ENCLOSURE 1

PROPOSED TECHNICAL SPECIFICATION

BROWNS FERRY NUCLEAR PLANT

UNIT 2

(TVA BFN TS 285)

UNIT 2
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SAFETY LIMIT	LIMITING SAFETY SYSTEM SETTING

1.1 FUEL CLADDING INTEGRITY

Applicability

Applies to the interrelated variables associated with fuel thermal behavior.

Objective

To establish limits which ensure the integrity of the fuel cladding.

Specifications

A. Thermal Power Limits

1. Reactor Pressure >800 psia and Core Flow > 10% of Rated.

   When the reactor pressure is greater than 800 psia, the existence of a minimum critical power ratio (MCPR) less than 1.07 shall constitute violation of the fuel cladding integrity safety limit.

2.1 FUEL CLADDING INTEGRITY

Applicability

Applies to trip settings of the instruments and devices which are provided to prevent the reactor system safety limits from being exceeded.

Objective

To define the level of the process variables at which automatic protective action is initiated to prevent the fuel cladding integrity safety limit from being exceeded.

Specifications

The limiting safety system settings shall be as specified below:

A. Neutron Flux Trip Settings

1. APRM Flux Scram Trip Setting (RUN Mode) (Flow Biased)

   a. When the Mode Switch is in the RUN position, the APRM flux scram trip setting shall be:

SAFETY LIMIT	LIMITING SAFETY SYSTEM SETTING
	2.1.A Neutron Flux Trip Settings

2.1.A.1.a (Cont'd)

$$S \leq (0.58W + 62\%)$$

where:

S = Setting in percent of rated thermal power (3293 MWt)

W = Loop recirculation flow rate in percent of rated (rated loop recirculation flow rate equals 34.2×10^6 lb/hr)

b. For no combination of loop recirculation flow rate and core thermal power shall the APRM flux scram trip setting be allowed to exceed 120% of rated thermal power.

SAFETY LIMIT

LIMITING SAFETY SYSTEM SETTING

2.1.A Neutron Flux Trip Settings

2.1.A.1.b. (Cont'd)

NOTE: These settings assume operation within the basic thermal hydraulic design criteria. These criteria are LHGR $\leq$13.4 kW/ft and MCPR within limits of Specification 3.5.K. If it is determined that either of these design criteria is being violated during operation, action shall be initiated within 15 minutes to restore operation within prescribed limits. Surveillance requirements for APRM scram setpoint are given in Specification 4.5.L.

c. The APRM Rod Block trip setting shall be:

$$S_{RB} \leq (0.58W + 50\%)$$

where:

S_{RB} = Rod Block setting in percent of rated thermal power (3293 MWt)

W = Loop recirculation flow rate in percent of rated (rated loop recirculation flow rate equals 34.2×10^6 lb/hr)

SAFETY LIMIT

1.1.A Thermal Power Limits

2. Reactor Pressure $\leq$800 psia or Core Flow $\leq$10% of rated.

When the reactor pressure is $\leq$800 psia or core flow is $\leq$10% of rated, the core thermal power shall not exceed 823 MWt (25% of rated thermal power).

LIMITING SAFETY SYSTEM SETTING

2.1.A Neutron Flux Trip Settings (Cont'd)

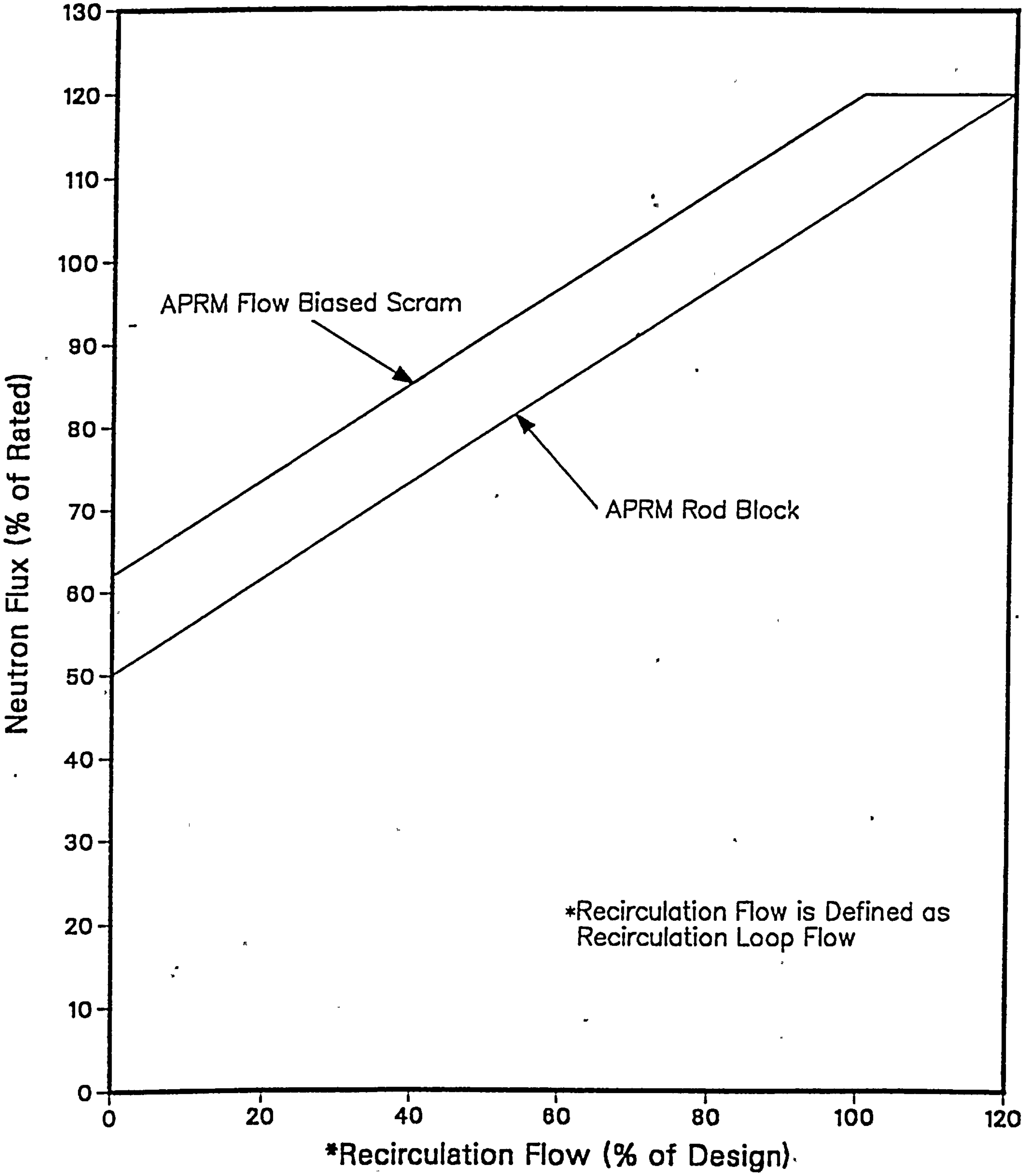
d. Fixed High Neutron Flux Scram Trip Setting--When the mode switch is in the RUN position, the APRM fixed high flux scram trip setting shall be:

S$\leq$120% power.

2. APRM and IRM Trip Settings (Startup and Hot Standby Modes).

a. APRM--When the reactor mode switch is in the STARTUP position, the APRM scram shall be set at less than or equal to 15% of rated power.

b. IRM--The IRM scram shall be set at less than or equal to 120/125 of full scale.



APRM Flow Reference Scram and APRM Rod Block Settings
Fig. 2.1-1

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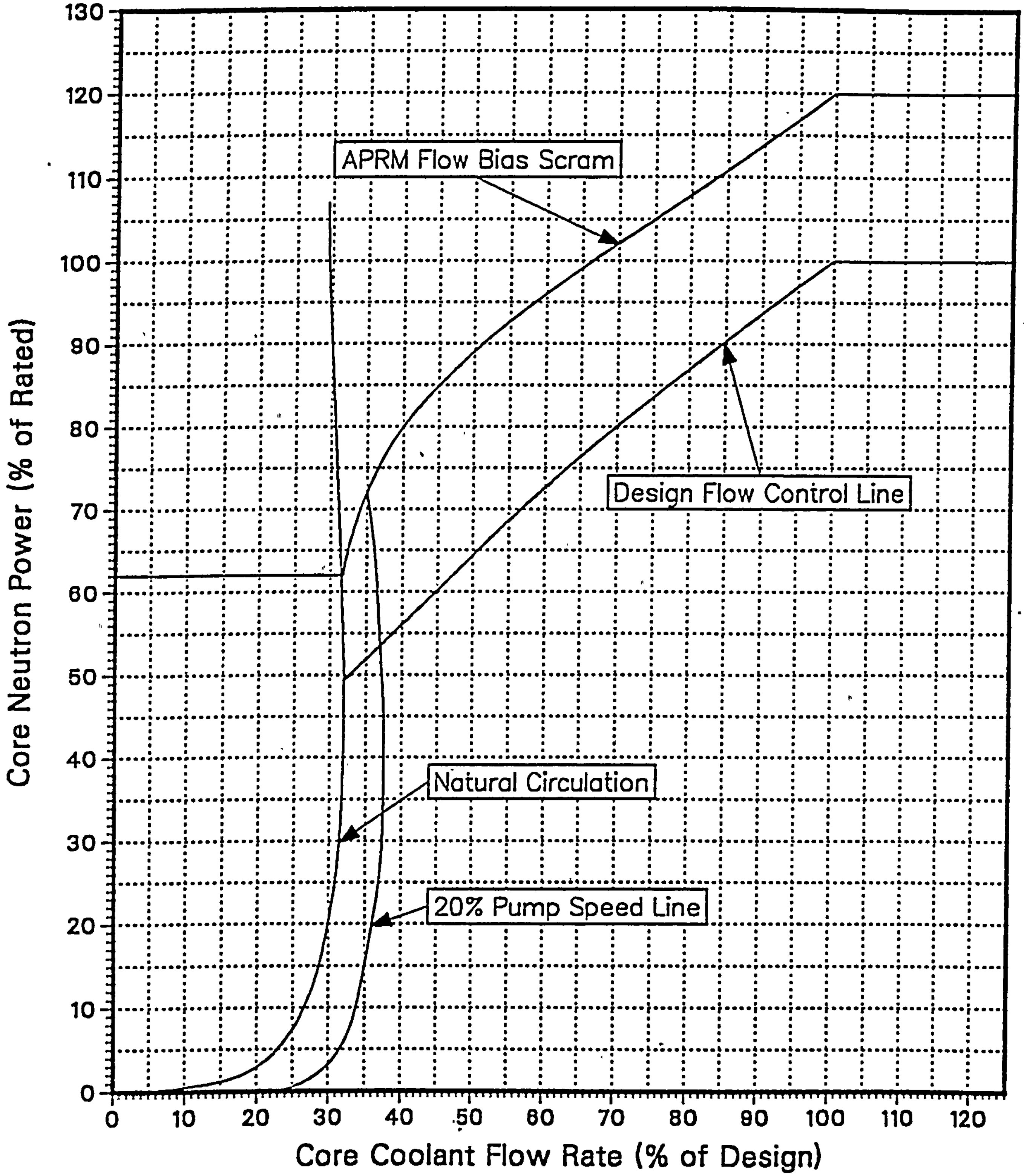

APRM Flow Bias Scram vs. Reactor Core Flow
Fig. 2.1-2

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In summary

1. The licensed maximum power level is 3,293 MWt.

2. Analyses of transients employ adequately conservative values of the controlling reactor parameters.

3. The abnormal operational transients were analyzed to a power level of 3,440 MWt.

4. The analytical procedures now used result in a more logical answer than the alternative method of assuming a higher starting power in conjunction with the expected values for the parameters.

The bases for individual setpoints are discussed below:

A. Neutron Flux Scram

1. APRM Flow-Biased High Flux Scram Trip Setting (RUN Mode)

The average power range monitoring (APRM) system, which is calibrated using heat balance data taken during steady-state conditions, reads in percent of rated power (3,293 MWt). Because fission chambers provide the basic input signals, the APRM system responds directly to core average neutron flux.

During power increase transients, the instantaneous fuel surface heat flux is less than the instantaneous neutron flux by an amount depending upon the duration of the transient and the fuel time constant. For this reason, the flow-biased scram APRM flux signal is passed through a filtering network with a time constant which is representative of the fuel time constant. As a result of this filtering, APRM flow-biased scram will occur only if the neutron flux signal is in excess of the setpoint and of sufficient time duration to overcome the fuel time constant and result in an average fuel·surface heat flux which is equivalent to the neutron flux trip setpoint. This setpoint is variable up to 120 percent of rated power based on recirculation drive flow according to the equations given in Section 2.1.A.1 and the graph in Figure 2.1-2. For the purpose of licensing transient analysis, neutron flux scram is assumed to occur at 120 percent of rated power. Therefore, the flow biased scram provides additional margin to the thermal limits for slow transients such as loss of feedwater heating. No safety credit is taken for flow-biased scrams.

2.1 BASES (Cont'd)

Analyses of the limiting transients show that no scram adjustment is required to assure MCPR > 1.07 when the transient is initiated from MCPR limits specified in Specification 3.5.k.

2. APRM Flux Scram Trip Setting (Refuel or Start & Hot Standby Mode)

For operation in the startup mode while the reactor is at low pressure, the APRM scram setting of 15 percent of rated power provides adequate thermal margin between the setpoint and the safety limit, 25 percent of rated. The margin is adequate to accommodate anticipated maneuvers associated with power plant startup. Effects of increasing pressure at zero or low void content are minor, cold water from sources available during startup is not much colder than that already in the system, temperature coefficients are small, and control rod patterns are constrained to be uniform by operating procedures backed up by the rod worth minimizer and the Rod Sequence Control System. Thus, of all possible sources of reactivity input, uniform control rod withdrawal is the most probable cause of significant power rise. Because the flux distribution associated with uniform rod withdrawals does not involve high local peaks, and because several rods must be moved to change power by a significant percentage of rated power, the rate of power rise is very slow. Generally, the heat flux is in near equilibrium with the fission rate. In an assumed uniform rod withdrawal approach to the scram level, the rate of power rise is no more than five percent of rated power per minute, and the APRM system would be more than adequate to assure a scram before the power could exceed the safety limit. The 15 percent APRM scram remains active until the mode switch is placed in the RUN position. This switch occurs when reactor pressure is greater than 850 psig.

3. IRM Flux Scram Trip Setting

The IRM System consists of eight chambers, four in each of the reactor protection system logic channels. The IRM is a five-decade instrument which covers the range of power level between that covered by the SRM and the APRM. The five decades are covered by the IRM by means of a range switch and the five decades are broken down into 10 ranges, each being one-half of a decade in size. The IRM scram setting of 120 divisions is active in each range of the IRM. For example, if the instrument were on range 1, the scram setting would be at 120 divisions for that range; likewise if the instrument was on range 5, the scram setting would be 120 divisions on that range.

IRM Flux Scram Trip Setting (Continued)

Thus, as the IRM is ranged up to accommodate the increase in power level, the scram setting is also ranged up. A scram at 120 divisions on the IRM instruments remains in effect as long as the reactor is in the startup mode. In addition, the APRM 15 percent scram prevents higher power operation without being in the RUN mode. The IRM scram provides protection for changes which occur both locally and over the entire core. The most significant sources of reactivity change during the power increase are due to control rod withdrawal. For insequence control rod withdrawal, the rate of change of power is slow enough due to the physical limitation of withdrawing control rods that heat flux is in equilibrium with the neutron flux. An IRM scram would result in a reactor shutdown well before any safety limit is exceeded. For the case of a single control rod withdrawal error, a range of rod withdrawal accidents was analyzed. This analysis included starting the accident at various power levels. The most severe case involves an initial condition in which the reactor is just subcritical and the IRM system is not yet on scale. This condition exists at quarter rod density. Quarter rod density is illustrated in paragraph 7.5.5 of the FSAR. Additional conservatism was taken in this analysis by assuming that the IRM channel closest to the withdrawn rod is bypassed. The results of this analysis show that the reactor is scrammed and peak power limited to one percent of rated power, thus maintaining MCPR above 1.07. Based on the above analysis, the IRM provides protection against local control rod withdrawal errors and continuous withdrawal of control rods in sequence.

4. Fixed High Neutron Flux Scram Trip

The average power range monitoring (APRM) system, which is calibrated using heat balance data taken during steady-state conditions, reads in percent of rated power (3,293 MWt). The APRM system responds directly to neutron flux. Licensing analyses have demonstrated that with a neutron flux scram of 120 percent of rated power, none of the abnormal operational transients analyzed violate the fuel safety limit and there is a substantial margin from fuel damage.

B. APRM Control Rod Block

Reactor power level may be varied by moving control rods or by varying the recirculation flow rate. The APRM system provides a control rod block to prevent rod withdrawal beyond a given point at constant recirculation flow rate and thus to protect against the condition of a MCPR less than 1.07. This rod block trip setting, which is automatically varied with recirculation loop flow rate, prevents an increase in the reactor power level to excess values due to control rod withdrawal. The flow variable trip setting provides substantial margin from fuel damage, assuming a steady-state operation at the trip setting over the entire power/flow domain, including above the rated rod line (Reference 3). The margin to the Safety Limit increases as the flow decreases for the specified trip setting versus flow relationship; therefore, the worst case MCPR which could occur during steady-state operation is at 108 percent of rated thermal power because of the APRM rod block trip setting. The actual power distribution in the core is established by specified control rod sequences and is monitored continuously by the incore LPRM system.

C. Reactor Water Low Level Scram and Isolation (Except Main Steamlines)

The setpoint for the low level scram is above the bottom of the separator skirt. This level has been used in transient analyses dealing with coolant inventory decrease. The results reported in FSAR Subsection 14.5 show that scram and isolation of all process lines (except main steam) at this level adequately protects the fuel and the pressure barrier, because MCPR is greater than 1.07 in all cases, and system pressure does not reach the safety valve settings. The scram setting is sufficiently below normal operating range to avoid spurious scrams.

D. Turbine Stop Valve Closure Scram

The turbine stop valve closure trip anticipates the pressure, neutron flux and heat flux increases that would result from closure of the stop valves. With a trip setting of 10 percent of valve closure from full open, the resultant increase in heat flux is such that adequate thermal margins are maintained even during the worst case transient that assumes the turbine bypass valves remain closed. (Reference 2)

E. Turbine Control Valve Fast Closure or Turbine Trip Scram

Turbine control valve fast closure or turbine trip scram anticipates the pressure, neutron flux, and heat flux increase that could result from control valve fast closure due to load rejection or control valve closure due to turbine trip; each without bypass valve capability. The reactor protection system initiates a scram in less than 30 milliseconds after the start of control valve fast closure due to load rejection or control valve closure due to turbine trip. This scram is achieved by rapidly reducing hydraulic control oil pressure at the main turbine control valve actuator disc dump valves. This loss of pressure is sensed by pressure switches whose contacts form the one-out-of-two-twice logic input to the reactor protection system. This trip setting, a nominally 50 percent greater closure time and a different valve characteristic from that of the turbine stop valve, combine to produce transients very similar to that for the stop valve. No significant change in MCPR occurs. Relevant transient analyses are discussed in References 2 and 3 of the Final Safety Analysis Report. This scram is bypassed when turbine steam flow is below 30 percent of rated, as measured by turbine first state pressure.

2.1 BASES (Cont'd)

F. (Deleted)

G. & H. Main Steamline Isolation on Low Pressure and Main Steam Line Isolation Scram

The low pressure isolation of the main steamlines at 825 psig was provided to protect against rapid reactor depressurization and the resulting rapid cooldown of the vessel. The scram feature that occurs when the main steamline isolation valves close shuts down the reactor so that high power operation at low reactor pressure does not occur, thus providing protection for the fuel cladding integrity safety limit. Operation of the reactor at pressures lower than 825 psig requires that the reactor mode switch be in the STARTUP position, where protection of the fuel cladding integrity safety limit is provided by the IRM and APRM high neutron flux scrams. Thus, the combination of main steamline low pressure isolation and isolation valve closure scram assures the availability of neutron flux scram protection over the entire range of applicability of the fuel cladding integrity safety limit. In addition, the isolation valve closure scram anticipates the pressure and flux transients that occur during normal or inadvertent isolation valve closure. With the scrams set at 10 percent of valve closure, neutron flux does not increase.

I.J.& K. Reactor Low Water Level Setpoint for Initiation of HPCI and RCIC Closing Main Steam Isolation Valves, and Starting LPCI and Core Spray Pumps.

These systems maintain adequate coolant inventory and provide core cooling with the objective of preventing excessive clad temperatures. The design of these systems to adequately perform the intended function is based on the specified low level scram setpoint and initiation setpoints. Transient analyses reported in Section 14 of the FSAR demonstrate that these conditions result in adequate safety margins for both the fuel and the system pressure.

L. References

1. "BWR Transient Analysis Model Utilizing the RETRAN Program," TVA-TR81-01-A.

2. Generic Reload Fuel Application, Licensing Topical Report NEDE-20411-P-A, and Addenda.

3. Browns Ferry Nuclear Plant Unit 2, Cycle 6, Licensing Report, Extended Load Line Limit Analysis, TVA-BFE-052, April, 1990.

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TABLE 3.2.C
INSTRUMENTATION THAT INITIATES ROD BLOCKS

Minimum Operable Channels Per Trip Function (5)	Function	Trip Level Setting
4(1)	APRM Upscale (Flow Bias)	≤0.58W + 50% (2)
4(1)	APRM Upscale (Startup Mode) (8)	≤12%
4(1)	APRM Downscale (9)	≥3%
4(1)	APRM Inoperative	(10b)
2(7)	RBM Upscale (Flow Bias)	≤0.66W + 40% (2)(13)
2(7)	RBM Downscale (9)	≥3%
2(7)	RBM Inoperative	(10c)
6(1)	IRM Upscale (8)	≤108/125 of full scale
6(1)	IRM Downscale (3)(8)	≥5/125 of full scale
6(1)	IRM Detector not in Startup Position (8)	(11)
6(1)	IRM Inoperative (8)	(10a)
3(1) (6)	SRM Upscale (8)	≤ $1X10^5$ counts/sec.
3(1) (6)	SRM Downscale (4)(8)	≥3 counts/sec.
3(1) (6)	SRM Detector not in Startup Position (4)(8)	(11)
3(1) (6)	SRM Inoperative (8)	(10a)
2(1)	Flow Bias Comparator	≤10% difference in recirculation flows
2(1)	Flow Bias Upscale	≤115% recirculation flow
1	Rod Block Logic	N/A
2(1)	RCSC Restraint (PS85-61A,B)	147 psig turbine first stage pressure
1(12)	High Water Level in West Scram Discharge Tank (LS-85-45L)	≤25 gal.
1(12)	High Water Level in East Scram Discharge Tank (LS-85-45M)	≤25 gal.

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LIMITING CONDITIONS FOR OPERATION

3.5 Core and Containment Cooling Systems

L. APRM Setpoints

1. Whenever the core thermal power is $\geq$ 25% of rated, the ratio of FRP/CMFLPD shall be $\geq$ 1.0, or the APRM scram and rod block setpoint equations listed in Section 2.1.A shall be multiplied by FRP/CMFLPD as follows:

$$S \leq (0.58W + 62\%) \left(\frac{FRP}{CMFLPD}\right)$$

$$S_{RB} \leq (0.58W + 50\%) \left(\frac{FRP}{CMFLPD}\right)$$

2. When it is determined that 3.5.L.1 is not being met, 6 hours is allowed to correct the condition.

3. If 3.5.L.1 and 3.5.L.2 cannot be met, the reactor power shall be reduced to $\leq$ 25% of rated thermal power within 4 hours.

M. Core Thermal-Hydraulic Stability

1. The reactor shall not be operated at a thermal power and core flow inside of Regions I and II of Figure 3.5.M-1.

2. If Region I of Figure 3.5.M-1 is entered, immediately initiate a manual scram.

3. If Region II of Figure 3.5.M-1 is entered:

SURVEILLANCE REQUIREMENTS

4.5 Core and Containment Cooling Systems

L. APRM Setpoints

FRP/CMFLPD shall be determined daily when the reactor is $\geq$ 25% of rated thermal power.

M. Core Thermal-Hydraulic Stability

1. Verify that the reactor is outside of Region I and II of Figure 3.5.M-1:

   a. Following any increase of more than 5% rated thermal power while initial core flow is less than 45% of rated, and

   b. Following any decrease of more than 10% rated core flow while initial thermal power is greater than 40% of rated.

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENTS
3.5 Core and Containment Cooling Systems	4.5 Core and Containment Cooling Systems

3.5.M.3. (Cont'd)

a. Immediately initiate action and exit the region within 2 hours by inserting control rods or by increasing core flow (starting a recirculation pump to exit the region is not an appropriate action), and

b. While exiting the region, immediately initiate a manual scram if thermal-hydraulic instability is observed, as evidenced by APRM oscillations which exceed 10 percent peak-to-peak of rated or LPRM oscillations which exceed 30 percent peak-to-peak of scale. If periodic LPRM upscale or downscale alarms occur, immediately check the APRM's and individual LPRM's for evidence of thermal-hydraulic instability.

AMENDMENT NO. 174

ENCLOSURE 2

SUMMARY OF CHANGES

1a. Revision to Limiting Safety System Setting (LSSS) 2.1.A.1.a.

Existing LSSS 2.1.A.1.a reads:

"When the Mode Switch is in the RUN position, the APRM flux scram trip setting shall be:

$$S \leq (0.66W + 54\%)"$$

Proposed change to LSSS 2.1.A.1.a would read:

"When the Mode Switch is in the RUN position, the APRM flux scram trip setting shall be:

$$S \leq (0.58W + 62\%)"$$

b. Revision to LSSS 2.1.A.1.c.

Existing LSSS 2.1.A.1.c reads:

"The APRM Rod Block trip setting shall be:

$$S_{RB} \leq (0.66W + 42\%)"$$

Proposed change to LSSS 2.1.A.1.c would read:

"The APRM Rod Block trip setting shall be:

$$S_{RB} \leq (0.58W + 50\%)"$$

2. Replace Figures 2.1-1 and 2.1-2 with the enclosed revisions.

3a. Revision to Bases Section 2.1.A.1 (APRM Flow-Biased High Flux Scram Trip Setting [RUN Mode]).

Replace ". . . During transients, the instantaneous fuel surface heat flux . . ." with ". . . During power increase transients, the instaneous fuel surface heat flux . . .".

b. Revision to Bases Section 2.1.A.1 (APRM Flow-Biased High Flux Scram Trip Setting [RUN Mode]).

Replace ". . . Therefore, the flow-biased provides . . ." with ". . . Therefore, the flow-biased scram provides . . .".

c. Revision to Bases Section 2.1.A.3 (IRM Flux Scram Trip Setting)

Replace ". . . heat flux is in equilibrium with the neutron flux, and an IRM scram would result in a reactor shutdown . . ." with ". . . heat flux is in equilibrium with the neutron flux. An IRM scram would result in a reactor shutdown . . .".

d. Revision to Bases Section 2.1.B (APRM Control Rod Block).

Replace ". . . over the entire recirculation flow range." with ". . . over the entire power/flow domain, including above the rated rod line (Reference 3)."

e. Revision to Bases Section 2.1.G & H (Main Steam Line Isolation on Low Pressure and Main Steam Line Isolation Scram)

Replace ". . . Advantage is taken of the scram feature that occurs when the main steam line isolation valves are closed to provide for reactor shutdown so that high power operation . . ." with " . . . The scram feature that occurs when the main steam line isolation valves close shuts down the reactor so that high power operation . . .".

f. Revision to Bases Section 2.1.L (References).

Add the following reference in Section L:

"3. Browns Ferry Nuclear Plant Unit 2, Cycle 6, Licensing Report, Extended Load Line Limit Analysis, TVA-BFE-052, April, 1990."

4. Revision to Table 3.2.C (Instrumentation that Initiates Rod Blocks).

Change the APRM Upscale (Flow Bias) trip level setting from "$\leq$0.66W + 42%" to "$\leq$0.58W + 50%."

5. Revision to Limiting Condition for Operation (LCO) 3.5.L.1.

Existing LCO 3.5.L.1 reads:

"1. Whenever . . . the ratio of . . . the APRM scram and rod block setpoint equations listed in Sections 2.1.A and 2.1.B shall be multiplied by FRP/CMFLPD as follows:

$$S \leq (0.66W + 54\%)\ (FRP/CMFLPD)$$

$$S_{RB} \leq (0.66W + 42\%)\ (FRP/CMFLPD)"$$

Proposed change to LCO 3.5.L.1 would read

"1. Whenever . . . the ratio of . . . the APRM scram and rod block setpoint equations listed in Section 2.1.A shall be multiplied by FRP/CMFLPD as follows:

$$S \leq (0.58W + 62\%)\ (FRP/CMFLPD)$$

$$S_{RB} \leq (0.58W + 50\%)\ (FRP/CMFLPD)"$$

ENCLOSURE 3

REASON AND JUSTIFICATION FOR THE PROPOSED CHANGES

REASON FOR CHANGE

The BFN unit 2 technical specifications, as described below, are being revised to allow operation in the region bounded by the power/flow line defined by 0.58Wd + 50%, the rated power line, and the rated load line. Specifically:

1. Limiting Safety System Settings (LSSS) 2.1.A.1.a and 2.1.A.1.c are being revised to specify new equations for the flow-biased APRM scram and rod block setpoints.

2. Figures 2.1-1 and 2.1-2 are being changed to show the revised flow-biased scram and rod block lines.

3. The Bases Section 2.1 is being revised to include the increased power/flow domain for which it is applicable, to reference the supporting licensing report, to correct typographical errors, and to make editorial changes in the text.

4. Table 3.2.C is being changed to show the revised APRM upscale trip level setting.

5. Limiting Condition for Operation (LCO) 3.5.L.1 is being revised to correct a typographical error and to specify new equations for the APRM flow-biased scram and rod block setpoints as modified by the ratio of fraction of rated power (FRP) to core maximum fraction of limiting power density (CMFLPD).

JUSTIFICATION FOR CHANGES

The current Browns Ferry FSAR and reload licensing amendment justify operation in a region bounded by the rated power line up to 100% power. LSSS 2.1.A.1.a (flow-biased APRM scram) and LSSS 2.1.A.1.c (flow-biased APRM rod block) constrain operation at less than rated conditions such that the safety analyses initiated from the licensing basis conditions (104.3% power at 105% flow) are bounding for operation in the defined power/flow operating domain. LCO 3.5.L.1 further restricts operation by reducing the flow-biased scram and rod block setpoints by the ratio of FRP/CMFLPD to compensate for increased power peaking at off-rated conditions such as during startup.

Although the flow-biased rod block and scram setpoints constrain operation, no credit is taken for the flow-biased scram in the reference licensing analyses. That is, transient events initiated from less than rated conditions are assumed to be ultimately terminated by the fixed 120% flux scram or other safety-grade scram signals. Previous sensitivity studies have shown that events initiated from less than rated conditions are less severe than events initiated from the licensing basis conditions.

The proposed changes to the LSSSs and the LCO are justified by the extended load line limit analysis (ELLLA) (Enclosure 5). This analysis shows that operation within the extended load line region is either bounded by the reference licensing safety analyses or the results are less than the design

safety limits. The proposed change to the flow-biased scram setpoint equation is being made to maintain the same margin between the rod block and scram setpoints that currently exists. Since no credit is taken for the flow-biased scram in the referenced licensing analyses or the ELLLA, this change will not impact any margin of safety.

Additionally, typographical errors in Bases Section 2.1.A.1 and LCO 3.5.L.1 are being corrected. The error in the bases is described in item 3.b of Enclosure 2. The error in the LCO refers back to the flow-biased scram and rod block equations in Sections 2.1.A and 2.1.B, however both equations are contained in Section 2.1.A. Editorial changes are made to the text of Bases Section 2.1 which do not affect the intent. The Bases are also being revised to reference the licensing report which supports this change. That report is included in Enclosure 5.

ENCLOSURE 4

PROPOSED DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATION

DESCRIPTION OF PROPOSED TECHNICAL SPECIFICATION AMENDMENT

BFN unit 2 technical specifications (TSs) are being revised to allow operation in an expanded power/flow region as follows:

1. Limiting Safety System Settings (LSSS) 2.1.A.1.a and 2.1.A.1.c are being revised to specify new equations for the flow-biased APRM scram and rod block setpoints.

2. Figures 2.1-1 and 2.1-2 are being changed to show the revised flow-biased scram and rod block lines.

3. The Bases Section 2.1 is being revised to include the increased power/flow domain for which it is applicable, to reference the supporting licensing report, to correct typographical errors, and to make editorial changes in the text.

4. Table 3.2.C is being changed to show the revised APRM upscale trip level setting.

5. Limiting Condition for Operation (LCO) 3.5.L.1 is being revised to correct a typographical error and to specify new equations for the APRM flow-biased scram and rod block setpoints as modified by the ratio of fraction of rated power (FRP) to core maximum fraction of limiting power density (CMFLPD).

BASIS FOR PROPOSED NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

NRC has provided standards for determining whether a significant hazards consideration exists as stated in 10 CFR 50.92(c). A proposed amendment to an operating license involves no significant hazards consideration if operation of the facility in accordance with the proposed amendment would not (1) involve a significant increase in the probability or consequences of an accident previously evaluated, or (2) create the possibility of a new or different kind of accident from an accident previously evaluated, or (3) involve a significant reduction in margin of safety.

1. The proposed change does not involve a significant increase in the probability or consequences of accident previously evaluated.

   The proposed change will expand the operating domain to allow operation in a region of higher core power versus core flow up to rated power conditions. The extended load line limit analyses (ELLLA) considered the effects of the change on previously evaluated accidents. The ELLLA showed that the results of these events meet the limiting safety design criteria. Furthermore, the proposed change will not affect the operability of safety-related equipment necessary to mitigate the effects of design basis accidents. Therefore, the change will not significantly increase the probability or consequences of accidents previously evaluated.

2. The proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

   The proposed change does not require the addition of any new equipment to the plant design or require any existing equipment to operate in a different manner from which it was designed to operate. The plant operating domain is being expanded slightly by changing the APRM flow-biased rod block and scram setpoints. However, the plant design basis, 105% steam flow at 100% core flow, is not changed.

3. The proposed change does not involve a significant reduction in a margin of safety.

   The proposed change does not affect the ability of the plant safety related trips or equipment to perform their intended functions. Although the flow-biased APRM scram setpoint is being slightly increased, no credit for this scram is considered in the licensing basis or the ELLLA. The APRM flow-biased scram serves as an additional scram over and above those required to maintain the margin of safety.

TVA-BFE-052
April, 1990

ENCLOSURE 5

TENNESSEE VALLEY AUTHORITY

BROWNS FERRY NUCLEAR PLANT

UNIT 2, CYCLE 6

LICENSING REPORT

EXTENDED LOAD LINE LIMIT ANALYSIS

CONTENTS

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FIGURES

1. SUMMARY

This report justifies the expansion of the operating region of the power/flow map for Unit 2 of Browns Ferry Nuclear Plant (BFNP-2). The operating envelope is modified to include the extended operating region bounded by the power/flow line defined by 0.58Wd + 50%*, the rated power line, and the rated load line, as shown in Figure 1.

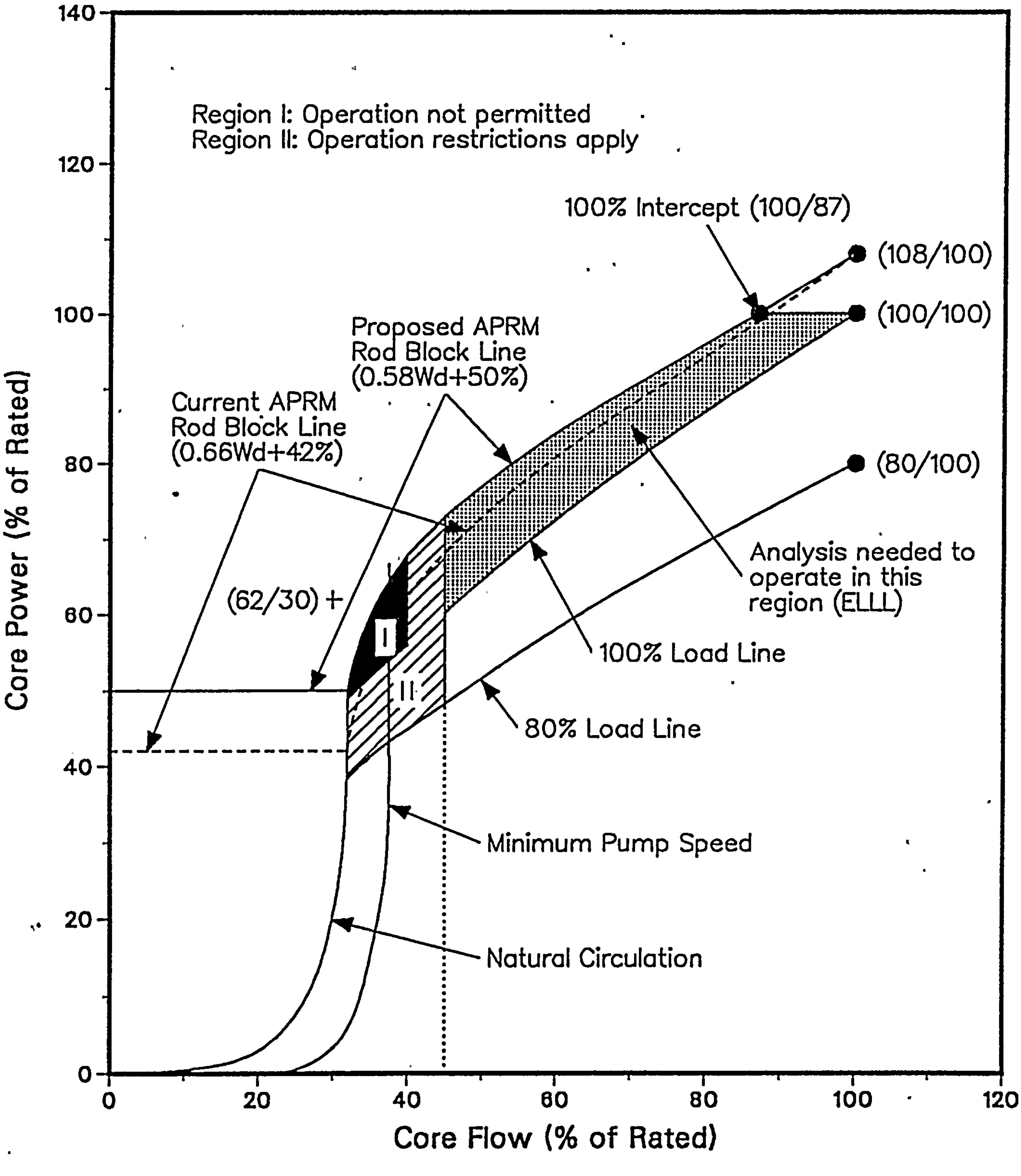
The technical analysis contained in this report is referred to as the Extended Load Line Limit (ELLL) analysis and the shaded area in Figure 1 is referred to as the ELLL region.

The discussion and analyses presented show that events initiated from within the ELLL region meet the applicable design criteria and the reference licensing basis operating limits remain valid.

Therefore, the safety analyses confirm that BFNP-2 Cycle 6 can be safely operated in the ELLL region.

*Wd is the recirculation drive flow in percent of rated.

Figure 1
BFNP-2 Power/Flow Map



2. INTRODUCTION

The flexibility of a Boiling Water Reactor (BWR) during power ascension 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, the difference in equilibrium Xenon concentrations will result in less than rated power at rated core flow. Second, fuel pellet-cladding-interaction considerations (for non-barrier fuel types) inhibit control rod withdrawals at high power levels; thus reactivity compensation for changing Xenon concentrations may not be allowed under the Preconditioning Interim Operating Management Recommendations (PCIOMRs). The combination of these two factors can cause difficulty in attaining rated core power in a reasonable time period.

These limitations can be overcome by allowing operation with a rod pattern that requires fewer adjustments when ascending to full power. This requires an expansion of the current power/flow map to allow operation above the rated load line.

The technical analysis contained in this report is referred to as the Extended Load Line Limit (ELLL) analysis and the shaded area in Figure 1 is referred to as the ELLL region. The ELLL operating region is bounded by the power/flow line defined by 0.58Wd + 50% (ELLL load line), the rated power line, and the rated load line.

The purpose of this report is to present the results of the ELLL analyses which were performed for BFNP-2, Cycle 6.

3. DISCUSSION

3.1 BACKGROUND

Operation of BFNP-2 utilizing the standard power/flow map is described in Chapter 3 of the BFN FSAR (Reference 1). This section of the FSAR describes the basic operating envelope (FSAR Figure 3.7-1) within which normal reactor operations are conducted and provides the basic philosophy behind the power/flow curve. Reference 2 presents the safety analysis for the standard operating region of the power/flow map.

The ELLL analysis expands the operating domain to include the ELLL region between the rated load line and the ELLL load line. Rated power operation at any core flow between 87% and 100% is acceptable. The expanded operating map is shown in Figure 1.

3.2 ANALYTICAL BASIS

A modified power/flow curve has been derived to provide relief from the operating restrictions inherently imposed during ascension to power by the existing power/flow curve and PCIOMRs. Five design basis objectives were specified in deriving this operating curve (Reference 3):

a. For those transients and accidents that are sensitive to variations in power and flow, the licensing basis power/flow point must be shown to be a more limiting condition than any condition within the ELLL region (i.e., the shaded region of Figure 1). Otherwise, revised operating limits for the ELLL region must be defined.

b. In no instance shall the ratio of power to flow intentionally exceed the ratio defined by the ELLL load line.

c. The slope of the ELLL load line must be such that flow increases are capable of compensating for xenon buildup while increasing reactor power to rated power at rated core flow.

d. The consequences of all accidents and transients analyzed in the FSAR and subsequent amendments and the reload licensing submittals must remain within the limits normally specified for such events.

e. Reactor power ascension from minimum recirculation pump speed to full power shall be directly attainable through combined control rod movement and recirculation flow increase without violation of either the ELLL load line or PCIOMRs.

3.3 ANALYSIS AND RESULTS

An evaluation was performed by General Electric (GE) to support operation of BFNP-2 in the ELLL region (Reference 4). This evaluation determined the potential impacts on reactor stability, containment dynamic loadings, vessel internals' structural integrity, emergency core cooling system performance and anticipated transients without scram performance. The potential impact on fuel thermal limits, with the exception of transient considerations, was also addressed in this evaluation. The systems analyses results show that operation in the ELLL region is within allowable design limits for stability, loss-of-coolant accident, containment, reactor internals and anticipated transient without scram events.

Transient analyses to support BFNP-2 operation in the ELLL region were performed by Tennessee Valley Authority (Reference 5). Guidelines for performing transient analyses for the ELLL region were provided by GE in Reference 4. These transient analyses, in combination with the analyses performed by GE, constitute the full scope of analysis required for ELLL operations.

3.3.1 Abnormal Operating Transients

The following Abnormal Operating Transients were reevaluated in the ELLL region (Reference 5). They are:

a. Generator Load Rejection Without Bypass (GLRWOB)

b. Feedwater Flow Controller Failure (FWCF)

These two transients are the most limiting pressurization events, and thus are the most likely to impact the critical power ratio (CPR) operating limits. The other pressurization and non-pressurization transients were determined not to impact the operating limits. The reevaluation was performed at the limiting power/flow condition of Figure 1 (rated power/87% core flow). The initial conditions are presented in Table 1.

The computer model described in Reference 6 was used to simulate both the Generator Load Rejection Without Bypass and Feedwater Controller Failure events. The transient peak value results and CPR results for the two cases analyzed are summarized in Tables 2 and 3. The transient responses are presented in Figures 2 through 9. The results of this evaluation show that the delta-CPR results for all the cases analyzed in the ELLL region are equal to

or bounded by the current Technical Specification limits. No change in operating limits are therefore required.

The rationale for the selection of 100% rated power for the transient analysis condition is that ELLL operation is intended to provide additional maneuvering flexibility and does not represent a change to plant design. Therefore, it must be demonstrated that this additional flexibility is within the plant design which is bounded by the reference transient analysis. This has been the previous GE licensing position for all plants incorporating the ELLL analysis. However, loss-of-coolant accident (LOCA) analysis was performed at 102% rated power to satisfy the requirements of 10CFR50 Appendix K.

3.3.2 ASME Pressure Vessel Code Compliance

The main steam isolation valve (MSIV) closure with an indirect (flux) scram event is used to determine compliance to the American Society of Mechanical Engineers (ASME) pressure vessel code. This event was analyzed at the 100% power/87% core flow point for BFNP-2 cycle 6 (Reference 5). The results are compared to those for the reference licensing basis analysis in Table 4. As shown, the peak vessel pressures are well below the 1375 psig design limit and are bounded by the reference licensing basis analysis for BFNP-2.

3.3.3 Rod Withdrawal Error

The rod block monitor setpoint is a function of drive flow. The RWE event initiating from the ELLL region was reevaluated (Reference 5), and found to be bounded by the reference licensing basis analysis because at the lower core flow, control rod withdrawal will be blocked earlier by the flow biased rod block monitor system. Thus, the reference licensing basis evaluation at the rated power and flow condition is conservative for operation in the ELLL region.

3.3.4 Slow Flow Runout Event and Kf Bases

The purpose of Kf is to define MCPR operating limits at off-rated flow conditions. In particular, Kf is designed to maintain core thermal margins in the event of a slow flow runout event. The Kf curves currently in the BFNP-2 Technical Specifications were derived generically by the fuel vendor. In order to ensure that these curves adequately bound the slow flow runout event for operation in the ELLL region, the event was reevaluated on a comparable basis to the generically derived curves.

Detailed evaluations of the slow flow runout event initiated from the limiting point in the ELLL region verified that the existing Kf curves are acceptable for. ELLL operation. Although the flow will runout along a steeper rod line than would occur in the normal operating domain, the change in core power and MCPR from a given initial core flow will be limited by the recirculation system characteristics such that the Kf curves based on the normal power/flow map bound the ELLL results.

3.3.5 Thermal-Hydraulic Stability

General Electric has established stability criteria to demonstrate compliance to the requirements set forth in 10CFR50 Appendix A, General Design Criteria (GDC) 10 and 12. These stability compliance criteria consider potential limit cycle response within the limits of safety system or operator intervention and assure that for GE BWR fuel designs this operating mode does not result in specified acceptable fuel design limits being exceeded. The stability compliance of all licensed GE BWR fuel designs, including those contained in the General Electric Standard Application for Reactor Fuel (GESTAR II, Reference 7), is demonstrated on a generic basis in Reference 8 (for operation in the normal as well as the ELLL region). The BFNP-2 Cycle 6 core contains licensed GE BWR fuel and, hence, the generic evaluation in Reference 8 is applicable. In addition, the BFNP-2 Cycle 6 core contains four Westinghouse QUAD+ demonstration assemblies. Westinghouse Electric Corporation has confirmed the stability compliance of the QUAD+ assembly in the Westinghouse Reference Safety Report for BWR Fuel (Reference 9).

Recent stability concerns have resulted in the development of interim stability corrective actions by the BWR Owner's Group (BWROG). These recommendations apply to all GE BWRs and were developed including consideration of those plants which have expanded operating regions, such as ELLL. These interim actions have been reviewed and accepted by the NRC as documented in Reference 10. One of the stated modifications applies to BWR/4 which have a simulated thermal power monitor (a filtered average power range monitor flow-biased scram). For these plants, which include BFNP-2, the NRC requires that the BWROG corrective actions be supplemented with a requirement to manually scram the reactor upon the occurrence of a dual recirculation pump trip while the reactor is in the RUN mode. These corrective actions have been incorporated into the BFNP-2 Technical Specifications (References 11 and 12) and are more than adequate to reduce potential thermal-hydraulic concerns when operating in the ELLL region (Reference 4).

3.3.6 Loss-of-Coolant Accident

Based on the discussion of the Loss-of-Coolant Accident in Reference 4, it is concluded that the current LOCA analysis for BFNP-2 is applicable for operation in the ELLL region.

The results and conclusions regarding the effects of core flow on LOCA analyses for all operating plants (Reference 13) have been presented to and were approved by the NRC (Reference 14). These analyses were performed using an approved LOCA analytical model in accordance with 10CFR50 Appendix K basis (References 15 and 16).

Reference 13 shows that 251-inch BWR/4 plants like BFNP-2 have the smallest effect of core flows on LOCA analysis of all the BWR/4 designs because of the smallest "effective break size" (ratio of largest break area to water inventory in the reactor primary system). This ratio determines how rapidly the reactor will depressurize during a LOCA, and more importantly, the minimum transient core flow dip during the first second following the break. The smaller this minimum core flow dip is, the less probable that early boiling transition (EBT) is likely to occur in the highest power plane. These plants also have a relatively early reflooding time which allows a relatively high MAPLHGR (maximum average planar linear heat generation rate). The Reference 13 analyses also demonstrate that the peak clad temperature (PCT) reduction due to low power levels more than compensates for early loss of nucleate boiling in low flow analyses for even the largest break case. The Reference 13 analyses also took no credit for the flow dependent MCPR multiplier, Kf, in determining whether or not EBT would occur, therefore assuming the bundle was closer to EBT than actually allowed by the Technical Specifications. Regardless of the limiting break size or location, there is no required MAPLHGR multiplier for application at low core flow condition for 251-inch BWR/4s, including BFNP-2.

Therefore, the LOCA analysis for BFNP-2 (Reference 17) is applicable to plant operation in the ELLL region. The MAPLHGRs or peak clad temperatures calculated in the Reference 17 LOCA analysis remain applicable for the ELLL region.

3.3.7 Containment Analysis

Rated power operation at less than rated core flow conditions causes the coolant pressure and temperature within the reactor to drop slightly from the rated values. The downcomer temperature is slightly lower at lower flows because the percentage of cool feedwater in the downcomer

increases relative to the rated condition. The reactor pressure and internal differential pressures are slightly lower because of the lower core flow. Subsequently, if a LOCA is postulated at these conditions, the initial break flow will be slightly higher than at the rated power/flow condition.

The short term LOCA containment pressure and temperature response were evaluated in Reference 4 using the NRC-approved containment response model of Reference 18. The major parameters which characterize the containment response are: the peak drywell pressure, peak wetwell pressure, peak drywell temperature, peak wetwell (airspace) temperature, and peak suppression pool water temperature. The major containment dynamic loads which occur in a Mark I plant during a design basis LOCA include pool swell, vent thrust, condensation oscillation and chugging. These loads are controlled by the containment thermal hydraulic response during the LOCA.

Based on the discussion of these major containment response parameters and containment dynamic loads in Reference 4, a LOCA while operating in the ELLL region for BFNP-2 would produce a containment response within design limits.

3.3.8 Reactor Internals Integrity

For a recirculation pump runout event initiating in the ELLL region, the resulting reactor power increase will be higher than that for the same event initiating from the rated rod line. The flow runout is limited by the scoop tube mechanical stop which is assumed to be set at 2.5% above the maximum allowable core flow of 105% of rated. The higher power/flow condition reached at the end of this type of event may impact the loadings across the reactor internal components.

Based on the analysis of reactor internals integrity in Reference 4, plant operation in the ELLL region for BFNP-2 will produce core plate, channel, shroud and shroud head differential pressures that are bounded by the Reference 19 results and therefore are within the design limits. The shroud support pressure drop is higher than the value in Reference 19 (32.9 vs 32.7 psi). However, this loading is less than the limiting load for this component (53 psi, Reference 19), and therefore is acceptable.

3.3.9 ATWS Evaluation

Based on the discussion of Anticipated Transient without Scram (ATWS) events in Reference 4, an ATWS event initiated from operation in the ELLL region would produce a response

within design limits. The conclusions reached in Reference 20 that the BWR can adequately mitigate the ATWS events have been shown to also be true when the events are initiated at reduced core flow.

The event considered for the evaluation is the MSIV closure since this event gives the most conservative results. The maximum vessel bottom pressure increased from 1296 psig to 1367 psig but is still well within the limits of the emergency stress level of 1500 psig. The maximum fuel cladding temperature will increase only slightly as evidenced by the small increase from 143% to 147% for the maximum heat flux. The temperature will remain far below the limit of coolable geometry. The maximum pressure suppression pool temperature decreased substantially due to the reduction in vessel water level. It decreased from 186 degrees Fahrenheit (F) to 158 degrees F which is, of course, below the historical maximum guideline of 190 degrees F.

3.4 CONCLUSIONS

This report justifies the expansion of the operating region of the power/flow map for Unit 2 of Browns Ferry Nuclear Plant, Cycle 6. The operating envelope is modified to include the extended operating region bounded by the power/flow line defined by 0.58Wd + 50%, the rated power line, and the rated load line.

The discussion and analyses presented show that events initiated from within the ELLL region meet the applicable design criteria and the reference licensing basis operating limits remain valid.

Therefore, the safety analyses confirm that BFNP-2 Cycle 6 can be safely operated in the ELLL region.

Table 1
Input Parameters and Initial Conditions for Transient Analysis

		Reference Analysis (104.3% P/105% F)	ELLLA (100% P/87% F)
1.	Thermal Power, MWt	3436	3293
2.	Steam Flow, Mlb/hr	14.05	13.37
3.	Core Flow, Mlb/hr	107.6	89.2
4.	Feedwater Flow Rate, lb/sec	3902.8	3712.5
5.	Feedwater Temperature, degrees F	379.6	375.3
6.	Vessel Dome Pressure, psia	1035	1019.9
7.	Core Exit Pressure, psig	1031	1015
8.	Turbine Bypass Capacity, % NBR	26.2	26.2
9.	Core Coolant Inlet Enthalpy, Btu/lb	524.39	517.96
10.	Turbine Inlet Pressure, psig	974	963
11.	Fuel Lattice	P8x8R	P8x8R
12.	Core Leakage Flow, % Core flow	11.18	11.16
13.	MCPR Safety Limit for Incidents of Moderate Frequency	1.07	1.07

Table 2
Summary of Pressurization Transient Results

	Generator Load Rejection Without Bypass		Feedwater Controller Failure	
	Reference Analysis	ELLLA	Reference Analysis	ELLLA
Initial Core Power, % Rated	104.3	100.0	104.3	100.0
Core Flow, % Rated	105.0	87.0	105.0	87.0
Peak Power, % Rated	403.4	252.1	234.8	172.3
Peak Heat Flux, % Rated	121.6	109.4	115.5	108.2
Peak Vessel Pressure, psia	1235.3	1226.5	1215.1	1195.2

Table 3
Summary of CPR Results

	Generator Load Rejection Without Bypass		Feedwater Controller Failure	
	Reference Analysis	ELLLA	Reference Analysis	ELLLA(1)
Initial Core Power, % Rated	104.3	100.0	104.3	100.0
Core Flow, % Rated	105.0	87.0	105.0	87.0
Operating Limit MCPR (a)	1.35	1.31	1.27	1.27
delta-CPR (a)	0.28	0.24	0.20	0.20
Operating Limit MCPR (b)	1.26	1.21	1.23	1.23
delta-CPR (b)	0.19	0.14	0.16	0.16

(a) Option A adders included.

(b) Option B adders included.

(1) The ELLL analysis of FWCF includes the 0.03 delta-CPR adder determined for the reference conditions to bound potential increases due to initial conditions more severe than 105% steam flow.

Table 4
ASME Pressure Vessel Code Compliance:
MSIV Closure (Flux Scram)

	Reference Analysis	ELLLA
Initial Core Power, % Rated	104.3	100.0
Core Flow, % Rated	105.0	87.0
Peak Power, % Rated	527.2	397.1
Peak Heat Flux, % Rated	141.5	127.7
Peak Vessel Pressure, psia	1281.0	1254.2

Figure 2
BFNP-2 Cycle 6 Generator Load Rejection without Bypass
Net Reactivity Response

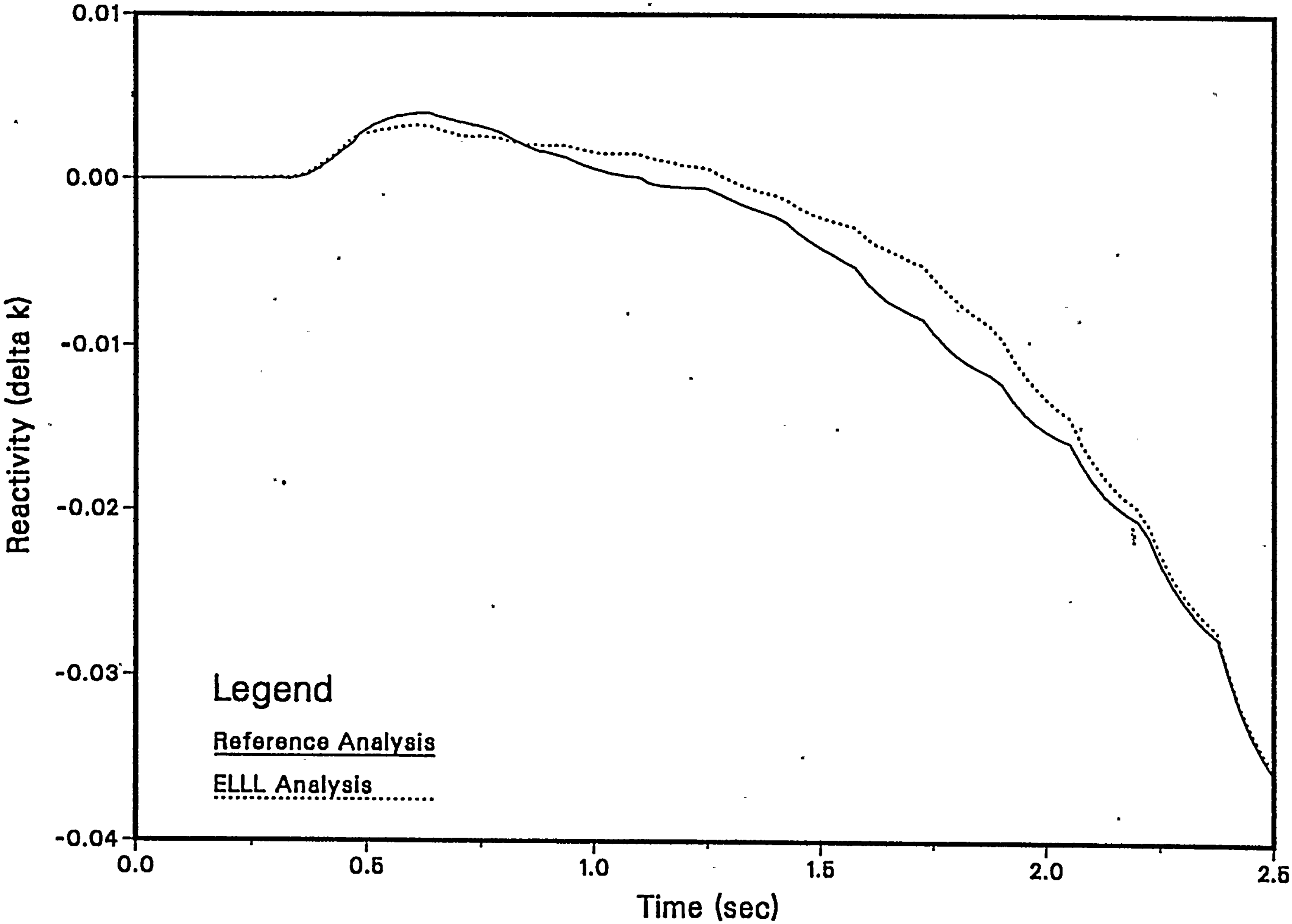


Figure 3
BFNP-2 Cycle 6 Generator Load Rejection without Bypass
Thermal Power Response

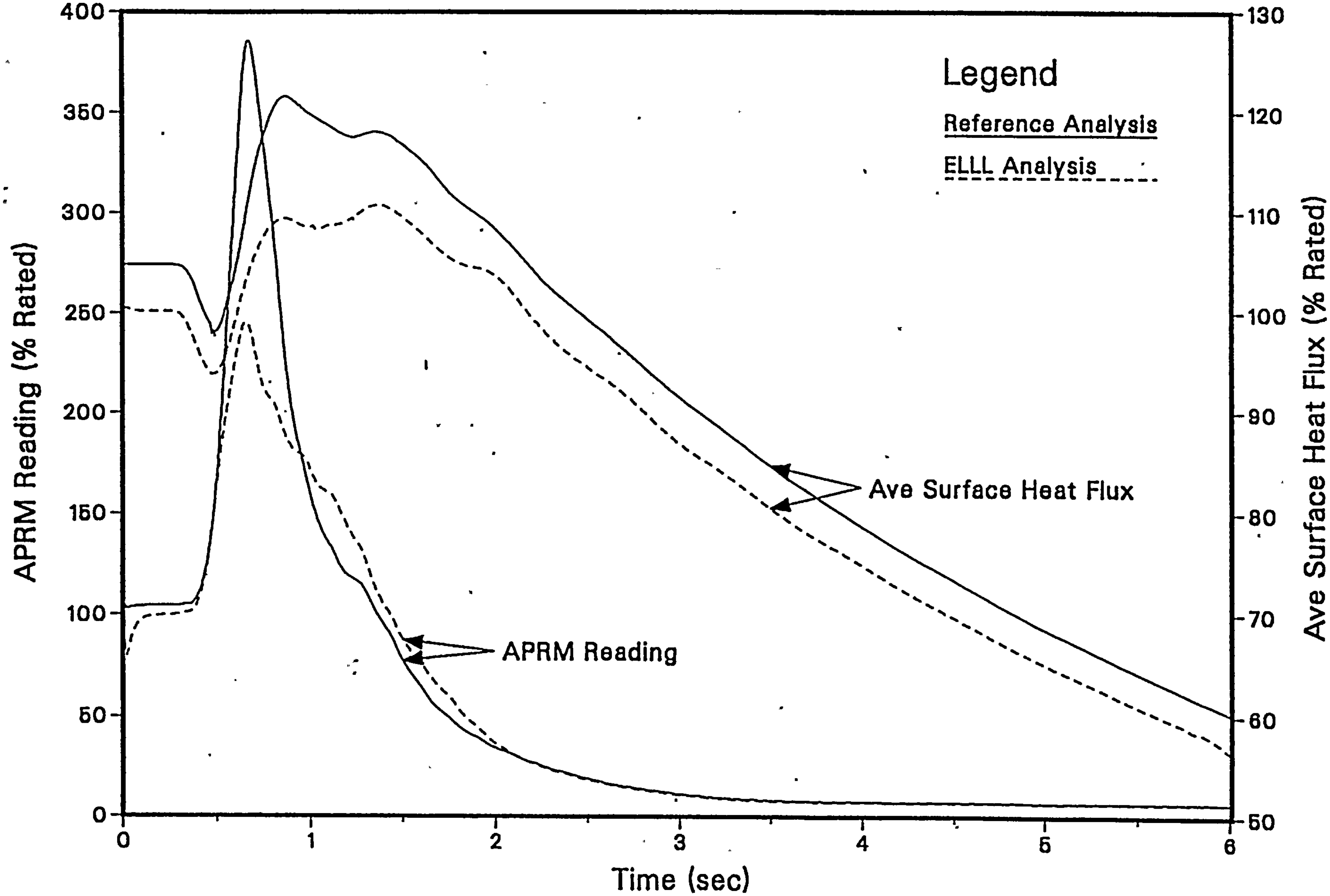


Figure 4
BFNP-2 Cycle 6 Generator Load Rejection without Bypass
Vessel Flow Rates

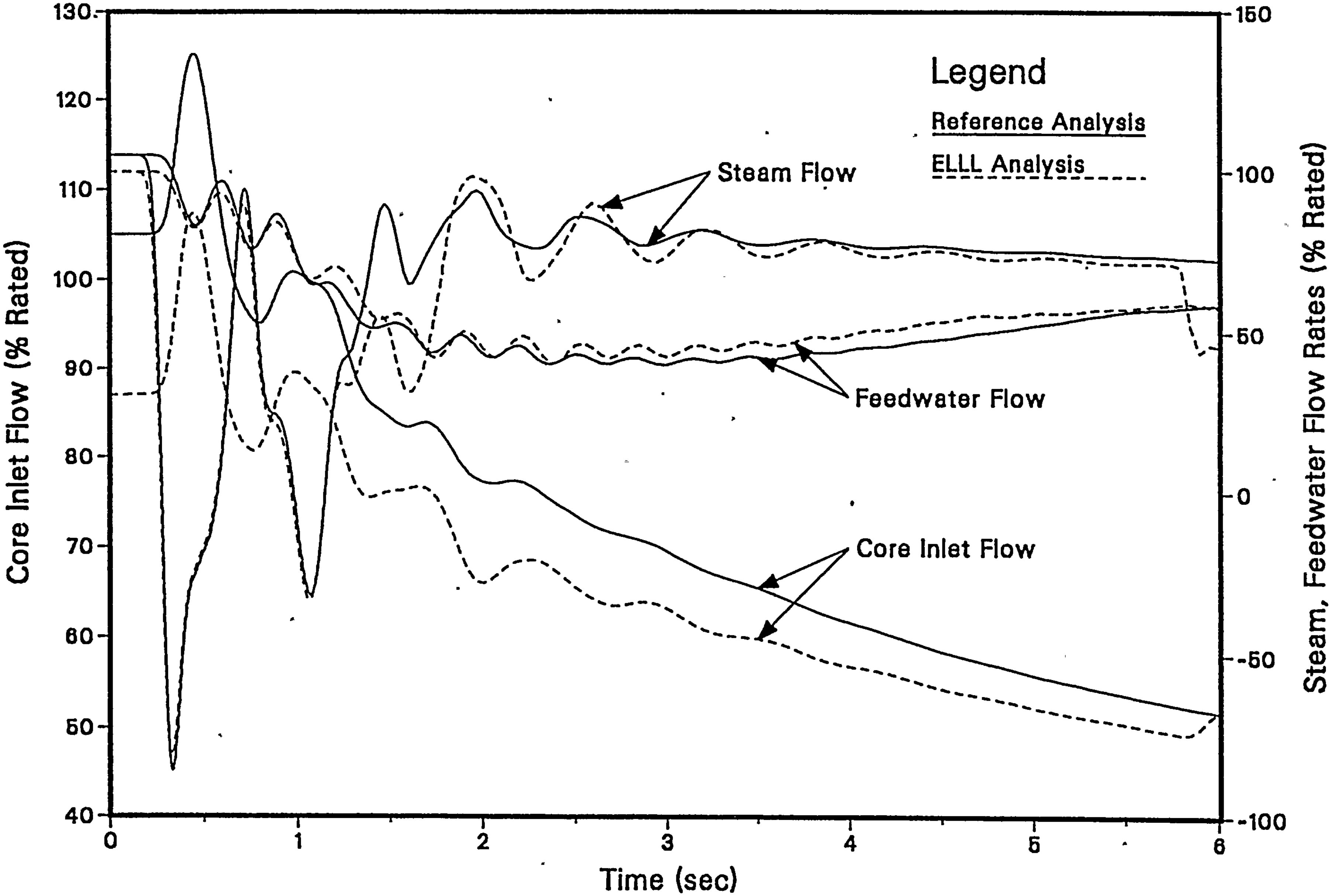


Figure 5
BFNP-2 Cycle 6 Generator Load Rejection without Bypass
Reactor Pressure and Level Response

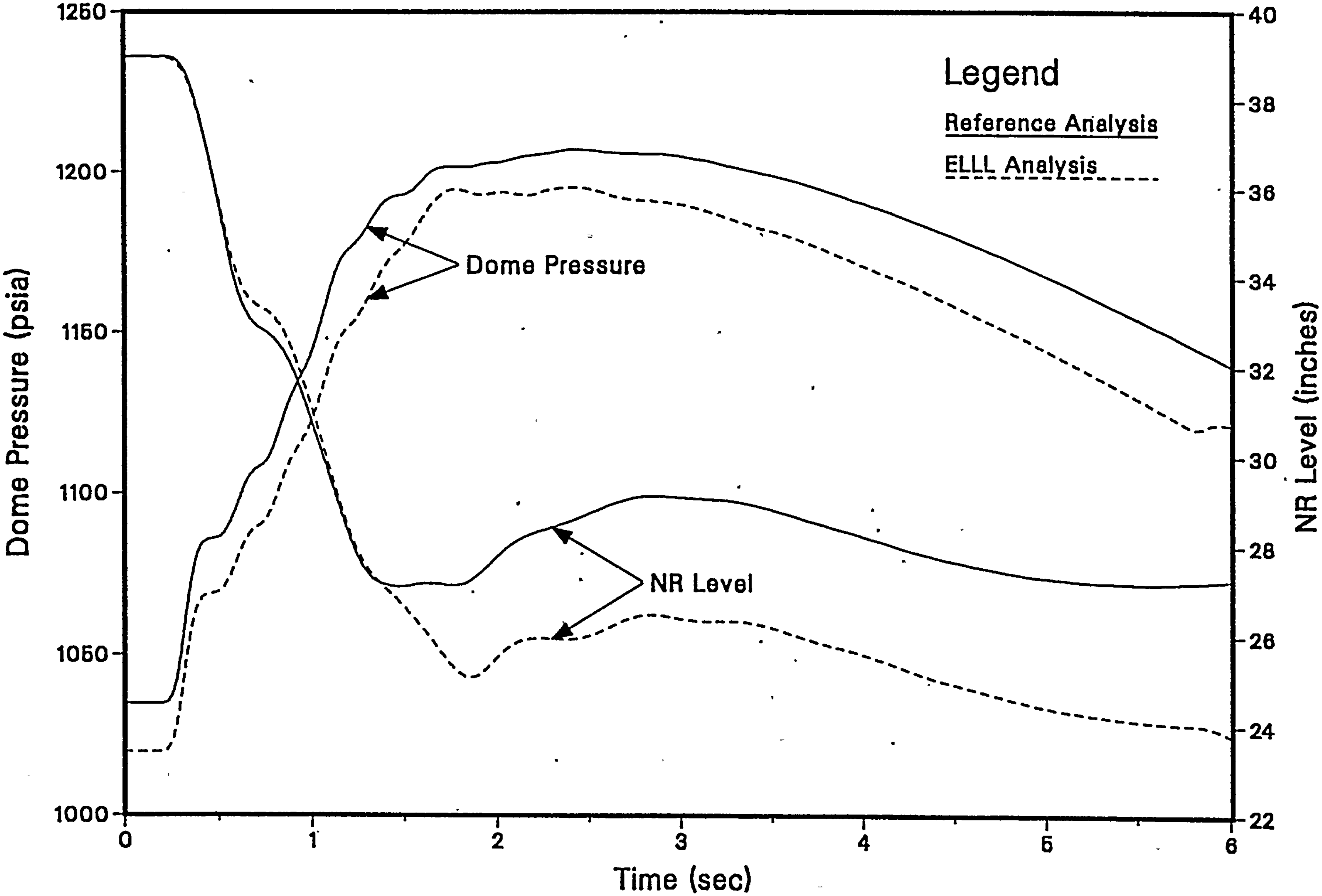


Figure 6
BFNP-2 Cycle 6 Feedwater Controller Failure
Net Reactivity Response

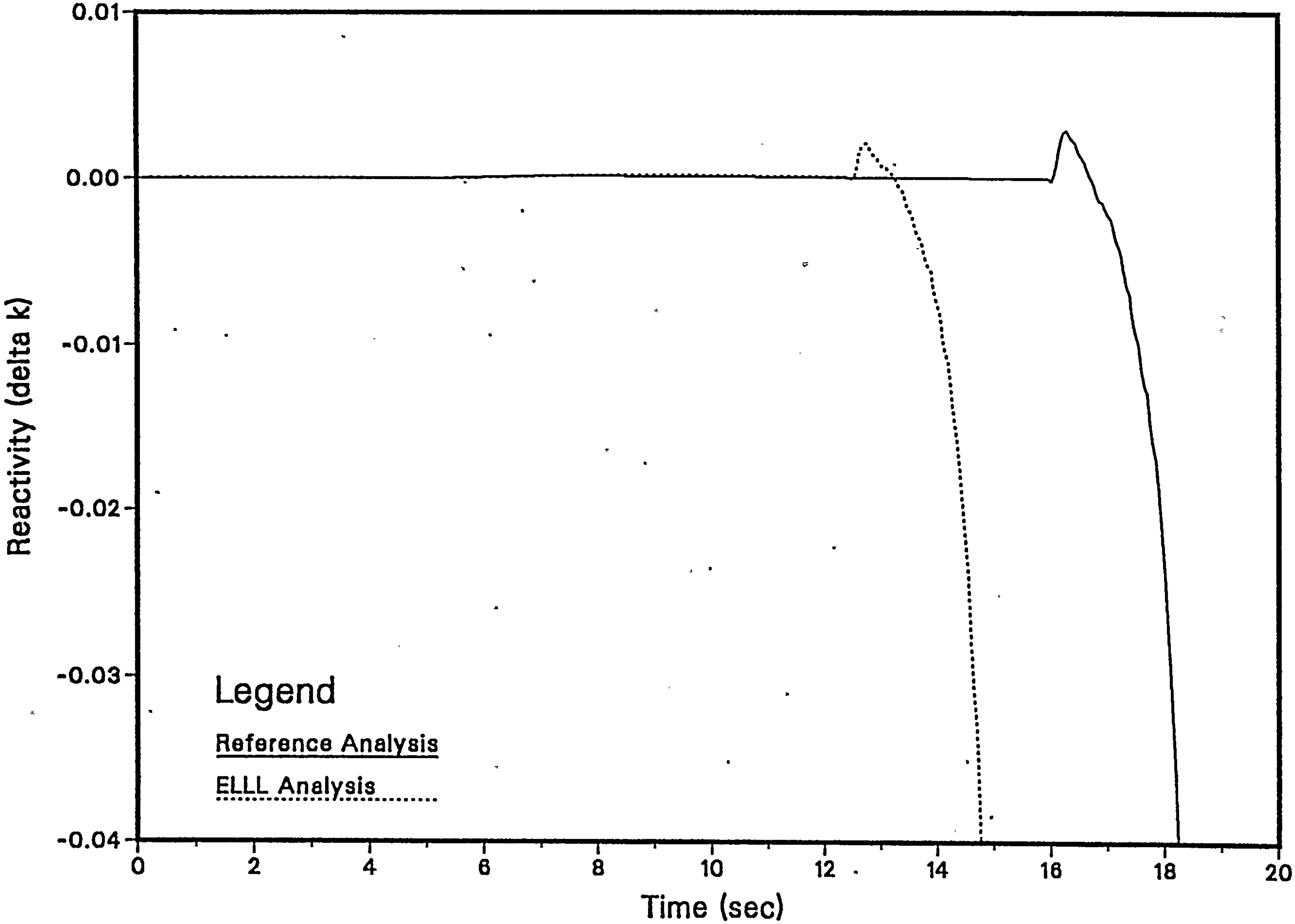


Figure 7
BFNP-2 Cycle 6 Feedwater Controller Failure
Thermal Power Response

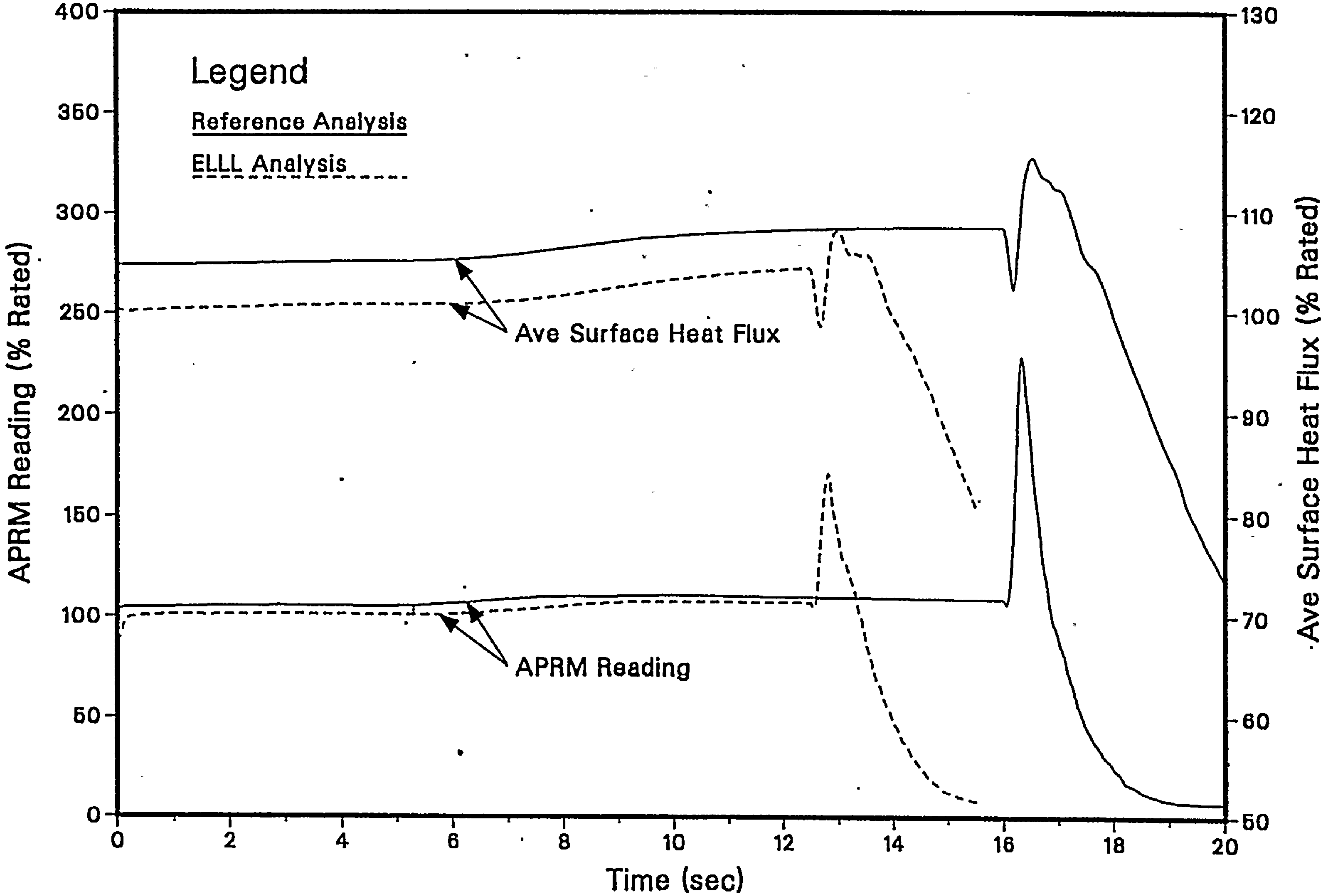


Figure 8
BFNP-2 Cycle 6 Feedwater Controller Failure
Vessel Flow Rates

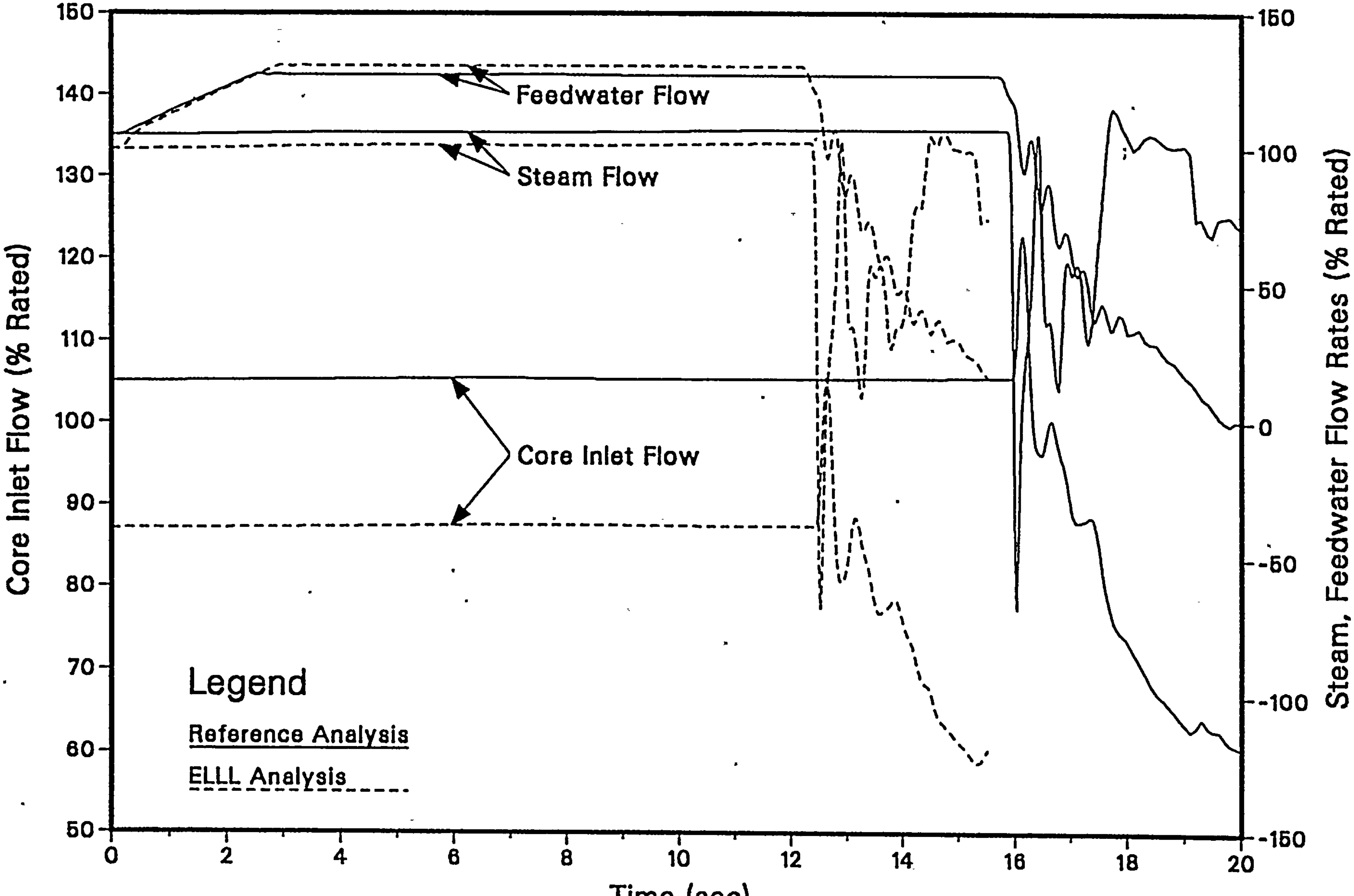
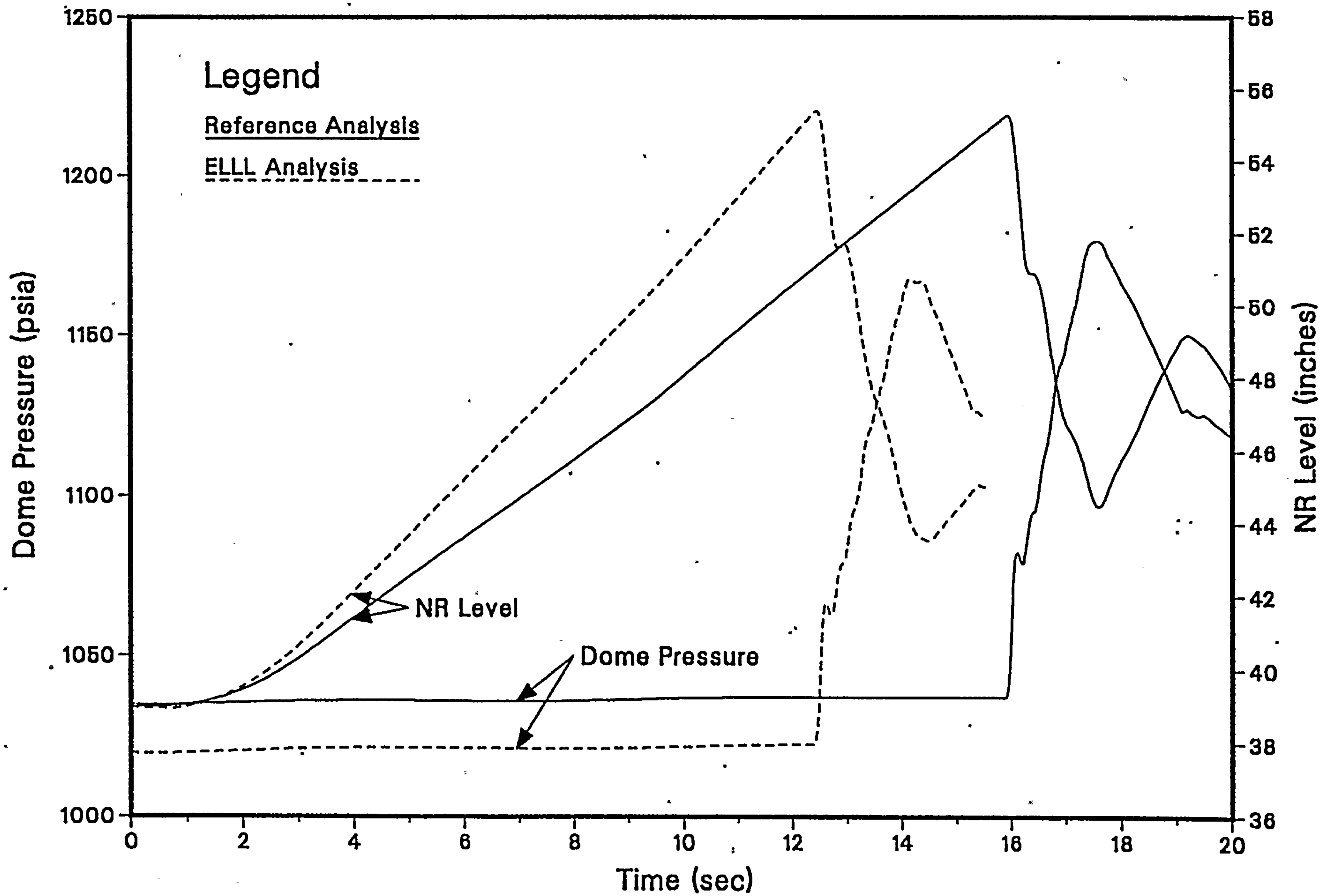


Figure 9
BFNP-2 Cycle 6 Feedwater Controller Failure
Reactor Pressure and Level Response



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