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Maximum Extended Load Line Limit and Increased Core Flow for Cooper Nuclear Station

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ABSTRACT

A safety evaluation has been performed to demonstrate that Cooper Nuclear Station (CNS) can operate in the Maximum Extended Load Line Limit (MELLL) and Increased Core Flow (ICF) operating domains. The MELLL domain extends the rated rod line to the 121% rod line up to rated power. The ICF domain extends the core flow to 105% of rated up to rated power. Results of the evaluations show that CNS can safely operate in the MELLL and ICF operating domains without risk to the public health and safety.

ACRONYMS

Term	Definition
ADS	Automatic Depressurization System
AOO	Anticipated Operational Occurrence
APRM	Average Power Range Monitor
ARI	Alternate Rod Insertion
ARTS	APRM/RBM/Technical Specification
ATWS	Anticipated Transient Without Scram
BOC	Beginning of Cycle
BWR	Boiling Water Reactor
CNS	Cooper Nuclear Station
CPR	Critical Power Ratio
CRD	Control Rod Drive
ECCS	Emergency Core Cooling System
EOC	End of Cycle
EOC20	End of Cycle 20
ELLL	Extended Line Load Limit
FIV	Flow-Induced Vibration
FWCF	Feedwater Controller Failure
GDC	General Design Criteria
GE	General Electric
ICF	Increased Core Flow
ITS	Improved Technical Specifications
JPSL	Jet Pump Sensing Line
LFWH	Loss of Feedwater Heater
LOCA	Loss-of-Coolant Accident
LPRM	Local Power Range Monitor
LRNBP	Load Rejection with No Bypass

ACRONYMS
(Continued)

Term	Definition
MELLL	Maximum Extended Load Line Limit
MCPR	Minimum Critical Power Ratio
MOC	Middle of Cycle
MSIV	Main Steamline Isolation Valve
NPPD	Nebraska Public Power District
NBR	Nuclear Boiler Rated
OFS	Orificed Fuel Support
OLMCPR	Operating Limit Minimum Critical Power
PCT	Peak Clad Temperature
PULD	Plant Unique Load Definition
RBM	Rod-Block Monitor
RHR	Residual Heat Removal
RIPD	Reactor Internal Pressure Differences
RLB	Recirculation Line Break
RPT	Recirculation Pump Trip
RPV	Reactor Pressure Vessel
RRS	Reactor Recirculation System
SLCS	Standby Liquid Control System
SLMCPR	Safety Limit Minimum Critical Power Ratio
SRV	Safety-Relief Valve
SSV	Spring-Safety Valve
TTNBP	Turbine Trip with No Bypass
USAR	Updated Safety Analysis Report

ABBREVIATIONS

Term	Definition
BTU/lb	British Thermal Unit/Pound
c	Cents of Reactivity
ft^3	Cubic Feet
°F	Degrees Fahrenheit
gpm	Gallons per Minute
lb_f	Pounds Force
lb_m	Pounds Mass
Mlb/hr	Million Pounds/Hour
MWt	Megawatts Thermal
psid	Pounds per Square Inch, Differential
psig	Pounds per Square Inch, Gage

TABLE OF CONTENTS

TABLE OF CONTENTS
(Continued)

LIST OF TABLES

LIST OF FIGURES

1.0 INTRODUCTION AND SUMMARY

Changes in the plant-operating domain can improve the operating flexibility of the Boiling Water Reactor (BWR) nuclear power plant. Two changes in the operating domain are proposed for the Cooper Nuclear Station (CNS): Maximum Extended Load Line Limit (MELLL) to replace the current CNS Extended Load Line Limit (ELLL) [Reference 1] and Increased Core Flow (ICF). CNS currently operates with an operating domain bounded by ELLL and 100% rated flow.

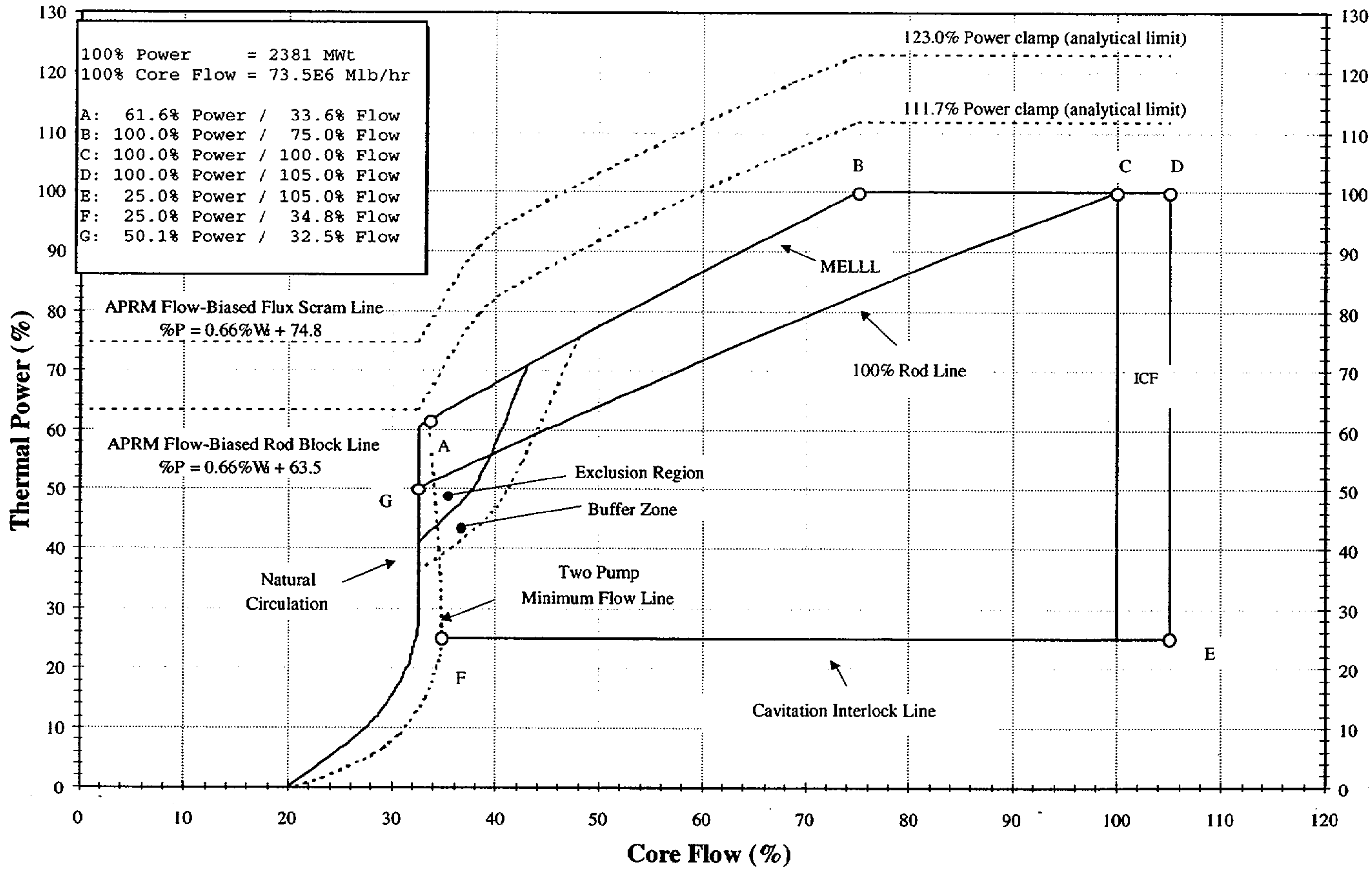
Extending plant operation at rated power with less than rated core flow improves the power ascension capability by reducing the number of adjustments made to compensate for reactivity changes due to xenon effects and fuel burnup. Also, full power operation at less than rated core flow allows for flow control spectral shift operation, which improves fuel cycle economics. The current operating domain will be modified to include the extended operating region bounded by the rod line which passes through the 100% power and 75% core flow point (~121% rod line).

Increasing the core flow will allow CNS to maintain rated core thermal power after reaching the end-of-cycle all-control-rods-out exposure by slowly increasing core flow up to 105% of the rated value. In addition, during power coastdown in the ICF domain, CNS would maintain a constant recirculation flow rate profile, consistent with the maximum ICF value.

This report presents the results of the safety and system response evaluations performed for operation of CNS at both MELLL and ICF conditions. The revised power/flow region is shown in Figure 1-1. The core-wide Anticipated Operational Occurrences (AOOs) and Vessel Overpressure Protection Safety Analyses is documented in this report. The Emergency Core Cooling System (ECCS) analyses were performed for the limiting fuel design

All of the analyses presented in this report demonstrate that the plant can safely operate with MELLL and ICF. Specific operating Minimum Critical Power Ratio (MCPR) limits are provided for CNS Cycle 20. The impact on future operating cycles must be determined separately for those cycles.

Figure 1-1 CNS Power/Flow Map



2.0 ANTICIPATED OPERATIONAL OCCURRENCES

The core-wide AOOs were analyzed to support operation in both the MELLL and ICF domains. Primarily, the rod withdrawal limit setpoint, the initial control rod pattern and the error rod position affect the transient response of the localized rod withdrawal error event. Since neither MELLL nor ICF impacts these parameters, only core-wide AOOs are included in this report. The purpose of these evaluations is to establish the operating minimum critical power ratio (MCPR) limit for operation of CNS with MELLL and ICF for Cycle 20. AOO analysis for future cycles will be performed with the reload analysis.

2.1 Analysis Approach and Inputs

The core-wide AOO analyses for MELLL and ICF were performed for the limiting CNS Cycle 20 reload transients. These transient events include:

- Generator Load Rejection with No Bypass (LRNBP)
- Turbine Trip with No Bypass (TTNBP)
- Feedwater Controller Failure (FWCF) maximum demand
- Loss of 100°F Feedwater Heater (LFWH)

The analytical methods, as well as the input assumptions, such as reactor protection system setpoints and plant configurations, are consistent with the reload analysis.

The core-wide rapid pressurization events (LRNBP, TTNBP and FWCF) and the LFWH events are limiting for these two operating domains because the other potentially limiting events, such as mislocated bundle and rotated bundle, were analyzed and determined to be extremely mild.

Changes in the Cycle 20 initial conditions due to MELLL and ICF are given in Table 2-1. The analyses were performed at various powers and flows. For the LRNBP and TTNBP analyses, it is assumed that the turbine bypass is out of service and the safety-relief valves (SRVs) have a relaxed tolerance. The FWCF was analyzed assuming both the turbine bypass operable and one turbine bypass inoperable.

2.2 AOO Results

The peak values for neutron flux, core average heat flux, steamline pressure, vessel pressure and Uncorrected Change in Critical Power Ratio (ΔCPR) for each event analyzed in the MELLL, Rated, or ICF operating domain is given in Table 2-2 for the limiting fuel design, GE14. The MCPR operating limits associated with these two operating domains are given in Table 2-3. Key system responses for these events are shown in Figures 2-1 through 2-6.

These results show that the MCPR operating limits for 100% power/100% flow and MELLL are bounded by the ICF operating condition. The LRNBP is the limiting AOO event. Thus, future reload analyses will be bounded by the 100% power/105% flow initial condition.

Table 2-1
AOO Analysis Input and Initial Conditions

Parameter	Cycle 20 Licensing Basis	Cycle 20 MELLL	Cycle 20 ICF
	Power/Flow "A"	Power/Flow "B"	Power/Flow "C"
Thermal Power, MWt/% rated	2381	2381	2381
Core Flow, Mlb/hr/% rated	73.50	55.10	77.18
Steam Flow, Mlb/hr	9.56	9.54	9.56
Feedwater Temperature, °F	367.1	367.1	367.1
Core Inlet Enthalpy, BTU/lb	520.4	511.3	521.7
Dome Pressure, psig	1005	1005	1005
Core Average Void Fraction, %	33.46	39.79	32.39

Table 2-2

Core-Wide Transient Analysis Results for CNS Cycle 20

Transient AOO	Initial Power /Flow	Peak Neutron Flux (%NBR)	Peak Heat Flux (%NBR)	Peak Steamline Pressure (psig)	Peak Vessel Pressure (psig)	ΔCPR GE14
Equipment In Service						
TTNBP	"C"	288	115	1156	1194	0.32
LRNBP	"C"	298	115	1156	1193	0.33
FWCF	"C"	203	116	1129	1162	0.28
TTNBP	"B"	240	112	1158	1188	0.28
LRNBP	"B"	238	111	1158	1188	0.28
FWCF	"B"	169	111	1131	1157	0.22
LFWH	"B"	—	—	—	—	0.12
TTNBP	"A"	289	114	1156	1192	0.32
LRNBP	"A"	297	114	1156	1192	0.32
FWCF	"A"	197	115	1129	1160	0.27
1 Turbine Bypass out of Service						
FWCF	"C"	236	119	1139	1175	0.32

Table 2-3
MCPR Operating Limits for CNS Cycle 20

AOO	Power/Flow	Exposure	Option A GE14	Option B GE14
TTNBP	"B"	EOC-2K	1.48	1.37
LRNBP	"B"	EOC-2K	1.48	1.37
FWCF	"B"	EOC-2K	1.42	1.31
TTNBP	"B"	EOC(b)	1.59	1.42
LRNBP	"B"	EOC	1.58	1.41
FWCF	"B"	EOC	1.53	1.36
TTNBP	"A"	EOC-2K	1.51	1.40
LRNBP	"A"	EOC-2K	1.51	1.40
FWCF	"A"	EOC-2K	1.47	1.36
TTNBP	"A"	EOC(b)	1.63	1.46
LRNBP	"A"	EOC	1.63	1.46
FWCF	"A"	EOC	1.58	1.41
TTNBP	"C"	EEOC	1.63	1.46
LRNBP	"C"	EEOC	1.63	1.46
FWCF	"C"	EEOC	1.59	1.42

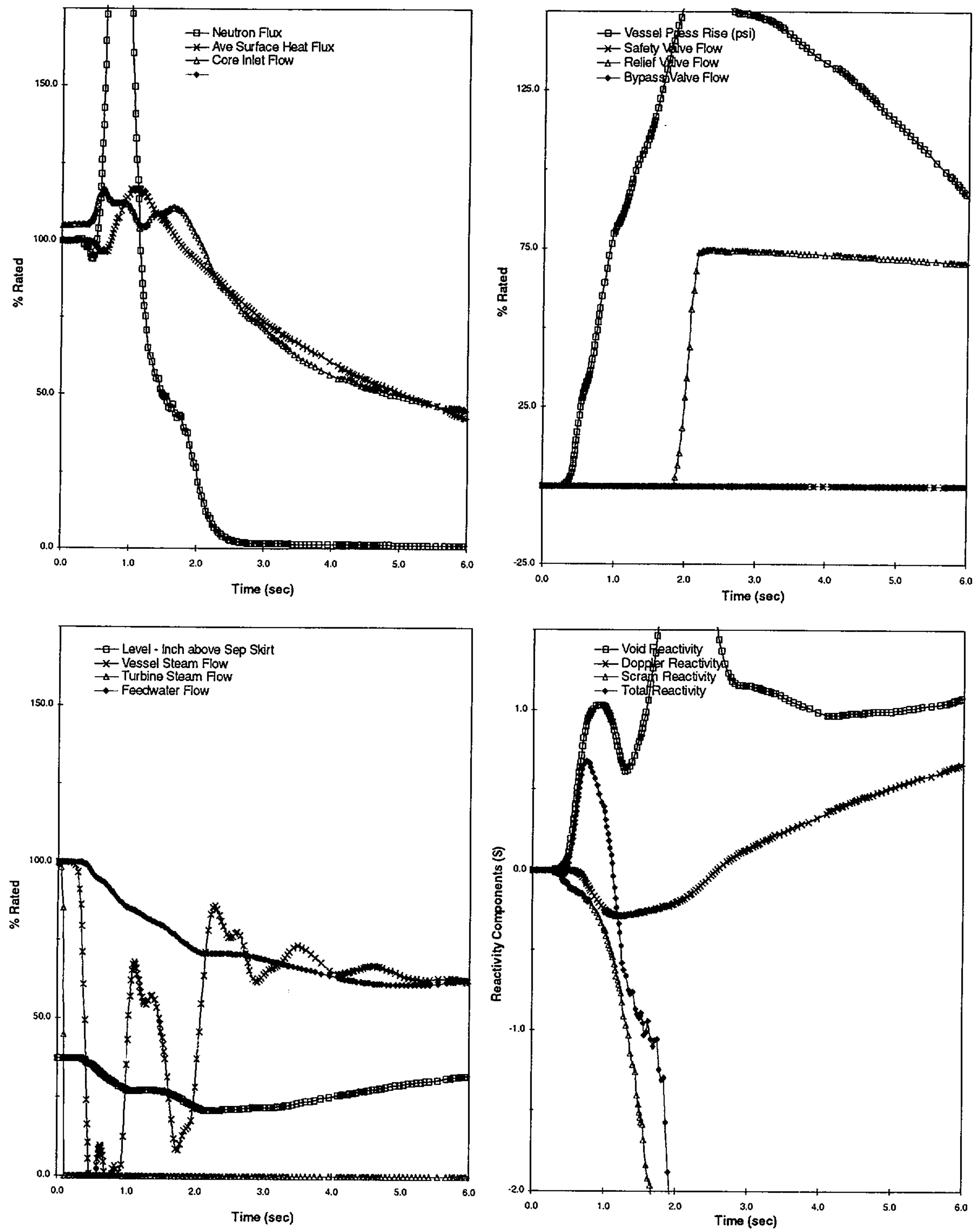


Figure 2-1 Plant Response to Turbine Trip w/o Bypass (EOC, 100% Power, 105% Flow)

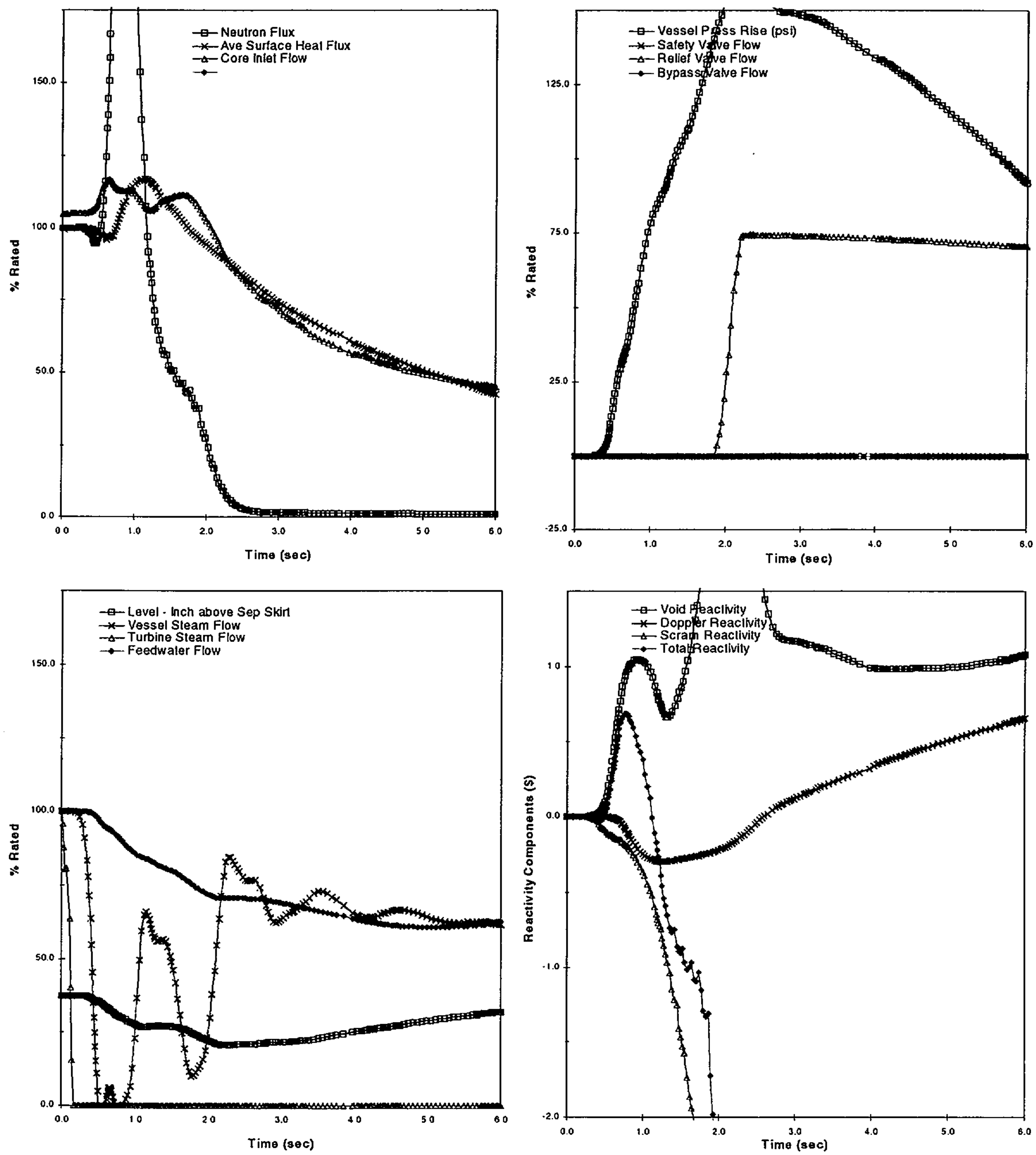


Figure 2-2 Plant Response to Load Rejection w/o Bypass (EOC, 100% Power, 105% Flow)

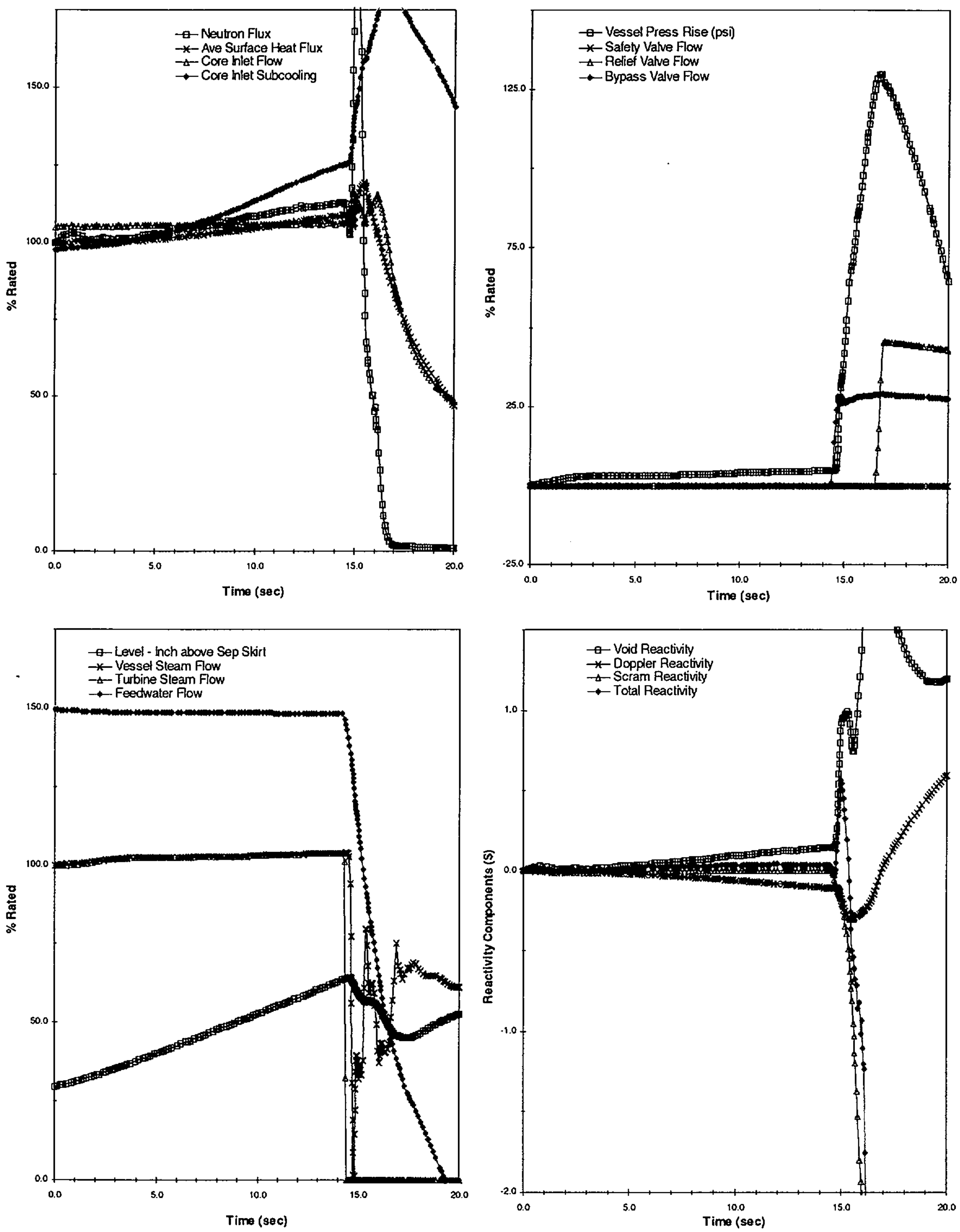


Figure 2-3 Plant Response to Feedwater Controller Failure (EOC, 100% Power, 105% Flow)

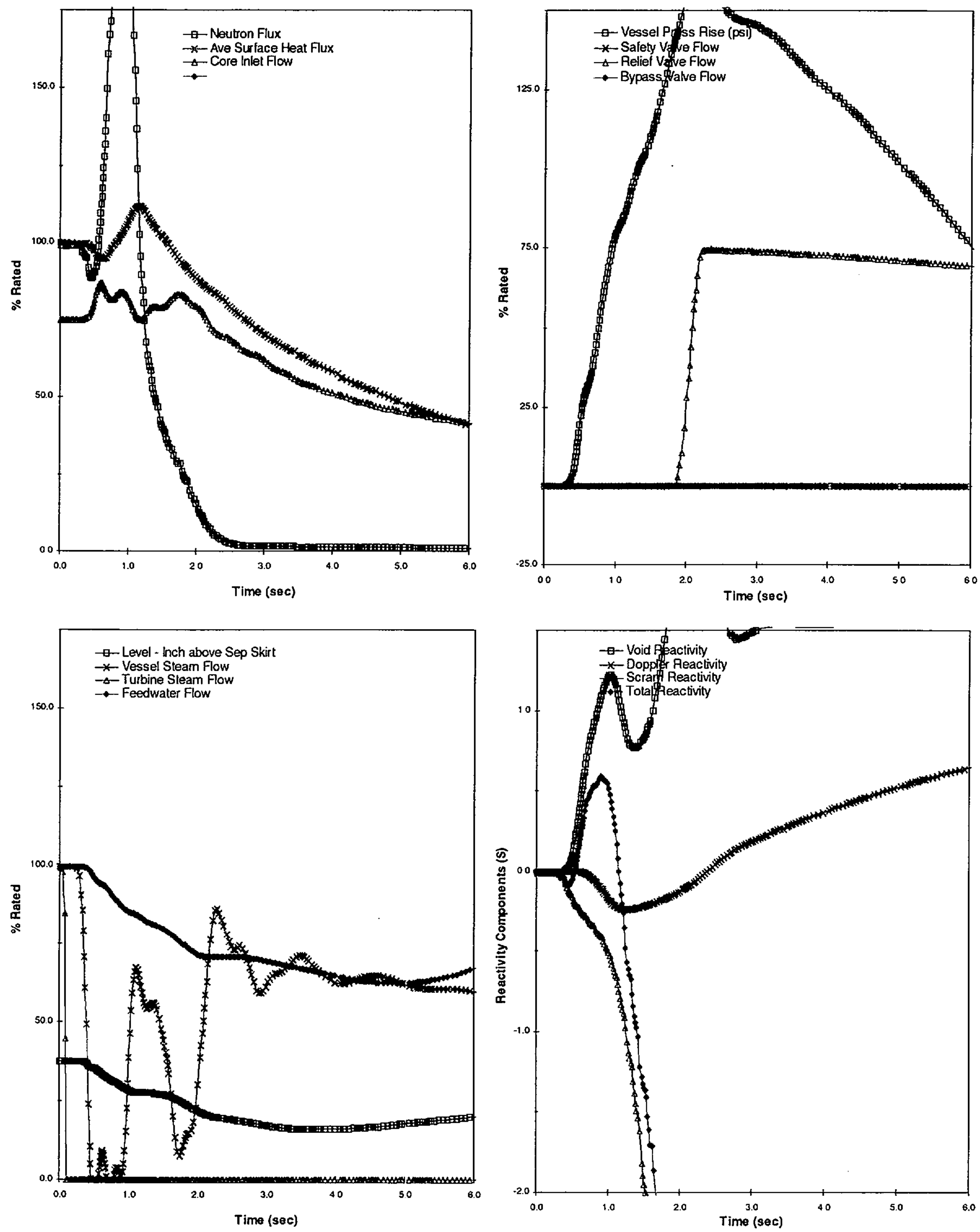


Figure 2-4 Plant Response to Turbine Trip w/o Bypass (EOC, 100% Power, 75% Flow

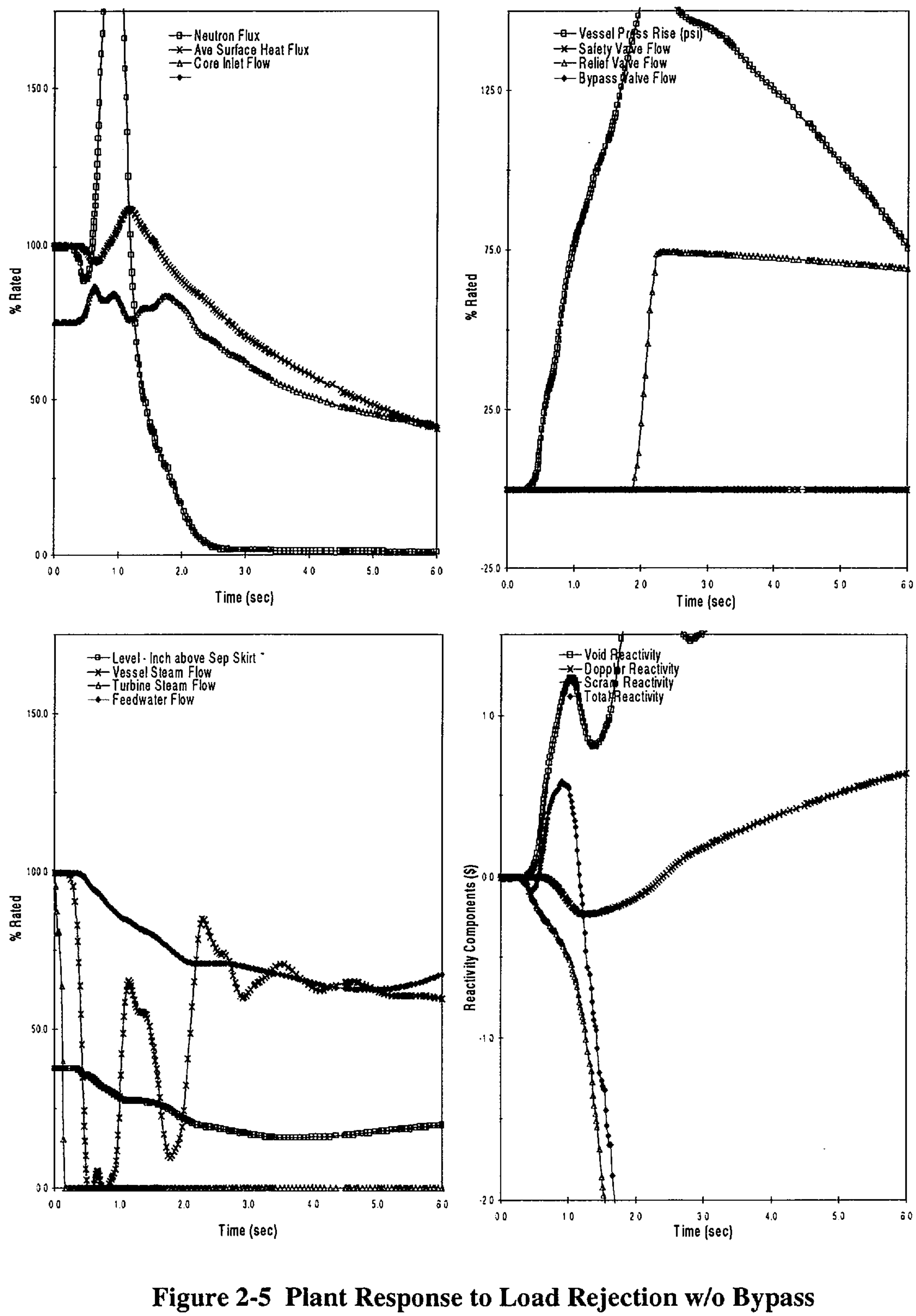


Figure 2-5 Plant Response to Load Rejection w/o Bypass (EOC, 100% Power, 75% Flow)

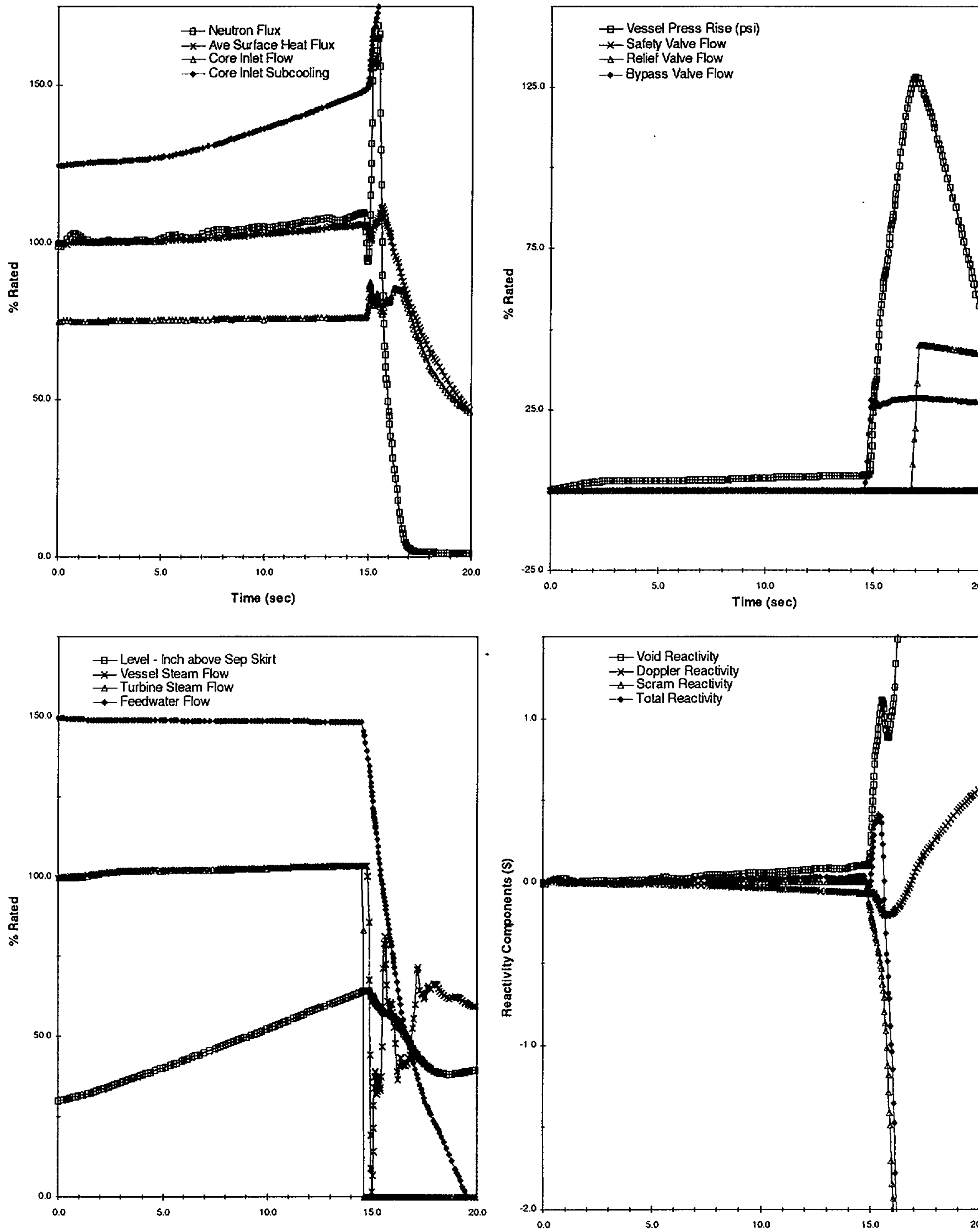


Figure 2-6 Plant Response to Feedwater Controller Failure (EOC, 100% Power, 75% Flow)

3.0 VESSEL OVERPRESSURE PROTECTION ANALYSIS

The Main Steamline Isolation Valve (MSIV) closure with a flux scram event is used to determine the compliance to the ASME Pressure Vessel Code. This event was analyzed at ICF conditions only.

3.1 Analysis Approach and Inputs

The input assumptions, such as reactor protection system setpoints and plant configurations, are consistent with reload analysis.

3.2 Overpressure Protection Results

The results of the analysis for Cycle 20 are shown in Table 3-1. As shown in this table, the peak vessel pressure is 1246 psig with the 8 SRVs and 3 SSVs set 3% above their nominal value. This pressure is well below the 1375 psig ASME code limit. Key system responses for this event are shown in Figure 3-1.

Table 3-1

ASME Pressure Vessel Code Compliance MSIV Closure (Flux Scram)

Initial Power/Flow (% NBR)	Peak Steam Line Pressure (psig)	Peak Vessel Pressure (psig)
"D"	1222	1246

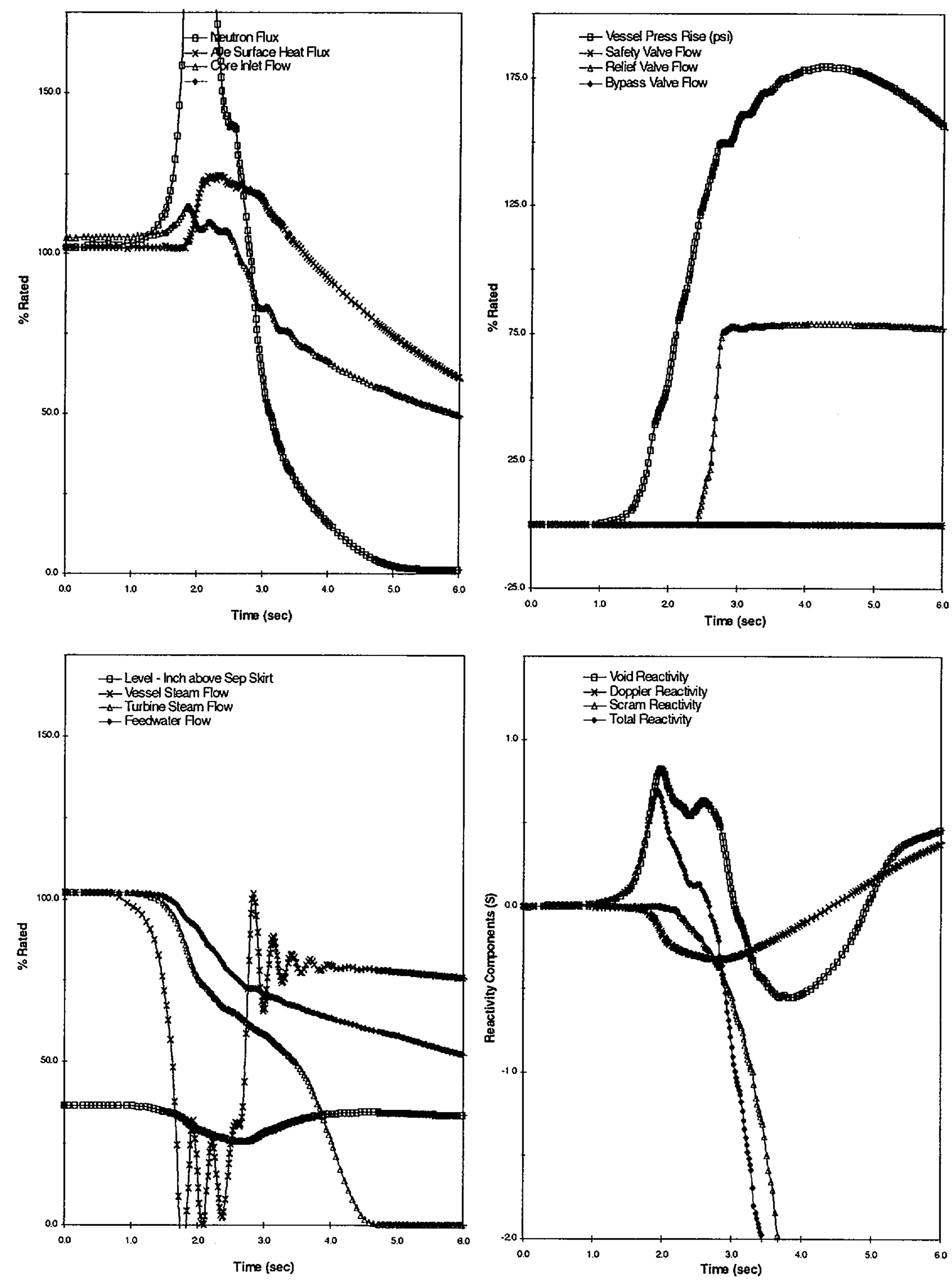


Figure 3-1 Plant Response to MSIV Closure – Flux Scram (EOC, 102% Power, 105% Flow)

4.0 LOSS-OF-COOLANT ACCIDENT ANALYSIS

The Loss-of-Coolant Accident (LOCA) was evaluated for the impact due to MELLL and ICF operation. The change in core flow due to MELLL and ICF operation primarily affects the time and depth of boiling transition in the fuel bundle which, in turn, affects the Peak Cladding Temperature (PCT) response for the event. The lower core flow associated with MELLL may lead to an earlier and deeper boiling transition in the fuel bundle, resulting in a higher PCT. Therefore, the limiting LOCA event was analyzed at the MELLL conditions. The MELLL results are expected to be bounding for ICF operation.

5.0 CONTAINMENT RESPONSE

Operation in the MELLL domain changes some of the conditions assumed for the containment evaluation documented in the CNS Updated Safety Analysis Report (USAR) [Reference 3].

Long-term heatup of the suppression pool following a LOCA is governed by the capability of the residual heat removal system to remove decay heat and sensible energy in the vessel and piping. The decay heat depends upon the reactor rated power level, which remains unchanged with either MELLL or ICF. Therefore, the current long-term containment response documented in Reference 3 is applicable for operation in both the MELLL and ICF domains.

The LOCA containment dynamic loads analysis is based upon the results of the short-term LOCA analysis. The LOCA dynamic loads considered for MELLL and ICF operation include pool swell, condensation oscillation, chugging, SRV discharge, and vent system thrust loads.

5.1 Analysis Approach and Inputs

Short-term containment temperature and pressure response analyses and the containment dynamic loads analysis use the currently approved methods documented in Reference 4. The other analysis inputs are consistent with the original analysis documented in Reference 3.

5.2 Containment Analysis Results

Results of the containment dynamic loads evaluation show that all containment loads remain within the limits previously defined in the CNS Plant Unique Load Definition (PULD) [Reference 5].

6.0 REACTOR INTERNAL PRESSURE DIFFERENCES

Operation in either the MELLL or ICF domain affects the pressure differences across reactor internal components.

Operation with ICF results in higher initial flow velocities relative to rated flow conditions. Thus, ICF causes increased pressure differentials across the reactor internal components for normal, transient (Upset), emergency, and accident (Faulted) conditions.

The internal components in the reactor vessel are subject to the pressure loadings resulting from the hydraulic resistance against the coolant flowing across those components.

6.1 Analysis Approach and Inputs

The impact of MELLL on reactor internal pressure differences (RIPD) is bounded by ICF. Because of higher initial flow velocities, ICF will result in higher Normal conditions RIPD.

Analyses of Normal operating conditions were performed with the steady-state thermal-hydraulic model at 100% power / 105% flow. The inputs used for this analysis are consistent with the original CNS RIPD with the assumption of a full core of the limiting fuel (GE14) for pressure drop consideration.

For Upset conditions, the steady-state (Normal condition) values are conservatively adjusted to obtain the limiting AOO RIPDs. However, the initial steady-state pressure differences at the low flow conditions (MELLL and ELLL) would be smaller than for ICF at the same power level because of the lower initial flow velocity. Consequently, it is bounding to apply the Upset condition adjustment factors to the conservative ICF steady-state results.

Emergency RIPDs were obtained by using the LAMB model to analyze the limiting All Automatic Depressurization System (ADS) Valves Actuation event. The LAMB computer code is documented in Reference 2.

Faulted RIPD values are obtained using the LAMB computer code to analyze the limiting steamline break accident. No MELLL specific calculation is required. This is because MELLL is bounded by ICF. The Faulted condition RIPD calculation for CNS also includes an evaluation at the low power cavitation interlock point (22.5% power / 110% flow).

6.2 RIPD Analysis Results

The results of the RIPD analyses are shown in Table 6-1. These results are used as inputs to the reactor internal structural integrity evaluation. The analysis evaluates the stresses and overall impact on the mechanical integrity of the reactor internal components for the extended operating domain.

Table 6-1

RIPD Results for Normal, Upset, Emergency and Faulted Conditions for CNS in MELLL and ICF Domain

Internal Components	ΔP (psid) ICF Condition	
Core Plate & Guide Tube	22.3	Normal
	24.7	Upset
	24.5	Emergency
	29.0	Faulted
Shroud Support Ring & Lower Shroud	30.1	Normal
	32.5	Upset
	35.0	Emergency
	53.0	Faulted
Upper Shroud	7.9	Normal
	11.8	Upset
	15.7	Emergency
	31.0	Faulted
Shroud Head	8.1	Normal
	12.2	Upset
	14.7	Emergency
	31.0	Faulted
Shroud Head to Water Level, Irreversible ΔP	10.8	Normal
	16.2	Upset
	16.3	Emergency
	32.0	Faulted

Table 6-1 (Continued)

Internal Components	ΔP (psid)	ICF Condition
Shroud Head to Water Level,	1.1	Normal
Elevation ΔP	1.7	Upset
	1.6	Emergency
	2.5	Faulted
Channel Wall,	8.3	Normal
Core Average Power Bundle	11.2	Upset
	9.1	Emergency
	11.0	Faulted
Channel Wall,	10.9	Normal
Maximum Power Bundle	13.8	Upset
	N/A	Emergency
	N/A	Faulted
Channel Wall,	9.2	Normal
Average Central Power Bundle	12.1	Upset
	N/A	Emergency
	N/A	Faulted
Top Guide	0.7	Normal
	1.2	Upset
	1.5	Emergency
	3.7	Faulted
Steam Dryer	0.3	Normal
	0.4	Upset
	N/A	Emergency
	4.0	Faulted

7.0 REACTOR INTERNALS STRUCTURAL INTEGRITY

Changes in loading conditions due to MELLL and ICF operating conditions are evaluated to determine the impact on the structural integrity of the reactor internal components.

7.1 Analysis Approach and Inputs

The evaluation of the key reactor internal Core Support and Non-Core Support Structure components was performed to assess the component structural integrity for the load changes associated with the MELLL and ICF operating conditions.

The original design basis geometry/configuration was assumed for all the components unless the component has undergone permanent structural changes. If the component was permanently changed, then the latest documentation for that component was reviewed and used as the design basis for the component in this assessment

7.2 Structural Evaluation Results

The structural adequacy of the reactor internal components were assessed for the load changes associated with MELLL and ICF, using the original/existing analysis as the design basis. All of the loads and/or resultant stresses at MELLL and ICF are within the design basis allowable values for each of the components reviewed.

8.0 REACTOR INTERNAL VIBRATION

Evaluations of the changes in the flow-induced vibration (FIV) response of critical reactor components within the RPV due to operation in the ICF domain were performed to assure that these vibration responses were within established criteria. Changes in plant conditions associated with MELLL were also considered.

All safety-related reactor internal components, except for two JPSLs, that were evaluated had stresses less than the acceptance criteria at the increased core flow rate condition. Two JPSLs, one per recirculation loop, were found to have a remote possibility for a second natural frequency that could be near the recirculation pump vane passing frequency if five as-built lengths were at their extreme design value tolerances. Therefore, it is highly improbable that these two sensing lines actually have natural frequencies near the maximum pump speed. In any case, a single JPSL is sufficient to detect any jet pump anomaly in a jet pump pair. Thus, failure of one JPSL is not a safety issue.

9.0 REACTOR RECIRCULATION SYSTEM EVALUATION

The Reactor Recirculation System (RRS) provides forced circulation of reactor coolant water up through the reactor core. The scope of this portion of the study is to evaluate the RRS capability to support the MELLL and ICF condition for CNS.

The evaluation concluded that MELLL and ICF will affect the RRS operating pressures and temperatures only to a small extent.

10.0 THERMAL-HYDRAULIC STABILITY

GE has established stability criteria to demonstrate compliance to the requirements set forth in 10CFR50 Appendix A, General Design Criteria (GDC) 10 and 12. The stability compliance of all GE fuel designs is demonstrated on a generic basis for operation in both the MELLL and ICF domains.

CNS is currently operating under the requirements of reactor stability Long-Term Solution Option I-D. Option I-D provides an administratively controlled exclusion region to prevent normal operation where an instability could be expected to occur. The solution also includes a buffer zone defined to be 5% of rated power and rated core flow outside of the exclusion region.

10.1 Analysis Methods and Inputs

The Option I-D calculations consist of two parts: (1) the exclusion region calculation and (2) the SLMCPR protection calculation.

The Option I-D initial application for CNS is documented in Reference 6. The same NRC approved methodology was used for the CNS Cycle 20 reload core, including the GE14 fuel design.

10.2 Stability Analysis Results

Instability Regions: The endpoints of the exclusion region are defined on the MELLL line and on the natural circulation line using the frequency domain analysis methodology. The endpoints of the buffer zone are defined as 5% of rated flow higher along the MELLL line and 5% of rated power lower along the natural circulation line. The endpoints are provided in Table 10-1. The region boundaries are defined using the Generic Shape Function [Reference 6]. The regions are shown on the CNS power/flow map in Figure 1-1.

Minimum Critical Power Ratio Safety Limit (MCPRSL) Protection: The inputs to the hot bundle oscillation magnitude calculation are provided in Table 10-2. The differences from the Reference 6 calculation are the nominal reactor power at natural circulation for the flow-biased APRM flux trip and the average power on rated rod line at natural circulation. This results in a normalized statistical hot bundle oscillation magnitude of 0.959. The corresponding stability-based OLMCPRs are provided in Table 10-3.

Table 10-1
Exclusion Region and Buffer Zone Endpoints

Point	Core Flow (% Rated)	Reactor Power (% Rated)
Exclusion Region:		
MELLL Line	43.2	70.9
Natural Circulation Line	32.5	41.0
Buffer Zone:		
MELLL Line	48.2	75.7
Natural Circulation Line	32.5	36.0

Table 10-2
CNS Cycle 20 Inputs to Hot Bundle Oscillation Magnitude Calculation

Variable	Input Value
Core Size:	548 bundles
Trip System:	Flow-biased APRM
Trip Logic:	One-out-of-two, taken twice
Oscillation Mode:	Core-wide
APRM Channel Failure:	Most responsive APRM channel assumed to fail in 100% of trials
LPRM Failures:	Random per χ^2 distribution
Oscillation Period:	Random per χ^2 distribution
Growth Rate:	Random per χ^2 distribution
Oscillation Overshoot:	Random per distribution
Average Power on Rated Rod Line at Natural Circulation:	50.1 % rated power
Nominal APRM Trip Level at Natural Circulation:	70.0 % rated power
Total Scram Delay Time:	854 msec.
Total Number of LPRMs:	124

Table 10-3

CNS Cycle 20 Stability-Based OLMCPR

Stability-based OLMCPR		**Limiting Plant OLMCPR**	
OLMCPR (2 pump trip)	1.20*(Peak Hot Excess) 1.20*(End of Cycle)	OLMCPR (100/100)	1.23 (GE9 Based)
OLMCPR (steady state)	1.38*	OLMCPR(100/45)	1.45

NOTE: Stability is not limiting as long as the stability based OLMCPR is lower than the limiting plant OLMCPR.

*Value based on Generic DIVOM curve for core wide mode oscillations, Figure 7-3 of Reference 7.

11.0 ANTICIPATED TRANSIENTS WITHOUT SCRAM

Plant Anticipated Transients Without Scram (ATWS) requirements are defined in 10CFR50.62. It requires the plant to have (1) an automatic ATWS recirculation pump trip, (2) an automatic Alternate Rod Insertion (ARI) system, and (3) an 86 gpm equivalent Standby Liquid Control System (SLCS). Criteria have been defined to demonstrate that these requirements are met for plant changes which can affect plant response to ATWS events. The specific criteria are as follows:

- Reactor vessel integrity is maintained (peak vessel pressure is less than ASME Service Level C limit).
- Containment integrity is maintained (peak suppression pool temperature is less than peak suppression pool temperature limit for containment analysis and peak containment pressure is less than containment design pressure).
- Fuel integrity is maintained (peak cladding temperature and peak cladding oxidation are below the corresponding 10CFR50.46 limits).

11.1 Analysis Methods and Inputs

Four events are analyzed:

- Closure of all MSIVs.
- Pressure regulator failure to maximum demand.
- Loss of auxiliary power.
- Inadvertent opening of one SRV.

The first two events have been determined to be limiting for ATWS analysis with ODYN. The two limiting events are analyzed at the limiting MELLL state point for beginning-of-cycle (BOC), middle-of-cycle (MOC) and end-of-cycle (EOC) conditions, since void coefficient changes throughout the cycle affect the plant response to ATWS events..

11.2 ATWS Analysis Results

The calculated results are all within the corresponding ATWS acceptance criteria. Therefore, the associated plant modifications are acceptable against the plant ATWS requirements of 10CFR50.62. Specifically:

- The peak cladding temperature from the bounding ATWS events is well below the 10CFR50.46 limit of 2200°F. Cladding oxidation is not explicitly calculated, but for this PCT and duration, is well below the 10CFR50.46 limit of 17%. Therefore, there is substantial margin relative to maintaining fuel integrity.
- The peak vessel pressure is below the ASME Service Level C limit of 1500 psig and meets the ATWS overpressure criteria.
- The peak suppression pool temperature is below the maximum containment temperature limit of 281°F and below the long-term maximum suppression pool temperature calculated for LOCA conditions. The peak containment pressure is also well below the containment design pressure of 56 psig.

Table 11-1

Key Initial Operating Conditions for ATWS Analysis

Parameter	Value
Dome Pressure, psia	1020
Nuclear Boiler Rated (NBR) Core Flow, Mlb/hr	73.5
Lowest Core Flow at Rated Power, Mlb/hr / %NBR	55.1 / 75
Core Thermal Power, MWt / %NBR	2381 / 100
Steam/Feed Flow, Mlb/hr / %NBR	9.56 / 100
High Dome Pressure RPT Setpoint, Upper Tech. Spec. Limit (psig)	1120
Number of SRV / SSV Operational	8 / 3

12.0 REFERENCES

1. "Extended Load Line Limit and ARTS Improvement Program Analyses for Cooper Nuclear Station Cycle 14", NEDC-31892P, Class III, Rev.1, May 1991.

2. GESTAR II "General Electric Standard Application for Reactor Fuel", NEDE-24011-P-A, Class III, Rev. 13, August 20, 1996.

3. Nebraska Public Power District, Cooper Nuclear Station, Updated Safety Analysis Report (USAR), Docket Number 50-298 [CD-ROM containing the CNS USAR and the Improved Technical Specifications (ITS) as of 5/19/99 enclosed with Letter from Paul Ballinger to Erik Stromqvist dated May 20, 1999].

4. "Generic Evaluations of General Electric Boiling Water Reactor Power Uprate," NEDC-31984P, Class III, July 1991, and Supplements 1 and 2.

5. "Mark I Containment Program Plant Unique Load Definition – Cooper Nuclear Station," NEDO-24573, Revision 2, April 1982.

6. "Application of the 'Regional Exclusion With Flow-Biased APRM Neutron Flux Scram' Stability Solution (Option I-D) to the Cooper Nuclear Station," GENE-A13-00395-01, November 1996.

7. "Reactor Stability Detect & Suppress Solutions Licensing Basis Methodology for Reload Applications," NEDO-32465-A, August 1996.

ATTACHMENT 3

Cooper Nuclear Station, NRC Docket 50-298, DPR-46

General Electric Nuclear Energy Affidavit
Regarding Withholding from Public Disclosure

GE Nuclear Energy

General Electric Company
175 Curtner Avenue, San Jose CA 95125

Feb. 2, 2000		**Action Requested by:**	Feb. 9, 2000
GE-MIG-1H69L-050		**Response to:**	N/A
DRF L12-00867-00		**Project Deliverable:**	Final SAR for MELLL and ICF

cc: K. Cole
J. Fox
D. McNeil

To: Paul Ballinger (NPPD)
From: Erik Stromqvist (GE)
Author: J. Fox (GE)
Subject: MIG Project Task 1104: Maximum Extended Load Line Limit and Increased Core Flow for Cooper Nuclear Station, Revision 0, NEDC-32914P and NEDO-32914

References: 1. NPPD Task Authorization 419 dated May 12, 1999
2. MIG Project Work Plan Rev. 1 dated Sept. 30, 1999

Dear Paul,

In accordance with Reference 1 and 2, this letter documents completion of the subject CNS MIG Project Deliverable. Enclosed please find the subject Report in four copies per your request. Your approval of this report is requested by 2/9/00.

This transmittal contains GE proprietary information, which is provided under the NPPD/GE proprietary information agreement. GE customarily maintains this information in confidence and withholds it from public disclosure.

The attached affidavit identifies that the designated information has been handled and classified as proprietary to GE. Along with the affidavit this information is suitable for review by the NRC. GE hereby requests that the designated information be withheld from public disclosure in accordance with the provisions of 10 CFR 2.790. The sections of the report containing GE-proprietary information are identified with a bar on the right margin.

Included with this transmittal is also a non-proprietary version of the subject Report in four copies per your request.

A signed copy of this Letter is included in DRF L12-00867-00 and the supplement DRF L12-00867-14. Supporting technical information and evidence of verification for the attached reports are contained in DRF L12-00867-14.

Sincerely yours,

Erik Stromqvist

GE MIG Project Manager

Attachment: MELLL and ICF for Cooper Nuclear Station, Revision 0, NEDC-32914P and associated affidavit in original. Non-proprietary version MELLL and ICF for Cooper Nuclear Station, Revision 0, NEDO-32914

General Electric Company

AFFIDAVIT

I, **George B. Stramback,** being duly sworn, depose and state as follows:

(1) I am Project Manager, Regulatory Services, General Electric Company ("GE") and have been delegated the function of reviewing the information described in paragraph (2) which is sought to be withheld, and have been authorized to apply for its withholding.

(2) The information sought to be withheld is contained in the GE proprietary report NEDC-32914P, *Maximum Extended Load Line Limit and Increased Core Flow for Cooper Nuclear Station*, Revision 0, Class III (GE Proprietary Information), dated January 2000. The proprietary information is delineated by bars marked in the margin adjacent to the specific material.

(3) In making this application for withholding of proprietary information of which it is the owner, GE relies upon the exemption from disclosure set forth in the Freedom of Information Act ("FOIA"), 5 USC Sec. 552(b)(4), and the Trade Secrets Act, 18 USC Sec. 1905, and NRC regulations 10 CFR 9.17(a)(4), 2.790(a)(4), and 2.790(d)(1) for "trade secrets and commercial or financial information obtained from a person and privileged or confidential" (Exemption 4). The material for which exemption from disclosure is here sought is all "confidential commercial information", and some portions also qualify under the narrower definition of "trade secret", within the meanings assigned to those terms for purposes of FOIA Exemption 4 in, respectively, Critical Mass Energy Project v. Nuclear Regulatory Commission, 975F2d871 (DC Cir. 1992), and Public Citizen Health Research Group v. FDA, 704F2d1280 (DC Cir. 1983).

(4) Some examples of categories of information which fit into the definition of proprietary information are:

a. Information that discloses a process, method, or apparatus, including supporting data and analyses, where prevention of its use by General Electric's competitors without license from General Electric constitutes a competitive economic advantage over other companies;

b. Information which, if used by a competitor, would reduce his expenditure of resources or improve his competitive position in the design, manufacture, shipment, installation, assurance of quality, or licensing of a similar product;

c. Information which reveals cost or price information, production capacities, budget levels, or commercial strategies of General Electric, its customers, or its suppliers;

d. Information which reveals aspects of past, present, or future General Electric customer-funded development plans and programs, of potential commercial value to General Electric;

e. Information which discloses patentable subject matter for which it may be desirable to obtain patent protection.

The information sought to be withheld is considered to be proprietary for the reasons set forth in both paragraphs (4)a. and (4)b., above.

(5) The information sought to be withheld is being submitted to NRC in confidence. The information is of a sort customarily held in confidence by GE, and is in fact so held. The information sought to be withheld has, to the best of my knowledge and belief, consistently been held in confidence by GE, no public disclosure has been made, and it is not available in public sources. All disclosures to third parties including any required transmittals to NRC, have been made, or must be made, pursuant to regulatory provisions or proprietary agreements which provide for maintenance of the information in confidence. Its initial designation as proprietary information, and the subsequent steps taken to prevent its unauthorized disclosure, are as set forth in paragraphs (6) and (7) following.

(6) Initial approval of proprietary treatment of a document is made by the manager of the originating component, the person most likely to be acquainted with the value and sensitivity of the information in relation to industry knowledge. Access to such documents within GE is limited on a "need to know" basis.

(7) The procedure for approval of external release of such a document typically requires review by the staff manager, project manager, principal scientist or other equivalent authority, by the manager of the cognizant marketing function (or his delegate), and by the Legal Operation, for technical content, competitive effect, and determination of the accuracy of the proprietary designation. Disclosures outside GE are limited to regulatory bodies, customers, and potential customers, and their agents, suppliers, and licensees, and others with a legitimate need for the information, and then only in accordance with appropriate regulatory provisions or proprietary agreements.

(8) The information identified in paragraph (2), above, is classified as proprietary because it contains detailed results of analytical models, methods and processes, including computer codes, which GE has developed, obtained NRC approval of, and applied to perform evaluations of maximum extended load line limits (MELLL) and increased core flow (ICF) for BWRs.

The development and approval of the BWR analysis computer codes used in this analysis was achieved at a significant cost, on the order of several million dollars, to GE.

The development of the evaluation process along with the interpretation and application of the analytical results is derived from the extensive experience database that constitutes a major GE asset.

(9) Public disclosure of the information sought to be withheld is likely to cause substantial harm to GE's competitive position and foreclose or reduce the availability of profit-making opportunities. The information is part of GE's comprehensive BWR safety and technology base, and its commercial value extends beyond the original development cost. The value of the technology base goes beyond the extensive physical database and analytical methodology and includes development of the expertise to determine and apply the appropriate evaluation process. In addition, the technology base includes the value derived from providing analyses done with NRC-approved methods.

The research, development, engineering, analytical and NRC review costs comprise a substantial investment of time and money by GE.

The precise value of the expertise to devise an evaluation process and apply the correct analytical methodology is difficult to quantify, but it clearly is substantial.

GE's competitive advantage will be lost if its competitors are able to use the results of the GE experience to normalize or verify their own process or if they are able to claim an equivalent understanding by demonstrating that they can arrive at the same or similar conclusions.

The value of this information to GE would be lost if the information were disclosed to the public. Making such information available to competitors without their having been required to undertake a similar expenditure of resources would unfairly provide competitors with a windfall, and deprive GE of the opportunity to exercise its competitive advantage to seek an adequate return on its large investment in developing these very valuable analytical tools.

STATE OF CALIFORNIA)
) ss:
COUNTY OF SANTA CLARA)

George B. Stramback, being duly sworn, deposes and says:

That he has read the foregoing affidavit and the matters stated therein are true and correct to the best of his knowledge, information, and belief.

Executed at San Jose, California, this 2nd day of February 2000.

George B. Stramback
General Electric Company

Subscribed and sworn before me this 2ND day of FEBRUARY 2000.

Notary Public, State of California

ANNA HANLIN
Commission # 1184507
Notary Public - California
Santa Clara County
My Comm. Expires Jun 19, 2002