APPENDIX 15A

15A.1 PURPOSE

The purpose of this appendix is to present assumptions and computer codes used in the analysis of certain accidents treated in Section 15.

15A.2 FORMAT

Each accident discussed will be presented in the order used in Section 15. Within each section, the following topics will be discussed: mathematical model, transport assumptions, computer codes used, and dose assumptions.

Table 15A-1 presents the isotopic core inventory for various decay times. Table 15A-2 presents the radionuclide concentrations used in the accident analyses.

15A.3 CONTROL ROD DROP ACCIDENT (Section 15.4.9)

This Section has been deleted. Equations 15A-1 through 15A-24 have been deleted. Pages 15A-2 through 15A-10 have been deleted.

15A.4 INSTRUMENT LINE FAILURE ACCIDENT (Section 15.6.2)

15A.4.1 Mathematical Model

15A.4.1.1 Design Basis Analysis

The design basis analysis is presented in Section 15.6.2.5.2.

15A.4.2 Transport Assumptions

15A.4.2.1 Design Basis Analysis

The transport assumptions are discussed in Section 15.6.2.5.2.2.

15A.4.2.2 Deleted

15A.4.3 Computer Codes Used

The computer code used to model transport in the Reactor Building and to the environment is the RADTRAD 3.02 computer code (NUREG/CR-6604).

15A.4.4 Dose Assumptions

The breathing rates used in this analysis of this accident are taken from Regulatory Guide 1.183, Revision 0. Specifically, they are:

0 to 8 hours	3.50E-4 m^3/s
8 to 24 hours	1.80E-4 m^3/s
greater than 24 hours	2.30E-4 m^3/s

The iodine dose conversion factors used for the thyroid inhalation and whole body doses for an adult were taken from Federal Guidance Reports (FGR) 11 and 12 respectively.

15A.5 STEAM LINE BREAK ACCIDENT (Section 15.6.4)

15A.5.1 Deleted

15A.5.2 Transport Assumptions

The transport assumptions are described in Sections 15.6.4.5.2.

15A.5.3 Computer Codes Used

The computer code used to model transport to the environment is the RADTRAD 3.02 computer code (NUREG/CR-6604).

15A.5.4 Dose Assumptions

The dose assumptions described in Section 15A.4.4 apply to this accident as well.

15A.6 LOSS-OF-COOLANT ACCIDENT (Section 15.6.5)

15A.6.1 Transport Assumptions

The reactor building exhaust rate is modeled as a step function. The value assumed for each step is the value the function would have at the beginning of the time period. The approximation continues until time t = 8 hours, at which time the exhaust rate is assumed to be a constant value of 3324 cfm.

The exhaust rate for each time step is doubled to account for 50% mixing in the Reactor Building and 10% is added to account for flow variation.

Initially, the FRVS exhaust rate is 19,800 cfm.

The ESF leakage is modeled assuming a constant leakage rate of 2 gpm. It is assumed that 10% (that is, 0.2 gpm) of the leakage becomes airborne.

15A.6.2 Computer Code Used

The RADTRAD 3.02 computer code (NUREG/CR 6604) was used to calculate the off-site dose consequences of primary containment leakage, ESF leakage outside the primary containment, and MSIV leakage.

15A.6.3 Dose Assumptions

The breathing rates used in this analysis of this accident are taken from Regulatory Guide 1.183, Revision 0. Specifically, they are:

0 to 8 hours	3.5E-4 m^3/s
8 to 24 hours	1.8E-4 m^3/s
greater than 24 hours	2.3E-4 m^3/s

15A.7 FEEDWATER LINE BREAK ACCIDENT (Section 15.6.6)

15A.7.1 Mathematical Model

15A.7.1.1 Design Basis Analysis

The design basis analysis is presented in Section 15.6.6.

15A.7.2 Transport Assumptions

15A.7.2.1 Design Basis Analysis

All assumptions are described in Section 15.6.6.5.

15A.7.3 Computer Codes Used

The computer code used to model the activity transport to the environment is RADTRAD 3.02 (NUREG/CR 6604).

15A.7.4 Dose Assumptions

The dose assumptions described in Section 15A.4.4 apply to this accident as well.

15A.8 WASTE GAS SYSTEM FAILURE ACCIDENT (SECTION 15.7.1)

All information is presented in Section 15.7.1.

15A.9 LIQUID RADWASTE TANK FAILURE ACCIDENT (SECTION 15.7.3)

All information is presented in Section 15.7.3.

15A.10 FUEL HANDLING ACCIDENT (SECTION 15.7.4)

15A.10.1 Mathematical Model

The model describing the transport of activity is described in Section 15.7.4.9.2.

15A.10.2 Transport Assumptions

The transport assumptions are described in Sections 15.7.4.9.1 and 15.7.4.9.2.

15A.10.3 Computer Codes Used

The RADTRAD computer code (Version 3.02) was used to calculate the off-site radiological consequences of a fuel handling accident.

15A.10.4 Dose Assumptions

The dose assumptions described in Section 15A.4.4 apply to this accident as well.

TABLE 15A-1

CORE INVENTORIES FOLLOWING SHUTDOWN, Ci

Isotope	0 Min	30 Min	1440 Min

Security Related Information
Table withheld Under 10 CFR 2.390

TABLE 15A-2

RADIONUCLIDE CONCENTRATIONS IN REACTOR COOLANT AND MAIN STEAM(1), $\mu C/g$ (FROM GALE CODE)

Isotope	Reactor Coolant Design Basis(2)	Reactor Steam Design Basis(2)
Noble Gases		

Security Related Information
Table withheld Under 10 CFR 2.390

---

(1) The reactor coolant concentration is specified at the nozzle where reactor water leaves the reactor vessel. Similarly, the reactor steam concentration is specified at time 0 at the nozzle.

(2) Design basis concentrations correspond to 350,000, μCi/s @ 30 min.

(3) All iodine concentrations have been adjusted lower to account for the reduced I-131 source term, which was reported in Revision 1 of NUREG-0016.

APPENDIX 15B

SPECIAL ANALYSIS

15B.1 INTRODUCTION

An analysis of the transient caused by continuous control rod withdrawal in the startup range (Section 15.4.1.2) was performed to demonstrate that the licensing-basis criterion for fuel failure will not be exceeded when an out of sequence control rod is withdrawn at the maximum allowable normal drive speed. The sequence and timing assumed in this special analysis is shown in Table 15B-1.

The rod worth minimizer (RWM) constraints on rod sequence will prevent the continuous withdrawal of an out of sequence rod. This analysis was performed to demonstrate that, even for the unlikely event where the RWM fails to block the continuous withdrawal of an out of sequence rod, the licensing basis criterion for fuel failure is still satisfied.

The methods and design basis used for performing the detailed analysis for this event are similar to those previously approved for the control rod drop accident (CRDA) (References 15B-1, 15B-2, and 15B-3). Additional, simplified point model kinetics calculations were performed to evaluate the dependence of peak fuel enthalpy on the control blade worth.

The licensing basis criterion for fuel failure is that the contained energy of a fuel pellet located in the peak power region of the core shall not exceed 170 cal/g-UO_2.

15B.2 METHODS OF ANALYSIS

Since the rod worth calculations using the approved design-basis methods (References 15B-1, 15B-2, and 15B-3) use three dimensional geometry, it is not practical to do a detailed analysis of this event by parameterizing control rod worths. Therefore, the methods of analysis employed were to perform a detailed evaluation of this event for a typical BWR and control rod worth (1.6 percent ΔK) and to use a point model kinetics calculation to evaluate the results over the expected ranges of out of sequence control rod worths. The detailed calculations are performed to demonstrate 1) the consequences of this event over the expected power operating range and 2) the validity of the approximate point-model kinetics calculation. The point model kinetics calculation demonstrates that the licensing criterion for fuel failure is easily satisfied over the range of expected out of sequence control rod worths. These methods are described in more detail below.

The methods used to perform the detailed calculation are identical to those used to perform the design basis CRDA with the following exceptions:

1. The rod withdrawal rate is 3.6 ips (0.3 fps) rather than the blade drop velocity of 3.11 fps. Although faster withdrawal rates are possible, it would require the failure of the associated control rod drive mechanism or hydraulic control unit (as described in Section 4.6.2) in addition to the assumed failure of the RWM. If the associated control rod drive mechanism or hydraulic control unit were assumed to be the worst single failure, then the RWM would terminate the event prior to the full rod withdrawal, or even prior to control rod movement.

2. Scram is initiated either by the intermediate range monitor (IRM) or by a 15 percent power scram initiated by the average power range monitor (APRM) in the startup range. The IRM system is assumed to be in the worst bypass condition allowed by technical specifications.

3. The blade being withdrawn is inserted along with remaining drives at technical specification insertion rates upon initiation of the scram signal.

Examination of a number of rod withdrawal transients in the low power startup range using a two-dimensional R/Z model has shown clearly that a higher fuel enthalpy addition would result from transients starting at the 1 percent power level rather than from lower power levels. The analysis further shows that for continuous rod withdrawal from these initial power levels (1 percent range), the APRM 15 percent power-level scram is likely to be reached as soon as the degraded (worst bypass condition) IRM scram. Consequently, credit is taken for either the IRM or APRM 15 percent power scram in meeting the consequences of this event. The transients for this response were initiated at 1 percent of power and were performed using the APRM 15 percent power scram.

An initial point kinetics calculation was run to determined the time required to scram based on an APRM scram setpoint of 15 percent power and an initial power level of 1 percent. From this time and the maximum allowable rod withdrawal speed, itis possible to show the degree of rod withdrawal before reinsertion due to the scram. From this information, Figure 15B-1, showing the modified effective reactivity shape, was constructed.

The point model kinetics calculations use the same equations employed in the adiabatic approximation described on Page 4-1 of Reference 15B-1. The rod reactivity characteristics and scram reactivity functions are input identically to the adiabatic calculations, and the Doppler reactivity is input as a function of core average fuel enthalpy. The Doppler reactivity feedback function used in the point model kinetics calculations was derived from the detailed analysis of the 1.6 percent rod worth case described above. This is a conservative assumption for higher rod worths since the power peaking and hence spatial Doppler feedback will be larger for higher rod worths. As will be seen in the results section, maximum enthalpies resulted from cases initiated at 1 percent of rated power. In this power range, the APRM will initiate scram at 15 percent of power; hence, the APRM 15 percent power scram was used for these calculations thereby eliminating the

need to perform the spatial analysis required for the IRM scram. All other inputs are consistent with the detailed transient calculation.

The point model kinetics calculations result in core-average enthalpies. The peak enthalpies were calculated using the following equation:

$$h = h_o + (P/A)_T (\bar{h}_f - h_o),$$

where:

h = final peak fuel enthalpy,

h_o = initial fuel enthalpy,

$\bar{h}_f$ = final core average fuel enthalpy, and

$(P/A)_T$ = total peaking factor (radial peaking) x (axial peaking) x (local fuel pin peaking)ç.

For these calculations, the radial and axial peaking factors were obtained as a function of rod worth from the calculations performed in Section 3.6 of Reference 15B-2 and are shown in Figure 15B-2. It was conservatively assumed that no power flattening due to Doppler feedback occurred during the course of the transient.

15B.3 RESULTS

The reactivity insertion resulting from moving the control rod is shown in Figure 15B-1 for the point model kinetics calculations. The core-average power versus time and the global peaking factors from Section 3.6 of Reference 15B-2 are shown in Figures 15B-3 and 15B-2, respectively. The results of the point model kinetics calculation are summarized in Table 15B-2 along with the results of the detailed analysis.

From Figure 15B-3 and Table 15B-2, it is shown that the core average energy deposition is insensitive to control rod worth; therefore, the only change in peak enthalpy as a function of rod worth will result from differences in the global peaking, which increases with rod worth. Comparison of the global peaking factors shown in Figure 15B-2 with the values used in the detailed calculation demonstrates that the Reference 15B-2 values are reasonable for their application in this study. For all cases, the peak fuel enthalpy is well below the licensing basis criterion of 170 cal/g.

Cases 4 and 5 of Table 15B-2 show that the point model kinetics calculations give conservative results relative to the detailed evaluations. The primary difference is that the global peaking will flatten during the transient due to Doppler feedback. This is accounted for in the detailed calculation, but the point model kinetics calculations conservatively assumed that the peaking remains constant at its initial value.

The differences in core-average and peak enthalpy between cases 1 and 5 are due to the fact that for case 1 the scram was initiated by the APRM 15 percent power scram setpoint; whereas, in case 5 the scram was initiated by the IRMs. As can be seen by Figure 15B-4, this would occur at a core average power of 21 percent. Since the APRM trip point will be reached first, it is reasonable to take credit for the APRM scram.

15B.4 This Section Deleted

15B.5 CONCLUSIONS

The above evaluations of continuous withdrawal of a control rod in the startup range indicate that the peak fuel enthalpies due to the continuous withdrawal of an out of sequence rod in the startup range will be much less than the licensing basis criterion of 170 cal/gm. In light of the conservative nature of these evaluations and the markedly different fuel designs and vendor methodologies, the substantial margins to 170 cal/gm limit support a generic conclusion that the peak fuel enthalpy associated with continuous withdrawal of a control rod in the startup range in the HCGS core will remain below 170 cal/gm.

15B.5 REFERENCES

15B-1 C. J. Paone, et al, "Rod Drop Accident Analysis For Large Boiling Water Reactors", NEDO-10527, March 1972.

15B-2 R. C. Stirn, et al, "Rod Drop Accident Analysis For Large Boiling Water Reactors", NEDO-10527, Supplement 1, July 1972.

15B-3 R. C. Stirn, "Rod Drop Accident Analysis For Large Boiling Water Reactors, Addendum No. 2, Exposed Cores", NEDO-10527, Supplement 2, January 1973.

15B-4 R. C. Stirn, J. F. Klapproth, "Continuous Rod Withdrawal Transient in the Startup Range", NEDO-23842, April 1978.

15B-5 Deleted.

TABLE 15B-1

SEQUENCE OF EVENTS FOR CONTINUOUS ROD WITHDRAWAL
DURING REACTOR STARTUP

Time (s)		Event
0	1.	The reactor is critical and operating in the startup range.
>0	2.	The operator selects and withdraws an out-of-sequence control rod at the maximum normal drive speed of 3.6 ips.
4	3.	Either the RWM or the second qualified verifier fail to block the selection (selection error) and continuous withdrawal (withdraw error) of the out-of-sequence rod.
4-8	4.	The reactor scram is initiated by the IRM system or the APRM system.
5-9	5.	The prompt power burst is terminated by a combination of Doppler and/or scram feedback.
10	6.	The transient is finally terminated by the scram of all rods, including the control rod being withdrawn. Scram insertion times are assumed to be 5 seconds to 90 percent insertion.

TABLE 15B-2

SUMMARY OF RESULTS FOR DETAILED AND POINT MODEL KINETICS CALCULATIONS OF CONTINUOUS ROD WITHDRAWAL IN THE STARTUP RANGE

Case	Control Rod Worth ($ΔK)	$\bar{h}_f$ (cal/g)	$(P/A)_G$ (2)	h (cal/g)
1	1.6	17.3	24.2	42.7
2	2.0	17.3	30.9	50.0
3	2.5	17.2	46.0	58.5
4	1.6 (1)	18.3	19.7 (3)	56.2
5	1.6 (4)	18.3	19.7	59.6

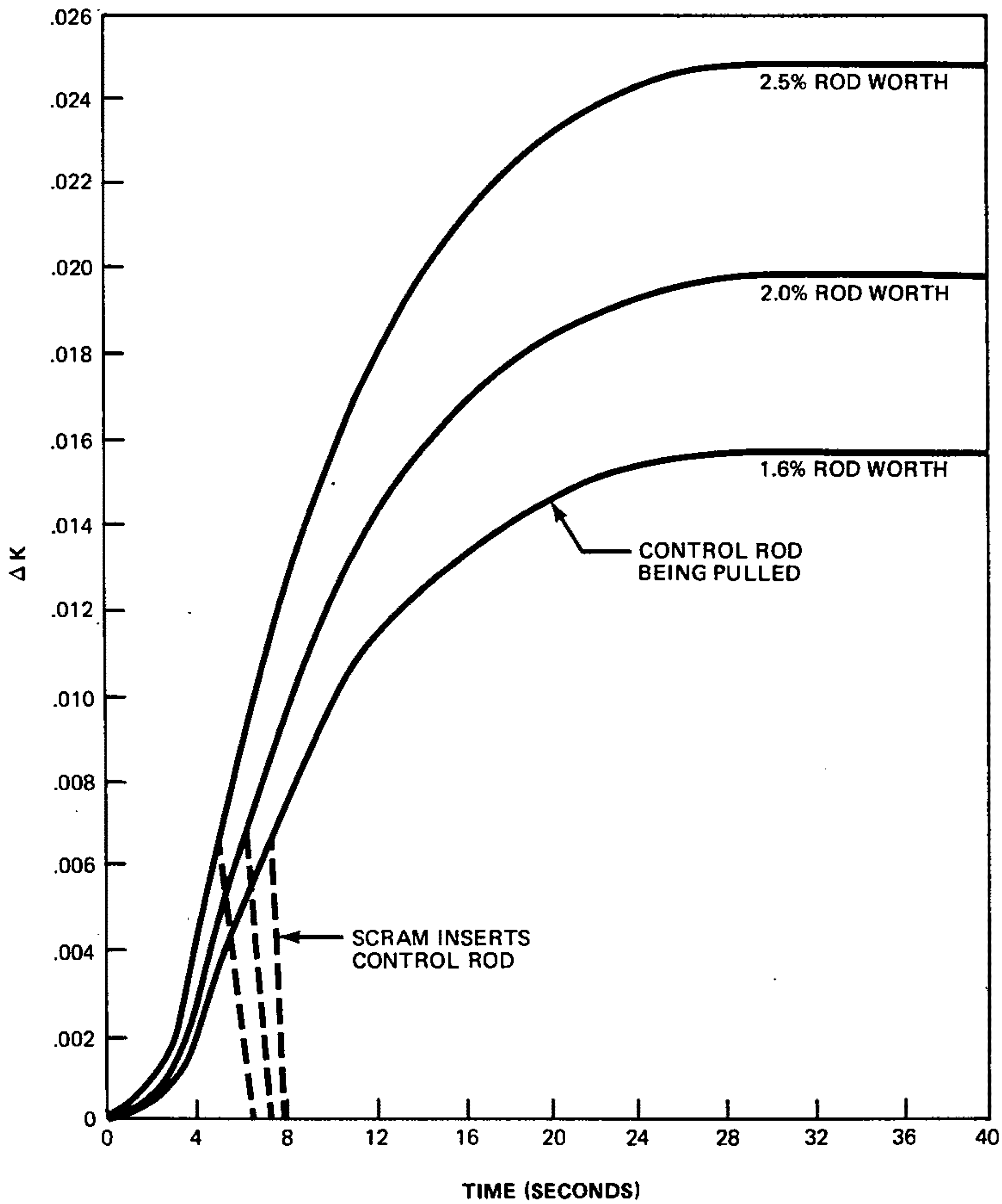
---

(1) Detailed transient calculation. All other data reported are for point model kinetics calculations.

(2) $(P/A)_G$ = global peaking factor (radial x axial).

(3) The $(P/A)_G$ = 19.7 is the initial value. For the detailed analysis, this value will decrease during the course of the transient since the power shape will flatten due to Doppler feedback.

(4) Point model kinetics calculation with an IRM-initiated scram and 3-D simulator global peaking.



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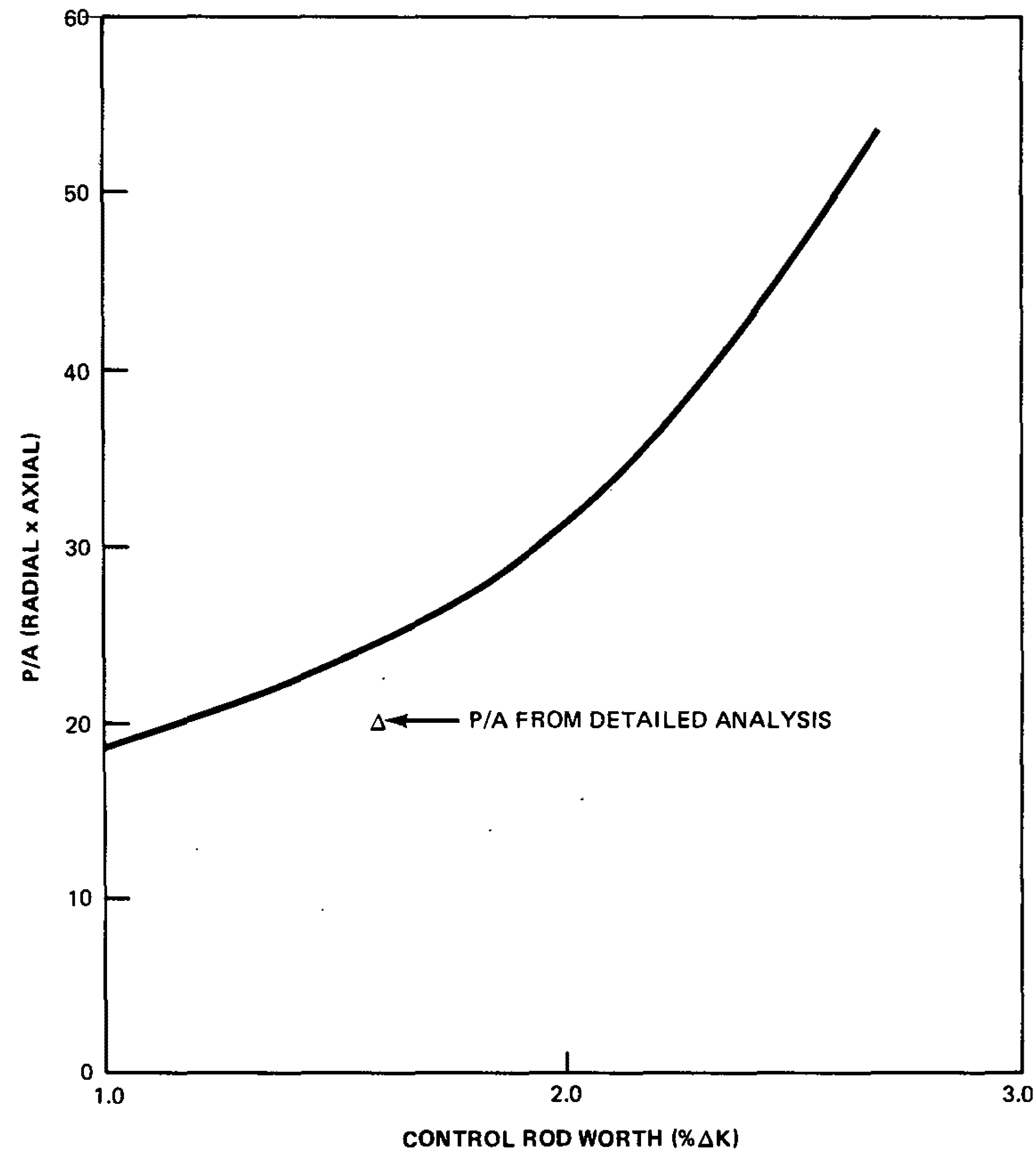
POINT KINETICS CONTROL ROD
REACTIVITY INSERTION (4)

UPDATED FSAR

FIGURE 15B-1



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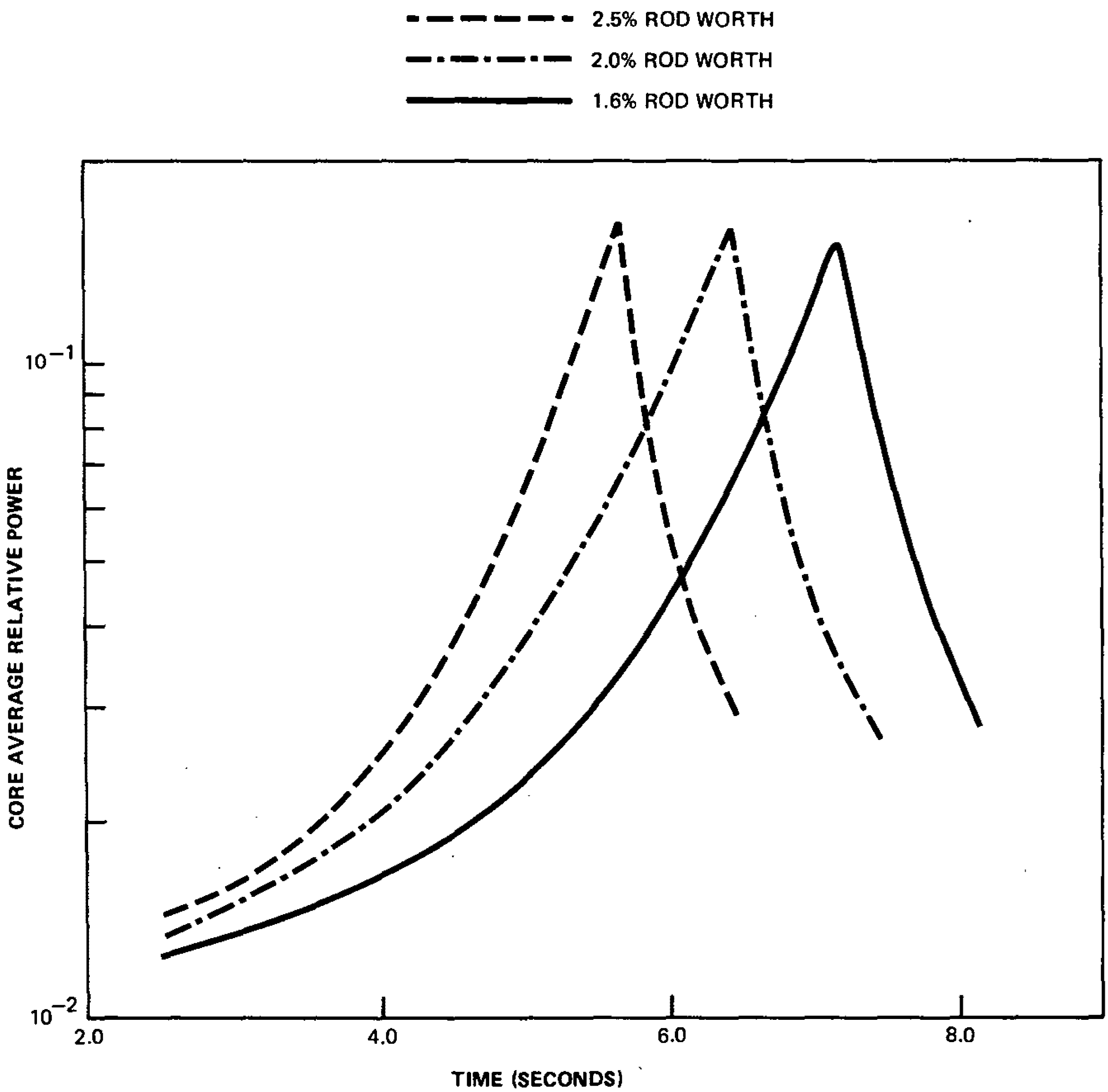
P/A vs ROD WORTH NEDO-10527 SUPPLEMENT 1 (2) AND DETAILED ANALYSIS (4)

UPDATED FSAR

FIGURE 15B-2



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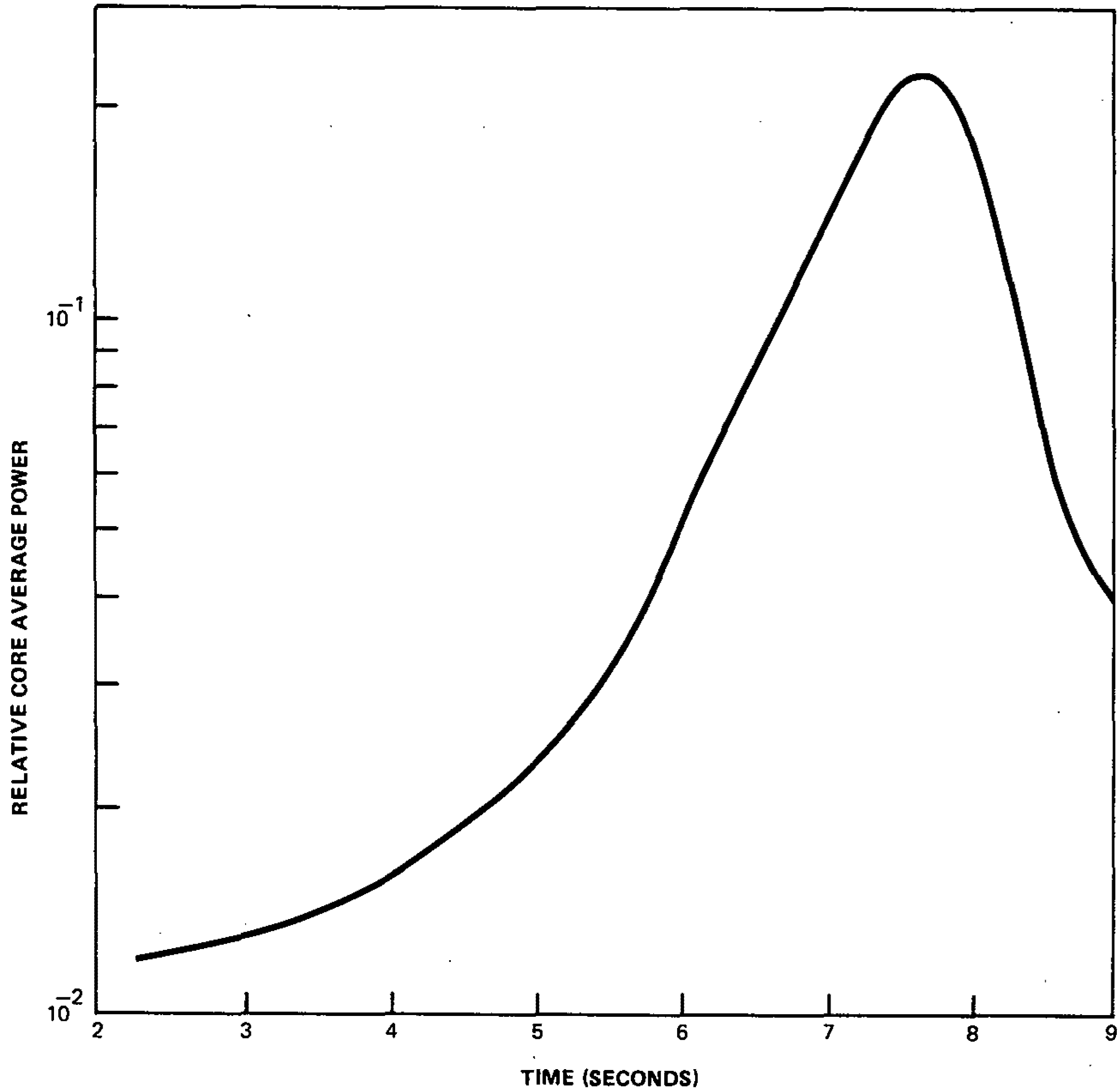
CONTINUOUS RWE IN THE STARTUP RANGE CORE AVERAGE POWER vs TIME FOR 1.6%, 2.0% AND 2.5% WORTH'S (POINT MODEL KINETICS) (4)

UPDATED FSAR

FIGURE 15B-3



ASSUMPTIONS:

1. 1.6% Δk ROD
2. 0.3 fps WITHDRAWAL VELOCITY
3. IRM SCRAM FOR WORST BYPASS CONDITION
4. $P_o = 10^{-2}$ OF RATED
5. 1967 PRODUCT LINE TECH SPEC SCRAM RATE
6. EXPOSURE = 0.0 GWD/T

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CONTINUOUS CONTROL ROD WITHDRAWAL FROM HOT STARTUP (4)

UPDATED FSAR

FIGURE 15B-4

APPENDIX 15C

HOPE CREEK SINGLE LOOP OPERATION ANALYSIS

FEBRUARY 1986

Prepared for

PUBLIC SERVICE ELECTRIC AND GAS COMPANY

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APPENDIX 15.C

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15.C RECIRCULATION SYSTEMS SINGLE-LOOP OPERATION

The information presented in Appendix 15C is historical in nature. The single-loop operation (SLO) required operating limits are confirmed or determined on a reload basis in accordance with the requirements in Reference 15.C.8-6. In addition, SLO has been determined to be acceptable for CPPU operating conditions as described in Reference 15.C.8-12.

15.C.1 INTRODUCTION AND SUMMARY

Single-loop operation (SLO) at reduced power is highly desirable in the event recirculation pump or other component maintenance renders one loop inoperative. To justify single-loop operation, accidents and abnormal operational transients associated with power operations, as presented in Sections 6.2 and 6.3 and the main text of Chapter 15.0, were reviewed for the single-loop case with only one pump in operation. This appendix presents the results of the safety evaluation for the operation of the Hope Creek Generating Station (HCGS) with single recirculation loop inoperable. This safety evaluation was performed for GE and ABB fuel in Hope Creek. The analysis shows that the transient consequences for SLO (ΔCPR) are bounded by the full power analysis results given in the FSAR. The conclusion drawn from the transient analysis results presented in this report is applicable to reload cycle operation.

Increased uncertainties in the core total flow and Traversing In-Core Probe (TIP) readings result in an incremental increase in the Minimum Critical Power Ratio (MCPR) fuel-cladding integrity safety limit during single-loop operation. No increase in rated MCPR operating limit and no change in the power or flow dependent MCPR limit is required because all abnormal operational transients analyzed for single-loop operation indicated that there is more than enough MCPR margin to compensate for this increase in MCPR safety limit. The recirculation flow rate dependent rod block and scram setpoint equation given in Chapter 16 (Technical Specifications) are adjusted for one-pump operation.

Thermal-hydraulic stability was evaluated for its adequacy with respect to General Design Criteria 12 (10CFR50, Appendix A). It is shown that this stability criterion is satisfied during SLO. It is further shown that the increase in neutron noise observed during SLO is independent of system stability margin.

To prevent potential control oscillations from occurring in the recirculation flow control system, the operation mode of the recirculation flow control system must be restricted to operation in the manual control mode for single-loop operation.

The Maximum Average Planar Linear Heat Generation Rate (MAPLHGR) for single-loop operation is reduced to accommodate the impact of SLO on the LOCA analysis.

The impact of single-loop operation on the FSAR specifications for containment response including the containment dynamic loads was evaluated. It was confirmed that the containment response under SLO is within the present design values.

The impact of single-loop operation on the Anticipated Transient Without Scram (ATWS) analysis was evaluated. It is found that all ATWS acceptance criteria are met during SLO.

The fuel thermal and mechanical duty for transient events occurring during SLO is found to be bounded by the fuel design bases. The Average Power Range Monitor (APRM) fluctuation should not exceed a flux amplitude of ±15% of rated and the core plate-differential pressure fluctuation should not exceed 3.2 psi peak to peak to be consistent with the fuel rod and assembly design bases.

A recirculation pump drive flow limit is imposed for SLO. The highest drive flow that meets acceptable vessel internal vibration criteria is the drive flow limit for SLO. The pump speed at Hope Creek Generating Station should be limited to 90% of rated under single-loop operating conditions.

15.C.2 MCPR FUEL CLADDING INTEGRITY SAFETY LIMIT

Except for core total flow and TIP reading, the uncertainties used in the statistical analysis to determine the MCPR fuel cladding integrity safety limit are not dependent on whether coolant flow is provided by one or two recirculation pumps. A 6% core flow measurement uncertainty has been established for single-loop operation (compared to 2.5% for two-loop operation). At shown below, this value conservatively reflects the one standard deviation (one sigma) accuracy of the core flow measurement system documented in Reference 15.C.8-1.

The random noise component of the TIP reading uncertainty was revised for single recirculation loop operation to reflect the operating plant test results given in Subsection 15.C.2.2. This revision resulted in a single-loop operation process computer effective TIP uncertainty of 6.8% of initial cores and 9.1% for reload cores. Comparable two-loop process computer uncertainty values are 6.3% for initial cores and 8.7% for reload cores. This represents a 4.6% increase in process computer determination relative assembly power.

The net effect of these two revised uncertainties is an incremental increase in the required MCPR fuel cladding integrity safety limit.

15.C.2.1 Core Flow Uncertainty

15.C.2.1.1 Core Flow Measurement During Single-Loop Operation

The jet pump core flow measurement system is calibrated to measure core flow when both sets of jet pumps are in forward flow; total core flow is the sum of the indicated loop flows. For single-loop operation, however, some inactive jet pumps will be backflowing (at active pump speeds above approximately 40%). Therefore, the measured flow in the backflowing jet pumps must be subtracted from the measured flow in the active loop to obtain the total core flow. In addition, the jet pump coefficient is different for reverse flow than for forward flow, and the measurement of reverse flow must be modified to account for this difference.

In single-loop operation, the total core flow is derived by the following formula:

$$\begin{pmatrix} TotalCore \\ Flow \end{pmatrix} = \begin{pmatrix} ActiveLoop \\ IndicatedFlow \end{pmatrix} - C \begin{pmatrix} InactiveLoop \\ IndicatedFlow \end{pmatrix}$$

The coefficient C (=0.95) is defined as the ratio of "Inactive Loop True Flow" to "Inactive Loop Indicated Flow". "Loop Indicated Flow" is the flow measured by the jet pump "single-tap" loop flow summers and indicators, which are set to read forward flow correctly.

The 0.95 factor was the result of a conservative analysis to appropriately modify the single-tap flow coefficient for reverse flow.* If a more exact, less conservative, core flow is required, special in-reactor calibration tests can be made. Such calibration tests would involve: calibrating core support plate ΔP versus core flow during one-pump and two-pump operation along with 100% flow control line and calculating the correct value of C based on the core support plate ΔP and the loop flow indicator readings.

15.C.2.1.2 <u>Core Flow Uncertainty Analysis</u>

The uncertainty analysis procedure used to establish the core flow uncertainty for one-pump operation is essentially the same as for two-pump operation with some exceptions. The core flow uncertainty analysis is described in Reference 15.C.8-1. The analysis of one-pump core flow uncertainty is summarized below.

---

*The analytical expected value of the "C" coefficient for HCGS is 0.84.

For single-loop operation, the total core flow can be expressed as follows (refer to Figure 15.C.2-1):

$$W_C = W_A - W_I$$

where:

W_C = total core flow,

W_A = active loop flow, and

W_I = inactive loop (true) flow.

By applying the "propagation of errors" method to the above equation, the variance of the total flow uncertainty can be approximated by:

$$\sigma_{W_C}^2 = \sigma_{W_{sys}}^2 + \left(\frac{1}{1-a}\right)^2 \bullet \sigma_{W_{A_{rand}}}^2 + \left(\frac{a}{1-a}\right)^2 \bullet \left(\sigma_{W_{I_{rand}}}^2 + \sigma_C^2\right)$$

where:

σ_{W_C} = uncertainty of total core flow;

$\sigma_{W_{sys}}$ = uncertainty systematic to both loops;

$\sigma_{W_{A_{rand}}}$ = random uncertainty of active loop only;

$\sigma_{W_{I_{rand}}}$ = random uncertainty of inactive loop only;

σ_C = uncertainty of "C" coefficient; and

a = ratio of inactive loop flow (W_I) to active loop flow (W_A).

From an uncertainty analysis, the conservative, bounding values of

$\sigma_{W_{sys}}$, $\sigma_{W_{A_{rand}}}$, $\sigma_{W_{I_{rand}}}$ and σ_C are 1.6%, 2.6%, 3.5%, and 2.8%, respectively.

Based on the above uncertainties and a bounding value of 0.36* for "a", the variance of the total flow uncertainty is approximately:

*This flow split ratio varies from about 0.13 to 0.36. The 0.36 value is a conservative bounding value. The analytical expected value of the flow split ratio for HCGS is ~ 0.33.

$$\sigma^2_{W_C} = (1.6)^2 + \left(\frac{1}{1 - 0.36}\right)^2 \bullet (2.6)^2 + \left(\frac{0.36}{1 - 0.36}\right)^2 \bullet \left((3.5)^2 + (2.8)^2\right)$$

$$= (5.0\%)^2$$

When the effect of 4.1% core bypass flow split uncertainty at 12% (bounding case) bypass flow fraction is added to the total core flow uncertainty, the active-coolant flow uncertainty is:

$$\sigma^2_{\substack{\text{active}\\\text{coolant}}} = (5.0\%)^2 + \left(\frac{0.12}{1 - 0.12}\right)^2 \bullet (4.1\%)^2 = (5.1\%)^2$$

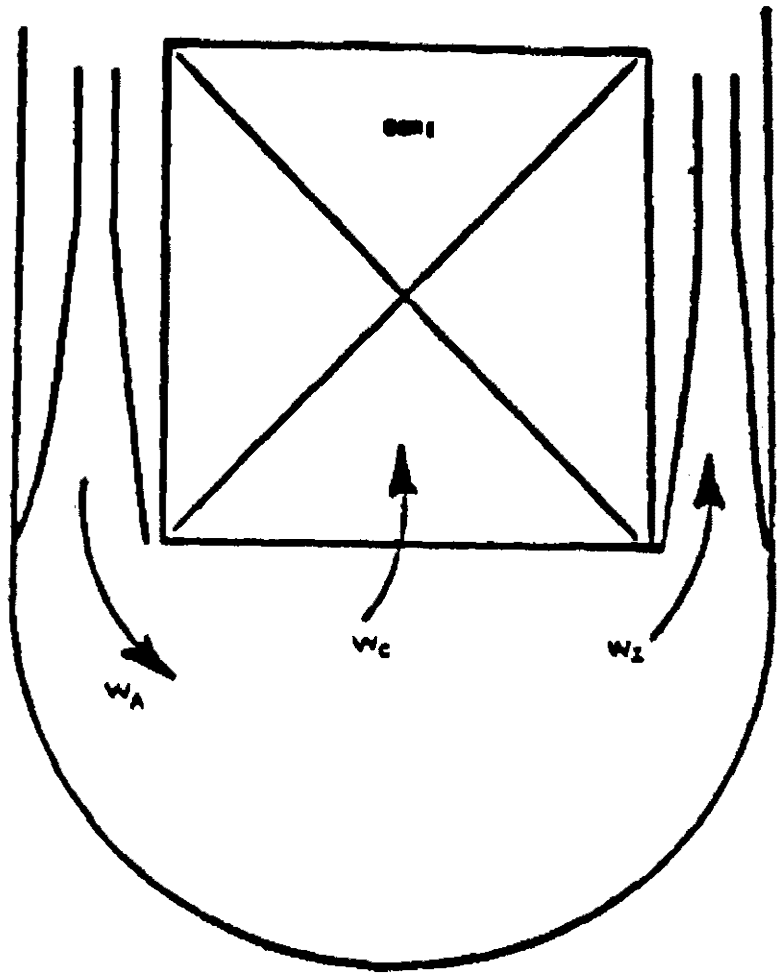
which is less than the 6% flow uncertainty assumed in the statistical analysis.

In summary, core flow during one-pump operation is measured in a conservative way and its uncertainty has been conservatively evaluated.

15.C.2.2 TIP Reading Uncertainty

To ascertain the TIP noise uncertainty for single recirculation loop operation, a test was performed at an operating BWR. The test was performed at a power level 59.3% of rated with a single recirculation pump in operation (core flow 46.3% of rated). A rotationally symmetric control rod pattern existed during the test.

Five consecutive traverses were made with each of five TIP machines, giving a total of 25 traverses. Analysis of this data resulted in a nodal TIP noise of 2.85%. Use of this TIP noise value as a component of the process computer total uncertainty results in a one-sigma process computer total effect TIP uncertainty value for single-loop operation of 6.8% for initial cores and 9.1% for reload cores.



W_C = Total Core Flow
W_A = Active Loop Flow
W_I = Inactive Loop Flow

PSE&G	ILLUSTRATION OF SINGLE RECIRCULATION LOOP OPERATION FLOWS	FIGURE 15.C.2-1

15.C.3 MCPR OPERATING LIMIT

15.C.3.1 Abnormal Operational Transients

Operating with one recirculation loop results in a maximum power output which is about 30% below that which is attainable for two-pump operation. Therefore, the consequences of abnormal operational transients from one-loop operation will be considerably less severe than those analyzed for two-loop operation. For pressurization, flow increase, flow decrease, and cold water injection transients, the results presented in Chapter 15 bound both the thermal and overpressure consequences of one-loop operation.

The consequences of flow decrease transients are bounded by the full power analysis. A single pump trip from one-loop operation is less severe than a two-pump trip from full power because of the reduced initial power level.

The worst flow increase transient results from a recirculation flow controller failure, and the worst cold water injection transient results from the loss of feedwater heating. For the former event, the impact on CPR is derived assuming both recirculation loop controllers fail. This condition produces the maximum possible power increase and hence maximum ΔCPR for transients initiated from less than rated power and flow. During operation with only one recirculation loop, the flow and power increase associated with this failure with only one loop will be less than that associated with both loops; therefore, the impact on CPR of the worst flow increase event derived with the two-pump assumption is conservative for single-loop operation.

The latter event, loss of feedwater heating, is generally the most severe cold water event with respect to increase in core power. This power increase is caused by positive reactivity insertion from increased core inlet subcooling and it is relatively insensitive to initial power level. A generic statistical loss of feedwater heater analysis using different

initial power levels and other core design parameters concluded one-pump operation with lower initial power level is conservatively bounded by the full power two-pump analysis. The conclusions regarding the consequences of the inadvertent restart of the idle recirculation pump in Chapter 15.4.4 are still applicable for single-loop operation.

Assessments of the relative impact on the limiting pressurization transients for single-loop and two-loop conditions show that the consequences for single-loop conditions are bounded by the two-loop results. The following sections provide examples of these assessments and confirm the generic nature of the conclusions.

15.C.3.1.1 Feedwater Controller Failure - Maximum Demand (Cycle 1)

This event is postulated on the basis of a single failure of a master feedwater control device, specifically one which can directly cause an increase in coolant inventory by increasing the total feedwater flow. The most severe applicable event is a feedwater controller failure during maximum flow demand. The feedwater controller is assumed to fail to its upper limit at the beginning of the event.

A feedwater controller failure during maximum flow demand at 75% power and 60% flow during single recirculation loop operation produces the sequence of events listed in Table 15.C.3-2. Figure 15.C.3-1 shows the changes in important variables during this transient. References to percent power, percent of rated, etc., contained in the text, figures, and tables describing this event are relative to the Cycle 1 licensed power level of 3293 MW_{th}.

The computer model described in Reference 15.C.8-2 was used to simulate this event.

The analysis has been performed with the plant conditions tabulated in Table 15.C.3-1. with the initial vessel water level at Level 4 (instead of normal water level) for conservatism. By lowering the initial water level, more cold feedwater will be injected before Level 8 is reached resulting in higher heat fluxes.

The safety analysis condition is at 75% rated thermal power and 60% rated core flow, which represents single recirculation loop operation at 100% pump speed on the 105% rod line. End of cycle (all rod out) scram characteristics are assumed. The safety-relief valve action is conservatively assumed to occur with higher than nominal setpoints. The transient is simulated by programming an upper limit failure in the feedwater system such that 159% of rated feedwater flow occurs at the reactor dome pressure of 973 psig, and 135% of rated feedwater flow would occur at the design pressure of 1060 psig.

The simulated feedwater controller transient is shown in Figure 15.C.3-1. The high water level turbine trip and feedwater pump trip are initiated at approximately 6.1 seconds. Scram occurs simultaneously from stop valve closure, and limits the neutron flux peak and fuel thermal transient. The turbine bypass system opens to limit peak pressures in the steam supply system. Events caused by low water level trips, including initiation of HPCS and RCIC core cooling system functions are not included in the simulation. Should these events occur, they will follow sometime after the primary effects have occurred, and are expected to be less severe than those already experienced by the system.

Table 15.C.3-4 gives a summary of the transient analysis results. The calculated MCPR is 1.17, which is well above the safety limit MCPR of 1.07 so no fuel failure due to boiling transition is predicted. The peak vessel pressure predicted is 1121 psig and is will below the ASME limit of 1375 psig.

15.C.3.1.2 <u>Generator Load Rejection with Bypass Failure (Cycle 1)</u>

Fast closure of the turbine control valves (TCV) is initiated wherever electrical grid disturbances occur which result in significant loss of electrical load on the generator. The turbine control valves are required to close as rapidly as possible to prevent excessive overspeed of the turbine-generator (T-G) rotor. Closure of the main turbine control valves will increase system pressure.

A loss of generator electrical load with bypass failure at 75% power and 60% flow during single recirculation loop operation produces the sequence of events listed in Table 15.C.3-3. Figure 15.C.3-2 shows the changes in important variables during this transient. References to percent power, percent of rated, etc., contained in the text, figures, and tables describing this event are relative to the Cycle 1 licensed power level of 3293 MW_{th}.

Generator load rejection causes turbine control valve (TCV) fast closure which initiates a scram trip signal for power levels greater than 40% NB rated. In addition, recirculation pump trip is initiated. Both of these trip signals satisfy single failure criterion and credit is taken for these protection features.

The pressure relief system which operates the relief valves independently when system pressure exceeds relief valve instrumentation setpoints is assumed to function normally during the time period analyzed.

All plant control systems maintain normal operation unless specifically designated to the contrary.

The computer model described in Reference 15.C.8-2 was used to simulate this event.

The analysis has been performed with the plant conditions tabulated in Table 15.C.3-1, except that the turbine bypass function is assumed to fail.

The safety analysis condition is at 75% rated thermal power and 60% rated core flow, which represents single recirculation loop operation at 100% pump speed on the 105% rod line.

The turbine electro-hydraulic control system (EHC) power/load unbalance device detects load rejection before a measurable speed change takes place.

The closure characteristics of the turbine control valves are assumed such that the valves operate in the full arc (FA) mode and have a full stroke closure time, from fully open to fully closed, of 0.15 second.

Auxiliary Power would normally be independent of any turbine-generator over-speed effects and continuously be supplied at rated frequency as automatic fast transfer to auxiliary power supplies occurs.

The simulated generator load rejection with bypass failure is shown in Figure 15.C.3-2.

Events caused by low water level trips, including initiation of HPCI and RCIC core cooling system functions are not included in this simulation. If these events occur, they will follow sometime after the primary concerns of fuel margin and overpressure effects have passed, and will result in effects less severe than those already experienced by the reactor system, and will provide long-term reactor inventory control.

Table 15.C.3-4 summarizes the transient analysis results. The peak neutron flux reaches about 120% of rated and average surface heat flux peaks at about 104% of its initial value. The peak vessel pressure predicted is 1162 psig and is well below the ASME limit of 1375 psig. The calculated MCPR is 1.16 which is considerably above the cycle 1 safety limit MCPR of 1.07.

15C.3.1.3 Evaluation for ABB Fuel

The impact of pressurization transients for single-loop operation (SLO) conditions relative to two-loop conditions has also been evaluated for the limiting pressurization events in a mixed 8X8-4 and SVEA-96+ core. These calculations were performed with the ABB licensing analysis methodology in Reference 15.C.8-10. The calculations show that MCPR operating limits established by the limiting two loop transients are conservatively applicable to transients initiated from SLO conditions. This conclusion accommodates the fact that the SVEA-96+ and 8x8-4 SLMCPR for SLO is increased by an increment appropriate to accommodate the increased SLO uncertainties discussed in Section 15.C.2. These results provide further confirmation that MCPR operating limits establishes by the limiting pressurization events based on the two loop evaluations will conservatively protect the fuel during postulated limiting pressurization transients initiated from SLO conditions.

Appendix 15D provides more information on the SLO analysis that is performed during the reload.

15.C.3.1.4 Summary and Conclusions

The discussion in section 15C.3.1.1 through 15.C.3.1.2 illustrates the conclusion that the operating limit MCPRs is established by pressurization transients for two-pump operation are also applicable to single-loop operation conditions.

For pressurization, Table 15.C.3-4 indicates that the peak pressures are well below the ASME code value of 1375 psig. Hence, it is concluded that the pressure barrier integrity is maintained under single-loop operation.

15.C.3.2 Rod Withdrawal Error

The rod withdrawal error at rated power is given in the FSAR. These analyses are performed to demonstrate, even if the operator ignores all instrument indications and the alarm which could occur during the course of the transient, the rod block system will stop rod withdrawal at a minimum critical power ratio (MCPR) which is higher than the fuel cladding integrity safety limit. For ARTS/MELLLA analyses, the RWE is conservatively performed without a rod block and ensures the MCPR is higher than the fuel cladding integrity safety limit. Modification of the rod block equation (below) and lower power assures the MCPR safety limit is not violated.

One-pump operation results in backflow through 10 of the 20 jet pumps while the flow is being supplied into the lower plenum from the 10 active jet pumps. Because of the backflow through the inactive jet pumps, the present rod block equation was conservatively modified for use during one-pump operation because the direct active-loop flow measurement may not indicate actual flow above about 40% core flow without correction.

A procedure has been established for correcting the rod block equation to account for the discrepancy between actual flow and indicated flow in the active loop. This preserves the original relationship between rod block and actual effective drive flow when operating with a single-loop.

The two-pump rod block equation is:

$$RB = mW + RB_{100} - m(100)$$

The one-pump equation becomes:

$$RB = mW + RB_{100} - m(100) - m\Delta W$$

where

ΔW	=	difference between two-loop and single-loop effective drive flow at the same core flow. This value is expected to be 8% of rated (to be determined by PSE&G).
RB	=	power at rod block in %;
m	=	flow reference slope for the rod block monitor (RBM)
W	=	drive flow in % of rated.
RB_{100}	=	top level rod block at 100% flow.

If the rod block setpoint (RB_{100}) is changed, the equation must be recalculated using the new value.

The APRM trip settings are flow biased in the same manner as the rod block monitor trip setting. Therefore, the APRM rod block and scram trip settings are subject to the same procedural changes as the rod block monitor trip settings discussed above.

15.C.3.3 Operating MCPR Limit

For single-loop operation, the operating, MCPR limit remains unchanged from the normal two-loop operation limit. Although the increased uncertainties in core flow and TIP readings resulted in an incremental increase in MCPR fuel cladding integrity safety limit during single-loop operation (Section 15.C.2), the results in Section 15.C.3 indicate that there is more than enough MCPR margin during single-loop operation to compensate for this increase in safety limit. For single-loop operation at lower flows, the steady-state operating MCPR limit is established by reduced flow operating MCPRs. This ensures the 99.9% statistical limit requirement is always satisfied for any postulated abnormal operational occurrence.

Since the maximum core flow runout during single loop operation is only about 60% of rated, the current reduced flow MCPRs which are generated based on the flow runout up to rated core flow are also adequate to protect the flow runout events during single-loop operation.

TABLE 15.C.3-1

INPUT PARMETERS AND INITIAL CONDITIONS

1.	Thermal Power Level, MWt	2470
2.	Steam Flow, lb per hr	10.17×10^6
3.	Core Flow, lb per hr	60.00×10^6
4.	Feedwater Flow Rate, lb per sec	2824
5.	Feedwater Temperature, °F	390
6.	Vessel Dome Pressure, psig	973
7.	Vessel Core Pressure, psig	978
8.	Turbine Bypass Capacity, % NBR	25
9.	Core Coolant Inlet Enthalpy, Btu per lb	512.1
10.	Turbine Inlet Pressure, psig	944
11.	Fuel Lattice	C(P8x8R)
12.	Core Average Gap Conductance, Btu/sec-ft^2 - °F	0.1744
13.	Core Bypass Flow, %	11.27
14.	Required Initial MCPR	1.28**
15.	MCPR Safety Limit	1.07
16.	Doppler Coefficient, ¢/°F	*
17.	Void Coefficient, ¢/% Rated Voids	*
18.	Core Average Rated Fraction, %	45.1
19.	Scram Reactivity, $ΔK	*
20.	Control Rod Drive Speed Position versus Time	Figure 15.0-1

* This value is calculated within the computer code (Reference 15.C.8-2) for end of Cycle 1 conditions based on input from the CRUNCH file.

** K_f times the Rated Operating Limit MCPR

TABLE 15.C.3-1 (Cont.)

INPUT PARAMETERS AND INITIAL CONDITIONS

21.	Fuel Exposure	End of Cycle 1
22.	Jet Pump Ratio, M	3.56
23.	Safety/Relief Valve Capacity, % NBR	
	@ 1121 psig	85.8
	Manufacturer	Target Rock
	Quantity Installed	14
24.	Relief Function Delay, seconds	0.4
25.	Relief Function Response Time Constant, seconds	0.15
26.	Setpoints for Safety/Relief Valves	
	Safety/Relief Function, psig	1121, 1131, 1141
27.	Number of Valve Groupings Simulated	
	Safety/Relief Function	3
28.	Safety/Relief Valve Reclosure Setpoints, psig	1087, 1097, 1107
29.	High Flux Trip Setpoint, % NBR (121 x 1.043)	126.2
30.	High Pressure Scram Setpoint, psig	1071
31.	Vessel Level Trips, Feet Above Separator Skirt Bottom	
	Level 8 - (L8), feet	6.042
	Level 4 - (L4), feet	3.625
	Level 3 - (L3), feet	1.792
	Level 2 - (L2), feet	-3.708
32.	APRM Simulated Thermal Power Trip Setpoint, % NBR (117 x 1.043)	122.0
33.	Recirculation Pump Trip Delay, seconds	0.175
34.	Inertia Time-Constant of Recirculation Pump Trip (maximum), seconds	4.5
35.	Total Steamline Volume, ft^3	6619
36.	Pressure Setpoint of ATWS Recirculation Pump Trip, psig	1101

TABLE 15.C.3-2

SEQUENCE OF EVENTS FOR FIGURE 15.C.3-1
FEEDWATER CONTROLLER FAILURE, MAXIMUM DEMAND

Time-sec	Event
0	Initiate simulated failure to the upper limit on feedwater flow.
6.1	L8 vessel level setpoint trips main turbine and feedwater pumps. Turbine bypass operation initiated.
6.1	Reactor scram trip actuated from main turbine stop valve position switches.
6.1	Recirculation pump trip (RPT) actuated by stop valve position switches.
6.2	Main turbine stop valves closed and turbine bypass valves start to open.
6.3	Recirculation pump motor circuit breaker opens causing decrease in core flow.

TABLE 15.C.3-3

SEQUENCE OF EVENTS FOR FIGURE 15.C.3-2
GENERATOR LOAD REJECTION WITH BYPASS FAILURE

Time-sec	Event
(-)0.015 (approx.)	Turbine-generator detects loss of electrical load.
0	Turbine-generator load rejection sensing devices trip to initiate turbine control valve fast closure.
0	Turbine bypass valves fail to operate.
0	Fast control valve closure (FCV) initiates scram trip and recirculation pump trip (RPT).
0.07	Turbine control valves closed.
0.175	Recirculation pump motor circuit breaker opens causing decrease in core flow.
2.2	Group 1 relief valves actuated.
2.6	Group 2 relief valves actuated.
2.8	Group 3 relief valves actuated.
4.4	Group 3 relief valves start to close.
9.5	All relief valves are closed.

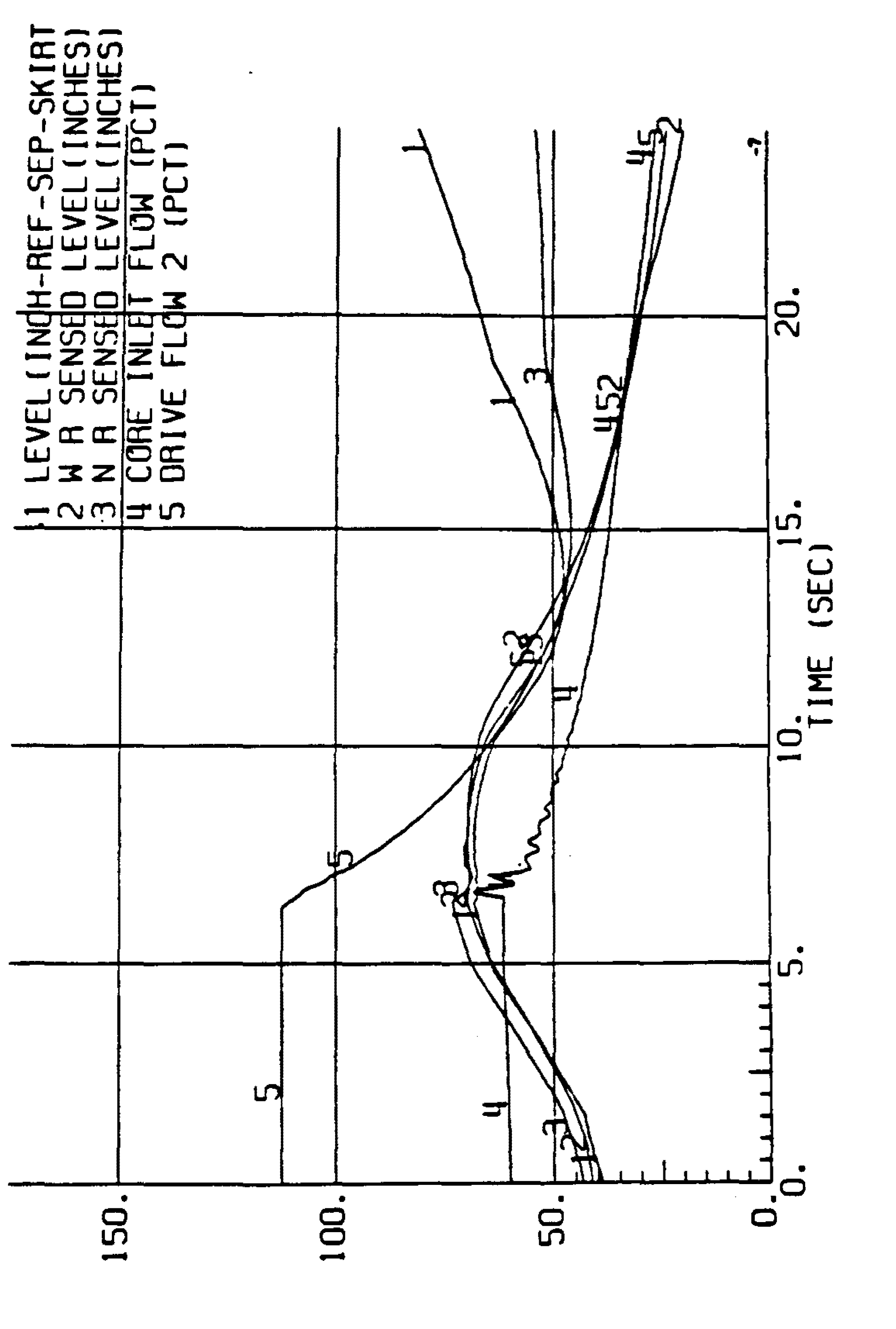
TABLE 15.C.3-4

SUMMARY OF TRANSIENT PEAK VALUE AND CPR RESULTS

	FWCF	LRBPF
Initial Power/Flow (% Rated)	75/60	75/60
Peak Neutron Flux (% Rated)	91.2	119.7
Peak Heat Flux (% Initial)	103.3	103.9
Peak Dome Pressure (psig)	1107	1148
Peak Vessel Bottom Pressure (psig)	1121	1162
Required Initial MCPR*	1.28	1.28
Transient MCPR**	1.17	1.16
Safety Limit MCPR (for SLO)	1.07	1.07
Margin to Safety Limit	0.10	0.09

*K_f times the Rated Operating Limit MCPR.

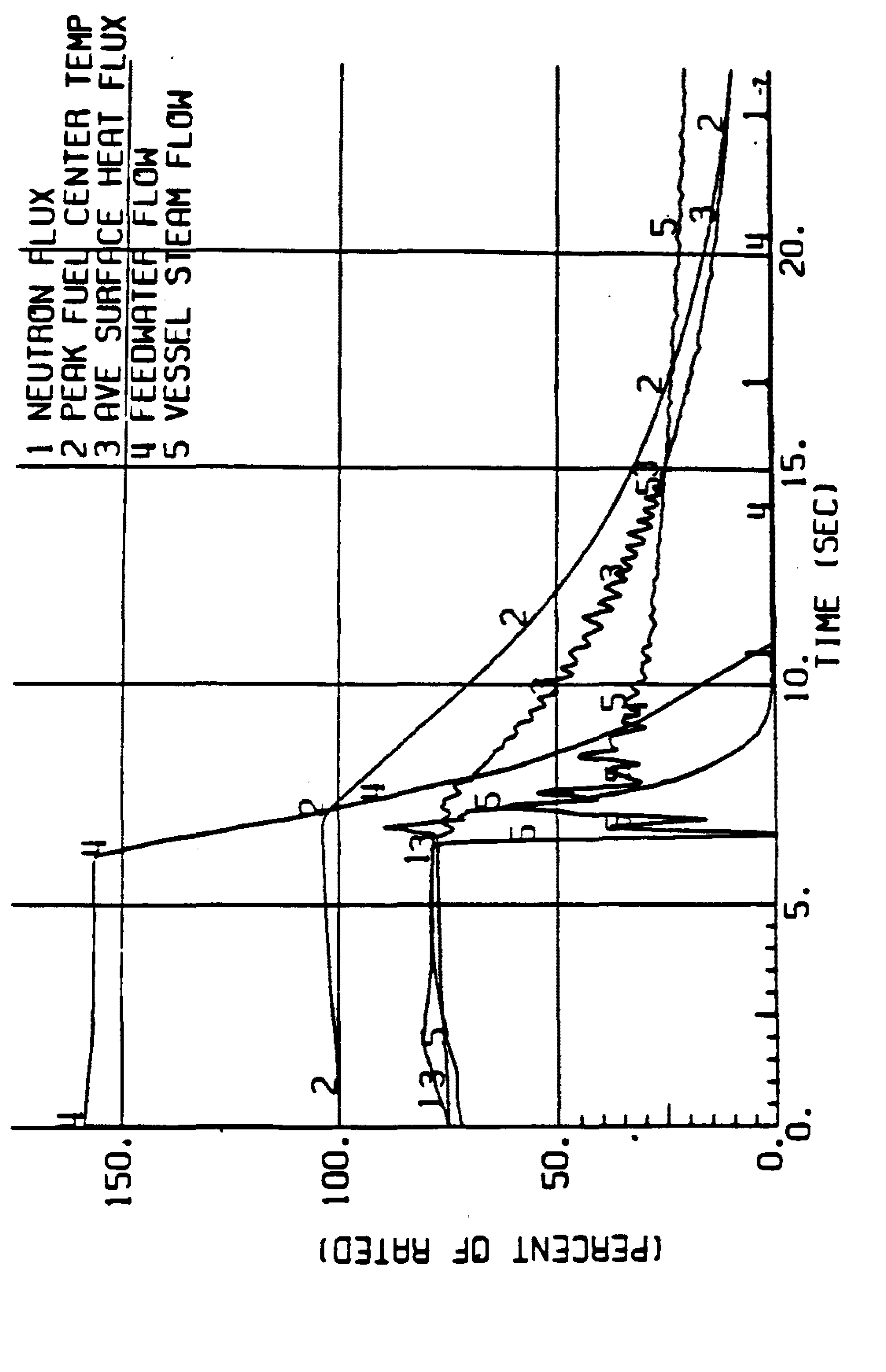
**Includes Option A adder.



PSE&G

FEEDWATER CONTROLLER FAILURE - MAXIMUM DEMAND
75% POWER/60% CORE FLOW

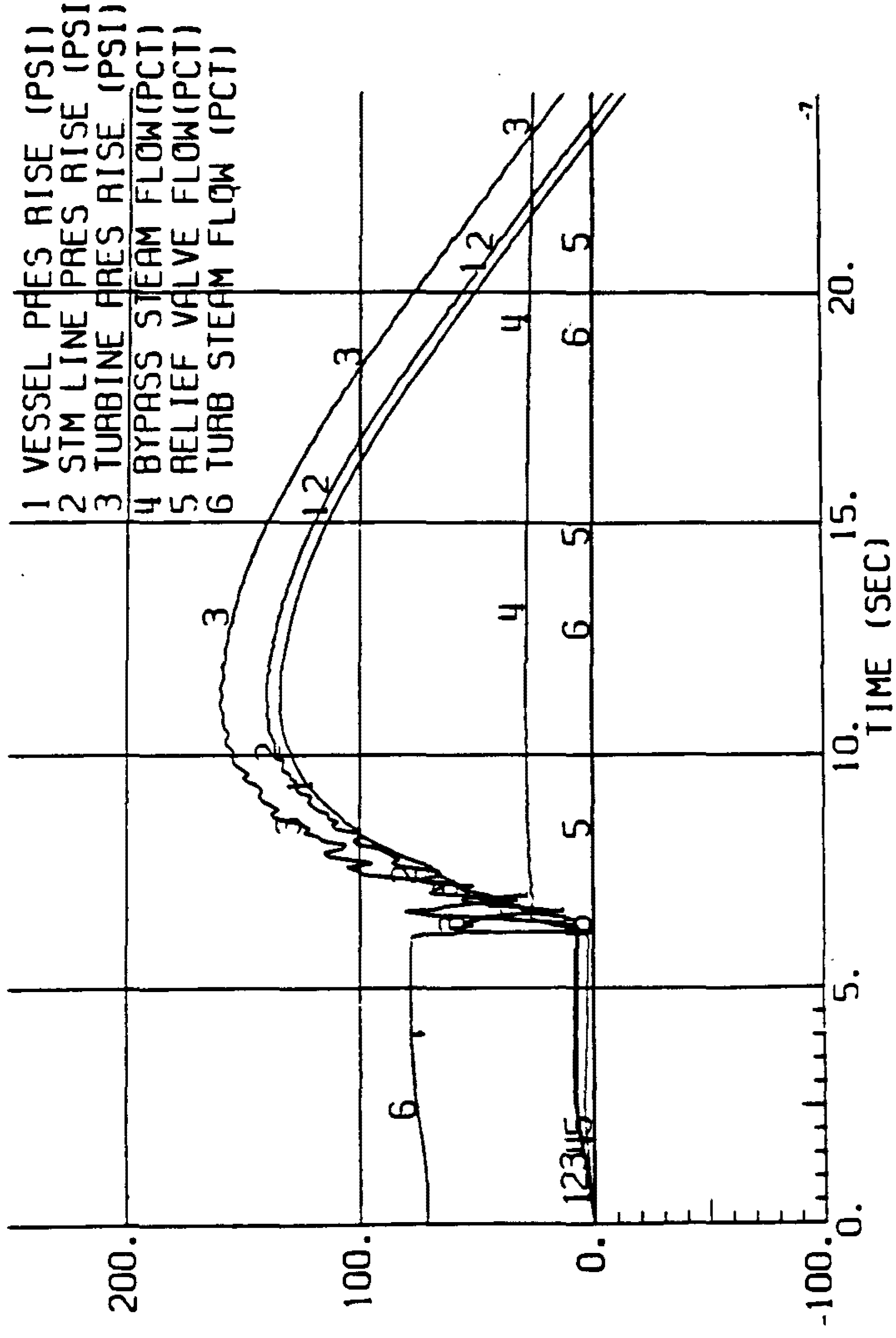
FIGURE 15.C.3-1



FEEDWATER CONTROLLER FAILURE - MAXIMUM DEMAND
75% POWER/60% CORE FLOW

FIGURE 15.C.3-1 CONT'D.

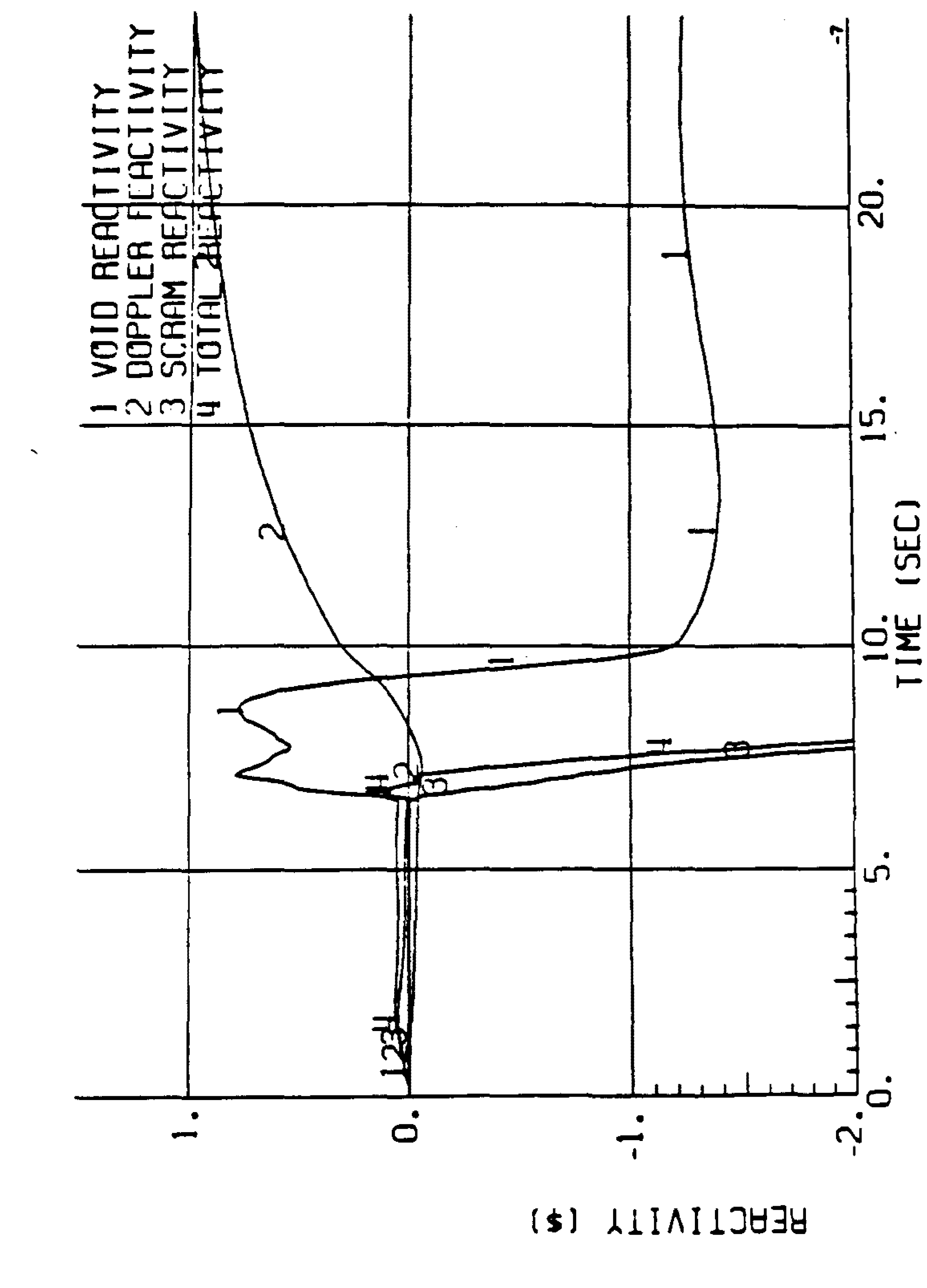
PSE&G



FEEDWATER CONTROLLER FAILURE - MAXIMUM DEMAND
75% POWER/60% CORE FLOW

FIGURE 15.C.3-1 CONT'D

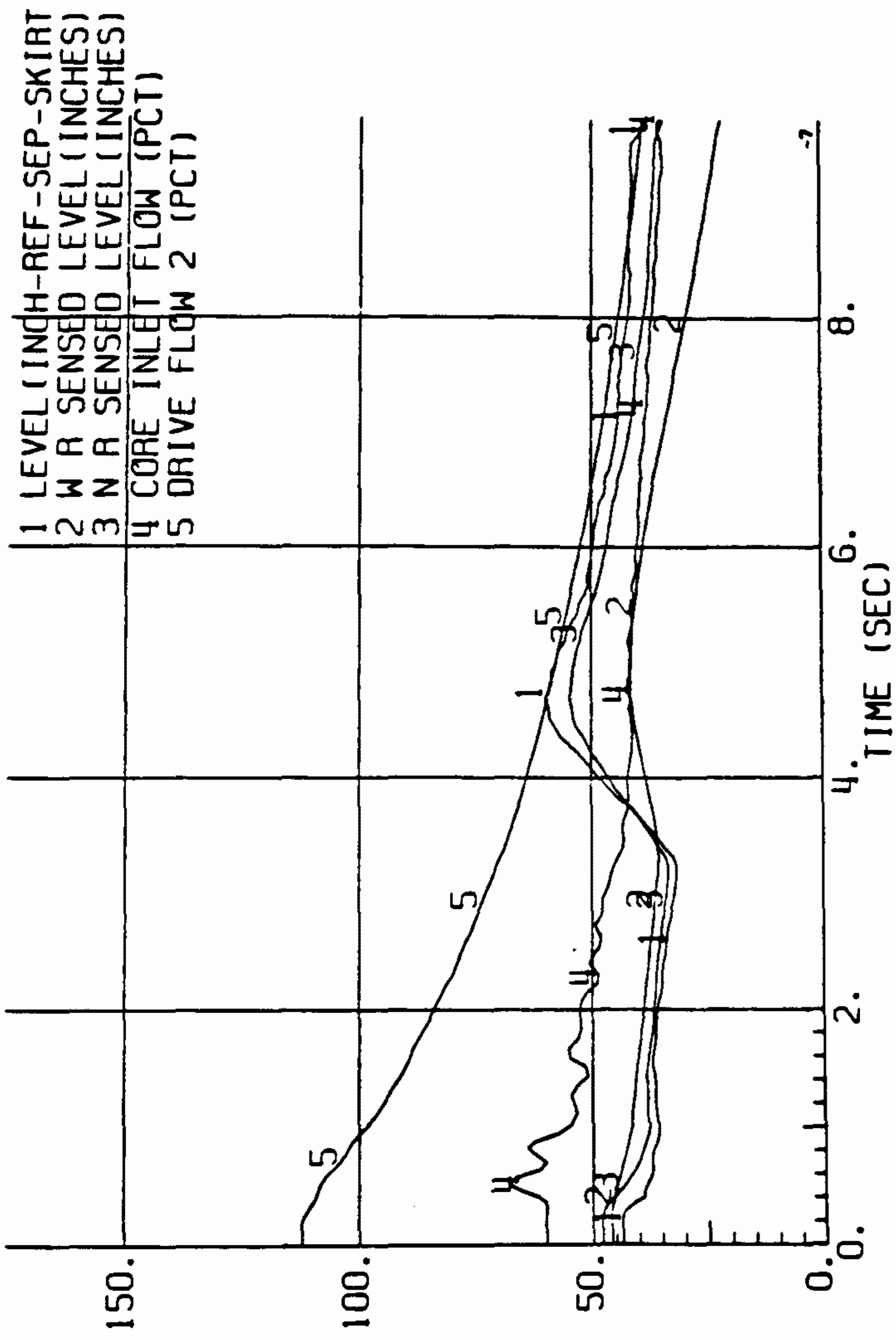
PSE&G



FEEDWATER CONTROLLER FAILURE - MAXIMUM DEMAND
75% POWER/60% CORE FLOW

PSE&G

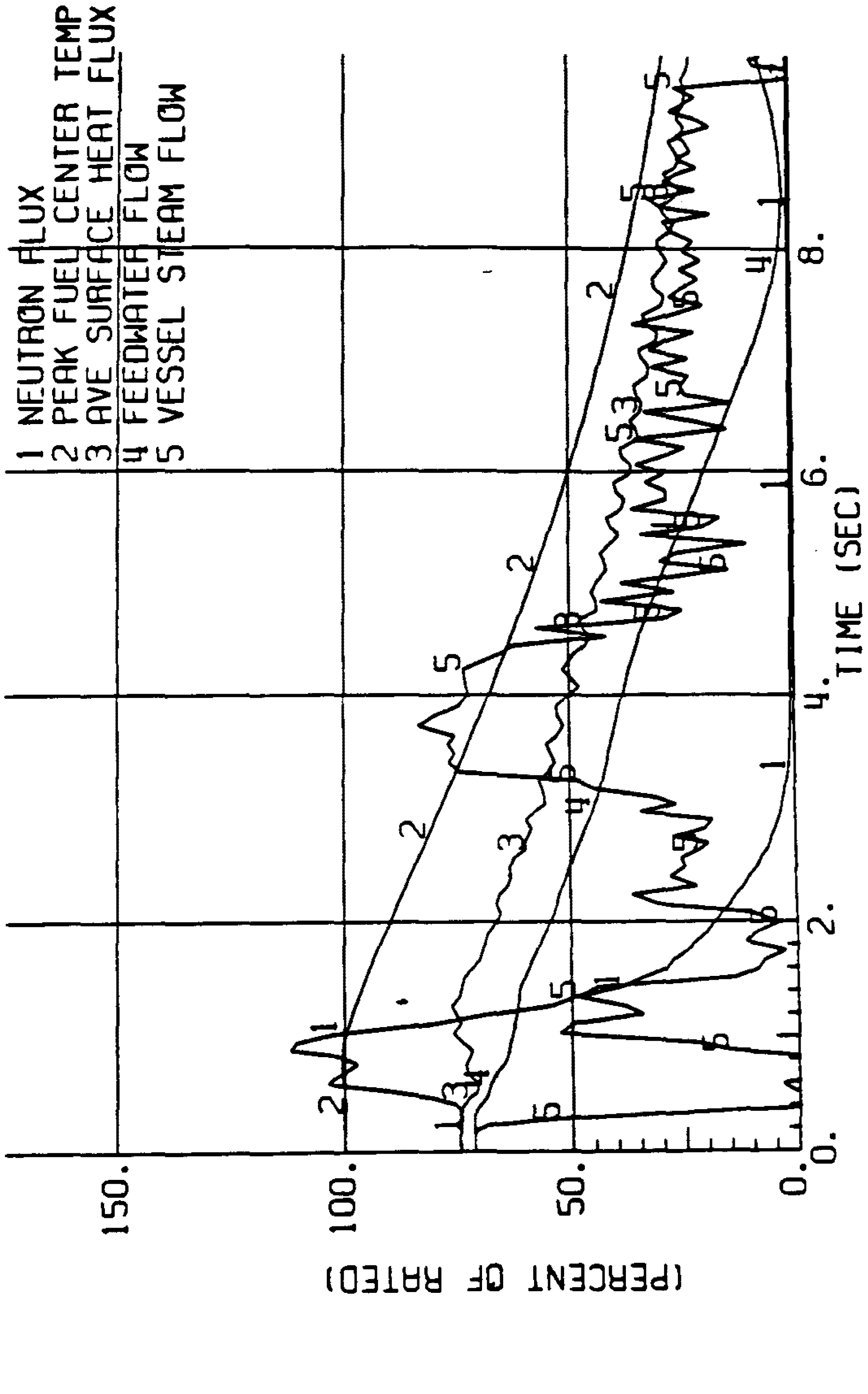
FIGURE 15.C.3-1 CONT'D.



GENERATOR LOAD REJECTION WITH BYPASS FAILURE, 75% POWER/60% CORE FLOW

FIGURE 15.C.3.2

PSE&G



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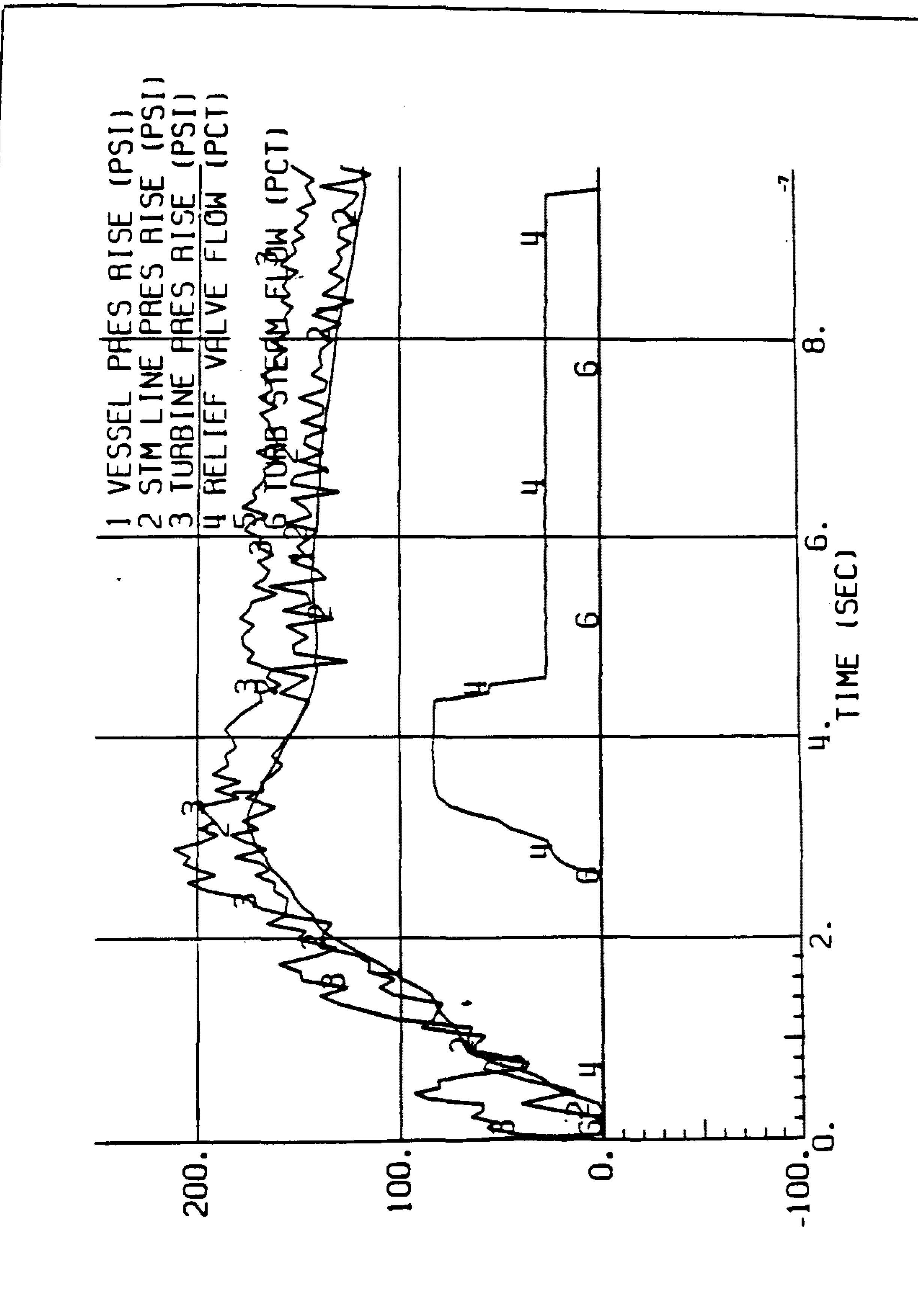
GENERATOR LOAD REJECTION WITH BYPASS FAILURE, 75% POWER/60% CORE FLOW

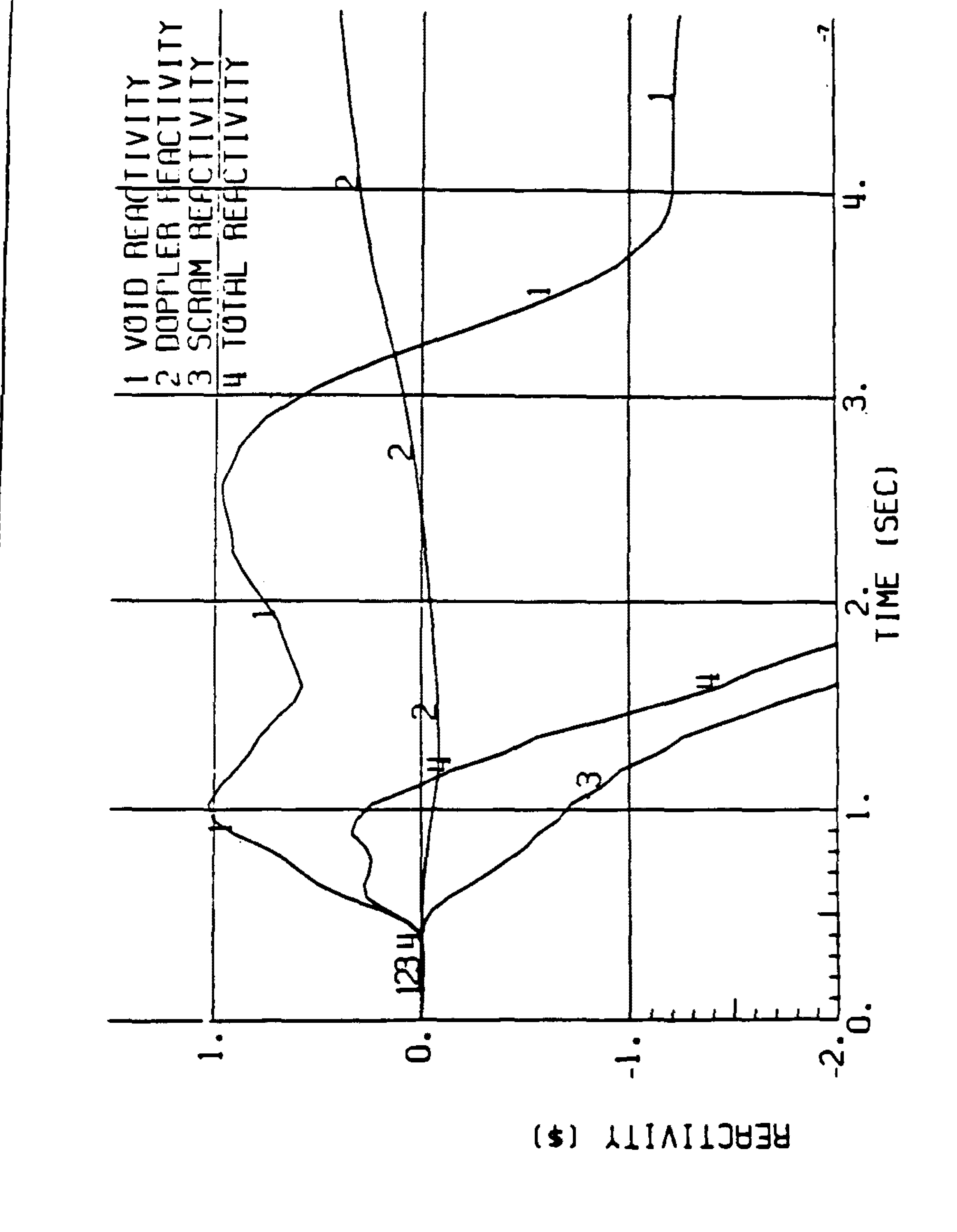
FIGURE 15.C.3.2 CONT'D.



GENERATOR LOAD REJECTION WITH BYPASS FAILURE, 75% POWER/60% CORE FLOW

FIGURE 15.C.3.2 CONT'D.

PSE&G



PSE&G

GENERATOR LOAD REJECTION WITH BYPASS FAILURE, 75% POWER/60% CORE FLOW

FIGURE 15.C.3.2 CONT'D

15.C.4 STABILITY ANALYSIS

15.C.4.1 Phenomena

The primary contributing factors to the stability performance with one recirculation loop not in service are the power/flow ratio and the recirculation loop characteristics. As forced circulation with only one recirculation loop in operation, the reactor core stability is influenced by the inactive recirculation loop. As core flow increases in SLO, the inactive jet pump forward flow decreases because the driving head across the inactive jet pumps decreases with increasing core flow. The reduced flow in the inactive loop reduces the resistance that the recirculation loops impose on reactor core flow perturbations thereby adding a destabilizing effect. At the same time the increased core flow results in a lower power/flow ratio which is a stabilizing effect. These two countering effects result in slightly decreased stability margin (higher decay ratio) initially as core flow is increased (from minimum) in SLO and then an increase in stability margin (lower decay ratio) as core flow is increased further and reverse flow in the inactive loop is established.

As core flow is increased further during SLO and substantial reverse flow is established in the inactive loop an increase in jet pump flow, core flow and neutron noise is observed. A cross flow is established in the annular downcomer region near the jet pump suction entrance caused by the reverse flow of the inactive recirculation loop. This cross flow interacts with the jet pump suction flow of the active recirculation loop and in-creases the jet pump flow noise. This effect increases the total core flow noise which tends to drive the neutron flux noise.

To determine if the increased noise is being caused by reduced stability margin as SLO core flow was increased, an evaluation was performed which phenomenologically accounts for single-loop operation effects on stability, as summarized in Reference 15.C.8-3. The model predictions were initially

compared with test data and showed very good agreement for both two-loop and single-loop test conditions. An evaluation was performed to determine the effect of reverse flow on stability during SLO. With increasing reverse flow, SLO exhibited slightly lower decay ratios than two-loop operation. However, at core flow conditions with no reverse flow, SLO was slightly less stable. This is consistent with observed behavior in stability tests at operating BWRs (Reference 15.C.8-4).

In addition to the above analyses, the cross flow established during reverse flow conditions was simulated analytically and shown to cause an increase in the individual and total jet pump flow noise, which is consistent with test data (Reference 15.C.8-3). The results of these analyses and tests indicate that the stability characteristics are not significantly different from two-loop operation. At low core flow, SLO may be slightly less stable than two-loop operation but as core flow is increased and reverse flow is established the stability performance is similar. At higher core flow with substantial reverse flow in the inactive recirculation loop, the effect of cross flow on the flow noise results in an increase in system noise (jet pump, core flow and neutron flux noise).

15.C.4.2 Compliance to Stability Criteria

Compliance with the stability licensing criteria set forth in 10CFR50 Appendix A, General Design Criterion (GDC-12), is achieved by either preventing stability-related neutron flux oscillations or detecting and suppressing the oscillations prior to exceeding Specified Acceptable Fuel Design Limits. The BWR Owners' Group (BWROG) has developed long-term solutions, which incorporate either prevention or detection and suppression features, or a combination of both features, to ensure compliance with GDC-12. Methodologies have been developed to support the licensing of these long-term solutions.

The BWROG has also developed guidelines (Reactor Stability Interim Corrective Actions) for the licensee to use prior to the licensee's successful implementation of a Long Term Stability Solution. These guidelines expand the interim corrective actions identified in NRC Bulletin 88-07, Supplement 1.

The expanded guidelines primarily accommodate the experience gained from plant stability events as well as conclusions based on recent analytical studies supporting the Long Term Stability solution. Based on the communications between the NRC and the BWROG, these guidelines fully satisfy the Bulletin 88-07, Supplement 1, requirements.

HCGS has implemented Reactor Stability Interim Corrective Actions based on the BWROG's recommendations to reduce the potential for unacceptable oscillations associated with the single-loop operation prior to the implementation of the Long Term Stability Solution.

15.C.5 LOSS-OF-COOLANT ACCIDENT ANALYSES

If two recirculation loops are operating and a pipe break occurs in one of the two recirculation loops, the pump in the unbroken loop is assumed to immediately trip and begin to coast down. The decaying core flow due to the pump coastdown results in very effective heat transfer (nucleate boiling) during the initial phase of the blowdown. Typically, nucleate boiling will be sustained during the first 5 to 9 seconds after the accident, for the design basis accident (DBA).

If only one recirculation loop is operating, and the break occurs in the operating loop, continued core flow is provided only by natural circulation because the vessel is blowing down to the reactor containment through both sections of the broken loop. The core flow decreases more rapidly than in the two-loop operating case, and the departure from nucleate boiling for the high power node might occur 1 or 2 seconds after the postulated accident, resulting in more severe cladding heatup for the one-loop operating case.

In addition to changing the blowdown heat transfer characteristics, losing recirculation pump coastdown flow can also affect the system inventory and reflooding phenomena. Of particular interest are the changes in the high-power node uncovery and reflooding times, the system pressure and the time of rated core spray for different break sizes. One-loop operation results in small changes in the high-power node uncovery times and times of rated spray. The effect of the reflooding times for various break sizes is also generally small.

Analyses single recirculation loop operation using the models and assumptions documented in References 15.C.8-9 or 15.C.8-10, as appropriate, are performed for HCGS. Using the appropriate methods, limiting pipe breaks are identified.

The single loop LOCA evaluation results in maximum planar linear heat generation rate (MAPLHGR) curves specific to single loop operation which assume that LOCA acceptance criteria in 10CFR50.46 are satisfied.

FIGURE 15.C.5-1 HAS BEEN DELETED

15.C.6 CONTAINMENT ANALYSIS

The range of power/flow conditions which are included in the SLO operating domain for Hope Creek were investigated to determine if there would be any impact on the FSAR specifications for containment response, including the containment dynamic loads. The SLO operating conditions were confirmed to be within the range of operating conditions which have previously been considered in defining the containment pressure and temperature response and containment dynamic loads for two-loop operation. Therefore, the containment response for Hope Creek with single-loop operation has been confirmed to be within the present design values.

15.C.7 MISCELLANEOUS IMPACT EVALUATION

15.C.7.1 Anticipated Transient Without Scram (ATWS) Impact Evaluation

The principal difference between single-loop operation (SLO) and normal two-loop operation (TLO) affecting Anticipated Transient Without Scram (ATWS) performance is that of initial reactor conditions. Since the SLO initial power flow condition is less than the rated condition used for TLO ATWS analysis, the transient response is less severe and therefore bounded by the TLO analyses.

It is concluded that if an ATWS event were initiated at HCGS from the SLO conditions, the results would be less severe than if it were initiated from rated conditions.

15.C.7.2 Fuel Mechanical Performance

Component pressure differential and fuel rod overpower values for anticipated operational occurrences initiated from SLO conditions have been found to be bounded by those applied in the fuel rod and assembly design bases.

It is observed that due to the substantial reverse flow established during SLO both the Average Power Range Monitor (APRM) noise and core plate differential pressure noise are slightly increased. An analysis has been carried out to determine that the APRM fluctuation should not exceed a flux amplitude of ±15% of rated and the core plate differential pressure fluctuation should not exceed 3.2 psi peak to peak to be consistent with the fuel rod and assembly design bases.

15.C.7.3 Vessel Internal Vibration

Vibration tests for SLO were performed during the startup of two BWR 4-251 plants. An extensive vibration test was conducted at a prototype BWR 4-251 plant, Browns Ferry 1, the results of which are used as a standard for comparison. A confirmatory vibration test was performed at the Peach Bottom 2 & 3 plants.

The Browns Ferry 1 test data demonstrates that all instrumented vessel internals components vibrations are within the allowable criteria. The highest measured vibration in terms of percent criteria for single-loop operation was 70%. This was measured at a jet pump riser brace during cold flow conditions at 100% of rated pump speed.

The Peach Bottom vibration test data shows that vessel internals vibration levels are within the allowable criteria for all test conditions. The highest measured vibration in terms of percent criteria for single-loop operation was 96%. This was measured at a jet pump elbow location during 68% power condition at 92% of rated pump speed. This vibration amplitude is the highest, in terms of percent criteria, experienced in vessel internals for the BWR 4-251 plants studied.

The conclusion is that under all operating conditions, the vibration level is acceptable. However, due to the high vibration levels recorded, it is recommended that Hope Creek not perform single-loop operation with pump speed exceeding 90% of rated pump speed. The same recommendation has been accepted by the Browns Ferry and Peach Bottom plants.

This analysis is conservative because the criteria are developed by assuming that the plant operates on a steady state single loop operations throughout the plant life.

15.C.8 REFERENCES

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I5.C.8-2 "Qualification of the One-Dimensional Core Transient Model for Boiling Water Reactors", NEDO-24154-A, August 1986.

15.C.8-3 Letter, H.C. Pfefferlen (GE) to C.O. Thomas (NRC), "Submittal of Response to Stability Action Item from NRC Concerning Single-Loop Operation," September 1983.

15.C.8-4 S.F. Chen and R.O. Niemi, "Vermont Yankee Cycle 8 Stability and Recirculation Pump Trip Test Report", General Electric Company, August 1982 (NEDE-25445, Proprietary Information).

15-C-8-5 G.A. Watford, "Compliance of the General Electric Boiling Water Reactor Fuel Designs to Stability Licensing Criteria", General Electric Company, October 1984 (NEDE-22277-P-1, Proprietary Information).

15.C.8-6 "General Electric Standard Application for Reactor Fuel", NEDE-24011-P-A, and "General Electric Standard Application for Reactor Fuel (Supplement for United States)," NEDE-24011-P-A-US, latest revision.

15.C.8-7 "BWR Core Thermal Hydraulic Stability". General Electric Company, February 10, 1984 (Service Information Letter-380, Revision 1).

15.C.8-8 Letter, C.O. Thomas (NRC) to H.C. Pfefferlen (GE), "Acceptance for Referencing of Licensing Topical Report NEDE-24011, Rev. 6, Amendment 8, Thermal Hydraulic Stability Amendment to GESTAR II," April 24, 1985.

15.C.8-9 "*SAFER/GESTR-LOCA Loss-of-Coolant Accident Analysis for Hope Creek Generating Station at Power Uprate,"* NEDC-33172P, March 2005.

15.C.8-10 ABB Combustion Engineering Nuclear Power, "Reference Safety Report for Boiling Water Reactor Reload Fuel," ABB Report CENPD-300-P-A (proprietary), July 1996.

15.C.8-11 Latest BWROG recommendations for "interim Stability Solution".

15.C.8-12 NEDC-33076P, Rev. 2, "Safety Analysis Report for Hope Creek Constant Pressure Power Uprate", August 2006.

APPENDIX 15D

CYCLE 21 RELOAD ANALYSIS RESULTS

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Cycle 21 Reload Analysis Results

15D.1 INTRODUCTION AND PURPOSE

During each reload, fresh fuel assemblies are loaded into the core. A change in fuel design and core configuration has the potential to affect the results of the Section 15 events. Therefore an analysis of the potentially limiting events is performed on a cycle-to-cycle basis. This analysis is known as the reload licensing analysis. This appendix to Section 15 represents the results of cycle specific reload licensing analysis.

The purpose of this appendix is to summarize the cycle specific reload licensing analysis. This appendix is referenced throughout Section 15 for the results of the appropriate events. It is also referenced in Section 5.2.2.

15D.2 RELOAD METHODOLOGY

The NRC-approved reload methodology is documented in GESTAR II (Reference 15D.5-1).

The reload methodology is used to perform an evaluation of the potentially limiting events. The potentially limiting events can be divided into three groups: Anticipated Operational Occurrences (AOOs), Design Basis Accidents (DBAs) and Special Events. The AOOs are:

- Loss of Feedwater Heating (LOFH): See Section 15.1.1
- Feedwater Controller Failure Maximum Demand (FWCF): See Section 15.1.2
- Generator Load Rejection, No Bypass(GLRNB): See Section 15.2.2
- Turbine Trip, No Bypass (TTNB): See Section 15.2.3
- Rod Withdrawal Error at Power (RWE): See Section 15.4.2

The DBAs are:

- Loss of Coolant Accident (LOCA): See Section 15.6.5
- Misloaded Fuel Bundle Accident (Mislocated or Misoriented): See Section 15.4.7
- Control Rod Drop Accident (CRDA): See Section 15.4.9
- Fuel Handling Accident: See Section 15.7.4

The special events are:

- Shutdown without Control Rods: (none identified)
- Core Thermal-Hydraulic Stability: (none identified)
- ASME Over-Pressurization: See Section 5.2.2
- Anticipated Transient Without Scram (ATWS): See Section 15.8

In addition to the aforementioned events, an assessment is made to re-confirm that the results of the events evaluated for two recirculation loop operation bounds the single recirculation loop configuration, or specific single loop operation limits are established.

15D.3 RELOAD ANALYSIS RESULTS

The results of the Cycle 21 reload analysis are presented within this section, or can be found in Reference 15D.5-2.

15D.3.1 Loss of Feedwater Heating

The description of the Loss of Feedwater Heating (LOFH) is found in Section 15.1.1.

The results presented in this section assume that the plant is operating in manual flow control mode.

15D.3.1.1 Initial Conditions

The analysis has been performed with the conditions tabulated in Table 15D-1.

15D.3.1.2 Sequence of Events

The LOFH event is analyzed with a three-dimensional core simulator (Reference 15D.5-1). This is a conservative steady state analysis for the determination of the appropriate power distribution limits during the event. Since it is not a dynamic simulation, no sequence of events is available.

The event can be initiated by closure of an extraction line to a feedwater heater or by bypassing one or more feedwater heaters. No subsequent operator action to mitigate plant response to the loss of feedwater heating is assumed.

15D.3.1.3 Results

The initiation of the LOFH event is an assumed 110°F reduction in feedwater temperature. The analysis results for the LOFH in the manual flow control mode are summarized in Table 15D-2 and in Reference 15D.5-2.

15D.3.2 Feedwater Controller Failure - Maximum Demand

The description of the Feedwater Controller Failure - Maximum Demand (FWCF) is found in Section 15.1.2.

15D.3.2.1 Initial Conditions

The analysis has been performed with the conditions tabulated in Table 15D-1. The FWCF event has the potential to be the limiting event.

15D.3.2.2 Sequence of Events

The sequences of events for the FWCF analysis are listed in Table 15D-3.

15D.3.2.3 Results

Analysis results for the FWCF events are presented in Figure 15D-1. This figure presents the transient variation of various important system parameters (Reference 15D.5-2).

15D.3.3 Generator Load Rejection, No Bypass

The description of the Generator Load Rejection, No Bypass (GLRNB) is found in Section 15.2.2.

15D.3.3.1 Initial Conditions

The analysis has been performed with the conditions tabulated in Table 15D-1. The values tabulated in Table 15D-1 represent analysis assumptions, which were established as design input for this event as described in GESTAR (Reference 15D.5-1).

15D.3.3.2 Sequence of Events

The sequence of events for the GLRNB analysis is listed in Table 15D-4.

15D.3.3.3 Results

The analysis results for the GLRNB are presented in Figure 15D-2. This figure presents the transient variation of various important system parameters (Reference 15D.5-2).

15D.3.4 Turbine Trip, No Bypass

The description of the Turbine Trip, No Bypass (TTNB) is found in Section 15.2.3.

The TTNB event is similar to the GLRNB event. Although the two events have different initiating faults, the TTNB event parameter responses follow the same trend as the GLRNB event response. The TTNB event was analyzed for Cycle 21.

15D.3.4.1 Initial Conditions

The analysis has been performed with the conditions tabulated in Table 15D-1.

The values tabulated in Table 15D-1 represent analysis assumptions, which were established as design input for this event as described in GESTAR (Reference 15D.5-1). These analysis assumptions are the same as for the GLRNB event.

15D.3.4.2 Sequence of Events

The sequence of events for the TTNB analysis is similar to the GLRNB in Table 15D-4.

15D.3.4.3 Results

The analysis results for the TTNB are presented in Figure 15D-3. This figure presents the transient variation of various important system parameters (Reference 15D.5-2).

15D.3.5 Rod Withdrawal Error

The description of the Rod Withdrawal Error (RWE) is found in section 15.4.2.

15D.3.5.1 Initial Conditions

The analysis has been performed with the conditions tabulated in Table 15D-1.

The values tabulated in Table 15D-1 represent analysis assumptions, which were established as conservative design input for this event as described in GESTAR (Reference 15D.5-1).

15D.3.5.2 Sequence of Events

The RWE event is analyzed with a three-dimensional core simulator (see Reference 15D.5-3). This is a conservative steady state analysis for the determination of the appropriate power distribution limits during the event. Since, it is not a dynamic simulation, no sequence of events is available.
An operator is assumed to erroneously select and continuously withdraw a control rod at its maximum withdrawal rate at rated conditions until rod withdrawal is terminated by the Rod Block Monitor system.

15D.3.5.3 Results

The ARTS based rod withdrawal error is an unblocked basis. The unblocked rod withdrawal error results in a ΔCPR of 0.31 (cycle OLMCPR of 1.39).

15D.3.6 Section 15D.3.6 deleted.

15D.3.7 Loss of Coolant Accident

The description of the loss of coolant accident (LOCA) is found in Section 15.6.5.

The LOCA is a design bases accident. The GE14 fuel was analyzed for the LOCA and the results are summarized in Reference 15D.5-5. The GNF2 fuel was analyzed for the LOCA and the results are summarized in Reference 15D.5-14.

The consequences of a design basis LOCA are evaluated for each unique reload fuel design to support the establishment of core operating limits for that fuel design. This evaluation establishes appropriate Maximum Average Planar Linear Heat Generation rate (MAPLHGR) limits for the reload fuel (Reference 15D.5-2). The operation of the core within these established MAPLHGR limits ensures that the ECCS LOCA requirements are met. The MAPLHGR operating limits for reload fuel are presented and controlled in the cycle's Core Operating Limit Report (COLR).

15D.3.8 Misloaded Fuel Bundle Accident

The description of the Misloaded Fuel Bundle Accident is found in Section 15.4.7.

The reload licensing methodology analyzes two events in this category: the mislocated fuel bundle event and the misoriented fuel bundle event (Reference 15D.5-1). Although both events are classified as accidents, each is analyzed as an operating transient (AOO) in accordance with GESTAR (Reference 15D.5-1).

15D.3.8.1 Mislocated Fuel Bundle

This design basis accident involves the mislocation of a fuel assembly into the wrong core location and the subsequent operation of the reactor with the mislocated assembly.

The sequence of events for the mislocated fuel bundle accident is presented in Table 15D-7.

The results of the mislocated fuel bundle accident are presented in terms of cycle Operating Limit MCPR in Table 15D-8. (Reference 15D.5-2)

15D.3.8.2 Misoriented Fuel Bundle

This design basis event involves the misorientation (rotation) of a fuel assembly relative to the orientation assumed in the reference core design.

The sequence of events for the misloaded fuel bundle accident is presented in Table 15D-7.

The results of the misoriented fuel bundle accident are presented in terms of cycle Operating Limit MCPR in Table 15D-9. (Reference 15D.5-2)

15D.3.9 Control Rod Drop Accident

The description of the Control Rod Drop Accident is described in Section 15.4.9.

HCGS is a Banked Position Withdrawal Sequence (BPWS) plant, and therefore, in accordance with GESTAR II (Reference 15D.5-1), does not need to analyze the control rod drop accident (CRDA) each reload. The results of a generic analysis of the event are provided in Section 15.4.9.

15D.3.10 Fuel Handling Accident

The current HCGS licensing analysis bounds the consequences of any fuel handling accident (see Section 15.7.4).

15D.3.11 Shutdown Without Control Rods

The Standby Liquid Control System (SLCS) shutdown capability has been evaluated at a moderator temperature of 160°C as a function of exposure for a core Boron concentration equivalent to 660 ppm at 20°C. The minimum shutdown margin for the Hope Creek Cycle 21 core is 0.023 Δk (Reference 15D.5-2).

15D.3.12 Core Thermal-Hydraulic Stability

GE SIL-380 recommendations, BWROG Interim Corrective Actions (Reference 15D.5-11) and Backup Stability Protection for Inoperable Option III Solution (Reference 15D.5-10) have been included in the Hope Creek Cycle 21 operating procedures. Regions of restricted operation defined in Attachment 1 to NRC Bulletin No. 88-07, Supplement 1, "Power Oscillations in Boiling Water Reactors (BWRs)" and expanded in BWROG Interim Corrective Actions (Reference 15D.5-11) and Backup Stability Protection for Inoperable Option III Solution (Reference 15D.5-10) are used for Hope Creek Cycle 21 backup stability protection evaluation (Reference 15D.5-2). The standard interim corrective action stability regions are expanded as appropriate to offer stability protection per Reference 15D.5-12 and Reference 15D.5-10 for Hope Creek Cycle 21 operation.

Hope Creek has implemented BWROG Long Term Stability Solution Option III using the Oscillation Power Range Monitor (OPRM) as described in Reference 15D.5-6. The cycle-specific stability OPRM Amplitude Setpoint and Successive Confirmation Count (SCC) Setpoint are confirmed for each cycle using the Option III Solution with GS3 methodology described in Reference 15D.5-6 and reported in Reference 15D.5-2.

All stability calculations for Hope Creek Cycle 21 include the limitations presented in Reference 15D.5-13.

15D.3.13 ASME Over-Pressurization

The ASME over-pressurization analysis is performed to evaluate margin to the vessel pressure safety limit. The basis for this event is described in Section 5.2.2.

MSIV closure with flux scram was found to be the most limiting event in terms of vessel pressure. The results are summarized as follows:

- Maximum Vessel Pressure 1289 psig
- Maximum Steam Dome Pressure 1268 psig
- Maximum Steam Line Pressure 1263 psig

The scram on MSIV position is not credited for this event. The maximum pressures during the event are below the ASME upset code limit of 1375 psig, which is 110% of the reactor vessel design pressure. Furthermore the maximum steam dome pressure predicted during the event is below the Technical Specification steam dome pressure safety limit of 1325 psig. (Reference 15D.5-2)

15D.3.14 Section Deleted

15D.4 Single Loop Operation

GNF has confirmed that the basis for single loop operation (SLO) presented in Appendix 15C remains valid for the current cycle. The confirmation involves an analysis at core power consistent with, or bounding the Technical Specification limit. Reference 15D.5-2 identifies that for Cycle 21, for single loop operation, the safety limit MCPR will be 1.11 and the LHGR and MAPLHGR multiplier will be 0.80. The cycle-specific stability OPRM-setpoint values, confirmed per Reference 15D.5-6 and reported in Reference 15D.5-2, support SLO per Reference 15D.5-6.

15D.5 References

15D.5-1 General Electric Company, "General Electric Standard Application for Reactor Fuel," NEDE-24011-P-A, and "General Electric Standard Application for Reactor Fuel (Supplement for United States)," NEDE-24011-P-A-US, latest revision.

15D.5-2 Global Nuclear Fuel - Americas, "Supplemental Reload Licensing Report for Hope Creek Reload 20 Cycle 21," 002N6864, Revision 0, September 2016.

15D.5-3 "*Steady-State Nuclear Methods*," NEDE-30130-P-A, April 1985.

15D.5-4 Deleted

15D.5-5 "*SAFER/GESTR-LOCA Loss-of-Coolant Accident Analysis for Hope Creek Generating Station at Power Uprate*," NEDC-33172P, March 2005.

15D.5-6 "GEH Simplified Stability Solution (GS3)," NEDE-33766P-A, Revision 1, March 2015.

15D.5-7 Deleted.

15D.5-8 Deleted.

15D.5-9 Deleted.

15D.5-10 "Backup Stability Protection (BSP) for Inoperable Option III Solution," OG-02-0119-260, July 2002.

15D.5-11 "BWR Owners' Group Guidelines for Stability Interim Corrective Action," BWROG-94079, June 1994.

15D.5-12 "Review of BWR Owners' Group Guidelines for Stability Interim Corrective Action," BWROG-02072, November 2002.

15D.5-13 "Applicability of GE Methods to Expanded Operating Domains," Licensing Topical Report, NEDC-33173P-A, Revision 4, November 2012.

15D.5-14 "Hope Creek Generating Station GNF2 ECCS-LOCA Evaluation", 002N5176-R0, Revision 0, August 2016

TABLE 15D-1

INPUT PARAMETERS AND INITIAL CONDITIONS FOR RELOAD LICENSING ANALYSIS

1. Loss of Feedwater Heating:

Power (% of Rated)	100
Core Flow (% of Rated)	94.8
Feedwater Temperature (°F) minus 110°F	321.6
Core Mid-Plane Pressure (psia)	1034.1
Core Coolant Inlet Enthalpy (BTU/lbm)	523.8
Core Average Void Fraction (%)	50.7
Cycle Exposure	MOC

2. Feedwater Controller Failure – Maximum Demand

Power (% of Rated)	100
Core Flow (% of Rated)	105
Steam Flow (Mlbm/Hr)	14.7
Feedwater Flow Rate (Mlbm/Hr)	14.7
Feedwater Temperature (°F)	329.6
Steam Dome Pressure (psig)	984.3
Core Exit Pressure (psig)	996.3
Core Coolant Inlet Enthalpy (BTU/lbm)	511.0
Core Average Void Fraction (%)	39.9
Cycle Exposure	EOC with ICF
EOC-RPT Operable	Yes

3. Generator Load Rejection, No Bypass; and Turbine Trip, No Bypass

Power (% of Rated)	100
Core Flow (% of Rated)	105
Steam Flow (Mlbm/Hr)	16.8
Feedwater Flow Rate (Mlbm/Hr)	16.8
Feedwater Temperature (°F)	431.6
Steam Dome Pressure (psig)	1005.3
Core Exit Pressure (psig)	1018.3
Core Coolant Inlet Enthalpy (BTU/lbm)	526.3
Core Average Void Fraction (%)	44.9
Cycle Exposure	EOC with ICF
EOC-RPT Operable	Yes

4. Rod Withdrawal Error

Power (% of Rated)	100
Core Flow (% of Rated)	100
Steam Flow (Mlbm/Hr)	16.8
Feedwater Flow Rate (Mlbm/Hr)	16.8
Feedwater Temperature (°F)	431.6
Core Mid-Plane Pressure (psig)	1020.3
Core Coolant Inlet Enthalpy (BTU/lbm)	525.2
Core Average Void Fraction (%)	47.7
Limiting Rod Pattern (for MCPR Determination)	Yes
Cycle Exposure	7,500 MWd/ST

TABLE 15D-2

RESULT SUMMARY FOR LOSS OF FEEDWATER HEATING

MANUAL CONTROL

Final Conditions:

Power (% of Rated)	118.6
Core Flow (% of Rated)	96.8
Feedwater Temperature (°F)	321.6
Core Mid-Plane Pressure (psig)	1026.2
Core Exit Pressure (psig)	N/A
Core Coolant Inlet Enthalpy (BTU/lbm)	503.5
Core Average Void Fraction (%)	49.6
Peak Neutron Flux (% of Rated)[(1)]	N/A

(1) The APRM simulated thermal Flux scram is not credited.

TABLE 15D-3

SEQUENCE OF EVENTS FOR FEEDWATER CONTROLLER FAILURE MAXIMUM DEMAND

Time (Seconds)	Event Descriptions
0	A simulated failure to the Feedwater pump runout flow
11.9	L8 vessel level setpoint trips main turbine and feedwater pumps. Turbine bypass operation is initiated.
11.9	Reactor scram actuated from TSV position switches.
14.9	Relief valves start to open.

TABLE 15D-4

SEQUENCE OF EVENTS FOR GENERATOR LOAD REJECTION WITHOUT BYPASS OPERATION

Time (Seconds)	Event Descriptions
< 0.0	Loss of electrical load detected by the turbine generator.
0.0	Turbine generator load rejection sensing devices trip to initiate turbine control valve fast closure.
0.0	Turbine bypass valves fail to operate. Fast closure of TCVs initiates scram and RPT.
0.07	TCVs are closed.
1.6	Group 1 SRVs are actuated in relief mode.

TABLE 15D-5

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TABLE 15D-6

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TABLE 15D-7

SEQUENCE OF EVENTS FOR MISLOADED FUEL BUNDLE ACCIDENT

1. During core loading operation, a fuel bundle is placed in the wrong location (or misoriented in the proper core location).

2. Subsequently, the bundle intended for this location is placed in the assigned location of the previously misplaced bundle (Not Applicable for misoriented bundle accident)

3. During core verification procedure, these errors are not observed

4. The plant is brought to full power operation without detecting misplaced bundles

5. The plant continues to operate

TABLE 15D-8

Mislocated Fuel Assembly Results

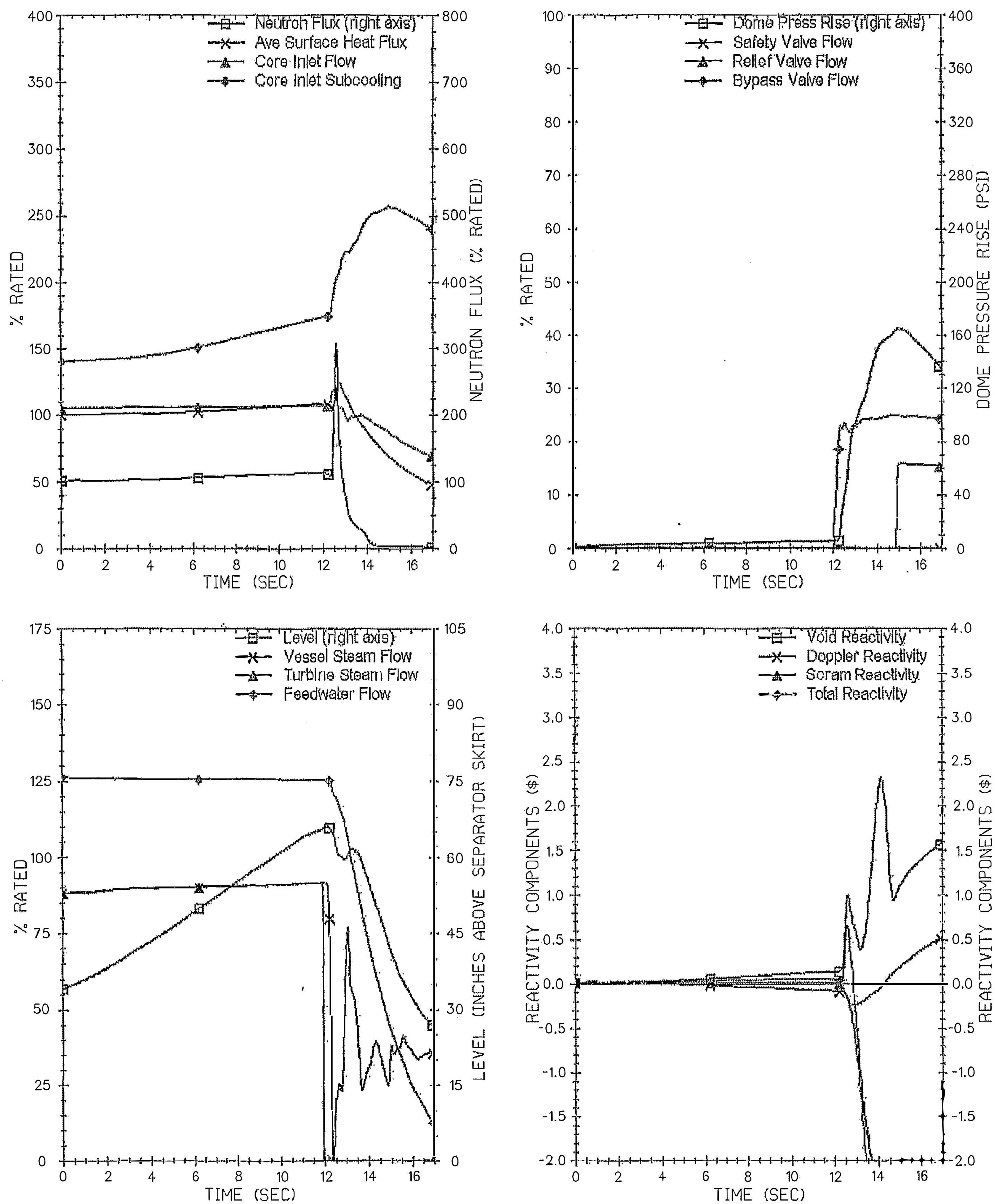
Burnup Range	**OLMCPR**
	GNF2/GE14
BOC13-EOC13	1.21
BOC21-EOC21	Non-Limiting

The Mislocated Fuel Loading Error was determined to be non-limiting for Cycle 21 based on Cycle 13 results and the experience and procedural basis for the limiting fuel type, GNF2. The Cycle 21 "Non-Limiting" disposition for GNF2 also applies for the GE14 fuel.

TABLE 15D-9

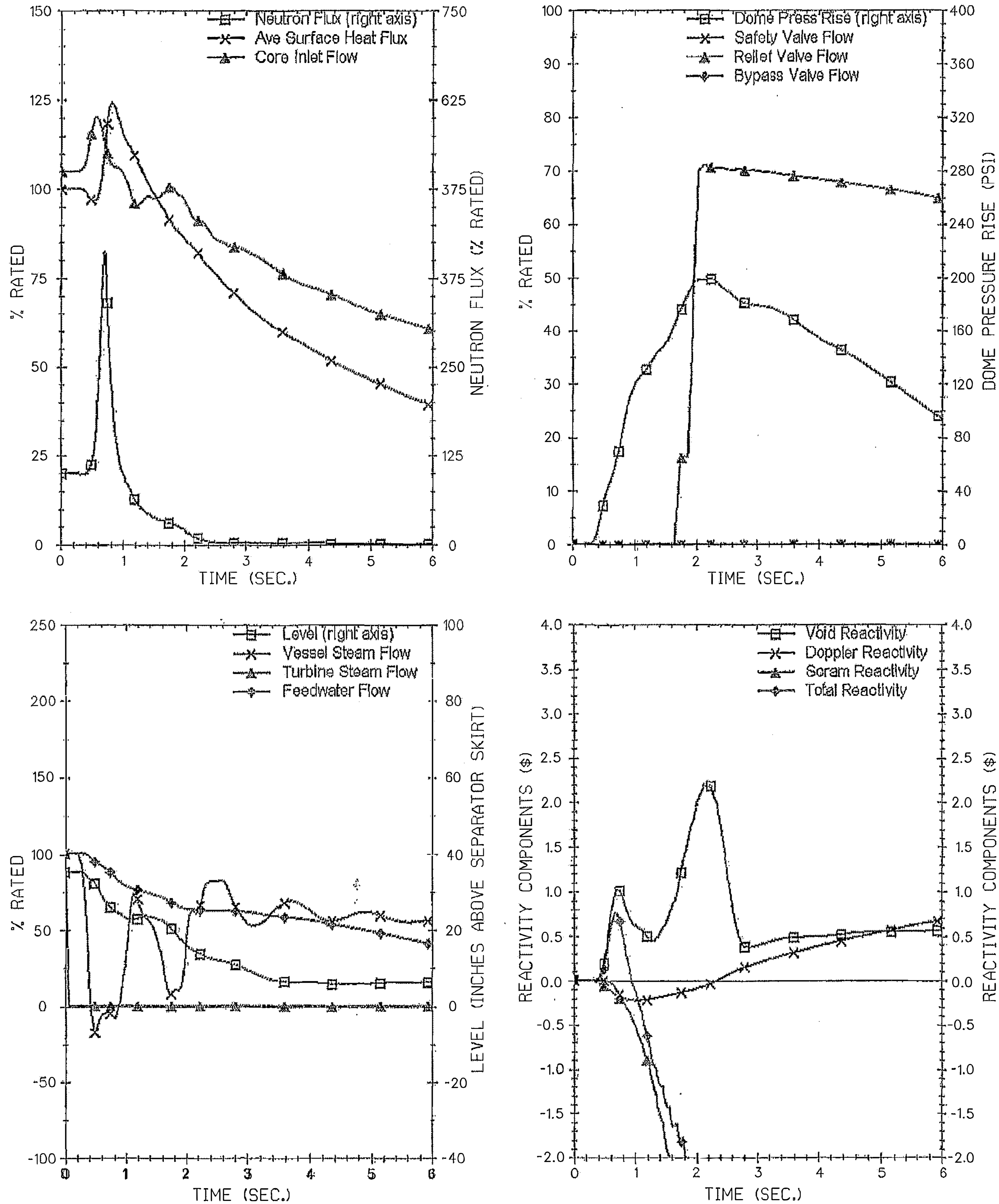
Misoriented Fuel Assembly Results

Burnup Range	**OLMCPR**	
	GNF2	**GE14**
BOC21-EOC21	1.21	1.24



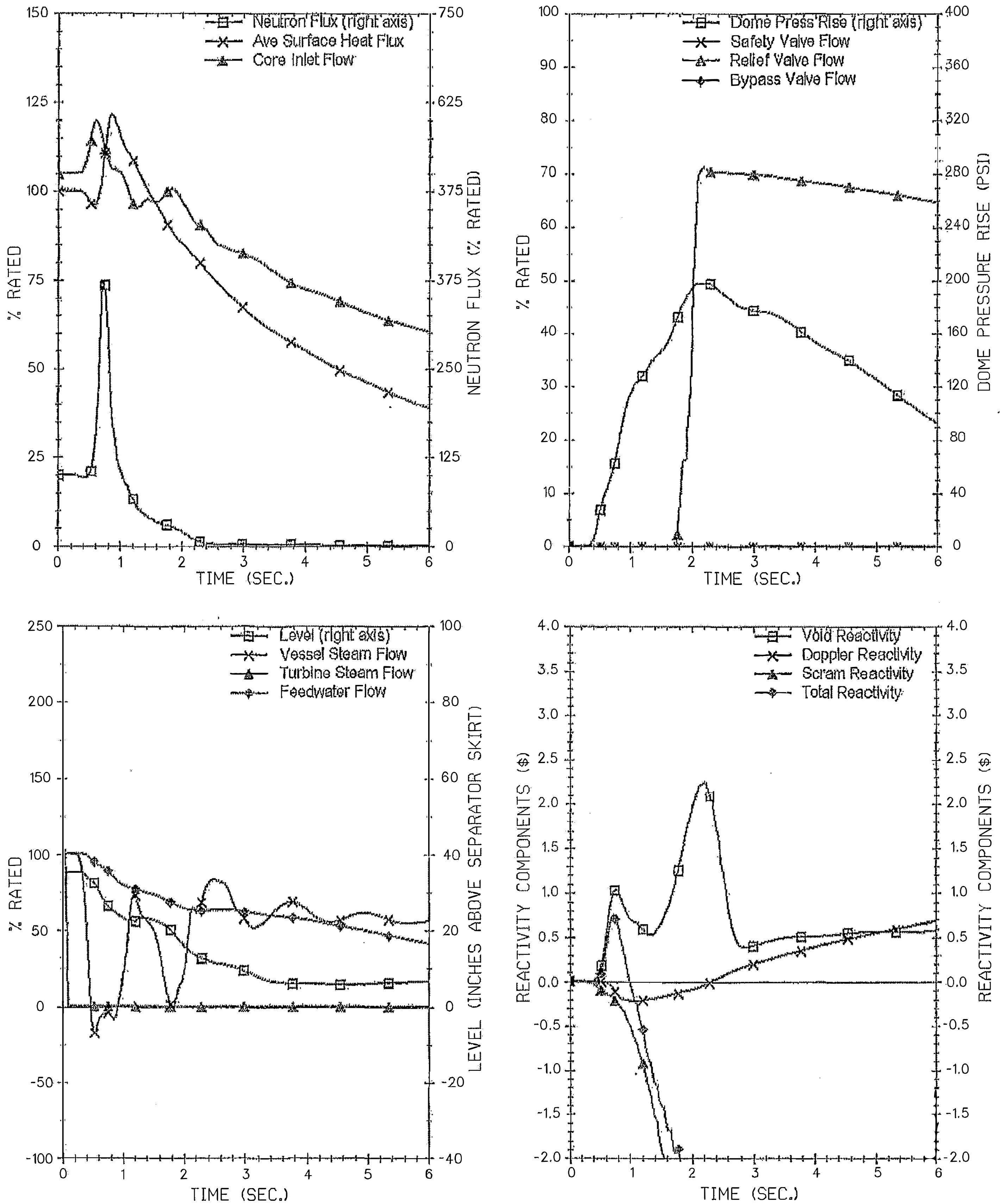
REVISION 22, MAY 9, 2017

PSEG Nuclear, LLC HOPE CREEK NUCLEAR GENERATING STATION	Hope Creek Nuclear Generating Station PLANT RESPONSE TO FW CONTROLLER FAILURE (EOC WITH ICF & FWTR)
	Updated FSAR — Figure 15D-1



REVISION 22, MAY 9, 2017

PSEG Nuclear, LLC HOPE CREEK NUCLEAR GENERATING STATION	Hope Creek Nuclear Generating Station PLANT RESPONSE TO LOAD REJECTION W/O BYPASS (EOC WITH ICF)
	Updated FSAR Figure 15D-2



REVISION 22, MAY 9, 2017

PSEG Nuclear, LLC HOPE CREEK NUCLEAR GENERATING STATION	Hope Creek Nuclear Generating Station PLANT RESPONSE TO LOAD REJECTION W/O BYPASS (EOC WITH ICF) Updated FSAR Figure 15D-3

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PSEG NUCLEAR L.L.C. HOPE CREEK GENERATING STATION	
HOPE CREEK UFSAR - REV 14	SHEET 1 OF 1
July 26, 2005	F15D-4

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PSEG NUCLEAR L.L.C.
HOPE CREEK GENERATING STATION

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PSEG NUCLEAR L.L.C. HOPE CREEK GENERATING STATION	
HOPE CREEK UFSAR - REV 14	SHEET 1 OF 1
July 26, 2005	F15D-6

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PSEG NUCLEAR L.L.C.
HOPE CREEK GENERATING STATION

HOPE CREEK UFSAR - REV 14 SHEET 1 OF 1

July 26, 2005 F15D-7

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PSEG NUCLEAR L.L.C. HOPE CREEK GENERATING STATION	
HOPE CREEK UFSAR - REV 14	SHEET 1 OF 1
July 26, 2005	F15D-8

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PSEG NUCLEAR L.L.C. HOPE CREEK GENERATING STATION	
HOPE CREEK UFSAR - REV 14	SHEET 1 OF 1
July 26, 2005	F15D-9

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PSEG NUCLEAR L.L.C.
HOPE CREEK GENERATING STATION

HOPE CREEK UFSAR - REV 14	SHEET 1 OF 1
July 26, 2005	F15D-10

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PSEG NUCLEAR L.L.C.
HOPE CREEK GENERATING STATION

HOPE CREEK UFSAR - REV 14 SHEET 1 OF 1
July 26, 2005 F15D-11

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HOPE CREEK UFSAR - REV 14	SHEET 1 OF 1
July 26, 2005	F15D-12

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PSEG NUCLEAR L.L.C.
HOPE CREEK GENERATING STATION

HOPE CREEK UFSAR - REV 14	SHEET 1 OF 1
July 26, 2005	F15D-13

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PSEG NUCLEAR L.L.C.
HOPE CREEK GENERATING STATION

HOPE CREEK UFSAR - REV 14	SHEET 1 OF 1
July 26, 2005	F15D-14

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PSEG NUCLEAR L.L.C.
HOPE CREEK GENERATING STATION

HOPE CREEK UFSAR - REV 14	SHEET 1 OF 1
July 26, 2005	F15D-15

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PSEG NUCLEAR L.L.C.
HOPE CREEK GENERATING STATION

HOPE CREEK UFSAR - REV 14 SHEET 1 OF 1

July 26, 2005 F15D-16

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July 26, 2005	F15D-17

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July 26, 2005	F15D-18

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HOPE CREEK UFSAR - REV 14	SHEET 1 OF 1
July 26, 2005	F15D-19