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SUBJECT: Forwards application for proposed Amend 39 to License NPF-22, revising Tech Specs to support Cycle 2 reload. Fee paid.

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Bruce D. Kenyon
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JUN 19 1986

Director of Nuclear Reactor Regulation
Attention: Ms. E. Adensam, Project Director
BWR Project Directorate No. 3
Division of BWR Licensing
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

SUSQUEHANNA STEAM ELECTRIC STATION
PROPOSED AMENDMENT 39 TO LICENSE NO. NPF-22
PLA-2661 FILES R41-2, A7-8C Docket No. 50-388

Dear Ms. Adensam:

The purpose of this letter is to propose changes to the Susquehanna SES Unit 2 Technical Specifications in support of the ensuing Cycle 2 reload. Changes to the following Technical Specifications are requested:

| | |
|---|---|
| | Index |
| 1.0 | Definitions |
| 3/4.1.2 | Reactivity Anomalies |
| 3/4.2.1 | Average Planar Linear Heat Generation Rate |
| 3/4.2.2 | APRM Setpoints |
| 3/4.2.3 | Minimum Critical Power Ratio |
| 3/4.2.4 | Linear Heat Generation Rate |
| 3/4.3.4.2 | End-of-Cycle Recirculation Pump Trip System Instrumentation |
| 3/4.4.1.1.2 | Recirculation Loops - Single Loop Operation |
| 3/4.7.8 | Main Turbine Bypass System |
| 5.3.1 | Fuel Assemblies |
| B 2.1 | Safety Limits |
| B 3/4.1.1 | Shutdown Margin |
| B 3/4.1.2 | Reactivity Anomalies |
| B 3/4.1.3 | Control Rods |
| B 3/4.1.4 | Control Rod Program Controls |
| B 3/4.2.1 | Average Planar Linear Heat Generation Rate |
| B 3/4.2.2 | APRM Setpoints |
| B 3/4.2.3 | Minimum Critical Power Ratio |
| B 3/4.2.4 | Linear Heat Generation Rate |
| B 3/4.4.1 | Recirculation System |
| B 3/4.7.8 | Main Turbine Bypass System |

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As discussed in a telecon held with your staff on June 16, 1986, and in the attached reload summary report, this submittal does not contain Minimum Critical Power Ratio (MCPR) Technical Specification Limits. The methodology which will be used to derive these limits is being provided at this time for your review; the actual values will be supplied in mid-July.

The following attachments to this letter are provided to illustrate and technically support each of the changes:

- Marked-up Technical Specification Changes
- No Significant Hazards Considerations
- Susquehanna SES Unit 2 Cycle 2 Reload Summary Report
- XN-NF-86-60, "Susquehanna Unit 2 Cycle 2 Reload Analysis," May, 1986
- XN-NF-86-55, "Susquehanna Unit 2 Cycle 2 Plant Transient Analysis," May, 1986
- XN-NF-86-65, "Susquehanna LOCA-ECCS Analysis MAPLHGR Results for 9X9 fuel," May, 1986
- Susquehanna SES Unit 2 Cycle 2 Proposed Startup Physics Tests Summary Description, May, 1986

Please note that with respect to thermal hydraulic stability of the Exxon Nuclear Company 9X9 fuel being inserted during this reload, PP&L has already submitted (PLA-2637, dated April 30, 1986) a stability test program which will supplement the results presented in the pertinent analyses attached. Certain other supplementary data will also be submitted to the NRC at their request in accordance with our discussions on May 30, 1986. It is not our intent to treat any of this supplementary information as revisions to this proposal.

Also, sufficient analysis has not been completed to support Single Loop Operation (SLO) with the 9X9 fuel design. The Technical Specifications have been altered accordingly, and we will provide a separate submittal on this issue based on appropriate analysis when it is available. Again, this submittal is not to be considered a revision to this proposed amendment.

Susquehanna SES Unit 2 is currently scheduled to be shutdown for refueling and inspection on August 2, 1986 and to restart as early as October 3, 1986.

We request that your approval be conditioned to become effective upon startup after this outage, and will keep you informed of any schedule changes.

Any questions with respect to this proposed amendment should be directed to Mr. R. Sgarro at (215) 770-7855. Pursuant to 10CFR170, the appropriate fee is enclosed.

Very truly yours,

B. D. Kenyon
Senior Vice President-Nuclear

Attachments

cc: M. J. Campagnone - USNRC
R. H. Jacobs - USNRC

T. M. Gerusky
Bureau of Radiation Protection
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XN-NF-86-55
Issue Date: 5/15/86

# SUSQUEHANNA UNIT 2 CYCLE 2 PLANT TRANSIENT ANALYSIS

Prepared by: T.H. Keheley, Team Leader
BWR Safety Analysis

Concur: R.E. Collingham, Manager
BWR Safety Analysis

Concur: J.N. Morgan, Manager
Customer Services Engineering

Concur: G.N. Ward, Manager
Reload Licensing

Approve: H.E. Williamson, Manager
Licensing & Safety Engineering

Approve: G.L. Ritter, Manager
Fuel Engineering & Technical Services

thk/mln

EXXON NUCLEAR COMPANY, INC.

.8606240301

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## 1.0 INTRODUCTION

This report presents the results of Exxon Nuclear Company's (ENC's) evaluation of system transient events for Susquehanna Unit 2 during Cycle 2 operation with a reload of ENC 9x9 BWR fuel. This evaluation together with an evaluation of core transient events determines the necessary thermal margin (MCPR limits) to protect against the occurrence of boiling transition during the most limiting anticipated transient. Thermal margins are calculated for operation within the allowed regions of the power/flow operating map up to the full power/full flow operating condition. Resulting Thermal margin analyses are also presented for operation in the Increased Core Flow (ICF) region of the power/flow operating map and for operation with a Final Feedwater Temperature Reduction (FFTR). Analyses are also reported for operation with the Recirculation Pump Trip (RPT) out of service and with the turbine bypass capability inoperable. An evaluation is also made to demonstrate the vessel integrity for the most limiting pressurization event. The bases for these analyses have been provided in Reference 1.

## 2.0 SUMMARY

Using ENC methodology and considering Cycle 2 fuels, the most limiting plant system transient with regard to thermal margin at rated power and flow conditions was determined to be the generator Load Rejection Without Bypass (LRWB). The Minimum Critical Power Ratio (MCPR) limits for potentially limiting plant system transient events are shown in Table 2.1 for comparison. The values in Table 2.1 were determined assuming bounding conditions in the analyses. These transients were evaluated with all co-resident fuel types modeled and the most limiting condition was used to determine the reported MCPRs. The Control Rod Withdrawal Error (CRWE) analysis and Cycle 2 MCPR operating limit are reported in Reference 2.

Maximum system pressure has been calculated for the containment isolation event, which is a rapid closure of all main steam isolation valves, using the scenario as specified by the ASME Pressure Vessel Code. This analysis shows that during Cycle 2 the safety valves of Susquehanna Unit 2 have sufficient capacity and performance to prevent the pressure from reaching the established transient pressure safety limit of 110% of design pressure (1.1 x 1250 = 1375 psig). The analysis also assumed six safety relief valves out of service. The maximum system pressures predicted during the event are shown in Table 2.1.

Results for RPT out of service are reported in Section 3.2.1, and results for operation at ICF and FFTR are reported in Section 4.

Table 2.1 Transient Analysis Results at Design Basis Conditions*

| Transient | Δ CPR/MCPR** | |
|---|---|---|
| | ENC 9x9 | GE 8x8 |
| Load Rejection Without Bypass | 0.17/1.23 | 0.16/1.22 |
| Feedwater Controller Failure | 0.15/1.21 | 0.14/1.20 |
| Loss of Feedwater Heating | NA /1.14 | NA /1.14 |

Maximum Pressure (psig)

| Transient | Vessel Dome | Vessel Lower Plenum | Steam Line |
|---|---|---|---|
| MSIV Closure | 1301 | 1315 | 1305 |

* 104% power/100% flow.

** Based on a safety limit MCPR of 1.06.

## 3.0 TRANSIENT ANALYSIS FOR THERMAL MARGIN

### 3.1 Design Basis

Consistent with the FSAR plant transient analysis, thermal margin operating MCPR limits are determined based on the 104% power/100% flow operating point. This thermal margin operating MCPR limit is then modified as a function of power and flow as required to protect against boiling transition resulting from transients occurring from allowed conditions on the power/flow operating map. The plant conditions for the 104% power/100% flow point are as shown in Table 3.1. The most limiting point in Cycle 2 has been determined to be at end of full power capability when control rods are fully withdrawn from the core. The thermal margin limit established for end of full power conditions is conservative for cases where control rods are partially inserted. Following requirements established in the Plant Operating License and associated Technical Specifications, observance of a MCPR limit of 1.23 for 9x9 fuel and 1.22 for 8x8 fuel or greater conservatively protects against boiling transition during anticipated plant systems transients from design basis conditions for Susquehanna Unit 2 Cycle 2.

The calculational models used to determine thermal margin include ENC's plant transient and core thermal-hydraulic codes as described in previous documentation[1,4-7]. Fuel pellet-to-clad gap conductances used in the analyses are based on calculations with RODEX2[8]. Table 3.2 summarizes the values used for important parameters to provide a bounding analysis. Recirculation Pump Trip (RPT) coastdown was input based on measured Susquehanna Unit 2 startup test data. To confirm the neutronics as required by the SER issued for the supplements of Reference 1 the Susquehanna system transient model was benchmarked to appropriate Susquehanna Unit 2 startup test data. All transients were analyzed on a bounding basis using the COTRANSA hot channel delta CPR model as described in Reference 9.

### 3.2 Anticipated Transients

ENC considers eight categories of potential system transient occurrences for Jet Pump BWRs in XN-NF-79-71[1]. The loss of feedwater heating transient has been analyzed on a generic basis as reported in Reference 10. Results shown for this transient are from the ENC generic analysis.

The two most limiting transients are described here in detail to show the thermal margin for Cycle 2 of Susquehanna Unit 2. These transients are:

- Load Rejection Without Bypass (LRWB)

- Feedwater Controller Failure (FWCF)

A summary of the transient analyses is shown in Table 3.3. Other plant transient events are inherently nonlimiting or clearly bounded by one of the above events.

#### 3.2.1 Load Rejection Without Bypass

This event is the most limiting of the class of transients characterized by rapid vessel pressurization. The generator load rejection causes a turbine control valve trip, which initiates a reactor scram and RPT. The compression wave produced by the fast control valve closure travels through the steam lines into the vessel and creates the vessel pressurization. Turbine bypass flow, which could mitigate the pressurization effect, is not allowed. The excursion of core power due to void collapse is primarily terminated by reactor scram and void growth due to RPT. Figures 3.1 and 3.2 depict the time variance of critical reactor and plant parameters during the load rejection transient calculation with bounding assumptions. The bounding assumptions are consistent with ENC's COTRANSA code uncertainties analysis methodology as reported in XN-NF-79-71(P) Rev. 2, Supplements 1-3 and approved by NRC. The bounding assumptions include:

- Technical Specification minimum control rod speed
- Technical Specification maximum scram delay time
- integral power increased by 10%

At design basis conditions (104% power/100% flow) this results in a delta CPR of 0.17 for the load rejection without bypass when RPT is operable for ENC 9x9 fuel. The corresponding delta CPR for GE 8x8 fuel is 0.16.

The load rejection was then analyzed assuming the same bounding conditions but with both RPT and bypass inoperable. This resulted in delta CPRs of 0.31 for both ENC 9x9 and GE 8x8 fuel.

### 3.2.2 Feedwater Controller Failure

Failure of the feedwater control system is postulated to lead to a maximum increase in feedwater flow into the vessel. As the excessive feedwater flow subcools the recirculating water returning to the reactor core, the core power will rise and attain a new equilibrium if no other action is taken. Eventually, the inventory of water in the downcomer will rise until the high level vessel trip setting is exceeded. To protect against spillover of subcooled water to the turbine, the turbine trips, closing the turbine stop valves and initiating a reactor scram. The compression wave that is created, though mitigated by bypass flow, pressurizes the core and causes a power excursion. The power increase is terminated by reactor scram, RPT, and pressure relief from the bypass valves opening.

The evaluation of the flow event at design basis conditions was performed with bounding values and resulted in a delta CPR of 0.15 for ENC 9x9 fuel and 0.14 for GE 8x8 fuel. Figures 3.3 and 3.4 present key variables for this feedwater controller failure event. This event was also examined for reduced power conditions at full flow. The results for the FWCF transients

from reduced power conditions are shown in Table 3.4. The calculated results show that FWCF delta CPRs vary with decreasing power at full flow conditions. The highest delta CPRs were calculated at the 40% power/100% flow conditions.

This transient event at full power and full flow conditions was also analyzed assuming bounding conditions and failure of the bypass valves to open. This resulted in a delta CPR of 0.18 for ENC 9x9 fuel and 0.17 for GE 8x8 fuel.

### 3.2.3 Loss of Feedwater Heating

The loss of feedwater heating leads to a gradual increase in the subcooling of the water in the reactor lower plenum. Reactor power slowly rises to the thermal power monitor system trip setpoint. The gradual power change allows fuel thermal response to maintain pace with the increase in neutron flux. ENC has analyzed the loss of feedwater heating event on a generic basis as described in Reference 10. Based on the generic analysis and the Cycle 2 safety limit of 1.06, the MCPR limit for Susquehanna Unit 2 Cycle 2 will be 1.14 for both the ENC 9x9 fuel and the GE 8x8 fuel for the loss of feedwater heating event.

The bypass valves do not significantly affect the loss of feedwater heating results. Thus, this MCPR limit is applicable whether the bypass valves are operable or not.

## 3.3 Calculational Model

The plant transient code used to evaluate the generator load rejection and feedwater flow increase was ENC's code COTRANSA[1]. The axial one-dimensional neutronics model predicted reactor power shifts toward the core middle and top as pressurization occurred. This was accounted for explicitly in determining thermal margin changes in the transient. The loss of feedwater heating event

was evaluated generically because rapid pressurization and void collapse do not occur in this event. Appendix A[3] of the Susquehanna Unit 1 Cycle 2 analysis delineates the changes made to COTRANSA[1] to merge the PTSBWR3 code with the COTRANSA code, to refine numerical techniques and to improve input. Appendix A of Reference 9 describes the refinement made to the hot channel model to calculate the delta CPR's during the transient. Appendix B of Reference 3 delineates the plant related changes made to these codes for the Susquehanna Units 1 and 2 analyses.

## 3.4 Safety Limit

The safety limit is the minimum value of the critical power ratio (CPR) at which the fuel could be operated where the expected number of rods in boiling transition would not exceed 0.1% of the fuel rods in the core. The safety limit is the MCPR which would be permitted to occur during the limiting anticipated operational occurrence. The safety limit for all fuel types in Susquehanna Unit 2 Cycle 2 was determined by the methodology presented in Reference 4 to have a value of 1.06. The input parameters and uncertainties used to establish the safety limit are presented in Appendix A of this report.

Table 3.1 Reactor Design and Plant Conditions
Susquehanna Unit 2

| | |
|---|---|
| Reactor Thermal Power (104%) | 3439 MWt |
| Total Core Flow (100%) | 100.0 Mlb/hr |
| Core In-Channel Flow | 89.7 Mlb/hr |
| Core Bypass Flow | 10.3 Mlb/hr |
| Core Inlet Enthalpy | 518.0 Btu/lbm |
| Vessel Pressures | |
| Steam Dome | 1031 psia |
| Upper Plenum | 1049 psia |
| Core | 1058 psia |
| Lower Plenum | 1067 psia |
| Turbine Pressure | 974.7 psia |
| Feedwater/Steam Flow | 14.15 Mlb/hr |
| Feedwater Enthalpy | 360.8 Btu/lbm |
| Recirculation Pump Flow (per pump) | 15.7 Mlb/hr |

Table 3.2 Significant Parameter Values Used in Analysis
Susquehanna Unit 2

| | |
|---|---|
| High Neutron Flux Trip | 125.3% |
| Control Rod Insertion Time | 3.5 sec/90% inserted |
| Control Rod Worth | nominal |
| Void Reactivity Feedback | nominal |
| Time to Deenergized Pilot Scram Solenoid Valves | 200 msec (maximum) |
| Time to Sense Fast Turbine Control Valve Closure | 30 msec |
| Time from High Neutron Flux Time to Control Rod Motion | 290 msec |
| Turbine Stop Valve Stroke Time | 100 msec |
| Turbine Stop Valve Position Trip | 90% open |
| Turbine Control Valve Stroke Time (Total) | 70 msec |
| Fuel/Cladding Gap Conductance Core Average (Constant) | 443.8 Btu/hr-ft2-F |
| Safety/Relief Valve Performance Settings | Technical Specifications |
| Relief Valve Capacity | 225.4 lbm/sec (1110 psig) |
| Pilot Operated Valve Delay/Stroke | 400/150 msec |

Table 3.2 Significant Parameter Values Used in Analysis (Cont.)
Susquehanna Unit 2

| | |
|---|---|
| MSIV Stroke Time | 3.0 sec |
| MSIV Position Trip Setpoint | 90% open |
| Turbine Bypass Valve Performance | |
| Total Capacity | 936.11 lbm/sec |
| Delay to Opening (80% open) | 300 msec |
| Fraction of Energy Generated in Fuel | 0.965 |
| Vessel Water Level (above Separator Skirt) | |
| High Level Trip | 58.7 in |
| Normal | 36.5 in |
| Low Level Trip | 8 in |
| Maximum Feedwater Runout Flow | |
| Three Pumps | 4118 lbm/sec |
| Recirculation Pump Trip Setpoint | 1170 psig Vessel Pressure |

Table 3.2 Significant Parameter Values Used in Analysis (Cont.)
Susquehanna Unit 2

Control Characteristics

| | |
|---|---|
| Sensor Time Constants | |
| Pressure | 500 msec |
| Others | 250 msec |
| Feedwater Control Mode | Three-Element |
| Feedwater Master Controller | |
| Proportional Gain | 50.0 (%/%) (%/ft) |
| Reset Rate | 1.70 (%/sec/ft) |
| Feedwater 100% Mismatch | |
| Water Level Error | 48 in |
| Steam Flow Equiv. | 100% |
| Flow Control Mode | Manual |
| Pressure Regulator Settings | |
| Lead | 3.0 sec |
| Lag | 7.0 sec |
| Gain | 3.33%/psid |

Table 3.3 Results of System Plant Transient Analyses

| Event | Maximum Neutron Flux % Rated | Maximum Core Average Heat Flux % Rated | Maximum System Pressure (psia) | Δ CPR |
|---|---|---|---|---|
| Load Rejection Without Bypass | 274 | 114.3 | 1213 | .17 |
| Feedwater Controller Failure | 245 | 114.7 | 1180 | .15 |
| MSIV Closure with Flux Scram | 368 | 130.7 | 1330 | NA |

Note: All events are bounding case at 104% power/100% flow.

Table 3.4

Feedwater Controller Failure Analysis Results at 100% Flow

| % Power | Delta CPR | |
|---|---|---|
| | GE 8x8 | ENC 9x9 |
| 100 | .14 | .15 |
| 80 | .22 | .24 |
| 65 | .23 | .25 |
| 40 | .26 | .29 |

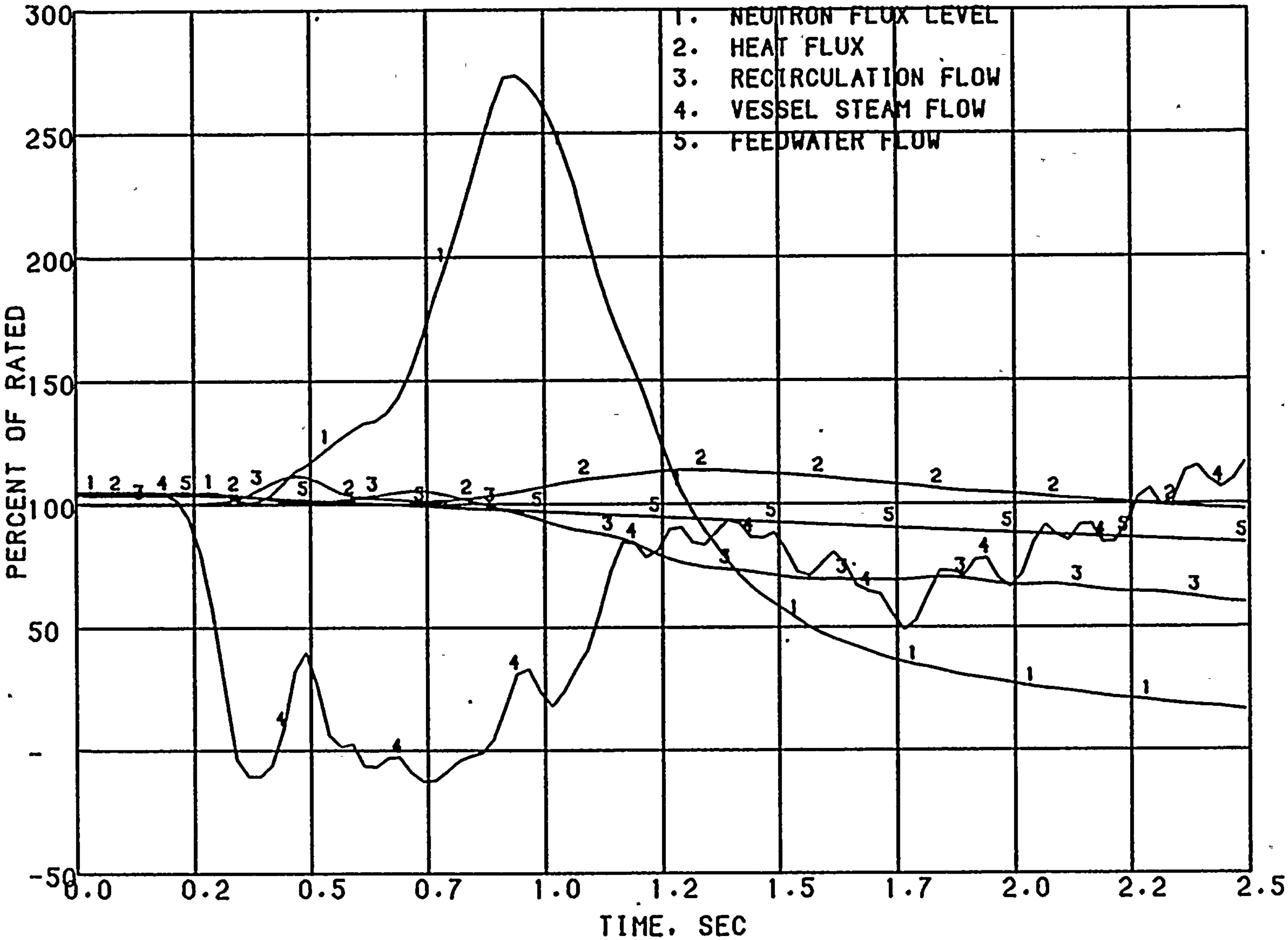


Figure 3.1 Load Rejection Without Bypass

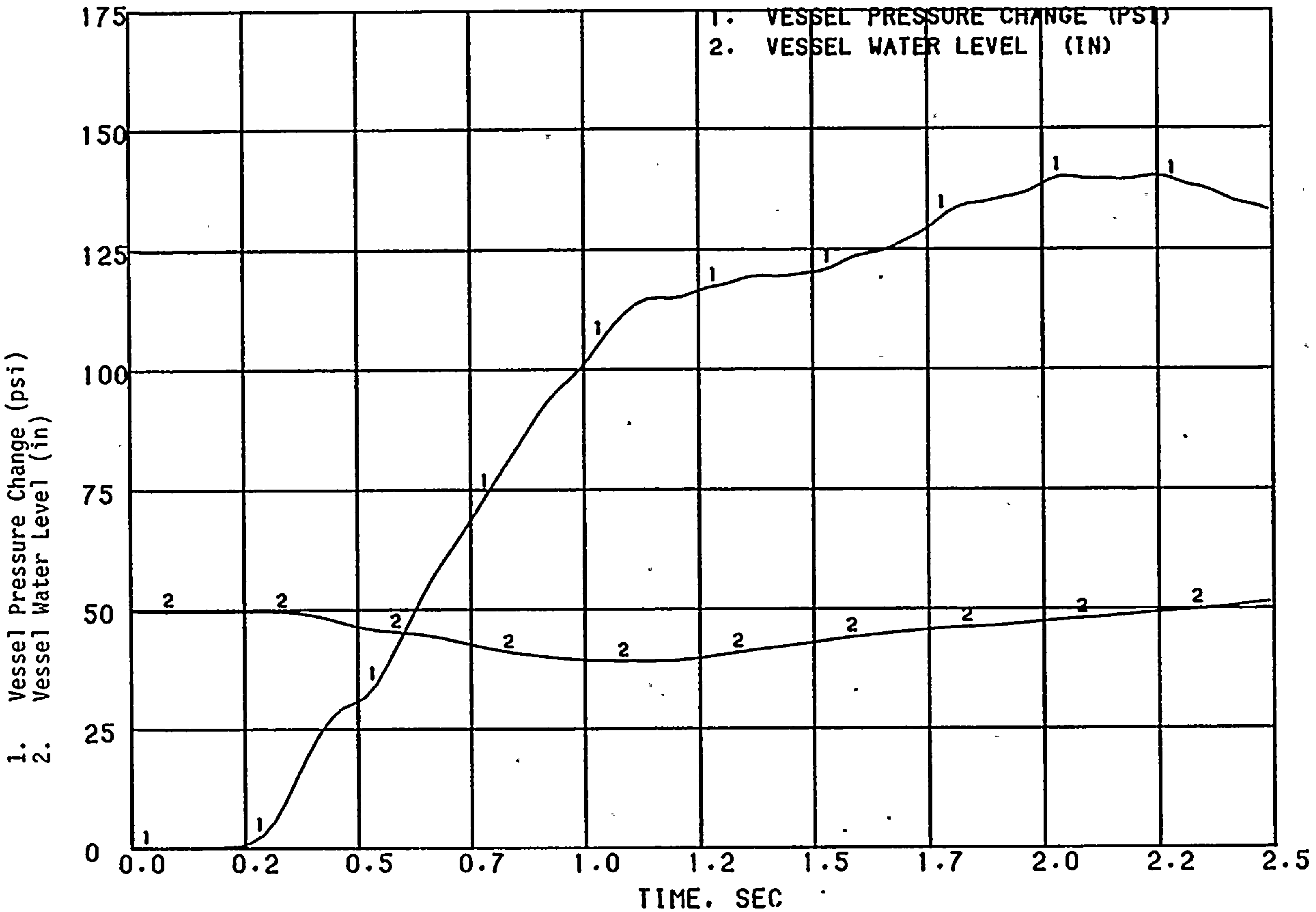


Figure 3.2 Load Rejection Without Bypass

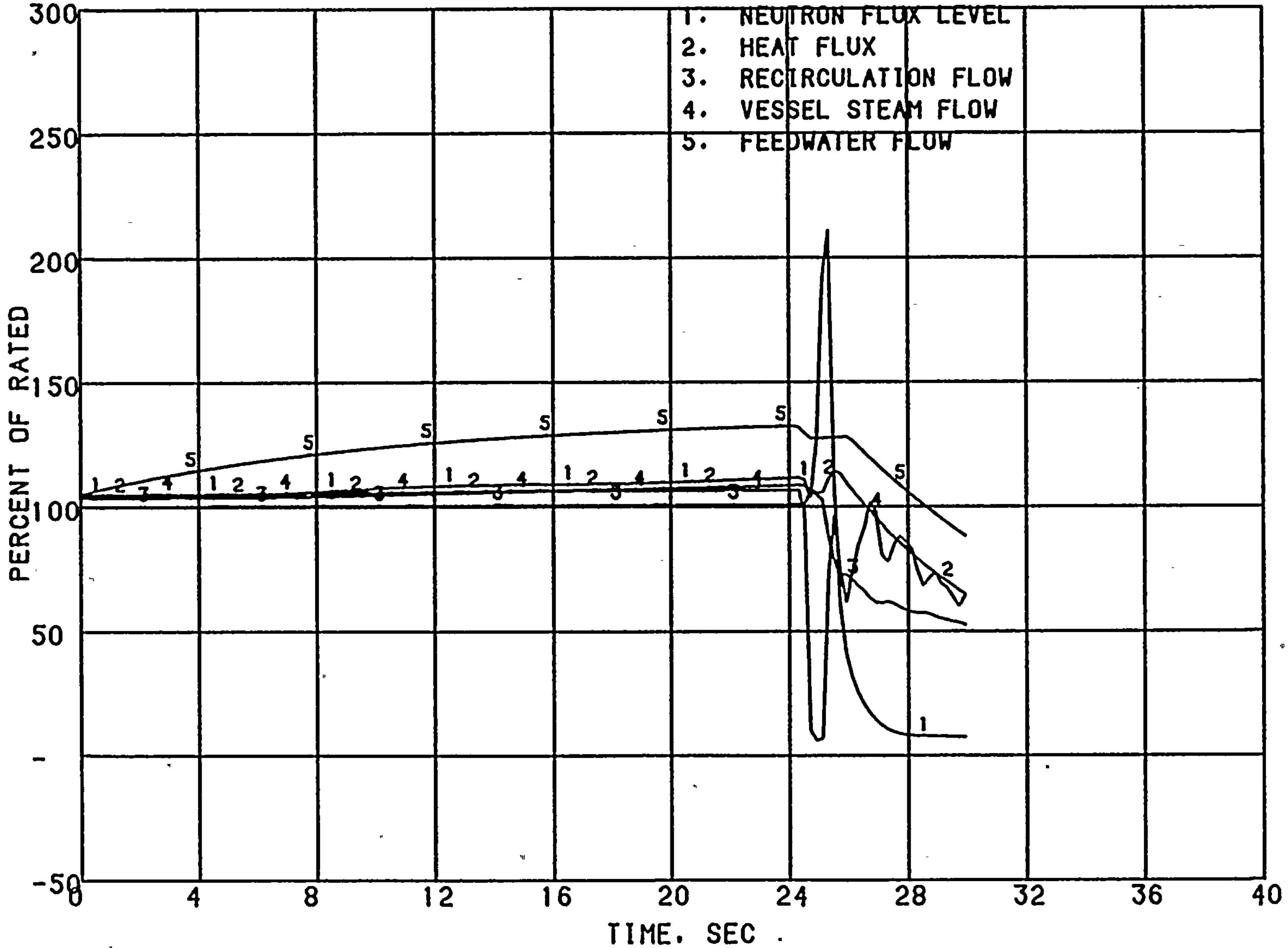


Figure 3.3 Feedwater Controller Failure

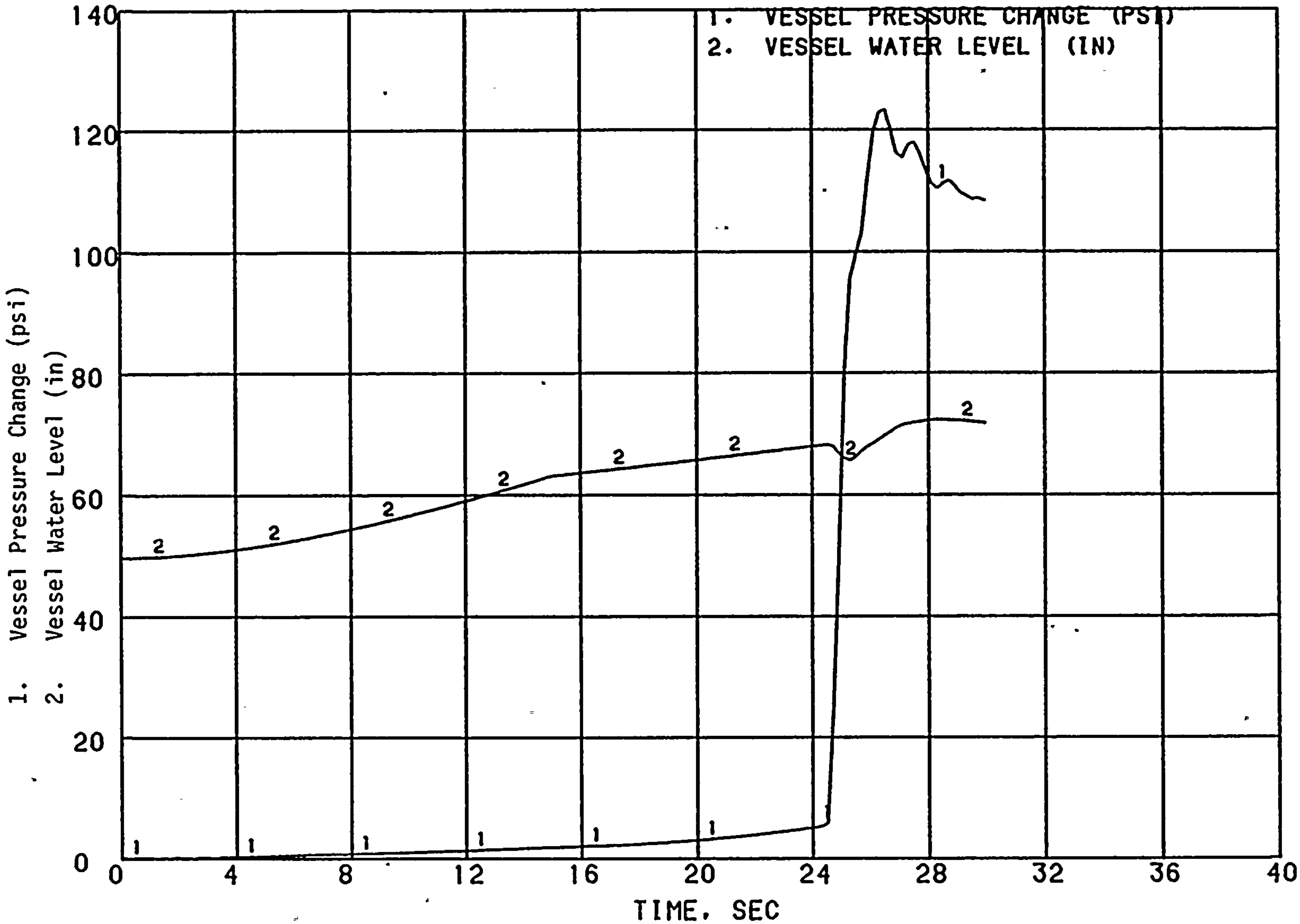


Figure 3.4 Feedwater Controller Failure

## 4.0 ANALYSES WITH INCREASED CORE FLOW (ICF) AND FINAL FEEDWATER TEMPERATURE REDUCTION (FFTR)

As part of the Susquehanna Unit 2 licensing analysis, ENC evaluated transients for operation in the Increased Core Flow (ICF) operating region up to 108% of rated flow. Transient analyses were also performed for a feedwater temperature reduction of up to 65 degrees F at both nominal flow and increased core flow conditions at the end of the operating cycle. This 65 degree F temperature reduction was conservatively held constant at all power levels evaluated. A summary of the transient analyses is shown in Table 4.1. Comparison of the results in Table 2.1 and 4.1 indicate that ICF had no significant effect on the LRWB delta CPR results and FFTR condition slightly reduced the impact of this document. The corresponding maximum overpressurization event is discussed in Section 5.0 and the pump run-up analysis is reported in Section 6.0.

The effects of the final feedwater temperature reduction were evaluated by analyzing the FWCF transient over the allowed power range for both nominal feedwater temperature and a 65 degree F final feedwater temperature reduction. Calculations were performed for both the 100% core flow and for the 108% core flow conditions. The results of these calculations are shown in Table 4.2.

The calculated FWCF transient delta CPR generally increases with decreasing power at both flow conditions, and an increased MCPR limit is indicated for low power operating conditions. Thus, for increased core flow operation, increased MCPR limits are indicated. A further, but small, delta CPR increase is generally indicated to operate with reduced feedwater temperature for both rated core flow and increased core flow.

Table 4.1 Results of System Plant Transient Analysis at ICF and at FFTR

Load Rejection Without Bypass

| | Maximum Neutronic Flux (% rated) | Minimum Core Average (% rated) | Maximum System Pressure (psia) | Delta CPR |
|---|---|---|---|---|
| 104/100 (FFTR) | 253 | 112.9 | 1191 | 0.15 |
| 104/108 | 241 | 112.1 | 1210 | 0.17 |
| 104/108 (FFTR) | 222 | 110.8 | 1187 | 0.15 |

ASME Overpressure (MSIV Closure) (psig)

| | Vessel Dome | Vessel Lower Plenum | Steam Line |
|---|---|---|---|
| 104/100 (FFTR) | 1264 | 1279 | 1265 |
| 104/108 | 1290 | 1307 | 1296 |
| 104/108 (FFTR) | 1257 | 1274 | 1259 |

Table 4.2 Feedwater Controller Failure
Delta CPR Results of ICF and FFTR Analyses

| | Nominal Feedwater Temp. | | FFTR* | |
|---|---|---|---|---|
| % Power/% Flow | GE 8X8 | ENC 9x9 | GE 8X8 | ENC 9X9 |
| 100 / 100 | 0.14 | 0.15 | 0.16 | 0.17 |
| 80 / 100 | 0.22 | 0.24 | 0.20 | 0.22 |
| 65 / 100 | 0.23 | 0.25 | 0.24 | 0.26 |
| 40 / 100 | 0.26 | 0.29 | 0.26 | 0.29 |
| 100 / 108 | 0.15 | 0.16 | 0.16 | 0.17 |
| 80 / 108 | 0.20 | 0.22 | 0.20 | 0.22 |
| 65 / 108 | 0.23 | 0.25 | 0.24 | 0.26 |
| 40 / 108 | 0.27 | 0.30 | 0.26 | 0.30 |

*65°F reduction in Feedwater Temperature.

## 5.0 MAXIMUM OVERPRESSURIZATION

Maximum system pressure has been calculated for the containment isolation event (rapid closure of all main steam isolation valves) with an adverse scenario as specified by the ASME Pressure Vessel Code. This analysis showed that the safety valves of Susquehanna Unit 2 have sufficient capacity and performance to prevent pressure from reaching the established transient pressure safety limit of 110% of the design pressure. The maximum system pressures predicted during the event are shown in Table 2.1. This analysis also assumed six safety relief valves out of service.

### 5.1 Design Basis

The reactor conditions used in the evaluation of the maximum pressurization event are those shown in Table 3.1. The most critical active component (scram on MSIV closure) was assumed to fail during the transient. The calculation was performed with ENC's advanced plant simulator code COTRANSA[1], which includes an axial one-dimensional neutronics model.

### 5.2 Pressurization Transients

ENC has evaluated several pressurization events and has determined that closure of all Main Steam Isolation Valves (MSIVs) without direct scram is the most limiting. Though the closure rate of the MSIVs is substantially slower than the turbine stop valves or turbine control valves, the compressibility of the additional fluid in the steam lines causes the severity of these faster closures to be less. Essentially, the rate of steam velocity reduction is concentrated toward the end of the valve stroke, generating a substantial compression wave. Once the containment is isolated the subsequent core power production must be absorbed in a smaller volume than if a turbine isolation had occurred. Calculations have determined that the overall result is to cause isolation (MSIV closures) to be more limiting for system pressure than turbine isolations.

## 5.3 Closure of All Main Steam Isolation Valves

This calculation assumed that six relief valves were out of service and that all four steam isolation valves were isolated at the containment boundary within 3 seconds. At about 5.5 seconds, the reactor scram is initiated by reaching the high flux trip setpoints. Since scram performance was degraded to its Technical Specification limit, effective power shutdown is delayed until after 7.1 seconds. Substantial thermal power production enhances pressurization. Pressures reach the recirculation pump trip setpoint (1170 psig) before the pressurization has been reversed. Loss of coolant flow leads to enhanced steam production as less subcooled water is available to absorb core thermal power. The maximum pressure calculated in the steam lines was 1305 psig occurring near the vessel at about 10.1 seconds. The maximum vessel pressure was 1315 psig occurring in the lower plenum at about 10.0 seconds.

The analysis was repeated for ICF and FFTR conditions and the results are summarized in Table 4.1. Compaison of the results in Table 2.1 and Table 4.1 show that the design basis conditions are more limiting than ICF or FFTR conditions. At about 5.5 seconds, the reactor scram is initiated by reaching the high flux trip setpoints. Since scram performance was degraded to its Technical Specification limit, effective power shutdown is delayed until after 6.5 seconds. Substantial thermal power production enhances pressurization. Pressures reach the recirculation pump trip setpoint (1170 psig) before the pressurization has been reversed. Loss of coolant flow leads to enhanced steam production as less subcooled water is available to absorb core thermal power. The maximum pressure calculated in the steam lines was 1296 psig occurring near the vessel at about 10.2 seconds. The maximum vessel pressure was 1307 psig occurring in the lower plenum at about 9.8 seconds.

## 6.0 RECIRCULATION PUMP RUN-UP

Analysis of pump run-up events for operation at less than rated recirculation pump capacity demonstrates the need for an augmentation of the full flow MCPR operating limit for lower flow conditions. This is due to the potential for large reactor power increases should an uncontrolled pump flow increase occur.

This section discusses pump excursions when the plant is in manual flow control operation mode. Based on the results obtained from previous analyses which showed two pump excursions were the limiting pump run-up event, only two pump excursions are evaluated for Susquehanna Unit 2 Cycle 2. These results indicate that MCPR would decrease below the safety limit if the full flow reference MCPR was observed at initial conditions. Thus, an augmented MCPR is needed for partial flow operation to protect the two pump excursion event.

The evaluation of the two recirculation pump flow excursion for Susquehanna Unit 2 showed that establishment of MCPR limits for this event which prevents boiling transition will also bound single pump runups. The analysis of the two pump flow excursion indicates that the limiting event scenario is a gradual quasi-steady run-up due to the inlet enthalpy lag associated with a more rapid run-up.

The Susquehanna Unit 2 Cycle 2 analysis conservatively assumed the run-up event initiated at 57% power/40% flow and reached 111% rated power at 110% rated flow. 110% flow is consistent with increased core flow analysis. This power to flow relationship bounds that calculated by XTGBWR for the constant Xenon assumption.

The results of the two pump run-up analyses for manual flow control are presented in Figure 6.1. The cycle specific MCPR limit for Susquehanna Unit 2 Cycle 2 shall be the maximum of the reduced flow MCPR operating limit and the full flow MCPR operating limit.

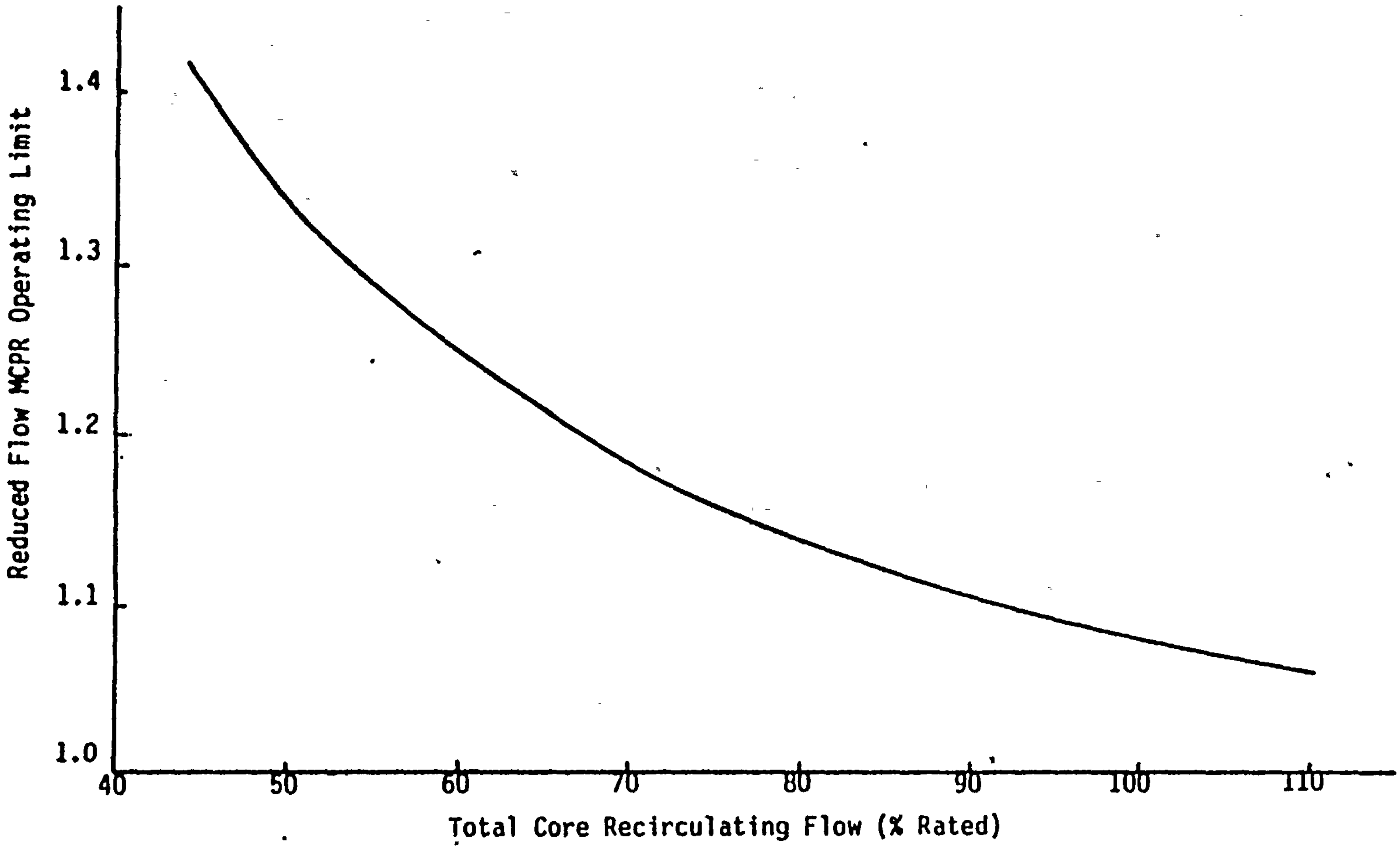


Figure 6.1 Reduced Flow MCPR Operating Limit

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# APPENDIX A

## MCPR SAFETY LIMIT

### A.1 INTRODUCTION

The MCPR fuel cladding integrity safety limit was calculated using the methodology and uncertainties described in Reference A.1. In this methodology, a Monte Carlo procedure is used to evaluate plant measurement and power predictions uncertainties such that during sustained operation at the MCPR Cladding Integrity Safety Limit, at least 99.9% of the fuel rods in the core would be expected to avoid boiling transition. This appendix describes the calculation and presents the analytical results

## A.2 CONCLUSIONS

During sustained operation at a MCPR of 1.06 with the design basis power distribution described below, at least 99.9% of the fuel rods in the core are expected to avoid boiling transition at a confidence level of 95%.

## A.3 DESIGN BASIS POWER DISTRIBUTION

Predicted power distributions were extracted from the fuel management analysis for Susquehanna Unit 2 Cycle 2. These radial power distributions were evaluated for performance as the design basis radial power map, and the distribution at 10,500 MWD/MT cycle exposure was selected as the most severe expected distribution for the cycle. The distribution was skewed toward higher power factors by the addition of bundles with a radial peaking factor approximating an operating MCPR level of 1.26 at full power.

The resulting design basis radial power distribution is shown in Figure A.3-1.

The fuel management analysis indicated that the maximum power ENC bundle in the core at this statepoint was predicted to be operating at an exposure level of 12,600 MWD/MT, so a local power distribution typical of a nodal exposure of 15,000 MWD/MT was selected as the design basis local power distribution. This distribution is shown in Figure A.3-2.

A boundingly flat local power distribution was selected for the co-resident G.E. Fuel. This distribution is shown in Figure A.3-3.

Because the predicted power distributions during the cycle were not all characterized by bottom peaked axial distributions, representative safety limit evaluations were performed at several representative cycle burnup statepoints throughout the cycle, including all points at which the power was skewed toward the upper half of the core. These analyses confirmed the that most severe power distribution conditions were those which are predicted to exist at the end of Cycle 2. The 1.06 safety limit was confirmed at all the points evaluated.

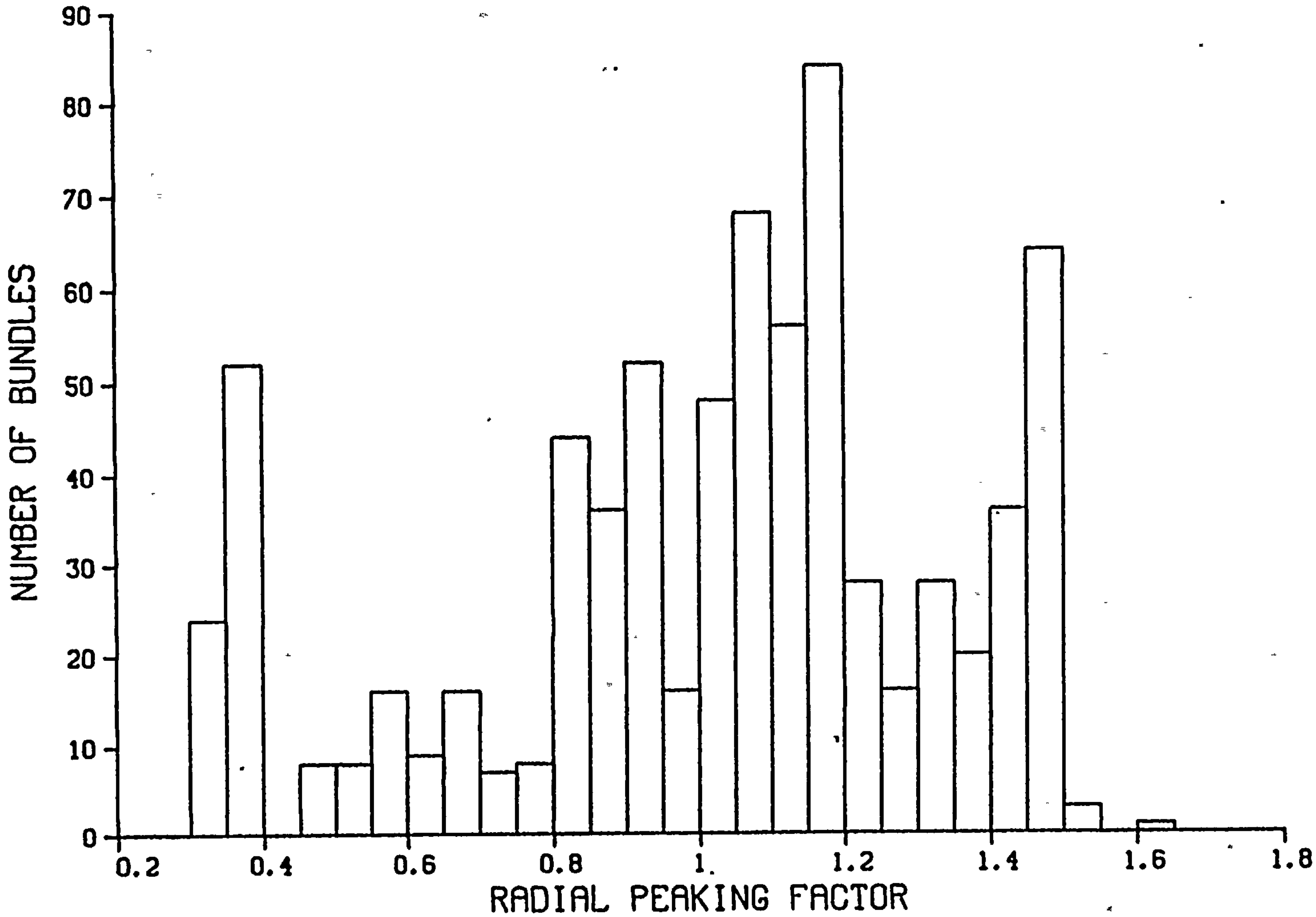


Figure A.3-1 Design Basis Radial Power Histogram

| | | | | | | | | |
|---|---|---|---|---|---|---|---|---|
| 0.97 | 1.01 | 0.97 | 1.04 | 1.04 | 1.05 | 0.97 | 1.02 | 0.97 |
| 1.01 | 0.94 | 0.97 | 1.07 | 1.06 | 0.95 | 1.00 | 0.95 | 1.02 |
| 0.97 | 0.99 | 1.04 | 1.05 | 1.05 | 1.02 | 1.06 | 1.00 | 0.97 |
| 1.04 | 0.93 | 1.05 | 1.01 | 0.97 | 0.00 | 1.02 | 0.95 | 1.05 |
| 1.03 | 1.05 | 1.03 | 1.00 | 0.00 | 0.97 | 1.05 | 1.06 | 1.04 |
| 1.04 | 0.94 | 1.04 | 1.00 | 1.00 | 1.01 | 1.05 | 1.07 | 1.04 |
| 0.97 | 0.98 | 0.90 | 1.04 | 1.03 | 1.05 | 1.04 | 0.97 | 0.97 |
| 0.91 | 0.94 | 0.98 | 0.94 | 1.05 | 0.93 | 0.99 | 0.94 | 1.01 |
| 0.88 * | 0.91 | 0.97 | 1.04 | 1.03 | 1.04 | 0.97 | 1.01 | 0.97 |

Figure A.3-2
DESIGN BASIS LOCAL POWER DISTRIBUTION
ENC XN-1 9X9 FUEL

* Rod adjacent to control blade corner location

| | | | | | | | |
|---|---|---|---|---|---|---|---|
| 1.03 | 1.00 | 0.99 | 0.99 | 0.99 | 0.99 | 1.00 | 1.03 |
| 1.00 | 0.99 | 1.03 | 1.02 | 0.99 | 0.99 | 0.97 | 1.00 |
| 0.99 | 1.03 | 0.91 | 1.02 | 1.01 | 0.98 | 0.99 | 0.99 |
| 0.99 | 1.03 | 1.02 | 0.00 | 1.02 | 1.01 | 0.99 | 0.99 |
| 0.99 | 1.02 | 1.01 | 0.91 | 0.00 | 1.02 | 1.02 | 0.99 |
| 0.99 | 0.99 | 1.02 | 1.01 | 1.02 | 0.91 | 1.03 | 0.99 |
| 1.00 | 0.97 | 0.99 | 1.02 | 1.03 | 1.03 | 0.99 | 1.00 |
| 1.03 * | 1.00 | 0.99 | 0.99 | 0.99 | 0.99 | 1.00 | 1.03 |

Figure A.3-3
DESIGN BASIS LOCAL POWER DISTRIBUTION
G.E. 8X8 FUEL

* Rod adjacent to control blade corner location

## A.4 CALCULATION OF THE NUMBER OF RODS IN BOILING TRANSITION

The SAFTLIM computer code was used to analyze the number of fuel rods in boiling transition. The XN-3 correlation[A-2] was used to predict critical heat flux phenomena. Five hundred Monte Carlo trials were performed to support the MCPR safety limit. Non-parametric tolerance limits[A-3] were used in lieu of Pearson curve fitting. The uncertainties used in the analysis for normal conditions were those identified in Reference A-1. At least 99.9% of the fuel rods in the core were expected to avoid boiling transition with a confidence level of 95%.

## A.5 REFERENCES

A-1. "Exxon Nuclear Critical Power Methodology for Boiling Water Reactors", Revision 1, XN-NF-524(A), Exxon Nuclear Company, Richland, WA (November 1983).

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# SUSQUEHANNA UNIT 2 CYCLE 2 PLANT TRANSIENT ANALYSIS

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