 fuel transition have confirmed that the operating envelope defined by the Technical Specifications continues to be bounded by the revised analytical basis, which in no case exceeds the acceptance limits. Therefore, the margin of safety provided by the analyses in accordance with these acceptance limits is maintained and not reduced.

Conclusion

Based upon the preceding information, it has been determined that the proposed changes to the Technical Specifications do not involve a significant increase in the probability or consequences of an accident previously evaluated, create the possibility of a new or different kind of accident from any accident previously evaluated or involve a significant reduction in a margin of safety. Therefore, it is concluded that the proposed changes meet the requirements of 10 CFR 50.92 (c) and do not involve a significant hazards consideration.

Enclosure 3

Vogtle Electric Generating Plant Units 1 and 2

Request for Technical Specifications Changes

VANTAGE-5 Fuel Design

Instructions for Incorporation

ENCLOSURE 3

VOGTLE ELECTRIC GENERATING PLANT
REQUEST FOR TECHNICAL SPECIFICATIONS CHANGES
VANTAGE-5 FUEL DESIGN

INSTRUCTIONS FOR INCORPORATION

The proposed amendment to the Technical Specifications would be incorporated as follows:

Phase 1

Attachments 1a and 1b: Effective following the Vogtle 1 Cycle 3 Shutdown
(Effective as of Vogtle 1 Cycle 4 Startup)

Remove Page	Insert Page
III and IV*	III and IV*
V and VI*	V and VI*
XV and XVI*	XV and XVI*
XXIII and XXIV*	XXIII and XXIV*
2-1 and 2-2	2-1 and 2-2, 2-2a
2-3 and 2-4	2-3 and 2-4
2-5 and 2-6	2-5 and 2-6
2-7 and 2-8	2-7 and 2-8
2-9 and 2-10	2-9 and 2-10
2-11	2-11, 2-12 and 2-13, 2-14 and 2-15, 2-16 and 2-17, 2-18 and 2-19
B 2-1 and B 2-2	B 2-1 and B 2-1a, B 2-2
3/4 1-11 and 3/4 1-12	3/4 1-11 and 3/4 1-12
3/4 1-13 and 3/4 1-14*	3/4 1-13 and 3/4 1-14*
3/4 1-19 and 3/4 1-20*	3/4 1-19 and 3/4 1-20*
3/4 2-1 and 3/4 2-2	3/4 2-1 and 3/4 2-1a, 3/4 2-2
3/4 2-3 and 3/4 2-4	3/4 2-3 and 3/4 2-4
3/4 2-5 and 3/4 2-6	3/4 2-5 and 3/4 2-6
3/4 2-7 and 3/4 2-8*	3/4 2-7 and 3/4 2-7a, 3/4 2-8*
3/4 2-13	3/4 2-13
3/4 3-33 and 3/4 3-34*	3/4 3-33 and 3/4 3-34*
3/4 5-1 and 3/4 5-2*	3/4 5-1 and 3/4 5-2*
3/4 5-9* and 3/4 5-10	3/4 5-9* and 3/4 5-10
B 3/4 2-1 and B 3/4 2-2	B 3/4 2-1 and B 3/4 2-2
B 3/4 2-3 and B 3/4 2-4	B 3/4 2-3 and B 3/4 2-4
B 3/4 2-5 and B 3/4 2-6	B 3/4 2-5 and B 3/4 2-6, B 3/4 2-7 and B 3/4 2-8, B 3/4 2-9 and B 3/4 2-10
B 3/4 4-1 and B 3/4 4-2*	B 3/4 4-1 and B 3/4 4-2*
B 3/4 5-1* and B 3/4 5-2	B 3/4 5-1* and B 3/4 5-2
6-21 and 6-21a	6-21 and 6-21a, 6-21b

* Overleaf page containing no changes

ENCLOSURE 3
(Cont'd)

VOGTLE ELECTRIC GENERATING PLANT
REQUEST FOR TECHNICAL SPECIFICATIONS CHANGES
VANTAGE-5 FUEL DESIGN

INSTRUCTIONS FOR INCORPORATION

The proposed amendment to the Technical Specifications would be incorporated as follows:

Phase 2

Attachments 2a and 2b: Effective following the Vogtle 2 Cycle 2 Shutdown (Effective as of Vogtle 2 Cycle 3 Startup)

Remove Page	Insert Page
III and IV*	III and IV*
V and VI*	V and VI*
XV and XVI*	XV and XVI*
XXIII and XXIV*	XXIII and XXIV*
2-1 and 2-2, 2-2a	2-1 and 2-2
2-3 and 2-4	2-3 and 2-4
2-5 and 2-6	2-5 and 2-6
2-7 and 2-8	2-7 and 2-8
2-9 and 2-10	2-9 and 2-10
2-11, 2-12 and 2-13, 2-14 and 2-15, 2-16 and 2-17, 2-18 and 2-19	2-11
B 2-1 and B2-1a, B 2-2*	B 2-1 and B 2-2*
3/4 1-11 and 3/4 1-12	3/4 1-11 and 3/4 1-12
3/4 1-13 and 3/4 1-14*	3/4 1-13 and 3/4 1-14*
3/4 1-19 and 3/4 1-20*	3/4 1-19 and 3/4 1-20*
3/4 2-1 and 3/4 2-1a, 3/4 2-2	3/4 2-1 and 3/4 2-2
3/4 2-3* and 3/4 2-4	3/4 2-3* and 3/4 2-4
3/4 2-5 and 3/4 2-6	3/4 2-5 and 3/4 2-6
3/4 2-7 and 3/4 2-7a, 3/4 2-8*	3/4 2-7 and 3/4 2-8*
3/4 2-13	3/4 2-13
3/4 3-33 and 3/4 3-34*	3/4 3-33 and 3/4 3-34*
3/4 5-1 and 3/4 5-2*	3/4 5-1 and 3/4 5-2*
3/4 5-9* and 3/4 5-10	3/4 5-9* and 3/4 5-10
B 3/4 2-1 and B 3/4 2-2	B 3/4 2-1 and B 3/4 2-2
B 3/4 2-3 and B 3/4 2-4	B 3/4 2-3 and B 3/4 2-4
B 3/4 2-5 and B 3/4 2-6, B 3/4 2-7 and B 3/4 2-8, B 3/4 2-9 and B 3/4 2-10	
B 3/4 5-1* and B 3/4 5-2	B 3/4 5-1* and B 3/4 5-2
6-21 and 6-21a, 6-21b	6-21 and 6-21a

* Overleaf page containing no changes

Attachment 1a

Vogtle Electric Generating Plant Units 1 and 2

Request for Technical Specifications Changes

VANTAGE-5 Fuel Design

Technical Specifications Marked-Up Pages

Effective following the Vogtle 1 Cycle 3 Shutdown
(Effective as of Vogtle 1 Cycle 4 Startup)

FIGURE 2.1-1a REACTOR CORE SAFETY LIMIT (UNIT 2) 2-2a

INDEX

SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

BASES

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

BASES

SECTION PAGE

ADMINISTRATIVE CONTROLS

SECTION

2.0 SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS*

2.1 SAFETY LIMITS

REACTOR CORE

2.1.1 The combination of THERMAL POWER (NI-0041, NI-0042, NI-0043, NI-0044), pressurizer pressure (PI-0455A, B&C, PI-0456 & PI-0456A, PI-0457 & PI-0457A, PI-0458 & PI-0458A), and the highest operating loop coolant temperature (T_{avg}) (TI-0412, TI-0422, TI-0432, TI-0442) shall not exceed the limits shown in Figure 2.1-1 (UNIT 1) or Figure 2.1-1a (UNIT 2).

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 6.6.1.

REACTOR COOLANT SYSTEM PRESSURE

2.1.2 The Reactor Coolant System pressure (PI-0408, PI-0418, PI-0428, PI-0438) 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 6.6.1.

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 6.6.1.

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*Where specific instrument numbers are provided in parentheses they are for information only, and apply to each unit unless specifically noted (to assist in identifying associated instrument channels or loops) and are not intended to limit the requirements to the specific instruments associated with the number.

THIS PAGE APPLICABLE TO UNIT 1 ONLY

INSERT 7 HERE

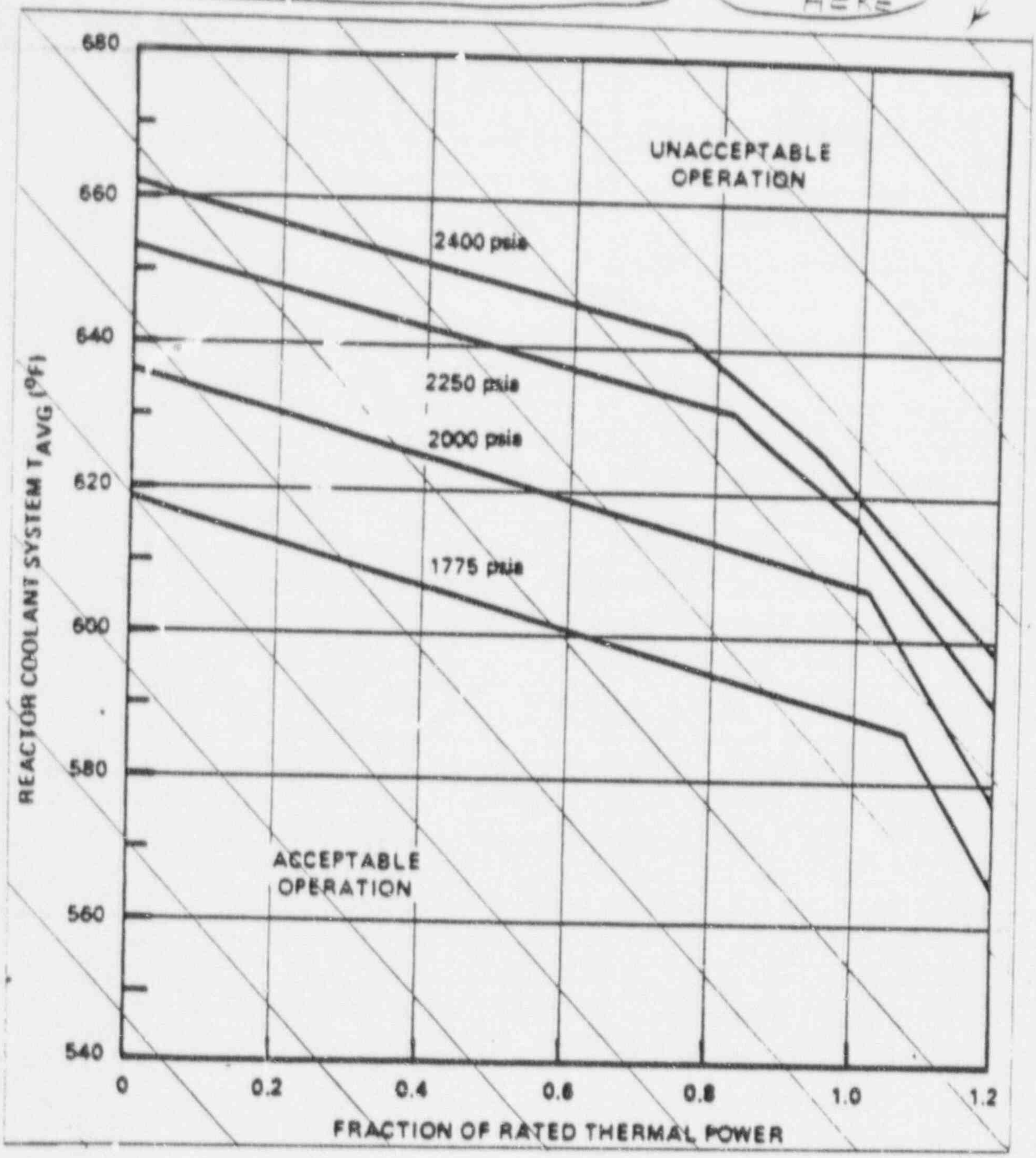
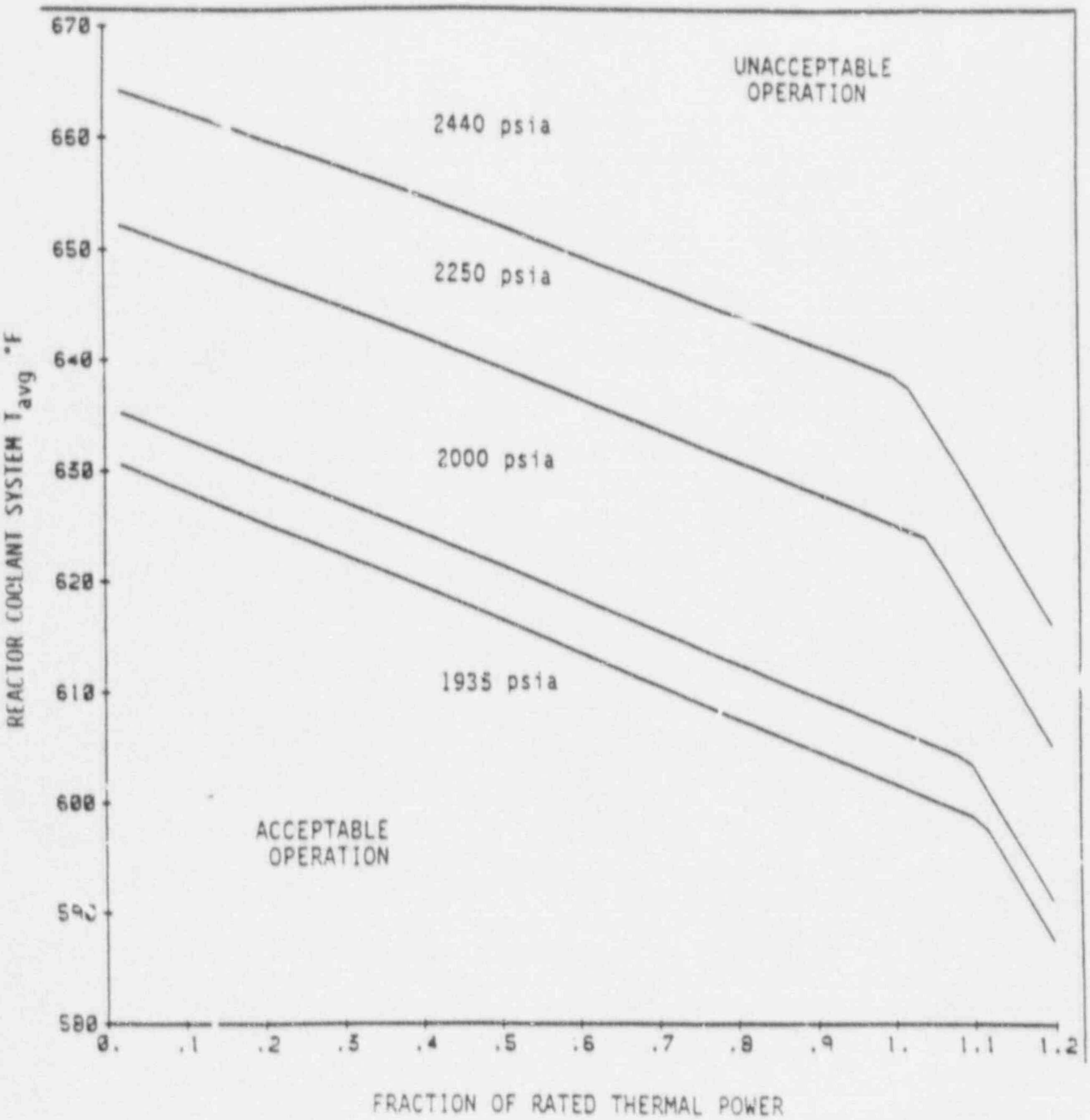


FIGURE 2.1-1

REACTOR CORE SAFETY LIMIT - UNIT 1

INSERT 7


UNACCEPTABLE
OPERATION
2440 psia
2250 psia
2000 psia
1935 psia
ACCEPTABLE
OPERATION
670
660
650
640
630
620
610
600
590
580
0.
.1
.2
.3
.4
.5
.6
.7
.8
.9
1.
1.1
1.2
REACTOR COOLANT SYSTEM T_{avg} °F
FRACTION OF RATED THERMAL POWER

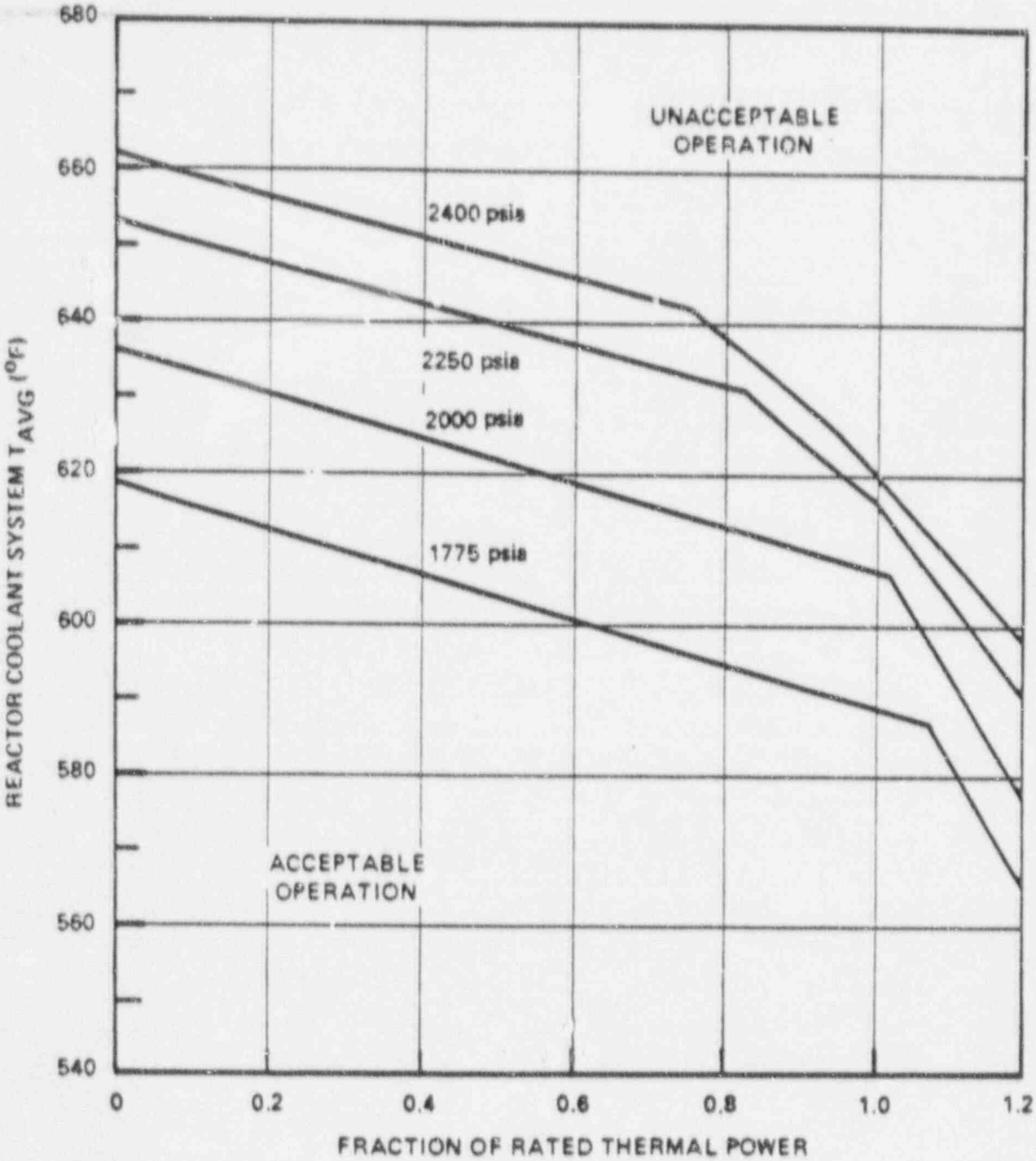


FIGURE 2.1-1 a

REACTOR CORE SAFETY LIMIT - UNIT 2

2.2 LIMITING SAFETY SYSTEM SETTINGS

REACTOR TRIP SYSTEM INSTRUMENTATION SETPOINTS

2.2.1 The Reactor Trip System Instrumentation and Interlock Setpoints shall be set consistent with the Trip Setpoint values shown in Table 2.2-1 (UNIT 1) or Table 2.2-1a (UNIT 2).

APPLICABILITY: As shown for each channel in Table 3.3-1.

ACTION:

a. With a Reactor Trip System Instrumentation or Interlock Setpoint less conservative than the value shown in the Trip Setpoint column but more conservative than the value shown in the Allowable Value column of Table 2.2-1 or Table 2.2-1a, adjust the Setpoint consistent with the Trip Setpoint value.

b. With the Reactor Trip System Instrumentation or Interlock Setpoint less conservative than the value shown in the Allowable Values column of Table 2.2-1 or Table 2.2-1a, either:

1. Adjust the Setpoint consistent with the Trip Setpoint value of Table 2.2-1 or Table 2.2-1a and determine within 12 hours that Equation 2.2-1 was satisfied for the affected channel, or

2. Declare the channel inoperable and apply the applicable ACTION statement requirement of Specification 3.3.1 until the channel is restored to OPERABLE status with its Setpoint adjusted consistent with the Trip Setpoint value.

Equation 2.2-1 $$Z + R + S \leq TA$$

Where:

Z = The value from Column Z of Table 2.2-1 or Table 2.2-1a for the affected channel,

R = The "as measured" value (in percent span) of rack error for the affected channel,

S = Either the "as measured" value (in percent span) of the sensor error, or the value from Column S (Sensor Error) of Table 2.2-1 or Table 2.2-1a for the affected channel, and

TA = The value from Column TA (Total Allowance) of Table 2.2-1 or Table 2.2-1a for the affected channel.

THIS PAGE APPLICABLE TO UNIT 1 ONLY

TABLE 2.2-1 - UNIT 1

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TOTAL ALLOWANCE (TA)	Z	SENSOR ERROR (S)	TRIP SETPOINT	ALLOWABLE VALUE
1. Manual Reactor Trip	N.A.	N.A.	N.A.	N.A.	N.A.
2. Power Range, Neutron Flux (NI-0041B&C, NI-0042B&C, NI-0043B&C, NI-0044B&C)					
a. High Setpoint	7.5	4.56	0	<109% of RTP#	<111.3% of RTP#
b. Low Setpoint	8.3	4.56	0	<25% of RTP#	<27.3% of RTP#
3. Power Range, Neutron Flux, High Positive Rate (NI-0041B&C, NI-0042B&C, NI-0043B&C, NI-0044B&C)	1.6	0.50	0	<5% of RTP# with a time constant >2 seconds	<6.3% of RTP# with a time constant >2 seconds
DELETED	Note: This change was previously requested by letter ELV-01761				
4. ~~Power Range, Neutron Flux, High Negative Rate (NI-0041B&C, NI-0042B&C, NI-0043B&C, NI-0044B&C)~~	~~1.6~~	~~0.50~~	~~0~~	~~<5% of RTP# with a time constant >2 seconds~~	~~<6.3% of RTP# with a time constant >2 seconds~~
5. Intermediate Range, Neutron Flux (NI-0035B, NI-0036B)	17.0	8.41	0	<25% of RTP#	<31.1% of RTP#
6. Source Range, Neutron Flux (NI-0031B, NI-0032B)	17.0	10.01	0	$<10^5$ cps	$<1.4 \times 10^5$ cps
7. Overtemperature ΔT (TDI-411C, TDI-421C, TDI-431C, TDI-441C)	10.7 (UNIT 1) ~~6.6~~	7.04 (UNIT 1) ~~3.37~~	1.96 ~~1.95~~ +1.17 (UNIT 1) ~~+ 0.50~~	See Note 1	See Note 2
8. Overpower ΔT (TDI-411B, TDI-421B, TDI-431B, TDI-441B)	4.3 (UNIT 1) ~~4.9~~	1.54	1.96 (UNIT 1) ~~1.95~~	See Note 3	See Note 4

#RTP = RATED THERMAL POWER

THIS PAGE APPLICABLE TO UNIT 1 ONLY

TABLE 2.2-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS - UNIT 1

FUNCTIONAL UNIT	TOTAL ALLOWANCE (TA)	Z	SENSOR ERROR (S)	TRIP SETPOINT	ALLOWABLE VALUE
9. Pressurizer Pressure-Low (PI-0455A,B&C, PI-0456 & PI-0456A, PI-0457 & PI-0457A, PI-0458 & PI-0458A)	3.1	0.71	1.67	≥1960 psig**	≥1950 psig
10. Pressurizer Pressure-High (PI-0455A,B&C, PI-0456 & PI-0456A, PI-0457 & PI-0457A, PI-0458 & PI-0458A)	3.1	0.71	1.67	≤2385 psig	≤2395 psig
11. Pressurizer Water Level-High (LI-0459A, LI-0460A, LI-0461)	8.0	2.18	1.67	≤92% of instrument span	≤93.9% of instrument span
12. Reactor Coolant Flow-Low (LOOP1 LOOP2 LOOP3 LOOP4 FI-0414 FI-0424 FI-0434 FI-0444 FI-0415 FI-0425 FI-0435 FI-0445 FI-0416 FI-0426 FI-0436 FI-0446)	2.5	1.87	0.60	≥90% of loop design flow*	≥89.4% of loop design flow*
13. Steam Generator Water Level Low-Low (LOOP1 LOOP2 LOOP3 LOOP4 LI-0517 LI-0527 LI-0537 LI-0547 LI-0518 LI-0528 LI-0538 LI-0548 LI-0519 LI-0529 LI-0539 LI-0549 LI-0551 LI-0552 LI-0553 LI-0554)	18.5 (21.8)***	17.18 (18.21)***	1.67	≥18.5% (37.8)*** of narrow range instrument span	≥17.8% (35.9)*** of narrow range instrument span
14. Undervoltage - Reactor Coolant Pumps	6.0	0.58	0	≥9600 volts (70% bus voltage)	≥9481 volts (69% bus voltage)
15. Underfrequency - Reactor Coolant Pumps	3.3	0.50	0	≥57.3 Hz	≥57.1 Hz

*Loop design flow = 95,700 gpm

**Time constants utilized in the lead-lag controller for Pressurizer Pressure-Low are 10 seconds for lead and 1 second for lag. CHANNEL CALIBRATION shall ensure that these time constants are adjusted to these values.

***The value stated inside the parenthesis is for instrument that has the lower tap at elevation 333"; the value stated outside the parenthesis is for instrumentation that has the lower tap at elevation 438".

THIS PAGE APPLICABLE TO UNIT 1 ONLY

TABLE 2.2-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS - UNIT 1

FUNCTIONAL UNIT	TOTAL ALLOWANCE (TA)	Z	SENSOR ERROR (S)	TRIP SETPOINT	ALLOWABLE VALUE
16. Turbine Trip					
a. Low Fluid Oil Pressure (PT-6161, PT-6162, PT-6163)	N.A.	N.A.	N.A.	$\geq$580 psig	$\geq$500 psig
b. Turbine Stop Valve Closure	N.A.	N.A.	N.A.	$\geq$96.7% open	$\geq$96.7% open
17. Safety Injection Input from ESF	N.A.	N.A.	N.A.	N.A.	N.A.
18. Reactor Trip System Interlocks					
a. Intermediate Range Neutron Flux, P-6 (NI-0035B, NI-0036B)	N.A.	N.A.	N.A.	$\geq 1 \times 10^{-10}$ amp	$\geq 6 \times 10^{-11}$ amp
b. Low Power Reactor Trips Block, P-7					
1) P-10 input (NI-0041B&C, NI-0042B&C, NI-0043B&C, NI-0044B&C)	N.A.	N.A.	N.A.	$\leq$10% of RTP#	$\leq$12.3% of RTP#
2) P-13 input (PI-0505, PI-0506)	N.A.	N.A.	N.A.	$\leq$10% RTP# Turbine Impulse Pressure Equivalent	$\leq$12.3% RTP# Turbine Impulse Pressure Equivalent
c. Power Range Neutron Flux, P-8 (NI-0041B&C, NI-0042B&C, NI-0043B&C, NI-0044B&C)	N.A.	N.A.	N.A.	$\leq$48% of RTP#	$\leq$50.3% of RTP#

#RTP = RATED THERMAL POWER

THIS PAGE APPLICABLE TO UNIT 1 ONLY

TABLE 2.2-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS — UNIT 1

FUNCTIONAL UNIT	TOTAL ALLOWANCE (TA)	Z	SENSOR ERROR (S)	TRIP SETPOINT	ALLOWABLE VALUE
d. Power Range Neutron Flux, P-9 (NI-0041B&C, NI-0042B&C, NI-0043B&C, NI-0044B&C)	N.A.	N.A.	N.A	≤50% of RTP#	≤52.3% of RTP#
e. Power Range Neutron Flux, P-10 (NI-0041B&C, NI-0042B&C, NI-0043B&C, NI-0044B&C)	N.A.	N.A.	N.A	≥10% of RTP#	≥7.7% of RTP#
f. Turbine Impulse Chamber Pressure, P-13 (PI-0505, PI-0506)	N.A.	N.A.	N.A.	<10% RTP# Turbine Impulse Pressure Equivalent	<12.3% RTP# Turbine Impulse Pressure Equivalent
19. Reactor Trip Breakers	N.A.	N.A.	N.A	N.A.	N.A.
20. Automatic Trip and Interlock Logic	N.A.	N.A.	N.A.	N.A.	N.A.

#RTP = RATED THERMAL POWER

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TABLE 2.2-1 (Continued)

TABLE NOTATIONS - UNIT 1

NOTE 1: OVERTEMPERATURE ΔT

$$\Delta T \frac{(1 + \tau_1 S)}{(1 + \tau_2 S)} \left(\frac{1}{1 + \tau_3 S}\right) \leq \Delta T_0 \left\{K_1 - K_2 \frac{(1 + \tau_4 S)}{(1 + \tau_5 S)} \left[T \left(\frac{1}{1 + \tau_6 S}\right) - T'\right] + K_3(P - P') - f_1(\Delta I)\right\}$$

Where:

ΔT	=	Measured ΔT ~~by RTD Manifold Instrumentation~~; (UNIT 1)
$\frac{1 + \tau_1 S}{1 + \tau_2 S}$	=	Lead-lag compensator on measured ΔT;
τ_1, τ_2	=	Time constants utilized in lead-lag compensator for ΔT, $\tau_1 \geq 8$ s, $\tau_2 \leq 3$ s;
$\frac{1}{1 + \tau_3 S}$	=	Lag compensator on measured ΔT;
τ_3	=	Time constants utilized in the lag compensator for ΔT, $\tau_3 = 0$ s;
ΔT_0	=	Indicated ΔT at RATED THERMAL POWER;
K_1	$\leq$	~~1.10~~; 1.12 (UNIT 1)
K_2	=	~~0.012/°F~~; 0.0224/°F (UNIT 1)
$\frac{1 + \tau_4 S}{1 + \tau_5 S}$	=	The function generated by the lead-lag compensator for T_{avg} dynamic compensation;
τ_4, τ_5	=	Time constants utilized in the lead-lag compensator for T_{avg}, $\tau_4 \geq 28$ s, $\tau_5 \leq 4$ s;
T	=	Average temperature, °F;
$\frac{1}{1 + \tau_6 S}$	=	Lag compensator on measured T_{avg};
τ_6	=	Time constant utilized in the measured T_{avg} lag compensator, $\tau_6 = 0$ s;

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TABLE 2.2-1 (Continued)

TABLE NOTATIONS (Continued) - UNIT 1

NOTE 1: (Continued)

588.4°F (UNIT 1) operating temperature

T' ≤ ~~588.5°F~~ (Nominal T_{avg} ~~at RATED THERMAL POWER~~);

K_3 = ~~0.00056/psig~~; 0.00115/psig (UNIT 1)

P = Pressurizer pressure, psig;

P' = 2235 psig (Nominal RCS operating pressure);

S = Laplace transform variable, s^{-1};

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

-32.0% (UNIT 1) +11.0% (UNIT 1)

(1) For $q_t - q_b$ between ~~-33.5%~~ and ~~+6.5%~~, $f_1(\Delta I) = 0$, 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;

-32.0% (UNIT 1)

(2) For each percent that the magnitude of $q_t - q_b$ exceeds ~~-33.5%~~, the ΔT Trip Setpoint shall be automatically reduced by ~~1.27%~~ of its value at RATED THERMAL POWER; and

3.25% (UNIT 1) +11.0% (UNIT 1)

(3) For each percent that the magnitude of $q_t - q_b$ exceeds ~~+6.5%~~, the ΔT Trip Setpoint shall be automatically reduced by ~~0.83%~~ of its value at RATED THERMAL POWER.

1.97% (UNIT 1)

NOTE 2: The channel's maximum Trip Setpoint shall not exceed its computed Trip Setpoint by more than ~~2.5%~~.

3.1% (UNIT 1) of ΔT span.

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TABLE 2 -1 (Continued)

TABLE NOTATIONS (Continued) - UNIT 1

NOTE 3: OVERPOWER ΔT

$$\Delta T \frac{(1 + \tau_1 S)}{(1 + \tau_2 S)} \left(\frac{1}{1 + \tau_3 S}\right) \leq \Delta T_0 \{K_4 - K_5 \left(\frac{\tau_7 S}{1 + \tau_7 S}\right) \left(\frac{1}{1 + \tau_6 S}\right) T - K_6 [T \left(\frac{1}{1 + \tau_6 S}\right) - T''] - f_2(\Delta I)\}$$

Where:

ΔT = Measured ΔT ~~by RTD manifold instrumentation~~; (UNIT 1)

$\frac{1 + \tau_1 S}{1 + \tau_2 S}$ = Lead-lag compensator on measured ΔT;

τ_1, τ_2 = Time constants utilized in lead-lag compensator for ΔT, $\tau_1 \geq 8$ S, $\tau_2 \leq 3$ s;

$\frac{1}{1 + \tau_3 S}$ = Lag compensator on measured ΔT;

τ_3 = Time constants utilized in the lag compensator for ΔT, $\tau_3 = 0$s;

ΔT_0 = Indicated ΔT at RATED THERMAL POWER;

K_4 $\leq$ $\boxed{1.089}$ 1.08 (UNIT 1)

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

$\frac{\tau_7 S}{1 + \tau_7 S}$ = The function generated by the rate-lag compensator for T_{avg} dynamic compensation,

τ_7 = Time constants utilized in the rate-lag compensator for T_{avg}, $\tau_7 \geq 10$ s,

$\frac{1}{1 + \tau_6 S}$ = Lag compensator on measured T_{avg};

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TABLE 2.2-1 (Continued)

TABLE NOTATIONS (Continued) - UNIT 1

NOTE 3: (Continued)

τ_6 = Time constant utilized in the measured T_{avg} lag compensator, $\tau_6 = 0$ s;

0.0020/°F (UNIT 1)

K_6 $\geq$ $\boxed{\cancel{0.0013/°F}}$ for $T > T''$ and $K_6 = 0$ for $T \leq T''$,

T = Average Temperature, °F;

T'' = Indicated T_{avg} at RATED THERMAL POWER (Calibration temperature for ΔT instrumentation, $\leq$ $\boxed{\cancel{588.5°F}}$), 588.4°F (UNIT 1)

S = Laplace transform variable, s^{-1}; and

$f_2(\Delta I)$ = 0 for all ΔI.

NOTE 4: The channel's maximum Trip Setpoint shall not exceed its computed Trip Setpoint by more than $\boxed{2.4\%}$ of ΔT span.

1.9% (UNIT 1)

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TABLE 2.2-1* (a - UNIT 2)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TOTAL ALLOWANCE (TA)	Z	SENSOR ERROR (S)	TRIP SETPOINT	ALLOWABLE VALUE
1. Manual Reactor Trip	N.A.	N.A.	N.A.	N.A.	N.A.
2. Power Range, Neutron Flux (NI-0041B&C, NI-0042B&C, NI-0043B&C, NI-0044B&C)					
a. High Setpoint	7.5	4.56	0	≤109% of RTP#	≤111.3% of RTP#
b. Low Setpoint	8.3	4.56	0	≤25% of RTP#	≤27.3% of RTP#
3. Power Range, Neutron Flux, High Positive Rate (NI-0041B&C, NI-0042B&C, NI-0043B&C, NI-0044B&C)	1.6	0.50	0	≤5% of RTP# with a time constant ≥2 seconds	≤6.3% of RTP# with a time constant ≥2 seconds
4. ~~Power Range, Neutron Flux, High Negative Rate (NI-0041B&C, NI-0042B&C, NI-0043B&C, NI-0044B&C)~~ DELETED	~~1.6~~	~~0.50~~	~~0~~	~~≤5% of RTP# with a time constant ≥2 seconds~~	~~≤6.3% of RTP# with a time constant ≥2 seconds~~
5. Intermediate Range, Neutron Flux (NI-0035B, NI-0036B)	17.0	8.41	0	≤25% of RTP#	≤31.1% of RTP#
6. Source Range, Neutron Flux (NI-0031B, NI-0032B)	17.0	10.01	0	≤10^5 cps	≤1.4×10^5 cps
7. Overtemperature ΔT (TDI-411C, TDI-421C, TDI-431C, TDI-441C)	6.6 (UNIT 2)	3.37 (UNIT 2)	1.95 + 0.50 (UNIT 2)	See Note 1	See Note 2
8. Overpower ΔT (TDI-411B, TDI-421B, TDI-431B, TDI-441B)	4.9 (UNIT 2)	1.54	1.95 (UNIT 2)	See Note 3	See Note 4

NOTE: This change was previously requested by letter ELV-01761

#RTP = RATED THERMAL POWER

THIS PAGE APPLICABLE TO UNIT 2 ONLY

(a)

TABLE 2.2-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS -UNIT 2

FUNCTIONAL UNIT	TOTAL ALLOWANCE (TA)	Z	SENSOR ERROR (S)	TRIP SETPOINT	ALLOWABLE VALUE
9. Pressurizer Pressure-Low (PI-0455A,B&C, PI-0456 & PI-0456A, PI-0457 & PI-0457A, PI-0458 & PI-0458A)	3.1	0.71	1.67	≥1960 psig**	≥1950 psig
10. Pressurizer Pressure-High (PI-0455A,B&C, PI-0456 & PI-0456A, PI-0457 & PI-0457A, PI-0458 & PI-0458A)	3.1	0.71	1.67	≤2385 psig	≤2395 psig
11. Pressurizer Water Level-High (LI-0459A, LI-0460A, LI-0461)	8.0	2.18	1.67	≤92% of instrument span	≤93.9% of instrument span
12. Reactor Coolant Flow-Low (LOOP1 FI-0414 FI-0415 FI-0416; LOOP2 FI-0424 FI-0425 FI-0426; LOOP3 FI-0434 FI-0435 FI-0436; LOOP4 FI-0444 FI-0445 FI-0446)	2.5	1.87	0.60	≥90% of loop design flow*	≥89.4% of loop design flow*
13. Steam Generator Water Level Low-Low (LOOP1 LI-0517 LI-0518 LI-0519 LI-0551; LOOP2 LI-0527 LI-0528 LI-0529 LI-0552; LOOP3 LI-0537 LI-0538 LI-0539 LI-0553; LOOP4 LI-0547 LI-0548 LI-0549 LI-0554)	18.5 (21.8)***	17.18 (18.21)***	1.67	≥18.5% (37.8)*** of narrow range instrument span	≥17.8% (35.9)*** of narrow range instrument span
14. Undervoltage - Reactor Coolant Pumps	6.0	0.58	0	≥9600 volts (70% bus voltage)	≥9481 volts (69% bus voltage)
15. Underfrequency - Reactor Coolant Pumps	3.3	0.50	0	≥57.3 Hz	≥57.1 Hz

(13)

*Loop design flow = 95,700 gpm

**Time constants utilized in the lead-lag controller for Pressurizer Pressure-Low are 10 seconds for lead and 1 second for lag. CHANNEL CALIBRATION shall ensure that these time constants are adjusted to these values.

***The value stated inside the parenthesis is for instrument that has the lower tap at elevation 333"; the value stated outside the parenthesis is for instrumentation that has the lower tap at elevation 438".

THIS PAGE APPLICABLE TO UNIT 2 ONLY

(a)

TABLE 2.2-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS - UNIT 2

FUNCTIONAL UNIT	TOTAL ALLOWANCE (TA)	Z	SENSOR ERROR (S)	TRIP SETPOINT	ALLOWABLE VALUE
16. Turbine Trip					
a. Low Fluid Oil Pressure (PT-6161, PT-6162, PT-6163)	N.A.	N.A.	N.A.	≥580 psig	≥500 psig
b. Turbine Stop Valve Closure	N.A.	N.A.	N.A.	≥96.7% open	≥96.7% open
17. Safety Injection Input from ESF	N.A.	N.A.	N.A.	N.A.	N.A.
18. Reactor Trip System Interlocks					
a. Intermediate Range Neutron Flux, P-6 (NI-0035B, NI-0036B)	N.A.	N.A.	N.A.	$\geq 1 \times 10^{-10}$ amp	$\geq 6 \times 10^{-11}$ amp
b. Low Power Reactor Trips Block, P-7					
1) P-10 input (NI-0041B&C, NI-0042B&C, NI-0043B&C, NI-0044B&C)	N.A.	N.A.	N.A.	≤10% of RTP#	≤12.3% of RTP#
2) P-13 input (PI-0505, PI-0506)	N.A.	N.A.	N.A.	<10% RTP# Turbine Impulse Pressure Equivalent	<12.3% RTP# Turbine Impulse Pressure Equivalent
c. Power Range Neutron Flux, P-8 (NI-0041B&C, NI-0042B&C, NI-0043B&C, NI-0044B&C)	N.A.	N.A.	N.A.	≤48% of RTP#	≤50.3% of RTP#

2-8

(14)

#RTP = RATED THERMAL POWER

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a

TABLE 2.2-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS - UNIT 2

FUNCTIONAL UNIT	TOTAL ALLOWANCE (TA)	Z	SENSOR ERROR (S)	TRIP SETPOINT	ALLOWABLE VALUE
d. Power Range Neutron Flux, P-9 (NI-0041B&C, NI-0042B&C, NI-0043B&C, NI-0044B&C)	N.A.	N.A.	N.A.	≤50% of RTP#	≤52.3% of RTP#
e. Power Range Neutron Flux, P-10 (NI-0041B&C, NI-0042B&C, NI-0043B&C, NI-0044B&C)	N.A.	N.A.	N.A	≥10% of RTP#	≥7.7% of RTP#
f. Turbine Impulse Chamber Pressure, P-13 (PI-0505, PI-0506)	N.A.	N.A.	N.A.	<10% RTP# Turbine Impulse Pressure Equivalent	<12.3% RTP# Turbine Impulse Pressure Equivalent
19. Reactor Trip Breakers	N.A.	N.A.	N.A	N.A.	N.A.
20. Automatic Trip and Interlock Logic	N.A.	N.A.	N.A.	N.A.	N.A.

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#RTP = RATED THERMAL POWER

THIS PAGE APPLICABLE TO UNIT 2 ONLY

(a)

TABLE 2.2-1 (Continued)

TABLE NOTATIONS - UNIT 2

NOTE 1: OVERTEMPERATURE ΔT

$$\Delta T \frac{(1 + \tau_1 S)}{(1 + \tau_2 S)} \left(\frac{1}{1 + \tau_3 S}\right) \leq \Delta T_o \left\{K_1 - K_2 \frac{(1 + \tau_4 S)}{(1 + \tau_5 S)} \left[T \left(\frac{1}{1 + \tau_6 S}\right) - T'\right] + K_3(P - P') - f_1(\Delta I)\right\}$$

Where:

ΔT	=	Measured ΔT by RTD Manifold Instrumentation; (UNIT-2)
$\frac{1 + \tau_1 S}{1 + \tau_2 S}$	=	Lead-lag compensator on measured ΔT;
τ_1, τ_2	=	Time constants utilized in lead-lag compensator for ΔT, $\tau_1 \geq 8$ s, $\tau_2 \leq 3$ s;
$\frac{1}{1 + \tau_3 S}$	=	Lag compensator on measured ΔT;
τ_3	=	Time constants utilized in the lag compensator for ΔT, $\tau_3 = 0$ s;
ΔT_o	=	Indicated ΔT at RATED THERMAL POWER;
K_1	$\leq$	1.10;
K_2	=	0.012/°F; (UNIT 2)
$\frac{1 + \tau_4 S}{1 + \tau_5 S}$	=	The function generated by the lead-lag compensator for T_{avg} dynamic compensation;
τ_4, τ_5	=	Time constants utilized in the lead-lag compensator for T_{avg}, $\tau_4 \geq 28$ s, $\tau_5 \leq 4$ s;
T	=	Average temperature, °F;
$\frac{1}{1 + \tau_6 S}$	=	Lag compensator on measured T_{avg};
τ_6	=	Time constant utilized in the measured T_{avg} lag compensator, $\tau_6 = 0$ s;

THIS PAGE APPLICABLE TO UNIT 2 ONLY

(9)

TABLE 2.2-1 (Continued)

TABLE NOTATIONS (Continued) UNIT 2

NOTE 1: (Continued)

T'	$\leq$	588.5°F (Nominal T_{avg} ~~at RATED THERMAL POWER~~ operating temperature);
K_3	=	0.00056/psig; (UNIT 2)
P	=	Pressurizer pressure, psig;
P'	=	2235 psig (Nominal RCS operating pressure);
S	=	Laplace transform variable, s^{-1};

and $f_1(\Delta I)$ is a function of the indicated difference between top and bottom detectors of the power-range neutron ion chambers; with gains to be selected based on measured instrument response during plant startup tests such that: (UNIT 2)

(1) For $q_t - q_b$ between -33.5% and + 6.5%, $f_1(\Delta I) = 0$, 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;

(UNIT 2)

(2) For each percent that the magnitude of $q_t - q_b$ exceeds - 33.5%, the ΔT Trip Setpoint shall be automatically reduced by 1.27% of its value at RATED THERMAL POWER; and

(UNIT 2)

(3) For each percent that the magnitude of $q_t - q_b$ exceeds + 6.5%, the ΔT Trip Setpoint shall be automatically reduced by 0.83% of its value at RATED THERMAL POWER.

(UNIT 2)

(UNIT 2)

NOTE 2: The channel's maximum Trip Setpoint shall not exceed its computed Trip Setpoint by more than 2.5%.

(17)

THIS PAGE APPLICABLE TO UNIT 2 ONLY

TABLE . .-1 (Continued) (a)

TABLE NOTATIONS (Continued) — UNIT 2

NOTE 3: OVERPOWER ΔT

$$\Delta T \frac{(1 + \tau_1 S)}{(1 + \tau_2 S)} \left(\frac{1}{1 + \tau_3 S}\right) \leq \Delta T_o \{K_4 - K_5 \left(\frac{\tau_7 S}{1 + \tau_7 S}\right) \left(\frac{1}{1 + \tau_6 S}\right) T - K_6 [T \left(\frac{1}{1 + \tau_6 S}\right) - T''] - f_2(\Delta I)\}$$

Where: ΔT = Measured ΔT by RTD manifold instrumentation; (UNIT 2)

$\frac{1 + \tau_1 S}{1 + \tau_2 S}$ = Lead-lag compensator on measured ΔT;

τ_1, τ_2 = Time constants utilized in lead-lag compensator for ΔT, $\tau_1 \geq 8$ S, $\tau_2 \leq 3$ s;

$\frac{1}{1 + \tau_3 S}$ = Lag compensator on measured ΔT;

τ_3 = Time constants utilized in the lag compensator for ΔT, τ_3=0s;

ΔT_o = Indicated ΔT at RATED THERMAL POWER;

K_4 $\leq$ 1.089, (UNIT 2)

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

$\frac{\tau_7 S}{1 + \tau_7 S}$ = The function generated by the rate-lag compensator for T_{avg} dynamic compensation,

τ_7 = Time constants utilized in the rate-lag compensator for T_{avg}, $\tau_7 \geq 10$ s,

$\frac{1}{1 + \tau_6 S}$ = Lag compensator on measured T_{avg};

THIS PAGE APPLICABLE TO UNIT 2 ONLY

a

TABLE 2.2-1 (Continued)

TABLE NOTATIONS (Continued) - UNIT 2

NOTE 3: (Continued)

τ_6	=	Time constant utilized in the measured T_{avg} lag compensator, τ_6 = 0 s; (UNIT 2)
K_6	$\geq$	0.0013/°F for T > T" and K_6 = 0 for T $\leq$ T",
T	=	Average Temperature, °F;
T"	=	Indicated T_{avg} at RATED THERMAL POWER (Calibration temperature for ΔT instrumentation, $\leq$ 588.5°F), (UNIT 2)
S	=	Laplace transform variable, s^{-1}; and
$f_2(\Delta I)$	=	0 for all ΔI.

NOTE 4: The channel's maximum Trip Setpoint shall not exceed its computed Trip Setpoint by more than 2.4% of ΔT span. (UNIT 2)

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THIS PAGE APPLICABLE TO UNIT 1 ONLY

2.1 SAFETY LIMITS

BASES

2.1.1 REACTOR CORE - UNIT 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.

correlations which have

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 (R Grid) correlation. The W-3 (R Grid) DNB correlation has~~ been developed to predict the DNB flux and the location of DNB for axially uniform and nonuniform heat flux distributions. The local DNB heat flux ratio (DNBR) is defined as the ratio of the heat flux that would cause DNB at a particular core location to the local heat flux and 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% probability at a 95% confidence level that DNB will not occur and is chosen as an appropriate margin to DNB for all operating conditions.

Replace with Insert 1

The curves of Figure 2.1-1 show reactor core safety limits ~~which are determined~~ for a range of ~~reactor operating conditions. The core limits represent the loci of points of~~ THERMAL POWER, REACTOR COOLANT SYSTEM pressure and average temperature which satisfy the following criteria:

less than

A. The average enthalpy at the vessel exit is ~~equal to~~ the enthalpy of saturated liquid (far left line segment in each curve).

B. The minimum DNBR is not less than the design limit (all the other line segments in each curve).

C. The hot channel exit quality is not greater than the upper limit of the quality range of the W-3 (R-Grid) correlation which is 15% (middle line segment on Reactor Coolant System pressure curves, 2400 psia and 2250 psia; this is not a limiting criterion for this plant).

Replace with Insert 2

INSERT 1

The DNB thermal design criterion is that the probability that DNB will not occur on the most limiting rod is at least 95% (at a 95% confidence level) for any Condition I or II event.

In meeting the DNB design criterion, uncertainties in plant operating parameters, nuclear and thermal parameters, fuel fabrication parameters and computer codes must be considered. As described in the FSAR, the effects of these uncertainties have been statistically combined with the correlation uncertainty. Design limit DNBR values have been determined that satisfy the DNB design criterion.

Additional DNBR margin is maintained by performing the safety analyses to a higher DNBR limit. This margin between the design and safety analysis limit DNBR values is used to offset known DNBR penalties (e.g., rod bow and transition core) and to provide DNBR margin for operating and design flexibility.

INSERT 2

B. The minimum DNBR satisfies the DNB design criterion (all the other line segments in each curve). The VANTAGE 5 fuel is analyzed using the WRB-2 correlation with design limit DNBR values of 1.24 and 1.23 for the typical cell and thimble cells, respectively. The LOPAR fuel is analyzed using the WRB-1 correlation with design limit DNBR values of 1.23 and 1.22 for the typical and thimble cells, respectively.

C. The hot channel exit quality is not greater than the upper limit of the quality range (including the effect of uncertainties) of the DNB correlations. This is not a limiting criterion for this plant.

THIS PAGE APPLICABLE TO UNIT 2 ONLY

2.1.1 REACTOR CORE - UNIT 2

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 (R Grid) correlation. The W-3 (R Grid) DNB correlation has been developed to predict the DNB flux and the location of DNB for axially uniform and nonuniform heat flux distributions. The local DNB heat flux ratio (DNBR) is defined as the ratio of the heat flux that would cause DNB at a particular core location to the local heat flux and 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% probability at a 95% confidence level that DNB will not occur and is chosen as an appropriate margin to DNB for all operating conditions.

The curves of Figure 2.1-1 (a) show reactor core safety limits which are determined for a range of reactor operating conditions. The core limits represent the loci of points of THERMAL POWER, REACTOR COOLANT SYSTEM pressure and average temperature which satisfy the following criteria:

A. The average enthalpy at the vessel exit is equal to the enthalpy of saturated liquid (far left line segment in each curve).

B. The minimum DNBR is not less than the design limit (all the other line segments in each curve).

C. The hot channel exit quality is not greater than the upper limit of the quality range of the W-3 (R-Grid) correlation which is 15% (middle line segment on Reactor Coolant System pressure curves, 2400 psia and 2250 psia; this is not a limiting criterion for this plant).

REACTOR CORE (Continued)

These curves are based on an enthalpy hot channel factor, $F^{RTP}_{\Delta H}$ ~~$F^{N}_{\Delta H}$ of 1.55~~ 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:

$$F^{N}_{\Delta H} = F^{RTP}_{\Delta H}\ \cancel{1.55}\ [1 + PF_{\Delta H}\ \cancel{0.3}\ (1-P)]$$

Where: $F^{RTP}_{\Delta H}$ is the limit at RATED THERMAL POWER (RTP) specified in the CORE OPERATING LIMITS REPORT (COLR),
$PF_{\Delta H}$ is the Power Factor Multiplier for $F^{N}_{\Delta H}$ specified in the COLR, and

~~Where~~ P is the fraction of ~~RATED THERMAL POWER~~ RTP.

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 (ΔI) function of the Overtemperature trip. When the axial power imbalance is not within the tolerance, the axial power imbalance effect on the Overtemperature ΔT trips will reduce the Setpoints to provide protection consistent with core Safety Limits.

2.1.2 REACTOR COOLANT SYSTEM PRESSURE

The restriction of this Safety Limit protects the integrity of the Reactor Coolant System (RCS) from overpressurization and thereby prevents the release of radionuclides contained in the reactor coolant from reaching the containment atmosphere.

The reactor vessel, pressurizer, and the RCS piping, valves and fittings are designed to Section III of the ASME Code for Nuclear Power Plants which permits a maximum transient pressure of 110% (2735 psig) of design pressure. The Safety Limit of 2735 psig is therefore consistent with the design criteria and associated Code requirements.

The entire RCS is hydrotested at 125% (3107 psig) of design pressure, to demonstrate integrity prior to initial operation.

BORATED WATER SOURCE - SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.1.2.5 As a minimum, one of the following borated water sources shall be OPERABLE:

a. A Boric Acid Storage Tank with:

1) A minimum contained borated water volume of 9504 gallons (19% of instrument span) (LI-102A, LI-104A),

2) A boron concentration between 7000 ppm and 7700 ppm, and

3) A minimum solution temperature of 65°F (TI-0103).

b. The refueling water storage tank (RWST) with:

1) A minimum contained borated water volume of 99404 gallons (9% of instrument span) (LI-0990A&B, LI-0991A&B, LI-0992A, LI-0993A),

2) A boron concentration between 2400 ppm and 2600 ppm, and

3) A minimum solution temperature of $\boxed{54°F}$ (TI-10982).

44°F (UNIT 1) or 54°F (UNIT 2)

APPLICABILITY: MODES 5 and 6.

ACTION:

With no borated water source OPERABLE, suspend all operations involving CORE ALTERATIONS or positive reactivity changes.

SURVEILLANCE REQUIREMENTS

4.1.2.5 The above required borated water source shall be demonstrated OPERABLE:

a. At least once per 7 days by:

1) Verifying the boron concentration of the water,

2) Verifying the contained borated water volume, and

3) When the boric acid storage tank is the source of borated water and the ambient temperature of the boric acid storage tank room (TISL-20902, TISL-20903) is $\leq$72°F, verify the boric acid storage tank solution temperature is $\geq$ 65°F.

b. At least once per 24 hours by verifying the RWST temperature (TI-10982) when it is the source of borated water and the outside air temperature is less than $\boxed{50°F}$.

40°F (UNIT 1) or 50°F (UNIT 2)

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 Tank with:

   1) A minimum contained borated water volume of 36674 gallons (81% of instrument span) (LI-102A, LI-104A),

   2) A boron concentration between 7000 ppm and 7700 ppm, and

   3) A minimum solution temperature of 65°F (TI-0103).

b. The refueling water storage tank (RWST) with:

   1) A minimum contained borated water volume of 631478 gallons (86% of instrument span) (LI-0990A&B, LI-0991A&B, LI-0992A, LI-0993A),

   2) A boron concentration between 2400 ppm and 2600 ppm,

   3) A minimum solution temperature of $\boxed{54°F}$,
   44°F (UNIT 1) or 54°F (UNIT 2)

   4) A maximum solution temperature of 116°F (TI-10982), and

   5) RWST Sludge Mixing Pump Isolation Valves capable of closing on RWST low-level.

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

ACTION:

a. With the Boric Acid Storage Tank inoperable and being used as one of the above required borated water sources, restore the tank to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and borated to a SHUTDOWN MARGIN as required by Figure 3.1-2 at 200°F; restore the Boric Acid Storage Tank to OPERABLE status within the next 7 days or be in COLD SHUTDOWN within the next 30 hours.

b. With the RWST inoperable, except for the Sludge Mixing Pump Isolation Valves, restore the tank to OPERABLE status within 1 hour or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

ACTION (Continued)

c. With a Sludge Mixing Pump Isolation Valve(s) inoperable, restore the valve(s) to OPERABLE status within 24 hours or isolate the sludge mixing system by either closing the manual isolation valves or deenergizing the OPERABLE solenoid pilot valve within 6 hours and maintain closed.

SURVEILLANCE REQUIREMENTS

4.1.2.6 Each borated water source shall be demonstrated OPERABLE:

a. At least once per 7 days by:

1) Verifying the boron concentration in the water,

2) Verifying the contained borated water volume of the water source, and

3) When the boric acid storage tank is the source of borated water and the ambient temperature of the boric acid storage tank room (TISL-20902, TISL-20903) is $\leq$ 72°F, verify the boric acid storage tank solution temperature is $\geq$ 65°F.

b. At least once per 24 hours by verifying the RWST temperature (TI-10982) when the outside air temperature is less than ~~50°F~~.

c. At least once per 18 months by verifying that the Sludge Mixing Pump Isolation Valves automatically close upon RWST low-level test signal.

40°F (UNIT 1) or 50°F (UNIT 2)

ROD DROP TIME

2.7 (UNIT 1) or 2.2 (UNIT 2)

LIMITING CONDITION FOR OPERATION

3.1.3.4 The individual shutdown and control rod drop time from the physical fully withdrawn position shall be less than or equal to $\boxed{2.2}$ seconds from beginning of decay of stationary gripper coil voltage to dashpot entry with:

a. T_{avg} (TI-0412, TI-0422, TI-0432, TI-0442) 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 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 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 Section 3/4.2.1 for Unit 1 Only with the next page.

3/4.2 POWER DISTRIBUTION LIMITS

3/4.2.1 AXIAL FLUX DIFFERENCE

LIMITING CONDITION FOR OPERATION

3.2.1 The indicated (NI-0041B, NI-0042B, NI-0043B, NI-0044B) AXIAL FLUX DIFFERENCE (AFD) shall be maintained within the target band (flux difference units) about the target flux difference. The target band is specified in the CORE OPERATING LIMITS REPORT (COLR).

The indicated AFD may deviate outside the required target band at greater than or equal to 50% but less than 90% of RATED THERMAL POWER provided the indicated AFD is within the Acceptable Operation Limits specified in the COLR and the cumulative penalty deviation time does not exceed 1 hour during the previous 24 hours.

The indicated AFD may deviate outside the required target band at greater than 15% but less than 50% of RATED THERMAL POWER provided the cumulative penalty deviation time does not exceed 1 hour during the previous 24 hours.

APPLICABILITY: MODE 1, above 15% of RATED THERMAL POWER.* #

ACTION:

a. With the indicated AFD outside of the required target band and with THERMAL POWER greater than or equal to 90% of RATED THERMAL POWER, within 15 minutes either:

   1. Restore the indicated AFD to within the target band limits, or

   2. Reduce THERMAL POWER to less than 90% of RATED THERMAL POWER.

b. With the indicated AFD outside of the required target band for more than 1 hour of cumulative penalty deviation time during the previous 24 hours or outside the Acceptable Operation Limits specified in the COLR and with THERMAL POWER less than 90% but equal to or greater than 50% of RATED THERMAL POWER, reduce:

   1. THERMAL POWER to less than 50% of RATED THERMAL POWER within 30 minutes, and

   2. The Power Range Neutron Flux* - High Setpoints to less than or equal to 55% of RATED THERMAL POWER within the next 4 hours.

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* See Special Test Exceptions Specification 3.10.2.

\# Surveillance testing of the Power Range Neutron Flux Channel may be performed (below 90% of RATED THERMAL POWER) pursuant to Specification 4.3.1.1 provided the indicated AFD is maintained within the Acceptable Operation Limits specified in the COLR. A total of 16 hours operation may be accumulated with the AFD outside of the above required target band during testing without penalty deviation.

3/4.2 POWER DISTRIBUTION LIMITS

3/4.2.1 A IAL FLUX DIFFERENCE - UNIT 1

Replacement Section 3/4.2.1 for UNIT 1 Only.

LIMITING CONDITION FOR OPERATION

3.2.1 The indicated (NI-0041B, NI-0042B, NI-0043B, NI-0044B) AXIAL FLUX DIFFERENCE (AFD) shall be maintained within the limits specified in the CORE OPERATING LIMITS REPORT (COLR).

APPLICABILITY: MODE 1 ABOVE 50 PERCENT RATED THERMAL POWER*.

ACTION:

a. With the indicated AXIAL FLUX DIFFERENCE outside of the limits specified in the COLR,

   1. Either restore the indicated AFD to within the 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 percent of RATED THERMAL POWER within the next 4 hours.

b. THERMAL POWER shall not be increased above 50% of RATED THERMAL POWER unless the indicated AFD is within the limits specified in the COLR.

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:

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

   2) At least once per hour until the AFD Monitor Alarm is updated after restoration to OPERABLE status.

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.

c. The provisions of Specification 4.0.4 are not applicable.

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 its limits.

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*See Special Test Exceptions Specification 3.10.2.

THIS PAGE APPLICABLE TO UNIT 2 ONLY

3/4.2 POWER DISTRIBUTION LIMITS

3/4.2.1 AXIAL FLUX DIFFERENCE - UNIT 2

LIMITING CONDITION FOR OPERATION

3.2.1 The indicated (NI-0041B, NI-0042B, NI-0043B, NI-0044B) AXIAL FLUX DIFFERENCE (AFD) shall be maintained within the target band (flux difference units) about the target flux difference. The target band is specified in the CORE OPERATING LIMITS REPORT (COLR).

The indicated AFD may deviate outside the required target band at greater than or equal to 50% but less than 90% of RATED THERMAL POWER provided the indicated AFD is within the Acceptable Operation Limits specified in the COLR and the cumulative penalty deviation time does not exceed 1 hour during the previous 24 hours.

The indicated AFD may deviate outside the required target band at greater than 15% but less than 50% of RATED THERMAL POWER provided the cumulative penalty deviation time does not exceed 1 hour during the previous 24 hours.

APPLICABILITY: MODE 1, above 15% of RATED THERMAL POWER.* #

ACTION:

a. With the indicated AFD outside of the required target band and with THERMAL POWER greater than or equal to 90% of RATED THERMAL POWER, within 15 minutes either:

   1. Restore the indicated AFD to within the target band limits, or

   2. Reduce THERMAL POWER to less than 90% of RATED THERMAL POWER.

b. With the indicated AFD outside of the required target band for more than 1 hour of cumulative penalty deviation time during the previous 24 hours or outside the Acceptable Operation Limits specified in the COLR and with THERMAL POWER less than 90% but equal to or greater than 50% of RATED THERMAL POWER, reduce:

   1. THERMAL POWER to less than 50% of RATED THERMAL POWER within 30 minutes, and

   2. The Power Range Neutron Flux* - High Setpoints to less than or equal to 55% of RATED THERMAL POWER within the next 4 hours.

---

* See Special Test Exceptions Specification 3.10.2.

Surveillance testing of the Power Range Neutron Flux Channel may be performed (below 90% of RATED THERMAL POWER) pursuant to Specification 4.3.1.1 provided the indicated AFD is maintained within the Acceptable Operation Limits specified in the COLR. A total of 16 hours operation may be accumulated with the AFD outside of the above required target band during testing without penalty deviation.

THIS PAGE APPLICAB = TO UNIT 2 ONLY

LIMITING CONDITION FOR OPERATION - UNIT 2

ACTION (Continued)

c. With the indicated AFD outside of the required target band for more than 1 hour of cumulative penalty deviation time during the previous 24 hours and with THERMAL POWER less than 50% but greater than 15% of RATED THERMAL POWER, the THERMAL POWER shall not be increased equal to or greater than 50% of RATED THERMAL POWER until the indicated AFD is within the required target band and the cumulative penalty deviation has been reduced to less than 1 hour in the previous 24 hours.

SURVEILLANCE REQUIREMENTS

4.2.1.1 The indicated AFD 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 until the AFD Monitor Alarm is updated after restoration to OPERABLE status.

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 target band when two or more OPERABLE excore channels are indicating the AFD to be outside the target band. Penalty deviation outside of the required target band shall be accumulated on a time basis of:

a. One minute penalty deviation for each 1 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 1 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 Specification 4.2.1.3 above or by linear interpolation between the most recently measured value and 0% at the end of the cycle life. The provisions of Specification 4.0.4 are not applicable.

FIGURE 3.2-1 (DELETED)

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:

$$F_Q(Z) \leq \frac{F_Q^{RTP}}{P} [K(Z)] \text{ for } P > 0.5$$

$$F_Q(Z) \leq \frac{F_Q^{RTP}}{0.5} [K(Z)] \text{ for } P \leq 0.5$$

Where: F_Q^{RTP} = the F_Q limit at RATED THERMAL POWER (RTP) specified in the CORE OPERATING LIMITS REPORT (COLR),

Where: $P = \frac{\text{THERMAL POWER}}{\text{RATED THERMAL POWER}}$, and

K(Z) = the normalized $F_Q(Z)$ as a function of core height specified in the COLR.

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 ΔT Trip Setpoints have been reduced at least 1% (in ΔT span) for each 1% $F_Q(Z)$ exceeds the limit; and (Value of K4)

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

FIGURE 3.2-2 (DELETED)

Replace F_{xy} Surveillance Requirement Sections 4.2.2.1, 4.2.2.2 and 4.2.2.3 with new F_Q Surveillance Requirement Sections 4.2.2.1, 4.2.2.2 and 4.2.2.3 for UNIT 1 ONLY (Pages 2-4, 2-5, 2-6)

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 before exceeding 75% of RATED THERMAL POWER following each fuel loading.

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 Specification 4.2.2.2b., above to:

1) The F_{xy} limits for RATED THERMAL POWER (F_{xy}^{RTP}) for the appropriate measured core planes given in Specification 4.2.2.2e. and f., below, and

2) The relationship:

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

Where F_{xy}^L is the limit for fractional THERMAL POWER operation expressed as a function of F_{xy}^{RTP}, PF_{xy} is the power factor multiplier for F_{xy} specified in the COLR, 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 either:

a) 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 Effective Full Power Days (EFPD), whichever occurs first.

Replacement page for F_Q Surveillance for UNIT 1 Only

SURVEILLANCE REQUIREMENTS - UNIT 1

4.2.2.1 The provisions of Specifications 4.0.4 are not applicable.

4.2.2.2 $F_Q(Z)$ shall be evaluated to determine if it 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. Determining the computed heat flux hot channel factor, $F_Q^C(Z)$, as follows:

Increase the measured $F_Q(Z)$ obtained from the power distribution map by 3% to account for manufacturing tolerances and further increase the value by 5% to account for measurement uncertainties.

c. Verifying that $F_Q^C(Z)$, obtained in Specification 4.2.2.2b above, satisfies the relationship in Specification 3.2.2.

d. Satisfying the following relationship:

$$F_Q^C(Z) \leq \frac{F_Q^{RTP} \times K(Z)}{P \times W(Z)} \text{ for } P > 0.5$$

$$F_Q^C(Z) \leq \frac{F_Q^{RTP} \times K(Z)}{0.5 \times W(Z)} \text{ for } P \leq 0.5$$

Where $F_Q^C(Z)$ is obtained in Specification 4.2.2.2b above, F_Q^{RTP} is the F_Q limit, K(Z) is the normalized $F_Q(Z)$ as a function of core height, P is the fraction of RATED THERMAL POWER, and W(Z) is the cycle dependent function that accounts for power distribution transients encountered during normal operation. F_Q^{RTP}, K(Z), and W(Z) are specified in the CORE OPERATING LIMITS REPORT as per Specification 6.8.1.6.

e. Measuring $F_Q(Z)$ according to the following schedule:

1. Upon achieving equilibrium conditions after exceeding by 20% or more of RATED THERMAL POWER, the THERMAL POWER at which FQ(Z) was last determined*, or

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

*During power escalation after each fuel loading, power level may be increased until equilibrium conditions at any power level greater than or equal to 50% of RATED THERMAL POWER have been achieved and a power distribution map obtained.

Replacement page for F_Q Surveillance for UNIT 1 Only

THIS PAGE APPLICABLE TO UNIT 1 ONLY

SURVEILLANCE REQUIREMENTS (Continued) - UNIT 1

f. With measurements indicating

$$\underset{\text{over } Z}{\text{maximum}} \left(\frac{F_Q^C(Z)}{K(Z)} \right)$$

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

1) Increase $F_Q^C(Z)$ by 2% and verify that this value satisfies the relationship in Specification 4.2.2.2d, or

2) $F_Q^C(Z)$ shall be measured at least once per 7 Effective Full Power Days until two successive maps indicate that

$$\underset{\text{over } Z}{\text{maximum}} \left(\frac{F_Q^C(Z)}{K(Z)} \right) \quad \text{is not increasing.}$$

g. With the relationships specified in Specification 4.2.2.2d above not being satisfied:

1) Calculate the percent $F_Q(Z)$ exceeds its limits by the following expression:

$$\left\{ \left(\underset{\text{over } Z}{\text{maximum}} \left[\frac{F_Q^C(Z) \times W(Z)}{\frac{F_Q^{RTP}}{P} \times K(Z)} \right] \right) - 1 \right\} \times 100 \text{ for } P > 0.5$$

$$\left\{ \left(\underset{\text{over } Z}{\text{maximum}} \left[\frac{F_Q^C(Z) \times W(Z)}{\frac{F_Q^{RTP}}{0.5} \times K(Z)} \right] \right) - 1 \right\} \times 100 \text{ for } P \le 0.5, \text{ and}$$

2) The following action shall be taken.

Within 15 minutes, control the AFD to within new AFD limits which are determined by reducing the AFD limits specified in the CORE OPERATING LIMITS REPORT by 1% AFD for each percent $F_Q(Z)$ exceeds its limits as determined in Specification 4.2.2.2g.1. Within 8 hours, reset the AFD alarm setpoints to these modified limits.

Replacement page for F_Q Surveillance for UNIT 1 Only

POWER DISTRIBUTION LIMITS

SURVEILLANCE REQUIREMENTS (Continued) - UNIT 1

h. The limits specified in Specification 4.2.2.2c are applicable in all core plane regions, i.e. 0 - 100%, inclusive.

i. The limits specified in Specifications 4.2.2.2d, 4.2.2.2f, and 4.2.2.2g 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 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.

THIS PAGE APPLICABLE TO UNIT 2 ONLY

SURVEILLANCE REQUIREMENTS - UNIT 2

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 before exceeding 75% of RATED THERMAL POWER following each fuel loading.

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 Specification 4.2.2.2b., above to:

1) The F_{xy} limits for RATED THERMAL POWER (F_{xy}^{RTP}) for the appropriate measured core planes given in Specification 4.2.2.2e. and f., below, and

2) The relationship:

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

Where F_{xy}^L is the limit for fractional THERMAL POWER operation expressed as a function of F_{xy}^{RTP}, PF_{xy} is the power factor multiplier for F_{xy} specified in the COLR, 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 either:

a) 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 Effective Full Power Days (EFPD), whichever occurs first.

THIS PAGE APPLICABLE TO UNIT 2 ONLY

SURVEILLANCE REQUIREMENTS (Continued) — UNIT 2

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 used in the Constant Axial Offset Control analysis for RATED THERMAL POWER (F_{xy}^{RTP}) shall be specified for all core planes containing Bank "D" control rods and all unrodded core planes in the COLR per Specification 6.8.1.6;

f. The F_{xy} limits of Specification 4.2.2.2e., 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, and

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.

2199 psig* (UNIT 1) or 2224 psig* (UNIT 2)

3/4.2.5 DNB PARAMETERS

LIMITING CONDITION FOR OPERATION

3.2.5 The following DNB-related parameters shall be maintained within the limits:

a. Reactor Coolant System T_{avg} (TI-0412, TI-0422, TI-0432, TI-0442), ≤ ~~591°F~~
592.5°F (UNIT 1) or 591°F (UNIT 2)

b. Pressurizer Pressure (PI-0455A,B&C, PI-0456 & PI-0456A, PI-0457 & PI-0457A, PI-0458 & PI-0458A), ≥ ~~2224 psig*~~

c. Reactor Coolant System Flow (FI-0414, FI-0415, FI-0416, FI-0424, FI-0425, FI-0426, FI-0434, FI-0435, FI-0436, FI-0444, FI-0445, FI-0446) ≥~~396,198 gpm **~~
391,225 gpm** (UNIT 1) or 396,198 gpm** (UNIT 2)

APPLICABILITY: MODE 1.

ACTION:

With any of the above parameters exceeding its limit, restore the parameter to within its limit within 2 hours or reduce THERMAL POWER to less than 5% of RATED THERMAL POWER within the next 4 hours.

SURVEILLANCE REQUIREMENTS

4.2.5.1 Reactor Coolant System T_{avg} and Pressurizer Pressure shall be verified to be within their limits at least once per 12 hours. RCS flow rate shall be monitored for degradation at least once per 12 hours. In the event of flow degradation, RCS flow rate shall be determined by precision heat balance within 7 days of detection of flow degradation.

4.2.5.2 The RCS flow rate indicators shall be subjected to CHANNEL CALIBRATION at each fuel loading and at least once per 18 months.

4.2.5.3 After each fuel loading, the RCS flow rate shall be determined by precision heat balance prior to operation above 75% RATED THERMAL POWER. The RCS flow rate shall also be determined by precision heat balance at least once per 18 months. Within 7 days prior to performing the precision heat balance flow measurement, the instrumentation used for performing the precision heat balance shall be calibrated. The provisions of 4.0.4 are not applicable for performing the precision heat balance flow measurement.

---

*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.

**Includes a ~~3.5%~~ flow measurement uncertainty.
2.2% (UNIT 1) or 3.5% (UNIT 2)

TABLE 3.3-3 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TOTAL ALLOWANCE (TA)	Z	SENSOR ERROR (S)	TRIP SETPOINT	ALLOWABLE VALUE
7. Semi-Automatic Switchover to Containment Emergency Sump (Continued)					
b. RWST Level--Low-Low Coincident With Safety Injection (LI-0990A&B, LI-0991A&B, LI-0992A, LI-0993A)	3.5	0.71	1.67	≥275.3 in. from tank base (≥39.1% of instrument span)	≥264.9 in. from tank base (≥37.4% of instrument span)
8. Loss of Power to 4.16 kV ESF Bus					
a. 4.16 kV ESF Bus Undervoltage-Loss of Voltage	N.A.	N.A.	N.A.	≥2975 volts with a ≤ 0.8 second time delay.	≥2912 volts with a ≤ 0.8 second time delay.
b. 4.16 kV ESF Bus Undervoltage-Degraded Voltage	N.A.	N.A.	N.A.	≥3746 volts with a ≤20 second time delay.	≥3683 volts with a ≤20 second time delay.
9. Engineered Safety Features Actuation System Interlocks					
a. Pressurizer Pressure, P-11 (PI-0455A,B&C, PI-0456 & PI-0456A, PI-0457 & PI-0457A, PI-0458 & PI-0458A)	N.A.	N.A.	N.A.	≤ 2000psig (UNIT 1) ≤ 1970 psig (UNIT 2)	≤ 2010 psig (Unit 1) ≤ 1980 psig (UNIT 2)
b. Reactor Trip, P-4	N.A.	N.A.	N.A.	N.A.	N.A.

3/4.5.1 ACCUMULATORS

b. UNIT 1: A contained borated water volume of between 6555 (29.2% of instrument span) and 6909 gallons (70.7% of instrument span) (LI-0950, LI-0951, LI-0952, LI-0953, LI-0954, LI-0955, LI-0956, LI-0957),

LIMITING CONDITION FOR OPERATION

3.5.1 Each Reactor Coolant System (RCS) accumulator shall be OPERABLE with:

a. The isolation valve open,

UNIT 2:

$\boxed{\text{b.}}$ A contained borated water volume of between 6616 (36% of instrument span) and 6854 gallons (64% of instrument span) (LI-0950, LI-0951, LI-0952, LI-0953, LI-0954, LI-0955, LI-0956, LI-0957),

c. A boron concentration of between 1900 ppm and 2600 ppm, and

d. A nitrogen cover-pressure of between 617 and 678 psig. (PI-0960A&B, PI-0961A&B, PI-0962A&B, PI-0963A&B, PI-0964A&B, PI-0965A&B, PI-0966A&B, PI-0967A&B).

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 1 hour or be in at least HOT STANDBY within the next 6 hours and reduce pressurizer pressure to less than 1000 psig 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 6 hours and reduce pressurizer pressure to less than 1000 psig within the following 6 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 (HV-8808A, B, C, D).

*Pressurizer pressure above 1000 psig.

BORON INJECTION SYSTEM

3/4.5.4 REFUELING WATER STORAGE TANK

LIMITING CONDITION FOR OPERATION

3.5.4 The refueling water storage tank (RWST) shall be OPERABLE with:

a. A minimum contained borated water volume of 631,478 gallons (86% of instrument span) (LI-0990A&B, LI-0991A&B, LI-0992A, LI-0993A).

b. A boron concentration of between 2400 ppm and 2600 ppm of boron,

c. A minimum solution temperature of $\boxed{54^\circ F}$, and 44°F (UNIT 1) or 54°F (UNIT 2)

d. A maximum solution temperature of 116°F (TI-10982)

e. RWST Sludge Mixing Pump Isolation valves capable of closing on RWST low-level.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

a. With the RWST inoperable except for the Sludge Mixing Pump Isolation Valves, restore the tank to OPERABLE status within 1 hour or be in at least HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With a Sludge Mixing Pump Isolation Valve(s) inoperable, restore the valve(s) to OPERABLE status within 24 hours or isolate the sludge mixing system by either closing the manual isolation valves or deenergizing the OPERABLE solenoid pilot valve within 6 hours and maintain closed.

SURVEILLANCE REQUIREMENTS

4.5.4 The RWST shall be demonstrated OPERABLE:

a. At least once per 7 days by:
   1) Verifying the contained borated water volume in the tank, and
   2) Verifying the boron concentration of the water.

b. At least once per 24 hours by verifying the RWST temperature when the outside air temperature is less than $\boxed{50^\circ F}$.

c. At least once per 18 months by verifying that the sludge mixing pump isolation valves automatically close upon an RWST low-level test signal.

40°F (UNIT 1) or 50°F (UNIT 2)

THIS PAGE APPLICABLE TO UNIT 1 ONLY

meeting the DNB design criterion

BASES

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 minimum DNBR in the core greater than or equal to 1.30~~ 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; and

$F^N_{\Delta H}$ 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~~; and~~.

~~$F_{xy}(Z)$ 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

The limits on AXIAL FLUX DIFFERENCE (AFD) assure that the $F_Q(Z)$ upper bound envelope of the F_Q limit specified in the CORE OPERATING LIMITS REPORT (COLR) times K(Z) 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 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.~~

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AXIAL FLUX DIFFERENCE (Continued)

~~Although it is intended that the plant will be operated with the AFD within the target band required by Specification 3.2.1 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 specified in the COLR 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 1-minute average of each of the OPERABLE excore detector outputs and provides an alarm message immediately if the AFD for two 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.~~

50%

allowed AI power operating space for RAOC operation specified within the COLR.

~~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 NUCLEAR ENTHALPY RISE HOT CHANNEL FACTOR - $F^N_{\Delta H}$

The limits on heat flux hot channel factor 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.

(3)

is

, (2) the DNB design criterion is met,

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 ensure that the limits are maintained provided:

a. Control rods in a single group move together with no individual rod insertion differing by more than ± 12 steps, indicated, from the group demand position;

b. Control rod banks are sequenced with a constant tip-to-tip distance between banks as ~~defined by Figure 3.1-3.~~

described in Specification 3.1.3.6.

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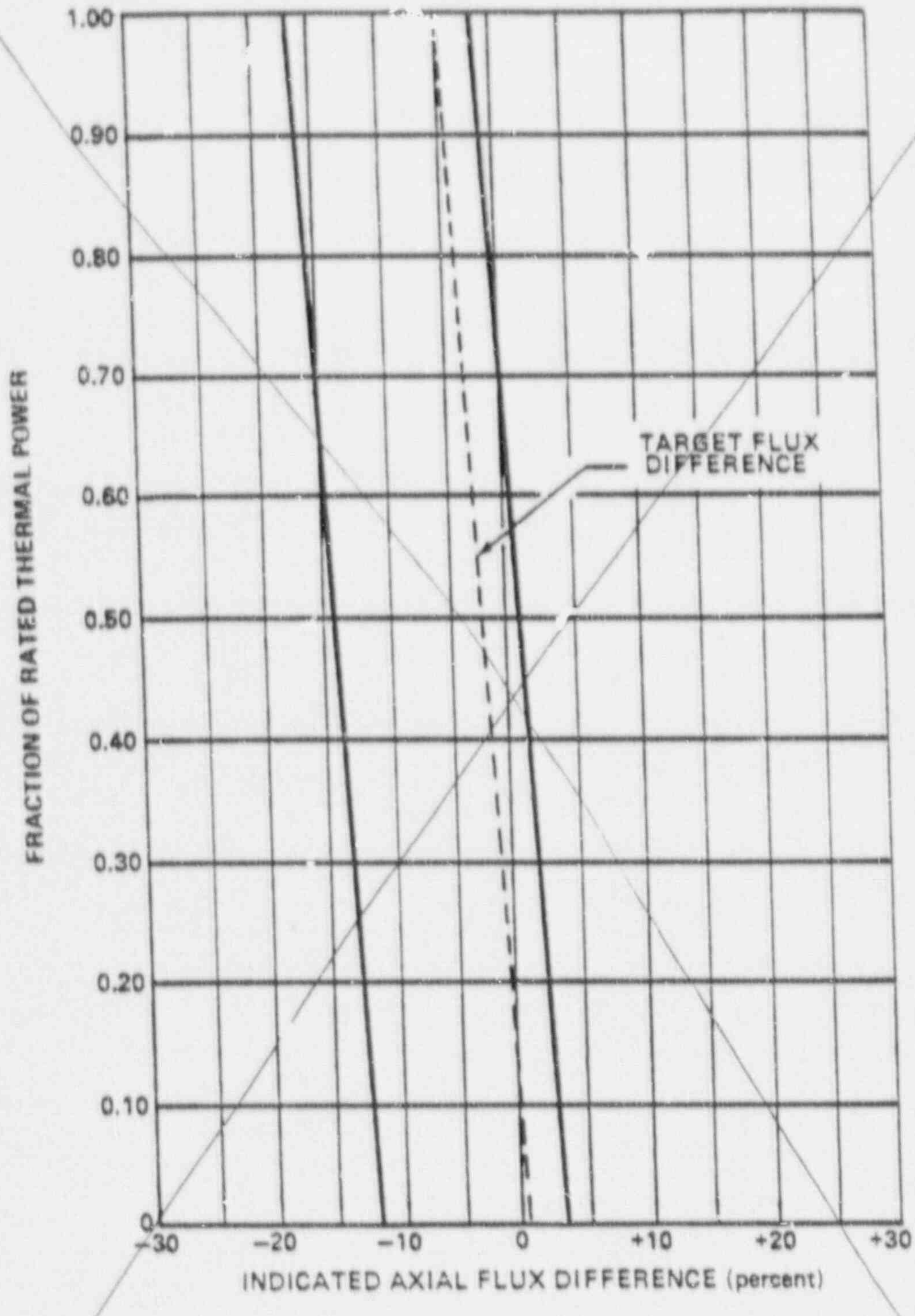

FIGURE B 3/4 2-1

TYPICAL INDICATED AXIAL FLUX DIFFERENCE VERSUS THERMAL POWER

HEAT FLUX HOT CHANNEL FACTOR 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; and

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. 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.

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.

INSERT S HERE

When $F^N_{\Delta H}$ is measured, (i.e., inferred), measurement uncertainty (i.e., the appropriate uncertainty on the incore inferred hot rod peaking factor) must be allowed for and 4% is the appropriate allowance for a full core map taken with the incore detection system.

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

~~a. Design limit DNBR of 1.30 vs 1.28,~~

~~b. Grid Spacing (K_s) of 0.046 vs 0.059,~~

~~c. Thermal Diffusion Coefficient of 0.038 vs 0.059,~~

~~d. DNBR Multiplier of 0.86 vs 0.88, and~~

~~e. Pitch reduction.~~

~~The applicable values of rod bow penalties are referenced in the FSAR.~~

INSERT 5

The heat flux hot channel factor $F_Q(Z)$ is measured periodically and increased by a cycle and height dependent power factor appropriate to RAOC operation, W(Z), to provide assurance that the limit on the heat flux hot channel factor, $F_Q(Z)$ is met. W(Z) accounts for the effects of normal operation transients within the AFD band and was determined from expected power control manuevers over the full range of burnup conditions in the core. The W(Z) function for normal operation and the AFD band are provided in the CORE OPERATING LIMITS REPORT per Specification 6.8.1.6.

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

~~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 F_{xy} limit for RATED THERMAL POWER (F_{xy}^{RTP}) as specified in the COLR per Specification 6.8.1.6 was determined from expected power control manuevers over the full range of burnup conditions in the core.~~

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 limit of 1.02 was selected to provide an allowance for the uncertainty associated with the indicated power tilt.

The 2-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% for each percent of tilt in excess of 1.

For purposes of monitoring QUADRANT POWER TILT RATIO when one excore detector is inoperable, the moveable 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 four symmetric thimbles. The two sets of four symmetric thimbles is a unique set of eight detector locations. 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

meet the DNB design criterion

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 with the initial FSAR assumptions and have been analytically demonstrated adequate to ~~maintain a minimum DNBR of 1.30~~ throughout each analyzed transient. The indicated T_{avg} value of ~~591°F~~ and the indicated pressurizer pressure value of 2199 ~~2224~~ psig correspond to analytical limits of ~~592.5°F~~ and ~~2205~~ psig respectively, with allowance for measurement uncertainty.

2185

592.5°F

594.4°F

THIS PAGE APPLICABLE TO UNIT 1 ONLY

3/4.2.5 DNB PARAMETERS (Continued)

The 12-hour periodic surveillance of these parameters through instrument readout is sufficient to ensure that the parameters are restored within their limits following load changes and other expected transient operation. The 18 month periodic measurement of the RCS total flow rate is adequate to detect flow degradation and ensure correlation of the flow indication channels with measured flow such that the indicated percent flow will provide sufficient verification of the flow rate degradation on a 12 hour basis. A change in indicated percent flow which is greater than the instrument channel inaccuracies and parallax errors is an appropriate indication of RCS flow degradation.

BASES

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 minimum DNBR in the core greater than or equal to 1.30 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; and

$F_{xy}(Z)$ 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

The limits on AXIAL FLUX DIFFERENCE (AFD) assure that the $F_Q(Z)$ upper bound envelope of the F_Q limit specified in the CORE OPERATING LIMITS REPORT (COLR) times K(Z) 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 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.

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AXIAL FLUX DIFFERENCE (Continued)

Although it is intended that the plant will be operated with the AFD within the target band required by Specification 3.2.1 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 specified in the COLR 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 1-minute average of each of the OPERABLE excore detector outputs and provides an alarm message immediately if the AFD for two 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 NUCLEAR ENTHALPY RISE HOT CHANNEL FACTOR - $F^N_{\Delta H}$

The limits on heat flux hot channel factor 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 ensure that the limits are maintained provided:

a. Control rods in a single group move together with no individual rod insertion differing by more than ± 12 steps, indicated, from the group demand position;

b. Control rod banks are sequenced with a constant tip-to-tip distance between banks as ~~defined by Figure 3.1-3.~~ described in Specification 3.1.3.6;

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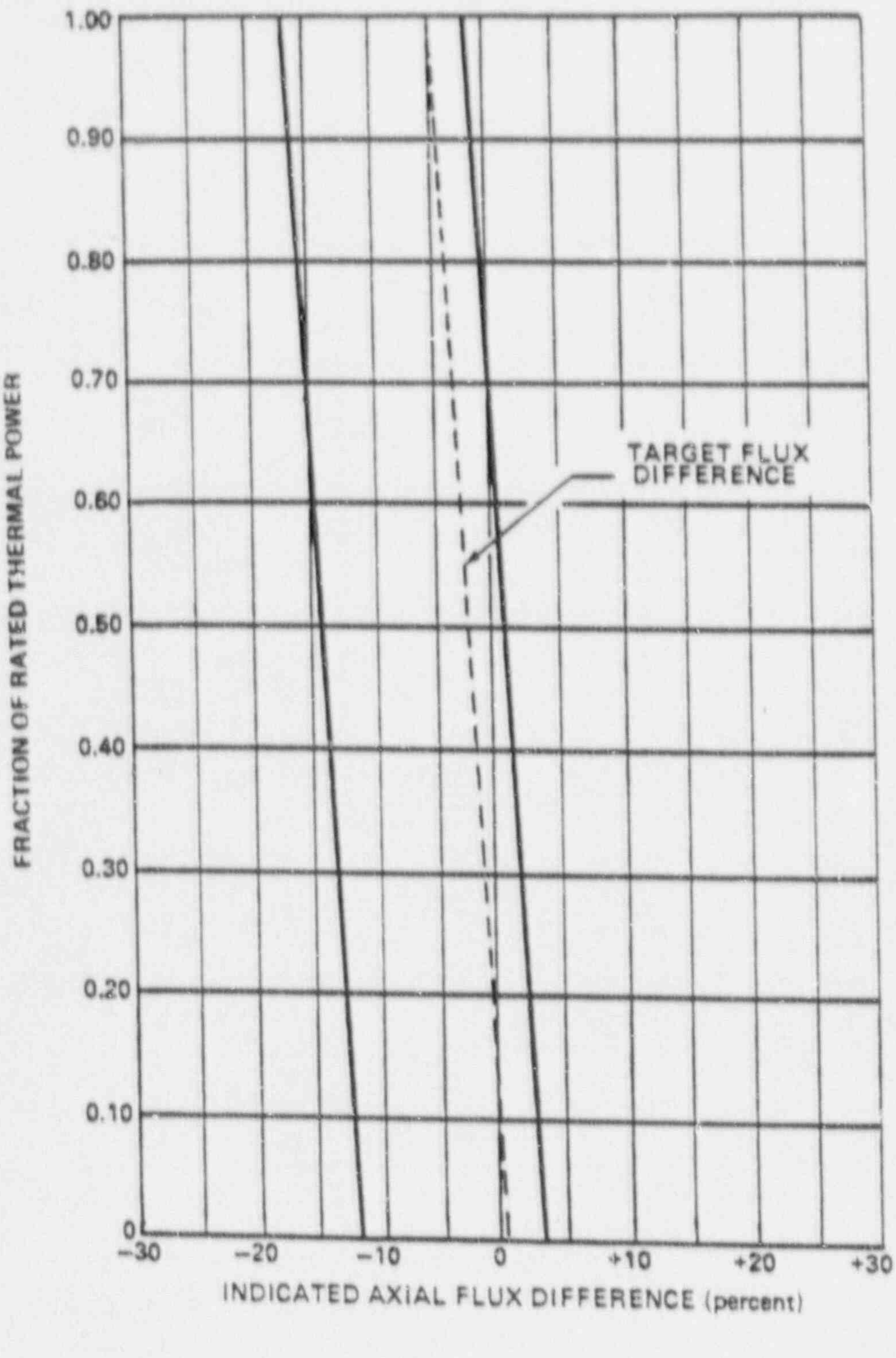

FIGURE B 3/4 2-1

TYPICAL INDICATED AXIAL FLUX DIFFERENCE VERSUS THERMAL POWER

HEAT FLUX HOT CHANNEL FACTOR 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; and

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. 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.

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.

When $F^N_{\Delta H}$ is measured, (i.e., inferred), measurement uncertainty (i.e., the appropriate uncertainty on the incore inferred hot rod peaking factor) must be allowed for and 4% is the appropriate allowance for a full core map taken with the incore detection system.

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

a. Design limit DNBR of 1.30 vs 1.28,

b. Grid Spacing (K_s) of 0.046 vs 0.059,

c. Thermal Diffusion Coefficient of 0.038 vs 0.059,

d. DNBR Multiplier of 0.86 vs 0.88, and

e. Pitch reduction.

The applicable values of rod bow penalties are referenced in the FSAR.

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

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 F_{xy} limit for RATED THERMAL POWER (F_{xy}^{RTP}) as specified in the COLR per Specification 6.8.1.6 was determined from expected power control manuevers over the full range of burnup conditions in the core.

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 cor[illegible] action is required, provides DNB and linear heat generation rate p[illegible]on with x-y plane power tilts. A limit of 1.02 was selected to pr[illegible] an allowance for the uncertainty associated with the indicated po[illegible]tilt.

The 2-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% for each percent of tilt in excess of 1.

For purposes of monitoring QUADRANT POWER TILT RATIO when one excore detector is inoperable, the moveable 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 four symmetric thimbles. The two sets of four symmetric thimbles is a unique set of eight detector locations. 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 with the initial FSAR assumptions and have been analytically demonstrated adequate to maintain a minimum DNBR of 1.30 throughout each analyzed transient. The indicated T_{avg} value of 591°F and the indicated pressurizer pressure value of 2224 psig correspond to analytical limits of 592.5°F and 2205 psig respectively, with allowance for measurement uncertainty.

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3/4.2.5 DNB PARAMETERS (Continued)

The 12-hour periodic surveillance of these parameters through instrument readout is sufficient to ensure that the parameters are restored within their limits following load changes and other expected transient operation. The 18 month periodic measurement of the RCS total flow rate is adequate to detect flow degradation and ensure correlation of the flow indication channels with measured flow such that the indicated percent flow will provide sufficient verification of the flow rate degradation on a 12 hour basis. A change in indicated percent flow which is greater than the instrument channel inaccuracies and parallax errors is an appropriate indication of RCS flow degradation.

3/4.4 REACTOR COOLANT SYSTEM

meet the DNB design criterion

BASES

3/4.4.1 REACTOR COOLANT LOOPS AND COOLANT CIRCULATION

The plant is designed to operate with all reactor coolant loops in operation and ~~maintain DNBR above 1.30~~ during all normal operations and anticipated transients. In MODES 1 and 2 with one reactor coolant loop not in operation this specification requires that the plant be in at least HOT STANDBY within 6 hours.

In MODE 3, two reactor coolant loops provide sufficient heat removal capability for removing core decay heat even in the event of a bank withdrawal accident; however, a single reactor coolant loop provides sufficient heat removal capacity if a bank withdrawal accident can be prevented, i.e., by opening the Reactor Trip System breakers.

In MODE 4, and in MODE 5 with reactor coolant loops filled, a single reactor coolant loop or RHR train provides sufficient heat removal capability for removing decay heat; but single failure considerations require that at least two trains/loops (either RHR or RCS) be OPERABLE.

In MODE 5 with reactor coolant loops not filled, a single RHR train provides sufficient heat removal capability for removing decay heat; but single failure considerations, and the unavailability of the steam generators as a heat removing component, require that at least two RHR trains be OPERABLE. The locking closed of the required valves in Mode 5 (with the loops not filled) precludes the possibility of uncontrolled boron dilution of the filled portion of the Reactor Coolant System. This action prevents flow to the RCS of unborated water by closing flowpaths from sources of unborated water. These limitations are consistent with the initial conditions assumed for the boron dilution accident in the safety analysis.

The operation of one reactor coolant pump (RCP) or one RHR pump provides adequate flow to ensure mixing, prevent stratification and produce gradual reactivity changes during boron concentration reductions in the Reactor Coolant System. The reactivity change rate associated with boron reduction will, therefore, be within the capability of operator recognition and control.

The restrictions on starting an RCP with one or more RCS cold legs less than or equal to 350°F are provided to prevent RCS pressure transients, caused by energy additions from the Secondary Coolant System, which could exceed the limits of Appendix G to 10 CFR Part 50. The RCS will be protected against overpressure transients and will not exceed the limits of Appendix G by restricting starting of the RCPs to when the secondary water temperature of each steam generator is less than 50°F above each of the RCS cold leg temperatures.

NOTE: This change was previously requested by letter ELV-01901

In MODE 4 the starting of an RCP, when no other RCP is operating, is restricted to a range of temperatures that are consistent with the analysis assumptions used to demonstrate that the RHR design pressure is not exceeded when RHR relief valves are used for RCS overpressure protection.

ECCS SUBSYSTEMS (Continued)

The limitation for all safety injection pumps to be inoperable below 350°F provides assurance that a mass addition pressure transient can be relieved by the operation of a single PORV.

The Surveillance Requirements provided to ensure OPERABILITY of each component ensures that at a minimum, the assumptions used in the safety analyses are met and that subsystem OPERABILITY is maintained. Surveillance Requirements for throttle valve position stops and flow balance testing provide assurance that proper ECCS flows will be maintained in the event of a LOCA. Maintenance of proper flow resistance and pressure drop in the piping system to each injection point is necessary to: (1) prevent total pump flow from exceeding runout conditions when the system is in its minimum resistance configuration, (2) provide the proper flow split between injection points in accordance with the assumptions used in the ECCS-LOCA analyses, (3) provide an acceptable level of total ECCS flow to all injection points equal to or above that assumed in the ECCS-LOCA analyses and (4) to ensure that centrifugal charging pump injection flow which is directed through the seal injection path is less than or equal to the amount assumed in the safety analysis. The surveillance requirements for leakage testing of ECCS check valves ensure a failure of one valve will not cause an intersystem LOCA. In MODE 3, with either HV-8809A or B closed for ECCS check valve leak testing, adequate ECCS flow for core cooling in the event of a LOCA is assured.

3/4.5.4 REFUELING WATER STORAGE TANK

The OPERABILITY of the Refueling Water Storage Tank (RWST) as part of the ECCS ensures that sufficient negative reactivity is injected into the core to counteract any positive increase in reactivity caused by RCS cooldown. RCS cooldown can be caused by inadvertent depressurization, a loss-of-coolant accident, or a steam line rupture.

The limits on RWST minimum volume and boron concentration ensure that 1) sufficient water is available within containment to permit recirculation cooling flow to the core, 2) the reactor will remain subcritical in the cold condition following a small LOCA or steamline break, assuming complete mixing of the RWST, RCS, and ECCS water volumes with all control rods inserted except the most reactive control assembly (ARI-1), and 3) the reactor will remain subcritical in the cold condition following a large break LOCA ~~(break flow > 3.0 ft²)~~ assuming complete mixing of the RWST, RCS, ECCS water and other sources of water that may eventually reside in the sump, post-LOCA with all control rods assumed to be out.

The contained water volume limit includes an allowance for water not usable because of tank discharge line location or other physical characteristics.

(UNIT 2) or all control rods inserted except for the two most reactive control assemblies (UNIT 1).

SEMIANNUAL RADIOACTIVE EFFLUENT RELEASE REPORT (Continued)

The Semiannual Radioactive Effluent Release Reports shall also include the following: an explanation as to why the inoperability of liquid or gaseous effluent monitoring instrumentation was not corrected within the time specified in Specification 3.3.3.9 or 3.3.3.10, respectively; and description of the events leading to liquid holdup tanks or gas storage tanks exceeding the limits of Specification 3.11.1.4 or 3.11.2.6, respectively.

MONTHLY OPERATING REPORTS

6.8.1.5 Routine reports of operating statistics and shutdown experience, including documentation of all challenges to the PORVs 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 Administrator of the Regional Office of the NRC, no later than the 15th of each month following the calendar month covered by the report.

CORE OPERATING LIMITS REPORT – UNIT 1

6.8.1.6 Core operating limits shall be established and documented in the CORE OPERATING LIMITS REPORT (COLR) before each reload cycle or any remaining part of a reload cycle for the following:

a. SHUTDOWN MARGIN LIMIT FOR MODES 1 and 2 for Specification 3/4.1.1.1,

b. SHUTDOWN MARGIN LIMITS FOR MODES 3, 4 and 5 for Specification 3/4.1.1.2,

c. Moderator temperature coefficient BOL and EOL limits and 300 ppm surveillance limit for Specification 3/4.1.1.3,

d. Shutdown Rod Insertion Limit for Specification 3/4.1.3.5,

e. Control Rod Insertion Limits for Specification 3/4.1.3.6,

f. Axial Flux Difference Limits~~, and target band~~ for Specification 3/4.2.1,

g. Heat Flux Hot Channel Factor, K(Z)~~, the Power Factor Multiplier~~ and W(Z) ~~F_{xy}^{RTP}~~ for Specification 3/4.2.2,

h. Nuclear Enthalpy Rise Hot Channel Factor Limit and the Power Factor Multiplier for Specification 3/4.2.3.

The analytical methods used to determine the core operating limits shall be those previously approved by the NRC in:

CORE OPERATING LIMITS REPORT (Continued) — UNIT 1

a. WCAP-9272-P-A, "WESTINGHOUSE RELOAD SAFETY EVALUATION METHODOLOGY", July 1985 (W Proprietary). (Methodology for Specifications 3.1.1.3 - Moderator Temperature Coefficient, 3.1.3.5 - Shutdown Bank Insertion Limit, 3.1.3.6 - Control Bank Insertion Limits, ~~3.2.1 - Axial Flux Difference, 3.2.2 - Heat Flux Hot Channel Factor,~~ and 3.2.3 - Nuclear Enthalpy Rise Hot Channel Factor.)

~~b. WCAP-8385, "POWER DISTRIBUTION CONTROL AND LOAD FOLLOWING PROCEDURES - TOPICAL REPORT", September 1974 (W Proprietary). (Methodology for Specification 3.2.1 - Axial Flux Difference [Constant Axial Offset Control].)~~

~~c. T. M. Anderson to K. Kniel (Chief of Core Performance Branch, NRC) January 31, 1980--Attachment: Operation and Safety Analysis Aspects of an Improved Load Follow Package. (Methodology for Specification 3.2.1 - Axial Flux Difference [Constant Axial Offset Control].)~~

~~d. NUREG-0800, Standard Review Plan, U. S. Nuclear Regulatory Commission, Section 4.3, Nuclear Design, July 1981. Branch Technical Position CPB 4.3-1, Westinghouse Constant Axial Offset Control (CAOC), Rev. 2, July 1981. (Methodology for Specification 3.2.1 - Axial Flux Difference [Constant Axial Offset Control].)~~

c. ~~e.~~ WCAP-9220-P-A, Rev. 1, "WESTINGHOUSE ECCS EVALUATION MODEL-1981 VERSION", February 1982 (W Proprietary). (Methodology for Specification 3.2.2 - Heat Flux Hot Channel Factor.)

The core operating limits shall be determined so that all applicable limits (e.g., fuel thermal-mechanical limits, core thermal-hydraulic limits, ECCS limits, nuclear limits such as shutdown margin, and transient and accident analysis limits) of the safety analysis are met.

The CORE OPERATING LIMITS REPORT, including any mid-cycle revisions or supplements thereto, shall be provided upon issuance, for each reload cycle, to the NRC Document Control Desk with copies to the Regional Administrator and Resident Inspector.

~~SPECIAL REPORTS~~

~~6.8.2 Special reports shall be submitted to the Regional Administrator of the Regional Office of the NRC within the time period specified for each report.~~

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b. WCAP-10216-P-A, "RELAXATION OF CONSTANT AXIAL OFFSET CONTROL FQ SURVEILLANCE TECHNICAL SPECIFICATION", June 1983 (W Proprietary).
(Methodology for Specifications 3.2.1 - Axial Flux Difference (Relaxed Axial Offset Control) and 3.2.2 - Heat Flux Hot Channel Factor (W(Z) surveillance requirements for F_Q Methodology).)

SEMIANNUAL RADIOACTIVE EFFLUENT RELEASE REPORT (Continued)

The Semiannual Radioactive Effluent Release Reports shall also include the following: an explanation as to why the inoperability of liquid or gaseous effluent monitoring instrumentation was not corrected within the time specified in Specification 3.3.3.9 or 3.3.3.10, respectively; and description of the events leading to liquid holdup tanks or gas storage tanks exceeding the limits of Specification 3.11.1.4 or 3.11.2.6, respectively.

MONTHLY OPERATING REPORTS

6.8.1.5 Routine reports of operating statistics and shutdown experience, including documentation of all challenges to the PORVs 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 Administrator of the Regional Office of the NRC, no later than the 15th of each month following the calendar month covered by the report.

CORE OPERATING LIMITS REPORT - UNIT 2

6.8.1.6 Core operating limits shall be established and documented in the CORE OPERATING LIMITS REPORT (COLR) before each reload cycle or any remaining part of a reload cycle for the following:

a. SHUTDOWN MARGIN LIMIT FOR MODES 1 and 2 for Specification 3/4.1.1.1,

b. SHUTDOWN MARGIN LIMITS FOR MODES 3, 4 and 5 for Specification 3/4.1.1.2,

c. Moderator temperature coefficient BOL and EOL limits and 300 ppm surveillance limit for Specification 3/4.1.1.3,

d. Shutdown Rod Insertion Limit for Specification 3/4.1.3.5,

e. Control Rod Insertion Limits for Specification 3/4.1.3.6,

f. Axial Flux Difference Limits, and target band for Specification 3/4.2.1,

g. Heat Flux Hot Channel Factor, K(Z), the Power Factor Multiplier and F_{xy}^{RTP}, for Specification 3/4.2.2,

h. Nuclear Enthalpy Rise Hot Channel Factor Limit and the Power Factor Multiplier for Specification 3/4.2.3.

The analytical methods used to determine the core operating limits shall be those previously approved by the NRC in:

CORE OPERATING LIMITS REPORT (Continued) - UNIT 2

a. WCAP-9272-P-A, "WESTINGHOUSE RELOAD SAFETY EVALUATION METHODOLOGY", July 1985 (W Proprietary).
(Methodology for Specifications 3.1.1.3 - Moderator Temperature Coefficient, 3.1.3.5 - Shutdown Bank Insertion Limit, 3.1.3.6 - Control Bank Insertion Limits, 3.2.1 - Axial Flux Difference, 3.2.2 - Heat Flux Hot Channel Factor, and 3.2.3 - Nuclear Enthalpy Rise Hot Channel Factor.)

b. WCAP-8385, "POWER DISTRIBUTION CONTROL AND LOAD FOLLOWING PROCEDURES - TOPICAL REPORT", September 1974 (W Proprietary).
(Methodology for Specification 3.2.1 - Axial Flux Difference [Constant Axial Offset Control].)

c. T. M. Anderson to K. Kniel (Chief of Core Performance Branch, NRC) January 31, 1980--Attachment: Operation and Safety Analysis Aspects of an Improved Load Follow Package.
(Methodology for Specification 3.2.1 - Axial Flux Difference [Constant Axial Offset Control].)

d. NUREG-0800, Standard Review Plan, U. S. Nuclear Regulatory Commission, Section 4.3, Nuclear Design, July 1981. Branch Technical Position CPB 4.3-1, Westinghouse Constant Axial Offset Control (CAOC), Rev. 2, July 1981.
(Methodology for Specification 3.2.1 - Axial Flux Difference [Constant Axial Offset Control].)

e. WCAP-9220-P-A, Rev. 1, "WESTINGHOUSE ECCS EVALUATION MODEL-1981 VERSION", February 1982 (W Proprietary).
(Methodology for Specification 3.2.2 - Heat Flux Hot Channel Factor.)

The core operating limits shall be determined so that all applicable limits (e.g., fuel thermal-mechanical limits, core thermal-hydraulic limits, ECCS limits, nuclear limits such as shutdown margin, and transient and accident analysis limits) of the safety analysis are met.

The CORE OPERATING LIMITS REPORT, including any mid-cycle revisions or supplements thereto, shall be provided upon issuance, for each reload cycle, to the NRC Document Control Desk with copies to the Regional Administrator and Resident Inspector.

SPECIAL REPORTS

6.8.2 Special reports shall be submitted to the Regional Administrator of the Regional Office of the NRC within the time period specified for each report.

Attachment 1b

Vogtle Electric Generating Plant Units 1 and 2

Request for Technical Specifications Changes

VANTAGE-5 Fuel Design

Technical Specifications Typed Pages

Effective following the Vogtle 1 Cycle 3 Shutdown
(Effective as of Vogtle 1 Cycle 4 Startup)

SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

BASES

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

BASES

ADMINISTRATIVE CONTROLS

2.0 SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS*

2.1 SAFETY LIMITS

REACTOR CORE

2.1.1 The combination of THERMAL POWER (NI-0041, NI-0042, NI-0043, NI-0044), pressurizer pressure (PI-0455A, B&C, PI-0456 & PI-0456A, PI-0457 & PI-0457A, PI-0458 & PI-0458A), and the highest operating loop coolant temperature (T_{avg}) (TI-0412, TI-0422, TI-0432, TI-0442) shall not exceed the limits shown in Figure 2.1-1 (Unit 1) or Figure 2.1-1a (Unit 2).

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 6.6.1.

REACTOR COOLANT SYSTEM PRESSURE

2.1.2 The Reactor Coolant System pressure (PI-0408, PI-0418, PI-0428, PI-0438) 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 6.6.1.

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 6.6.1.

*Where specific instrument numbers are provided in parentheses they are for information only, and apply to each unit unless specifically noted (to assist in identifying associated instrument channels or loops) and are not intended to limit the requirements to the specific instruments associated with the number.

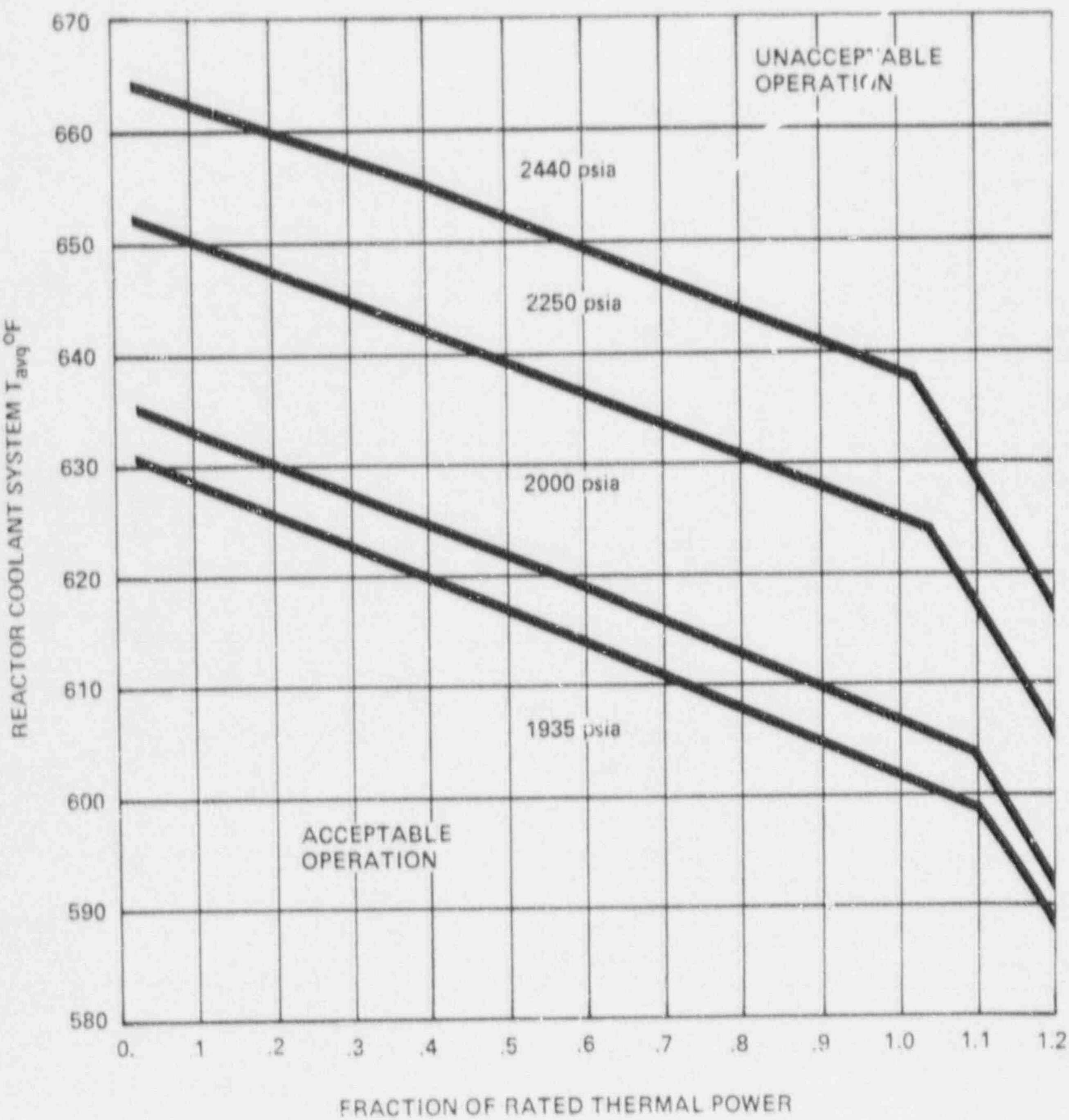

FIGURE 2.1-1
REACTOR CORE SAFETY LIMIT - UNIT 1

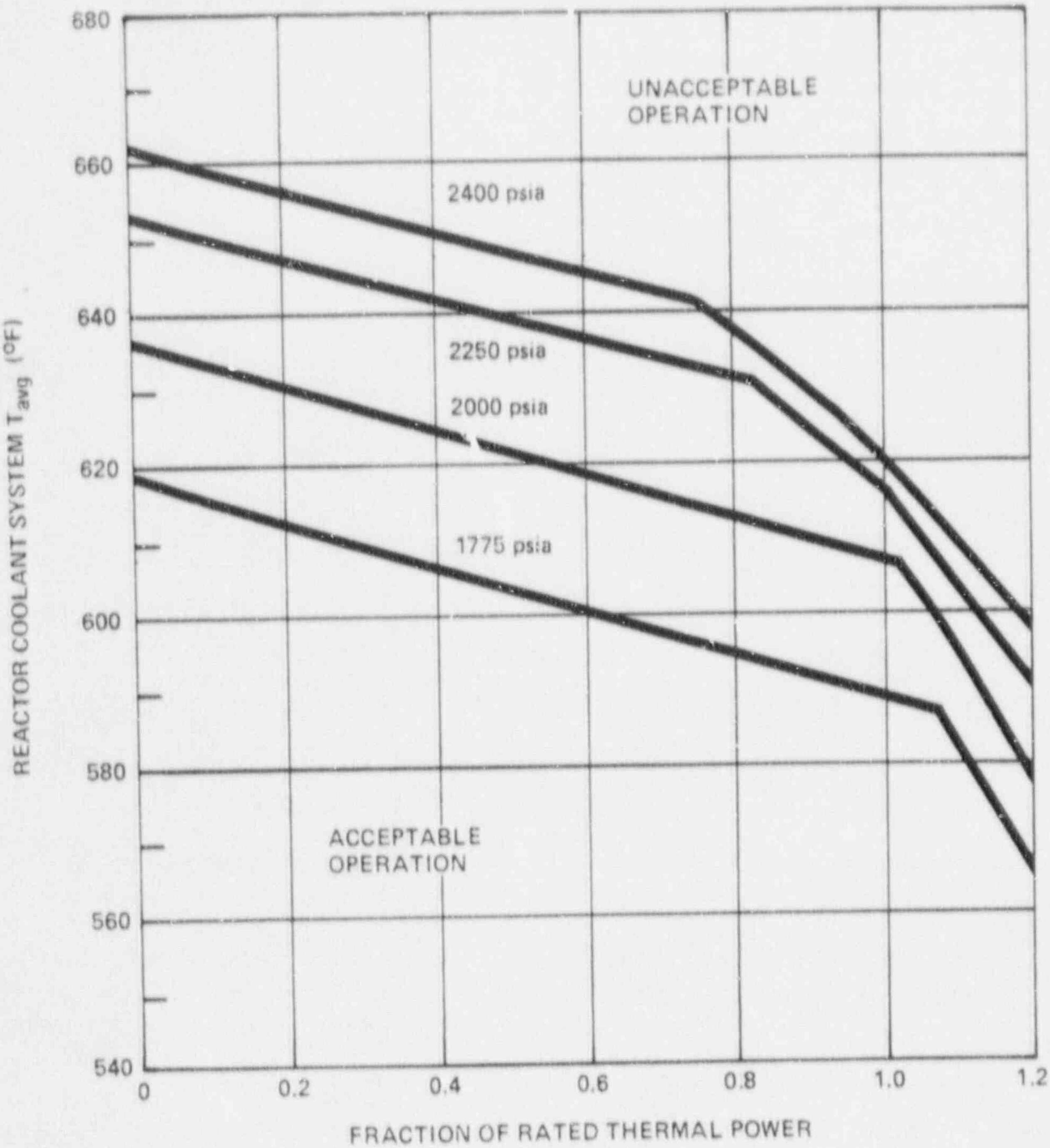

FIGURE 2.1-1a

REACTOR CORE SAFETY LIMIT - UNIT 2

2.2 LIMITING SAFETY SYSTEM SETTINGS

REACTOR TRIP SYSTEM INSTRUMENTATION SETPOINTS

2.2.1 The Reactor Trip System Instrumentation and Interlock Setpoints shall be set consistent with the Trip Setpoint values shown in Table 2.2-1 (Unit 1) or Table 2.2-1a (Unit 2).

APPLICABILITY: As shown for each channel in Table 3.3-1.

ACTION:

a. With a Reactor Trip System Instrumentation or Interlock Setpoint less conservative than the value shown in the Trip Setpoint column but more conservative than the value shown in the Allowable Value column of Table 2.2-1 or Table 2.2-1a, adjust the Setpoint consistent with the Trip Setpoint value.

b. With the Reactor Trip System Instrumentation or Interlock Setpoint less conservative than the value shown in the Allowable Values column of Table 2.2-1 or Table 2.2-1a, either:

   1. Adjust the Setpoint consistent with the Trip Setpoint value of Table 2.2-1 or Table 2.2-1a and determine within 12 hours that Equation 2.2-1 was satisfied for the affected channel, or

   2. Declare the channel inoperable and apply the applicable ACTION statement requirement of Specification 3.3.1 until the channel is restored to OPERABLE status with its Setpoint adjusted consistent with the Trip Setpoint value.

Equation 2.2-1 $$Z + R + S \leq TA$$

Where:

Z = The value from Column Z of Table 2.2-1 or Table 2.2-1a for the affected channel,

R = The "as measured" value (in percent span) of rack error for the affected channel,

S = Either the "as measured" value (in percent span) of the sensor error, or the value from Column S (Sensor Error) of Table 2.2-1 or Table 2.2-1a for the affected channel, and

TA = The value from Column TA (Total Allowance) of Table 2.2-1 or Table 2.2-1a for the affected channel.

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TABLE 2.2-1 - UNIT 1

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TOTAL ALLOWANCE (TA)	Z	SENSOR ERROR (S)	TRIP SETPOINT	ALLOWABLE VALUE
1. Manual Reactor Trip	N.A.	N.A.	N.A.	N.A.	N.A.
2. Power Range, Neutron Flux (NI-0041B&C, NI-0042B&C, NI-0043B&C, NI-0044B&C)					
a. High Setpoint	7.5	4.56	0	<109% of RTP#	≤111.3% of RTP#
b. Low Setpoint	8.3	4.56	0	25% of RTP#	≤27.3% of RTP#
3. Power Range, Neutron Flux, High Positive Rate (NI-0041B&C, NI-0042B&C, NI-0043B&C, NI-0044B&C)	1.6	0.50	0	≤5% of RTP# with a time constant ≥2 seconds	≤6.3% of RTP# with a time constant ≥2 seconds
4. Deleted.					
5. Intermediate Range, Neutron Flux (NI-0035B, NI-0036B)	17.0	8.41	0	≤25% of RTP#	≤31.1% of RTP#
6. Source Range, Neutron Flux (NI-0031B, NI-0032B)	17.0	10.01	0	$\leq 10^5$ cps	$\leq 1.4 \times 10^5$ cps
7. Overtemperature ΔT (TDI-411C, TDI-421C, TDI-431C, TDI-441C)	10.7 (Unit 1)	7.04 (Unit 1)	1.96 + 1.17 (Unit 1)	See Note 1	See Note 2
8. Overpower ΔT (TDI-411B, TDI-421B, TDI-431B, TDI-441B)	4.3 (Unit 1)	1.54	1.96 (Unit 1)	See Note 3	See Note [illegible]

#RTP = RATED THERMAL POWER

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TABLE 2.2-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS - UNIT 1

FUNCTIONAL UNIT	TOTAL ALLOWANCE (TA)	Z	SENSOR ERROR (S)	TRIP SETPOINT	ALLOWABLE VALUE
9. Pressurizer Pressure-Low (PI-0455A,B&C, PI-0456 & PI-0456A, PI-0457 & PI-0457A, PI-0458 & PI-0458A)	3.1	0.71	1.67	≥1960 psig**	≥1950 psig
10. Pressurizer Pressure-High (PI-0455A,B&C, PI-0456 & PI-0456A, PI-0457 & PI-0457A, PI-0458 & PI-0458A)	3.1	0.71	1.67	≤2385 psig	≤2395 psig
11. Pressurizer Water Level-High (LI-0459A, LI-0460A, LI-0461)	8.0	2.18	1.67	≤92% of instrument span	≤93.9% of instrument span
12. Reactor Coolant Flow-Low (LOOP1 FI-0414 FI-0415 FI-0416; LOOP2 FI-0424 FI-0425 FI-0426; LOOP3 FI-0434 FI-0435 FI-0436; LOOP4 FI-0444 FI-0445 FI-0446)	2.5	1.87	0.60	≥90% of loop design flow*	≥89.4% of loop design flow*
13. Steam Generator Water Level Low-Low (LOOP1 LI-0517 LI-0518 LI-0519 LI-0551; LOOP2 LI-0527 LI-0528 LI-0529 LI-0552; LOOP3 LI-0537 LI-0538 LI-0539 LI-0553; LOOP4 LI-0547 LI-0548 LI-0549 LI-0554)	18.5 (21.8)***	17.18 (18.21)***	1.67	≥18.5% (37.8)*** of narrow range instrument span	≥17.8% (35.9)*** of narrow range instrument span
14. Undervoltage - Reactor Coolant Pumps	6.0	0.58	0	≥9600 volts (70% bus voltage)	≥9481 volts (69% bus voltage)
15. Underfrequency - Reactor Coolant Pumps	3.3	0.50	0	≥57.3 Hz	≥57.1 Hz

*Loop design flow = 95,700 gpm

**Time constants utilized in the lead-lag controller for Pressurizer Pressure-Low are 10 seconds for lead and 1 second for lag. CHANNEL CALIBRATION shall ensure that these time constants are adjusted to these values.

***The value stated inside the parenthesis is for instrumentation that has the lower tap at elevation 333"; the value stated outside the parenthesis is for instrumentation that has the lower tap at elevation 438".

THIS PAGE APPLICABLE TO UNIT 1 ONLY

TABLE 2.2-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS - UNIT 1

FUNCTIONAL UNIT	TOTAL ALLOWANCE (TA)	Z	SENSOR ERROR (S)	TRIP SETPOINT	ALLOWABLE VALUE
16. Turbine Trip					
a. Low Fluid Oil Pressure (PI-6161, PT-6162, PT-6163)	N.A.	N.A.	N.A.	≥580 psig	≥500 psig
b. Turbine Stop Valve Closure	N.A.	N.A.	N.A.	≥96.7% open	≥96.7% open
17. Safety Injection Input from ESF	N.A.	N.A.	N.A.	N.A.	N.A.
18. Reactor Trip System Interlocks					
a. Intermediate Range Neutron Flux, P-6 (NI-0035B, NI-0036B)	N.A.	N.A.	N.A.	$\geq 1 \times 10^{-10}$ amp	$\geq 6 \times 10^{-11}$ amp
b. Low Power Reactor Trips Block, P-7					
1) P-10 input (NI-0041B&C, NI-0042B&C, NI-0043B&C, NI-0044B&C)	N.A.	N.A.	N.A.	≤10% of RTP#	≤12.3% of RTP#
2) P-13 input (PI-0505, PI-0506)	N.A.	N.A.	N.A.	<10% RTP# Turbine Impulse Pressure Equivalent	<12.3% RTP# Turbine Impulse Pressure Equivalent
c. Power Range Neutron Flux, P-8 (NI-0041B&C, NI-0042B&C, NI-0043B&C, NI-0044B&C)	N.A.	N.A.	N.A.	≤48% of RTP#	≤50.3% of RTP#

#RTP = RATED THERMAL POWER

THIS PAGE APPLICABLE TO UNIT 1 ONLY

TABLE 2.2-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS - UNIT 1

FUNCTIONAL UNIT	TOTAL ALLOWANCE (TA)	Z	SENSOR ERROR (S)	TRIP SETPOINT	ALLOWABLE VALUE
d. Power Range Nuetron Flux, P-9 (NI-0041B&C, NI-0042B&C, NI-0043B&C, NI-0044B&C)	N.A.	N.A.	N.A.	≤50% of RTP#	≤52.3% of RTP#
e. Power Range Neutron Flux, P-10 (NI-0041B&C, NI-0042B&C, NI-0043B&C, NI-0044B&C)	N.A.	N.A.	N.A.	≥10% of RTP#	≥7.7% of RTP#
f. Turbine Impulse Chamber Pressure, P-13 (PI-0505, PI-0506)	N.A.	N.A.	N.A.	≤10% RTP# Turbine Impulse Pressure Equivalent	≤12.3% RTP# Turbine Impulse Pressure Equivalent
19. Reactor Trip Breakers	N.A.	N.A.	N.A.	N.A.	N.A.
20. Automatic Trip and Interlock Logic	N.A.	N.A.	N.A.	N.A.	N.A.

#RTP = RATED THERMAL POWER

THIS PAGE APPLICABLE TO UNIT 1 ONLY

TABLE 2.2-1 (Continued)

TABLE NOTATIONS - UNIT 1

NOTE 1: OVERTEMPERATURE ΔT

$$\Delta T \frac{(1 + \tau_1 S)}{(1 + \tau_2 S)} \left(\frac{1}{1 + \tau_3 S}\right) \leq \Delta T_0 \{K_1 - K_2 \frac{(1 + \tau_4 S)}{(1 + \tau_5 S)} [T \left(\frac{1}{1 + \tau_6 S}\right) - T'] + K_3(P - P') - f_1(\Delta I)\}$$

Where: ΔT = Measured ΔT (Unit 1);

$\frac{1 + \tau_1 S}{1 + \tau_2 S}$ = Lead-lag compensator on measured ΔT;

τ_1, τ_2 = Time constants utilized in lead-lag compensator for ΔT, $\tau_1 \geq 8$ s, $\tau_2 \leq 3$ s;

$\frac{1}{1 + \tau_3 S}$ = Lag compensator on measured ΔT;

τ_3 = Time constants utilized in the lag compensator for ΔT, $\tau_3 = 0$ s;

ΔT_0 = Indicated ΔT at RATED THERMAL POWER;

K_1 ≤ 1.12 (Unit 1);

K_2 = 0.0224/°F (Unit 1);

$\frac{1 + \tau_4 S}{1 + \tau_5 S}$ = The function generated by the lead-lag compensator for T_{avg} dynamic compensation;

τ_4, τ_5 = Time constants utilized in the lead-lag compensator for T_{avg}, $\tau_4 \geq 28$ s, $\tau_5 \leq 4$ s;

T = Average temperature, °F;

$\frac{1}{1 + \tau_6 S}$ = Lag compensator on measured T_{avg};

τ_6 = Time constant utilized in the measured T_{avg} lag compensator, $\tau_6 = 0$ s;

TABLE 2.2-1 (Continued)

TABLE NOTATIONS (Continued) - UNIT 1

NOTE 1: (Continued)

T' $\leq$ 588.4°F (Unit 1) (Nominal T_{avg} operating temperature;)

K_3 = 0.00115/psig (Unit 1);

P = Pressurizer pressure, psig;

P' = 2235 psig (Nominal RCS operating pressure);

S = Laplace transform variable, s^{-1};

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

(1) For $q_t - q_b$ between -32.0% (Unit 1) and + 11.0% (Unit 1), $f_1(\Delta I) = 0$, 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;

(2) For each percent that the magnitude of $q_t - q_b$ exceeds - 32.0% (Unit 1), the ΔT Trip Setpoint shall be automatically reduced by 3.25% (Unit 1) of its value at RATED THERMAL POWER; and

(3) For each percent that the magnitude of $q_t - q_b$ exceeds + 11.0% (Unit 1), the ΔT Trip Setpoint shall be automatically reduced by 1.97% (Unit 1) of its value at RATED THERMAL POWER.

NOTE 2: The channel's maximum Trip Setpoint shall not exceed its computed Trip Setpoint by more than 3.1% (Unit 1) of ΔT span.

TABLE 2.2-1 (Continued)

TABLE NOTATIONS (Continued) - UNIT 1

NOTE 3: OVERPOWER ΔT

$$\Delta T \frac{(1 + \tau_1 S)}{(1 + \tau_2 S)} \left(\frac{1}{(1 + \tau_3 S)} \right) \leq \Delta T_0 \left\{ K_4 - K_5 \left(\frac{\tau_7 S}{(1 + \tau_7 S)} \right) \left(\frac{1}{(1 + \tau_6 S)} \right) T - K_6 \left[T \left(\frac{1}{(1 + \tau_6 S)} \right) - T'' \right] - f_2(\Delta I) \right\}$$

Where: ΔT = Measured ΔT (Unit 1);

$\frac{1 + \tau_1 S}{1 + \tau_2 S}$ = Lead-lag compensator on measured ΔT;

τ_1, τ_2 = Time constants utilized in lead-leg compensator for ΔT, $\tau_1 \geq 8$ S, $\tau_2 \leq 3$ s;

$\frac{1}{1 + \tau_3 S}$ = Lag compensator on measured ΔT;

τ_3 = Time constants utilized in the lag compensator for ΔT, τ_3=0s;

ΔT_0 = Indicated ΔT at RATED THERMAL POWER;

K_4 ≤ 1.08 (Unit 1),

K_5 ≥ 0.02/°F for increasing average temperature and ≥ 0 for decreasing average temperature,

$\frac{\tau_7 S}{1 + \tau_7 S}$ = The function generated by the rate-lag compensator for T_{avg} dynamic compensation,

τ_7 = Time constants utilized in the rate-lag compensator for T_{avg}, $\tau_7 \geq 10$ s,

$\frac{1}{1 + \tau_6 S}$ = Lag compensator on measured T_{avg};

THIS PAGE APPLICABLE TO UNIT 1 ONLY

TABLE 2.2-1 (Continued)

TABLE NOTATIONS (Continued) - UNIT 1

NOTE 3: (Continued)

τ_6	=	Time constant utilized in the measured T_{avg} lag compensator, $\tau_6 = 0$ s;
K_6	$\geq$	0.0020/°F (Unit 1) for $T > T"$ and $K_6 = 0$ for $T \leq T"$,
T	=	Average Temperature, °F;
T"	=	Indicated T_{avg} at RATED THERMAL POWER (Calibration temperature for ΔI instrumentation, $\leq$ 588.4°F (Unit 1)),
S	=	Laplace transform variable, s^{-1}; and
$f_2(\Delta I)$	=	0 for all ΔI.

NOTE 4: The channel's maximum Trip Setpoint shall not exceed its computed Trip Setpoint by more than 1.9% (Unit 1) of ΔT span.

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TABLE 2.2-1a - UNIT 2

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TOTAL ALLOWANCE (TA)	Z	SENSOR ERROR (S)	TRIP SETPOINT	ALLOWABLE VALUE
1. Manual Reactor Trip	N.A.	N.A.	N.A.	N.A.	N.A.
2. Power Range, Neutron Flux (NI-0041B&C, NI-0042B&C, NI-0043B&C, NI-0044B&C)					
a. High Setpoint	7.5	4.56	0	$\leq$109% of RTP#	$\leq$111.3% of RTP#
b. Low Setpoint	8.3	4.56	0	$\leq$25% of RTP#	$\leq$27.3% of RTP#
3. Power Range, Neutron Flux, High Positive Rate (NI-0041B&C, NI-0042B&C, NI-0043B&C, NI-0044B&C)	1.6	0.50	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. Deleted.					
5. Intermediate Range, Neutron Flux (NI-0035B, NI-0036B)	17.0	8.41	0	$\leq$25% of RTP#	$\leq$31.1% of RTP#
6. Source Range, Neutron Flux (NI-0031B, NI-0032B)	17.0	10.01	0	$\leq 10^5$ cps	$\leq 1.4 \times 10^5$ cps
7. Overtemperature ΔT (TDI-411C, TDI-421C, TDI-431C, TDI-441C)	6.6 (Unit 2)	3.37 (Unit 2)	1.95 + 0.50 (Unit 2)	See Note 1	See Note 2
8. Overpower ΔT (TDI-411B, TDI-421B, TDI-431B, TDI-441B)	4.9 (Unit 2)	1.54	1.95 (Unit 2)	See Note 3	See Note 4

#RTP = RATED THERMAL POWER

THIS PAGE APPLICABLE TO UNIT 2 ONLY

TABLE 2.2-1a (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS - UNIT 2

FUNCTIONAL UNIT	TOTAL ALLOWANCE (TA)	Z	SENSOR ERROR (S)	TRIP SETPOINT	ALLOWABLE VALUE
9. Pressurizer Pressure-Low (PI-0455A,B&C, PI-0456 & PI-0456A, PI-0457 & PI-0457A, PI-0458 & PI-0458A)	3.1	0.71	1.67	≥1960 psig**	≥1950 psig
10. Pressurizer Pressure-High (PI-0455A,B&C, PI-0456 & PI-0456A, PI-0457 & PI-0457A, PI-0458 & PI-0458A)	3.1	0.71	1.67	≤2385 psig	≤2395 psig
11. Pressurizer Water Level-High (LI-0459A, LI-0460A, LI-0461)	8.0	2.18	1.67	≤92% of instrument span	≤93.9% of instrument span
12. Reactor Coolant Flow-Low (LOOP1 LOOP2 LOOP3 LOOP4 / FI-0414 FI-0424 FI-0434 FI-0444 / FI-0415 FI-0425 FI-0435 FI-0445 / FI-0416 FI-0426 FI-0436 FI-0446)	2.5	1.87	0.60	≥90% of loop design flow*	≥89.4% of loop design flow*
13. Steam Generator Water Level Low-Low (LOOP1 LOOP2 LOOP3 LOOP4 / LI-0517 LI-0527 LI-0537 LI-0547 / LI-0518 LI-0528 LI-0538 LI-0548 / LI-0519 LI-0529 LI-0539 LI-0549 / LI-0551 LI-0552 LI-0553 LI-0554)	18.5 (21.8)***	17.18 (18.21)***	1.67	≥18.5% (37.8)*** of narrow range instrument span	≥17.8% (35.9)*** of narrow range instrument span
14. Undervoltage - Reactor Coolant Pumps	6.0	0.58	0	≥9600 volts (70% bus voltage)	≥9481 volts (69% bus voltage)
15. Underfrequency - Reactor Coolant Pumps	3.3	0.50	0	≥57.3 Hz	≥57.1 Hz

*Loop design flow = 95,700 gpm

**Time constants utilized in the lead-lag controller for Pressurizer Pressure-Low are 10 seconds for lead and 1 second for lag. CHANNEL CALIBRATION shall ensure that these time constants are adjusted to these values.

***The value stated inside the parenthesis is for instrumentation that has the lower tap at elevation 333"; the value stated outside the parenthesis is for instrumentation that has the lower tap at elevation 438".

THIS PAGE APPLICABLE TO UNIT 2 ONLY

TABLE 2.2-1a (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS - UNIT 2

FUNCTIONAL UNIT	TOTAL ALLOWANCE (TA)	Z	SENSOR ERROR (S)	TRIP SETPOINT	ALLOWABLE VALUE
16. Turbine Trip					
a. Low Fluid Oil Pressure (PT-6161, PT-6162, PT-6163)	N.A.	N.A.	N.A.	≥580 psig	≥500 psig
b. Turbine Stop Valve Closure	N.A.	N.A.	N.A.	≥96.7% open	≥96.7% open
17. Safety Injection Input from ESF	N.A.	N.A.	N.A.	N.A.	N.A.
18. Reactor Trip System Interlocks					
a. Intermediate Range Neutron Flux, P-6 (NI-0035B, NI-0036B)	N.A.	N.A.	N.A.	≥1 x 10^{-10} amp	≥6 x 10^{-11} amp
b. Low Power Reactor Trips Block, P-7					
1) P-10 input (NI-0041B&C, NI-0042B&C, NI-0043B&C, NI-0044B&C)	N.A.	N.A.	N.A.	≤10% of RTP#	≤12.3% of RTP#
2) P-13 input (PT-0505, PT-0506)	N.A.	N.A.	N.A.	≤10% RTP# Turbine Impulse Pressure Equivalent	≤12.3% RTP# Turbine Impulse Pressure Equivalent
c. Power Range Neutron Flux, P-8 (NI-0041B&C, NI-0042B&C, NI-0043B&C, NI-0044B&C)	N.A.	N.A.	N.A.	≤48% of RTP#	≤50.3% of RTP#

#RTP = RATED THERMAL POWER

TABLE 2.2-1a (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS - UNIT 2

FUNCTIONAL UNIT	TOTAL ALLOWANCE (TA)	Z	SENSOR ERROR (S)	TRIP SETPOINT	ALLOWABLE VALUE
d. Power Range Nuetron Flux, P-9 (NI-0041B&C, NI-0042B&C, NI-0043B&C, NI-0044B&C)	N.A.	N.A.	N.A.	≤50% of RTP#	≤52.3% of RTP#
e. Power Range Neutron Flux, P-10 (NI-0041B&C, NI-0042B&C, NI-0043B&C, NI-0044B&C)	N.A.	N.A.	N.A.	≥10% of RTP#	≥7.7% of RTP#
f. Turbine Impulse Chamber Pressure, P-13 (PI-0505, PI-0506)	N.A.	N.A.	N.A.	≤10% RTP# Turbine Impulse Pressure Equivalent	≤12.3% RTP# Turbine Impulse Pressure Equivalent
19. Reactor Trip Breakers	N.A.	N.A.	N.A.	N.A.	N.A.
20. Automatic Trip and Interlock Logic	N.A.	N.A.	N.A.	N.A.	N.A.

#RTP = RATED THERMAL POWER

TABLE 2.2-1a (Continued)

TABLE NOTATIONS - UNIT 2

NOTE 1: OVERTEMPERATURE ΔT

$$\Delta T \frac{(1 + \tau_1 S)}{(1 + \tau_2 S)} \frac{1}{1 + \tau_3 S} \leq \Delta T_0 \{K_1 - K_2 \frac{(1 + \tau_4 S)}{(1 + \tau_5 S)} [T \frac{1}{1 + \tau_6 S} - T'] + K_3(P - P') - f_1(\Delta I)\}$$

Where: ΔT = Measured ΔT by RTD Manifold Instrumentation (Unit 2);

$\frac{1 + \tau_1 S}{1 + \tau_2 S}$ = Lead-lag compensator on measured ΔT;

τ_1, τ_2 = Time constants utilized in lead-lag compensator for ΔT, $\tau_1 \geq 8$ s, $\tau_2 \leq 3$ s;

$\frac{1}{1 + \tau_3 S}$ = Lag compensator on measured ΔT;

τ_3 = Time constants utilized in the lag compensator for ΔT, $\tau_3 = 0$ s;

ΔT_0 = Indicated ΔT at RATED THERMAL POWER;

K_1 $\leq$ 1.10 (Unit 2);

K_2 = 0.012/°F (Unit 2);

$\frac{1 + \tau_4 S}{1 + \tau_5 S}$ = The function generated by the lead-lag compensator for T_{avg} dynamic compensation;

τ_4, τ_5 = Time constants utilized in the lead-lag compensator for T_{avg}, $\tau_4 \geq 28$ s, $\tau_5 \leq 4$ s;

T = Average temperature, °F;

$\frac{1}{1 + \tau_6 S}$ = Lag compensator on measured T_{avg};

τ_6 = Time constant utilized in the measured T_{avg} lag compensator, $\tau_6 = 0$ s;

TABLE 2.2-1a (Continued)

TABLE NOTATIONS (Continued) - UNIT 2

NOTE 1: (Continued)

T' $\leq$ 588.5 F (Unit 2) (Nominal T_{avg} operating temperature);

K_3 = 0.00056/psig (Unit 2);

P = Pressurizer pressure, psig;

P' = 2235 psig (Nominal RCS operating pressure);

S = Laplace transform variable, s^{-1};

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

(1) For $q_t - q_b$ between -33.5% (Unit 2) and + 6.5% (Unit 2), $f_1(\Delta I) = 0$, 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;

(2) For each percent that the magnitude of $q_t - q_b$ exceeds - 33.5% (Unit 2), the ΔI Trip Setpoint shall be automatically reduced by 1.27% (Unit 2) of its value at RATED THERMAL POWER; and

(3) For each percent that the magnitude of $q_t - q_b$ exceeds + 6.5% (Unit 2), the ΔI Trip Setpoint shall be automatically reduced by 0.83% (Unit 2) of its value at RATED THERMAL POWER.

NOTE 2: The channel's maximum Trip Setpoint shall not exceed its computed Trip Setpoint by more than 2.5% (Unit 2).

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TABLE 2.2-1a (Continued)

TABLE NOTATIONS (Continued) - UNIT 2

NOTE 3: OVERPOWER ΔT

$$\Delta T \frac{(1 + \tau_1 S)}{(1 + \tau_2 S)} \left(\frac{1}{1 + \tau_3 S}\right) \leq \Delta T_0 \left\{K_4 - K_5 \left(\frac{\tau_7 S}{1 + \tau_7 S}\right)\left(\frac{1}{1 + \tau_6 S}\right) T - K_6 \left[T \left(\frac{1}{1 + \tau_6 S}\right) - T''\right] - f_2(\Delta I)\right\}$$

Where: ΔT = Measured ΔT by RTD manifold instrumentation (Unit 2);

$\frac{1 + \tau_1 S}{1 + \tau_2 S}$ = Lead-lag compensator on measured ΔT;

τ_1, τ_2 = Time constants utilized in lead-leg compensator for ΔT, $\tau_1 \geq 8$ S, $\tau_2 \leq 3$ s;

$\frac{1}{1 + \tau_3 S}$ = Lag compensator on measured ΔT;

τ_3 = Time constants utilized in the lag compensator for ΔT, τ_3=0s;

ΔT_0 = Indicated ΔT at RATED THERMAL POWER;

K_4 $\leq$ 1.089 (Unit 2),

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

$\frac{\tau_7 S}{1 + \tau_7 S}$ = The function generated by the rate-lag compensator for T_{avg} dynamic compensation,

τ_7 = Time constants utilized in the rate-lag compensator for T_{avg}, $\tau_7 \geq 10$ s,

$\frac{1}{1 + \tau_6 S}$ = Lag compensator on measured T_{avg};

TABLE 2.2-1a (Continued)

TABLE NOTATIONS (Continued) - UNIT 2

NOTE 3: (Continued)

τ_6 = Time constant utilized in the measured T_{avg} lag compensator, $\tau_6 = 0$ s;

K_6 $\geq$ 0.0013/°F (Unit 2) for $T > T''$ and $K_6 = 0$ for $T \leq T''$.

T = Average Temperature, °F;

T'' = Indicated T_{avg} at RATED THERMAL POWER (Calibration temperature for ΔT instrumentation, $\leq$ 588.5°F (Unit 2)),

S = Laplace transform variable, s^{-1}; and

$f_2(\Delta I)$ = 0 for all ΔI.

NOTE 4: The channel's maximum Trip Setpoint shall not exceed its computed Trip Setpoint by more than 2.4% (Unit 2) of ΔT span.

2.1 SAFETY LIMITS

BASES

2.1.1 REACTOR CORE - UNIT 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 correlations which have been developed to predict the DNB flux and the location of DNB for axially uniform and nonuniform heat flux distributions. The local DNB heat flux ratio (DNBR) is defined as the ratio of the heat flux that would cause DNB at a particular core location to the local heat flux and is indicative of the margin to DNB.

The DNB thermal design criterion is that the probability that DNB will not occur on the most limiting rod is at least 95% (at a 95% confidence level) for any Condition I or II event.

In meeting the DNB design criterion, uncertainties in plant operating parameters, nuclear and thermal parameters, fuel fabrication parameters, and computer codes must be considered. As described in the FSAR, the effects of these uncertainties have been statistically combined with the correlation uncertainty. Design limit DNBR values have been determined that satisfy the DNB design criterion.

Additional DNBR margin is maintained by performing the safety analyses to a higher DNBR limit. This margin between the design and safety analysis limit DNBR values is used to offset known DNBR penalties (e.g., rod bow and transition core) and to provide DNBR margin for operating and design flexibility.

The curves of Figure 2.1-1 show reactor core safety limits for a range of THERMAL POWER, REACTOR COOLANT SYSTEM pressure, and average temperature which satisfy the following criteria:

A. The average enthalpy at the vessel exit is less than the enthalpy of saturated liquid (far left line segment in each curve).

B. The minimum DNBR satisfies the DNB design criterion (all the other line segments in each curve). The VANTAGE 5 fuel is analyzed using the WRB-2 correlation with design limit DNBR values of 1.24 and 1.23 for the typical cell and thimble cells, respectively. The LOPAR fuel is analyzed using the WRB-1 correlation with design limit DNBR values of 1.23 and 1.22 for the typical and thimble cells, respectively.

C. The hot channel exit quality is not greater than the upper limit of the quality range (including the effect of uncertainties) of the DNB correlations. This is not a limiting criterion for this plant.

2.1 SAFETY LIMITS

BASES

2.1.1 REACTOR CORE - UNIT 2

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 (R Grid) correlation. The W-3 (R Grid) DNB correlation has been developed to predict the DNB flux and the location of DNB for axially uniform and nonuniform heat flux distributions. The local DNB heat flux ratio (DNBR) is defined as the ratio of the heat flux that would cause DNB at a particular core location to the local heat flux and 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% probability at a 95% confidence level that DNB will not occur and is chosen as an appropriate margin to DNB for all operating conditions.

The curves of Figure 2.1-1a show reactor core safety limits which are determined for a range of reactor operating conditions. The core limits represent the loci of points of THERMAL POWER, REACTOR COOLANT SYSTEM pressure and average temperature which satisfy the following criteria:

A. The average enthalpy at the vessel exit is equal to the enthalpy of saturated liquid (far left line segment in each curve).

B. The minimum DNBR is not less than the design limit (all the other line segments in each curve).

C. The hot channel exit quality is not greater than the upper limit of the quality range of the W-3 (R-Grid) correlation which is 15% (middle line segment on Reactor Coolant System pressure curves, 2400 psia and 2250 psia; this is not a limiting criterion for this plant).

REACTOR CORE (Continued)

These curves are based on an enthalpy hot channel factor, $F_{\Delta H}^{RTP}$, and a reference cosine with a peak of 1.55 for axial power shape. An allowance is included for an increase in $F_{\Delta H}^{N}$ at reduced power based on the expression:

$$F_{\Delta H}^{N} = F_{\Delta H}^{RTP}[1+ PF_{\Delta H}\ (1-P)]$$

Where: $F_{\Delta H}^{RTP}$ is the limit at RATED THERMAL POWER (RTP) specified in the CORE OPERATING LIMITS REPORT (COLR).

$PF_{\Delta H}$ is the Power Factor Multiplier for $F_{\Delta H}^{N}$ [illegible]ied in the COLR, and

P is the fraction of RTP.

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 (ΔI) function of the Overtemperature trip. When the axial power imbalance is not within the tolerance, the axial power imbalance effect on the Overtemperature ΔT trips will reduce the Setpoints to provide protection consistent with core Safety Limits.

2.1.2 REACTOR COOLANT SYSTEM PRESSURE

The restriction of this Safety Limit protects the integrity of the Reactor Coolant System (RCS) from overpressurization and thereby prevents the release of radionuclides contained in the reactor coolant from reaching the containment atmosphere.

The reactor vessel, pressurizer, and the RCS piping, valves and fittings are designed to Section III of the ASME Code for Nuclear Power Plants which permits a maximum transient pressure of 110% (2735 psig) of design pressure. The Safety Limit of 2735 psig is therefore consistent with the design criteria and associated Code requirements.

The entire RCS is hydrotested at 125% (3107 psig) of design pressure, to demonstrate integrity prior to initial operation.

BORATED WATER SOURCE - SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.1.2.5 As a minimum, one of the following borated water sources shall be OPERABLE:

a. A Boric Acid Storage Tank with:

1) A minimum contained borated water volume of 9504 gallons (19% of instrument span) (LI-102A, LI-104A),

2) A boron concentration between 7000 ppm and 7700 ppm, and

3) A minimum solution temperature of 65°F (TI-0103).

b. The refueling water storage tank (RWST) with:

1) A minimum contained borated water volume of 99404 gallons (9% of instrument span) (LI-0990A&B, LI-0991A&B, LI-0992A, LI-0993A),

2) A boron concentration between 2400 ppm and 2600 ppm, and

3) A minimum solution temperature of 44°F (Unit 1) or 54°F (Unit 2) (TI-10982).

APPLICABILITY: MODES 5 and 6.

ACTION:

With no borated water source OPERABLE, suspend all operations involving CORE ALTERATIONS or positive reactivity changes.

SURVEILLANCE REQUIREMENTS

4.1.2.5 The above required borated water source shall be demonstrated OPERABLE:

a. At least once per 7 days by:

1) Verifying the boron concentration of the water,

2) Verifying the contained borated water volume, and

3) When the boric acid storage tank is the source of borated water and the ambient temperature of the boric acid storage tank room (TISL-20902, TISL-20903) is ≤72°F, verify the boric acid storage tank solution temperature is >65°F.

b. At least once per 24 hours by verifying the RWST temperature (TI-10982) when it is the source of borated water and the outside air temperature is less than 40°F (Unit 1) or 50°F (Unit 2).

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 Tank with:

   1) A minimum contained borated water volume of 36674 gallons (81% of instrument span) (LI-102A, LI-104A),

   2) A boron concentration between 7000 ppm and 7700 ppm, and

   3) A minimum solution temperature of 65°F (TI-0103).

b. The refueling water storage tank (RWST) with:

   1) A minimum contained borated water volume of 631478 gallons (86% of instrument span) (LI-0990A&B, LI-0991A&B, LI-0992A, LI-0993A),

   2) A boron concentration between 2400 ppm and 2600 ppm,

   3) A minimum solution temperature of 44°F (Unit 1) or 54°F (Unit 2),

   4) A maximum solution temperature of 116°F (TI-10982), and

   5) RWST Sludge Mixing Pump Isolation Valves capable of closing on RWST low-level.

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

ACTION:

a. With the Boric Acid Storage Tank inoperable and being used as one of the above required borated water sources, restore the tank to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and borated to a SHUTDOWN MARGIN as required by Figure 3.1-2 at 200°F; restore the Boric Acid Storage Tank to OPERABLE status within the next 7 days or be in COLD SHUTDOWN within the next 30 hours.

b. With the RWST inoperable, except for the Sludge Mixing Pump Isolation Valves, restore the tank to OPERABLE status within 1 hour or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

LIMITING CONDITION FOR OPERATION (Continued)

ACTION (Continued)

c. With a Sludge Mixing Pump Isolation Valve(s) inoperable, restore the valve(s) to OPERABLE status within 24 hours or isolate the sludge mixing system by either closing the manual isolation valves or deenergizing the OPERABLE solenoid pilot valve within 5 hours and maintain closed.

SURVEILLANCE REQUIREMENTS

4.1.2.6 Each borated water source shall be demonstrated OPERABLE:

a. At least once per 7 days by:

1) Verifying the boron concentration in the water,

2) Verifying the contained borated water volume of the water source, and

3) When the boric acid storage tank is the source of borated water and the ambient temperature of the boric acid storage tank room (TISL-20902, TISL-20903) is ≤ 72°F, verify the boric acid storage tank solution temperature is ≥ 65°F.

b. At least once per 24 hours by verifying the RWST temperature (TI-10982) when the outside air temperature is less than 40°F (Unit 1) or 50°F (Unit 2).

c. At least once per 18 months by verifying that the Sludge Mixing Pump Isolation Valves automatically close upon RWST low-level test signal.

ROD DROP TIME

LIMITING CONDITION FOR OPERATION

3.1.3.4 The individual shutdown and control rod drop time from the physical fully withdrawn position shall be less than or equal to 2.7 (Unit 1) or 2.2 (Unit 2) seconds from beginning of decay of stationary gripper coil voltage to dashpot entry with:

a. T_{avg} (TI-0412, TI-0422, TI-0432, TI-0442) 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 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 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.

3/4.2 POWER DISTRIBUTION LIMITS

3/4.2.1 AXIAL FLUX DIFFERENCE - UNIT 1

LIMITING CONDITION FOR OPERATION

3.2.1 The indicated (NI-0041B, NI-0042B, NI-0043B, NI-0044B) AXIAL FLUX DIFFERENCE (AFD) shall be maintained within the limits specified in the CORE OPERATING LIMITS REPORT (COLR).

APPLICABILITY: MODE 1 ABOVE 50 PERCENT RATED THERMAL POWER*.

ACTION:

a. With the indicated AXIAL FLUX DIFFERENCE outside of the limits specified in the COLR,

1. Either restore the indicated AFD to within the 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 percent of RATED THERMAL POWER within the next 4 hours.

b. THERMAL POWER shall not be increased above 50% of RATED THERMAL POWER unless the indicated AFD is within the limits specified in the COLR.

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:

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

2) At least once per hour until the AFD Monitor Alarm is updated after restoration to OPERABLE status.

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.

c. The provisions of Specification 4.0.4 are not applicable.

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 its limits.

*See Special Test Exceptions Specification 3.10.2.

3/4.2 POWER DISTRIBUTION LIMITS

3/4.2.1 AXIAL FLUX DIFFERENCE - UNIT 2

LIMITING CONDITION FOR OPERATION

3.2.1 The indicated (NI-0041B, NI-0042B, NI-0043B, NI-0044B) AXIAL FLUX DIFFERENCE (AFD) shall be maintained within the target band (flux difference units) about the target flux difference. The target band is specified in the CORE OPERATING LIMITS REPORT (COLR).

The indicated AFD may deviate outside the required target band at greater than or equal to 50% but less than 90% of RATED THERMAL POWER provided the indicated AFD is within the Acceptable Operation Limits specified in the COLR and the cumulative penalty deviation time does not exceed 1 hour during the previous 24 hours.

The indicated AFD may deviate outside the required target band at greater than 15% but less than 50% of RATED THERMAL POWER provided the cumulative penalty deviation time does not exceed 1 hour during the previous 24 hours.

APPLICABILITY: MODE 1, above 15% of RATED THERMAL POWER.* #

ACTION:

a. With the indicated AFD outside of the required target band and with THERMAL POWER greater than or equal to 90% of RATED THERMAL POWER, within 15 minutes either:

   1. Restore the indicated AFD to within the target band limits, or

   2. Reduce THERMAL POWER to less than 90% of RATED THERMAL POWER.

b. With the indicated AFD outside of the required target band for more than 1 hour of cumulative penalty deviation time during the previous 24 hours or outside the Acceptable Operation Limits specified in the COLR and with THERMAL POWER less than 90% but equal to or greater than 50% of RATED THERMAL POWER, reduce:

   1. THERMAL POWER to less than 50% of RATED THERMAL POWER within 30 minutes, and

   2. The Power Range Neutron Flux* - High Setpoints to less than or equal to 55% of RATED THERMAL POWER within the next 4 hours.

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*See Special Test Exceptions Specification 3.10.2.

#Surveillance testing of the Power Range Neutron Flux Channel may be performed (below 90% of RATED THERMAL POWER) pursuant to Specification 4.3.1.1 provided the indicated AFD is maintained within the Acceptable Operation Limits specified in the COLR. A total of 16 hours operation may be accumulated with the AFD outside of the above required target band during testing without penalty deviation.

LIMITING CONDITION FOR OPERATION - UNIT 2

ACTION (Continued)

c. With the indicated AFD outside of the required target band for more than 1 hour of cumulative penalty deviation time during the previous 24 hours and with THERMAL POWER less than 50% but greater than 15% of RATED THERMAL POWER, the THERMAL POWER shall not be increased equal to or greater than 50% of RATED THERMAL POWER until the indicated AFD is within the required target band and the cumulative penalty deviation has been reduced to less than 1 hour in the previous 24 hours.

SURVEILLANCE REQUIREMENTS

4.2.1.1 The indicated AFD 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 until the AFD Monitor Alarm is updated after restoration to OPERABLE status.

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 target band when two or more OPERABLE excore channels are indicating the AFD to be outside the target band. Penalty deviation outside of the required target band shall be accumulated on a time basis of:

a. One minute penalty deviation for each 1 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 1 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 Specification 4.2.1.3 above or by linear interpolation between the most recently measured value and 0% at the end of the cycle life. The provisions of Specification 4.0.4 are not applicable.

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:

$$F_Q(Z) \leq \frac{F_Q^{RTP}}{P} [K(Z)] \text{ for } P > 0.5$$

$$F_Q(Z) \leq \frac{F_Q^{RTP}}{0.5} [K(Z)] \text{ for } P \leq 0.5$$

Where: F_Q^{RTP} = the F_Q limit at RATED THERMAL POWER (RTP) specified in the CORE OPERATING LIMITS REPORT (COLR),

Where: $P = \frac{\text{THERMAL POWER}}{\text{RATED THERMAL POWER}}$, and

$K(Z)$ = the normalized $F_Q(Z)$ as a function of core height specified in the COLR.

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 ΔT Trip Setpoints (value of K_4) have been reduced at least 1% (in ΔT span) for each 1% $F_Q(Z)$ exceeds the limit; and

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

SURVEILLANCE REQUIREMENTS - UNIT 1

4.2.2.1 The provisions of Specifications 4.0.4 are not applicable.

4.2.2.2 $F_Q(Z)$ shall be evaluated to determine if it 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. Determining the computed heat flux hot channel factor, $F_Q^C(Z)$, as follows:

Increase the measured $F_Q(Z)$ obtained from the power distribution map by 3% to account for manufacturing tolerances and further increase the value by 5% to account for measurement uncertainties.

c. Verifying that $F_Q^C(Z)$, obtained in Specification 4.2.2.2b above, satisfies the relationship in Specification 3.2.2.

d. Satisfying the following relationship:

$$F_Q^C(Z) \leq \frac{F_Q^{RTP} \quad x\ K(Z)}{P\ x\ W(Z)} \text{ for } P > 0.5$$

$$F_Q^C(Z) \leq \frac{F_Q^{RTP} \quad x\ K(Z)}{0.5\ x\ W(Z)} \text{ for } P \leq 0.5$$

Where $F_Q^C(Z)$ is obtained in Specification 4.2.2.2b above, F_Q^{RTP} is the F_Q limit, K(Z) is the normalized $F_Q(Z)$ as a function of core height, P is the fraction of RATED THERMAL POWER, and W(Z) is the cycle dependent function that accounts for power distribution transients encountered during normal operation. F_Q^{RTP}, K(Z), and W(Z) are specified in the CORE OPERATING LIMITS REPORT as per Specification 6.8.1.6.

e. Measuring $F_Q(Z)$ according to the following schedule:

1. Upon achieving equilibrium conditions after exceeding by 20% or more of RATED THERMAL POWER, the THERMAL POWER at which FQ(Z) was last determined*, or

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

*During power escalation after each fuel loading, power level may be increased until equilibrium conditions at any power level greater than or equal to 50% of RATED THERMAL POWER have been achieved and a power distribution map obtained.

f. With measurements indicating

$$\underset{\text{over } Z}{\text{maximum}} \left(\frac{F_Q^C(Z)}{K(Z)} \right)$$

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

1) Increase $F_Q^C(Z)$ by 2% and verify that this value satisfies the relationship in Specification 4.2.2.2d, or

2) $F_Q^C(Z)$ shall be measured at least once per 7 Effective Full Power Days until two successive maps indicate that

$$\underset{\text{over } Z}{\text{maximum}} \left(\frac{F_Q^C(Z)}{K(Z)} \right) \quad \text{is not increasing.}$$

g. With the relationships specified in Specification 4.2.2.2d above not being satisfied:

1) Calculate the percent $F_Q(Z)$ exceeds its limits by the following expression:

$$\left\{ \left(\underset{\text{over } Z}{\text{maximum}} \left[\frac{F_Q^C(Z) \times W(Z)}{\frac{F_Q^{RTP}}{P} \times K(Z)} \right] \right) - 1 \right\} \times 100 \text{ for } P > 0.5$$

$$\left\{ \left(\underset{\text{over } Z}{\text{maximum}} \left[\frac{F_Q^C(Z) \times W(Z)}{\frac{F_Q^{RTP}}{0.5} \times K(Z)} \right] \right) - 1 \right\} \times 100 \text{ for } P \leq 0.5, \text{ and}$$

2) The following action shall be taken.

Within 15 minutes, control the AFD to within new AFD limits which are determined by reducing the AFD limits specified in the CORE OPERATING LIMITS REPORT by 1% AFD for each percent $F_Q(Z)$ exceeds its limits as determined in Specification 4.2.2.2g.1. Within 8 hours, reset the AFD alarm setpoints to these modified limits.

SURVEILLANCE REQUIREMENTS (Continued) - UNIT 1

h. The limits specified in Specification 4.2.2.2c are applicable in all core plane regions, i.e. 0 - 100%, inclusive.

i. The limits specified in Specifications 4.2.2.2d, 4.2.2.2f, and 4.2.2.2g 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 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.

SURVEILLANCE REQUIREMENTS - UNIT 2

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 before exceeding 75% of RATED THERMAL POWER following each fuel loading.

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 Specification 4.2.2.2b. above to:

1) The F_{xy} limits for RATED THERMAL POWER (F_{xy}^{RTP}) for the appropriate measured core planes given in Specification 4.2.2.2e. and f. below, and

2) The relationship:

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

Where F_{xy}^L is the limit for fractional THERMAL POWER operation expressed as a function of F_{xy}^{RTP}, PF_{xy} is the power factor multiplier for F_{xy} specified in the COLR, 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 either:

a) 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 Effective Full Power Days (EFPD), whichever occurs first.

SURVEILLANCE REQUIREMENTS (Continued) - UNIT 2

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 used in the Constant Axial Offset Control analysis for RATED THERMAL POWER (F_{xy}^{RTP}) shall be specified for all core planes containing Bank "D" control rods and all unrodded core planes in the COLR per Specification 6.8.1.6;

f. The F_{xy} limits of Specification 4.2.2.2e. 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, and

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.

3/4.2.5 DNB PARAMETERS

LIMITING CONDITION FOR OPERATION

3.2.5 The following DNB-related parameters shall be maintained within the limits:

a. Reactor Coolant System T_{avg} (TI-0412, TI-0422, TI-0432, TI-0442), $\leq$ 592.5°F (Unit 1) or $\leq$ 591°F (Unit 2).

b. Pressurizer Pressure (PI-0455A,B&C, PI-0436 & PI-0456A, PI-0457 & PI-0457A, PI-0458 & PI-0458A), $\geq$ 2199 psig* (Unit 1) or 2224 psig* (Unit 2).

c. Reactor Coolant System Flow (FI-0414, FI-0415, FI-0416, FI-0424, FI-0425, FI-0426, FI-0434, FI-0435, FI-0436, FI-0444, FI-0445, FI-0446) $\geq$391,225 gpm** (Unit 1) or 396,198 gpm** (Unit 2).

APPLICABILITY: MODE 1.

ACTION:

With any of the above parameters exceeding its limit, restore the parameter to within its limit within 2 hours or reduce THERMAL POWER to less than 5% of RATED THERMAL POWER within the next 4 hours.

SURVEILLANCE REQUIREMENTS

4.2.5.1 Reactor Coolant System T_{avg} and Pressurizer Pressure shall be verified to be within their limits at least once per 12 hours. RCS flow rate shall be monitored for degradation at least once per 12 hours. In the event of flow degradation, RCS flow rate shall be determined by precision heat balance within 7 days of detection of flow degradation.

4.2.5.2 The RCS flow rate indicators shall be subjected to CHANNEL CALIBRATION at each fuel loading and at least once per 18 months.

4.2.5.3 After each fuel loading, the RCS flow rate shall be determined by precision heat balance prior to operation above 75% RATED THERMAL POWER. The RCS flow rate shall also be determined by precision heat balance at least once per 18 months. Within 7 days prior to performing the precision heat balance flow measurement, the instrumentation used for performing the precision heat balance shall be calibrated. The provisions of 4.0.4 are not applicable for performing the precision heat balance flow measurement.

*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.

**Includes a 2.2% (Unit 1) or 3.5% (Unit 2) flow measurement uncertainty.

TABLE 3.[illegible] (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TOTAL ALLOWANCE (TA)	Z	SENSOR ERROR (S)	TRIP SETPOINT	ALLOWABLE VALUE
7. Semi-Automatic Switchover to Containment Emergency Sump (Continued)					
b. RWST Level--Low-Low Coincident With Safety Injection (LI-0990A&B, LI-0991A&B, LI-0992A, LI-0993A)	3.5	0.71	1.67	≥ 275.3 in. from tank base (≥ 39.1% of instrument span)	≥ 264.9 in. from tank base (≥ 37.4% of instrument span)
8. Loss of Power to 4.16 kV ESF Bus					
a. 4.16 kV ESF Bus Undervoltage-Loss of Voltage	N.A.	N.A.	N.A.	≥ 2975 volts with a ≤ 0.8 second time delay.	≥ 2912 volts with a ≤ 0.8 second time delay.
b. 4.16 kV ESF Bus Undervoltage-Degraded Voltage	N.A.	N.A.	N.A.	≥ 3746 volts with a ≤ 20 second time delay.	≥ 3683 volts with a ≤ 20 second time delay.
9. Engineered Safety Features Actuation System Interlocks					
a. Pressurizer Pressure, P-11 (PI-0455A,B&C, PI-0456 & PI-0456A, PI-0457 & PI-0457A, PI-0458 & PI-0458A).	N.A.	N.A.	N.A.	≤ 2000 psig (Unit 1) ≤ 1970 psig (Unit 2)	≤ 2010 psig (Unit 1) ≤ 1980 psig (Unit 2)
b. Reactor Trip, P-4	N.A.	N.A.	N.A.	N.A	N.A.

3/4.5 EMERGENCY CORE COOLING SYSTEMS

3/4.5.1 ACCUMULATORS

LIMITING CONDITION FOR OPERATION

3.5.1 Each Reactor Coolant System (RCS) accumulator shall be OPERABLE with:

a. The isolation valve open,

b. Unit 1: A contained borated water volume of between 6555 (29.2% of instrument span) and 6909 gallons (70.7% of instrument span) (LI-0950, LI-0951, LI-0952, LI-0953, LI-0954, LI-0955, LI-0956, LI-0957),

Unit 2: - A contained borated water volume of between 6616 (36% of instrument span) and 6854 gallons (64% of instrument span) (LI-0950, LI-0951, LI-0952, LI-0953, LI-0954, LI-0955, LI-0956, LI-0957),

c. A boron concentration of between 1900 ppm and 2600 ppm, and

d. A nitrogen cover-pressure of between 617 and 678 psig. (PI-0960A&B, PI-0961A&B, PI-0962A&B, PI-0963A&B, PI-0964A&B, PI-0965A&B, PI-0966A&B, PI-0967A&B)

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 1 hour or be in at least HOT STANDBY within the next 6 hours and reduce pressurizer pressure to less than 1000 psig 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 6 hours and reduce pressurizer pressure to less than 1000 psig within the following 6 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 (HV-8808A, B, C, D).

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*Pressurizer pressure above 1000 psig.

3/4.5.4 REFUELING WATER STORAGE TANK

LIMITING CONDITION FOR OPERATION

3.5.4 The refueling water storage tank (RWST) shall be OPERABLE with:

a. A minimum contained borated water volume of 631,478 gallons (86% of instrument span) (LI-0990A&B, LI-0991A&B, LI-0992A, LI-0993A).

b. A boron concentration of between 2400 ppm and 2600 ppm of boron,

c. A minimum solution temperature of 44°F (Unit 1) or 54°F (Unit 2), and

d. A maximum solution temperature of 116°F (TI-10982).

e. RWST Sludge Mixing Pump Isolation valves capable of closing on RWST low-level.

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

ACTION:

a. With the RWST inoperable except for the Sludge Mixing Pump Isolation Valves, restore the tank to OPERABLE status within 1 hour or be in at least HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With a Sludge Mixing Pump Isolation Valve(s) inoperable, restore the valve(s) to OPERABLE status within 24 hours or isolate the sludge mixing system by either closing the manual isolation valves or deenergizing the OPERABLE solenoid pilot valve within 6 hours and maintain closed.

SURVEILLANCE REQUIREMENTS

4.5.4 The RWST shall be demonstrated OPERABLE:

a. At least once per 7 days by:

1) Verifying the contained borated water volume in the tank, and

2) Verifying the boron concentration of the water.

b. At least once per 24 hours by verifying the RWST temperature when the outside air temperature is less than 40°F (Unit 1) or 50°F (Unit 2).

c. At least once per 18 months by verifying that the sludge mixing pump isolation valves automatically close upon an RWST low-level test signal.

3/4.2 POWER DISTRIBUTION LIMITS - UNIT 1

BASES

The specifications of this section provide assurance of fuel integrity during Condition I (Normal Operation) and II (Incidents of Moderate Frequency) events by: (1) meeting the DNB design criterion 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; and

$F^N_{\Delta H}$ 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.

3/4.2.1 AXIAL FLUX DIFFERENCE

The limits on AXIAL FLUX DIFFERENCE (AFD) assure that the $F_Q(Z)$ upper bound envelope of the F_Q limit specified in the CORE OPERATING LIMITS REPORT (COLR) times K(Z) is not exceeded during either normal operation or in the event of xenon redistribution following power changes.

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 1-minute average of each of the OPERABLE excore detector outputs and provides an alarm message immediately if the AFD for two or more OPERABLE excore channels are outside the allowed ΔI power operating space for RAOC operation specified within the COLR and the THERMAL POWER is greater than 50% of RATED THERMAL POWER.

AXIAL FLUX DIFFERENCE (Continued)

3/4.2.2 and 3/4.2.3 HEAT FLUX HOT CHANNEL FACTOR AND NUCLEAR ENTHALPY RISE HOT CHANNEL FACTOR - $F^N_{\Delta H}$

The limits on heat flux hot channel factor and nuclear enthalpy rise hot channel factor ensure that: (1) the design limit on peak local power density is not exceeded, (2) the DNB design criterion is met, and (3) 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 ensure that the limits are maintained provided:

a. Control rods in a single group move together with no inidivdual rod insertion differing by more than ± 12 steps, indicated, from the group demand position;

b. Control rod groups are sequenced with a constant tip-to-tip distance between banks as described in Specification 3.1.3.6;

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

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. 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.

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.

The heat flux hot channel factor $F_Q(Z)$ is measured periodically and increased by a cycle and height dependent power factor appropriate to RAOC operation, W(Z), to provide assurance that the limit on the heat flux hot channel factor, $F_Q(Z)$, is met. W(Z) accounts for the effects of normal operation transients within the AFD band and was determined from expected power control maneuvers over the full range of burnup conditions in the core. The W(Z) function for normal operation and the AFD band are provided in the CORE OPERATING LIMITS REPORT per Specification 6.8.1.6.

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

When $F^N_{\Delta H}$ is measured, (i.e., inferred), measurement uncertainty (i.e., the appropriate uncertainty on the incore inferred hot rod peaking factor) must be allowed for and 4% is the appropriate allowance for a full core map taken with the incore detection system.

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 limit of 1.02 was selected to provide an allowance for the uncertainty associated with the indicated power tilt.

The 2-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% for each percent of tilt in excess of 1.

For purposes of monitoring QUADRANT POWER TILT RATIO when one excore detector is inoperable, the moveable 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 four symmetric thimbles. The two sets of four symmetric thimbles is a unique set of eight detector locations. 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 with the initial FSAR assumptions and have been analytically demonstrated adequate to meet the DNB design criterion throughout each analyzed transient. The indicated T_{avg} value of 592.5°F and the indicated pressurizer pressure value of 2199 psig correspond to analytical limits of 594.4°F and 2185 psig respectively, with allowance for measurement uncertainty.

3/4.2.5 DNB PARAMETERS (Continued)

The 12-hour periodic surveillance of these parameters through instrument readout is sufficient to ensure that the parameters are restored within their limits following load changes and other expected transient operation. The 18 month periodic measurement of the RCS total flow rate is adequate to detect flow degradation and ensure correlation of the flow indication channels with measured flow such that the indicated percent flow will provide sufficient verification of the flow rate degradation on a 12 hour basis. A change in indicated percent flow which is greater than the instrument channel inaccuracies and parallax errors is an appropriate indication of RCS flow degradation.

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 minimum DNBR in the core greater than or equal to 1.30 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;

$F_{\Delta H}^N$ 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; and

$F_{xy}(Z)$ 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

The limits on AXIAL FLUX DIFFERENCE (AFD) assure that the $F_Q(Z)$ upper bound envelope of the F_Q limit specified in the CORE OPERATING LIMITS REPORT (COLR) times K(Z) 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 conditons. The 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.

AXIAL FLUX DIFFERENCE (Continued)

Although it is intended that the plant will be operated with the AFD within the target band required by Specification 3.2.1 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 specified in the COLR 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 1-minute average of each of the OPERABLE excore detector outputs and provides an alarm message immediately if the AFD for two 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 NUCLEAR ENTHALPY RISE HOT CHANNEL FACTOR - $F^N_{\Delta H}$

The limits on heat flux hot channel factor 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 ensure that the limits are maintained provided:

a. Control rods in a single group move together with no individual rod insertion differing by more than ± 12 steps, indicated, from the group demand position;

b. Control rod groups are sequenced with a constant tip-to-tip distance between banks as described in Specification 3.1.3.6;

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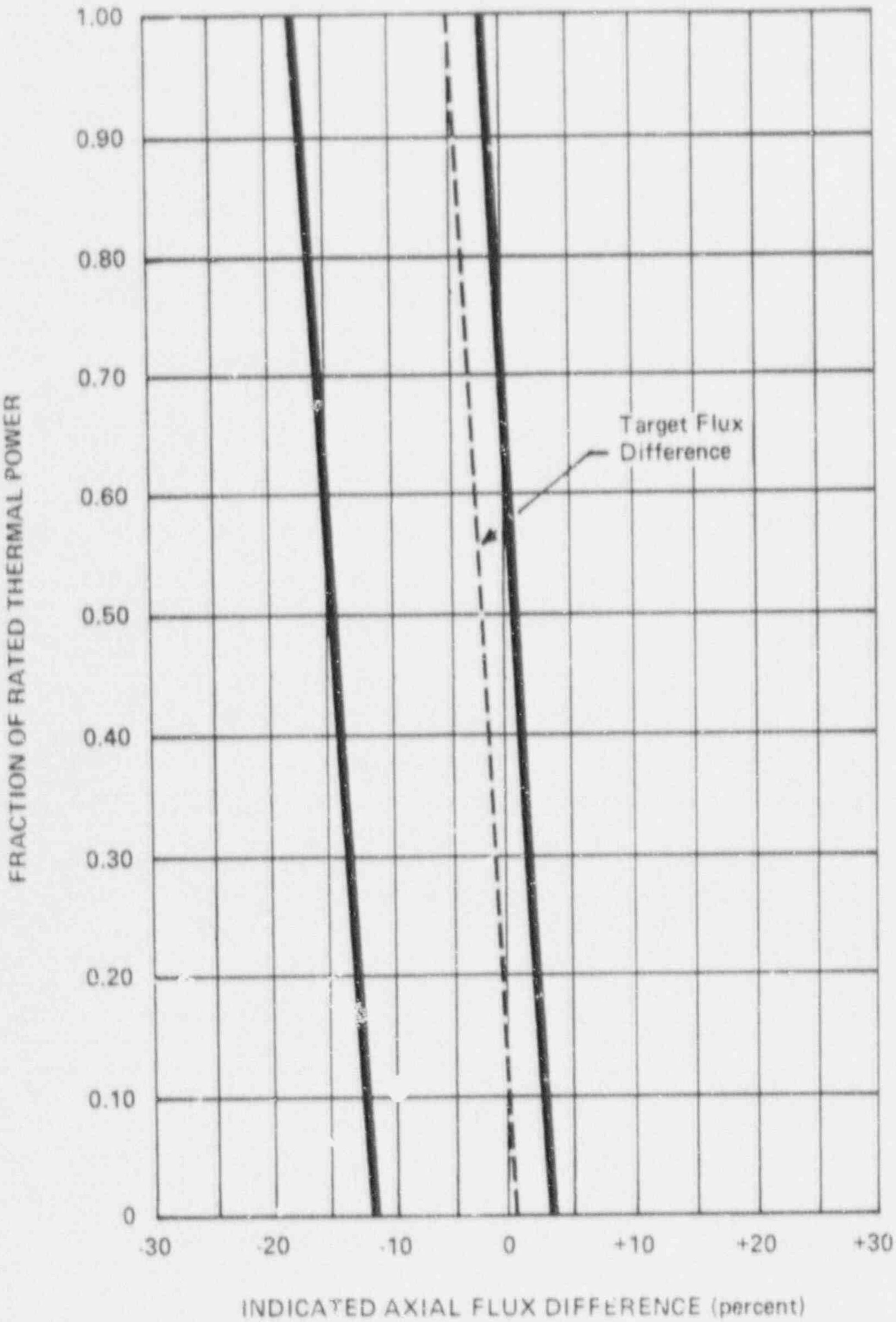

FIGURE B 3/4 2-1
TYPICAL INDICATED AXIAL FLUX DIFFERENCE VERSUS THERMAL POWER

HEAT FLUX HOT CHANNEL FACTOR 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; and

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. 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.

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.

When $F^N_{\Delta H}$ is measured, (i.e., inferred), measurement uncertainty (i.e., the appropriate uncertainty on the incore inferred hot rod peaking factor) must be allowed for and 4% is the appropriate allowance for a full core map taken with the incore detection system.

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

a. Design limit DNBR of 1.30 vs 1.28,

b. Grid Spacing (K_s) of 0.046 vs 0.059,

c. Thermal Diffusion Coefficient of 0.038 vs 0.059,

d. DNBR Multiplier of 0.86 vs 0.88, and

e. Pitch reduction.

The applicable values of rod bow penalties are referenced in the FSAR.

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

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 F_{xy} limit for RATED THERMAL POWER (F_{xy}^{RTP}) as specified in the COLR per Specification 6.8.1.6 was determined from expected power control maneuvers over the full range of burnup conditions in the core.

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 limit of 1.02 was selected to provide an allowance for the uncertainty associated with the indicated power tilt.

The 2-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% for each percent of tilt in excess of 1.

For purposes of monitoring QUADRANT POWER TILT RATIO when one excore detector is inoperable, the moveable 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 four symmetric thimbles. The two sets of four symmetric thimbles is a unique set of eight detector locations. 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 with the initial FSAR assumptions and have been analytically demonstrated adequate to maintain a minimum DNBR of 1.30 throughout each analyzed transient. The indicated T_{avg} value of 591°F and the indicated pressurizer pressure value of 2224 psig correspond to analytical limits of 592.5°F and 2205 psig respectively, with allowance for measurement uncertainty.

3/4.2.5 DNB PARAMETERS (Continued)

The 12-hour periodic surveillance of these parameters through instrument readout is sufficient to ensure that the parameters are restored within their limits following load changes and other expected transient operation. The 18 month periodic measurement of the RCS total flow rate is adequate to detect flow degradation and ensure correlation of the flow indication channels with measured flow such that the indicated percent flow will provide sufficient verification of the flow rate degradation on a 12 hour basis. A change in indicated percent flow which is greater than the instrument channel inaccuracies and parallax errors is an appropriate indication of RCS flow degradation.

3/4.4.1 REACTOR COOLANT LOOPS AND COOLANT CIRCULATION

The plant is designed to operate with all reactor coolant loops in operation and meet the DNB design criterion during all normal operations and anticipated transients. In MODES 1 and 2 with one reactor coolant loop not in operation this specification requires that the plant be in at least HOT STANDBY within 6 hours.

In MODE 3, two reactor coolant loops provide sufficient heat removal capability for removing core decay heat even in the event of a bank withdrawal accident; however, a single reactor coolant loop provides sufficient heat removal capacity if a bank withdrawal accident can be prevented, i.e., by opening the Reactor Trip System breakers.

In MODE 4, and in MODE 5 with reactor coolant loops filled, a single reactor coolant loop or RHR train provides sufficient heat removal capability for removing decay heat; but single failure considerations require that at least two trains/loops (either RHR or RCS) be OPERABLE.

In MODE 5 with reactor coolant loops not filled, a single RHR train provides sufficient heat removal capability for removing decay heat; but single failure considerations, and the unavailability of the steam generators as a heat removing component, require that at least two RHR trains be OPERABLE. The locking closed of the required valves in Mode 5 (with the loops not filled) precludes the possibility of uncontrolled boron dilution of the filled portion of the Reactor Coolant System. These actions prevent flow to the RCS of unborated water in excess of that analyzed. These limitations are consistent with the initial conditions assumed for the boron dilution accident in the safety analysis.

The operation of one reactor coolant pump (RCP) or one RHR pump provides adequate flow to ensure mixing, prevent stratification, and produce gradual reactivity changes during boron concentration reductions in the Reactor Coolant System. The reactivity change rate associated with boron reduction will, therefore, be within the capability of operator recognition and control.

The restrictions on starting an RCP with one or more RCS cold legs less than or equal to 350°F are provided to prevent RCS pressure transients, caused by energy additions from the Secondary Coolant System, which could exceed the limits of Appendix G to 10 CFR Part 50. The RCS will be protected against overpressure transients and will not exceed the limits of Appendix G by restricting starting of the RCPs to when the secondary water temperature of each steam generator is less than 50°F above each of the RCS cold leg temperatures. In MODE 4 the starting of an RCP, when no other RCP is operating, is restricted to a range of temperatures that are consistent with analysis assumptions used to demonstrate that the RHR design pressure is not exceeded when RHR relief valves are used for RCS overpressure protection.

ECCS SUBSYSTEMS (Continued)

The limitation for all safety injection pumps to be inoperable below 350°F provides assurance that a mass addition pressure transient can be relieved by the operation of a single PORV.

The Surveillance Requirements provided to ensure OPERABILITY of each component ensure that at a minimum, the assumptions used in the safety analyses are met and that subsystem OPERABILITY is maintained. Surveillance Requirements for throttle valve position stops and flow balance testing provide assurance that proper ECCS flows will be maintained in the event of a LOCA. Maintenance of proper flow resistance and pressure drop in the piping system to each injection point is necessary to: (1) prevent total pump flow from exceeding runout conditions when the system is in its minimum resistance configuration, (2) provide the proper flow split between injection points in accordance with the assumptions used in the ECCS-LOCA analyses, (3) provide an acceptable level of total ECCS flow to all injection points equal to or above that assumed in the ECCS-LOCA analyses and (4) to ensure that centrifugal charging pump injection flow which is directed through the seal injection path is less than or equal to the amount assumed in the safety analysis. The surveillance requirements for leakage testing of ECCS check valves ensure a failure of one valve will not cause an intersystem LOCA. In MODE 3, with either HV-8809A or B closed for ECCS check valve leak testing, adequate ECCS flow for core cooling in the event of a LOCA is assured.

3/4.5.4 REFUELING WATER STORAGE TANK

The OPERABILITY of the Refueling Water Storage Tank (RWST) as part of the ECCS ensures that sufficient negative reactivity is injected into the core to counteract any positive increase in reactivity caused by RCS cooldown. RCS cooldown can be caused by inadvertent depressurization, a loss-of-coolant accident, or a steam line rupture.

The limits on RWST minimum volume and boron concentration ensure that 1) sufficient water is available within containment to permit recirculation cooling flow to the core, 2) the reactor will remain subcritical in the cold condition following a small LOCA or steamline break, assuming complete mixing of the RWST, RCS, and ECCS water volumes with all control rods inserted except the most reactive control assembly (ARI-1), and 3) the reactor will remain subcritical in the cold condition following a large break LOCA assuming complete mixing of the RWST, RCS, ECCS water and other sources of water that may eventually reside in the sump, post-LOCA with all control rods assumed to be out (Unit 2) or all control rods inserted except for the two most reactive control assemblies (Unit 1).

The contained water volume limit includes an allowance for water not usable because of tank discharge line location or other physical characteristics.

SEMIANNUAL RADIOACTIVE EFFLUENT RELEASE REPORT (Continued)

The Semiannual Radioactive Effluent Release Reports shall also include the following: an explanation as to why the inoperability of liquid or gaseous effluent monitoring instrumentation was not corrected within the time specified in Specification 3.3.3.9 or 3.3.3.10, respectively; and description of the events leading to liquid holdup tanks or gas storage tanks exceeding the limits of Specification 3.11.1.4 or 3.11.2.6, respectively.

MONTHLY OPERATING REPORTS

6.8.1.5 Routine reports of operating statistics and shutdown experience, including documentation of all challenges to the PORVs 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 Administrator of the Regional Office of the NRC, no later than the 15th of each month following the calendar month covered by the report.

CORE OPERATING LIMITS REPORT - UNIT 1

6.8.1.6 Core operating limits shall be established and documented in the CORE OPERATING LIMITS REPORT (COLR) before each reload cycle or any remaining part of a reload cycle for the following:

a. SHUTDOWN MARGIN LIMIT FOR MODES 1 and 2 for Specification 3/4.1.1.1,

b. SHUTDOWN MARGIN LIMITS FOR MODES 3, 4, and 5 for Specification 3/4.1.1.2,

c. Moderator temperature coefficient BOL and EOL limits and the 300-ppm surveillance limit for Specification 3/4.1.1.3,

d. Shutdown Rod Insertion Limit for Specification 3/4.1.3.5,

e. Control Rod Insertion Limits for Specification 3/4.1.3.6,

f. Axial Flux Difference Limits for Specification 3/4.2.1,

g. Heat Flux Hot Channel Factor, K(Z) and W(Z) for Specification 3/4.2.2,

h. Nuclear Enthalpy Rise Hot Channel Factor Limit and the Power Factor Multiplier for Specification 3/4.2.3.

The analytical methods used to determine the core operating limits shall be those previously approved by the NRC in:

CORE OPERATING LIMITS REPORT (Continued) - UNIT 1

a. WCAP-9272-P-A, "WESTINGHOUSE RELOAD SAFETY EVALUATION METHODOLOGY," July 1985 (W Proprietary). (Methodology for Specification 3.1.1.3 - Moderator Temperature Coefficient, 3.1.3.5 - Shutdown Bank Insertion Limit, 3.1.3.6 - Control Bank Insertion Limits, and 3.2.3 - Nuclear Enthalpy Rise Hot Channel Factor.)

b. WCAP-10216-P-A, "RELAXATION OF CONSTANT AXIAL OFFSET CONTROL F_Q SURVEILLANCE TECHNICAL SPECIFICATION," June 1983 (W Proprietary). (Methodology for Specifications 3.2.1 - Axial Flux Difference (Relaxed Axial Offset Control) and 3.2.2 - Heat Flux Hot Channel Factor (W(Z) surveillance requirements for F_Q Methodology).)

c. WCAP-9220-P-A, Rev. 1, "WESTINGHOUSE ECCS EVALUATION MODEL-1981 VERSION," February 1982 (W Proprietary). (Methodology for Specification 3.2.2 - Heat Flux Hot Channel Factor.)

The core operating limits shall be determined so that all applicable limits (e.g., fuel thermal-mechanical limits, core thermal-hydraulic limits, ECCS limits, nuclear limits such as shutdown margin, and transient and accident analysis limits) of the safety analysis are met.

The CORE OPERATING LIMITS REPORT, including any mid-cycle revisions or supplements thereto, shall be provided upon issuance, for each reload cycle, to the NRC Document Control Desk with copies to the Regional Administrator and Resident Inspector.

CORE OPERATING LIMITS REPORT - UNIT 2

6.8.1.6 Core operating limits shall be established and documented in the CORE OPERATING LIMITS REPORT (COLR) before each reload cycle or any remaining part of a reload cycle for the following:

a. SHUTDOWN MARGIN limit for MODES 1 and 2 for Specification 3/4.1.1.1,

b. SHUTDOWN MARGIN limits for MODES 3, 4, and 5 for Specification 3/4.1.1.2,

c. Moderator temperature coefficient BOL and EOL limits and the 300-ppm surveillance limit for Specification 3/4.1.1.3,

d. Shutdown Rod Insertion Limits for Specification 3/4.1.3.5,

e. Control Rod Insertion Limits for Specification 3/4.1.3.6,

f. Axial Flux Difference Limits, and target band for Specification 3/4.2.1,

g. Heat Flux Hot Channel Factor, K(Z), the Power Factor Multiplier and F_{xy}^{RTP}, for Specification 3/4.2.2,

h. Nuclear Enthalpy Rise Hot Channel Factor Limit and the Power Factor Multiplier for Specification 3/4.2.3.

The analytical methods used to determine the core operating limits shall be those previously approved by the NRC in:

CORE OPERATING LIMITS REPORT (Continued) - UNIT 2

a. WCAP-9272-P-A, "WESTINGHOUSE RELOAD SAFETY EVALUATION METHODOLOGY," July 1985 (W Proprietary).
(Methodology for Specification 3.1.1.3 - Moderator Temperature Coefficient, 3.1.3.5 - Shutdown Bank Insertion Limit, 3.1.3.6 - Control Bank Insertion Limits, 3.2.1 - Axial Flux Difference, 3.2.2 - Heat Flux Hot Channel Factor, and 3.2.3 - Nuclear Enthalpy Rise Hot Channel Factor.)

b. WCAP-8385, "POWER DISTRIBUTION CONTROL AND LOAD FOLLOWING PROCEDURES - TOPICAL REPORT," September 1974 (W Proprietary).
(Methodology for Specification 3.2.1 - Axial Flux Difference [Constant Axial Offset Control].)

c. T. M. Anderson to K. Kniel (Chief of Core Performance Branch, NRC) January 31, 1980 -- Attachment: Operation and Safety Analysis Aspects of an Improved Load Follow Package.
(Methodology for Specification 3.2.1 - Axial Flux Difference [Constant Axial Offset Control].)

d. NUREG-0800, Standard Review Plan, U. S. Nuclear Regulatory Commission, Section 4.3, Nuclear Design, July 1981. Branch Technical Position CPB 4.3-1, Westinghouse Constant Axial Offset Control (CAOC), Rev.2, July 1981.
(Methodology for Specification 3.2.1 - Axial Flux Difference [Constant Axial Offset Control].)

e. WCAP-9220-P-A, Rev. 1, "WESTINGHOUSE ECCS EVALUATION MODEL-1981 VERSION," February 1982 (W Proprietary).
(Methodology for Specification 3.2.2 - Heat Flux Hot Channel Factor.)

The core operating limits shall be determined so that all applicable limits (e.g., fuel thermal-mechanical limits, core thermal-hydraulic limits, ECCS limits, nuclear limits such as shutdown margin, and transient and accident analysis limits) of the safety analysis are met.

The CORE OPERATING LIMITS REPORT, including any mid-cycle revisions or supplements thereto, shall be provided upon issuance, for each reload cycle, to the NRC Document Control Desk with copies to the Regional Administrator and Resident Inspector.

SPECIAL REPORTS

6.8.2 Special reports shall be submitted to the Regional Administrator of the Regional Office of the NRC within the time period specified for each report.

Attachment 2a

Vogtle Electric Generating Plant Units 1 and 2

Request for Technical Specifications Changes

VANTAGE-5 Fuel Design

Technical Specifications Marked-Up Pages

Effective following the Vogtle 2 Cycle 2 Shutdown
(Effective as of Vogtle 2 Cycle 3 Startup)

INDEX

SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

BASES

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

BASES

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ADMINISTRATIVE CONTROLS

2.0 SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS*

2.1 SAFETY LIMITS

REACTOR CORE

2.1.1 The combination of THERMAL POWER (NI-0041, NI-0042, NI-0043, NI-0044), pressurizer pressure (PI-0455A, B&C, PI-0456 & PI-0456A, PI-0457 & PI-0457A, PI-0458 & PI-0458A), and the highest operating loop coolant temperature (T_{avg}) (TI-0412, TI-0422, TI-0432, TI-0442) shall not exceed the limits shown in Figure 2.1-1 ~~(Unit 1) or Figure 2.1-1a (Unit 2)~~.

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 6.6.1.

REACTOR COOLANT SYSTEM PRESSURE

2.1.2 The Reactor Coolant System pressure (PI-0408, PI-0418, PI-0428, PI-0438) 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 6.6.1.

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 6.6.1.

*Where specific instrument numbers are provided in parentheses they are for information only, and apply to each unit unless specifically noted (to assist in identifying associated instrument channels or loops) and are not intended to limit the requirements to the specific instruments associated with the number.

~~THIS PAGE APPLICABLE TO UNIT 1 ONLY~~

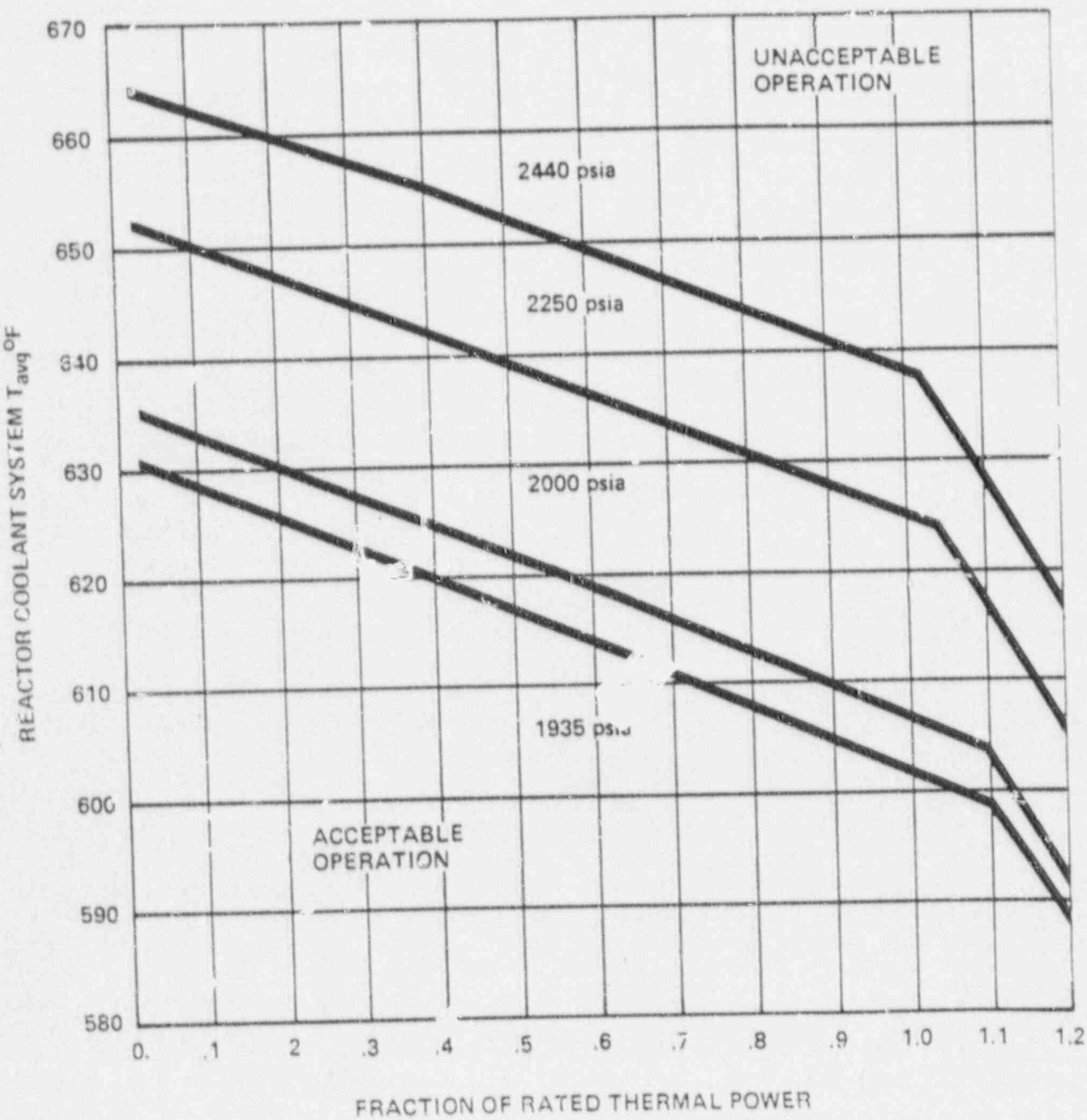

FIGURE 2.1-1

REACTOR CORE SAFETY LIMIT ~~- UNIT 1~~

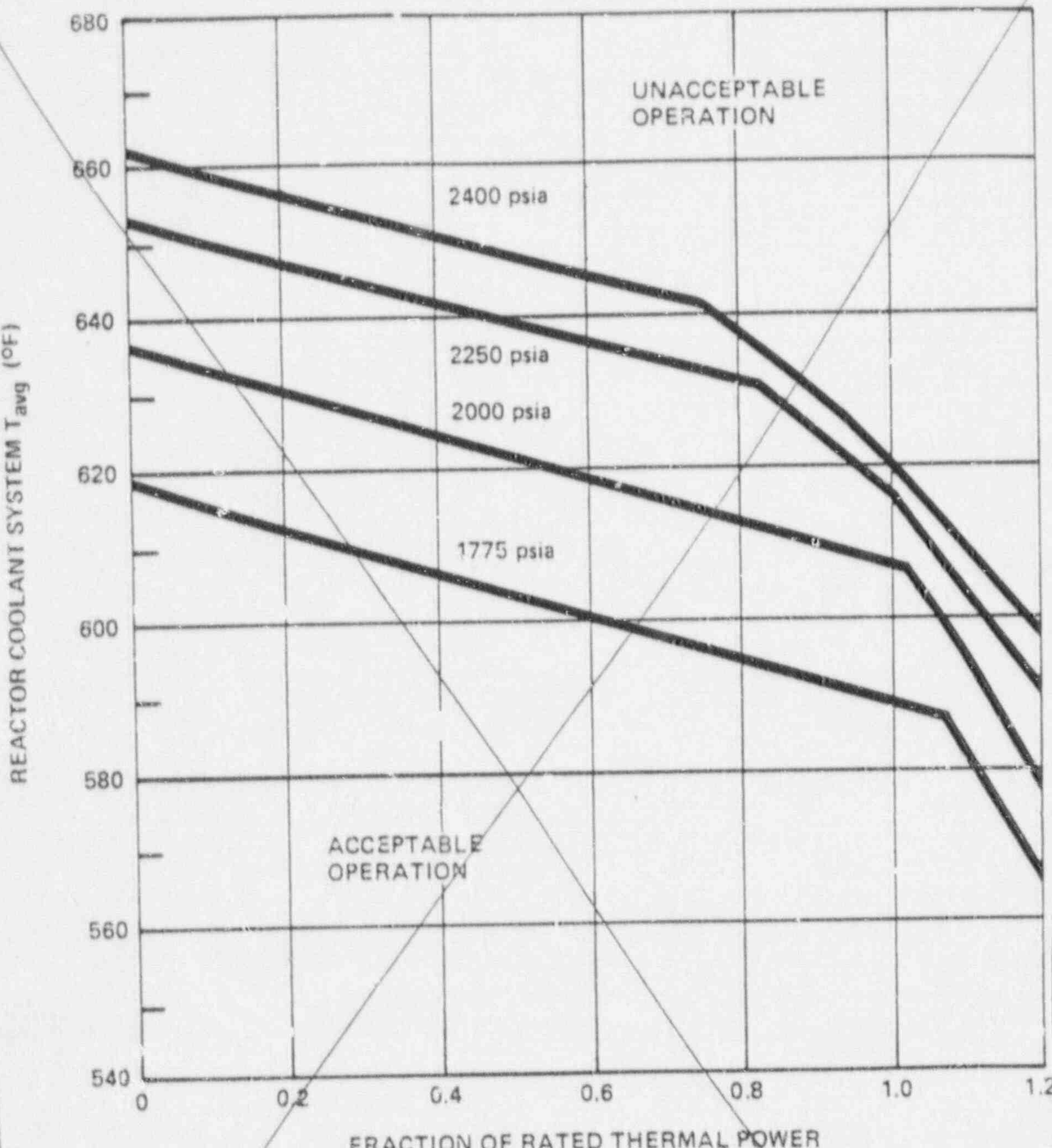

FIGURE 2.1-1a
REACTOR CORE SAFETY LIMIT - UNIT 2

2.2 LIMITING SAFETY SYSTEM SETTINGS

REACTOR TRIP SYSTEM INSTRUMENTATION SETPOINTS

2.2.1 The Reactor Trip System Instrumentation and Interlock Setpoints shall be set consistent with the Trip Setpoint values shown in Table 2.2-1 ~~(Unit 1) or Table 2.2-1a (Unit 2)~~.

APPLICABILITY: As shown for each channel in Table 3.3-1.

ACTION:

a. With a Reactor Trip System Instrumentation or Interlock Setpoint less conservative than the value shown in the Trip Setpoint column but more conservative than the value shown in the Allowable Value column of Table 2.2-1 ~~or Table 2.2-1a~~, adjust the Setpoint consistent with the Trip Setpoint value.

b. With the Reactor Trip System Instrumentation or Interlock Setpoint less conservative than the value shown in the Allowable Values column of Table 2.2-1 ~~or Table 2.2-1a~~, either:

   1. Adjust the Setpoint consistent with the Trip Setpoint value of Table 2.2-1 ~~or Table 2.2-1a~~ and determine within 12 hours that Equation 2.2-1 was satisfied for the affected channel, or

   2. Declare the channel inoperable and apply the applicable ACTION statement requirement of Specification 3.3.1 until the channel is restored to OPERABLE status with its Setpoint adjusted consistent with the Trip Setpoint value.

Equation 2.2-1 $$Z + R + S \leq TA$$

Where:

Z = The value from Column Z of Table 2.2-1 ~~or Table 2.2-1a~~ for the affected channel,

R = The "as measured" value (in percent span) of rack error for the affected channel,

S = Either the "as measured" value (in percent span) of the sensor error, or the value from Column S (Sensor Error) of Table 2.2-1 ~~or Table 2.2-1a~~ for the affected channel, and

TA = The value from Column TA (Total Allowance) of Table 2.2-1 ~~or Table 2.2-1a~~ for the affected channel.

~~THIS PAGE APPLIES TO UNIT 1 ONLY~~

TABLE 2.2-1 - ~~UNIT 1~~

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TOTAL ALLOWANCE (TA)	Z	SENSOR ERROR (S)	TRIP SETPOINT	ALLOWABLE VALUE
1. Manual Reactor Trip	N.A.	N.A.	N.A.	N.A.	N.A.
2. Power Range, Neutron Flux (NI-0041B&C, NI-0042B&C, NI-0043B&C, NI-0044B&C)					
a. High Setpoint	7.5	4.56	0	≤109% of RTP#	≤111.3% of RTP#
b. Low Setpoint	8.3	4.56	0	≤25% of RTP#	≤27.3% of RTP#
3. Power Range, Neutron Flux, High Positive Rate (NI-0041B&C, NI-0042B&C, NI-0043B&C, NI-0044B&C)	1.6	0.50	0	≤5% of RTP# with a time constant ≥2 seconds	≤6.3% of RTP# with a time constant ≥2 seconds
4. Deleted.					
5. Intermediate Range, Neutron Flux (NI-0035B, NI-0036B)	17.0	8.41	0	≤25% of RTP#	≤31.1% of RTP#
6. Source Range, Neutron Flux (NI-0031B, NI-0032B)	17.0	10.01	0	$\leq 10^5$ cps	$\leq 1.4 \times 10^5$ cps
7. Overtemperature ΔT (TDI-411C, TDI-421C, TDI-431C, TDI-441C)	10.7 ~~(Unit 1)~~	7.04 ~~(Unit 1)~~	1.96 + 1.17 ~~(Unit 1)~~	See Note 1	See Note 2
8. Overpower ΔT (TDI-411B, TDI-421B, TDI-431B, TDI-441B)	4.3 ~~(Unit 1)~~	1.54	1.96 ~~(Unit 1)~~	See Note 3	See Note 4

#RTP - RATED THERMAL POWER

~~THIS PAGE APPLICABLE TO UNIT 1 ONLY~~

TABLE 2.2-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS ~~UNIT 1~~

FUNCTIONAL UNIT	TOTAL ALLOWANCE (TA)	Z	SENSOR ERROR (S)	TRIP SETPOINT	ALLOWABLE VALUE
9. Pressurizer Pressure-Low (PI-0455A,B&C, PI-0456 & PI-0456A, PI-0457 & PI-0457A, PI-0458 & PI-0458A)	3.1	0.71	1.67	≥1960 psig**	≥1950 psig
10. Pressurizer Pressure-High (PI-0455A,B&C, PI-0456 & PI-0456A, PI-0457 & PI-0457A, PI-0458 & PI-0458A)	3.1	0.71	1.67	≤2385 psig	≤2395 psig
11. Pressurizer Water Level-High (LI-0459A, LI-0460A, LI-0461)	8.0	2.18	1.67	≤92% of instrument span	≤93.9% of instrument span
12. Reactor Coolant Flow-Low (LOOP1 FI-0414 FI-0415 FI-0416; LOOP2 FI-0424 FI-0425 FI-0426; LOOP3 FI-0434 FI-0435 FI-0436; LOOP4 FI-0444 FI-0445 FI-0446)	2.5	1.87	0.60	≥90% of loop design flow*	≥89.4% of loop design flow*
13. Steam Generator Water Level Low-Low (LOOP1 LI-0517 LI-0518 LI-0519 LI-0551; LOOP2 LI-0527 LI-0528 LI-0529 LI-0552; LOOP3 LI-0537 LI-0538 LI-0539 LI-0553; LOOP4 LI-0547 LI-0548 LI-0549 LI-0554)	18.5 (21.8)***	17.18 (19.21)***	1.67	≥18.5% (37.8)*** of narrow range instrument span	≥17.8% (35.9)*** of narrow range instrument span
14. Undervoltage - Reactor Coolant Pumps	6.0	0.58	0	≥9600 volts (70% bus voltage)	≥9481 volts (69% bus voltage)
15. Underfrequency - Reactor Coolant Pumps	3.3	0.50	0	≥57.3 Hz	≥57.1 Hz

*Loop design flow = 95,700 gpm

**Time constants utilized in the lead-lag controller for Pressurizer Pressure-Low are 10 seconds for lead and 1 second for lag. CHANNEL CALIBRATION shall ensure that these time constants are adjusted to these values.

***The value stated inside the parenthesis is for instrumentation that has the lower tap at elevation 333"; the value stated outside the parenthesis is for instrumentation that has the lower tap at elevation 438".

~~THIS PAGE APPLICABLE TO UNIT 1 ONLY~~

TABLE 2.2-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS ~~- UNIT 1~~

FUNCTIONAL UNIT	TOTAL ALLOWANCE (TA)	Z	SENSOR ERROR (S)	TRIP SETPOINT	ALLOWABLE VALUE
16. Turbine Trip					
a. Low Fluid Oil Pressure (PT-6161, PT-6162, PT-6163)	N.A.	N.A.	N.A.	$\geq$580 psig	$\geq$500 psig
b. Turbine Stop Valve Closure	N.A.	N.A.	N.A.	$\geq$96.7% open	$\geq$96.7% open
17. Safety Injection Input from ESF	N.A.	N.A.	N.A.	N.A.	N.A.
18. Reactor Trip System Interlocks					
a. Intermediate Range Neutron Flux, P-6 (NI-0035B, NI-0036B)	N.A.	N.A.	N.A.	$\geq 1 \times 10^{-10}$ amp	$\geq 6 \times 10^{-11}$ amp
b. Low Power Reactor Trips Block, P-7					
1) P-10 input (NI-0041B&C, NI-0042B&C, NI-0043B&C, NI-0044B&C)	N.A.	N.A.	N.A.	$\leq$10% of RTP#	$\leq$12.3% of RTP#
2) P-13 input (PT-0505, PT-0506)	N.A.	N.A.	N.A.	$\leq$10% RTP# Turbine Impulse Pressure Equivalent	$\leq$12.3% RTP# Turbine Impulse Pressure Equivalent
c. Power Range Neutron Flux, P-8 (NI-0041B&C, NI-0042B&C, NI-0043B&C, NI-0044B&C)	N.A.	N.A.	N.A.	$\leq$48% of RTP#	$\leq$50.3% of RTP#

#RTP = RATED THERMAL POWER

~~THIS PAGE APPLICABLE TO UNIT 1 ONLY~~

TABLE 2.2-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS ~~- UNIT 1~~

FUNCTIONAL UNIT	TOTAL ALLOWANCE (TA)	Z	SENSOR ERROR (S)	TRIP SETPOINT	ALLOWABLE VALUE
d. Power Range Nuetron Flux, P-9 (NI-0041B&C, NI-0042B&C, NI-0043B&C, NI-0044B&C)	N.A.	N.A.	N.A.	≤50% of RTP#	≤52.3% of RTP#
e. Power Range Neutron Flux, P-10 (NI-0041B&C, NI-0042B&C, NI-0043B&C, NI-0044B&C)	N.A.	N.A.	N.A.	≥10% of RTP#	≥7.7% of RTP#
f. Turbine Impulse Chamber Pressure, P-13 (PI-0505, PI-0506)	N.A.	N.A.	N.A.	≤10% RTP# Turbine Impulse Pressure Equivalent	≤12.3% RTP# Turbine Impulse Pressure Equivalent
19. Reactor Trip Breakers	N.A.	N.A.	N.A.	N.A.	N.A.
20. Automatic Trip and Interlock Logic	N.A.	N.A.	N.A.	N.A.	N.A.

#RTP = RATED THERMAL POWER

~~THIS PAGE APPLICABLE TO UNIT 1 ONLY~~

TABLE 2.2-1 (Continued)

TABLE NOTATIONS ~~- UNIT 1~~

NOTE 1: OVERTEMPERATURE ΔT

$$\Delta T \frac{(1 + \tau_1 S)}{(1 + \tau_2 S)} \left(\frac{1}{1 + \tau_3 S}\right) \leq \Delta T_0 \{K_1 - K_2 \frac{(1 + \tau_4 S)}{(1 + \tau_5 S)} [T \left(\frac{1}{1 + \tau_6 S}\right) - T'] + K_3(P - P') - f_1(\Delta I)\}$$

Where: ΔT = Measured ΔT ~~(Unit 1)~~;

$\frac{1 + \tau_1 S}{1 + \tau_2 S}$ = Lead-lag compensator on measured ΔT;

τ_1, τ_2 = Time constants utilized in lead-lag compensator for ΔT, $\tau_1 \geq 8$ s, $\tau_2 \leq 3$ s;

$\frac{1}{1 + \tau_3 S}$ = Lag compensator on measured ΔT;

τ_3 = Time constants utilized in the lag compensator for ΔT, $\tau_3 = 0$ s;

ΔT_0 = Indicated ΔT at RATED THERMAL POWER;

K_1 ≤ 1.12 ~~(Unit 1)~~;

K_2 = 0.0224/°F ~~(Unit 1)~~;

$\frac{1 + \tau_4 S}{1 + \tau_5 S}$ = The function generated by the lead-lag compensator for T_{avg} dynamic compensation;

τ_4, τ_5 = Time constants utilized in the lead-lag compensator for T_{avg}, $\tau_4 \geq 28$ s, $\tau_5 \leq 4$ s;

T = Average temperature, °F;

$\frac{1}{1 + \tau_6 S}$ = Lag compensator on measured T_{avg};

τ_6 = Time constant utilized in the measured T_{avg} lag compensator, $\tau_6 = 0$ s;

~~THIS PAGE APPLICABLE TO UNIT 1 ONLY~~

TABLE 2.2-1 (Continued)

TABLE NOTATIONS (Continued) ~~- UNIT 1~~

NOTE 1: (Continued)

T' ≤ 588.4°F ~~(Unit 1)~~ (Nominal T_{avg} operating temperature;)

K_3 = 0.00115/psig ~~(Unit 1)~~;

P = Pressurizer pressure, psig;

P' = 2235 psig (Nominal RCS operating pressure);

S = Laplace transform variable, s^{-1};

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

(1) For $q_t - q_b$ between -32.0% ~~(Unit 1)~~ and + 11.0% ~~(Unit 1)~~, $f_1(\Delta I) = 0$, 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;

(2) For each percent that the magnitude of $q_t - q_b$ exceeds - 32.0% ~~(Unit 1)~~, the ΔT Trip Setpoint shall be automatically reduced by 3.25% ~~(Unit 1)~~ of its value at RATED THERMAL POWER; and

(3) For each percent that the magnitude of $q_t - q_b$ exceeds + 11.0% ~~(Unit 1)~~, the ΔT Trip Setpoint shall be automatically reduced by 1.97% ~~(Unit 1)~~ of its value at RATED THERMAL POWER.

NOTE 2: The channel's maximum Trip Setpoint shall not exceed its computed Trip Setpoint by more than 3.1% ~~(Unit 1)~~ of ΔT span.

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TABLE 2.2-1 (Continued)

TABLE NOTATIONS (Continued) ~~UNIT 1~~

NOTE 3: OVERPOWER ΔT

$$\Delta T \frac{(1 + \tau_1 S)}{(1 + \tau_2 S)} \left(\frac{1}{1 + \tau_3 S} \right) \leq \Delta T_0 \left\{ K_4 - K_5 \left(\frac{\tau_7 S}{1 + \tau_7 S} \right) \left(\frac{1}{1 + \tau_6 S} \right) T - K_6 \left[T \left(\frac{1}{1 + \tau_6 S} \right) - T'' \right] - f_2(\Delta I) \right\}$$

Where: ΔT = Measured ΔT ~~(Unit 1)~~;

$\frac{1 + \tau_1 S}{1 + \tau_2 S}$ = Lead-lag compensator on measured ΔT;

τ_1, τ_2 = Time constants utilized in lead-leg compensator for ΔT, $\tau_1 \geq 8$ S, $\tau_2 \leq 3$ s;

$\frac{1}{1 + \tau_3 S}$ = Lag compensator on measured ΔT;

τ_3 = Time constants utilized in the lag compensator for ΔT, τ_3=0s;

ΔT_0 = Indicated ΔT at RATED THERMAL POWER;

K_4 $\leq$ 1.08 ~~(Unit 1)~~.

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

$\frac{\tau_7 S}{1 + \tau_7 S}$ = The function generated by the rate-lag compensator for T_{avg} dynamic compensation,

τ_7 = Time constants utilized in the rate-lag compensator for T_{avg}, $\tau_7 \geq 10$ s,

$\frac{1}{1 + \tau_6 S}$ = Lag compensator on measured T_{avg};

~~THIS PAGE APPLICABLE TO UNIT 1 ONLY~~

TABLE 2.2-1 (Continued)

TABLE NOTATIONS (Continued) ~~UNIT 1~~

NOTE 3: (Continued)

τ_6	=	Time constant utilized in the measured T_{avg} lag compensator, τ_6 = 0 s;
K_6	$\geq$	0.0020/°F ~~(Unit 1)~~ for $T > T''$ and $K_6 = 0$ for $T \leq T''$.
T	=	Average Temperature, °F;
T''	=	Indicated T_{avg} at RATED THERMAL POWER (Calibration temperature for ΔI instrumentation, $\leq$ 588.4°F ~~(Unit 1)~~),
S	=	Laplace transform variable, s^{-1}; and
$f_2(\Delta I)$	=	0 for all ΔI.

NOTE 4: The channel's maximum Trip Setpoint shall not exceed its computed Trip Setpoint by more than 1.9% ~~(Unit 1)~~ of ΔT span.

TABLE 2.2-1a - UNIT 2

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TOTAL ALLOWANCE (TA)	Z	SENSOR ERROR (S)	TRIP SETPOINT	ALLOWABLE VALUE
1. Manual Reactor Trip	N.A.	N.A.	N.A.	N.A.	N.A.
2. Power Range, Neutron Flux (NI-0041B&C, NI-0042B&C, NI-0043B&C, NI-0044B&C)					
a. High Setpoint	7.5	4.56	0	≤109% of RTP#	≤111.3% of RTP#
b. Low Setpoint	8.3	4.56	0	≤25% of RTP#	≤27.3% of RTP#
3. Power Range, Neutron Flux, High Positive Rate (NI-0041B&C, NI-0042B&C, NI-0043B&C, NI-0044B&C)	1.6	0.50	0	≤5% of RTP# with a time constant ≥2 seconds	≤6.3% of RTP# with a time constant ≥2 seconds
4. Deleted.					
5. Intermediate Range, Neutron Flux (NI-0035B, NI-0036B)	17.0	8.41	0	≤25% of RTP#	≤31.1% of RTP#
6. Source Range, Neutron Flux (NI-0031B, NI-0032B)	17.0	10.01	0	≤10^5 cps	≤1.4×10^5 cps
7. Overtemperature ΔT (TDI-411C, TDI-421C, TDI-431C, TDI-441C)	6.6 (Unit 2)	3.37 (Unit 2)	1.95 + 0.50 (Unit 2)	See Note 1	See Note 2
8. Overpower ΔT (TDI-411B, TDI-421B, TDI-431B, TDI-441B)	4.9 (Unit 2)	1.54	1.95 (Unit 2)	See Note 3	See Note 4

#RTP = RATED THERMAL POWER

TABLE 2.2-1a (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS - UNIT 2

FUNCTIONAL UNIT	TOTAL ALLOWANCE (TA)	Z	SENSOR ERROR (S)	TRIP SETPOINT	ALLOWABLE VALUE
9. Pressurizer Pressure-Low (PI-0455A,B&C, PI-0456 & PI-0456A, PI-0457 & PI-0457A, PI-0458 & PI-0458A)	3.1	0.71	1.67	≥1960 psig**	≥1950 psig
10. Pressurizer Pressure-High (PI-0455A,B&C, PI-0456 & PI-0456A, PI-0457 & PI-0457A, PI-0458 & PI-0458A)	3.1	0.71	1.67	≤2385 psig	≤2395 psig
11. Pressurizer Water Level-High (LI-0459A, LI-0460A, LI-0461)	8.0	2.18	1.67	≤92% of instrument span	≤93.9% of instrument span
12. Reactor Coolant Flow-Low (LOOP1 LOOP2 LOOP3 LOOP4 FI-0414 FI-0424 FI-0434 FI-0444 FI-0415 FI-0425 FI-0435 FI-0445 FI-0416 FI-0426 FI-0436 FI-0446)	2.5	1.87	0.60	≥90% of loop design flow*	≥89.4% of loop design flow*
13. Steam Generator Water Level Low-Low (LOOP1 LOOP2 LOOP3 LOOP4 LI-0517 LI-0527 LI-0537 LI-0547 LI-0518 LI-0528 LI-0538 LI-0548 LI-0519 LI-0529 LI-0539 LI-0549 LI-0551 LI-0552 LI-0553 LI-0554)	18.5 (21.8)***	17.18 (18.21)***	1.67	≥18.5% (37.8)*** of narrow range instrument span	≥17.8% (35.9)*** of narrow range instrument span
14. Undervoltage - Reactor Coolant Pumps	6.0	0.58	0	≥9600 volts (70% bus voltage)	≥9481 volts (69% bus voltage)
15. Underfrequency - Reactor Coolant Pumps	3.3	0.50	0	≥57.3 Hz	≥57.1 Hz

*Loop design flow = 95,700 gpm

**Time constants utilized in the lead-lag controller for Pressurizer Pressure-Low are 10 seconds for lead and 1 second for lag. CHANNEL CALIBRATION shall ensure that these time constants are adjusted to these values.

***The value stated inside the parenthesis is for instrumentation that has the lower tap at elevation 333"; the value stated outside the parenthesis is for instrumentation that has the lower tap at elevation 438".

VOGTLE UNITS - 1 & 2

2-14

TABLE 2.2-1a (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS - UNIT 2

FUNCTIONAL UNIT	TOTAL ALLOWANCE (TA)	Z	SENSOR ERROR (S)	TRIP SETPOINT	ALLOWABLE VALUE
16. Turbine Trip					
a. Low Fluid Oil Pressure (PT-6161, PT-6162, PT-6163)	N.A.	N.A.	N.A.	≥580 psig	≥500 psig
b. Turbine Stop Valve Closure	N.A.	N.A.	N.A.	≥96.7% open	≥96.7% open
17. Safety Injection Input from ESF	N.A.	N.A.	N.A.	N.A.	N.A.
18. Reactor Trip System Interlocks					
a. Intermediate Range Neutron Flux, P-6 (NI-0035B, NI-0036B)	N.A.	N.A.	N.A.	$\geq 1 \times 10^{-10}$ amp	$\geq 6 \times 10^{-11}$ amp
b. Low Power Reactor Trips Block, P-7					
1) P-10 input (NI-0041B&C, NI-0042B&C, NI-0043B&C, NI-0044B&C)	N.A.	N.A	N.A.	≤10% of RTP#	≤12.3% of RTP#
2) P-13 input (PI-0505, PI-0506)	N.A.	N.A.	N.A.	<10% RTP# Turbine Impulse Pressure Equivalent	<12.3% RTP# Turbine Impulse Pressure Equivalent
c. Power Range Neutron Flux, P-8 (NI-0041B&C, NI-0042B&C, NI-0043B&C, NI-0044B&C)	N.A.	N.A.	N.A.	<48% of RTP#	≤50.3% of RTP#

#RTP - RATED THERMAL POWER

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TABLE 2.2-1a (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS - UNIT 2

FUNCTIONAL UNIT	TOTAL ALLOWANCE (TA)	Z	SENSOR ERROR (S)	TRIP SETPOINT	ALLOWABLE VALUE
d. Power Range Nuetron Flux, P-9 (NI-0041B&C, NI-0042B&C, NI-0043B&C, NI-0044B&C)	N.A.	N.A.	N.A.	$\leq$50% of RTP#	$\leq$52.3% of RTP#
e. Power Range Neutron Flux, P-10 (NI-0041B&C, NI-0042B&C, NI-0043B&C, NI-0044B&C)	N.A.	N.A.	N.A.	$\geq$10% of RTP#	$\geq$7.7% of RTP#
f. Turbine Impulse Chamber Pressure, P-13 (PI-0505, PI-0506)	N.A.	N.A.	N.A.	<10% RTP# Turbine Impulse Pressure Equivalent	<12.3% RTP# Turbine Impulse Pressure Equivalent
19. Reactor Trip Breakers	N.A.	N.A.	N.A.	N.A.	N.A.
20. Automatic Trip and Interlock Logic	N.A.	N.A.	N.A.	N.A.	N.A.

#RTP = RATED THERMAL POWER

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TABLE 2.2-1a (Continued)

TABLE NOTATIONS - UNIT 2

NOTE 1: OVERTEMPERATURE ΔT

$$\Delta T \frac{(1 + \tau_1 S)}{(1 + \tau_2 S)} \frac{1}{1 + \tau_3 S} \leq \Delta T_0 \{K_1 - K_2 \frac{(1 + \tau_4 S)}{(1 + \tau_5 S)} [T \frac{1}{1 + \tau_6 S} - T'] + K_3 (P - P') - f_1(\Delta I)\}$$

Where: ΔT = Measured ΔT by RTD Manifold Instrumentation ~~(Unit 2)~~;

$\frac{1 + \tau_1 S}{1 + \tau_2 S}$ = Lead-lag compensator on measured ΔT;

τ_1, τ_2 = Time constants utilized in lead-lag compensator for ΔT, $\tau_1 \geq 8$ s, $\tau_2 \leq 3$ s;

$\frac{1}{1 + \tau_3 S}$ = Lag compensator on measured ΔT;

τ_3 = Time constants utilized in the lag compensator for ΔT, $\tau_3 = 0$ s;

ΔT_0 = Indicated ΔT at RATED THERMAL POWER;

K_1 $\leq$ 1.10 (Unit 2);

K_2 = 0.012/°F (Unit 2);

$\frac{1 + \tau_4 S}{1 + \tau_5 S}$ = The function generated by the lead-lag compensator for T_{avg} dynamic compensation;

τ_4, τ_5 = Time constants utilized in the lead-lag compensator for T_{avg}, $\tau_4 \geq 28$ s, $\tau_5 \leq 4$ s;

T = Average temperature, °F;

$\frac{1}{1 + \tau_6 S}$ = Lag compensator on measured T_{avg};

τ_6 = Time constant utilized in the measured T_{avg} lag compensator, $\tau_6 = 0$ s;

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TABLE 2.2-1a (Continued)

TABLE NOTATIONS (Continued) - UNIT 2

NOTE 1: (Continued)

T' _ 588.5 F (Unit 2) (Nominal T_{avg} operating temperature);

K_3 = 0.00056/psig (Unit 2);

P - Pressurizer pressure, psig;

P' = 2235 psig (Nominal RCS operating pressure);

S = Laplace transform variable, s^{-1};

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

(1) For $q_t - q_b$ between -33.5% (Unit 2) and + 6.5% (Unit 2), $f_1(\Delta I) = 0$, 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;

(2) For each percent that the magnitude of $q_t - q_b$ exceeds - 33.5% (Unit 2), the ΔT Trip Setpoint shall be automatically reduced by 1.27% (Unit 2) of its value at RATED THERMAL POWER; and

(3) For each percent that the magnitude of $q_t - q_b$ exceeds + 6.5% (Unit 2), the ΔT Trip Setpoint shall be automatically reduced by 0.83% (Unit 2) of its value at RATED THERMAL POWER.

NOTE 2: The channel's maximum Trip Setpoint shall not exceed its com... ..d Trip Setpoint by more than 2.5% (Unit 2).

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TABLE 2.2-1a (Continued)

TABLE NOTATIONS (Continued) - UNIT 2

NOTE 3: OVERPOWER ΔT

$$\Delta T \frac{(1 + \tau_1 S)}{(1 + \tau_2 S)} \left(\frac{1}{1 + \tau_3 S}\right) \leq \Delta T_0 \{K_4 - K_5 \left(\frac{\tau_7 S}{1 + \tau_7 S}\right) \left(\frac{1}{1 + \tau_6 S}\right) T - K_6 [T \left(\frac{1}{1 + \tau_6 S}\right) - T''] - f_2(\Delta I)\}$$

Where: ΔT = Measured ΔT by RTD manifold instrumentation (Unit 2);

$\frac{1 + \tau_1 S}{1 + \tau_2 S}$ = Lead-lag compensator on measured ΔT;

τ_1, τ_2 = Time constants utilized in lead-leg compensator for ΔT, $\tau_1 \geq 8$ S, $\tau_2 \leq 3$ s;

$\frac{1}{1 + \tau_3 S}$ = Lag compensator on measured ΔT;

τ_3 = Time constants utilized in the lag compensator for ΔT, τ_3=0s;

ΔT_0 = Indicated ΔT at RATED THERMAL POWER;

K_4 ≤ 1.089 (Unit 2),

K_5 ≥ 0.02/°F for increasing average temperature and ≥ 0 for decreasing average temperature,

$\frac{\tau_7 S}{1 + \tau_7 S}$ = The function generated by the rate-lag compensator for T_{avg} dynamic compensation,

τ_7 = Time constants utilized in the rate-lag compensator for T_{avg}, $\tau_7 \geq 10$ s,

$\frac{1}{1 + \tau_6 S}$ = Lag compensator on measured T_{avg};

TABLE 2.2-1a (Continued)

TABLE NOTATIONS (Continued) - UNIT 2

NOTE 3: (Continued)

τ_6 = Time constant utilized in the measured T_{avg} lag compensator, $\tau_6 = 0$ s;

K_6 $\geq$ 0.0013/°F (Unit 2) for $T > T"$ and $K_6 = 0$ for $T \leq T"$.

T = Average Temperature, °F;

T" = Indicated T_{avg} at RATED THERMAL POWER (Calibration temperature for ΔT instrumentation, $\leq$ 588.5°F (Unit 2)),

S = Laplace transform variable, s^{-1}; and

$f_2(\Delta I)$ = 0 for all ΔI.

NOTE 4: The channel's maximum Trip Setpoint shall not exceed its computed Trip Setpoint by more than 2.4% (Unit 2) of ΔT span.

2.1 SAFETY LIMITS

BASES

2.1.1 REACTOR CORE ~~- UNIT 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 correlations which have been developed to predict the DNB flux and the location of DNB for axially uniform and nonuniform heat flux distributions. The local DNB heat flux ratio (DNBR) is defined as the ratio of the heat flux that would cause DNB at a particular core location to the local heat flux and is indicative of the margin to DNB.

The DNB thermal design criterion is that the probability that DNB will not occur on the most limiting rod is at least 95% (at a 95% confidence level) for any Condition I or II event.

In meeting the DNB design criterion, uncertainties in plant operating parameters, nuclear and thermal parameters, fuel fabrication parameters, and computer codes must be considered. As described in the FSAR, the effects of these uncertainties have been statistically combined with the correlation uncertainty. Design limit DNBR values have been determined that satisfy the DNB design criterion.

Additional DNBR margin is maintained by performing the safety analyses to a higher DNBR limit. This margin between the design and safety analysis limit DNBR values is used to offset known DNBR penalties (e.g., rod bow and transition core) and to provide DNBR margin for operating and design flexibility.

The curves of Figure 2.1-1 show reactor core safety limits for a range of THERMAL POWER, REACTOR COOLANT SYSTEM pressure, and average temperature which satisfy the following criteria:

A. The average enthalpy at the vessel exit is less than the enthalpy of saturated liquid (far left line segment in each curve).

B. The minimum DNBR satisfies the DNB design criterion (all the other line segments in each curve). The VANTAGE 5 fuel is analyzed using the WRB-2 correlation with design limit DNBR values of 1.24 and 1.23 for the typical cell and thimble cells, respectively. The LOPAR fuel is analyzed using the WRB-1 correlation with design limit DNBR values of 1.23 and 1.22 for the typical and thimble cells, respectively.

C. The hot channel exit quality is not greater than the upper limit of the quality range (including the effect of uncertainties) of the DNB correlations. This is not a limiting criterion for this plant.

2.1 SAFETY LIMITS

BASES

2.1.1 REACTOR CORE - UNIT 2

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 (R Grid) correlation. The W-3 (R Grid) DNB correlation has been developed to predict the DNB flux and the location of DNB for axially uniform and nonuniform heat flux distributions. The [illegible] heat flux ratio (DNBR) is defined as the ratio of the heat flux t[illegible] cause DNB at a particular core location to the local heat flux a[illegible] 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% probability at a 95% confidence level that DNB will not occur and is chosen as an appropriate margin to DNB for all operating conditions.

The curves of Figure 2.1-1a show reactor core safety limits which are determined for a range of reactor operating conditions. The core limits represent the loci of points of THERMAL POWER, REACTOR COOLANT SYSTEM pressure and average temperature which satisfy the following criteria:

A. The average enthalpy at the vessel exit is equal to the enthalpy of saturated liquid (far left line segment in each curve).

B. The minimum DNBR is not less than the design limit (all the other line segments in each curve).

C. The hot channel exit quality is not greater than the upper limit of the quality range of the W-3 (R-Grid) correlation which is 15% (middle line segment on Reactor Coolant System pressure curves, 2400 psia and 2250 psia; this is not a limiting criterion for this plant).

BORATED WATER SOURCE - SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.1.2.5 As a minimum, one of the following borated water sources shall be OPERABLE:

a. A Boric Acid Storage Tank with:

1) A minimum contained borated water volume of 9504 gallons (19% of instrument span) (LI-102A, LI-104A),

2) A boron concentration between 7000 ppm and 7700 ppm, and

3) A minimum solution temperature of 65°F (TI-0103).

b. The refueling water storage tank (RWST) with:

1) A minimum contained borated water volume of 99404 gallons (9% of instrument span) (LI-0990A&B, LI-0991A&B, LI-0992A, LI-0993A),

2) A boron concentration between 2400 ppm and 2600 ppm, and

3) A minimum solution temperature of 44°F ~~(Unit 1) or 54°F (Unit 2)~~ (TI-10982).

APPLICABILITY: MODES 5 and 6.

ACTION:

With no borated water source OPERABLE, suspend all operations involving CORE ALTERATIONS or positive reactivity changes.

SURVEILLANCE REQUIREMENTS

4.1.2.5 The above required borated water source shall be demonstrated OPERABLE:

a. At least once per 7 days by:

1) Verifying the boron concentration of the water,

2) Verifying the contained borated water volume, and

3) When the boric acid storage tank is the source of borated water and the ambient temperature of the boric acid storage tank room (TISL-20902, TISL-20903) is ≤72°F, verify the boric acid storage tank solution temperature is ≥65°F.

b. At least once per 24 hours by verifying the RWST temperature (TI-10982) when it is the source of borated water and the outside air temperature is less than 40°F ~~(Unit 1) or 50°F (Unit 2)~~.

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 Tank with:

1) A minimum contained borated water volume of 36674 gallons (81% of instrument span) (LI-102A, LI-104A),

2) A boron concentration between 7000 ppm and 7700 ppm, and

3) A minimum solution temperature of 65°F (TI-0103).

b. The refueling water storage tank (RWST) with:

1) A minimum contained borated water volume of 631478 gallons (86% of instrument span) (LI-0990A&B, LI-0991A&B, LI-0992A, LI-0993A),

2) A boron concentration between 2400 ppm and 2600 ppm,

3) A minimum solution temperature of 44°F ~~(Unit 1) or 54°F (Unit 2)~~,

4) A maximum solution temperature of 116°F (TI-10982), and

5) RWST Sludge Mixing Pump Isolation Valves capable of closing on RWST low-level.

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

ACTION:

a. With the Boric Acid Storage Tank inoperable and being used as one of the above required borated water sources, restore the tank to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and borated to a SHUTDOWN MARGIN as required by Figure 3.1-2 at 200°F; restore the Boric Acid Storage Tank to OPERABLE status within the next 7 days or be in COLD SHUTDOWN within the next 30 hours.

b. With the RWST inoperable, except for the Sludge Mixing Pump Isolation Valves, restore the tank to OPERABLE status within 1 hour or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

LIMITING CONDITION FOR OPERATION (Continued)

ACTION (Continued)

c. With a Sludge Mixing Pump Isolation Valve(s) inoperable, restore the valve(s) to OPERABLE status within 24 hours or isolate the sludge mixing system by either closing the manual isolation valves or deenergizing the OPERABLE solenoid pilot valve within 6 hours and maintain closed.

SURVEILLANCE REQUIREMENTS

4.1.2.6 Each borated water source shall be demonstrated OPERABLE:

a. At least once per 7 days by:

1) Verifying the boron concentration in the water,

2) Verifying the contained borated water volume of the water source, and

3) When the boric acid storage tank is the source of borated water and the ambient temperature of the boric acid storage tank room (TISL-20902, TISL-20903) is ≤ 72°F, verify the boric acid storage tank solution temperature is ≥ 65°F.

b. At least once per 24 hours by verifying the RWST temperature (TI-10982) when the outside air temperature is less than 40°F ~~(Unit 1) or 50°F (Unit 2)~~

c. At least once per 18 months by verifying that the Sludge Mixing Pump Isolation Valves automatically close upon RWST low-level test signal.

ROD DROP TIME

LIMITING CONDITION FOR OPERATION

3.1.3.4 The individual shutdown and control rod drop time from the physical fully withdrawn position shall be less than or equal to 2.7 ~~(Unit 1) or 2.2 (Unit 2)~~ seconds from beginning of decay of stationary gripper coil voltage to dashpot entry with:

a. T_{avg} (TI-0412, TI-0422, TI-0432, TI-0442) greater than or equal to $551^\circ F$, and

b. All reactor coolant pumps operating.

APPLICABILITY: MODES 1 and 2.

ACTION:

With the drop time of any 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 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.

3/4.2 POWER DISTRIBUTION LIMITS

3/4.2.1 AXIAL FLUX DIFFERENCE ~~- UNIT 1~~

LIMITING CONDITION FOR OPERATION

3.2.1 The indicated (NI-0041B, NI-0042B, NI-0043B, NI-0044B) AXIAL FLUX DIFFERENCE (AFD) shall be maintained within the limits specified in the CORE OPERATING LIMITS REPORT (COLR).

APPLICABILITY: MODE 1 ABOVE 50 PERCENT RATED THERMAL POWER*.

ACTION:

a. With the indicated AXIAL FLUX DIFFERENCE outside of the limits specified in the COLR,

   1. Either restore the indicated AFD to within the 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 percent of RATED THERMAL POWER within the next 4 hours.

b. THERMAL POWER shall not be increased above 50% of RATED THERMAL POWER unless the indicated AFD is within the limits specified in the COLR.

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:

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

   2) At least once per hour until the AFD Monitor Alarm is updated after restoration to OPERABLE status.

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.

c. The provisions of Specification 4.0.4 are not applicable.

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 its limits.

*See Special Test Exceptions Specification 3.10.2.

~~3/4.2 POWER DISTRIBUTION LIMITS~~

~~3/4.2.1 AXIAL FLUX DIFFERENCE - UNIT 2~~

~~LIMITING CONDITION FOR OPERATION~~

~~3.2.1 The indicated (NI-0041B, NI-0042B, NI-0043B, NI-0044B) AXIAL FLUX DIFFERENCE (AFD) shall be maintained within the target band (flux difference units) about the target flux difference. The target band is specified in the CORE OPERATING LIMITS REPORT (COLR).~~

~~The indicated AFD may deviate outside the required target band at greater than or equal to 50% but less than 90% of RATED THERMAL POWER provided the indicated AFD is within the Acceptable Operation Limits specified in the COLR and the cumulative penalty deviation time does not exceed 1 hour during the previous 24 hours.~~

~~The indicated AFD may deviate outside the required target band at greater than 15% but less than 50% of RATED THERMAL POWER provided the cumulative penalty deviation time does not exceed 1 hour during the previous 24 hours.~~

~~APPLICABILITY: MODE 1, above 15% of RATED THERMAL POWER.* #~~

~~ACTION:~~

a. ~~With the indicated AFD outside of the required target band and with THERMAL POWER greater than or equal to 90% of RATED THERMAL POWER, within 15 minutes either:~~

   1. ~~Restore the indicated AFD to within the target band limits, or~~

   2. ~~Reduce THERMAL POWER to less than 90% of RATED THERMAL POWER.~~

b. ~~With the indicated AFD outside of the required target band for more than 1 hour of cumulative penalty deviation time during the previous 24 hours or outside the Acceptable Operation Limits specified in the COLR and with THERMAL POWER less than 90% but equal to or greater than 50% of RATED THERMAL POWER, reduce:~~

   1. ~~THERMAL POWER to less than 50% of RATED THERMAL POWER within 30 minutes, and~~

   2. ~~The Power Range Neutron Flux* - High Setpoints to less than or equal to 55% of RATED THERMAL POWER within the next 4 hours.~~

~~*See Special Test Exceptions Specification 3.10.2.~~

~~#Surveillance testing of the Power Range Neutron Flux Channel may be performed (below 90% of RATED THERMAL POWER) pursuant to Specification 4.3.1.1 provided the indicated AFD is maintained within the Acceptable Operation Limits specified in the COLR. A total of 16 hours operation may be accumulated with the AFD outside of the above required target band during testing without penalty deviation.~~

POWER DISTRIBUTION LIMITS

LIMITING CONDITION FOR OPERATION - UNIT 2

ACTION (Continued)

c. With the indicated AFD outside of the required target band for more than 1 hour of cumulative penalty deviation time during the previous 24 hours and with THERMAL POWER less than 50% but greater than 15% of RATED THERMAL POWER, the THERMAL POWER shall not be increased equal to or greater than 50% of RATED THERMAL POWER until the indicated AFD is within the required target band and the cumulative penalty deviation has been reduced to less than 1 hour in the previous 24 hours.

SURVEILLANCE REQUIREMENTS

4.2.1.1 The indicated AFD 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 until the AFD Monitor Alarm is updated after restoration to OPERABLE status.

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 target band when two or more OPERABLE excore channels are indicating the AFD to be outside the target band. Penalty deviation outside of the required target band shall be accumulated on a time basis of:

a. One minute penalty deviation for each 1 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 1 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 Specification 4.2.1.3 above or by linear interpolation between the most recently measured value and 0% at the end of the cycle life. The provisions of Specification 4.0.4 are not applicable.

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SURVEILLANCE REQUIREMENTS ~~- UNIT 1~~

4.2.2.1 The provisions of Specifications 4.0.4 are not applicable.

4.2.2.2 $F_Q(Z)$ shall be evaluated to determine if it 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. Determining the computed heat flux hot channel factor, $F_Q^C(Z)$, as follows:

Increase the measured $F_Q(Z)$ obtained from the power distribution map by 3% to account for manufacturing tolerances and further increase the value by 5% to account for measurement uncertainties.

c. Verifying that $F_Q^C(Z)$, obtained in Specification 4.2.2.2b above, satisfies the relationship in Specification 3.2.2.

d. Satisfying the following relationship:

$$F_Q^C(Z) \leq \frac{F_Q^{RTP} \times K(Z)}{P \times W(Z)} \text{ for } P > 0.5$$

$$F_Q^C(Z) \leq \frac{F_Q^{RTP} \times K(Z)}{0.5 \times W(Z)} \text{ for } P \leq 0.5$$

Where $F_Q^C(Z)$ is obtained in Specification 4.2.2.2b above, F_Q^{RTP} is the F_Q limit, K(Z) is the normalized $F_Q(Z)$ as a function of core height, P is the fraction of RATED THERMAL POWER, and W(Z) is the cycle dependent function that accounts for power distribution transients encountered during normal operation. F_Q^{RTP}, K(Z), and W(Z) are specified in the CORE OPERATING LIMITS REPORT as per Specification 6.8.1.6.

e. Measuring $F_Q(Z)$ according to the following schedule:

1. Upon achieving equilibrium conditions after exceeding by 20% or more of RATED THERMAL POWER, the THERMAL POWER at which FQ(Z) was last determined*, or

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

---

*During power escalation after each fuel loading, power level may be increased until equilibrium conditions at any power level greater than or equal to 50% of RATED THERMAL POWER have been achieved and a power distribution map obtained.

f. With measurements indicating

$$\underset{\text{over } Z}{\text{maximum}} \left(\frac{F_Q^C(Z)}{K(Z)} \right)$$

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

1) Increase $F_Q^C(Z)$ by 2% and verify that this value satisfies the relationship in Specification 4.2.2.2d, or

2) $F_Q^C(Z)$ shall be measured at least once per 7 Effective Full Power Days until two successive maps indicate that

$$\underset{\text{over } Z}{\text{maximum}} \left(\frac{F_Q^C(Z)}{K(Z)} \right) \quad \text{is not increasing.}$$

g. With the relationships specified in Specification 4.2.2.2d above not being satisfied:

1) Calculate the percent $F_Q(Z)$ exceeds its limits by the following expression:

$$\left\{ \left(\underset{\text{over } Z}{\text{maximum}} \left[\frac{F_Q^C(Z) \times W(Z)}{\frac{F_Q^{RTP}}{P} \times K(Z)} \right] \right) - 1 \right\} \times 100 \text{ for } P > 0.5$$

$$\left\{ \left(\underset{\text{over } Z}{\text{maximum}} \left[\frac{F_Q^C(Z) \times W(Z)}{\frac{F_Q^{RTP}}{0.5} \times K(Z)} \right] \right) - 1 \right\} \times 100 \text{ for } P \leq 0.5\text{, and}$$

2) The following action shall be taken.

Within 15 minutes, control the AFD to within new AFD limits which are determined by reducing the AFD limits specified in the CORE OPERATING LIMITS REPORT by 1% AFD for each percent $F_Q(Z)$ exceeds its limits as determined in Specification 4.2.2.2g.1. Within 8 hours, reset the AFD alarm setpoints to these modified limits.

h. The limits specified in Specification 4.2.2.2c are applicable in all core plane regions, i.e. 0 - 100%, inclusive.

i. The limits specified in Specifications 4.2.2.2d, 4.2.2.2f, and 4.2.2.2g 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 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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POWER DISTRIBUTION LIMITS

SURVEILLANCE REQUIREMENTS - UNIT 2

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 before exceeding 75% of RATED THERMAL POWER following each fuel loading.

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 Specification 4.2.2.2b. above to:

1) The F_{xy} limits for RATED THERMAL POWER (F_{xy}^{RTP}) for the appropriate measured core planes given in Specification 4.2.2.2e. and f. below, and

2) The relationship:

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

Where F_{xy}^L is the limit for fractional THERMAL POWER operation expressed as a function of F_{xy}^{RTP}, PF_{xy} is the power factor multiplier for F_{xy} specified in the COLR, 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 either:

a) 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 Effective Full Power Days (EFPD), whichever occurs first.

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 used in the Constant Axial Offset Control analysis for RATED THERMAL POWER (F_{xy}^{RTP}) shall be specified for all core planes containing Bank "D" control rods and all unrodded core planes in the COLR per Specification 6.8.1.6;

f. The F_{xy} limits of Specification 4.2.2.2e. 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, and

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.

3/4.2.5 DNB PARAMETERS

LIMITING CONDITION FOR OPERATION

3.2.5 The following DNB-related parameters shall be maintained within the limits:

a. Reactor Coolant System T_{avg} (TI-0412, TI-0422, TI-0432, TI-0442), ≤ 592.5°F ~~(Unit 1) or ≤ 591°F (Unit 2)~~.

b. Pressurizer Pressure (PI-0455A,B&C, PI-0456 & PI-0456A, PI-0457 & PI-0457A, PI-0458 & PI-0458A), ≥ 2199 psig* ~~(Unit 1) or 2224 psig* (Unit 2)~~.

c. Reactor Coolant System Flow (FI-0414, FI-0415, FI-0416, FI-0424, FI-0425, FI-0426, FI-0434, FI-0435, FI-0436, FI-0444, FI-0445, FI-0446) ≥391,225 gpm** ~~(Unit 1) or 396,198 gpm** (Unit 2)~~.

APPLICABILITY: MODE 1.

ACTION:

With any of the above parameters exceeding its limit, restore the parameter to within its limit within 2 hours or reduce THERMAL POWER to less than 5% of RATED THERMAL POWER within the next 4 hours.

SURVEILLANCE REQUIREMENTS

4.2.5.1 Reactor Coolant System T_{avg} and Pressurizer Pressure shall be verified to be within their limits at least once per 12 hours. RCS flow rate shall be monitored for degradation at least once per 12 hours. In the event of flow degradation, RCS flow rate shall be determined by precision heat balance within 7 days of detection of flow degradation.

4.2.5.2 The RCS flow rate indicators shall be subjected to CHANNEL CALIBRATION at each fuel loading and at least once per 18 months.

4.2.5.3 After each fuel loading, the RCS flow rate shall be determined by precision heat balance prior to operation above 75% RATED THERMAL POWER. The RCS flow rate shall also be determined by precision heat balance at least once per 18 months. Within 7 days prior to performing the precision heat balance flow measurement, the instrumentation used for performing the precision heat balance shall be calibrated. The provisions of 4.0.4 are not applicable for performing the precision heat balance flow measurement.

*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.

**Includes a 2.2% ~~(Unit 1) or 3.5% (Unit 2)~~ flow measurement uncertainty.

TABLE 3.[illegible] (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TOTAL ALLOWANCE (TA)	Z	SENSOR ERROR (S)	TRIP SETPOINT	ALLOWABLE VALUE
7. Semi-Automatic Switchover to Containment Emergency Sump (Continued)					
b. RWST Level--Low-Low Coincident With Safety Injection (LI-0990A&B, LI-0991A&B, LI-0992A, LI-0993A)	3.5	0.71	1.67	≥ 275.3 in. from tank base (≥ 39.1% of instrument span)	≥ 264.9 in. from tank base (≥ 37.4% of instrument span)
8. Loss of Power to 4.16 kV ESF Bus					
a. 4.16 kV ESF Bus Undervoltage-Loss of Voltage	N.A.	N.A.	N.A.	≥ 2975 volts with a ≤ 0.8 second time delay.	≥ 2912 volts with a ≤ 0.8 second time delay.
b. 4.16 kV ESF Bus Undervoltage-Degraded Voltage	N.A.	N.A.	N.A.	≥ 3746 volts with a ≤ 20 second time delay.	≥ 3683 volts with a ≤ 20 second time delay.
9. Engineered Safety Features Actuation System Interlocks					
a. Pressurizer Pressure, P-11 (PI-0455A,B&C, PI-0456 & PI-0456A, PI-0457 & PI-0457A, PI-0458 & PI-0458A),	N.A.	N.A.	N.A.	≤ 2000 psig ~~(Unit 1) ≤ 1970 psig (Unit 2)~~	≤ 2010 psig ~~(Unit 1) ≤ 1980 psig (Unit 2)~~
b. Reactor Trip, P-4	N.A.	N.A.	N.A.	N.A	N.A.

3/4.5.1 ACCUMULATORS

LIMITING CONDITION FOR OPERATION

3.5.1 Each Reactor Coolant System (RCS) accumulator shall be OPERABLE with:

a. The isolation valve open,

b. ~~Unit 1:~~ A contained borated water volume of between 6555 (29.2% of instrument span) and 6909 gallons (70.7% of instrument span) (LI-0950, LI-0951, LI-0952, LI-0953, LI-0954, LI-0955, LI-0956, LI-0957),

~~Unit 2: - A contained borated water volume of between 6616 (36% of instrument span) and 6854 gallons (64% of instrument span) (LI-0950, LI-0951, LI-0952, LI-0953, LI-0954, LI-0955, LI-0956, LI-0957),~~

c. A boron concentration of between 1900 ppm and 2600 ppm, and

d. A nitrogen cover-pressure of between 617 and 678 psig. (PI-0960A&B, PI-0961A&B, PI-0962A&B, PI-0963A&B, PI-0964A&B, PI-0965A&B, PI-0966A&B, PI-0967A&B)

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 1 hour or be in at least HOT STANDBY within the next 6 hours and reduce pressurizer pressure to less than 1000 psig 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 6 hours and reduce pressurizer pressure to less than 1000 psig within the following 6 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 (HV-8808A, B, C, D).

---

*Pressurizer pressure above 1000 psig.

3/4.5.4 REFUELING WATER STORAGE TANK

LIMITING CONDITION FOR OPERATION

3.5.4 The refueling water storage tank (RWST) shall be OPERABLE with:

a. A minimum contained borated water volume of 631,478 gallons (86% of instrument span) (LI-0990A&B, LI-0991A&B, LI-0992A, LI-0993A).

b. A boron concentration of between 2400 ppm and 2600 ppm of boron,

c. A minimum solution temperature of 44°F ~~(Unit 1) or 54°F (Unit 2)~~, and

d. A maximum solution temperature of 116°F (TI-10982).

e. RWST Sludge Mixing Pump Isolation valves capable of closing on RWST low-level.

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

ACTION:

a. With the RWST inoperable except for the Sludge Mixing Pump Isolation Valves, restore the tank to OPERABLE status within 1 hour or be in at least HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With a Sludge Mixing Pump Isolation Valve(s) inoperable, restore the valve(s) to OPERABLE status within 24 hours or isolate the sludge mixing system by either closing the manual isolation valves or deenergizing the OPERABLE solenoid pilot valve within 6 hours and maintain closed.

SURVEILLANCE REQUIREMENTS

4.5.4 The RWST shall be demonstrated OPERABLE:

a. At least once per 7 days by:

1) Verifying the contained borated water volume in the tank, and

2) Verifying the boron concentration of the water.

b. At least once per 24 hours by verifying the RWST temperature when the outside air temperature is less than 40°F ~~(Unit 1) or 50°F (Unit 2)~~.

c. At least once per 18 months by verifying that the sludge mixing pump isolation valves automatically close upon an RWST low-level test signal.

3/4.2 POWER DISTRIBUTION LIMITS ~~- UNIT 1~~

BASES

The specifications of this section provide assurance of fuel integrity during Condition I (Normal Operation) and II (Incidents of Moderate Frequency) events by: (1) meeting the DNB design criterion 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; and

$F_{\Delta H}^N$ 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.

3/4.2.1 AXIAL FLUX DIFFERENCE

The limits on AXIAL FLUX DIFFERENCE (AFD) assure that the $F_Q(Z)$ upper bound envelope of the F_Q limit specified in the CORE OPERATING LIMITS REPORT (COLR) times K(Z) is not exceeded during either normal operation or in the event of xenon redistribution following power changes.

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 1-minute average of each of the OPERABLE excore detector outputs and provides an alarm message immediately if the AFD for two or more OPERABLE excore channels are outside the allowed ΔI power operating space for RAOC operation specified within the COLR and the THERMAL POWER is greater than 50% of RATED THERMAL POWER.

AXIAL FLUX DIFFERENCE (Continued)

3/4.2.2 and 3/4.2.3 HEAT FLUX HOT CHANNEL FACTOR AND NUCLEAR ENTHALPY RISE HOT CHANNEL FACTOR - $F^N_{\Delta H}$

The limits on heat flux hot channel factor and nuclear enthalpy rise hot channel factor ensure that: (1) the design limit on peak local power density is not exceeded, (2) the DNB design criterion is met, and (3) 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 ensure that the limits are maintained provided:

a. Control rods in a single group move together with no inidivdual rod insertion differing by more than ± 12 steps, indicated, from the group demand position;

b. Control rod groups are sequenced with a constant tip-to-tip distance between banks as described in Specification 3.1.3.6;

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

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. 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.

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.

The heat flux hot channel factor $F_Q(Z)$ is measured periodically and increased by a cycle and height dependent power factor appropriate to RAOC operation, W(Z), to provide assurance that the limit on the heat flux hot channel factor, $F_Q(Z)$, is met. W(Z) accounts for the effects of normal operation transients within the AFD band and was determined from expected power control maneuvers over the full range of burnup conditions in the core. The W(Z) function for normal operation and the AFD band are provided in the CORE OPERATING LIMITS REPORT per Specification 6.8.1.6.

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

When $F^N_{\Delta H}$ is measured, (i.e., inferred), measurement uncertainty (i.e., the appropriate uncertainty on the incore inferred hot rod peaking factor) must be allowed for and 4% is the appropriate allowance for a full core map taken with the incore detection system.

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 limit of 1.02 was selected to provide an allowance for the uncertainty associated with the indicated power tilt.

The 2-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% for each percent of tilt in excess of 1.

For purposes of monitoring QUADRANT POWER TILT RATIO when one excore detector is inoperable, the moveable 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 four symmetric thimbles. The two sets of four symmetric thimbles is a unique set of eight detector locations. 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 with the initial FSAR assumptions and have been analytically demonstrated adequate to meet the DNB design criterion throughout each analyzed transient. The indicated T_{avg} value of 592.5°F and the indicated pressurizer pressure value of 2199 psig correspond to analytical limits of 594.4°F and 2185 psig respectively, with allowance for measurement uncertainty.

3/4.2.5 DNB PARAMETERS (Continued)

The 12-hour periodic surveillance of these parameters through instrument readout is sufficient to ensure that the parameters are restored within their limits following load changes and other expected transient operation. The 18 month periodic measurement of the RCS total flow rate is adequate to detect flow degradation and ensure correlation of the flow indication channels with measured flow such that the indicated percent flow will provide sufficient verification of the flow rate degradation on a 12 hour basis. A change in indicated percent flow which is greater than the instrument channel inaccuracies and parallax errors is an appropriate indication of RCS flow degradation.

3/4.2 POWER DISTRIBUTION LIMITS - UNIT 2

BASES

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 minimum DNBR in the core greater than or equal to 1.30 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; and

$F_{xy}(Z)$ 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

The limits on AXIAL FLUX DIFFERENCE (AFD) assure that the $F_Q(Z)$ upper bound envelope of the F_Q limit specified in the CORE OPERATING LIMITS REPORT (COLR) times K(Z) 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 conditons. The 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.

AXIAL FLUX DIFFERENCE (Continued)

Although it is intended that the plant will be operated with the AFD within the target band required by Specification 3.2.1 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 specified in the COLR 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 1-minute average of each of the OPERABLE excore detector outputs and provides an alarm message immediately if the AFD for two 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 NUCLEAR ENTHALPY RISE HOT CHANNEL FACTOR - $F^N_{\Delta H}$

The limits on heat flux hot channel factor 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 ensure that the limits are maintained provided:

a. Control rods in a single group move together with no individual rod insertion differing by more than ± 12 steps, indicated, from the group demand position;

b. Control rod groups are sequenced with a constant tip-to-tip distance between banks as described in Specification 3.1.3.6;

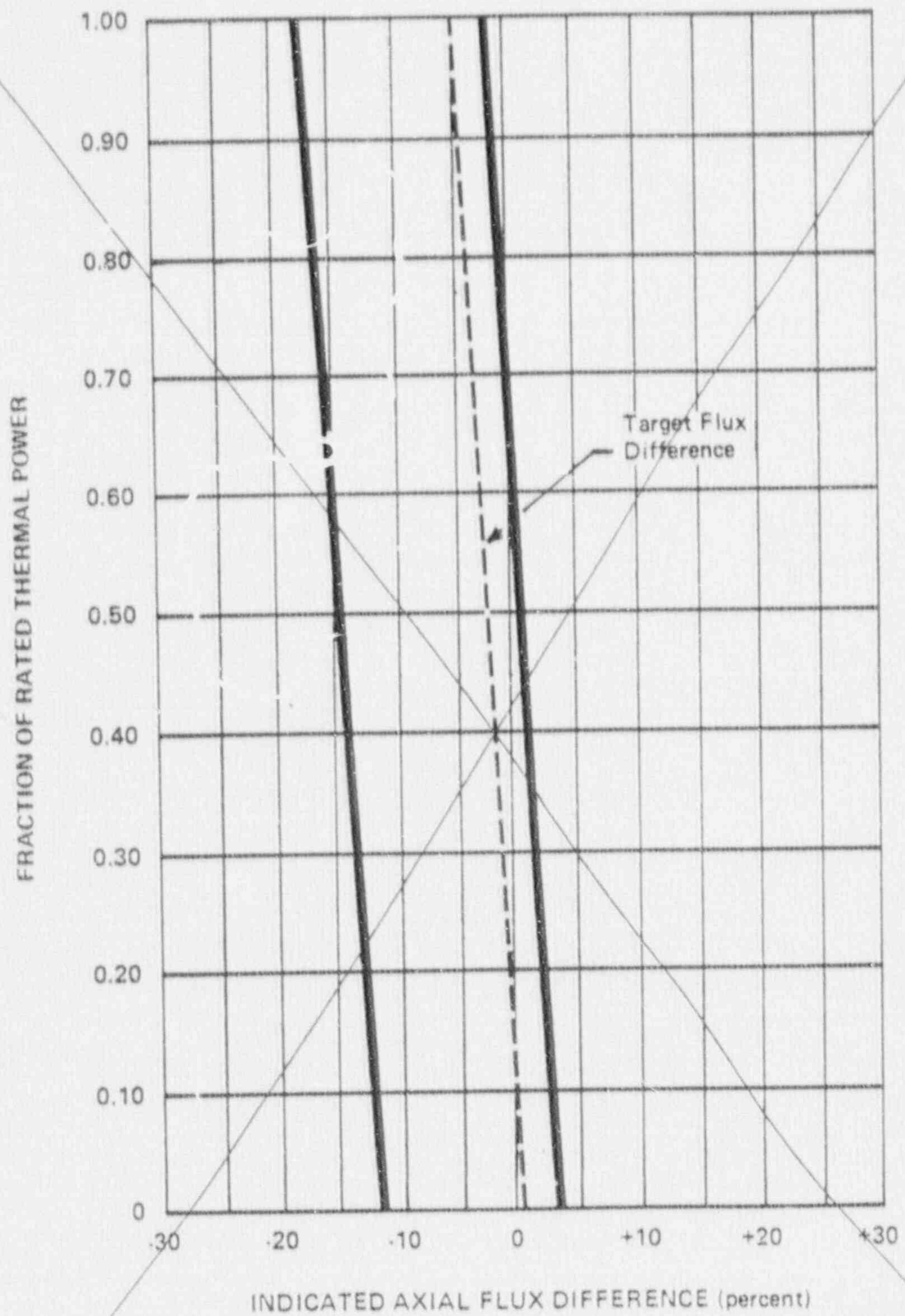

FIGURE B 3/4 2-1

TYPICAL INDICATED AXIAL FLUX DIFFERENCE VERSUS THERMAL POWER

HEAT FLUX HOT CHANNEL FACTOR 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; and

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. 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.

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.

When $F^N_{\Delta H}$ is measured, (i.e., inferred), measurement uncertainty (i.e., the appropriate uncertainty on the incore inferred hot rod peaking factor) must be allowed for and 4% is the appropriate allowance for a full core map taken with the incore detection system.

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

a. Design limit DNBR of 1.30 vs 1.28,

b. Grid Spacing (K_5) of 0.046 vs 0.059,

c. Thermal Diffusion Coefficient of 0.038 vs 0.059,

d. DNBR Multiplier of 0.86 vs 0.88, and

e. Pitch reduction.

The applicable values of rod bow penalties are referenced in the FSAR.

BASES

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

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 F_{xy} limit for RATED THERMAL POWER (F_{xy}^{RTP}) as specified in the COLR per Specification 6.8.1.6 was determined from expected power control maneuvers over the full range of burnup conditions in the core.

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 limit of 1.02 was selected to provide an allowance for the uncertainty associated with the indicated power tilt.

The 2-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% for each percent of tilt in excess of 1.

For purposes of monitoring QUADRANT POWER TILT RATIO when one excore detector is inoperable, the moveable 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 four symmetric thimbles. The two sets of four symmetric thimbles is a unique set of eight detector locations. 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 with the initial FSAR assumptions and have been analytically demonstrated adequate to maintain a minimum DNBR of 1.30 throughout each analyzed transient. The indicated T_{avg} value of 591°F and the indicated pressurizer pressure value of 2224 psig correspond to analytical limits of 592.5°F and 2205 psig respectively, with allowance for measurement uncertainty.

3/4.2.5 DNB PARAMETERS (Continued)

The 12-hour periodic surveillance of these parameters through instrument readout is sufficient to ensure that the parameters are restored within their limits following load changes and other expected transient operation. The 18 month periodic measurement of the RCS total flow rate is adequate to detect flow degradation and ensure correlation of the flow indication channels with measured flow such that the indicated percent flow will provide sufficient verification of the flow rate degradation on a 12 hour basis. A change in indicated percent flow which is greater than the instrument channel inaccuracies and parallax errors is an appropriate indication of RCS flow degradation.

ECCS SUBSYSTEMS (Continued)

The limitation for all safety injection pumps to be inoperable below 350°F provides assurance that a mass addition pressure transient can be relieved by the operation of a single PORV.

The Surveillance Requirements provided to ensure OPERABILITY of each component ensure that at a minimum, the assumptions used in the safety analyses are met and that subsystem OPERABILITY is maintained. Surveillance Requirements for throttle valve position stops and flow balance testing provide assurance that proper ECCS flows will be maintained in the event of a LOCA. Maintenance of proper flow resistance and pressure drop in the piping system to each injection point is necessary to: (1) prevent total pump flow from exceeding runout conditions when the system is in its minimum resistance configuration, (2) provide the proper flow split between injection points in accordance with the assumptions used in the ECCS-LOCA analyses, (3) provide an acceptable level of total ECCS flow to all injection points equal to or above that assumed in the ECCS-LOCA analyses and (4) to ensure that centrifugal charging pump injection flow which is directed through the seal injection path is less than or equal to the amount assumed in the safety analysis. The surveillance requirements for leakage testing of ECCS check valves ensure a failure of one valve will not cause an intersystem LOCA. In MODE 3, with either HV-8809A or B closed for ECCS check valve leak testing, adequate ECCS flow for core cooling in the event of a LOCA is assured.

3/4.5.4 REFUELING WATER STORAGE TANK

The OPERABILITY of the Refueling Water Storage Tank (RWST) as part of the ECCS ensures that sufficient negative reactivity is injected into the core to counteract any positive increase in reactivity caused by RCS cooldown. RCS cooldown can be caused by inadvertent depressurization, a loss-of-coolant accident, or a steam line rupture.

The limits on RWST minimum volume and boron concentration ensure that 1) sufficient water is available within containment to permit recirculation cooling flow to the core, 2) the reactor will remain subcritical in the cold condition following a small LOCA or steamline break, assuming complete mixing of the RWST, RCS, and ECCS water volumes with all control rods inserted except the most reactive control assembly (ARI-1), and 3) the reactor will remain subcritical in the cold condition following a large break LOCA assuming complete mixing of the RWST, RCS, ECCS water and other sources of water that may eventually reside in the sump, post-LOCA with ~~all control rods assumed to be out (Unit 2) or~~ all control rods inserted except for the two most reactive control assemblies ~~(Unit 1)~~.

The contained water volume limit includes an allowance for water not usable because of tank discharge line location or other physical characteristics.

SEMIANNUAL RADIOACTIVE EFFLUENT RELEASE REPORT (Continued)

The Semiannual Radioactive Effluent Release Reports shall also include the following: an explanation as to why the inoperability of liquid or gaseous effluent monitoring instrumentation was not corrected within the time specified in Specification 3.3.3.9 or 3.3 3.10, respectively; and description of the events leading to liquid holdup tanks or gas storage tanks exceeding the limits of Specification 3.11.1.4 or 3.11.2.6, respectively.

MONTHLY OPERATING REPORTS

6.8.1.5 Routine reports of operating statistics and shutdown experience, including documentation of all challenges to the PORVs 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 Administrator of the Regional Office of the NRC, no later than the 15th of each month following the calendar month covered by the report.

CORE OPERATING LIMITS REPORT - ~~UNIT 1~~

6.8.1.6 Core operating limits shall be established and documented in the CORE OPERATING LIMITS REPORT (COLR) before each reload cycle or any remaining part of a reload cycle for the following:

a. SHUTDOWN MARGIN LIMIT FOR MODES 1 and 2 for Specification 3/4.1.1.1,

b. SHUTDOWN MARGIN LIMITS FOR MODES 3, 4, and 5 for Specification 3/4.1.1.2,

c. Moderator temperature coefficient BOL and EOL limits and the 300-ppm surveillance limit for Specification 3/4.1.1.3,

d. Shutdown Rod Insertion Limit for Specification 3/4.1.3.5,

e. Control Rod Insertion Limits for Specification 3/4.1.3.6,

f. Axial Flux Difference Limits for Specification 3/4.2.1,

g. Heat Flux Hot Channel Factor, K(Z) and W(Z) for Specification 3/4.2.2,

h. Nuclear Enthalpy Rise Hot Channel Factor Limit and the Power Factor Multiplier for Specification 3/4.2.3.

The analytical methods used to determine the core operating limits shall be those previously approved by the NRC in:

CORE OPERATING LIMITS REPORT (Continued) ~~UNIT 1~~

a. WCAP-9272-P-A, "WESTINGHOUSE RELOAD SAFETY EVALUATION METHODOLOGY," July 1985 (W Proprietary).
(Methodology for Specification 3.1.1.3 - Moderator Temperature Coefficient, 3.1.3.5 - Shutdown Bank Insertion Limit, 3.1.3.6 - Control Bank Insertion Limits, and 3.2.3 - Nuclear Enthalpy Rise Hot Channel Factor.)

b. WCAP-10216-P-A, "RELAXATION OF CONSTANT AXIAL OFFSET CONTROL F_Q SURVEILLANCE TECHNICAL SPECIFICATION," June 1983 (W Proprietary).
(Methodology for Specifications 3.2.1 - Axial Flux Difference (Relaxed Axial Offset Control) and 3.2.2 - Heat Flux Hot Channel Factor (W(Z) surveillance requirements for F_Q Methodology).)

c. WCAP-9220-P-A, Rev. 1, "WESTINGHOUSE ECCS EVALUATION MODEL-1981 VERSION," February 1982 (W Proprietary).
(Methodology for Specification 3.2.2 - Heat Flux Hot Channel Factor.)

The core operating limits shall be determined so that all applicable limits (e.g., fuel thermal-mechanical limits, core thermal-hydraulic limits, ECCS limits, nuclear limits such as shutdown margin, and transient and accident analysis limits) of the safety analysis are met.

The CORE OPERATING LIMITS REPORT, including any mid-cycle revisions or supplements thereto, shall be provided upon issuance, for each reload cycle, to the NRC Document Control Desk with copies to the Regional Administrator and Resident Inspector.

CORE OPERATING LIMITS REPORT - UNIT 2

6.8.1.6 Core operating limits shall be established and documented in the CORE OPERATING LIMITS REPORT (COLR) before each reload cycle or any remaining part of a reload cycle for the following:

a. SHUTDOWN MARGIN limit for MODES 1 and 2 for Specification 3/4.1.1.1,

b. SHUTDOWN MARGIN limits for MODES 3, 4, and 5 for Specification 3/4.1.1.2,

c. Moderator temperature coefficient BOL and EOL limits and the 300-ppm surveillance limit for Specification 3/4.1.1.3,

d. Shutdown Rod Insertion Limits for Specification 3/4.1.3.5,

e. Control Rod Insertion Limits for Specification 3/4.1.3.6,

f. Axial Flux Difference Limits, and target band for Specification 3/4.2.1,

g. Heat Flux Hot Channel Factor, K(Z), the Power Factor Multiplier and F_{xy}^{RTP}, for Specification 3/4.2.2,

h. Nuclear Enthalpy Rise Hot Channel Factor Limit and the Power Factor Multiplier for Specification 3/4.2.3.

The analytical methods used to determine the core operating limits shall be those previously approved by the NRC in:

CORE OPERATING LIMITS REPORT (Continued) - UNIT 2

a. WCAP-9272-P-A, "WESTINGHOUSE RELOAD SAFETY EVALUATION METHODOLOGY," July 1985 (W Proprietary).
(Methodology for Specification 3.1.1.3 - Moderator Temperature Coefficient, 3.1.3.5 - Shutdown Bank Insertion Limit, 3.1.3.6 - Control Bank Insertion Limits, 3.2.1 - Axial Flux Difference, 3.2.2 - Heat Flux Hot Channel Factor, and 3.2.3 - Nuclear Enthalpy Rise Hot Channel Factor.)

b. WCAP-8385, "POWER DISTRIBUTION CONTROL AND LOAD FOLLOWING PROCEDURES - TOPICAL REPORT," September 1974 (W Proprietary).
(Methodology for Specification 3.2.1 - Axial Flux Difference [Constant Axial Offset Control].)

c. T. M. Anderson to K. Kniel (Chief of Core Performance Branch, NRC) January 31, 1980 -- Attachment: Operation and Safety Analysis Aspects of an Improved Load Follow Package.
(Methodology for Specification 3.2.1 - Axial Flux Difference [Constant Axial Offset Control].)

d. NUREG-0800, Standard Review Plan, U. S. Nuclear Regulatory Commission, Section 4.3, Nuclear Design, July 1981. Branch Technical Position CPB 4.3-1, Westinghouse Constant Axial Offset Control (CAOC), Rev.2, July 1981.
(Methodology for Specification 3.2.1 - Axial Flux Difference [Constant Axial Offset Control].)

e. WCAP-9220-P-A, Rev. 1, "WESTINGHOUSE ECCS EVALUATION MODEL-1981 VERSION," February 1982 (W Proprietary).
(Methodology for Specification 3.2.2 - Heat Flux Hot Channel Factor.)

The core operating limits shall be determined so that all applicable limits (e.g., fuel thermal-mechanical limits, core thermal-hydraulic limits, ECCS limits, nuclear limits such as shutdown margin, and transient and accident analysis limits) of the safety analysis are met.

The CORE OPERATING LIMITS REPORT, including any mid-cycle revisions or supplements thereto, shall be provided upon issuance, for each reload cycle, to the NRC Document Control Desk with copies to the Regional Administrator and Resident Inspector.

SPECIAL REPORTS

6.8.2 Special reports shall be submitted to the Regional Administrator of the Regional Office of the NRC within the time period specified for each report.

Attachment 2b

Vogtle Electric Generating Plant Units 1 and 2

Request for Technical Specifications Changes

VANTAGE-5 Fuel Design

Technical Specifications Typed Pages

Effective following the Vogtle 2 Cycle 2 Shutdown
(Effective as of Vogtle 2 Cycle 3 Startup)

SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

BASES

THIS PAGE BECOMES APPLICABLE FOLLOWING SHUTDOWN FROM UNIT 2 CYCLE 2 OPERATION.

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

THIS PAGE BECOMES APPLICABLE FOLLOWING SHUTDOWN FROM UNIT 2 CYCLE 2 OPERATION.

BASES

THIS PAGE BECOMES APPLICABLE FOLLOWING SHUTDOWN FOR UNIT 2 CYCLE 2 OPERATION.

ADMINISTRATIVE CONTROLS

THIS PAGE BECOMES APPLICABLE FOLLOWING SHUTDOWN FROM UNIT 2 CYCLE 2 OPERATION.

2.0 SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS*

2.1 SAFETY LIMITS

REACTOR CORE

2.1.1 The combination of THERMAL POWER (NI-0041, NI-0042, NI-0043, NI-0044), pressurizer pressure (PI-0455A, B&C, PI-0456 & PI-0456A, PI-0457 & PI-0457A, PI-0458 & PI-0458A), and the highest operating loop coolant temperature (T_{avg}) (TI-0412, TI-0422, TI-0432, TI-0442) shall not exceed the limits shown in Figure 2.1-1.

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 6.6.1.

REACTOR COOLANT SYSTEM PRESSURE

2.1.2 The Reactor Coolant System pressure (PI-0408, PI-0418, PI-0428, PI-0438) 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 6.6.1.

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 6.6.1.

*Where specific instrument numbers are provided in parentheses they are for information only, and apply to each unit unless specifically noted (to assist in identifying associated instrument channels or loops) and are not intended to limit the requirements to the specific instruments associated with the number.

THIS PAGE BECOMES APPLICABLE FOLLOWING SHUTDOWN FROM UNIT 2 CYCLE 2 OPERATION.

THIS PAGE BECOMES APPLICABLE FOLLOWING SHUTDOWN FROM UNIT 2 CYCLE 2 OPERATION.

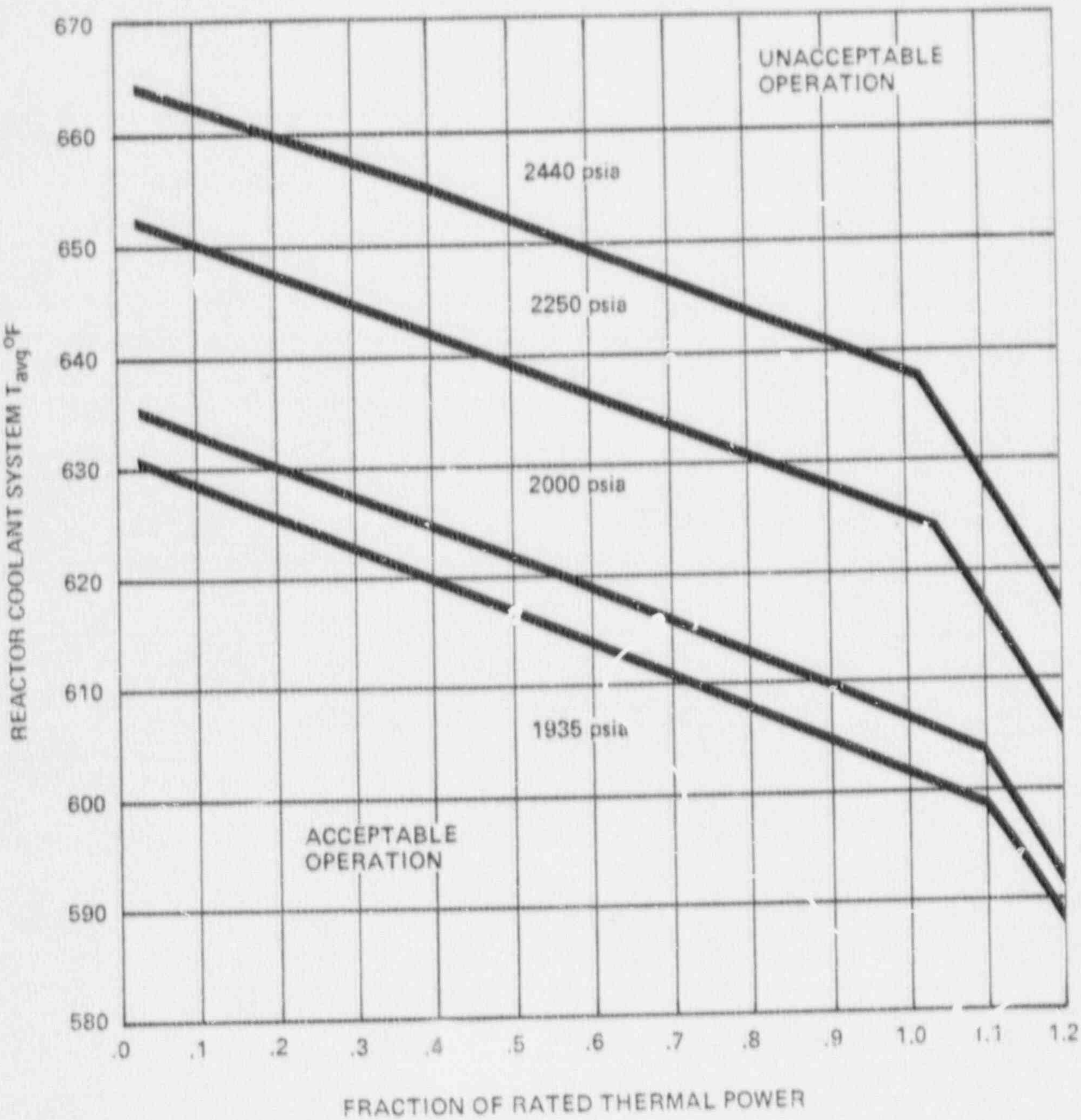

FIGURE 2.1-1
REACTOR CORE SAFETY LIMIT

2.2 LIMITING SAFETY SYSTEM SETTINGS

REACTOR TRIP SYSTEM INSTRUMENTATION SETPOINTS

2.2.1 The Reactor Trip System Instrumentation and Interlock Setpoints shall be set consistent with the Trip Setpoint values shown in Table 2.2-1.

APPLICABILITY: As shown for each channel in Table 3.3-1.

ACTION:

a. With a Reactor Trip System Instrumentation or Interlock Setpoint less conservative than the value shown in the Trip Setpoint column but more conservative than the value shown in the Allowable Value column of Table 2.2-1, adjust the Setpoint consistent with the Trip Setpoint value.

b. With the Reactor Trip System Instrumentation or Interlock Setpoint less conservative than the value shown in the Allowable Values column of Table 2.2-1, either:

   1. Adjust the Setpoint consistent with the Trip Setpoint value of Table 2.2-1 and determine within 12 hours that Equation 2.2-1 was satisfied for the affected channel, or

   2. Declare the channel inoperable and apply the applicable ACTION statement requirement of Specification 3.3.1 until the channel is restored to OPERABLE status with its Setpoint adjusted consistent with the Trip Setpoint value.

   Equation 2.2-1 $Z + R + S \leq TA$

   Where:

   Z = The value from Column Z of Table 2.2-1 for the affected channel,

   R = The "as measured" value (in percent span) of rack error for the affected channel,

   S = Either the "as measured" value (in percent span) of the sensor error, or the value from Column S (Sensor Error) of Table 2.2-1 for the affected channel, and

   TA = The value from Column TA (Total Allowance) of Table 2.2-1 for the affected channel.

THIS PAGE BECOMES APPLICABLE FOLLOWING SHUTDOWN FROM UNIT 2 CYCLE 2 OPERATION.

TABLE 2.2-1

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TOTAL ALLOWANCE (TA)	Z	SENSOR ERROR (S)	TRIP SETPOINT	ALLOWABLE VALUE
1. Manual Reactor Trip	N.A.	N.A.	N.A.	N.A.	N.A.
2. Power Range, Neutron Flux (NI-0041B&C, NI-0042B&C, NI-0043B&C, NI-0044B&C)					
a. High Setpoint	7.5	4.56	0	$\leq$109% of RTP#	$\leq$111.3% of RTP#
b. Low Setpoint	8.3	4.56	0	$\leq$25% of RTP#	$\leq$27.3% of RTP#
3. Power Range, Neutron Flux, High Positive Rate (NI-0041B&C, NI-0042B&C, NI-0043B&C, NI-0044B&C)	1.6	0.50	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. Deleted.					
5. Intermediate Range, Neutron Flux (NI-0035B, NI-0036B)	17.0	8.41	0	$\leq$25% of RTP#	$\leq$31.1% of RTP#
6. Source Range, Neutron Flux (NI-0031B, NI-0032B)	17.0	10.01	0	$\leq 10^5$ cps	$\leq 1.4 \times 10^5$ cps
7. Overtemperature ΔT (TDI-411C, TDI-421C, TDI-431C, TDI-441C)	10.7	7.04	1.96 + 1.17	See Note 1	See Note 2
8. Overpower ΔT (TDI-411B, TDI-421B, TDI-431B, TDI-441B)	4.3	1.54	1.96	See Note 3	See Note 4

#RTP = RATED THERMAL POWER

THIS PAGE BECOMES APPLICABLE FOLLOWING SHUTDOWN FROM UNIT 2 CYCLE 2 OPERATION.

TABLE 2.2-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TOTAL ALLOWANCE (TA)	Z	SENSOR ERROR (S)	TRIP SETPOINT	ALLOWABLE VALUE
9. Pressurizer Pressure-Low (PI-0455A,B&C, PI-0456 & PI-0456A, PI-0457 & PI-0457A, PI-0458 & PI-0458A)	3.1	0.71	1.67	≥1960 psig**	≥1950 psig
10. Pressurizer Pressure-High (PI-0455A,B&C, PI-0456 & PI-0456A, PI-0457 & PI-0457A, PI-0458 & PI-0458A)	3.1	0.71	1.67	≤2385 psig	≤2395 psig
11. Pressurizer Water Level-High (LI-0459A, LI-0460A, LI-0461)	8.0	2.18	1.67	≤92% of instrument span	≤93.9% of instrument span
12. Reactor Coolant Flow-Low (LOOP1 LOOP2 LOOP3 LOOP4 FI-0414 FI-0424 FI-0434 FI-0444 FI-0415 FI-0425 FI-0435 FI-0445 FI-0416 FI-0426 FI-0436 FI-0446)	2.5	1.87	0.60	≥90% of loop design flow*	≥89.4% of loop design flow*
13. Steam Generator Water Level Low-Low (LOOP1 LOOP2 LOOP3 LOOP4 LI-0517 LI-0527 LI-0537 LI-0547 LI-0518 LI-0528 LI-0538 LI-0548 LI-0519 LI-0529 LI-0539 LI-0549 LI-0551 LI-0552 LI-0553 LI-0554)	18.5 (21.8)***	17.18 (18.21)***	1.67	≥18.5% (37.8)*** of narrow range instrument span	≥17.8% (35.9)*** of narrow range instrument span
14. Undervoltage - Reactor Coolant Pumps	6.0	0.58	0	≥9600 volts (70% bus voltage)	≥9481 volts (69% bus voltage)
15. Underfrequency - Reactor Coolant Pumps	3.3	0.50	0	≥57.3 Hz	≥57.1 Hz

*Loop design flow = 95,700 gpm

**Time constants utilized in the lead-lag controller for Pressurizer Pressure-Low are 10 seconds for lead and 1 second for lag. CHANNEL CALIBRATION shall ensure that these time constants are adjusted to these values.

***The value stated inside the parenthesis is for instrumentation that has the lower tap at elevation 333"; the value stated outside the parenthesis is for instrumentation that has the lower tap at elevation 438".

THIS PAGE BECOMES APPLICABLE FOLLOWING SHUTDOWN FROM UNIT 2 CYCLE 2 OPERATION.

TABLE 2.2-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TOTAL ALLOWANCE (TA)	Z	SENSOR ERROR (S)	TRIP SETPOINT	ALLOWABLE VALUE
16. Turbine Trip					
a. Low Fluid Oil Pressure (PT-6161, PT-6162, PT-6163)	N.A.	N.A.	N.A.	≥580 psig	≥500 psig
b. Turbine Stop Valve Closure	N.A.	N.A.	N.A.	≥96.7% open	≥96.7% open
17. Safety Injection Input from ESF	N.A.	N.A.	N.A.	N.A.	N.A.
18. Reactor Trip System Interlocks					
a. Intermediate Range Neutron Flux, P-6 (NI-0035B, NI-0036B)	N.A.	N.A.	N.A.	≥1 x 10^{-10} amp	≥6 x 10^{-11} amp
b. Low Power Reactor Trips Block, P-7					
1) P-10 input (NI-0041B&C, NI-0042B&C, NI-0043B&C, NI-0044B&C)	N.A.	N.A.	N.A.	≤10% of RTP#	≤12.3% of RTP#
2) P-13 input (PI-0505, PI-0506)	N.A.	N.A.	N.A.	≤10% RTP# Turbine Impulse Pressure Equivalent	≤12.3% RTP# Turbine Impulse Pressure Equivalent
c. Power Range Neutron Flux, P-8 (NI-0041B&C, NI-0042B&C, NI-0043B&C, NI-0044B&C)	N.A.	N.A.	N.A.	≤48% of RTP#	≤50.3% of RTP#

#RTP = RATED THERMAL POWER

THIS PAGE BECOMES APPLICABLE FOLLOWING SHUTDOWN FROM UNIT 2 CYCLE 2 OPERATION.

TABLE 2.2-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TOTAL ALLOWANCE (TA)	Z	SENSOR ERROR (S)	TRIP SETPOINT	ALLOWABLE VALUE
d. Power Range Nuetron Flux, P-9 (NI-0041B&C, NI-0042B&C, NI-0043B&C, NI-0044B&C)	N.A.	N.A.	N.A.	$\leq$50% of RTP#	$\leq$52.3% of RTP#
e. Power Range Neutron Flux, P-10 (NI-0041B&C, NI-0042B&C, NI-0043B&C, NI-0044B&C)	N.A.	N.A.	N.A.	$\geq$10% of RTP#	$\geq$7.7% of RTP#
f. Turbine Impulse Chamber Pressure, P-13 (PI-0505, PI-0506)	N.A.	N.A.	N.A.	$\leq$10% RTP# Turbine Impulse Pressure Equivalent	$\leq$12.3% RTP# Turbine Impulse Pressure Equivalent
19. Reactor Trip Breakers	N.A.	N.A.	N.A.	N.A.	N.A.
20. Automatic Trip and Interlock Logic	N.A.	N.A.	N.A.	N.A.	N.A.

#RTP = RATED THERMAL POWER

THIS PAGE BECOMES APPLICABLE FOLLOWING SHUTDOWN FROM UNIT 2 CYCLE 2 OPERATION.

TABLE 2.2-1 (Continued)

TABLE NOTATIONS

NOTE 1: OVERTEMPERATURE ΔT

$$\Delta T \frac{(1 + \tau_1 S)}{(1 + \tau_2 S)} \left(\frac{1}{1 + \tau_3 S}\right) \leq \Delta T_0 \{K_1 - K_2 \frac{(1 + \tau_4 S)}{(1 + \tau_5 S)} [T \left(\frac{1}{1 + \tau_6 S}\right) - T'] + K_3(P - P') - f_1(\Delta I)\}$$

Where: ΔT = Measured ΔT

$\frac{1 + \tau_1 S}{1 + \tau_2 S}$ = Lead-lag compensator on measured ΔT;

τ_1, τ_2 = Time constants utilized in lead-lag compensator for ΔT, $\tau_1 \geq 8$ s, $\tau_2 \leq 3$ s;

$\frac{1}{1 + \tau_3 S}$ = Lag compensator on measured ΔT;

τ_3 = Time constants utilized in the lag compensator for ΔT, $\tau_3 = 0$ s;

ΔT_0 = Indicated ΔT at RATED THERMAL POWER;

K_1 $\leq$ 1.12;

K_2 = 0.0224/°F;

$\frac{1 + \tau_4 S}{1 + \tau_5 S}$ = The function generated by lead-lag compensator for T_{avg} dynamic compensation;

τ_4, τ_5 = Time constants utilized in the lead-lag compensator for T_{avg}, $\tau_4 \geq 28$ s, $\tau_5 \leq 4$ s;

T = Average temperature, °F;

$\frac{1}{1 + \tau_6 S}$ = Lag compensator on measured T_{avg};

τ_6 = Time constant utilized in the measured T_{avg} lag compensator, $\tau_6 = 0$ s;

THIS PAGE BECOMES APPLICABLE FOLLOWING SHUTDOWN FROM UNIT 2 CYCLE 2 OPERATION.

TABLE 2.2-1 (Continued)

TABLE NOTATIONS (Continued)

NOTE 1: (Continued)

T' $\leq$ 588.4°F (Nominal T_{avg} operating temperature);

K_3 = 0.00115/psig;

P = Pressurizer pressure, psig;

P' = 2235 psig (Nominal RCS operating pressure);

S = Laplace transform variable, s^{-1};

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

(1) For $q_t - q_b$ between -32.0% and + 11.0%, $f_1(\Delta I) = 0$, 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;

(2) For each percent that the magnitude of $q_t - q_b$ exceeds - 32.0%, the ΔT Trip Setpoint shall be automatically reduced by 3.25% of its value at RATED THERMAL POWER; and

(3) For each percent that the magnitude of $q_t - q_b$ exceeds + 11.0%, the ΔT Trip Setpoint shall be automatically reduced by 1.97% of its value at RATED THERMAL POWER.

NOTE 2: The channel's maximum Trip Setpoint shall not exceed its computed Trip Setpoint by more than 3.1% of ΔT span.

THIS PAGE BECOMES APPLICABLE FOLLOWING SHUTDOWN FROM UNIT 2 CYCLE 2 OPERATION.

TABLE 2.2-1 (Continued)

TABLE NOTATIONS (Continued)

NOTE 3: OVERPOWER ΔT

$$\Delta T \frac{(1 + \tau_1 S)}{(1 + \tau_2 S)} \left(\frac{1}{1 + \tau_3 S}\right) \leq \Delta T_0 \{K_4 - K_5 \left(\frac{\tau_7 S}{1 + \tau_7 S}\right)\left(\frac{1}{1 + \tau_6 S}\right) T - K_6 [T \left(\frac{1}{1 + \tau_6 S}\right) - T''] - f_2(\Delta I)\}$$

Where: ΔT = Measured ΔT;

$\frac{1 + \tau_1 S}{1 + \tau_2 S}$ = Lead-lag compensator on measured ΔT;

τ_1, τ_2 = Time constants utilized in lead-leg compensator for ΔT, $\tau_1 \geq 8$ S, $\tau_2 \leq 3$ s;

$\frac{1}{1 + \tau_3 S}$ = Lag compensator on measured ΔT;

τ_3 = Time constants utilized in the lag compensator for ΔT, τ_3=0s;

ΔT_0 = Indicated ΔT at RATED THERMAL POWER;

K_4 $\leq$ 1.08,

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

$\frac{\tau_7 S}{1 + \tau_7 S}$ = The function generated by the rate-lag compensator for T_{avg} dynamic compensation,

τ_7 = Time constants utilized in the rate-lag compensator for T_{avg}, $\tau_7 \geq 10$ s,

$\frac{1}{1 + \tau_6 S}$ = Lag compensator on measured T_{avg};

THIS PAGE BECOMES APPLICABLE FOLLOWING SHUTDOWN FROM UNIT 2 CYCLE 2 OPERATION.

TABLE 2.2-1 (Continued)

TABLE NOTATIONS (Continued)

NOTE 3: (Continued)

τ_6	=	Time constant utilized in the measured T_{avg} lag compensator, $\tau_6 = 0$ s;
K_6	$\geq$	0.0020/°F for $T > T''$ and $K_6 = 0$ for $T \leq T''$.
T	=	Average Temperature, °F;
T"	=	Indicated T_{avg} at RATED THERMAL POWER (Calibration temperature for ΔT instrumentation, $\leq$ 588.4°F),
S	=	Laplace transform variable, s^{-1}; and
$f_2(\Delta I)$	=	0 for all ΔI.

NOTE 4: The channel's maximum Trip Setpoint shall not exceed its computed Trip Setpoint by more than 1.9% of ΔT span.

THIS PAGE BECOMES APPLICABLE FOLLOWING SHUTDOWN FROM UNIT 2 CYCLE 2 OPERATION.

2.1 SAFETY LIMITS

BASES

2.1.1 REACTOR CORE

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 correlations which have been developed to predict the DNB flux and the location of DNB for axially uniform and nonuniform heat flux distributions. The local DNB heat flux ratio (DNBR) is defined as the ratio of the heat flux that would cause DNB at a particular core location to the local heat flux and is indicative of the margin to DNB.

The DNB thermal design criterion is that the probability that DNB will not occur on the most limiting rod is at least 95% (at a 95% confidence level) for any Condition I or II event.

In meeting the DNB design criterion, uncertainties in plant operating parameters, nuclear and thermal parameters, fuel fabrication parameters, and computer codes must be considered. As described in the FSAR, the effects of these uncertainties have been statistically combined with the correlation uncertainty. Design limit DNBR values have been determined that satisfy the DNB design criterion.

Additional DNBR margin is maintained by performing the safety analyses to a higher DNBR limit. This margin between the design and safety analysis limit DNBR values is used to offset known DNBR penalties (e.g., rod bow and transition core) and to provide DNBR margin for operating and design flexibility.

The curves of Figure 2.1-1 show reactor core safety limits for a range of THERMAL POWER, REACTOR COOLANT SYSTEM pressure, and average temperature which satisfy the following criteria:

A. The average enthalpy at the vessel exit is less than the enthalpy of saturated liquid (far left line segment in each curve).

B. The minimum DNBR satisfies the DNB design criterion (all the other line segments in each curve). The VANTAGE 5 fuel is analyzed using the WRB-2 correlation with design limit DNBR values of 1.24 and 1.23 for the typical cell and thimble cells, respectively. The LOPAR fuel is analyzed using the WRB-1 correlation with design limit DNBR values of 1.23 and 1.22 for the typical and thimble cells, respectively.

C. The hot channel exit quality is not greater than the upper limit of the quality range (including the effect of uncertainties) of the DNB correlations. This is not a limiting criterion for this plant.

THIS PAGE BECOMES APPLICABLE FOLLOWING SHUTDOWN FROM UNIT 2 CYCLE 2 OPERATION.

BORATED WATER SOURCE - SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.1.2.5 As a minimum, one of the following borated water sources shall be OPERABLE:

a. A Boric Acid Storage Tank with:

1) A minimum contained borated water volume of 9504 gallons (19% of instrument span) (LI-102A, LI-104A),

2) A boron concentration between 7000 ppm and 7700 ppm, and

3) A minimum solution temperature of 65°F (TI-0103).

b. The refueling water storage tank (RWST) with:

1) A minimum contained borated water volume of 99404 gallons (9% of instrument span) (LI-0990A&B, LI-0991A&B, LI-0992A, LI-0993A),

2) A boron concentration between 2400 ppm and 2600 ppm, and

3) A minimum solution temperature of 44°F (TI-10982).

APPLICABILITY: MODES 5 and 6.

ACTION:

With no borated water source OPERABLE, suspend all operations involving CORE ALTERATIONS or positive reactivity changes.

SURVEILLANCE REQUIREMENTS

4.1.2.5 The above required borated water source shall be demonstrated OPERABLE:

a. At least once per 7 days by:

1) Verifying the boron concentration of the water,

2) Verifying the contained borated water volume, and

3) When the boric acid storage tank is the source of borated water and the ambient temperature of the boric acid storage tank room (TISL-20902, TISL-20903) is ≤72°F, verify the boric acid storage tank solution temperature is ≥65°F.

b. At least once per 24 hours by verifying the RWST temperature (TI-10982) when it is the source of borated water and the outside air temperature is less than 40°F.

THIS PAGE BECOMES APPLICABLE FOLLOWING SHUTDOWN FROM UNIT 2 CYCLE 2 OPERATION.

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 Tank with:

1) A minimum contained borated water volume of 36674 gallons (81% of instrument span) (LI-102A, LI-104A),

2) A boron concentration between 7000 ppm and 7700 ppm, and

3) A minimum solution temperature of 65°F (TI-0103).

b. The refueling water storage tank (RWST) with:

1) A minimum contained borated water volume of 631478 gallons (86% of instrument span) (LI-0990A&B, LI-0991A&B, LI-0992A, LI-0993A),

2) A boron concentration between 2400 ppm and 2600 ppm,

3) A minimum solution temperature of 44°F,

4) A maximum solution temperature of 116°F (TI-10982), and

5) RWST Sludge Mixing Pump Isolation Valves capable of closing on RWST low-level.

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

ACTION:

a. With the Boric Acid Storage Tank inoperable and being used as one of the above required borated water sources, restore the tank to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and borated to a SHUTDOWN MARGIN as required by Figure 3.1-2 at 200°F; restore the Boric Acid Storage Tank to OPERABLE status within the next 7 days or be in COLD SHUTDOWN within the next 30 hours.

b. With the RWST inoperable, except for the Sludge Mixing Pump Isolation Valves, restore the tank to OPERABLE status within 1 hour or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

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LIMITING CONDITION FOR OPERATION (Continued)

ACTION (Continued)

c. With a Sludge Mixing Pump Isolation Valve(s) inoperable, restore the valve(s) to OPERABLE status within 24 hours or isolate the sludge mixing system by either closing the manual isolation valves or deenergizing the OPERABLE solenoid pil valve within 6 hours and maintain closed.

SURVEILLANCE REQUIREMENTS

4.1.2.6 Each borated water source shall be demonstrated OPERABLE:

a. At least once per 7 days by:

1) Verifying the boron concentration in the water,

2) Verifying the contained borated water volume of the water source, and

3) When the boric acid storage tank is the source of borated water and the ambient temperature of the boric acid storage tank room (TISL-20902, TISL-20903) is $\leq$ 72°F, verify the boric acid storage tank solution temperature is $\geq$ 65°F.

b. At least once per 24 hours by verifying the RWST temperature (TI-10982) when the outside air temperature is less than 40°F.

c. At least once per 18 months by verifying that the Sludge Mixing Pump Isolation Valves automatically close upon RWST low-level test signal.

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REACTIVITY CONTROL SYSTEMS

ROD DROP TIME

LIMITING CONDITION FOR OPERATION

3.1.3.4 The individual shutdown and control rod drop time from the physical fully withdrawn position shall be less than or equal to 2.7 seconds from beginning of decay of stationary gripper coil voltage to dashpot entry with:

a. T_{avg} (TI-0412, TI-0422, TI-0432, TI-0442) 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 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 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.

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3/4.2 POWER DISTRIBUTION LIMITS

3/4.2.1 AXIAL FLUX DIFFERENCE

LIMITING CONDITION FOR OPERATION

3.2.1 The indicated (NI-0041B, NI-0042B, NI-0043B, NI-0044B) AXIAL FLUX DIFFERENCE (AFD) shall be maintained within the limits specified in the CORE OPERATING LIMITS REPORT (COLR).

APPLICABILITY: MODE 1 ABOVE 50 PERCENT RATED THERMAL POWER*.

ACTION:

a. With the indicated AXIAL FLUX DIFFERENCE outside of the limits specified in the COLR,

   1. Either restore the indicated AFD to within the 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 percent of RATED THERMAL POWER within the next 4 hours.

b. THERMAL POWER shall not be increased above 50% of RATED THERMAL POWER unless the indicated AFD is within the limits specified in the COLR.

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:

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

   2) At least once per hour until the AFD Monitor Alarm is updated after restoration to OPERABLE status.

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.

c. The provisions of Specification 4.0.4 are not applicable.

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 its limits.

*See Special Test Exceptions Specification 3.10.2.

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SURVEILLANCE REQUIREMENTS

4.2.2.1 The provisions of Specifications 4.0.4 are not applicable.

4.2.2.2 $F_Q(Z)$ shall be evaluated to determine if it 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. Determining the computed heat flux hot channel factor, $F_Q^C(Z)$, as follows:

Increase the measured $F_Q(Z)$ obtained from the power distribution map by 3% to account for manufacturing tolerances and further increase the value by 5% to account for measurement uncertainties.

c. Verifying that $F_Q^C(Z)$, obtained in Specification 4.2.2.2b above, satisfies the relationship in Specification 3.2.2.

d. Satisfying the following relationship:

$$F_Q^C(Z) \leq \frac{F_Q^{RTP} \times K(Z)}{P \times W(Z)} \text{ for } P > 0.5$$

$$F_Q^C(Z) \leq \frac{F_Q^{RTP} \times K(Z)}{0.5 \times W(Z)} \text{ for } P \leq 0.5$$

Where $F_Q^C(Z)$ is obtained in Specification 4.2.2.2b above, F_Q^{RTP} is the F_Q limit, $K(Z)$ is the normalized $F_Q(Z)$ as a function of core height, P is the fraction of RATED THERMAL POWER, and $W(Z)$ is the cycle dependent function that accounts for power distribution transients encountered during normal operation. F_Q^{RTP}, $K(Z)$, and $W(Z)$ are specified in the CORE OPERATING LIMITS REPORT as per Specification 6.8.1.6.

e. Measuring $F_Q(Z)$ according to the following schedule:

1. Upon achieving equilibrium conditions after exceeding by 20% or more of RATED THERMAL POWER, the THERMAL POWER at which FQ(Z) was last determined*, or

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

*During power escalation after each fuel loading, power level may be increased until equilibrium conditions at any power level greater than or equal to 50% of RATED THERMAL POWER have been achieved and a power distribution map obtained.

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f. With measurements indicating

$$\underset{\text{over } Z}{\text{maximum}} \left(\frac{F_Q^C(Z)}{K(Z)} \right)$$

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

1) Increase $F_Q^C(Z)$ by 2% and verify that this value satisfies the relationship in Specification 4.2.2.2d, or

2) $F_Q^C(Z)$ shall be measured at least once per 7 Effective Full Power Days until two successive maps indicate that

$$\underset{\text{over } Z}{\text{maximum}} \left(\frac{F_Q^C(Z)}{K(Z)} \right) \text{ is not increasing.}$$

g. With the relationships specified in Specification 4.2.2.2d above not being satisfied:

1) Calculate the percent $F_Q(Z)$ exceeds its limits by the following expression:

$$\left\{ \left(\underset{\text{over } Z}{\text{maximum}} \left[\frac{F_Q^C(Z) \times W(Z)}{\frac{F_Q^{RTP}}{P} \times K(Z)} \right] - 1 \right\} \times 100 \text{ for } P > 0.5$$

$$\left\{ \left(\underset{\text{over } Z}{\text{maximum}} \left[\frac{F_Q^C(Z) \times W(Z)}{\frac{F_Q^{RTP}}{0.5} \times K(Z)} \right] - 1 \right\} \times 100 \text{ for } P \leq 0.5, \text{ and}$$

2) The following action shall be taken:

Within 15 minutes, control the AFD to within new AFD limits which are determined by reducing the AFD limits specified in the CORE OPERATING LIMITS REPORT by 1% AFD for each percent $F_Q(Z)$ exceeds its limits as determined in Specification 4.2.2.2g.1. Within 8 hours, reset the AFD alarm setpoints to these modified limits.

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h. The limits specified in Specification 4.2.2.2c are applicable in all core plane regions, i.e., 0 - 100%, inclusive.

i. The limits specified in Specifications 4.2.2.2d, 4.2.2.2f, and 4.2.2.2g 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 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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3/4.2.5 DNB PARAMETERS

LIMITING CONDITION FOR OPERATION

3.2.5 The following DNB-related parameters shall be maintained within the limits:

a. Reactor Coolant System T_{avg} (TI-0412, TI-0422, TI-0432, TI-0442), $\leq$ 592.5°F

b. Pressurizer Pressure (PI-0455A,B&C, PI-0456 & PI-0456A, PI-0457 & PI-0457A, PI-0458 & PI-0458A), $\geq$ 2199 psig*

c. Reactor Coolant System Flow (FI-0414, FI-0415, FI-0416, FI-0424, FI-0425, FI-0426, FI-0434, FI-0435, FI-0436, FI-0444, FI-0445, FI-0446) $\geq$ 391,225 gpm**

APPLICABILITY: MODE 1.

ACTION:

With any of the above parameters exceeding its limit, restore the parameter to within its limit within 2 hours or reduce THERMAL POWER to less than 5% of RATED THERMAL POWER within the next 4 hours.

SURVEILLANCE REQUIREMENTS

4.2.5.1 Reactor Coolant System T_{avg} and Pressurizer Pressure shall be verified to be within their limits at least once per 12 hours. RCS flow rate shall be monitored for degradation at least once per 12 hours. In the event of flow degradation, RCS flow rate shall be determined by precision heat balance within 7 days of detection of flow degradation.

4.2.5.2 The RCS flow rate indicators shall be subjected to CHANNEL CALIBRATION at each fuel loading and at least once per 18 months.

4.2.5.3 After each fuel loading, the RCS flow rate shall be determined by precision heat balance prior to operation above 75% RATED THERMAL POWER. The RCS flow rate shall also be determined by precision heat balance at least once per 18 months. Within 7 days prior to performing the precision heat balance flow measurement, the instrumentation used for performing the precision heat balance shall be calibrated. The provisions of 4.0.4 are not applicable for performing the precision heat balance flow measurement.

*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.

**Includes a 2.2% flow measurement uncertainty.

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TABLE 3.3-3 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TOTAL ALLOWANCE (TA)	Z	SENSOR ERROR (S)	TRIP SETPOINT	ALLOWABLE VALUE
7. Semi-Automatic Switchover to Containment Emergency Sump (Continued)					
b. RWST Level--Low-Low Coincident With Safety Injection (LI-0990A&B, LI-0991A&B, LI-0992A, LI-0993A)	3.5	0.71	1.67	≥ 275.3 in. from tank base (≥ 39.1% of instrument span)	≥ 264.9 in. from tank base (≥ 37.4% of instrument span)
8. Loss of Power to 4.16 kV ESF Bus					
a. 4.16 kV ESF Bus Undervoltage-Loss of Voltage	N.A.	N.A.	N.A.	≥ 2975 volts with a ≤ 0.8 second time delay.	≥ 2912 volts with a ≤ 0.8 second time delay.
b. 4.16 kV ESF Bus Undervoltage-Degraded Voltage	N.A.	N.A.	N.A.	≥ 3746 volts with a ≤ 20 second time delay.	≥ 3683 volts with a ≤ 20 second time delay.
9. Engineered Safety Features Actuation System Interlocks					
a. Pressurizer Pressure, P-11 (PI-0455A,B&C, PI-0456 & PI-0456A, PI-0457 & PI-0457A, PI-0458 & PI-0458A).	N.A.	N.A.	N.A.	≤ 2000 psig	≤ 2010 psig
b. Reactor Trip, P-4	N.A.	N.A.	N.A.	N.A	N.A.

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3/4.5 EMERGENCY CORE COOLING SYSTEMS

3/4.5.1 ACCUMULATORS

LIMITING CONDITION FOR OPERATION

3.5.1 Each Reactor Coolant System (RCS) accumulator shall be OPERABLE with:

a. The isolation valve open,

b. A contained borated water volume of between 6555 (29.2% of instrument span) and 6909 gallons (70.7% of instrument span) (LI-0950, LI-0951, LI-0952, LI-0953, LI-0954, LI-0955, LI-0956, LI-0957),

c. A boron concentration of between 1900 ppm and 2600 ppm, and

d. A nitrogen cover-pressure of between 617 and 678 psig. (PI-0960A&B, PI-0961A&B, PI-0962A&B, PI-0963A&B, PI-0964A&B, PI-0965A&B, PI-0966A&B, PI-0967A&B)

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 1 hour or be in at least HOT STANDBY within the next 6 hours and reduce pressurizer pressure to less than 1000 psig 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 6 hours and reduce pressurizer pressure to less than 1000 psig within the following 6 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 (HV-8808A, B, C, D).

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*Pressurizer pressure above 1000 psig.

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3/4.5.4 REFUELING WATER STORAGE TANK

LIMITING CONDITION FOR OPERATION

3.5.4 The refueling water storage tank (RWST) shall be OPERABLE with:

a. A minimum contained borated water volume of 631,478 gallons (86% of instrument span) (LI-0990A&B, LI-0991A&B, LI-0992A, LI-0993A).

b. A boron concentration of between 2400 ppm and 2600 ppm of boron,

c. A minimum solution temperature of 44°F, and

d. A maximum solution temperature of 116°F (TI-10982).

e. RWST Sludge Mixing Pump Isolation valves capable of closing on RWST low-level.

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

ACTION:

a. With the RWST inoperable except for the Sludge Mixing Pump Isolation Valves, restore the tank to OPERABLE status within 1 hour or be in at least HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With a Sludge Mixing Pump Isolation Valve(s) inoperable, restore the valve(s) to OPERABLE status within 24 hours or isolate the sludge mixing system by either closing the manual isolation valves or deenergizing the OPERABLE solenoid pilot valve within 6 hours and maintain closed.

SURVEILLANCE REQUIREMENTS

4.5.4 The RWST shall be demonstrated OPERABLE:

a. At least once per 7 days by:

1) Verifying the contained borated water volume in the tank, and

2) Verifying the boron concentration of the water.

b. At least once per 24 hours by verifying the RWST temperature when the outside air temperature is less than 40°F.

c. At least once per 18 months by verifying that the sludge mixing pump isolation valves automatically close upon an RWST low-level test signal.

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3/4.2 POWER DISTRIBUTION LIMITS

BASES

The specifications of this section provide assurance of fuel integrity during Condition I (Normal Operation) and II (Incidents of Moderate Frequency) events by: (1) meeting the DNB design criterion 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; and

$F^N_{\Delta H}$ 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

3/4.2.1 AXIAL FLUX DIFFERENCE

The limits on AXIAL FLUX DIFFERENCE (AFD) assure that the $F_Q(Z)$ upper bound envelope of the F_Q limit specified in the CORE OPERATING LIMITS REPORT (COLR) times K(Z) is not exceeded during either normal operation or in the event of xenon redistribution following power changes.

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 1-minute average of each of the OPERABLE excore detector outputs and provides an alarm message immediately if the AFD for two or more OPERABLE excore channels are outside the allowed ΔI power operating space for RAOC operation specified within the COLR and the THERMAL POWER is greater than 50% of RATED THERMAL POWER.

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AXIAL FLUX DIFFERENCE (Continued)

3/4.2.2 and 3/4.2.3 HEAT FLUX HOT CHANNEL FACTOR AND NUCLEAR ENTHALPY RISE HOT CHANNEL FACTOR - $F^N_{\Delta H}$

The limits on heat flux hot channel factor and nuclear enthalpy rise hot channel factor ensure that: (1) the design limit on peak local power density is not exceeded, (2) the DNB design criterion is met, and (3) 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 ensure that the limits are maintained provided:

a. Control rods in a single group move together with no inidivdual rod insertion differing by more than ± 12 steps, indicated, from the group demand position;

b. Control rod groups are sequenced with a constant tip-to-tip distance between banks as described in Specification 3.1.3.6;

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

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. 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.

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.

The heat flux hot channel factor $F_Q(Z)$ is measured periodically and increased by a cycle and height dependent power factor appropriate to RAOC operation, W(Z), to provide assurance that the limit on the heat flux hot channel factor, $F_Q(Z)$, is met. W(Z) accounts for the effects of normal operation transients within the AFD band and was determined from expected power control maneuvers over the full range of burnup conditions in the core. The W(Z) function for normal operation and the AFD band are provided in the CORE OPERATING LIMITS REPORT per Specification 6.8.1.6.

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

When $F_{\Delta H}^{N}$ is measured, (i.e., inferred), measurement uncertainty (i.e., the appropriate uncertainty on the incore inferred hot rod peaking factor) must be allowed for and 4% is the appropriate allowance for a full core map taken with the incore detection system.

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 limit of 1.02 was selected to provide an allowance for the uncertainty associated with the indicated power tilt.

The 2-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% for each percent of tilt in excess of 1.

For purposes of monitoring QUADRANT POWER TILT RATIO when one excore detector is inoperable, the moveable 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 four symmetric thimbles. The two sets of four symmetric thimbles is a unique set of eight detector locations. 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 with the initial FSAR assumptions and have been analytically demonstrated adequate to meet the DNB design criterion throughout each analyzed transient. The indicated T_{avg} value of 592.5°F and the indicated pressurizer pressure value of 2199 psig correspond to analytical limits of 594.4°F and 2185 psig respectively, with allowance for measurement uncertainty.

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3/4.2.5 DNB PARAMETERS (Continued)

The 12-hour periodic surveillance of these parameters through instrument readout is sufficient to ensure that the parameters are restored within their limits following load changes and other expected transient operation. The 18 month periodic measurement of the RCS total flow rate is adequate to detect flow degradation and ensure correlation of the flow indication channels with measured flow such that the indicated percent flow will provide sufficient verification of the flow rate degradation on a 12 hour basis. A change in indicated percent flow which is greater than the instrument channel inaccuracies and parallax errors is an appropriate indication of RCS flow degradation.

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ECCS SUBSYSTEMS (Continued)

The limitation for all safety injection pumps to be inoperable below 350°F provides assurance that a mass addition pressure transient can be relieved by the operation of a single PORV.

The Surveillance Requirements provided to ensure OPERABILITY of each component ensure that at a minimum, the assumptions used in the safety analyses are met and that subsystem OPERABILITY is maintained. Surveillance Requirements for throttle valve position stops and flow balance testing provide assurance that proper ECCS flows will be maintained in the event of a LOCA. Maintenance of proper flow resistance and pressure drop in the piping system to each injection point is necessary to: (1) prevent total pump flow from exceeding runout conditions when the system is in its minimum resistance configuration, (2) provide the proper flow split between injection points in accordance with the assumptions used in the ECCS-LOCA analyses, (3) provide an acceptable level of total ECCS flow to all injection points equal to or above that assumed in the ECCS-LOCA analyses and (4) to ensure that centrifugal charging pump injection flow which is directed through the seal injection path is less than or equal to the amount assumed in the safety analysis. The surveillance requirements for leakage testing of ECCS check valves ensure a failure of one valve will not cause an intersystem LOCA. In MODE 3, with either HV-8809A or B closed for ECCS check valve leak testing, adequate ECCS flow for core cooling in the event of a LOCA is assured.

3/4.5.4 REFUELING WATER STORAGE TANK

The OPERABILITY of the Refueling Water Storage Tank (RWST) as part of the ECCS ensures that sufficient negative reactivity is injected into the core to counteract any positive increase in reactivity caused by RCS cooldown. RCS cooldown can be caused by inadvertent depressurization, a loss-of-coolant accident, or a steam line rupture.

The limits on RWST minimum volume and boron concentration ensure that 1) sufficient water is available within containment to permit recirculation cooling flow to the core, 2) the reactor will remain subcritical in the cold condition following a small LOCA or steamline break, assuming complete mixing of the RWST, RCS, and ECCS water volumes with all control rods inserted except the most reactive control assembly (ARI-1), and 3) the reactor will remain subcritical in the cold condition following a large break LOCA assuming complete mixing of the RWST, RCS, ECCS water and other sources of water that may eventually reside in the sump, post-LOCA with all control rods inserted except for the two most reactive control assemblies.

The contained water volume limit includes an allowance for water not usable because of tank discharge line location or other physical characteristics.

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SEMIANNUAL RADIOACTIVE EFFLUENT RELEASE REPORT (Continued)

The Semiannual Radioactive Effluent Release Reports shall also include the following: an explanation as to why the inoperability of liquid or gaseous effluent monitoring instrumentation was not corrected within the time specified in Specification 3.3.3.9 or 3.3 3.10, respectively; and description of the events leading to liquid holdup tanks or gas storage tanks exceeding the limits of Specification 3.11.1.4 or 3.11.2.6, respectively.

MONTHLY OPERATING REPORTS

6.8.1.5 Routine reports of operating statistics and shutdown experience, including documentation of all challenges to the PORVs 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 Administrator of the Regional Office of the NRC, no later than the 15th of each month following the calendar month covered by the report.

CORE OPERATING LIMITS REPORT

6.8.1.6 Core operating limits shall be established and documented in the CORE OPERATING LIMITS REPORT (COLR) before each reload cycle or any remaining part of a reload cycle for the following:

a. SHUTDOWN MARGIN LIMIT FOR MODES 1 and 2 for Specification 3/4.1.1.1,

b. SHUTDOWN MARGIN LIMITS FOR MODES 3, 4, and 5 for Specification 3/4.1.1.2,

c. Moderator temperature coefficient BOL and EOL limits and the 300-ppm surveillance limit for Specification 3/4.1.1.3,

d. Shutdown Rod Insertion Limits for Specification 3/4.1.3.5,

e. Control Rod Insertion Limits for Specification 3/4.1.3.6,

f. Axial Flux Difference Limits for Specification 3/4.2.1,

g. Heat Flux Hot Channel Factor, K(Z) and W(Z), for Specification 3/4.2.2,

h. Nuclear Enthalpy Rise Hot Channel Factor Limit and the Power Factor Multiplier for Specification 3/4.2.3.

The analytical methods used to determine the core operating limits shall be those previously approved by the NRC in:

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CORE OPERATING LIMITS REPORT (Continued)

a. WCAP-9272-P-A, "WESTINGHOUSE RELOAD SAFETY EVALUATION METHODOLOGY," July 1985 (W Proprietary).
(Methodology for Specifications 3.1.1.3 - Moderator Temperature Coefficient, 3.1.3.5 - Shutdown Bank Insertion Limit, 3.1.3.6 - Control Bank Insertion Limits, and 3.2.3 - Nuclear Enthalpy Rise Hot Channel Factor.)

b. WCAP-10216-P-A, "RELAXATION OF CONSTANT AXIAL OFFSET CONTROL F_Q SURVEILLANCE TECHNICAL SPECIFICATION," June 1983 (W Proprietary).
(Methodology for Specifications 3.2.1 - Axial Flux Difference (Relaxed Axial Offset Control) and 3.2.2 - Heat Flux Hot Channel Factor (W(Z) surveillance requirements for F_Q Methodology).)

c. WCAP-9220-P-A, Rev. 1, "WESTINGHOUSE ECCS EVALUATION MODEL-1981 VERSION," February 1982 (W Proprietary).
(Methodology for Specification 3.2.2 - Heat Flux Hot Channel Factor.)

The core operating limits shall be determined so that all applicable limits (e.g., fuel thermal-mechanical limits, core thermal-hydraulic limits, ECCS limits, nuclear limits such as shutdown margin, and transient and accident analysis limits) of the safety analysis are met.

The CORE OPERATING LIMITS REPORT, including any mid-cycle revisions or supplements thereto, shall be provided upon issuance, for each reload cycle, to the NRC Document Control Desk with copies to the Regional Administrator and Resident Inspector.

SPECIAL REPORTS

6.8.2 Special reports shall be submitted to the Regional Administrator of the Regional Office of the NRC within the time period specified for each report.

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Enclosure 4

Vogtle Electric Generating Plant Units 1 and 2

Request for Technical Specifications Changes

VANTAGE-5 Fuel Design

Safety Evaluation Report

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1.0 INTRODUCTION AND CONCLUSIONS

The Vogtle Electric Generating Plant (VEGP) Units 1 and 2 are currently operating with a Westinghouse 17x17 low-parasitic (LOPAR) fueled core. For subseouent cycles, it is planned to refuel and operate the VEGP Units 1 and 2 with the Westinghouse VANTAGE 5 improved fuel design. As a result, future core loadings would range from approximately 55% LOPAR, and 45% VANTAGE 5 transition cores 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 1.

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 2. The initial irradiation of a fuel region containing all the VANTAGE 5 design features occurred in the Callaway Plant in November 1987. The Callaway VANTAGE 5 licensing submittal was made to the NRC on March 31, 1987 (ULNRC-1470, Docket No. 50-483). NRC approval was received in October 1987. 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 operating in Westinghouse plants. The VEGP Units 1 and 2 will be operating in reload Cycles 4 and 3, respectively, with fuel containing the following features: Integral Fuel Burnable Absorbers (IFBAs), Intermediate Flow Mixers (IFM) grids, Reconstitutable Top Nozzles (RTN), (currently operating in both VEGP Units 1 and 2), and fuel assemblies modified for extended burnup (currently operating in both VEGP Units 1 and 2). In addition, both the VEGP Units 1 and 2 fuel assemblies are currently operating with the Debris Filter Bottom Nozzle (DFBN). The axial blankets are optional on the first transition Cycle, i.e., Vogtle Unit 1, Cycle 4 and Vogtle Unit 2, Cycle 3.

A brief summary of the VANTAGE 5 design features and the major advantages of the improved fuel design are given below. These features and figures illustrating the VANTAGE 5 design are presented in more detail in Section 2.0.

Integral Fuel Burnable Absorber (IFBA) - The IFBA features a thin boride coating on the fuel pellet surface in the central portion of the enriched UO_2 pellets. In a typical reload core, approximately forty percent 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) Grid - Three IFM grids located between the four upper most zircaloy grids provide increased DNB margin. Increased margin permits an increase in the design basis $F_{\Delta H}$ and F_Q.

Reconstitutable Top Nozzle (RTN) - 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. For VANTAGE 5 reload cores, low leakage loading patterns (burned blankets) are shown to further improve uranium utilization and provide additional pressurized thermal shock margin.

This submittal is to serve as a reference safety evaluation/analysis report for the region-by-region reload transition from the present VEGP LOPAR fueled core to an all VANTAGE 5 fueled core. The submittal examines the differences between the VANTAGE 5 and LOPAR fuel assembly designs and evaluates the effect of these differences on the core performance during the transition to an all VANTAGE 5 core. The VANTAGE 5 core evaluation/analyses were performed at a core thermal power level of 3565 MWt for Units 1 and 2 with the following conservative assumptions made in the safety evaluations: a full power $F_{\Delta H}$ of 1.65 for the VANTAGE 5 fuel and 1.57 for the LOPAR fuel, and 1.70 both for the VANTAGE 5 fuel non-LOCA analyses and the small break LOCA analysis, an increase in the maximum F_Q to 2.50, 10% plant total steam generator tube plugging for both Units 1 and 2, and a core bypass flow of 8.4% with thimble plugs removed. The analysis assumption of core bypass flow with thimble plugs removed is conservative for operation with thimble plugs and/or Wet Annular Burnable Absorber (WABA) rods. The axial offset strategy will be the licensed Relaxed Axial Offset Control (RAOC) with F_Q surveillance. RAOC uses a +10/-20% Axial Flux Difference (AFD) band at 100% RTP for the safety evaluations.

The standard reload design methods described in Reference 4 will be used as a basic reference document in support of future VEGP Units 1 and 2 Reload Safety Evaluations (RSE) with VANTAGE 5 fuel reloads. Sections 2.0 through 5.0 summarize the Mechanical, Nuclear, Thermal and Hydraulic, and Accident Evaluations, respectively. Section 6.0 gives a summary of the Technical Specifications changes needed.

Consistent with the Westinghouse standard reload methodology, Reference 4, 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 RTSR.

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 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 V. C. Summer 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 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 which was completed in November 1988 (EOC burnup 46,000 MWD/MTU). The observed behavior of the four assemblies was as good as that observed at the end of the first and second cycles 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 contained 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 was irradiated for three reactor cycles and showed good mechanical integrity.

The following plants are currently operating with regions of VANTAGE 5 fuel assemblies: Callaway, V. C. Summer, Shearon Harris, Diablo Canyon and Byron/Braidwood.

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

1. The Westinghouse VANTAGE 5 reload fuel assemblies for the VEGP Units 1 and 2 are mechanically compatible with the current LOPAR fuel assemblies, control rods, and reactor internals interfaces. The VANTAGE 5/LOPAR fuel assemblies satisfy the current design BASES for the VEGP Units 1 and 2.

2. The structural integrity of the 17x17 VANTAGE 5 fuel assembly design for seismic/LOCA loadings has been evaluated for both VEGP Units 1 and 2. Evaluation of the 17x17 VANTAGE 5 fuel assembly component stresses and grid impact forces due to postulated faulted condition accidents verified that the VANTAGE 5 fuel assembly design is structurally acceptable.

3. Taking credit for Leak-Before-Break technology, it has been demonstrated that RCCA insertion will occur during a LOCA with a limited displacement break. This allows for consideration of RCCA insertion when performing the evaluation to demonstrate that the core will remain subcritical during the post-LOCA long-term core cooling phase of the event.

4. Change. in the nuclear characteristics due to the transition from LOPAR to VANTAGE 5 fuel will be typical of the normal cycle-to-cycle variations experienced as loading patterns change.

5. The reload VANTAGE 5 fuel assemblies are hydraulically compatible with the LOPAR fuel assemblies from previous cycles of operation.

6. The core design and safety analyses results documented in this report show the core's capability for operating safely for the rated VEGP Units 1 and 2 design thermal power with $F_{\Delta H}$ of 1.65 for the VANTAGE 5 fuel and 1.57 for the LOPAR fuel, F_Q of 2.50 and steam generator tube plugging levels up to 10%. The analysis was performed at 3565 MWt which is conservative compared to the current rated thermal power, 3411 MWt.

7. The steam generator tube rupture analysis to support the transition to VANTAGE 5 fuel shows that offsite doses for the VEGP Units 1 and 2 are well within the allowable guidelines specified in the SRP (NUREG-0800), FSAR Chapter 15.6.3 and 10CFR100.

8. The projected increase in the fuel burnup levels with the use of VANTAGE 5 fuel has a negligible effect on the radiological consequences of accidents due to the very small changes in the core inventory of fission products.

9. The previously reviewed licensing basis continues to be met when the VEGP Units 1 and 2 are reloaded with VANTAGE 5 fuel. Plant operating limitations given in the Technical Specifications will be satisfied with the proposed changes noted in Enclosure 3 of this submittal. A reference is established upon which to base Westinghouse reload safety evaluations for future reloads with VANTAGE 5 fuel.

?.O MECHANICAL EVALUATION

Introduction and Summary

This section evaluates the mechanical design and the compatibility of the 17x17 VANTAGE 5 fuel assembly with the current low-parasitic (LOPAR) fuel assemblies during the transition through mixed-fueled 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 refueling equipment. The VANTAGE 5 design is intended to replace and be compatible with cores containing fuel of the LOPAR design. The VANTAGE 5 design dimensions are essentially equivalent to the current VEGP Units 1 and 2 LOPAR fuel assembly design from an exterior assembly envelope and reactor internals interface standpoint. References in this section are made to WCAP-10444-P-A, "VANTAGE 5 Fuel Assembly Reference Core Report," Reference 2, and to WCAP-9500-A, "Reference Core Report 17x17 Optimized Fuel Assembly," Reference 1.

The significant new mechanical features of the VANTAGE 5 design relative to the initial core/Cycle 1 LOPAR fuel design for both VEGP Units 1 and 2 include the following:

- o Integral Fuel Burnable Absorber (IFBA)
- o Intermediate Flow Mixer (IFM) Grids
- o Reconstitutable Top Nozzle (RTN)
- o Extended burnup capability including slightly longer fuel rods
- o Axial blankets
- o Replacement of six intermediate inconel grids with zircaloy grids
- o Reduction in fuel rod, guide thimble and instrumentation tube diameter
- o Redesigned fuel rod bottom end plug to facilitate reconstitution capability
- o Snag-resistant inconel grids (top and bottom)
- o Debris Filter Bottom Nozzle (DFBN)

The RTN, DFBN, redesigned fuel rod bottom end plug, snag-resistant grid, and the fuel assembly extended burnup modification have been introduced previously in both VEGP Units 1 and 2. These features will continue to be utilized in the VANTAGE 5 design. Table 2-1 provides a comparison of the LOPAR and VANTAGE 5 fuel assembly design parameters.

Fuel Rod Performance

Fuel rod design evaluations for VEGP Units 1 and 2 were performed using the NRC approved models, References 5 and 6, and the extended burnup design methods in Reference 3. Fuel rod performance for all VEGP fuel is shown to satisfy the NRC Standard Review Plan (SRP) fuel rod design basis on a region by region basis. These same BASES are applicable to all fuel rod designs, including the LOPAR and VANTAGE 5 fuel designs, with the only difference being that the 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 2.

There is no effect from a fuel rod design standpoint 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 (such as the presence of axial blankets or changes in the fuel rod and plenum length, for example). Analysis of IFBA rods includes any geometry changes necessary to model the presence of the burnable absorber, and conservatively models the gas release from the zirconium diboride pellet coating.

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 evaluated for the mechanical design of a fuel region (for example an increase in the peaking factors) are addressed for all effected fuel regions as part of the reload safety evaluation process when the plant change is to be implemented.

TABLE 2-1

Comparison of 17x17 LOPAR
and
17x17 VANTAGE 5 Fuel Assembly Design Parameters

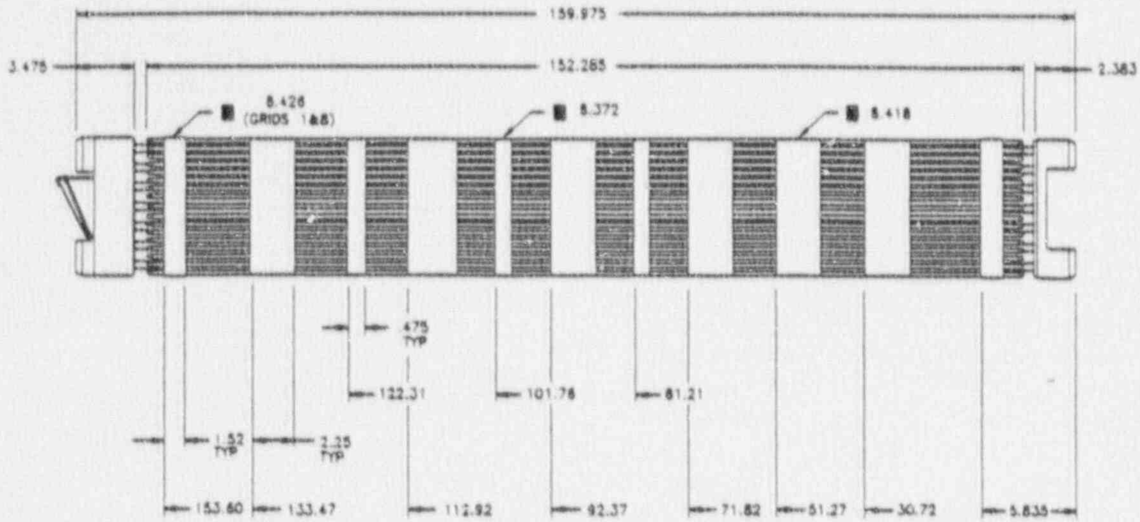
PARAMETER	17x17 LOPAR DESIGN	17x17 VANTAGE 5 DESIGN
Fuel Assy Length, in.	159.765	159.975
Fuel Rod Length, in.	151.560	152.285
Assembly Envelope, in.	8.426	8.426
Compatible with Core Internals	Yes	Yes
Fuel Rod Pitch, in.	0.496	0.496
Number of Fuel Rods/Assy.	264	264
Number/Guide Thimble Tubes/Assy.	24	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.	0.0225	0.0225
Fuel/Clad Gap. mil.	6.5	6.2
Fuel Pellet Diameter, in.	0.3225	0.3088
Fuel Pellet Length, in.	0.387	0.370

Grid Assemblies

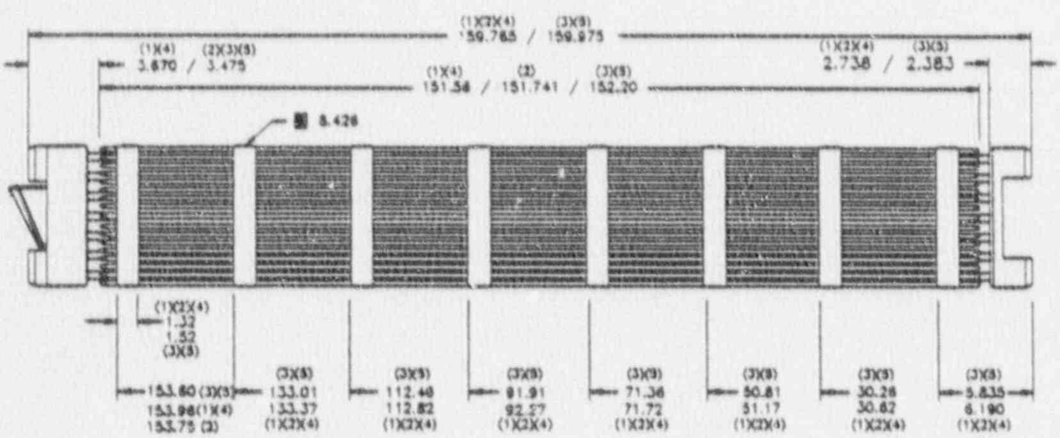
The top and bottom inconel (non-mixing vane) grids of the VANTAGE 5 fuel assemblies are similar in design to the inconel grids of the Cycle 1, LOPAR fuel assemblies for VEGP Units 1 and 2. The differences are: 1) the spring and dimple heights have been modified to accommodate the reduced diameter fuel rod, 2) the top grid spring force has been reduced to minimize rod bow, 3) the VANTAGE 5 top grid uses type 304L stainless steel sleeves instead of 304 stainless steel sleeves used for the LOPAR top grid, 4) the top and bottom grids have a snag-resistant design which minimizes assembly interactions during core loading/unloading, 5) the top and bottom grids have dimples which are rotated 90° to minimize fuel rod fretting and dimple cocking, and 6) the top and bottom grid heights have been increased to 1.522 inches. The snag-resistant design and the 304L stainless steel sleeves were introduced in Vogtle Unit 1, Cycle 2/Region 4 and in the Vogtle Unit 2 initial core/Cycle 1. Rotated dimples and 1.522 inch grid heights were introduced in Vogtle Unit 1, Cycle 3/Region 5 and Vogtle Unit 2, Cycle 2/Region 4. These features will continue to be utilized. The six intermediate (mixing vane) grids are made of zircaloy material rather than inconel which is currently used in the LOPAR design.

The IFM grids shown in Figure 2.1 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 to be 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. Impact loads during these



17X17 VANTAGE 5 FUEL ASSEMBLY



17X17 RECONSTITUTABLE LOPAR FUEL ASSEMBLY

(1) INITIAL CORE/ CYCLE 1 UNIT 1

(2) CYCLE 2/REGION 4 UNIT 1

(3) CYCLE 3/REGION 5 UNIT 1

(4) INITIAL CORE/CYCLE 1 UNIT 2

(5) CYCLE 2/REGION4 UNIT 2

Georgia Power — VOGTLE ELECTRIC GENERATING PLANT UNIT 1 AND UNIT 2	17X17 VANTAGE 5 / LOPAR FUEL ASSEMBLY COMPARISON
	FIGURE 2.1

events have been calculated for the IFM grids for typical Westinghouse reactors. A coolable geometry is, therefore, assured at the IFM grid elevation, as well as at the structural grid elevation.

Reconstitutable Top Nozzle and Debris Filter Bottom Nozzle

The RTN for the VANTAGE 5 fuel assembly differs from the welded top nozzle design in two ways: a groove is provided in each thimble thru-hole in the nozzle plate to facilitate removal, and the nozzle plate thickness is reduced to provide additional axial space for fuel rod growth. The RTN feature was previously introduced in Vogtle Unit 1, Cycle 2 and Vogtle Unit 2, Cycle 2 and will continue to be utilized.

In the VANTAGE 5 RTN design, a stainless steel nozzle insert is mechanically connected to the top nozzle adapter plate by means of a pre-formed circumferential bulge near the top of the insert. The insert engages a mating groove in the wall of the adapter plate thimble tube thru-hole. The insert has 4 equally spaced axial slots which allow the insert to deflect inwardly at the elevation of the bulge, thus permitting the installation or removal of the nozzle. The insert bulge is positively held in the adapter plate mating groove by placing a lock tube with a uniform ID identical to that of the thimble tube into the insert. The lock tube is secured in place by two means. First, a top flare creates a tight fit. Second, six non-yielding projections on the OD which interface with the concave side of the insert preclude escape during core component transfer.

The full complement of these joints comprises the structural connection (reconstitutable design feature) between the top nozzle and the remainder of the VANTAGE 5 fuel assembly. The nozzle insert-to-adapter plate bulge joints replace the uppermost grid sleeve-to-adapter plate welded joints found in current fuel assemblies. The nozzle insert-to-thimble tube multiple 4-lobe bulge joint located in the lower portion of the insert represents the structural connection between the insert and the remainder of the fuel assembly below the elevation of the insert. The uppermost grid sleeve is connected to the thimble tube by similar 4-lobe bulge joints.

To remove the top nozzle, a tool is first inserted through the lock tube and expanded radially to engage the bottom edge of the tube. An axial force is then exerted on the tool which overrides the 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 examination or replacement. Reconstitution is completed by the remounting of the nozzle and the insertion of new locked tubes. The design BASES and evaluation of the RTN are given in Section 2.3.2 in Reference 2.

The VANTAGE 5 design will include the use of the DFBN to reduce the possibility of fuel rod damage due to debris-induced fretting. For the DFBN design the relatively large flow holes in the LOPAR bottom nozzle are replaced with a new pattern of smaller flow holes. 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. The overall design of the DFBN is similar to the current design except it is shorter and has a thinner top plate to allow for fuel rod growth.

<u>Axial Blankets</u>

Although noted as a new mechanical feature of the VANTAGE 5 design and licensed in Reference 2, axial blankets have been and are currently operating in Westinghouse plants to reduce neutron leakage and improve fuel utilization. A description and design application of this feature are contained in Reference 2, Section 3.0. The axial blankets utilize a chamfered pellet physically different from the enriched pellet in the fuel stack to help prevent accidental mixing with the enriched pellet during manufacturing.

Mechanical Compatibility of Fuel Assemblies

Based on the evaluation of the VANTAGE 5/LOPAR design differences and hydraulic test results, References 1 and 2, 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 fuel assemblies.

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 7.

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 2, 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 2, 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 8, for the 17x17 OFA at EOL.

Seismic/LOCA Impact on Fuel Assemblies

An evaluation of the VANTAGE 5 fuel assembly structural integrity considering the lateral effects of LOCA and seismic loadings has been performed using time-history numerical techniques based on the Vogtle plant specific Safe Shutdown Earthquake (SSE).

The VANTAGE 5 fuel assembly is structurally equivalent to the LOPAR fuel design. The main differences between the two designs are six zircaloy grids replacing inconel mid-grids in the LOPAR design, three additional intermediate flow mixers, and optimized fuel rods. The load bearing capability for the zircaloy grids and flow mixers under the faulted condition loadings has been analyzed. The results indicated that 17x17 VANTAGE 5 grid loads are below the allowable grid strengths.

Based on the grid load results, the 17x17 VANTAGE 5 zircaloy grid is capable of maintaining the core coolable geometry under the SSE and asymmetric pipe rupture transients in either homogeneous or transition core operation. The 17x17 VANTAGE 5 fuel assembly is structurally acceptable for both VEGP Units 1 and 2. This is also true for a transition core composed of both VANTAGE 5 and LOPAR fuel assembly core configurations. The grids will not buckle due to the combined impact loads of a seismic and LOCA event. The coolable geometry requirement is met. The stresses in the fuel assembly components resulting from seismic and LOCA induced deflections are within acceptable limits.

Core Components

The core components are designed to be compatible with VANTAGE 5 and LOPAR fuel assemblies. The reduced diameter VANTAGE 5 thimble tube provides sufficient clearance for insertion of control rods, WABA rods, source rods or dually compatible thimble plugs to assure the proper operation of these core components.

3.0 NUCLEAR EVALUATION

The evaluation of the transition and equilibrium cycle VANTAGE 5 cores presented in Reference 2, as well as the VEGP specific transition core evaluation, demonstrate that the impact of implementing VANTAGE 5 does not cause a significant change to the physics characteristics of the VEGP cores beyond the normal range of variations seen from cycle-to-cycle.

The methods and core models used in the VEGP reload transition core evaluations are described in Reference 2, 4, 9, and 10. These licensed methods and models have been used for Vogtle and other previous Westinghouse reload designs using the OFA and VANTAGE 5 fuel. No changes to the nuclear design philosophy, methods, or models are necessary because of the transition to VANTAGE 5 fuel.

For the nuclear design area, the following VEGP Units 1 and 2 Technical Specifications changes are proposed:

1) Increased $F_{\Delta H}$ limits. These higher limits serve to increase nuclear design flexibility and allow loading patterns with reduced 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 (axial blankets and part length burnable absorbers).

3) Relaxed Axial Offset Control (RAOC) band implementation, and revised surveillance requirements on the heat flux hot channel factor, $F_Q(z)$. The methods used in the core analysis incorporating these Technical Specifications changes are licensed and described in Reference 9. Implementation of a RAOC band will increase plant operating flexibility, and performing surveillance directly on $F_Q(z)$, rather than $F_{xy}(z)$ the radial component of the total peaking factor, will more directly monitor the parameter of interest. 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. The Nuclear Design analysis was performed for a RAOC AFD band of +10/-20% AFD at 100% RTP and +30/-35% AFD at 50% RTP; the actual RAOC operating bands to be used for future specific reloads will be the same or narrower and will be specified in a Core Operating Limits Report.

4) Core Operating Limits Report. The AFD operating bands for RAOC operation will be specified in the Core Operating Limits Report, instead of in the Technical Specifications. This eliminates the necessity of Technical Specification amendments for future reload cycles, while providing assurance that the correct operating limits will be followed.

Power distributions and peaking factors show slight changes as a result of the incorporation of reduced length burnable absorbers, axial blankets, and increased peaking factors 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 the Vogtle reactor as it transitions to an all VANTAGE 5 core show little change relative to the range of parameters experienced for the all LOPAR fuel 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 report which serves as a basis for any significant changes which may require a future NRC review.

4.0 THERMAL AND HYDRAULIC EVALUATION

This section describes the calculational methods used for the thermal-hydraulic analysis, the DNB performance, and the hydraulic compatibility during the transition from mixed-fuel cores to an all VANTAGE 5 core. The Westinghouse transition core DNB methodology is given in References 1 and 11 and has been approved by the NRC via Reference 12. Using this methodology, transition cores are anlayzed as if the entire core consisted of one assembly type (full LOPAR or full VANTAGE 5), and the resultant DNBR values are reduced by the appropriate transition core penalty.

The LOPAR and the VANTAGE 5 fuel assemblies were shown to be hydraulically compatible in Reference 2.

The DNBR analyses for VEGP Units 1 and 2 were based on parameters which conservatively bound the licensing values. Table 4-1 summarizes the pertinent thermal and hydraulic design parameters used in the analyses as well as the licensing values.

The improved THINC-IV PWR design modeling method, Reference 13, was used for the DNBR analyses of the VANTAGE 5 and LOPAR fuel. For high core power density applications, the improved model yields more conservative values of minimum DNBR than the present model, Reference 14. No changes to the basic THINC-IV models and correlations were made for the improved core modeling scheme.

The DNBR analyses of the VANTAGE 5 fuel and LOPAR fuel are based on the Revised Thermal Design Procedure (RTDP), Reference 15. The primary DNB correlation used for the LOPAR fuel is the WRB-1 DNB correlation which is described in Reference 16. The VANTAGE 5 fuel DNBR analyses use the WRB-2 DNB correlation which is described in Reference 2. The WRB-2 DNB correlation takes credit for the reduced grid-to-grid spacing of the VANTAGE 5 fuel assembly mixing vane grids resulting from the use of the Intermediate Flow Mixer (IFM) grids. Both the WRB-1 and WRB-2 DNB correlations have a correlation limit of 1.17.

The W-3 DNB correlation, References 17 and 18, is used for both fuel types where the primary DNB correlations are not applicable. The WRB-1 and WRB-2 DNB correlations were developed based on mixing vane data and, therefore, are only applicable in the heated rod spans above the first mixing vane grid. The W-3 DNB correlation, which does not take credit for mixing vane grids, is used to calculate DNBR values in the heated region below the first mixing vane grid. In addition, the W-3 DNB correlation is applied in the analysis of accident conditions where the system pressure is below the range of the primary correlations. For system pressures in the range of 500 psia to 1000 psia, the W-3 DNB correlation limit is 1.45, Reference 19. For system pressures greater than 1000 psia, the W-3 DNB correlation limit is 1.30. A cold wall factor, Reference 20, is applied to the W-3 DNB correlation to account for the presence of the unheated thimble surfaces. Also, a 0.88 multiplier is applied to the W-3 DNB correlation to account for the 17x17 fuel rod diameter effect, Reference 21.

With the RTDP methodology, uncertainties in plant operating parameters, nuclear and thermal parameters, fuel fabrication parameters, computer codes and DNB correlation predictions are considered statistically to obtain DNB uncertainty factors. Based on the DNB uncertainty factors, RTDP design limit DNBR values are determined such that there is at least a 95 percent probability at a 95 percent confidence level that DNB will not occur on the most limiting fuel rod during normal operation and operational transients and during transient conditions arising from faults of moderate frequency (Condition I and II events as defined in ANSI N18.2).

Uncertainties in the plant operating parameters (pressurizer pressure, primary coolant temperature, reactor power, and reactor coolant system flow) have been evaluated for the VEGP Units 1 and 2 with Resistance Temperature Detector (RTD) bypass loops, Reference 22, and for the RTD bypass loops eliminated, Reference 23. In the DNBR analyses with RTDP, a set of plant operating parameter uncertainties were used which are bounding for operation with RTD bypass loops or for RTD bypass loops eliminated. Only the random portion of the plant operating parameter uncertainties is included in the statistical combination. Instrumentation bias is treated as a direct DNBR penalty. Since the parameter uncertainties are considered in determining the RTDP design limit DNBR values, the plant safety analyses are performed using input parameters at their nominal values.

TABLE 4-1
THERMAL AND HYDRAULIC DESIGN PARAMETERS
FOR VEGP UNITS 1 AND 2

Thermal and Hydraulic Design Parameters (Using RTDP)		Analysis Parameters	Licensing Parameters
Reactor Core Heat Output, MWt		3565	3411
Reactor Core Heat Output, 10^6 BTU/hr		12164	11639
Heat Generated in Fuel, %		97.4	97.4
Pressurizer Pressure, Nominal, psia		2250	2250
$F_{\Delta H}$, Nuclear Enthalpy Rise Hot Channel Factor	(LOPAR)	1.57[1+.3(1-P)]	1.57[1+.3(1-P)]
	(V-5)	1.65[1+.3(1-P)]	1.65[1+.3(1-P)]
Minimum DNBR at Nominal Conditions			
Typical Flow Channel	(LOPAR)	2.32	>2.32
	(V-5)	2.41	>2.41
Thimble (Cold Wall) Flow Channel	(LOPAR)	2.22	>2.22
	(V-5)	2.28	>2.28
Design Limit DNBR			
Typical Flow Channel	(LOPAR)	1.23	1.23
	(V-5)	1.24	1.24
Thimble (Cold Wall) Flow Channel	(LOPAR)	1.22	1.22
	(V-5)	1.23	1.23
DNB Correlation	(LOPAR)	WRB-1	WRB-1
	(V-5)	WRB-2	WRB-2

TABLE 4-1 (continued)
THERMAL AND HYDRAULIC DESIGN PARAMETERS
FOR VEGP UNITS 1 AND 2

HFP Nominal Coolant Conditions		Analysis Parameters*	Licensing Parameters**
Vessel Minimum Measured Flow Rate (including Bypass),			
10^6 lbm/hr		142.3(a)	145.1(b)
gpm		382,400	391,225
Vessel Thermal Design Flow Rate (including Bypass),			
10^6 lbm/hr		139.4	142.1
gpm		374,400	382,800
Core Flow Rate (excluding Bypass, based on Thermal Design Flow)			
10^6 lbm/hr		127.7	130.2
gpm		342,950	350,645
Fuel Assembly Flow Area for heat Transfer, ft^2+	(LOPAR)	51.08	51.08
	(V-5)	54.13	54.13
Core Inlet Mass Velocity, 10^6 lbm/hr-ft^2 (Based on TDF)+	(LOPAR)	2.50	2.55
	(V-5)	2.36	2.41

+Assumes all LOPAR or VANTAGE 5 Core

* Analysis flow rates are based on 10% steam generator tube plugging
** Licensing flow rates are based on 0% steam generator tube plugging
(a) Inlet temperature = 557.4°F
(b) Inlet temperature = 559.3°F

TABLE 4-1 (continued)
THERMAL AND HYDRAULIC DESIGN PARAMETERS
FOR VEGP UNITS 1 AND 2

Thermal and Hydraulic Design Parameters (Based on TDF)		Analysis Parameters	Licensing Parameters
Nominal Vessel/Core Inlet Temperature, °F		556.8	558.7
Vessel Average Temperature, °F		588.4	588.4
Core Average Temperature, °F		593.0	592.6
Vessel Outlet Temperature, °F		620.0	618.1
Average Temperature Rise in Vessel, °F		63.2	59.4
Average Temperature Rise in Core, °F		68.2	64.2
Heat Transfer			
Active Heat Transfer Surface Area, ft^{2+}	(LOPAR)	59,742	59,742
	(V-5)	57,505	57,505
Average Heat Flux, BTU/hr-ft^{2+}	(LOPAR)	198,370	189,800
	(V-5)	206,085	197,200
Average Linear Power, kw/ft^{++}		5.69	5.45
Peak Linear Power for Normal Operation, kw/ft^{+++}		14.2	13.6
Temperature Limit for Prevention of Centerline Melt, °F		4,700	4,700

\+ Assumes all LOPAR or VANTAGE 5 core

++ Based on densified active fuel length

+++ Based on 2.50 F_Q peaking factor

The RTDP design limit DNBR values are 1.24 and 1.23 for the typical and thimble cells respectively for VANTAGE 5 fuel, and 1.23 and 1.22 for the typical and thimble cells respectively for LOPAR fuel.

In addition to the above considerations, plant specific DNBR margin was maintained by performing the safety analyses to DNBR limits higher than the design limit DNBR values.

A fraction of the available DNBR margin is utilized to accommodate the transition core penalty. For VANTAGE 5 fuel, this transition core penalty is a function of the number of VANTAGE 5 fuel assemblies in the core as given in Reference 24. There is no transition core penalty for the LOPAR fuel. Additional margin is used to offset the rod bow DNBR penalty. Based on Reference 7, the fuel rod bow DNBR penalty is less than 1.5% for both LOPAR and VANTAGE 5 fuel in the 20 inch grid spans. No rod bow penalty is required in the 10 inch grid spans of the VANTAGE 5 fuel. The remaining DNBR margin, after consideration of these penalties, is available for operating and design flexibility.

The option of thimble plug removal has been included in all of the DNBR analyses performed for the VANTAGE 5 and LOPAR fuel. The primary effect of thimble plug removal is an increase in the core bypass flow. This increased core bypass flow is reflected in the core flow rates and the DNBR values presented in Table 4-1.

Operation with thimble plugs in place reduces the core bypass flow through the fuel assembly thimble tubes. The reduction in core bypass flow for operation with the thimble plugs in place is a DNBR benefit. The increased DNBR margin associated with the use of a full compliment of thimble plugs can be used to offset DNBR penalties.

The Standard Thermal Design Procedure (STDP) is used for those analyses where the RTDP methodology is not applicable. In the STDP method, the parameters used in DNBR analyses are treated in a conservative way to give the lowest minimum DNBR. Sufficient DNBR margin to cover appropriate DNBR penalties is preserved whenever the STDP is used.

The fuel temperatures used in safety analysis calculations for the VANTAGE 5 and LOPAR fuel were calculated with the improved fuel performance code, Reference 6. This code was used to perform both design and safety calculations. These fuel temperatures were used as initial conditions for LOCA and non-LOCA transients.

5.0 ACCIDENT EVALUATION

5.1 Non-LOCA Accidents

This section addresses the effects of the VANTAGE 5 design features and safety analysis assumptions for the VEGP Units 1 and 2 non-LOCA accident analyses.

5.1.1 VANTAGE 5 Design Features

The following VANTAGE 5 fuel assembly design features were considered in the non-LOCA analyses and evaluations:

- o Fuel Rod Dimensions
- o Intermediate Flow Mixer (IFM) Grids
- o Axial Blankets
- o Integral Fuel Burnable Absorbers (IFBAs)
- o Reconstitutable Top Nozzle (RTN)
- o Debris Filter Bottom Nozzle (DFBN)
- o Extended Burnup
- o Zircaloy Grids

A brief description of each of these and its consideration in the non-LOCA 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 the LOPAR and VANTAGE 5 fuel.

IFM Grids

The IFM grid feature of the VANTAGE 5 fuel design increases Departure from Nucleate Boiling (DNB) margin. The fuel safety analysis limit DNB values contain significant DNB margin (see Section 4.0).

The IFM grid feature of the VANTAGE 5 fuel design increases the core pressure drop. This results in an increased control rod scram time to the dashpot from 2.2 to 2.7 seconds. This increased drop time primarily affects the fast reactivity transients which were analyzed for this report. The revised control rod drop time was incorporated in all the analyzed events requiring this parameter and the remaining transients have been evaluated.

Axial Blankets and IFBAs

Axial blankets reduce power at the top and bottom of the rod which increases axial power peaking at the center of the rod. This effect is offset by the presence of part-length 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 the time in core 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 sufficiently conservative to accommodate the introduction of axial blankets and/or IFBA in the VEGP Units 1 and 2.

Reconstitutable Top Nozzle (RTN) and Debris Filter Bottom Nozzle (DFBN)

RTNs and DFBNs have been used extensively in Westinghouse designs. Analyses/tests were performed to confirm the hydraulic compatibility of the Westinghouse nozzle designs to the existing design; therefore, these components will not affect any parameters important to the non-LOCA safety analyses.

Extended Burnup

The VANTAGE 5 fuel assemblies were modified for extended burnups by reducing the thickness of both the top and bottom nozzle end plates, decreasing the height of the bottom nozzle and increasing the length of the fuel rod. The effects of these extended burnup features have been accounted for in the non-LOCA analyses.

Zircaloy Grids

Zircaloy grids have replaced inconel grids in the VANTAGE 5 fuel assembly with the exception of the top and bottom grids which remain inconel. The effects of Zircaloy grids have been incorporated in the safety analysis.

5.1.2 Safety Analysis Assumptions

Listed below are the safety analysis assumptions which represent a departure from those currently used for VEGP Units 1 and 2.

- o Increased Core Thermal Power
- o Reduced Thermal Design Flow
- o Revised Thermal Design Procedure (RTDP) for Appropriate DNB Events
- o Increased Uncertainties for Reactor Coolant System (RCS) Temperature and Pressure
- o 10% Steam Generator Tube Plugging
- o Relaxed Axial Offset Control (RAOC)
- o Increased End of Life Moderator Density Coefficient
- o Removal of Thimble Plugs
- o Increased Design Enthalpy Rise Hot Channel Factors ($F_{\Delta H}$ and F_Q)
- o Steam Generator tap relocation

A brief description of each of these assumptions follows.

Increased Core Thermal Power

An increase in the nominal core thermal power from 3411 MWt to 3565 MWt was considered in the non-LOCA safety analyses for the potential rerating of VEGP Units 1 and 2. The non-LOCA safety analyses performed at 3565 MWt will conservatively bound the current rated core thermal power level of 3411 MWt.

Reduced Thermal Design Flow

A decrease in the RCS thermal design flow from 382,800 gpm to 374,400 gpm was considered in the non-LOCA safety analyses for the potential rerating of VEGP Units 1 and 2. The non-LOCA safety analyses performed at 374,400 gpm will conservatively bound the current thermal design flow of 382,800 gpm.

Revised Thermal Design Procedure (RTDP)for Appropriate DNB Events

The calculational method utilized to meet the DNB design basis is the RTDP, which is described in Reference 15. Uncertainties in the plant operating parameters are statistically incorporated in the design limit DNBR value as discussed in Section 4.0. Since the parameter uncertainties are considered in determining the design DNBR value, the associated plant safety analyses are performed using nominal initial conditions.

Increased Uncertainties for RCS Temperature and Pressure

The RCS temperature uncertainity has been increased from ± 4°F to ± 6°F, and the RCS pressure uncertainty has been increased from ± 30 psi to ± 50 psi.

10% Steam Generator Tube Plugging

The nominal primary and secondary side conditions have been established which include up to 10% steam generator tube plugging (or the hydraulic equivalent of plugs and sleeves) in each steam generator. It is assumed that no one steam generator exceeds 10% tube plugging. These conditions were assumed for all of the analyses performed.

Relaxed Axial Offset Control (RAOC)

RAOC operation with a +10/-20% Axial Flux Difference (AFD) band at 100% Rated Thermal Power (RTP) operation formed the basis for the safety evaluations.

Increased End of Life Moderator Density Coefficient

In order to accommodate longer fuel cycles and extended fuel burnup, a moderator density coefficient of 0.50 $\Delta\rho$/gm/cc corresponding to end of cycle, full power conditions was conservatively incorporated into the safety analyses performed for this report.

Thimble Plug Removal

Thimble plug removal affects core pressure drops and increases core bypass flow. These effects have been conservatively incorporated into the non-LOCA analyses performed for this report. The analyses are applicable for either the thimble plugs in place or removed.

Increased $F_{\Delta H}$ and F_Q

The design $F_{\Delta H}$ for the LOPAR and VANTAGE 5 fuel is 1.57 and 1.65, respectively. The non-LOCA calculations applicable for the VANTAGE 5 core have assumed a full power $F_{\Delta H}$ of 1.70. This is a conservative safety analysis assumption for this report.

ihe increase in the Technical Specification maximum LOCA F_Q from 2.3 to 2.5 is bounded in the non-LOCA transients. A maximum F_Q of 2.55 was conservatively assumed in the non-LOCA safety analyses.

Steam Generator Tap Relocation

The non-LOCA analyses account for the relocation of the steam generator level tap. The tap was lowered from 438 inches to 333 inches from the top of the tubesheet.

5.1.3 Non-LOCA Safety Evaluation Methodology

The non-LOCA reload safety evaluation methodology is described in Reference 4. The methodology confirms that, if a core configuration is bounded by existing safety analyses, the applicable safety criteria are satisfied. The methodology systematically identifies both parameter changes on a cycle-by-cycle basis which may exceed existing safety analysis assumptions and the transients which require evaluation. This methodology is applicable to the evaluation of VANTAGE 5 transition and full cores.

Any required evaluation identified by the reload methodology is one of two types. If the identified parameter is only slightly out of bounds, or if 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 significant or not easily quantifiable effect on the transients, analyses are required.

The analysis approach will utilize Westinghouse codes and methods which have been accepted by the NRC and have been used in previous submittals to the NRC. These methods are those which have been presented to the NRC for a specific plant, reference SARs or reports for NRC approval. The analysis methods and codes are described in Appendix A.

The key safety parameters are documented in Reference 4. Values of these safety parameters which bound both fuel types (LOPAR and VANTAGE 5) were assumed in the non-LOCA safety analyses. For subsequent fuel reloads, the key safety parameters will be evaluated to determine if violations of these bounding values exist. An evaluation of the affected accidents will take place as described in Reference 4.

5.1.4 Conclusions

Descriptions of the non-LOCA accidents analyzed for this report, method of analysis, results, and conclusions are contained in Appendix A. Appendix A conforms to the format of the VEGP Units 1 and 2 FSAR. It was found that the appropriate safety criteria were met for each of the transients analyzed. In addition, an evaluation was performed regarding the effect of VANTAGE 5 fuel on the steamline 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 affected by the transition to VANTAGE 5 fuel.

Based on the plant operating limitations given in the Technical Specifications and the proposed Technical Specifications changes given in Enclosure 3 of this submittal, the results show that the transition from LOPAR to VANTAGE 5 fuel can be accommodated with margin to the applicable FSAR safety limits.

5.2 LOCA Accidents

This section addresses the effects of the VANTAGE 5 design features and modified safety analysis assumptions for the VEGP Units 1 and 2 LOCA analyses.

5.2.1 Large Break LOCA

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

The large break Loss-Of-Coolant Accident (LOCA) analysis for VEGP Units 1 and 2, applicable to a full core of VANTAGE 5 fuel assemblies, was performed to develop VEGP Units 1 and 2 specific peaking factor limits. This is consistent with the methodology employed in the Reference Core Report for 17x17 VANTAGE 5, Reference 2. The Westinghouse 1981 Evaluation Model with BASH, References 25 and 26, was utilized and a spectrum of cold leg breaks was analyzed for VEGP Units 1 and 2 that bound nominal operating conditions.

Other pertinent large break LOCA analysis assumptions include:

- An uprated core thermal power of 3565 MWt,
- 10% steam generator tubes plugged for each of the four steam generators,
- An $F_{\Delta H}$ of 1.65 for VANTAGE 5 and 1.57 for LOPAR, and a total peaking factor, F_Q of 2.5, for VANTAGE 5 and LOPAR fuel in the transition core
- A two line segment K(z) curve with a value for K(z) = 1.0 up to 6 ft. and a linear ramp up to a value of 0.92 at 12 ft,

- o A reduced thermal design flow of 93,600 gpm per loop,
- o A 10% reduction of ECCS safety injection flow,
- o A delay time increase to 40 seconds for the ECCS safety injection,
- o A RWST minimum water temperature of 40°F,
- o A RCS temperature operating band of ± 6°F,
- o A RCS pressure uncertainty of ± 50 psia,
- o A widened accumulator water level range of 860 cubic feet to 940 cubic feet with a nominal water level of 900 cubic feet,
- o The fuel temperature and rod internal pressure data based on the improved fuel thermal model, Reference 6,
- o Containment Mini Purge Isolation,
- o Thimble Plug Removal,
- o RAOC operation with a +10/-20% AFD band at 100% RTP,
- o Steam Generator tap relocation.

The VANTAGE 5 fuel features, as applied at the VEGP Units 1 and 2, result in a fuel assembly that is more limiting than the LOPAR fuel currently implemented at VEGP Units 1 and 2 with respect to large break LOCA ECCS performance, Reference 2. As such, VANTAGE 5 fuel has been analyzed herein.

5.2.1.2 Method of Analysis

The method used in analyzing the VEGP Units 1 and 2 for VANTAGE 5 fuel, including the computer codes used and the assumptions are described in detail in Appendix B, Section 15.6.5.

5.2.1.3 Results

The results of the large break LOCA analysis for VEGP Units 1 and 2, including tabular and plotted results of the break spectrum analyzed are provided in Appendix B, Section 15.6.5.

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

VEGP Units 1 and 2 are neither an Upper Head Injection (UHI) or Upper Plenum Injection (UPI) plant; so restriction 1 does not apply.

The VEGP Units 1 and 2 plant specific large break LOCA analysis considers both minimum and maximum ECCS safeguards to address restriction 2. The C_d=0.6 Double Ended Cold Leg Guillotine (DECLG) with minimum ECCS flows was found to result in the most limiting consequences.

Concerning restriction 3, a chopped cosine power shape was used in the large break LOCA analysis for the VEGP Units 1 and 2.

5.2.1.4 Conclusions

The large break LOCA analysis performed for the VEGP Units 1 and 2 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 amount of fuel cladding that reacts chemically with the water or steam does not exceed one percent of the total amount of fuel rod zircaloy cladding in the reactor.

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.

The large break LOCA results for all breaks analyzed is shown in Table 15.6.5-2 of Appendix B, Section 15.6.5.

The large break LOCA analysis for the VEGP Units 1 and 2 assuming a full core of VANTAGE 5 fuel, utilizing the 1981 Evaluation Model (EM) with the BASH calculational model, resulted in a peak cladding temperature of 2058°F for the limiting DECLG break at a total peaking factor of 2.50. The maximum local metal-water reaction was 5.62%, and the total core wide metal-water reaction was less than 0.3% for all cases analyzed. The clad temperature transients turn around at a time when the core geometry was still amenable to cooling.

Also the effect of the transition core cycles is conservatively evaluated to be at most 50°F higher for the calculated peak cladding temperature which would yield a transition core PCT of 2108°F. The transition core penalty can be accommodated by the margin to the 10CFR50.46, 2200°F limit. It can be determined from the results contained in Appendix B, Section 15.6.5 that the ECCS analysis for the VEGP Units 1 and 2 remain in compliance with the requirement of 10CFR50.46 including consideration for transition core configurations.

5.2.2 Small Break LOCA

5.2.2.1 Description of Analysis and Assumptions for 17x17 VANTAGE 5 Fuel

The small break LOCA was analyzed assuming a full core of VANTAGE 5 fuel to determine the peak cladding temperature. This is consistent with the methodology employed in WCAP-10444-P-A, Reference 2, for the 17x17 VANTAGE 5 transition. The currently approved NOTRUMP small break ECCS evaluation model, References 27 and 28, was utilized for a spectrum of cold leg breaks. Appendix B, Section 15.6.5, includes a full description of the analysis and assumptions utilized for the Westinghouse VANTAGE 5 ECCS LOCA analysis. Pertinent assumptions for the VEGP Units 1 and 2 small break LOCA analysis include:

- o An uprated core thermal power of 3565 MWt,
- o 10% steam generator tubes plugged for each of the four steam generators,
- o An $F_{\Delta H}$ of 1.70 for VANTAGE 5 and LOPAR fuel, and a total peaking factor for F_Q of 2.58,
- o A two line segment K(z) curve with a value for K(z) = 1.0 up to 6 ft. and a linear ramp to a value of 0.92 at 12 ft,
- o A reduced thermal design flow of 93,600 gpm per loop,
- o A 10% reduction of ECCS safety injection flow,
- o A delay time increase to 40 seconds for the ECCS safety injection,
- o A RWST minimum water temperature of 40°F,
- o A RCS temperature operating band of ± 6°F,
- o A RCS pressure uncertainty of ± 50 psia,
- o A widened accumulator water level range of 860 cubic feet to 940 cubic feet with a nominal water level of 900 cubic feet,
- o The fuel temperature and rod internal pressure data based on the improved fuel thermal model, Reference 6,
- o Thimble plug removal,
- o RAOC operation with a +10/-20% AFD band at 100% RTP,
- o Steam Generator tap relocation.

Sensitivity studies performed using the NOTRUMP small break evaluation model have demonstrated that VANTAGE 5 fuel is more limiting than the OFA fuel in the calculated ECCS performance. Similar studies using the WFLASH evaluation model have previously shown the OFA fuel is more limiting than the LOPAR fuel. For the small break LOCA, the effect of the fuel difference is most 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 the NOTRUMP evaluation model analysis. On this basis, only VANTAGE 5 fuel was analyzed, since it is the most limiting of the two types of fuel (LOPAR and VANTAGE 5) that would reside in the VEGP Units 1 and 2 cores.

5.2.2.2 Method of Analysis

The methods of analysis, including codes used and assumptions, are described in detail in Appendix B, Section 15.6.5.

5.2.2.3 Results

The results of the small break LOCA analysis, including tabular and plotted results of the break spectrum analyzed, are provided in Appendix B, Section 15.6.5.

5.2.2.4 Conclusions

The small break VANTAGE 5 LOCA analysis for the VEGP Units 1 and 2, utilizing the currently approved NOTRUMP Evaluation Model resulted in a calculated Peak Cladding Temperature (PCT) of 2056°F for the 3-inch diameter cold leg break. The analysis assumed a limiting small break power shape consistent with a LOCA $F_Q(z)$ envelope of 2.58 at the core midplane elevation and 2.368 at the top of the core. The maximum local-water reaction is 7.74%, and the total core metal-water reaction is less than 0.3% for all cases analyzed. The clad temperature transients turn around at a time when the core geometry is still amendable to cooling.

Analyses presented in Appendix B, Section 15.6.5 show that one centrifugal pump and one high head 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 and the peak cladding temperature results are well below the peak cladding temperatures calculated for the large break LOCA. Adequate protection is therefore afforded by the ECCS in the event of a small break LOCA.

5.2.3 Transition Core Effects on LOCA

When assessing the effect of transition cores on the large break LOCA analysis, it must be determined whether the transition core can have a greater calculated PCT than either a complete core of the LOPAR fuel assembly design or a complete core of the VANTAGE 5 design. For a given peaking factor, the only mechanism available to cause a transition core to have a greater calculated PCT than a full core of either fuel is the possibility of flow redistribution due to fuel assembly hydraulic resistance mismatch. Hydraulic resistance mismatch will exist only for a transition core and is the only unique difference between a complete core of either fuel type and the transition core.

5.2.3.1 Large Break LOCA

The large break LOCA analysis was performed with a full core of VANTAGE 5 and conservatively applies the blowdown results to transition cores. The VANTAGE 5 differs hydraulically from the LOPAR fuel assembly design it replaces. The differences in the total assembly hydraulic resistance between the two designs is approximately 10% higher for VANTAGE 5.

An evaluation of hydraulic mismatch of approximately 10% showed an insignificant effect on blowdown cooling during a LOCA. The SATAN-VI computer code models the crossflows between the average core flow channel (N-1 fuel assemblies) and the hot assembly flow channel (one flow assembly) during a blowdown. To better understand the transition core large break LOCA blowdown transient phenomena, conservative blowdown fuel clad heatup calculations have been performed to determine the clad temperature effect on the new fuel design for mixed core configurations. The effect was determined by reducing the axial flow in the hot assembly at the appropriate elevations to simulate the effects of the transition core hydraulic resistance mismatch. In addition, the Westinghouse blowdown evaluation model was modified to account for grid heat transfer enhancement during blowdown for this evaluation. The results of this evaluation have shown that no peak cladding temperature penalty is observed during blowdown for the mixed core. Therefore, it is not necessary to perform a blowdown calculation for the VANTAGE 5 transition core configuration because the evaluation model blowdown calculation performed for the full VANTAGE 5 core is conservative and bounding.

Since the overall resistance of the two types of fuel is essentially the same during blowdown, only the crossflows during core reflood due to the IFM grid need to be evaluated. The LOCA analysis uses the BASH computer code to calculate the reflood transient, Reference 25, which utilizes the BART code, Reference 29. A detailed description of the BASH code is given in Appendix B. Fuel assembly design specific analyses have been performed with a version of the BART computer code, which accurately models mixed core configurations during reflood. Westinghouse transition core designs, including specific 17x17 LOPAR to VANTAGE 5 transition core cases, were analyzed. For this case, BART modeled both fuel assembly types and predicted the reduction in axial flow at the appropriate elevations. As expected, the increase in hydraulic resistance for the VANTAGE 5 assembly was shown to produce a reduction in reflood steam flow rate for the VANTAGE 5 fuel at mixing vane grid elevations for transition core configurations. This reduction in steam flow rate is partially offset by the fuel grid heat transfer enhancement predicted by the BART code during reflood. The various fuel assembly specific transition core analyses performed resulted in peak cladding temperature increases of up to 50°F for core axial elevations that bound the location of the PCT. Therefore, the maximum PCT penalty possible for VANTAGE 5 fuel residing in a transition core is 50°F, Reference 2. Once a full core of VANTAGE 5 fuel is achieved, the large break LOCA analysis will apply without the transition core penalty.

5.2.3.2 Small Break LOCA

The NOTRUMP computer code, Reference 27, is used to model the core hydraulics during a small break event. Only one core flow channel is modeled in the NOTRUMP code, since the core flow during a small break is relatively slow, providing enough time to maintain flow equilibrium between fuel assemblies (i.e., no crossflow). Therefore, hydraulic resistance mismatch is not a factor for small break. Thus, it is not necessary to perform a small break evaluation for transition cores, and it is sufficient to reference the small break LOCA for the complete core of the VANTAGE 5 fuel design, as bounding for all transition cycles.

5.2.4 Blowdown Reactor Vessel and Loop Forces

The forces created by a hypothesized 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 significantly affected by a change in fuel at the VEGP Units 1 and 2 from 17x17 LOPAR to VANTAGE 5 fuel. The forces in the vicinity of the core are affected by the core flow area/volume. Since there will be no significant change in the core flow area/volume for VANTAGE 5 fuel, there will not be an adverse change in the forces calculated for a hypothesized LOCA. Forces acting in the RCS loop piping as a result of a hypothesized LOCA are not influenced by changes in fuel assembly design.

Reviewing LOCA hydraulic forcing functions used in the plant design for the VEGP Units, it has been determined that LOCA forcing functions for limited displacement inlet and outlet nozzle breaks are acceptable for the evaluation of VANTAGE 5 fuel. Taking credit for Leak-Before-Break (LBB) technology, it has been determined that any differences due to the implementation of VANTAGE 5 can be offset by the margin that exists between limited displacement vessel nozzle breaks and RCS branch line breaks. Thus the implementation of VANTAGE 5 fuel at VEGP Units 1 and 2 will not result in an increase of the calculated consequences of a hypothesized LOCA on the reactor vessel internals or RCS loop piping.

5.2.4.1 Evaluation of RCCA Insertion

In order to assess the feasibility of taking credit for the RCCA insertion during a postulated rupture of one square foot of the Reactor Pressure Vessel (RPV) inlet nozzle break and one square foot of the RPV outlet nozzle break, the dynamic loads that were calculated for these two breaks were combined with the cross-flow loads resulting from the RPV inlet and outlet nozzle breaks on the VEGP Units 1 and 2.

The evaluation showed that the maximum guide tube deflection for the RPV inlet and outlet nozzle breaks was within the allowable deflection limits which were established during the scram-deflection testing for the 17x17 guide tube. Consequently, the control rods in the VEGP Units 1 and 2 can be inserted following the postulated LOCA breaks described above.

5.2.5 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 VEGP Units 1 and 2 does not affect the assumptions for decay heat, core reactivity or boron concentration for sources of water residing in the containment sump Post-LOCA. Thus, the post-LOCA long term core cooling, ECCS flows, and switchover of the ECCS to hot leg recirculation are not significantly affected by the implementation of VANTAGE 5 fuel. Additionally Westinghouse during a specific reload design, performs an independent check on core subcriticality for each cycle operated at VEGP Units 1 and 2.

Currently, to show that the core temperature is reduced and decay heat is removed for an extended period of time, it has been determined that the core must remain subcritical (Keff <1.0) after a postulated LOCA. As a result of the effect of LOCA forces on the control rods (RCCAs) for breaks greater than 1.0 ft_2 in the Reactor Coolant System (RCS), an additional requirement has been needed to show that subcriticality can be maintained while taking no credit for the control rods. This set of circumstances led to a requirement that the borated ECCS water provides sufficient negative reactivity to keep the reactor core subcritical (Keff <1.0), All-Rods-Out (ARO), No Xenon, at the most reactive time in life for reactor coolant system temperatures $\leq$ 212°F.

This requirement has been easily satisfied for years by the Refueling Water Storage Tank (RWST) minimum Technical Specification requirement of 2000 ppm for the boron concentration. Consequently, much of the plant equipment design and qualification assumed an RWST boron concentration in the range of 2000 to 2600 ppm. The current trend to a longer core cycle life and a Positive Moderator Temperature Coefficient (PMTC) has required increased RWST boron concentrations to meet the post-LOCA long term subcriticality requirement, because the added reactivity from the RCCAs is not considered. Increasing the RWST boron concentration, in turn, increases the demands on related LOCA requirements and equipment.

Assuming that the control rods do not enter the core during the LOCA event is one area of conservatism identified in the current post-LOCA long term core cooling methodology. This is based on the assumption that the LOCA is initiated instantaneously and may cause enough damage to the upper internals that the control rods cannot enter the core. Currently, this assumption is made for reactor coolant system breaks greater than 1.0 ft^2 when performing the ECCS LOCA thermal analyses. For the postulated LOCA with breaks in the reactor coolant system smaller than 1.0 ft^2, credit for control rods is used in the ECCS thermal analysis of the small break LOCA.

However, taking credit for control rods during the long term core cooling phase of the LOCA is a departure from the licensing commitment stated in Section 4.5 of WCAP-8339, "Westinghouse ECCS Evaluation Model - Summary." Borderlon, F. M., et al., June 1974. This commitment states that the core will be maintained in a shutdown condition with borated water, with no mention of taking credit for control rods. The following information will provide a discussion on the use of LBB technology to satisfy the long term core cooling portion of the postulated LOCA and replace the licensing approach in WCAP-8339.

General Discussion

Using the LBB technology has demonstrated that safe shutdown of the reactor can be accomplished based on detection of a leak before a significant rupture in the RCS piping can occur. LBB technology has eliminated analyses of main RCS piping breaks in the evaluation of the mechanical and structural integrity of the reactor coolant system for a postulated LOCA and this has led to a reduction in break sizes considered in analyses to determine the LOCA hydraulic forcing functions on reactor vessel internals and reactor coolant system loop supports.

Generally, the analysis to generate the LOCA hydraulic forcing functions is analyzed or evaluated for reactor coolant system branch line breaks which are less than 1.0 ft^2 in break area. Since these breaks have been considered acceptable for use when LBB technology is applied to demonstrate the structural and mechanical integrity of the reactor coolant system, taking credit for a mechanical feature such as control rods (RCCAs) when determining the post-LOCA boron requirements should also be acceptable as long as control rod insertion can be demonstrated for branch line breaks. Using LBB technology to satisfy 10CFR50.46 Acceptance Criterion 5 for long term cooling would be an extension of the same approach used for the calculation of grid loads from a postulated LOCA to demonstrate a coolable geometry.

For post-LOCA long term core cooling, the mechanical integrity evaluation based on branch line break LOCA hydraulic forcing functions for the reactor vessel components, and specifically insertion of the control rods (RCCAs), has been evaluated in Section 5.2.4.1 to establish that credit for control rods can be taken for the post accident long term core cooling aspect of the postulated LOCA. However, for the large break ECCS LOCA thermal analyses, no credit for control rods will continue to be assumed to demonstrate compliance with the first three acceptance criteria of 10CFR50.46.

Conclusions

Large break LOCAs are considered hypothetical events and are analyzed to demonstrate the effectiveness of the ECCS and limit the core peaking factors. Analysis of the large break LOCA for VEGP Units 1 and 2 currently limits the core peaking factors to assure that the calculated PCT is less than 2200°F, a specific requirement of 10CFR50.46. Limiting the core peaking factors by the Large Break LOCA ECCS thermal analysis has not created conditions undesirable to safety. The large break LOCA ECCS analysis would still be performed without taking credit for RCCA insertion to determine acceptable thermal limits. To meet the post-LOCA long term core cooling requirement with RCCA insertion, the borated ECCS water should provide sufficient negative reactivity to keep the reactor core subcritical (K_{eff} <1.0, All-Rods-In minus 2, (ARI-2), No Xenon, most reactive time in life, at RCS temperatures ≤ 212°F). The ARI-2 requirement is to address the loss of one rod due to a rod ejection event which would create a LOCA and the failure of one rod to insert.

5.2.6 Steam Generator Tube Rupture

5.2.6.1 Introduction

Design basis analyses of a Steam Generator Tube Rupture (SGTR) event at the VEGP Units 1 and 2 have been performed to assess the effect of the transition to a core with VANTAGE 5 fuel assemblies. The analyses performed include a demonstration of margin to steam generator overfill in the event of a tube rupture and an analysis which demonstrates that the calculated offsite radiation doses are within the limits set forth in 10CFR100.

The analyses performed bound operation of VEGP Units 1 and 2 at an uprated NSSS power of 3579 MWt with a LOPAR/VANTAGE 5 fuel transition core, LOPAR fuel core or VANTAGE 5 fuel core with up to 10% uniform steam generator tube plugging. The analyses also considered a 90 second delay time for auxiliary feedwater flow delivery and the steam generator lower narrow range level tap relocation. Since the assumption that the initial primary coolant activity at the Standard Technical Specifications limit will not change for VEGP Units 1 and 2 due to the proposed change in fuel, the parameters which effect the offsite radiation doses calculated for the FSAR SGTR analysis are primary to secondary break flow and the steam released from the ruptured steam generator to the atmosphere. Therefore, the analyses to support the transition to VANTAGE 5 fuel assess the effect of the fuel change on primary to secondary break flow and steam released via the ruptured steam generator.

5.2.6.2 Methodology

The steam generator tube rupture analyses were performed for VEGP Units 1 and 2 using the methodology and the assumptions described in WCAP-11731, Reference 30.

Plant response to the SGTR event was modeled using the LOFTTR2 computer code with conservative assumptions of break size and location, condenser availability and initial secondary water mass in the ruptured steam generator. The analysis methodology includes the simulation of the operator actions for recovery from a steam generator tube rupture event based on VEGP Units 1 and 2 Emergency Operating Procedures, which were developed from the Westinghouse Owners Group Emergency Response Guidelines.

Since the limiting single failure is different for the overfill analysis and the offsite radiation dose analysis, the two analyses were performed using different single failure assumptions. For the margin to overfill analysis, the single failure was assumed to be that the auxiliary feedwater flow control valve located in the flow path from the turbine-driven auxiliary feedwater pump to the ruptured steam generator fails to close when the isolation of the ruptured steam generator is being performed. In the offsite radiation dose analysis, the ruptured steam generator PORV was assumed to fail open when the isolation of the ruptured steam generator was performed.

5.2.6.3 Margin to Overfill Analysis Results

The LOFTTR2 analysis to determine the margin to overfill was performed for the time period from the steam generator tube rupture until the primary and secondary pressures are equalized and break flow is terminated. The water volume in the secondary side of the ruptured steam generator was calculated as a function of time to demonstrate that overfill does not occur. The results of the analysis demonstrate that the transition to VANTAGE 5 fuel does not change the conclusion that there is margin to overfill calculated for VEGP Units 1 and 2 in the event of a tube rupture.

5.2.6.4 Offsite Radiation Dose Analysis Results

For the offsite radiation dose analysis, the primary to secondary break flow and the steam release to the atmosphere from both the ruptured and intact generators were calculated for use in determining the activity released to the atmosphere. The mass releases were calculated with the LOFTTR2 program from the initiation of the event until termination of the break flow. For the time period following break flow termination, steam releases from and feedwater flows to the ruptured and intact steam generators were determined from a mass and energy balance using the calculated RCS and steam generator conditions when primary to secondary tube leakage was terminated. The mass release information was used to calculate the radiation doses at the exclusion area boundary and at the low population zone assuming that the primary coolant activity is at the Standard Technical Specifications limit prior to the accident. The results of the analysis to support the transition to VANTAGE 5 fuel show that the offsite doses for VEGP Units 1 and 2 are well within the allowable guidelines specified in the Standard Review Plan, NUREG-0800, FSAR Chapter 15.6.3, and 10CFR100. Refer to Appendix C for the Offsite radiation dose analysis results.

A more complete description of the SGTR radiological assessment and results is provided in Appendix B.

5.2.7 Containment Integrity Mass and Energy Release

For the short term mass and energy release analysis presented in Section 6.2.1.4.3 of the Vogtle FSAR, there is no impact on this analysis, because changes in fuel type have a very small effect on the transient. For subcompartment analyses, approximately only the first 3 seconds of the blowdown are considered. Therefore, the current short term mass and energy release analysis remains valid and applicable.

For the long term mass and energy analysis presented in the Vogtle FSAR, there is no adverse effect on the analysis, because the plant T-avg remains essentially the same for the change in fuel. Additionally the VANTAGE 5 fuel closely resembles the 17x17 LOPAR fuel. Since the fuel type dimensionally remains basically the same, there is no difference in initial core stored energy, and hence no additional energy would be available for release to containment. Thus, the current analysis remains valid and applicable.

5.3 Radiological Evaluation

The effect of the transition to VANTAGE 5 fuel on the radiological source terms and, subsequently, on the releases, both normal and accidental, is primarily due to the extension of maximum fuel burnup levels for both the VANTAGE 5 and LOPAR fuel types. This safety evaluation and Appendix C address the fuel burnup to a maximum level of 60,000 MWD/MTU for the lead rod average. This safety evaluation and Appendix C are based on the current NRC position regarding extended fuel burnups as set forth in the Federal Register (Reference 31) and NUREG/CR-5009 (Reference 32). These reviews considered average burnups of up to 60,000 MWD/MTU for the lead rod average which bounds both the VANTAGE 5 and LOPAR fuel designs. The radiological consequences in the VEGP FSAR remain valid for both LOPAR and VANTAGE 5 fuel assembly designs to 60,000 MWD/MTU lead rod average burnups. The only exceptions are the revised results for the Fuel Handling Accident and SGTR Event which are presented in Appendix C. A reduction in the conservatism in the assumptions of Reg. Guide 1.25 was utilized for the Fuel Handling Accidents, however, the calculations of doses remain conservative. The results in Appendix C for these two accident scenarios are within the guideline values, as referenced in the NRC Standard Review Plan.

6.0 SUMMARY OF TECHNICAL SPECIFICATIONS CHANGES

Table 6-1 presents a list of the Technical Specifications changes and the justification for the changes. The changes noted in Table 6-1 are given in the proposed Technical Specifications page mark-ups (see Enclosure 3 of this submittal).

TABLE 6-1

SUMMARY AND JUSTIFICATION FOR THE VOGTLE UNITS 1 AND 2 TECHNICAL SPECIFICATIONS CHANGES FOR VANTAGE 5 FUEL

Page	Section	Description	Justification
2-2 2-4,2-8 2-9,2-10 2-11	2.1 2.2	Change to core limits, to the OTDT and OPDT Setpoints.	This change is a result of changes associated with the VANTAGE 5 fuel and to provide operational flexibility.
B 2-1 B 2-2 B 3/4 2-1 B 3/4 2-2 B 3/4 2-4 B 3/4 4-1	2.1.1 Basis 3/4.2 Basis 3/4.4 Basis	Addition of the WRB-1 and WRB-2 correlation	This change reflects the DNB correlation used in analyses.
3/4 1-11 3/4 1-12 3/4 1-13 3/4 5-10	3.1.2.5 3.1.2.6 3.5.4	RWST minimum solution temperature	This change is to allow operational flexibility.
B 3/4 5-2	3/4.5.4 Basis	RWST Bases	This change is to allow operational flexibility.
3/4 1-19	3.1.3.4	Revised rod drop time to less than or equal to 2.7 seconds	This change is a result of changes associated with the VANTAGE 5 fuel. The effect of this increase on the safety analysis has been considered.
3/4 2-1 3/4 2-2 3/4 2-4 3/4 2-6 3/4 2-7 B 3/4 2-1 B 3/4 2-2 B 3/4 2-4	3/4.2.1 3/4.2.2 B 3/4.2 Basis	Axial Flux Difference and $F_Q(Z)$ changes	This change is made to give the plant operating flexibility and also as a result of changes associated with the VANTAGE 5 fuel.
3/4 2-13 B 3/4 2-5	3/4.2.5 3/4.2.5 Basis	DNB parameter changes	This change is made to give the plant operating flexibility and also as a result of changes associated with the VANTAGE 5 fuel.
3/4 3-33	3/4.3.2	Pressurizer pressure trip setpoint	This change is to provide operational flexibility.
3/4 5-1	3/4.5.1	Accumulator Water Level Range	This change is to provide operational flexibility.
6-21 6-21a	6.8.1.6	Core Operating Limits	This change is to provide operational flexibility.

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Appendix A

Non-LOCA Accident Analyses

for the Vogtle Electric Generating Plant Units 1 and 2

Transition to Westinghouse 17x17 VANTAGE-5 Fuel Assemblies

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Appendix A

Non-LOCA Accident Analysis

15.0 Introduction

This section addresses the impact of the complete transition of the Vogtle Electric Generating Plant (VEGP) Units 1 and 2 from Westinghouse 17x17 LOPAR fuel to Westinghouse 17x17 VANTAGE 5 fuel on the FSAR Chapter 15 non-LOCA accident analyses. Section 15.0.11 of this report discusses the methods used for accident evaluation.

VEGP's licensing basis includes the analyses or evaluations of the non-LOCA accidents as listed below.

The evaluations of all non-LOCA accidents performed to determine the impact of the VANTAGE 5 fuel transition are documented in this report. The specific design features associated with the VANTAGE 5 fuel and the modified safety analysis assumptions considered in the non-LOCA safety analyses are described elsewhere in this report.

Accidents Analyzed

The transients affected by the VANTAGE 5 fuel design features or modified safety analysis assumptions as discussed elsewhere in this report that where analyzed are the following:

1. Feedwater System Malfunctions That Result in a Decrease in Feedwater Temperature (See Section 15.1.1)
2. Feedwater System Malfunctions that Result in an Increase in Feedwater Flow (See Section 15.1.2)
3. Excessive Increase in Secondary Steam Flow (See Section 15.1.3)
4. Turbine Trip (See Section 15.2.3)
5. Partial Loss of Forced Reactor Coolant Flow (See Section 15.3.1)
6. Complete Loss of Forced Reactor Coolant Flow (See Section 15.3.2)
7. Reactor Coolant Pump Shaft Seizure (Locked Rotor) (See Section 15.3.3)
8. Reactor Coolant Pump Shaft Break (See Section 15.3.4)
9. Uncontrolled Rod Cluster Control Assembly Bank Withdrawal from a Subcritical or Low-Power Startup Condition (See Section 15.4.1)
10. Uncontrolled Rod Cluster Control Assembly Bank Withdrawal at Power (See Section 15.4.2)
11. Rod Cluster Control Assembly Misalignment (System Malfunction or Operator Error) (See Section 15.4.3)

12. Startup of an Inactive Reactor Coolant Pump at an Incorrect Temperature (See Section 15.4.4)

13. Chemical and Volume Control System Malfunction That Results in a Decrease in the Boron Concentration in the Reactor Coolant (See Section 15.4.6)

14. Spectrum of Rod Cluster Control Assembly Ejection Accidents (See Section 15.4.8)

15. Steamline Break With Coincidental Rod Withdrawal at Power (See Section 15.4.9)

16. Inadvertent Operation of the Emergency Core Cooling System During Power Operation (See Section 15.5.1)

17. Inadvertent Opening of a Pressurizer Safety or Relief Valve (See Section 15.6.1)

Accidents Evaluated

The remaining non-LOCA accidents are as follows:

1. Inadvertent Opening of a Steam Generator Relief or Safety Valve (See Section 15.1.4)

2. Steam System Piping Failure (See Section 15.1.5)

3. Steam Pressure Regulator Malfunction or Failure That Results in Decreasing Steam Flow (See Section 15.2.1)

4. Loss of External Electrical Load (See Section 15.2.2)

5. Inadvertent Closure of Main Steam Isolation Valves (See Section 15.2.4)

6. Loss of Condenser Vacuum and Other Events Resulting in a Turbine Trip (See Section 15.2.5)

7. Loss of Nonemergency AC Power to the Plant Auxiliaries (See Section 15.2.6)

8. Loss of Normal Feedwater Flow (See Section 15.2.7)

9. Feedwater System Pipe Break (See Section 15.2.8)

10. Inadvertent Loading and Operation of a Fuel Assembly in an Improper Position (See Section 15.4.7)

11. Chemical and Volume Control System Malfunction That Increases Reactor Coolant Inventory (See Section 15.5.2)

Refer to the appropriate sections for a discussion of the analysis or evaluation for each of the accidents listed previously.

15.0.1 Classification of Plant Conditions

None of the VANTAGE 5 fuel design features or modified safety analysis assumptions as discussed elsewhere in this report affect this section.

15.0.2 Optimization of Control S,stems

None of the VANTAGE 5 fuel design features or modified safety analysis assumptions as discussed elsewhere in this report affect this section.

15.0.3 Plant Characteristics and Initial Conditions Assumed in the Accident Analyses

Table 15.0.3-1 lists the power rating values assumed in analyses performed for this report. The power rating values listed in Table 15.0.3-1 are based on the design nuclear steam supply system (NSSS) thermal power output which includes the thermal power generated by the reactor coolant pumps (RCPs).

The initial conditions employed in the accident analyses are conservative to bound a future plant uprating. The analyses of departure from nucleate boiling (DNB) accidents use the Revised Thermal Design Procedure (RTDP) to define the initial conditions. Initial conditions for other accidents are obtained by applying the maximum steady-state errors to rated values (this procedure is commonly known as Standard Thermal Design Procedure or STDP). The following steady-state errors are considered in the analyses:

o	Core power	$\pm$2 percent allowance for calorimetric error (note that this error is conservatively applied in the positive direction in non-LOCA accident analyses)
o	Average reactor coolant system temperature	$\pm$6°F allowance for deviation from programmed T_{avg} (includes measurement error)
o	Pressurizer pressure	$\pm$50 psi allowance for steady-state fluctuations and measurement errors

Accidents employing RTDP assume minimum measured flow (MMF); accidents employing STDP assume thermal design flow (TDF). Table 15.0.3-2 summarizes the initial conditions and computer codes used in the accident analyses.

The values of other pertinent plant parameters used in the accident analyses are given in Table 15.0.3-3.

15.0.4 Reactivity Coefficients Assumed in the Accident Analyses

The transient response of the reactor coolant system (RCS) depends on reactivity feedback effects, particularly the moderator temperature coefficient and the Doppler power coefficient.

In the analysis of all events, conservative values of beginning of life and end of life reactivity coefficients are used. Figure 15.0.4-1 shows the Doppler power coefficients, as a function of power, used in the transient analyses. Figure 15.0.4-2 shows the moderator density coefficient, as a function of temperature, used in the transient analyses. The values used in each event are given in Table 15.0.3-2.

15.0.5 <u>Rod Cluster Control Assembly Insertion Characteristics</u>

The negative reactivity insertion following a reactor trip is a function of the acceleration of the rod cluster control assemblies (RCCAs) and the variation in rod worth as a function of rod position.

With respect to accident analyses, the RCCA insertion time from the start of insertion up to the dashpot entry, approximately 85 percent of the rod cluster travel, is assumed to be 2.7 seconds. The RCCA position versus time is shown on Figure 15.0.5-1.

Figure 15.0.5-2 shows the fraction of total negative reactivity insertion versus normalized rod insertion. There is inherent conservatism in the use of this curve in that its basis is a bottom-skewed axial power distribution. For cases other than those associated with axial xenon oscillations, the more favorable axial power distribution existing before trip results in significant negative reactivity to be inserted.

The normalized RCCA negative reactivity insertion versus time used in the safety analysis is shown on Figure 15.0.5-3. The curve shown on this figure results from the combination of Figure 15.0.5-1 and Figure 15.0.5-2. The transient analyses, except where specifically noted otherwise, assume a total negative reactivity insertion following a trip of 4.0% $\Delta k/k$. This assumption is verified to be conservative with respect to the core design.

For analyses requiring the use of a multi-dimensional spatial neutron kinetics code (TWINKLE, Reference 1), the code directly calculates the negative reactivity insertion resulting from reactor trip which is not separable from other reactivity feedback effects. In this case, the code models the RCCA position versus time of Figure 15.0.5-1.

15.0.6 <u>Trip Points and Time Delays to Trip Assumed in Accident Analyses</u>

A reactor trip signal opens two trip breakers connected in series, which feeds power to the control rod drive mechanisms. The loss of power to the mechanism coils causes the mechanisms to release the 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 time delay from the time that the reactor reaches trip setpoint conditions to the time the rods are free to fall defines the total delay to trip. Limiting trip setpoints assumed in accident analyses and the time delay assumed for each trip function are given in Table 15.0.6-1.

Table 15.0.6-1 refers to the overtemperature ΔT (OTΔT) and the overpower ΔT (OPΔT) reactor trip setpoints shown on Figure 15.0.6-1. The e trip setpoints bound the transition cores and a full core of VANTAGE 5 fuel. The associated OTΔT f(ΔI) penalty is shown on Figure 15.0.6-2.

For all the reactor trips, the difference between the trip setpoints assumed in the analysis and the nominal trip setpoints account for instrumentation channel error and setpoint error. The plant Technical Specifications specify the nominal trip setpoints. The calibration of protection system channels and the periodic determination of instrument response times are in accordance with the plant Technical Specifications.

15.0.7 Instrumentation Drift and Calorimetric Errors, Power Range Neutron Flux

The VANTAGE 5 fuel design features and the modified safety analysis assumptions as discussed elsewhere in this report with respect to these changes are covered in WCAP-12460 and WCAP-12462.

15.0.8 Plant Systems and Components Available for Mitigation of Accident Effects

None of the VANTAGE 5 fuel design features or modified safety analysis assumptions as discussed elsewhere in this report affect plant systems and components available for mitigation of accident effects.

15.0.9 Fission Product Inventories

The VANTAGE 5 fuel design features and the modified safety analysis assumptions as discussed elsewhere in this report affect the fission product inventories and are addressed in Appendix C of this report.

15.0.10 Residual Decay Heat

None of the VANTAGE 5 fuel design features or modified safety analysis assumptions as discussed elsewhere in this report affect residual decay heat.

15.0.11 Computer Codes Utilized

Summary descriptions of the principal computer codes used in the transient analyses are given below. Table 15.0.3-2 lists the codes used in the analysis of each transient.

15.0.11.1 FACTRAN Computer Code

FACTRAN calculates the transient temperature distribution in a cross-section of a metal clad UO_2 fuel rod 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, density). The code uses a fuel model which simultaneously contains the following features:

- o A sufficiently large number of radial space increments to handle fast transients such as a rod ejection accidents
- o Material properties which are functions of temperature and a sophisticated fuel-to-clad gap heat transfer calculation
- o The necessary calculations to handle post-DNB transients: film boiling heat transfer correlations; zircaloy-water reaction; and partial melting of the fuel

FACTRAN is further discussed in Reference 2.

15.0.11.2 LOFTRAN Computer Code

Transient response studies of a pressurized water reactor (PWR) system to specified perturbations in process parameters use the LOFTRAN program. The LOFTRAN program models all four reactor coolant loops. This code simulates a multi-loop system by a model containing the reactor vessel, hot and cold leg piping, steam generators (tube and shell sides), and the pressurizer. The pressurizer heaters, spray, relief valves, and safety valves are also considered in the program. LOFTRAN also includes a point neutron kinetics model and reactivity effects of the moderator, fuel, boron, and rods. The secondary side of the steam generator uses a homogeneous, saturated mixture for the thermal transients and a water level correlation for indication and control. The code simulates the reactor protection system which includes reactor trips on high neutron flux, OTΔT, OPΔT, high and low pressurizer pressure, low flow, and high pressurizer level. Control systems are also simulated including rod control, steam dump, feedwater control, and pressurizer pressure control. The ECCS, including the accumulators, is also modeled.

LOFTRAN also can calculate the transient value of DNBR based on the input from the core thermal safety limits.

LOFTRAN is further discussed in Reference 3.

15.0.11.3 ANC Computer Code

ANC is an advanced nodal code capable of two-dimensional and three-dimensional neutronics calculations. ANC is the reference model for all safety analysis calculations, power distributions, peaking factors, critical boron concentrations, control rod worths, reactivity coefficients, etc. In addition, three-dimensional ANC validates one-dimensional and two-dimensional results and provides information about radial (x-y) peaking factors as a function of axial position. It can calculate discrete pin powers from nodal information as well.

ANC is further discussed in Reference 10.

15.0.11.4 TWINKLE Computer Code

The TWINKLE program is a multi-dimensional spatial neutron kinetics code. 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 multi-region 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. The code provides various output edits, e.g., channelwise power, axial offset, enthalpy, volumetric surge, pointwise power and fuel temperatures.

The TWINKLE code predicts the kinetic behavior of a reactor for transients which cause a major perturbation in the spatial neutron flux distribution.

TWINKLE is further described in Reference 1.

15.0.11.5 <u>THINC Computer Code</u>

The THINC computer program performs thermal-hydraulic calculations. This code calculates coolant density, mass velocity, enthalpy, void fractions, static pressure, and departure from nucleate boiling ratio (DNBR) distributions along flow channels within a reactor core. Safety Evaluation Section 4.0 describes the THINC code.

15.0.12 Limiting Single Failures

None of the VANTAGE 5 fuel design features or modified safety analysis assumptions as discussed elsewhere in this report affect the limiting single failures currently assumed in VEGP's safety analyses.

15.0.13 Operator Actions

None of the VANTAGE 5 fuel design features or modified safety analysis assumptions as discussed elsewhere in this report affect the operator actions.

Table 15.0.3-1

Nuclear Steam Supply System Power Ratings

Parameter	Value
Design NSSS thermal power output (MWt)	3579
Minimum thermal power generated by the RCPs (MWt)	14
Maximum NSSS thermal power output (MWt)	3585
Maximum thermal power generated by the RCPs (MWt)	20
Design core thermal power (MWt)	3565

Table 15.0.3-2 (Sheet 1 of 3)

Summary of Initial Conditions and Computer Codes Used

Section	Faults	Computer Codes Utilized	Reactivity Coefficients Assumed: Moderator Density $(\Delta k/gm/cm^3)$	Reactivity Coefficients Assumed: Doppler	Initial NSSS Thermal Power Output Assumed (MWt)
15.1	Increase in heat removal by the secondary system				
	Feedwater system malfunctions that result in an increase in feedwater flow	LOFTRAN (also refer to Section 15.4.1)	0.50	Lower (see Figure 15.0.4-1)	0 and 3579[a][f]
	Excessive increase in secondary steam flow	LOFTRAN	0.0 and 0.50	Lower and upper (see Figure 15.0.4-1)	3579[a][f]
15.2	Decrease in heat removal by the secondary system				
	Loss of external electrical load and/or turbine trip	LOFTRAN	Figure 15.0.4-2 and 0.50	Lower and upper (see Figure 15.0.4-1)	3579[a][f]
15.3	Decrease in reactor coolant system flowrate				
	Partial and complete loss of forced reactor coolant flow.	LOFTRAN, FACTRAN, THINC	Figure 15.0.4-2	Upper (see Figure 15.0.4-1)	3585[c][f]
	Reactor coolant pump shaft seizure (locked rotor)	LOFTRAN, FACTRAN	Figure 15.0.4-2	Upper (see Figure 15.0.4-1)	3657[c][d][e]

Table 15.0.3-2 (Sheet 2 of 3)

Summary of Initial Conditions and Computer Codes Used

Section	Faults	Computer Codes Utilized	Reactivity Coefficients Assumed: Moderator Density ($\Delta k/gm/cm^3$)	Reactivity Coefficients Assumed: Doppler	Initial NSSS Thermal Power Output Assumed (MWt)
15.4	Reactivity and power distribution anomalies				
	Uncontrolled RCCA bank withdrawal from a subcritical or low power startup condition	TWINKLE, FACTRAN, THINC	Refer to Section 15.4.1.2	Coefficient is consistent with Doppler Defect of -0.94% ΔK	0
	Uncontrolled RCCA bank withdrawal at power	LOFTRAN	Figure 15.0.4-2 and 0.50	Lower and upper (see Figure 15.0.4-1)	358.5, 2151, and 3585[c][f]
	RCCA misalignment	THINC, ANC	N/A	N/A	3565[b]
	Startup of an inactive reactor coolant pump at an incorrect temperature	LOFTRAN, FACTRAN, THINC	0.50	Lower (see Figure 15.0.4-1)	2577[a][f]
	Chemical and volume control system malfunction that results in a decrease in the boron concentration in the reactor	N/A	N/A	N/A	0 and 3565[b]
	Spectrum of RCCA ejection accidents	TWINKLE, FACTRAN, THINC	Refer to Section 15.4.8.2	Coefficient is consistent with Doppler Defect of -0.94% ΔK	0 and 3636[d]

Table 15.0.3-2 (Sheet 3 of 3)

Summary of Initial Conditions and Computer Codes Used

Section	Faults	Computer Codes Utilized	Reactivity Coefficients Assumed: Moderator Density ($\Delta k/gm/cm^3$)	Reactivity Coefficients Assumed: Doppler	Initial NSSS Thermal Power Output Assumed (MWt)
	Steamline break with coincidental rod withdrawal at power	LOFTRAN	0.50	Lower (see Figure 15.0.4-1)	3585[c][f]
15.5	Increase in reactor coolant inventory				
	Inadvertent operation of the emergency core cooling system during power operation	LOFTRAN	0.0	Lower (see Figure 15.0.4-1)	3565[b][f]
15.6	Decrease in reactor coolant inventory				
	Inadvertent opening of a pressurizer safety or relief valve	LOFTRAN	Figure 15.0.4-2	Upper (see Figure 15.0.4-1)	3585[c][f]

(a) Minimum pump heat of 14 MWt assumed
(b) No pump heat (core thermal power) assumed
(c) Maximum pump heat of 20 MWt assumed
(d) 2% margin applied
(e) Standard Thermal Design Procedure (STDP) with a thermal design flow (TDF) of 93600 gpm/loop assumed
(f) Revised Thermal Design Procedure (RTDP) with a minimum measured flow (MMF) of 95600 gpm/loop assumed

Table 15.0.3-3

Nominal Values of Pertinent Plant Parameters
Used in the Accident Analyses

Parameter	STDP Values	RTDP Values
Thermal output of NSSS (including minimum thermal power generated by the RCPs, MWt)	3579	3579
Core inlet temperature (°F)	556.8	557.4
Vessel average temperature (°F)	588.4	588.4
RCS pressure (psia)	2250	2250
Reactor coolant flow per loop (gpm)	93600[a]	95600[b]
Steam flow from NSSS, total (lb/hr)	15,920,000	15,920,000
Steam pressure at steam generator outlet (psia)	950	950
Maximum steam moisture content (%)	0.25	0.25
Assumed feedwater temperature at steam generator inlet (°F)	446	446
Average core heat flux (Btu/hr-ft^2)	206,085	206,085

(a) Thermal design flow (TDF)
(b) Minimum measured flow (MMF)

Table 15.0.6-1

Trip Points and Time Delays to Trip Assumed in Accident Analyses

Trip Function	Limiting Trip Point Assumed in Analyses	Time Delays (s)
Power range high neutron flux, high setting	118%	0.5
Power range high neutron flux, low setting	35%	0.5
High neutron flux, P-8	84%	0.5
Source range neutron flux	N/A	0.5
OTΔT	Variable (See Figure 15.0.6-1)	6.0[a]
OPΔT	Variable (See Figure 15.0.6-1)	6.0[a]
High pressurizer pressure	2425 psig	2.0
Low pressurizer pressure	1920 psig	2.0
Low reactor coolant flow (from loop flow detectors)	87% loop flow	1.0
RCP undervoltage trip	68% nominal	1.5
Turbine trip	N/A	2.0
Low-low steam generator level	16.0% of narrow range level span[b] 30.0% of narrow range level span[c]	2.0 2.0
High steam generator level trip of the feedwater pumps and closure of feedwater system valves and turbine trip	100% of narrow range level span	2.5[d] 7.0[e]

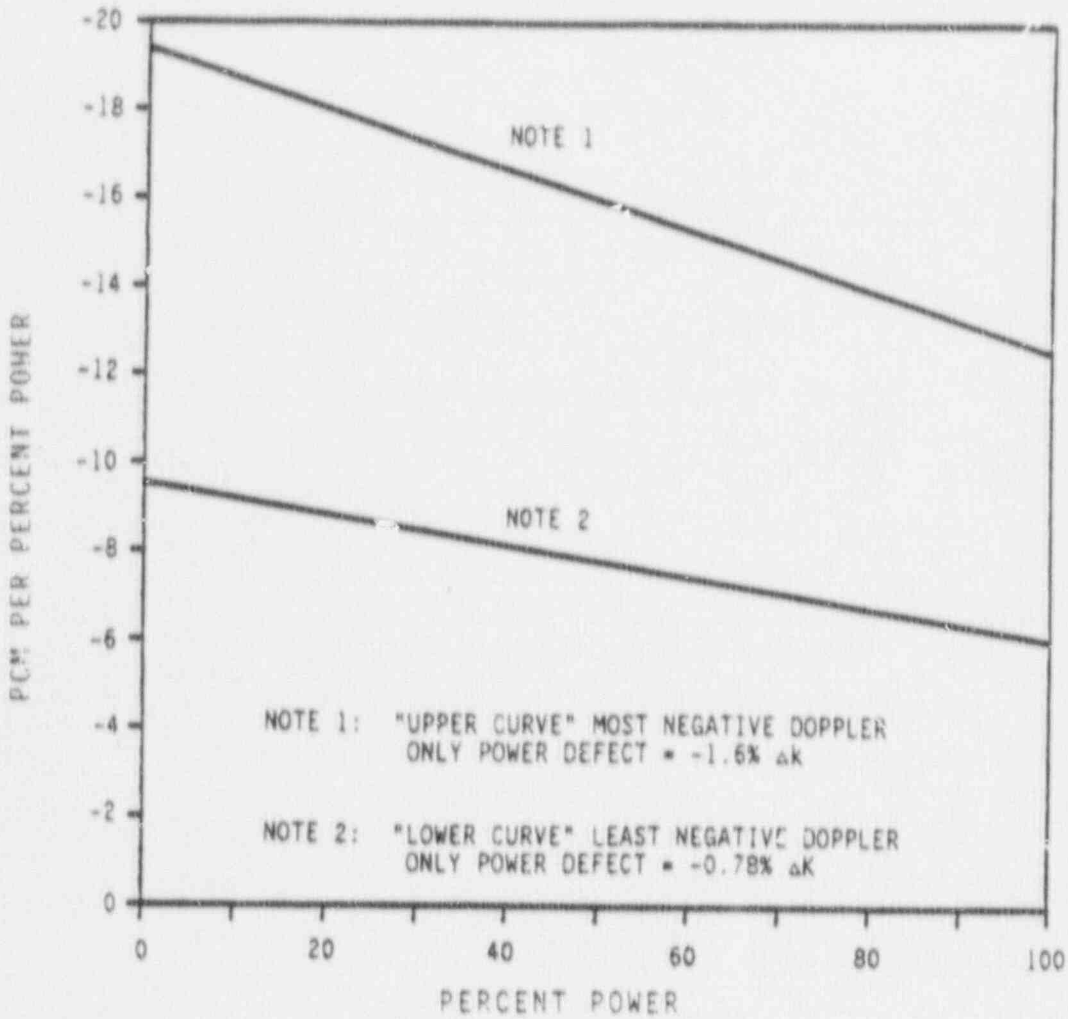
a. Total time delay (including resistance temperature detector (RTD) bypass loop fluid transport delay effect, bypass loop piping thermal capacity, RTD time response, and trip circuit channel electronics delay) from the time the temperature difference in the coolant loop exceeds the trip setpoint until the rods are free to fall.

b. Low-low level setpoint assumed for "Feedwater System Pipe Break" (15.2.8); the setpoint includes environmental errors.

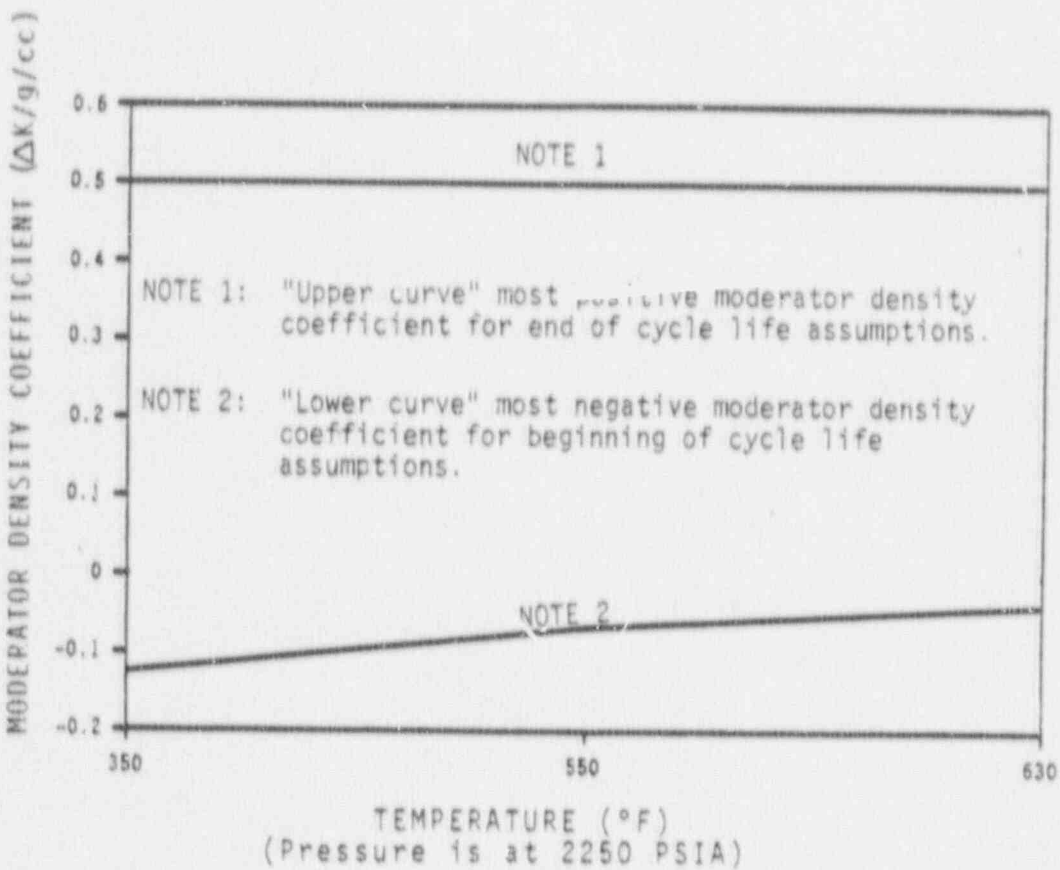
c. Low-low level setpoint assumed for "Loss of Non-Emergency AC Power to the Plant Auxiliaries" (15.2.6) and "Loss of Normal Feedwater Flow" (15.2.7); the setpoint does not include any environmental errors.

d. From time setpoint is reached to turbine trip.

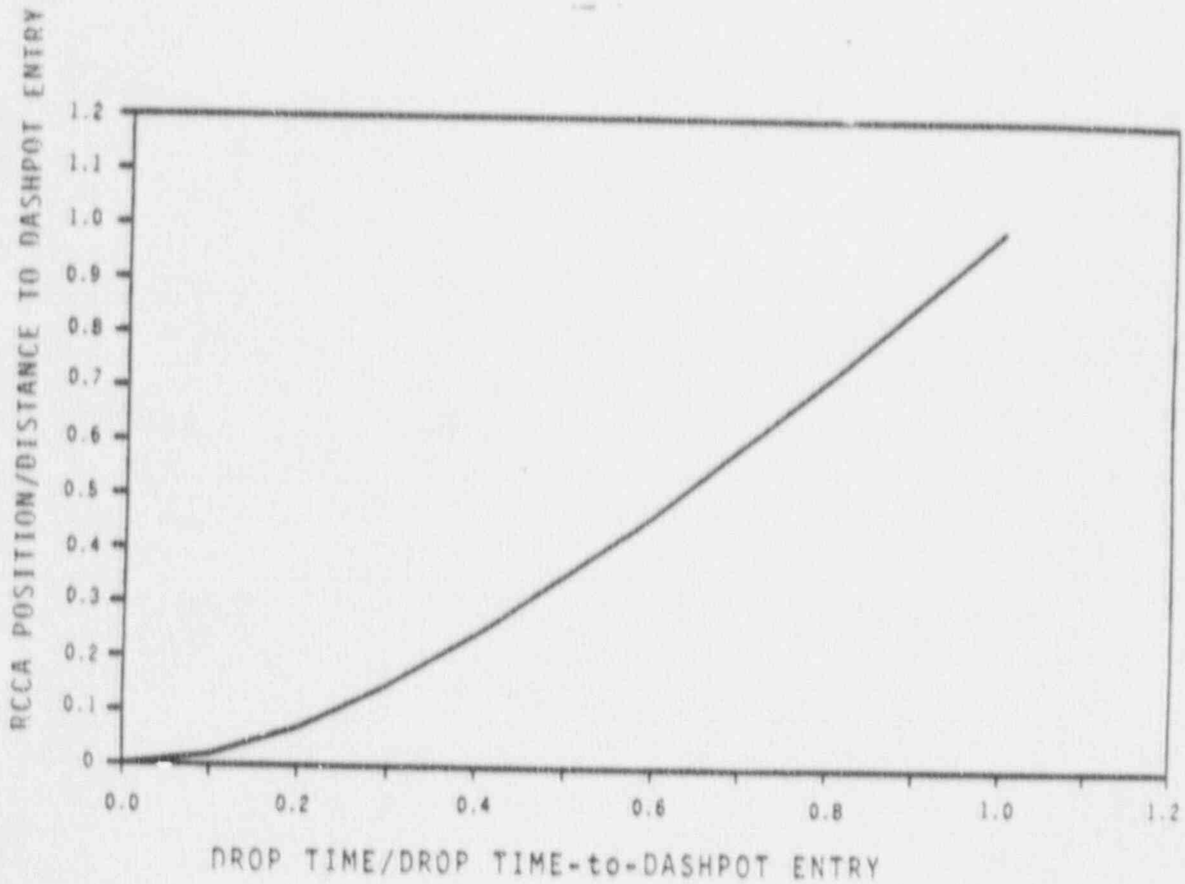
e. From time setpoint is reached to feedwater isolation.



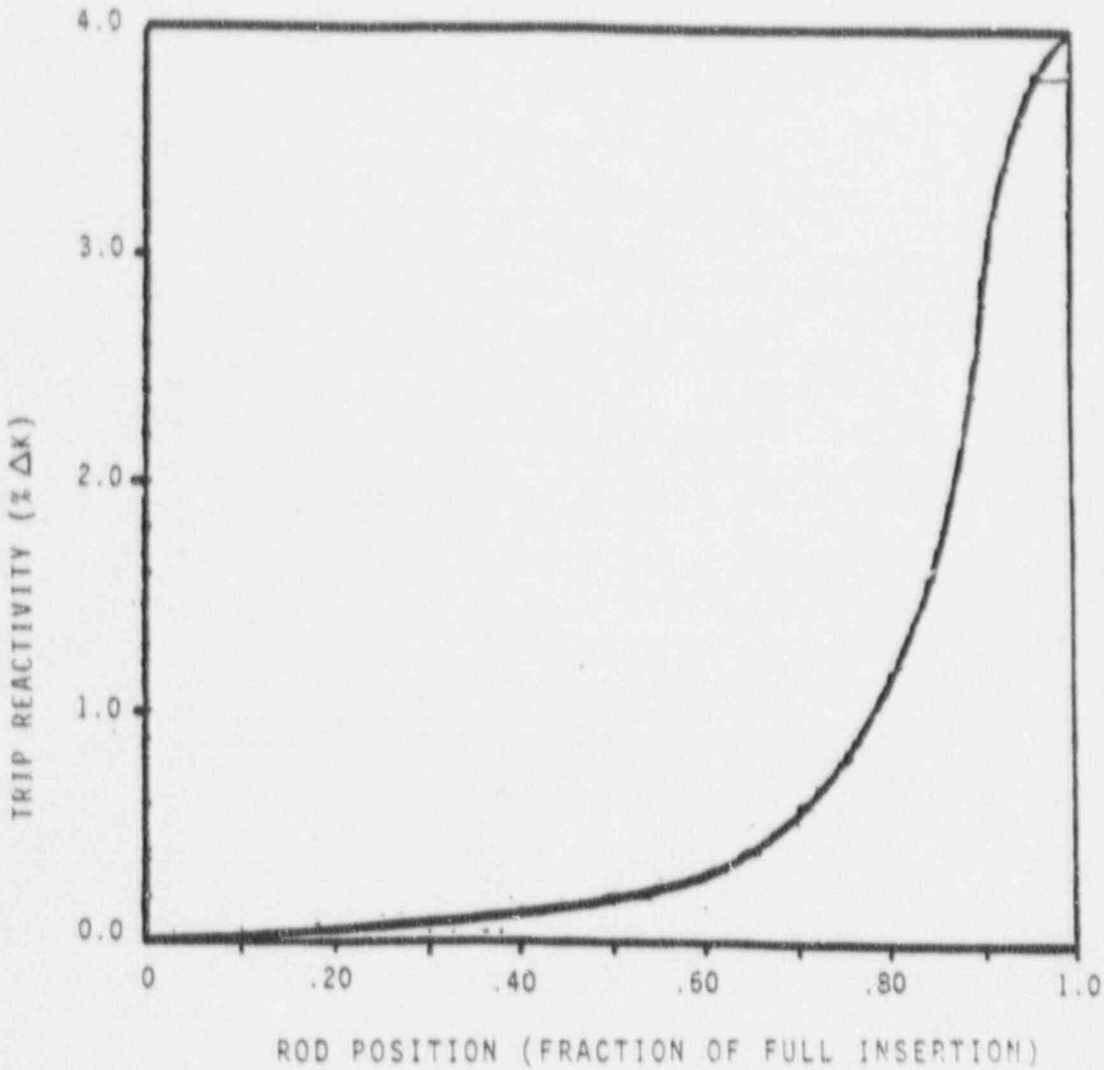
Georgia Power	VOGTLE ELECTRIC GENERATING PLANT UNIT 1 AND UNIT 2	DOPPLER POWER COEFFICIENT USED IN ACCIDENT ANALYSES
		FIGURE 15.0.4-1



Georgia Power	VOGTLE ELECTRIC GENERATING PLANT UNIT 1 AND UNIT 2	MODERATOR DENSITY COEFFICIENT USED IN ACCIDENT ANALYSES
		FIGURE 15.0.4-2



Georgia Power	VOGTLE ELECTRIC GENERATING PLANT UNIT 1 AND UNIT 2	ROD POSITION VS. TIME ON REACTOR TRIP FIGURE 15.0.5-1

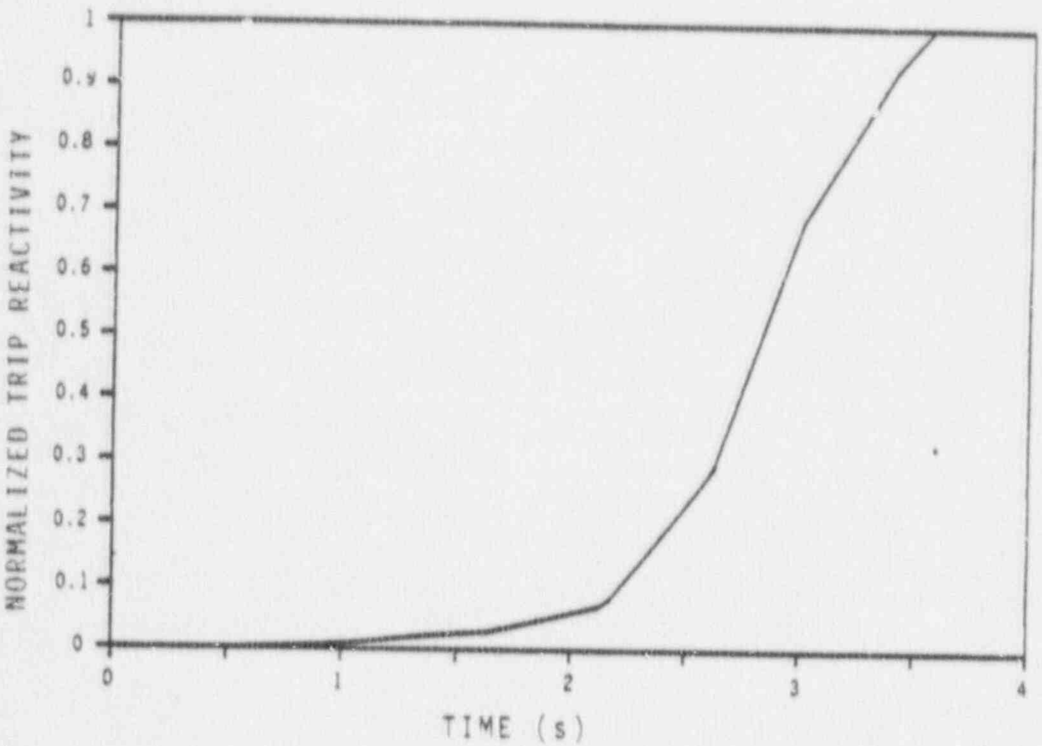


Georgia Power

VOGTLE
ELECTRIC GENERATING PLANT
UNIT 1 AND UNIT 2

RCCA REACTIVITY WORTH VS. FRACTION FULL INSERTION

FIGURE 15.0.5-2

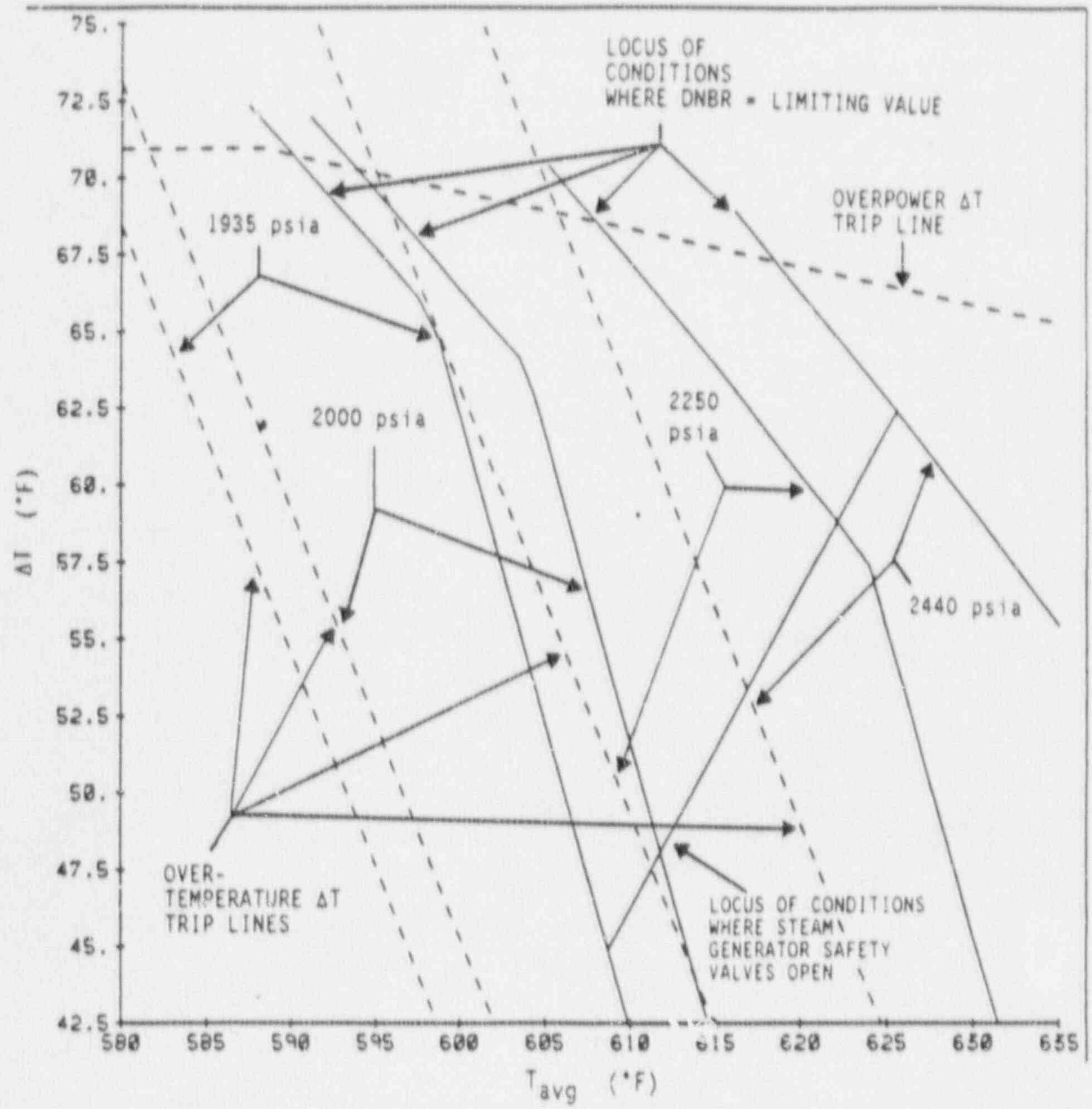


Georgia Power

VOGTLE
ELECTRIC GENERATING PLANT
UNIT 1 AND UNIT 2

NORMALIZED RCCA BANK WORTH VS. TIME AFTER TRIP

FIGURE 15.0.5-3

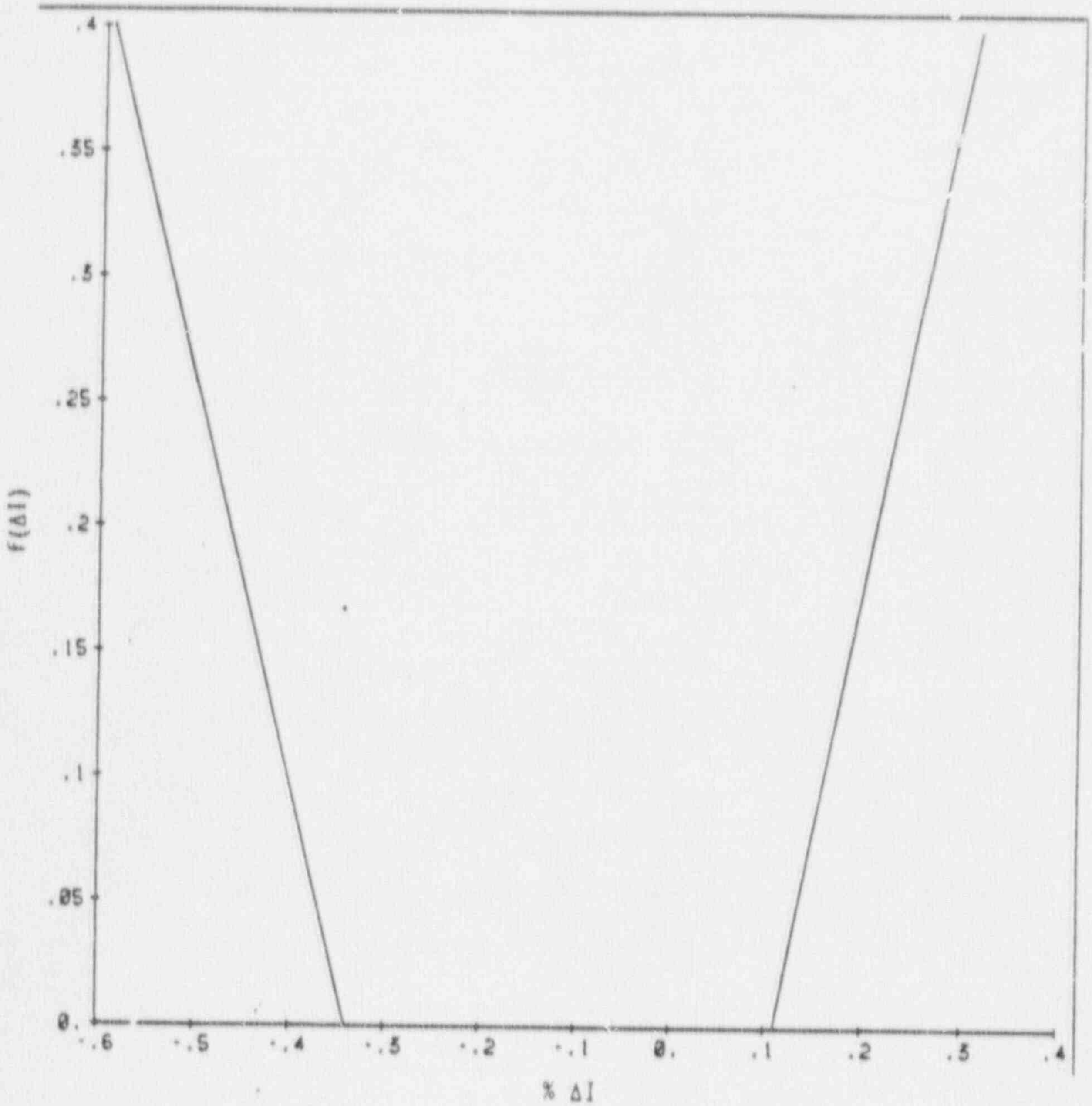


Georgia Power

VOGTLE
ELECTRIC GENERATING PLANT
UNIT 1 AND UNIT 2

OVERTEMPERATURE AND OVERPOWER
ΔT PROTECTION

FIGURE 15.0.C-1



Georgia Power

VOGTLE
ELECTRIC GENERATING PLANT
UNIT 1 AND UNIT 2

f(ΔI) PENALTY

FIGURE 15.0.6-2

15.1 Increase in Heat Removal by the Secondary System

Many events can result in an increase in heat removal from the RCS by the secondary system. This section presents several limiting cases of such events.

15.1.1 Feedwater System Malfunctions that Result in a Decrease in Feedwater Temperature

15.1.1.1 Introduction

Reductions in feedwater temperature will result in an increase in core power by initially decreasing reactor coolant temperature. The thermal capacity of the secondary plant and of the RCS attenuates such transients. The high neutron flux trip, OTΔT trip, and OPΔT trip prevent any power increase which could lead to a DNBR less than the limit value.

A low-pressure feedwater train or a high-pressure heater out of service may cause a reduction in feedwater temperature. If a spurious heater drain pump trips, there is a sudden reduction in feedwater inlet temperature to the steam generators. At power, this increased subcooling will create a greater load demand on the RCS.

With the plant at no-load conditions, the addition of cold feedwater will cause a decrease in RCS temperature and a reactivity insertion due to the effects of the negative moderator temperature coefficient of reactivity; however, the rate of energy change is reduced as load and feedwater flow decrease, so the transient is less severe than the full power case.

The net effect on the RCS due to a reduction in feedwater temperature is similar to the effect of increasing secondary steam flow; i.e., the reactor will reach a new equilibrium condition at a power level corresponding to the new steam generator ΔT.

This is an American Nuclear Society (ANS) Condition II incident.

15.1.1.2 Method of Analysis

This transient is analyzed by computing conditions at the feedwater pump inlet following the removal of a low-pressure feedwater train or a high-pressure heater from service. These feedwater conditions are then used to recalculate a heat balance for that RCS loop containing a reduced number of feedwater heaters in service. This heat balance gives the new feedwater conditions at the steam generator inlet.

The following assumptions are made:

A. Plant initial power level corresponds to design NSSS thermal output at the uprated condition

B. One string of feedwater heaters is isolated

This accident analysis employs the RTDP with the initial conditions shown in Tables 15.0.3-2 and 15.0.3-3.

No single active failure in any plant systems or equipment will adversely affect the consequences of the accident.

15.1.1.3 Results

Isolation of a string of low-pressure feedwater heaters causes a reduction in feedwater temperature, which increases the thermal load on the primary system. The calculated reduction in feedwater temperature is less than 30°F. This reduction in feedwater temperature results in an increase in heat load on the primary system of less than 10 percent of full power. Thus, increased thermal load due to a spurious heater drain pump trip would result in a transient very similar (but of a reduced magnitude) to that presented in Section 15.1.3 for an excessive increase in secondary steam flow transient. The consequences of a 10 percent step load increase are evaluated in Section 15.1.3; therefore, there is no presentation of the results of this analysis.

15.1.1.4 Conclusions

The decrease in feedwater temperature transient is less severe than the increase in secondary steam flow event (Section 15.1.3). Based on results presented in Section 15.1.2 and Section 15.1.3, the applicable acceptance criteria for the decrease in feedwater temperature event have been met. The conclusions presented in the FSAR remain valid.

15.1.2 Feedwater System Malfunctions that Result in an Increase in Feedwater Flow

15.1.2.1 Introduction

Addition of excessive feedwater will cause an increase in core power by decreasing reactor coolant temperature. The thermal capacity of the secondary plant and of the RCS attenuates such transients. The high neutron flux trip, OPΔT trip and OTΔT trip prevent any power increase which could lead to DNBR less than the minimum allowable value in the event that the steam generator high-high water level protection does not actuate.

The full opening of a feedwater control valve due to a feedwater control system malfunction or an operator error may cause excessive feedwater flow. At power conditions, 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 temperature coefficient of reactivity.

The steam generator high-high water level trip signal does the following:

- o Closes the feedwater valves
- o Closes the feedwater pump discharge valves
- o Trips the turbine

- o Trips the main feedwater pumps
- o Prevents continuous addition of excessive feedwater

The turbine trip signal initiates a reactor trip.

This is an ANS Condition II incident.

15.1.2.2 Method of Analysis

The analysis of the excessive heat removal due to a feedwater system malfunction transient uses the detailed digital computer code LOFTRAN (Reference 3). This code simulates the neutron kinetics, RCS, pressurizer, pressurizer relief and safety valves, pressurizer spray, steam generator, steam generator safety valves, and feedwater control system. The code computes pertinent plant variables including temperatures, pressures, and power level.

The purpose of the system analysis is to demonstrate acceptable plant behavior in the event of an excessive feedwater addition due to a control system malfunction or operator error which allows a feedwater control valve to open fully. Cases analyzed assuming a conservatively large negative moderator temperature coefficient follow:

- o Accidental full opening of a feedwater control valve with the reactor just critical at zero power conditions (Mode 3) with the reactor in manual rod control
- o Accidental full opening of a feedwater control valve with the reactor just critical at zero power conditions (Mode 2) with the reactor in both automatic and manual rod control
- o Accidental full opening of one feedwater control valve with the reactor at full power assuming automatic and manual rod control

The calculation of the reactivity insertion rate following a feedwater system malfunction uses the following assumptions:

A. For the feedwater control valve accident at full power, one feedwater control valve malfunctions, which results in a step increase to 157 percent of nominal feedwater flow to one steam generator.

B. For the feedwater control valve accident at no-load conditions (Mode 2), a feedwater control valve malfunction occurs which results in a step increase in flow to one steam generator from zero to 225 percent of the nominal full load value for one steam generator.

C. For the feedwater control valve accident at no-load conditions (Mode 3), a feedwater control valve malfunction occurs which results in a step increase in flow to one steam generator from zero to 50 percent of the nominal full load value for one steam generator. The flow is limited by a 6 inch pipe in the line from the condensate pump to the steam generators through which all the feedwater to the steam generators must pass.

D. For the no-load condition, feedwater temperature is at a conservatively low value of 32°F.

E. No credit is taken for the heat capacity of the RCS and steam generator thick metal in attenuating the resulting plant cooldown.

F. The feedwater flow resulting from a fully open control valve terminates by a steam generator high-high level trip signal which closes all feedwater control and isolation valves, trips the main feedwater pumps, and trips the turbine. The turbine trip signal initiates a reactor trip.

G. The analysis assumes the RCS flow equivalent to the operation of four RCPs.

The analysis uses RTDP methodology in the determination of initial reactor power, pressure, and RCS temperature (see Tables 15.0.3-2 and 15.0.3-3). The analysis does not require normal reactor control system and engineered safety systems to function. The reactor protection system function at full power trips the reactor due to overpower or turbine trip on high-high steam generator water level conditions.

No single active failure in any plant systems or equipment will adversely affect the consequences of the accident.

15.1.2.3 Results

In the case of an accidental full opening of one feedwater control valve with the reactor at zero power (Mode 3) and the above mentioned assumptions, the maximum reactivity insertion rate is less than the maximum reactivity insertion rate analyzed in Section 15.4.1 for uncontrolled rod cluster control assembly bank withdrawal from a subcritical or low-power startup condition; therefore, this section does not present the results of the analysis. Note that the analysis in Section 15.4.1 conservatively assumes only two RCPs in operation for the DNBR calculation. This assumption bounds the feedwater malfunction event with respect to the DNBR.

The analysis of the Mode 2 case uses the TWINKLE, FACTRAN, and THINC codes to evaluate the resulting reactivity insertion rate as described in Section 15.4.1. The Mode 2 analysis assumes all four RCPs to be in operation. The resulting minimum calculated DNBR is above the limit value and is less limiting than the analysis presented in Section 15.4.1; therefore, this section does not present the results of the analysis.

Note that if the incident occurs with the unit just critical at no-load, the reactor may trip by the power range high neutron flux trip (low setting) set at approximately 25 percent nominal full power.

The full power case (maximum reactivity feedback coefficients, manual rod control) results in the greatest power increase. Assuming automatic rod control results in a less severe transient.

When the steam generator water level in the faulted loop reaches the high-high level setpoint, the feedwater control valves and feedwater pump discharge valves are automatically closed and the main feedwater pumps trip. This prevents continuous addition of feedwater. A turbine trip and a resulting reactor trip are also initiated.

Transient results show the increase in nuclear power associated with the increased thermal load on the reactor (See Figure 15.1.2-1 and Figure 15.1.2-2). The DNBR does not drop below the limit value. Following the reactor trip, the plant approaches a stabilized condition; standard plant shutdown procedures then apply to further cool down the plant.

Since the power level rises during the excessive feedwater flow incident, the fuel temperatures will also rise until after reactor trip occurs. The core heat flux lags behind the neutron flux response due to the fuel rod thermal time constant; hence, the peak heat flux does not exceed 118 percent of its nominal value (i.e., the assumed high neutron flux trip setpoint). Thus, the peak fuel temperature will remain well below the fuel melting temperature.

The transient results show that DNB does not occur at any time during the excessive feedwater flow incident; thus, there is no reduction in the ability of the primary coolant to remove heat from the fuel rods. The fuel cladding temperature, therefore, does not rise significantly above its initial value during the transient.

The calculated sequence of events for the increase in feedwater flow for the full power case is shown in Table 15.1.2-1.

15.1.2.4 Conclusions

The results of the analysis show that the DNBRs encountered for an excessive feedwater addition at power are above the limit value. Additionally, for an excessive feedwater addition in Mode 3, the resulting reactivity insertion rate is less than the value used in the rod withdrawal from subcritical analysis (Section 15.4.1). For an excessive feedwater addition in Mode 2, the resulting reactivity insertion rate was analyzed and the minimum DNBR is above the limit value and is less limiting than the analysis presented in Section 15.4.1. The conclusions presented in the FSAR remain valid.

15.1.3 Excessive Increase in Secondary Steam Flow

15.1.3.1 Introduction

A rapid increase in steam flow that causes a power mismatch between the reactor core power and the steam generator load demand defines an excessive load increase incident. The reactor control system design accommodates a 10 percent step load increase and a 5 percent per minute ramp load increase in the range of 15 to 100 percent of full power. Any loading rate more than these values may cause a reactor trip actuated by the reactor protection system. Sections 15.1.4 and 15.1.5 discuss steam flow increases greater than 10 percent.

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.

Reactor coolant condition signals control turbine bypass to the condenser during power operation, i.e., high reactor coolant temperature indicates a need for turbine bypass by using steam dumps. A single controller malfunction does not cause turbine bypass; an interlock blocks the opening of the steam dump valves unless a large turbine load decrease or turbine trip occurs.

The following reactor protection system signals protect against an excessive load increase accident:

o OPΔT

o OTΔT

o Power range high neutron flux

o Low pressurizer pressure

This is an ANS Condition II incident.

15.1.3.2 <u>Method of Analysis</u>

The analysis of this accident uses the LOFTRAN code (Reference 3). This code simulates the neutron kinetics, RCS, pressurizer, pressurizer relief and safety valves, pressurizer spray, steam generator, steam generator safety valves, and feedwater system. The code computes pertinent plant variables including temperatures, pressures, and power level.

The analysis includes four cases that demonstrate the plant behavior following a 10 percent step load increase from rated load. These cases are as follows:

A. Manual rod control with minimum reactivity feedback

B. Manual rod control with maximum reactivity feedback

C. Automatic rod control with minimum reactivity feedback

D. Automatic rod control with maximum reactivity feedback

For the minimum moderator feedback cases, the analysis assumes that the core has a zero moderator temperature coefficient of reactivity and the least negative Doppler-only power coefficient curve. This results in the least inherent transient response capability. The zero moderator temperature coefficient of reactivity bounds a positive moderator temperature coefficient for this cooldown event. For the 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. For the cases with automatic rod control, no credit was taken for ΔT trips on overtemperature or overpower in order to demonstrate the inherent transient capability of the plant. Under actual operating conditions, such a trip may occur after which the plant would quickly stabilize.

The analysis assumes a 10 percent step increase in steam demand, and the analysis of all the cases does not take credit for pressurizer heaters. The analysis of this accident uses RTDP as described in Reference 7. The analysis assumes nominal values for the initial reactor power, pressure, and RCS temperature. The limit DNBR includes uncertainties in initial conditions. Tables 15.0.3-2 and 15.0.3-3 show the plant characteristics and initial conditions.

The analysis does not require normal reactor control systems and engineered safety systems to function. The analysis assumes the reactor protection system to be operable; however, due to the error allowances assumed in the setpoints, a reactor trip does not occur. No single active failure will prevent the reactor protection system from performing its intended function.

No single active failure in any plant systems or equipment will adversely affect the consequences of the accident.

15.1.3.3 Results

Figures 15.1.3-1 through 15.1.3-4 illustrate the transient with the reactor in the manual rod control mode. As expected, for the 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 maximum moderator feedback manual rod controlled case, there is a large increase in reactor power due to the moderator feedback. A reduction in DNBR occurs, but DNBR remains above the limit value.

Figures 15.1.3-5 through 15.1.3-8 illustrate the transient assuming the reactor is in the automatic rod control mode. Both the minimum and 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 where a reduction in power can occur by following normal plant operating procedures. Note that due to the measurement errors assumed in the setpoints, it is possible that reactor trip could occur for the automatic rod control cases. The plant would then reach a stabilized condition following the event.

The excessive load increase incident is an overpower transient for which the fuel temperatures will rise. Reactor trip does not occur for 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, there is no reduction in the ability of the primary coolant to remove heat from the fuel rod. Thus, the fuel cladding temperature does not rise significantly above its initial value during the transient.

The calculated sequence of events for the excessive load increase incident is shown in Table 15.1.2-1.

15.1.3.4 Conclusions

The analysis presented shows that for a 10 percent step load increase, the DNBR remains above the limit value. The plant reaches a stabilized condition following the load increase. The conclusions presented in the FSAR remain valid.

15.1.4 Inadvertent Opening of a Steam Generator Relief or Safety Valve

Section 15.1.4 of the FSAR describes the inadvertent opening of a steam generator relief or safety valve accident. Performance of the analysis is at hot zero power conditions with the control rods fully inserted in the core and the most reactive rod stuck out. Since the analysis of this accident does not use the RTDP, the only significant impact of the design changes associated with the VANTAGE 5 fuel reload transition is that of the increase in the $F^N_{\Delta H}$ peaking factor. The analysis performed for steam system piping failure (Section 15.1.5) which evaluates the increase in the $F^N_{\Delta H}$ peaking factor, bounds this transient. The safety analysis DNBR limit is met and the conclusions of the FSAR remain valid.

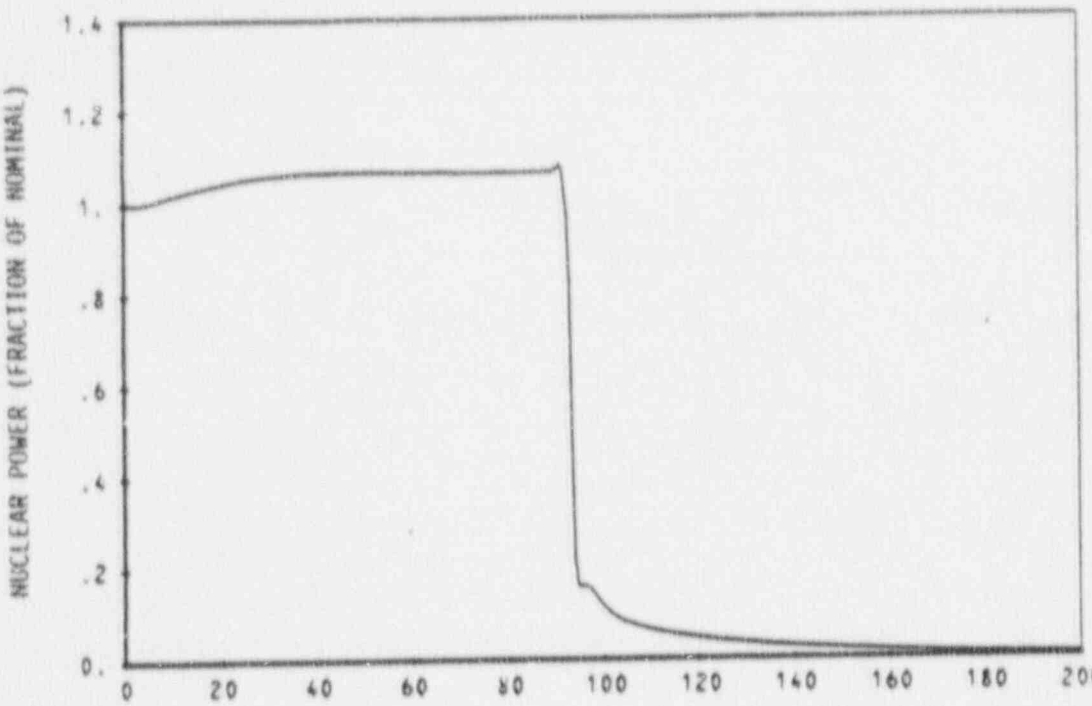
15.1.5 Steam System Piping Failure

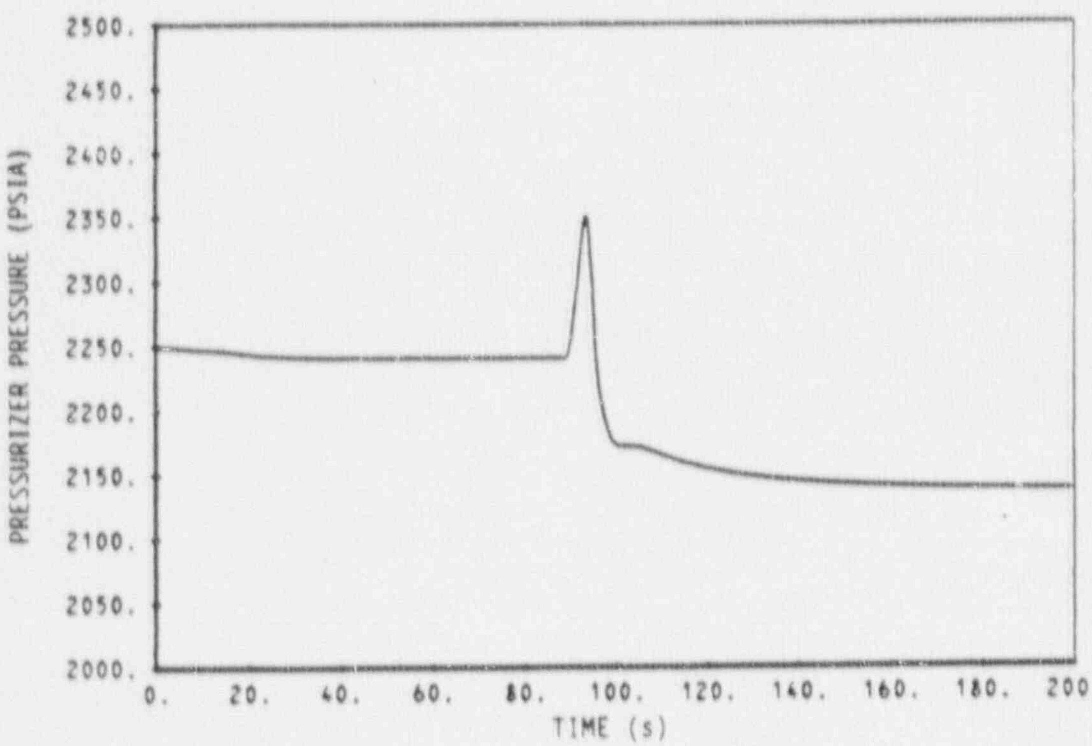
Section 15.1.5 of the FSAR describes the steamline break accident. The analysis is performed at hot zero power conditions with the control rods fully inserted in the core and the most reactive rod stuck out. Since the analysis of this accident does not use the RTDP, the only significant impact of the design changes associated with the VANTAGE 5 fuel reload transition is that of the increase in the $F^N_{\Delta H}$ peaking factor. The evaluation of the increase in the $F^N_{\Delta H}$ peaking factor concludes that the safety analysis DNBR limit is met; therefore, the conclusions of the FSAR remain valid.

Table 15.1.2-1

Time Sequence of Events for Incidents That Result in an Increase in Heat Removal by the Secondary System

Accident	Event	Time Delays (s)
Feedwater system malfunctions that result in an increase in feedwater flow (full power)	One main feedwater control valve fails fully open	0.0
	Minimum DNBR occurs	83.0
	High-high steam generator water level signal generated	86.2
	Turbine trip occurs due to high-high steam generator level	88.7
	Reactor trip occurs	90.7
	Feedwater isolation valves close automatically	93.2
Excessive increase in secondary steam flow		
1. Manual rod control (minimum moderator feedback)	10 percent step load increase	0.0
	Equilibrium conditions reached (approximate time only)	150
2. Manual rod control (maximum moderator feedback)	10 percent step load increase	0.0
	Equilibrium conditions reached (approximate time only)	50
3. Automatic rod control (minimum moderator feedback)	10 percent step load increase	0.0
	Equilibrium conditions reached (approximate time only)	125
4. Automatic rod control (maximum moderator feedback)	10 percent step load increase	0.0
	Equilibrium conditions reached (approximate time only)	50





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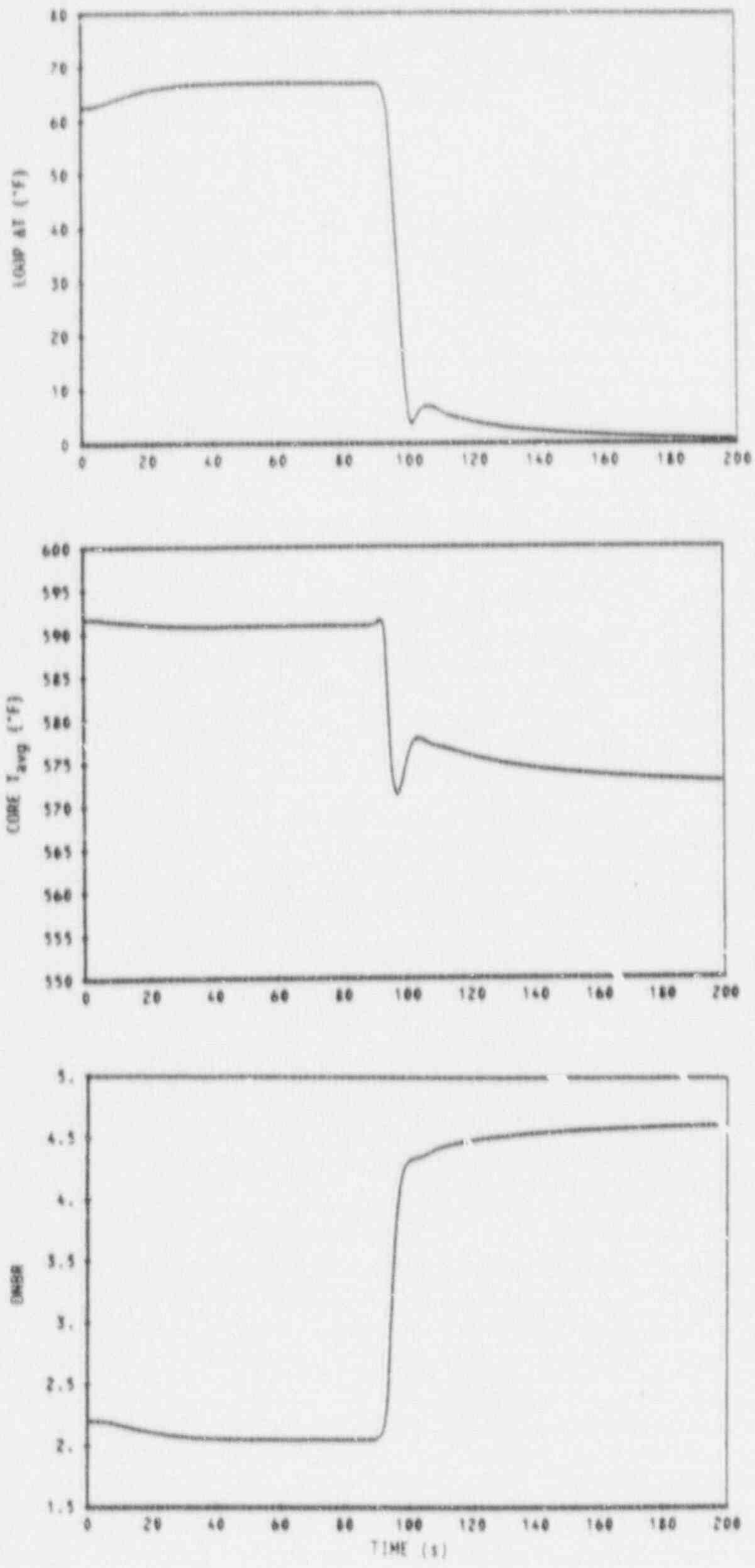
TRANSIENTS FOR FEEDWATER CONTROL VALVE MALFUNCTION

FIGURE 15.1.2-1



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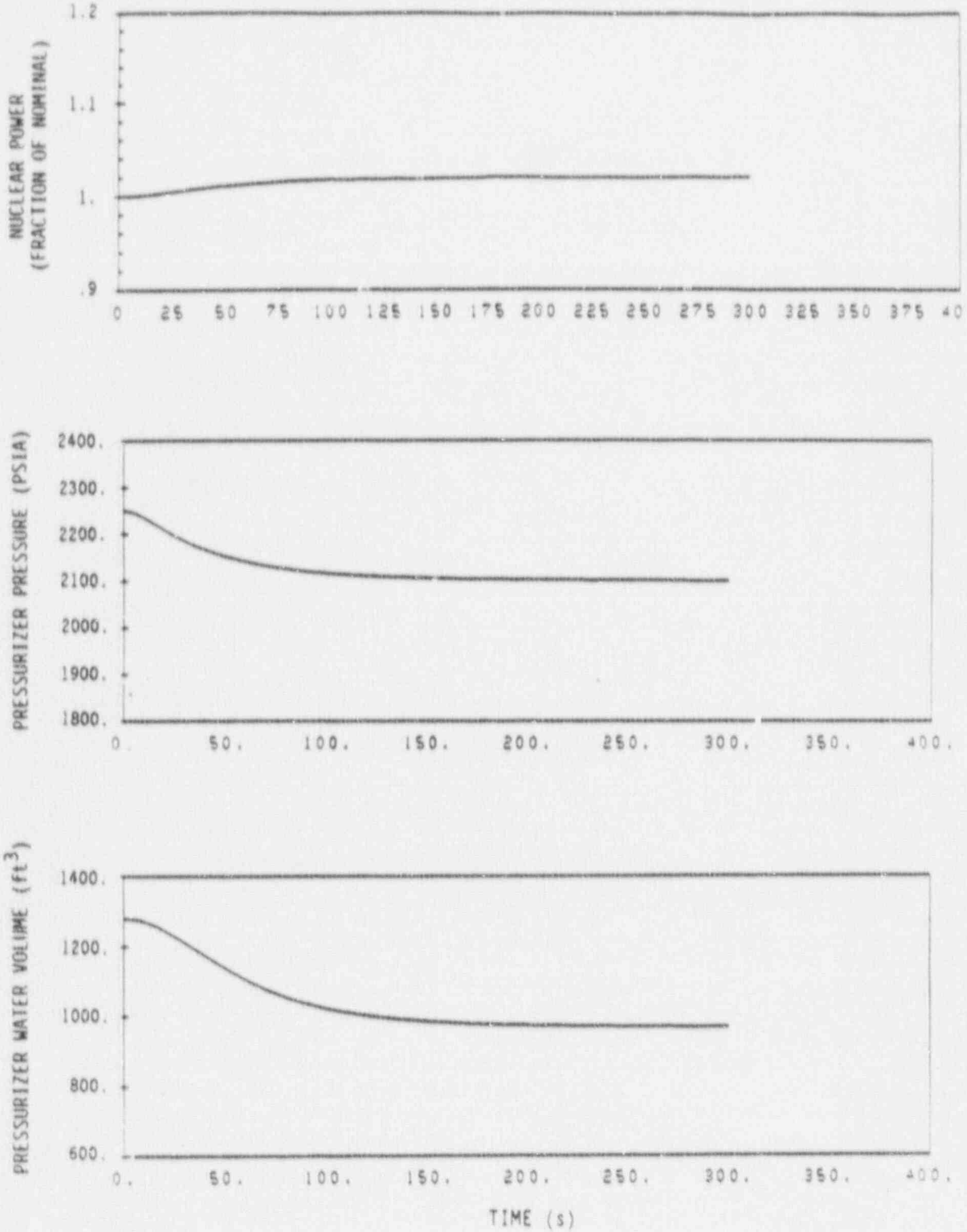
TRANSIENTS FOR FEEDWATER CONTROL VALVE MALFUNCTION

FIGURE 15.1.2-2



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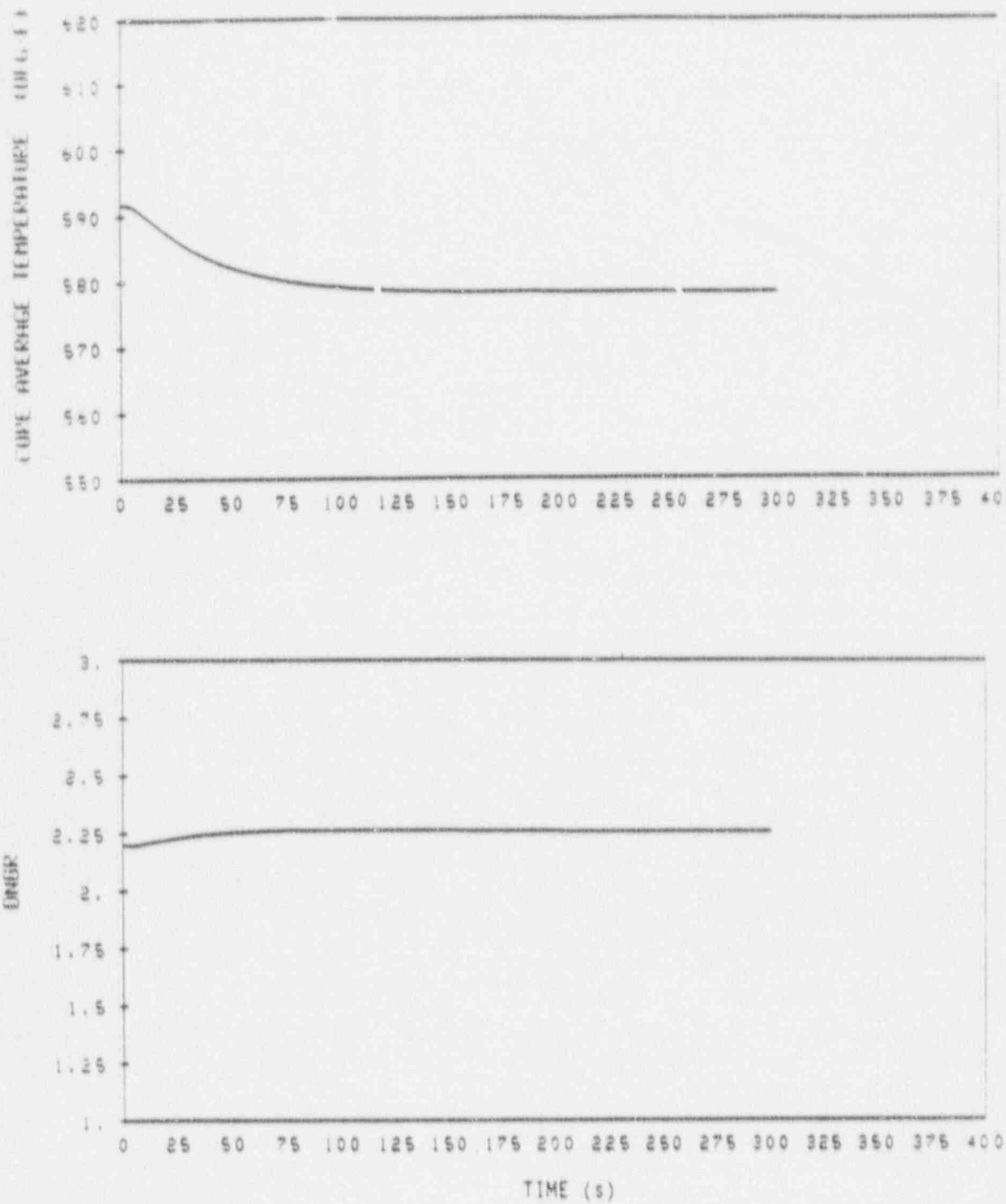
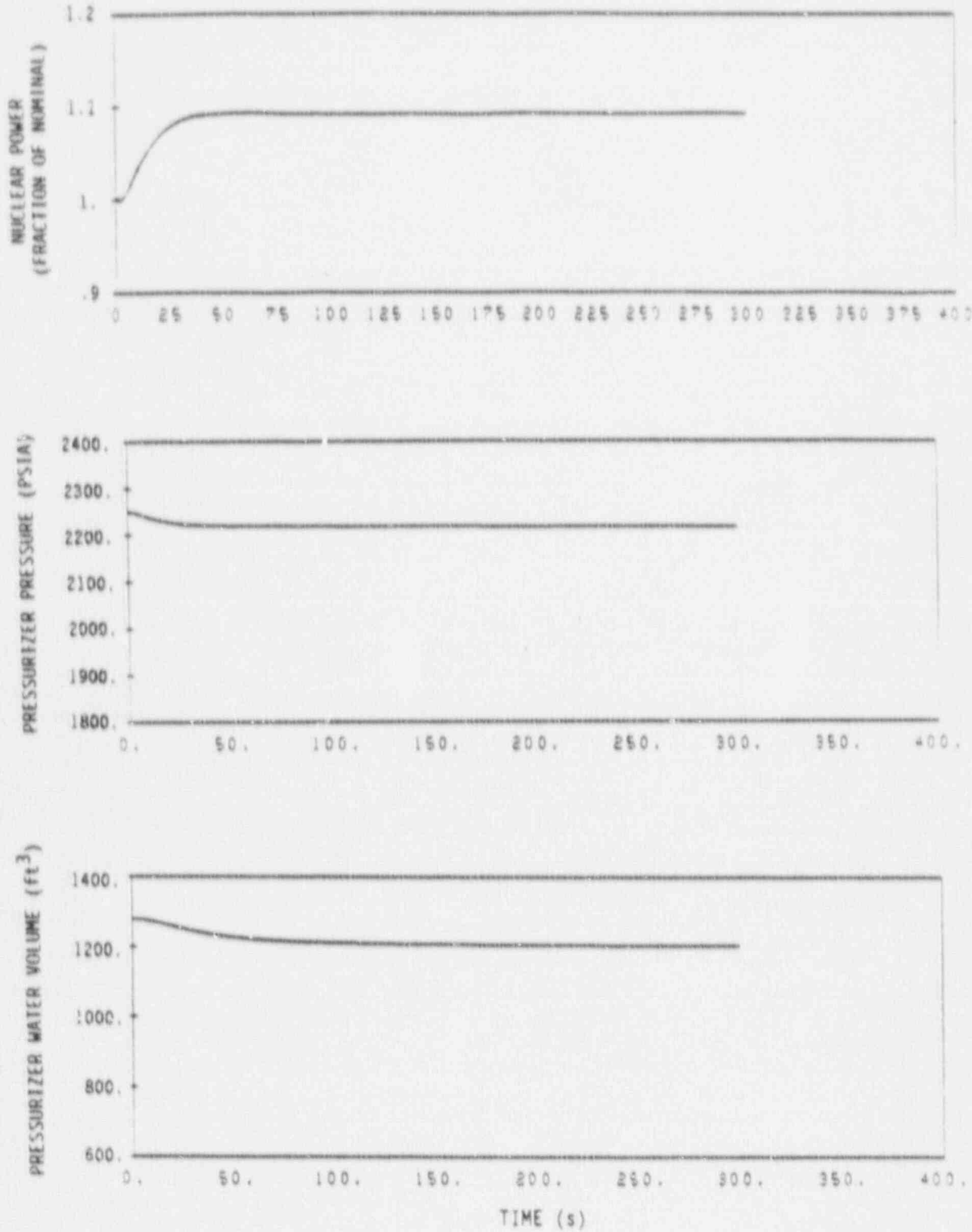
10% STEP LOAD INCREASE, MIN. MODERATOR FEEDBACK, MANUAL ROD CONTROL

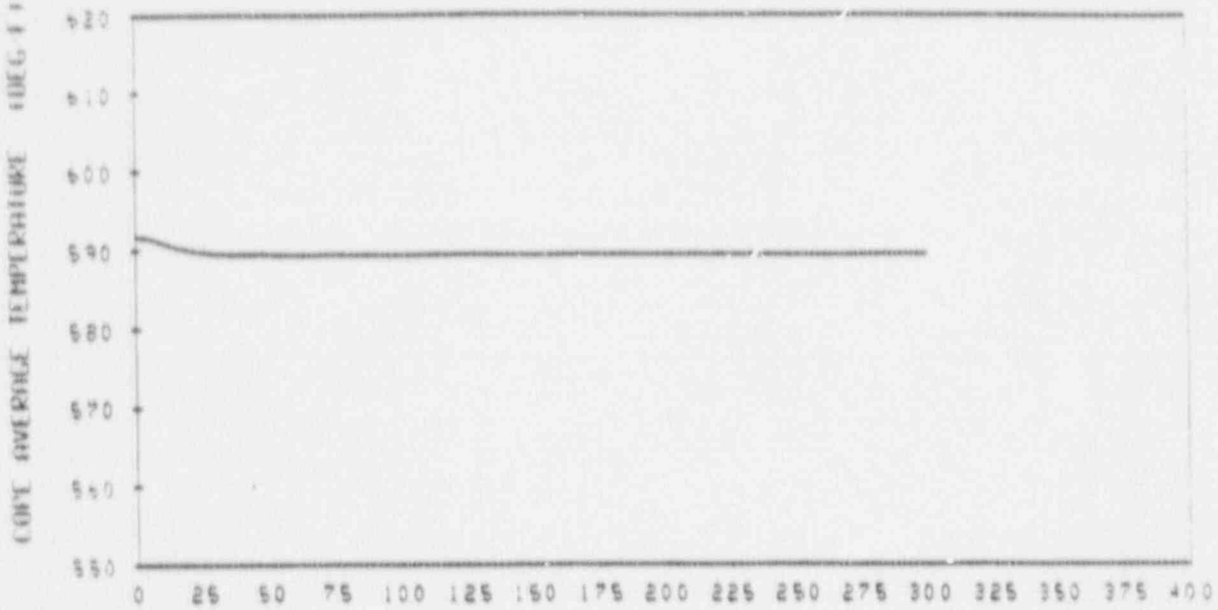
FIGURE 15.1.3-1

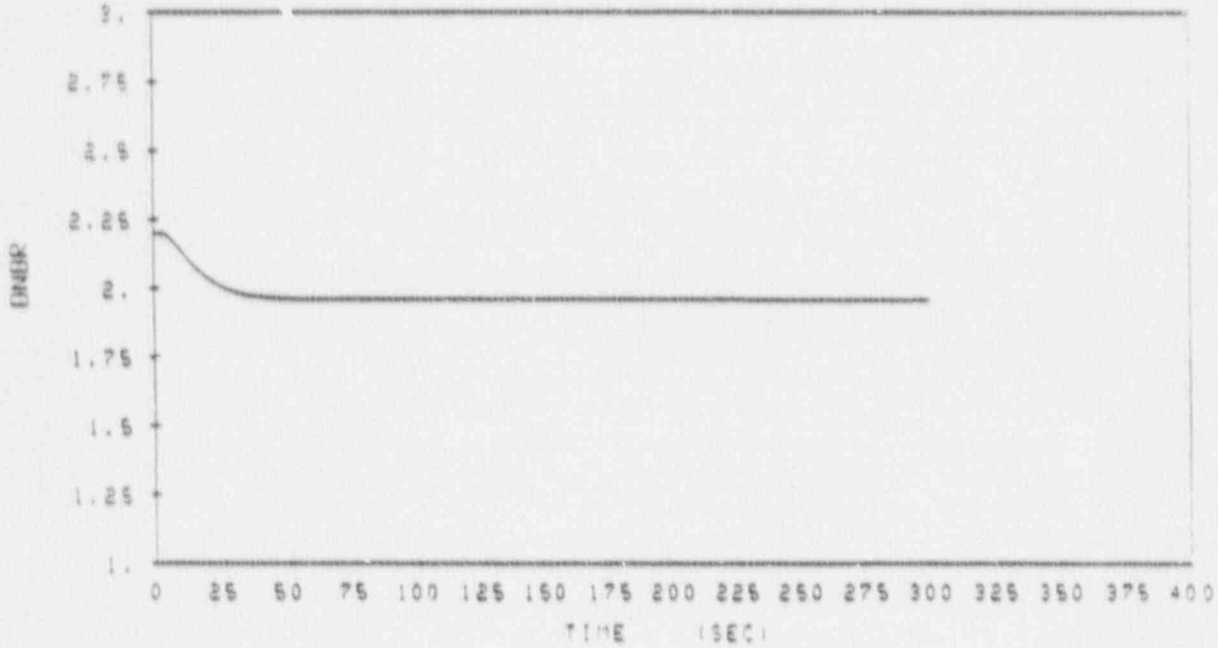


Georgia Power VOGTLE ELECTRIC GENERATING PLANT UNIT 1 AND UNIT 2	10% STEP LOAD INCREASE, MIN. MODERATOR FEEDBACK, MANUAL ROD CONTROL
	FIGURE 15.1.3-2



Georgia Power	VOGTLE ELECTRIC GENERATING PLANT UNIT 1 AND UNIT 2	10% STEP LOAD INCREASE, MAX. MODERATOR FEEDBACK, MANUAL ROD CONTROL
		FIGURE 15.1.3-3





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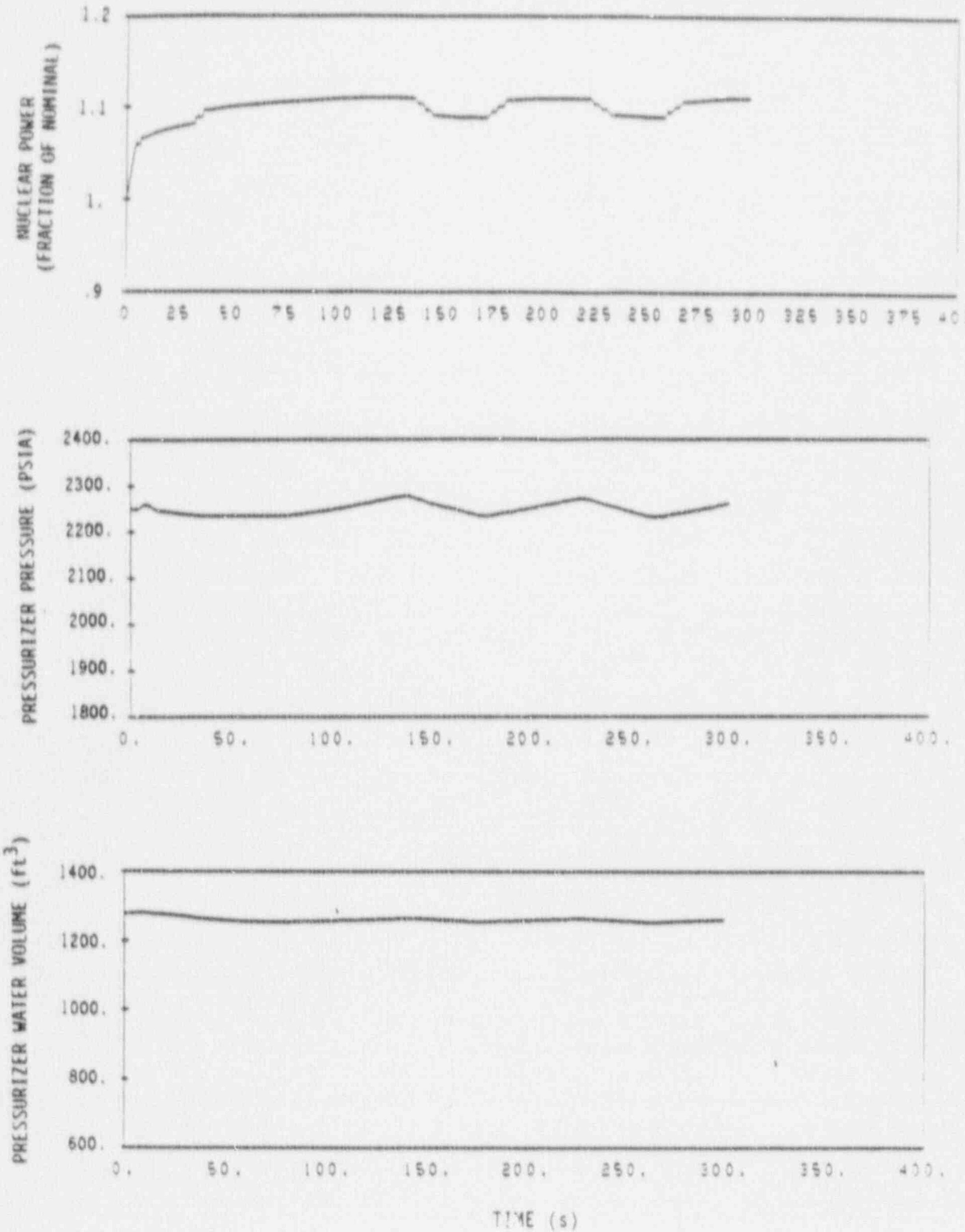
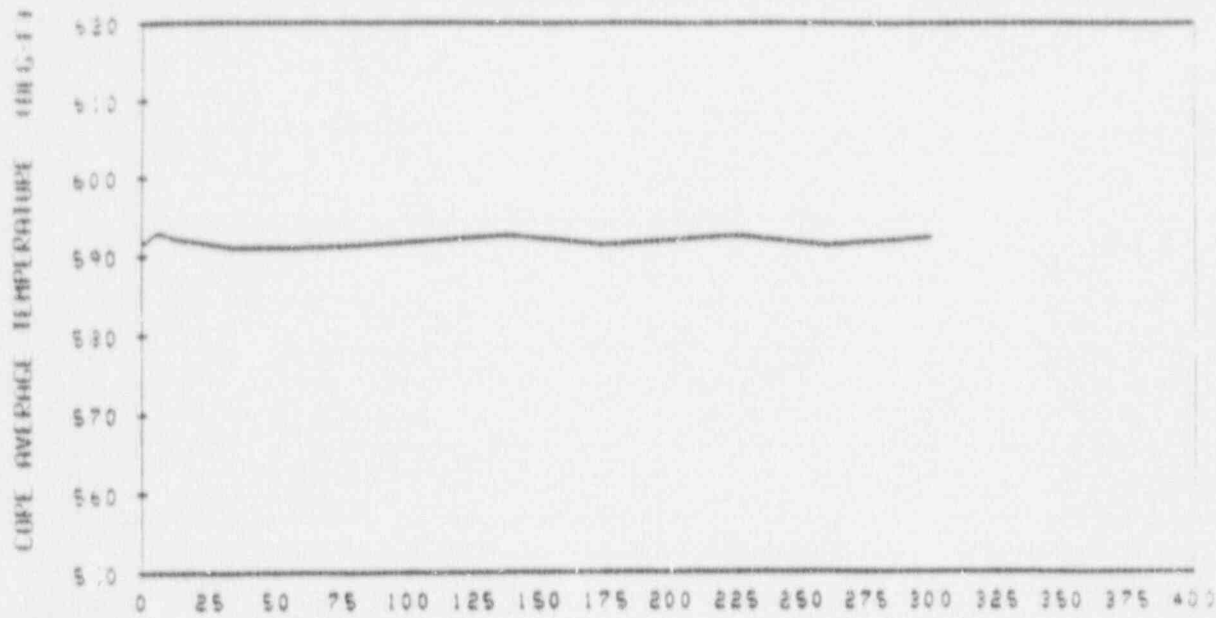
UNIT 1 AND UNIT 2

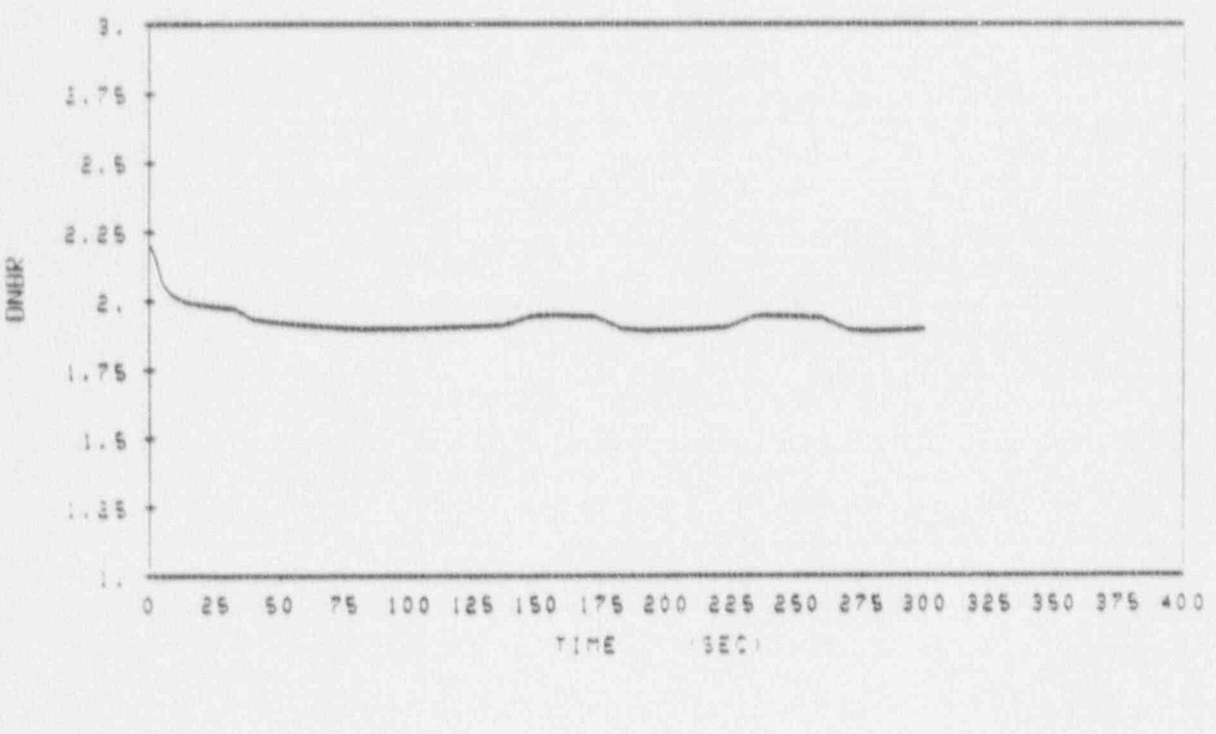
10% STEP LOAD INCREASE, MAX. MODERATOR FEEDBACK, MANUAL ROD CONTROL

FIGURE 15.1.3-4



Georgia Power VOGTLE ELECTRIC GENERATING PLANT UNIT 1 AND UNIT 2	10% STEP LOAD INCREASE, MIN. MODERATOR FEEDBACK, AUTOMATIC ROD CONTROL
	FIGURE 15.1.3-5





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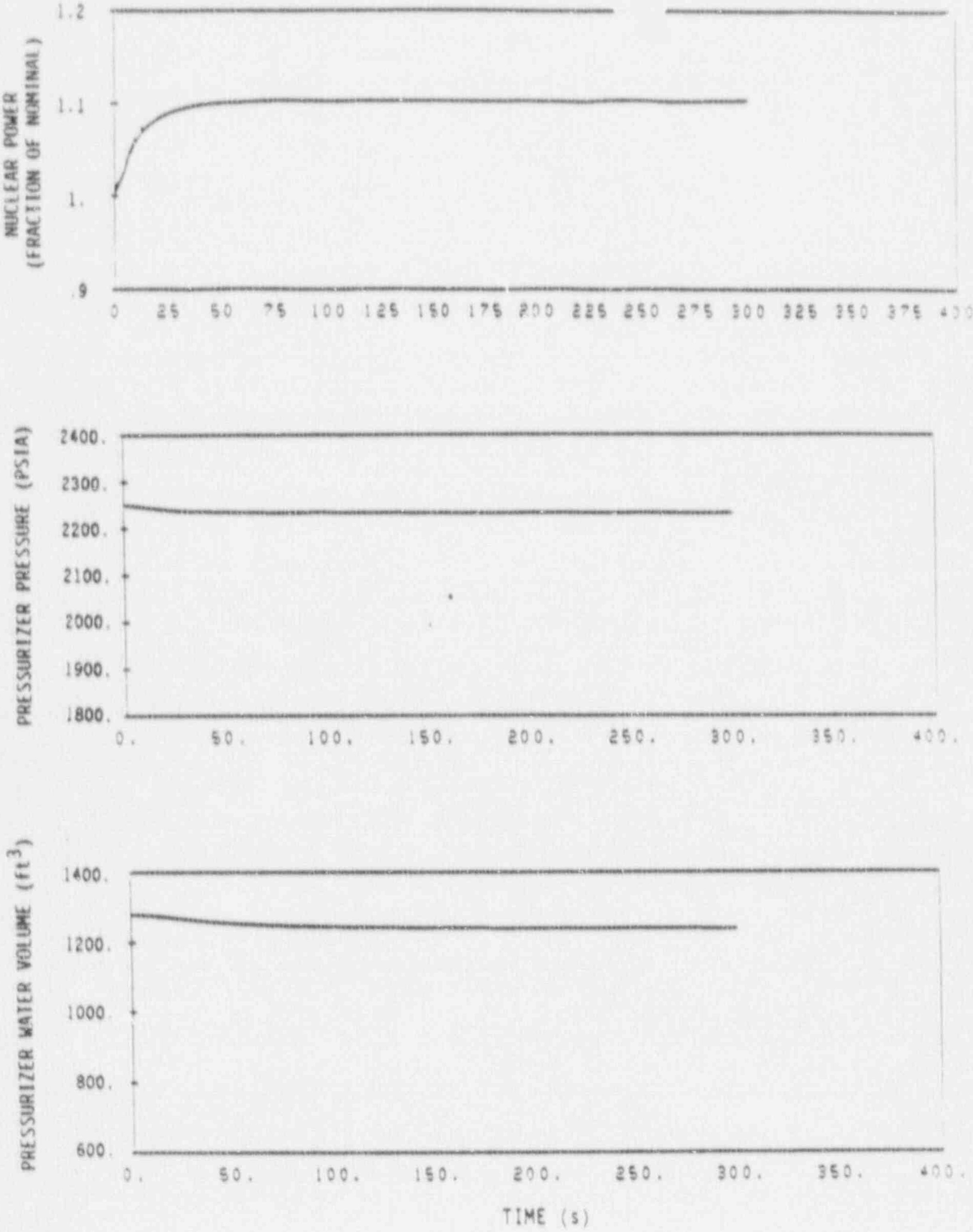
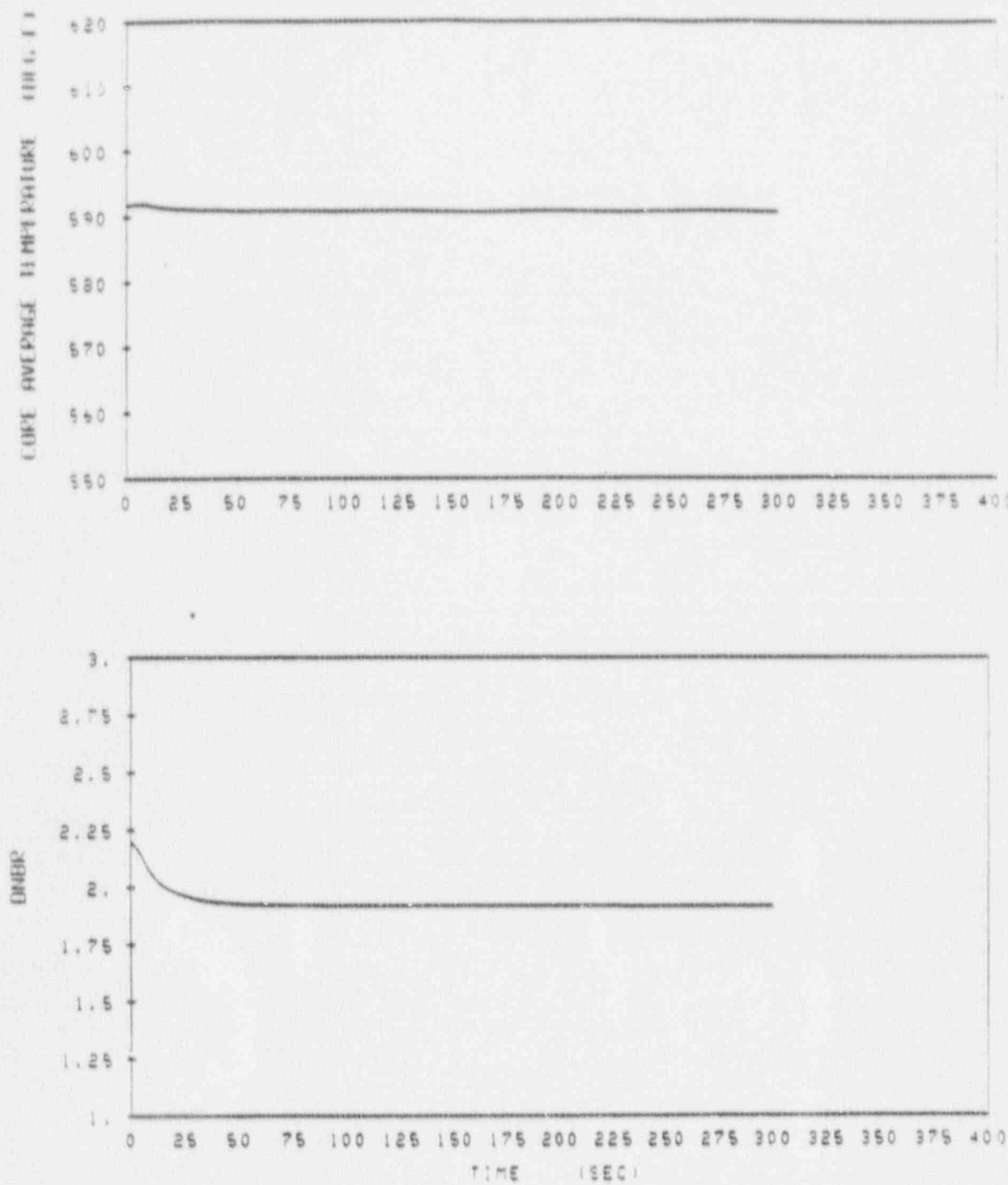
UNIT 1 AND UNIT 2

10% STEP LOAD INCREASE, MIN. MODERATOR FEEDBACK, AUTOMATIC ROD CONTROL

FIGURE 15.1.3-6



Georgia Power	VOGTLE ELECTRIC GENERATING PLANT UNIT 1 AND UNIT 2	10% STEP LOAD INCREASE, MAX. MODERATOR FEEDBACK, AUTOMATIC ROD CONTROL FIGURE 15.1.3-7



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ELECTRIC GENERATING PLANT
UNIT 1 AND UNIT 2

10% STEP LOAD INCREASE, MAX. MODERATOR FEEDBACK, AUTOMATIC ROD CONTROL

FIGURE 15.1.3-8

15.2 Decrease in Heat Removal by the Secondary System

Several postulated transients and accidents which result in a reduction of the capacity of the secondary system to remove heat generated in the RCS are discussed in this section.

15.2.1 Steam Pressure Regulator Malfunction or Failure that Results in Decreasing Steam Flow

As stated in the FSAR, there are no steam pressure regulators in either VEGP units whose failure or malfunction could cause a steam flow transient.

15.2.2 Loss of Electrical Load

As discussed in FSAR Section 15.2.2, the analysis for the turbine trip event bounds this transient. Section 15.2.3 describes the turbine trip event.

15.2.3 Turbine Trip

15.2.3.1 Introduction

For a turbine trip event, the reactor trips directly (unless below approximately 50 percent power) from a signal derived from the turbine stop emergency trip fluid pressure and turbine stop valves. The turbine stop valves close rapidly (typically in 0.1 second) on loss of trip fluid pressure actuated by one of several possible turbine trip signals. Turbine trip initiation signals include:

- o Generator trip
- o Low condenser vacuum
- o Loss of lubricating oil
- o Turbine thrust bearing failure
- o Turbine overspeed
- o Manual trip

Upon initiation of stop valve closure, steam flow to the turbine stops abruptly. Sensors on the stop valve detect the turbine trip and initiate turbine bypass through steam dump valves and, if above 50 percent power, a reactor trip. The loss of steam flow results in an almost immediate rise in secondary system temperature and pressure with a resultant increase in primary system temperature and pressure. A slightly more severe transient than the loss of electrical load event occurs for the turbine trip event due to a more rapid loss of steam flow caused by the more rapid valve closure. FSAR Sections 15.2.2 and 15.2.3 contain more information on the loss of load and turbine trip events.

A turbine trip accident is more limiting than the loss of external electrical load (Section 15.2.2), inadvertent closure of main steam isolation valves (Section 15.2.4), and loss of condenser vacuum and other events resulting in a turbine trip (Section 15.2.5); therefore, this event has been analyzed in detail.

This is an ANS Condition II incident.

15.2.3.2 Method of Analysis

In this analysis, evaluation of the behavior of the units is for a complete loss of steam load from nominal full power, with a turbine trip not causing a direct reactor trip. This demonstrates the adequacy of the pressure-relieving devices and the core protection margins. This assumption delays reactor trip until conditions in the RCS result in a trip due to other signals. Thus, the analysis models a worst-case transient. In addition, no credit is taken for the turbine bypass system. Main feedwater flow terminates at the time of turbine trip with no credit taken for auxiliary feedwater (except for long-term recovery) to mitigate the consequences of the transient.

The analysis of the turbine trip transients employs the detailed digital computer program LOFTRAN (Reference 3). The program simulates the neutron kinetics, RCS, pressurizer, pressurizer relief and safety valves, pressurizer spray, steam generator, and steam generator safety valves. LOFTRAN computes pertinent plant variables including temperatures, pressures, and power level.

The analysis of this accident uses RTDP methodology. Plant characteristics and initial conditions are shown in Tables 15.0.3-2 and 15.0.3-3.

The following summarizes the major assumptions used in the analysis:

A. Initial Operating Conditions

The analysis assumes nominal values of core power, reactor coolant average temperature, and nominal reactor coolant average pressure. The limit DNBR includes uncertainties in initial conditions as described in the Safety Evaluation (Section 4). Previous studies have shown that the peak pressurizer pressure reached for the turbine trip event is insensitive to the initial conditions of temperature and pressure, and the peak pressurizer pressure is only slightly sensitive to the initial power condition. Therefore, the use of these initial conditions is appropriate for this event.

B. Moderator and Doppler Coefficients of Reactivity

The analysis of the turbine trip is with both maximum and minimum reactivity feedback. With maximum feedback, the analysis assumes a large negative moderator temperature coefficient and the most-negative Doppler-only power coefficient. With minimum feedback, the analysis assumes the most positive moderator temperature coefficient and the least-negative Doppler-only power coefficient.

C. Rod Control

It is conservative to assume that the reactor is in manual rod control with respect to the maximum pressures attained. If the reactor were in automatic rod control, the control rod banks would move before the trip and reduce the severity of the transient.

D. 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 the safety valves limits secondary steam pressure at the setpoint value.

E. Pressurizer Spray and Power-Operated Relief Valves

The following analyses are the two cases for both the minimum and maximum reactivity feedback cases examined:

1. 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.

2. 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.

F. Feedwater Flow

The analysis assumes main feedwater flow to the steam generators to be lost at the time of turbine trip. No credit is taken for auxiliary feedwater flow since the plant will reach a stabilized condition before auxiliary feedwater initiation is normally assumed to occur; however, the auxiliary feedwater pumps will start on a trip of the main feedwater pumps. The auxiliary feedwater flow would remove core decay heat following plant stabilization.

G. Reactor Trip

Reactor trip actuates by the first reactor protection system trip setpoint reached, with no credit taken for direct reactor trip on turbine trip. Trip signals are expected due to high pressurizer pressure, OTΔT, high pressurizer water level, and low-low steam generator water level.

No single active failure in any plant systems or equipment will adversely affect the consequences of the accident.

15.2.3.3 <u>Results</u>

The transient responses for a turbine trip from nominal full power operation are shown for the following four cases: two cases with minimum reactivity feedback and two cases with maximum reactivity feedback (Figures 15.2.3-1 through 15.2.3-8).

Figures 15.2.3-1 and 15.2.3-2 show the transient responses for the turbine trip event with minimum reactivity feedback assuming full credit for the pressurizer spray and pressurizer power-operated relief valves. No credit is taken for the turbine bypass through the steam dumps. The reactor trips on the high pressurizer pressure trip channel. The minimum DNBR remains well above the limit value. The pressurizer safety valves and steam generator safety valves prevent overpressurization in the primary and secondary systems, respactively.

Figures 15.2.3-3 and 15.2.3-4 show the responses for the turbine trip event with maximum reactivity feedback. All other plant parameters are the same as the above. The DNBR increases throughout the transient and never drops below its initial value. The pressurizer power-operated relief valves and steam generator safety valves prevent overpressurization in the primary and secondary systems, respectively. The reactor trips on the high pressurizer pressure trip channel. The pressurizer safety valves do not actuate for this case.

The turbine trip accident was also studied assuming the plant to be initially operating at nominal full power with no credit taken for the pressurizer spray, pressurizer power-operated relief valves, or turbine bypass system. The reactor trips on the high pressurizer pressure signal. Figures 15.2.3-5 and 15.2.3-6 show the transient responses with minimum reactivity feedback. The neutron flux remains constant at nominal full power until the reactor trips. The DNBR never goes below its initial value throughout the transient. In this case the pressurizer safety valves and steam generator safety valves actuate to maintain the RCS and main steam system pressure below 110 percent of their respective design values.

Figures 15.2.3-7 and 15.2.3-8 show the transient responses with maximum reactivity feedback with the other assumptions being the same as in the preceding case. The reactor trips on the high pressurizer pressure signal and the DNBR increases throughout the transient. The pressurizer safety valves and steam generator safety valves actuate to limit primary and secondary system pressures, respectively.

The calculated sequence of events for the turbine trip event is shown in Table 15.2.3-1.

15.2.3.4 Conclusions

Results of the analyses show that the plant design is such that a turbine trip without a direct reactor trip does not present any 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 also maintained since the DNBR remains above the limit value. Thus, the conclusions presented in the FSAR remain valid for the turbine trip event.

15.2.4 Inadvertent Closure of Main Steam Isolation Valves

Inadvertent closure of the main steam isolation valves would result in a turbine trip with no credit taken for the turbine bypass system. Section 15.2.3 discusses turbine trips. The analysis performed for the turbine trip event applies to this category of events.

15.2.5 Loss of Condenser Vacuum and Other Events Resulting in a Turbine Trip

Loss of condenser vacuum is one of the events that can cause a turbine trip. Section 15.2.3 describes turbine trip initiating events. FSAR Section 15.2.5 provides additional information for this category. The analysis performed for the turbine trip event applies to this category of events.

15.2.6 Loss of Non-Emergency AC Power to the Plant Auxiliaries

Section 15.2.6 of the FSAR describes the loss of non-emergency AC power to the plant auxiliaries accident. This is a long-term heat removal event analyzed to determine if the auxiliary feedwater (AFW) heat removal capacity is sufficient to ensure that the peak RCS pressure does not exceed allowable limits and the natural circulation is sufficient to remove residual heat from the core. The increases in $F^N_{\Delta H}$ and F_Q do not affect system transients, and thus have no impact on these events. The remaining effects of the VANTAGE 5 fuel transition will also have no discernible impact on these transients; therefore, the results and conclusions presented in the FSAR remain valid for this event.

15.2.7 Loss of Normal Feedwater Flow

Section 15.2.7 of the FSAR describes the loss of normal feedwater flow accident. This is a long-term heat removal event analyzed to determine if the AFW heat removal capacity is sufficient in removing long term decay heat and preventing excessive heatup of the RCS with possible resultant RCS overpressurization or loss of RCS water. The increases in $F^N_{\Delta H}$ and F_Q do not affect system transients, and thus have no impact on these events. The remaining effects of the VANTAGE 5 fuel transition will also have no discernible impact on these transients; therefore, the results and conclusions presented in the FSAR remain valid for this event.

15.2.8 Feedwater System Pipe Break

Section 15.2.8 of the FSAR describes the feedwater system pipe break accident. This is a long-term heat removal event analyzed to determine if the AFW heat removal capacity is sufficient to ensure that the peak RCS pressure does not exceed allowable limits, and the core remains covered and in a coolable geometry. The increases in $F^N_{\Delta H}$ and F_Q do not affect system transients, and thus have no impact on these events. The remaining effects of the VANTAGE 5 fuel transition will also have no discernible impact on these transients; therefore, the results and conclusions presented in the FSAR remain valid for this event.

Table 15.2.3-1 (Sheet 1 of 2)

Time Sequence of Events for Incidents Which Result in a Decrease in Heat Removal by the Secondary System

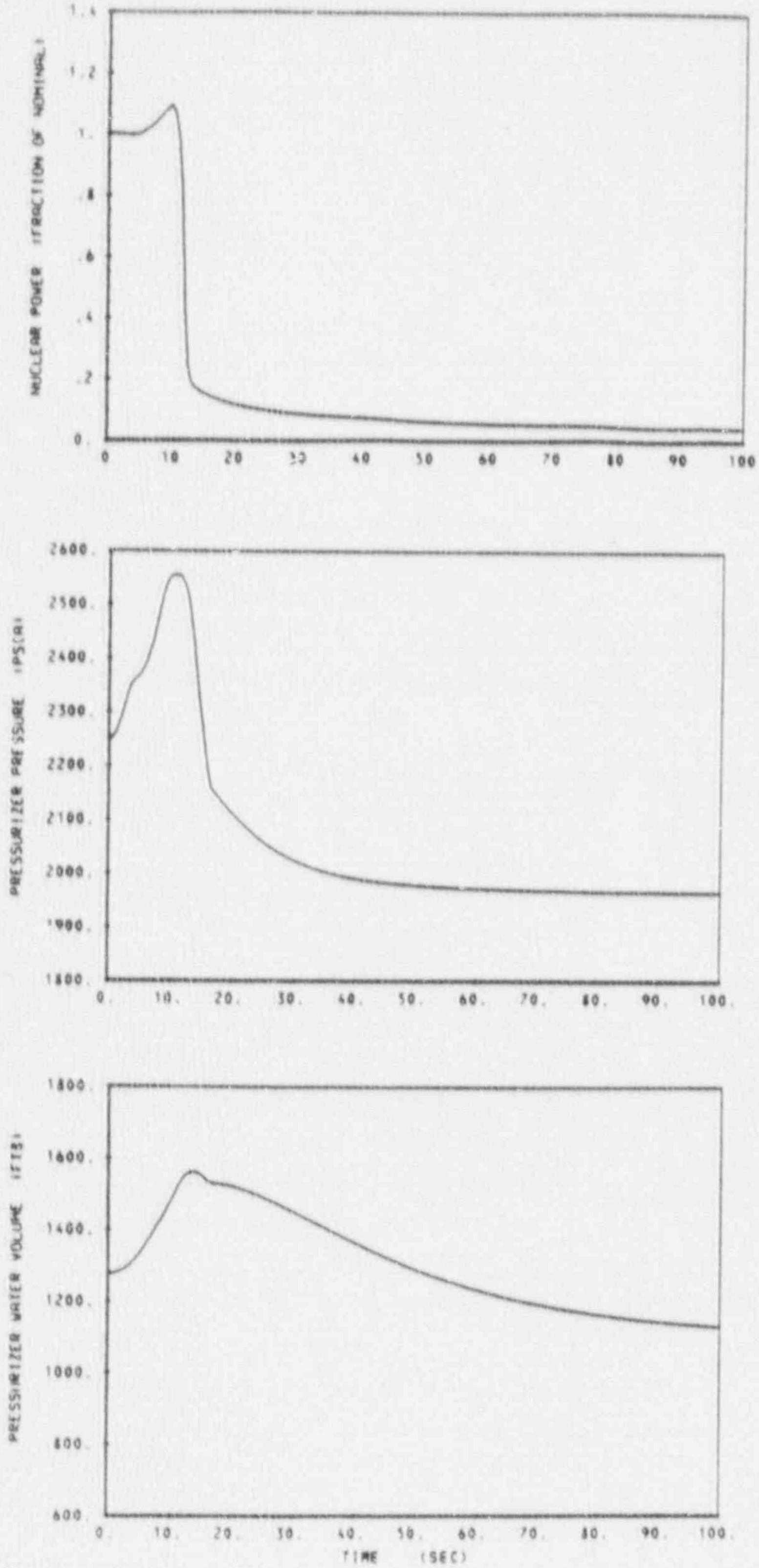
Accident	Event	Time Delays (s)
Turbine Trip		
1. With pressurizer control (minimum reactivity feedback)	Turbine trip; loss of main feed-water flow	0.0
	High pressurizer pressure reactor trip point reached	7.1
	Initiation of steam release from steam generator safety valves	7.5
	Rods begin to drop	9.1
	Peak pressurizer pressure occurs	10.5
	Minimum DNBR occurs	11.0
2. With pressurizer control (maximum reactivity feedback)	Turbine trip; loss of main feed-water flow	0.0
	Initiation of steam release from steam generator safety valves	7.5
	High pressurizer pressure reactor trip setpoint reached	8.4
	Peak pressurizer pressure occurs	9.0
	Rods begin to drop	10.4
	Minimum DNBR occurs	(a)
3. Without pressurizer control (minimum reactivity feedback)	Turbine trip; loss of main feed-water flow	0.0
	High pressurizer pressure reactor trip point reached	4.6
	Rods begin to drop	6.6
	Initiation of steam release from steam generator safety valves	7.5
	Peak pressurizer pressure occurs	8.0
	Minimum DNBR occurs	(a)

Table 15.2.3-1 (Sheet 2 of 2)

Time Sequence of Events for Incidents Which Result in a Decrease in Heat Removal by the Secondary System

Accident	Event	Time Delays (s)
4. Without pressurizer control (maximum reactivity feedback)	Turbine trip; loss of main feedwater flow	0.0
	High pressurizer pressure reactor trip point reached	4.6
	Rods begin to drop	6.6
	Peak pressurizer pressure occurs	7.5
	Initiation of steam release from steam generator safety valves	7.5
	Minimum DNBR occurs	(a)

a. DNBR does not decrease below its initial value.



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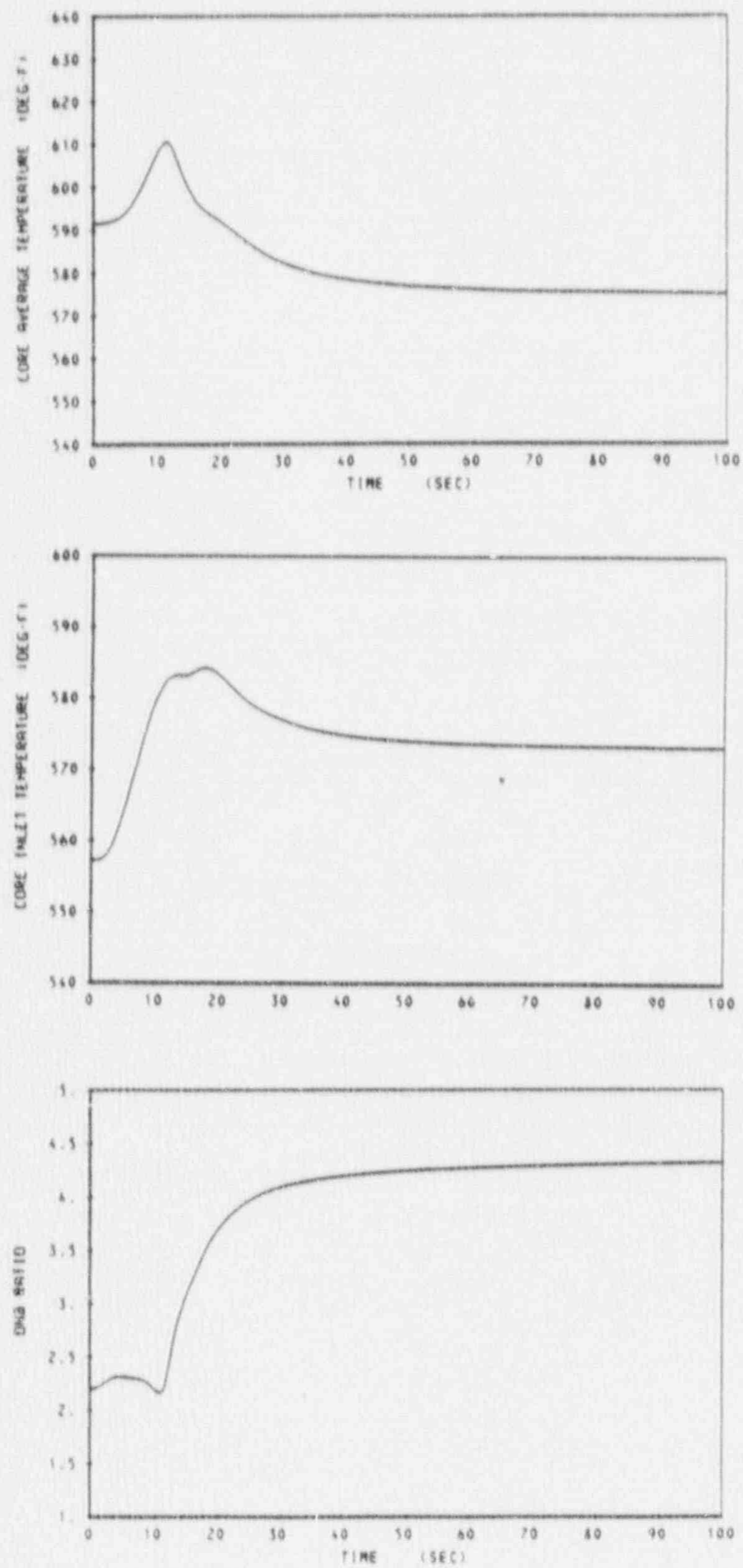
TURBINE TRIP ACCIDENT WITH PRESSURIZER SPRAY AND POWER-OPERATED RELIEF VALVES, MIN. MODERATOR FEEDBACK

FIGURE 15.2.3-1



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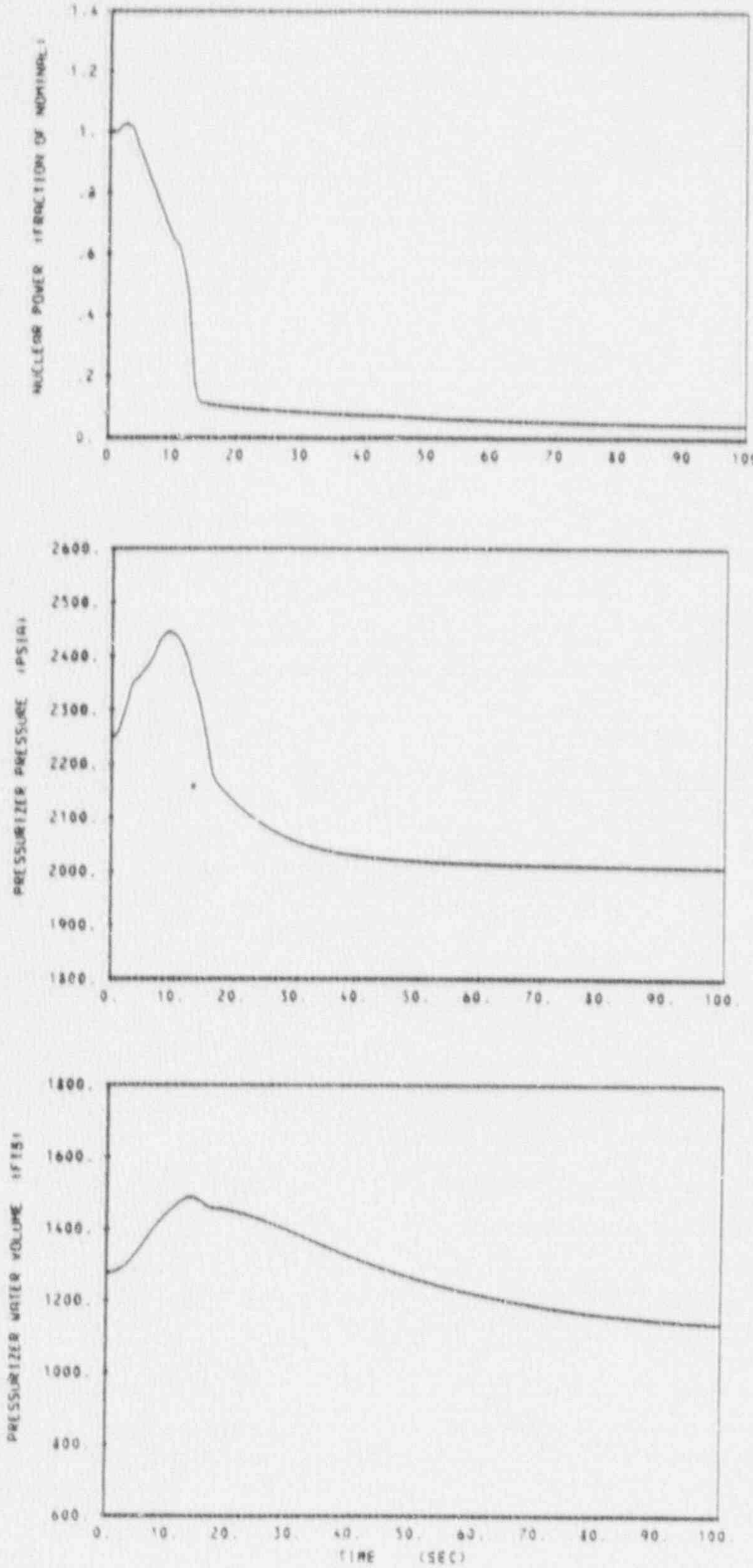
TURBINE TRIP ACCIDENT WITH PRESSURIZER SPRAY AND POWER-OPERATED RELIEF VALVES, MIN. MODERATOR FEEDBACK

FIGURE 15.2.3-2



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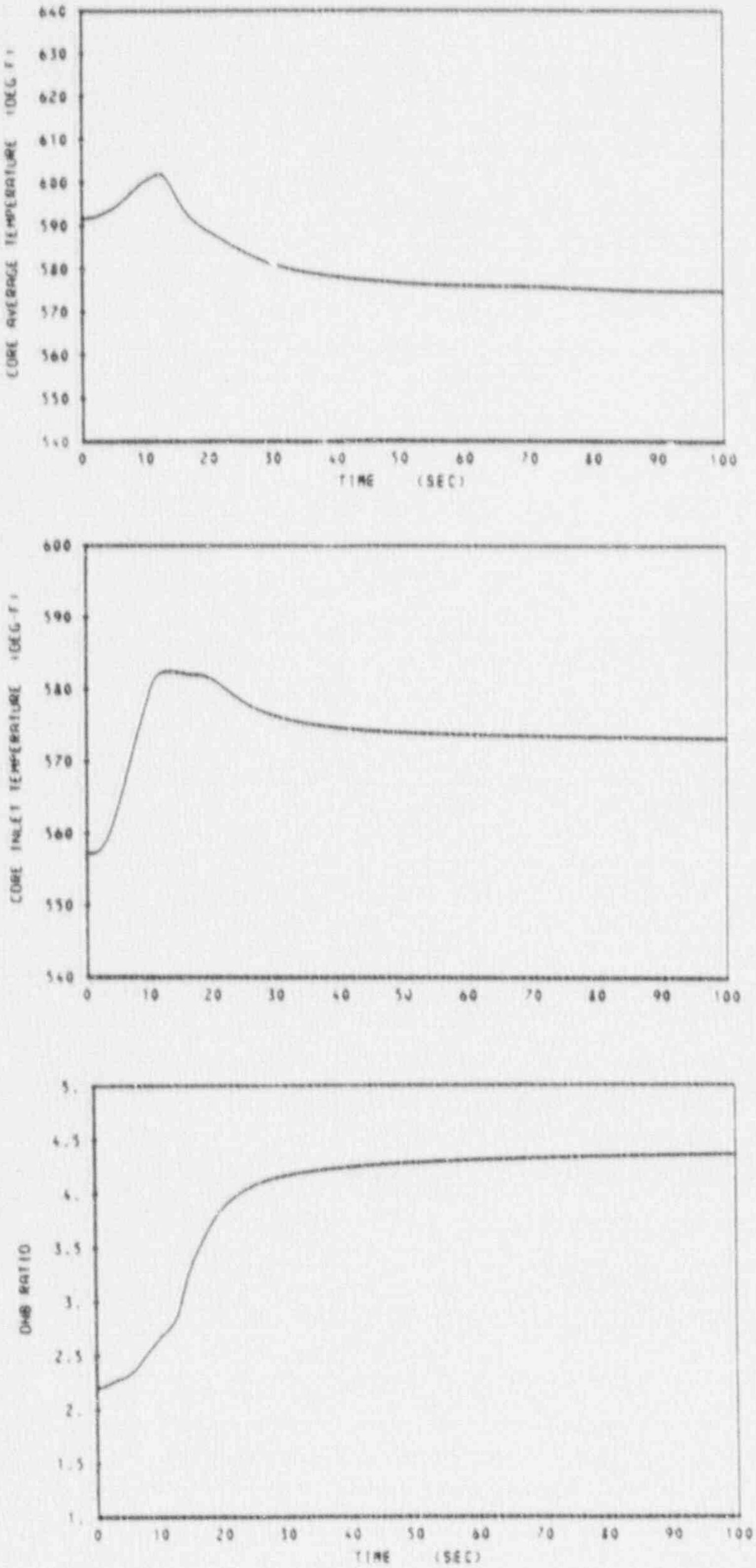
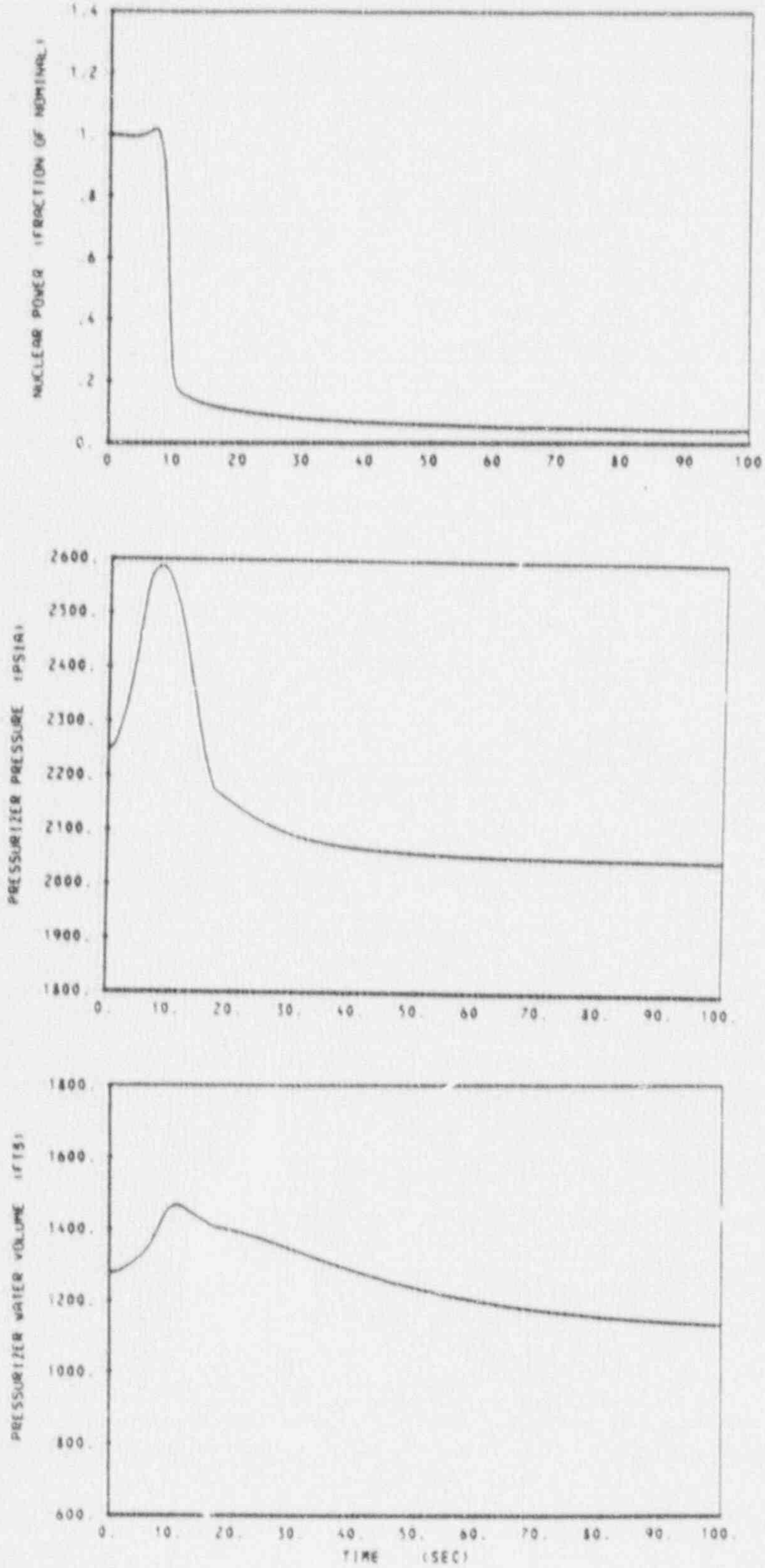
VOGTLE
ELECTRIC GENERATING PLANT
UNIT 1 AND UNIT 2

TURBINE TRIP ACCIDENT WITH PRESSURIZER SPRAY AND POWER-OPERATED RELIEF VALVES, MAX. MODERATOR FEEDBACK

FIGURE 15.2.3-3



Georgia Power	VOGTLE ELECTRIC GENERATING PLANT UNIT 1 AND UNIT 2	TURBINE TRIP ACCIDENT WITH PRESSURIZER SPRAY AND POWER-OPERATED RELIEF VALVES, MAX. MODERATOR FEEDBACK
		FIGURE 15.2.3-4



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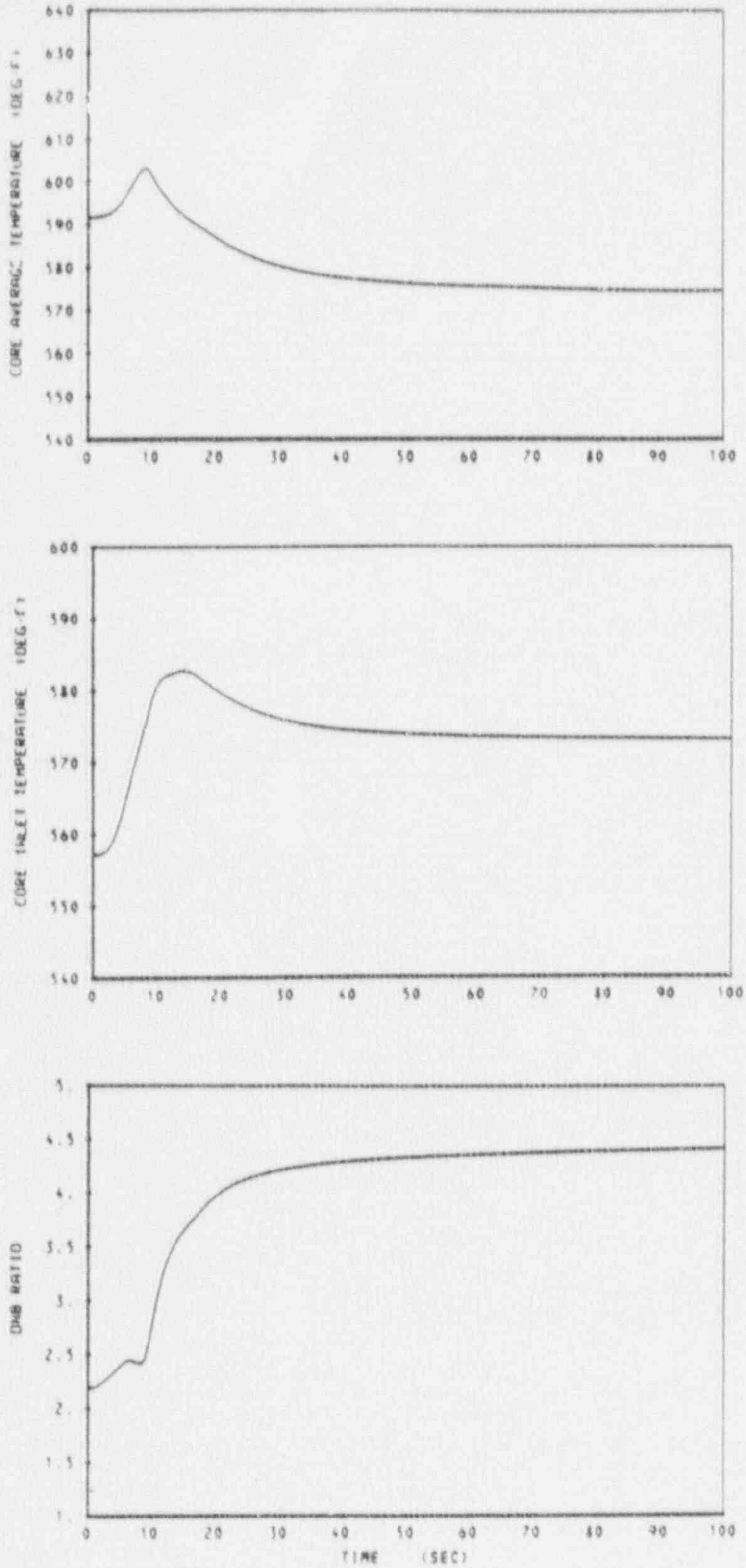
ELECTRIC GENERATING PLANT
UNIT 1 AND UNIT 2

TURBINE TRIP ACCIDENT WITHOUT PRESSURIZER SPRAY AND POWER-OPERATED RELIEF VALVES, MIN. MODERATOR FEEDBACK

FIGURE 15.2.3-5



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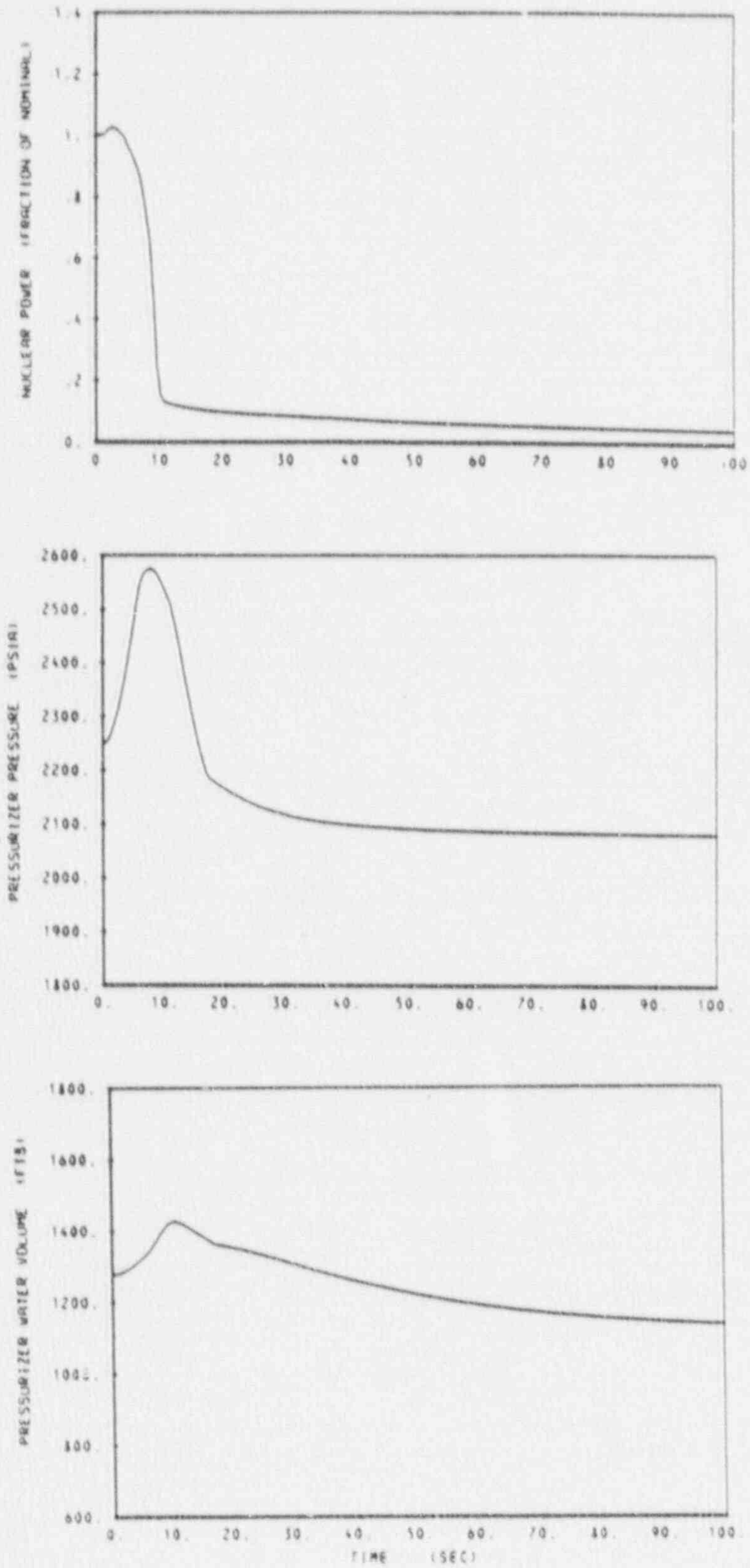
ELECTRIC GENERATING PLANT
UNIT 1 AND UNIT 2

TURBINE TRIP ACCIDENT WITHOUT PRESSURIZER SPRAY AND POWER-OPERATED RELIEF VALVES, MIN. MODERATOR FEEDBACK

FIGURE 15.2.3-6



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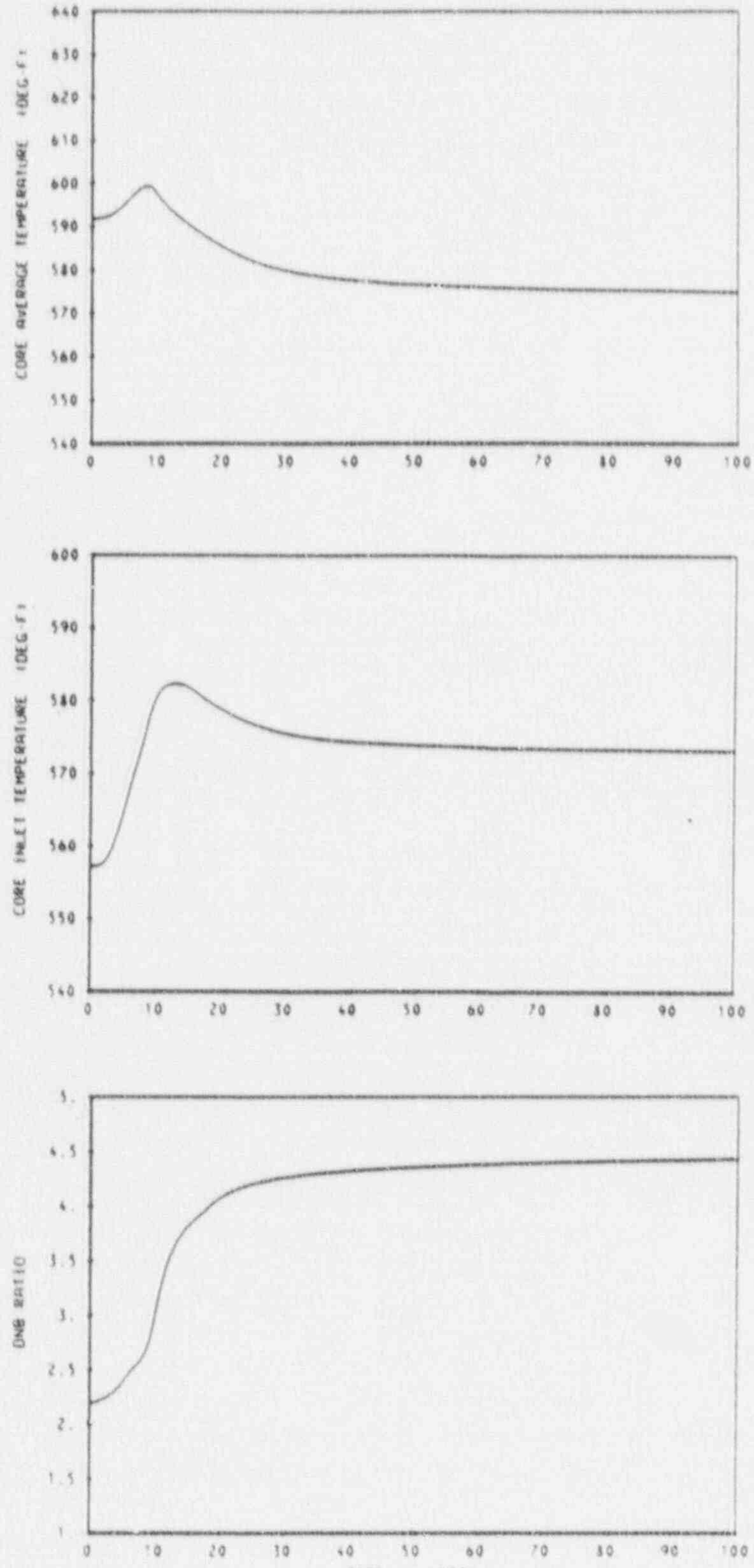
UNIT 1 AND UNIT 2

TURBINE TRIP ACCIDENT WITHOUT PRESSURIZER SPRAY AND POWER-OPERATED RELIEF VALVES, MAX. MODERATOR FEEDBACK

FIGURE 15.2.3-7



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UNIT 1 AND UNIT 2

TURBINE TRIP ACCIDENT WITHOUT PRESSURIZER SPRAY AND POWER-OPERATED RELIEF VALVES, MAX. MODERATOR FEEDBACK

FIGURE 15.2.3-8

15.3 Decrease in Reactor Coolant System Flowrate

Several faults can result in a decrease in the RCS flowrate. This section discusses these events.

15.3.1 Partial Loss of Forced Reactor Coolant Flow

15.3.1.1 Introduction

A partial loss-of-forced-reactor-coolant flow accident can result from a mechanical or electrical failure in an RCP or from a fault in the power supply to the pump or pumps supplied by an RCP bus. If the reactor is at power at the time of the accident, the immediate effect of the 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 does not trip promptly.

Two buses connected to the generators supply power to the pumps. When a generator trip occurs, the buses are automatically transferred to a transformer supplied from external power lines, and the pumps continue to operate. Following any turbine trip where there are no electrical faults which require tripping the generator from the network, the generator remains connected to the network for approximately 30 seconds. The RCPs remain connected to the generator, thus ensuring full flow for approximately 30 seconds after the reactor trip before any transfer is made.

The low primary coolant flow reactor trip signal, which actuates in any reactor coolant loop by two out of three low-flow signals, provides the necessary protection against this event. Above permissive P-8, low flow in any loop will actuate a reactor trip. Between approximately 10 percent power (permissive P-7) and the power level corresponding to permissive P-8, low flow in any two loops will actuate a reactor trip. Above permissive P-7, two or more RCP circuit breakers from the same bus will open which will actuate the corresponding undervoltage relays. This results in a reactor trip which serves as backup to the flow trip.

This is a ANS Condition II incident.

15.3.1.2 Method of Analysis

This analysis examines partial loss-of-forced-reactor-coolant flow involving loss of two pumps with four loops in operation.

This analysis uses three digital computer codes. First the LOFTRAN code (Reference 3) calculates the loop and core flow during the transient, the time of reactor trip based on the calculated flows, the nuclear power transient, and the primary system pressure and temperature transients. The FACTRAN code (Reference 2) then calculates the heat flux transient based on the nuclear power and flow from LOFTRAN. Finally, the THINC code calculates the DNBR during the transient based on the heat flux from FACTRAN and flow from LOFTRAN. The DNBR transients presented represent the minimum of the typical or thimble fuel assembly cell.

This analysis employs RTDP methodology; therefore, the initial conditions assume nominal values of power, reactor coolant average temperature, and RCS average pressure (see Tables 15.0.3-2 and 15.0.3-3). The limit DNBR includes uncertainties in the initial conditions.

The analysis assumes a conservatively large absolute value of the Doppler-only power coefficient (see Figure 15.0.4-1). This is equivalent to a total integrated Doppler reactivity from 0 to 100 percent power of 0.016 Δk.

The analysis assumes the most positive moderator temperature coefficient (minimum moderator density coefficient) since this results in the maximum core power during the initial part of the transient when the transient reaches minimum DNBR (see Figure 15.0.4-2).

These analyses use the curve of trip reactivity insertion versus time (Figure 15.0.5-3).

The basis for the flow coastdown analysis is 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.

No single active failure in any plant systems or equipment will adversely affect the consequences of the accident.

15.3.1.3 Results

Figures 15.3.1-1 through 15.3.1-4 show the transient response for the loss of power to two RCPs with four loops in operation. The reactor trips on the low-flow signal. Figure 15.3.1-4 shows the DNBR to be always greater than the safety analysis limit value for the most limiting fuel assembly cell.

Since DNB does not occur, the ability of the primary coolant to remove heat from the fuel rod is not significantly reduced. Thus, the average fuel and clad temperature do not increase significantly above their respective initial values.

The time sequence of events is shown in Table 15.3.1-1 for the partial loss of flow event.

15.3.1.4 Conclusions

The analysis shows that the minimum DNBR always remains above the limit value during the transient. Thus, all applicable acceptance criteria are met. The conclusions presented in the FSAR remain valid.

15.3.2 Complete Loss of Forced Reactor Coolant Flow

15.3.2.1 Introduction

A loss-of-forced-reactor-coolant flow may result from a simultaneous loss of electrical power to all RCPs. 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 DNB with subsequent adverse effects to the fuel if the reactor does not trip promptly. The reactor trip together with flow sustained by the inertia of the pump impeller will be sufficient to prevent RCS overpressurization and the DNBR from exceeding the limit values. The trip systems available to mitigate the consequences of this accident are the following:

- o RCP power supply bus undervoltage or underfrequency

- o Low reactor coolant loop flow

These trip functions are fully described in the FSAR.

This is an ANS Condition III incident.

15.3.2.2 Method of Analysis

The method of analysis and the assumptions made regarding initial operating conditions and reactivity coefficients are identical to those discussed in Section 15.3.1, except that following the loss of power supply to all pumps at power, a reactor trip actuates by either RCP power supply undervoltage or underfrequency.

15.3.2.3 Results

Figures 15.3.2-1 through 15.3.2-4 show the transient response for the loss of power to all RCPs with four loops in operation. The reactor trips on the undervoltage signal. Figure 15.3.2-4 shows the DNBR to be always greater than the safety analysis limit value for the most limiting fuel assembly cell.

Since DNB does not occur, the ability of the primary coolant to remove heat from the fuel rod is not significantly reduced. Thus, the average fuel and clad temperature do not increase significantly above their respective initial values.

Besides the complete loss-of-forced-reactor-coolant flow (loss of power to four pumps), an underfrequency event with a frequency decay rate of 5 Hz/sec was also analyzed. For this event, the reactor trip occurs on an underfrequency signal. The DNBR analysis of the underfrequency event verified that the DNBR remains above the safety analysis limit value. If the maximum grid frequency decay rate is less than approximately 2.5 Hz/sec, the low flow signal will actuate a reactor trip which will protect the core from underfrequency events.

The time sequence of events is shown in Table 15.3.1-1 for the complete loss of flow event.

15.3.2.4 Conclusions

The analysis shows that the minimum DNBR always remains above the limit value during the transient. Thus, the analysis does not predict any adverse fuel effects or clad rupture and all applicable acceptance criteria are met. The conclusions presented in the FSAR remain valid.

15.3.3 Reactor Coolant Pump Shaft Seizure (Locked Rotor)

15.3.3.1 Introduction

For the instantaneous seizure of an RCP rotor, flow through the affected reactor coolant loop is rapidly reduced, leading to a reactor trip on a low flow signal. Following the trip, heat stored in the fuel rods continues to be transferred into the core coolant, causing the coolant to expand. At the same time, heat transfer to the shell side of the steam generator reduces, 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 reduces to zero upon plant trip). The rapid expansion of the coolant in the reactor core, combined with the reduced heat transfer in the steam generator, causes an insurge into the pressurizer and a pressure increase throughout the RCS. The insurge into the pressurizer causes a pressure increase, which in turn actuates the automatic spray system, opens the power-operated relief valves, and opens the pressurizer safety valves in that sequence. The power-operated relief valves are safety grade and would be expected to function properly during an accident; however, for conservatism, the analysis does not use the pressure-reducing effect of the power-operated relief valves and the pressure-reducing effect of the spray.

The analysis of the locked rotor event demonstrates that overpressurization of the RCS does not occur and that the core remains in a coolable geometry. Included in the analysis are the design changes associated with the transition to VANTAGE 5 fuel and other modified safety analysis assumptions.

This is an ANS Condition IV incident.

15.3.3.2 Method of Analysis

The analysis of this transient uses two digital computer codes. The LOFTRAN code (Reference 3) calculates: 1) the resulting loop and core flow transients following the pump seizure; 2) the time of reactor trip based on the loop flow transients; 3) the nuclear power following reactor trip; and 4) the peak RCS pressure. The thermal behavior of the fuel located at the core hot spot is investigated using the FACTRAN code (Reference 2) based on the core flow and the nuclear power calculated by LOFTRAN. The FACTRAN code uses a film boiling heat transfer coefficient.

The plant characteristics and the initial conditions are shown in Table 15.0.3-2 and Table 15.0.3-3. The analysis evaluates the transient with and without offsite power available.

No single active failure in any plant systems or equipment will adversely affect the consequences of the accident.

15.3.3.2.2 Evaluation of the Pressure Transient

After pump seizure, the neutron flux is rapidly reduced by control rod insertion due to reactor trip on low coolant flow in the affected loop. Rod motion begins 1 second after the flow in the affected loop reaches 87 percent of nominal flow. No credit is taken for the pressure reducing effect of the pressurizer relief valves, pressurizer spray, steam dump or controlled feedwater flow after plant trip. Although these components will operate and will result in a lower peak RCS pressure, ignoring their effect provides an additional degree of conservatism.

The analysis conservatively bounds the pressurizer safety valves opening at 2500 psia and achieving rated flow at 2575 psia.

15.3.3.2.3 Evaluation of DNB in the Core During the Accident

Because DNB occurs in the core for this accident, there is an evaluation of the consequences with respect to fuel rod thermal transients. Results obtained from analyses of this "hot spot" condition represent the upper limit with respect to clad temperature and zirconium-water reaction.

In the evaluation, the rod power at the hot spot is conservatively assumed to be 2.55 times the average rod power (i.e., F_Q = 2.55) at the initial core power level.

15.3.3.2.4 Film Boiling Coefficient

To model the effect of DNB occurring, the FACTRAN code calculates the film boiling coefficient using the Bishop-Sandberg-Tong film boiling correlation. Fluid properties are evaluated at film temperatures (average between wall and bulk temperatures). The program calculates the film coefficient at every time step based upon the actual heat transfer conditions at the time. The neutron flux, system pressure, bulk density, and mass flowrate as a function of time are program inputs.

This analysis uses the initial values of the pressure and the bulk density throughout the transient since they are the most conservative with respect to clad temperature response. For conservatism, the analysis assumes DNB to start at the beginning of the accident to maximize the fuel rod thermal transient.

15.3.3.2.5 Fuel Clad Gap Coefficient

The magnitude and time dependence of the heat transfer coefficient between fuel and clad (gap coefficient) have a pronounced influence on the thermal results. The larger the value of the gap coefficient, the more heat transferred between pellet and clad. Based on investigations of the effect of the gap coefficient upon the maximum clad temperature during the transient, the analysis assumes the gap coefficient to increase from a steady-state value consistent with initial fuel temperature to 10,000 Btu/hr-ft^2-°F at the initiation of the transient. Thus, the large amount of energy stored in the fuel because of the small initial value releases to the clad at the initiation of the transient.

15.3.3.2.6 Zirconium-Steam Reaction

The zirconium-steam reaction can become significant above 1800°F (clad temperature). In order to take this phenomenon into account, the models (Reference 4) introduced the following correlation which defines the rate of the zirconium-steam reaction.

$$\frac{d(w^2)}{dt} = 33.3 \times 10^6 \times e^{-[(45000.)/(1.986\ T)]}$$

where: w = amount reacted, mg/cm^2

t = time, sec

T = temperature, °F

The reaction heat is 1510 cal/g

15.3.3.3 Results

The transient results for the locked rotor accident are shown in Figures 15.3.3-1 through 15.3.3-4. Table 15.3.3-1 also summarizes the results of the locked rotor calculations. The peak RCS pressure reached during the transient is less than that which would cause stresses to exceed the faulted condition stress limits of the American Society of Mechanical Engineers Code, Section III. Also, the peak clad temperature is considerably less than 2700°F. Note that the clad temperature was conservatively calculated assuming DNB occurs at the initiation of the transient. These results represent the most limiting conditions of the locked rotor event or RCP shaft break.

As a result of this accident, a fraction of the fuel rods will undergo DNB and will release gap inventory to the reactor coolant. Fewer than 5 percent of the fuel rods in the core will have clad damage. Appendix C discusses the evaluation of the radiological consequences of a postulated locked rotor accident.

The calculated sequence of events for the locked rotor event is shown in Table 15.3.1-1.

15.3.3.4 Conclusions

- o The maintenance of the integrity of the primary coolant system occurs because the peak RCS pressure reached during any of the transients is less than that which would cause stresses to exceed the faulted conditions stress limits.

- o Since the peak clad surface temperature calculated for the hot spot during the worst transient remains considerably less than 2700°F (the temperature at which clad embrittlement may occur), the core will remain in a coolable geometry.

The conclusions presented in the FSAR remain valid.

15.3.4 Reactor Coolant Pump Shaft Break

The analysis for the locked rotor event described in Section 15.3.3 represents the most limiting event with respect to a locked rotor or pump shaft break. With a failed shaft, the impeller could conceivably be free to spin in the reverse direction as opposed to the assumption of being in a fixed position as for a locked rotor. The effect of such reverse spinning is a slight decrease in the end point (steady-state) core flow when compared to the locked rotor. The analysis described in Section 15.3.3 modeled the most limiting conditions for the locked rotor and shaft break; therefore, this section does not present a separate analysis.

Table 15.3.1-1

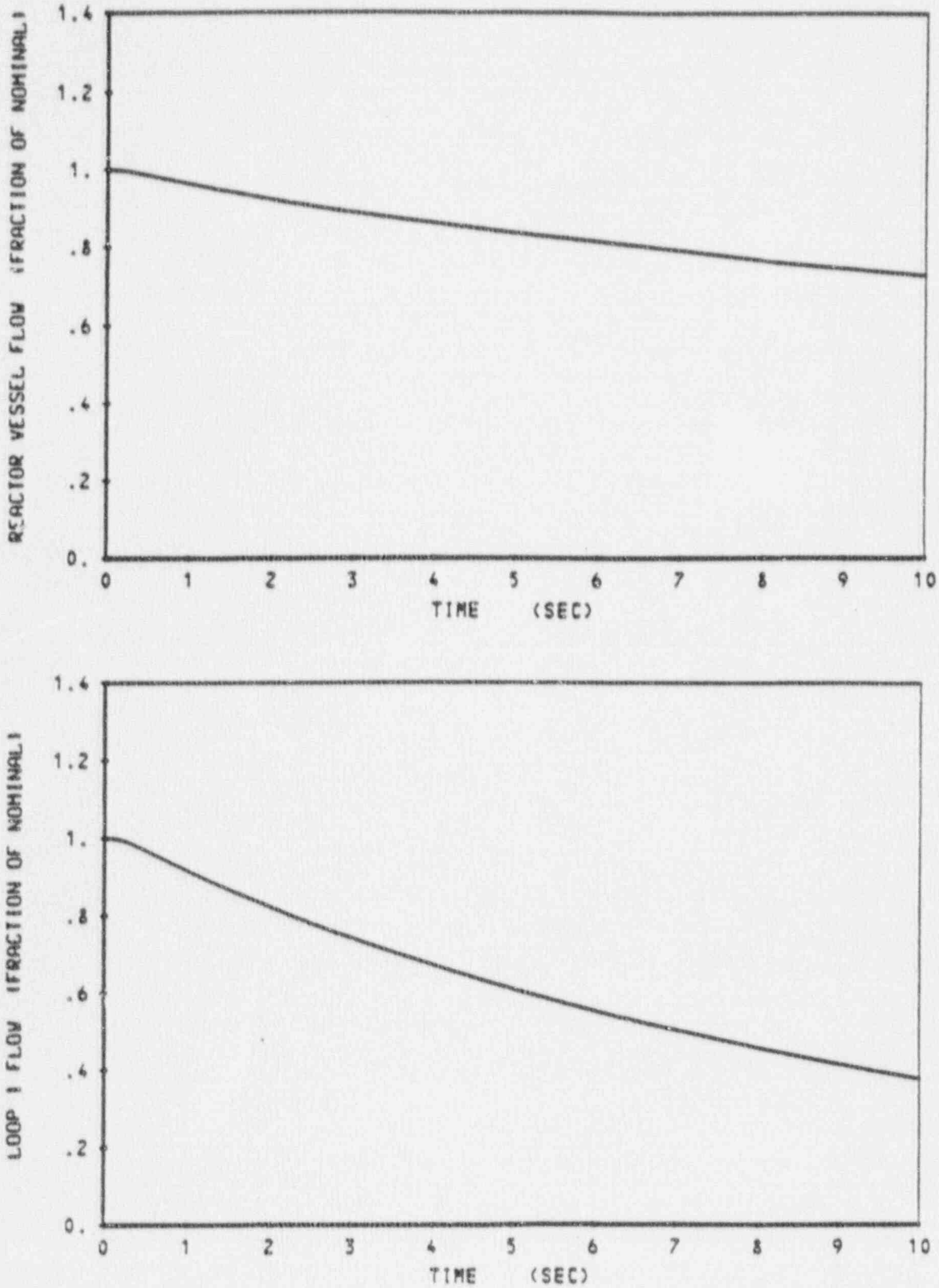
<u>Time Sequence of Events for Incidents Which Result in a Decrease in Reactor Coolant System Flowrate</u>

Accident	Event	Time Delays (s)
Partial loss of forced reactor coolant flow		
Loss of two pumps with four loops in operation	Coastdown begins	0.0
	Low-flow reactor trip	1.4
	Rods begin to drop	2.4
	Minimum DNBR occurs	3.6
Complete loss of forced reactor coolant flow		
Loss of four pumps with with four loops in operation	All operating pumps lose power and begin coasting down	0.0
	RCP undervoltage trip point reached	0.0
	Rods begin to drop	1.5
	Minimum DNBR occurs	3.2
RCP shaft seizure (locked rotor)		
One locked rotor with four loops in operation with offsite power available	Rotor on one pump locks	0.0
	Low-flow trip point reached	0.0
	Rods begin to drop	1.0
	Maximum RCS pressure occurs	3.3
	Maximum clad average temperature occurs	3.5
One locked rotor with four loops in operation without offsite power available	Rotor on one pump locks	0.0
	Low-flow trip point reached	0.0
	Rods begin to drop	1.0
	Maximum RCS pressure occurs	3.4
	Maximum clad average temperature occurs	3.6

Table 15.3 3-1

Summary of Results for the Locked Rotor Transient
(Four Loops Operating Initially)

Parameter	Value for With Offsite Power Available	Value for Without Offsite Power Available
Maximum RCS pressure (psia)	2669.0	2669.0
Maximum clad average temperature, core hot spot (°F)	2048.0	2054.0
$Zr-H_2O$ reaction, core hot spot (percent by weight)	0.6	0.7

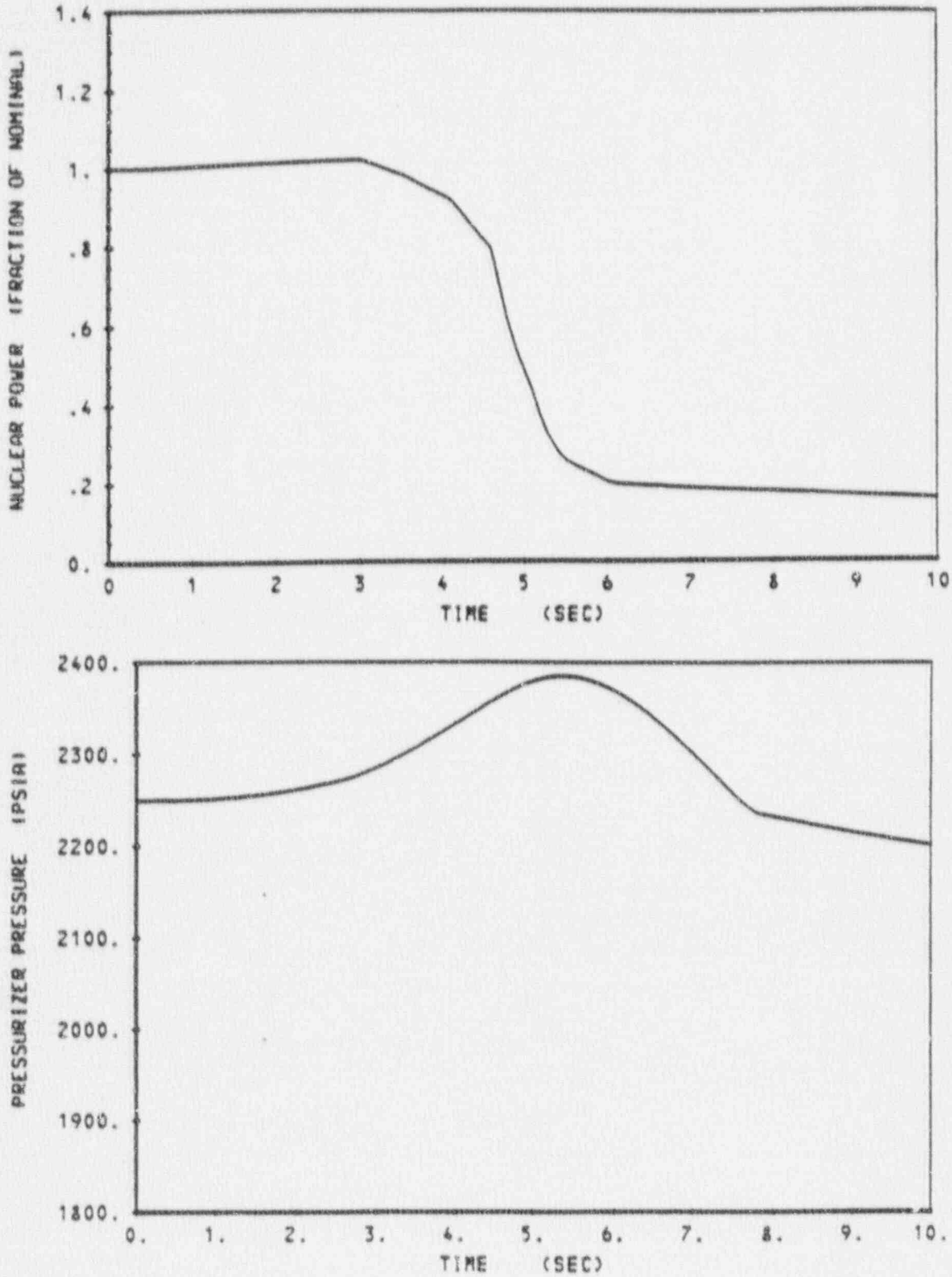


Georgia Power

VOGTLE
ELECTRIC GENERATING PLANT
UNIT 1 AND UNIT 2

FLOW TRANSIENTS FOR 4 LOOPS IN OPERATION, 2 PUMPS COASTING DOWN

FIGURE 15.3.1-1



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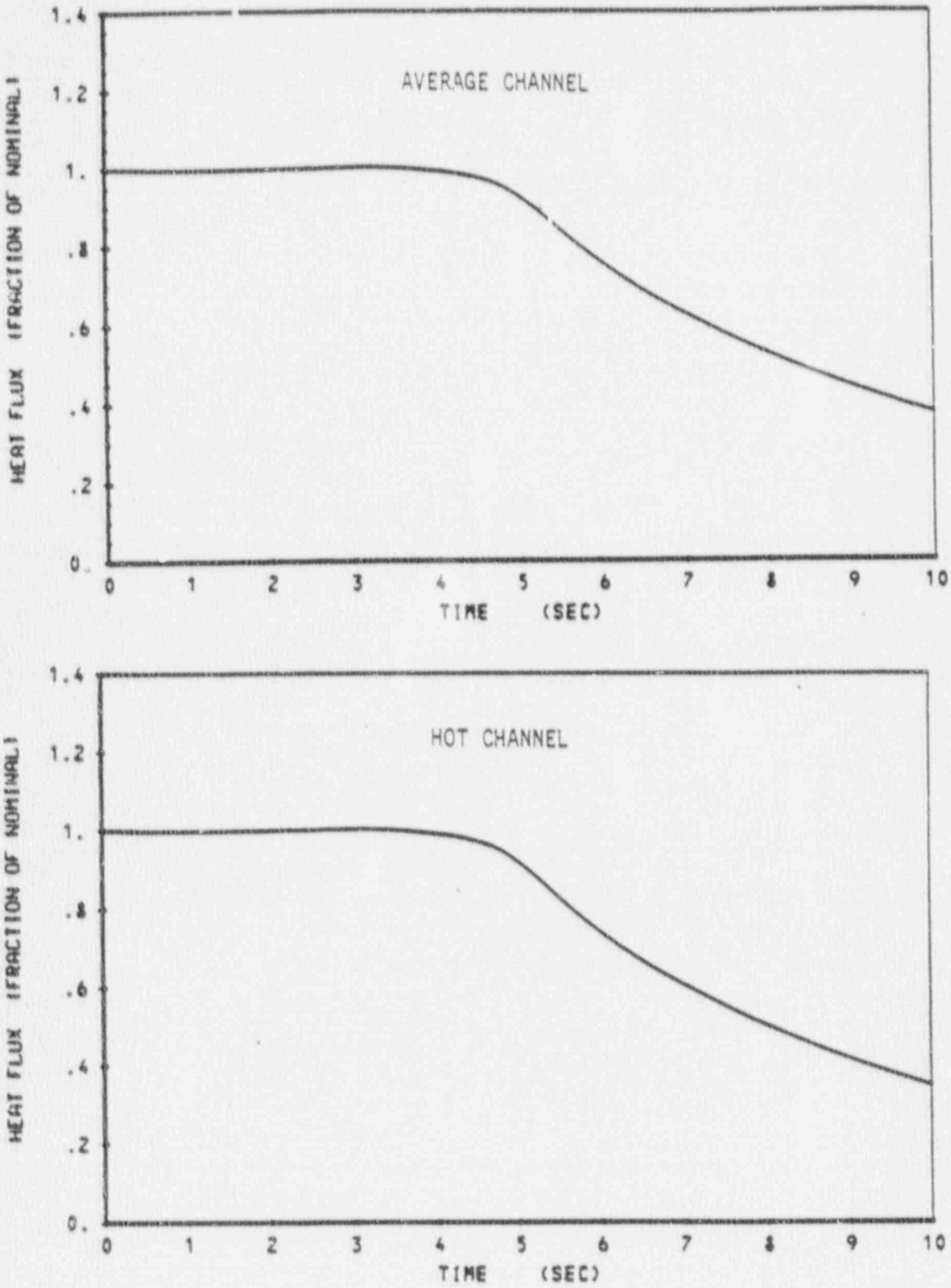
VOGTLE
ELECTRIC GENERATING PLANT
UNIT 1 AND UNIT 2

NUCLEAR POWER AND PRESSURIZER PRESSURE TRANSIENTS FOR 4 LOOPS IN OPERATION, 2 PUMPS COASTING DOWN

FIGURE 15.3.1-2



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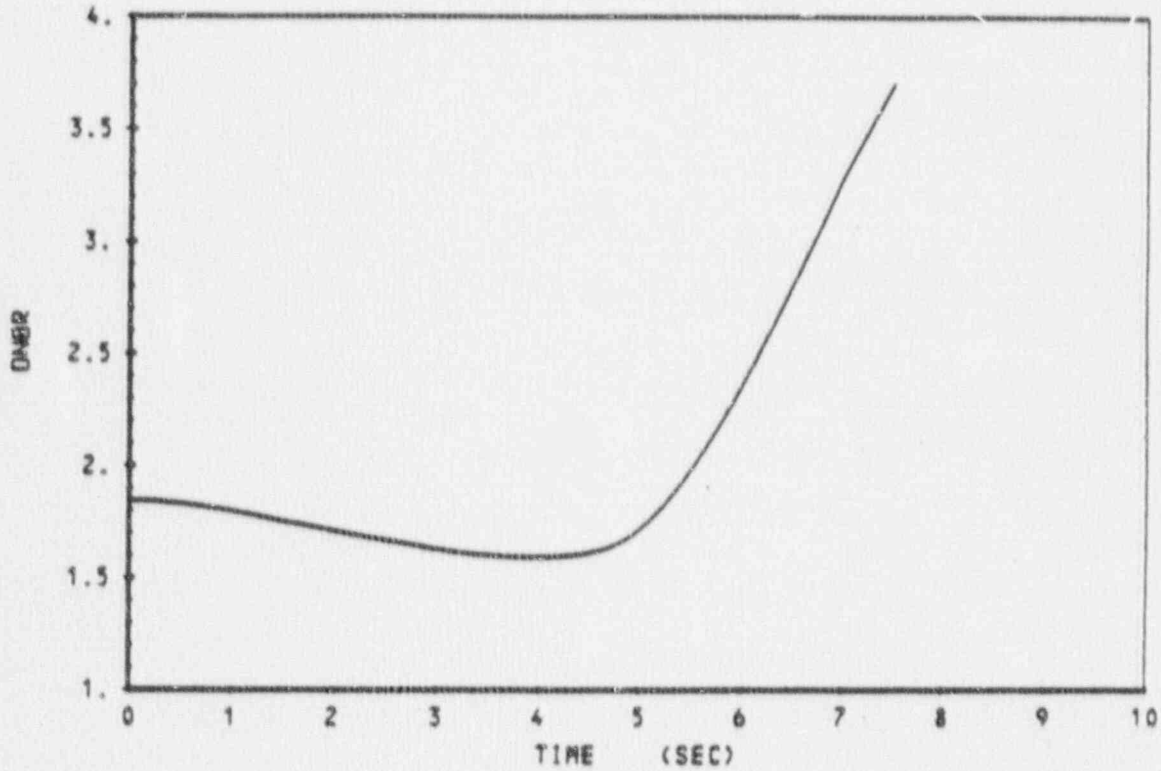
ELECTRIC GENERATING PLANT
UNIT 1 AND UNIT 2

AVERAGE AND HOT CHANNEL HEAT FLUX TRANSIENTS FOR 4 LOOPS IN OPERATION, 2 PUMPS COASTING DOWN

FIGURE 15.3.1-3



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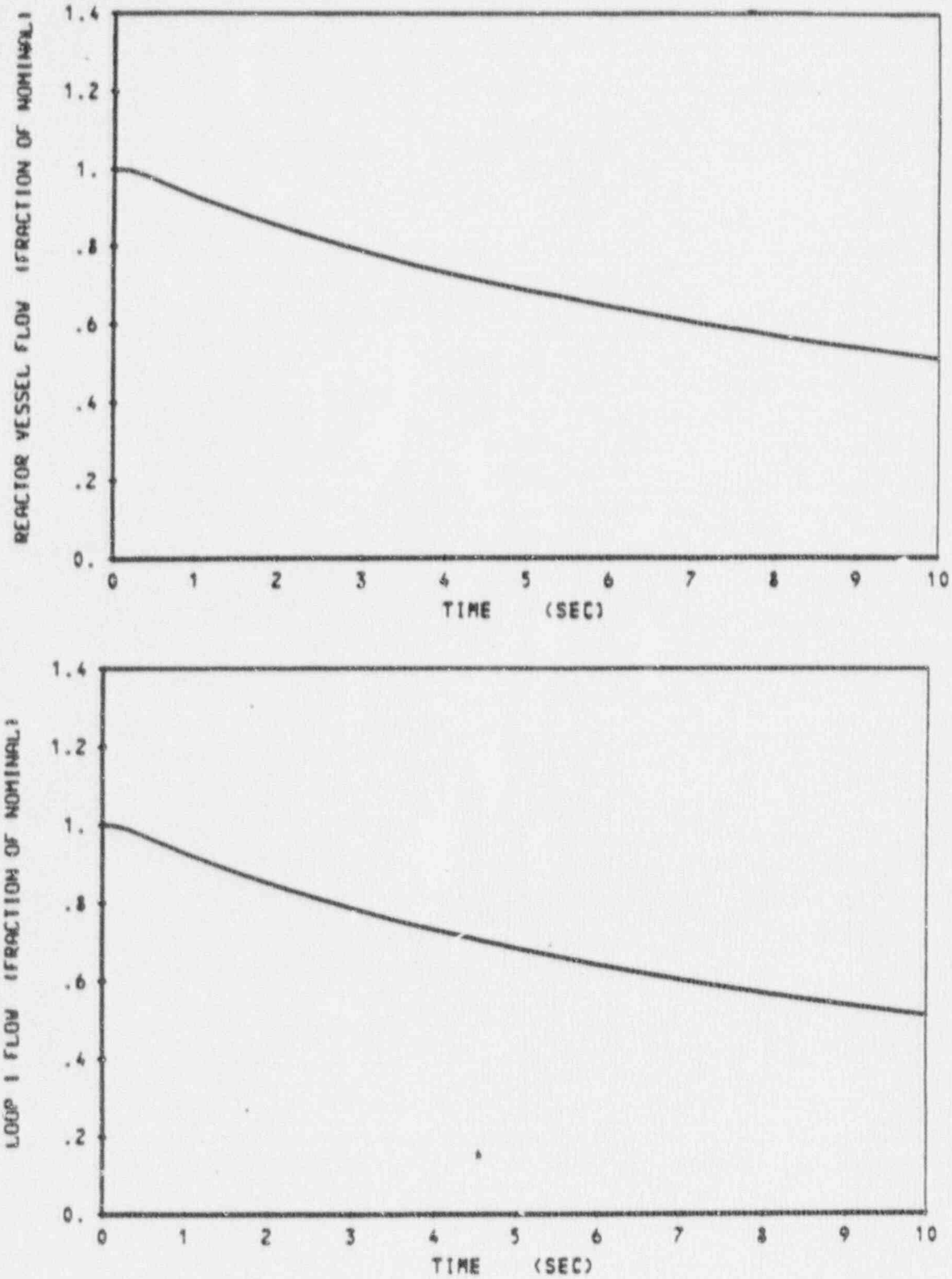
VOGTLE
ELECTRIC GENERATING PLANT
UNIT 1 AND UNIT 2

DNBR VERSUS TIME FOR 4 LOOPS IN OPERATION, 2 PUMPS COASTING DOWN

FIGURE 15.3.1-4



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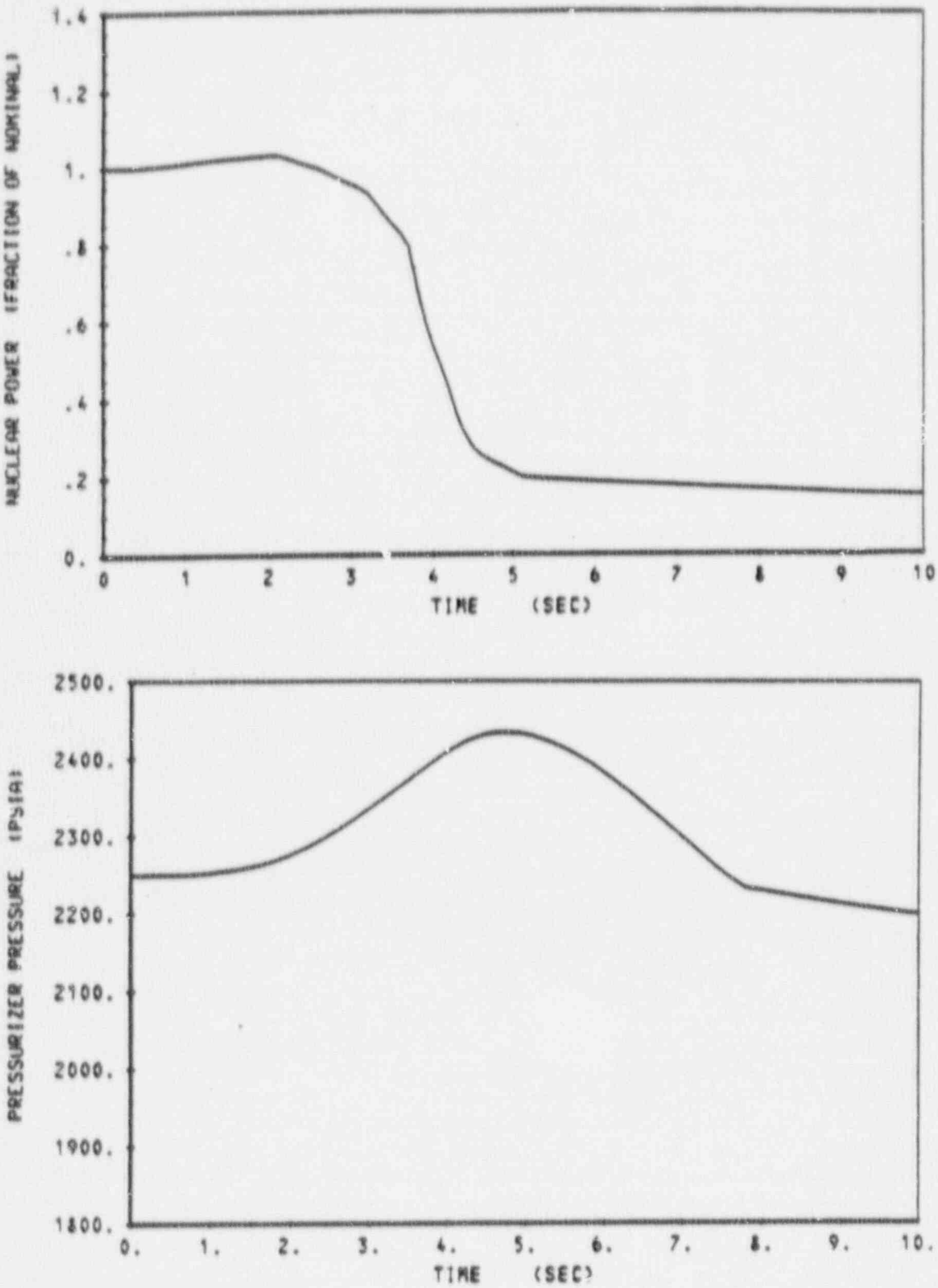
VOGTLE
ELECTRIC GENERATING PLANT
UNIT 1 AND UNIT 2

FLOW TRANSIENTS FOR 4 LOOPS IN OPERATION, 4 PUMPS COASTING DOWN

FIGURE 15.3.2-1



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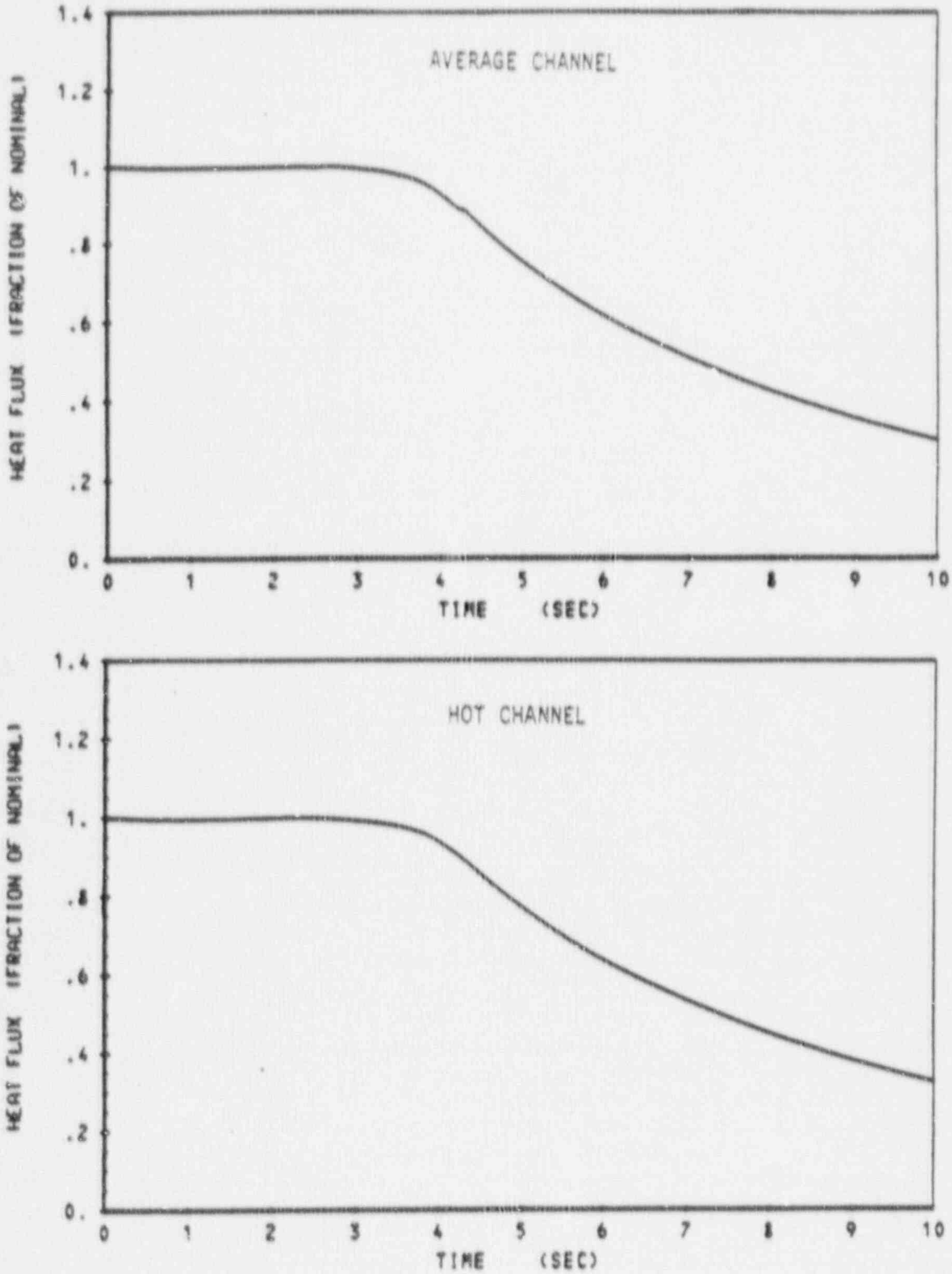
VOGTLE
ELECTRIC GENERATING PLANT
UNIT 1 AND UNIT 2

NUCLEAR POWER AND PRESSURIZER PRESSURE TRANSIENTS FOR 4 LOOPS IN OPERATION, 4 PUMPS COASTING DOWN

FIGURE 15.3.2-2



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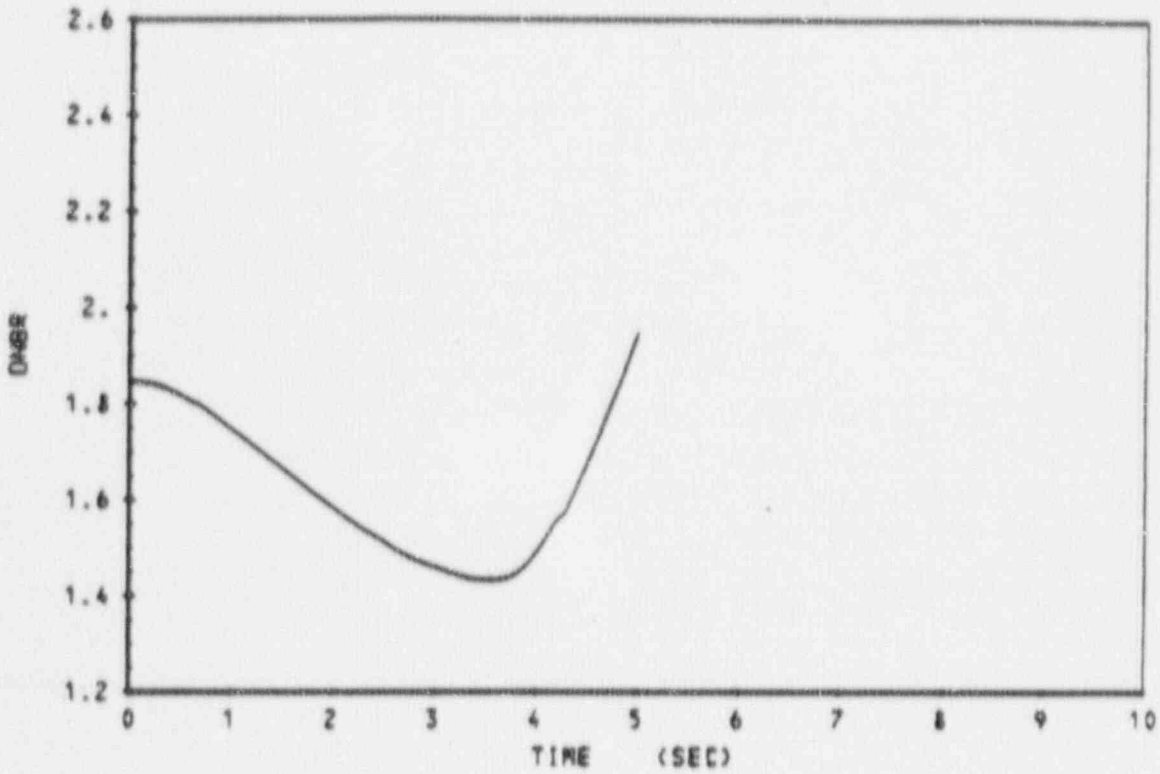
VOGTLE
ELECTRIC GENERATING PLANT
UNIT 1 AND UNIT 2

AVERAGE AND HOT CHANNEL HEAT FLUX TRANSIENTS FOR 4 LOOPS IN OPERATION, 4 PUMPS COASTING DOWN

FIGURE 15.3.2-3



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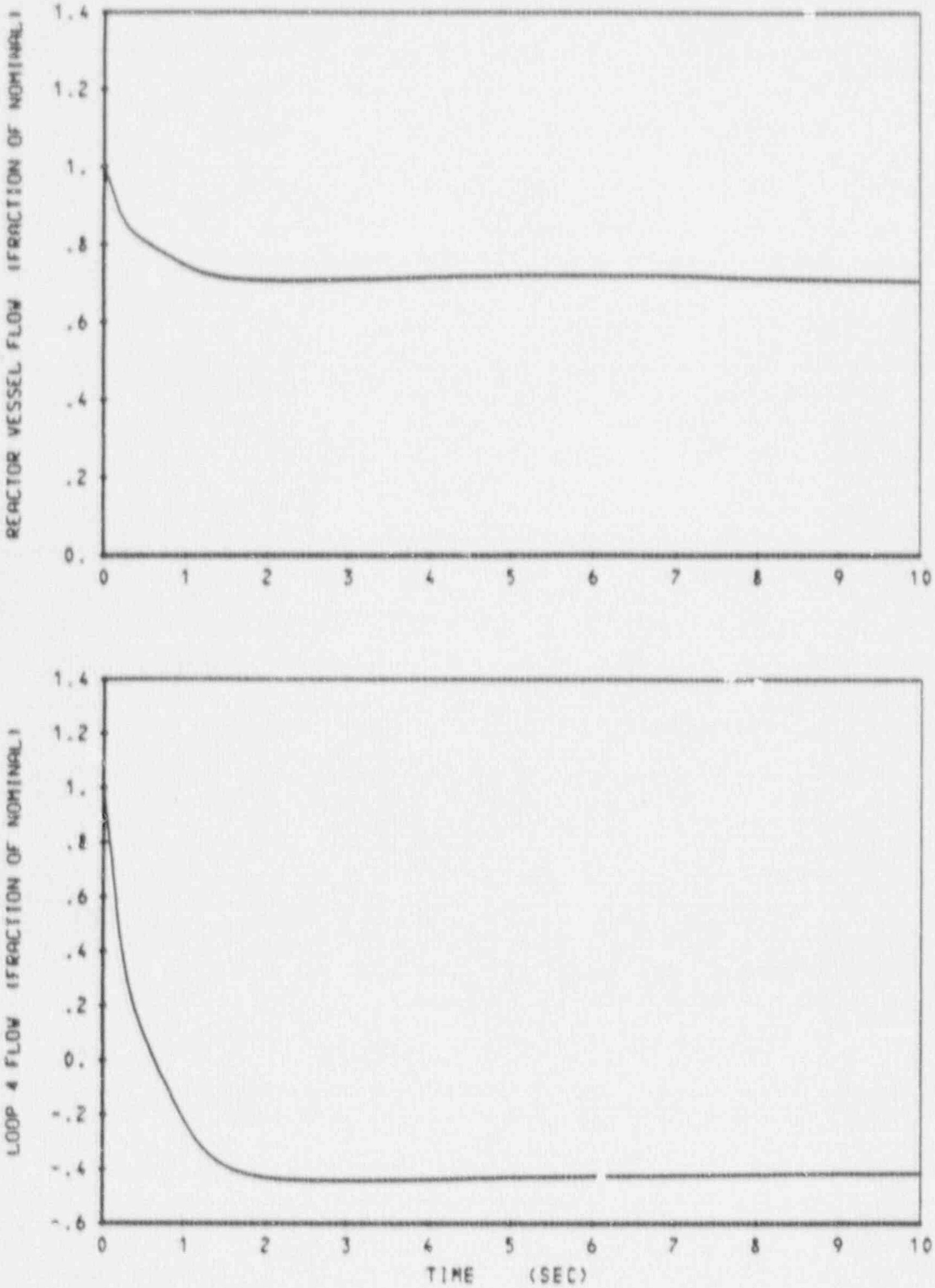
VOGTLE
ELECTRIC GENERATING PLANT
UNIT 1 AND UNIT 2

DNBR VS. TIME FOR 4 LOOPS IN OPERATION
4 PUMPS COASTING DOWN

FIGURE 15.3.2-4



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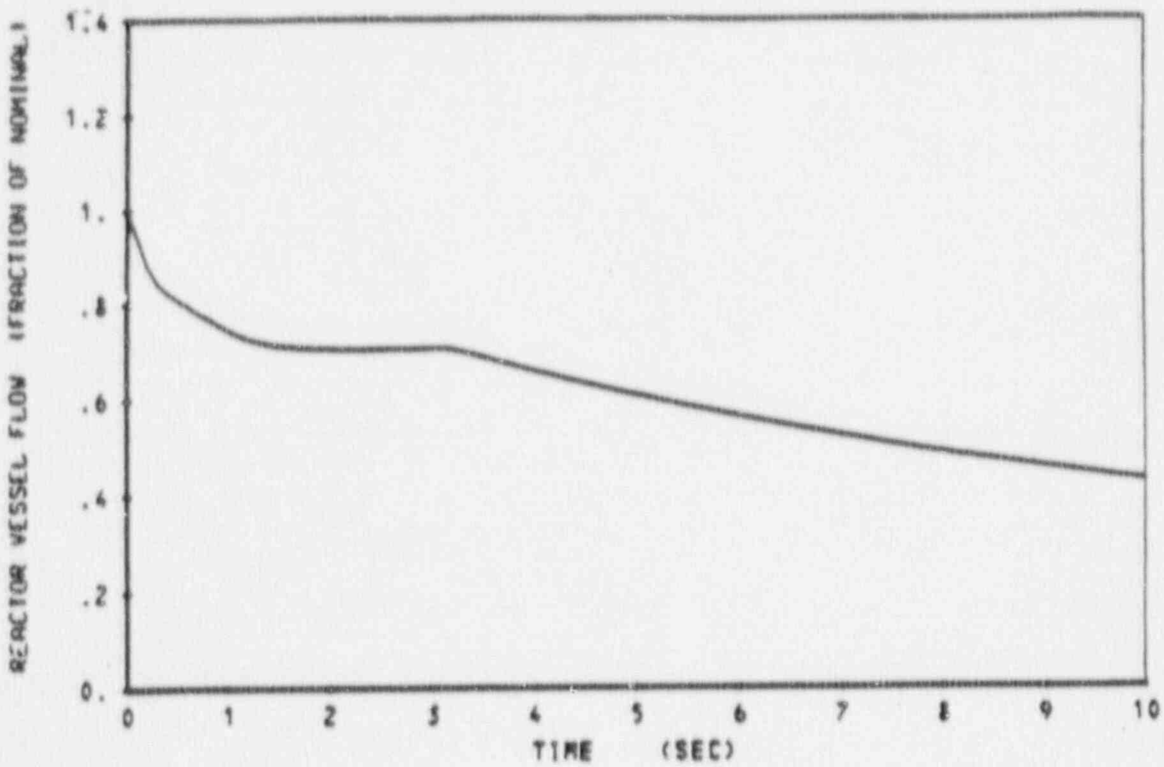
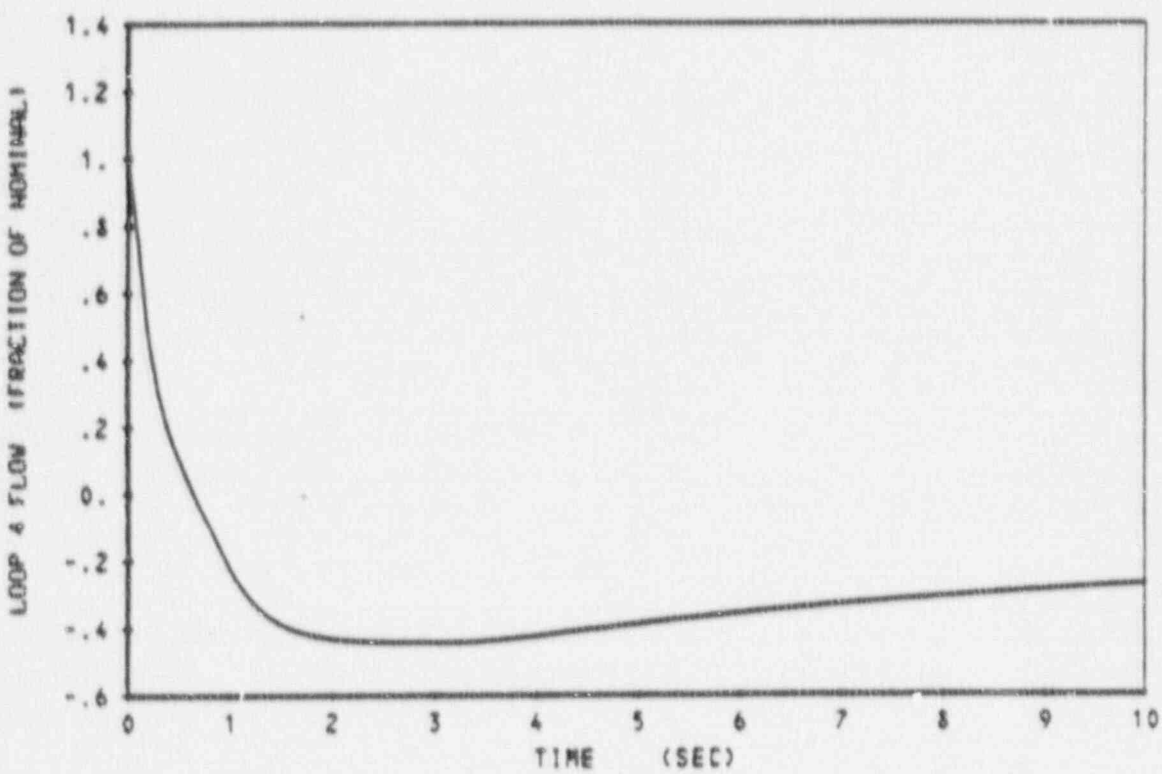
VOGTLE
ELECTRIC GENERATING PLANT
UNIT 1 AND UNIT 2

FLOW TRANSIENTS FOR 4 LOOPS IN OPERATION,
1 LOCKED ROTOR WITH OFFSITE POWER AVAILABLE

FIGURE 15.3.3-1a





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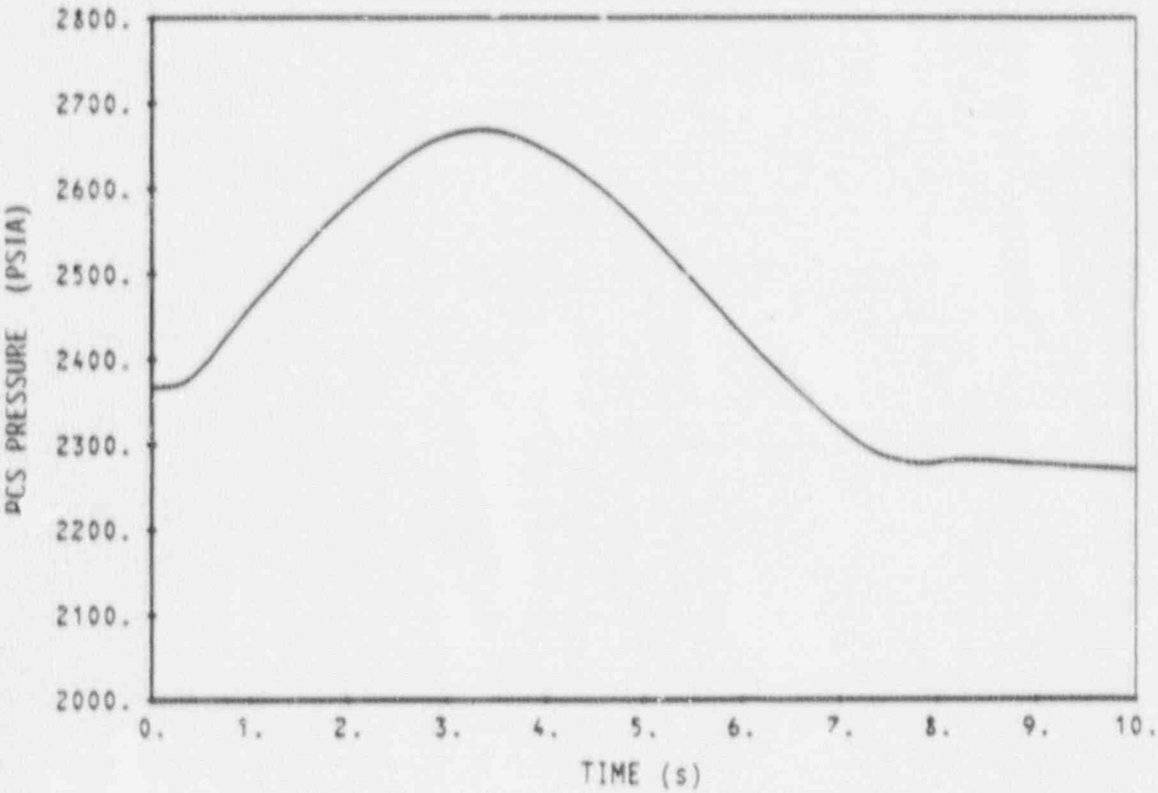
ELECTRIC GENERATING PLANT
UNIT 1 AND UNIT 2

FLOW TRANSIENTS FOR 4 LOOPS IN OPERATION,
1 LOCKED ROTOR WITHOUT OFFSITE POWER AVAILABLE

FIGURE 15.3.3-1b



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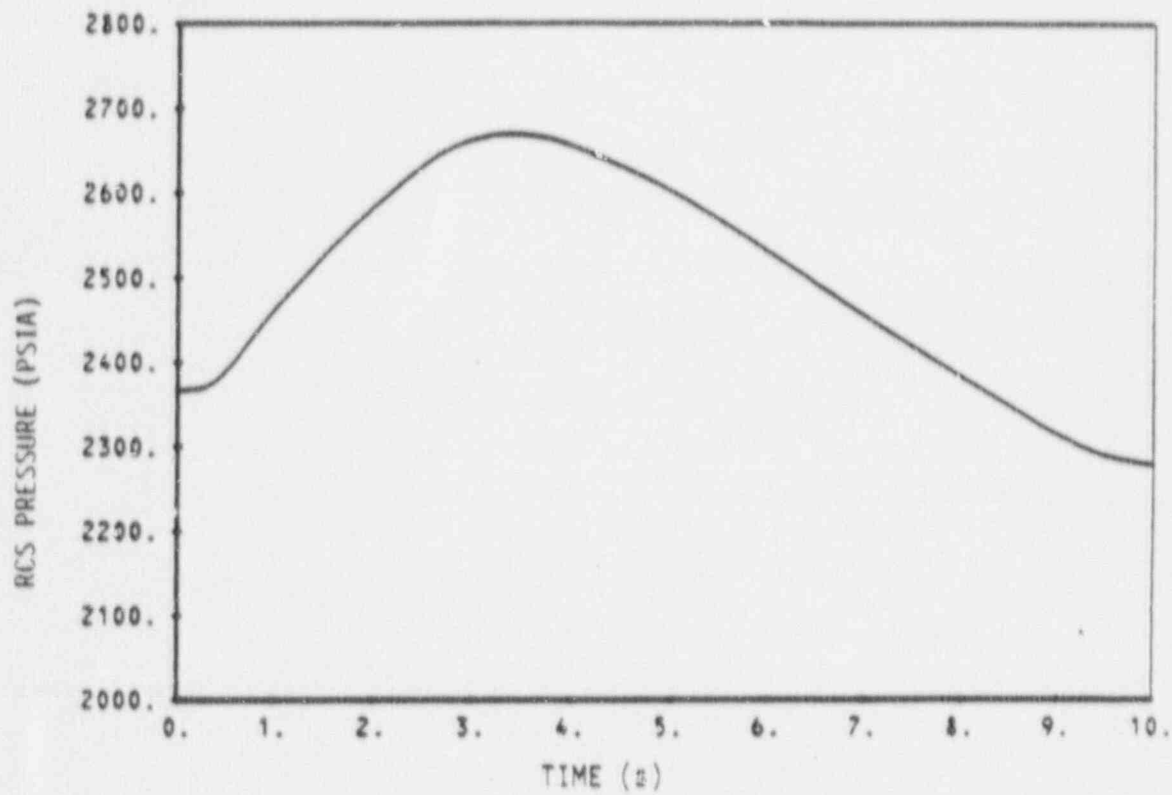
ELECTRIC GENERATING PLANT
UNIT 1 AND UNIT 2

PEAK REACTOR COOLANT PRESSURE FOR 4 LOOPS IN OPERATION,
1 LOCKED ROTOR WITH OFFSITE POWER AVAILABLE

FIGURE 15.3.3-2a



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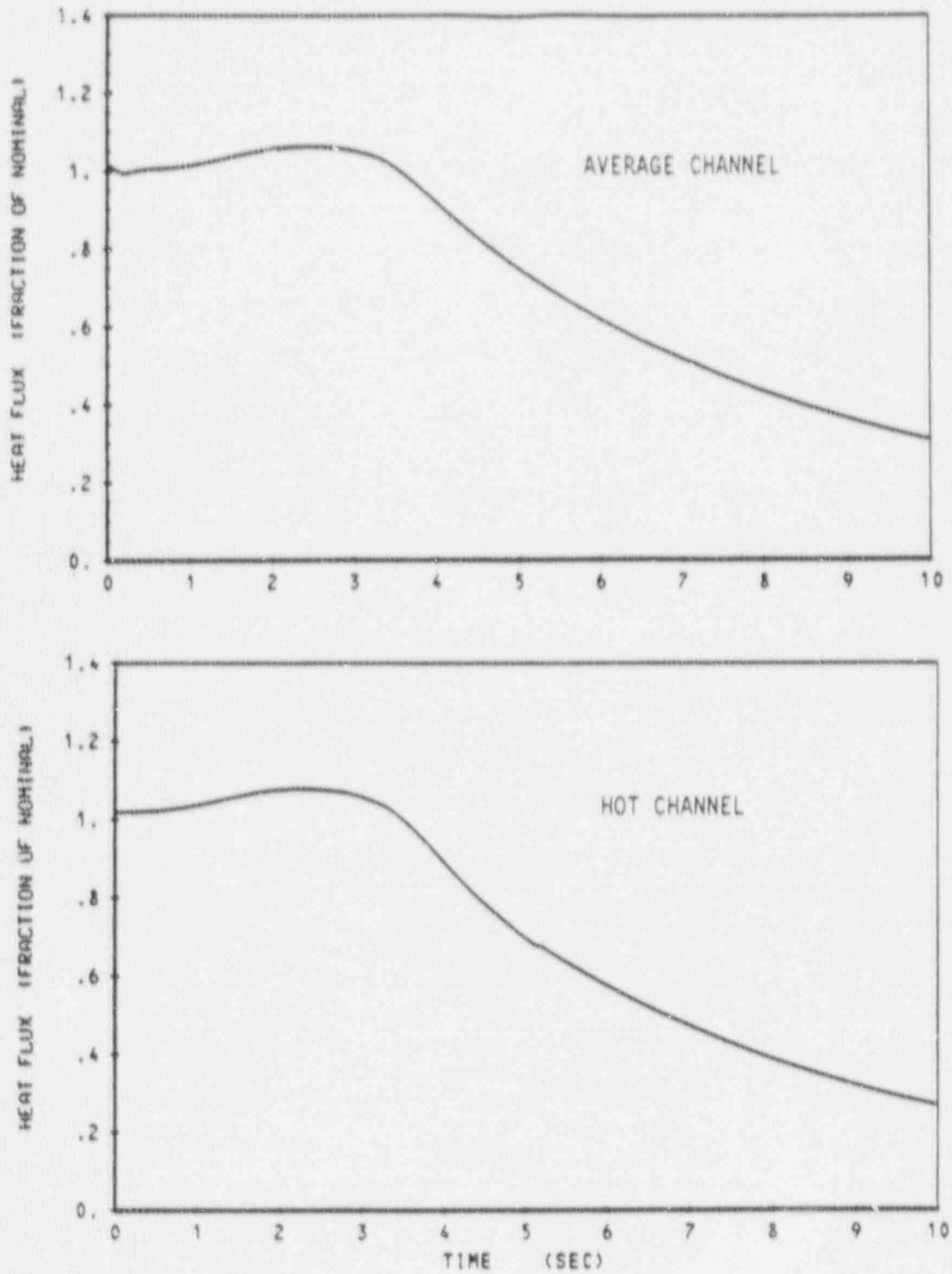
VOGTLE
ELECTRIC GENERATING PLANT
UNIT 1 AND UNIT 2

PEAK REACTOR COOLANT PRESSURE FOR 4 LOOPS IN OPERATION, 1 LOCKED ROTOR WITHOUT OFFSITE POWER AVAILABLE

FIGURE 15.3.3-2b



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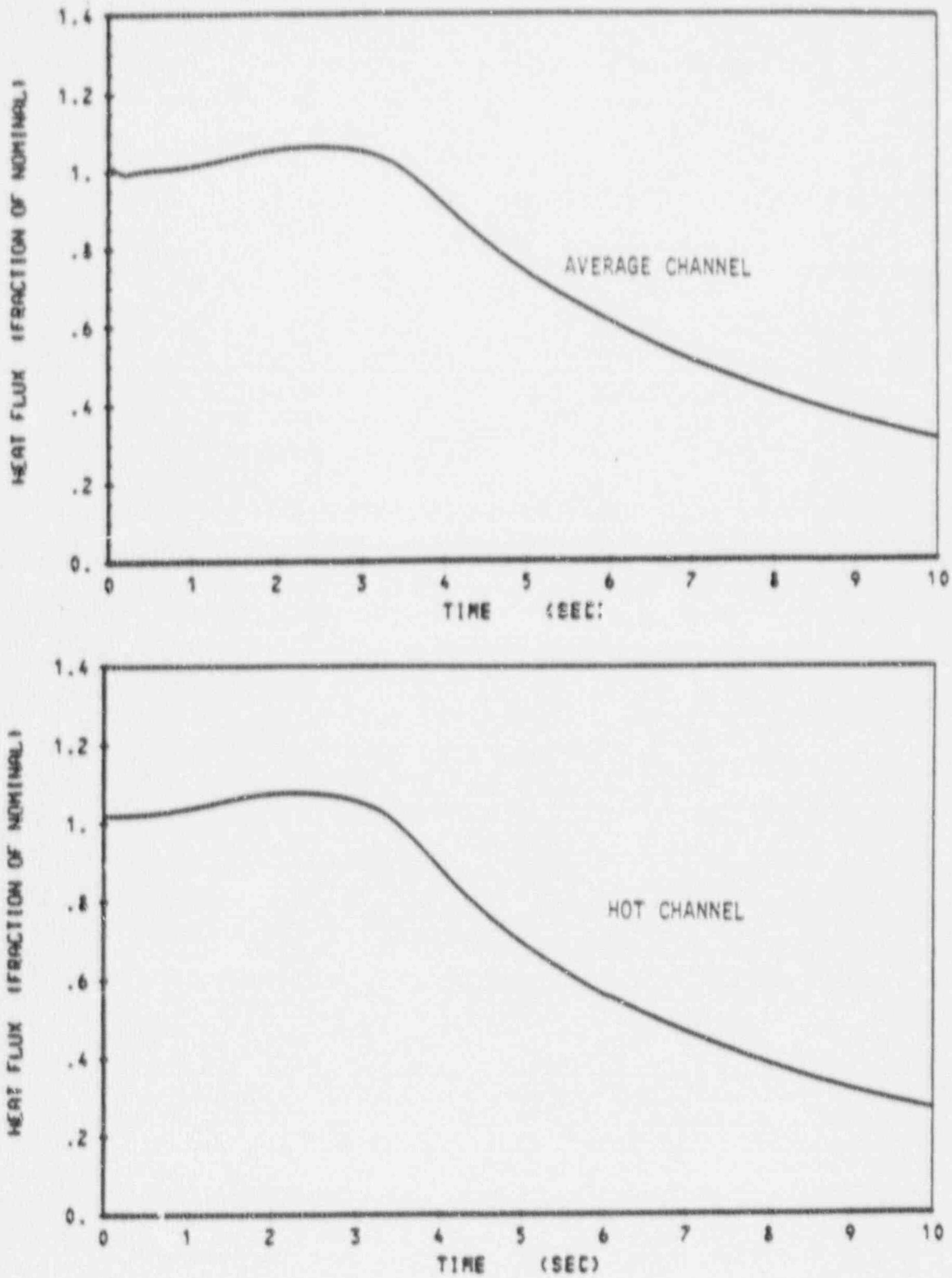
VOGTLE
ELECTRIC GENERATING PLANT
UNIT 1 AND UNIT 2

AVERAGE AND HOT CHANNEL HEAT FLUX TRANSIENTS FOR 4 LOOPS IN OPERATION, 1 LOCKED ROTOR WITH OFFSITE POWER AVAILABLE

FIGURE 15.3.3-3a



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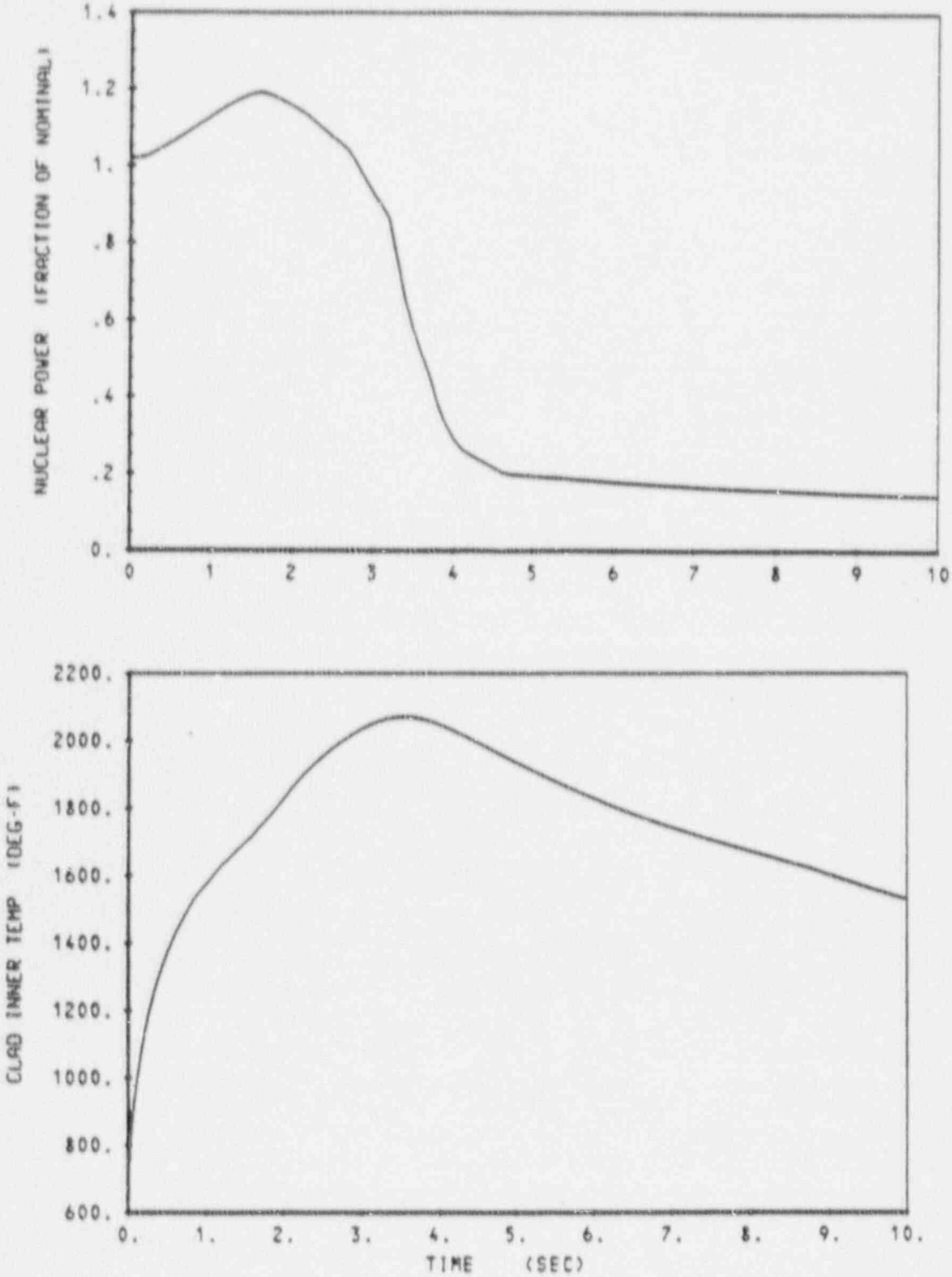
VOGTLE
ELECTRIC GENERATING PLANT
UNIT 1 AND UNIT 2

AVERAGE AND HOT CHANNEL HEAT FLUX TRANSIENTS FOR 4 LOOPS IN OPERATION, 1 LOCKED ROTOR WITHOUT OFFSITE POWER AVAILABLE

FIGURE 15.3.3-3b



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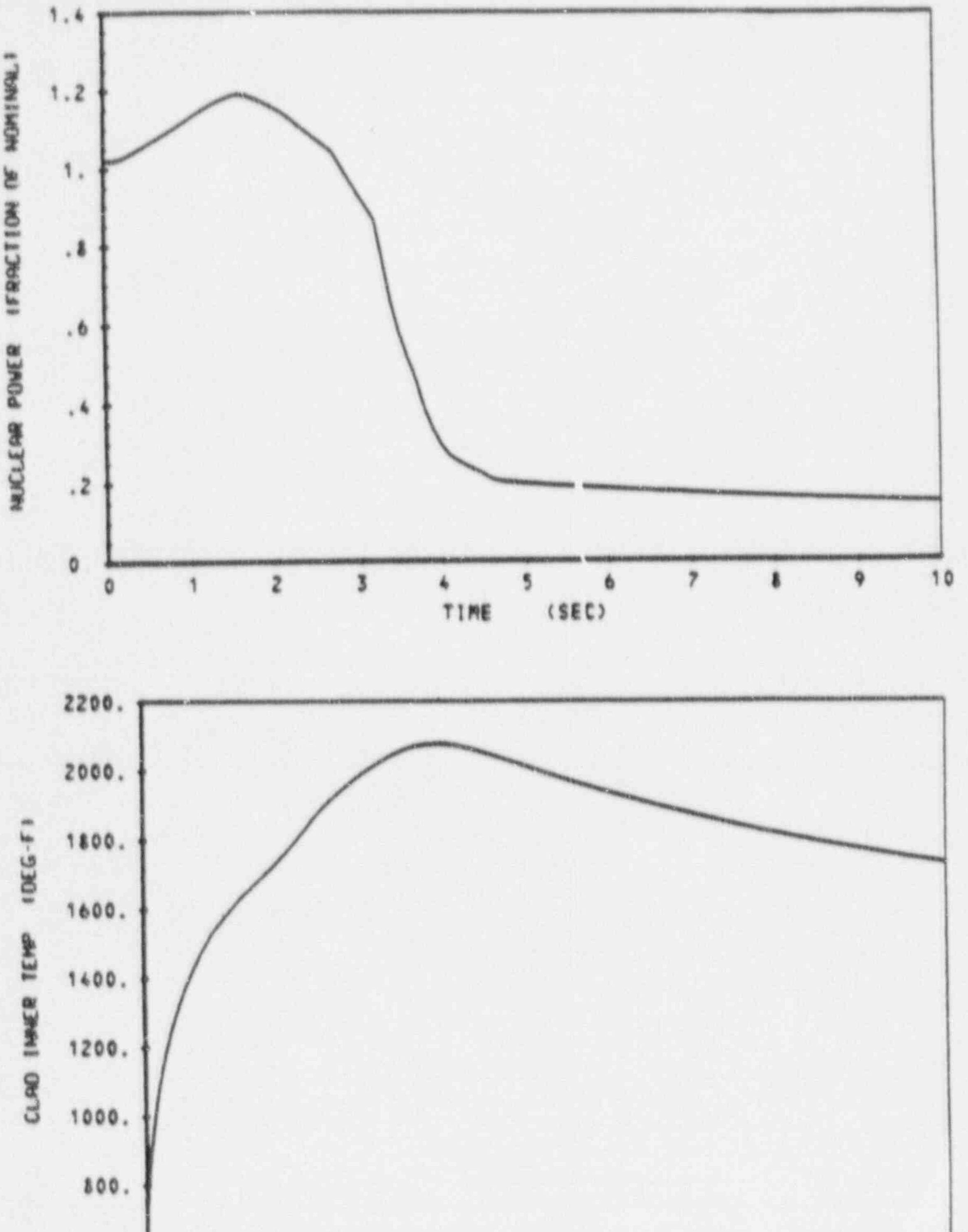
VOGTLE
ELECTRIC GENERATING PLANT
UNIT 1 AND UNIT 2

NUCLEAR POWER AND MAXIMUM CLAD TEMPERATURE AT HOT SPOT TRANSIENTS FOR 4 LOOPS IN OPERATION, 1 LOCKED ROTOR WITH OFFSITE POWER AVAILABLE

FIGURE 15.3.3-4a



Georgia Power VOGTLE ELECTRIC GENERATING PLANT UNIT 1 AND UNIT 2	NUCLEAR POWER AND MAXIMUM CLAD TEMPERATURE AT HOT SPOT TRANSIENTS FOR 4 LOOPS IN OPERATION, 1 LOCKED ROTOR WITHOUT OFFSITE POWER AVAILABLE
	FIGURE 15.3.3-4b

15.4 Reactivity and Power Distribution Anomalies

Several postulated faults can result in reactivity and power distribution anomalies. Control rod motion, control rod ejection, boron concentration changes, or addition of cold water to the RCS results in reactivity changes. Control rod motion, control rod misalignment, control rod ejection, or fuel assembly mislocation results in power distribution changes. This section discusses these events.

15.4.1 Uncontrolled Rod Cluster Control Assembly Bank Withdrawal from a Subcritical or Low-Power Startup Condition

15.4.1.1 Introduction

An RCCA withdrawal incident is an uncontrolled addition of reactivity to the reactor core caused by withdrawal of RCCA banks resulting in a power excursion. While the occurrence of a transient of this type is highly unlikely, a malfunction of the control rod drive system can cause such a transient. This could occur with the reactor either subcritical, low power startup, or at power. Section 15.4.2 discusses the "at power" case.

RCCA bank withdrawal adds reactivity at a prescribed and controlled rate to bring the reactor from a subcritical condition to a low power level during startup. Although the initial startup procedure uses the method of boron dilution, the normal startup is with RCCA bank withdrawal. RCCA bank motion can cause much faster changes in reactivity than can be made by changing boron concentration.

The control rod drive mechanisms wire into preselected banks which remain unchanged during the core life. The circuit design is such that RCCAs can not be withdrawn in other than their proper withdrawal sequence. Control of the power supplied to the rod banks is such that no more than two banks can be withdrawn at any time. The RCCA drive mechanism is the magnetic latch type, and the coil actuation sequencing provides variable speed travel. The analysis of the maximum reactivity insertion rate includes the assumption of the simultaneous withdrawal of the two sequential banks having the maximum combined worth at maximum speed.

Should a continuous control rod assembly withdrawal initiate, the transient will terminate by the following reactor trip functions:

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

- o Intermediate range high neutron reactor flux 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 are reading above approximately 10 percent of full power/flux level and is automatically reinstated when three of the four power range channels indicate a power/flux level below this setpoint.

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

- o Power range high neutron flux reactor trip (high setting) is actuated when two out of the four power range channels indicate a flux level above a preset setpoint. This trip function is always active.

- o High nuclear flux rate reactor trip is actuated when the positive rate of change of neutron flux on two out of four nuclear power range channels indicates a rate above the preset setpoint. It is always active.

In addition, control rod stops on high intermediate range flux (one out of two) and high power range flux (one out of four) serve to cease rod withdrawal and prevent the need to actuate the intermediate range flux trip and the power range flux trip, respectively.

This is an ANS Condition II incident.

15.4.1.2 Method of Analysis

The following three stages comprise the analysis of the uncontrolled RCCA bank withdrawal from subcritical accident: first, an average core nuclear power transient calculation; then, an average core heat transfer calculation; and finally, the DNBR calculation. The spatial neutron kinetics computer code TWINKLE (Reference 1) performs the average core calculation to determine the average power generation with time including the various total core feedback effects, i.e. Doppler reactivity and moderator reactivity. FACTRAN (Reference 2) performs a fuel rod transient heat transfer calculation to determine the average heat flux and temperature transients. The average heat flux is next used in THINC for transient DNBR calculations.

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

A. Since the magnitude of the neutron flux peak reached during the initial part of the transient for any given rate of reactivity insertion is strongly dependent on the Doppler power reactivity coefficient, the analysis employs a conservatively low value for Doppler power defect (-940 pcm).

B. The contribution of the moderator reactivity coefficient is negligible during the initial part of the transient because the heat transfer time constant between the fuel and the moderator is much longer than the neutron flux response time constant; however, after the initial neutron flux peak, the moderator temperature reactivity coefficient affects the succeeding rate of power increase. The analysis assumes a moderator temperature coefficient which is +7 pcm/°F at the zero power nominal temperature.

C. The analysis assumes the reactor to be at hot zero power (557°F). This assumption is more conservative than that of a lower initial system temperature. The higher initial system temperature yields a larger fuel-to-water heat transfer coefficient, a larger fuel-specific heat, and a less-negative (smaller absolute magnitude) Doppler coefficient; these reduce the Doppler feedback effect, thereby increasing the neutron flux peak. The high neutron flux peak combined with a high fuel specific heat and larger heat transfer coefficient yields a larger peak heat flux. The analysis assumes the initial effective multiplication factor (k_{eff}) to be 1.0 since this results in the maximum neutron flux peak.

D. The most adverse combination of instrumentation error, setpoint error, delay for trip signal actuation, and delay for control rod assembly release are taken into account. The analysis assumes a 10 percent increase in the power range flux trip setpoint, raising it from the nominal value of 25 percent to a value of 35 percent and not taking any credit for the source and intermediate range protection. Figure 15.4.1-1 shows that the rise in nuclear flux is so rapid that the effect of error in the trip setpoint on the actual time at which the rods release is negligible. Besides the above, the assumption that the highest worth control rod assembly is stuck in its fully withdrawn position is the basis of the rate of negative reactivity insertion corresponding to the reactor trip.

E. The maximum positive reactivity insertion rate assumed is greater than that for the simultaneous withdrawal of the two sequential control banks having the greatest combined worth at maximum speed (45 in./min).

F. The DNB analysis assumes the most limiting axial and radial power shapes associated with having the two highest combined worth banks in their high-worth position.

G. The analysis assumes the initial power level to be below the power level expected for any shutdown condition (10^{-9} fraction of nominal power). The combination of highest reactivity insertion rate and low initial power produces the highest peak heat flux.

H. The analysis assumes two RCPs to be in operation (Mode 3 Technical Specification allowed operation). This is conservative with respect to the DNB transient.

The accident analysis employs the STDP with the initial conditions shown in Tables 15.0.3-2 and 15.0.3-3.

No single active failure in any plant systems or equipment will adversely affect the consequences of the accident.

15.4.1.3 Results

The nuclear power, heat flux, fuel average temperature, and clad temperature versus time transient results are shown in Figures 15.4.1-1 through 15.4.1-3. In addition, Table 15.4.1-1 presents the time sequence of events.

For all regions of the core, the DNB design basis is met.

15.4.1.4 Conclusions

The minimum DNBR remains above the safety analysis limit value; therefore, the conclusions presented in the FSAR remain valid.

15.4.2 Uncontrolled Rod Cluster Control Assembly Bank Withdrawal At Power

15.4.2.1 Introduction

An uncontrolled RCCA withdrawal at power results in an increase in 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 reactor coolant temperature. Unless terminated by manual or automatic action, the power mismatch and resultant coolant temperature rise could eventually result in DNB; therefore, to avert damage to the fuel clad the reactor protection system is designed to terminate any such transient before the DNBR falls below the limit value.

The automatic features of the reactor protection system which prevent core damage in an RCCA bank withdrawal incident at power include the following:

- o Power range neutron flux instrumentation actuates a reactor trip on high neutron flux if two out of four channels exceed an overpower setpoint.
- o Reactor trip actuates if any two out of four ΔT channels exceed an OTΔT setpoint. This setpoint is automatically varied with axial power distribution, coolant average temperature, and coolant average pressure to protect against DNB.
- o Reactor trip actuates if any two out of four ΔT channels exceed an OPΔT setpoint. This setpoint is automatically varied with coolant average temperature so that the allowable heat generation rate (kW/ft) is not exceeded.
- o A high pressurizer pressure reactor trip, actuated from any two out of four pressure channels, is set at a fixed point. This set pressure is less than the set pressure for the pressurizer safety valves.
- o Any two out of three level channels when the reactor power is above approximately 10 percent (permissive P-7) will actuate a high pressurizer water level reactor trip.

Besides the above listed reactor trips, there are the following RCCA withdrawal blocks:

- o High neutron flux (one out of four)
- o OPΔT (two out of four)
- o OTΔT (two out of four)

FSAR Chapter 7 describes the manner in which the combination of OPΔT and OTΔT trips provide protection over the full range of RCS conditions. Figure 15.0.6-1 presents allowable reactor coolant loop average temperature and ΔT for the design power distribution and flow as a function of primary coolant pressure.

The purpose of this analysis is to demonstrate the manner in which the above protective systems function for various reactivity insertion rates from different initial conditions to prevent fuel damage. Reactivity insertion rates and initial conditions influence which protection function actuates first.

This is an ANS Condition II incident.

15.4.2.2 Method of Analysis

The analysis of this transient employs the LOFTRAN code (Reference 3). LOFTRAN uses the core limits as illustrated in Figure 15.0.6-1 as input to determine the minimum DNBR during the transient.

The analysis of this accident uses the RTDP described in Reference 7. Section 15.0.3 discusses the plant characteristics and initial conditions. For an uncontrolled RCCA bank withdrawal at power accident, the analysis assumes the following conservative assumptions:

A. Nominal values form the basis of the initial reactor power, pressure, and RCS temperature assumption. The limit DNBR includes uncertainties in initial conditions as described in Reference 7.

B. Reactivity coefficients -- two cases analyzed:

   1. A +7 pcm/°F moderator temperature coefficient of reactivity and a least-negative Doppler-only power coefficient form the basis of the beginning of life minimum reactivity feedback assumption.

   2. A conservatively large positive moderator density coefficient (corresponding to a large negative moderator temperature coefficient) and a most-negative Doppler-only power coefficient form the basis of the end of life maximum reactivity feedback assumption.

C. A conservative value of 118 percent of nominal full core power actuates the reactor trip on high neutron flux. The ΔT trips include all adverse instrumentation and setpoint errors while maximum values form the basis of the delays for the trip signal actuation assumption.

D. The assumption that the highest worth assembly is stuck in its fully withdrawn position forms the basis of the RCCA trip insertion characteristic.

E. The analysis examines a range of reactivity insertion rates. The maximum positive reactivity insertion rate is greater than that for the simultaneous withdrawal of the two control banks having the maximum combined worth at maximum speed assuming normal overlap.

No single active failure in any plant systems or equipment will adversely affect the consequences of the accident.

15.4.2.3 Results

Figures 15.4.2-1 through 15.4.2-3 show the transient response for a rapid RCCA bank withdrawal incident starting from full power with minimum feedback. Reactor trip on high neutron flux occurs shortly after the start of the accident. Because of the rapid reactor trip with respect to the thermal time constants of the plant, small changes in T_{avg} and pressure result, and the margin to DNB is maintained.

The transient response for a slow RCCA bank withdrawal from full power with minimum feedback is shown in Figures 15.4.2-4 through 15.4.2-6. Reactor trip on OTΔT occurs after a longer period and the rise in temperature and pressure is consequently larger than for rapid RCCA bank withdrawal. Again, the minimum DNBR is greater than the limit value.

Figure 15.4.2-7 shows the minimum DNBR as a function of reactivity insertion rate from initial full power operation for both minimum and maximum reactivity feedback. It can be seen that the two reactor trip functions (high neutron flux and OTΔT functions) provide DNB protection over the whole range of reactivity insertion rates. The minimum DNBR is always greater than the limit value.

Figures 15.4.2-8 and 15.4.2-9 show the minimum DNBR as a function of reactivity insertion rate for RCCA bank withdrawal incidents starting at 60 and 10 percent power, respectively. The results are similar to the 100 percent power case; however, as the initial power decreases, the range over which the OTΔT trip is effective is increased. In neither case does the DNBR fall below the limit value.

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

The reactor trips sufficiently fast during the RCCA bank withdrawal at power transient to ensure that there is not a reduction in the ability of the primary coolant to remove heat from the fuel rods. Thus, the fuel cladding temperature does not rise significantly above its initial value during the transient.

Table 15.4.1-1 shows the calculated sequence of events for the uncontrolled RCCA bank withdrawal at power incident.

15.4.2.4 Conclusions

The high neutron flux and OTΔT trip functions provide adequate protection over the entire range of possible reactivity insertion rates (i.e., the minimum value of DNBR is always larger than the limit value for all fuel types); therefore the conclusions presented in the FSAR remain valid.

15.4.3 Rod Cluster Control Assembly Misalignment (System Malfunction or Operator Error)

15.4.3.1 Introduction

RCCA misoperation accidents include the following:

o One or more dropped RCCAs within the same group

o A dropped RCCA bank

o Statically misaligned RCCA

o Withdrawal of a single RCCA

Each RCCA has a position indicator channel which displays the position of the assembly in a display grouping that is convenient to the operator. Fully inserted assemblies are also indicated by a rod at bottom signal which actuates a local alarm and a control room annunciator. Group demand position is also indicated.

RCCAs move in preselected banks, and the banks move in the same preselected sequence. Each bank of RCCAs consists of 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) withdraws the RCCA attached to the mechanism. Mechanical failures are in the direction of insertion or immobility.

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 event analyzed must result from multiple wiring failures, multiple significant operator errors, or subsequent and repeated operator disregard of event indication. The probability of such a combination of conditions is low such that the limiting consequences may include slight fuel damage.

The following indicators detect one or more dropped RCCAs, RCCA group, or RCCA bank:

o Sudden drop in the core power level as seen by the nuclear instrumentation system

o Asymmetric power distribution as seen on out-of-core neutron detectors or core exit thermocouples

o Rod at bottom signal

- o Rod deviation alarm
- o Rod position indication

The following indicators detect misaligned RCCAs:

- o Asymmetric power distribution as seen on out-of-core neutron detectors or core exit thermocouples
- o Rod deviation alarm
- o Rod position indicators

The resolution of the rod position indicator channel is ±5 percent of span (±7.5 in.). Deviation of any RCCA from its group by twice this distance (10 percent of span or 15 in.) will not cause power distributions worse than the design limits. The deviation alarm alerts the operator to rod deviation with respect to the group position in excess of 5 percent of span. If the rod deviation alarm is not operable, the Technical Specifications require the operator to take action.

If one or more rod position indicator channels is out of service, the operator must follow detailed operating instructions to ensure the alignment of the non-indicated RCCAs. The operator is also required to take action, as required by the Technical Specifications.

In the unlikely event of simultaneous electrical failures which could result in single RCCA withdrawal, the plant annunciator will display both the rod deviation and rod control urgent failure; and the rod position indicators will indicate the relative positions of the RCCAs 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 indication. The OTΔT reactor trip provides automatic protection for this event, although due to the increase in local power density, it is not possible to always provide assurance that the core safety limits will not be exceeded.

15.4.3.2 <u>Method of Analysis</u>

A. One or More Dropped RCCAs from the Same Group

The LOFTRAN computer code (Reference 3) calculates the transient system response for the evaluation of the dropped RCCA event. 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.

Calculated statepoints and nuclear models form the basis 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. The transient response analysis, nuclear peaking factor analysis, and performance of the DNB design basis confirmation are in accordance with the methodology described in Reference 5. Note that the analysis does not take credit for the negative flux rate reactor trip.

B. Dropped RCCA Bank

A dropped RCCA bank results in a symmetric power change in the core. As discussed in Reference 5, assumptions made for the dropped RCCA(s) analysis provide a bounding analysis for the dropped RCCA bank.

C. Statically Misaligned RCCA

FSAR Table 4.1-2 describes the computer codes used in the analysis of steady-state power distributions. The peaking factors are then used as input to the THINC code to calculate the DNBR. The analysis examines the case of the worst rod withdrawn from bank D inserted at the insertion limit with the reactor initially at full power. The analysis assumes this incident to occur at beginning of life since this results in the minimum value of moderator temperature coefficient. This assumption maximizes the power rise and minimizes the tendency of increased moderator temperature to flatten the power distribution.

D. Single RCCA Withdrawal at Full Power

FSAR Table 4.1-2 describes the computer codes used in the calculation of power distributions within the core. The peaking factors are then used by THINC to calculate the minimum DNBR for the event. The plant's analysis is for the case of the worst withdrawn rod from D bank inserted at the insertion limit, with the reactor initially at full power. The analysis assumes the transient to occur at beginning of life since this results in the minimum value of moderator temperature coefficient. This assumption maximizes the power rise and minimizes the tendency of increased moderator temperature to flatten the power distribution.

15.4.3.3 Results

A. One or More Dropped RCCAs

Single or multiple dropped RCCAs within the same group result in a negative reactivity insertion. The core is not adversely affected during this period, since power is decreasing rapidly. Either reactivity feedback or control bank withdrawal will reestablish power.

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. Figure 15.4.3-1 shows a typical transient response to a dropped RCCA (or RCCAs) in automatic rod control mode. In all cases, the minimum DNBR remains above the limit value.

Following plant stabilization, the operator may manually retrieve the RCCA by following approved operating procedures.

B. Dropped RCCA Bank

A dropped RCCA bank results in a negative reactivity insertion greater than 500 pcm. The core is not adversely affected during the insertion period, since power is decreasing rapidly. The transient will proceed as described in Part A; however, the return to power will be less due to the greater worth of the entire bank. The power transient for a dropped RCCA bank is symmetric. Following plant stabilization, normal procedures are followed.

C. 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 transient approaches the postulated conditions. The bank can be inserted to its insertion limit with any one assembly fully withdrawn without the DNBR falling below the limit value.

The insertion limits in the Technical Specifications may vary from time to time depending on several limiting criteria. The full-power insertion limits on control bank D must be chosen to be above that position which meets the minimum DNBR and peaking factors. The full power insertion limit is usually 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 limit value. The analysis of this case assumes that the initial reactor power, pressure, and RCS temperature are at their nominal values, with the increased radial peaking factor associated with the misaligned RCCA.

For RCCA misalignments with one RCCA fully inserted, the DNBR does not fall below the limit value. The analysis of this case assumes that initial reactor power, pressure, and RCS temperatures are at their nominal values, with the increased radial peaking factor associated with the misaligned RCCA.

DNB does not occur for the RCCA misalignment incident, thus there is no reduction in the ability of the primary coolant to remove heat from the fuel rod. 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 rate is well below that which would cause fuel melting.

After identifying an RCCA group misalignment condition, the operator must take action as required by the plant Technical Specifications and operating instructions.

D. The analysis of the single rod withdrawal event considers the following two events:

1. If the reactor is in the manual rod 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 withdrawing RCCA. Depending on initial bank insertion and location of the withdrawn RCCA, automatic reactor trip may not occur quickly enough to prevent the minimum DNBR from falling below the limit value. Evaluation of this case at the power and coolant conditions at which the OTΔT trip would trip the plant shows that an upper limit for the number of rods with a DNBR less than the limit value is 5 percent.

2. If the reactor is in the automatic rod control mode, the multiple failures that result in the withdrawal of a single RCCA cause immobility of the other RCCAs in the controlling bank. The transient will then proceed in the same manner as case 1 described above.

For such cases as above, a reactor trip will ultimately ensue, although not quickly enough in all cases to prevent a minimum DNBR in the core of less than the limit value. Following reactor trip, normal shutdown procedures are followed.

15.4.3.4 Conclusions

For cases of dropped RCCAs or dropped banks, the DNBR remains greater than the limit value; therefore, the DNB design criterion is met.

For all cases of any RCCA fully 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 limit value.

For the case of the accidental withdrawal of a single RCCA, with the reactor in the automatic or manual control mode and initially operating at full power with D bank at the insertion limit, an upper bound of the number of fuel rods experiencing DNBR is 5 percent of the total fuel rods in the core.

15.4.4 Startup of an Inactive Reactor Coolant Pump at an Incorrect Temperature

15.4.4.1 Introduction

Technical Specification 3/4.4.1 does not permit VEGP Units 1 and 2 operation in Modes 1 and 2 with less than four loops operating; however, this analysis assumes approximately 75 percent power in Mode 1 in order to bound Mode 3 operation where the Technical Specifications permit operation with less than four loops.

If the plant operates with one RCP out of service, there is reverse flow through the inactive loop due to the pressure difference across the reactor vessel. The cold leg temperature of the inactive loop is identical to the cold leg temperature of the active loops. If the reactor is operated at power and assuming there is no isolation of the secondary side of the steam generator in the inactive loop, 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.

If the startup of an inactive RCP accident occurs, the transient terminates 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 three-loop operation.

This is an ANS Condition II incident.

15.4.4.2 Method of Analysis

The analysis of this transient uses three digital computer codes. The LOFTRAN computer code (Reference 3) calculates the loop and core flow, nuclear power, and core pressure and temperature transients following the startup of an idle pump. FACTRAN (Reference 2) calculates the core heat flux transient based on core flow and nuclear power from LOFTRAN. The THINC code is then used to calculate the DNBR during the transient based on system conditions calculated by LOFTRAN and heat fluxes calculated by FACTRAN.

Section 15.0.3 discusses plant characteristics and initial conditions. In order to obtain conservative results for the startup of an inactive pump accident, the following assumptions are made (this analysis employed STDP):

A. 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 consistent with the maximum steady-state power level that would be permitted with three loops in operation. The high initial power gives the greatest temperature difference between the core inlet temperature and the inactive loop hot leg temperature.

B. Following initiation of startup of the idle pump, the inactive loop flow reverses and accelerates to its nominal full-flow value in approximately 9 seconds.

C. The analysis assumes a conservatively large negative moderator temperature coefficient.

D. The analysis assumes a least-negative Doppler-only power coefficient.

E. The initial reactor coolant loop flows are at the appropriate values for one pump out of service.

F. The reactor trip occurs on low coolant flow when the power range neutron flux exceeds the P-8 setpoint. The P-8 setpoint is conservatively assumed to be 84 percent of rated power, which corresponds to the nominal setpoint plus 9 percent for nuclear instrumentation errors.

No single active failure in any plant systems or equipment will adversely affect the consequences of the accident.

15.4.4.3 Results

The results following the startup of an idle pump with the above listed assumptions are shown in Figures 15.4.4-1 through 15.4.4-5. These curves show that 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 the core average temperature. The minimum DNBR during the transient is 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 the increase in reactor coolant flow and, as the inactive loop flow reverses, to the colder water entering the core from the hot leg side 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 in Figure 15.4.4-1.

The calculated sequence of events for this accident is shown in Table 15.4.1-1. The transient results illustrated in Figures 15.4.4-1 through 15.4.4-5 indicate that a stabilized plant condition, with the reactor tripped, is rapidly approached. By following normal shutdown procedures, the plant can subsequently achieve cooldown.

15.4.4.4 Conclusions

The transient results show that the core is not adversely affected. There is considerable margin to the safety analysis limit DNBRs; thus, the DNB design basis is met and the conclusions presented in the FSAR remain valid.

15.4.5 A Malfunction or Failure of the Flow Controller in a Boiling Water Reactor Loop that Results in an Increased Reactor Coolant Flowrate

This section is not applicable to VEGP Units 1 and 2.

15.4.6 Chemical and Volume Control System Malfunction that Results in a Decrease in the Boron Concentration in the Reactor Coolant

15.4.6.1 Introduction

Feeding primary grade water into the RCS via the reactor makeup portion of the chemical and volume control system (CVCS) adds reactivity to the core. 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 permits the operator to match the boron concentration of reactor coolant makeup water during normal charging to that in the RCS. Even under various postulated failure modes, the design of the CVCS limits the potential rate of dilution to a value which gives the operator sufficient time to correct the situation in a safe and orderly manner.

The opening of the primary water makeup control valve supplies water to the RCS which can dilute the reactor coolant. Inadvertent dilution can be readily terminated by closing this valve. In order to add makeup water to the RCS at pressure, at least one charging pump in addition to the primary makeup water pumps must be running. Normally, only one primary water supply pump is operating while the other is on standby.

The boric acid from the boric acid tank blends with primary grade water at the mixing tee, and the preset flowrates of boric acid and primary grade water on the control board determine the composition.

Information on the status of reactor coolant makeup is continuously available to the operator. Lights on the control board indicate the operating condition of pumps in the CVCS. Alarms actuate to warn the operator if boric acid or demineralized water flowrates deviate from preset values as a result of system malfunction.

This is an ANS Condition II incident.

15.4.6.2 Method of Analysis

To cover all phases of the plant operation, this analysis considers boron dilution during refueling, cold shutdown, hot shutdown, hot standby, startup, and power operation. The analysis assumes conservative values for the critical parameters, i.e., high RCS critical boron concentrations, high boron worths, minimum shutdown margins, and small RCS volumes. These result in conservative calculations of the time available for the operator to determine the cause of the addition and take corrective action before shutdown margin is lost.

A. Dilution During Refueling

This analysis evaluates boron dilution events during refueling (Mode 6). During refueling, a very small amount of unborated chemical solution is allowed to enter the RCS for water chemistry quality control. The opening of CVCS valves 176 and 177 provides the dilution flow path. The maximum flowrate possible through this flow path is less than 3.5 gpm. Any other chemical makeup solution required during refueling will be borated water supplied from the refueling water storage tank by the low head safety injection pumps.

Valves 175 and 183 in the CVCS will be locked closed during refueling operations. These valves will block additional flow paths which could allow more than 3.5 gpm of unborated chemical makeup water to reach the RCS.

B. Dilution During Cold Shutdown, Hot Shutdown, and Hot Standby

This analysis evaluates boron dilution events during cold shutdown (Mode 5), hot shutdown (Mode 4), and hot standby (Mode 3). The analysis uses failure modes and effects analysis, human error, and event tree analysis to identify credible boron dilution initiators and to evaluate the plant response to these events. For the initiators identified, the calculation of the time intervals from alarm to loss of shutdown margin helped determine the length of time available for operator response. These calculations depended on dilution flowrates, boron concentrations, and RCS volumes specific to the event and mode of operation. The technique modeled realistic plant conditions and responses, including both mechanical failure and human errors.

The analysis identified four events which are the most likely initiators. FSAR Section 15.4.6.2.1.2 provides details on the events.

The examination of Mode 5b (mid-loop operation) for the addition of small amounts of unborated chemical solution into the RCS for water quality chemistry control was included. The maximum flowrate possible for this flow path is approximately 3 percent of that associated with the limiting flow path for Modes 3, 4, and 5a (RCS loops filled).

C. Dilution During Full Power Operation, Including Startup

For the dilution during startup (Mode 2), the analysis assumes an initial maximum critical boron concentration of 1800 ppm based on the rods being inserted to the insertion limits. The analysis assumes the minimum change in the boron concentration from this initial condition to a hot zero power critical condition to be 300 ppm. The analysis also assumes full rod insertion to occur due to reactor trip, minus the most reactive stuck rod. The analysis assumes the dilution flow to be the combined capacity of the two primary water makeup pumps (approximately 242 gpm) and a minimum RCS water volume of 9583 ft^3. This volume corresponds to the active volume of the RCS minus the pressurizer and accounts for 10 percent steam generator tube plugging.

During power operation (Mode 1), the plant operates under either manual or automatic rod control. While the plant is in manual control, the analysis assumes the dilution flow to be a maximum of 242 gpm, which is the combined capacity of the two primary water makeup pumps. While in automatic control, the maximum letdown flow (approximately 125 gpm) limits the dilution flow. The analysis assumes an initial maximum critical boron concentration, corresponding to the rods inserted to the insertion limits at hot full power, of 1800 ppm. The analysis also assumes the minimum change in the boron concentration from this initial condition to a hot zero power critical condition to be 300 ppm. The analysis assumes full rod insertion to occur due to reactor trip, minus the most reactive stuck rod. The analysis uses a minimum water volume of 9583 ft^3 in the RCS, corresponding to the active volume of the RCS minus the pressurizer volume and accounts for 10 percent tube plugging.

No single active failure in any plant systems or equipment will adversely affect the consequences of the accident.

15.4.6.3 Results

During refueling, the maximum flowrate associated with the available dilution flow path is very small. The total time from the initiation of the event to the eventual complete loss of shutdown margin is significantly large compared to the minimum required operator action time; therefore, the operator has sufficient time to terminate the RCS water chemistry adjustments before the loss of shutdown margin. Additionally, assuming the availability of the high flux at shutdown alarm (set at 2.3 times background), it is shown that the Technical Specification shutdown margin requirement for Mode 6 is sufficient to ensure that the operator has 30 minutes from the time of alarm to terminate the dilution before shutdown margin is lost.

For dilution during cold shutdown (RCS loops filled), the Technical Specifications provide the required shutdown margin as a function of RCS boron concentration. The specified shutdown margin ensures that the operator has 15 minutes from the time of the high flux at shutdown alarm to the total loss of shutdown margin. For mid-loop operation during cold shutdown, the results are similar to those discussed above for refueling.

For dilution during hot shutdown and hot standby, the Technical Specifications provide the required shutdown margin as a function of RCS boron concentration. The specified shutdown margin ensures that the operator has 15 minutes from the time of the high flux at shutdown alarm to the total loss of shutdown margin.

In the event of an unplanned approach to criticality or dilution during power escalation while in the startup mode, the operator is alerted to the event by a source range reactor trip. After the reactor trip, there are more than 15 minutes for operator action prior to loss of shutdown margin.

During full power operation with the reactor in manual control, an OTΔT reactor trip alerts the operator to an uncontrolled dilution. At least 15 minutes are available after the reactor trip for operator action before the loss of shutdown margin.

During full power operation with the reactor in automatic control, the rod insertion limit alarms alert the operator to an uncontrolled dilution. At least 15 minutes are available after the low-low rod insertion limit alarm for operator action before the loss of shutdown margin.

15.4.6.4 Conclusions

The results presented above demonstrate that adequate time is available for the operator to manually terminate the source of dilution flow. Following termination of the dilution flow, the operator can initiate boration to establish adequate shutdown margin.

15.4.7 Inadvertent Loading and Operation of a Fuel Assembly in an Improper Position

Fuel type does not affect the ability of the in-core instrumentation to detect the inadvertent loading and operation of a fuel assembly in an improper position; therefore, the conclusions of the FSAR remain valid.

15.4.8 Spectrum of Rod Cluster Control Assembly Ejection Accidents

15.4.8.1 Introduction

This accident is a mechanical failure of a control rod mechanism pressure housing resulting in the ejection of an RCCA and drive shaft. The consequence of this mechanical failure is a rapid positive reactivity insertion together with an adverse core power distribution, possibly leading to localized fuel rod damage. FSAR Section 15.4.8 further discusses this accident.

The limiting criteria are as follows:

- o Average fuel pellet enthalpy at hot spot below 200 cal/g for unirradiated and irradiated fuel
- o Peak reactor coolant pressure less than that which could cause stresses to exceed the faulted condition stress limits

o Fuel melting less than 10 percent of the fuel volume at the hot spot even if the average fuel pellet enthalpy is below the limits of the first criterion listed above

Thi is an ANS Condition IV incident.

15.4.8.2 Method of Analysis

The performance of the calculation of the RCCA ejection transient is in two stages: first an average core channel calculation and then a hot region calculation. The average core calculation uses spatial neutron-kinetics methods to determine the average power generation as a function of time including the various total core feedback effects, i.e., Doppler reactivity and moderator reactivity. Enthalpy and temperature transients at 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 conservatively assumed to persist throughout the transient. Reference 6 provides a detailed discussion of the method of analysis.

Average Core Analysis

The average core transient analysis uses the spatial kinetics computer code, TWINKLE (Reference 1). This code solves the two group neutron diffusion theory kinetic equation 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 multi-region, transient fuel-clad-coolant heat transfer model for calculation of pointwise Doppler and moderator feedback effects. This analysis uses the code 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. 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. Further description of TWINKLE appears in Section 15.0.11.4.

Hot Spot Analysis

In the hot spot analysis, the initial heat flux is equal to the nominal value times the design hot channel factor. During the transient, the heat flux hot channel factor is linearly increased to the transient value in 0.1 second, the time for full ejection of the rod. Therefore, the assumption is made that the hot spots before and after ejection are coincident. This is conservative since the peak after ejection will occur in or adjacent to the assembly with the ejected rod, and before ejection the power in this region will be depressed due to the inserted rod.

The hot spot analysis uses the detailed fuel and clad transient heat transfer computer code, FACTRAN (Reference 2). 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 local coolant conditions. The zirconium-water reaction is explicitly represented, and all material properties are represented as a function of temperatures. The code uses a conservative radial power distribution within the fuel rod.

FACTRAN uses the Dittus-Boelter or Jens-Lottes correlation to determine the film heat transfer before DNB, and the Bishop-Sandberg-Tong correlation (Reference 8) to determine the film boiling coefficient after

DNB. The Bishop-Sandberg-Tong correlation is conservatively used assuming zero bulk fluid quality. The code does not calculate the DNBR, instead specifying a conservative DNB heat flux which forces the code into DNB. The code can calculate the gap heat transfer coefficient; however, it is adjusted to force the full power, steady-state temperature distribution to agree with fuel heat transfer design codes. Further description of FACTRAN appears in Section 15.0.11.1.

System Overpressure Analysis

Because the transient does not exceed the safety limits for fuel damage specified earlier, there is little likelihood of fuel dispersal into the coolant; therefore, the basis of the pressure surge calculation may be 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. A THINC calculation uses this heat flux data to determine the volume surge. Finally, the LOFTRAN computer code (Reference 3) simulates the volume surge. This code calculates the pressure transient taking into account fluid transport in the RCS and heat transfer to the steam generators. No credit is taken for the possible pressure reduction caused by the assumed failure of the control rod pressure housing.

15.4.8.3 Input Parameters

Input parameters for the analysis are conservatively selected on the basis of values calculated for this type of core. The discussion of the more important parameters is presented below. Table 15.4.8-1 presents the parameters used in this analysis.

1. Ejected Rod Worths and Hot Channel Factors

   The values for ejected rod worths and hot channel factors are calculated using either three-dimensional static methods or a synthesis method employing one-dimensional and two-dimensional calculations. The analysis uses standard nuclear design codes. 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. The analysis assumes adverse xenon distributions to provide worst case results.

   The ejected rod worth and hot channel factors include appropriate margins to account for any calculational uncertainties, including an allowance for nuclear power peaking due to densification.

   During initial plant startup physics testing, the measurement of ejected rod worths and power distributions is performed in the zero and part power configuration and compared to the values used in the analysis. Experience has shown that the ejected rod worth and power peaking factors are consistently overpredicted in the analysis.

2. Reactivity Feedback Weighting Factors

The largest temperature increases, and hence, the largest reactivity feedbacks, occur in channels where the power is higher than average. Since the weight of a region is dependent on flux, these regions have high weights. This means that the reactivity feedback is larger than that indicated by a simple channel analysis. Physics calculations have been performed for temperature changes with a flat temperature distribution and with a large number of axial and radial temperature distributions. The analysis compares reactivity changes and determines effective weighting factors. These weighting factors take the form of multipliers which, when applied to single channel feedbacks, correct them to effective whole-core feedbacks for the appropriate flux shape. Axial weighting is not necessary because this analysis employs a one-dimensional (axial) spatial kinetics method (i.e., the initial condition is made to match the ejected rod configuration). In addition, this analysis does not apply any weighting to the moderator feedback. The analysis applies a conservative radial weighting factor 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 have also been shown to be conservative compared to three-dimensional analysis (Reference 6).

3. Moderator and Doppler Coefficient

The nuclear code adjusts the critical boron concentrations at the beginning of life and end of life in order to obtain moderator density coefficient curves which are conservative when compared to the actual design conditions for the plant. As discussed above, these results do not have any weighting factor applied to them. The resulting moderator temperature coefficient is at least +7 pcm/°F at the appropriate zero or full power nominal average temperature for the beginning of life cases.

The calculation determines the Doppler reactivity defect as a function of power level using a one-dimensional steady-state computer code with a Doppler weighting factor of 1.0. The Doppler weighting factor will increase under accident conditions, as discussed above.

4. Delayed Neutron Fraction, β_{eff}

To allow for future cycles, the analysis used conservative β_{eff} estimates of 0.54 percent at beginning of life hot zero power; 0.57 percent at beginning of life hot full power; and 0.46 percent for both end of life cases.

5. Trip Reactivity Insertion

The trip reactivity insertion assumed is given in Table 15.4.8-1 and includes the effect of one stuck RCCA adjacent to the ejected rod. The reactivity of the ejected rod reduces these values. Dropping a rod of the required worth into the core simulates the shutdown reactivity. The start of rod motion occurred 0.5 seconds after reaching the high neutron flux trip point. It is assumed that insertion to dashpot does not occur until 3.3 seconds after the rods begin to fall. The choice of such a conservative insertion rate means that there is over 1 second after reaching the trip point before significant shutdown reactivity is inserted into the core. This is a particularly important conservatism for hot full power accidents.

The minimum design shutdown margin available for this plant at hot zero power (HZP) may only occur at end of life in the equilibrium cycle. This value includes an allowance for the worst stuck rod, an adverse xenon distribution, conservative Doppler and moderator defects, and an allowance for calculational uncertainties. Physics calculations have shown that two stuck RCCAs (one of which is the worst ejected rod) reduce the shutdown by about an additional 1% $\Delta k/k$. Therefore, following a reactor trip resulting from an RCCA ejection accident, the reactor will be subcritical when the core returns to HZP.

6. Reactor Protection

High neutron flux trip (high and low setting) and high positive rate of neutron flux increase trip, although the analysis only modeled the high neutron flux trip (high and low setting), provide reactor protection for a rod ejection. These protection functions are part of the reactor trip system. No single failure of the reactor trip system will negate the protection functions required for the rod ejection accident, or adversely affect the consequences of the accident.

No single active failure in any plant systems or equipment will adversely affect the consequences of the accident.

15.4.8.4 Results

Table 15.4.8-1 summarizes the results. The results of the analysis present cases for both beginning and end of life at zero and full power.

A. Beginning of Life, Full Power

This case assumed Control Bank D at its insertion limit. The worst ejected rod worth and hot channel factor were conservatively calculated to be 0.24% $\Delta k/k$ and 5.5, respectively. The peak hot spot fuel centerline temperature reached melting, conservatively assumed at 4900°F; however, fuel melting was well below the limiting criterion of 10 percent of the pellet volume at the hot spot.

B. Beginning of Life, Zero Power

For this condition, the analysis assumed Control Bank D to be fully inserted with Banks B and C at their insertion limits. The worst ejected rod is in Control Bank D and has a worth of 0.75% $\Delta k/k$ and a hot channel factor of 11.0. The fuel centerline temperature was 3985°F.

C. End of Life, Full Power

The analysis assumed Control Bank D at its insertion limit. The ejected rod worth and hot channel factors were conservatively calculated to be 0.25% $\Delta k/k$ and 6.0 respectively. The peak hot spot fuel centerline temperature reached melting, conservatively assumed at 4800°F; however, fuel melting was well below the limiting criterion of 10 percent of the pellet volume at the hot spot.

D. End of Life, Zero Power

The analysis of this case determines the ejected rod worth and hot channel factor by assuming Control Bank D to be fully inserted with Banks B and C at their insertion limits. The results were 0.84% $\Delta k/k$ and 26.0, respectively. The fuel centerline temperature was 3891°F. The Doppler weighting factor for this case is significantly higher than for the other cases due to the very large transient hot channel factor.

For all the cases analyzed, average fuel pellet enthalpy at the hot spot remained below 200 cal/g.

Table 15.4.1-1 presents the calculated sequence of events for the worst case rod ejection accidents. Figures 15.4.8-1 through 15.4.8-4 show the worst case rod ejection accident results. For all cases, rod insertion occurs after the nuclear power excursion is terminated by Doppler feedback. As discussed previously, the reactor remains subcritical following reactor trip.

Fission Product Release

The analysis conservatively assumes that the gaps of all rods entering DNB release fission products. In all cases considered, less than 10 percent of the rods entered DNB based on a detailed three-dimensional THINC analysis (Reference 6). Although the analysis predicts limited fuel melting at the hot spot for the full-power cases, in practice melting is not likely since the analysis conservatively assumed that the hot spots before and after ejection were coincident.

Pressure Surge

A detailed calculation of the pressure surge for an ejection rod worth of 1 dollar at beginning of life, hot full power, indicates that the peak pressure does not exceed that which would cause reactor pressure vessel stress to exceed the faulted condition stress limits (Reference 11). Since the severity of the present analysis does not exceed the worst case analysis, the accident for this plant will not result in an excessive pressure rise or further adverse effects to the RCS.

Lattice Deformations

A large temperature gradient exists 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 differential expansion tending to bow the midpoint of the rod toward the hotter side of the rod. Calculations indicate that this bowing results in a negative reactivity effect at the hot spot since Westinghouse cores are undermoderated, and bowing tends to increase the undermoderation at the hot spot. In practice, significant bowing is not expected since the structural rigidity of the core is more than sufficient to withstand the forces produced. Boiling in the hot spot region produces a net flow away from that region; however, the fuel releases heat to the water slowly, and it is considered inconceivable that cross flow is 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 produces a reduction in the total core moderator to fuel ratio and a large reduction in this ratio at the hot spot. Therefore, the net effect is negative feedback which leads to the conclusion 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 analysis.

Radiological Consequences

Appendix C contains the evaluation of the radiological consequences of a postulated control rod ejection accident.

15.4.8.5 Conclusions

The results of the analyses do not exceed the described fuel limits; therefore, there is no likelihood 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, there is no likelihood of further consequence to the RCS. The analyses have demonstrated that the fuel rods entering DNB are less than 10 percent of the fuel rods in the core; therefore, the assumption of 10 percent of the fuel rods in the core entering DNB for the fission product release calculation is conservative. The conclusions presented in the FSAR remain valid.

15.4.9 Steamline Break With Coincidental Rod Withdrawal at Power

15.4.9.1 Introduction

The coincidental and consequential occurrence of an uncontrolled RCCA bank withdrawal at power following a steamline break event is one of four potential interaction scenarios resulting from adverse environmental conditions (either inside or outside of containment) following a high energy line break; these scenarios are identified in "IE Information Notice 79-22." The premise of this concern is that during a high energy line break (such as steamline rupture), certain sensors used in the control systems could be exposed to an adverse environment. If the equipment is not qualified for the adverse environment, a control system malfunction might occur.

The automatic rod control system is one of the control systems that could malfunction. The rod control system relies on the measurement of T_{avg}, nuclear power, and turbine impulse pressure to determine if control rod motion is required. A small steamline rupture may occur outside containment near the turbine impulse pressure transmitters or inside containment in the vicinity of the excore detectors, thus exposing equipment used in the rod control system to an adverse environment. If this equipment is not properly qualified for these conditions, a consequential RCCA withdrawal following a steamline rupture may occur.

The steamline break affects the rod control system (via either an inside containment break near the excore detectors or an outside containment break near the turbine impulse transmitters) and causes the control rods to withdraw following the initiation of the transient. This causes an increase in reactor power and core heat flux to the point at which an OPΔT or OTΔT trip setpoint is reached. This trip terminates the most adverse part of the transient. The steamline break causes increased heat removal and subsequent decrease in primary pressure simultaneous with the increase in reactor power. Secondary pressure also decreases until the low steamline pressure setpoint is reached, initiating steamline and feedwater isolation.

Because of the lower RCS pressure coincident with the increase in reactor power, the consequences at the point of peak heat flux may be more adverse than the RCCA bank withdrawal at power transient analyzed in the FSAR.

The most limiting part of this transient pertinent to this study is immediately before reactor trip (i.e., rod motion). The most limiting case is that for the largest steamline break that trips on OPΔT prior to reaching a reactor trip on a safety injection signal (e.g., low steamline pressure). Therefore, the analysis assumes the largest steamline break size for which a low steamline pressure signal will not occur prior to the OPΔT reactor trip, and the analysis terminates 5 seconds after reactor trip. "Steam System Piping Failure" presented in FSAR Section 15.1.5 bounds the return to power following reactor trip and steamline isolation. If the low steamline pressure setpoint is reached, a reactor trip on safety injection actuation would result and terminate the event. Therefore, like the analysis performed for "Uncontrolled Rod Cluster Control Assembly Bank Withdrawal at Power" (Section 15.4.2), to demonstrate protection by the ΔT trips, only the applicable range of these trips needs to be considered. Also note that no credit is taken in the steamline break with coincident rod withdrawal at power analysis for a reactor trip via the high neutron flux overpower protection signal, since this trip function may be inoperable due to adverse environmental conditions associated with a steamline break inside containment.

The performance of the analysis for a steamline break with coincident withdrawal of the control rods due to an adverse environment demonstrates that the corresponding minimum DNBR does not decrease below the appropriate safety analysis limit DNBR value and no fuel or clad damage occurs. Additionally, no system overpressurization is expected since the steamline break results in a RCS depressurization as described above.

This is an ANS Condition III/IV incident.

15.4.9.2 <u>Method of Analysis</u>

The analysis of this transient uses the LOFTRAN computer code (Reference 3). The following assumptions were made for this transient:

A. The analysis employs RTDP methodology in determining initial conditions of maximum core power, reactor coolant average temperature, and minimum reactor coolant pressure.

B. End of life shutdown margin and equilibrium xenon conditions. The analysis assumes the most reactive RCCA stuck in its fully withdrawn position for conditions following reactor trip.

C. The analysis uses a negative moderator coefficient corresponding to the end of life unrodded core. This maximizes the reactivity insertion caused by the cooldown during the steamline break.

D. The analysis assumes the reactor trip setpoints on OPΔT and OTΔT at a conservative value. The ΔT trips include all adverse instrumentation and setpoint errors; the delays for trip actuation are at the maximum values.

E. The analysis bases the RCCA trip insertion characteristic on the assumption that the highest worth assembly is stuck in its fully withdrawn position.

F. The break size assumed for this transient is 2.24 ft^2 (0.560 ft^2 per steam generator). This is the largest break size for which a low steamline pressure signal will not occur before the reactor trip on OPΔT. Before the eventual steamline isolation on low steamline pressure, all four steam generators feed this break. Following steamline isolation, one steam generator feeds the break causing an asymmetric transient.

G. The calculation of the steam flow during a steamline break uses the Moody Curve for $fL/D = 0$.

H. The reactivity insertion rate is 900 pcm/min.

No single active failure in any plant systems or equipment will adversely affect the consequences of the accident.

15.4.9.3 <u>Results</u>

The calculated sequence of events for this transient is shown in Table 15.4.1-1.

Figures 15.4.9-1 and 15.4.9-2 show RCS transient and core heat flux following the steamline rupture with coincident RCCA bank withdrawal.

The steamline break affects the turbine impulse transmitters and causes the control rods to withdraw at the initiation of the transient. This causes an increase in reactor power and core heat flux to the point at which the OPΔT trip setpoint is reached. The reactor trip terminates the most adverse part of the transient. The steamline break causes increased heat removal and subsequent decrease in primary pressure simultaneous with the increase in reactor power. If the transient extends beyond post-reactor trip, secondary pressure will decrease until the low steamline pressure setpoint is reached, initiating steamline and feedwater isolation.

The analysis of the steamline break with coincident RCCA bank withdrawal demonstrates that the DNBR limit is met. The most limiting part of this transient pertinent to this study was immediately before reactor trip (i.e., rod motion). The transient for the steamline break presented in FSAR Section 15.1.5 bounds the return to power following reactor trip and steamline isolation. The other FSAR steamline break analysis assumed a larger break size and initial conditions corresponding to no-load temperatures (i.e., less stored energy in the RCS and reactor fuel).

The DNBR is always greater than the limit value. Figure 15.4.9-3 shows the DNBR as a function of time for this transient.

15.4.9.4 <u>Conclusions</u>

The analysis demonstrates that the DNBR does not decrease below the limit value and no fuel or clad damage occurs. Additionally, no system overpressurization will occur, thus all applicable safety criteria are met. As stated in the results, the large steamline break analysis presented in FSAR Section 15.1.5 bounds the return to power following a reactor trip and steamline isolation; therefore, there is adequate protection to ensure plant safety for this transient.

Table 15.4.1-1 (Sheet 1 of 3)

Time Sequence of Events for Incidents Which Result in Reactivity and Power Distribution Anomalies

Accident	Event	Time Delays (s)
Uncontrolled RCCA bank withdrawal from a subcritical or low-power startup condition		
	Initiation of uncontrolled rod withdrawal from 10^{-9} fraction of nominal power	0.0
	Power range high neutron flux low setpoint reached	12.5
	Peak nuclear power occurs	12.7
	Rods begin to fall into core	13.0
	Minimum DNBR occurs	14.5
	Peak heat flux occurs	14.9
	Peak average clad temperature occurs	15.1
	Peak average fuel temperature occurs	15.4
Uncontrolled RCCA bank withdrawal at power (full power with minimum feedback)		
1. Case A	Initiation of uncontrolled RCCA withdrawal at a high-reactivity insertion rate (80 pcm/sec)	0
	Power range high neutron flux high setpoint reached	1.4
	Rods begin to fall into core	1.9
	Minimum DNBR occurs	2.9
2. Case B	Initiation of uncontrolled RCCA withdrawal at a small reactivity insertion rate (3 pcm/sec)	0
	OTΔT setpoint reached	34.0
	Rods begin to fall into core	36.0
	Minimum DNBR occurs	37.1

Table 15.4.1-1 (Sheet 2 of 3)

Time Sequence of Events for Incidents Which Result in Reactivity and Power Distribution Anomalies

Accident	Event	Time Delays (s)
Startup of an inactive reactor coolant pump at an incorrect temperature		
	Initiation of pump startup	0.0
	Power reaches P-8 trip setpoint	3.5
	Rods begin to drop	4.0
	Minimum DNBR occurs	5.0
CVCS malfunction that results in a decrease in the boron concentration in the reactor coolant		
1. Dilution during startup	Power range - low setpoint reactor trip due to dilution	0
	Shutdown margin lost (if dilution continues after trip)	2010
2. Dilution during full-power operation		
a. Automatic reactor control	Operation receives low-low rod insertion limit alarm due to dilution	0
	Shutdown margin lost	3700
b. Manual reactor	Reactor trip on $OT\Delta T$ due to dilution	0
	Shutdown margin is lost (if dilution continues after trip)	1860

Table 15.4.1-1 (Sheet 3 of 3)

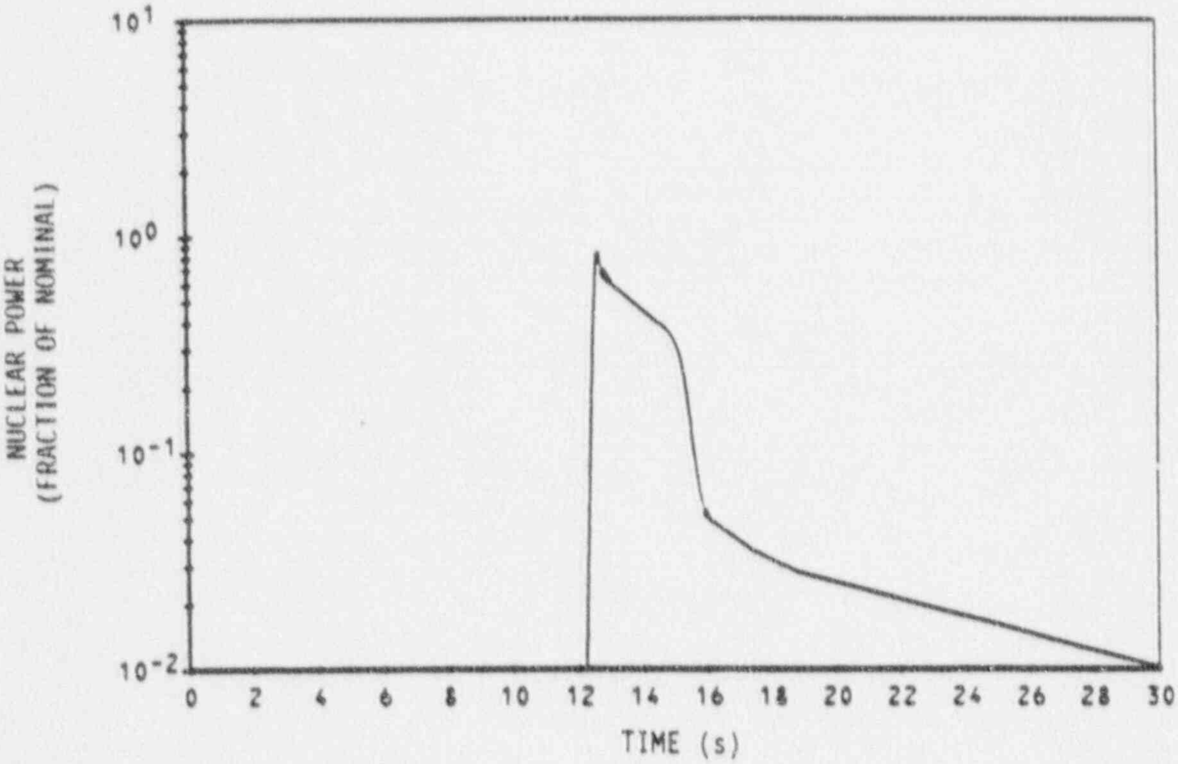
Time Sequence of Events for Incidents Which Result in Reactivity and Power Distribution Anomalies

Accident	Event	Time Delays (s)
RCCA ejection accident		
1. End of life, zero power	Initiation of rod ejection	0.0
	Power range high neutron flux low setpoint reached	0.22
	Peak nuclear power occurs	0.24
	Rods begin to fall into core	0.72
	Peak clad average temperature occurs	1.79
	Peak heat flux occurs	1.79
	Peak fuel average temperature occurs	1.98
2. Beginning of life, full power	Initiation of rod ejection	0.0
	Power range high neutron flux high setpoint reached	0.05
	Peak nuclear power occurs	0.13
	Rods begin to fall into core	0.55
	Peak fuel average temperature occurs	2.44
	Peak clad average temperature occurs	2.52
	Peak heat flux occurs	2.53
Steamline break with coincidental rod withdrawal at power		
	Steamline ruptures, RCCA bank begins to withdraw	0.0
	OPΔT reactor trip setpoint reached	8.8
	Rods begin to fall	10.8
	Minimum DNBR occurs	11.6

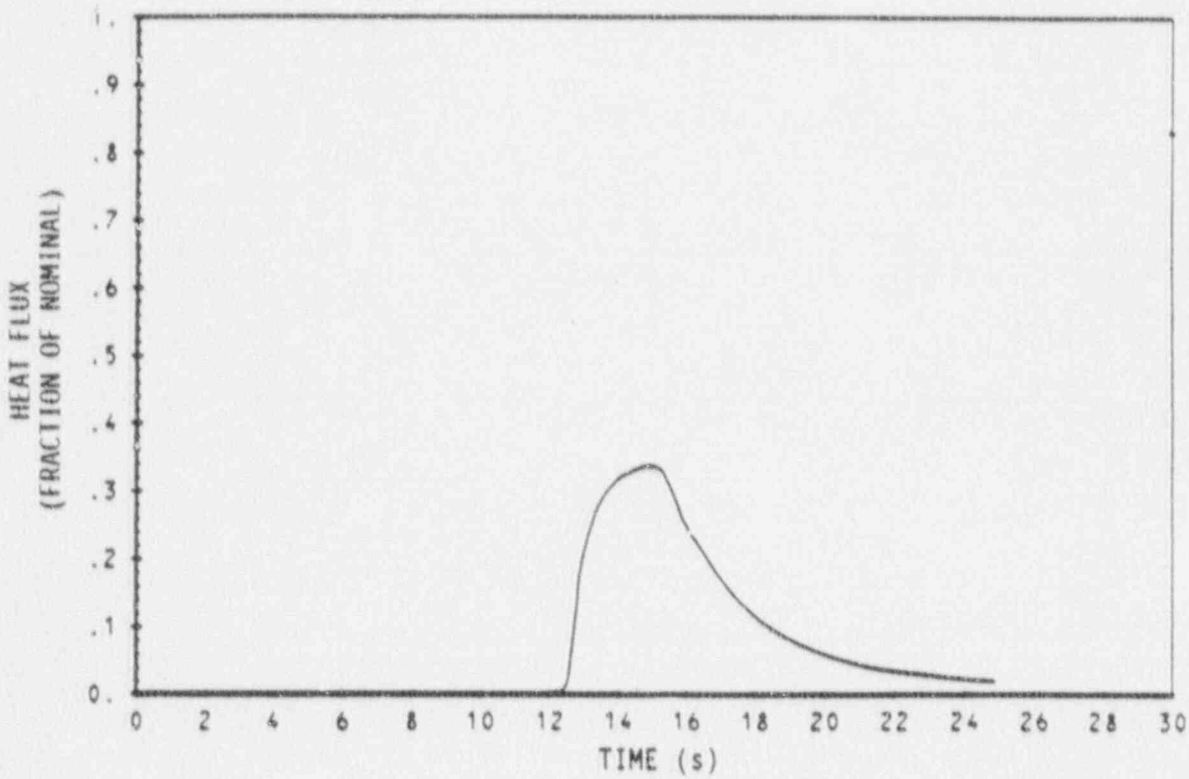
Table 15.4.8-1

Parameters Used in the Analysis of the Rod Cluster Control Assembly Ejection Accident

Time in Life	HZP Beginning	HFP Beginning	HZP End	HFP End
Power level (%)	0	102	0	102
Ejected rod worth (% ΔK)	0.75	0.24	0.84	0.25
Delayed neutron fraction (%)	0.54	0.57	0.46	0.46
Doppler feedback reactivity weighting	1.744	1.30	3.55	1.30
Trip reactivity (% ΔK)	2.0	4.0	2.0	4.0
F_Q before rod ejection		2.55		2.55
F_Q after rod ejection	11.0	5.5	26.0	6.0
Number of operational pumps	2	4	2	4
Maximum fuel pellet average temperature at the hot spot (°F)	3425	4091	3412	3901
Maximum fuel center temperature at the hot spot (°F)	3985	>4900	3891	>4800
Maximum fuel stored energy at the hot spot (cal/g)	144.9	179.2	144.2	169.2
Percent of fuel melt at the hot spot	0	<10	0	<10



Georgia Power	VOGTLE ELECTRIC GENERATING PLANT UNIT 1 AND UNIT 2	NEUTRON FLUX TRANSIENT FOR UNCONTROLLED ROD WITHDRAWAL FROM A SUBCRITICAL CONDITION FIGURE 15.4.1-1

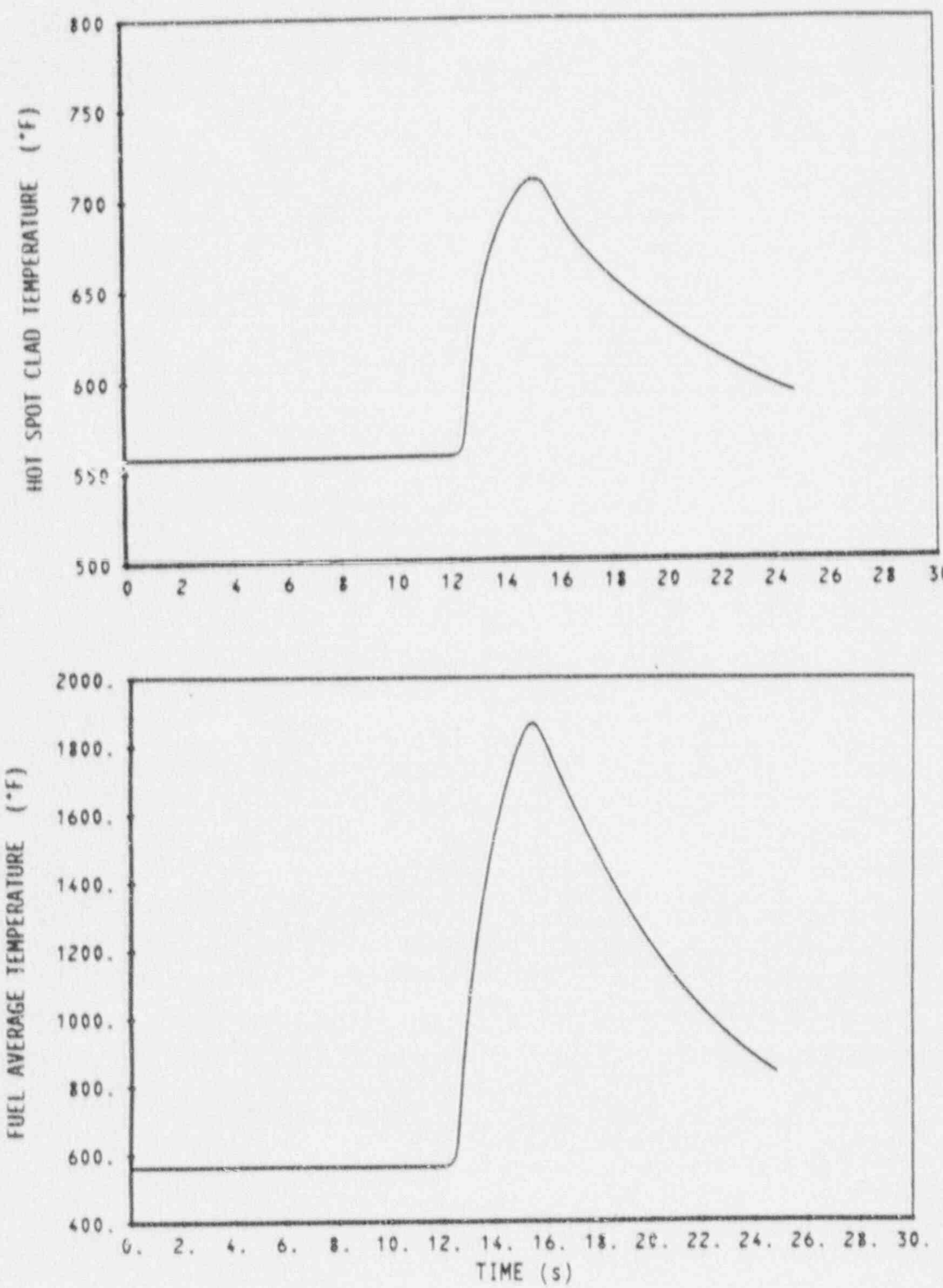


Georgia Power

VOGTLE
ELECTRIC GENERATING PLANT
UNIT 1 AND UNIT 2

THERMAL FLUX TRANSIENT FOR UNCONTROLLED ROD WITHDRAWAL FROM A SUBCRITICAL CONDITION

FIGURE 15.4.1-2

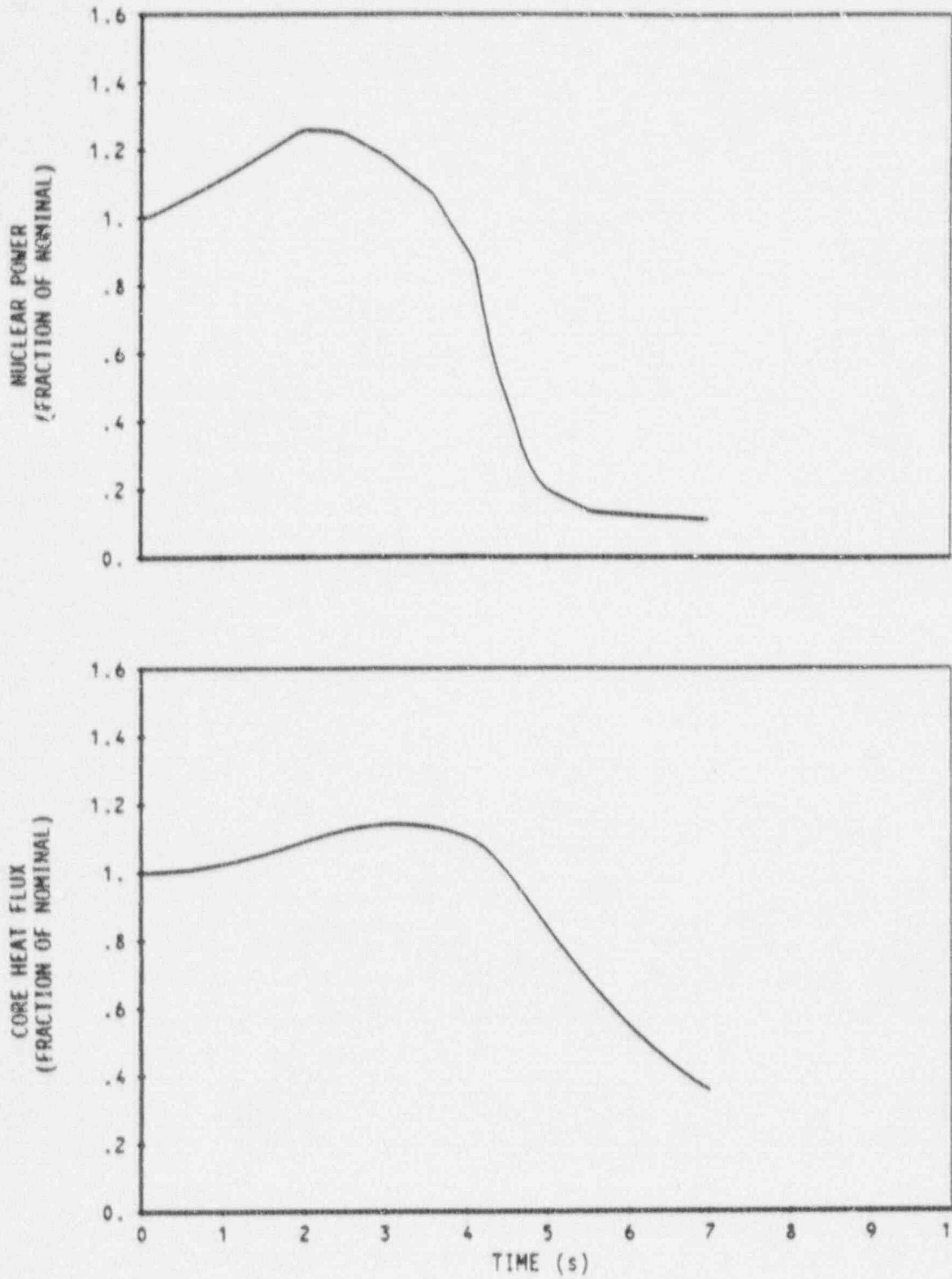


Georgia Power

VOGTLE
ELECTRIC GENERATING PLANT
UNIT 1 AND UNIT 2

FUEL AND CLAD TEMPERATURE FOR UNCONTROLLED ROD WITHDRAWAL FROM A SUBCRITICAL CONDITION

FIGURE 15.4.1-3

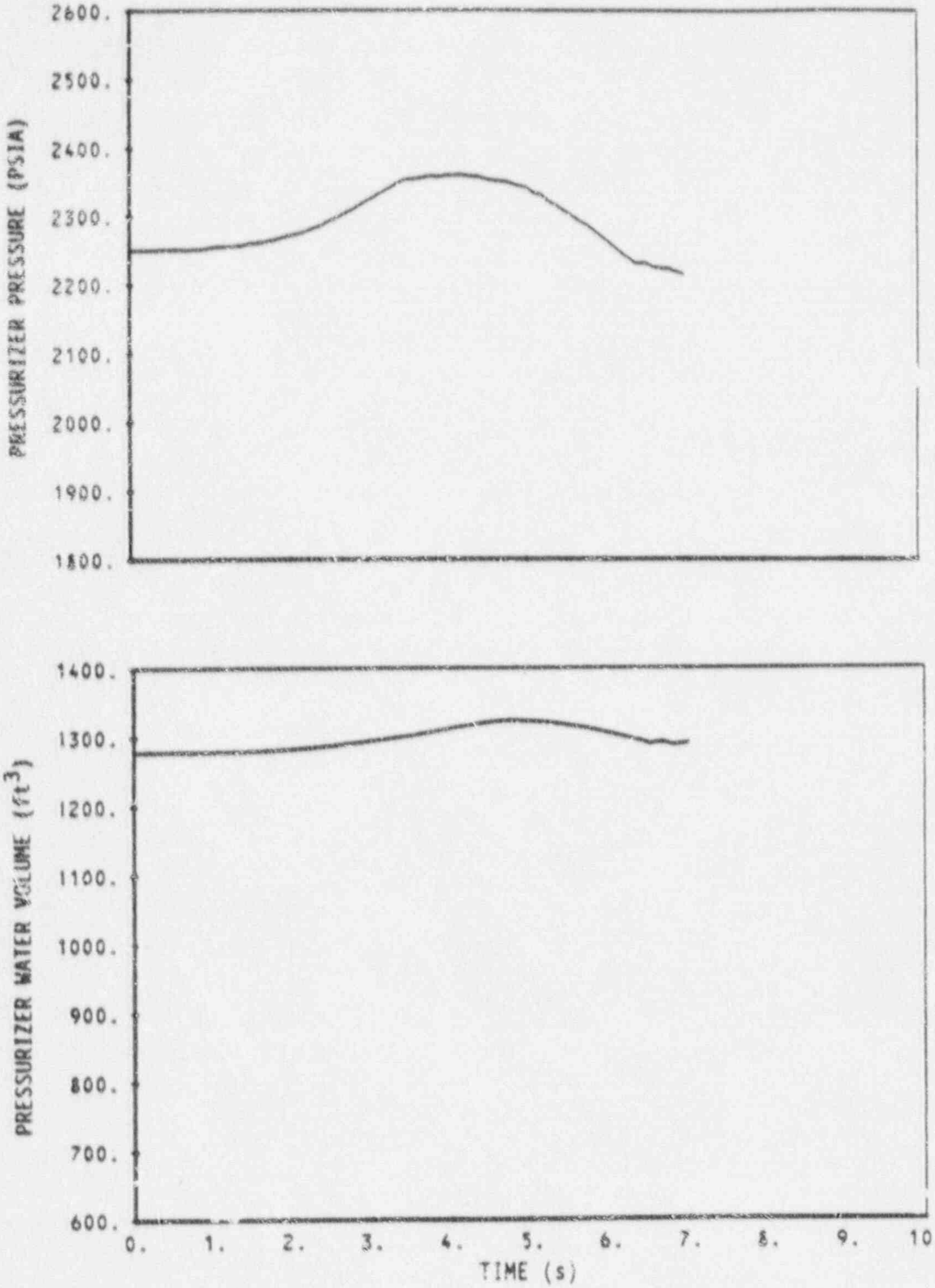


Georgia Power

VOGTLE
ELECTRIC GENERATING PLANT
UNIT 1 AND UNIT 2

UNCONTROLLED RCCA BANK WITHDRAWAL FROM FULL POWER WITH MINIMUM REACTIVITY FEEDBACK (80 PCM/S) WITHDRAWAL RATE)

FIGURE 15.4.2-1

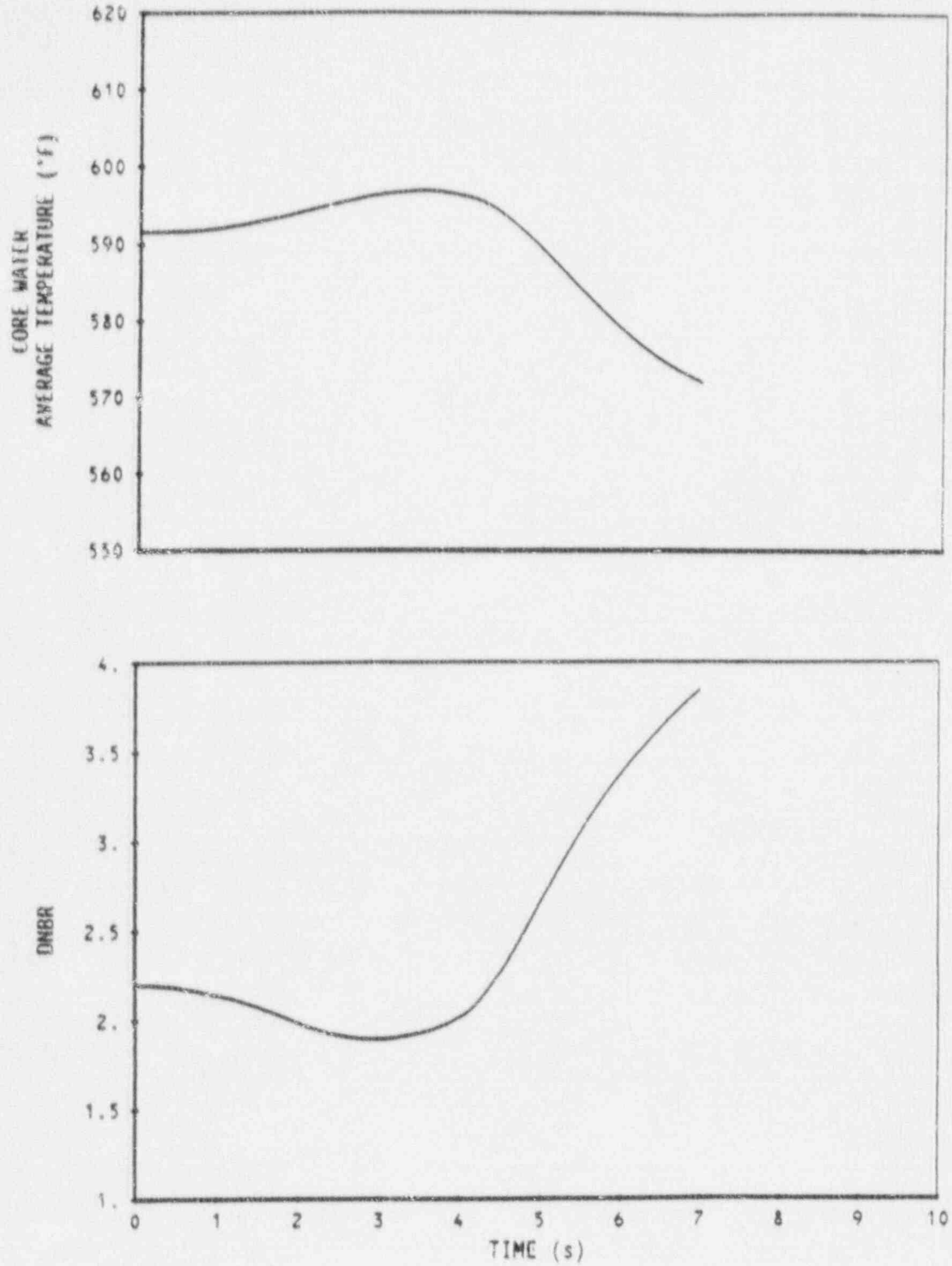


Georgia Power

VOGTLE
ELECTRIC GENERATING PLANT
UNIT 1 AND UNIT 2

UNCONTROLLED RCCA BANK WITHDRAWAL FROM FULL POWER WITH MINIMUM REACTIVITY FEEDBACK (80 PCM/S WITHDRAWAL RATE)

FIGURE 15.4.2-2

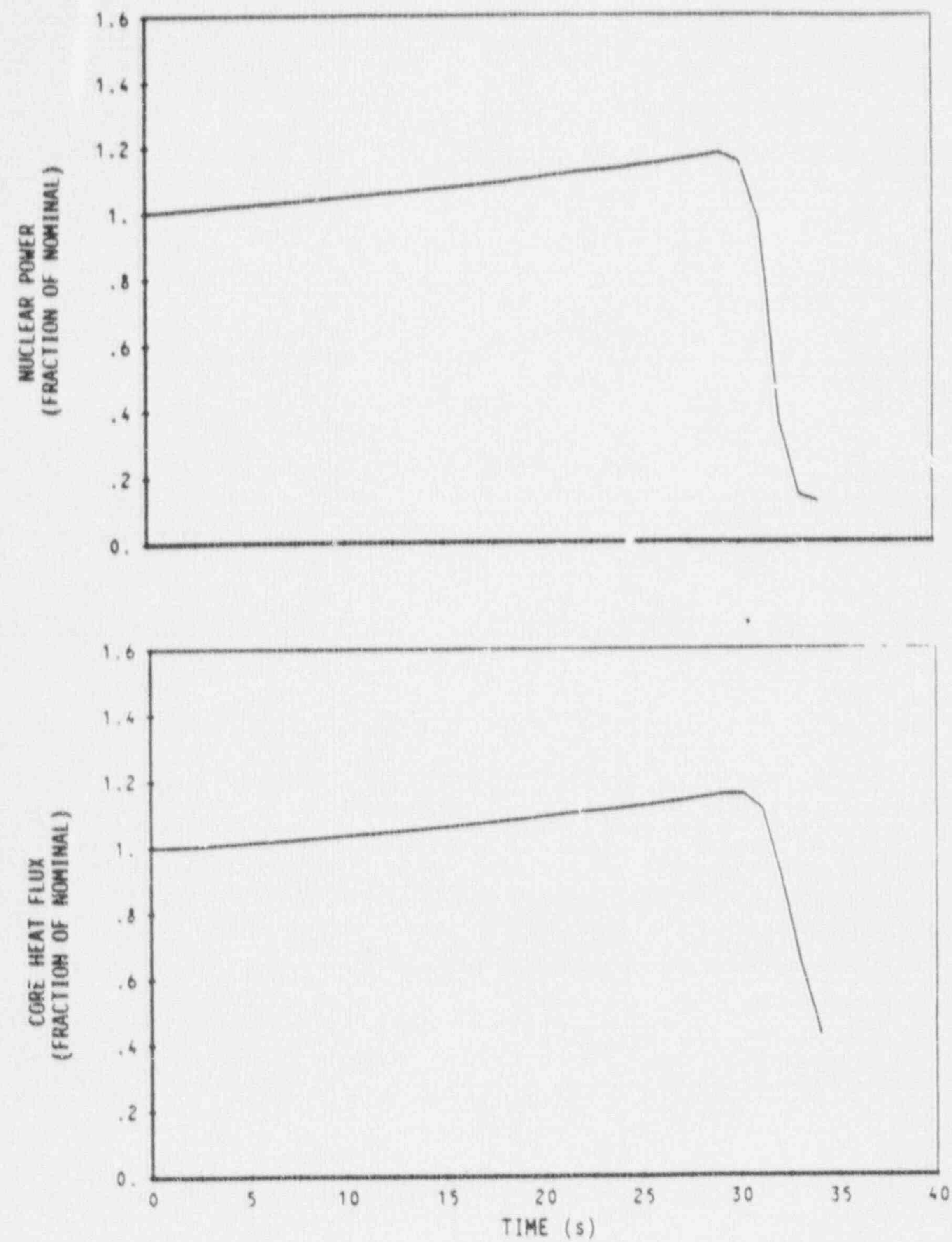


Georgia Power

VOGTLE
ELECTRIC GENERATING PLANT
UNIT 1 AND UNIT 2

UNCONTROLLED RCCA BANK WITHDRAWAL FROM FULL POWER WITH MINIMUM REACTIVITY FEEDBACK (80 PCM/S WITHDRAWAL RATE)

FIGURE 15.4.2-3

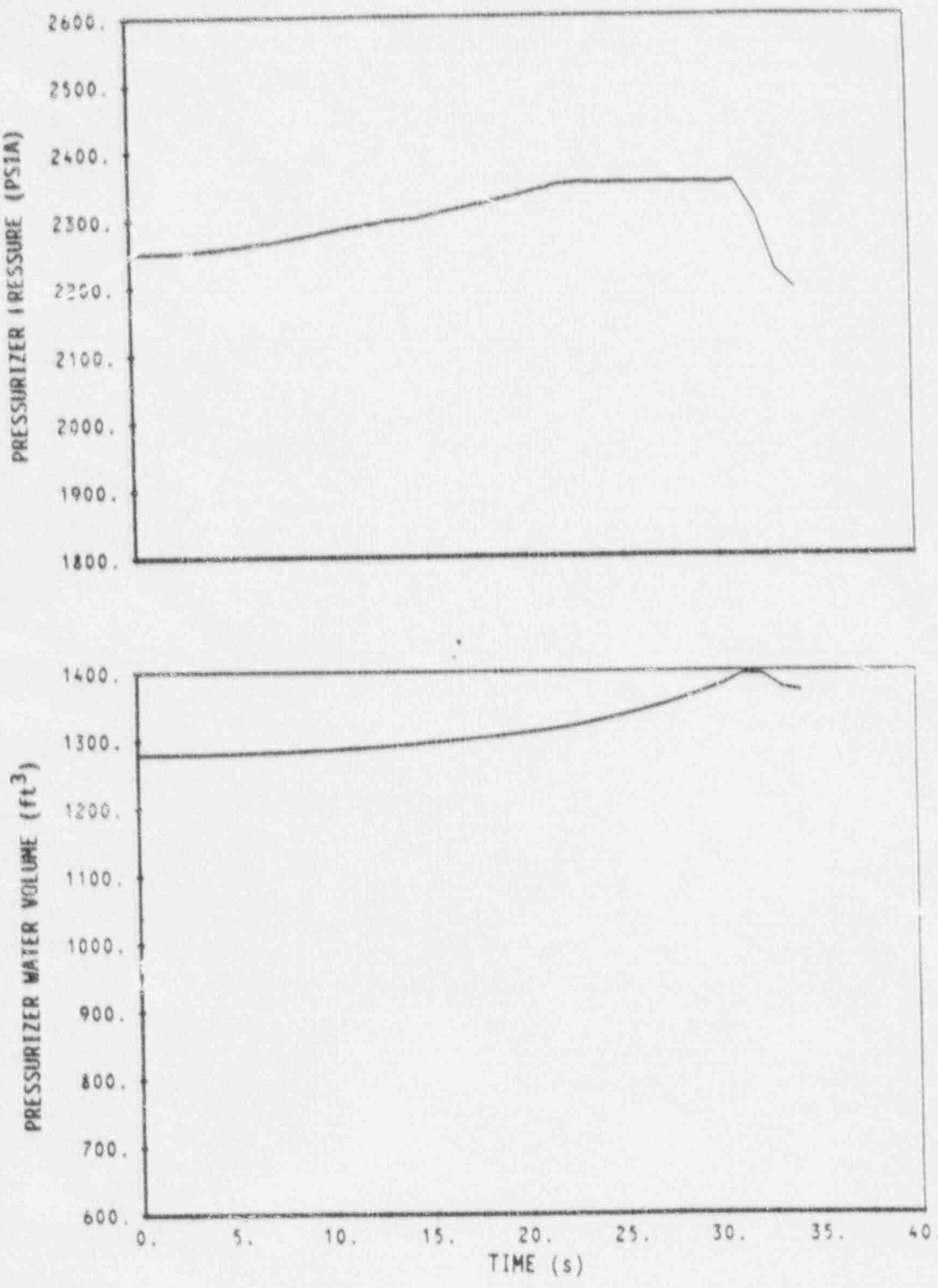


Georgia Power

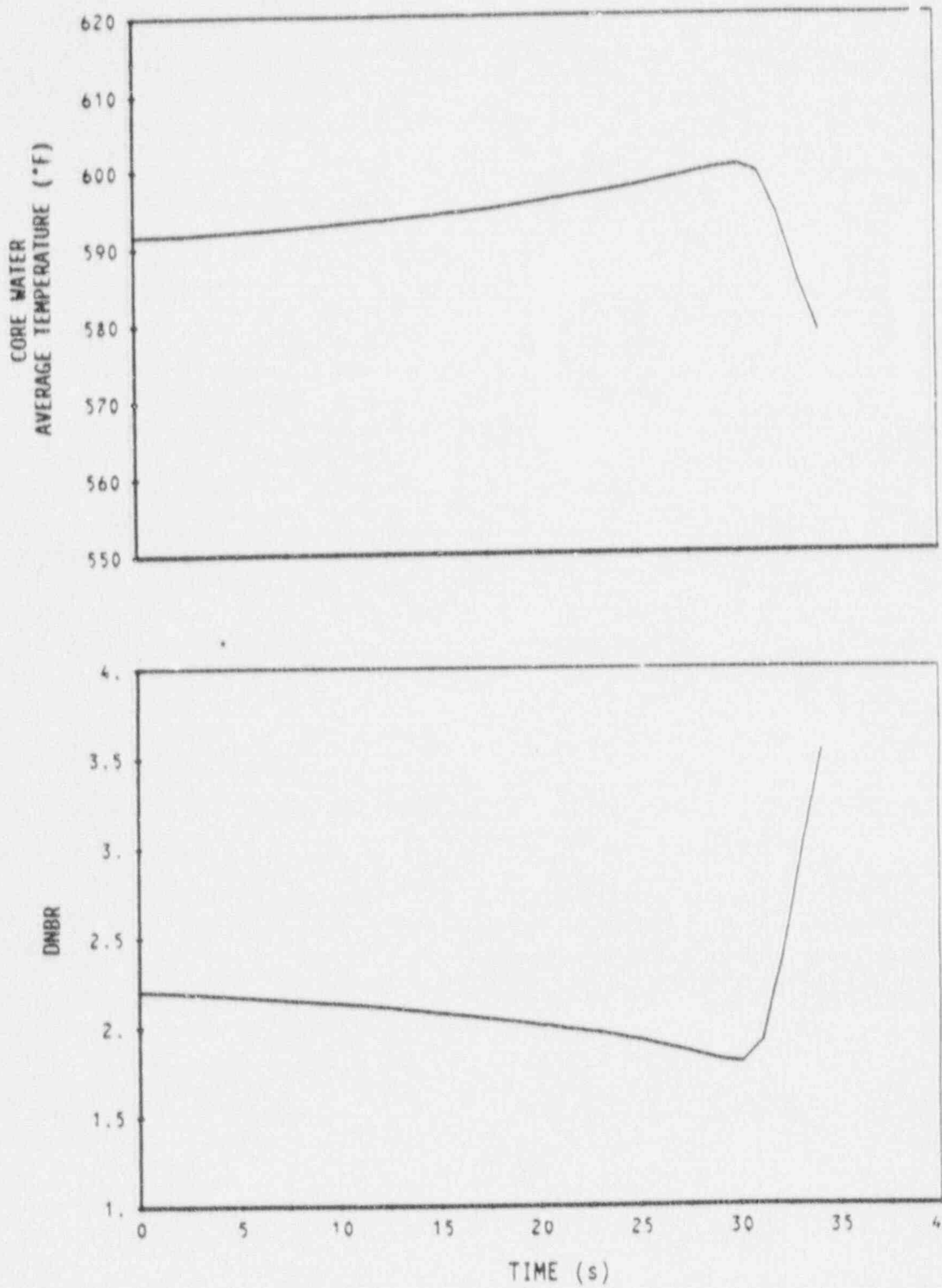
VOGTLE
ELECTRIC GENERATING PLANT
UNIT 1 AND UNIT 2

UNCONTROLLED RCCA BANK WITHDRAWAL FROM FULL POWER WITH MINIMUM REACTIVITY FEEDBACK (3 PCM/S WITHDRAWAL RATE)

FIGURE 15.4.2-4



Georgia Power	VOGTLE ELECTRIC GENERATING PLANT UNIT 1 AND UNIT 2	UNCONTROLLED RCCA BANK WITHDRAWAL FROM FULL POWER WITH MINIMUM REACTIVITY FEEDBACK (3 PCM/S WITHDRAWAL RATE) FIGURE 15.4.2-5

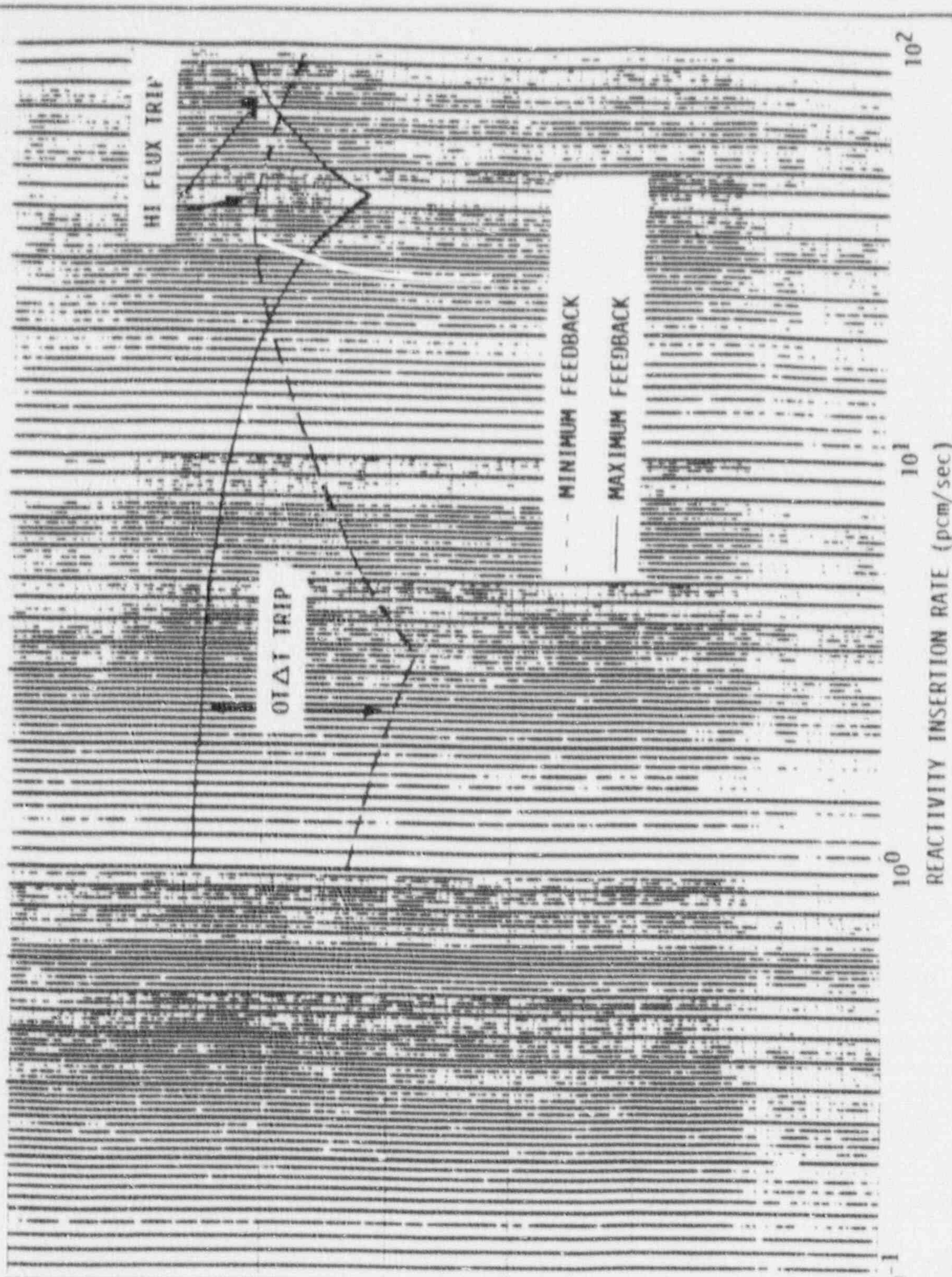


Georgia Power

VOGTLE
ELECTRIC GENERATING PLANT
UNIT 1 AND UNIT 2

UNCONTROLLED RCCA BANK WITHDRAWAL FROM FULL POWER WITH MINIMUM REACTIVITY FEEDBACK (3 PCM/S WITHDRAWAL RATE)

FIGURE 15.4.2-6



Georgia Power

VOGTLE
ELECTRIC GENERATING PLANT
UNIT 1 AND UNIT 2

MINIMUM DNBR VS. REACTIVITY INSERTION RATE FOR ROD WITHDRAWAL FROM 100% POWER

FIGURE 15.4.2-7

Georgia Power

VOGTLE
ELECTRIC GENERATING PLANT
UNIT 1 AND UNIT 2

MINIMUM DNBR VS. REACTIVITY INSERTION RATE FOR
ROD WITHDRAWAL FROM 60% POWER

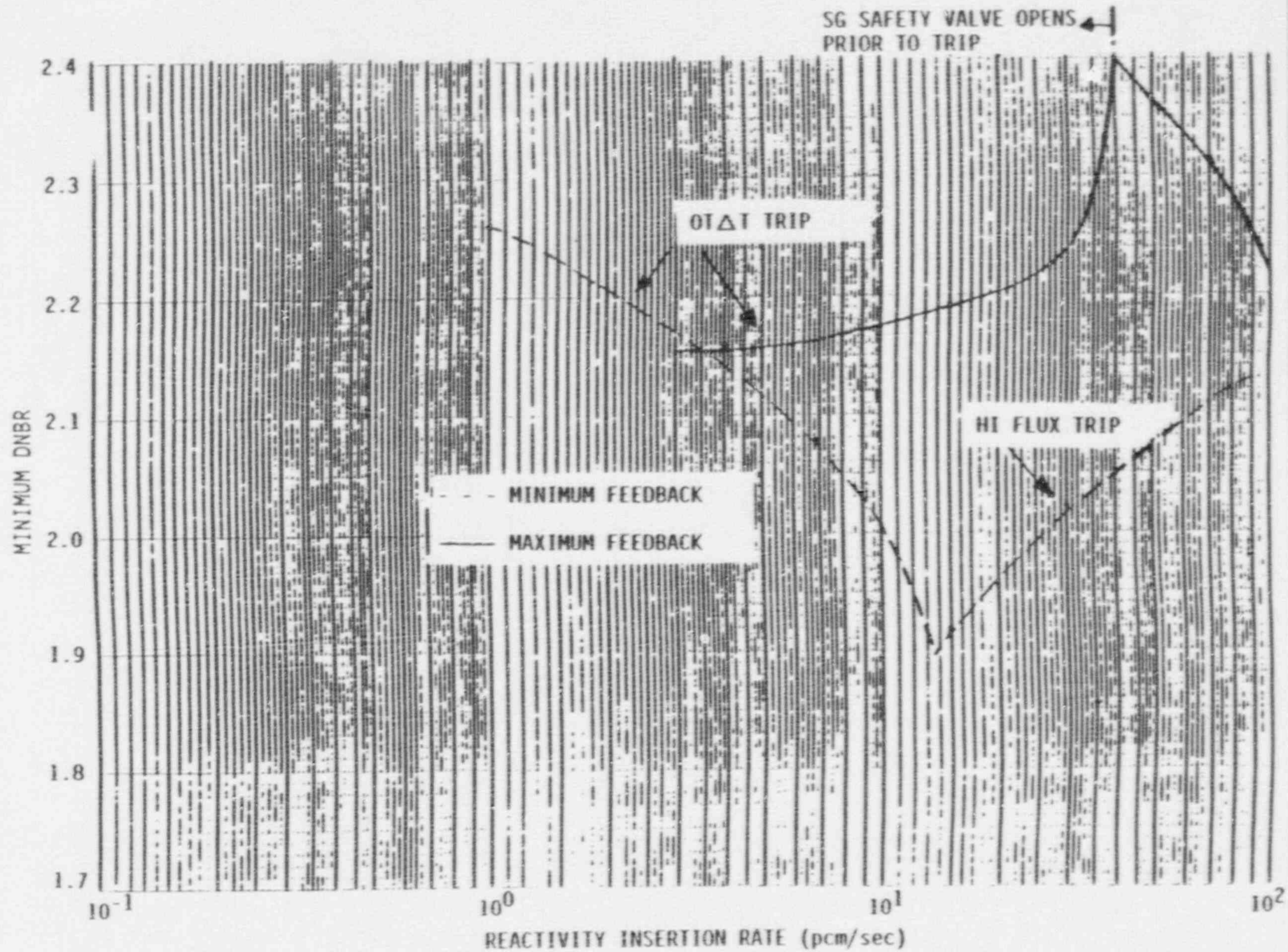
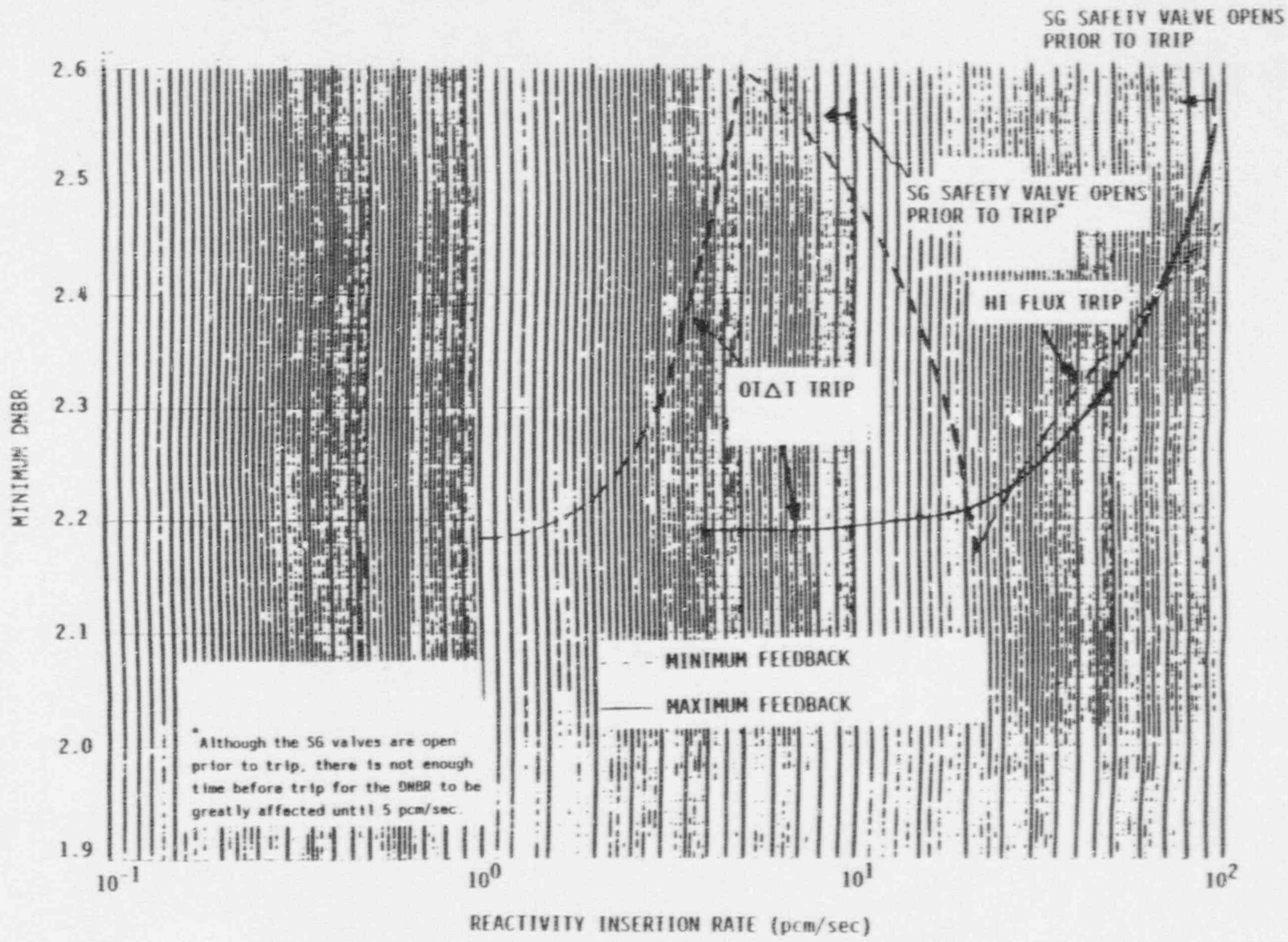


FIGURE 15.4.2-8

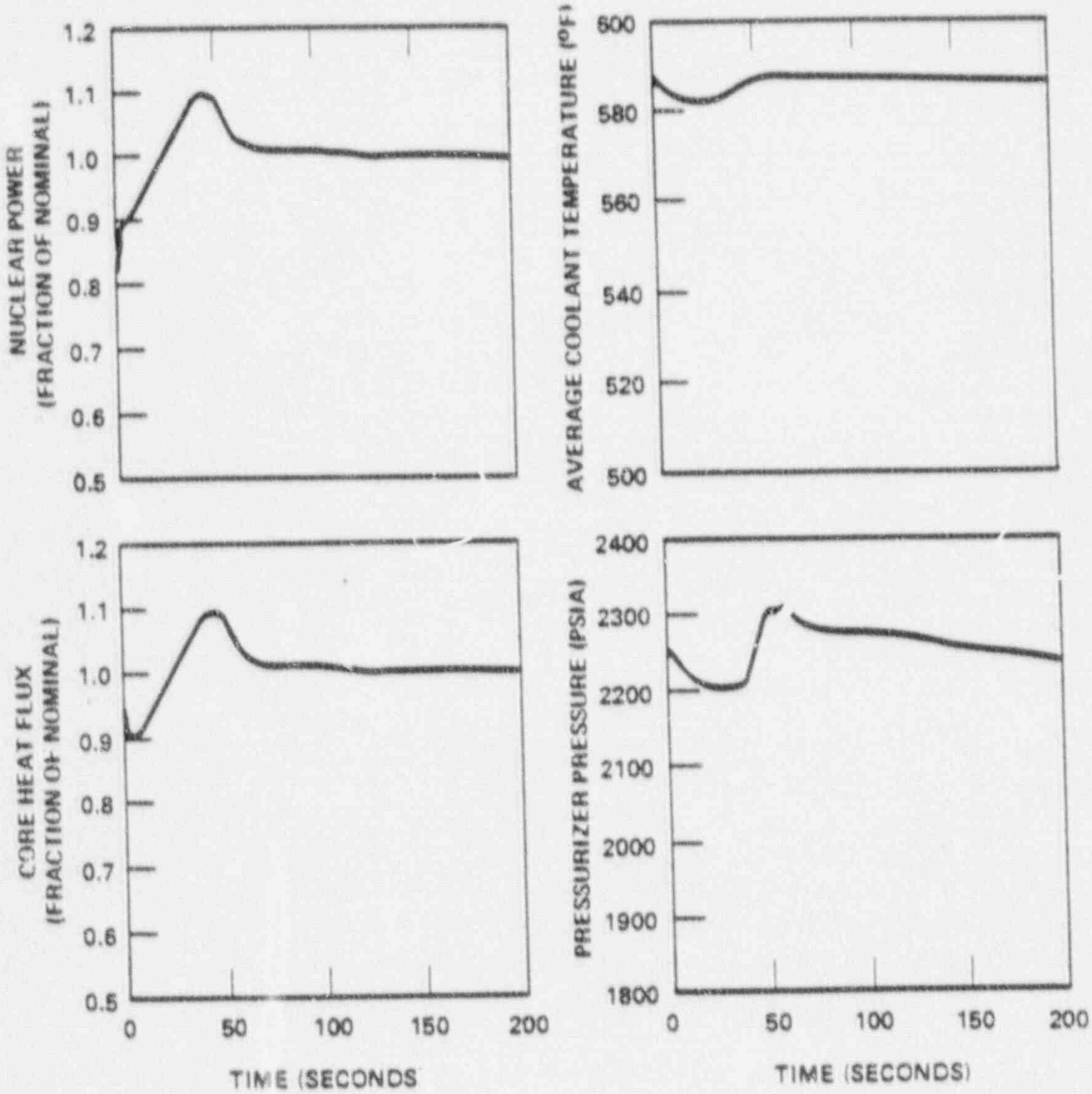


Georgia Power

VOGTLE
ELECTRIC GENERATING PLANT
UNIT 1 AND UNIT 2

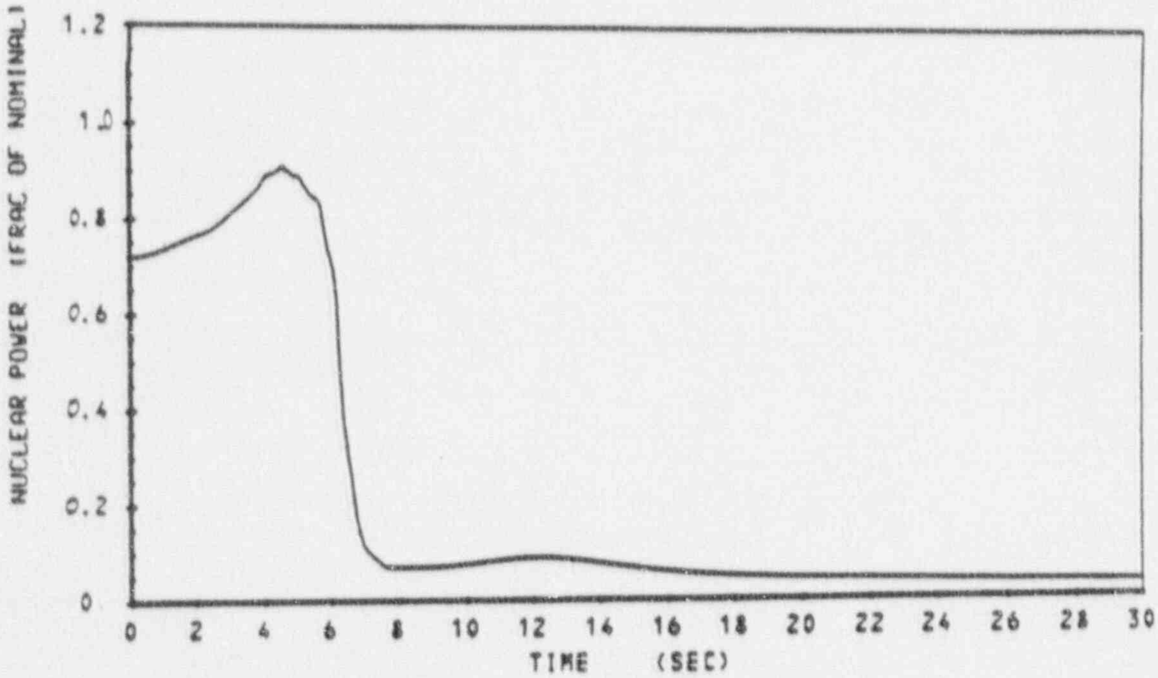
MINIMUM DNBR VS. REACTIVITY INSERTION RATE FOR ROD WITHDRAWAL FROM 10% POWER

FIGURE 15.4.2-9



Georgia Power	VOGTLE ELECTRIC GENERATING PLANT UNIT 1 AND UNIT 2	NUCLEAR POWER TRANSIENT AND CORE HEAT FLUX TRANSIENT FOR DROPPED RCCA FIGURE 15.4.3-1

433-9

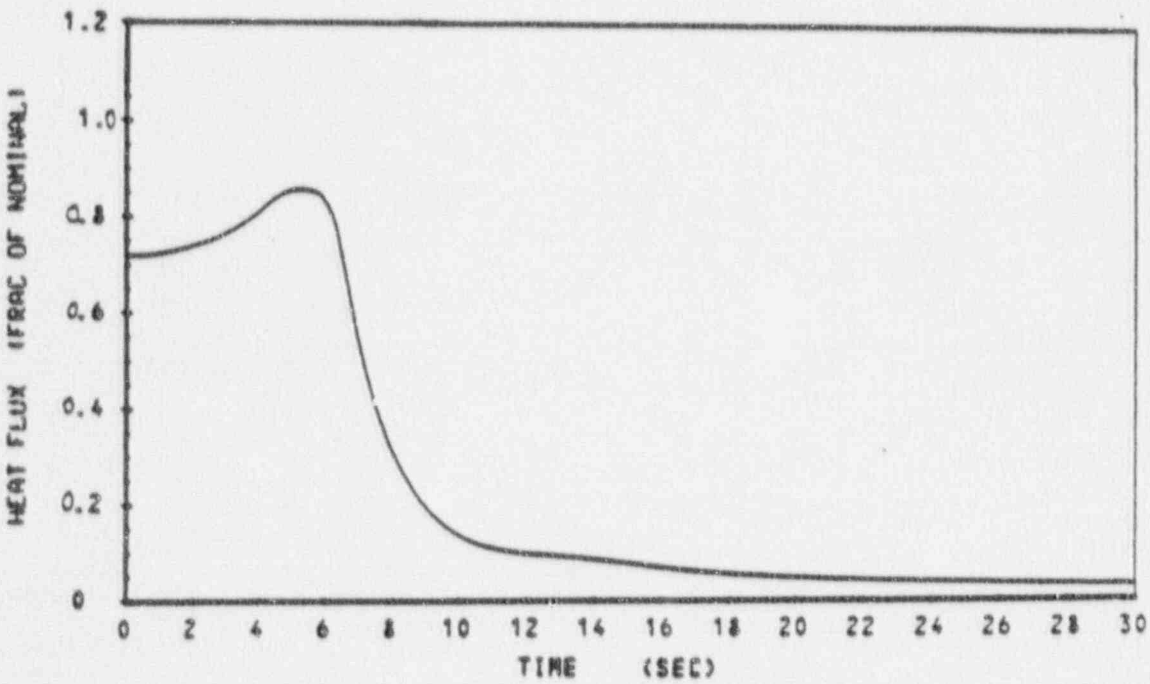


Georgia Power

VOGTLE
ELECTRIC GENERATING PLANT
UNIT 1 AND UNIT 2

IMPROPER STARTUP OF AN INACTIVE REACTOR COOLANT PUMP

FIGURE 15.4.4-1



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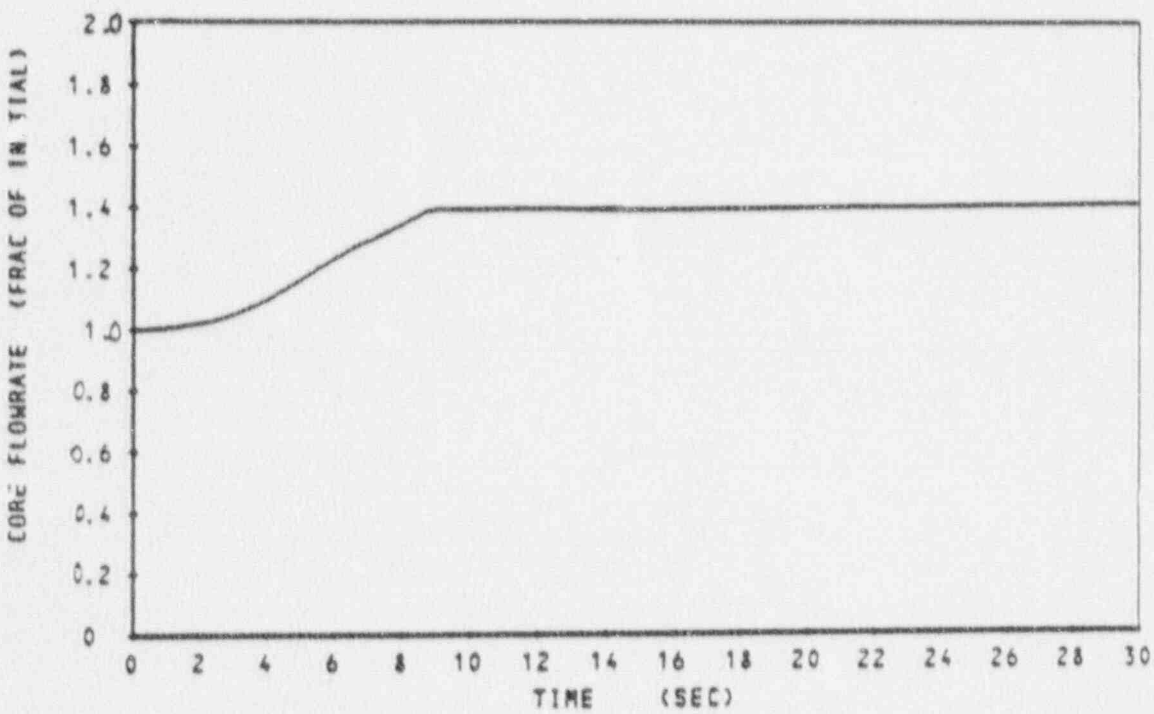
VOGTLE
ELECTRIC GENERATING PLANT
UNIT 1 AND UNIT 2

IMPROPER STARTUP OF AN INACTIVE REACTOR COOLANT PUMP

FIGURE ~~15~~.4.4-2

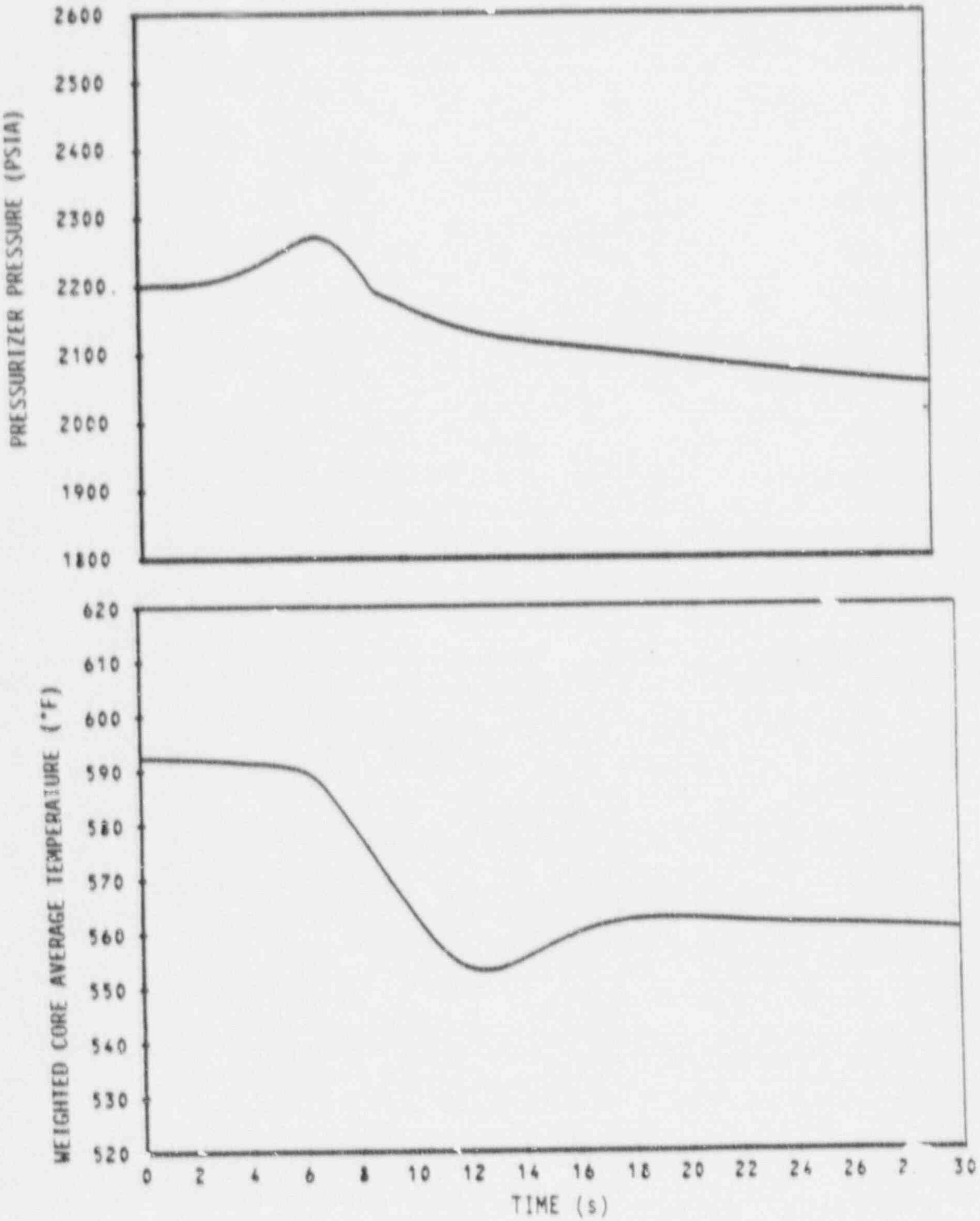


Georgia Power

VOGTLE
ELECTRIC GENERATING PLANT
UNIT 1 AND UNIT 2

IMPROPER STARTUP OF AN INACTIVE REACTOR COOLANT PUMP

FIGURE 15.4.4-3

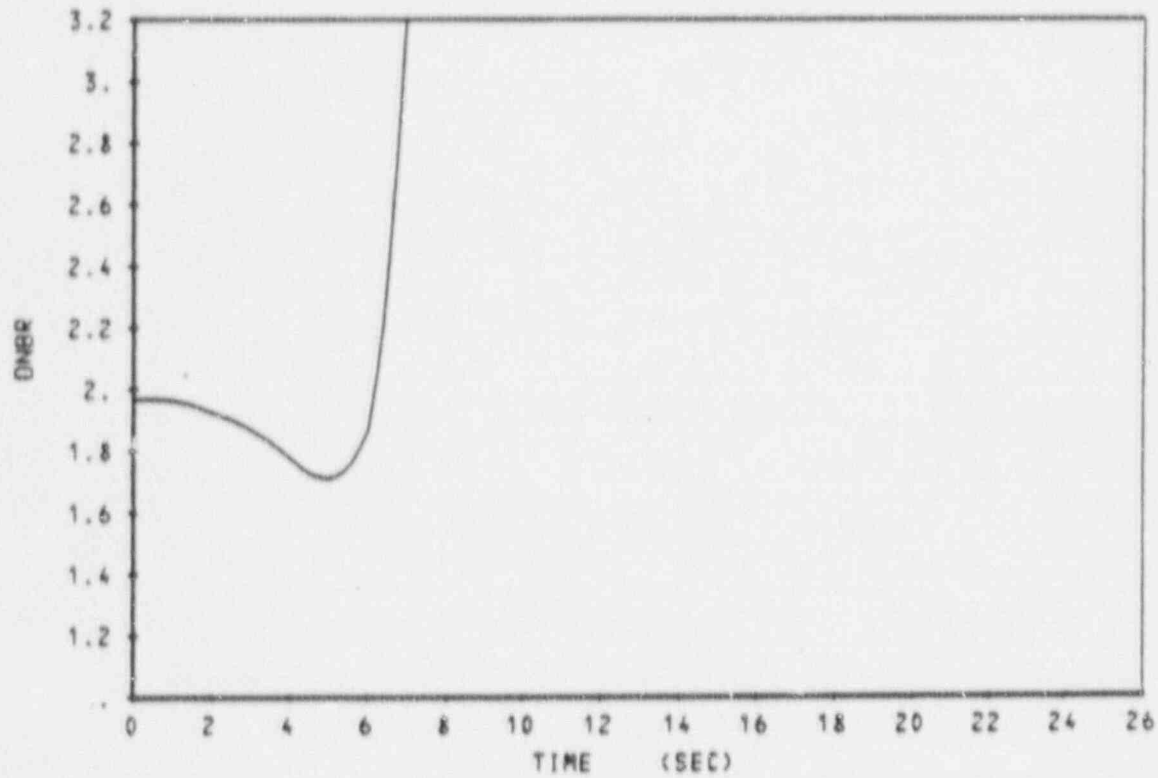


Georgia Power

VOGTLE
ELECTRIC GENERATING PLANT
UNIT 1 AND UNIT 2

IMPROPER STARTUP OF AN INACTIVE REACTOR COOLANT PUMP

FIGURE 15.4.4-4



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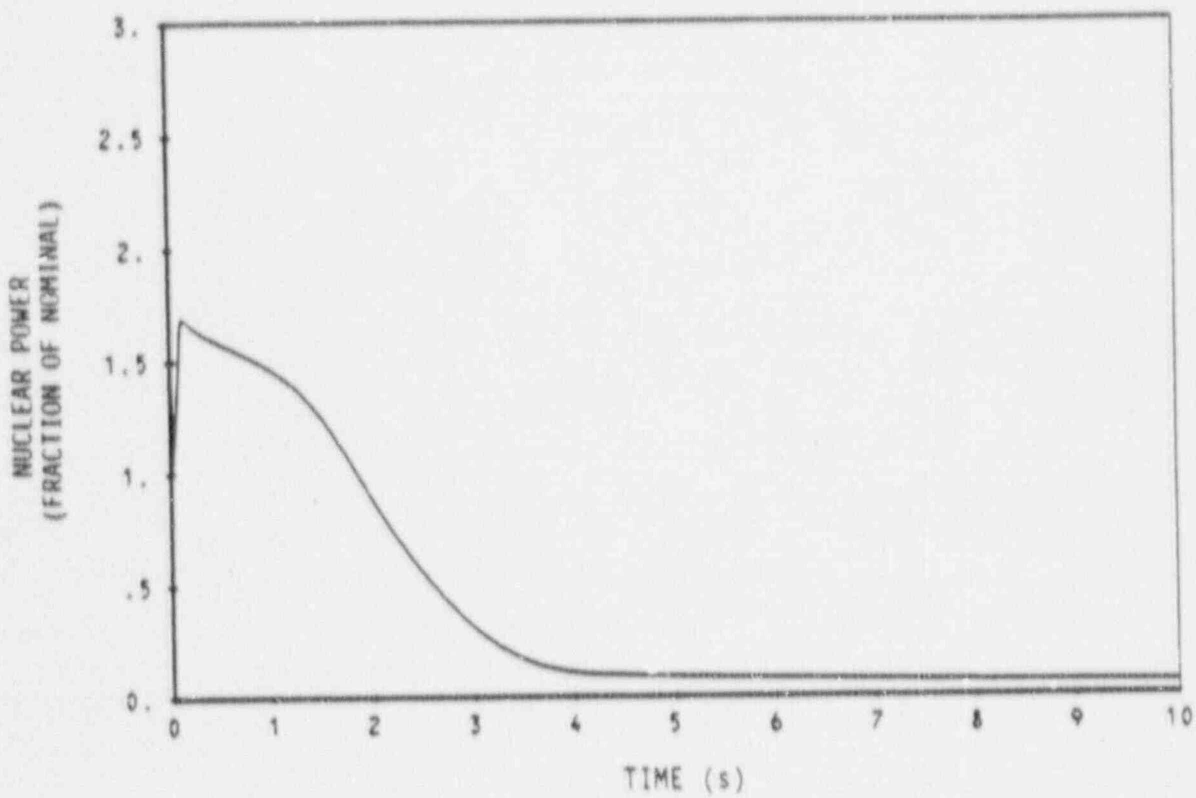
VOGTLE
ELECTRIC GENERATING PLANT
UNIT 1 AND UNIT 2

IMPROPER STARTUP OF AN INACTIVE REACTOR COOLANT PUMP

FIGURE 15.4.4-5

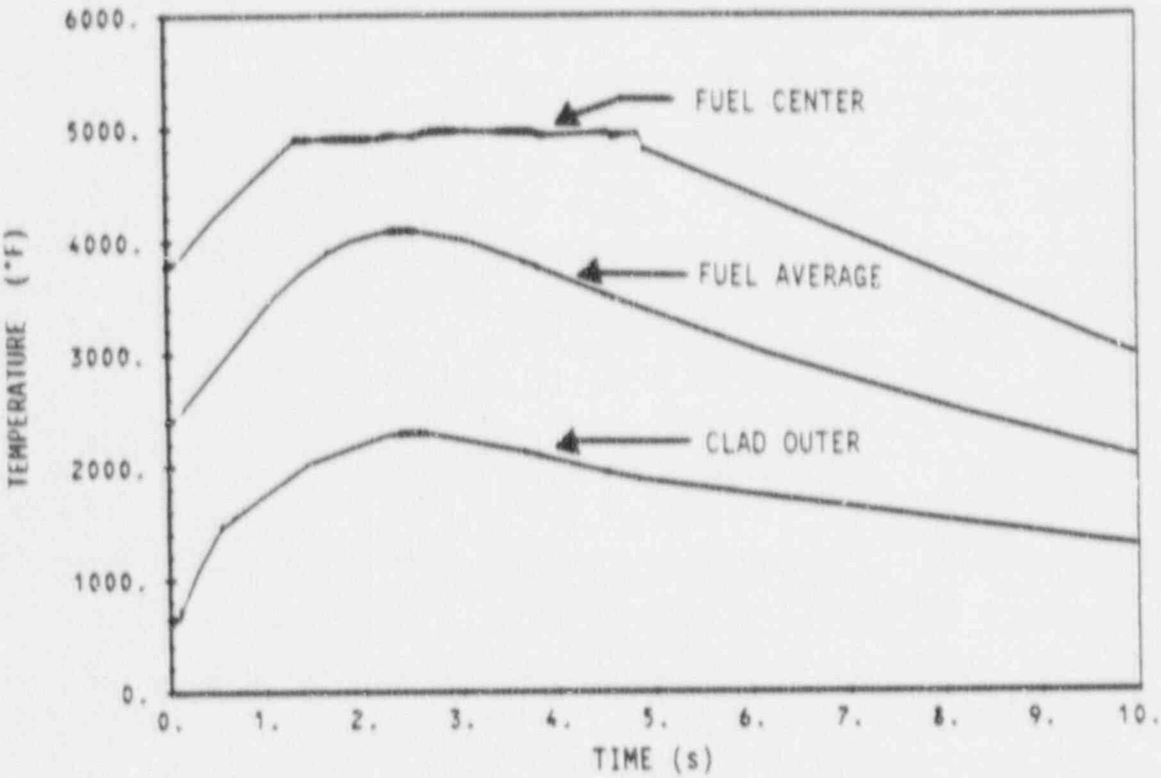


Georgia Power

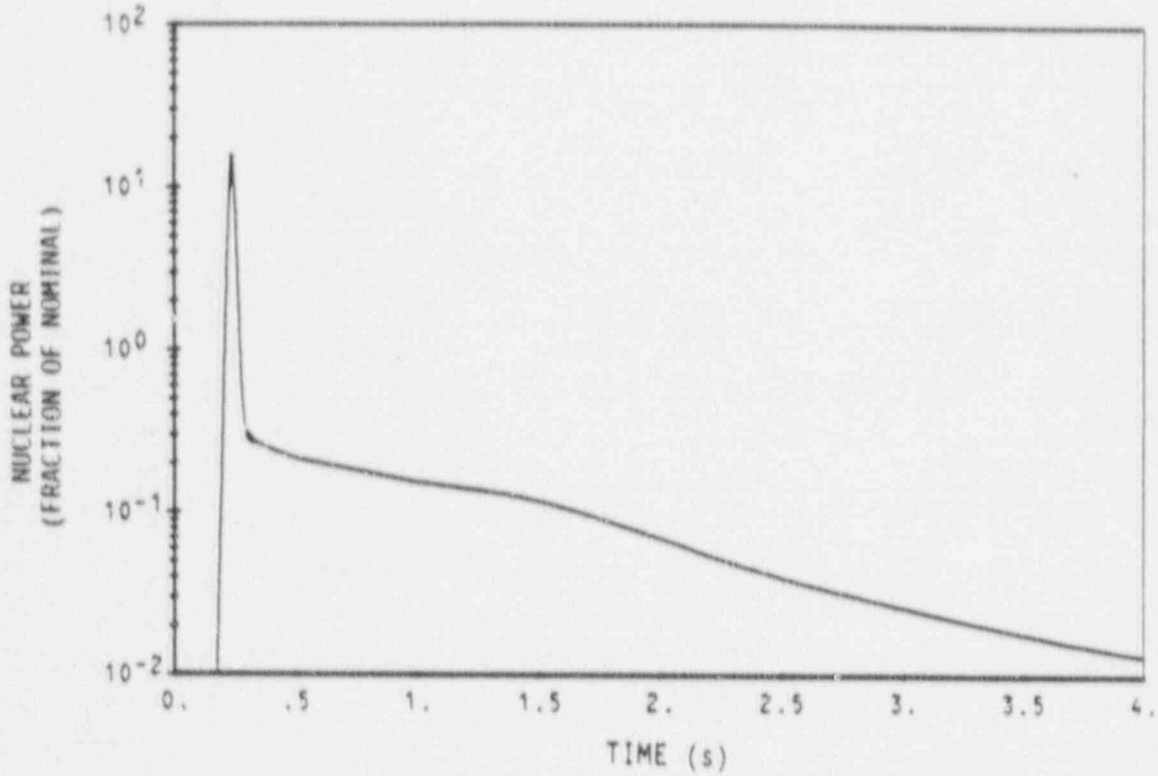
VOGTLE
ELECTRIC GENERATING PLANT
UNIT 1 AND UNIT 2

NUCLEAR POWER TRANSIENT BOL FULL POWER

FIGURE 15.4.8-1



Georgia Power	VOGTLE ELECTRIC GENERATING PLANT UNIT 1 AND UNIT 2	HOT SPOT FUEL & CLAD TEMP VS. TIME BOL FULL POWER FIGURE 15.4.8-2

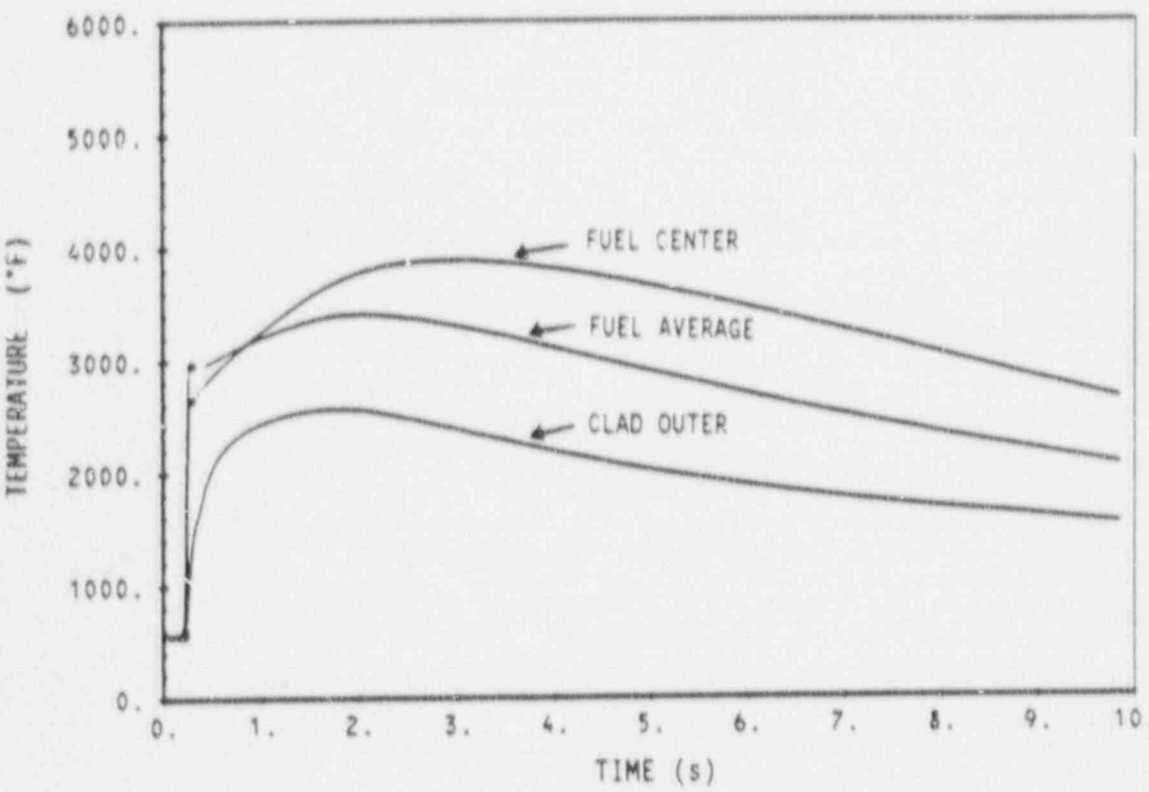


Georgia Power

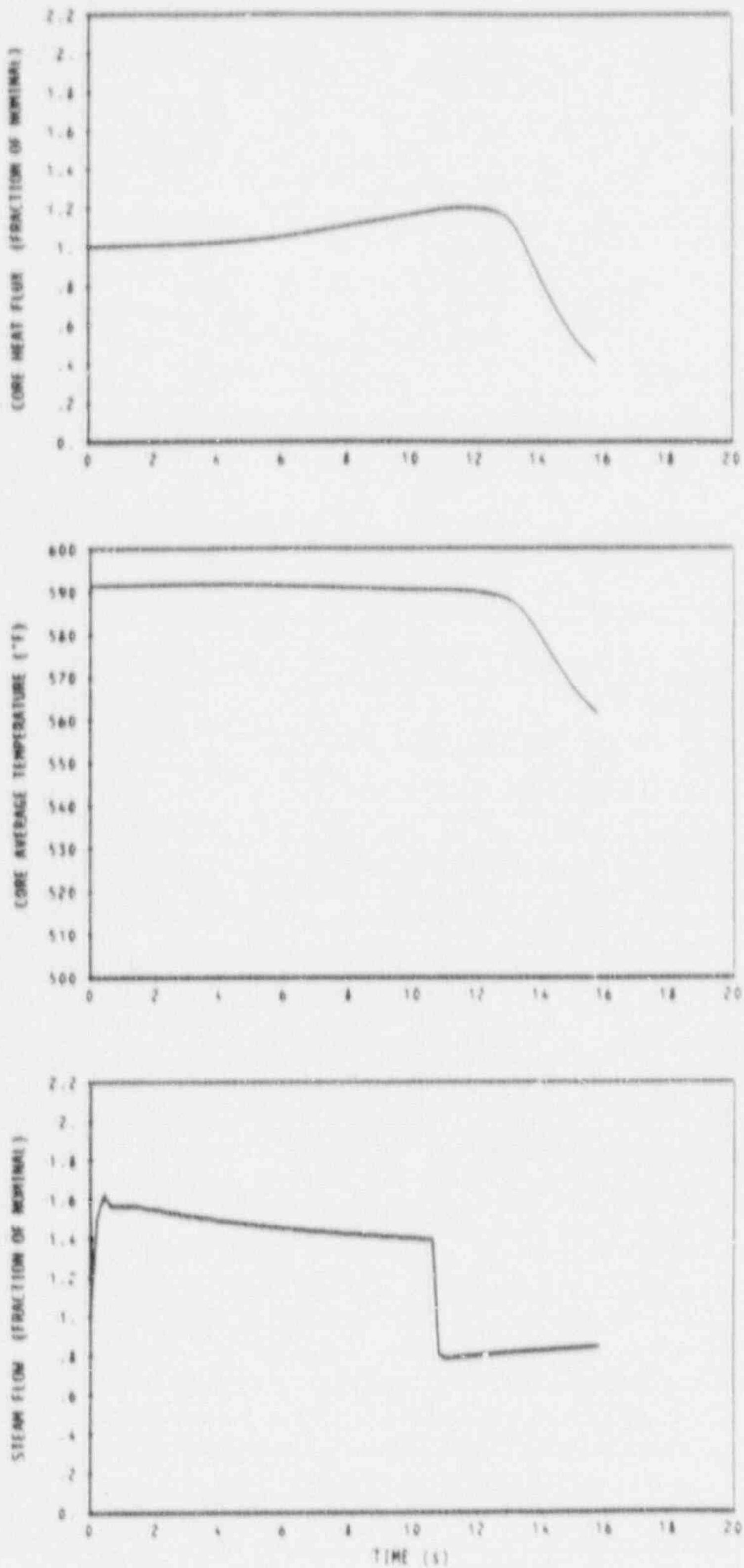
VOGTLE
ELECTRIC GENERATING PLANT
UNIT 1 AND UNIT 2

NUCLEAR POWER TRANSIENT EOL ZERO POWER

FIGURE 15.4.8-3



Georgia Power	VOGTLE ELECTRIC GENERATING PLANT UNIT 1 AND UNIT 2	HOT SPOT FUEL & CLAD TEMP VS. TIME EOL ZERO POWER
		FIGURE 15.4.8-4

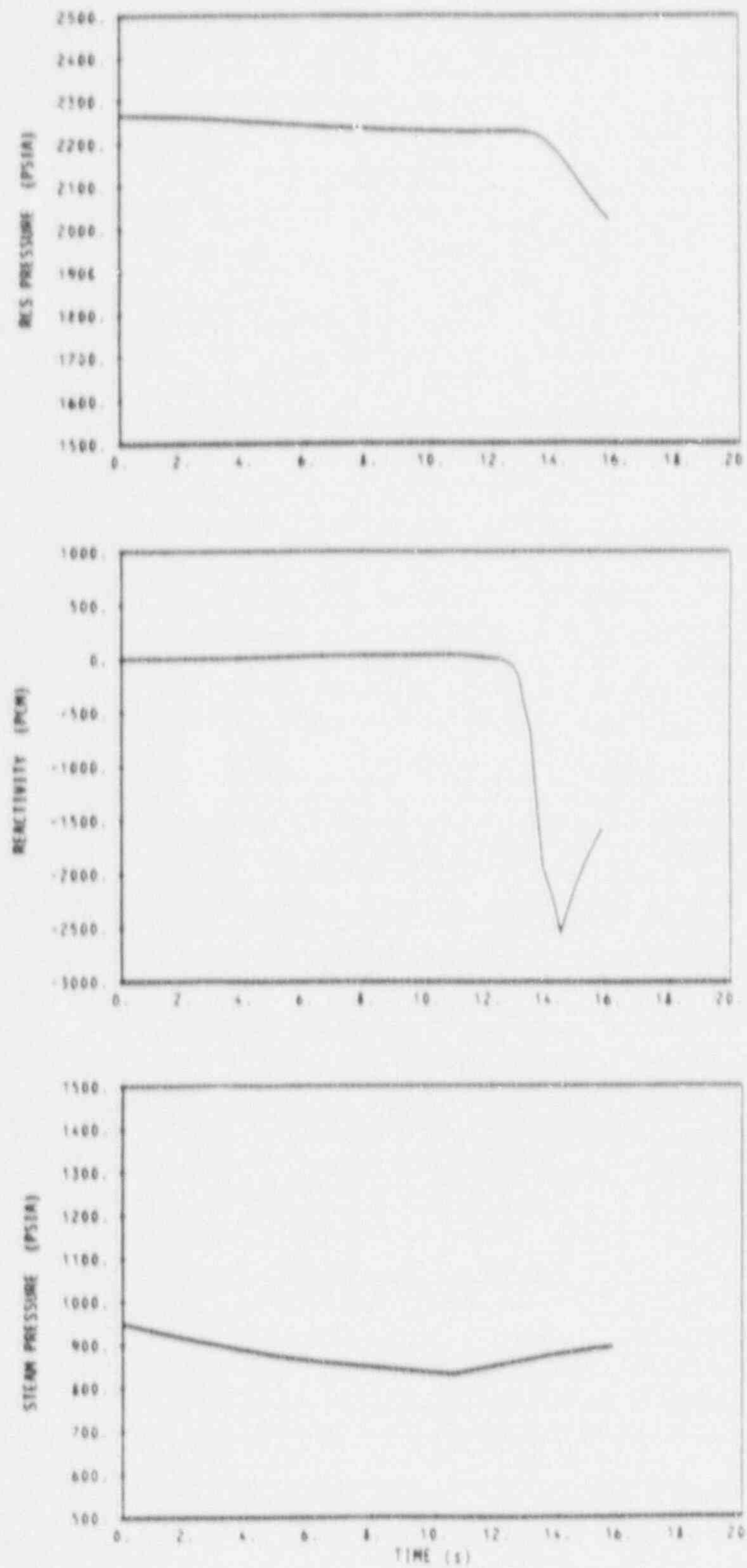


Georgia Power

VOGTLE
ELECTRIC GENERATING PLANT
UNIT 1 AND UNIT 2

STEAMLINE BREAK COINCIDENT WITH CONTROL ROD WITHDRAWAL: CORE HEAT FLUX, CORE AVERAGE TEMPERATURE, AND STEAM FLOW VS. TIME

FIGURE 15.4.9-1

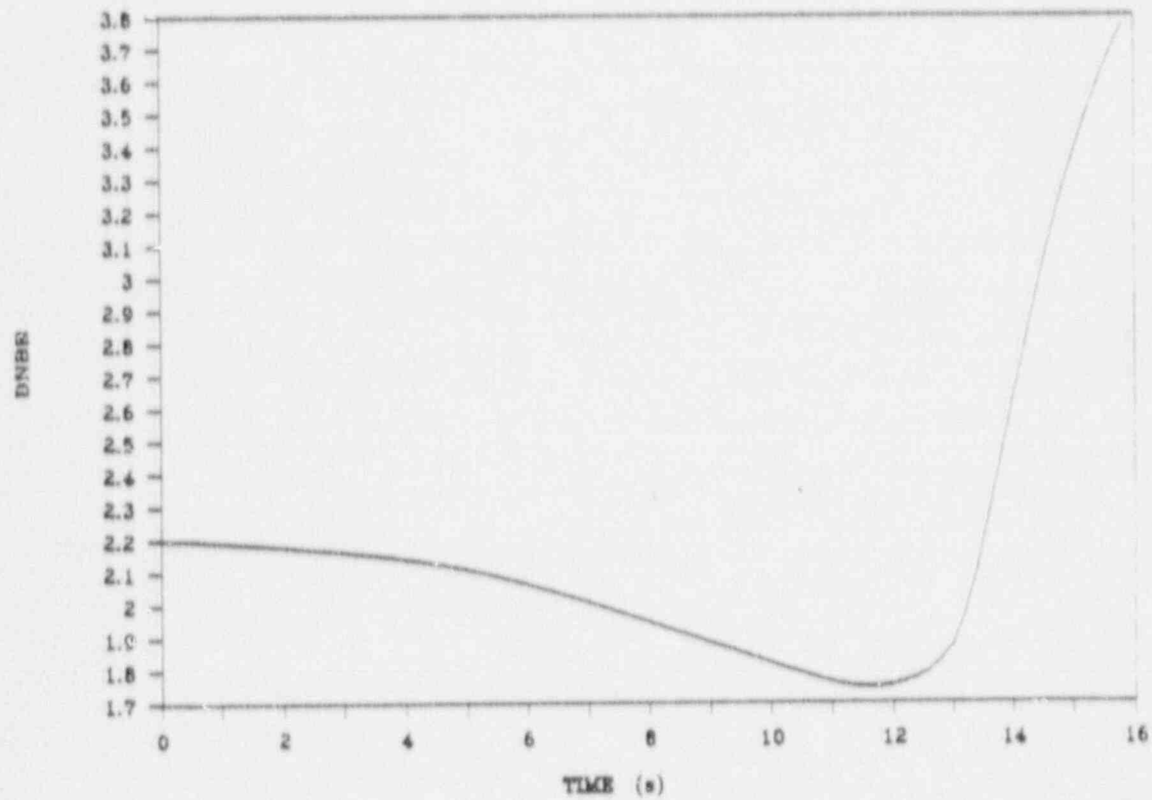


Georgia Power

VOGTLE
ELECTRIC GENERATING PLANT
UNIT 1 AND UNIT 2

STEAMLINE BREAK COINCIDENT WITH CONTROL ROD WITHDRAWAL: RCS PRESSURE, REACTIVITY AND STEAM PRESSURE VS. TIME

FIGURE 15.4.9-2



Georgia Power

VOGTLE
ELECTRIC GENERATING PLANT
UNIT 1 AND UNIT 2

STEAMLINE BREAK COINCIDENT WITH CONTROL ROD WITHDRAWAL: DNBR VS. TIME

FIGURE 15.4.9-3

15.5 Increase in Reactor Coolant Inventory

This section discusses the events which could result in an increase in the RCS inventory.

15.5.1 Inadvertent Operation of the Emergency Core Cooling System During Power Operation

15.5.1.1 Introduction

Operator error or a false electrical actuation signal can cause spurious ECCS operation at power. A spurious signal may originate from any of the safety injection (SI) actuation channels as described in FSAR Section 7.3.

The suction of the coolant charging pumps diverts from the volume control tank to the refueling water storage tank following the actuation signal. The valves isolating the charging pumps from the injection header then automatically open. The charging pumps force the borated water from the RWST through the header, the injection line, and into the cold leg of each loop. The SI pumps also start automatically, but they do not provide any flow when the RCS is at normal pressure. The passive accumulator tank safety injection system (SIS) and the low head system also do not provide any flow at normal RCS pressure.

An SI signal normally results in a reactor trip followed by a turbine trip; however, any single fault that actuates the SIS will not necessarily produce a reactor trip. If an SI signal generates a reactor trip, the operator should determine if the signal is spurious. If the SI signal is determined to be spurious, the operator should terminate SI and maintain the plant in the hot standby condition as determined by appropriate recovery procedures. If repair of the ECCS actuation instrumentation is necessary, future plant operation will be in accordance with the Technical Specifications.

If the reactor protection system does not produce an immediate trip as a result of the spurious SIS, the reactor experiences a negative reactivity excursion due to the injected boron, which causes a decrease in reactor power. The power mismatch causes a drop in T_{avg} and consequent coolant shrinkage. The pressurizer pressure and water level decrease. Load decreases due to the effect of reduced steam pressure on load after the turbine throttle valve is fully open. These effects will lessen until the rods have moved out of the core by using automatic rod control. The transient is eventually terminated by the reactor protection system low pressurizer pressure trip or by manual trip.

Initial operating conditions affect the time to trip. The initial conditions include the core burnup history which affects initial boron concentration, rate of change of boron concentration, and Doppler and moderator coefficients.

Recovery is made in the same manner as described for the case where the SI signal results directly in a reactor trip. The only difference is the lower T_{avg} and pressure associated with the power mismatch during the transient. The time at which reactor trip occurs does not affect this transient. At lower loads, coolant contraction will be slower which will result in a longer time to trip.

This is an ANS Condition II incident.

15.5.1.2 Method of Analysis

The analysis of the spurious operation of the ECCS employs the detailed digital computer program LOFTRAN (Reference 3). The code simulates the neutron-kinetics, RCS, pressurizer, pressurizer relief and safety valves, pressurizer spray, steam generator, steam generator safety valves, and 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. Previous analysis of several cases has shown that the results are independent of time to trip.

The analysis presents a typical transient representing minimum reactivity feedback. Results with maximum reactivity feedback indicate no significant transient. The analysis assumes zero injection line purge volume for calculational simplicity; thus, the boration transient begins immediately when the appropriate valves open.

The assumptions are as follows:

A. Initial Operating Conditions

The analysis of this transient employs RTDP methods. The analysis assumes nominal values consistent with the steady-state, full power operation for the initial reactor power and RCS temperature. This results in the maximum power difference for the load loss. The analysis assumes a nominal value consistent with steady-state, full power operation for initial RCS pressure. Tables 15.0.3-2 and 15.0.3-3 show the initial conditions assumed in the analysis.

B. Moderator and Doppler Coefficients of Reactivity

The analysis assumes a zero moderator temperature coefficient. The cooldown of the RCS would cause a faster decrease in reactivity and the reactor would trip sooner with the use of a positive moderator temperature coefficient. The analysis assumes a low (absolute value) Doppler power coefficient.

C. Reactor Control

The reactor is in manual rod control.

D. Pressurizer Heaters

Pressurizer heaters are inoperable. This assumption yields a higher rate of pressure decrease.

E. Boron Injection

At time zero, two charging pumps inject 2600 ppm borated water into the cold leg of each loop.

F. Turbine Load

Turbine load is constant until the governor drives the throttle valve wide open, and then drops as steam pressure drops.

G. Reactor Trip

Low pressurizer pressure initiates reactor trip.

No single active failure in any plant systems or equipment will adversely affect the consequences of the accident.

15.5.1.3 Results

Figures 15.5.1-1 through 15.5.1-3 show the transient response to inadvertent operations of the ECCS during power operation. Neutron flux starts decreasing immediately due to boron injection, but steam flow does not decrease until later in 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 reactor trips and control rods start moving into the core when the pressurizer pressure reaches the pressurizer low pressure trip setpoint. The DNBR increases throughout the transient.

Table 15.5.1-1 shows the calculated sequence of events. Pressure and temperature slowly rise after the reactor trip, because the turbine also trips and the reactor produces some power due to delayed neutron fissions and decay heat. Paragraph 15.5.1.1 discussed the recovery from this accident.

15.5.1.4 Conclusions

Results of the analysis show that spurious ECCS operation without immediate reactor trip does not present any hazard to the integrity of the RCS.

If the reactor does not trip immediately, the low pressurizer pressure reactor trip actuates. This trips the turbine and prevents excess cooldown, which expedites recovery from the incident.

The conclusions presented in the FSAR remain valid.

15.5.2 Chemical and Volume Control System Malfunction that Increases Reactor Coolant Inventory

Section 15.4.6, "Chemical and Volume Control System Malfunction that Results in a Decrease in Boron Concentration in the Reactor Coolant," analyzes an increase in reactor coolant inventory which results from the addition of cold, unborated water to the RCS. The preceding section, 15.5.1, analyzes an increase in reactor coolant inventory which results from the injection of highly borated water into the RCS.

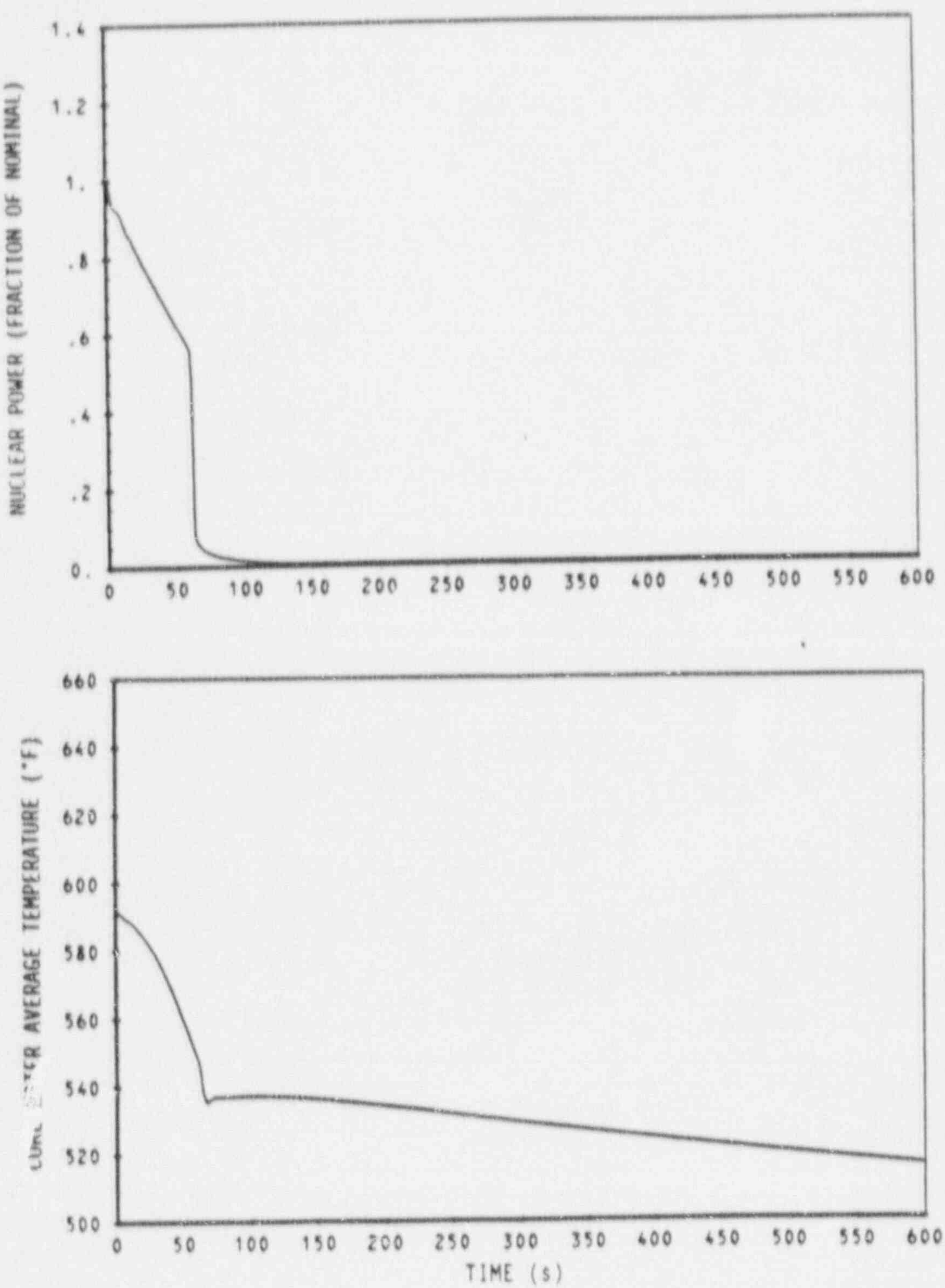
15.5.3 A Number of Boiling Water Reactor Transients

This section is not applicable to VEGP Units 1 and 2.

Table 15.5.1-1

Time Sequence of Events for Incidents Which Result in an Increase in Reactor Coolant Inventory

Accident	Event	Time Delays (s)
Inadvertent actuation of the ECCS during power operation		
	Spurious SI signal generated; two centrifugal charging pumps begin injecting borated water	0.0
	Low pressurizer pressure reactor trip setpoint reached	57.1
	Control rod motion begins	59.1
	Turbine throttle valve wide open; load begins to drop with steam pressure	60.0



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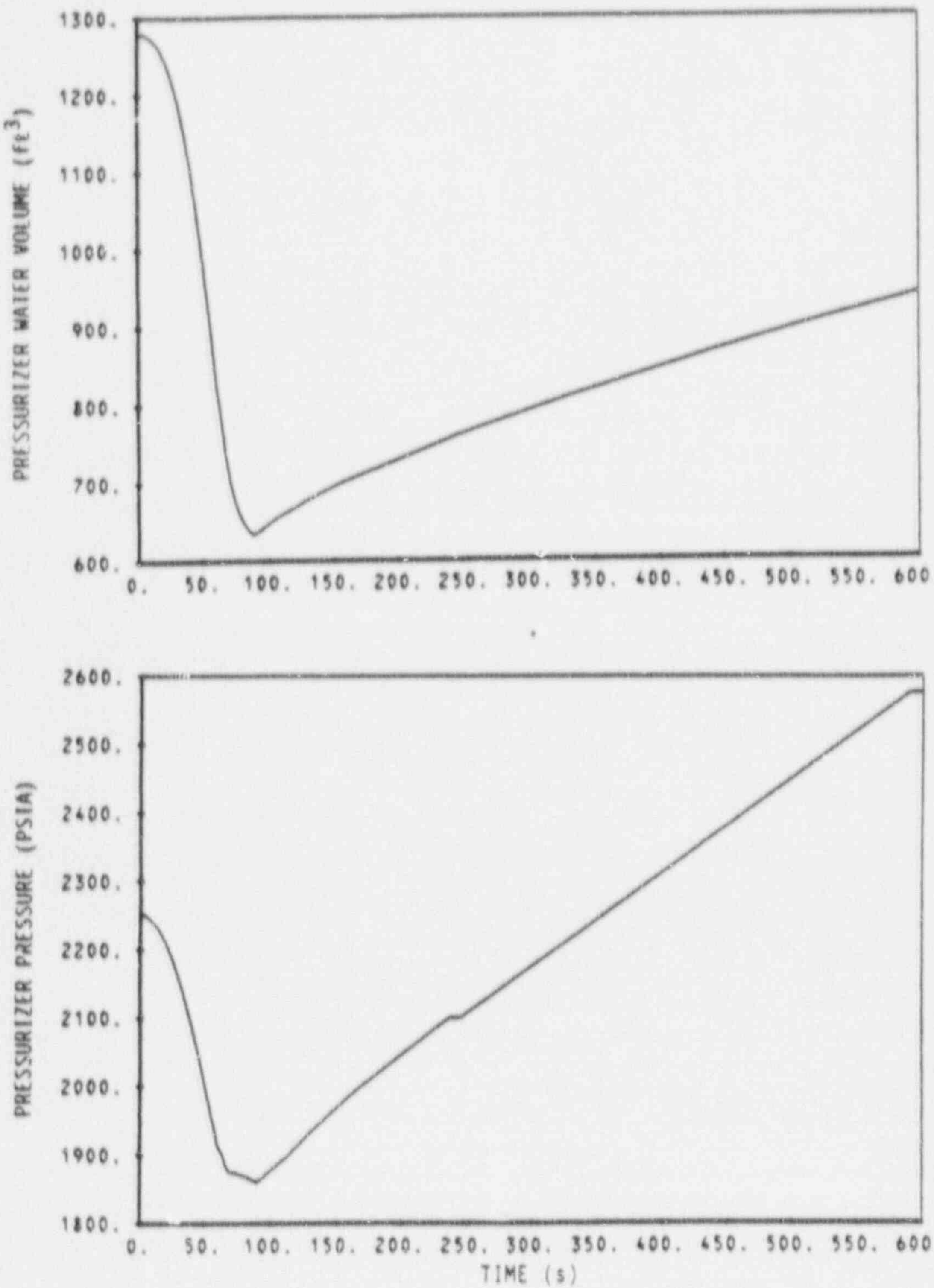
VOGTLE
ELECTRIC GENERATING PLANT
UNIT 1 AND UNIT 2

INADVERTENT OPERATION OF ECCS DURING POWER OPERATION

FIGURE 15.5.1-1

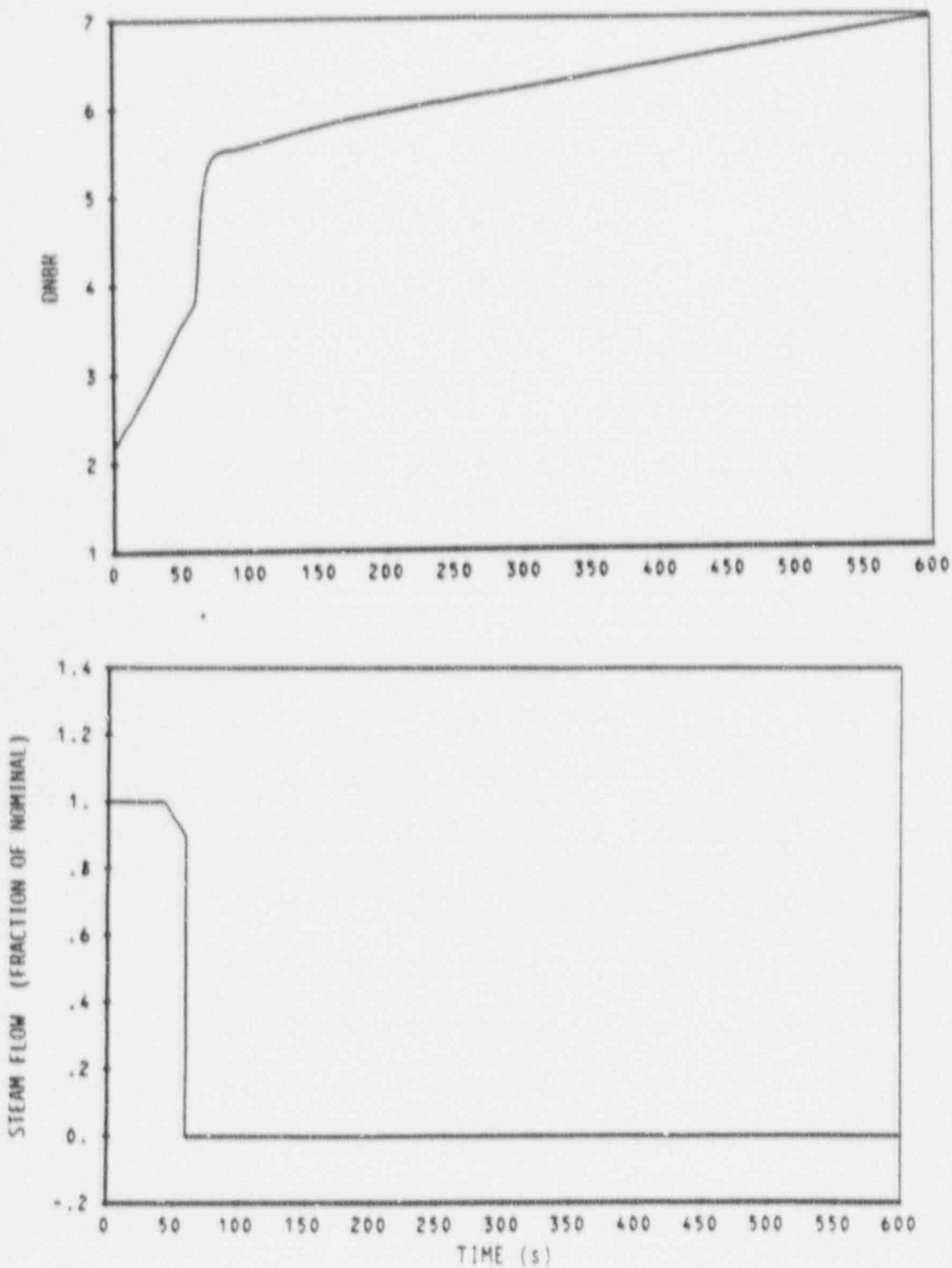


Georgia Power

VOGTLE
ELECTRIC GENERATING PLANT
UNIT 1 AND UNIT 2

INADVERTENT OPERATION OF ECCS DURING POWER OPERATION

FIGURE 15.5.1-2



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VOGTLE
ELECTRIC GENERATING PLANT
UNIT 1 AND UNIT 2

INADVERTENT OPERATION OF ECCS DURING POWER OPERATION

FIGURE 15.5.1-3

15.6 Decrease in Reactor Coolant Inventory

This section discusses the event which could result in a decrease in the RCS inventory.

15.6.1 Inadvertent Opening of a Pressurizer Safety or Relief Valve

15.6.1.1 Introduction

An accidental depressurization of the RCS could occur as a result of an inadvertent opening of a pressurizer relief or safety valve. Since a pressurizer safety valve is sized to relieve approximately twice the steam flowrate of a relief valve and will 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. The effect of the pressure decrease is to increase power via the moderator density feedback. The average coolant temperature remains approximately the same, but the pressurizer level increases until reactor trip.

The reactor may be tripped by the following reactor protection system signals:

o OTΔT

o Pressurizer low pressure

This is an ANS Condition II incident.

15.6.1.2 Method of Analysis

The accidental depressurization transient is analyzed by using the detailed digital computer code LOFTRAN (Reference 3). 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.

In order to produce conservative results in calculating the DNBR during the transient, the following assumptions are made:

A. Nominal values are assumed for the initial reactor power, pressure, and RCS temperatures. Uncertainties in initial conditions are included in the limit DNBR as described in Reference 7 (see Tables 15.0.3-2 and 15.0.3-3).

B. A most positive moderator temperature coefficient of reactivity is assumed. The spatial effect of voids resulting from local or subcooled boiling is not considered in the analysis with respect to reactivity feedback or core power shape.

C. A small (absolute value) Doppler coefficient of reactivity, such that the resultant amount of negative feedback is conservatively low, maximizes any power increase due to moderator feedback.

Normal reactor control systems are not required to function. The reactor protection system functions to trip the reactor on the appropriate signal. No single active failure will prevent the reactor protection system from functioning properly.

15.6.1.3 <u>Results</u>

The system responses to an inadvertent opening of a pressurizer safety valve are shown in Figures 15.6.1-1 and 15.6.1-2. Figure 15.6.1-1 illustrates the nuclear power transient following the depressurization. Nuclear power increases slowly until reactor trip occurs on OTΔT. The pressure decay transient and average temperature transient following are given in Figure 15.6.1-2. Pressure drops more rapidly while core heat generation is reduced via the trip and then slows once saturation temperature is reached in the hot leg. The DNBR decreases initially, but increases rapidly following the trip as shown in Figure 15.6.1-1. The DNBR remains above the limit value throughout the transient.

The calculated sequence of events is shown in Table 15.6.1-1.

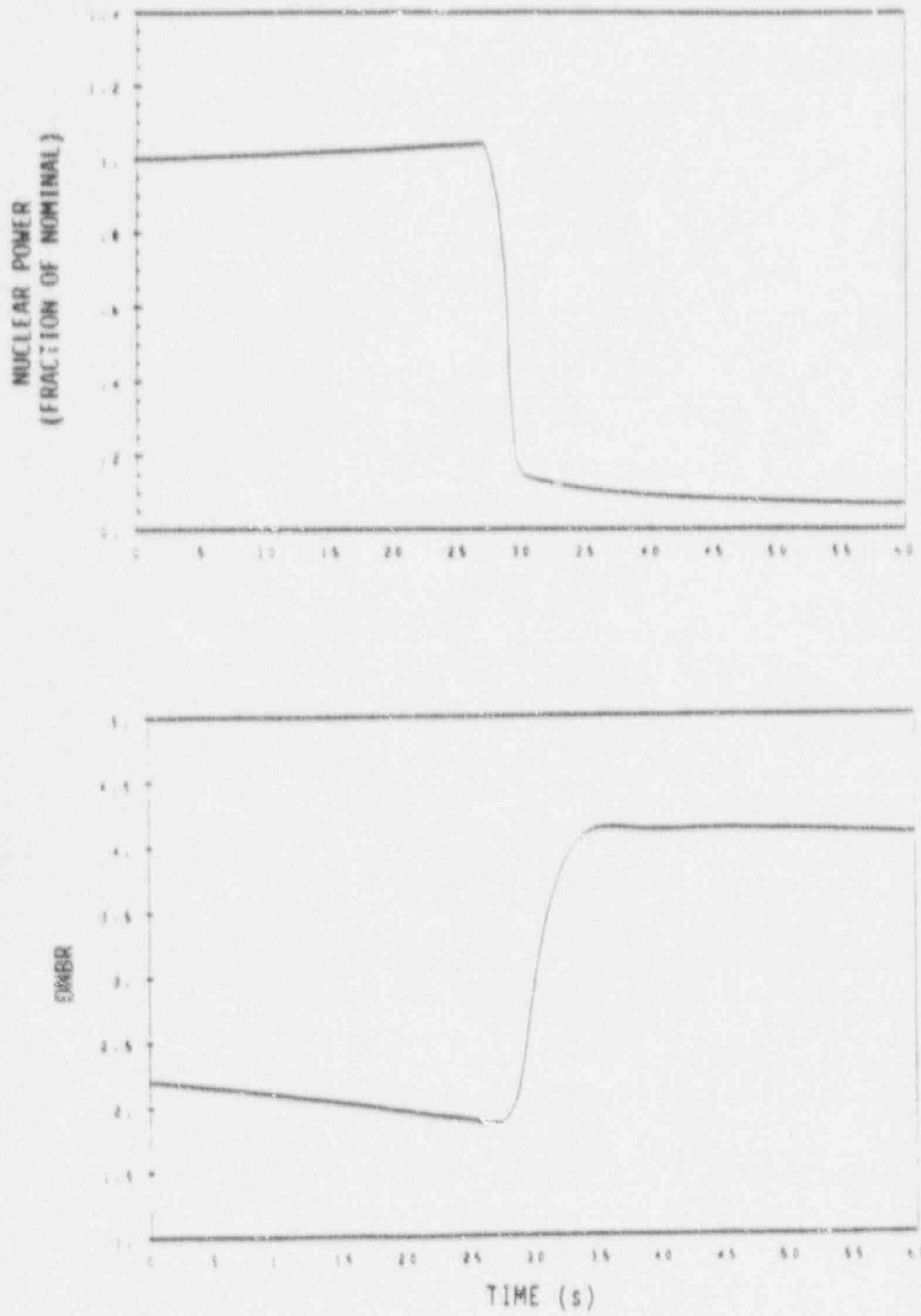
15.6.1.4 <u>Conclusions</u>

The results of the analysis show that the pressurizer low pressure and OTΔT reactor protection system signals provide adequate protection against the RCS depressurization event. The conclusions presented in the FSAR remain valid.

Table 15.6.1-1

Time Sequence of Events for Incidents Which Cause a Decrease in Reactor Coolant Inventory

Accident	Event	Time Delays (s)
Inadvertent opening of a pressurizer safety valve		
	Pressurizer safety valve opens fully	0.0
	OTΔT reactor trip setpoint reached	24.5
	Rods begin to drop	26.5
	Minimum DNBR occurs	27.0



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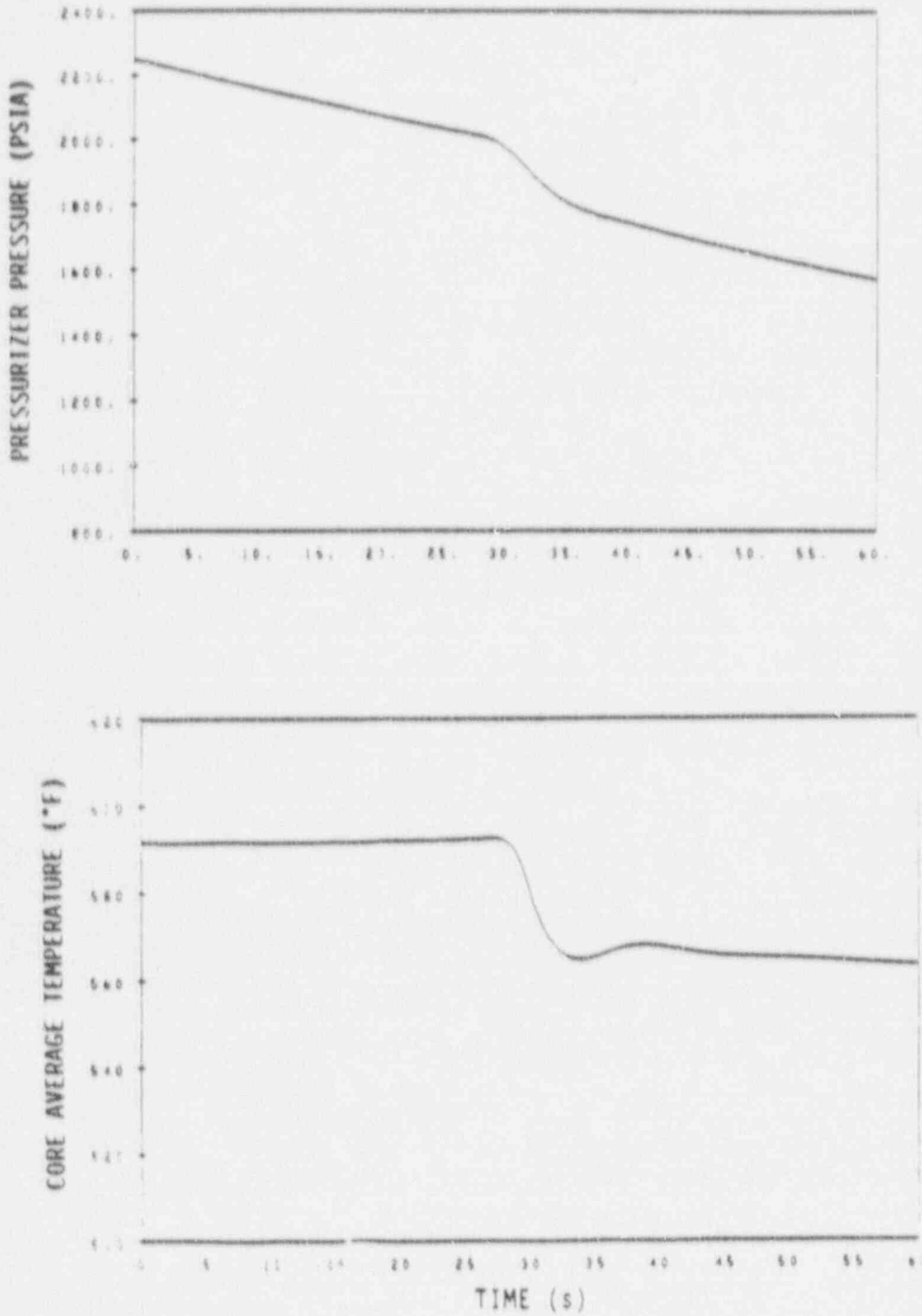
VOGTLE
ELECTRIC GENERATING PLANT
UNIT 1 AND UNIT 2

NUCLEAR POWER AND DNBR TRANSIENTS FOR INADVERTENT OPENING OF A PRESSURIZER SAFETY VALVE

FIGURE 15.6.1-1



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VOGTLE
ELECTRIC GENERATING PLANT
UNIT 1 AND UNIT 2

PRESSURIZER PRESSURE TRANSIENTS AND CORE AVG. TEMP. TRANSIENT FOR INADVERTENT OPENING OF A PRESSURIZER SAFETY VALVE

FIGURE 15.6.1-2

15.R References

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2. H. G. Hargrove, FACTRAN -- A FORTRAN-IV Code for Thermal Transients in a UO2 Fuel Rod, WCAP-7908-A, December 1989.

3. T. W. T. Burnett, et.al., LOFTRAN Code Description, WCAP-7907-A, April 1984.

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7. A. J. Friedland and S. Ray, Revised Thermal Design Procedure, WCAP-11397-P-A, April 1989.

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Appendix B

LOCA and SGTR Accident Analyses

for the Vogtle Electric Generating Plant Units 1 and 2

Transition to Westinghouse 17x17 VANTAGE-5 Fuel Assemblies

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LIST OF FIGURES (continued)

6.2.1.5 Minimum Containment Pressure Analysis for Performance Capability Studies on Emergency Core Cooling System

The containment backpressure and temperature and the containment wall condensing heat transfer coefficient, used for the limiting case CD = 0.6 double-ended cold leg guillotine break for the ECCS analysis found in Subsection 15.6.5, are presented in Figures 6.2.1-35, 36, and 37. The containment backpressure is calculated using the methods and assumptions described in Westinghouse Emergency Core Cooling System Evaluation Model - Summary, WCAP-8339, Appendix A. Input parameters including the containment initial conditions, net free containment volume, passive heat sink materials, thicknesses, surface areas, starting time, and number of containment cooling systems used in the analysis are described below. The inputs used remain the same as that presented in Chapter 6 of the FSAR except for the RWST water temperature which was changed from 50 to 40 degrees F.

6.2.1.5.1 Mass and Energy Release Data

The mass and energy releases to the containment during the reflood portions of the limiting break transient are presented in Table 6.2.1-69. No credit is taken for blowdown. The broken loop mass and energy releases to the containment for the limiting break are given in Table 6.2.1-70.

The mathematical models which calculate the mass and energy releases to the containment are described in Subsection 15.6.5. Since the requirements of Appendix K of 10 CFR 50 are very specific in regard to the modeling of the RCS during blowdown and since the models used are in conformance with Appendix K, no alterations to those models have been made in regard to the mass and energy releases. A break spectrum analysis is performed. (See the double-ended cold leg guillotines which affect the mass and energy released to the containment.) This effect is considered for each case analyzed. During refill, the mass and energy released to the containment is assumed to be zero, which minimizes the containment pressure. During reflood, the effect of steam-water mixing between the safety injection water and the steam flowing through the RCS intact loops reduces the available energy released to the containment vapor space and, therefore, tends to minimize containment pressure.

6.2.1.5.2 Initial Containment Internal Conditions

The following initial values were used in the analysis:

Containment pressure (psia)	14.7
Containment temperature (°F)	90
Refueling water storage tank temperature (°F)	40
NSCW temperature (°F)	40
Outside temperature (°F)	17

The containment initial conditions of 90°F and 14.7 psia are representatively low values anticipated during normal full-power operation. The initial relative humidity was conservatively assumed to be 99 percent.

6.2.1.5.3 Containment Volume

The volume used in the analysis is 2.95×10^6 ft^3.

6.2.1.5.4 Active Heat Sinks

The containment spray system and the containment fan coolers operate to remove heat from the containment. Pertinent data for these systems which were used in the analysis are presented in Table 6.2.1-71. The heat removal capability of each fan cooler is presented in Table 6.2.2-2.

Because the fan coolers use nuclear service cooling water (NSCW), the lowest normal NSCW temperature (40°F) was used in the analysis.

The containment sump temperature was not used in the analysis because the maximum peak cladding temperature occurs prior to initiation of the recirculation mode for the containment spray system. In addition, heat transfer between the sump water and the containment vapor space was not considered in the analysis.

6.2.1.5.5 Steam-Water Mixing

Water spillage rates from the broken loop accumulator are determined as part of the core reflooding calculation and are included in the containment (computer program COCO) code calculational model.

6.2.1.5.6 Passive Heat Sinks

The passive heat sinks used in the analysis, with their thermophysical properties, are given in Table 6.2.1-72. The passive heat sinks and the thermophysical properties were divided [illegible] compliance with Branch Technical Position CSB6-1, Minimum Containment Press[illegible] for Pressurized-Water Reactor (PWR) ECCS Performance Evaluation.

6.2.1.5.7 Heat Transfer to Passive Heat Sinks

The condensing heat transfer coefficients used for heat transfer to the steel containment structures are given in Figure 6.2.1-37 for the limiting break. The containment temperature transient for the limiting break is shown in Figure 6.2.1-36.

6.2.1.5.8 Other Parameters

The effect of having containment purge in operation at the onset of the double-ended cold leg guillotine break was evaluated. With a 5.0-s valve closure time and a 1.5-s signal delay time, containment pressure would be expected to drop less than 0.19 psi. In terms of the LOCA results presented in Chapter 15, this would indicate no penalty in FQ and only a small increase (less than 10°F) in peak clad temperature. This is well within the margin for conformance to Appendix K requirements. No other parameters have a substantial effect on the minimum containment pressure analysis.

6.2.1.5.9 Standard Revirw Plan Evaluation

The VEGP does not employ the heat transfer coefficients supplied in the Standard Review Plan.

The heat transfer coefficients were calculated in conformance with WCAP-8339, Appendix A, which has received Nuclear Regulatory Commission approval. This reference is for the COCO code and predates CSB6-1. Results using the heat transfer coefficients have been found acceptable in other plant applications.

TABLE 6.2.1-69

REFLOOD MASS AND ENERGY RELEASES
(DECLG BREAK/CD=0.6)

Time (s)	Mass (lbm/s)	Energy (Btu/s)
43.89	0.00	0.00
44.79	4.93	6404.72
45.14	4.81	6237.28
54.70	111.73	136141.56
73.40	121.20	147483.74
94.80	241.19	184279.88
117.50	276.88	188572.04
141.80	285.63	183843.10
174.20	340.96	191719.27

TABLE 6.2.1-70

BROKEN LOOP INJECTION SPILL DURING BLOWDOWN
(DECLG BREAK/CD=0.6)

Time (s)	Mass (lbm/s)	Energy (Btu/s)	Enthalpy (Btu/lbm)
0.000	4025.302	239988.489	59.620
1.010	3651.269	217688.648	59.620
2.010	3370.259	200934.819	59.620
3.010	3147.399	187647.928	59.620
4.010	2965.171	176783.501	59.620
5.010	2812.020	167652.612	59.620
6.010	2680.254	159796.745	59.620
7.010	2564.847	152916.207	59.620
8.010	2462.620	146821.384	59.620
9.010	2371.394	141382.530	59.620
10.010	2289.518	136501.055	59.620
11.010	2215.554	132091.329	59.620
12.010	2148.199	128075.601	59.620
13.010	2086.470	124395.354	59.620
14.010	2029.579	121003.522	59.620
15.010	1976.951	117865.795	59.620
16.010	1928.280	114964.045	59.620
17.010	1883.181	112275.229	59.620
18.010	1841.154	109769.580	59.620
19.010	1801.810	107423.889	59.620
20.010	1764.860	105220.971	59.620
21.010	1730.402	103166.590	59.620
22.010	1698.464	101262.404	59.620
23.010	1668.200	99458.108	59.620
24.010	1639.635	97755.014	59.620
25.010	111.567	895.577	8.027

TABLE 6.2.1-70 (Cont.)

BROKEN LOOP INJECTION SPILL DURING BLOWDOWN
(DECLG BREAK/CD=0.6)

Time (s)	Mass (lbm/s)	Energy (Btu/s)	Enthalpy (Btu/lbm)
26.010	111.756	897.097	8.027
27.010	111.954	898.688	8.027
28.010	112.160	900.337	8.027
29.010	112.373	902.046	8.027
30.010	112.582	903.729	8.027

TABLE 6.2.1-71

ACTIVE HEAT SINKS FOR MINIMUM CONTAINMENT PRESSURE ANALYSIS

Containment Spray Parameters	
Number of pumps operating	2
Maximum spray flow (gal/min)	6669
Fastest post-LOCA initiation of spray pumps, assuming offsite power loss and no diesel failure (s)	74.0
Containment Fan Coolers	
Number of fan coolers	8
Maximum CCWS flow (gal/min)	8975
Maximum NSCW water flow (gal/min)	9625
Fastest post-LOCA initiation of fan coolers assuming offsite power loss and no diesel failure (s)	41.1

TABLE 6.2.1-72 (SHEET 1 OF 2)

PASSIVE HEAT SINKS

Wall	Material	Thickness (ft)	Area (ft^2)
1	Epoxy	0.00025	32340.0
	Zinc coating	0.00020833	
	Carbon steel	0.020833	
	Concrete	0.5	
	Concrete	5.5	
2	Zinc coating	0.00020833	72043.0
	Carbon steel	0.020833	
	Concrete	0.5	
	Concrete	5.5	
3	Stainless steel	0.015833	11442.0
	Epoxy	0.001125	
	Concrete	0.5	
	Concrete	6.333	
4	Epoxy	0.0015417	15086.5
	Concrete	0.5	
	Concrete	2.25	
	Carbon steel	0.020833	
	Concrete	10.5	
5	Zinc	0.00020833	4957.0
	Carbon steel	0.020833	
	Concrete	0.5	
	Concrete	7.5	
6	Zinc	0.00020833	595.7
	Carbon steel	0.125	
7	Galvanization	0.00020833	316975.0
	Carbon Steel	0.004375	
8	Zinc	0.00020833	159120.9
	Carbon steel	0.055	
9	Epoxy	0.00025	3003.0
	Zinc	0.00020833	
	Carbon steel	0.020833	
10	Zinc	0.00020833	159134.0
	Carbon steel	0.020833	

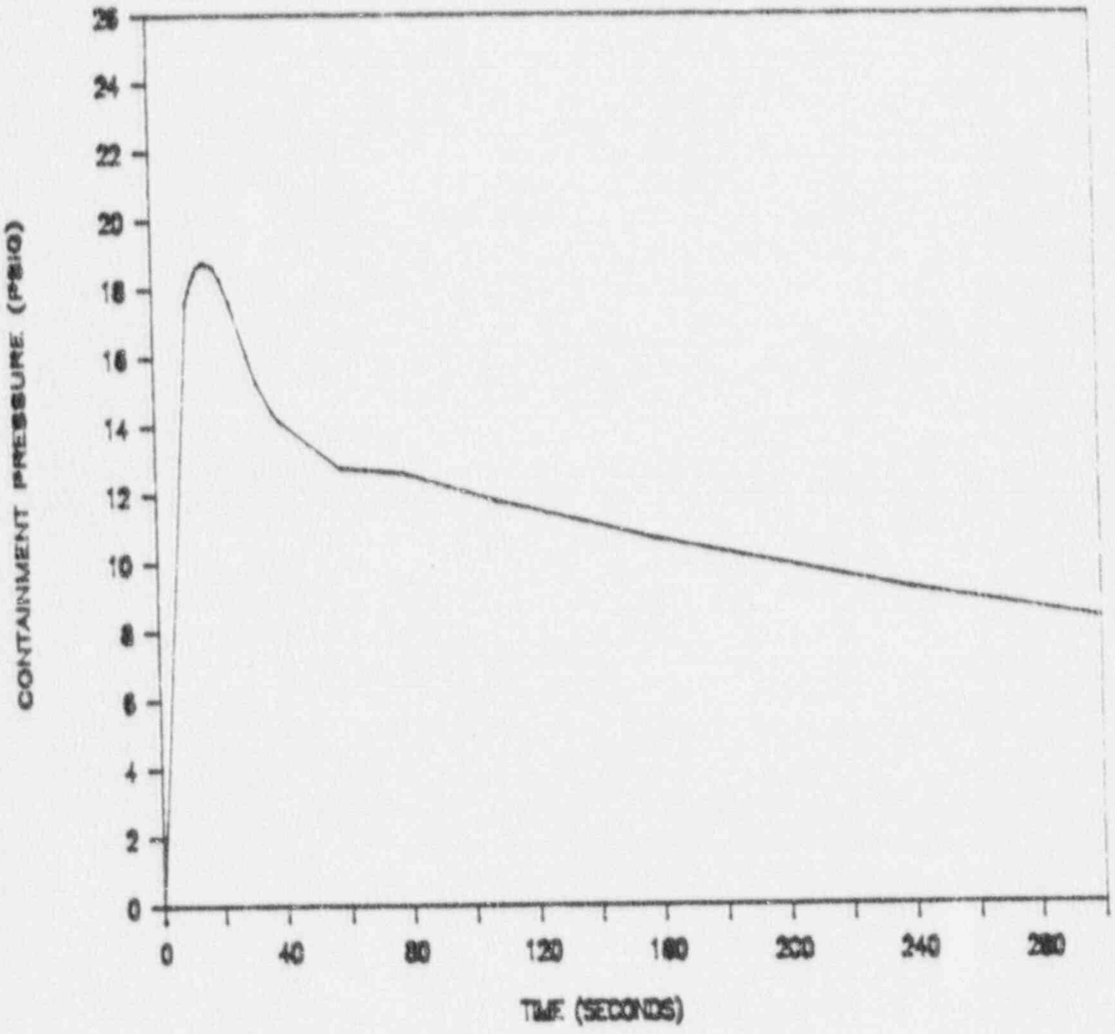
TABLE 6.2.1-72 (SHEET 2 OF 2)

Wall	Material	Thickness (ft)	Area (ft^2)
11	Zinc coating Carbon steel	0.00020833 0.041667	101266.4
12	Zinc coating Carbon steel	0.00020833 0.10417	30474.2
13	Stainless steel	0.008333	75041.0
14	Stainless steel	0.09375	232.0
15	Stainless steel	0.04425	1259.4
16	Epoxy Concrete Concrete	0.001125 0.5 6.833	44202.0
17	Epoxy Concrete Concrete	0.000875 0.5 3.5	19550.3
18	Zinc coating Carbon steel Concrete Concrete	0.00020833 0.08333 0.5 4.0	1546.8
19	Epoxy Concrete Concrete	0.001125 0.5 4.0	2215.4
20	- - - - -	- - - - -	- - - - -

TABLE 6.2.2-2

CONTAINMENT FAN COOLING HEAT REMOVAL CAPACITY
(POST-LOCA MODES)

Containment Temperature (°F)	Capacity (Btu/h)	NSCW Temperature (°F)
120	16.39E+06	40
150	25.92E+06	40
180	39.65E+06	40
210	53.57E+06	40
240	66.09E+06	40
270	77.97E+06	40

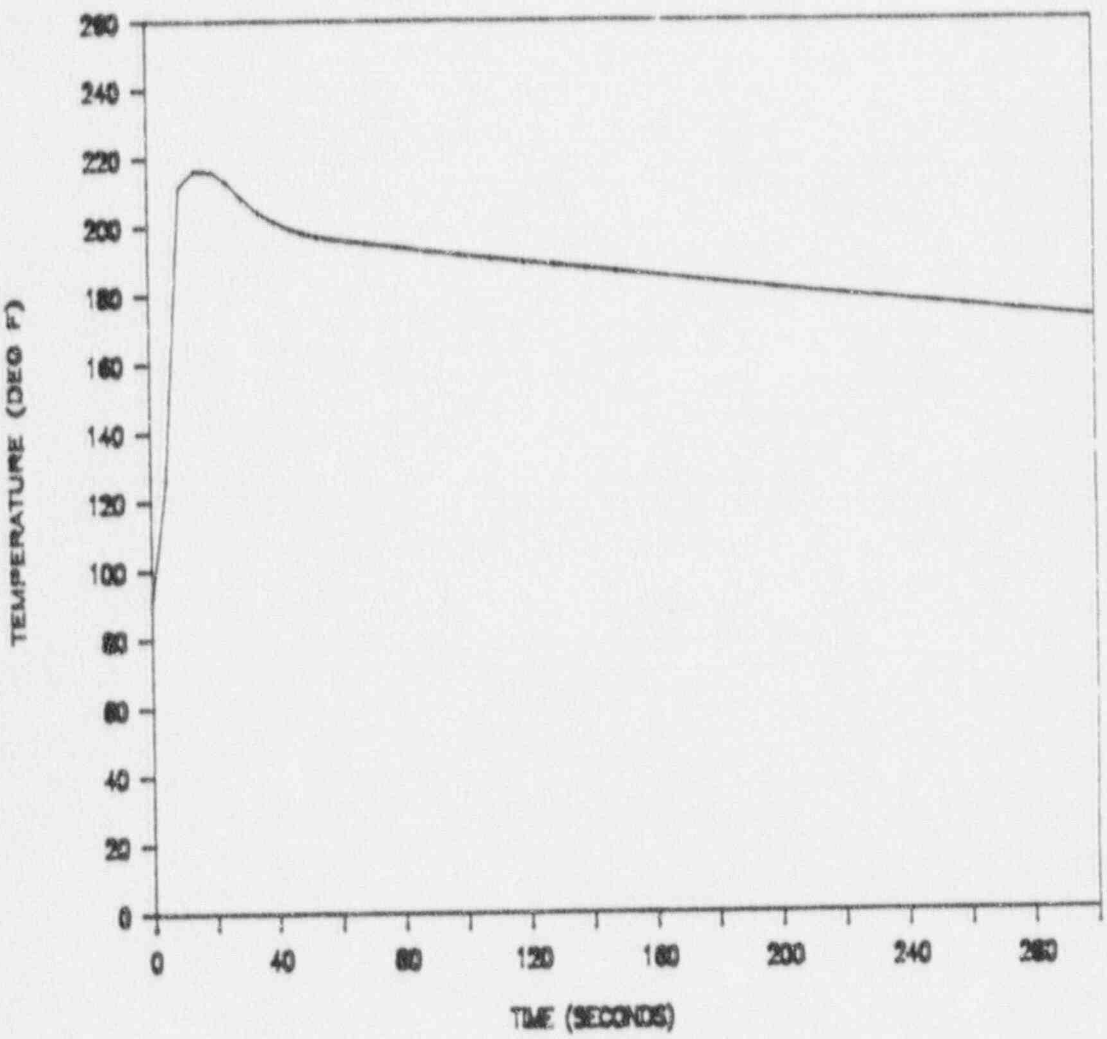


Georgia Power

VOGTLE
ELECTRIC GENERATING PLANT
UNIT 1 AND UNIT 2

CONTAINMENT PRESSURE
DECLG (CD = 0.6)

FIGURE 6.2.1-35

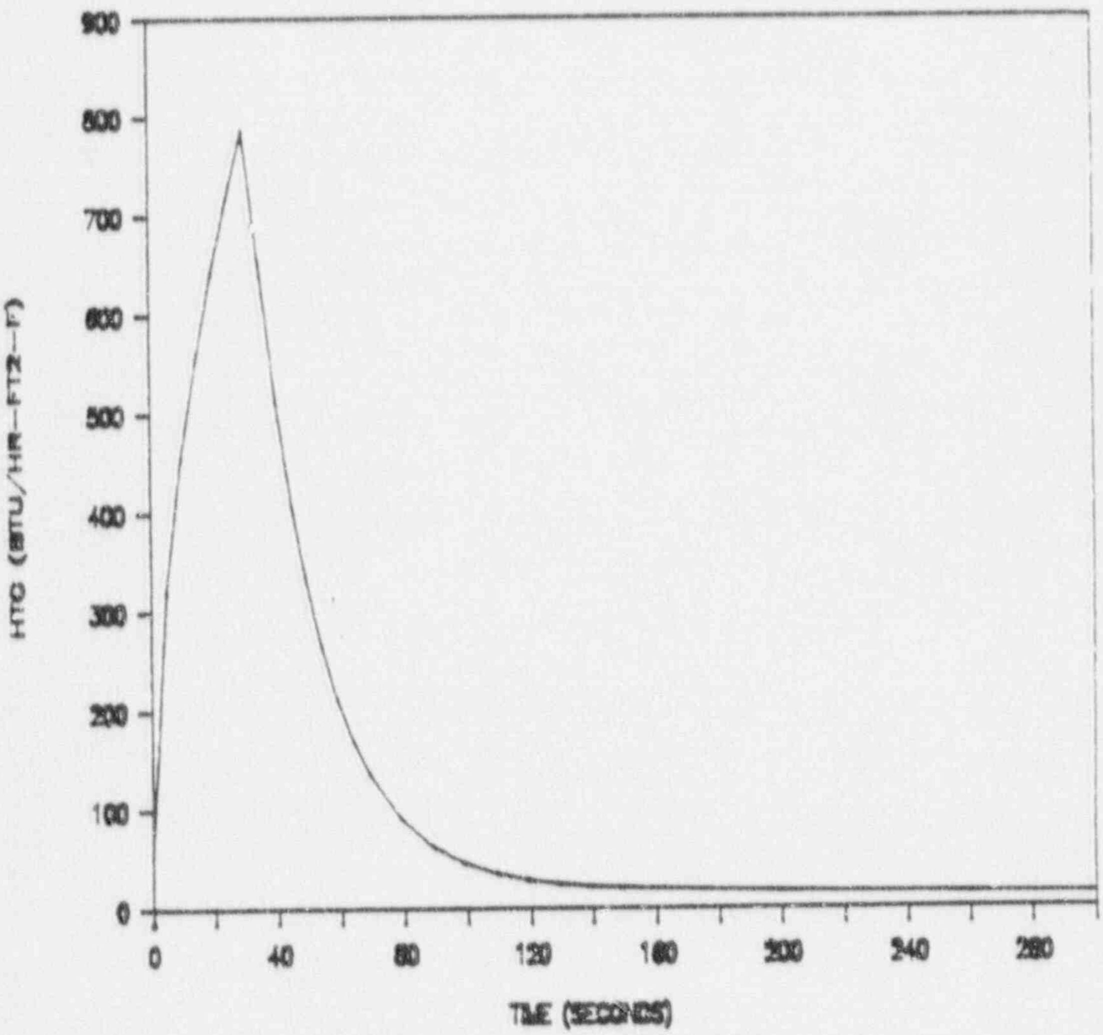


Georgia Power

VOGTLE
ELECTRIC GENERATING PLANT
UNIT 1 AND UNIT 2

CONTAINMENT TEMPERATURE
DECLG (CD = 0.6)

FIGURE 6.2.1-36



Georgia Power

VOGTLE
ELECTRIC GENERATING PLANT
UNIT 1 AND UNIT 2

CONTAINMENT WALL CONDENSING
HEAT TRANSFER COEFFICIENT
DECLG (CD = 0.6)

FIGURE 6.2.1-37

15.6.3 STEAM GENERATOR TUBE RUPTURE (SGTR)

15.6.3.1 Identification of Causes and Accident Description

The accident examined is the complete severance of a single steam generator tube. This event is considered an American Nuclear Society (ANS) Condition IV event, a limiting fault. (See subsection 15.0.1.) The accident is assumed to take place at full power with the reactor coolant contaminated with fission products corresponding to continuous operation with a limited number of defective fuel rods. The accident leads to an increase in contamination of the secondary system due to leakage of radioactive coolant from the reactor coolant system (RCS). In the event of a coincident loss of offsite power or failure of the condenser steam dump system, discharge of activity to the atmosphere takes place via the steam generator power-operated relief valves (and safety valves if their setpoint is reached).

Complete severance of a steam generator tube is considered a somewhat conservative assumption since the Inconel-600 tube material is highly ductile. The more probable mode of tube failure would be one or more minor leaks of undetermined origin. Activity in the steam and power conversion system is subject to continual surveillance, and an accumulation of minor leaks which exceed the limits established in the Technical Specifications is not permitted during the unit operation.

The operator is expected to determine that a steam generator tube rupture has occurred, to identify and isolate the faulty steam generator, and to complete the required recovery actions to stabilize the plant and terminate the primary to secondary break flow. These actions should be performed on a restricted time scale in order to minimize contamination of the secondary system and ensure termination of radioactive release to the atmosphere from the faulty unit. Consideration of the indications provided at the control board, together with the magnitude of the break flow, leads to the conclusion that the recovery procedure can be carried out on a time scale which ensures that break flow to the secondary system is terminated before water level in the affected steam generator rises into the main steam pipe. Sufficient indications and controls are provided to enable the operator to carry out these functions satisfactorily.

If normal operation of the various plant control systems is assumed, the following sequence of events is initiated by a tube rupture.

A. Pressurizer low-pressure and low-level alarms are actuated and charging pump flow increases in an attempt to maintain pressurizer level. On the secondary side, a steam flow/feedwater flow mismatch occurs, since feedwater flow to the affected steam generator is reduced as a result of primary coolant break flow to that unit.

B. The main steamline radiation monitors, the condenser air ejector radiation monitor, and/or the steam generator blowdown liquid monitor will alarm, indicating a sharp increase in radioactivity in the secondary system. The high radiation level alarm from the steam generator blowdown process monitor automatically isolates the system and terminates discharge. The high radiation level alarm from the air ejector monitor automatically diverts the air ejector and steam seal exhauster blower discharges through a filtration unit.

C. The decrease in RCS pressure due to continued loss of reactor coolant inventory leads to a reactor trip signal on low pressurizer pressure or overtemperature ΔT. Resultant plant cooldown following reactor trip leads to a rapid decrease in RCS pressure and pressurizer level, and a safety injection signal initiated by low pressurizer pressure follows soon after reactor trip. The safety injection signal automatically terminates normal feedwater supply and initiates auxiliary feedwater addition.

D. The reactor trip automatically trips the turbine; and if offsite power is available, the steam dump valves open, permitting steam dump to the condenser. In the event of a coincident loss of offsite power, the steam dump valves automatically close to protect the condenser. The steam generator pressure rapidly increases, resulting in steam discharge to the atmosphere through the steam generator power-operated relief valves (and safety valves if their setpoint is reached).

E. Following reactor trip and safety injection actuation, the continued action of the auxiliary feedwater supply and borated safety injection flow (supplied from the refueling water storage tank) provides a heat sink which absorbs some of the decay heat. This reduces the amount of steam bypass to the condenser, or in the case of loss of offsite power, steam relief to the atmosphere.

F. Safety injection flow results in increasing RCS pressure and pressurizer water level, and the RCS pressure trends toward the equilibrium value where the safety injection flow rate equals the break flow rate.

In the event of an SGTR, the plant operators must diagnose the SGTR and perform the required recovery actions to stabilize the plant and terminate the primary to secondary leakage. The operator actions for SGTR recovery are provided in the Emergency Operating Procedures. The major operator actions include identification and isolation of the ruptured steam generator, cooldown and depressurization of the RCS to restore inventory, and termination of SI to stop primary to secondary leakage. These operator actions are described below.

1. Identify the ruptured steam generator.

   High secondary side activity, as indicated by the main steamline radiation monitors, the condenser air ejector radiation monitor, or steam generator blowdown radiation monitors, typically will provide the first indication of an SGTR event. The ruptured steam generator can be identified by an unexpected increase in steam generator narrow range level or a high radiation indication on the corresponding main steamline radiation monitor. For an SGTR that results in a reactor trip at high power, the steam generator water level will decrease due to void collapse but is expected to remain in the narrow range for all of the steam generators. The AFW flow will begin to refill the steam generators, distributing approximately equal flow to each of the steam generators. Since primary to secondary leakage adds additional liquid inventory to the ruptured steam generator, the water level will increase more rapidly in that steam generator. This response, as indicated by the steam generator water level instrumentation, provides confirmation of an SGTR event and also identifies the ruptured steam generator.

2. Isolate the ruptured steam generator from the intact steam generators and isolate feedwater to the ruptured steam generator.

   Once a tube rupture has been identified, recovery actions begin by isolating steam flow from and stopping feedwater flow to the ruptured steam generator. In addition to minimizing radiological releases, this also reduces the possibility of overfilling the ruptured steam generator with water by 1) minimizing the accumulation of feedwater flow and 2) enabling the operator to establish a pressure differential between the ruptured and intact steam generators as a necessary step toward terminating primary to secondary leakage.

3. Cool down the RCS using the intact steam generators.

   After isolation of the ruptured steam generator, the RCS is cooled as rapidly as possible to less than the saturation temperature corresponding to the ruptured steam generator pressure by dumping steam from only the intact steam generators. This ensures adequate subcooling in the RCS after depressurization to the ruptured steam generator pressure in subsequent actions. If offsite power is available, the normal steam dump system to the condenser can be used to perform this cooldown. However, if offsite power is lost, the RCS is cooled using the power operated relief valves (PORVs) on the intact steam generators.

4. Depressurize the RCS to restore reactor coolant inventory.

   When the cooldown is completed, SI flow will increase RCS pressure until break flow matches SI flow. Consequently, SI flow must be terminated to stop primary to secondary leakage. However, adequate reactor coolant inventory must first be assured. This includes both sufficient reactor coolant subcooling and pressurizer inventory to maintain a reliable pressurizer level indication after SI flow is stopped. Since leakage from the primary side will continue after SI flow is stopped until RCS and ruptured steam generator pressures equalize, an "excess" amount of inventory is needed to ensure pressurizer level remains on span. The "excess" amount required depends on RCS pressure and reduces to zero when RCS pressure equals the pressure in the ruptured steam generator.

   The RCS depressurization is performed using normal pressurizer spray if the reactor coolant pumps (RCPs) are running. However, if offsite power is lost or the RCPs are not running for some other reason, normal pressurizer spray is not available. In this event, RCS depressurization can be performed using a pressurizer PORV or auxiliary pressurizer spray.

5. Terminate SI to stop primary to secondary leakage.

   The previous actions will have established adequate RCS subcooling, a secondary side heat sink, and sufficient reactor coolant inventory to ensure that SI flow is no longer needed. When these actions have been completed, SI flow must be stopped to terminate primary to secondary leakage. Primary to secondary leakage will continue after SI flow is stopped until RCS and ruptured steam generator pressures equalize. Charging flow, letdown, and pressurizer heaters will then be controlled to prevent repressurization of the RCS and reinitiation of leakage into the ruptured steam generator.

Following SI termination, the plant conditions will be stabilized, the primary to secondary break flow will be terminated, and all immediate safety concerns will have been addressed. At this time a series of operator actions are performed to prepare the plant for cool down to cold shutdown conditions. Subsequently, actions are performed to cool down and depressurize the RCS to cold shutdown conditions and to depressurize the ruptured steam generator.

15.6.3.2 Analysis of Effects and Consequences

An SGTR results in the leakage of contaminated reactor coolant into the secondary system and subsequent release of a portion of the activity to the atmosphere. Therefore, an analysis must be performed to assure that the offsite radiological consequences resulting from an SGTR are within the allowable guidelines. One of the major concerns for an SGTR is the possibility of steam generator overfill since this could potentially result in a significant increase in the offsite radiological consequences. Therefore, an analysis was performed to demonstrate margin to steam generator overfill similar to the analysis presented in Reference 1, assuming the limiting single failure relative to overfill. The results of this analysis demonstrated that there is margin to steam generator overfill for VEGP. An analysis was also performed to determine the offsite radiological consequences similar to the analysis presented in Reference 1, assuming the limiting single failure relative to offsite doses without steam generator overfill. Since steam generator overfill does not occur, the results of this analysis represent the limiting consequences for an SGTR for VEGP.

A thermal and hydraulic analysis was performed to determine the plant response for a design basis SGTR, and to determine the integrated primary to secondary break flow and the mass releases from the ruptured and intact steam generators to the condenser and to the atmosphere. This information was then used to calculate the quantity of radioactivity released to the environment and the resulting radiological consequences.

15.6.3.3 Thermal and Hydraulic Analysis

The plant response following an SGTR was analyzed with the LOFTTR2 (WCAP-11731) program until the primary to secondary break flow is terminated. The reactor protection system and the automatic actuation of the engineered safeguards systems were modeled in the analysis. The major operator actions which are required to terminate the break flow for an SGTR were also simulated in the analysis.

Analysis Assumptions

The accident modeled is a double-ended break of one steam generator tube located at the top of the tube sheet on the outlet (cold leg) side of the steam generator. It was assumed that the reactor is operating at full power at the time of the accident and the secondary level was assumed to correspond to operation at the nominal steam generator level minus an allowance for uncertainties. It was also assumed that a loss of offsite power occurs at the time of reactor trip and the highest worth control assembly was assumed to be stuck in its fully withdrawn position at reactor trip. Other important analysis assumptions include:

a) NSSS Power = 3579 MWt x 1.02 (uncertainty) = 3650.6 MWt,
b) Average RCS temperature = 588.4°F,
c) RCS Pressure = 2250 psia - 50 psia (uncertainty) = 2200,
d) Thermal Design Flow = 374400 gpm,
e) Pressurizer Level = 62% (includes uncertainty),
f) SG Tube Plugging = 10%,
g) Aux Feed Flow = 2110 gpm @ 1120 psig, 2340 gpm @ 1000 psig,
h) Aux Feed Flow Delay Time = 90 seconds for dose analysis.

The limiting single failure was assumed to be the failure of the PORV on the ruptured steam generator. Failure of this PORV in the open position will cause an uncontrolled depressurization of the ruptured steam generator which will increase primary to secondary leakage and the mass release to the atmosphere. It was assumed that the ruptured steam generator PORV fails open when the ruptured steam generator is isolated, and that the PORV is isolated by locally closing the associated block valve.

The major operator actions required for the recovery from an SGTR are discussed in Section 15.6.3.1 and these operator actions were simulated in the analysis. The operator action times used for the analysis were established in Reference 2 and are presented in Table 15.6.3-1. It is noted that the PORV on the ruptured steam generator was assumed to fail open at the time the ruptured steam generator was isolated. Before proceeding with the recovery operations, the failed open PORV on the ruptured steam generator was assumed to be isolated by locally closing the associated block valve. It was assumed that the ruptured steam generator PORV is isolated at 16 minutes after the valve was assumed to fail open. After the ruptured steam generator PORV was isolated, an additional delay time of 7 minutes (Table 15.6.3-1) was assumed for the operator action time to initiate the RCS cooldown.

Transient Description

The LOFTTR2 analysis results are described below. The sequence of events for this transient is presented in Table 15.6.3-2.

Following the tube rupture, reactor coolant flows from the primary into the secondary side of the ruptured steam generator since the primary pressure is greater than the steam generator pressure. In response to this loss of reactor coolant, pressurizer level decreases as shown in Figure 15.6.3-1. The RCS pressure also decreases as shown in Figure 15.6.3-2 as the steam bubble in the pressurizer expands. As the RCS pressure decreases due to the continued primary to secondary leakage, automatic reactor trip occurs on an overtemperature ΔT trip signal.

After reactor trip, core power rapidly decreases to decay heat levels. The turbine stop valves close and steam flow to the turbine is terminated. The steam dump system is designed to actuate following reactor trip to limit the increase in secondary pressure, but the steam dump valves remain closed due to the loss of condenser vacuum resulting from the assumed loss of offsite power at the time of reactor trip. Thus, the energy transfer from the primary system causes the secondary side pressure to increase rapidly after reactor trip until the steam generator PORVs (and safety valves if their setpoints are reached) lift to dissipate the energy, as shown in Figure 15.6.3-3. The main feedwater flow will be terminated and AFW flow will be automatically initiated following reactor trip and the loss of offsite power.

The RCS pressure decreases more rapidly after reactor trip as energy transfer to the secondary shrinks the reactor coolant and the tube rupture break flow continues to deplete primary inventory. Pressurizer level also decreases more rapidly following reactor trip. The decrease in RCS inventory results in a low pressurizer pressure SI signal. After SI actuation, the SI flow rate exceeds the tube rupture break flow rate and the pressurizer level begins to increase. This also results in an increase in the RCS pressure which trends toward the equilibrium value where the SI flow rate equals the break flow rate.

Since offsite power is assumed lost at reactor trip, the RCPs trip and a gradual transition to natural circulation flow occurs. Immediately following reactor trip the temperature differential across the core decreases as core power decays (see Figures 15.6.3-4 and 15.6.3-5), however, the temperature differential subsequently increases as natural circulation flow develops. The cold leg temperatures trend toward the steam generator temperature as the fluid residence time in the tube region increase. The intact steam generator loop temperatures continue to slowly decrease due to the continued AFW flow until operator actions are taken to control the AFW flow to maintain the specified level in the intact steam generators. The ruptured steam generator loop temperatures also continue to slowly decrease until the ruptured steam generator was isolated and the PORV was assumed to fail open.

Major Operator Actions

1. Identify and Isolate the Ruptured Steam Generator

   The ruptured steam generator was assumed to be identified and isolated at 12 minutes after the initiation of the SGTR or when the narrow range level recovers to 33%, whichever time is greater. However, at-power testing at VEGP with steam generator narrow range lower level tap relocation has shown that the steam generator narrow range level will not drop below 33% following a reactor trip. Therefore, it was conservatively assumed that the ruptured steam generator is isolated at 12 minutes. The ruptured steam generator PORV was also assumed to fail open at this time, and the failure was simulated at 720 seconds. The failure causes the ruptured steam generator to rapidly depressurize as shown in Figure 15.6.3.3, which results in an increase in primary to secondary leakage. The depressurization of the ruptured steam generator increases the break flow and energy transfer from primary to secondary which results in a decrease in the ruptured loop temperatures as shown in Figure 15.6.3-5. As noted previously, the intact steam generator loop temperatures also decrease, as shown in Figure 15.5.3-4, until the AFW flow to the intact steam generators is throttled. After this time, the heat transfer to the intact steam generators decreases and the temperature differential across the intact steam generators decreases. The decrease in the RCS temperatures results in an initial decrease in the pressurizer level and RCS pressure. However, the increased SI flow subsequently causes the pressurizer level and RCS pressure to increase again as shown in Figures 15.6.3-1 and 15.6.3-2, respectively. It was assumed that the time required for the operator to identify that the ruptured steam generator PORV is open and to locally close the associated block valve is 16 minutes. Thus, at 1686 seconds the depressurization of the ruptured steam generator was terminated.

2. Cool Down the RCS to establish Subcooling Margin

After the ruptured steam generator PORV block valve was closed, a 7 minute operator action time was imposed prior to initiation of cooldown. The depressurization of the ruptured steam generator affects the RCS cooldown target temperature since the temperature is dependent upon the pressure in the ruptured steam generator. Since offsite power was lost, the RCS was cooled by dumping steam to the atmosphere using the intact steam generator PORVs. The cooldown was continued until RCS subcooling at the ruptured steam generator pressure is 20°F plus an allowance of 24°F for instrument uncertainty. Because of the lower pressure in the ruptured steam generator, the associated temperature the RCS must be cooled to is also lower, which has the net effect of extending the time for cooldown. The cooldown was initiated at 2106 seconds and was completed at 2842 seconds.

The reduction in the intact steam generator pressures required to accomplish the cooldown is shown in Figure 15.6.3-3, and the effect of the cooldown on the RCS temperature is shown in Figure 15.6.3-4. The pressurizer level and RCS pressure also decrease during this cooldown process due to shrinkage of the reactor coolant, as shown in Figures 15.6.3-1 and 15.6.3-2.

3. Depressurize to Restore Inventory

After the RCS cooldown, a 2 minute operator action time was assumed prior to depressurization. The RCS was depressurized at 2960 seconds to assure adequate coolant inventory prior to terminating SI flow. With the RCPs stopped, normal pressurizer spray is not available and thus the RCS was depressurized by opening a pressurizer PORV. The depressurization was continued until any of the following conditions are satisfied: RCS pressure is less than the ruptured steam generator pressure and pressurizer level is greater than the allowance of 9% for pressurizer level uncertainty, or pressurizer level is greater than 69%, or RCS subcooling is less than the 24°F allowance for subcooling uncertainty. The effect of the RCS depressurization on the RCS pressure and the differential pressure between the RCS and the ruptured steam generator is shown in Figures 15.6.3-2 and 15.6.3-6. The RCS depressurization reduces the break flow as shown in Figure 15.6.3-7, and increases SI flow to refill the pressurizer as shown in Figure 15.6.3-1.

4. Terminate SI to Stop Primary to Secondary Leakage

The previous actions have established adequate RCS subcooling, verified a secondary side heat sink, and restored the reactor coolant inventory to ensure that SI flow is no longer needed. When these actions have been completed, the SI flow must be stopped to prevent repressurization of the RCS and to terminate primary to secondary leakage. The SI flow is terminated at this time if RCS subcooling is greater than the 24°F allowance for uncertainty, minimum AFW flow is available or at least one intact steam generator level is in the narrow range, the RCS pressure is increasing, and the pressurizer level is greater than the 9% allowance for uncertainty. To assure that the RCS pressure is increasing, SI was not terminated until the RCS pressure increases by at least 50 psi.

After depressurization was completed, an operator action time of 2 minutes was assumed prior to SI termination. Since the above requirements are satisfied, SI termination was performed at this time. After SI termination, the RCS pressure decreases as shown in Figure 15.6.3-2. The differential pressure between the RCS and the ruptured steam generator also decreases as shown in Figure 15.6.3-6. Figure 15.6.3-7 shows that the primary to secondary leakage continues after the SI flow is stopped until the RCS and ruptured steam generator pressures equalize.

The ruptured steam generator water volume is shown in Figure 15.6.3-8. It is noted that the water volume in the ruptured steam generator is significantly less than the total steam generator volume of 5906 ft^3 when the break flow is terminated. The mass of water in the ruptured steam generator is also shown as a function of time in Figure 15.6.3-9.

Mass Releases

The mass releases were determined for use in evaluating the exclusion area boundary and low population zone radiation exposure. The steam releases from the ruptured and intact steam generators, the feedwater flows to the ruptured and intact steam generators, and primary to secondary break flow into the ruptured steam generator were determined for the period from accident initiation until 2 hours after the accident and from 2 to 8 hours after the accident. The releases for 0-2 hours were used to calculate the radiation doses at the exclusion area boundary for a 2 hour exposure, and the releases for 0-8 hours were used to calculate the radiation doses at the low population zone for the duration of the accident.

The operator actions for the SGTR recovery up to the termination of primary to secondary leakage were simulated in the LOFTTR2 analysis. Thus, the steam releases from the ruptured and intact steam generators, the feedwater flows to the ruptured and intact steam generators, and the primary to secondary leakage into the ruptured steam generator were determined from the LOFTTR2 results for the period from the initiation of the accident until the leakage is terminated.

Following the termination of leakage, it was assumed that the actions are taken to cool down the plant to cold shutdown conditions. The PORVs for the intact steam generators were assumed to be used to cool down the RCS to the RHR system operating temperature of 350°F, at the maximum allowable cooldown rate of 100°F/hr. The steam releases and the feedwater flows for the intact steam generators for the period from leakage termination until 2 hours were determined from a mass and energy balance using the calculated RCS and intact steam generator conditions at the time of leakage termination and at 2 hours. The RCS cooldown was assumed to be continued after 2 hours until the RHR system in-service temperature of 350°F is reached. Depressurization of the ruptured steam generator was then assumed to be performed to the RHR in-service pressure of 390 psia via steam release from the ruptured steam generator PORV. The RCS pressure was also assumed to be reduced concurrently as the ruptured steam generator is depressurized. It was assumed that the continuation of the RCS cooldown and depressurization to RHR operating conditions are completed within 8 hours after the accident since there is ample time to complete the operations during this time period. The steam releases and feedwater flows from 2 to 8 hours were determined for the intact and ruptured steam generators from a mass and energy balance using the conditions at 2 hours and at the RHR system in-service conditions.

After 8 hours, it was assumed that further plant cooldown to cold shutdown as well as long-term cooling is provided by the RHR system. Therefore, the steam releases to the atmosphere were terminated after RHR in-service conditions were assumed to be reached at 8 hours.

For the time period from initiation of the accident until leakage termination, the releases were determined from the LOFTTR2 results for the time prior to reactor trip and following reactor trip. Since the condenser is in service until reactor trip, any radioactivity released to the atmosphere prior to reactor trip would be through the condenser air ejector. After reactor trip, the releases to the atmosphere were assumed to be via the steam generator PORVs. The mass release rates to the atmosphere from the LOFTTR2 analysis are presented in Figures 15.6.3-10 and 15.6.3-11 for the ruptured and intact steam generators, respectively, for the time period until leakage termination. The mass releases calculated from the time of leakage termination until 2 hours and from 2-8 hours are also assumed to be released to the atmosphere via the steam generator PORVs. The mass releases for the SGTR event for the 0-2 hour and 2-8 hour time intervals are presented in Table 15.6.3-3.

15.6.3.4 Offsite Radiation Dose Analysis

The evaluation of the radiological consequences of a steam generator tube rupture event assumes that the reactor has been operating at the maximum allowable Technical Specification limit for primary coolant activity and primary to secondary leakage for sufficient time to establish equilibrium concentrations of radionuclides in the reactor coolant and in the secondary coolant. Radionuclides from the primary coolant enter the steam generator, via the ruptured tube, and are released to the atmosphere through the steam generator PORVs (and safety valves) and via the condenser air ejector exhaust.

The quantity of radioactivity released to the environment, due to a SGTR, depends upon primary and secondary coolant activity, iodine spiking effects, primary to secondary break flow, break flow flashing fractions, attenuation of iodine carried by the flashed portion of the break flow, partitioning of iodine between the liquid and steam phases, the mass of fluid released from the generator, and liquid-vapor partitioning in the turbine condenser hot well. All of these parameters were conservatively evaluated in a manner consistent with the recommendations in Standard Review Plan 15.6.3.

1. Design Basis Analytical Assumptions

   The major assumptions and parameters used in the analysis are itemized in Table 15.6.3-4.

2. Source Term Calculations

   The radionuclide concentrations in the primary and secondary system, prior to and following the SGTR are determined as follows:

   a. The iodine concentrations in the reactor coolant will be based upon preaccident and accident initiated iodine spikes.

i. Accident Initiated Spike - The initial primary coolant iodine concentration is 1 μCi/gm of Dose Equivalent (D.E.) I-131. Following the primary system depressurization associated with the SGTR, an iodine spike is initiated in the primary system which increases the iodine release rate from the fuel to the coolant to a value 500 times greater than the release rate corresponding to the initial primary system iodine concentration.

ii. Preaccident Spike - A reactor transient has occurred prior to the SGTR and has raised the primary coolant iodine concentration from 1 to 60 μCi/gram of D.E. I-131.

b. The initial secondary coolant iodine concentration is 0.1 μCi/gram of D.E. I-131.

c. The chemical form of iodine in the primary and secondary coolant is assumed to be elemental.

d. The initial noble gas concentrations in the reactor coolant are based upon 1% fuel defects.

3. <u>Dose Calculations</u>

The iodine transport model utilized in this analysis was proposed by Postma and Tam (Reference 3). The model considers break flow flashing, droplet size, bubble scrubbing, steaming, and partitioning. The model assumes that a fraction of the iodine carried by the break flow becomes airborne immediately due to flashing and atomization. Removal credit is taken for scrubbing of iodine contained in the atomized coolant droplets when the rupture site is below the secondary water level. The fraction of primary coolant iodine which is not assumed to become airborne immediately mixes with the secondary water and is assumed to become airborne at a rate proportional to the steaming rate and the iodine partition coefficient. This analysis conservatively assumes an iodine partition coefficient of 100 between the steam generator liquid and steam phases. The model takes no scrubbing or mixing credit when the rupture site is above the secondary water level. Droplet removal by the dryers is conservatively assumed to be negligible. The iodine transport model is illustrated in Figure 15.6.3-12.

The following assumptions and parameters were used to calculate the activity released to the atmosphere and the offsite doses following a SGTR.

a. The mass of reactor coolant discharged into the secondary system through the rupture and the mass of steam released from the ruptured and intact steam generators to the atmosphere are presented in Table 15.6.3-3.

b. The time dependent fraction of rupture flow that flashes to steam and is immediately released to the environment is presented in Figure 15.6.3-13. The break flow flashing fraction was conservatively calculated assuming that 100 percent of the break flow comes from the hot leg side of the steam generator, whereas the break flow actually comes from both the hot leg and cold leg sides of the steam generator.

c. In the iodine transport model, the time dependent iodine removal efficiency for scrubbing of steam bubbles as they rise from the rupture site to the water surface conservatively assumes that the rupture is located at the intersection of the outer tube row and the upper anti-vibration bar. However, the tube rupture break flow was conservatively calculated assuming that the break is at the top of the tube sheet. The water level above the top of the tubes in the ruptured and intact steam generators is shown in Figure 15.6.3-14. The iodine removal efficiency is determined by the method suggested by Postma and Tam (Ref. 3). The iodine removal efficiencies are shown in Figure 15.6.3-15.

d. During the time period that the rupture (or leakage) site is uncovered, all of the activity carried by the break (leakage) flow is assumed to be directly released to the environment, i.e., the activity carried by the break (leakage) flow will neither mix with the secondary water nor partition. The rupture site is considered to be covered when the secondary water level is approximately 12 inches over the rupture location (approximately 10 inches over the apex of the tube bundle).

e. The total primary to secondary leak rate is assumed to be 1.0 gpm as allowed by the Technical Specifications. The leak rate is assumed to be 0.70 gpm to the three intact steam generators and 0.30 gpm to the ruptured steam generator.

f. The iodine partition factor between the liquid and steam of the ruptured and intact steam generators is assumed to be 100.

g. No credit was taken for radioactive decay during release and transport, or for cloud depletion by ground deposition during transport to the site boundary or outer boundary of the low population zone.

h. Short-term atmospheric dispersion factors (x/Qs) for accident analysis and breathing rates are provided in Table 15.6.3-8. The breathing rates were obtained from NRC Regulatory Guide 1.4, (Ref. 4).

4. Offsite Thyroid Dose Calculation Model

Offsite thyroid doses are calculated using the equation:

$$D_{Th} = \sum_i \left[DCF_i \left(\sum_j (IAR)_{ij} \quad (BR)_j \quad (x/Q)_j \right) \right]$$

where:

$(IAR)_{ij}$ = integrated activity of iodine nuclide i released during the time interval j in Ci*

$(BR)_j$ = breathing rate during time interval j in meter^3/second (Table 15.6.3-8)

$(x/Q)_j$ = atmospheric dispersion factor during time interval j in second/meter^3 (Table 15.6.3-8)

$(DCF)_i$ = thyroid dose conversion factor via inhalation for iodine nuclide i in rem/Ci (Table 15.6.3-9)

D_{Th} = thyroid dose via inhalation in rem

Offsite whole-body gamma doses are calculated using the equation:

$$D_\gamma = 0.25 \sum_i \left[\bar{E}_{\gamma i} \left(\sum_j (IAR)_{ij} \, (x/Q)_j \right) \right]$$

where:

$(IAR)_{ij}$ = integrated activity of noble gas nuclide i released during time interval j in Ci *

$(x/Q)_j$ = atmospheric dispersion factor during time interval j in seconds/m^3

---

* No credit is taken for cloud depletion by ground deposition or by radioactive decay during transport to the exclusion area boundary or to the outer boundary of the low-population zone.

$\bar{E}_{\gamma i}$ = average gamma energy for noble gas nuclide i in Mev/dis (Table 15.6.3-10)

D_{γ} = whole body gamma dose due to immersion in rem

Offsite beta-skin doses are calculated using the equation:

$$D_{\beta} = 0.23 \sum_{i} \left[\bar{E}_{\beta i} \left(\sum_{j} (IAR)_{ij} \ (x/Q)_{j} \right) \right]$$

where:

$(IAR)_{ij}$ = integrated activity of noble gas nuclide i released during time interval j in Ci *

$(x/Q)_{j}$ = atmospheric dispersion factor during time interval j in seconds/m^3

$\bar{E}_{\beta i}$ = average beta energy for noble gas nuclide i in Mev/dis (Table 15.6.3-10)

D_{β} = beta-skin dose due to immersion in rem

5. Results

Thyroid, whole-body gamma, and beta-skin doses at the Exclusion Area Boundary and Low Population Zone are presented in Table 15.6.3-11. All doses are well within the allowable guidelines as specified by Standard Review Plan 15.6.3 and 10CFR100.

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* No credit is taken for cloud depletion by ground deposition or by radioactive decay during transport to the exclusion area boundary or to the outer boundary of the low-population zone.

V. REFERENCES

1. Lewis, R. N., Mendler, O. J., Miller, T. A., and Rubin, K., "LOFTTR2 Analysis for A Steam Generator Tube Rupture Event for the Vogtle Electric Generating Plant Units 1 and 2," WCAP-11731 (Proprietary)/WCAP-11732 (Non-Proprietary), January 1988.

2. Southern Company Services, "SGTR Event Operator Action Times Using Vogtle Simulator", NCA-7002, February 1987.

3. Postma, A. K., Tam, P. S., "Iodine Behavior in a PWR Cooling System Following a Postulated Steam Generator Tube Rupture", NUREG-0409.

4. NRC Regulatory Guide 1.4, Rev. 2, "Assumptions Used for Evaluating the Potential Radiological Consequences of a LOCA for Pressurized Water Reactors", June 1974.

5. NRC Regulatory Guide 1.109, Rev. 1, "Calculation of Annual Doses to Man From Routine Releases of Reactor Effluents for the Purpose of Evaluating Compliance with 10 CFR Part 50 Appendix I", October 1977.

6. Bell, M. J., "ORIGEN - The ORNL Isotope Generation and Depletion Code", ORNL-4628, 1973.

TABLE 15.6.3-1

OPERATOR ACTION TIMES FOR DESIGN BASIS SGTR ANALYSIS

Action	Time (min)
Identify and isolate ruptured SG	12 min or LOFTTR2 calculated time to recover to 33% narrow range level in the ruptured SG, whichever is longer*
Operator action time to initiate cooldown	7
Cooldown	Calculated by LOFTTR2
Operator action time to initiate depressurization	2
Depressurization	Calculated by LOFTTR2
Operator action time to initiate SI termination	2
SI termination and pressure equalization	Calculated time for SI termination and equalization of RCS and ruptured SG pressures

*At-power testing at VEGP with steam generator narrow range lower level tap relocation has shown that the steam generator narrow range level will not drop below 33% following a reactor trip. Therefore, it was conservatively assumed that the ruptured SG is isolated at 12 minutes.

TABLE 15.6.3-2

SEQUENCE OF EVENTS

EVENT	TIME (sec)
SG Tube Rupture	0
Reactor Trip	43.8
SI Actuated	343
Ruptured SG Isolated	720
Ruptured SG PORV Fails Open	720
Ruptured SG PORV Block Valve Closed	1686
RCS Cooldown Initiated	2106
RCS Cooldown Terminated	2842
RCS Depressurization Initiated	2960
RCS Depressurization Terminated	3050
SI Terminated	3172
Break Flow Terminated	4620

TABLE 15.6.3-3
MASS RELEASES RESULTS

TOTAL MASS FLOW (POUNDS)

	0 - 2HRS	2- 8HRS
Ruptured SG		
- Condenser	49,900	0
- Atmosphere	105,400	32,800
- Feedwater	100,800	0
Intact SGs		
- Condenser	148,400	0
- Atmosphere	501,600	1,009,900
- Feedwater	827,500	1,026,100
Break Flow	162,000	0

TABLE 15.6.3-4
PARAMETERS USED IN EVALUATING
RADIOLOGICAL CONSEQUENCES

I.	Source Data	
	A. Core power level, MWt	3565
	B. Total steam generator tube leakage, prior to accident, gpm	1.0
	C. Reactor coolant iodine activity:	
	1. Accident Initiated Spike	The initial RC iodine activities based on 1 μCi/gram of D.E. I-131 are presented in Table 15.6.3-5. The iodine appearance rates assumed for the accident initiated spike are presented in Table 15.6.3-6.
	2. Pre-Accident Spike	Primary coolant iodine activities based on 60 μCi/gram of D.E. I-131 are presented in Table 15.6.3-5.
	3. Noble Gas Activity	The intial RC noble gas activities based on 1% fuel defects are presented in Table 15.6.3-7.

TABLE 15.6.3-4 (Sheet 2)

D.	Secondary system initial activity	Dose equivalent of 0.1 μCi/gm of I-131, presented in Table 15.6.3-5
E.	Reactor coolant mass, grams	2.3×10^{8}
F.	Initial Steam generator mass (each), grams	4.0×10^{7}
G.	Offsite power	Lost at time of reactor trip
H.	Primary-to-secondary leakage duration for intact SG, hrs.	8
I.	Species of iodine	100 percent elemental

II. Activity Release Data

A. Ruptured steam generator

1. Rupture flow	See Table 15.6.3-3
2. Rupture flow flashing fraction	See Figure 15.6.3-13
3. Iodine scrubbing efficiency	See Figure 15.6.3-15
4. Total steam release, lbs	See Table 15.6.3-3
5. Iodine partition factor	100

TABLE 15.6.3-4 (Sheet 3)

6. Location of tube rupture	Intersection of outer tube row and upper anti-vibration bar
B. Intact steam generators	
1. Total primary-to-secondary leakage, gpm	0.7
2. Total steam release, lbs	See Table 15.6.3-3
3. Iodine partition factor	100
C. Condenser	
1. Iodine partition factor	100
D. Atmospheric Dispersion Factors	See Table 15.6.3-8

TABLE 15.6.3-5

IODINE SPECIFIC ACTIVITIES

IN THE PRIMARY AND SECONDARY COOLANT

ON 1, 60 AND 0.1 μCi/gram OF D.E. I-131

	Specific Activity (μCi/gm)		
	Primary Coolant		Secondary Coolant
Nuclide	1 μCi/gm	60 μCi/gm	0.1 μCi/gm
I-131	0.76	45.6	0.076
I-132	0.76	45.6	0.076
I-133	1.14	68.4	0.011
I-134	0.20	11.7	0.020
I-135	0.63	37.8	0.063

TABLE 15.6.3-6

IODINE SPIKE APPEARANCE RATES

(CURIES/SECOND)

I-131	I-132	I-133	I-134	I-135
1.7	9.0	3.6	5.5	3.4

TABLE 15.6.3-7

NOBLE GAS SPECIFIC ACTIVITIES IN THE
REACTOR COOLANT BASED ON 1% FUEL DEFECTS

Gas	µCi/gm
Xe-133m	17.00
Xe-133	270.00
Xe-135m	0.48
Xe-135	7.20
Xe-138	0.64
Kr-85m	2.00
Kr-85	7.30
Kr-87	1.30
Kr-88	3.60

TABLE 15.6.3-8

ATMOSPHERIC DISPERSION FACTORS AND BREATHING RATES

Time (hours)	Exclusion Area Boundary x/Q (Sec/m^3)	Low Population Zone x/Q (Sec/m^3)	Breathing Rate (m^3/Sec) [Ref. 4]
0-2	1.8×10^{-4}	7.2×10^{-5}	3.47×10^{-4}
2-8	-	3.3×10^{-5}	3.47×10^{-4}

TABLE 15.6.3-9

THYROID DOSE CONVERSION FACTORS

(Rem/Curie) [Ref. 5]

Nuclide	Conversion Factor
I-131	1.49×10^6
I-132	1.43×10^4
I-133	2.69×10^5
I-134	3.73×10^3
I-135	5.60×10^4

TABLE 15.6.3-10

AVERAGE GAMMA AND BETA ENERGY FOR NOBLE GASES

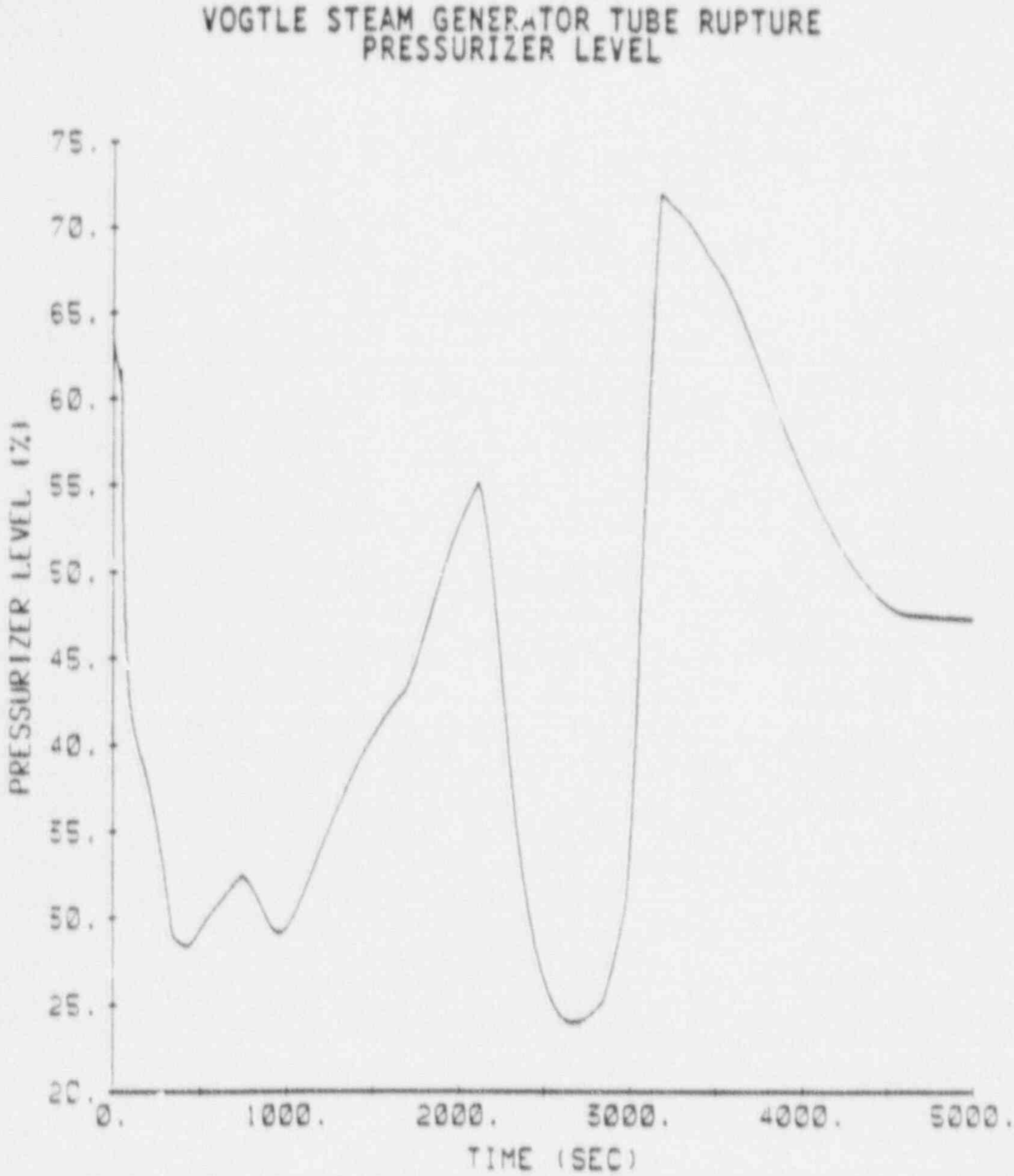
(Mev/dis) [Ref. 6]

Nuclide	$\bar{E}_{\gamma}$	$\bar{E}_{\beta}$
Xe-131m	0.0029	0.165
Xe-133m	0.02	0.212
Xe-133	0.03	0.153
Xe-135m	0.43	0.099
Xe-135	0.246	0.325
Xe-138	1.2	0.66
Kr-85m	0.156	0.253
Kr-85	0.0023	0.251
Kr-87	0.793	1.33
Kr-88	2.21	0.248

TABLE 15.6.3-11

OFFSITE RADIATION DOSES

	Doses (Rem) Calculated Value	Doses (Rem) Allowable Guideline Value
1. Accident Initiated Iodine Spike		
Exclusion Area Boundary (0-2 hr.)		
Thyroid Dose	4.8	30
Low Population Zone (0-8 hr.)		
Thyroid Dose	2.0	30
2. Pre-Accident Iodine Spike		
Exclusion Area Boundary (0-2 hr.)		
Thyroid Dose	33.1	300
Low Population Zone (0-8 hr.)		
Thyroid Dose	13.3	300
3. Whole-Body Gamma and Beta-Skin Dose		
Exclusion Area Boundary (0-2hr)		
Whole-Body Gamma Dose	0.07	2.5
Beta-Skin Dose	0.16	Not Specified
Low Population Zone (0-8hr.)		
Whole-Body Gamma Dose	0.03	2.5
Beta-Skin Dose	0.06	Not Specified



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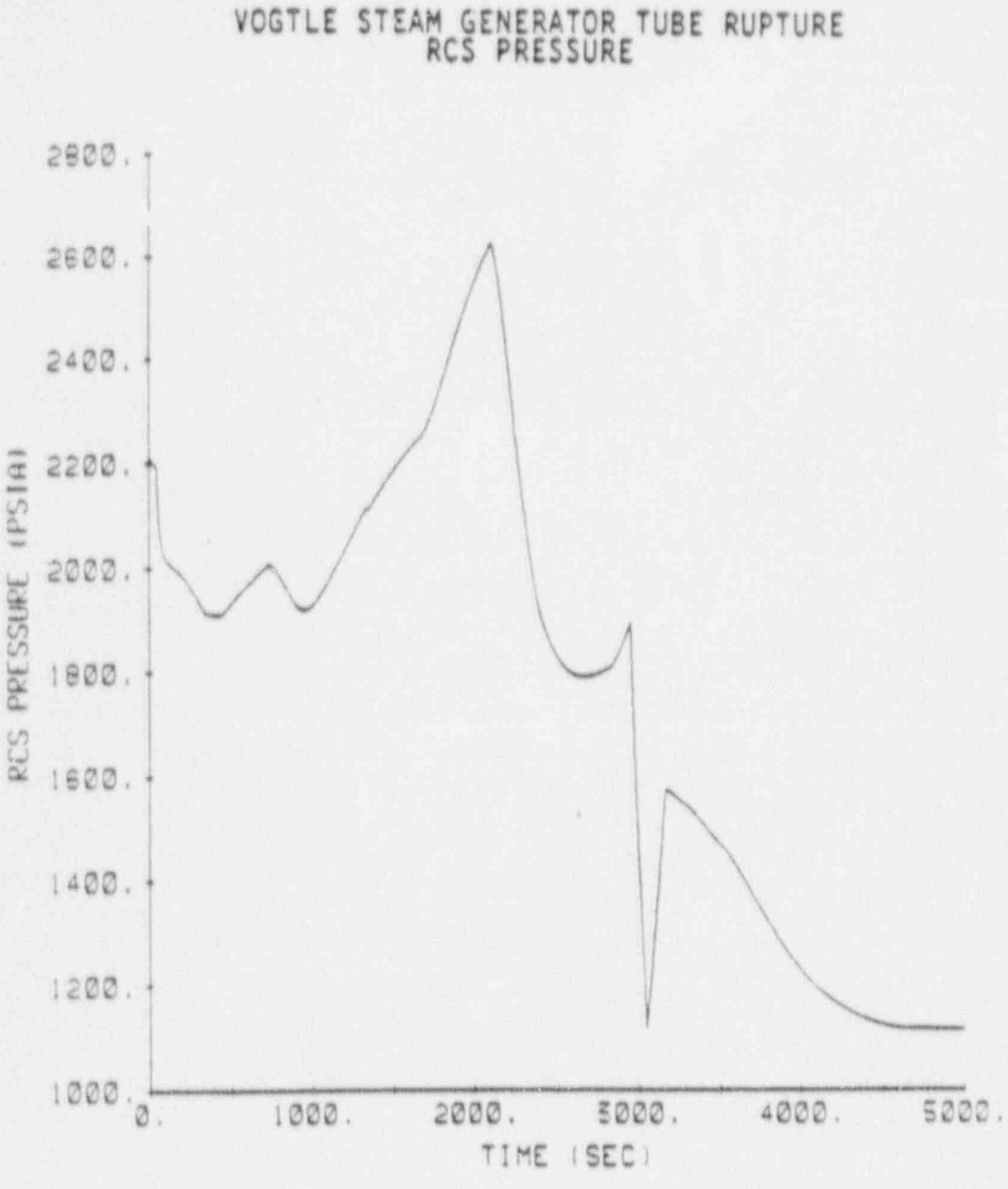
VOGTLE
ELECTRIC GENERATING PLANT
UNIT 1 AND UNIT 2

STEAM GENERATOR TUBE RUPTURE
PRESSURIZER LEVEL

FIGURE 15.6.3-1



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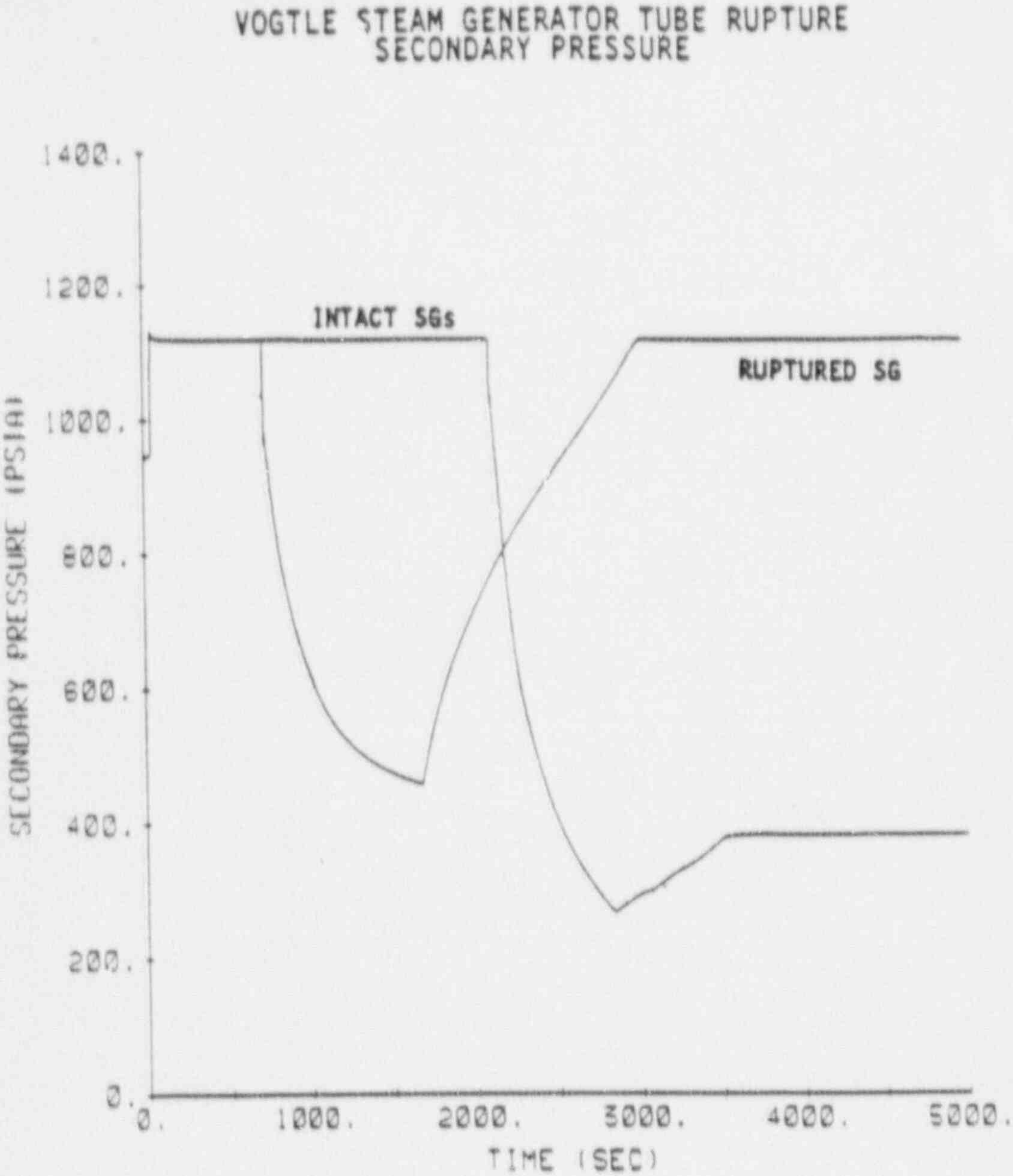
VOGTLE
ELECTRIC GENERATING PLANT
UNIT 1 AND UNIT 2

STEAM GENERATOR TUBE RUPTURE
RCS PRESSURE

FIGURE 15.6.3-2



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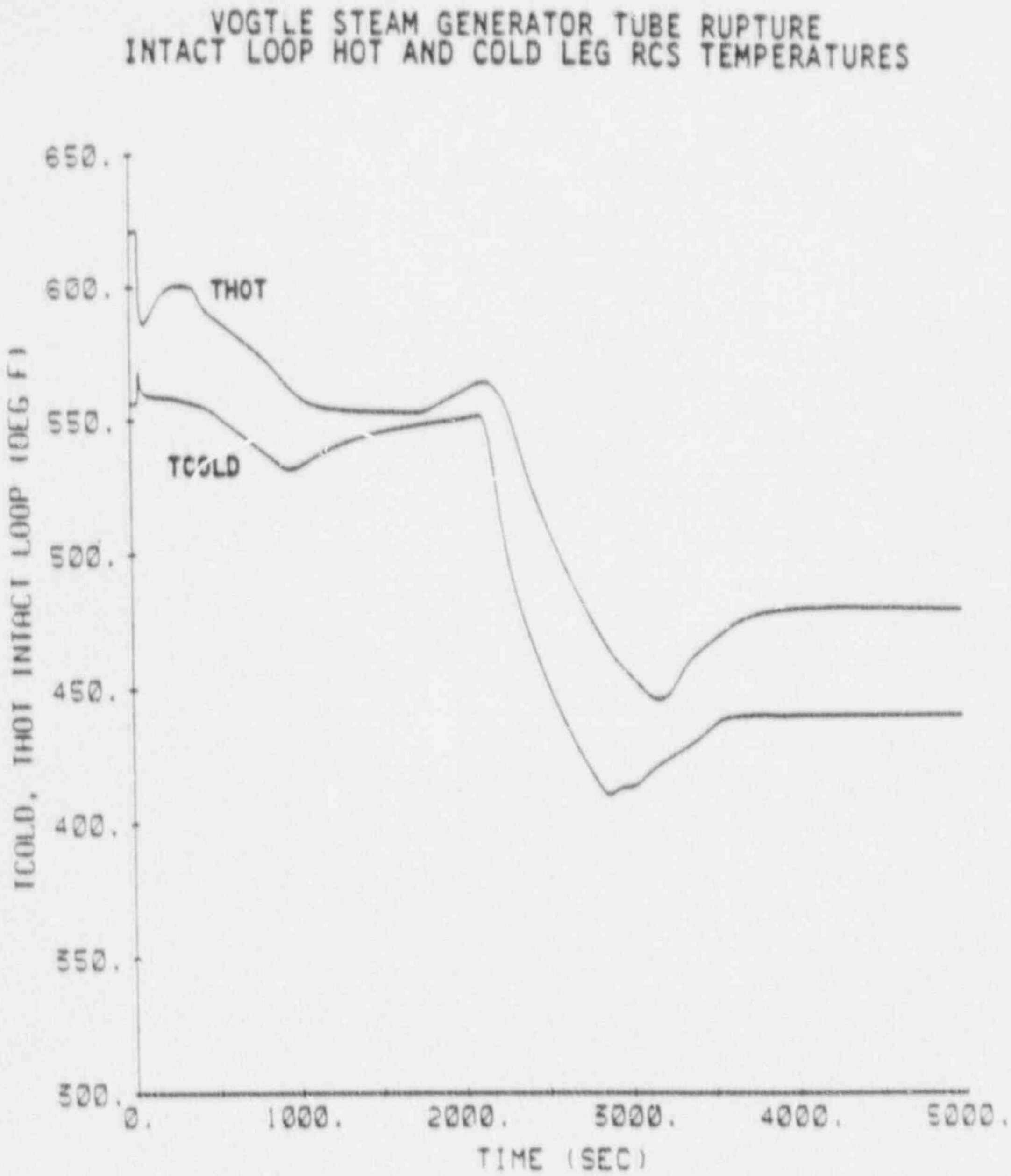
VOGTLE
ELECTRIC GENERATING PLANT
UNIT 1 AND UNIT 2

STEAM GENERATOR TUBE RUPTURE
SECONDARY PRESSURE

FIGURE 15.6.3-3



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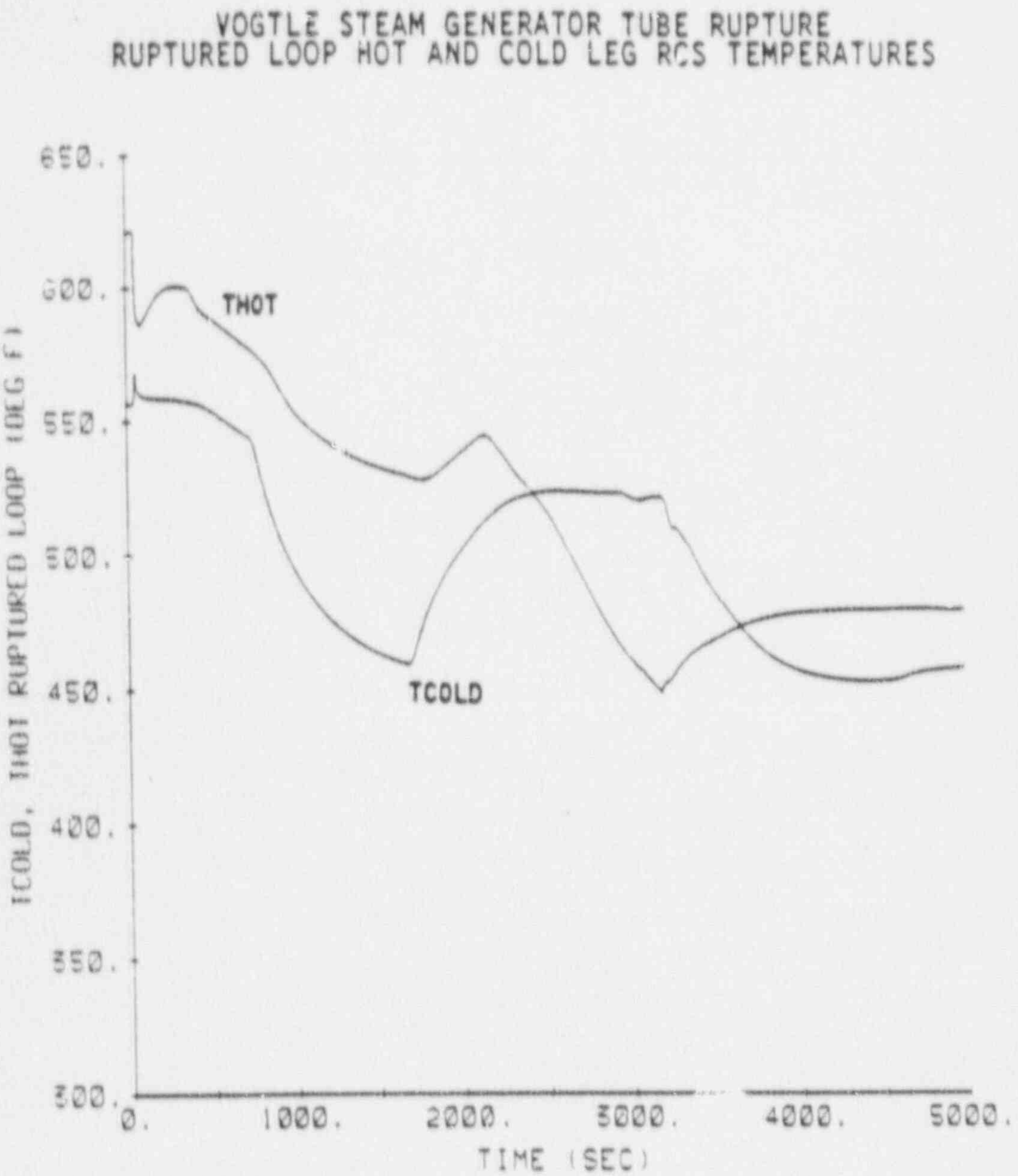
VOGTLE
ELECTRIC GENERATING PLANT
UNIT 1 AND UNIT 2

STEAM GENERATOR TUBE RUPTURE
INTACT LOOP Thot AND Tcold

FIGURE 15.6.3-4



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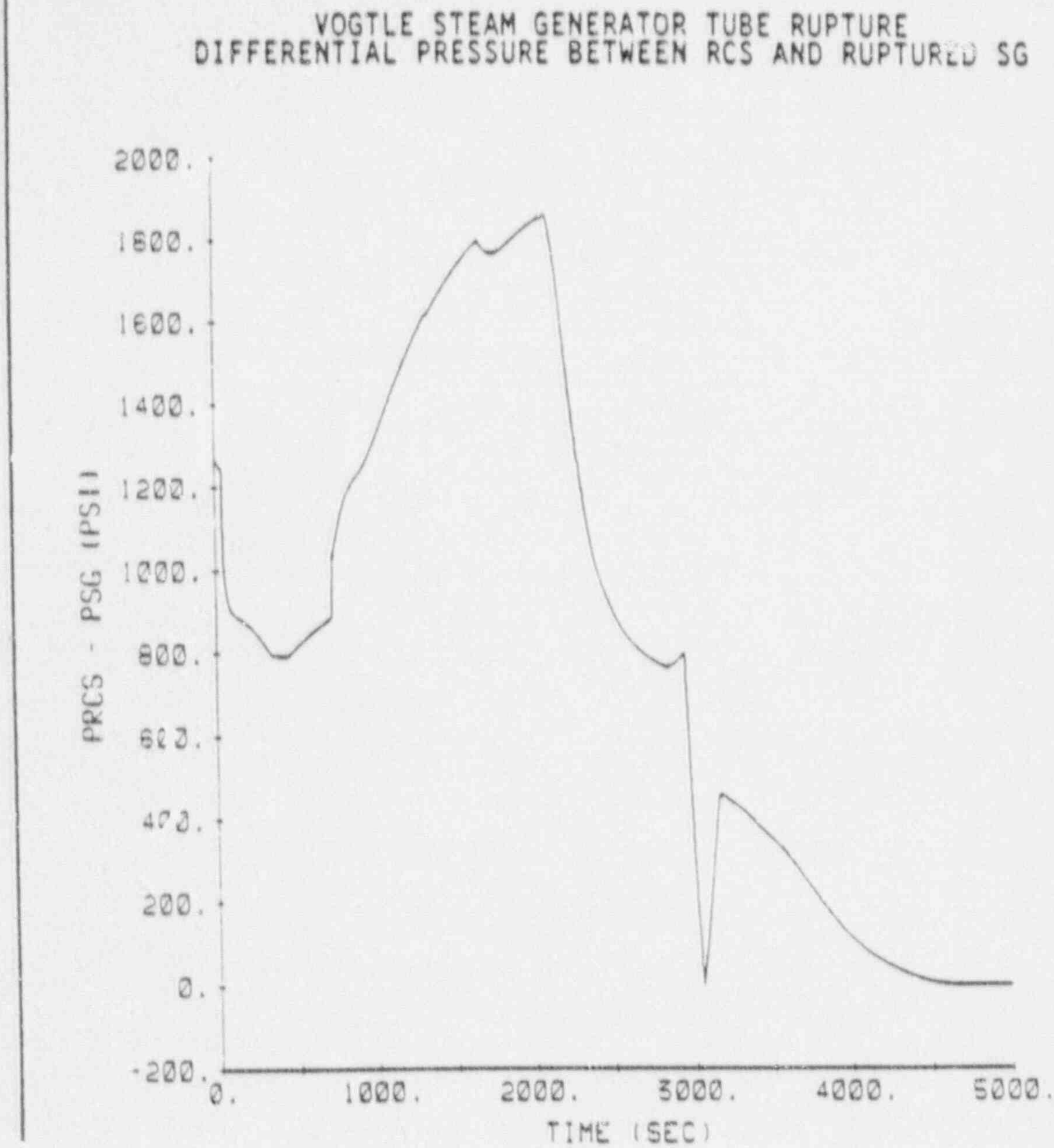
VOGTLE
ELECTRIC GENERATING PLANT
UNIT 1 AND UNIT 2

STEAM GENERATOR TUBE RUPTURE
RUPTURED LOOP Thot AND Tcold

FIGURE 15.6.3-5



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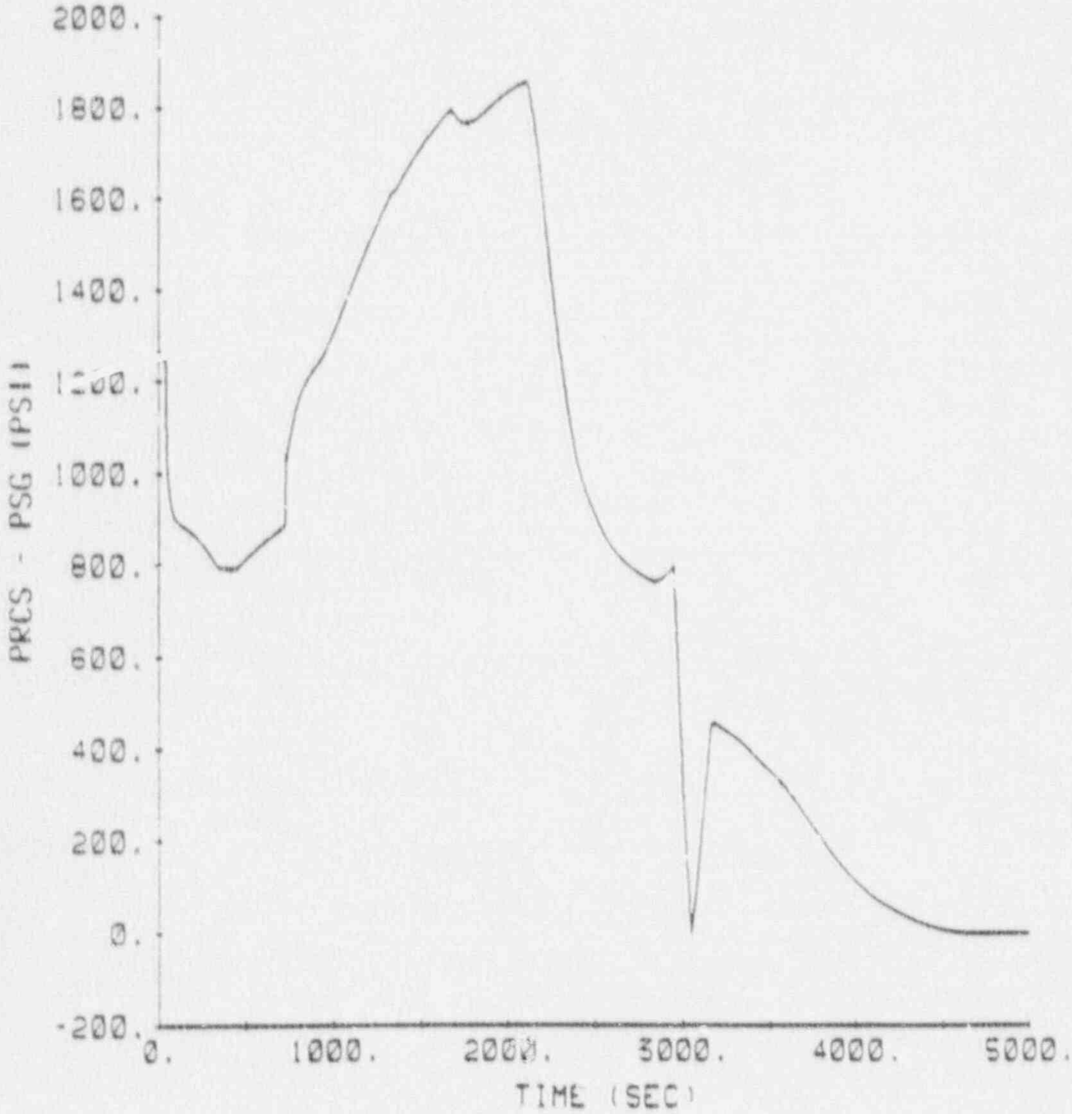
VOGTLE
ELECTRIC GENERATING PLANT
UNIT 1 AND UNIT 2

STEAM GENERATOR TUBE RUPTURE
DIFFERENTIAL PRESSURE

FIGURE 15.6.3-6

VOGTLE STEAM GENERATOR TUBE RUPTURE
DIFFERENTIAL PRESSURE BETWEEN RCS AND RUPTURED SG



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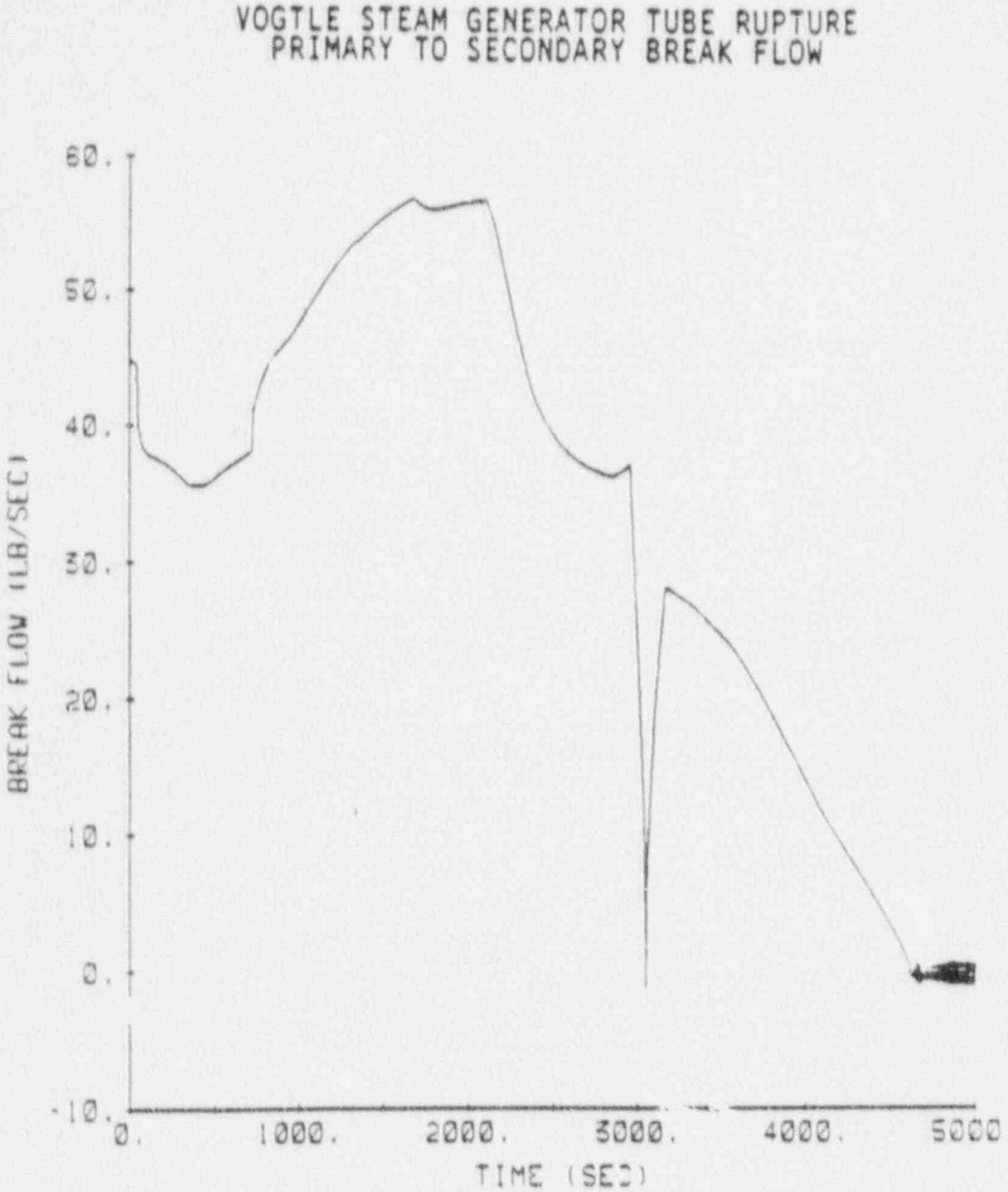
VOGTLE
ELECTRIC GENERATING PLANT
UNIT 1 AND UNIT 2

STEAM GENERATOR TUBE RUPTURE
DIFFERENTIAL PRESSURE

FIGURE 15.6.3-6



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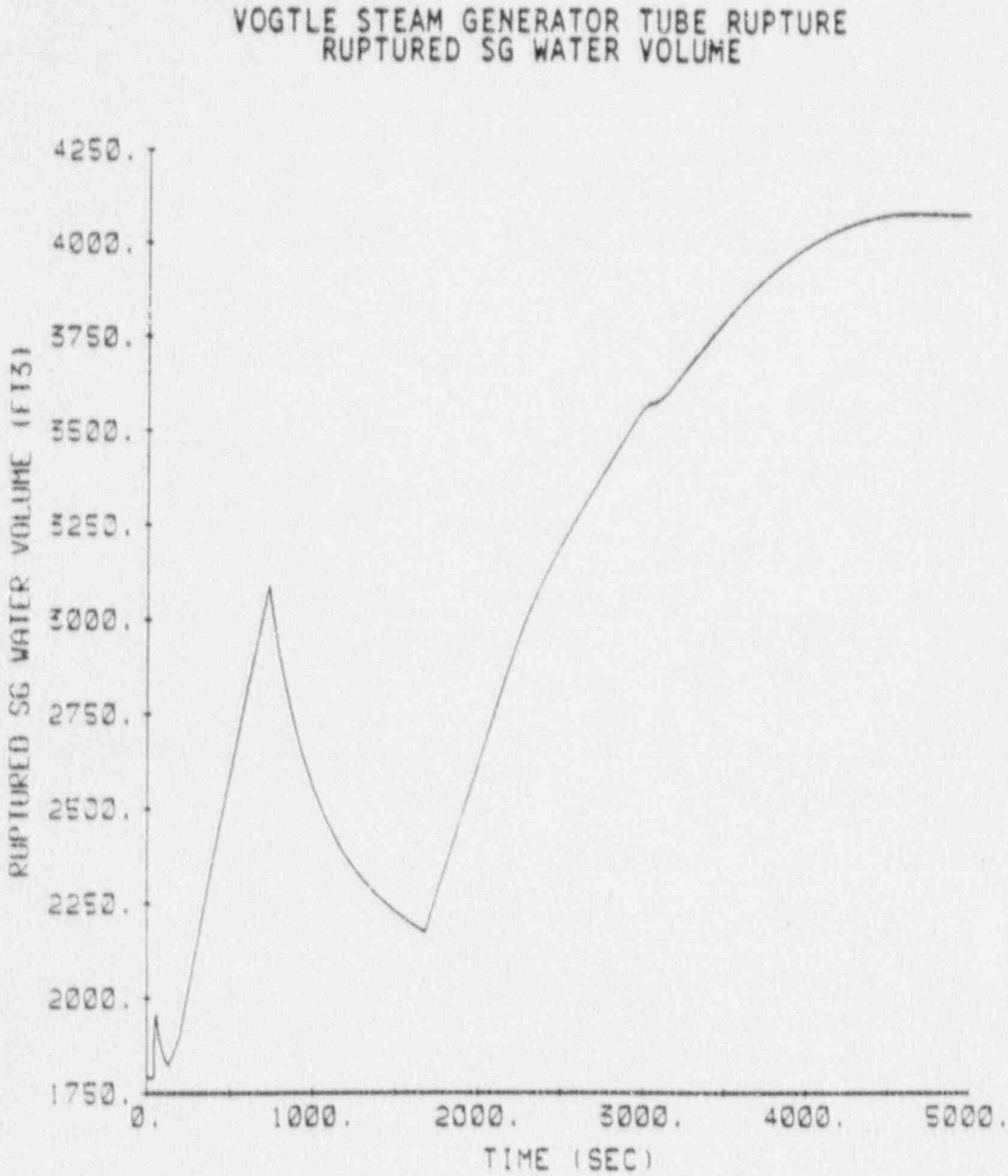
VOGTLE
ELECTRIC GENERATING PLANT
UNIT 1 AND UNIT 2

STEAM GENERATOR TUBE RUPTURE
PRIMARY TO SECONDARY BREAK FLOW

FIGURE 15.6.3-7



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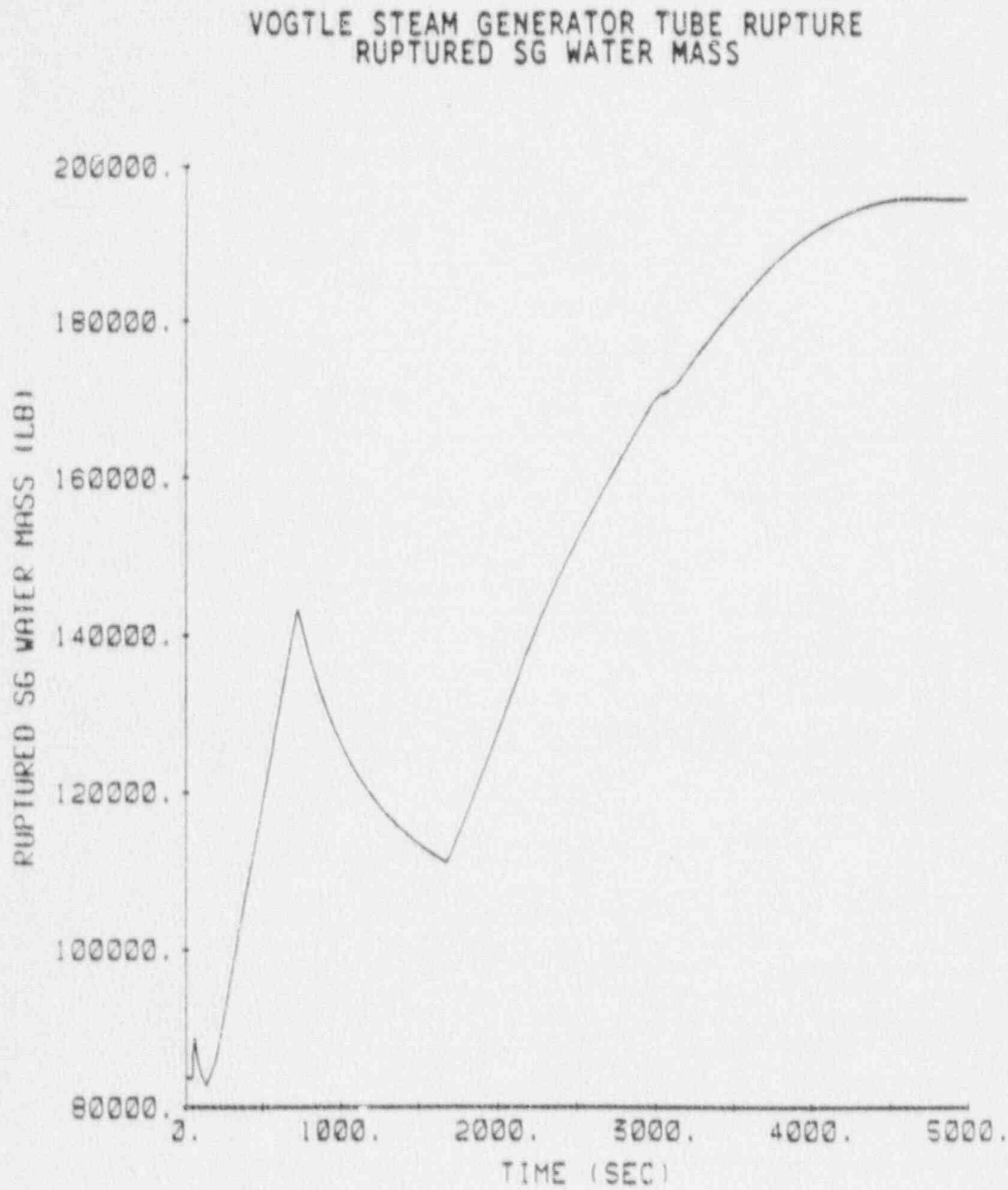
VOGTLE ELECTRIC GENERATING PLANT UNIT 1 AND UNIT 2

STEAM GENERATOR TUBE RUPTURE
RUPTURED SG WATER VOLUME

FIGURE 15.6.3-8



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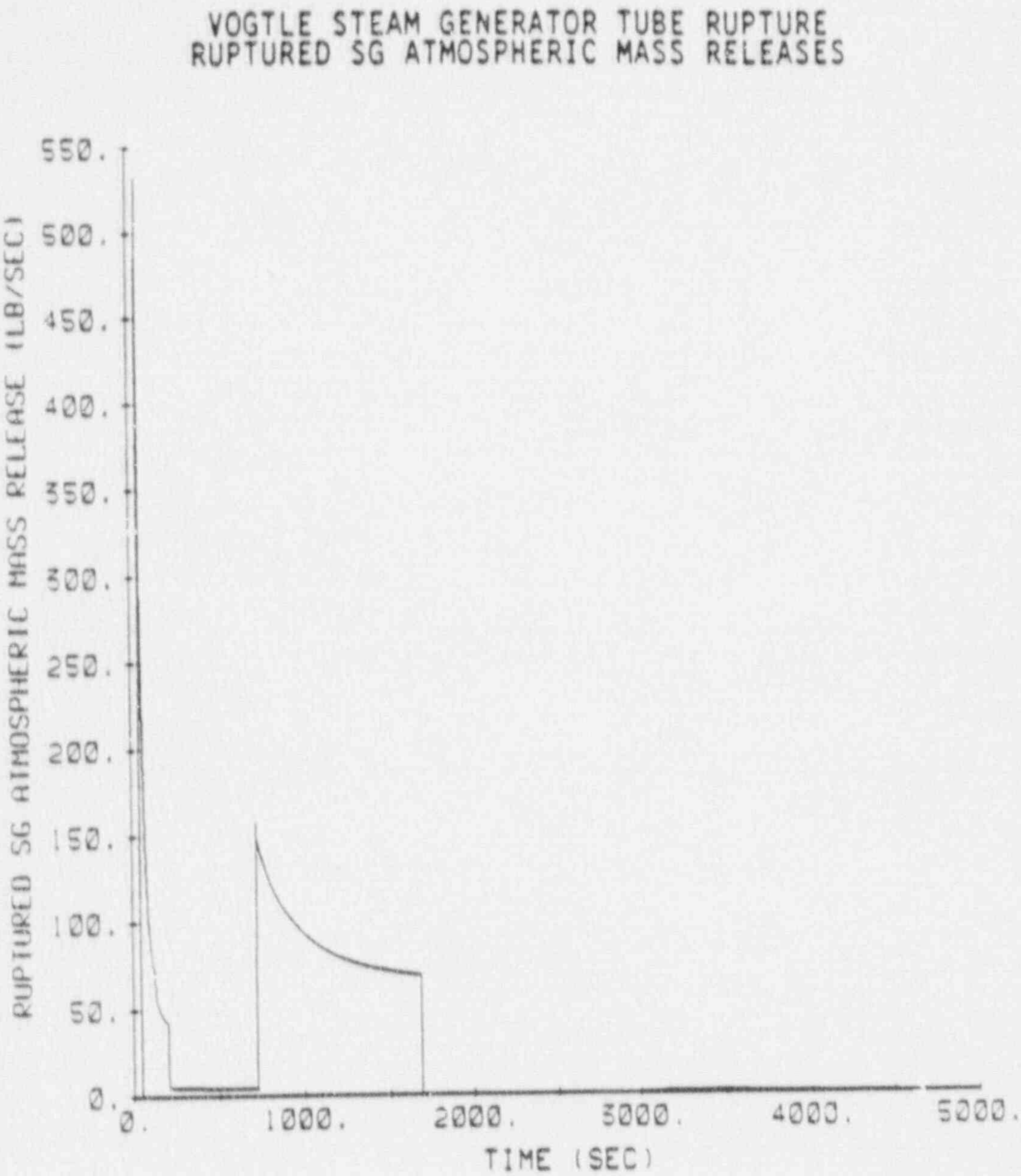
VOGTLE
ELECTRIC GENERATING PLANT
UNIT 1 AND UNIT 2

STEAM GENERATOR TUBE RUPTURE
RUPTURED SG WATER MASS

FIGURE 15.6.3-9



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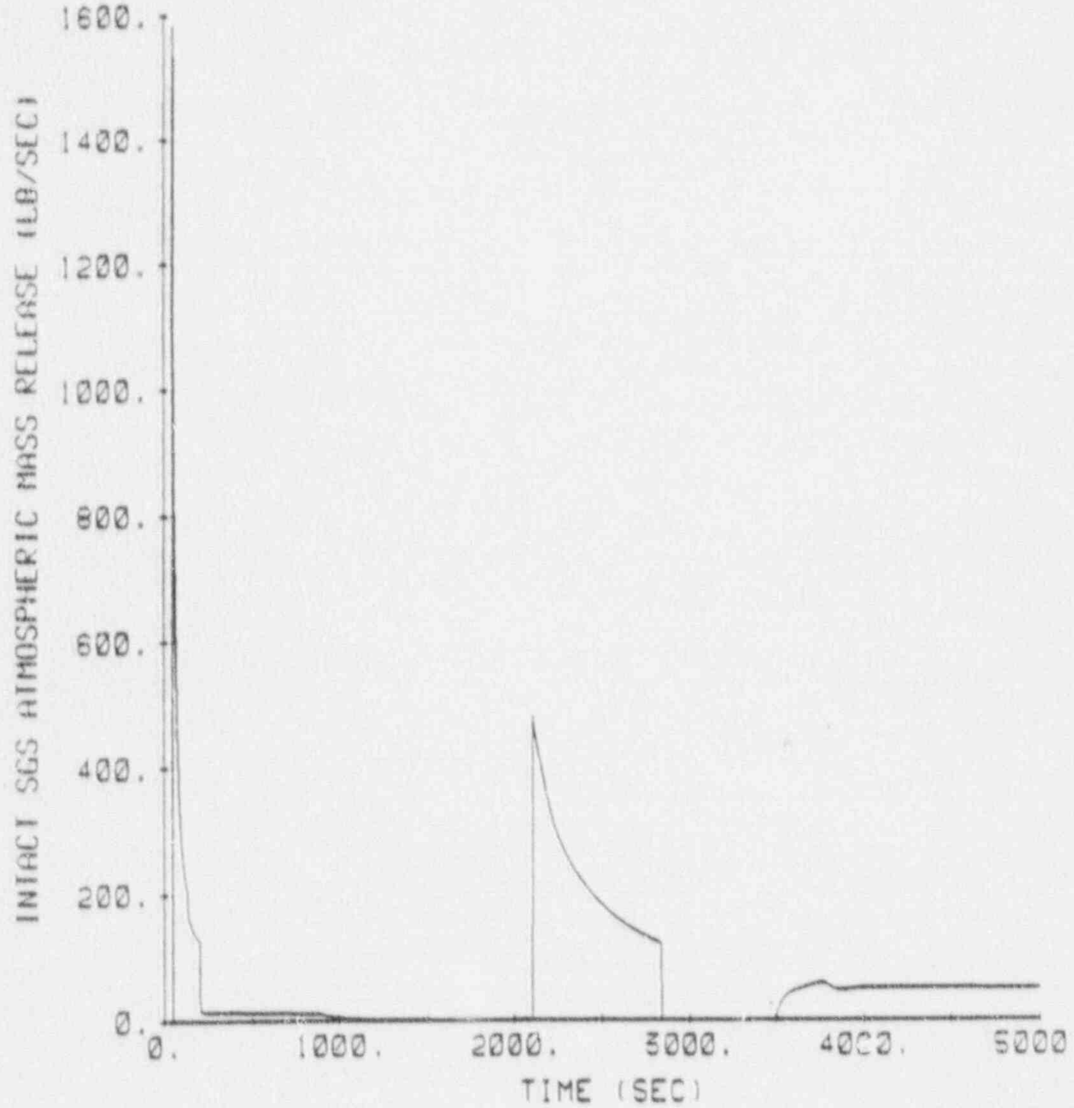
VOGTLE
ELECTRIC GENERATING PLANT
UNIT 1 AND UNIT 2

STEAM GENERATOR TUBE RUPTURE
RUPTURED SG ATMOS. MASS RELEASE

FIGURE 15.6.3-10

VOGTLE STEAM GENERATOR TUBE RUPTURE
INTACT SGS ATMOSPHERIC MASS RELEASE



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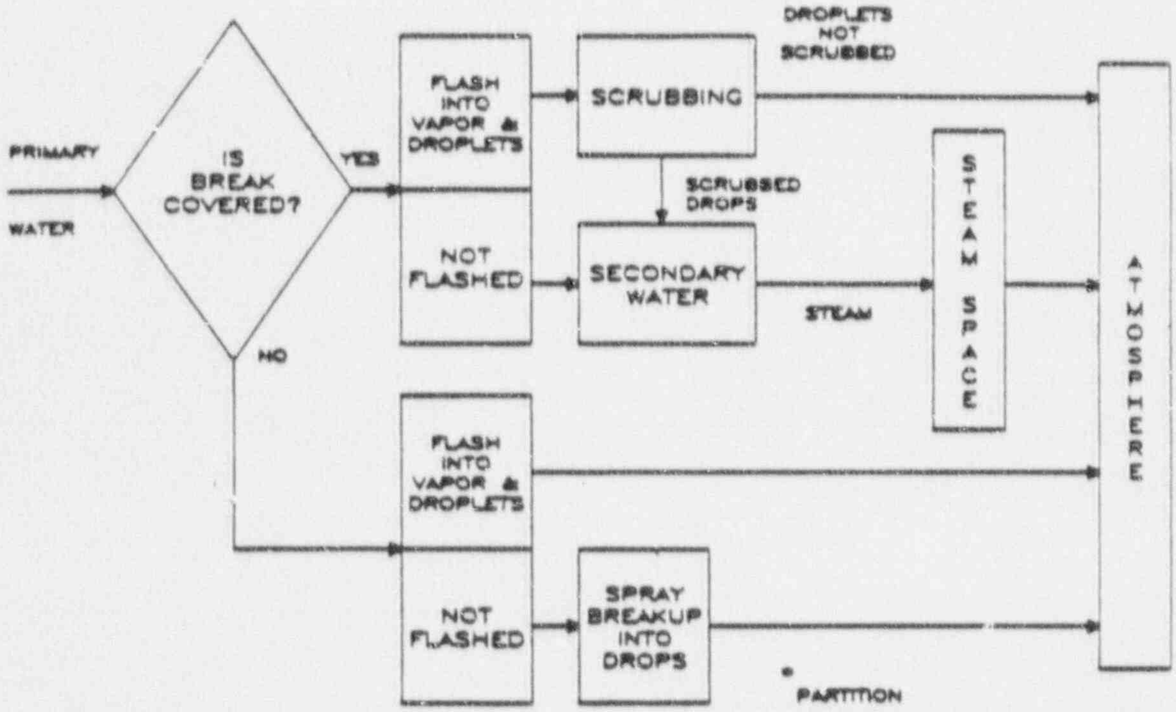
VOGTLE
ELECTRIC GENERATING PLANT
UNIT 1 AND UNIT 2

STEAM GENERATOR TUBE RUPTURE
INTACT SGs ATMOS. MASS RELEASE

FIGURE 15.6.3-11



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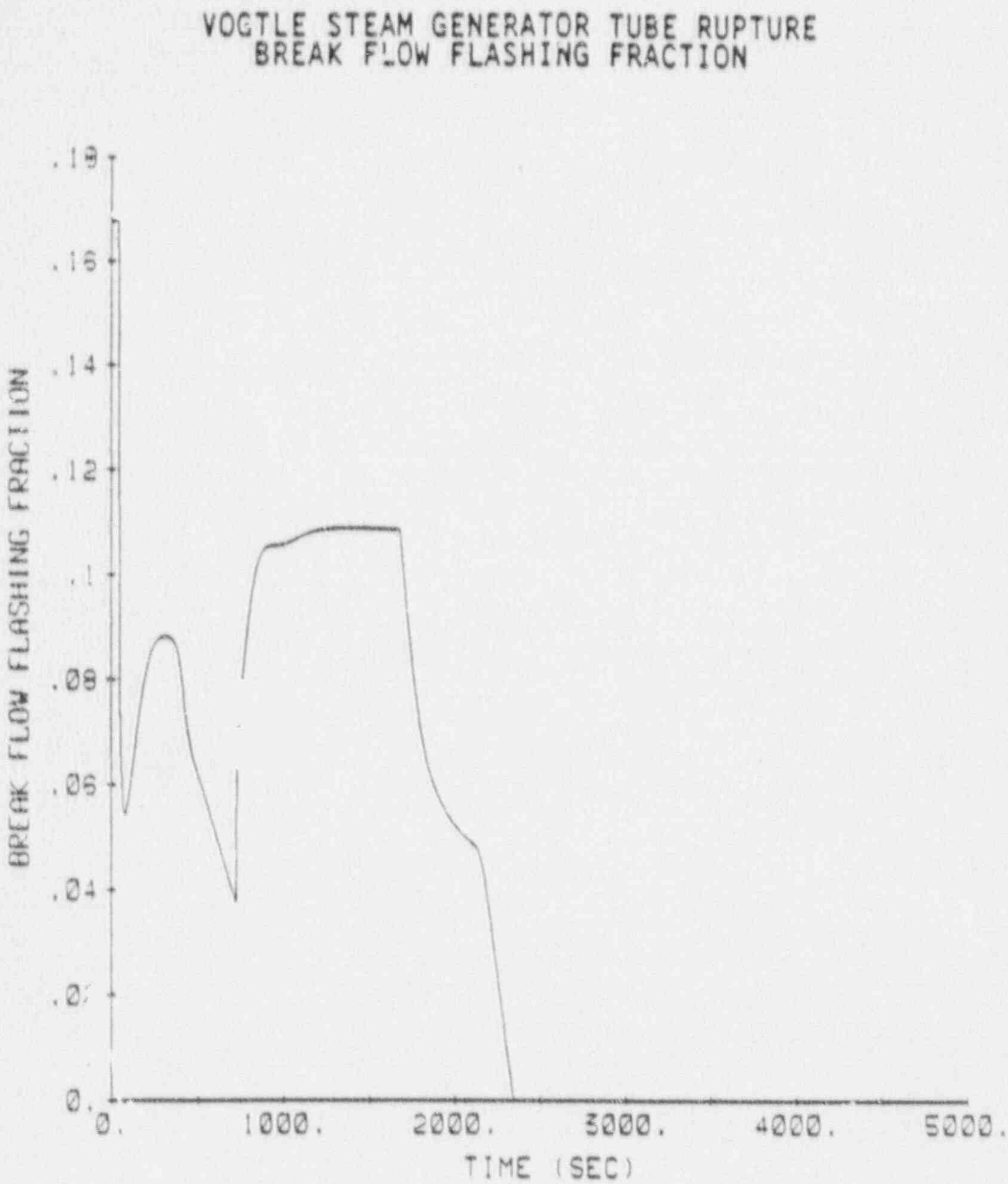
VOGTLE
ELECTRIC GENERATING PLANT
UNIT 1 AND UNIT 2

STEAM GENERATOR TUBE RUPTURE
IODINE TRANSPORT MODEL

FIGURE 15.6.3-12



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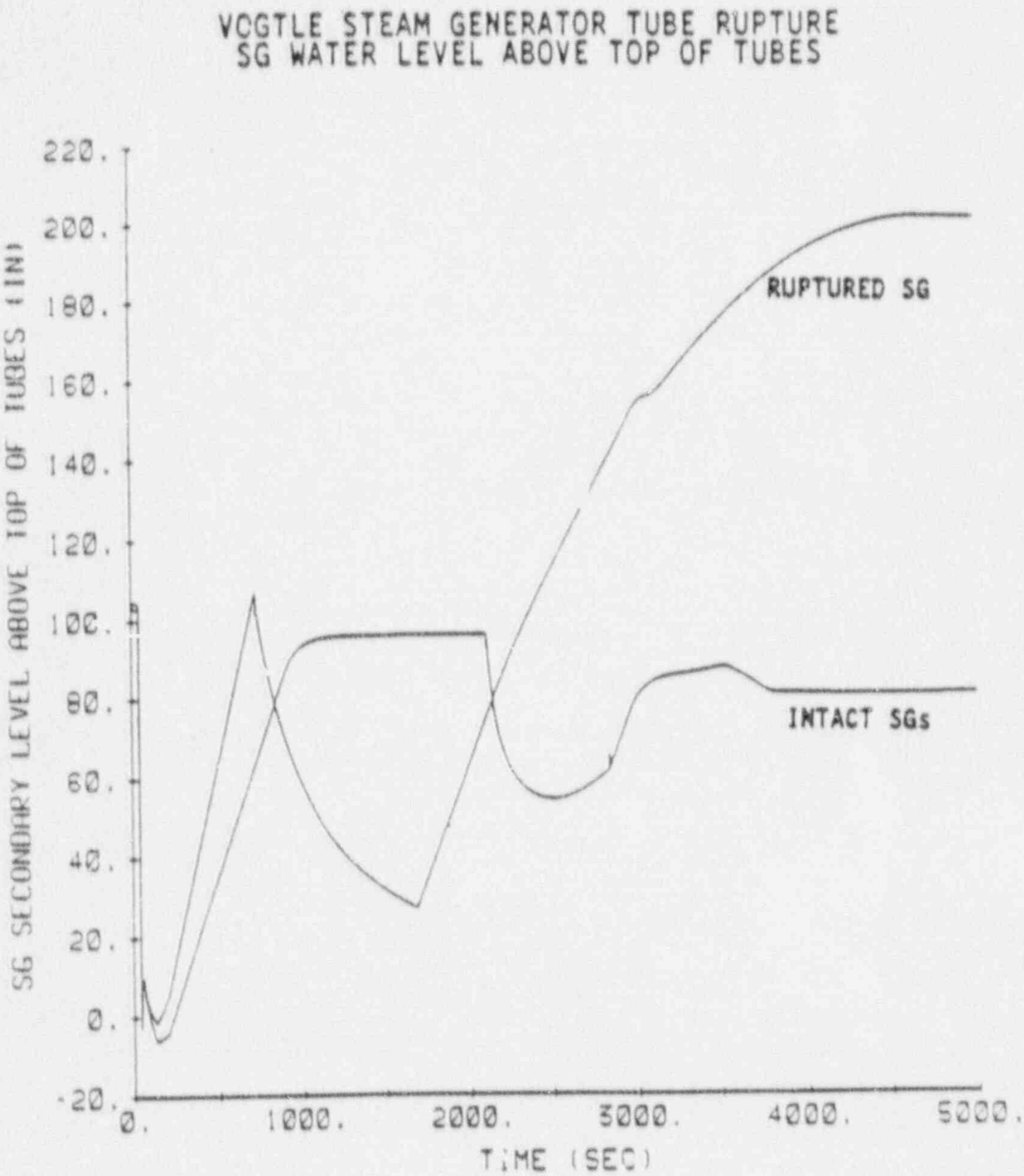
VOGTLE
ELECTRIC GENERATING PLANT
UNIT 1 AND UNIT 2

STEAM GENERATOR TUBE RUPTURE
BREAK FLOW FLASHING FRACTION

FIGURE 15.6.3-13



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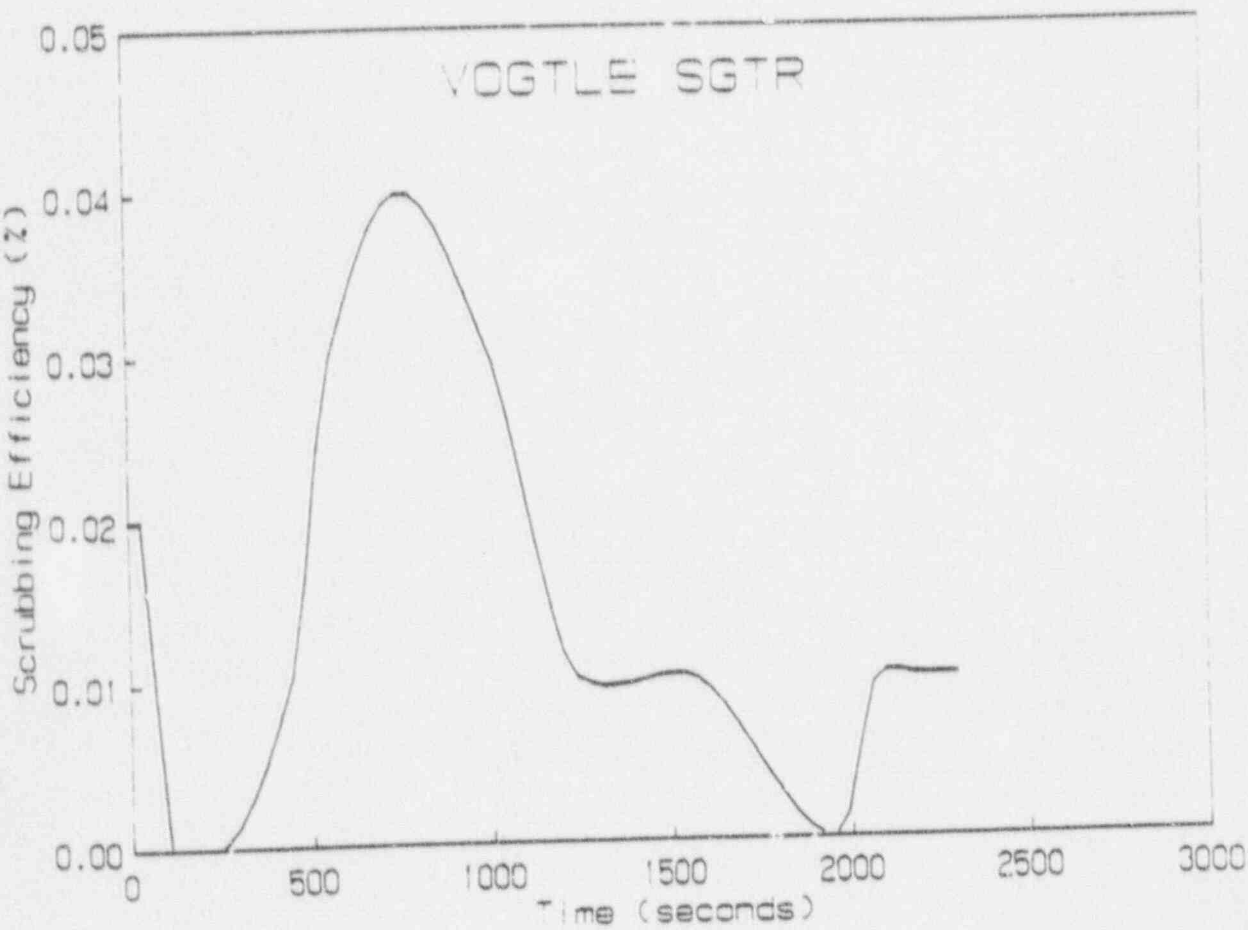
VOGTLE
ELECTRIC GENERATING PLANT
UNIT 1 AND UNIT 2

STEAM GENERATOR TUBE RUPTURE
SG WATER LEVEL ABOVE TOP OF TUBES

FIGURE 15.6.3-14



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VOGTLE
ELECTRIC GENERATING PLANT
UNIT 1 AND UNIT 2

STEAM GENERATOR TUBE RUPTURE
IODINE SCRUBBING EFFICIENCY

FIGURE 15.6.3-15

15.6.5 LOSS-OF COOLANT ACCIDENTS

15.6.5.1 Identification of Causes and Frequency Classification

A loss-of-coolant accident (LOCA) is the result of a pipe rupture of the reactor coolant system (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 a limiting fault, an American Nuclear Society (ANS) Condition IV event, in that it is not expected to occur during the lifetime of the plant but is postulated as a conservative design basis.

For large break LOCAs, the most limiting single failure is the one which produces the lowest containment pressure. The lowest containment pressure would be obtained only if all containment spray pumps and fan coolers operated subsequent to the postulated LOCA. Therefore, for the purposes of large break LOCA analyses, the most limiting single failure would be the loss of one RHR pump. However, the large break LOCA analyses assume both maximum containment safeguards (lowest containment pressure) and minimum ECCS safeguards (the loss of one complete train of ECCS components), which results in the minimum delivered ECCS flow available to the RCS.

A minor pipe break (small break), as considered in this subsection, is defined as a rupture of the reactor coolant pressure boundary with a total cross-sectional area less than 1.0 ft^2, in which the normally operating charging system flow is not sufficient to sustain pressurizer level and pressure. This is considered a Condition III event in that it is an infrequent fault that may occur during the life of the plant.

For small break LOCAs, the most limiting single active failure is of an emergency power train which results in loss of one complete train of ECCS components. The minimum delivered ECCS flow available to the RCS is based on this single failure.

The acceptance criteria for the LOCA described in 10 CFR 50.46 (Reference 1) are met as follows:

A. The calculated peak fuel element clad temperature is below the requirement of 2200°F.

B. 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.

C. The clad temperature transient is terminated at a time when the core geometry is still amenable to cooling. The localized cladding oxidation limits of 17 percent are not exceeded during or after quenching.

D. The core remains amenable to cooling during and after the break.

E. 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 a significant margin in emergency core cooling system (ECCS) performance following a LOCA. Reference 2 presents a recent study in regard to the probability of occurrence of RCS pipe ruptures.

In all cases, small breaks (less than 1.0 ft^2) yield results with more margin to the acceptance criteria limits than the limiting large break.

15.6.5.2 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:

A. Reactor trip and borated water injection complement 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 ECCS thermal analysis for boron content of the injection water. In addition, the insertion of control rods to shut down the reactor is neglected in the large-break ECCS thermal analysis.

B. Injection of borated water provides for heat transfer from the core and prevents excessive clad temperatures.

In the present Westinghouse design, the large break single failure is the loss of one RHR (low head) pump. This means that for a large break, credit could be taken for two high head charging pumps, two safety injection pumps, and one low head pump. However, the specific large break analysis for Vogtle has taken credit for one high head charging pump, one safety injection pump, and one [illegible] head pump.

The small break single failure is the loss of one ECCS Train. This means that for a small break credit could be taken for one high head charging pump, one safety injection pump, and one low head pump, which has been assumed for the specific small break analysis for Vogtle.

The current design for both small and large breaks assumes that at least one train is available for delivery of water to the RCS. This means that one pump in each subsystem delivers to the primary loop.

For the large break analysis, one ECCS train starts and delivers flow through the injection lines (one for each loop) with one branch injection line spilling to the containment backpressure. To minimize delivery to the reactor, the branch line chosen to spill is selected as the one with the minimum resistance.

For the small break analysis, one ECCS train starts and delivers flow through the injection lines (one for each loop) with one branch injection line spilling to the RCS backpressure, except for the charging pump injection lines, which has one injection line spilling to containment backpressure. To minimize delivery to the reactor, the branch line chosen to spill is selected as the one with the minimum resistance.

15.6.5.2.1 <u>Description of Large-Break LOCA Transient</u>

The sequence of events following a large-break LOCA are presented in Figure 15.6.5-1.

Before the break occurs, the unit is in an equilibrium condition; i.e., 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. Thereafter, the core heat transfer is based on local conditions with transition boiling and forced convection to steam as the major 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, 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 auxiliary feedwater system. The safety injection signal isolates the steam generators from normal feedwater flow and initiates emergency flow from the auxiliary feedwater system. The secondary flow aids in the reduction of RCS pressure.

When the RCS depressurizes to 615 psia, the accumulators begin to inject borated water into the reactor coolant loops. Since the loss of offsite power is assumed, the reactor coolant pumps 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 2300 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 bypassing of emergency core cooling water injected into the RCS are calculated not to be effective. At this time (end of bypass) refill of the reactor vessel lower plenum begins. Refill is complete when emergency core cooling water has filled the lower plenum of the reactor vessel which is bounded by the bottom of the fuel rods (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 later 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 safety injection pumps aid 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 long-term cooling. Core temperatures have been reduced to long-term, steady-state levels associated with dissipation of residual heat generation. After the water level of the refueling water storage tank reaches a minimum allowable value, coolant for long-term cooling of the core is obtained by semiautomatic switching to the cold leg recirculation mode of operation, in which spilled borated water is drawn from the containment emergency sumps by the residual heat removal pumps and returned to the RCS cold legs. The containment spray pumps are manually aligned to the containment emergency sumps and continue 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 to control the boric acid concentration in the reactor vessel. This time does not change due to the implementation of VANTAGE-5 fuel.

15.6.5.2.2 Description of Small-Break LOCA Transient

As contrasted with the large break, the blowdown phase of the small break occurs over a longer time period. Thus, for the small-break LOCA there are only three characteristic stages, i.e., a gradual blowdown with a decrease in water level and a partial core uncovery, core recovery, and long-term recirculation.

Should a small break occur, depressurization of the Reactor Coolant System causes fluid to flow into the loops from the pressurizer resulting in a pressure and level decrease in the pressurizer. Reactor trip occurs when the low pressurizer pressure trip setpoint is reached. During the earlier part of the small break transient, the effect of the break flow is not strong enough to overcome the flow maintained by the reactor coolant pumps through the core as they are coasting down following reactor trip. Upward flow through the core is maintained. However, the core flow is not sufficient to prevent a partial core uncovery. The ECCS is actuated when the appropriate setpoint is reached, and provides sufficient core flow to recover the core.

Before the break occurs the plant is in an equilibrium condition, i.e., 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 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, secondary system pressure increases and steam relief via the atmospheric relief and/or safety valves may occur. Makeup to the secondary side is automatically provided by the auxiliary feedwater pumps. The safety injection signal isolates normal feedwater flow by closing the main feedwater isolation valves and initiates auxiliary feedwater flow by starting the auxiliary feedwater pumps. The secondary flow aids in the reduction of Reactor Coolant System pressure.

When the RCS depressurizes to approximately 615 psia, the cold leg accumulators begin to inject borated water into the reactor coolant loops. However, the vessel mixture level starts to increase to cover the fuel with

ECCS pumped injection before the accumulator injection for most breaks. For all breaks, the accumulator injection provides enough water supply to bring the mixture level up to the upper plenum region where it is maintained. Due to the loss of offsite power assumption, the reactor coolant pumps are assumed to be tripped at the time of reactor trip during the accident and the effects of pump coastdown are included in the blowdown analyses.

15.6.5.3 Core and System Performance

15.6.5.3.1 Mathematical Model

The requirements of an acceptable ECCS evaluation model are presented in Appendix K of 10 CFR 50.

15.6.5.3.1.1 Large-Break LOCA Evaluation Model

The analysis of a large-break LOCA transient is divided into three phases: blowdown, refill, and 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.

The description of the various aspects of the LOCA analysis methodology is given in References 3 to 13 and 19 to 23. These documents describe the major phenomena modeled, the interfaces among the co puter co^es, and the features of the codes which ensure compliance with the Accep' .e Criteria. The SATAN-VI (Reference 4), WREFLOOD (Reference 5), COCO (Reference 6), BART (References 12, 13, and 23), BASH (Reference 10) and LOCBART (Reference 7 and 10) codes are used to assess the core heat transfer geometry and to determine if the core remains amenable to cooling throughout and subsequent to the blowdown, refill and reflood phases of the LOCA.

SATAN-VI is used to calculate 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 phase, these data are transferred to the WREFLOOD code. Also at the end-of-blowdown, the mass and energy release rates during blowdown are transferred to the COCO code for use in determination of the containment pressure response during the first phase of the LOCA. Additional SATAN-VI output data from the end-of-blowdown, including the core inlet flow rate and enthalpy, the core pressure, and the core power decay transient, are input to the LOCBART code.

WREFLOOD, using input from the SATAN-VI code, calculates the time to bottom of core recovery (BOC), RCS conditions at BOC and mass and energy release from the break during the reflood phase of the LOCA. Since the mass flow rate to the containment depends upon the core flooding rate and the local core pressure, which is a function of the Containment backpressure, the WREFLOOD and COCO codes are interactively linked. The BOC conditions calculated by WREFLOOD and the containment pressure transient calculated by COCO are used as input to the BASH code. Data from both the SATAN-VI code and the WREFLOOD code out to BOC are input to the LOCBART code which calculates core average conditions at BOC for use by the BASH code.

BASH provides a realistic thermal-hydraulic simulation of the reactor core and RCS during the reflood phase of a large break LOCA. Instantaneous values of the accumulator conditions and safety injection flow at the time of completion of lower plenum refill are provided to BASH by WREFLOOD. Figure 15.6.5-4 illustrates how BASH has been substituted for WREFLOOD in calculating transient values of core inlet flow, enthalpy, and pressure for the detailed fuel rod model, LOCBART. A detailed description of the BASH code is available in Reference 10. The BASH code provides a sophisticated treatment of steam/water flow phenomena in the reactor coolant system during core reflood. A dynamic interaction between core thermal-hydraulics and system behavior is expected, and experiments have shown this behavior. The BART code has been coupled with a loop model to form the BASH code and BART provides the entrainment rate for a given flooding rate. The loop model determines the loop flows and pressure drops in response to the calculated core exit flow determined by BART. The updated inlet flow calculated by the loop model is used by BART to calculate a new entrainment rate to be fed back to the loop code. This process of transferring data between BART, the loop code and back to BART forms the calculational process for analyzing the reflood transient. This coupling of the BART code with a loop code produces a dynamic flooding transient, which reflects the close coupling between core thermal-hydraulics and loop behavior.

The cladding heat-up transient is calculated by LOCBART which is a combination of the LOCTA code with BART. A more detailed description of the LOCBART code can be found in References 7 and 10. During reflood, the LOCBART code provides a significant improvement in the prediction of fuel rod behavior. In LOCBART the empirical FLECHT correlation has been replaced by the BART code. BART employs rigorous mechanistic models to generate heat transfer coefficients appropriate to the actual flow and heat transfer regimes experienced by the fuel rods.

The analysis in this section was performed with the upper head fluid temperature equal to the reactor coolant system cold leg fluid temperature, achieved by increasing the upper head cooling flow (Reference 17). Modeling features necessary to account for the reactor barrel-baffle region and the reactor fuel assembly thimbles were included in this analysis as presented in References 10, 20, and 23. The impact of a no single failure assumption for the ECCS was examined by re-analyzing the most limiting break with maximum ECCS flows as required by Reference 10.

15.6.5.3.1.2 Small-Break LOCA Evaluation Model

For loss-of-coolant accidents due to small breaks less than 1 square foot, the NOTRUMP, Reference 14, computer code 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 consisting of a number of advanced features. Among these features are the 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 system (ECCS) evaluation model was developed to determine the RCS response to design basis small break LOCAs and to address the NRC concerns expressed in NUREG-0611, "Generic Evaluation of Feedwater Transients and Small Break Loss-of-Coolant Accidents in Westinghouse-Designed Operating Plants."

In NOTRUMP, the RCS is nodalized into volumes interconnected by flowpaths. The broken loop is modeled explicitly, with the intact loops lumped into a second loop. The transient behavior of the system is determined from the governing conservation equations of mass, energy, and momentum applied throughout the system. A detailed description of the NOTRUMP code is provided in References 14 and 15.

The use of NOTRUMP in the analysis involves, among other things, the representation of the reactor core as heated control volumes with an associated bubble rise model to permit a transient mixture height calculation. The multinode capability of the program enables an explicit and detailed spatial representation of various system components. In particular, it enables a proper calculation of the behavior of the loop seal during a loss-of-coolant accident.

Clad thermal analyses are performed with a small break version of the LOCTA-IV code, Reference 7, which uses the RCS pressure, fuel rod power history, steam flow past the uncovered part of the core, and mixture height history from the NOTRUMP hydraulic calculations as input.

Figure 15.6.5-2 gives the safety injection flowrate for the small-break analysis.

Figure 15.6.5-3 presents the hot rod power shape utilized to perform the small-break analysis presented here. This power shape was chosen because it provides an appropriate distribution of power versus core height; also, local power is maximized in the upper regions of the reactor core (9 to 12 ft).

This power shape is skewed to the top of the core with the peak local power occurring at the higher core elevations.

This is limiting for the small-break analysis because of the core uncovery process for small breaks. As the core uncovers, the cladding in the upper elevation of the core heats up and is sensitive to the local power at that elevation. The cladding temperatures in the lower elevation of the core, below the two-phase mixture height, remains low. The peak clad temperature occurs above 9 ft.

Schematic representations of the computer code interface are given in Figure 15.6.5-5 for the Small Break LOCA Analysis.

The small-break analysis was performed with the approved NOTRUMP version of the Westinghouse ECCS evaluation model (References 7, 14, 15 and 16).

15.6.5.3.2 Input Parameters and Initial Conditions

Table 15.6.5-1 lists important input parameters and initial conditions used in the analysis.

The analysis was performed with the upper head fluid temperature equal to the RCS cold leg fluid temperature, achieved by increasing the upper head cooling flow (Reference 17).

The initial steady-state fuel pellet temperature and the fuel rod internal pressure used in the LOCA analysis has been generated with the PAD 3.4 Fuel Rod Design Code (Reference 18) which has been approved by the Nuclear Regulatory Commission.

A break spectrum sensitivity study for the large break LOCA is presented in Reference 19. For the small break LOCA a break spectrum sensitivity is presented in References 15 and 16. In addition, the analyses utilized the upflow barrel-baffle methodology described in Reference 20.

The bases used to select the numerical values that are input parameters to the analyses have been conservatively determined from extensive sensitivity studies (References 19, 21, and ... In addition, the requirements of Appendix K regarding specific mode ...eatures were met by selecting models which provide significant overall conservatism in the analysis. The assumptions made ...tain to the conditions of the reactor and associated safety system equ... at the time that the LOCA occurs and include such items as the core p...king factors, the containment pressure, and the performance of the ECCS system. Decay heat generated throughout the transient is also conservatively calculated.

15.6.5.3.3 Results

15.6.5.3.3.1 Large Break Results

Based on the results of the LOCA sensitivity studies, References 19, 21 and 22, 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 Table 15.6.5-2.

The mass and energy release data for the break resulting in the highest calculated peak clad temperature are discussed in Section 6.2.1.5.

Figures 15.6.5-6 through 15.6.5-36D present the parameters of principal interest from the large-break ECCS analyses. For all cases analyzed transients of the following parameters are presented:

- Hot spot clad temperature.
- Coolant pressure in the reactor core.
- Water level in the core and downcomer during reflood.
- Core reflooding rate.
- Thermal power during blowdown.
- Containment pressure.
- Core flow during blowdown (inlet and outlet).

- Core heat transfer coefficients.
- Hot spot fluid temperature.
- Mass released to containment during blowdown.
- Energy released to containment during blowdown.
- Fluid quality in the hot assembly during blowdown.
- Mass velocity during blowdown.
- Accumulator water flowrate during blowdown.
- Pumped safety injection water flow rate.

The peak cladding temperature (PCT) calculated for the large break LOCA to account for the VANTAGE-5 fuel is 2037°F. Added to the calculated PCT are 1) a 10°F increase due to containment mini-purge isolation, and 2) a 11°F increase due to a 6°F uncertainty band for the RCS operating average temperature. This brings the resultant PCT to 2058°F for a full core of VANTAGE 5 fuel, which is less than the acceptance criteria limit of 2200°F. During the transition cycles, the effect is conservatively evaluated to be at most a 50°F increase for the calculated PCT. This yields a PCT of 2108°F, which is still below the acceptance limit of 2200°F. The maximum local metal-water reaction is 5.62 percent, which is 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; and 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.

15.6.5.3.3.2 Small-Break Results

As noted previously, the calculated peak clad temperature resulting from a small-break LOCA is less than that calculated for the limiting large break. Based on the results of the LOCA sensitivity studies (References 16 and 19), the limiting small break was found to be less than a 10-in.-diameter rupture of the RCS cold leg. Therefore, a range of small-break analyses are presented which established the limiting small break. The results of these analyses are summarized in Table 15.6.5-3.

Figures 15.6.5-37 through 15.6.5-48 present the principal parameters of interest for the small-break ECCS analyses. For all cases analyzed the following transient parameters are presented:

- RCS pressure.
- Core mixture height.
- Hot spot clad temperature.

For the limiting break analyzed, the following additional transient parameters are presented:

- Core steam flowrate.
- Core heat transfer coefficient.
- Hot spot fluid temperature.

The peak cladding temperature (PCT) calculated for the small break LOCA to account for the VANTAGE-5 fuel is 2037°F. Added to the calculated PCT is 1) a 15°F increase due to steam generator level instrumentation modifications, and 2) a 4°F increase due to a 6°F uncertainty band for the RCS operating average temperature. This brings the resultant PCT to 2056°F, which is less than the acceptance criteria limit of 2200°F. The maximum local metal-water reaction is 7.74 percent, which is below the acceptance criteria limit of 17 percent. The total core metal-water reaction is less than 0.3 percent for all breaks, as compared with the 1 percent acceptance criteria. These results are below all acceptance criteria limits of 10 CFR 50.46.

15.6.5.4 REFERENCES

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TABLE 15.6.5-1

INPUT PARAMETERS USED IN THE ECCS ANALYSIS

Licensed core power (MWt) [a]	3565	
Peak linear power, includes 102% factor (kW/ft)	Large Break Small Break	14.511 at 6.0 ft. 14.331 at 9.5 ft.
Total peaking factor (FQ)	Large Break Small Break	2.5 2.58
$F_{\Delta H}$	Large Break Small Break	1.65 1.70
Power shape (small/large break)	Fig. 15.6.5-3/ Chopped Cosine, F_Z=1.5151	
Fuel assembly array	17 x 17 VANTAGE 5	
Accumulator water volume, nominal (ft^3/accumulator)	900	
Accumulator tank volume, nominal (ft^3/accumulator)	1350	
Accumulator gas pressure, minimum (psia)	615	
Safety injection pumped flow (small/large break)	Figure 15.6.5-2A/ Figure 15.6.5-36A + 36B	
Containment parameters	Paragraph 6.2.1.5	
Initial loop flow (lb/s)	9712.6	
Vessel inlet temperature (°F)	555.0	
Vessel outlet temperature (°F)	619.59	
Reactor coolant pressure (psia)	2300.0	
Steam pressure (psia)	935.91	
Steam generator tube plugging level (%)	10.0	

a. Two percent is added to this power for calorimetric uncertainty.

TABLE 15.6.5-2 (SHEET 1 OF 2)
LARGE-BREAK RESULTS

Calculation

Number of safety injection pumps operating	1CCP, 1HHSI, 1RHR
Steam generator tube plugging level (%)	10

Results	DECLG C_D=0.8	DECLG C_D=0.6	DECLG Maximum Safety Injection C_D=0.6	DECLG C_D=0.4
Peak clad temperature (°F)	1736	2058*	1830	1623
Peak clad location (ft)	8.75	8.75	6.25	7.0
Max local Zr/H2O reaction (%)	3.70	5.62	2.88	1.97
Local Zr/H_2O location (ft)	5.5	8.75	7.0	7.0
Total Zr/H_2O reaction (%)	<0.3	<0.3	<0.3	<0.3
Hot rod burst time (s)	41.55	42.73	42.73	90.15
Hot rod burst location	5.5	6.0	6.0	7.0

*Includes,
1) a 10°F increase due to containment mini-purge isolation
2) a 11°F increase due to a 6°F uncertainty band for the RCS operating average temperature
3) an additional 50°F should be added during transition cycles of VANTAGE-5

TABLE 15.6.5-2 (SHEET 2 OF 2)

Results (Seconds)	DECLG C_D=0.8	DECLG C_D=0.6	DECLG Maximum Safety Injection C_D=0.6	DECLG C_D=0.6
Start	0.0	0.0	0.0	0.0
Reactor trip signal	0.436	0.442	0.442	0.454
Safety injection signal	1.50	1.72	1.72	2.14
Accumulator injection (CL)	12.10	14.80	14.8	19.80
End of Bypass	25.513	30.839	30.839	37.622
End of Blowdown	25.516	30.839	30.839	37.622
Pump Injection	36.50	41.72	41.72	42.14
Bottom of core recovery	37.892	43.890	43.72	51.196
Accumulator empty	48.133	52.736	53.497	58.196
PCT Time	150.75	192.79	55.70	90.15

TABLE 15.6.5-3

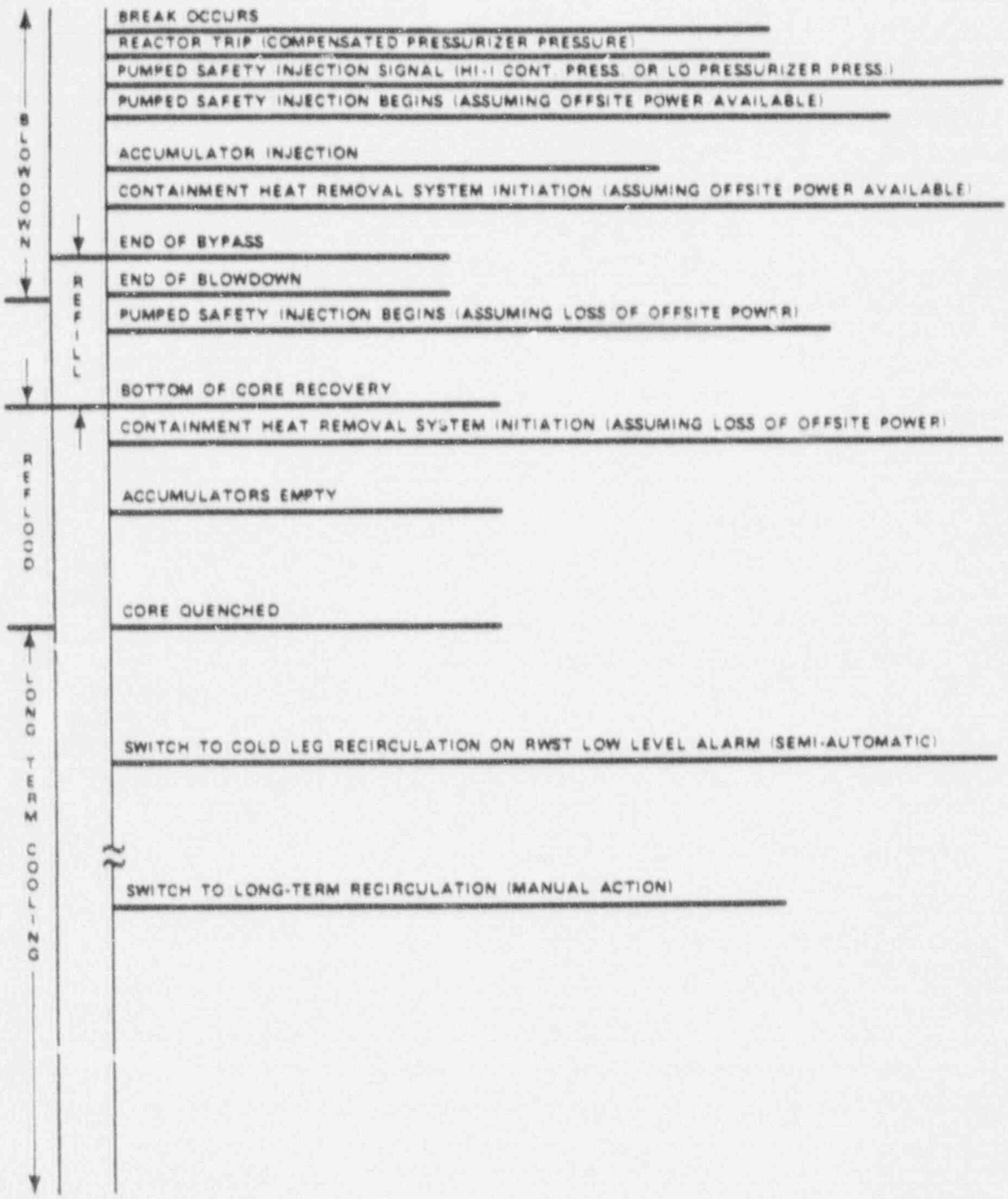
SMALL-BREAK RESULTS

	2 in.	3 in	4 in.
Peak clad temperature (°F)	1088	2056*	1236
Peak clad location (ft)	11.5	12.0	11.5
Local Zr/H_2O reaction maximum (%)	.083	7.74	.062
Local Zr/H_2O location (ft)	11.5	12.0	11.5
Total Zr/H_2O reaction (%)	<0.3	<0.3	<0.3
Hot rod burst time (s)	N/A	N/A	N/A
Hot rod burst location (ft)	N/A	N/A	N/A

*Includes:

1) a 15°F increase due to SG level instrumentation modification
2) a 4°F increase due to a 6°F uncertainty band for the RCS operating average temperature.

Sequence of Events (Seconds)	2 in.	3 in.	4 in.
Start	0.0	0.0	0.0
Reactor trip signal	41.84	16.976	9.704
Safety Injection signal	74.202	32.289	23.300
Safety Injection Begins	114.202	72.289	63.300
Loop Seal Venting Begins	1234.50	486.09	278.70
Top of core uncovered	2205.0	914.5	763.7
Accumulator injection begins	N/A	1772.5	1006.4
Peak clad temperature occurs	2552.0	1579.7	1027.0
Top of core covered	3977.0	2866.3	1628.9



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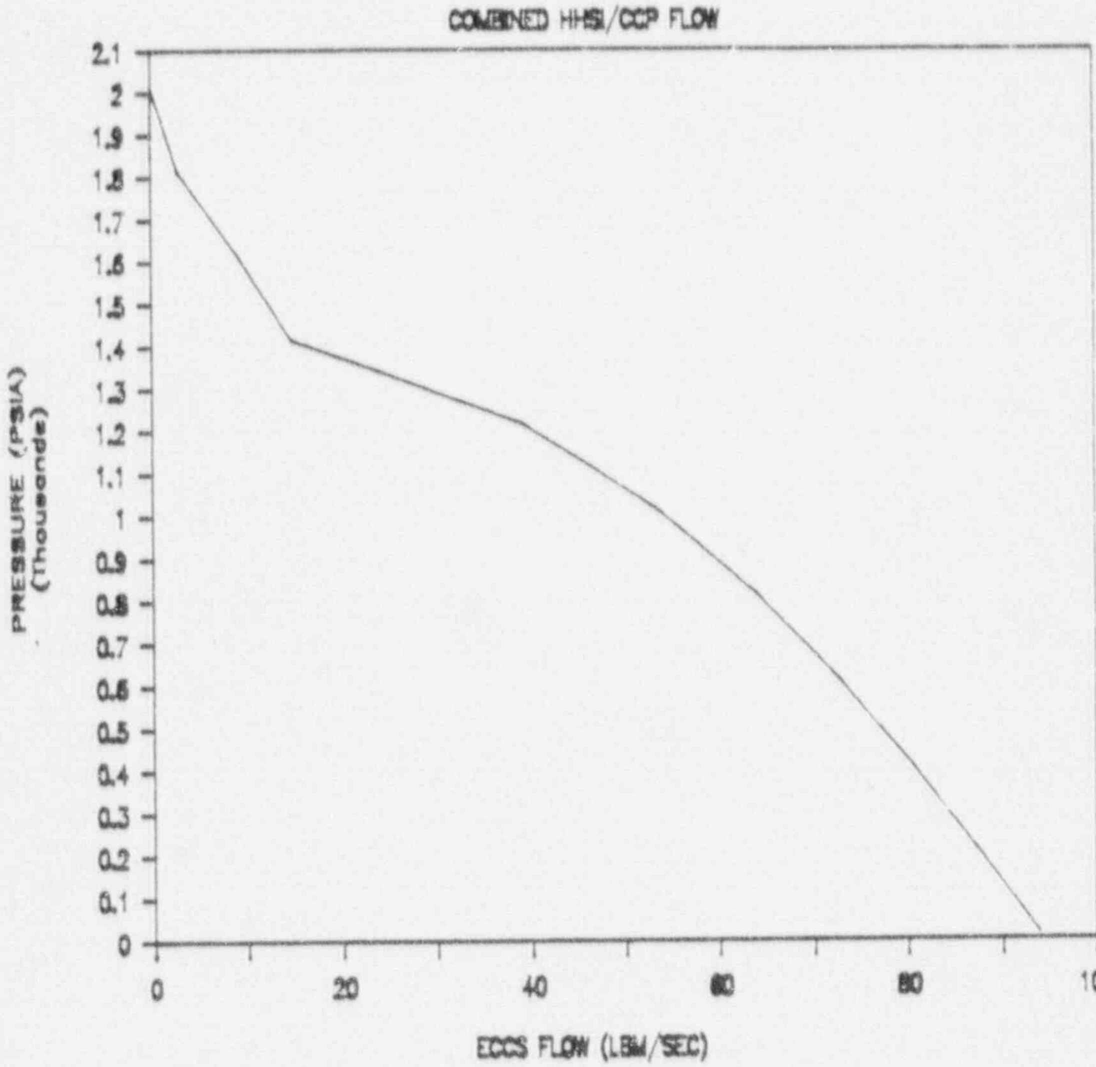
VOGTLE
ELECTRIC GENERATING PLANT
UNIT 1 AND UNIT 2

SEQUENCE OF EVENTS FOR
LARGE BREAK LOCA ANALYSIS

FIGURE 15.6.5-1



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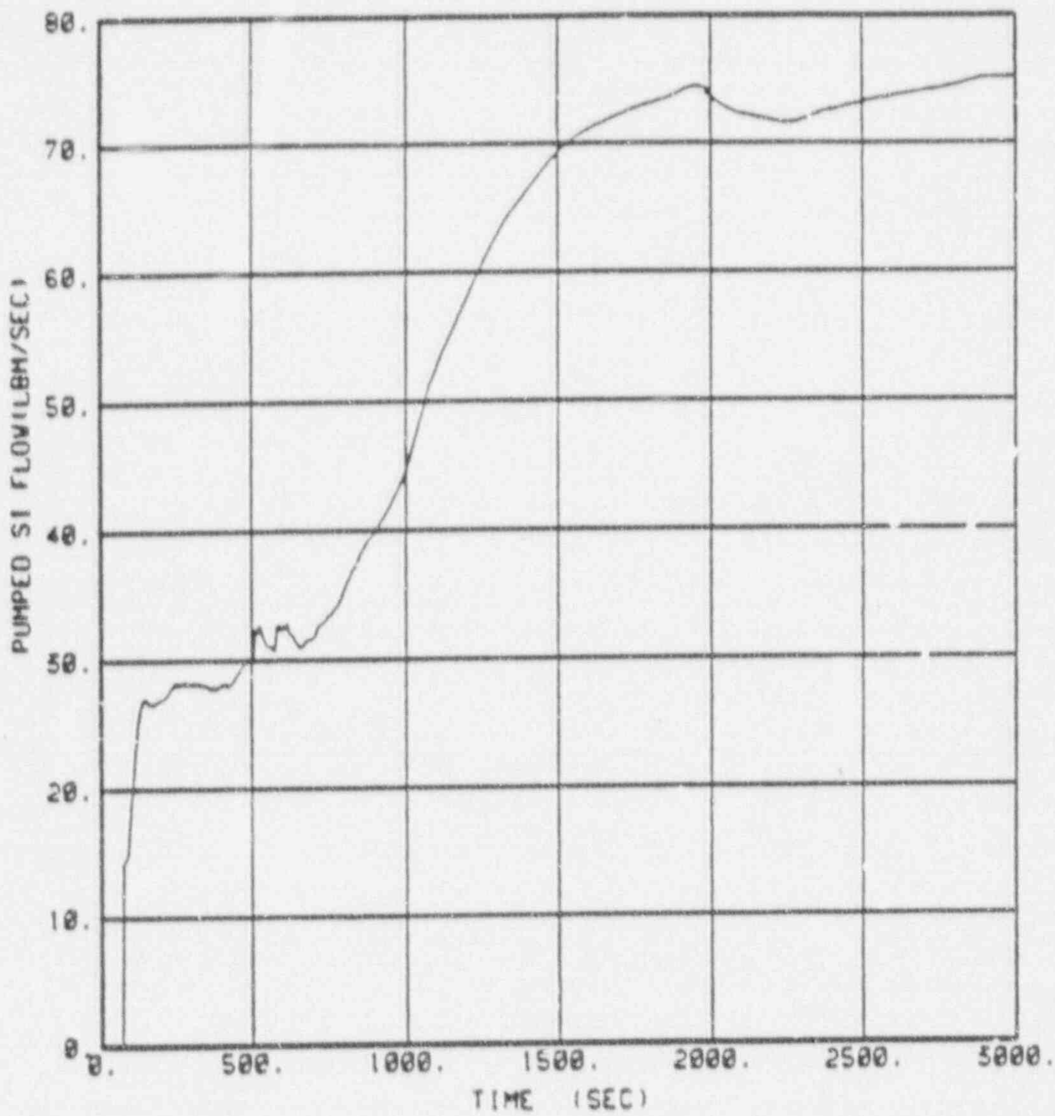
VOGTLE
ELECTRIC GENERATING PLANT
UNIT 1 AND UNIT 2

SMALL BREAK SAFFTY INJECTION FLOWRATE

FIGURE 15.6.5-2A

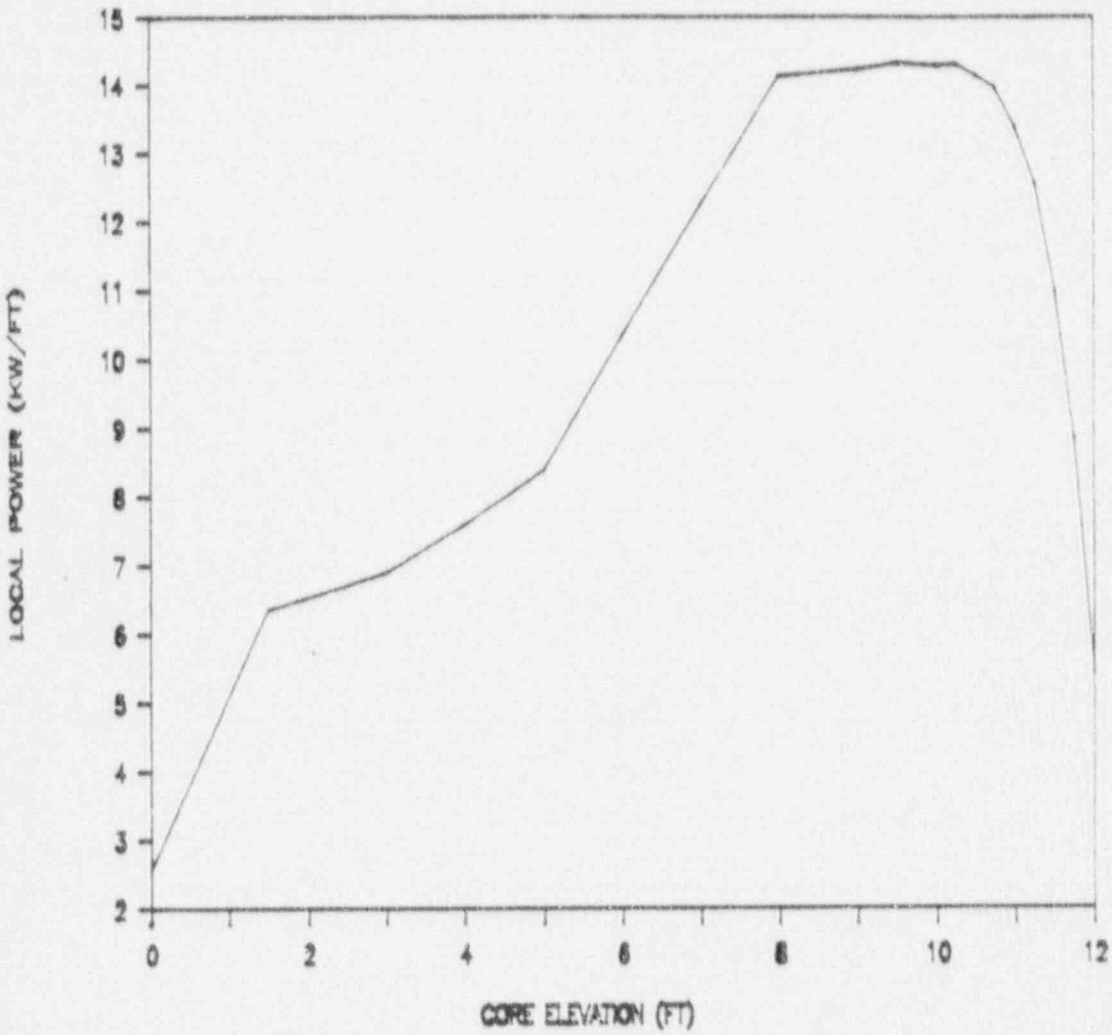


Georgia Power

VOGTLE
ELECTRIC GENERATING PLANT
UNIT 1 AND UNIT 2

ECCS PUMPED SAFETY INJECTION
DURING 3-INCH SMALL BREAK

FIGURE 15.6.5-2B

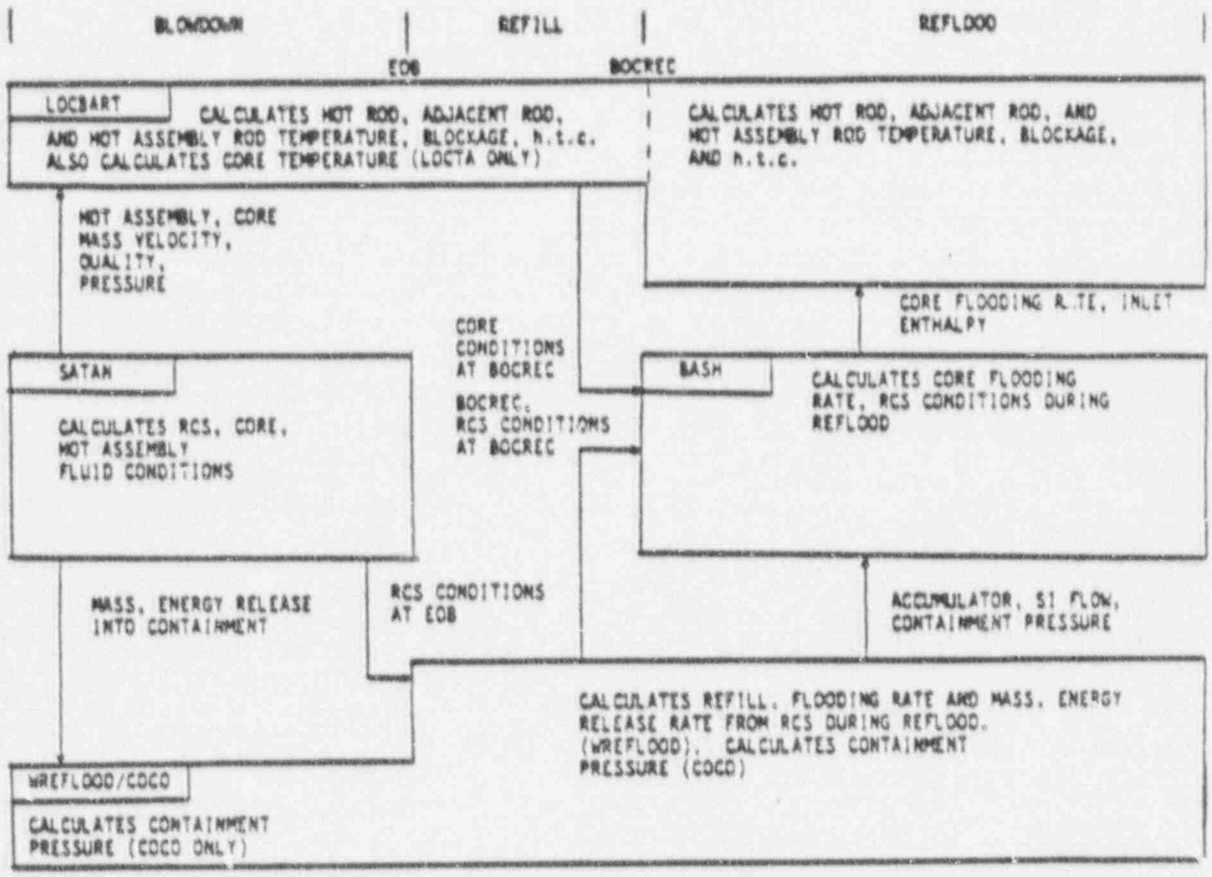


Georgia Power

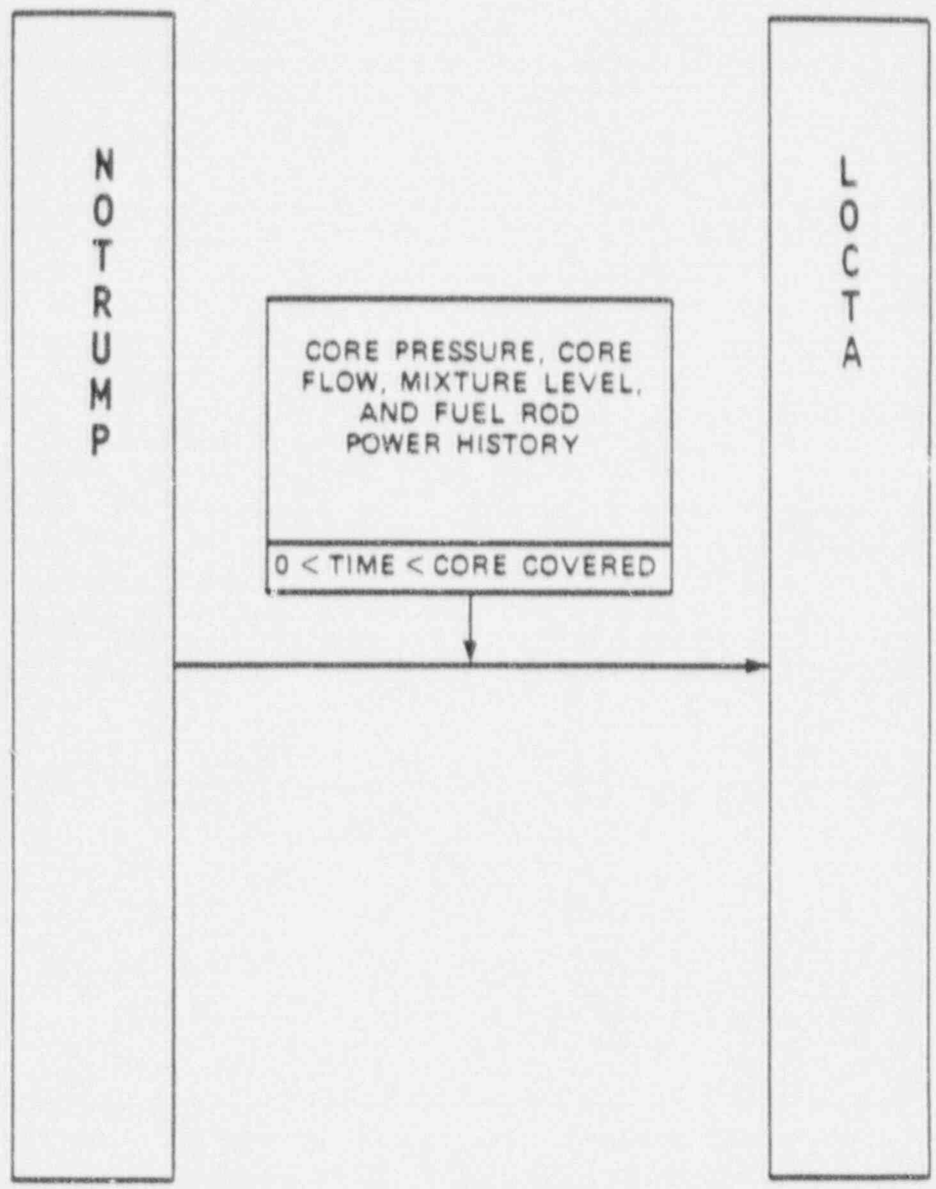
VOGTLE
ELECTRIC GENERATING PLANT
UNIT 1 AND UNIT 2

SMALL BREAK HOT ROD POWER SHAPE

FIGURE 15.6.5-3



Georgia Power	VOGTLE ELECTRIC GENERATING PLANT UNIT 1 AND UNIT 2	CODE INTERFACE DESCRIPTION FOR LARGE BREAK MODEL FIGURE 15.6.5-4

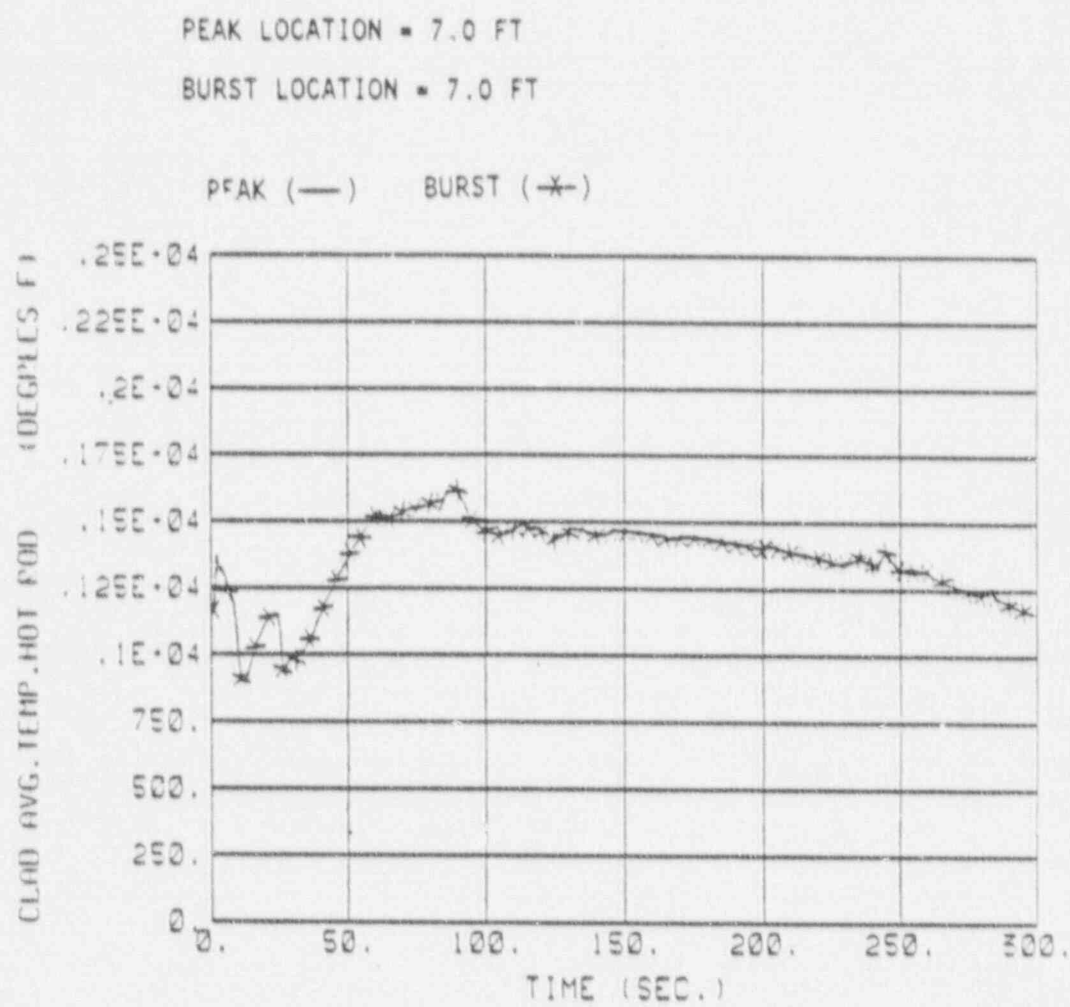


Georgia Power

VOGTLE
ELECTRIC GENERATING PLANT
UNIT 1 AND UNIT 2

CODE INTERFACE DESCRIPTION
FOR SMALL BREAK MODEL

FIGURE 15.6.5-5

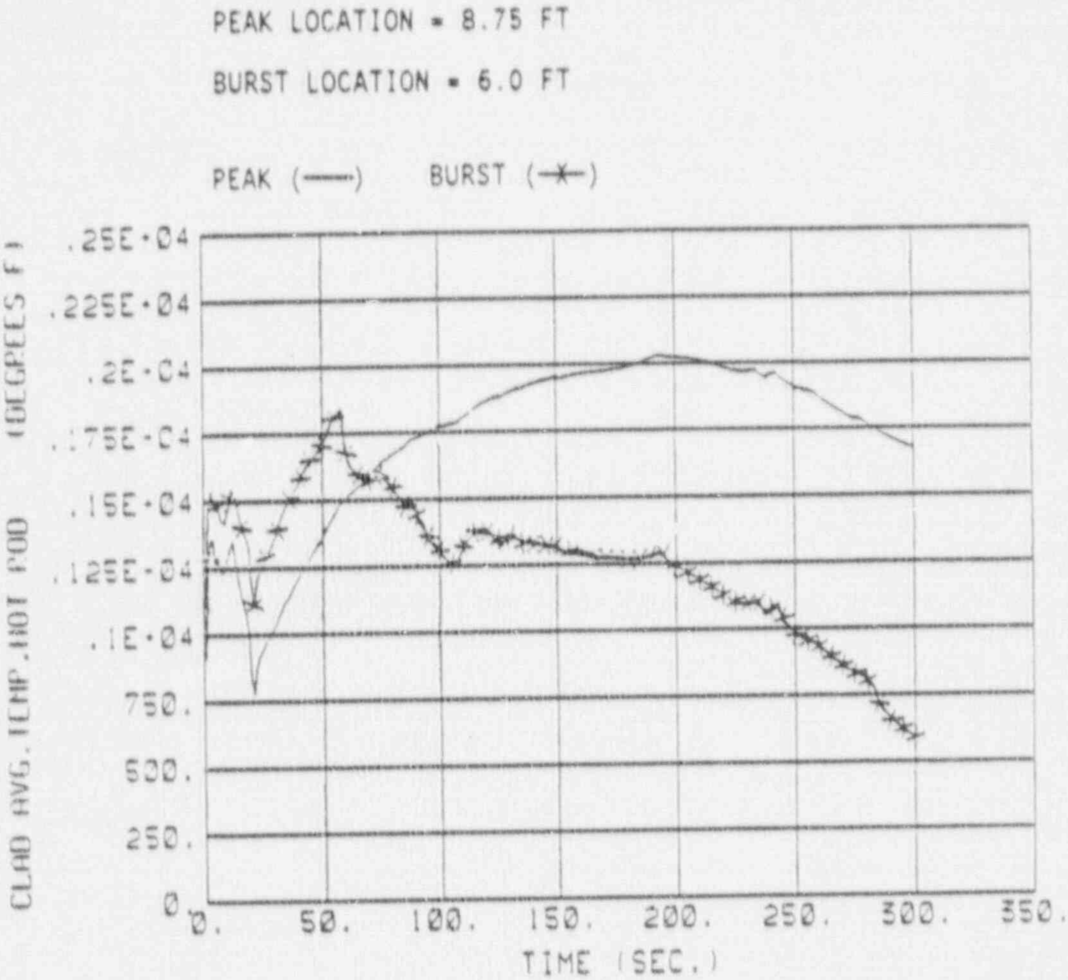


Georgia Power

VOGTLE
ELECTRIC GENERATING PLANT
UNIT 1 AND UNIT 2

PEAK CLAD TEMPERATURE
DECLG (CD = 0.4)

FIGURE 15.6.5-6

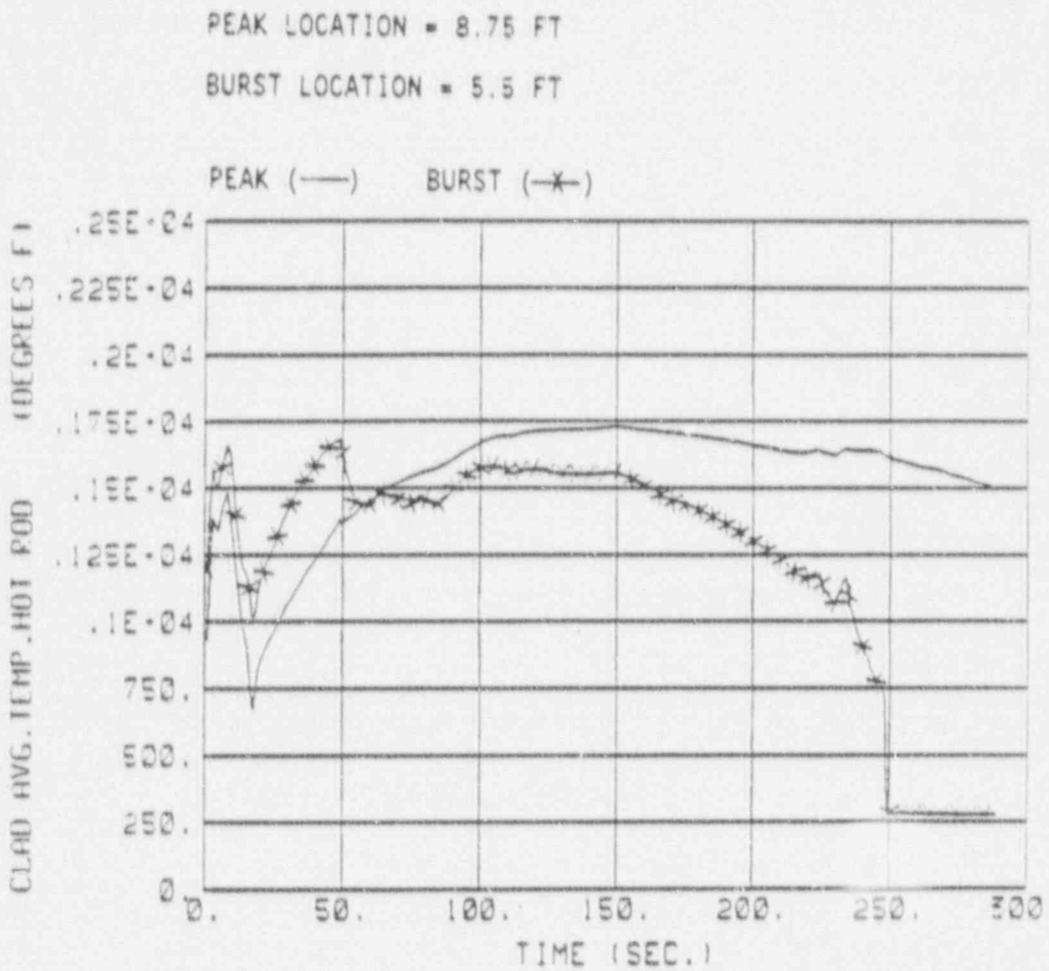


Georgia Power

VOGTLE
ELECTRIC GENERATING PLANT
UNIT 1 AND UNIT 2

PEAK CLAD TEMPERATURE
DECLG (CD = 0.6)

FIGURE 15.6.5-7

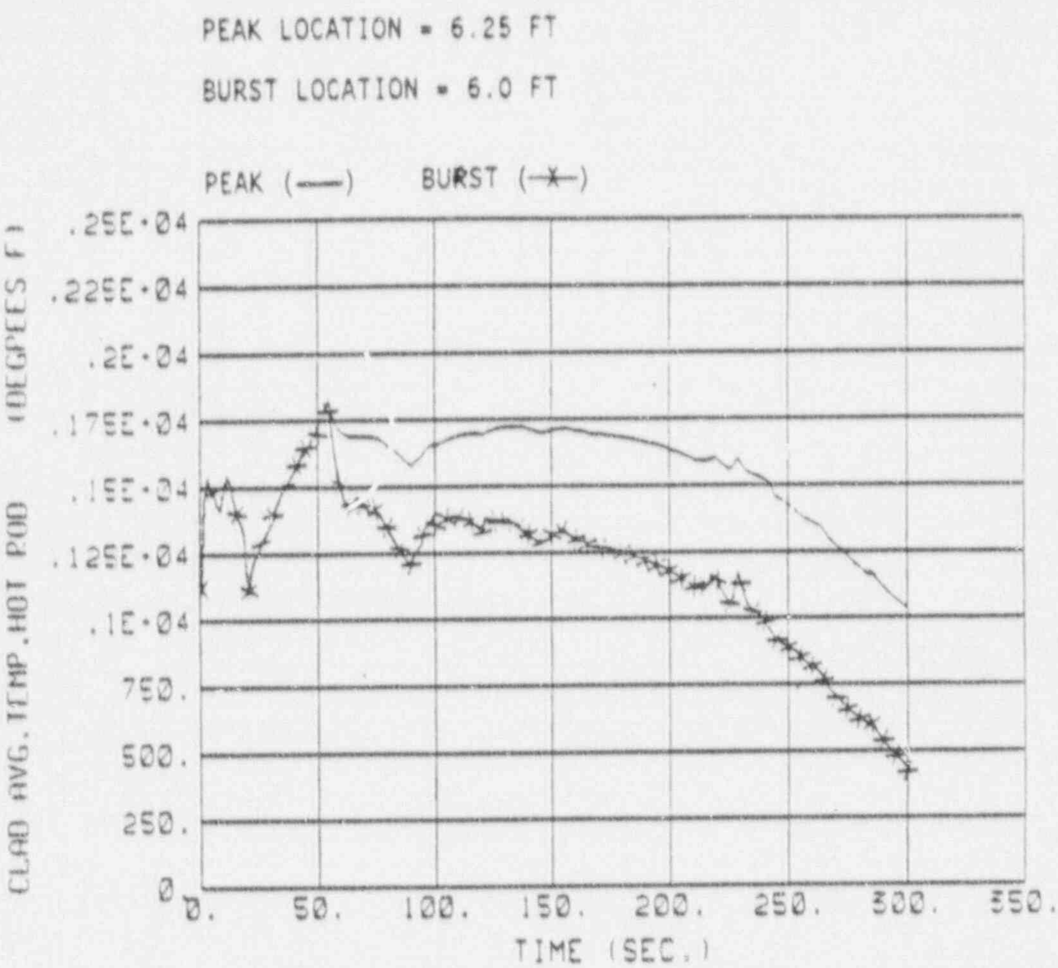


Georgia Power

ALVIN W. VOGTLE
ELECTRIC GENERATING PLANT
UNIT 1 AND UNIT 2

PEAK CLAD TEMPERATURE
DECLG (CD = 0.8)

FIGURE 15.6.5-8

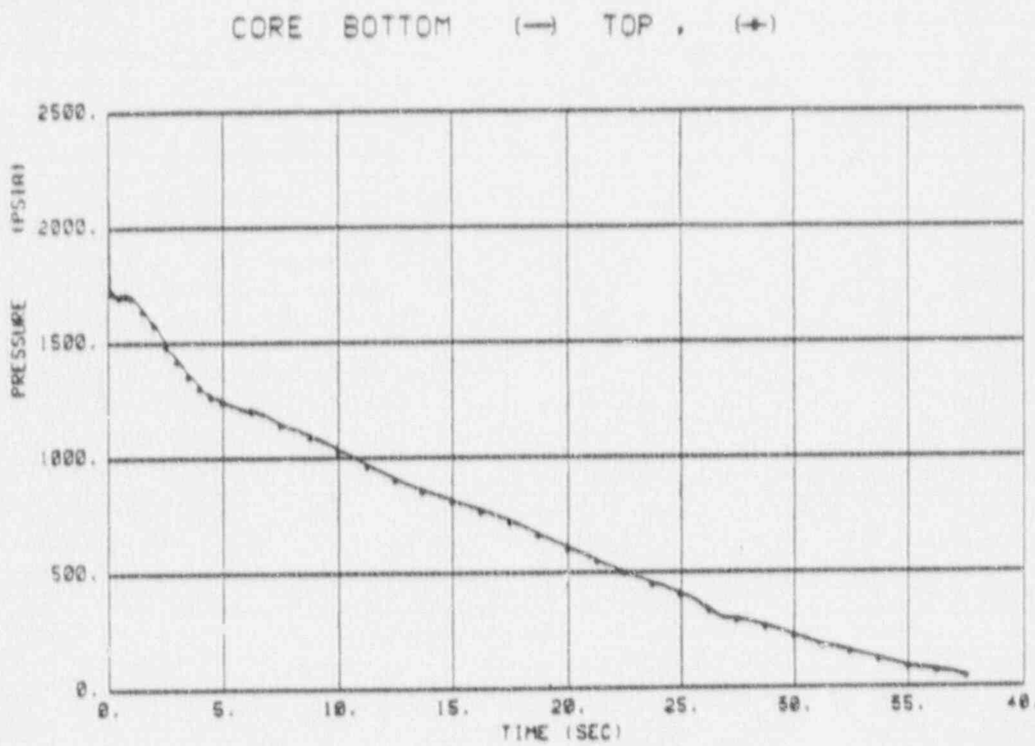


Georgia Power

VOGTLE
ELECTRIC GENERATING PLANT
UNIT 1 AND UNIT 2

PEAK CLAD TEMPERATURE
DECLG (CD = 0.6 MAXIMUM
SAFETY INJECTION)

FIGURE 15.6.5-9

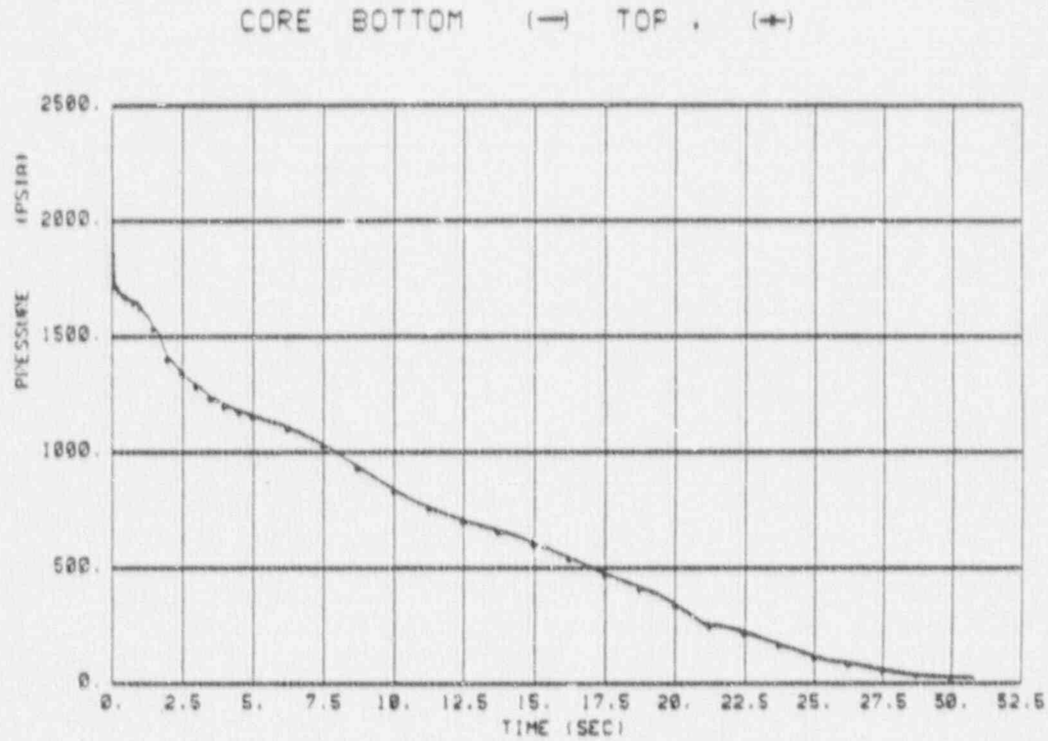


Georgia Power

VOGTLE
ELECTRIC GENERATING PLANT
UNIT 1 AND UNIT 2

CORE PRESSURE
DECLG (CD = 0.4)

FIGURE 15.6.5-10

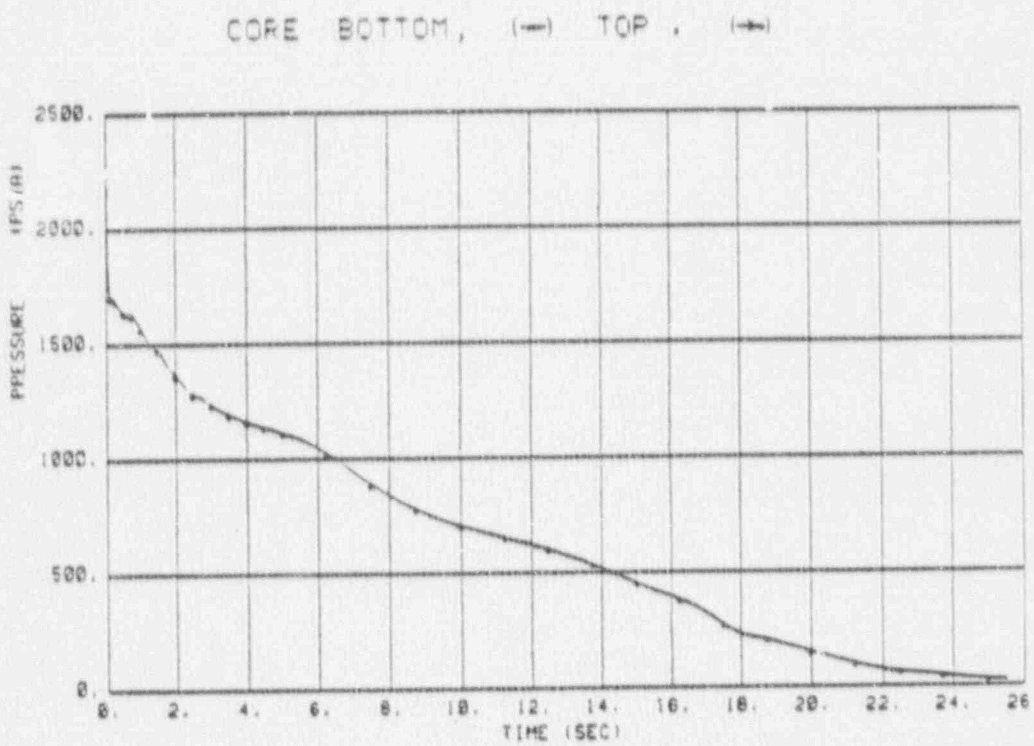


Georgia Power

VOGTLE
ELECTRIC GENERATING PLANT
UNIT 1 AND UNIT 2

CORE PRESSURE
DECLG (CD = 0.6)

FIGURE 15.6.5-11

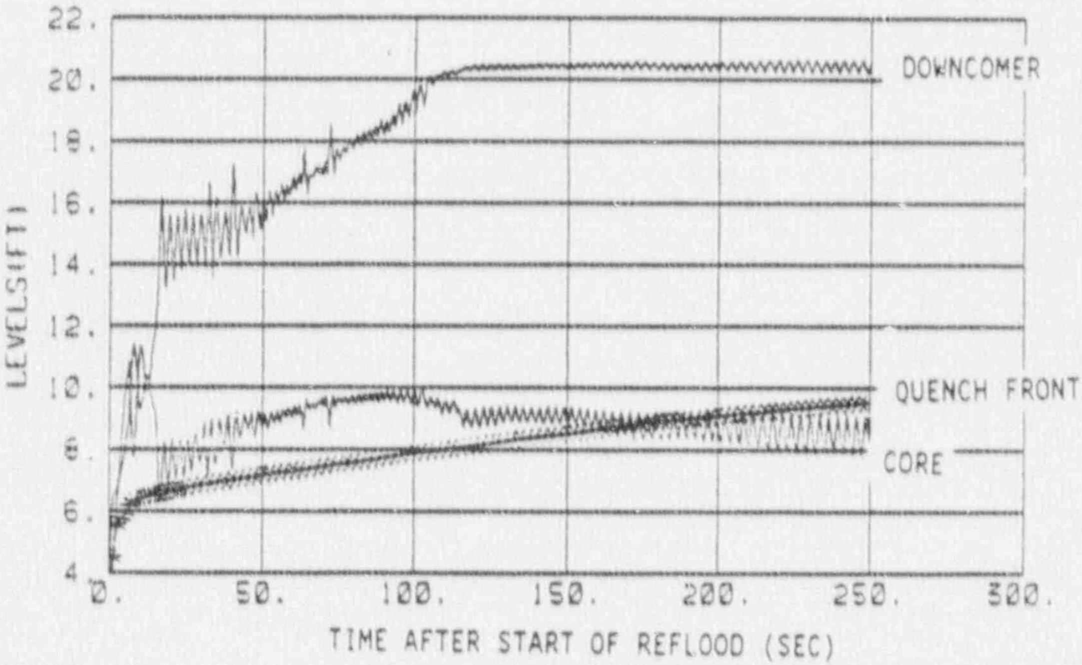


Georgia Power

VOGTLE
ELECTRIC GENERATING PLANT
UNIT 1 AND UNIT 2

CORE PRESSURE
DECLG (CD = 0.8)

FIGURE 15.6.5-12

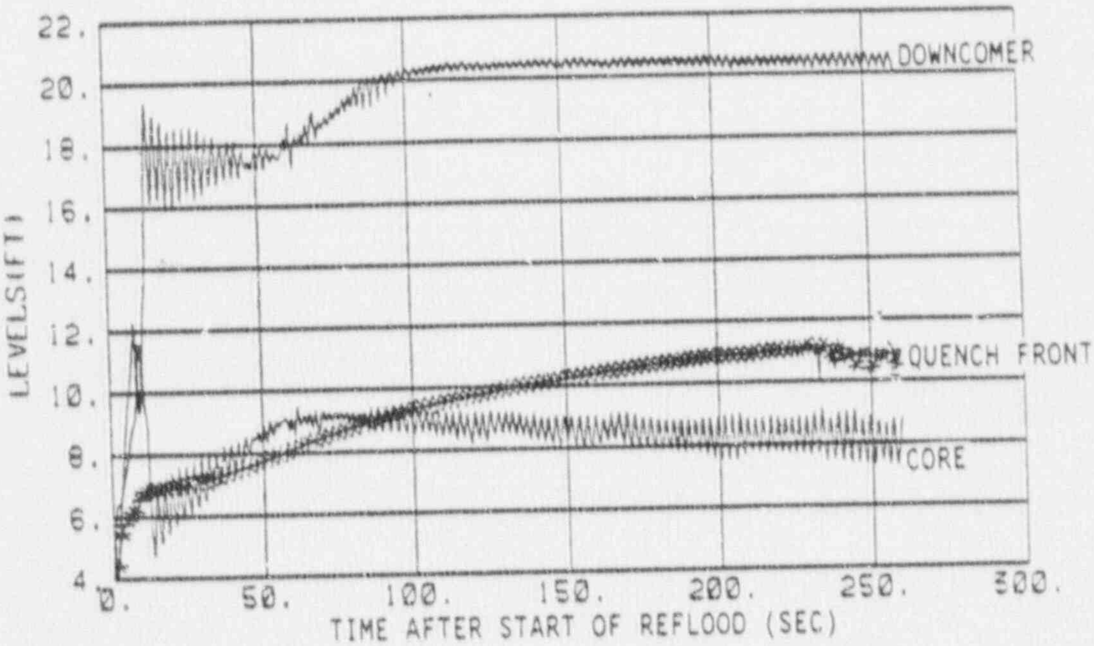


Georgia Power

VOGTLE
ELECTRIC GENERATING PLANT
UNIT 1 AND UNIT 2

REFLOOD TRANSIENT - CORE AND DOWNCOMER WATER LEVELS
DECLG (CD = 0.4)

FIGURE 15.6.5-13

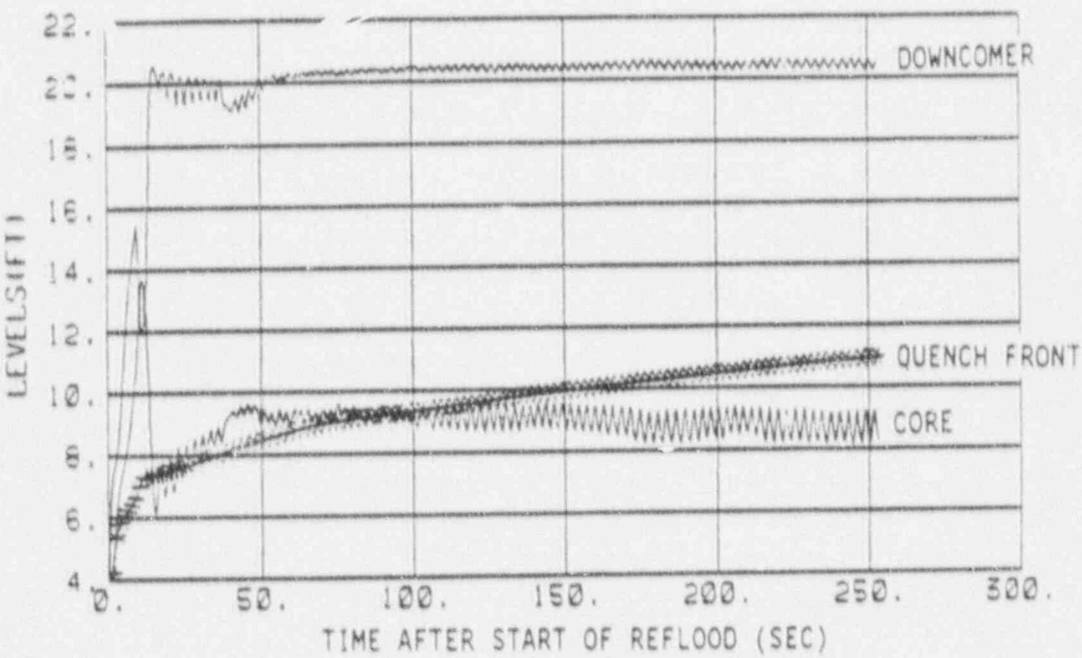


Georgia Power

VOGTLE
ELECTRIC GENERATING PLANT
UNIT 1 AND UNIT 2

REFLOOD TRANSIENT - CORE AND
DOWNCOMER WATER LEVELS
DECLG (CD = 0.6)

FIGURE 15.6.5-14

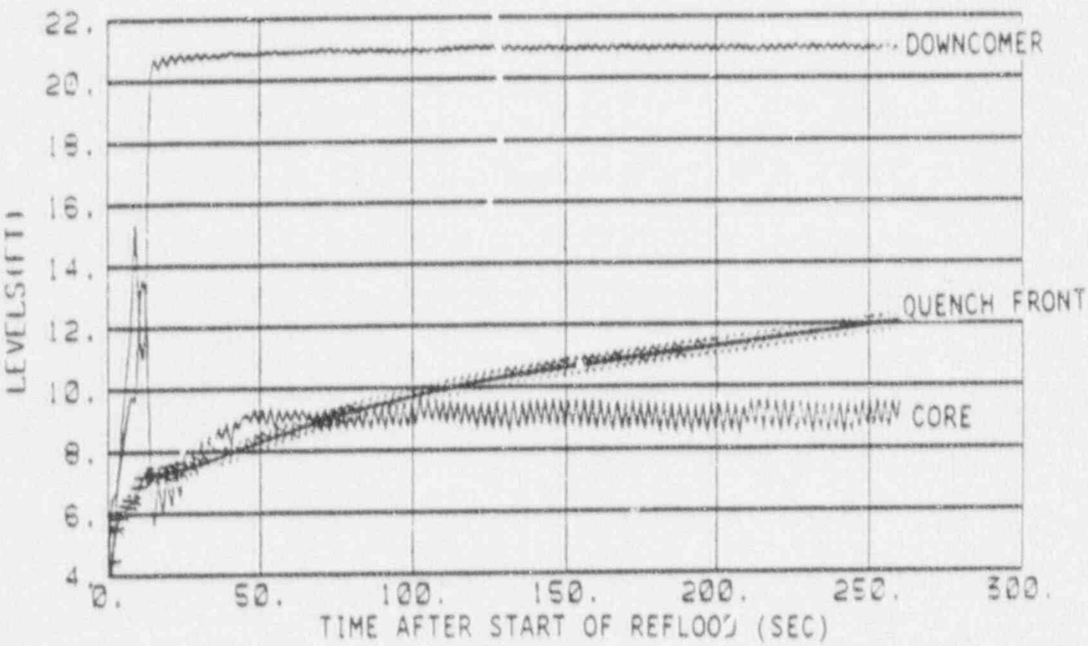


Georgia Power

VOGTLE
ELECTRIC GENERATING PLANT
UNIT 1 AND UNIT 2

REFLOOD TRANSIENT - CORE AND
DOWNCOMER WATER LEVELS
DECLG (CD = 0.8)

FIGURE 15.6.5-15

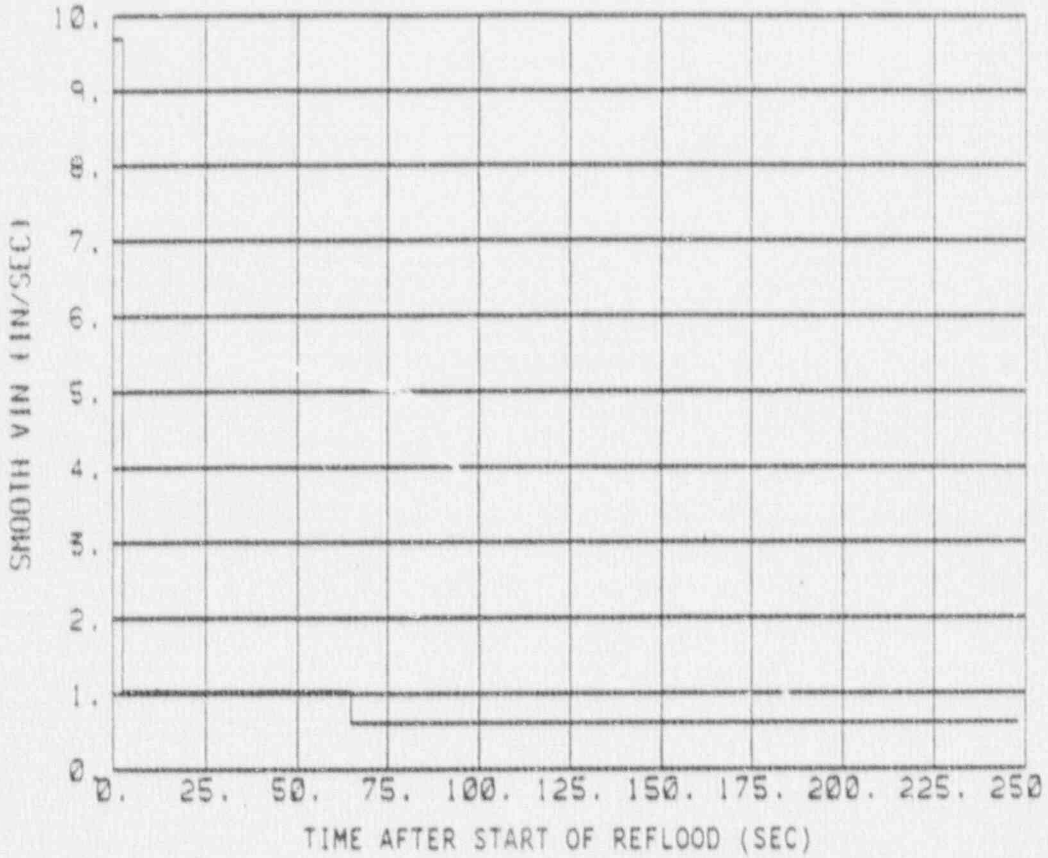


Georgia Power

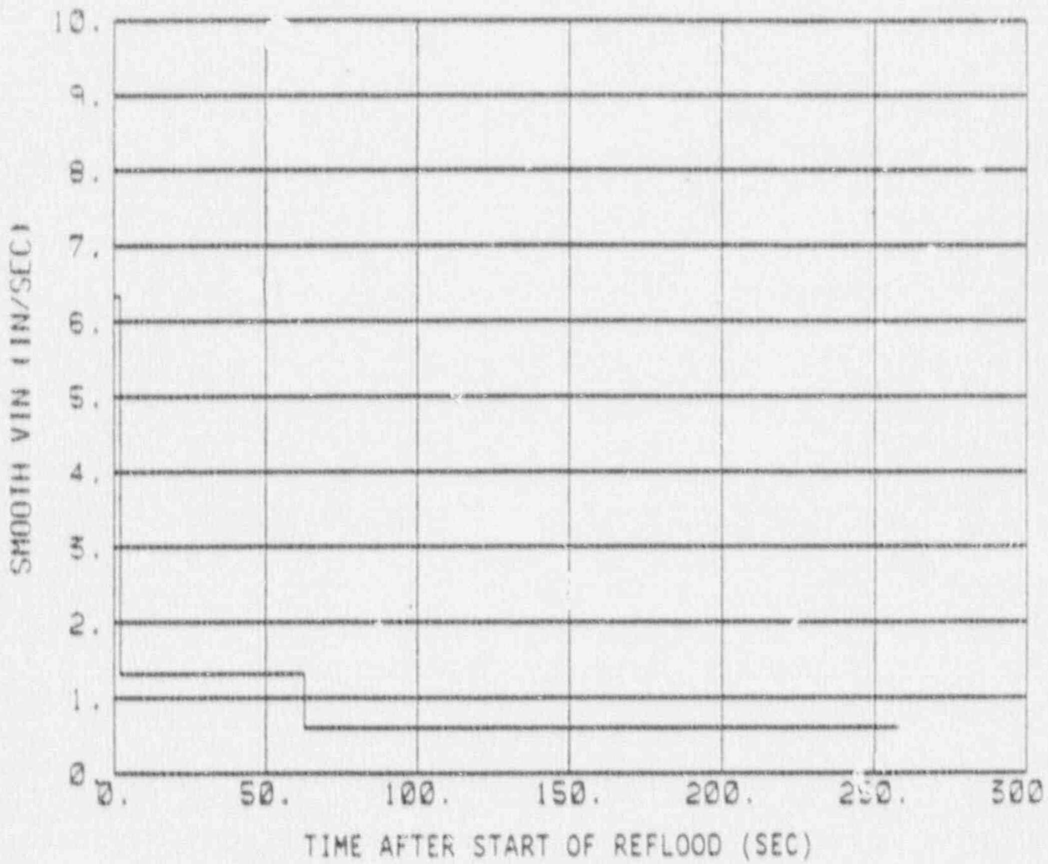
VOGTLE
ELECTRIC GENERATING PLANT
UNIT 1 AND UNIT 2

REFLOOD TRANSIENT - CORE AND DOWNCOMER WATER LEVELS DECLG (CD = 0.6 MAXIMUM SAFETY INJECTION)

FIGURE 15.6.5-16



Georgia Power	VOGTLE ELECTRIC GENERATING PLANT UNIT 1 AND UNIT 2	REFLOOD TRANSIENT CORE INLET VELOCITY DECLG (CD = 0.4)
		FIGURE 15.6.5-17

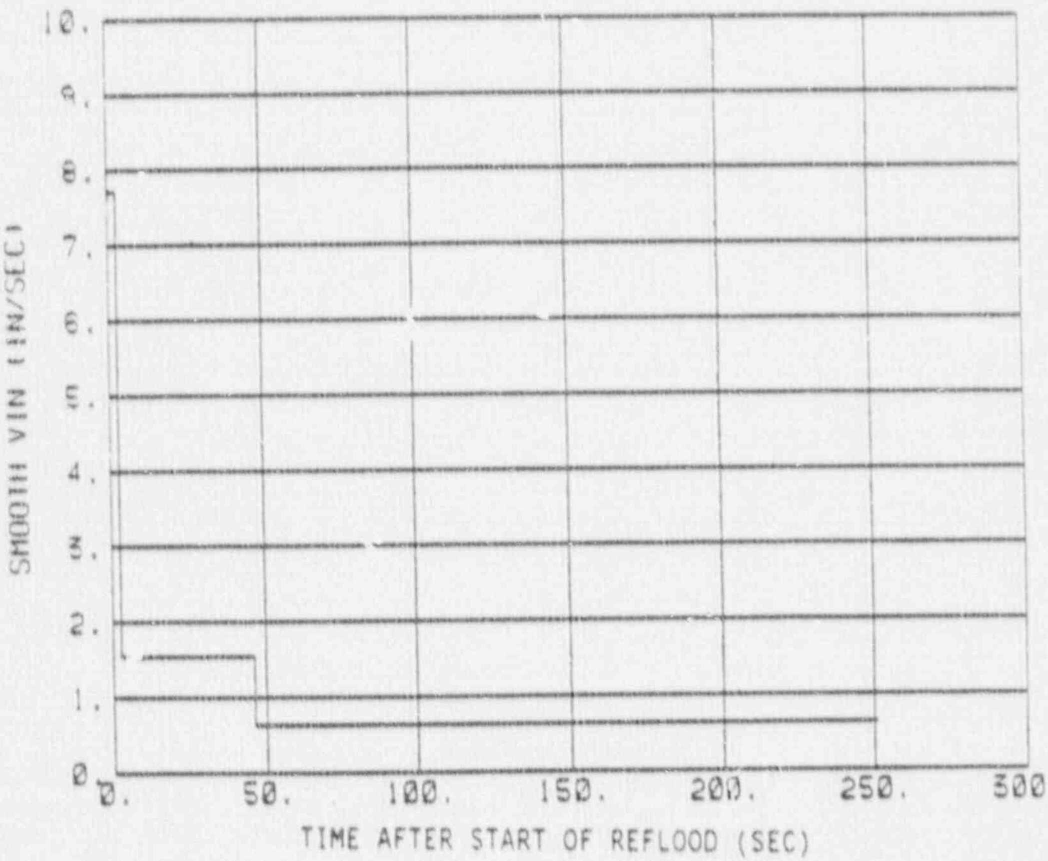


Georgia Power

VOGTLE
ELECTRIC GENERATING PLANT
UNIT 1 AND UNIT 2

REFLOOD TRANSIENT CORE INLET
VELOCITY DECLG (CD = 0.6)

FIGURE 15.6.5-18

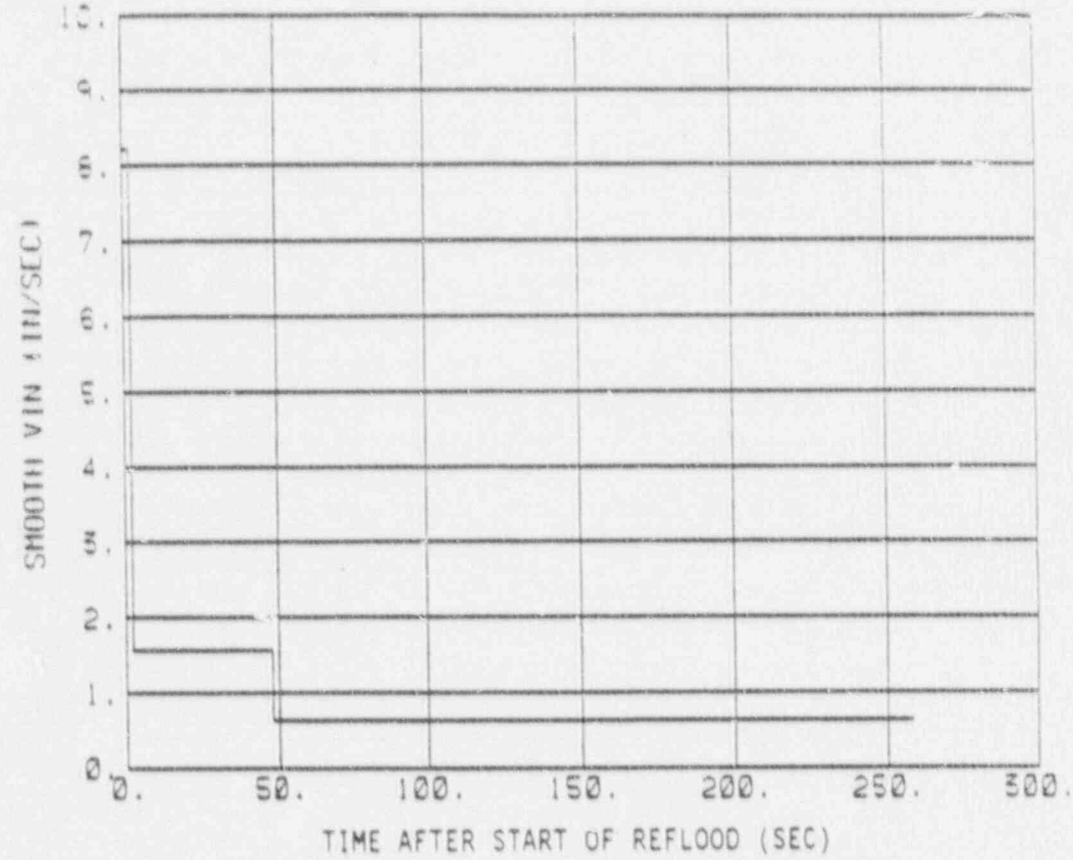


Georgia Power

VOGTLE
ELECTRIC GENERATING PLANT
UNIT 1 AND UNIT 2

REFLOOD TRANSIENT CORE INLET
VELOCITY DECLG (CD = 0.8)

FIGURE 15.6.5-19

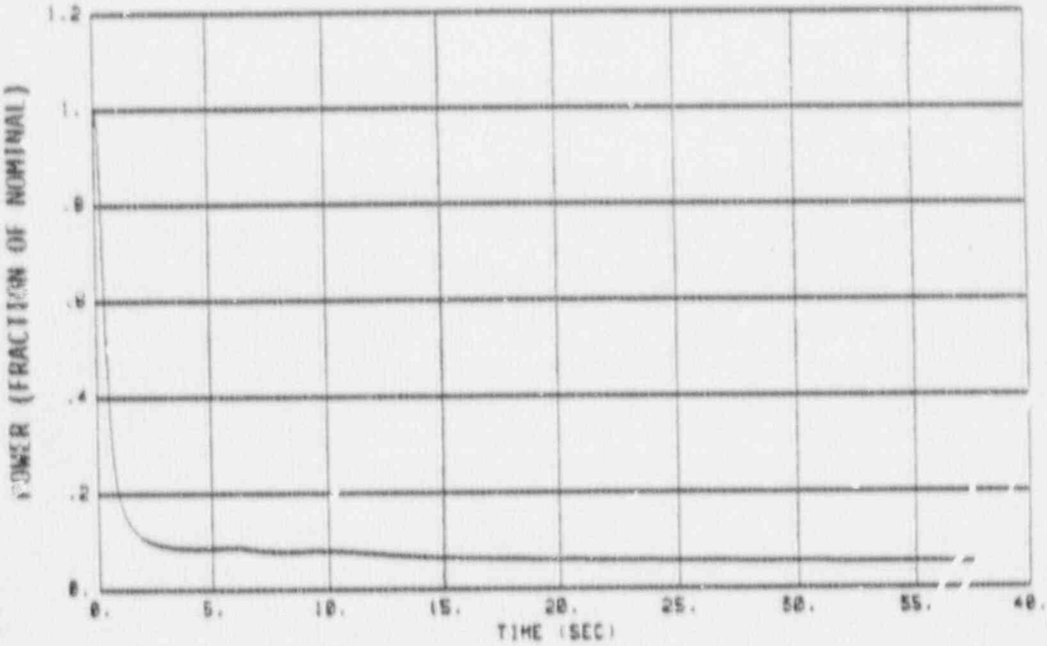


Georgia Power

VOGTLE
ELECTRIC GENERATING PLANT
UNIT 1 AND UNIT 2

REFLOOD TRANSIENT CORE INLET VELOCITY DECLG (CD = 0.6 MAXIMUM SAFETY INJECTION)

FIGURE 15.6.5-20

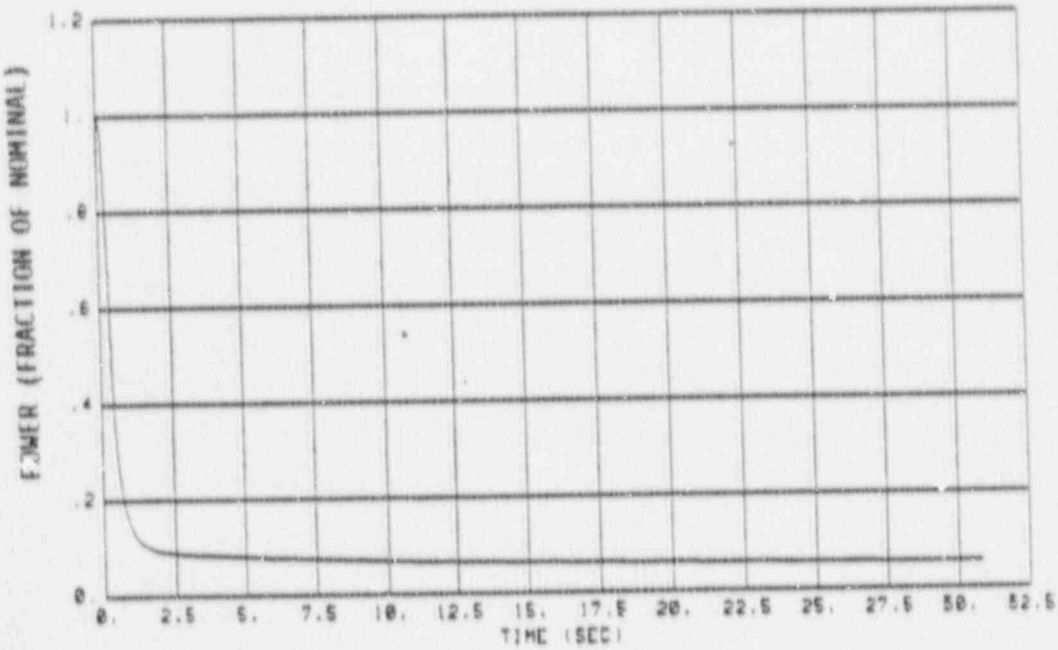


Georgia Power

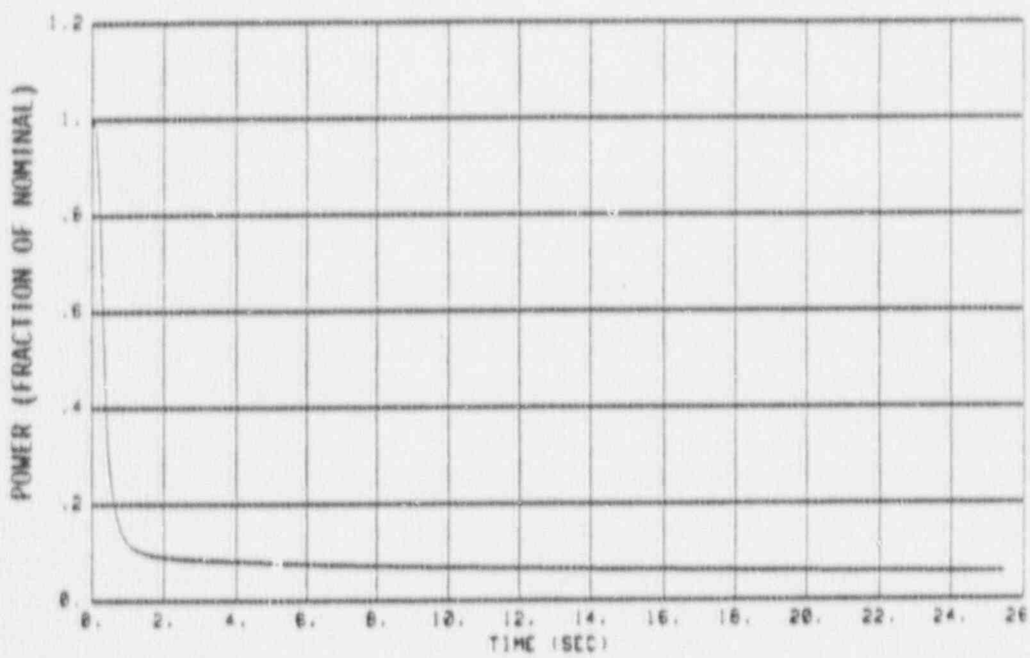
VOGTLE
ELECTRIC GENERATING PLANT
UNIT 1 AND UNIT 2

CORE POWER TRANSIENT
DECLG (CD = 0.4)

FIGURE 15.6.5-21



Georgia Power	VOGTLE ELECTRIC GENERATING PLANT UNIT 1 AND UNIT 2	CORE POWER TRANSIENT DECLG (CD = 0.6)
		FIGURE 15.6.5-22

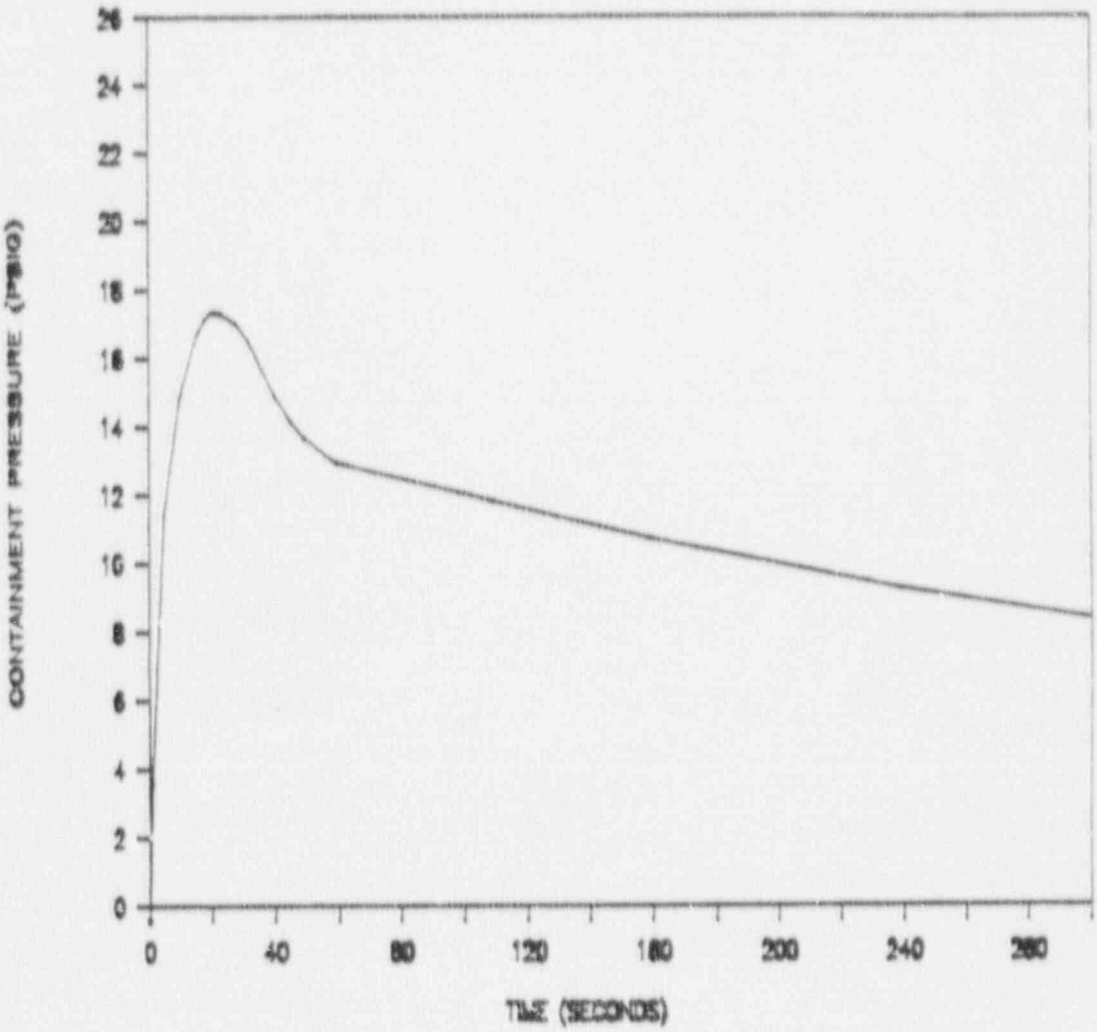


Georgia Power

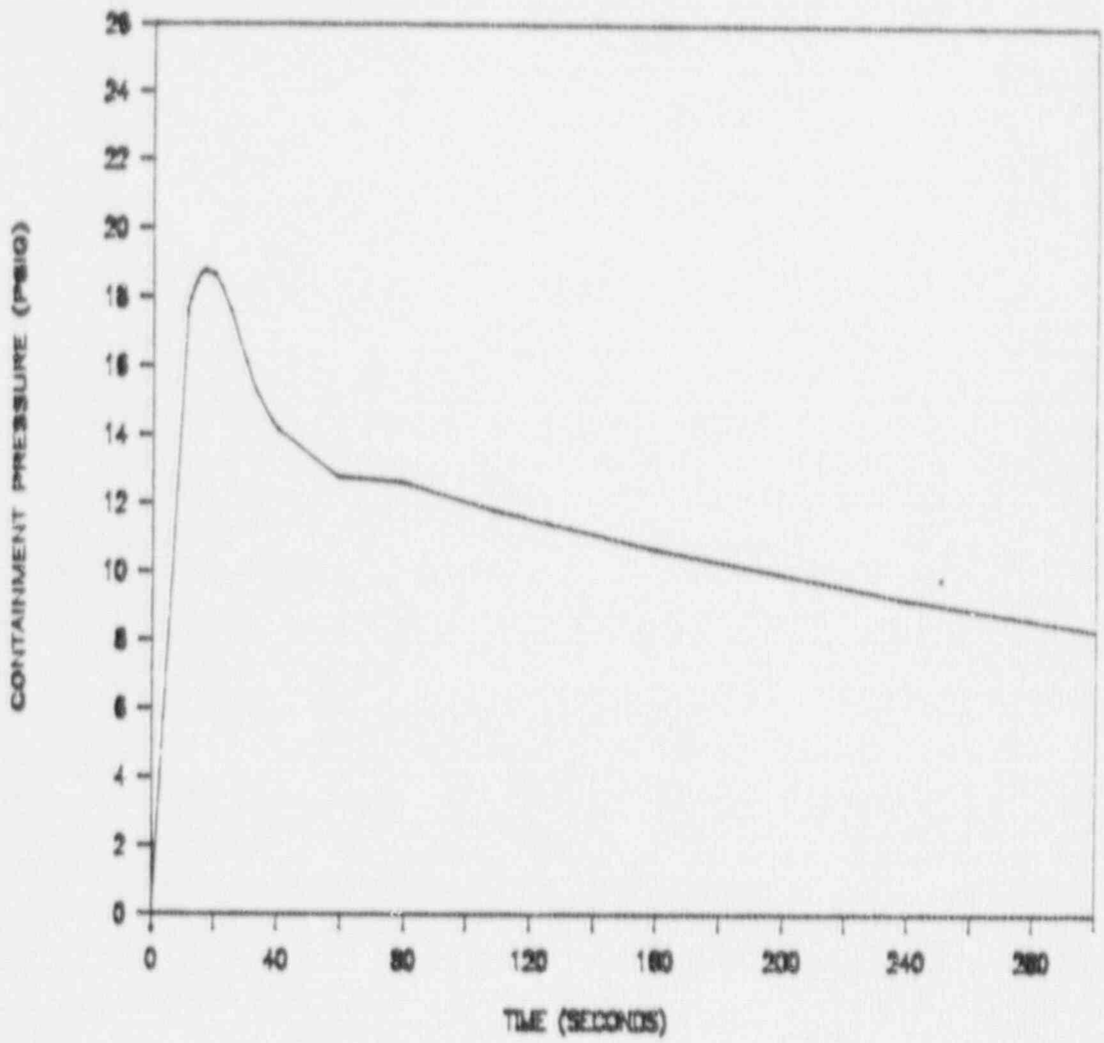
VOGTLE
ELECTRIC GENERATING PLANT
UNIT 1 AND UNIT 2

CORE POWER TRANSIENT
DECLG (CD = 0.8)

FIGURE 15.6.5-23



Georgia Power VOGTLE ELECTRIC GENERATING PLANT UNIT 1 AND UNIT 2	CONTAINMENT PRESSURE TRANSIENT (CD = 0.4)
	FIGURE 15.6.5-24

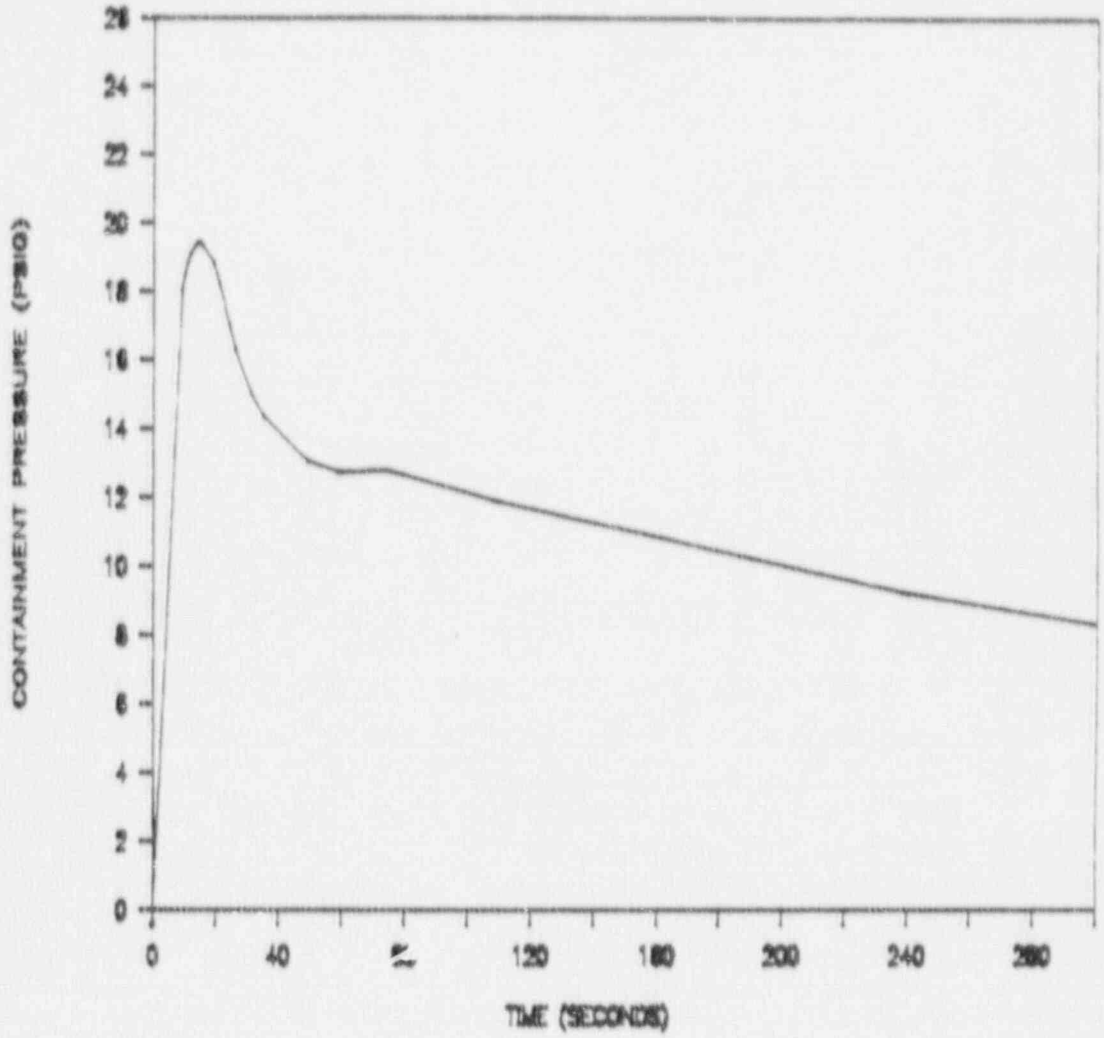


Georgia Power

VOGTLE
ELECTRIC GENERATING PLANT
UNIT 1 AND UNIT 2

CONTAINMENT PRESSURE TRANSIENT
(CD = 0.6)

FIGURE 15.6.5-25

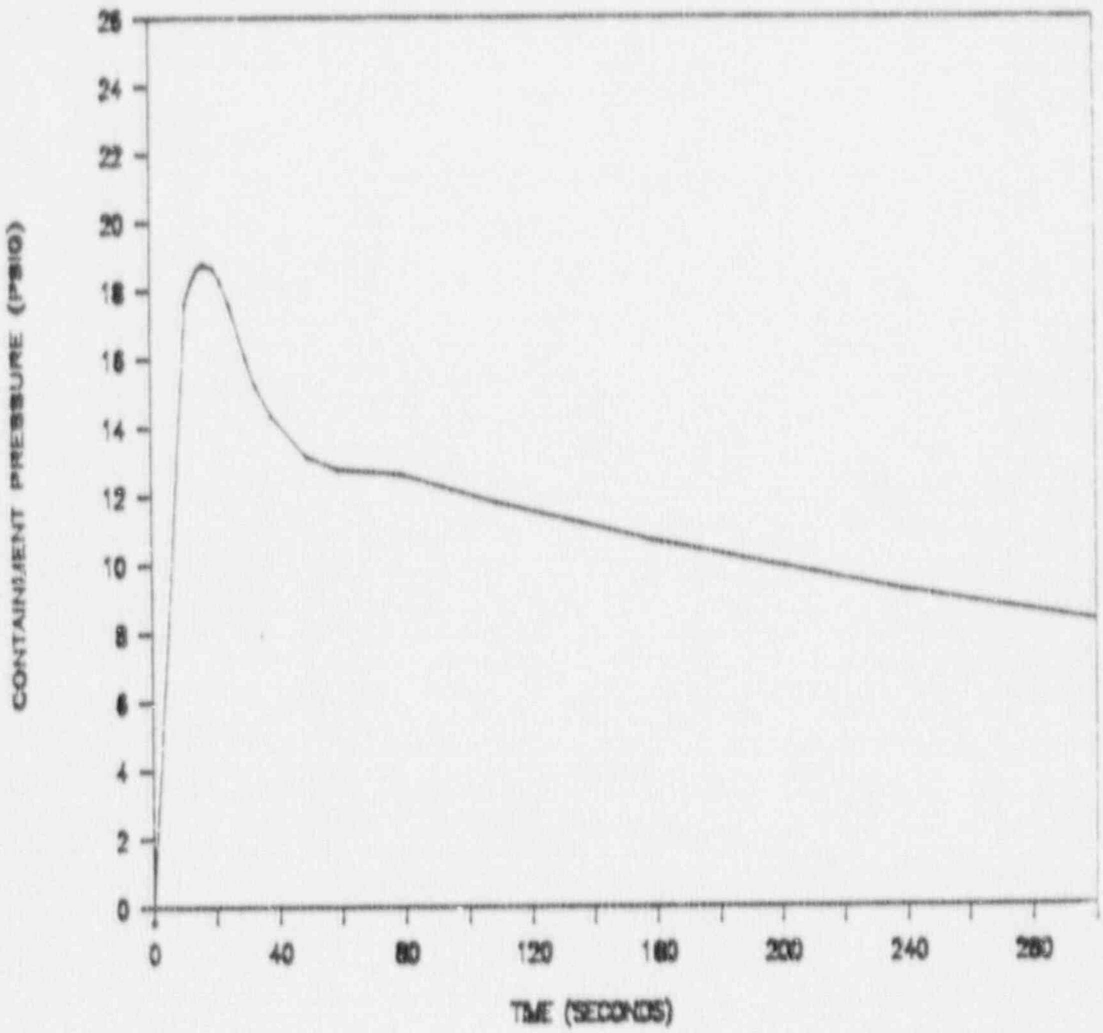


Georgia Power

VOGTLE
ELECTRIC GENERATING PLANT
UNIT 1 AND UNIT 2

CONTAINMENT PRESSURE TRANSIENT
(CD = 0.8)

FIGURE 15.6.5-26

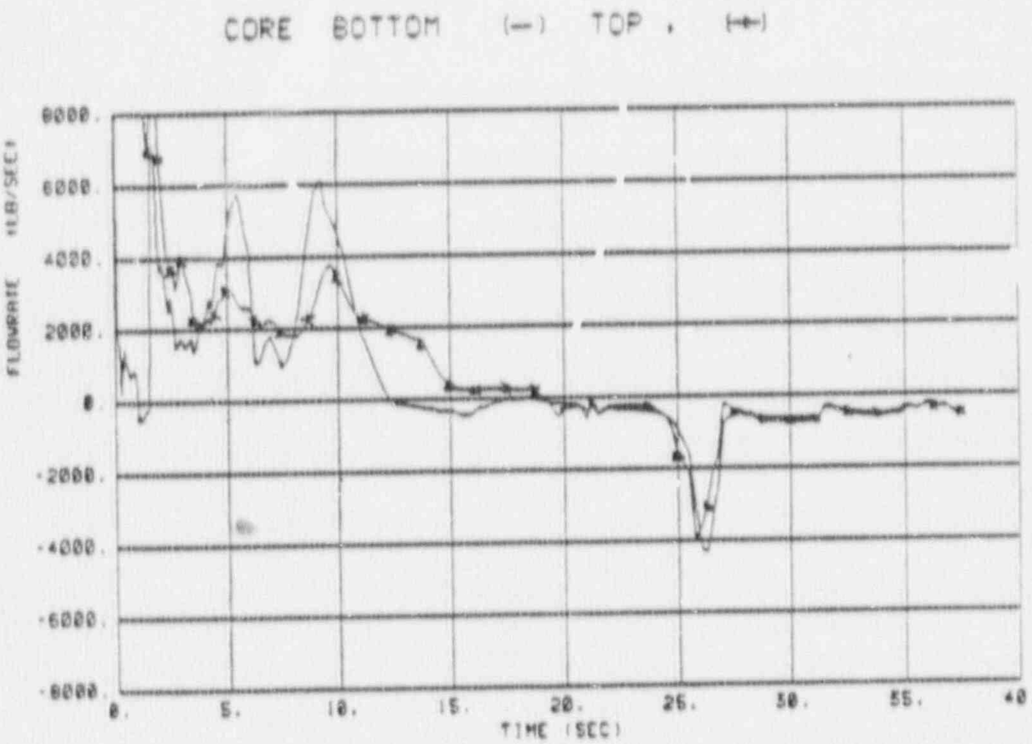


Georgia Power

VOGTLE
ELECTRIC GENERATING PLANT
UNIT 1 AND UNIT 2

CONTAINMENT PRESSURE TRANSIENT
(CD = 0.6 MAXIMUM SAFETY
INJECTION)

FIGURE 15.6.5-27

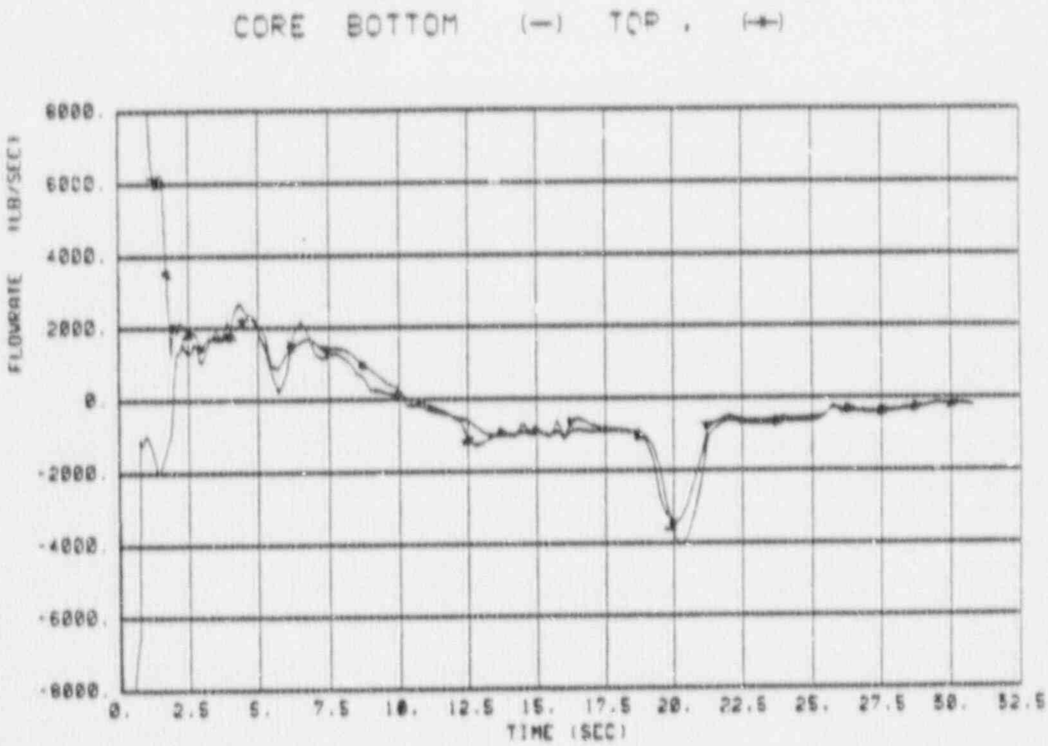


Georgia Power

VOGTLE
ELECTRIC GENERATING PLANT
UNIT 1 AND UNIT 2

CORE FLOW (TOP AND BOTTOM)
DECLG (CD = 0.4)

FIGURE 15.6.5-.8A

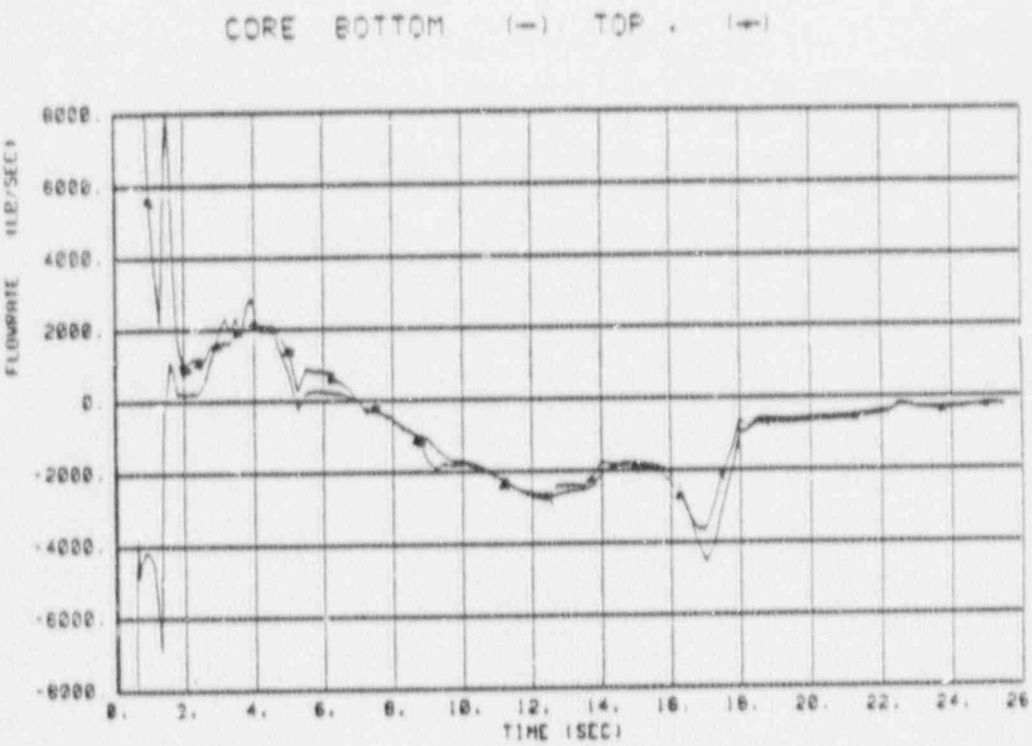


Georgia Power

VOGTLE
ELECTRIC GENERATING PLANT
UNIT 1 AND UNIT 2

CORE FLOW (TOP AND BOTTOM)
DECLG (CD = 0.6)

FIGURE 15.6.5-28B

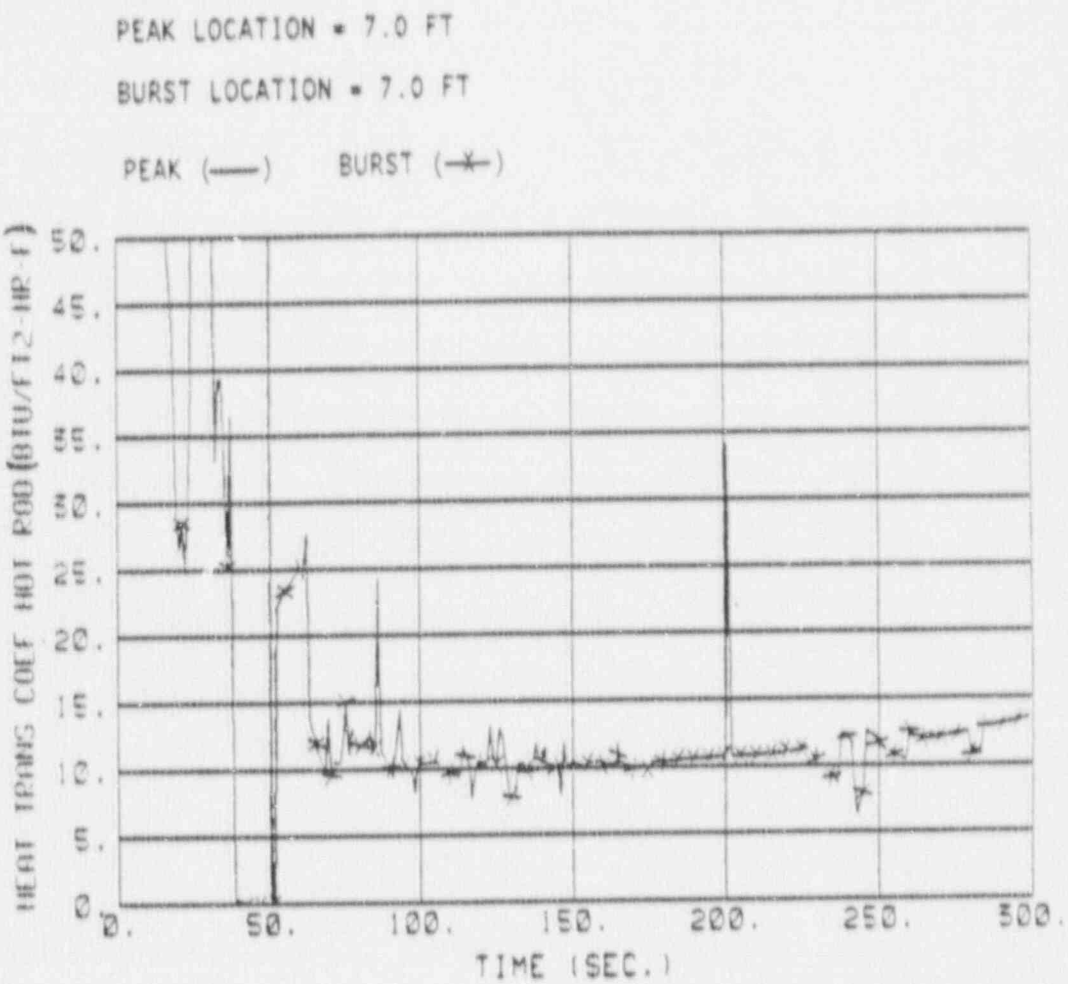


Georgia Power

VOGTLE
ELECTRIC GENERATING PLANT
UNIT 1 AND UNIT 2

CORE FLOW (TOP AND BOTTOM)
DECLG (CD = 0.8)

FIGURE 15.6.5-28C

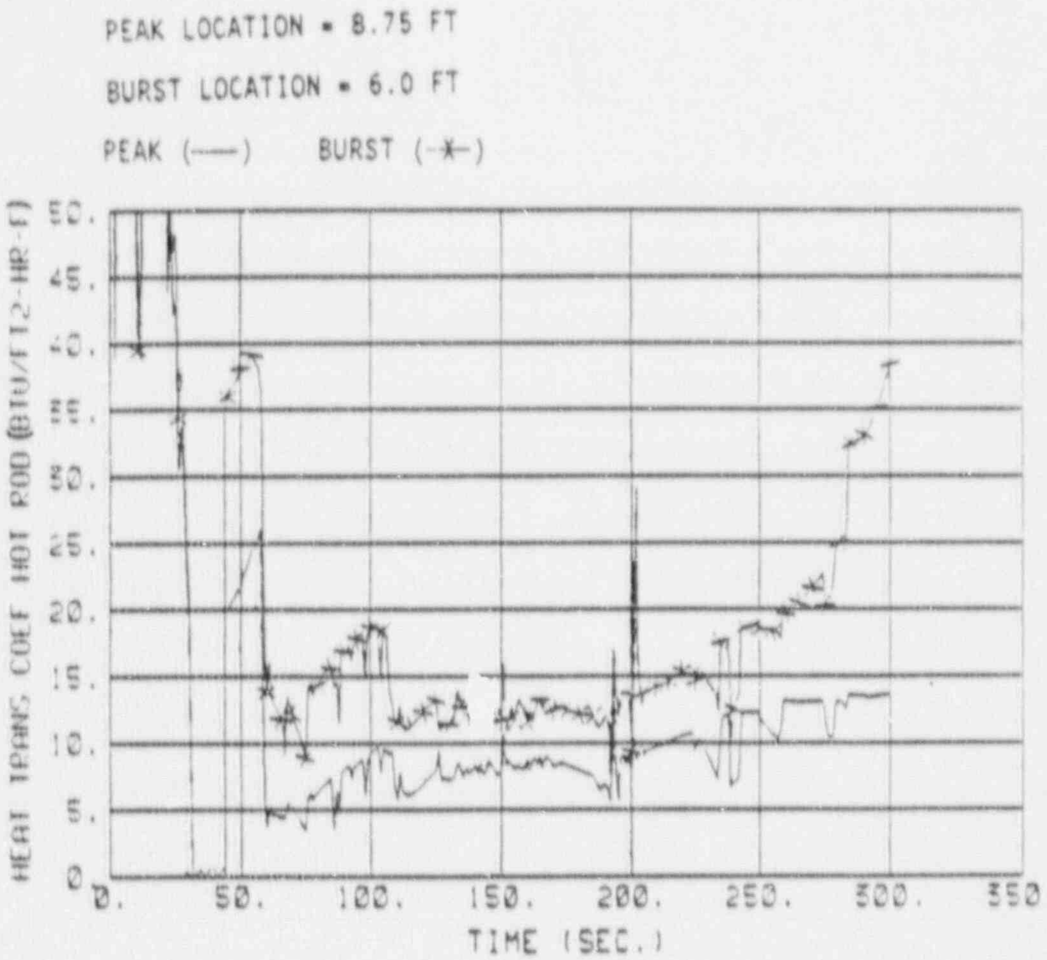


Georgia Power

ALVIN W. VOGTLE
ELECTRIC GENERATING PLANT
UNIT 1 AND UNIT 2

HEAT TRANSFER COEFFICIENT
DECLG (CD = 0.4)

FIGURE 15.6.5-29A

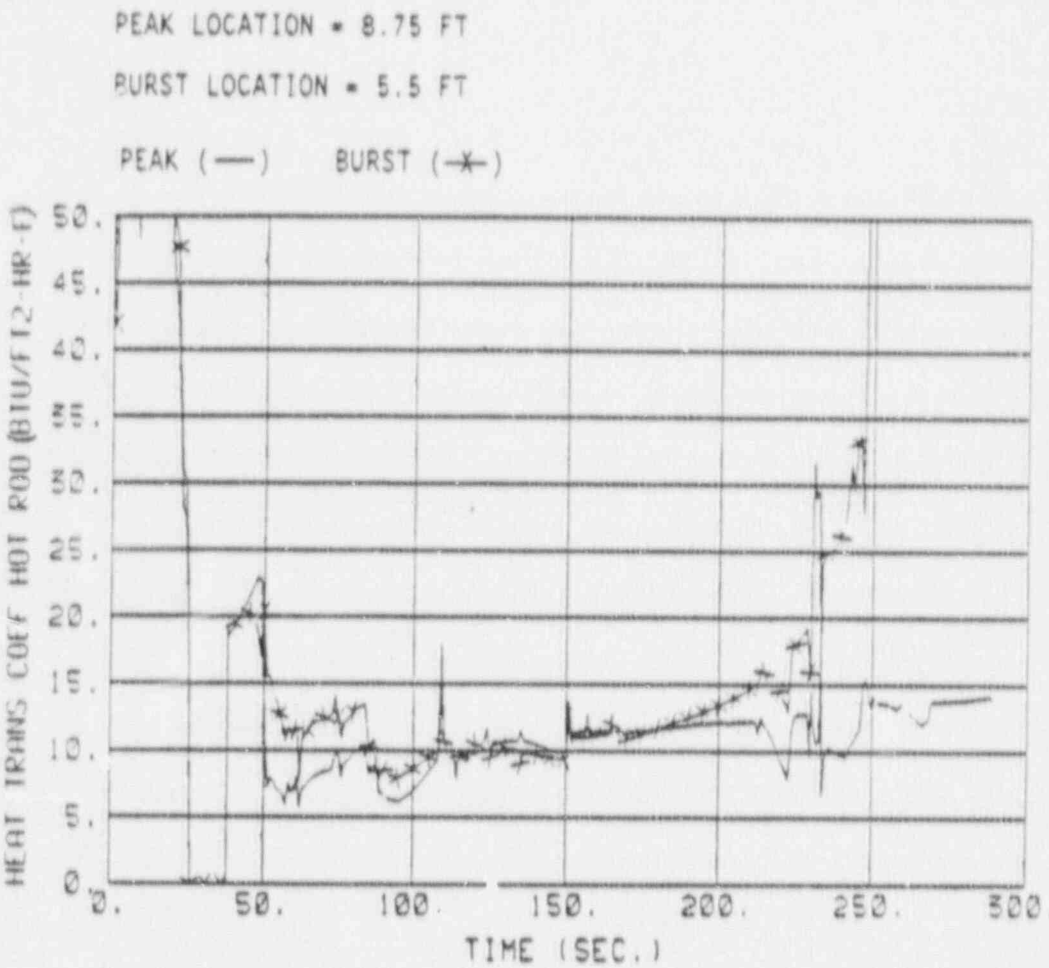


Georgia Power

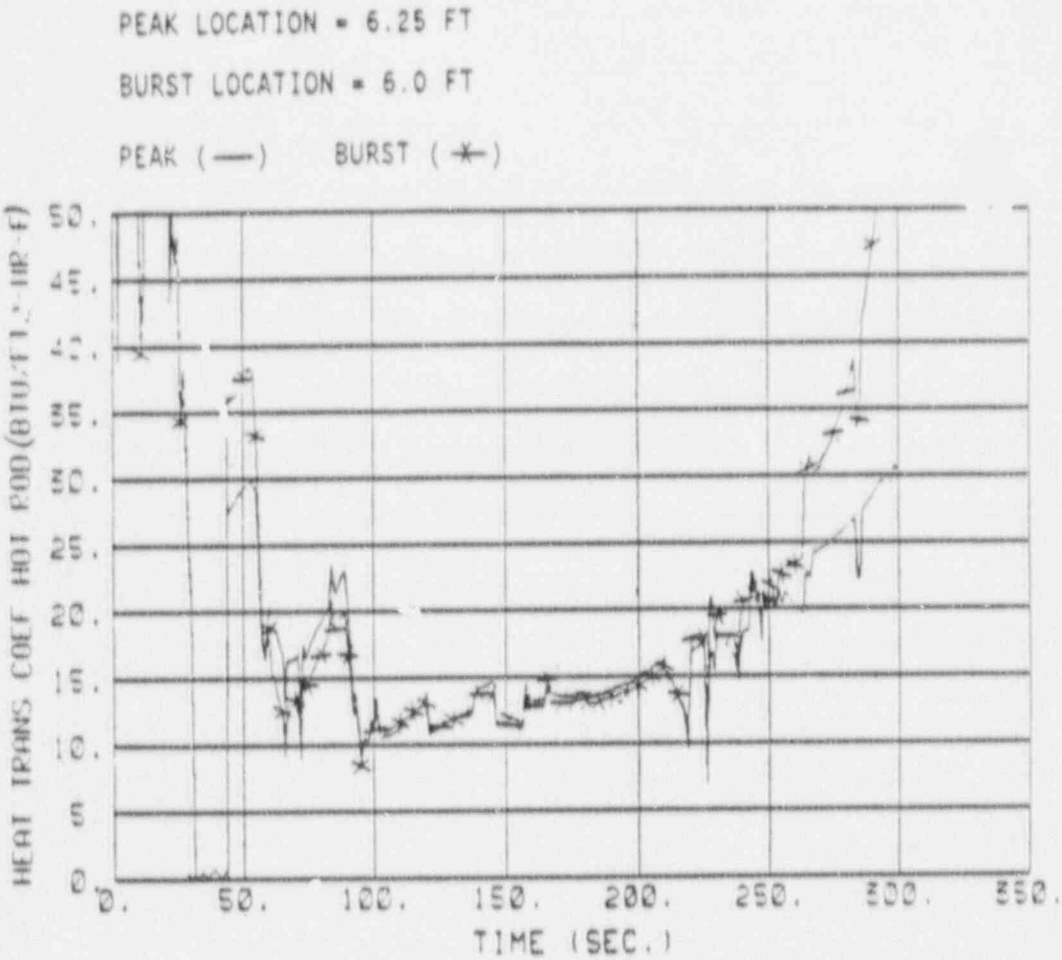
VOGTLE
ELECTRIC GENERATING PLANT
UNIT 1 AND UNIT 2

HEAT TRANSFER COEFFICIENT
DECLG (CD = 0.6)

FIGURE 15.6.5-29B



Georgia Power — VOGTLE ELECTRIC GENERATING PLANT UNIT 1 AND UNIT 2	HEAT TRANSFER COEFFICIENT DECLG (CD = 0.8)
	FIGURE 15.6.5-29C

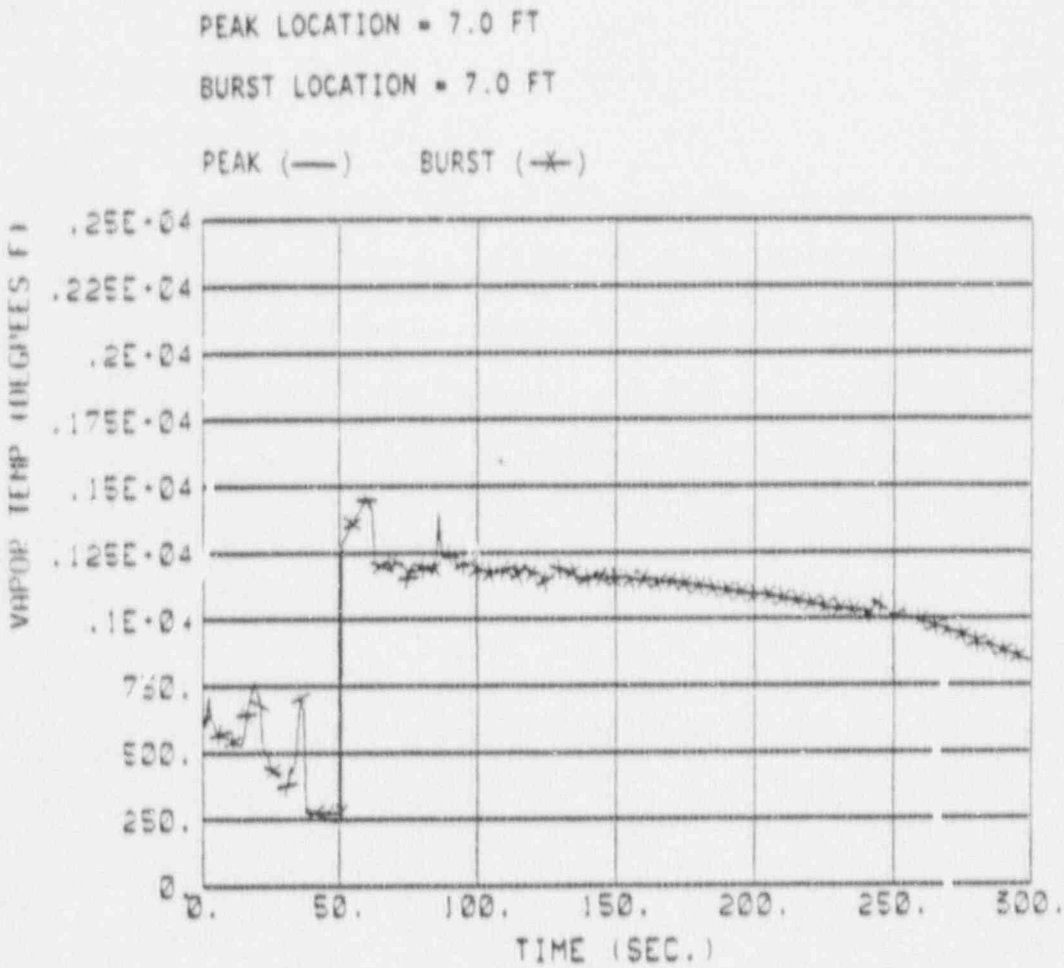


Georgia Power

VOGTLE
ELECTRIC GENERATING PLANT
UNIT 1 AND UNIT 2

HEAT TRANSFER COEFFICIENT
DECLG (CD = 0.6
MAXIMUM SAFETY INJECTION

FIGURE 15.6.5-29D

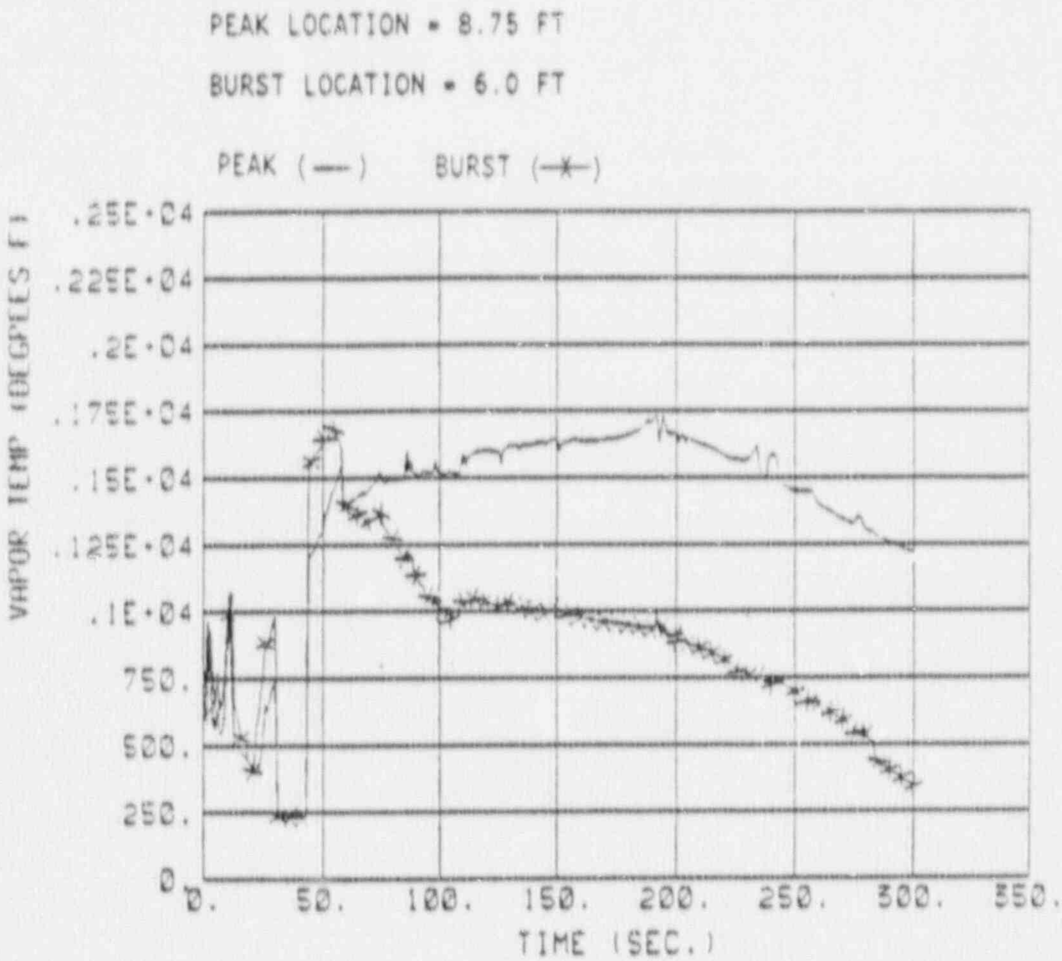


Georgia Power

VOGTLE
ELECTRIC GENERATING PLANT
UNIT 1 AND UNIT 2

FLUID TEMPERATURE
DECLG (CD = 0.4)

FIGURE 15.6.5-30A

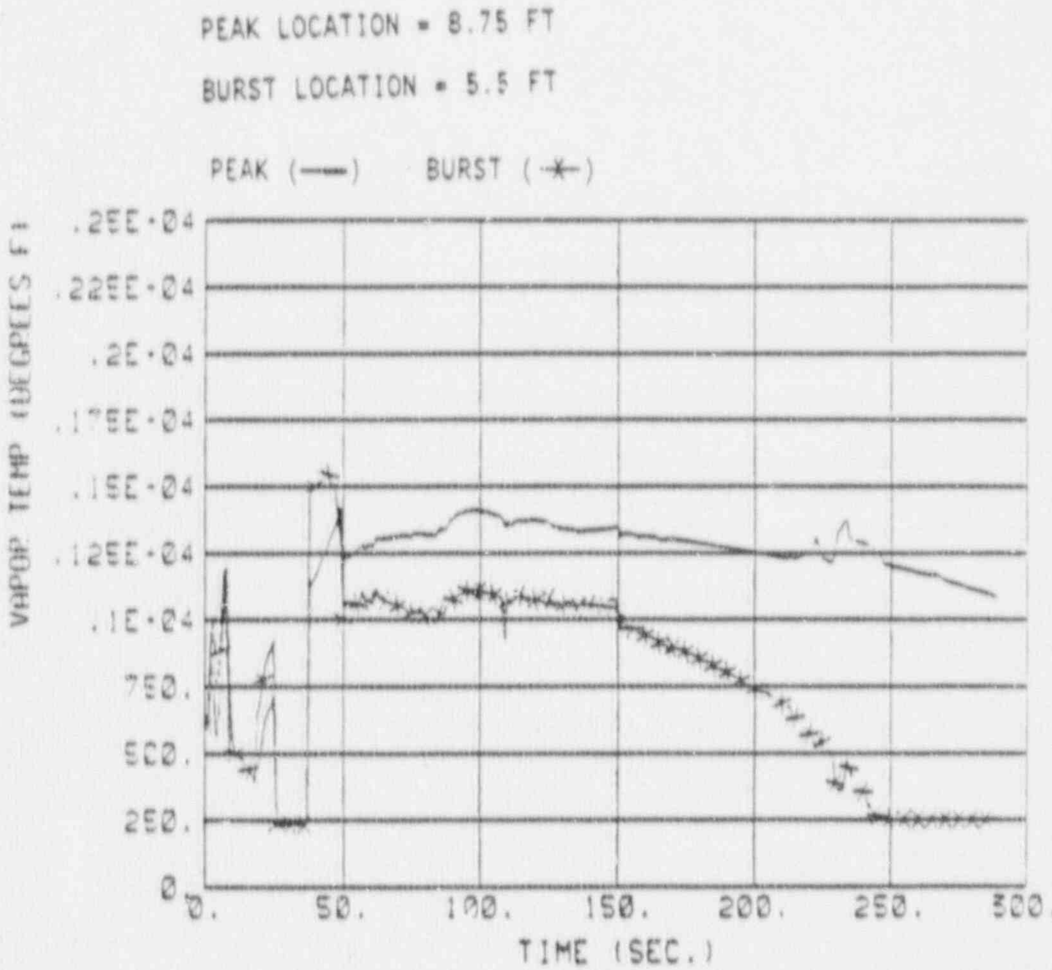


Georgia Power

VOGTLE
ELECTRIC GENERATING PLANT
UNIT 1 AND UNIT 2

FLUID TEMPERATURE
DECLG (CD = 0.6)

FIGURE 15.6.5-30B

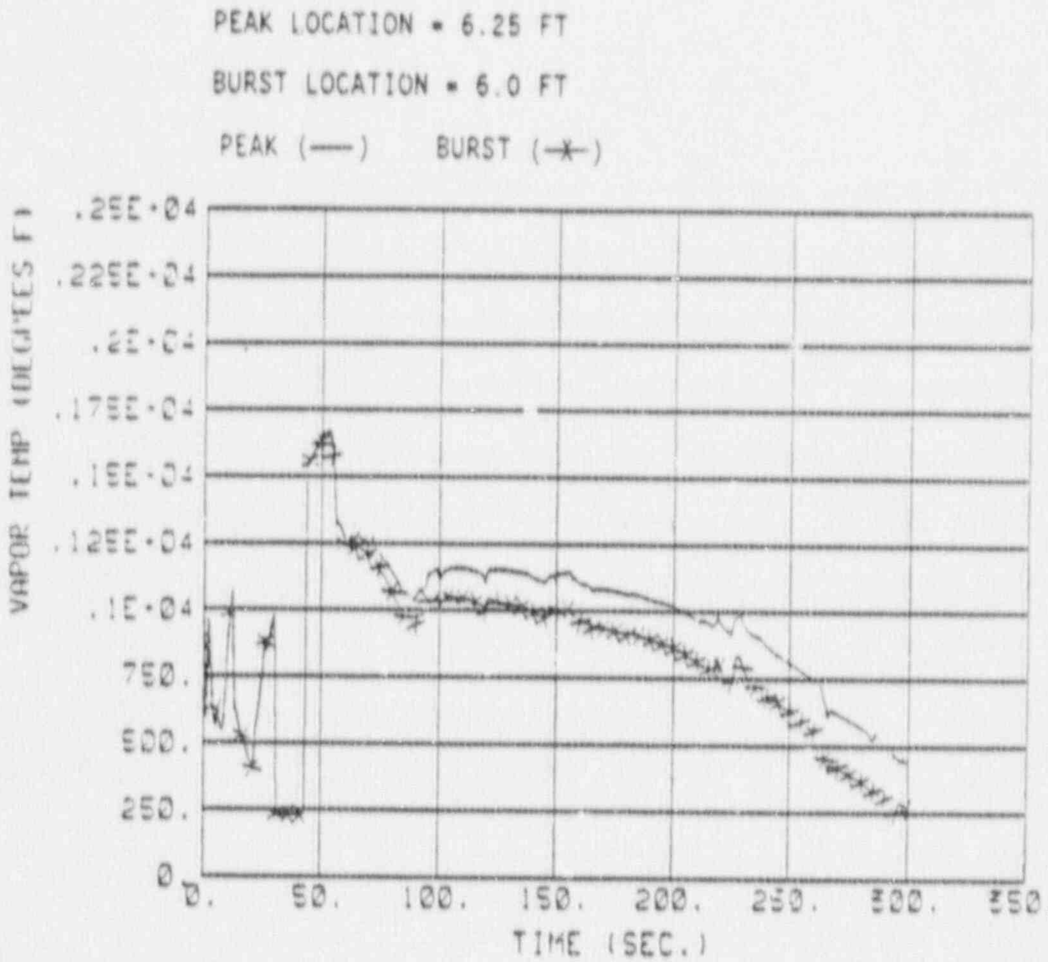


Georgia Power

VOGTLE
ELECTRIC GENERATING PLANT
UNIT 1 AND UNIT 2

FLUID TEMPERATURE
DECLG (CD = 0.8)

FIGURE 15.6.5-30C

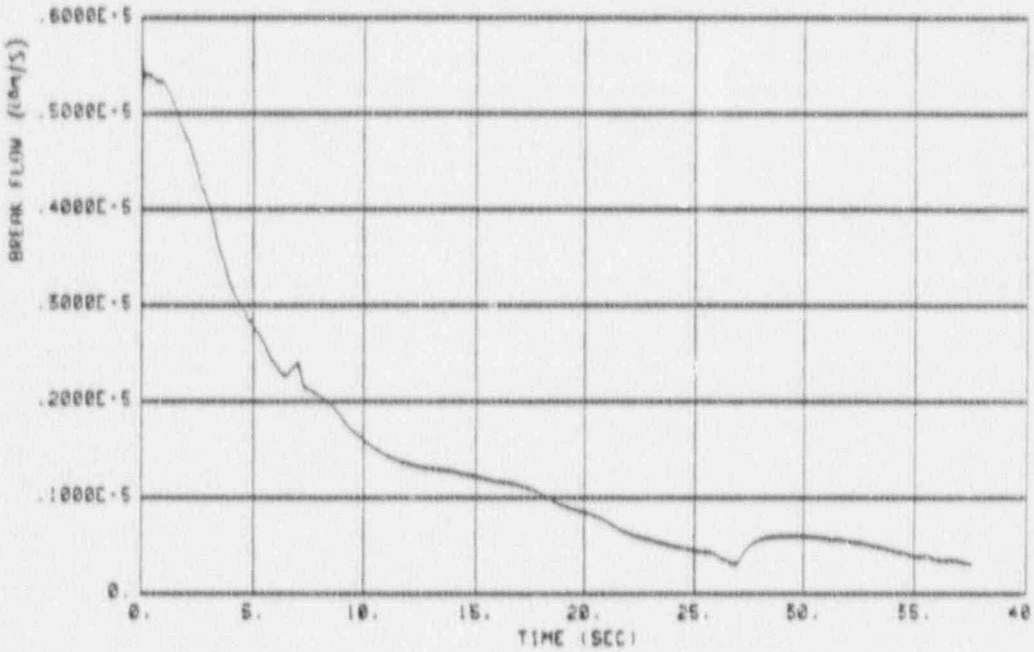


Georgia Power

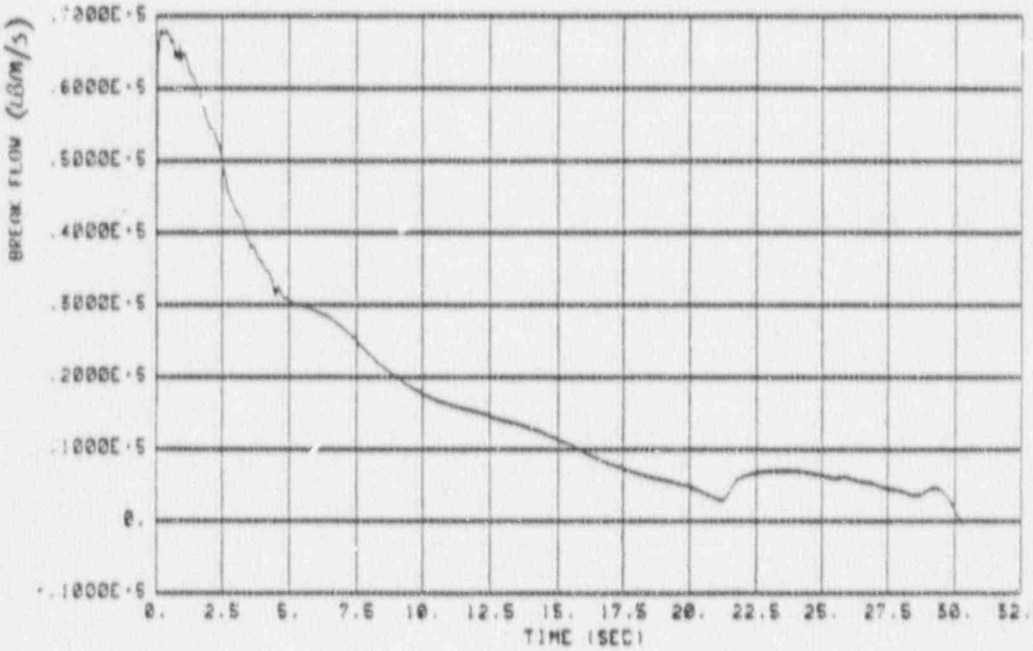
VOGTLE
ELECTRIC GENERATING PLANT
UNIT 1 AND UNIT 2

FLUID TEMPERATURE
DECLG (CD = 0.6
MAXIMUM SAFETY INJECTION)

FIGURE 15.6.5-30D



Georgia Power	VOGTLE ELECTRIC GENERATING PLANT UNIT 1 AND UNIT 2	BREAK FLOWRATE DECLG (CD = 0.4)
		FIGURE 15.6.5-31A

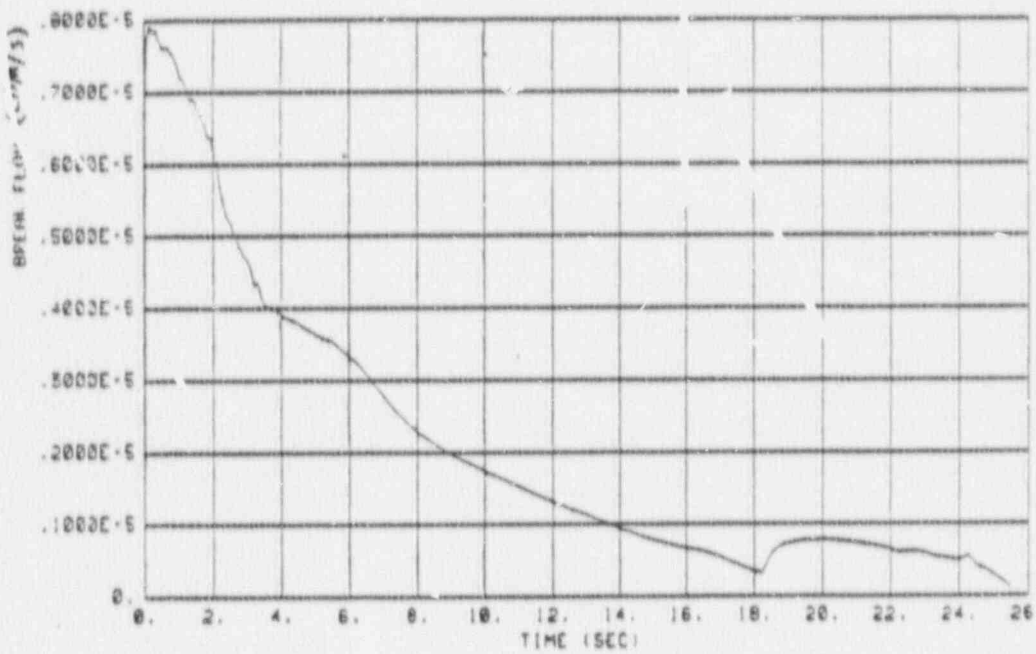


Georgia Power

VOGTLE
ELECTRIC GENERATING PLANT
UNIT 1 AND UNIT 2

BREAK FLOWRATE
DECLG (CD = 0.6)

FIGURE 15.6.5-31B

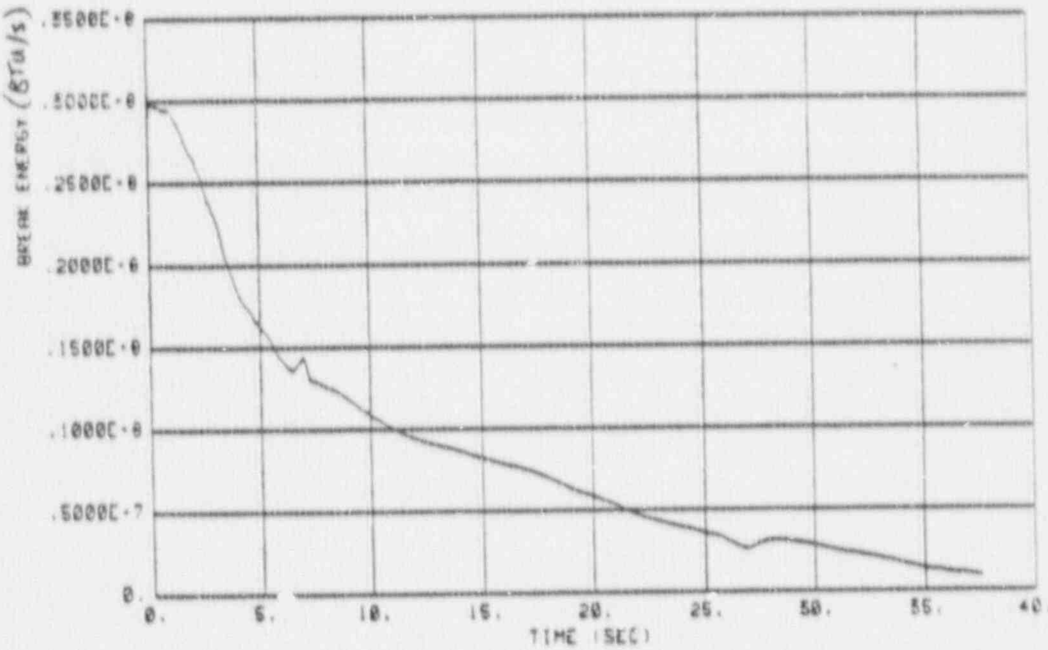


Georgia Power

VOGTLE
ELECTRIC GENERATING PLANT
UNIT 1 AND UNIT 2

BREAK FLOWRATE
DECLG (CD = 0.8)

FIGURE 15.6.5-31C

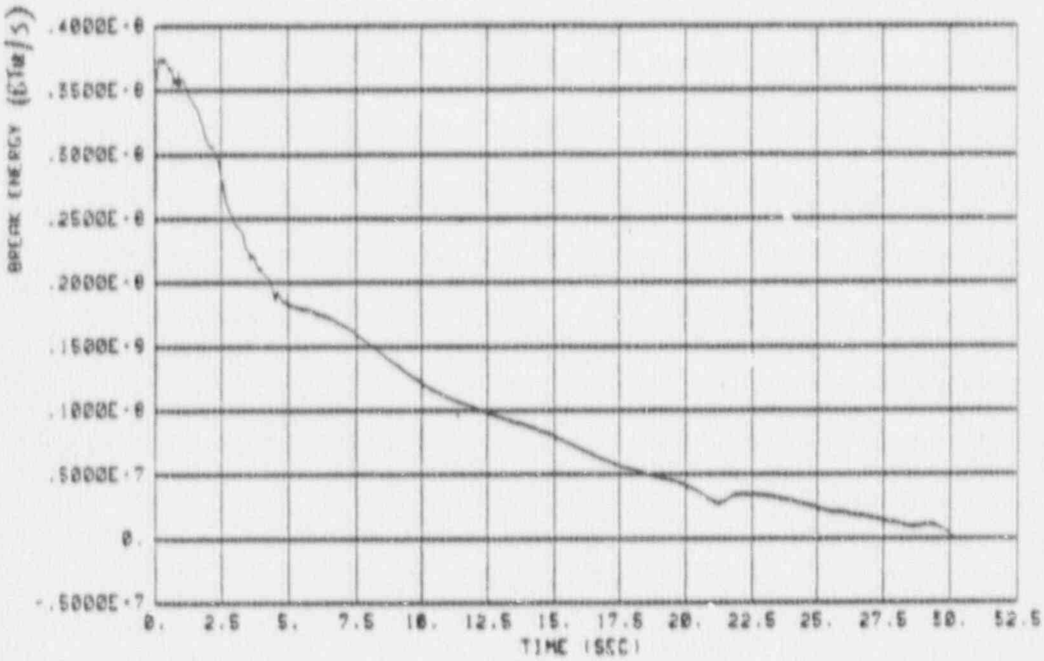


Georgia Power

VOGTLE
ELECTRIC GENERATING PLANT
UNIT 1 AND UNIT 2

BREAK ENERGY RELEASED TO CONTAINMENT (CD = 0.4)

FIGURE 15.6.5-32A

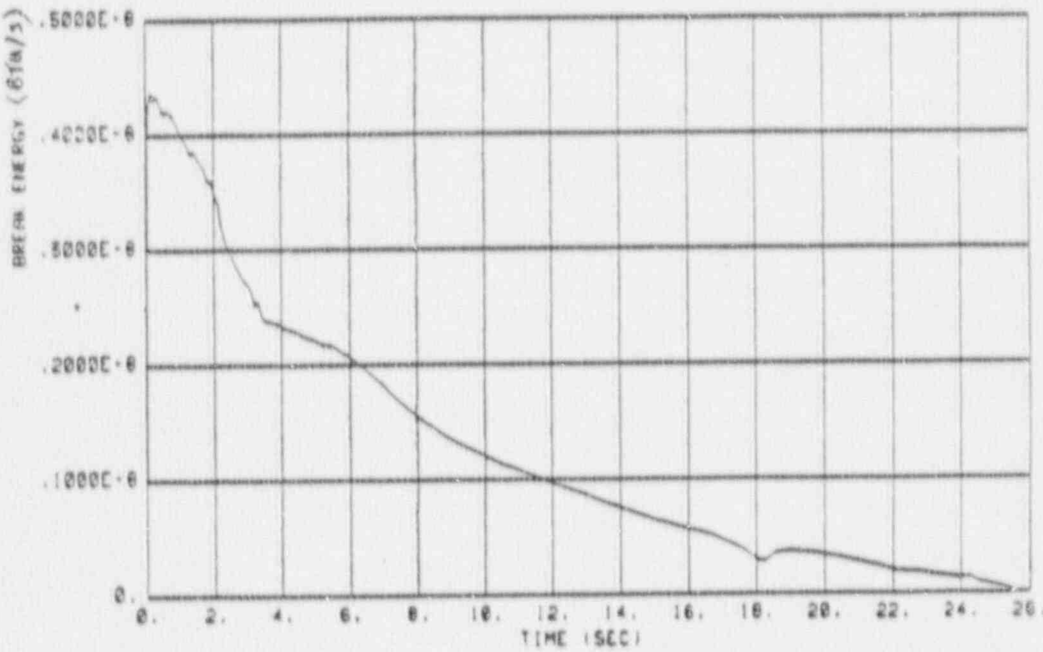


Georgia Power

VOGTLE
ELECTRIC GENERATING PLANT
UNIT 1 AND UNIT 2

BREAK ENERGY RELEASED TO CONTAINMENT (CD = 0.6)

FIGURE 15.6.5-32B

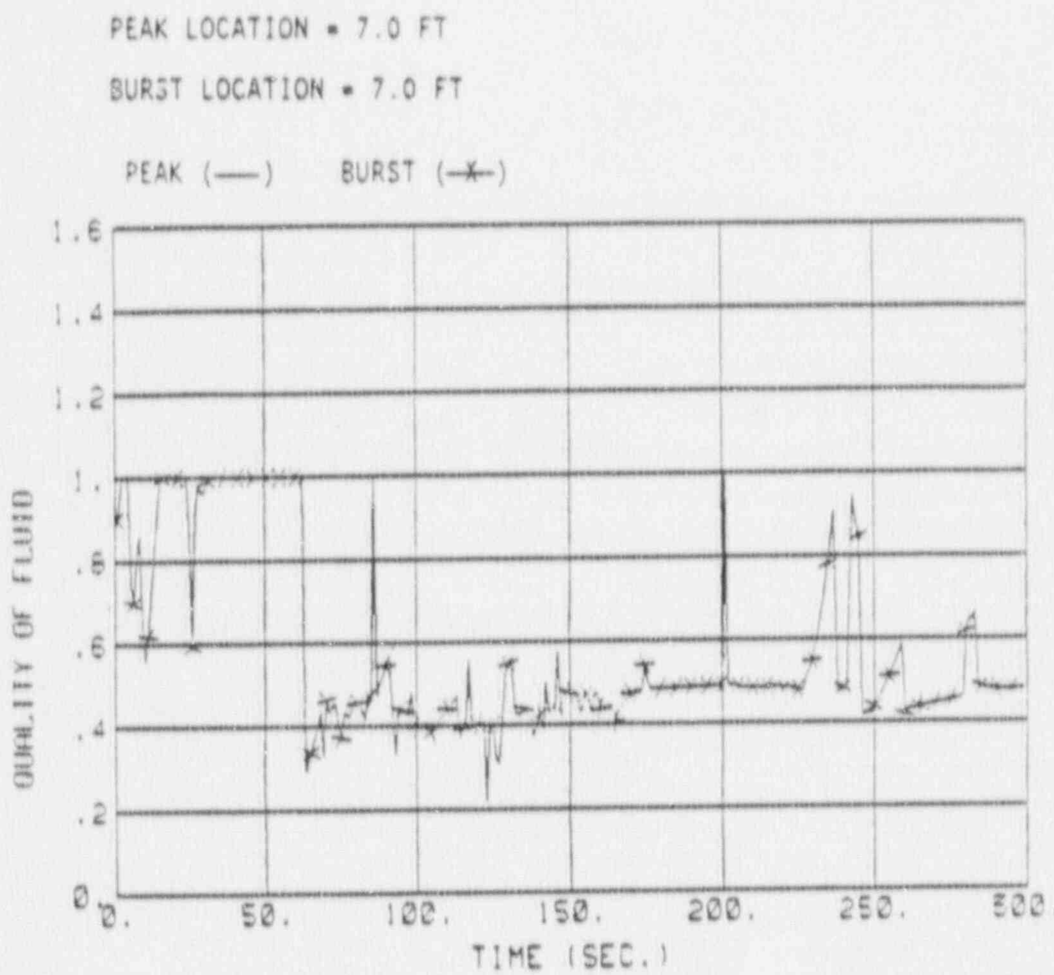


Georgia Power

VOGTLE
ELECTRIC GENERATING PLANT
UNIT 1 AND UNIT 2

BREAK ENERGY RELEASED TO CONTAINMENT (CD = 0.8)

FIGURE 15.6.5-32C

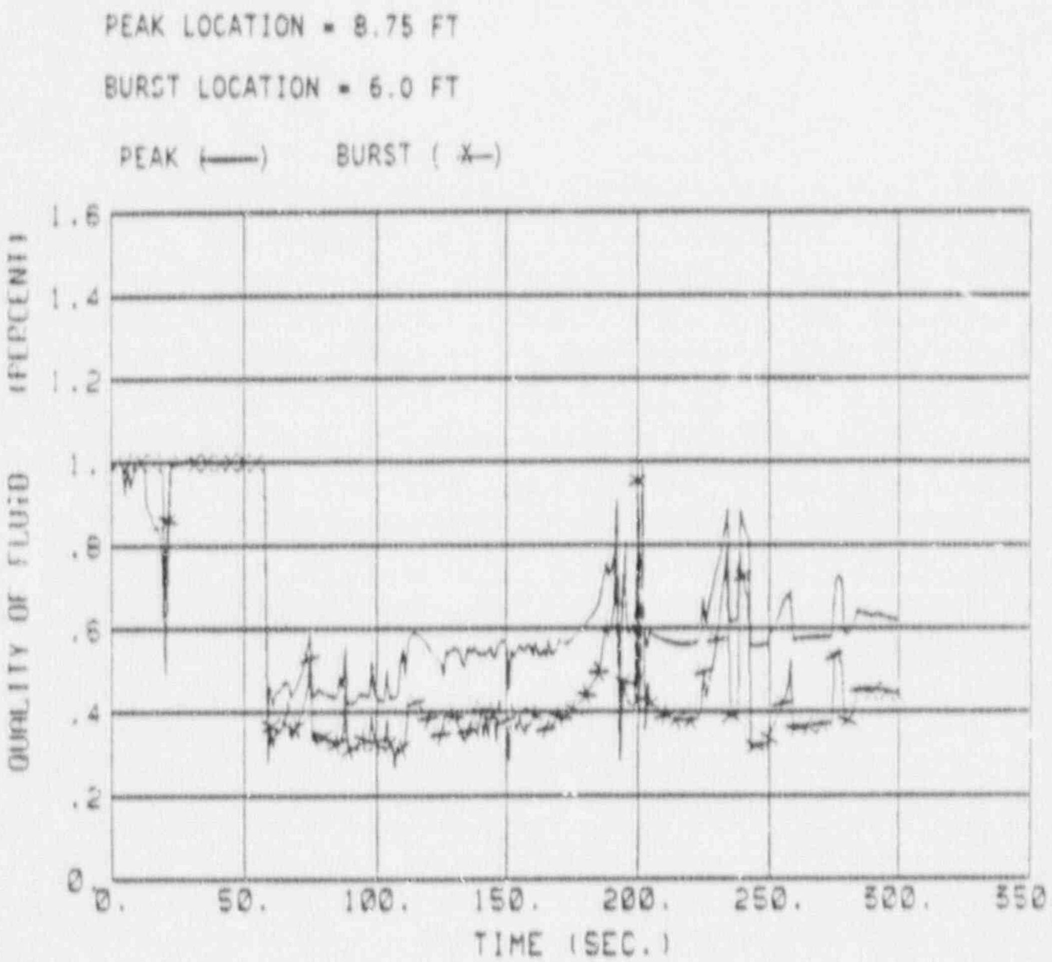


Georgia Power

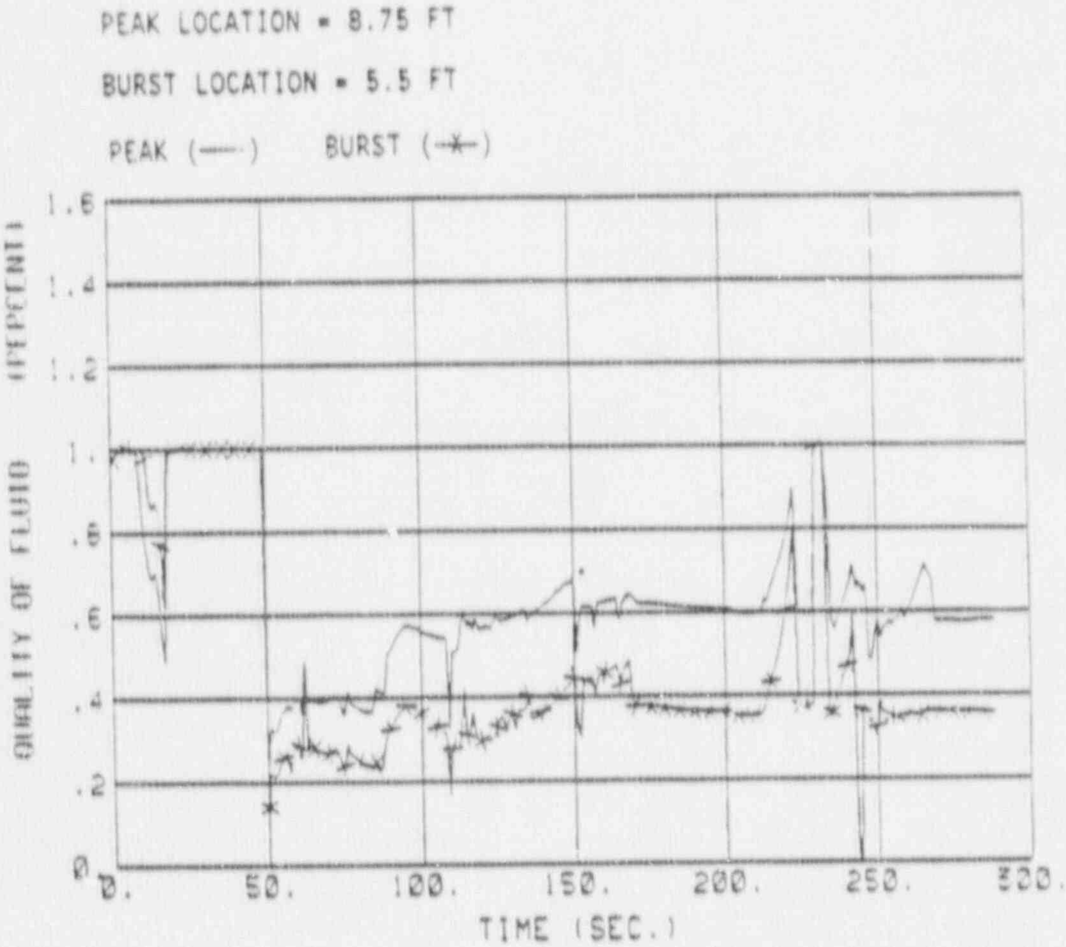
VOGTLE
ELECTRIC GENERATING PLANT
UNIT 1 AND UNIT 2

FLUID QUALITY
DECLG (CD = 0.4)

FIGURE 15.6.5-33A



Georgia Power	VOGTLE ELECTRIC GENERATING PLANT UNIT 1 AND UNIT 2	FLUID QUALITY DECLG (CD = 0.6)
		FIGURE 15.6.5-33B

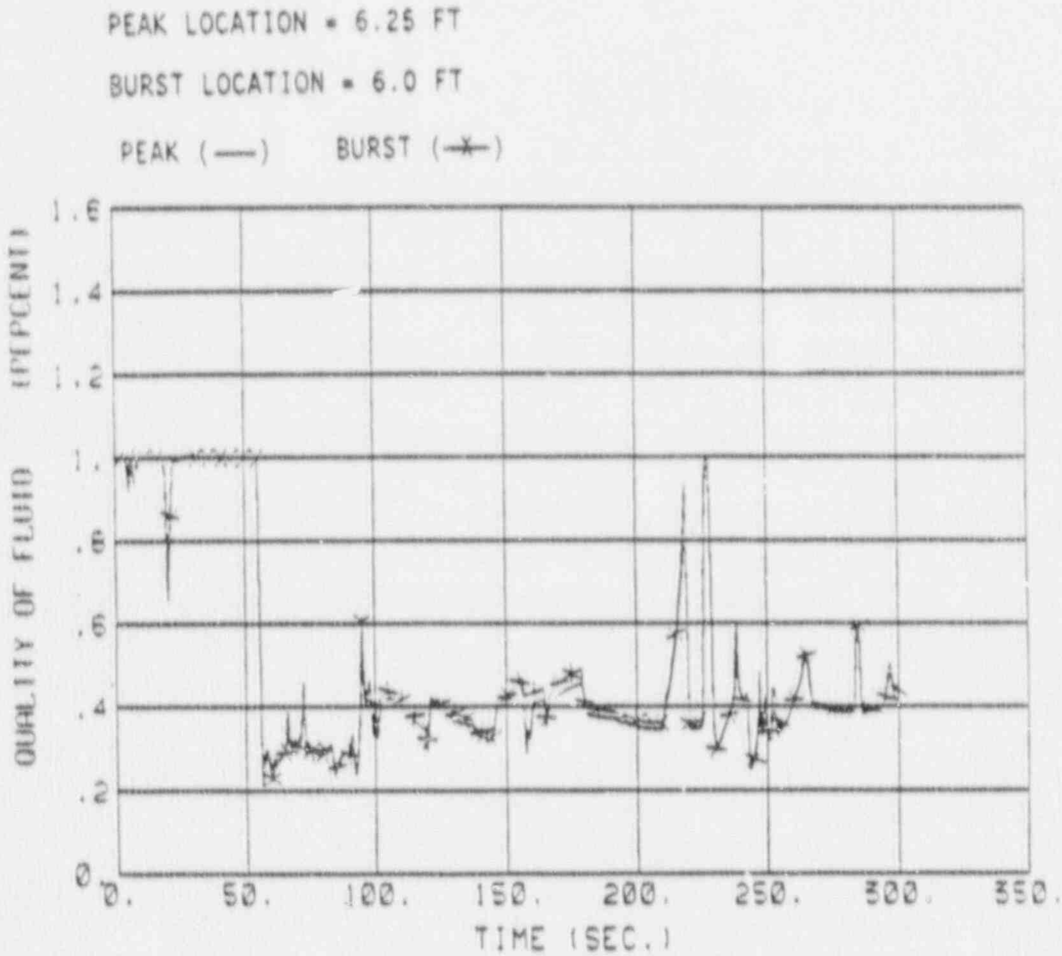


Georgia Power

VOGTLE
ELECTRIC GENERATING PLANT
UNIT 1 AND UNIT 2

FLUID QUALITY
DECLG (CD = 0.8)

FIGURE 15.6.5-33C

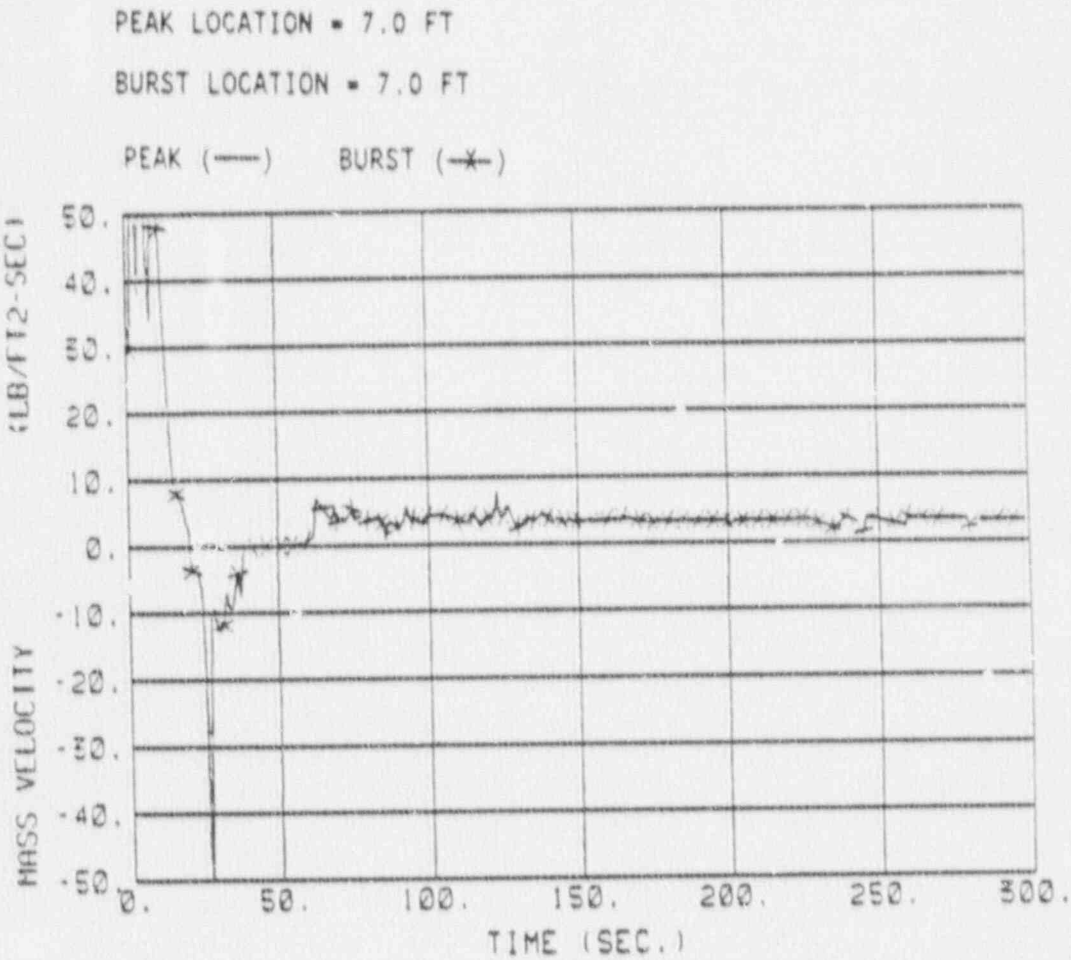


Georgia Power

VOGTLE
ELECTRIC GENERATING PLANT
UNIT 1 AND UNIT 2

FLUID QUALITY
DECLG (CD = 0.6
MAXIMUM SAFETY INJECTION)

FIGURE 15.6.5-33D

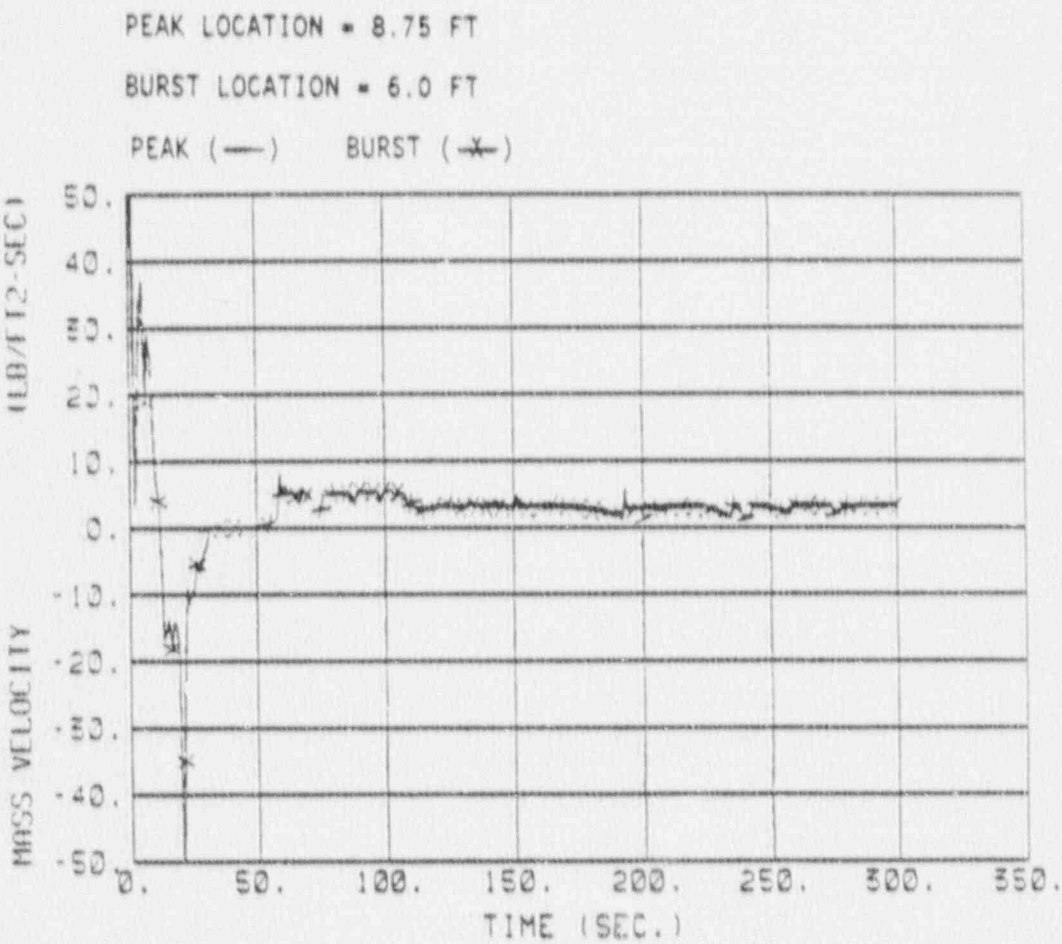


Georgia Power

VOGTLE
ELECTRIC GENERATING PLANT
UNIT 1 AND UNIT 2

MASS VELOCITY
DECLG (CD = 0.4)

FIGURE 15.6.5-34A

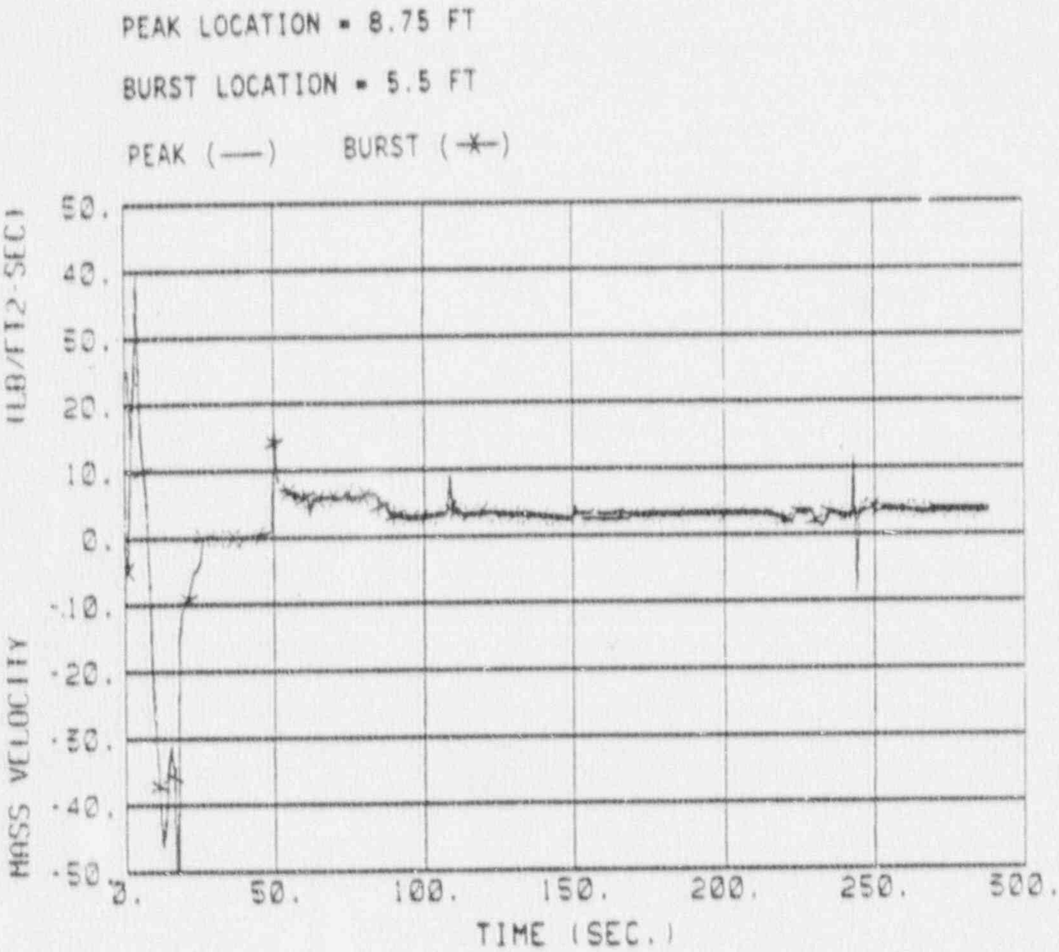


Georgia Power

VOGTLE
ELECTRIC GENERATING PLANT
UNIT 1 AND UNIT 2

MASS VELOCITY
DECLG (CD = 0.6)

FIGURE 15.6.5-34B

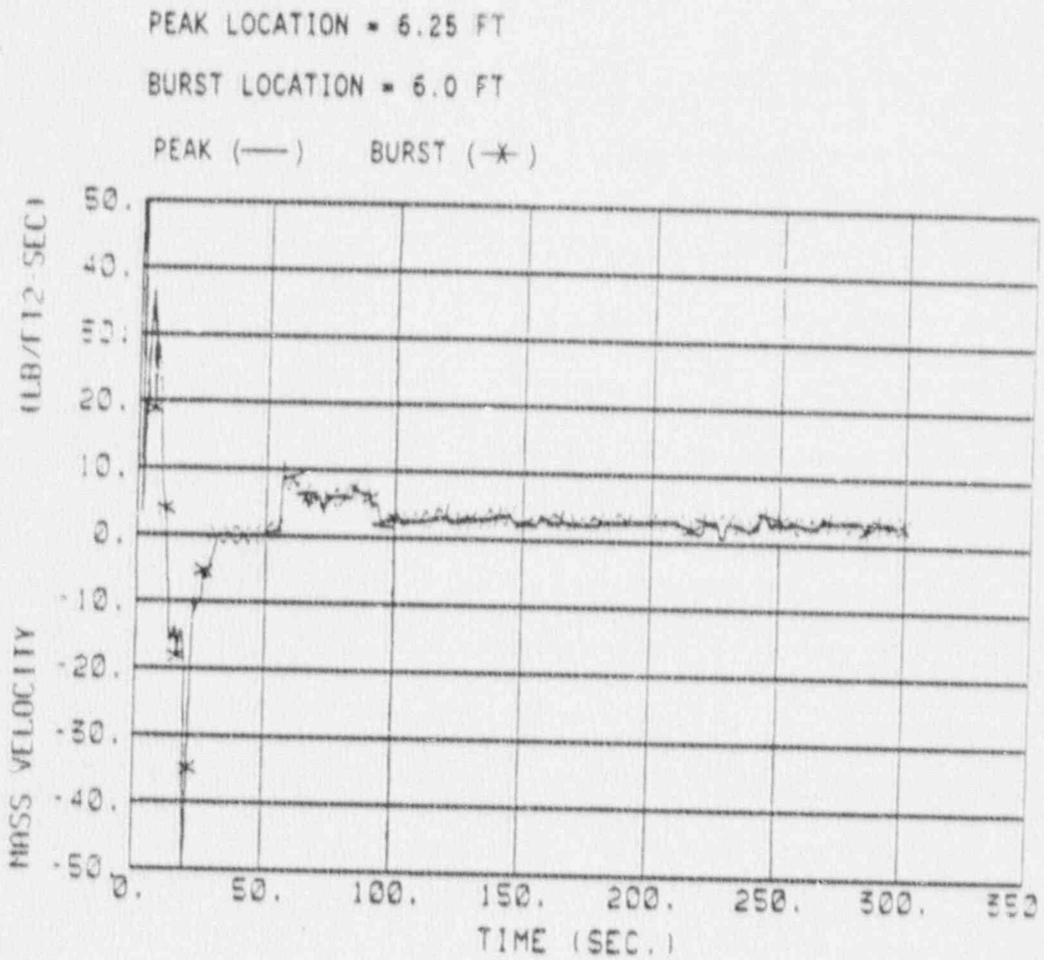


Georgia Power

VOGTLE
ELECTRIC GENERATING PLANT
UNIT 1 AND UNIT 2

MASS VELOCITY
DECLG (CD = 0.8)

FIGURE 15.6.5-34C

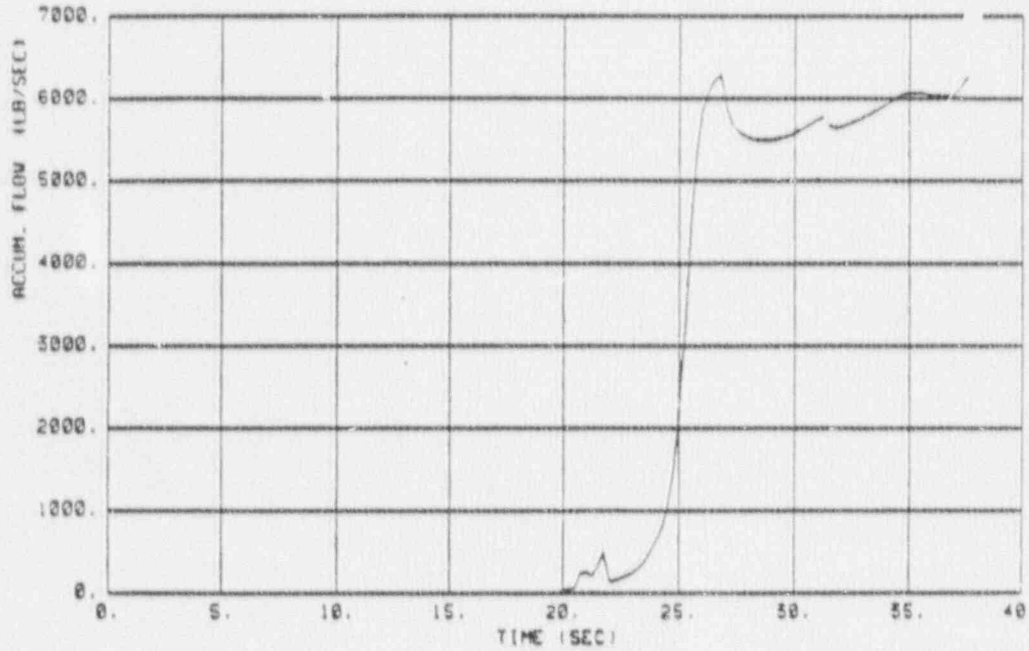


Georgia Power

VOGTLE
ELECTRIC GENERATING PLANT
UNIT 1 AND UNIT 2

MASS VELOCITY
DECLG (CD = 0.6
MAXIMUM SAFETY INJECTION

FIGURE 15.6.5-34D

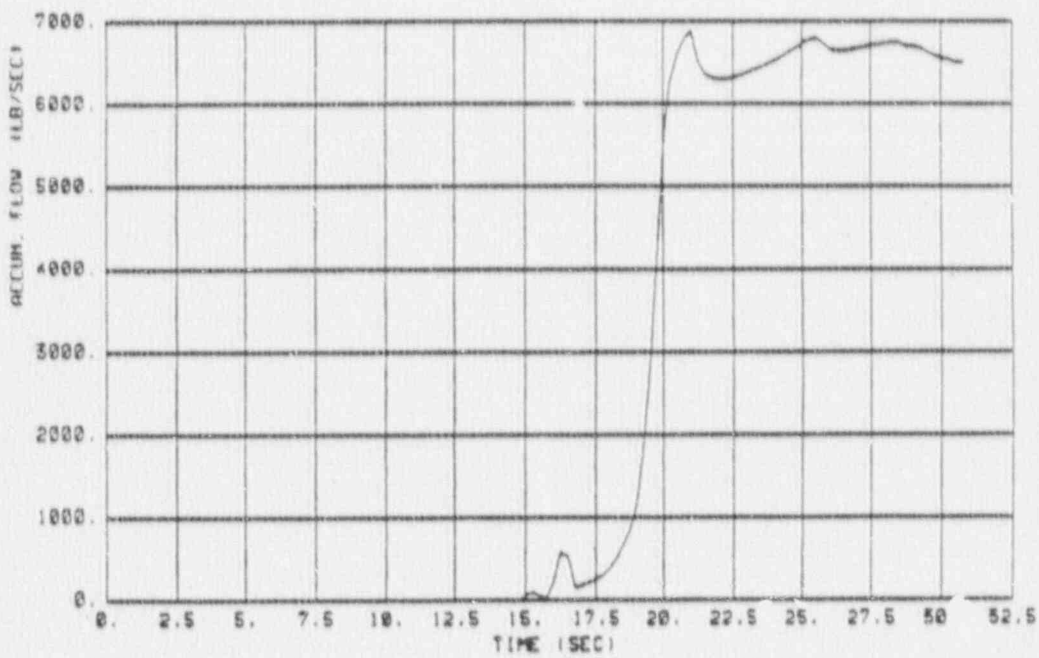


Georgia Power

VOGTLE
ELECTRIC GENERATING PLANT
UNIT 1 AND UNIT 2

ACCUMULATOR FLOW (BLOWDOWN)
DECLG (CD = 0.4)

FIGURE 15.6.5-35A

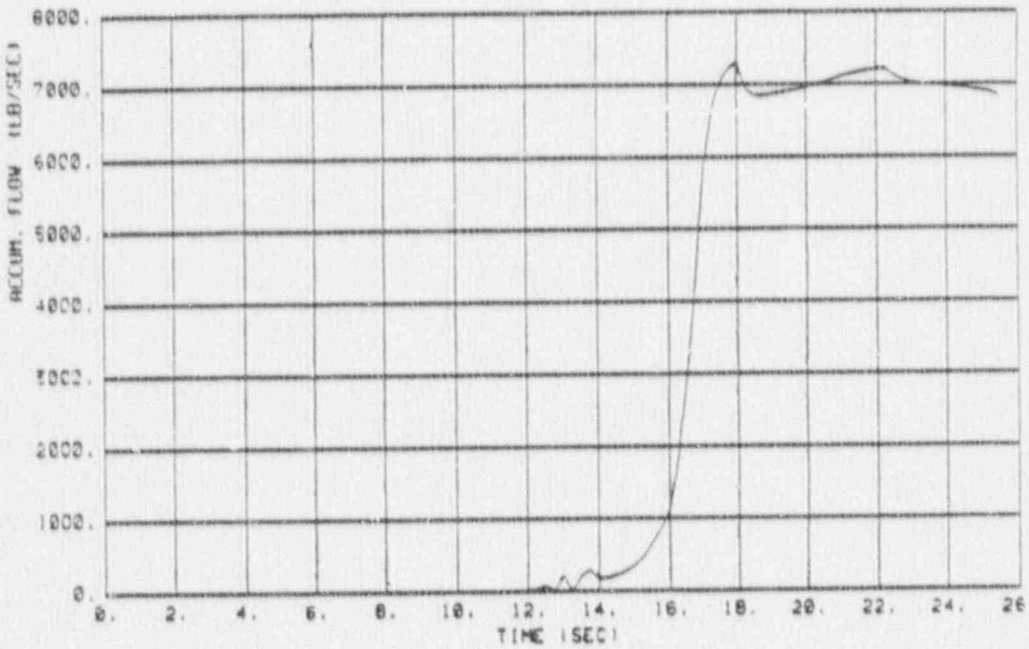


Georgia Power

VOGTLE
ELECTRIC GENERATING PLANT
UNIT 1 AND UNIT 2

ACCUMULATOR FLOW (BLOWDOWN)
DECLG (CD = 0.6)

FIGURE 15.6.5-35B

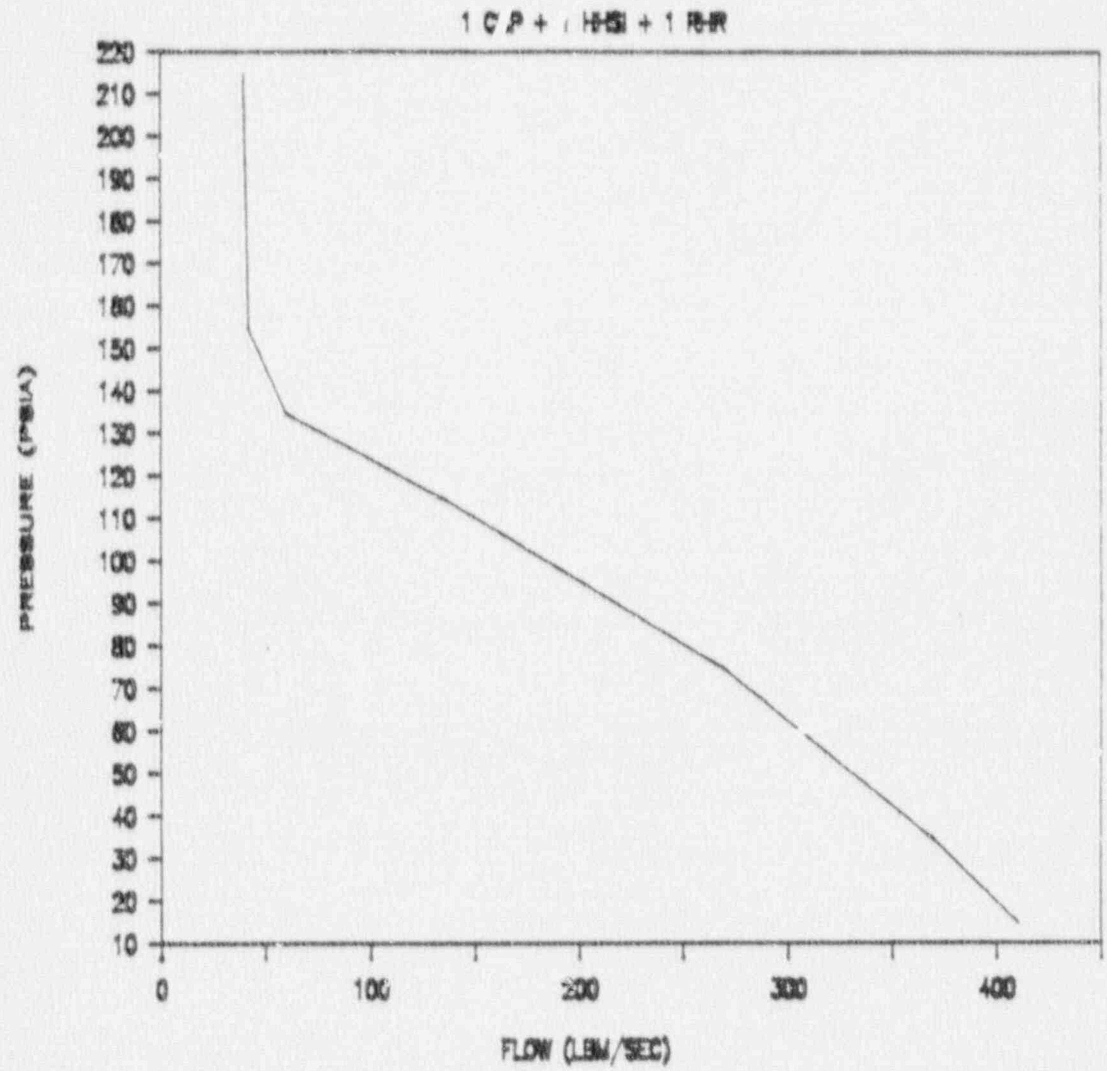


Georgia Power

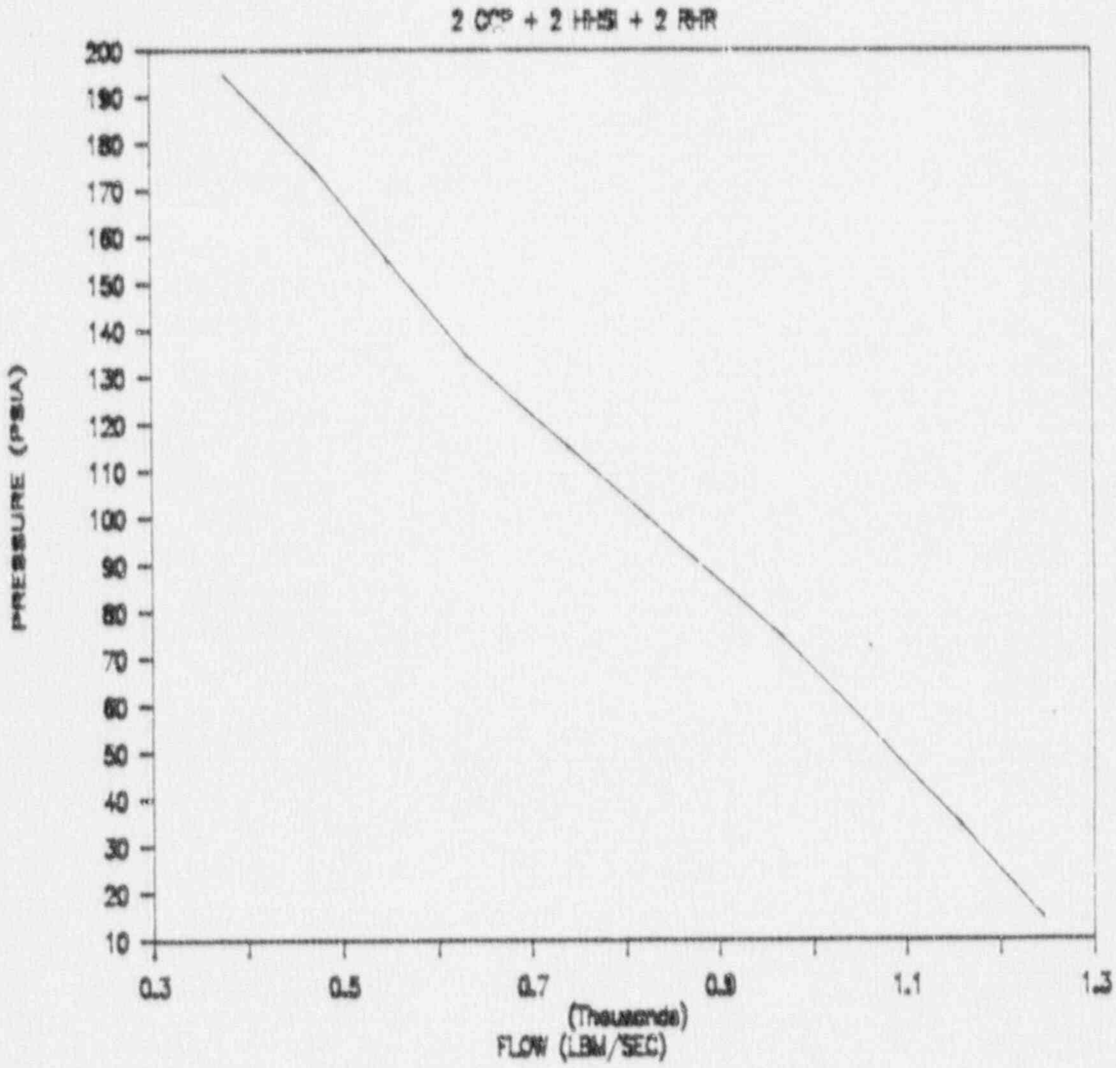
VOGTLE
ELECTRIC GENERATING PLANT
UNIT 1 AND UNIT 2

ACCUMULATOR FLOW (BLOWDOWN)
DECLG (CD = 0.8)

FIGURE 15.6.5-35C



Georgia Power VOGTLE ELECTRIC GENERATING PLANT UNIT 1 AND UNIT 2	MINIMUM SAFETY INJECTION (PRESSURE VERSUS FLOW) FIGURE 15.6.5-36A

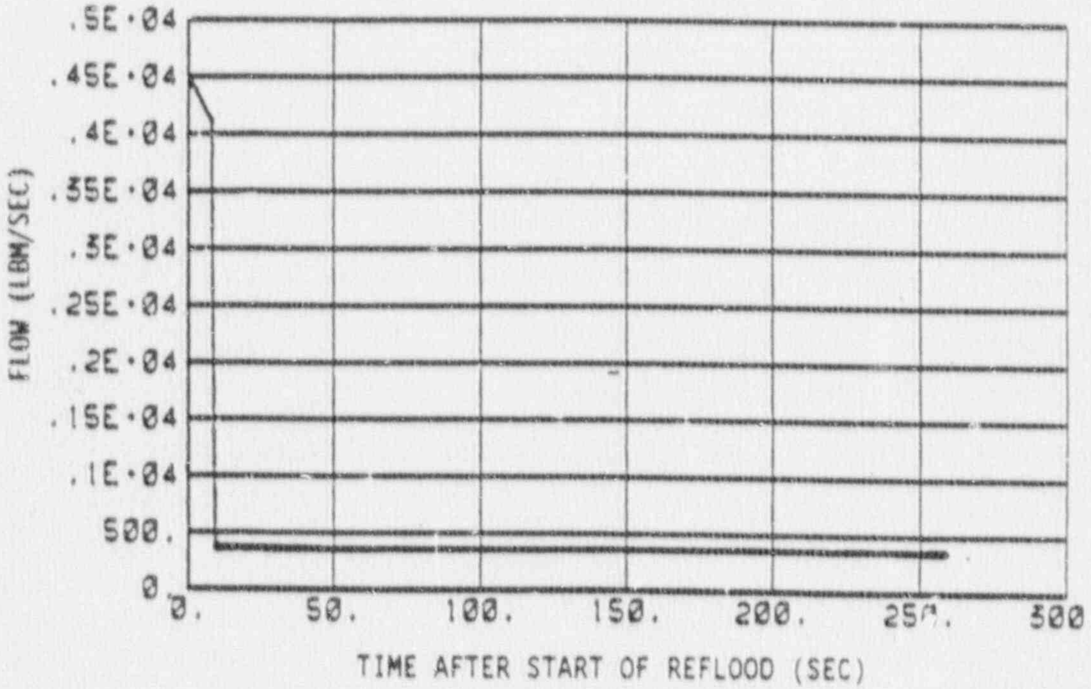


Georgia Power

VOGTLE
ELECTRIC GENERATING PLANT
UNIT 1 AND UNIT 2

MAXIMUM SAFETY INJECTION
(PRESSURE VERSUS FLOW)

FIGURE 15.6.5-36B

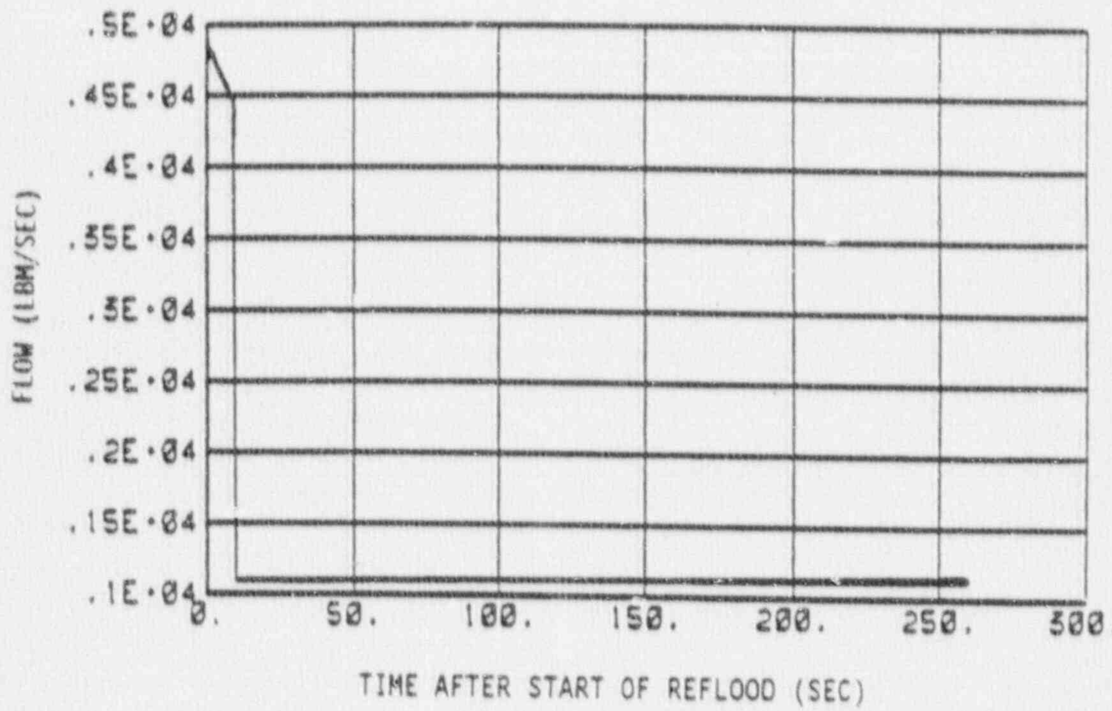


Georgia Power

VOGTLE ELECTRIC GENERATING PLANT UNIT 1 AND UNIT 2

COMBINED ACCUMULATOR AND ECCS PUMPED SAFETY INJECTION (CD = 0.6 MINIMUM SAFEGUARDS)

FIGURE 15.6.5-36C

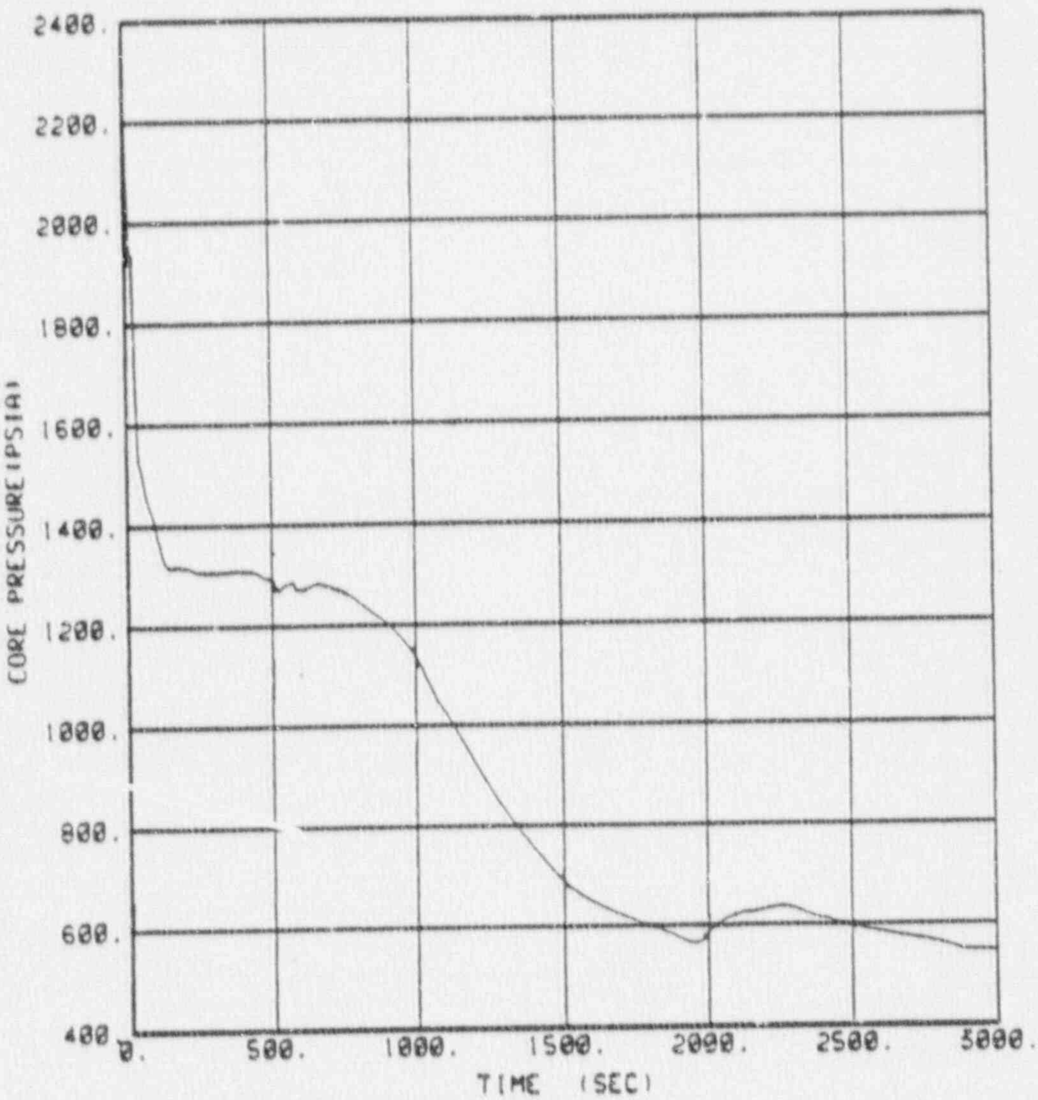


Georgia Power

VOGTLE
ELECTRIC GENERATING PLANT
UNIT 1 AND UNIT 2

COMBINED ACCUMULATOR AND ECCS PUMPED SAFETY INJECTION (CD = 0.6 MAXIMUM SAFEGUARDS)

FIGURE 15.6.5-36D

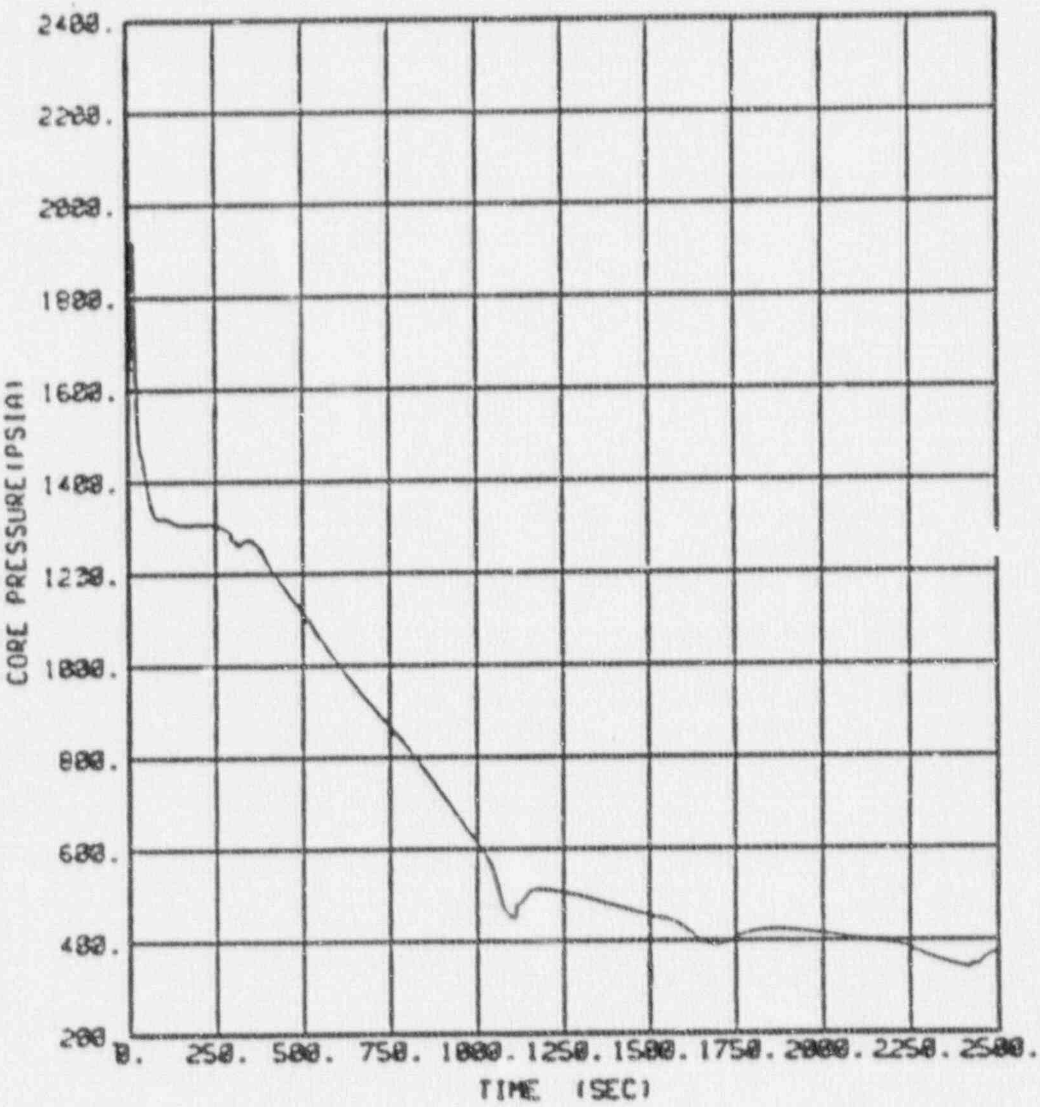


Georgia Power

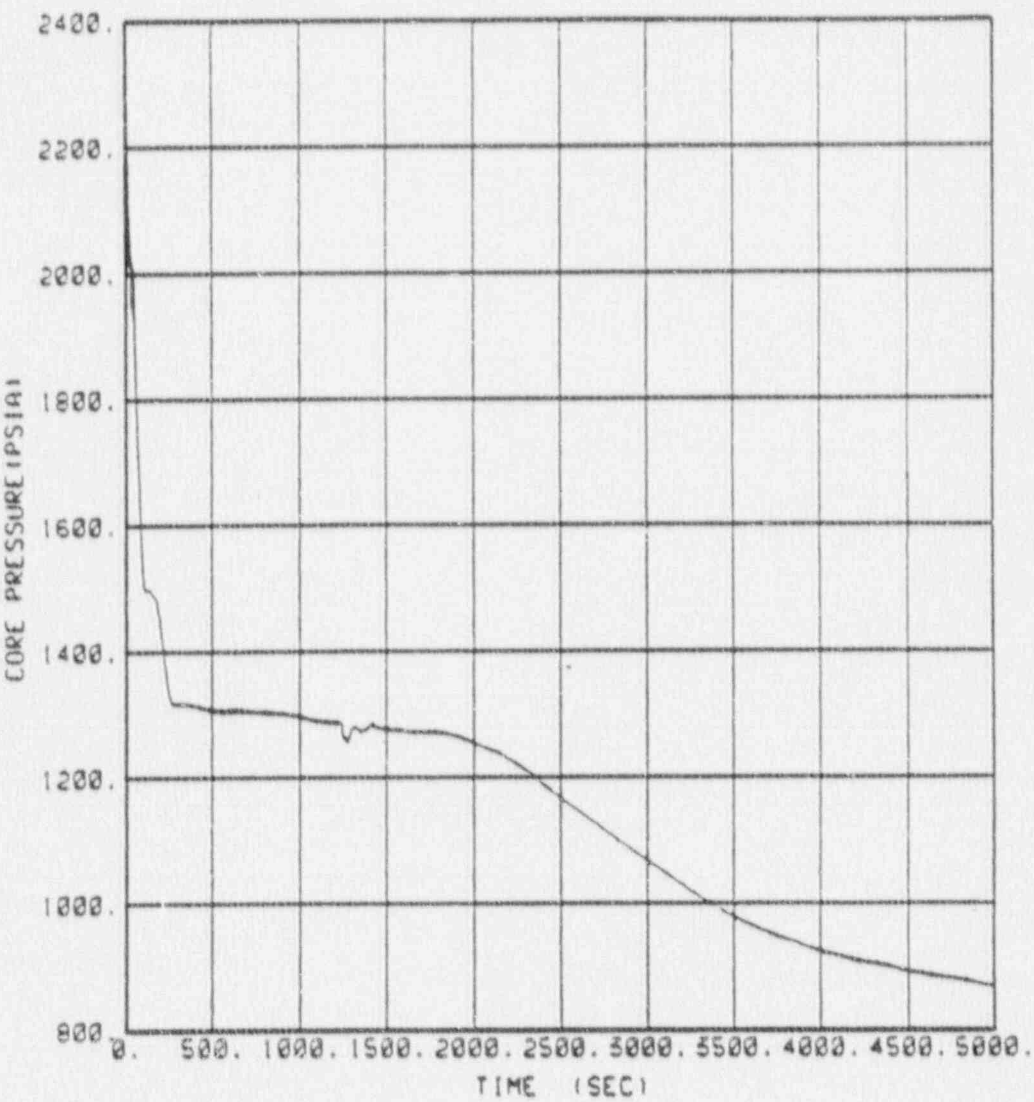
VOGTLE
ELECTRIC GENERATING PLANT
UNIT 1 AND UNIT 2

RCS D. RESSURIZATION TRANSIENT
(3-INCH BREAK)

FIGURE 15.6.5-37



Georgia Power	VOGTLE ELECTRIC GENERATING PLANT UNIT 1 AND UNIT 2	RCS DEPRESSURIZATION TRANSIENT (4-INCH BREAK)
		FIGURE 15.6.5-38

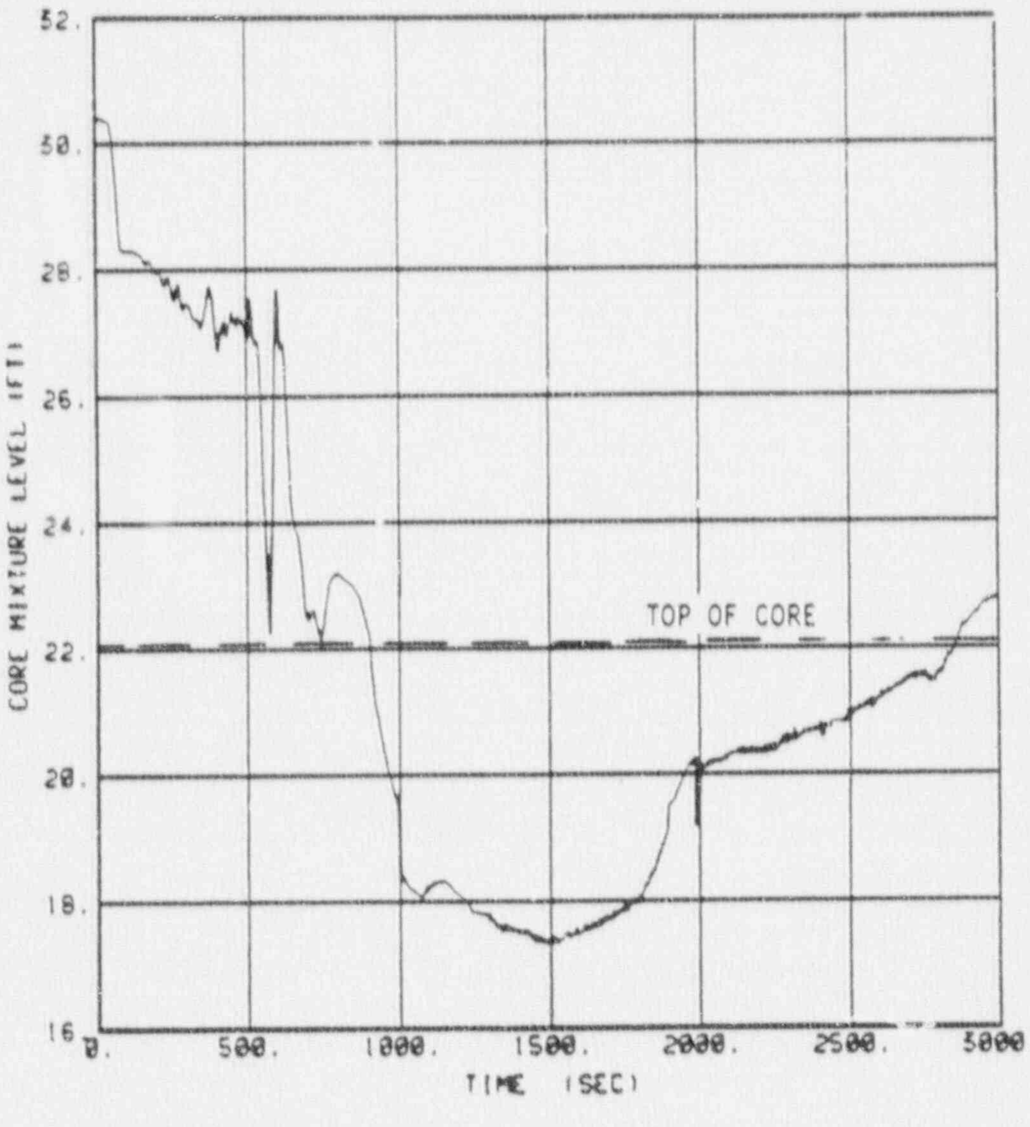


Georgia Power

VOGTLE
ELECTRIC GENERATING PLANT
UNIT 1 AND UNIT 2

RCS DEPRESSURIZATION TRANSIENT
(2-INCH BREAK)

FIGURE 15.6.5-39

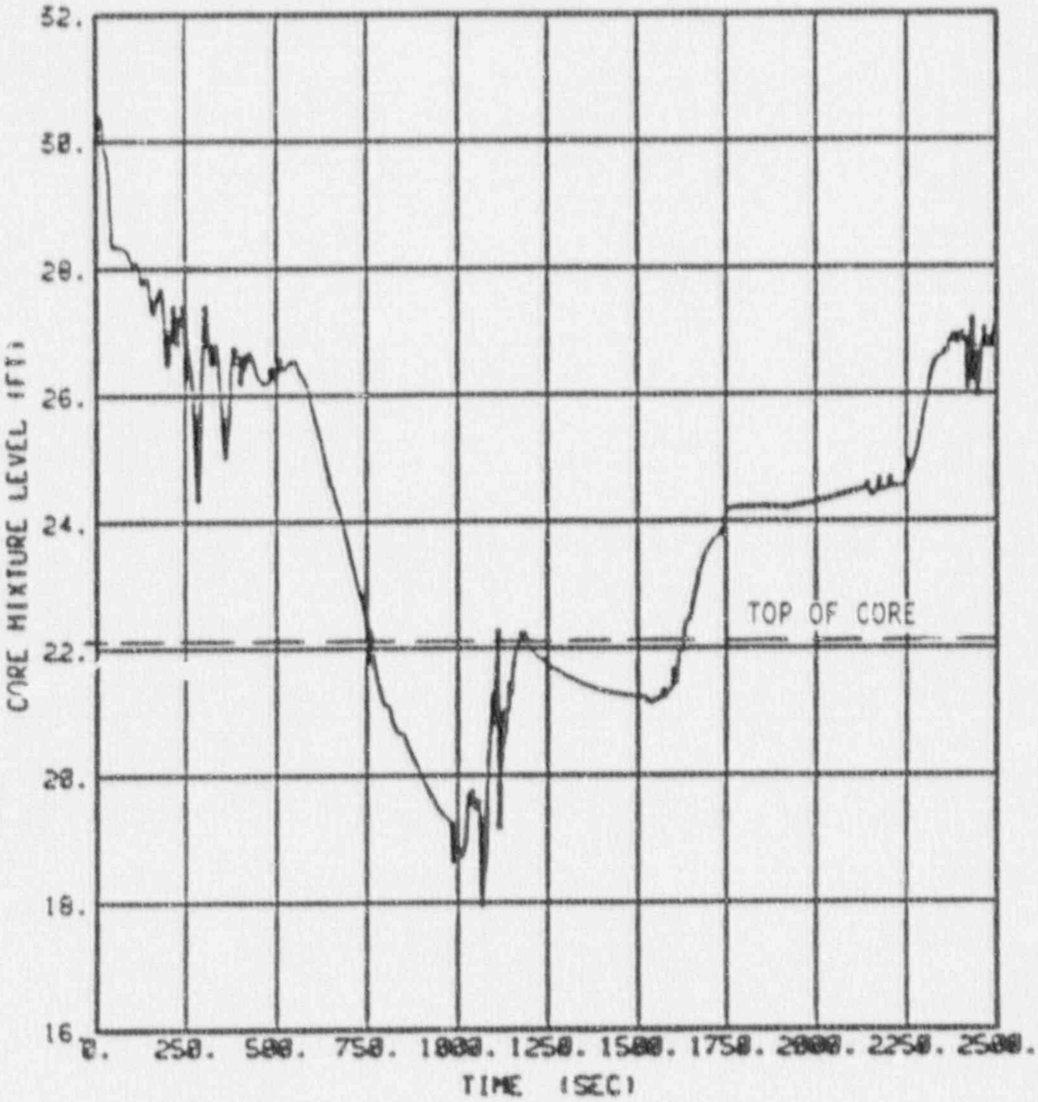


Georgia Power

VOGTLE
ELECTRIC GENERATING PLANT
UNIT 1 AND UNIT 2

CORE MIXTURE LEVEL
(3-INCH BREAK)

FIGURE 15.6.5-40

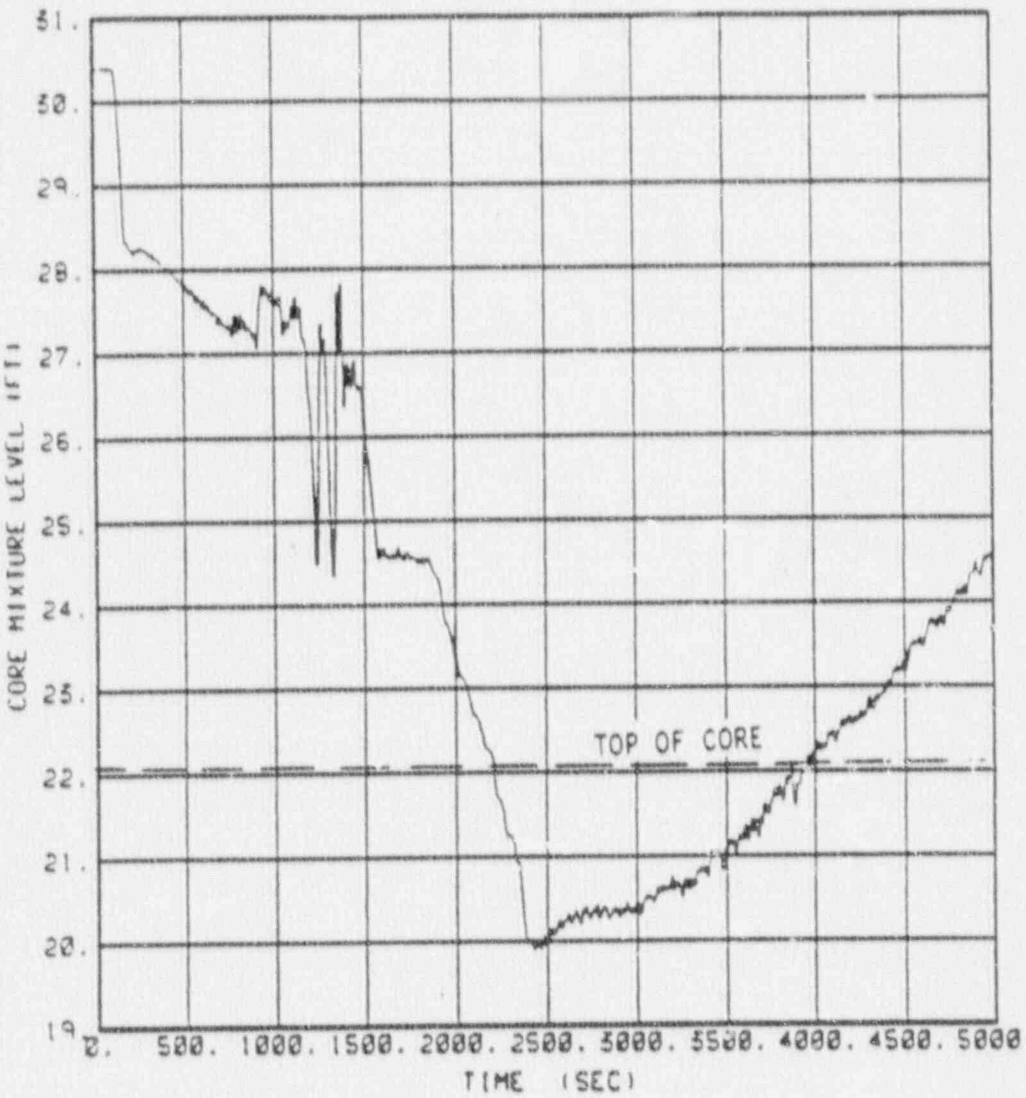


Georgia Power

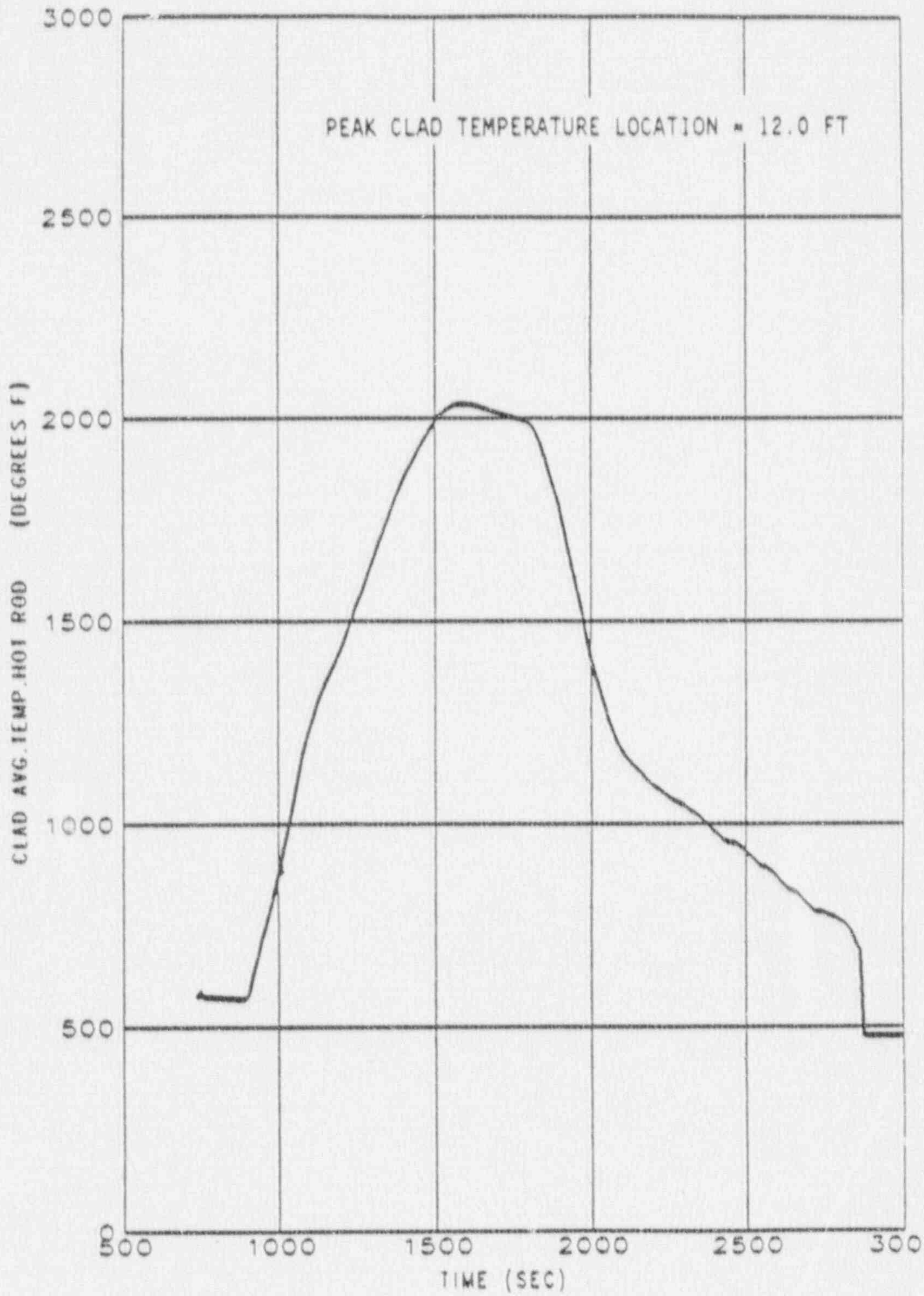
VOGTLE
ELECTRIC GENERATING PLANT
UNIT 1 AND UNIT 2

CORE MIXTURE LEVEL
(4-INCH BREAK)

FIGURE 15.6.5-41



Georgia Power VOGTLE ELECTRIC GENERATING PLANT UNIT 1 AND UNIT 2	CORE MIXTURE LEVEL (2-INCH BREAK)
	FIGURE 15.6.5-42

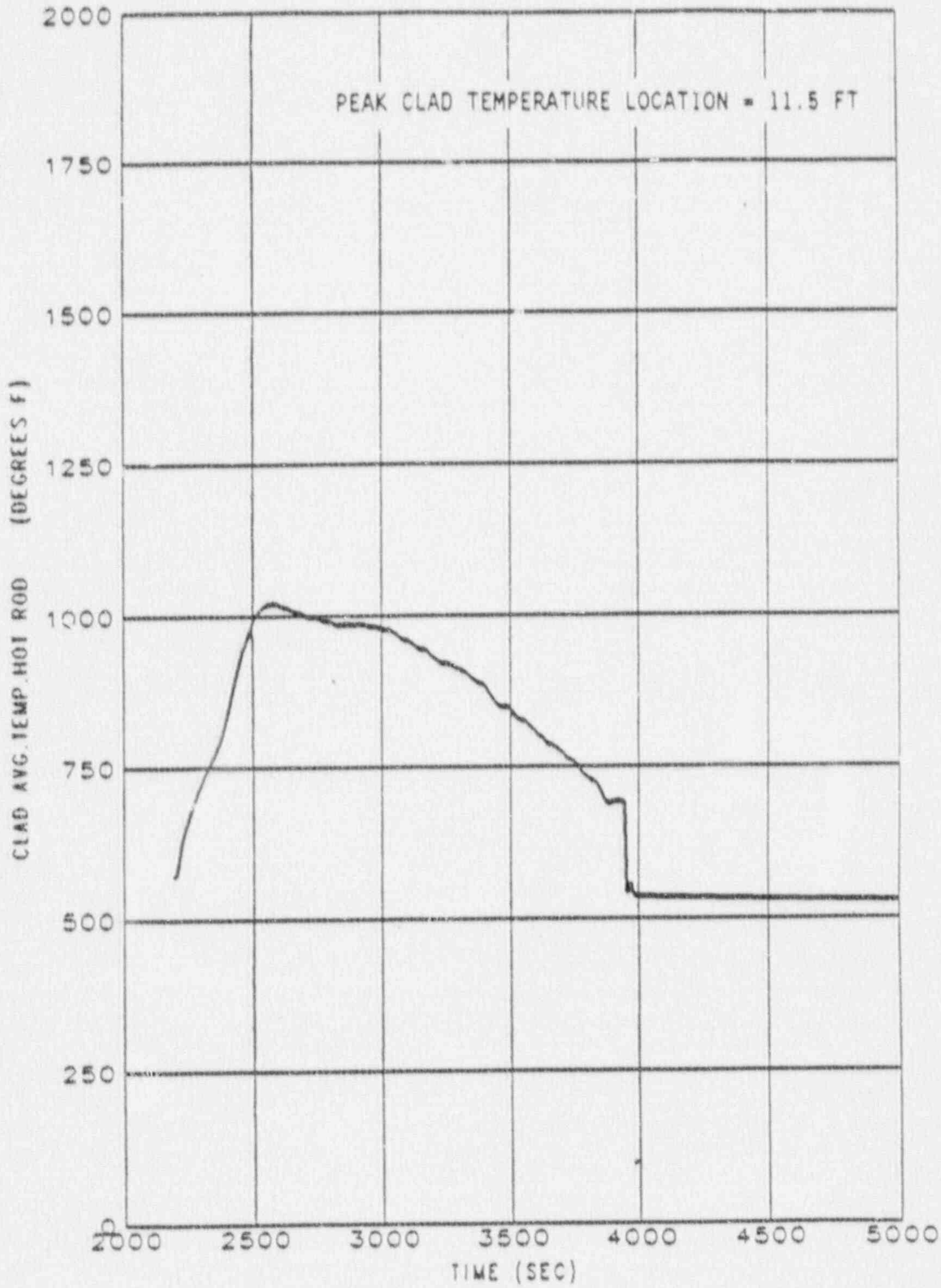


GEORGIA POWER

VOGTLE
ELECTRIC GENERATING PLANT
UNITS 1 AND 2

CLAD AVERAGE TEMPERATURE
THREE INCH BREAK - HOT ROD

FIGURE 15.6.5-43

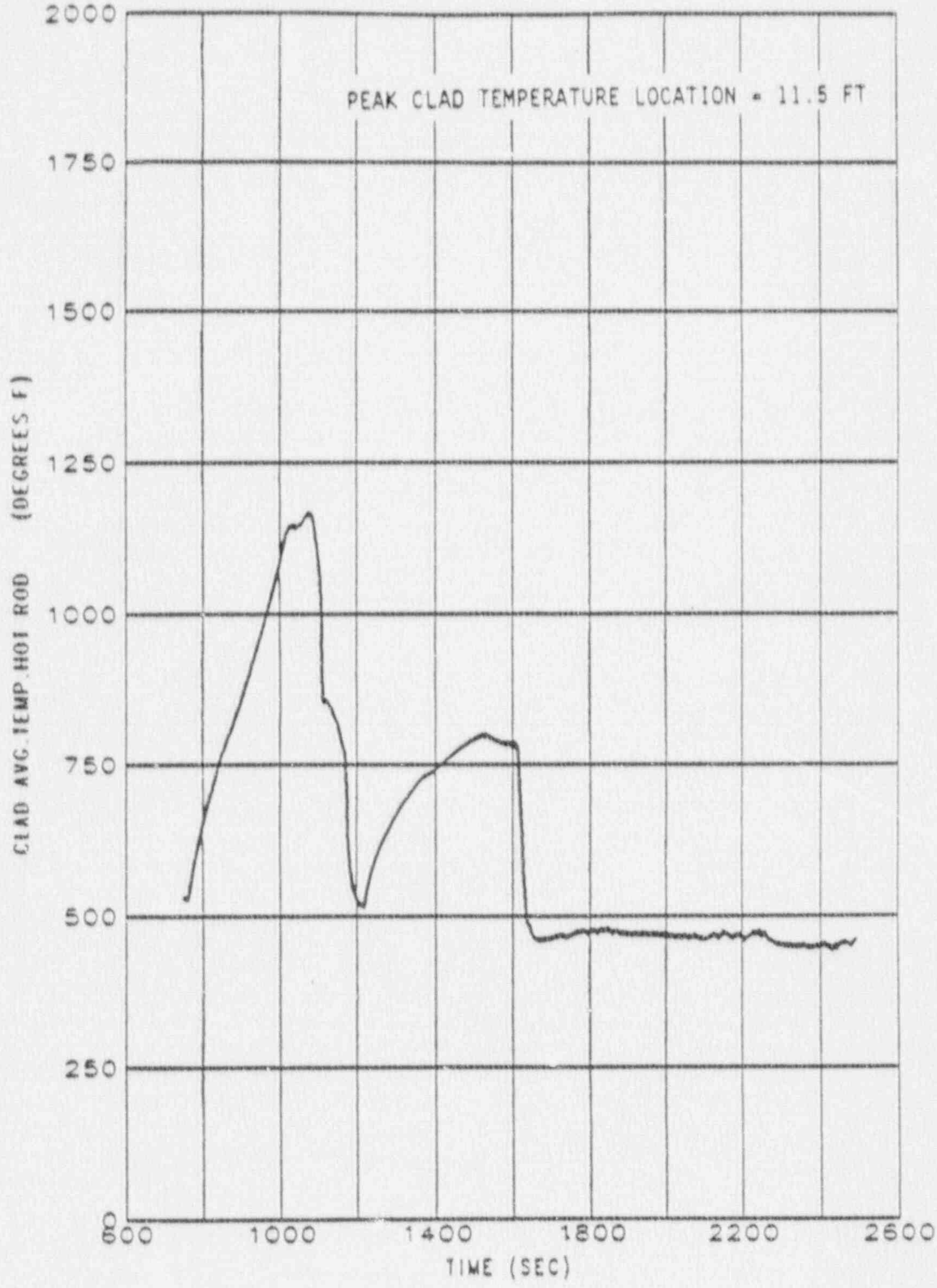


GEORGIA POWER

VOGTLE
ELECTRIC GENERATING PLANT
UNITS 1 AND 2

CLAD AVERAGE TEMPERATURE
TWO INCH BREAK - HOT ROD

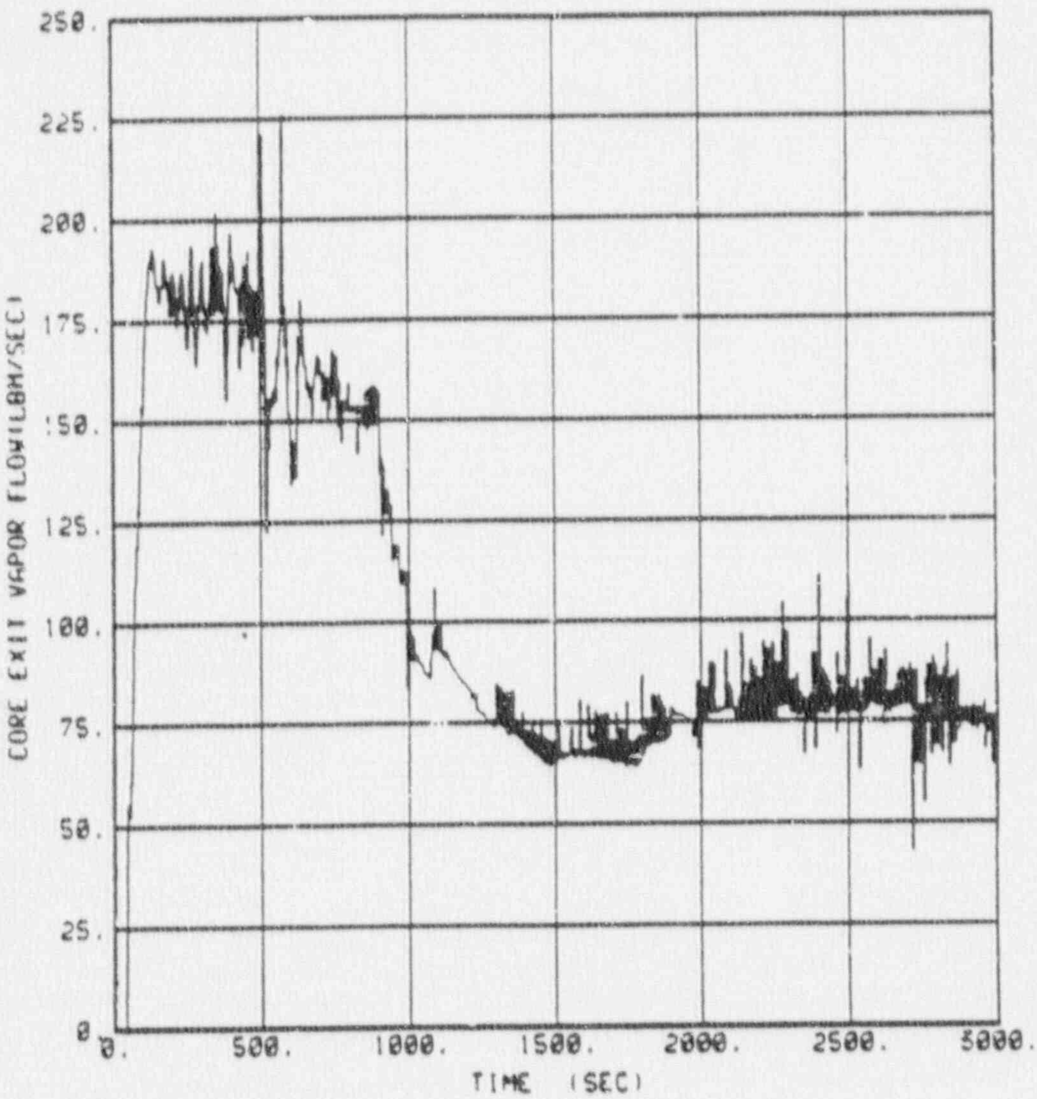
FIGURE 15.6.5-44



GEORGIA POWER
VOGTLE
ELECTRIC GENERATING PLANT
UNITS 1 AND 2

CLAD AVERAGE TEMPERATURE
FOUR INCH BREAK - HOT ROD

FIGURE 15.6.5-45

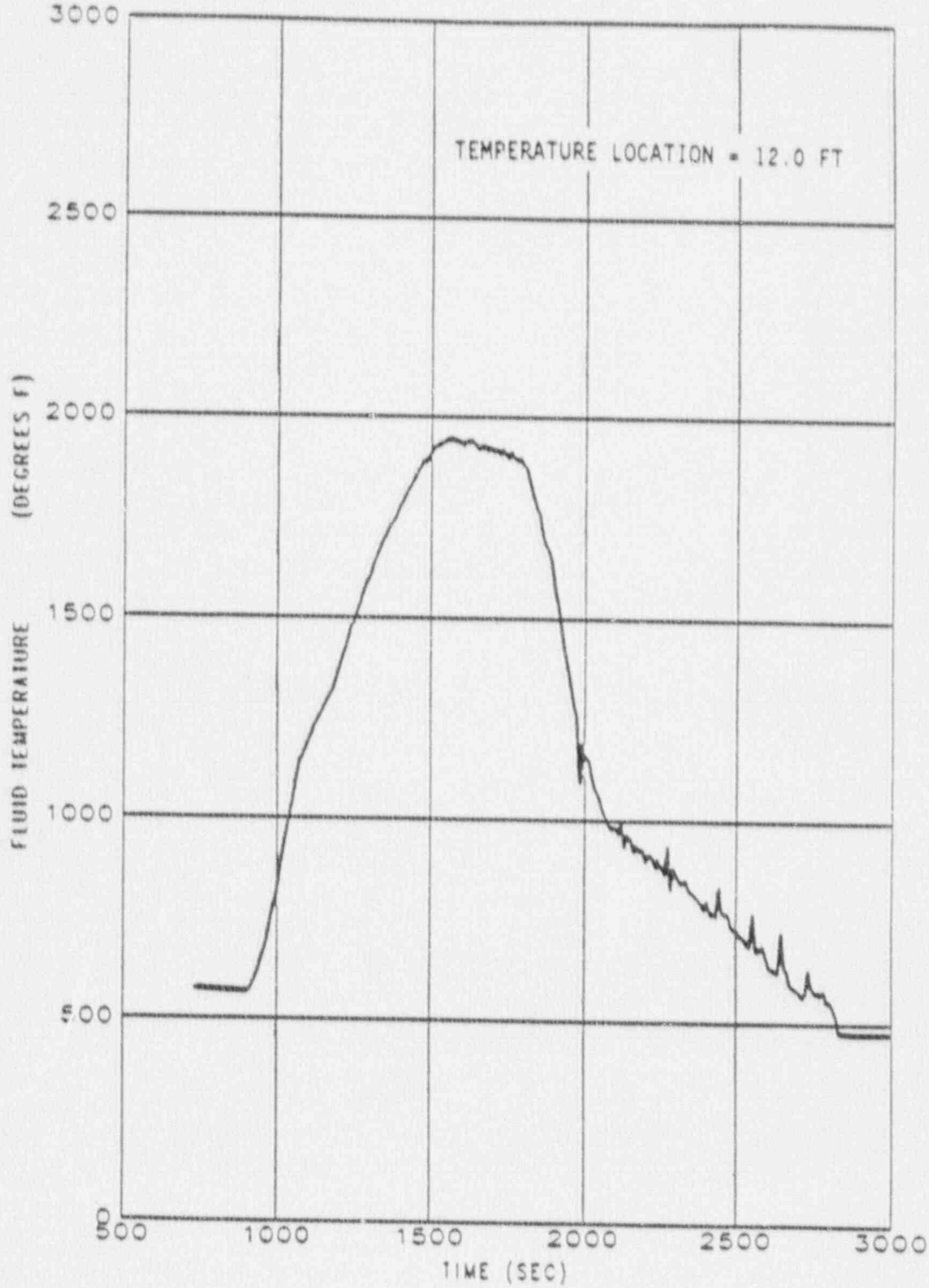


Georgia Power

VOGTLE
ELECTRIC GENERATING PLANT
UNIT 1 AND UNIT 2

CORE EXIT STEAM FLOW
(3-INCH BREAK)

FIGURE 15.6.5-46

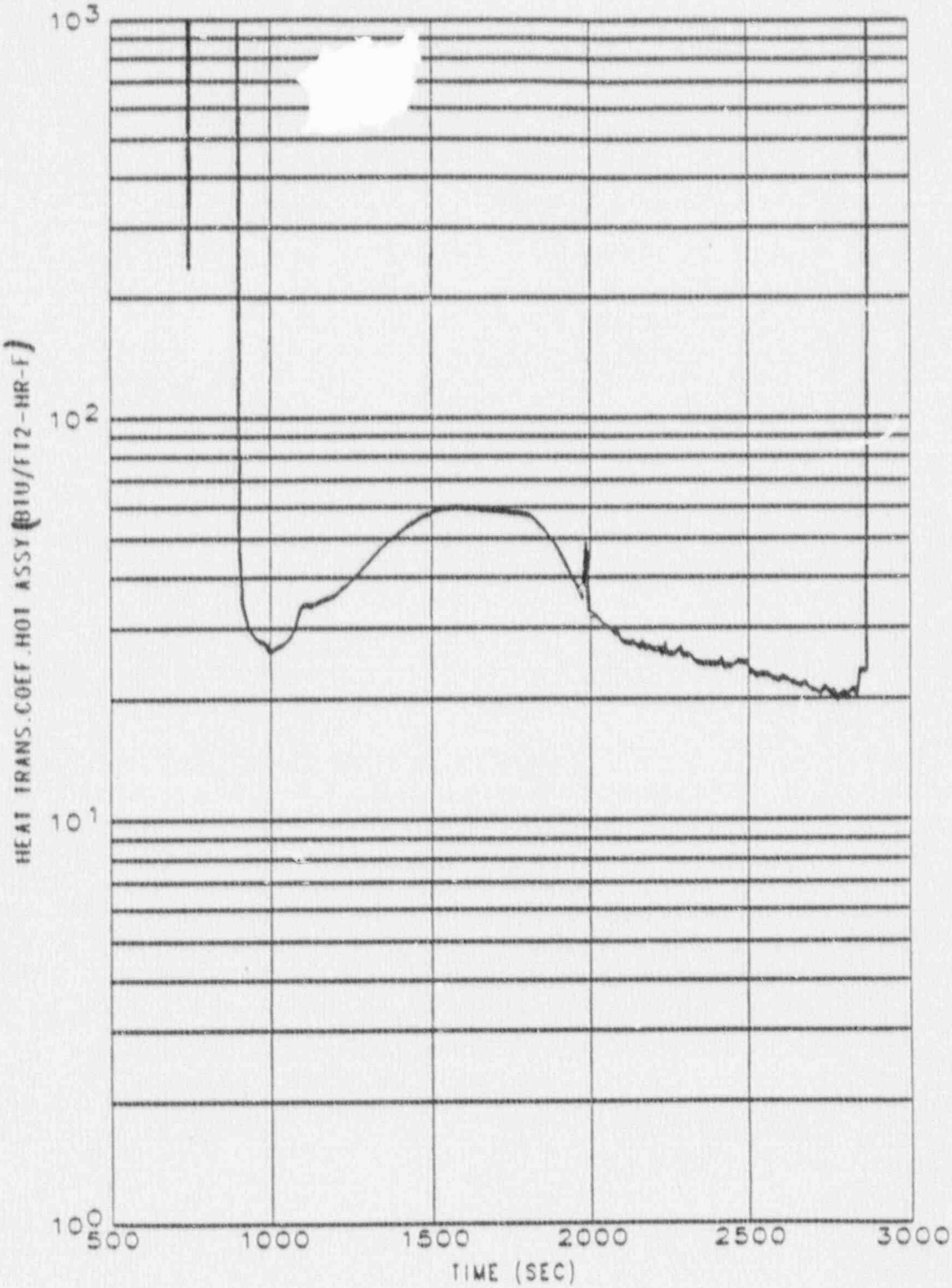


GEORGIA POWER

VOGTLE
ELECTRIC GENERATING PLANT
UNITS 1 AND 2

FLUID TEMPERATURE
THREE INCH BREAK - HOT SPOT

FIGURE 15.6.5-47



GEORGIA POWER

VOGTLE
ELECTRIC GENERATING PLANT
UNITS 1 AND 2

HEAT TRANSFER COEFFICIENT
THREE INCH BREAK - HOT ASSY

FIGURE 15.6.5-48

Appendix C

Radiological Assessment

for the Vogtle Electric Generating Plant Units 1 and 2

Transition to Westinghouse 17x17 VANTAGE-5 Fuel Assemblies

Radiological Assessment

This Appendix addresses the radiological effect of the change to extended fuel burnup associated with the use of VANTAGE 5 fuel and the other changes being implemented at this time in the VEGP Units 1 and 2.

1.0 Effect of Extended Fuel Burnup on Accident Doses

The extension of fuel burnup has been shown to have negligible effect on the core inventory of radioactive isotopes which are of concern in evaluating the radiological consequences of accidents (i.e., the short half-life noble gases and iodines). This has been documented both by Westinghouse, Reference 1, and the NRC, References 2 and 3. These reviews considered average burnups of up to 60,000 MWD/MTU for the lead rod which bounds the VANTAGE 5 and LOPAR fuel design to be provided for the VEGP Units 1 and 2.

Also of concern in evaluating the effect of extended fuel burnup on the radiological consequences of accidents is the fraction of core activity that is assumed to migrate out of the fuel matrix and into the fuel-clad gap region (i.e., the fuel-clad gap fraction) and thus be available for release in the event the cladding is breached. The accident analyses for the VEGP Units 1 and 2 have used the gap fractions recommended by Regulatory Guide 1.25. As indicated in Reference 1, these gap fractions remain extremely conservative for extended burnup fuel. Reference 2 agrees that the extended burnup fuel gap fractions for most isotopes of concern do not exceed the gap fractions specified in the Regulatory Guides.

The radiological consequences of accidents in the VEGP FSAR remain valid for both the VANTAGE 5 and LOPAR fuel assembly designs to 60,000 MWD/MTU lead rod, average burnups. The only exceptions are the Fuel Handling Accident (FHA) and the Steam Generator Tube Rupture (SGTR) doses which are discussed below.

2.0 Fuel Handling Accident (FHA)

References 2 and 3 make an exception in the case of I-131 for the Fuel Handling Accident (FHA) and specify a gap fraction of 12% which is a 20% increase over the Regulatory Guide 1.25 value. This increase is different from the analysis performed by Westinghouse which modeled the anticipated fuel management and determined a maximum gap fraction for I-131 of less than 2%.

Despite the fact that, as discussed above, no increase in the I-131 gap fraction is expected, the FHA doses have been recalculated to reflect the 20% increase in I-131 gap fraction as specified by References 2 and 3. Additionally, the recalculated doses reflect a conservative increase in radial peaking factor from 1.65 to 1.70 to assure that all future fuel designs are bounded by the analysis. In the new analysis, a reduction was made in the level of calculational conservatism; the pool scrubbing Decontamination Factor (DF) for elemental iodine was increased from 133 to 200. This revised pool scrubbing DF is a deviation from the guidance of Regulatory Guide 1.25 but is still conservative relative to Westinghouse tests (Reference 4) which support a pool scrubbing DF of 580. Additional conservatisms which remain in the FHA dose analysis are:

1. The gap fraction used for I-131 is 12% although the calculated peak gap fraction is less than 2%.

2. Every pin in the dropped assembly and 20% of an additional assembly are assumed to have cladding broken so as to release the gap inventory. Industry experience indicates that very little, if any, activity release would occur.

3. The thyroid doses for the FHA in the Fuel Handling Building are calculated assuming no credit for the charcoal filters. These filters are maintained in operable condition and would be expected to substantially reduce the releases of all forms of iodine.

4. The FHA analysis uses the Reg. Guide 1.25 assumption that 0.25% of the iodine in the assembly gap is in the organic form which is not subject to scrubbing removal in the pool. No organic iodine is expected to be present in the fuel.

5. In the Safety Evaluation Report, Reference 1, on the Westinghouse topical report for high burnup fuel, NRC stated that there would be about a 90% plateout of iodine in the gap (page 28 of Section A of Reference 1).

Taking into account the increased radial peaking factor, the increase in I-131 gap fraction, and the reduced conservatism in elemental iodine scrubbing by the pool, the revised FHA doses are as follows:

	Current FSAR Dose (rem)	Reanalysis Dose (rem)	SRP Acceptance Dose Limit (rem)
FHA Inside Fuel Building			
Exclusion Area Boundary			
Thyroid	71	73	75
Whole Body	0.27	0.29	6
Low Population Zone			
Thyroid	28	29	75
Whole Body	0.11	0.11	6

	Current FSAR Dose (rem)	Reanalysis Dose (rem)	SRP Acceptance Dose Limit (rem)
FHA Inside Containment			
Exclusion Area Boundary			
Thyroid	0.42	0.44	75
Whole Body	0.0016	0.0017	6
Low Population Zone			
Thyroid	0.17	0.18	75
Whole Body	0.00065	0.00069	6

3.0 Steam Generator Tube Rupture (SGTR) Event

The other plant changes proposed at this time (e.g., 10% tube plugging limit, etc.) do not affect any of the accident radiological consequence analyses except for the Steam Generator Tube Rupture (SGTR). The SGTR doses were reanalyzed to address the revised accident break flows and steam releases (Appendix B). The thyroid doses increased slightly and the gamma body and beta skin doses decreased slightly. All doses maintained a large margin to the acceptance criteria specified in Section 15.6.3 of the NRC's Standard Review Plan.

A summary of the doses is provided below:

	Current FSAR Dose (rem)	Reanalysis Dose (rem)	SRP Acceptance Dose Limit (rem)
Exclusion Area Boundary			
Thyroid (accident initiated spike)	4.8	4.8	30
Thyroid (pre-existing iodine spike)	31.4	33.1	300
Whole Body	0.07	0.07	2.5
Skin	0.17	0.16	Not Specified
Low Population Zone			
Thyroid (accident initiated spike)	1.9	2.0	30
Thyroid (pre-existing iodine spike)	12.6	13.3	300
Whole Body	0.03	0.03	2.5
Skin	0.07	0.06	Not Specified

4.0 References

1. WCAP-10125-P-A, "Extended Burnup Evaluation of Westinghouse Fuel," December 1985.

2. NUREG/CR-5009, "Assessment of the Use of Extended Burnup Fuel in Light Water Power Reactors," February 1988.

3. Federal Register/Vol. 53 No. 39/Monday, February 29, 1988/Pages 6040-6043.

4. WCAP-7828, "Radiological Consequences of a Fuel Handling Accident," December 1971 (non-proprietary).

Enclosure 5

Vogtle Electric Generating Plant Units 1 and 2

Request for Technical Specifications Changes

VANTAGE-5 Fuel Design

Environmental Evaluation

ENCLOSURE 5

VOGTLE ELECTRIC GENERATING PLANT
REQUEST FOR TECHNICAL SPECIFICATIONS CHANGES
VANTAGE-5 FUEL DESIGN

ENVIRONMENTAL EVALUATION

The generic environmental effects of the uranium fuel cycle are provided in Table S-3 of 10 CFR 51.51, "Uranium Fuel Cycle Environmental Data," and the environmental impact of transportation of the fuel and waste to and from a reference reactor is provided in Table S-4 of 10 CFR 51.52, "Environmental Effects of Transportation of Fuel and Waste." These regulations currently limit fuel burnup to 33 GWD/MTU and fuel enrichment to 4 weight percent U-235. By converting to VANTAGE 5 fuel, GPC anticipates extended fuel burnups to greater than 33 GWD/MTU and the use of fuel with initial enrichments to 4.55 weight percent U-235. The use of fuel enriched to 4.55 weight percent U-235 at VEGP Units 1 and 2 was approved by the NRC in License Amendments 24 and 5, respectively, dated October 10, 1989.

The environmental effects of extended burnup and higher initial enrichments have previously been addressed by the NRC. A notice published in the Federal Register on February 29, 1988 (53 FR 6054), states that the NRC's environmental assessment of extended fuel burnup is complete and that the environmental impacts summarized in Table S-3 of 10 CFR 51.51 and in Table S-4 of 10 CFR 51.52 bound the corresponding impacts for burnup levels up to 60 GWD/MTU and enrichments up to 5 weight percent U-235.

The environmental impacts of transportation resulting from the use of extended burnup and higher enrichment fuel were further addressed in the Federal Register on August 11, 1988 (53 FR 30355). This notice reiterated the conclusion stated in 53 FR 6054 and further concluded that there are no significant adverse radiological or nonradiological impacts associated with the use of extended burnup and/or increased enrichment, and that burnup levels to 60 GWD/MTU and enrichments to 5 weight percent U-235 will not significantly affect the quality of the human environment. Moreover, pursuant to 10 CFR 51.31, "Determinations Based on Environmental Assessment," the NRC determined that an environmental impact statement need not be prepared for this action.

As stated in 53 FR 30355, the NRC is in the process of revising the regulations in 10 CFR 51.52 to reflect the findings published in 53 FR 6054. In the interim, in connection with its review of proposed licensed amendments to permit use of fuel enriched to 5 weight percent U-235 and burnup levels to 60 GWD/MTU, and pursuant to 10 CFR 51.52(b), the NRC has elected to accept the analysis of environmental effects of the transportation of such fuel and waste provided in 53 FR 30355 until such time as the revision to the rule is issued.

10 CFR 51.22(c)(9) provides criterion for and identification of licensing and regulatory actions eligible for categorical exclusion. Such actions do not require an environmental assessment or an environmental impact statement. In accordance with criterion (c)(9), a proposed amendment to an operating license does not require an environmental assessment or an environmental impact statement provided that operation of the facility in accordance with the proposed amendment: (1) involves no significant hazards consideration, (2) there is no significant change in the types or significant increase in the amounts of any effluents that may be released offsite, and (3) there is no significant increase in individual or cumulative occupational radiation exposure. Based on the generic environmental evaluations provided by the NRC in the above cited Federal Register Notices, and the Significant Hazards Evaluation provided by GPC in Enclosure 2 to this submittal, GPC has determined that the proposed amendment meets the criteria for categorical exclusion setforth in 10 CFR 51.22(c)(9). Therefore, pursuant to 10 CFR 51.22 (b) an environmental assessment or environmental impact statement is not required in connection with the proposed license amendment.