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September 12, 2002
2130-02-20236

U.S. Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, DC 20555-0001

Subject: Response To Request For Additional Information – Technical Specification Change Request No. 308, General Electric Stability Analysis Methodology (TAC NO. MB4960)

Oyster Creek Generating Station (Oyster Creek)
Facility Operating License No. DPR-16
NRC Docket No. 50-219

This letter provides additional information (Enclosures 1, 2, & 3) in response to NRC requests for additional information (RAI) dated August 20, 2002, and June 17, 2002; and NRC February 12, 1998 Safety Evaluation Report (NEDC-32339P, Supplement 2, Revision 1) plant-specific actions, respectively; regarding Oyster Creek Technical Specification Request No. 308, submitted to NRC for review on April 26, 2002.

If any additional information is needed, please contact David J. Distel (610) 765-5517.

I declare under penalty of perjury that the foregoing is true and correct.

Very truly yours,

09-12-02
Executed On

Michael P. Gallagher
Director, Licensing & Regulatory Affairs
Mid-Atlantic Regional Operating Group

A001

Enclosures:
1) Response to Request for Additional Information - NRC RAI dated August 20, 2002
2) Response to Request for Additional Information - NRC RAI dated June 17, 2002
3) Response to NRC SER Plant-Specific Actions
4) GE Report MDE-63-0386, "Extended Load Line Limit Analysis for Oyster Creek Nuclear Generating Station," March 1986
5) Summary of Regulatory Commitments

cc: H. J. Miller, USNRC Administrator, Region I
P. S. Tam, USNRC Senior Project Manager, Oyster Creek
R. J. Summers, USNRC Senior Resident Inspector, Oyster Creek
File No. 02033

ENCLOSURE 1

OYSTER CREEK

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

DATED AUGUST 20, 2002

TECHNICAL SPECIFICATION CHANGE REQUEST No. 308

GENERAL ELECTRIC STABILITY ANALYSIS METHODOLOGY

1. **NRC Question**

Provide the results and conclusions of the analysis for operation of Oyster Creek in the extended operating region (reference General Electric (GE) Report MDE-63-0386, "Extended Load Line Limit Analysis for Oyster Creek Nuclear Generating Station," dated March 1986). Also, provide the extended load line limit analysis (ELLLA) region in Attachments 1 and 2, and identify the region which is currently restricted from full use of the ELLLA region of the power/flow map because of the flow biased rod block set points.

Response

A copy of the original Oyster Creek ELLLA report is provided as Enclosure 4. This report documents the basis for the acceptability of operation in the ELLLA domain. ELLLA is a standard operating flexibility option described in the General Electric Standard Application for Reactor Fuel (GESTAR II: NEDO-24011-P-A-US). Operation in the ELLLA domain was implemented at Oyster Creek during operating Cycle 11. The Oyster Creek ELLLA report documents the results of analyses of the impact of ELLLA operation on both cycle independent events (e.g., LOCA) and cycle-dependent events (e.g., transients). Subsequent cycle specific reload analyses have confirmed the continued acceptability of operation in the ELLLA domain.

Updated versions of Attachments 1 and 2 are enclosed. The ELLLA operating domain is indicated by the shaded areas. As indicated in Attachment 1, the currently unavailable ELLLA area represents approximately 1/3 of the analyzed area. This area is unavailable due to the impact of setpoint methodology on the Rod Block Nominal Trip Setpoint and general instrument process noise. More importantly, only 1/2 of the analyzed 'flow window" of 85% flow to 100% flow at full power is available. This constraint has a significant negative impact on plant operations and fuel cycle efficiency. The proposed change will eliminate this constraint by providing margin between the Rod Block Technical Specification Setpoint and the analyzed ELLLA region.

It should be noted that the power/flow maps illustrated in Attachments 1 and 2 are engineering figures and do not represent the operational power/flow maps contained in plant procedures. The operational power/flow maps do not illustrate the natural circulation line. The APRM Scram and Rod Block setpoint lines, the stability exclusion region and an administrative 'buffer region' are all extended down to zero flow on the operational maps. Operation in the low flow region of the power/flow map is excluded by Technical Specification requirements and plant operating procedures. Oyster Creek is not licensed to operate with less than three (3) recirculation loops in operation. This ensures that operation above the natural circulation line is maintained.

2. **NRC Question**

Provide justification for Oyster Creek's transition to the Generic Boiling Water Reactor Owners Group (BWROG) Long Term Stability Solution Option II, and identify the differences between the current method and the new one recommended by GE for Cycle 19 operation. Also, identify any differences between the BWROG Long Term Stability Solution Option II and NEDC-33065P, and hardware modifications due to this proposed change.

Response

The current Oyster Creek Long Term Stability Solution, originally developed and licensed by GPU Nuclear, is based on early analyses performed by the BWROG prior to the approval of the Generic BWROG Long Term Stability Solutions. These early analyses considered only 8 X 8 fuel designs. Oyster Creek is transitioning to GE11 fuel for Cycle 19. GE11 fuel is a 9 X 9 design. Therefore, the current Oyster Creek Long Term Stability Solution is not applicable to Cycle 19 operation due to the presence of GE11 fuel. The Generic BWROG Long Term Stability Solution Option II is a previously approved solution which bounds 9 X 9 fuel designs and is currently being utilized at a US BWR similar to Oyster Creek. The Generic BWROG Long Term Stability Solution Option II is more conservative compared to the current Oyster Creek Long Term Stability Solution. This is illustrated by the more restrictive Scram setpoints in the low flow range of the proposed operating map (Attachment 2) as compared to the current operating map (Attachment 1). Thus, the Generic BWROG Long Term Stability Solution Option II is an appropriate and acceptable replacement for the current Oyster Creek Long Term Stability Solution.

There are no significant differences between the Option II stability solution implemented by Oyster Creek as described in NEDC-33065P, including hardware modifications, and the Generic BWROG Long Term Stability Solution Option II documented in NEDO-32465-A (methodology) and NEDC-32339P (hardware).

3. **NRC Question**

Describe any impact on the BWROG Long Term Stability Solution Option II due to the 10 CFR Part 21 issue on the Generic DIVOM curve for regional mode oscillations, since Option II method uses quadrant-based average power range monitor scram to detect and suppress both in-phase and out-of-phase oscillations.

Response

The 10 CFR Part 21 notification on the BWROG Long Term Stability Solution Generic DIVOM curve for regional mode oscillations identified that the Generic DIVOM curve may not be bounding for all plants. An evaluation performed for Oyster Creek (NEDC-33065P, Rev. 0), based on the 10 CFR Part 21 'Figure-of-Merit' guidance, determined that the Generic DIVOM curve was bounding for Oyster Creek, which is a low power density plant. Therefore, the BWROG Long Term Stability Solution Option II continues to be an acceptable solution for Oyster Creek. If future evaluations/analyses performed by

the BWROG determine that the Generic DIVOM curve, or the BWROG Long Term Stability Solution Option II, is no longer acceptable, Oyster Creek will follow the recommendations of the BWROG revised (1994) stability Interim Corrective Actions (ICAs) until an alternate Long Term Stability Solution is approved.

4. **NRC Question**

Provide quantitative results from review of the Updated Final Safety Analysis Report (UFSAR) transient events such as rod withdrawal event, loss of feedwater heating, main steam isolation valve closure, and startup of an inactive loop at an incorrect temperature and show that the old UFSAR results are still valid.

Response

Qualitative and quantitative results (as applicable) of the UFSAR transient evaluations are presented in Enclosure 1 (pg. 11) of the OC TSCR #308 submittal (TAC MB4960). These results demonstrate that the impact of revised APRM Rod Block and Scram set points is acceptable. The following additional information is provided for each event.

A. Rod Withdrawal Error (RWE)

The current RWE event analysis assumes an APRM Rod Block Setpoint of 110%. With this setpoint, the error rod is typically blocked at a withdrawal position of 10.5ft - 11.0ft (nearly fully withdrawn). The Cycle 18 analysis result for the RWE event at Oyster Creek, using the current approved reload analysis methods, is a ΔCPR of 0.22. Additional analyses were performed utilizing the approved methods and assuming the proposed APRM Rod Block Setpoint of 115%. The results of these analyses indicate that the error rod becomes fully withdrawn prior to reaching the Rod Block Setpoint. However, the maximum impact of the rod withdrawal error on fuel thermal limits (MCPR, LHGR) occurs at an intermediate withdrawal position. The additional rod withdrawal associated with the higher Rod Block Setpoint does not increase the consequences of the RWE event. Furthermore, the RWE event has been explicitly evaluated as part of the Cycle 19 reload licensing analyses utilizing the revised Rod Block Setpoint. This ensures that Cycle 19 operation will continue to meet all applicable design and licensing criteria.

B. Loss of Feedwater Heating (LOFH)

The current LOFH analysis basis conservatively assumes that reactor power increases to 116.7% power based on the current APRM Scram Setpoint of 115%, plus some additional margin. The Cycle 18 analysis result for the LOFH event at Oyster Creek, using the current approved reload analysis methods, is a ΔCPR of 0.14. Additional analyses were performed utilizing the approved methods to support the proposed APRM Scram Setpoint of 120%. These analyses were performed to calculate the actual reactor power increase associated with a 100°F LOFH event. This analysis demonstrated that the actual maximum reactor power for this event is approximately 111% of rated power. Thus, the existing analysis is

based on a conservative power level and remains bounding for the proposed Scram Setpoint. Furthermore, the LOFH event has been explicitly evaluated as part of the Cycle 19 reload licensing analyses. This ensures that Cycle 19 operation will continue to meet all applicable design and licensing criteria.

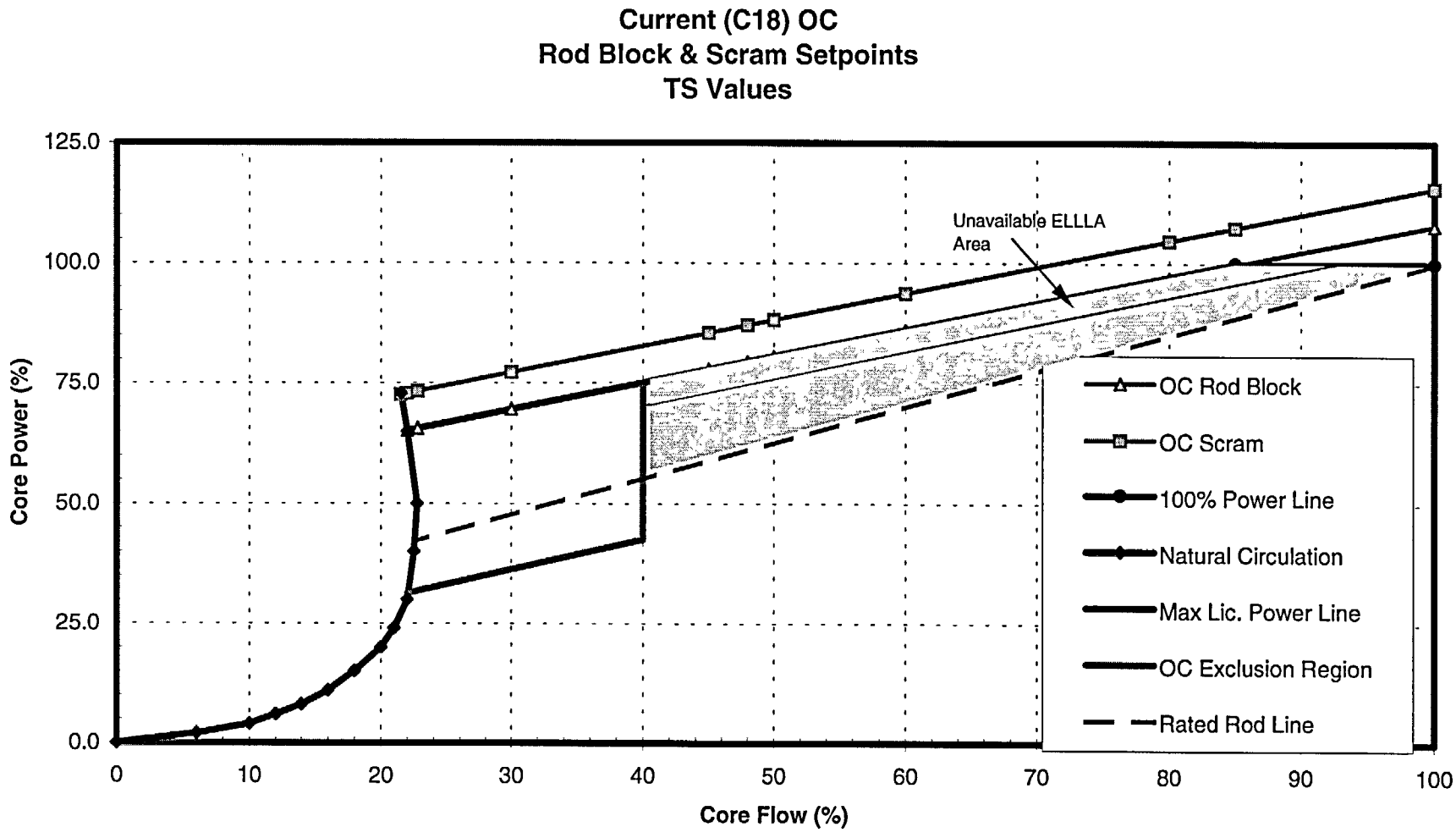
C. MSIV Closure (MSIVC)

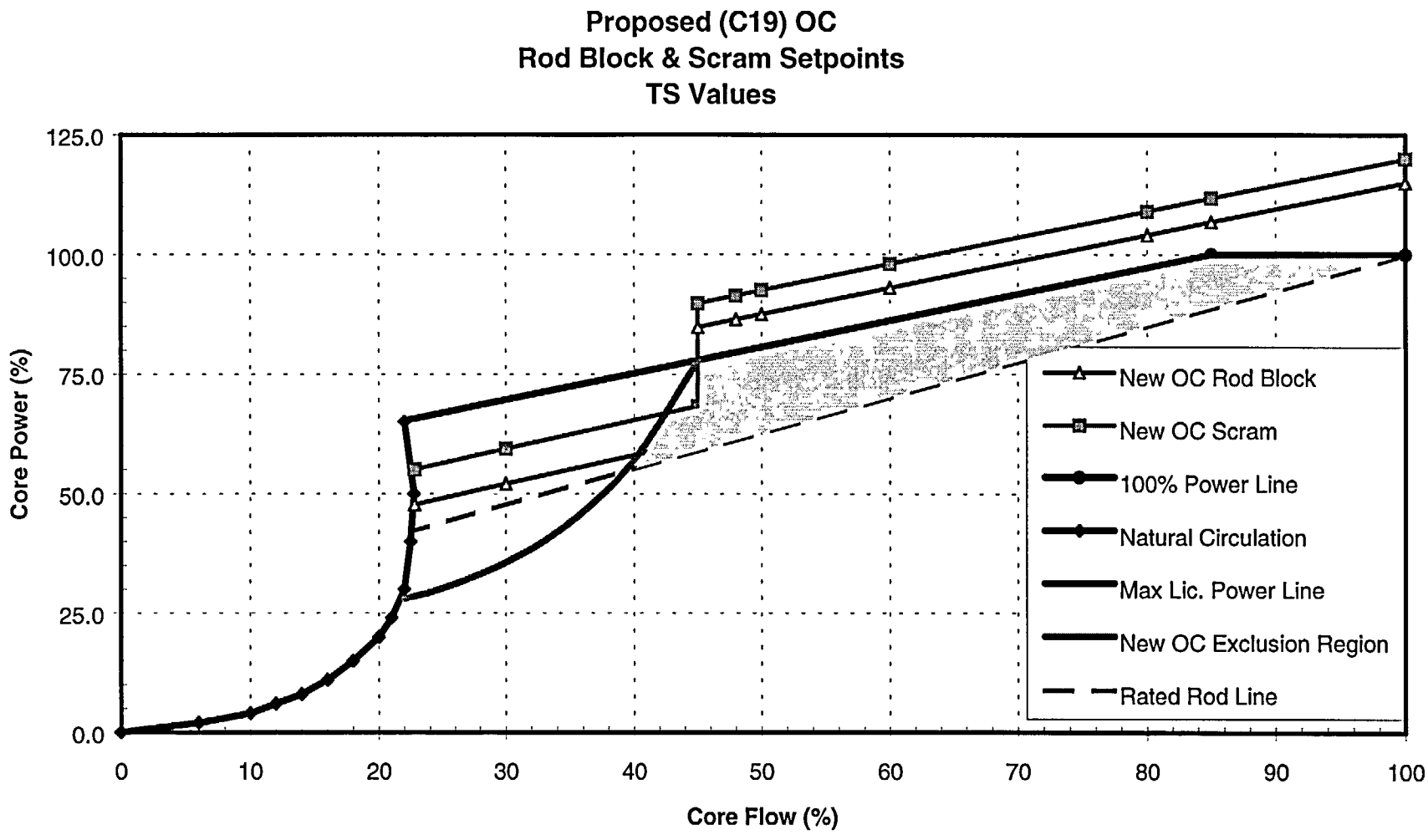
The current MSIVC analysis basis conservatively assumes an APRM Scram Setpoint of 120.0%, which is greater than the actual Scram Setpoint. This setpoint is consistent with the proposed Scram Setpoint, but does not include any additional margin. The Cycle 18 analysis result for the MSIVC event at Oyster Creek, using the current approved reload analysis methods, is a peak reactor vessel pressure of 1333 psig, which is significantly less than the acceptance criterion of 1375 psig. A qualitative evaluation, based on past experience evaluating the sensitivity of the MSICV event results to scram initiation time, indicates that a higher APRM Flux Scram setpoint (i.e., 122%) would delay the calculated scram time by a small amount. This small delay would result in a small increase in the calculated peak reactor pressure, but well within the available pressure margin in the analysis. Thus, the existing analysis remains bounding for the proposed APRM Scram Setpoint. Furthermore, the MSIVC event has been explicitly evaluated as part of the Cycle 19 reload licensing analyses utilizing the revised Scram Setpoint, with additional margin. This ensures that Cycle 19 operation will continue to meet all applicable design and licensing criteria.

D. Startup of an Inactive Loop

An explicit quantitative evaluation of the Startup of an Inactive Loop event is presented in Enclosure 1 (pg. 11) of the OC TSCR #308 submittal (TAC MB4960). The results of this evaluation demonstrate that while the consequences of the Startup of an Inactive Loop event increase (i.e., ΔCPR increases from a value of 0.09 to a value of 0.12), this event remains significantly bounded by the limiting transient events (i.e., turbine trip without bypass ΔCPR = 0.27). As a non-limiting event, this event was not evaluated as part of the Cycle 19 reload licensing analyses.

2130-02-20236
Attachment 1




Proposed (C19) OC
Rod Block & Scram Setpoints
TS Values
125.0
100.0
75.0
50.0
25.0
0.0
Core Power (%)
0
10
20
30
40
50
60
70
80
90
100
Core Flow (%)
New OC Rod Block
New OC Scram
100% Power Line
Natural Circulation
Max Lic. Power Line
New OC Exclusion Region
Rated Rod Line

ENCLOSURE 2

OYSTER CREEK

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

DATED JUNE 17, 2002

TECHNICAL SPECIFICATION CHANGE REQUEST No. 308

GENERAL ELECTRIC STABILITY ANALYSIS METHODOLOGY

1. **NRC Question**

Please identify the manufacturer, model of the digital flow control trip reference (FCTR) cards, and the vendor documentation that is available for the cards.

Response

The requested information is provided below:

Manufacturer: General Electric
Model No.: 148C7112G002
Option: II
Vendor Documents: GEK-109842, dated July 2002, 'Option II – Flow Control Trip Reference (FCTR) Card 148C7112G002

2. **NRC Question**

Enclosure 1 of your application states that "New Digital FCTR cards are being installed under 10 CFR 50.59 ....." Please provide the 10 CFR 50.59 screening report and evaluation for the FCTR cards.

Response

Oyster Creek 10 CFR 50.59 Evaluation No. OC-2002-E-0003, entitled, "ECR OC 01-01193 Increased Core Flow Implementation," has been completed to support plant modification and installation of the new FCTR cards for end-of-cycle 18 operation with the existing Oyster Creek stability solution. The digital FCTR cards implement setpoints in compliance with existing Technical Specifications until the proposed Technical Specification amendment is approved and incorporated for Cycle 19 operation. This evaluation is available onsite for review.

These new FCTR cards will be utilized to support plant operation with the GE BWROG Stability Option II Solution as described in Technical Specification Change Request No. 308.

3. **NRC Question**

Please identify the process used to ensure that procured digital FCTR cards under 10 CFR 50.59 are similar in design, specification and functionality as those for the product line previously approved by the NRC staff. If modifications have been made by the vendor or AmerGen, please provide the details of the changes.

Response

Oyster Creek is an Option II plant, as is Nine Mile Point Unit 1. The FCTR cards installed at Oyster Creek (Model No. 148C7112G002) are identical to those installed at Nine Mile Point Unit 1 (NMP-1) (Model No. 148C7112G001), with the following minor plant-specific differences:

a. The output range of the APRMs at Oyster Creek in percent reactor power is 0 to 150, whereas the range used at NMP-1 is 0 to 125.

b. The EPROMs used at Oyster Creek are different than those installed at NMP-1 only in that Oyster Creek has different setpoints. However, with regard to the methodology used in establishing design requirements and verification, the FCTR cards are the same. Oyster Creek has reviewed the methodology used for NMP-1 design and confirmed that it is the same methodology as used for Oyster Creek. General Electric uses baseline generic methods for programming in the EPROMS. These are generic programs used for many applications. The difference in each application is modified and verified by General Electric as part of their software management program specified verification/validation and baseline reviews.

Differences between the enhanced E1A cards and the Option II cards are addressed in GE NEDC-32696P, which is available onsite for NRC review.

With respect to the design process, General Electric provided performance specifications and validation test requirements and procedures to NMP-1. These documents were revised to address the Oyster Creek specific application requirements. The design process used by General Electric for the Oyster Creek FCTR cards is the same as that used for the NMP-1 application. The only modification in the software utilized for Oyster Creek from the software utilized at NMP-1 is the programmed setpoints. All other algorithms remain unchanged.

The functionality of the FCTR cards is assured as follows:

In addition to the tests and verification performed by General Electric as stated above, Oyster Creek will perform post-modification testing to ensure that the FCTR cards operate properly. The new setpoint curves for Oyster Creek consist of three linear segments, plus one step-change. An adequate number of points will be tested on each segment to ensure proper functionality in these regions. In addition, the functional testing will include the points of transition for the step-change.

This testing along with the factory tests performed by General Electric is adequate to demonstrate the functionality of the FCTR cards.

4. **NRC Question**

Please provide the environmental "qualification parameters" (mentioned in Enclosure 1, page 8) that the FCTR cards are being designed to.

Response

As noted in the Oyster Creek Technical Specification Change Request submittal, dated April 26, 2002, the APRM analog trip biased units are being replaced during the current operating Cycle 18 with new digital FCTR cards under 10 CFR 50.59. As part of the modification process, it was confirmed that the card's environmental qualification parameters enveloped plant-specific environmental conditions regarding temperature, humidity, pressure, seismic, and electromagnetic compatibility (Reference General Electric eDRF# 000-0004-1116, Option II Qualification Summary Report, dated July 13, 2002).

The detailed Oyster Creek environmental qualification requirements for the FCTR cards are described in the response to Action #2 in Enclosure 3.

5. **NRC Question**

Please confirm: (1) for your plant-specific location, electromagnetic interference (EMI) and radio frequency interference (RFI) susceptibility qualification of the new digital FCTR cards was determined based on guidance of Electric Power Research Institute Report TR-102323, and (2) the new digital hardware will not become a source of conducted and/or radiated EMI and RFI for either safety-related circuits.

Response

The FCTR card is qualified for electromagnetic compatibility by type testing and analysis. The electrostatic discharge test requirements are defined by IEC 801-2. Actual tests were performed by General Electric (GE Test Report EFR A00-02506-5). Per GE NEDC-32696P, the EMI testing performed eliminates the need for the licensee to perform in-plant electromagnetic environment surveys in accordance with EPRI guidelines.

Radiated Susceptibility:

Per Section 9.2.1 of GE Performance Specification 25A593NW, which is part of GEK-109842,when used as part of the APRM, the FCTR card is not susceptible to electromagnetic disturbance from neighboring cards.

Radiated Emissions:

Per Section 9.2.2 of GE Performance Specification 25A593NW, which is part of GEK-109842, when used as part of the APRM, the FCTR card does not cause electromagnetic disturbances in neighboring cards.

The new Oyster Creek cards are mounted within the existing cabinets of the neutron monitoring system (NMS) and used as part of the existing APRMs. The tests performed for the NMP-1 installation are applicable to Oyster Creek.

6. **NRC Question**

Please provide any reports or evaluation of the "burn-in process" mentioned in Enclosure 1, page 8, for the digital FCTR cards.

Response

The new FCTR cards were burned in for a period of 72 hours. The General Electric contracted test report (GE Purchase Order No. 52802027460) for the "burn-in" of the new FCTR cards installed at Oyster Creek is available onsite for review. The report is General Electric proprietary.

As a result of the burn-in process, two diodes failed to meet performance requirements and were replaced. The response to Question No. 8 provides additional details.

7. **NRC Question**

Please provide any reports or evaluations for the "shakedown period" that occurred at the end of cycle 18 for the digital FCTR cards.

Response

As discussed above, the digital FCTR cards are operational for the remainder of Cycle 18 and implement setpoints in compliance with existing Technical Specifications. If any design changes are identified as a result of the Cycle 18 "shakedown period" an evaluation or report of such changes will be provided for information to NRC.

8. **NRC Question**

Please report any failures or abnormal operation of the FCTR cards during the burn-in or shakedown process. Provide proposed or performed plant-specific changes that have or are planned to occur as a result of the burn-in and shakedown process.

Response

Prior to the actual burn-in, General Electric had to replace the U21 component on Card 8. Component U21 is used for voltage reference. Immediately following burn-in, General Electric had to replace diode VR3 on Cards 5 & 8 as they failed to meet performance specifications. This component is for filtered drive flow out, which is a non-critical function, and is not used in the Oyster Creek application. Neither of these failures required any modifications to be performed to the plant or to the FCTR cards.

Plant-specific changes, if needed, as a result of the shakedown period will be identified to NRC as described in response to Question 7 above.

9. **NRC Question**

If your in-house setpoint calculation methodology was previously approved by the NRC, please cite the approval document. If your methodology was not previously reviewed and approved by the NRC, please confirm that it is based on NRC-approved Industry standards, and meets the 95/95 confidence level requirement.

Response

Response to this question was provided in AmerGen letter to the NRC dated July 11, 2002 (2130-02-20187).

ENCLOSURE 3

OYSTER CREEK

RESPONSE TO NEDC-32339-A, SUPPLEMENT 2, REVISION 1 NRC SAFETY EVALUATION REPORT PLANT-SPECIFIC ACTIONS

TECHNICAL SPECIFICATION CHANGE REQUEST No. 308

GENERAL ELECTRIC STABILITY ANALYSIS METHODOLOGY

1. **Action**

Describe the plant-specific applicability of NEDC-32339P, Supplement 2, including clarifications and reconciled differences between the specific plant design and the topical report design descriptions.

Response

Differences between NEDC-32339P, Supplement 2, and Option II FCTR cards are addressed in NEDC-32696P. Oyster Creek is an Option II plant. The section regarding Period-Based Detection System does not apply to Oyster Creek as Oyster Creek does not have a period based detection system.

Design requirements such as single failure and software verification have been addressed during the engineering and design process. Existing channel independence and isolation is not affected by installing the FCTR cards at Oyster Creek.

2. **Action**

Provide the plant-specific design-basis environmental conditions (temperature, humidity, pressure, seismic, radiological, and electromagnetic compatibility), and confirm these conditions are enveloped by the E1A equipment environmental qualification values.

Response

The new FCTR cards are located in the neutron monitoring system (NMS) control panels located in the Control Room. Anticipated Control Room environments have been reviewed.

Temperature – The FCTR card is qualified to operate between 0°C and 70°C or between 32°F and 158°F. This envelopes the expected temperature ranges in the Control Room.

Humidity – The FCTR card is qualified to operate between 20% RH and 90% RH. This meets or exceeds the qualification of the APRM trip bias units that the FCTR cards are replacing. Based on further discussion with General Electric, the FCTR cards are manufactured with hardware that is similar to the NUMAC line. Per General Electric, the FCTR hardware is not sensitive to humidity. There are no control room design values with regard to relative humidity.

Pressure – The FCTR card is qualified to operate between 13 psia and 16 psia ambient pressure conditions. The Oyster Creek Control Room pressure during normal operation is 14.7 psia, which is enveloped by the FCTR card specification. Control room pressure is not considered as a design parameter for accident conditions.

Radiological – The Control Room is a considered a mild environment. Therefore, radiation exposure is not an issue.

Electromagnetic Compatibility – Per Section 3.3.1 of GE report NEDC-32339P-A Supp 2 Rev. 1 for Option E1A, "the EMI testing performed eliminates the need for utilities to perform in-plant electromagnetic environment surveys in accordance with EPRI guidelines." Also, Section 3.3.1 of GE report NEDC-32696P for Option II, states the same as above. In addition, both documents concluded that the FCTR cards will not become a source of conducted and/or radiated EMI or RFI for other safety related circuits. General Electric has performed testing with regard to electromagnetic compatibility. A summary of this testing is provided in GEK-109842 and described in the response to Question No. 5 in Enclosure 2.

Seismic – The new FCTR cards are enveloped by the Oyster Creek seismic requirements.

3. **Action**

Confirm that administrative controls will be provided for manually bypassing E1A channels or protective functions.

Response

The new FCTR cards perform essentially the same functions as the APRM trip bias units that they are replacing. Administrative controls with regard to bypassing channels manually remain unchanged.

4. **Action**

Confirm that any changes to the plant operator's control panel will receive human factors reviews per plant-specific procedures.

Response

The FCTR cards are not within the scope of Control Room instrumentation covered by NUREG 0737, Supplement 1. However, the application of the new FCTR cards and their impact on plant operations has been reviewed.

The new FCTR cards are being installed into the same slots as the APRM trip bias units that they are replacing. In this regard, there are no front panel changes in the control room.

Removal of the intermediate rod blocks does affect the layout on the front of Panel 4F of the Control Room. Any changes to the control room front panels are subjected to Operations review and approval.

An additional trip unit (Z11) is being removed from the APRM electronic drawers as a result of the elimination of the intermediate rod blocks. This does not require any human factors review.

ENCLOSURE 4

OYSTER CREEK

GE REPORT MDE-63-0386,

"EXTENDED LOAD LINE LIMIT ANALYSIS FOR OYSTER CREEK NUCLEAR GENERATING STATION,"

MARCH 1986

MDE-63-0386
DRF-L12-00725
March 1986

EXTENDED LOAD LINE LIMIT ANALYSIS FOR OYSTER CREEK NUCLEAR GENERATING STATION CYCLE 10

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GENERAL ELECTRIC

IMPORTANT NOTICE REGARDING
CONTENTS OF THIS REPORT

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ABSTRACT

The Extended Load Line Limit Analysis (ELLLA) was performed for Oyster Creek Nuclear Generating Plant to verify safe plant operation in a specified region above the rated load line on the power/flow map. The expansion of the operating region of the power/flow map allows 100% power operation at 85% flow and will assist more readily attaining rated power during ascension operation.

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1. INTRODUCTION

The flexibility of a boiling water reactor (BWR) during power ascension in proceeding from the low-power/low-core-flow condition to the high-power/high-core-flow condition is limited by two factors. First, if the rated load line control rod pattern is maintained as core flow is increased, changing equilibrium xenon concentrations will result in less than rated power at rated core flow. Second, fuel pellet-cladding interaction considerations inhibit withdrawal of control rods at high power levels. The combination of these two factors can result in the inability to attain rated core power directly.

These limitations can be overcome by allowing operation with a rod pattern that requires few adjustments when ascending to full power. This requires an expansion of the operating region of the current power/flow map. The operating envelope is modified to include the extended operating region bounded by the 108% Average Power Range Monitor (APRM) rod block line, the rated power line, and the rated load line, as shown in Figure 1.1. The expansion of the power/flow map allows Oyster Creek to operate at 100% power/85% flow.

The technical analysis contained in this report is referred to as the Extended Load Line Limit Analysis (ELLLA) and the entire shaded area in Figure 1-1 is referred to as the ELLLA Region. The analyses in this report were performed for Oyster Creek Cycle 10. Future reload submittals will incorporate the use of this extended load line in the analysis.

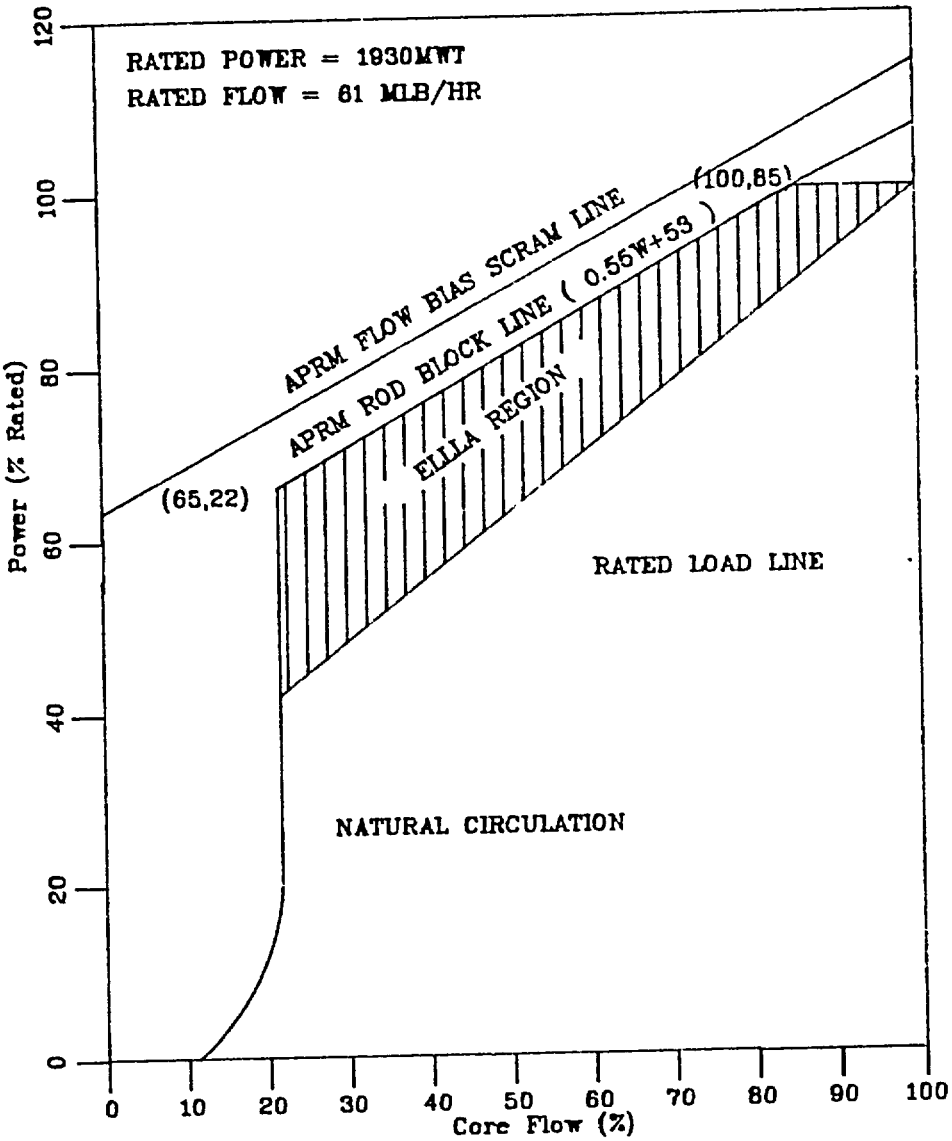

Figure 1-1 Oyster Creek Operating Power/Flow Map

2. SUMMARY

Analyses were performed to justify expansion of the operating region of the power/flow map for Oyster Creek Nuclear Generating Plant, Cycle 10. The analyses provide support to allow plant operation at 100% power and 85% flow. The discussion and analyses presented show that:

-The APRM flow bias scram line is set constant at 115.7% for above rated flow and is set parallel to the rod block line for below rated flow.

-The stability results are within the bounds of the ultimate perfomance criteria (1.0 decay ratio at all attainable conditions).

-The current Maximum Average Planar Linear Heat Generation Rate (MAPLHGR) limits are applicable for operation in the ELLLA region.

-The containment response is unaffected at the limiting operating condition of 102% power and 85% flow.

-The results of the transients at 100% power/85% flow are bounded by the same transients for the licensing basis at 100% power/100% flow except for the Loss-of-Feedwater Heating (LFWH) event. However, because this event is not limiting, the Technical Specification Minimum Critical Power Ratio (MCPR) operating limit for Oyster Creek Cycle 10 is unaffected.

-The peak vessel pressure is within the ASME code limit of 1375 psig for the limiting event (Main Steam Isolation Valve (MSIV) Closure Without Scram) initiated at 100% power and 85% flow.

-The Rod Withdrawal Error (RWE) results for ELLLA conditions are bounded by the Cycle 10 licensing basis results.

Therefore, it is concluded that all safety bases normally applied to Oyster Creek are satisfied throughout Cycle 10 for operation within the ELLLA region.

3. APRM SYSTEM IMPROVEMENTS

The APRM system has both scram and rod block functions. The APRM system serves several purposes.

a. generate trip signals which will automatically scram the reactor during bulk neutron flux level transients before the actual bulk neutron flux level exceeds 115.7% of the rated value,

b. prevent fuel damage from single operator errors or equipment malfunctions, and

c. provide an indication of the bulk thermal power level of the reactor in the power range.

The ELLLA provides justification for extending the operating region of the power/flow map to allow rated power operation at core flows varying between 85% flow to 100% flow. The APRM rod block line is redefined by the ELLLA as 0.55 W + 53, where W is the percent of rated core flow. To maintain the same margin between the APRM rod block line and the APRM flow bias scram line, the APRM flow bias scram line must be modified. This is to assure that events initiated within the ELLLA region will exhibit scram delay times similar to those for the scram event initiated at rated conditions. It is recommended that the APRM flow bias scram line be modified such that:

a. above 100% flow the scram setpoint is maintained constant at 115.7%.

b. below 100% flow the APRM flow bias scram line is set parallel to the APRM rod block line to maintain a power operating margin as pre-ELLLA.

1 The modification in the APRM flow-biased scram line is justified because:

a. the APRM flow-biased scram line has no effect on LOCA analysis results or on pressurized transient (TTNBP, LRNBP, FWCF) results since these events do not depend on a reactor scram on high neutron flux.

b. for slow transients such as LFWH or slow flow run out, these events may be terminated by flux scram. In such cases the same margin between the APRM rod block line and the APRM flow bias scram line is maintained as prior to ELLLA.

c. for off-rated events below the upper power/flow limit, the Technical Specifications require lowering the APRM trip when the design peaking is exceeded. If design peaking is not exceeded, then lower core power will assure increased thermal margin.

4. STABILITY ANALYSIS

The stability analysis for the Oyster Creek Reload 9/Cycle 10 reload licensing analysis (Reference 1) was performed at the intercept of the 100% rod line and the natural circulation line. During operation in the ELLLA region with the new APRM rod block line, the reactor could arrive at a much higher power level at natural circulation condition following a trip of all recirculation pumps than was previously obtained. Therefore, a stability analysis was performed at 65% power/22% flow (the extrapolated APRM rod block line/natural circulation curve intercept point) to evaluate the impact of ELLLA operation on reactor stability.

Both the channel hydrodynamic stability and the reactor core stability were examined utilizing a linearized analytical model (Licensing basis model described in References 1 and 2). The analysis was performed at the most limiting exposure condition during the cycle. The results from both the Cycle 10 reload licensing base case (Reference 1) and the new ELLLA intercept point (65% power/22% flow) case are tabulated in Table 4-1 and shown in Figure 4-1. The results show that the decay ratio of the new ELLLA intercept point is slightly higher than that of the reload licensing case but is still well within the ultimate performance criteria of 1.0 decay ratio.

The reason for the higher decay ratio at the new ELLLA intercept point is that the higher natural circulation to APRM/rod block intercept power level produces a higher core average void fraction. This has two effects on the stability results:

a. the core two phase pressure drop is increased, which tends to increase both the channel hydrodynamic and core decay ratio.

b. the higher core average void fraction increases the negative reactivity void-feedback effect, which tends to increase core decay ratio.

Since the core and channel stability of Oyster Creek in the ELLLA region is well below the ultimate performance criteria 1.0 decay ratio, and Oyster Creek is a BWR/2 there are no special stability- oriented technical specifications required nor is it required to perform cycle specific stability analyses for future reloads in the absence of significant plant or hydraulic fuel type changes (Reference 3).

TABLE 4-1

STABILITY RESULTS

Rod Line Analyzed:	Natural Circulation Condition	
	Extrapolated Rod Block Intercept	100% Rod Line Intercept
Reactor Core Stability Decay Ratio X_2/X_0:	0.67	0.61
Channel Hydrodynamic Performance Decay Ratio X_2/X_0:		
Channel Type		
P8x8R	0.35	0.15
EX8*	0.36	0.15

* EX8 fuel was analyzed per Reference 1.

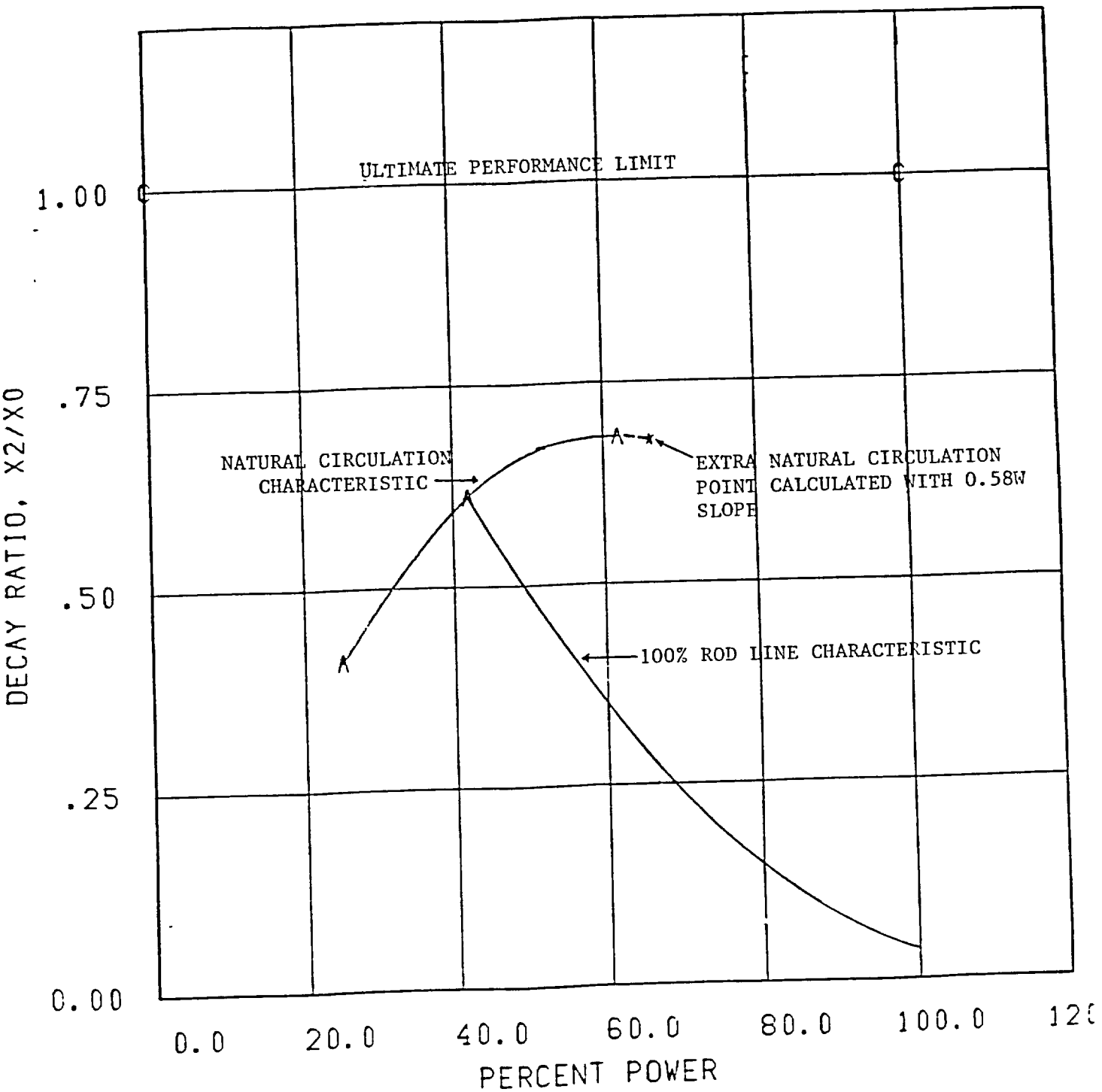

Figure 4-1. Reactor Core Stability Decay Ratio

5. LOSS-OF-COOLANT ACCIDENT (LOCA) ANALYSIS

The extended load line limit option will allow operation at rated power down to 85% core flow. Reload licensing LOCA analyses were performed at 102% power and 100% initial core flow. The potential effects of this reduced core flow on the consequences of a postulated LOCA are:

a. lower initial core flow can affect the coastdown response and may yield an earlier boiling transition time.

b. higher initial voiding in the bundle due to reduced core flow may result in a slightly earlier dryout time.

The effects of a reduced initial core flow on LOCA analyses over the entire LOCA break spectrum are addressed for both 4- and 5-recirculation loop operation to demonstrate conformance with the ECCS acceptance criteria of 10CFR50.46. The LOCA break spectrum is divided into three classes: small, intermediate and large breaks.

5.1 Small Breaks ($A<0.3\ ft^2$)

In this region the vessel depressurizes relatively slow. There will be no significant effect on small break severity due to the lower initial core flow because the peak clad temperature (PCT) is significantly more sensitive to the vessel inventory than to the initial core flow. By the time the break uncovers for the limiting small break, the core inlet flow will have completely coasted down. The length of the time prior to core uncovery also eliminates the effect of higher initial core voiding. Current Maximum Average Planar Linear Heat Generation Rate (MAPLHGR) limits for small break are not affected by the ELLLA option for both 4- and 5-loop operation.

5.2 Large Breaks (>40% DBA)

For the large breaks greater than 40% Design Break Accident (DBA) credit is not taken for coastdown flow due to the rapid decrease in core inlet flow. Therefore, the PCT calculations for large breaks are not affected by the lower initial flow for ELLLA operation. Dryout time occurs slightly earlier because of the higher initial voiding, but the effect is small as shown in Reference 5. The initial voiding in the core is not dependent on the number of the recirculation loops but on the initial core flow; therefore, the results are applicable for both 4- and 5-loop operation.

5.3 Intermediate Breaks

The lower initial core flow can affect the PCT for the intermediate breaks because the onset of boiling transition and the uncovery of the high power axial plane occur somewhat earlier. Calculation of the LOCA using the same conservative LOCA models which were used for the original Oyster Creek analysis (Reference 4) would predict higher PCTs for the intermediate breaks during ELLLA operation at 100% power/85% flow. This could cause the intermediate break to be limiting for certain exposures. However, incorporation of the NRC approved modified Bromley film boiling correlation (Reference 5) into the intermediate break analysis results in PCT of at least 70 degrees below the limiting DBA PCT. Therefore, the small break will still remain limiting at low exposures and the large (DBA) break limiting at higher exposures. Note that PCT constraints set MAPLHGR limits for the small break and maximum oxidation fraction constraints set MAPLHGR limits for the large break (Reference 1).

5.4 LOCA Summary

Based on the above discussion, the current MAPLHGR values (Reference 1) are applicable for ELLLA region for both 4- and 5-loop operation.

6. CONTAINMENT

The impact of plant operation in the ELLLA region on the containment LOCA response was evaluated. The ELLLA region for Oyster Creek allows plant operation at 100% power and 85% flow. Reducing core flow (while maintaining rated power) lowers the enthalpy (or alternately increase the subcooling) in the lower plenum and recirculation loops. In the event of a recirculation line break, this increase in subcooling can result in a slightly higher break mass flow rate, and subsequently, the potential for a higher drywell pressurization rate. The containment LOCA response of Reference 8 was re-analyzed at 102% power and 85% flow. The analysis used the NRC approved model and assumptions consistent with the Mark I Containment Load Definition Report (Reference 6). The plant conditions at the initiation of the pipe break are summarized in Table 6-1.

The containment temperature and pressure responses are presented in Figures 6-1 to 6-4 for both the standard case (Reference 7) and ELLLA case, respectively. Immediately following the pipe break, the drywell pressure increases and the water initially in the submerged portion of the vent system accelerates into the suppression pool. Until the water is completely cleared from the vent system, the air pressure in the wetwell does not change significantly. The vent system is cleared in 0.295 seconds after the initial pipe break for the ELLLA case as compared to 0.30 seconds for the standard case. This minor difference is due to a slightly larger break mass flow rate at the 102% power/85% flow condition. As the vents clear, both drywell and wetwell pressures increase. The blowdown continues into the drywell and flow continues from the drywell through the vent system and into the wetwell. Point A (of Figure 6-1) is characterized by the depletion of the fluid inventory initially occupying the recirculation loop. This point is reached in 0.60 seconds for the ELLLA case and 0.65 seconds for the standard case.

The drywell pressure increases more rapidly than the wetwell pressure until the mass flow rate into the drywell drops sharply. At approximately point B (Figure 6-1), the liquid level inside the reactor has dropped below the elevation of the pipe break and the blowdown (flow from the reactor), which was previously all liquid, becomes a steam-liquid mixture. The flow through the vent system now begins to decrease and results in a corresponding decrease in the vent system pressure. This permits the drywell and wetwell pressures to converge until they differ only by the hydrostatic pressure determined by the vent system submergence. This description of the timing of events applied to both temperature and pressure transients in the drywell and wetwell.

Due to reduced core flow, the peak drywell pressure and temperature at the ELLLA operating condition (102% power/85% flow) are slightly higher than those at standard condition (102% power/100% flow) as shown in Table 6-1. The maximum drywell pressurization rate at the ELLLA condition is 52.1 psi/sec as compared to 51.6 psi/sec at normal condition (Reference 7). However, the maximum drywell pressurization rate is still well below the limit of 58.2 psi/sec defined by plant-unique testing for defining LOCA related pool swell loads (Reference 8). The results indicate no impact of plant operation in the ELLLA region on containment response.

TABLE 6-1

PLANT CONDITIONS USED IN CONTAINMENT ANALYSIS

	Standard Case	ELLLA Case
Reactor Power (MWt)	1971	1971
Core Flow (Mlb/hr)	61.0	58.5
Initial Suppression Pool Temperature (°F)	77.5	77.5
Downcomer Submergence (ft)	3.53	3.53
Airspace Volume (ft^3)		
Drywell	180,000	180,000
Wetwell	126,400	126,400
Airspace Pressure (psig)		
Drywell	0.0	0.0
Wetwell	0.0	0.0
Pressurization Rate (psi/sec)	51.6	52.1
Vent Clearing Time	0.3	0.295
Pipe Inventory Depletion Time (sec)	0.65	0.6
Peak Drywell Pressure (psig)	48.3	51.4
Peak Drywell Temperature (°F)	294.4	297.3

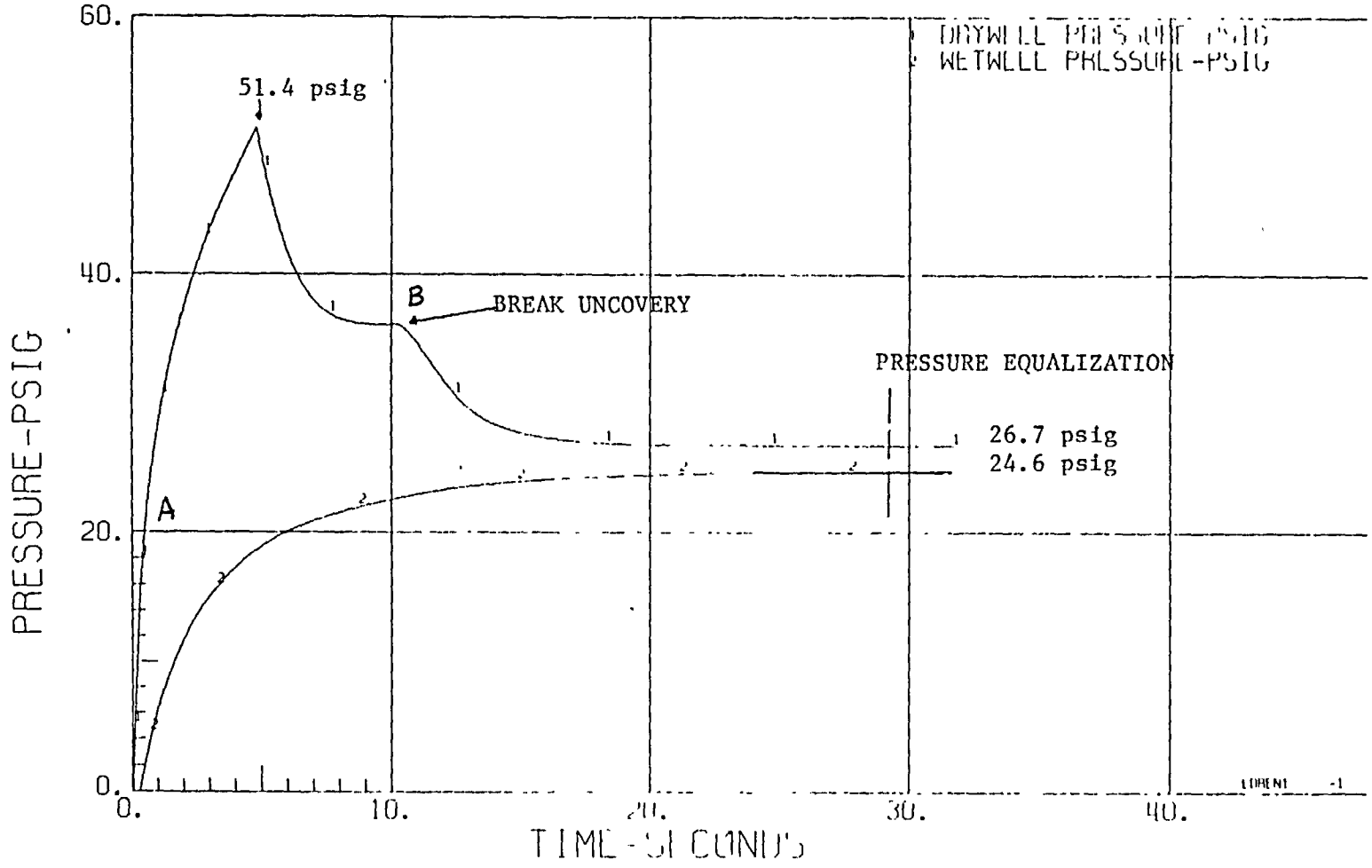

Figure 6-1. Drywell and Wetwell Pressure Response - 102% Power/85% Flow

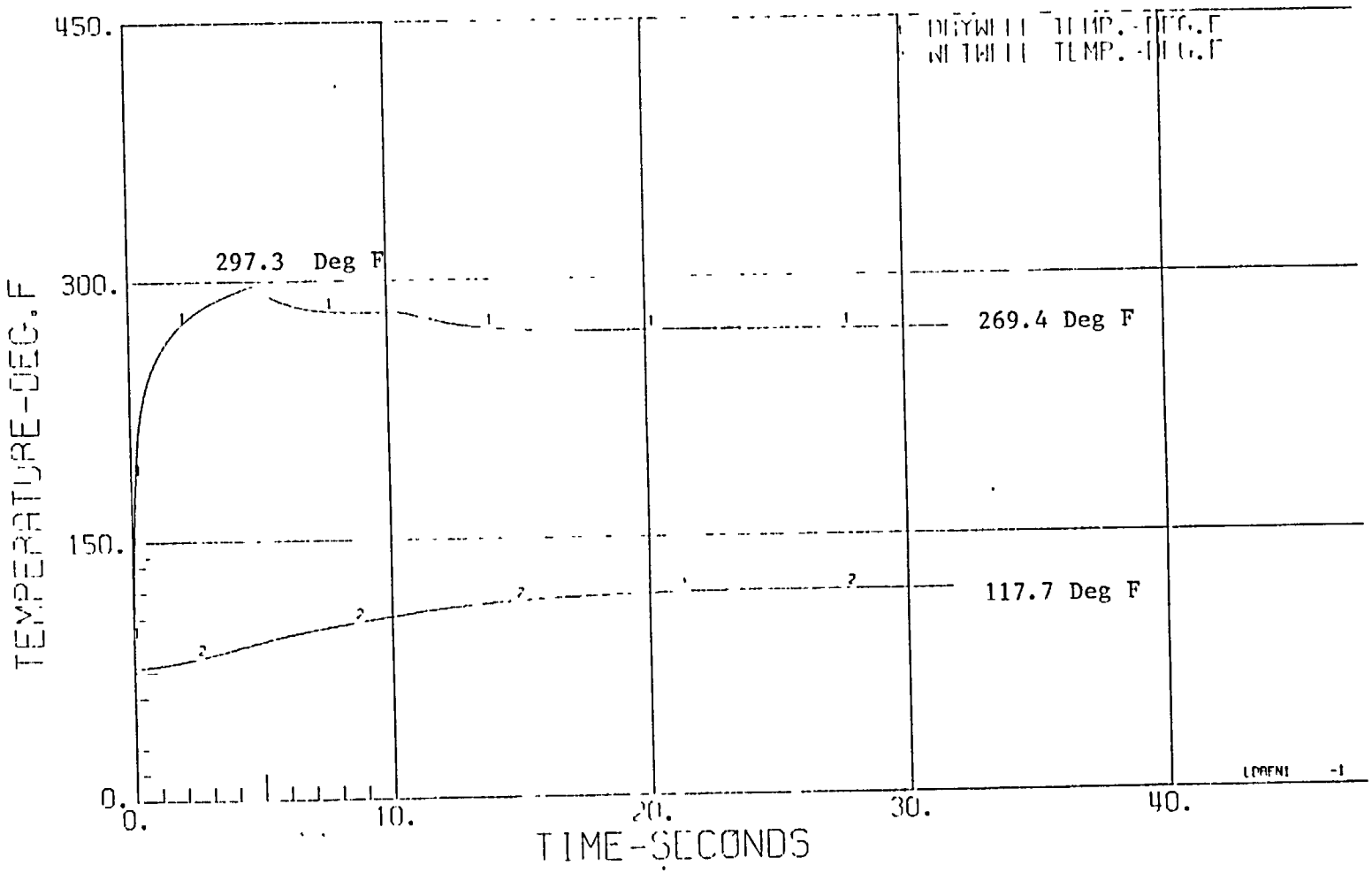

Figure 6-2. Drywell and Wetwell Temperature Response - 102% Power/85% Flow

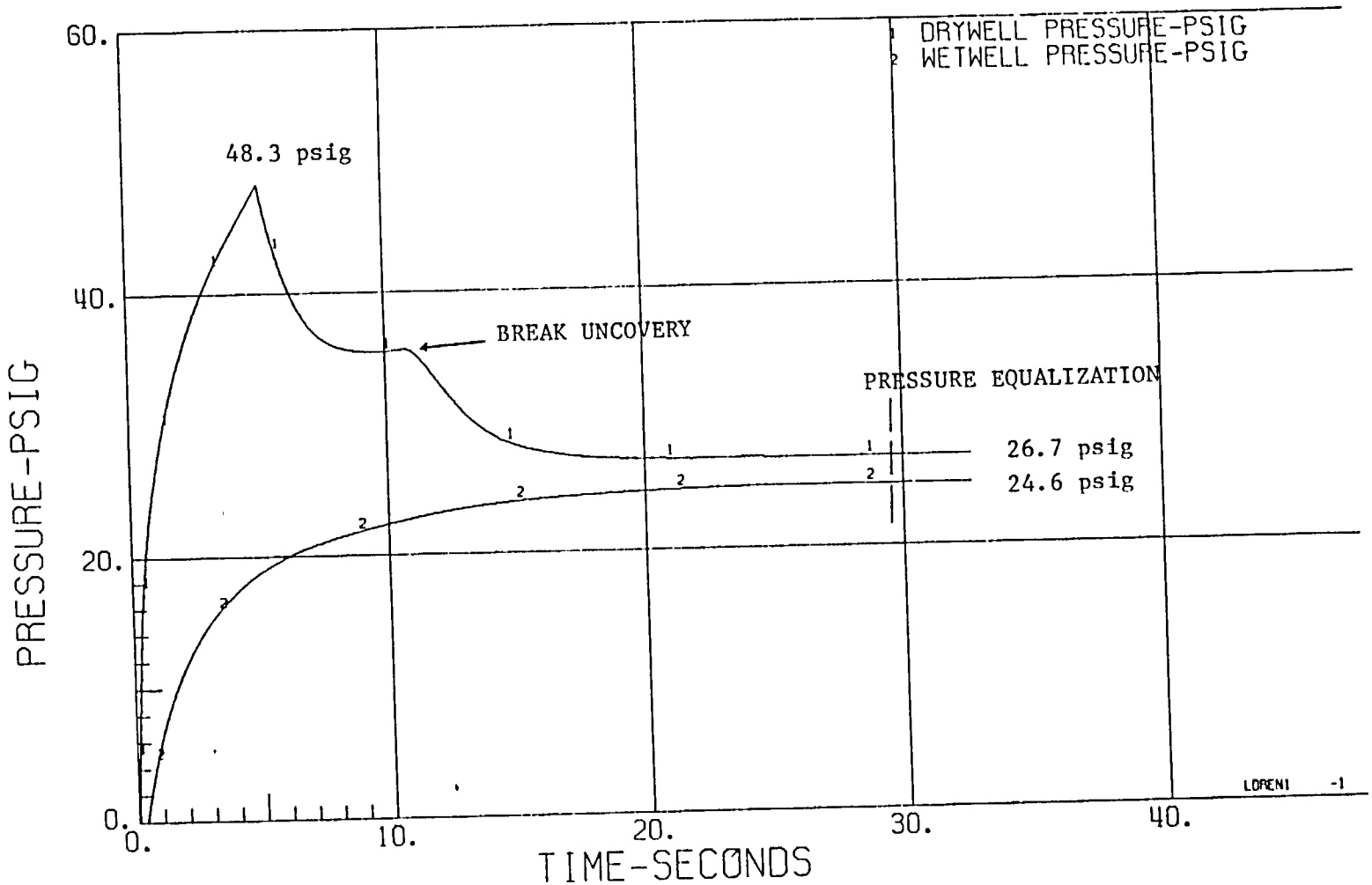

Figure 6-3. Drywell and Wetwell Pressure Response - 102% Power/100% Flow

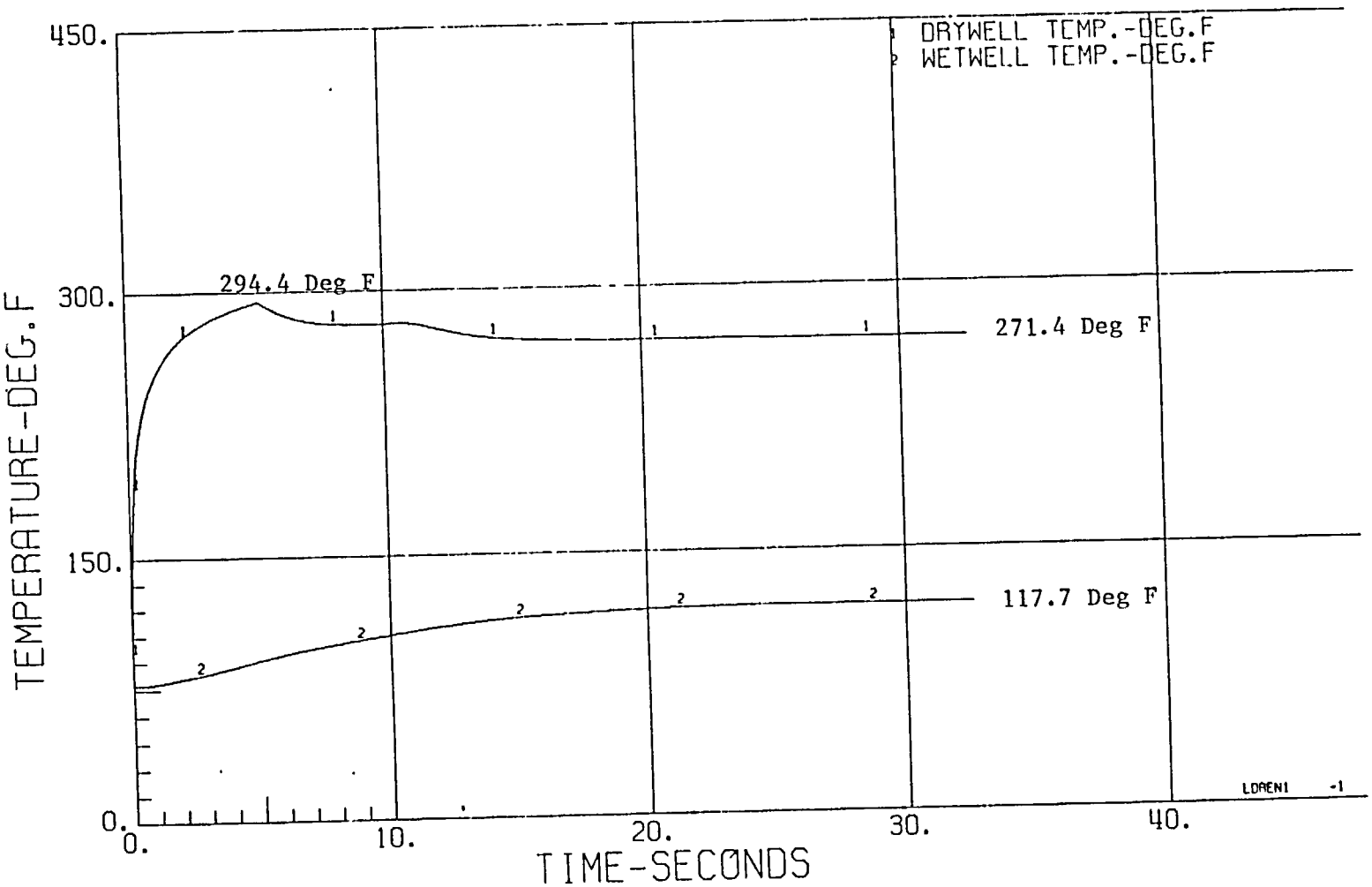

Figure 6-4. Drywell and Wetwell Temperature Response - 102% Power/100% Flow

7. TRANSIENTS

The transient events from the reload licensing analysis (Reference 4) were reanalyzed under ELLLA conditions. The goal is to show that the reload licensing results remain bounding for expected transient performances in the extended operating domain.

As shown in Reference 1, the most limiting transient event for Oyster Creek Cycle 10 is the Turbine Trip without Bypass event. The following transient events were reanalyzed at the ELLLA 100% power intercept point (100%P/85%F): Turbine Trip without Bypass (TTNBP), Feedwater Control Failure (FWCF), and Loss of Feedwater Heater (LFWH). These events were analyzed using the nuclear parameters resulting from the End of Cycle (EOC) target exposure shape.

Core-wide rapid pressurization events (TTNBP, FWCF) are analyzed using the system model code ODYN (Reference 9). This system model code is a one-dimensional representation of the reactor core which is coupled to the recirculation and control system model. The integrated model is based on one-dimensional reactor kinetics, multi-noded thermal-hydraulic equipment. ODYN contains a refined reactor core description and a detailed steamline model to simulate pressure dynamics during a transient.

For slower core-wide transients, such as LFWH, the REDY transient model is used. REDY has a point reactor kinetic representation rather than one-dimensional simulation (Reference 10).

The core wide results are then used to evaluate the transient MCPR using the single channel improved SCAT code (Reference 11).

The application of these computer codes for transient analysis and MCPR evaluation in the ELLLA region follows the same procedures used in the reload licensing analysis. The transient inputs for ELLLA are similar to the reload licensing analysis cases with the exception to the thermal-hydraulic data related to the off-rated power/flow condition (see Tables 7-1 and 7-2).

The results from both the licensing basis case and the reduced core flow case (ELLLA) are shown in Figures 7-1 through 7-6 and summarized in Table 7-3. As shown in Table 7-3, the (100P/85F) transient results are bounded by the licensing basis case (100P/100F) except for the Loss of Feedwater Heater event. However, this does not affect the minimum critical power ratio (MCPR) operating limit for Oyster Creek Cycle 10. These transients are discussed in more detail in the following subsection.

7.1 Turbine Trip No Bypass Event

A variety of turbine or nuclear system malfunctions can initiate a turbine trip. For such an event the turbine stop valve closure occurs over a period of 0.10 second. Position switches at the stop valves sense the turbine trip and initiate the reactor scram. The turbine stop valve closes causing a sudden reduction in steam flow which results in a nuclear system pressure increase. The sharp pressurization in the core collapses the voids in the core, resulting in a rapid neutron flux increase. When the pressure rises to the pressure relief setpoint, the safety/relief valves (S/RVs) open discharging steam to the suppression pool. This action limits the reactor pressure rise. The neutron flux reaches a maximum of 396.5% of rated for the ELLLA case as compared to 494.5% rated for the reload licensing basis case. The peak fuel surface heat flux reaches a maximum of 114.2% of its initial value as compared to 118.7% in the reload licensing basis case. The MCPRs of the 100% power/85% flow case are bounded by the 100% power/100% flow case. This is because the reduced core flow results in a higher void content in the reactor and causes the core axial power shape to shift down to a bottom-peaked profile. With the peak power node closer to the vessel bottom than in the licensing basis case, the control rod reactivity is more effective and limits the peak neutron flux and surface heat flux to a lower value.

7.2 Feedwater Controller Failure - Maximum Demand

This event is postulated on a basis of a single failure of the feedwater controller which is forced to its upper limit at the beginning of the event causing an increase in coolant inventory by increasing the feedwater flow. The influx of excess feedwater flow results in an increase core subcooling which reduces the void fraction and thus induces an increase in reactor power. The excess feedwater flow also results in a rise in the reactor water level which eventually leads to high water level trip, subsequently resulting in the main turbine and feedwater turbine trips and opening the turbine bypass valves. Reactor scram trip is actuated from main turbine stop valve position switches. The S/RVs open as steamline pressure reaches the setpoints.

The neutron flux and heat flux for the ELLLA case reach a maximum of 242.7% of rated and 114.1% of its initial value, respectively, which are below those of the licensing base case of 305.1% and 118.4%. The maximum vessel pressure is 1172 psig for 100% power/85% flow as compared to 1179 psig for the licensing basis case. The ΔCPR from the ELLLA is bounded by the licensing basis case as shown in Table 7-3.

As shown in Figure 7-3 and 7-4, the core inlet subcooling is higher at 100% power/85% flow case than in the reload basis case. As a result of higher inlet core subcooling the reactor power is slightly higher during the early stage of the event. However, the reduced core flow causes the axial power shape to shift down to bottom-peaked profile. With the peak power node closer to the vessel bottom than in the licensing basis case, the control rod reactivity is more effective and thus limits the peak neutron flux and surface heat flux to a lower value.

7.3 Loss-of-Feedwater Heater

The Loss of Feedwater Heating (LFWH) event is an abnormal operating transient that results from loss of feedwater heating due to the loss of one or more feedwater heaters. The event causes a decrease in the temperature of feedwater entering the reactor vessel. This results in

an increase in core inlet subcooling which collapsed voids and thus increases core average power and shifts the axial power distribution towards the bottom of the core. Because of this shift, voids begin to build up at the bottom which tends to offset the power increase. The LFWH transient behaves smoothly and slowly relative to other event.

For the LFWH event, the subcooling effect on the void reactivity is more pronounced at lower core flow because of higher subcooling. The core inlet subcooling change at reduced core flow is higher than at rated core flow as seen in Figures 7-5 and 7-6. However, the collapse in void fraction at 85% core flow is slightly higher than in the licensing basis case . Thus, the peak neutron flux, surface heat flux and ΔCPR for the ELLLA show only a small increase. However, this does not affect the MCPR operating limit for Oyster Creek Cycle 10.

7.4 Conclusion

The transient results show that the consequences of the TTNBP and FWCF initiated from within the ELLLA region are bounded by the consequences of the same events initiated from the licensing basis condition for Oyster Creek Cycle 10. The LFWH transient initiated from within the ELLLA region results in slightly higher MCPR than the same event initiated from the licensing basis condition. However, this event is not limiting and thus there is no change to the MCPR operating limit for Cycle 10.

TABLE 7-1

TRANSIENT INPUT DATA AND OPERATING CONDITIONS

	LICENSING BASIS POINT (100P/100F)	100% INTERCEPT PT. (100P/85F)
THERMAL POWER (MWt/%)	1930/100	1930/100
STEAM FLOW (Mlb/HR/%)	7.25/100	7.25/100
CORE FLOW (Mlb/HR/%)	61.0/100	51.85/85
DOME PRESSURE (PSIG)	1021	1020
TURBINE PRESSURE (PSIG)	957	956
RELIEF VALVES (NO./% NBR)	5/38.5	5/38.5
LOW SETPOINT (PSIG)	1070	1070
SPRING SAFETY VALVES (NO./% NBR)	16/144.56	16/144.56
LOW SETPOINT (PSIG)	1224	1224

TABLE 7-2

GETAB ANALYSIS INITIAL CONDITIONS

	LICENSING BASIS POINT (100P/100F)		100% INTERCEPT PT. (100P/85F)	
CORE POWER (MWt)	1930		1930	
CORE FLOW (Mlb/HR)	61.0		51.85	
CORE PRESSURE (PSIG)	1035		1032.5	
INLET ENTHALPY (BTU/lb)	517.5		511.8	
NONFUEL POWER FRACTION	0.035		0.035	
AXIAL PEAKING FACTOR	1.40		1.40	
	P8X8R	EX8*	P8X8	EX8*
LOCAL PEAKING FACTOR	1.20	1.28	1.20	1.28
RADIAL PEAKING FACTOR	1.738	1.650	1.754	1.672
R-FACTOR	1.051	1.098	1.051	1.098
BUNDLE POWER (MWt)	5.839	5.553	5.884	5.620
BUNDLE FLOW (10^3LB/HR)	91.13	90.75	76.48	76.36

* EX8 fuel was analyzed per Reference 1.

TABLE 7-3

PRESSURIZATION TRANSIENT RESULTS

	INITIAL POWER/ FLOW (%NBR)	PEAK NEUTRON FLUX (%NBR)	PEAK HEAT FLUX (%NBR)	PEAK STEAM LINE PRESS (psig)	PEAK VESSEL PRESSURE (psig)	DELTA CPR P8X8R	DELTA CPR EX8*
TURBINE TRIP WITHOUT BYPASS	100/100	494.5	118.70	1281	1296	0.25	0.22
	100/85	396.9	114.20	1270	1285	0.17	0.14
FEEDWATER CONTROLLER FAILURE	100/100	305.1	118.40	1149	1179	0.20	0.18
	100/85	242.7	114.10	1146	1172	0.13	0.11
LOSS OF FEEDWATER HEATER	100/100	114.30	113.50	1019	1052	0.12	0.11
	100/85	115.30	114.60	1019	1050	0.13	0.12

* EX8 fuel was analyzed per Reference 1.

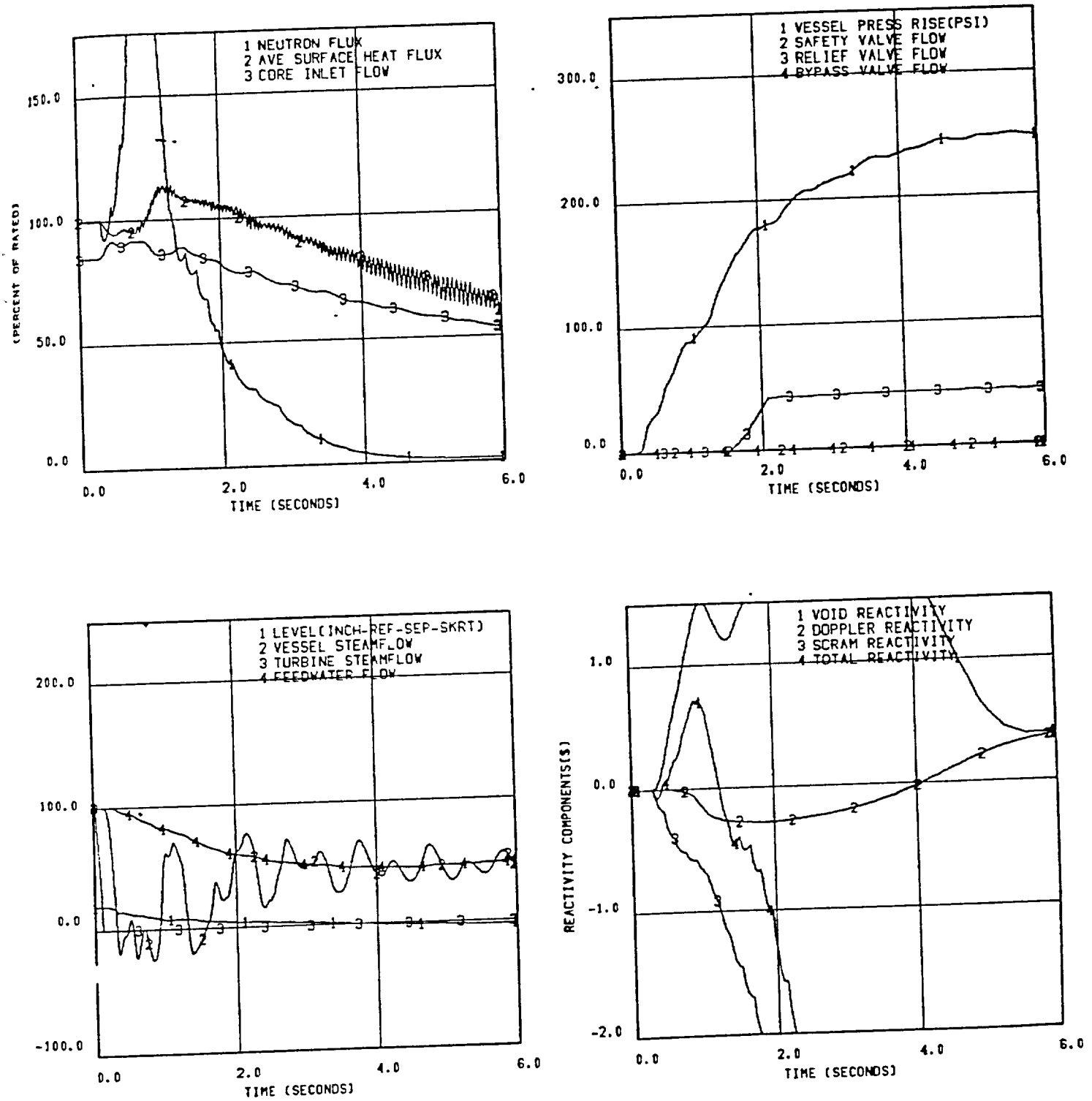

Figure 7-1. Plant Response to Turbine Trip without Bypass- 100% Power/85% Flow

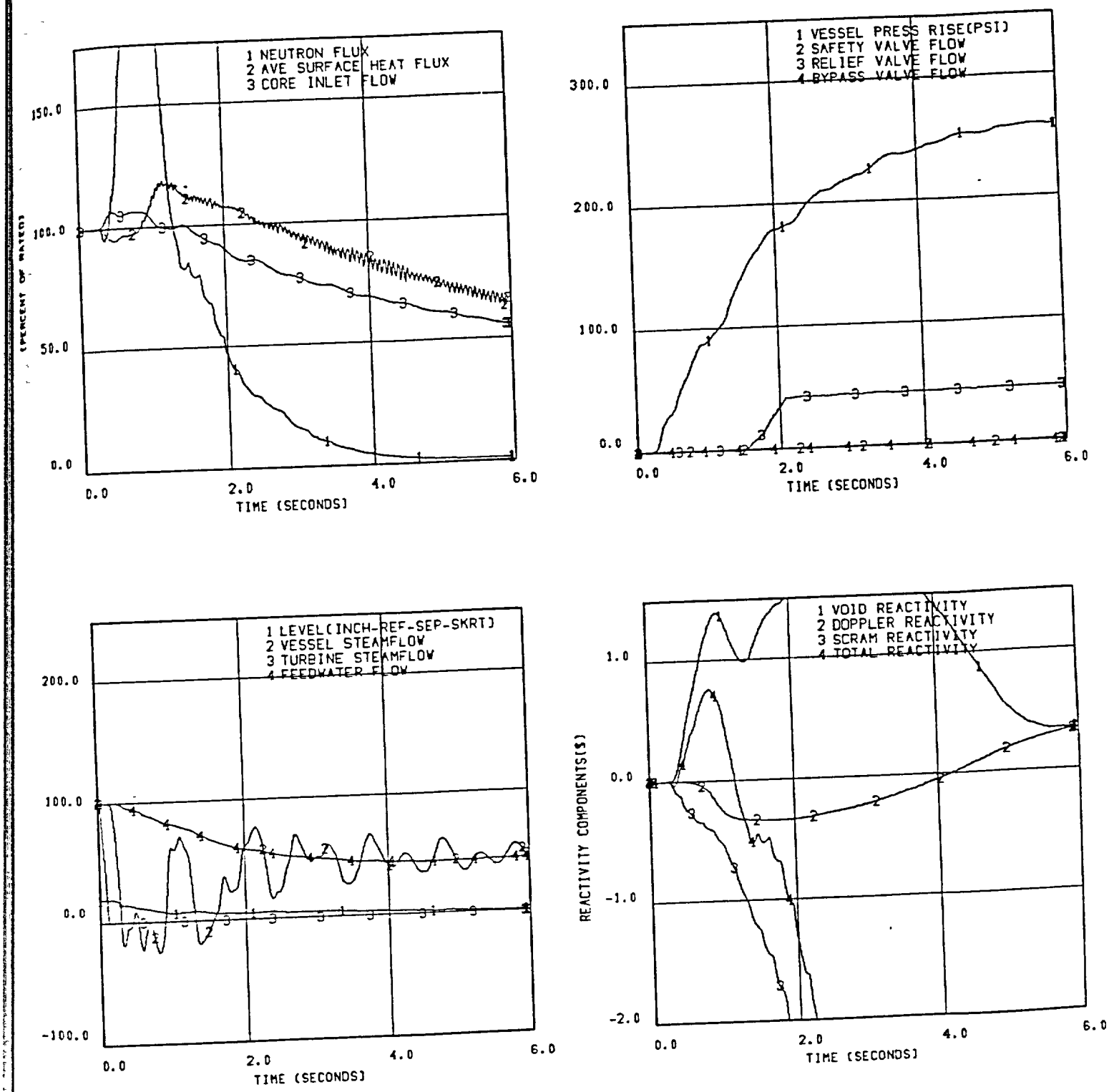

Figure 7-2. Plant Response to Turbine Trip without Bypass– 100% Power/100% Flow

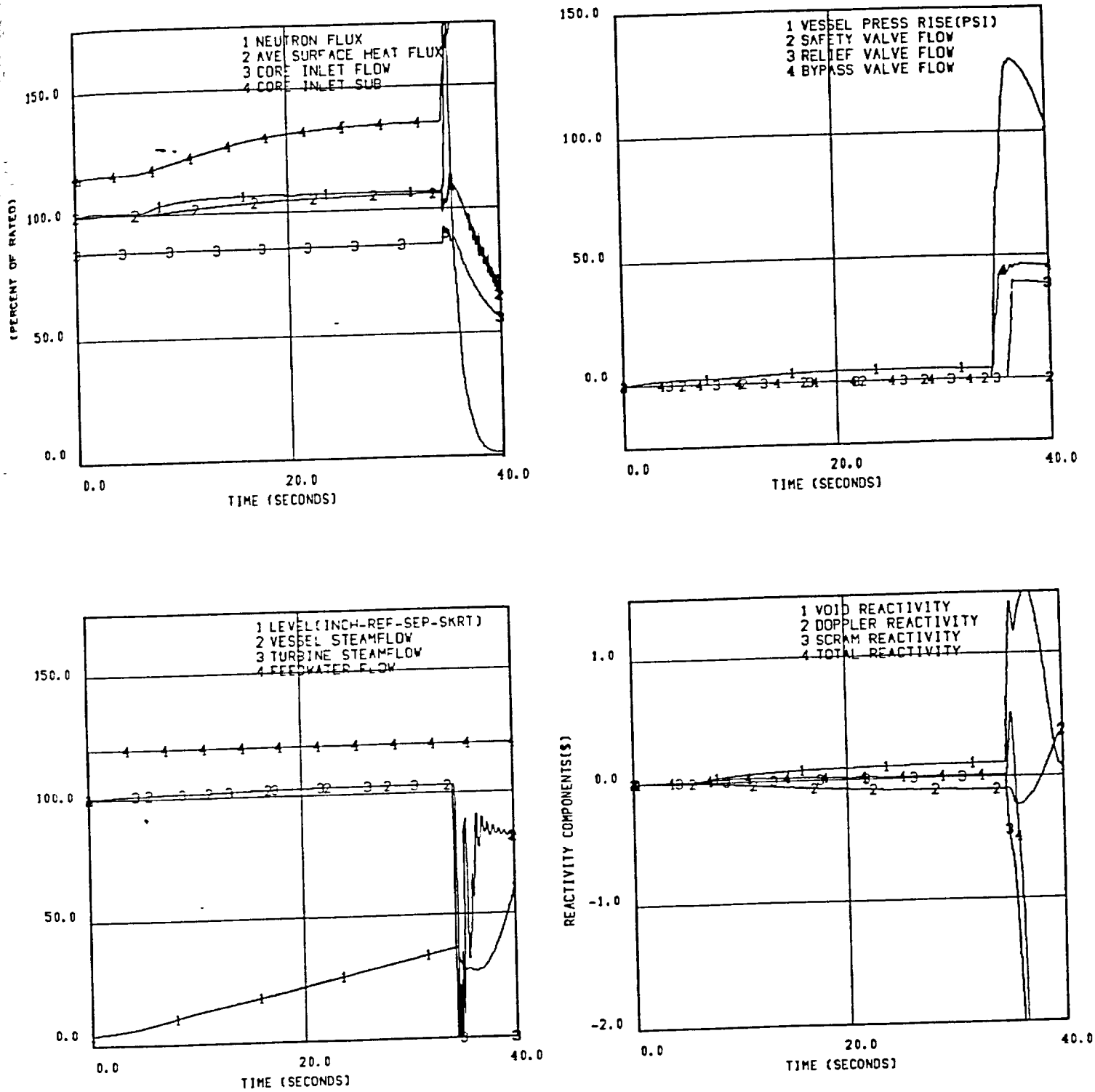

Figure 7-3. Plant Response to Feedwater Controller Failure
Maximum Demand - 100% Power/85% Flow

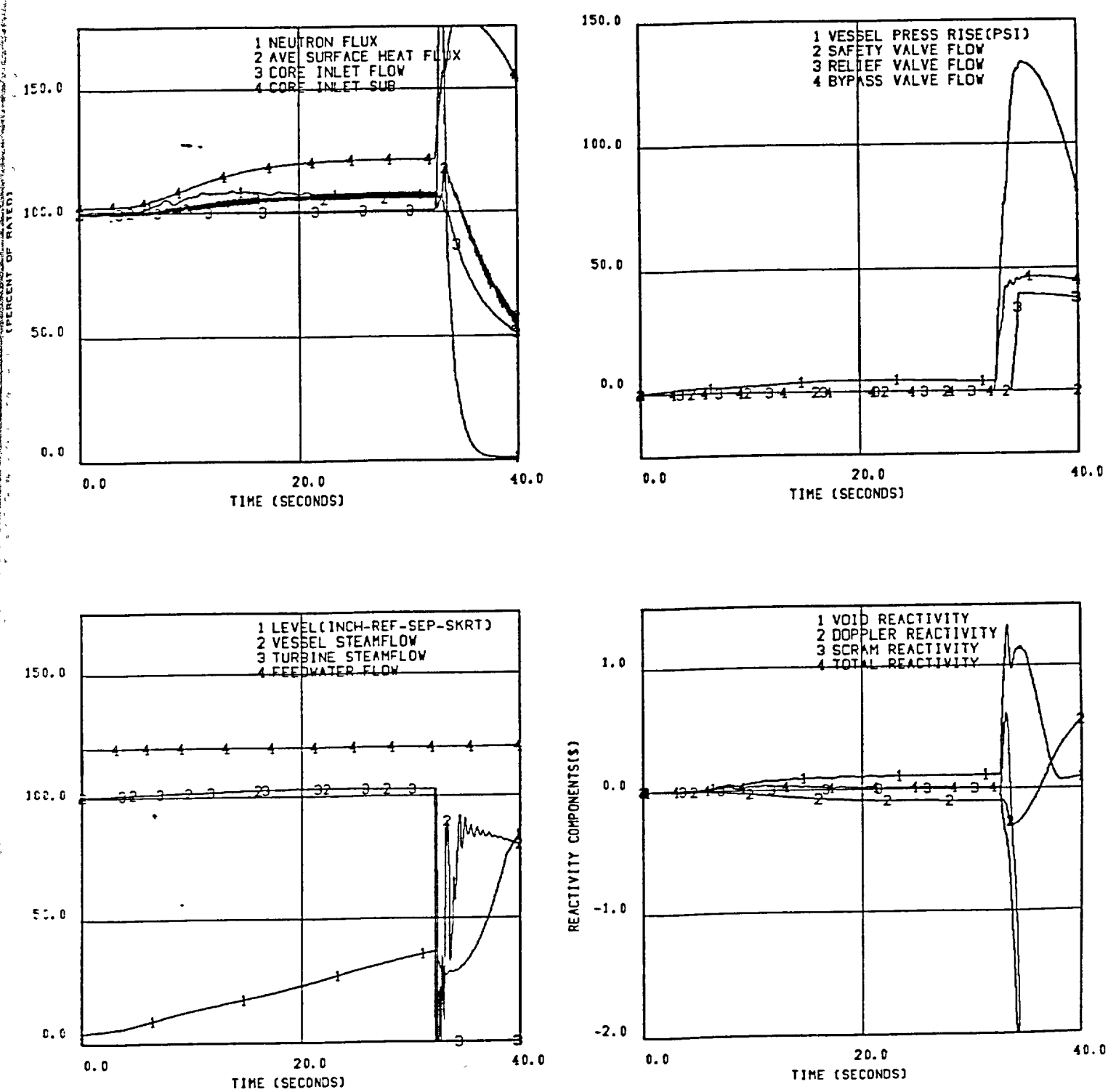

Figure 7-4. Plant Response to Feedwater Controller Failure - Maximum Demand - 100% Power/100% Flow

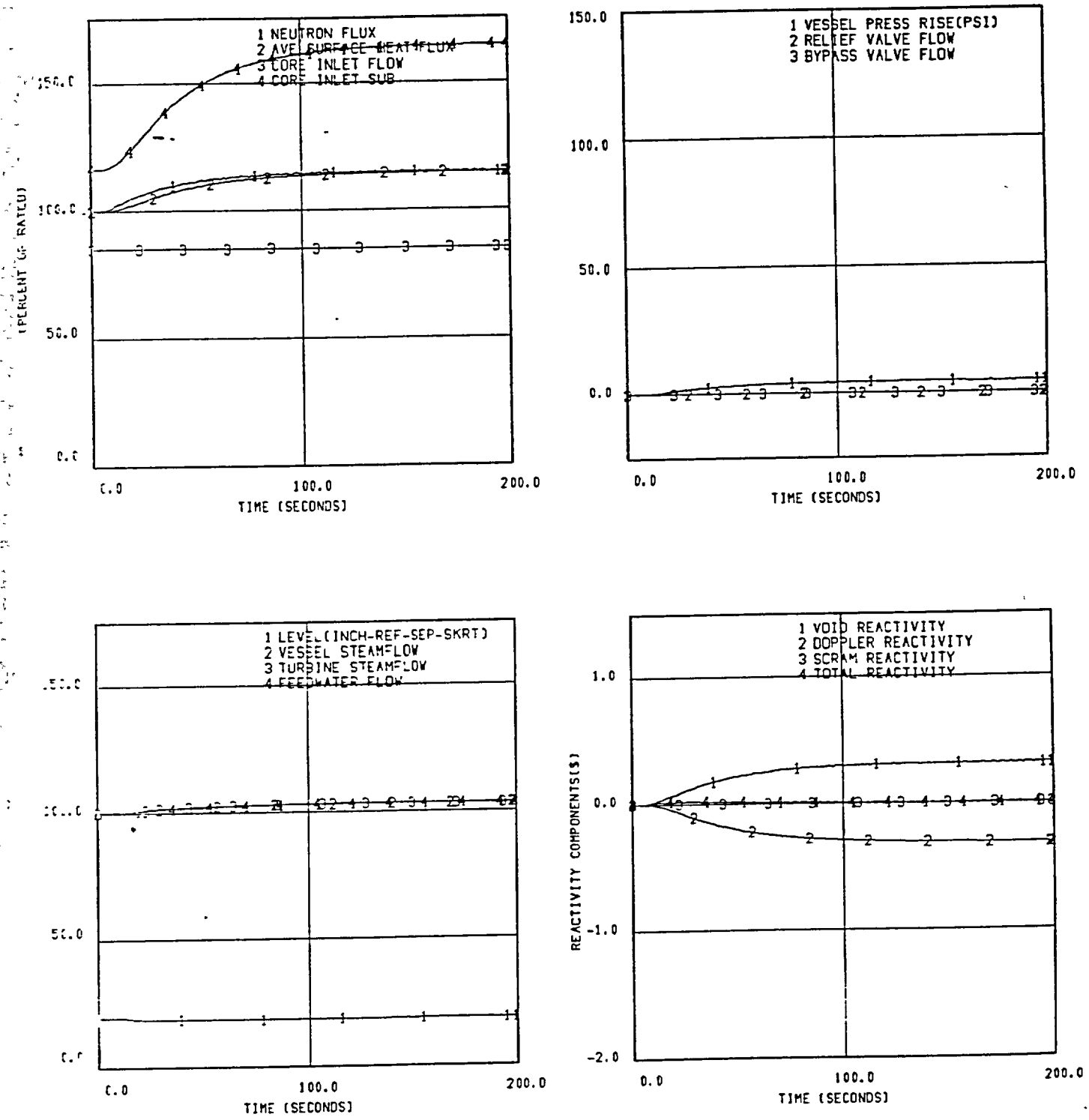

Figure 7-5. Plant Response to Loss of 100 Deg F Feedwater Heating - 100% Power/85% Flow

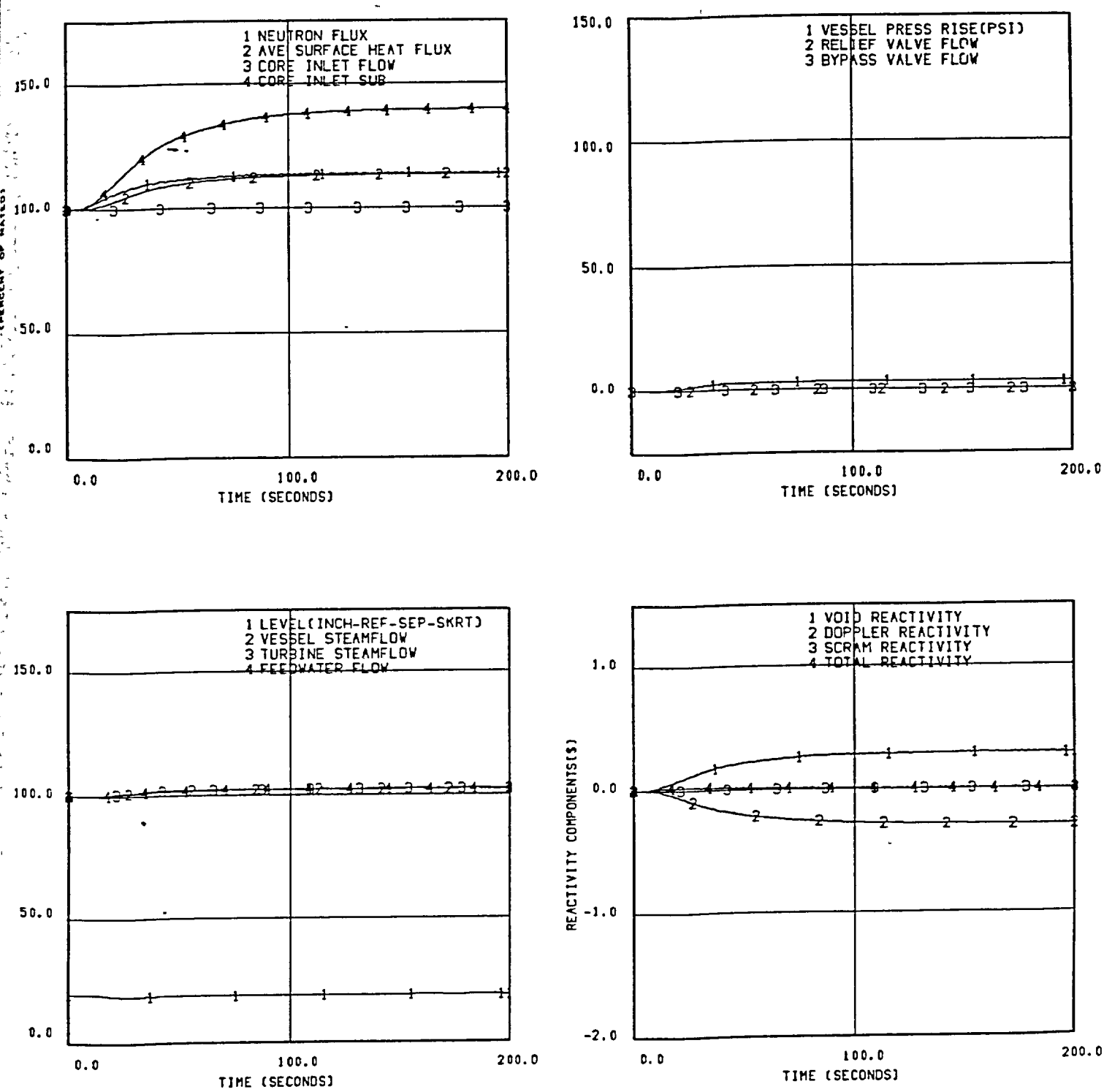

Figure 7-6. Plant Response to Loss of 100 Deg F Feedwater Heating - 100% Power/100% Flow

8. ASME PRESSURE VESSEL CODE COMPLIANCE

The Main Steam Isolation Valve (MSIV) closure with no scram event is used to determine compliance to the American Society of Mechanical Engineers (ASME) pressure vessel code. This event was analyzed at 100% power intercept point (100/85) using the nuclear parameters resulting from the EOC target exposure shape consistent with rated operation. For the case of a reactor isolation event, operation at low core flow results in higher initial void fraction in the reactor than at normal operation. This results in an increase in neutron flux and subsequently higher vessel pressure (Figures 8-1 and 8-2). The net increase in peak vessel pressure is only 1 psi (Table 8-1) and is still well below the design pressure limit of 1375 psig.

The key results of the MSIV closure with no scram event is the peak vessel pressure. The vessel pressure is primarily dependent on system energy volume and relief capacity, while fuel exposure dependence plays a minor role on the vessel pressure. A sensitivity study has shown that the variation of peak vessel pressure is less than 5 psi for different core exposures (i.e., beginning of cycle, mid of cycle, and end of cycle). As shown in the ELLLA evaluation, the net increase in peak vessel pressure is only 1 psi at reduced core flow. Thus, it is concluded that the overpressure at EOC for ELLLA has sufficient margin to show compliance with the ASME pressure vessel code.

TABLE 8-1

ASME PRESSURE VESSEL CODE COMPLIANCE: MSIV CLOSURE (NO SCRAM)

INITIAL POWER/ FLOW (% NBR)	PEAK STEAMLINE PRESS. (PSIG)	PEAK VESSEL PRESS. (PSIG)
100/100	1261	1298
100/85	1263	1299

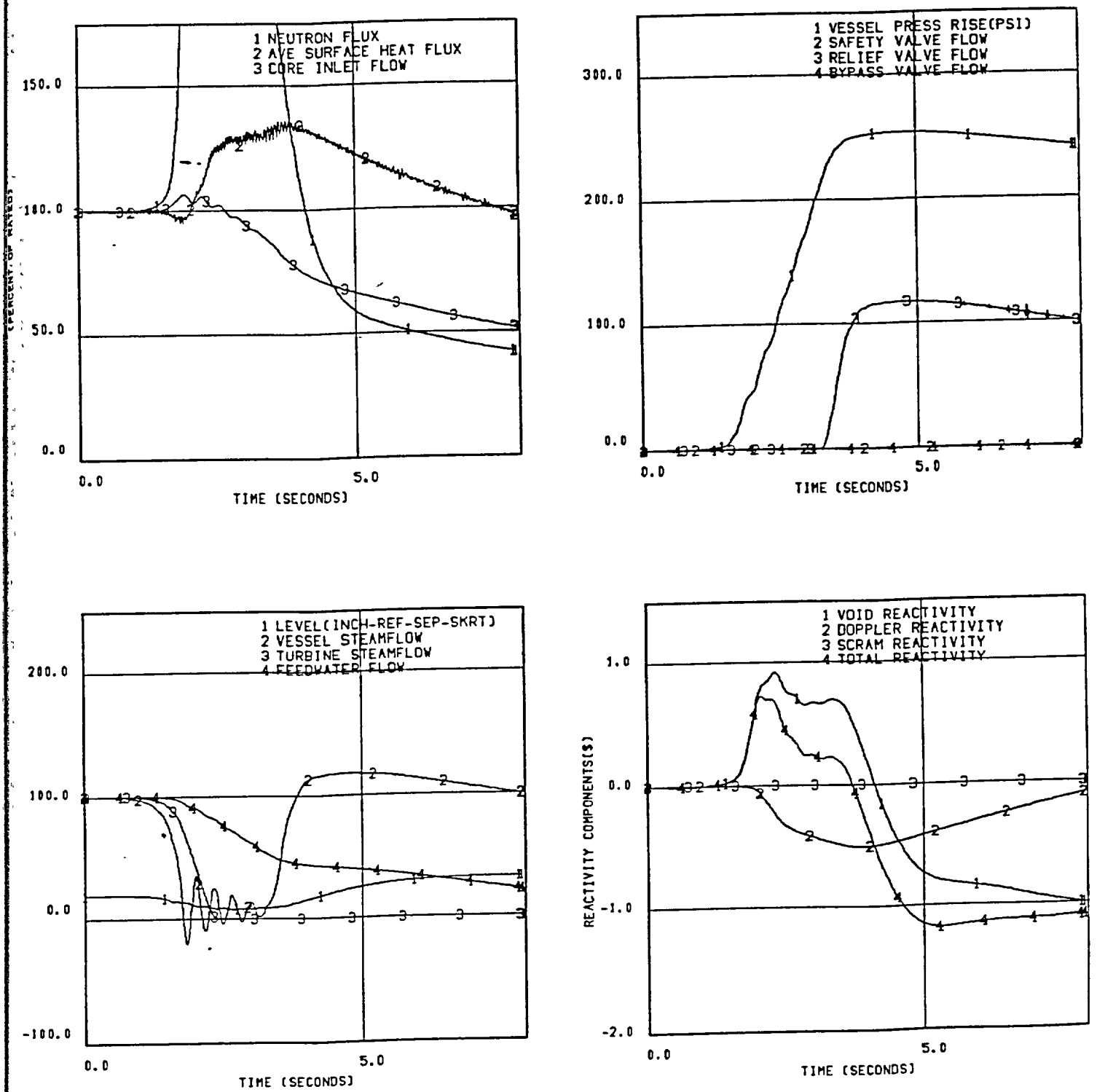

Figure 8-2. Plant Response to Main Steam Isolation Valve Closure with No Scram - 100% Power/100% Flow

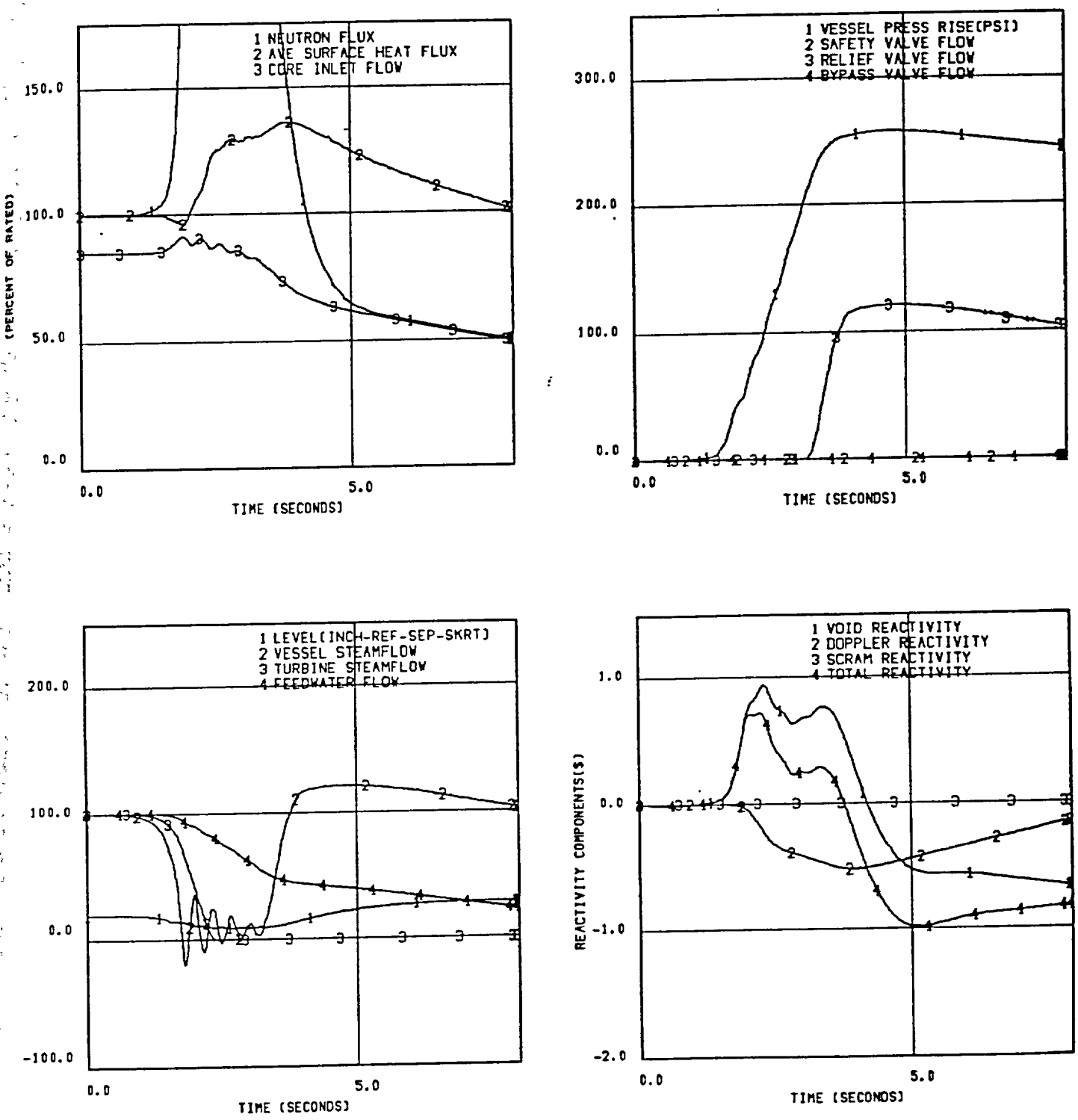

Figure 8-1. Plant Response to Main Steam Isolation Valve Closure with No Scram - 100% Power/85% Flow

9. ROD WITHDRAWAL ERROR

The Oyster Creek Cycle 10 Rod Withdrawal Error (RWE) analysis (Reference 1) calculates the same ΔCPR values for both 106% and 108% setpoints as summarized in Table 9-1 (Reference 1). The 108% APRM setpoint corresponds to the ELLLA condition. The RWE evaluation determines the maximum core reactivity state (i.e., Mid of Cycle) to allow maximum power distribution flexibility. The RWE analysis is thus not performed at 100% power/85% flow condition since this would result in less core reactivity. Operating in the ELLLA region is closer to the APRM rod block, as compared to previously analyzed states, and results in a reduction of rod motion prior to a rod block. This results in a reduction of the MCPR at each rod block setpoint. Therefore, the MCPRs of the 100% power/100% flow state will bound the 100% power/85% flow state.

TABLE 9-1

LOCAL ROD WITHDRAWAL ERROR (WITH LIMITING INSTRUMENT FAILURE) TRANSIENT SUMMARY

Reactor Power (%)	Rod Position (Feet Withdrawn)	ΔCPR		MLHGR (kW/ft)	
		P8x8R	Ex8*	P8x8R	Ex8*
104	6.5	0.23	0.29	15.2	15.6
105	7.5	0.25	0.31	14.7	15.4
106	8.5	0.27	0.33	14.9	15.8
107	9.0	0.27	0.33	15.3	16.4
108	9.0	0.27	0.33	15.3	16.4
109	9.5	0.28	0.34	15.8	16.9
110	10.0	0.29	0.34	16.2	17.5

*EX8 fuel was analyzed per Reference 1.

10. REFERENCES

1. NEDO-24195, "General Electric Reload Fuel Application for Oyster Creek Nuclear Generating Station,". August 1979, E&A No. 7, May 1984.

2. NEDO-21506, "Stability and Dynamic Performance of the General Electric Boiling Water Reactor," General Electric Company Licensing Topical Report, January 1977.

3. 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

4. Letter from R.L. Gridley (GE) to D.G. Eisenhut (NRC), "Review of Low Core Flow Effects on LOCA Analysis for Operating BWRs," May 8, 1984.

5. NEDO-20566-1-A, "Calculation of Low Flow Film Boiling Heat Transfer for BWR LOCA Analysis," October 1982.

6. NEDO-21888, Rev.2, "Mark I Containment Program Load Definition Report," November 1981.

7. NEDO-24572, Rev.2, "Mark I Containment Program Plant-Unique Load Definition Oyster Creek Nuclear Generating Station," July 1982.

8. NEDE-21944P, Volume 1, "Mark I Containment Program Quarter Scale Plant Unique Test," April 1979.

9. NEDO-24154, "Qualification of the One-Dimensional Core Transient Model for BWRs" October 1978.

10. NEDO-10802, "Analytical Methods of Plant Transient Evaluations for the General Electric Boiling Water Reactor," General Electric Company, February 1973.

11. NEDE-20566-P and NEDE-20566, "Analytical Model for Loss-of-Coolant Analyses in Accordance with 10CFR50 Appendix K," General Electric, January 1976.

ENCLOSURE 5

OYSTER CREEK

SUMMARY OF REGULATORY COMMITMENTS

SUMMARY OF REGULATORY COMMITMENTS

The following table summarizes those regulatory commitments established in this document. Any other actions discussed in the submittal represent intended or planned actions by AmerGen. They are described to the NRC for the NRC's information and are not regulatory commitments.

	COMMITMENT	COMMITTED DATE OR "OUTAGE"
1.	Response to NRC Question 7, Enclosure 2: If any design changes are identified as a result of the Cycle 18 "shakedown period" an evaluation or report of such changes will be provided for information to NRC.	1R19
2.	Response to NRC Question 8, Enclosure 2: Identify any plant-specific changes needed as a result of the shakedown period to the NRC.	1R19