GE Nuclear Energy

24A5172
Revision 0
Class I
February, 1995

# 24A5172, Rev. 0

# Supplemental Reload Licensing Report

# for

# Pilgrim Nuclear Power Station

# Reload 10 Cycle 11

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**Important Notice Regarding**

**Contents of This Report**

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# Acknowledgement

The engineering and reload licensing analyses, which form the technical basis of this Supplemental Reload Licensing Report, were performed by D.C. Serell, A. Alzaben, and T.A. Terrio, with assistance from F.T. Bolger and G.A. Galloway. The Supplemental Reload Licensing Report was prepared by D.C. Serell. This document has been verified by C.W. Smith.

The basis for this report is *General Electric Standard Application for Reactor Fuel*, NEDE-24011-P-A-10, February 1991; and the U.S. Supplement, NEDE-24011-P-A-10-US, March 1991.

## 1. Plant-unique Items

Appendix A: Analysis Conditions and Bases
Appendix B: Increased Core Flow
Appendix C: Decrease in Core Coolant Temperature Events

## 2. Reload Fuel Bundles

| Fuel Type | Cycle Loaded | Number |
|---|---|---|
| Irradiated: | | |
| BP8DRB300 (BP8x8R) | 8 | 136 |
| GE8B-P8DQB323-10GZ-80M-4WR-145-T (GE8x8EB) | 9 | 168 |
| GE10-P8HXB355-11GZ-100M-145-T (GE8x8NB-3) | 10 | 140 |
| New: | | |
| GE11-P9HUB378-15GZ-100T-141-T (GE11) | 11 | 136 |
| Total | | 580 |

## 3. Reference Core Loading Pattern

| | |
|---|---|
| Nominal previous cycle core average exposure at end of cycle: | 24245 MWd/MT (21994 MWd/ST) |
| Minimum previous cycle core average exposure at end of cycle from cold shutdown considerations: | 23804 MWd/MT (21594 MWd/ST) |
| Assumed reload cycle core average exposure at beginning of cycle: | 17222 MWd/MT (15623 MWd/ST) |
| Assumed reload cycle core average exposure at end of cycle: | 28520 MWd/MT (25873 MWd/ST) |
| Reference core loading pattern: | Figure 1 |

## 4. Calculated Core Effective Multiplication and Control System Worth – No Voids, 20°C

| | |
|---|---|
| Beginning of Cycle, $k_{effective}$ | |
| Uncontrolled | 1.103 |
| Fully controlled | 0.960 |
| Strongest control rod out | 0.986 |
| R, Maximum increase in cold core reactivity with exposure into cycle, $\Delta k$ | 0.003 |

## 5. Standby Liquid Control System Shutdown Capability

| Boron (ppm) | Shutdown Margin ($\Delta k$) (20°C, Xenon Free) |
|---|---|
| 675 | 0.042 |

## 6. Reload Unique GETAB Anticipated Operational Occurrences (AOO) Analysis Initial Condition Parameters[1]

| Exposure: BOC11 to EOC11–5787 MWd/MT (5250 MWd/ST) ANALYZED AT 102% CORE FLOW | | | | | | | |
|---|---|---|---|---|---|---|---|
| | Peaking Factors | | | | | | |
| Fuel Design | Local | Radial | Axial | R–Factor | Bundle Power (MWt) | Bundle Flow (1000 lb/hr) | Initial MCPR |
| GE11 | 1.45 | 1.85 | 1.31 | 1.035 | 6.209 | 91.9 | 1.30 |

| Exposure: EOC11–5787 MWd/MT (5250 MWd/ST) to EOC11 ANALYZED AT 107.5% CORE FLOW | | | | | | | |
|---|---|---|---|---|---|---|---|
| | Peaking Factors | | | | | | |
| Fuel Design | Local | Radial | Axial | R–Factor | Bundle Power (MWt) | Bundle Flow (1000 lb/hr) | Initial MCPR |
| GE11 | 1.45 | 1.75 | 1.18 | 1.035 | 5.859 | 101.2 | 1.37 |

## 7. Selected Margin Improvement Options

| | |
|---|---|
| Recirculation pump trip: | No |
| Rod withdrawal limiter: | No |
| Thermal power monitor: | No |
| Improved scram time: | Yes (ODYN Option B) |
| Measured scram time: | No |
| Exposure dependent limits: | Yes |
| Exposure points analyzed: | 2 |

1. The delta CPR response for GE11 bounds all other fuel types in the core.

## 8. Operating Flexibility Options

| | |
|---|---|
| Single–loop operation: | Yes[2] |
| Load line limit: | Yes |
| Extended load line limit: | Yes |
| Maximum extended load line limit: | Yes[3] |
| Increased core flow throughout cycle: | Yes[4] |
| Increased core flow at EOC: | Yes |
| BOC to 5000 MWD/STU Flow point analyzed: | 102 % |
| 5000 MWD/STU to EOC Flow point analyzed: | 107.5 % |
| Feedwater temperature reduction throughout cycle: | No |
| Final feedwater temperature reduction: | No |
| ARTS Program: | Yes[5] |
| Moisture separator reheater OOS: | No |
| Turbine bypass system OOS: | No |
| Safety/relief valves OOS: | No |
| ADS OOS: | No |
| One Main steam isolation valve OOS: | Yes[6] |

2. "Pilgrim Nuclear Power Station Single–Loop Operation", NEDC–24268,June,1980

3. H.X.Hoang,"Maximum Extended Load Line Limit Analyses for Pilgrim Nuclear Power Station Reload 9 Cycle 10", NEDC–32306P,MARCH 1994.

4. "Safety Review of Pilgrim Nuclear Power Station Unit No. 1 at Core Flow Conditions Above Rated Flow Throughout Cycle 6", NEDO–30242, August, 1983.

5. "ARTS Improvement Program Analysis for Pilgrim Nuclear Power Station", NEDO–31312P,September, 1987.

6. MSIV Out of Service Report, NSE–82–0982, DRF B21–00238, September 1982.

## 9. Core-wide AOO Analysis Results[7] [8]

**Methods used:** GEMINI; GEXL-PLUS

| **Exposure range: BOC11 to EOC11-5787 MWd/MT (5250 MWd/ST) ANALYZED AT 102% CORE FLOW** | | | | |
|---|---|---|---|---|
| | | | **Uncorrected ΔCPR** | |
| **Event** | **Flux (%NBR)** | **Q/A (%NBR)** | **GE11** | **Fig.** |
| FW Controller Failure | 225 | 115 | 0.23 | 2 |
| Load Reject w/o Bypass | 230 | 107 | 0.18 | 3 |

| **Exposure range: EOC11-5787 MWd/MT (5250 MWd/ST) to EOC11 ANALYZED AT 107.5% CORE FLOW** | | | | |
|---|---|---|---|---|
| | | | **Uncorrected ΔCPR** | |
| **Event** | **Flux (%NBR)** | **Q/A (%NBR)** | **GE11** | **Fig.** |
| FW Controller Failure | 304 | 123 | 0.30 | 4 |
| Load Reject w/o Bypass | 332 | 116 | 0.27 | 5 |

## 10. Local Rod Withdrawal Error (With Limiting Instrument Failure) AOO Summary[9]

Rod withdrawal error (RWE) is analyzed in General Electric BWR Licensing Report, *Average Power Range Monitor, Rod Block monitor and Technical Specification Improvement (ARTS) Program*, NEDC-30474-P, dated December 1983. A cycle-specific rod withdrawal analysis found the ΔMCPR is bounded by the generic RWE analysis reported in the referenced report. For a setpoint of up to 116%, the rated MCPR limit is 1.35.

7. The GE11 delta CPR response bounds all other fuel types in the core.

8. Analysis at 107.5% increased core flow was conservatively assumed for EOC11 analysis; 102% increased core flow was assumed as the BOC to EOC11-5250 MWd/ST early cycle analysis basis.

9. References: "ARTS Improvement Program Analysis for Pilgrim Nuclear Power Station", NEDC-31312-P, September, 1987 and H.X. Hoang, "ARTS Verification for Pilgrim Nuclear Power Station Reload 8 Cycle 9, GE-NE-187-11-0591, DRF A00-03980, June 1991. These documents were verified applicable to cycle 11.

## 11. Cycle MCPR Values[10 11 12]

Safety limit: 1.07

Single loop operation safety limit: 1.08

### Non–pressurization events:

| Exposure Range: BOC11 to EOC11 | |
|---|---|
| Rod Withdrawal Error (Setpoint can be selected up to 116%), All Fuels | **1.35** |
| Fuel Loading Error, GE11 Reload 10 Fuel | **1.12** |
| Fuel Loading Error, GE10 Reload 9 Fuel | **1.27** |

### Pressurization events:

| **Exposure range: BOC11 to EOC11–5787 MWd/MT (5250 MWd/ST) ANALYZED AT 102% CORE FLOW**<br>**Exposure point: EOC11–5787 MWd/MT (5250 MWd/ST)** | | |
|---|---|---|
| | **Option A** | **Option B** |
| | **GE11** | **GE11** |
| FW Controller Failure | 1.40 | 1.32 |
| Load Reject w/o Bypass | 1.35 | 1.27 |

| **Exposure range: EOC11–5787 MWd/MT (5250 MWd/ST) to EOC11 ANALYZED AT 107.5% CORE FLOW**<br>**Exposure point: EOC11** | | |
|---|---|---|
| | **Option A** | **Option B** |
| | **GE11** | **GE11** |
| FW Controller Failure | 1.45 | 1.39 |
| Load Reject w/o Bypass | 1.43 | 1.37 |

## 12. Overpressurization Analysis Summary

| Event | Psl (psig) | Pv (psig) | Plant Response |
|---|---|---|---|
| MSIV Closure (Flux Scram) | 1286 | 1302 | Figure 8 |

10. The minimum MCPR operating limit required by the SAFER/GESTR analysis is 1.20.

11. See Appendix C for discussion of decrease in core coolant temperature events.

12. For single–loop operation, the MCPR operating limit is not greater than the two–loop value.

## 13. Loading Error Results

From a misoriented bundle analysis with variable water gap, including a 0.02 penalty due to variable water gap R–factor uncertainty, the ΔMCPR for the fresh reload 10 GE11 fuel bundle is 0.05. The ΔMCPR for the reload 9 GE10 fuel bundle is 0.20.

## 14. Control Rod Drop Analysis Results

This is a banked position withdrawal sequence plant, therefore, the control rod drop accident analysis is not required. NRC approval is documented in NEDE–24011–P–A–US.

## 15. Stability Analysis Results

Pilgrim Nuclear Power Station is exempt from the current requirement to submit a cycle–specific stability analysis as documented in the letter, C. O. Thomas (NRC) to H. C. Pfefferlen (GE), *Acceptance for Referencing of Licensing Topical Report NEDE–24011 Rev. 6, Amendment 8, "Thermal Hydraulic Stability Amendment to GESTAR II,"* April 24, 1985.

Pilgrim Nuclear Power Station recognizes the issuance of NRC Bulletin No. 88–07, Supplement 1, *Power Oscillations in Boiling Water Reactors (BWRs)*, and will comply with the recommendations contained therein. Pilgrim Station is also complying with the NRC Bulletin No. 94–02, *Long–Term Solutions and Upgrade of Interim Operating Recommendations for Thermal–Hydraulic Instabilities in Boiling Water Reactors*. GE11 fuel has been demonstrated to have equivalent or better stability characteristics than BP8x8R fuel by the GESTAR Amendment 22 licensing analysis (Reference: NEDE 31917P, *GE11 Compliance with Amendment 22 of NEDE–24011–P–A (GESTRAR II), April 1991)*, and no unique or special actions are needed to comply with the above NRC Bulletins.

## 16. Loss–of–Coolant Accident Results

**LOCA method used:** SAFER/GESTR–LOCA

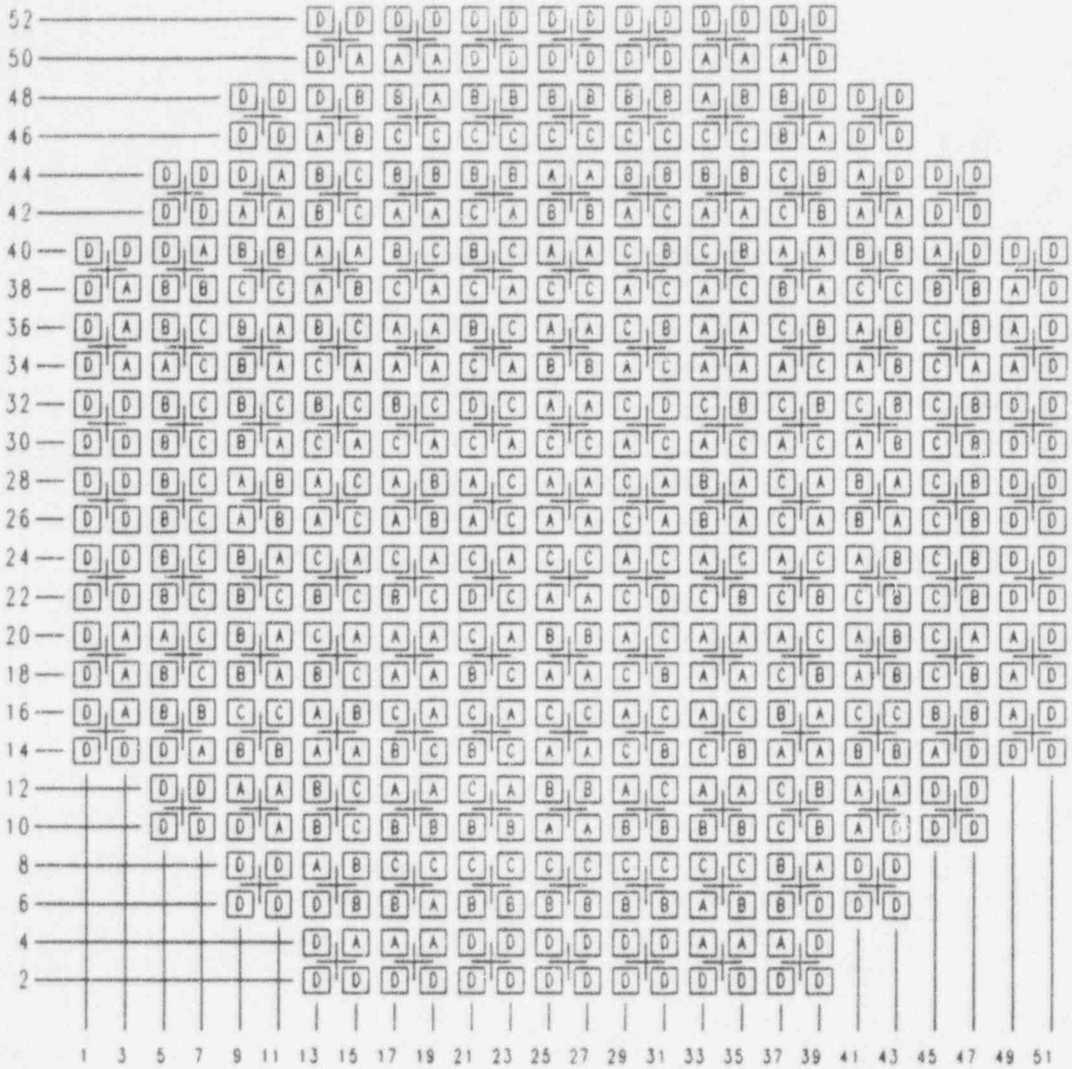
The LOCA analysis results are presented in Sections 5 and 6 of *Pilgrim Nuclear Power Station SAFER/GESTR–LOCA Loss–of–Coolant Accident Analysis*, NEDC–31852P, April, 1992 (Revision 1) as amended. The GE11 LOCA analysis for Pilgrim was performed using the same SAFER/GESTR analysis basis used for the previously analyzed BP/P8x8R and GE8x8EB/NB fuel types. Addition of the GE11 fuel will not significantly affect the overall system response of the plant for the various operating modes, and the GE11 analysis confirmed that the limiting break type and size and limiting ECCS failure (DBA recirculation suction line break with LPCIIV failure) do not change. The GE11 fuel analysis yielded a licensing basis peak PCT of 1815 °F and a peak local oxidation fraction of <0.3%, and all licensing basis criteria are met. The GE11 results are bounded by the 1825 °F licensing basis PCT for BP/P8x8R fuel and the overall licensing basis results reported in Table 6–1 of the Reference analysis.

The GE11 SAFER/GESTR results are applicable for a peak enriched lattice MAPLHGR of 12.16 kw/ft.,which bounds the MAPLHGRs for the reload 10 fuel. Therefore, the MAPLHGR limits reflect the thermal–mechanical limits for the reload fuel rather than LOCA/ECCS considerations. The most limiting and the least limiting MAPLHGRs for the new fuel are as follows:

## 16. Loss-of-Coolant Accident Results (cont)

Bundle Type: GE11-P9HUB378-15GZ-100T-141-T

| Average Planar Exposure | | MAPLHGR(kW/ft) | |
|---|---|---|---|
| (GWd/ST) | (GWd/MT) | Most Limiting | Least Limiting |
| 0.00 | 0.00 | 9.95 | 10.46 |
| 0.20 | 0.22 | 10.04 | 10.53 |
| 1.00 | 1.10 | 10.19 | 10.61 |
| 2.00 | 2.20 | 10.41 | 10.78 |
| 3.00 | 3.31 | 10.64 | 10.98 |
| 4.00 | 4.41 | 10.88 | 11.21 |
| 5.00 | 5.51 | 11.09 | 11.35 |
| 6.00 | 6.61 | 11.19 | 11.55 |
| 7.00 | 7.72 | 11.30 | 11.63 |
| 8.00 | 8.82 | 11.40 | 11.70 |
| 9.00 | 9.92 | 11.52 | 11.79 |
| 10.00 | 11.02 | 11.65 | 11.91 |
| 12.50 | 13.78 | 11.64 | 11.92 |
| 15.00 | 16.53 | 11.48 | 11.71 |
| 17.50 | 19.29 | 11.25 | 11.47 |
| 20.00 | 22.05 | 11.02 | 11.22 |
| 25.00 | 27.56 | 10.55 | 10.75 |
| 30.00 | 33.07 | 10.07 | 10.22 |
| 35.00 | 38.58 | 9.39 | 9.53 |
| 40.00 | 44.09 | 8.72 | 8.87 |
| 45.00 | 49.60 | 8.06 | 8.23 |
| 50.00 | 55.12 | 7.40 | 7.60 |
| 55.00 | 60.63 | 6.72 | 6.96 |
| 57.02 | 62.86 | 6.44 | 6.69 |
| 57.10 | 62.94 | — | 6.68 |
| 57.92 | 63.84 | — | 6.56 |
| 58.02 | 63.96 | — | 6.55 |



| Fuel Type | | | |
|---|---|---|---|
| A=GE8B-P8DQB323-10GZ-80M-4WR-145-T | (Cycle 9) | C=GE11-P9HUB378-15GZ-100T-141-T | (Cycle 11) |
| B=GE10-P8HXB355-11GZ-100M-145-T | (Cycle 10) | D=BP8DRB300 (BP8x8R) | (Cycle 8) |

**Figure 1 Reference Core Loading Pattern**

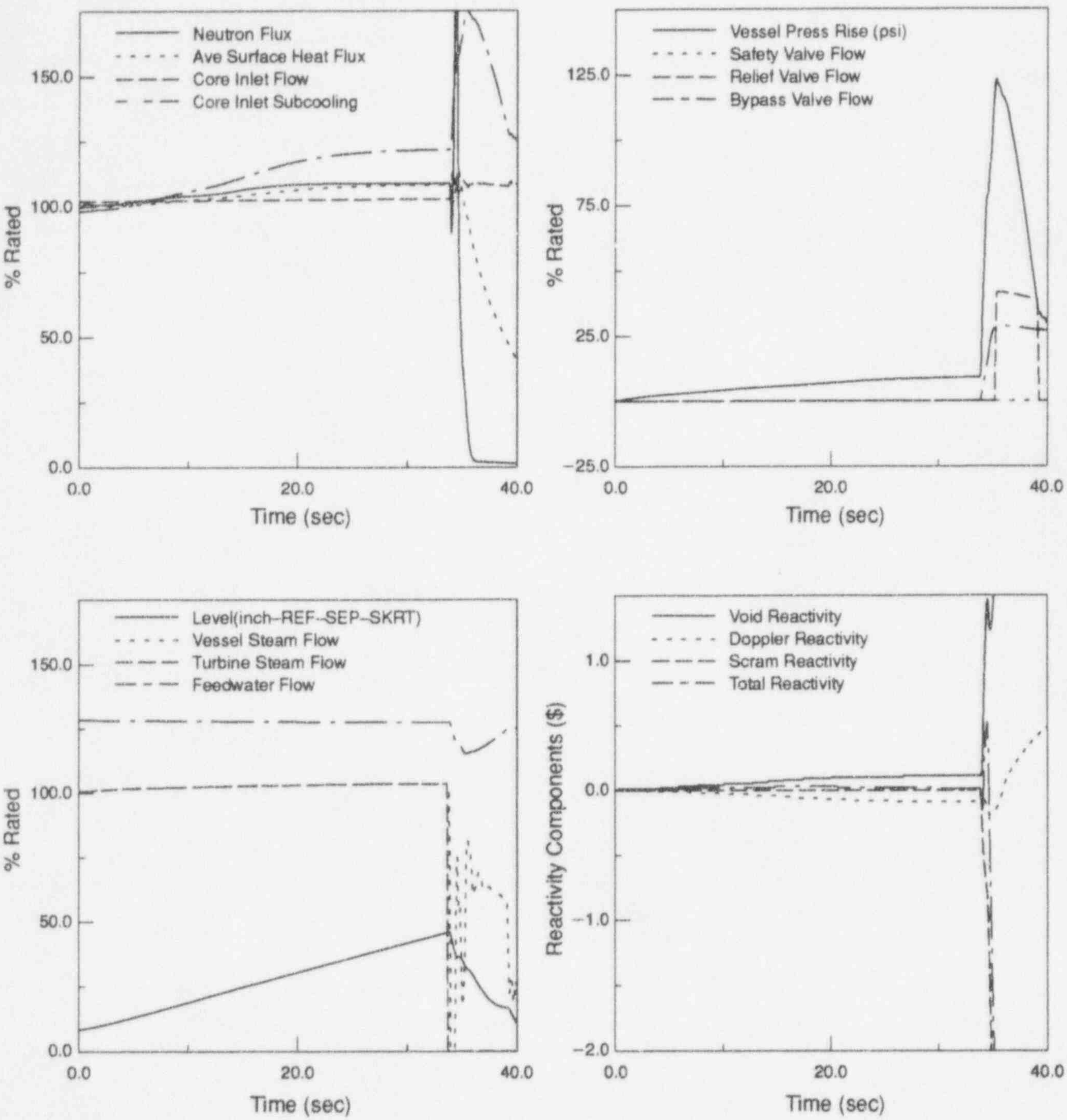



**Figure 2 Plant Response to FW Controller Failure (BOC11 to EOC11–5787 MWd/MT (5250 MWd/ST) ANALYZED AT 102% CORE FLOW)**

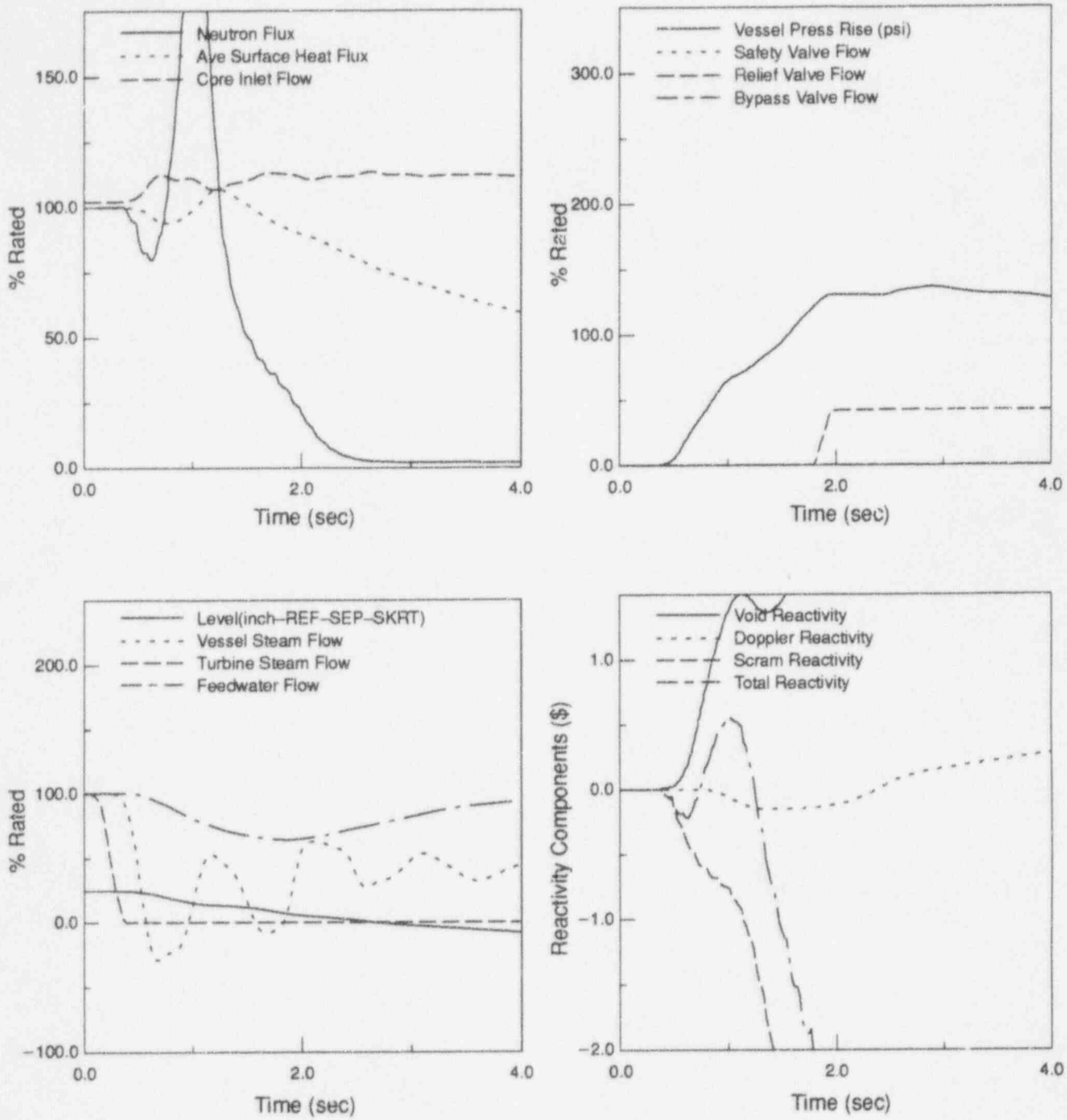



**Figure 3 Plant Response to Load Reject w/o Bypass (BOC11 to EOC11-5787 MWd/MT (5250 MWd/ST) ANALYZED AT 102% CORE FLOW)**

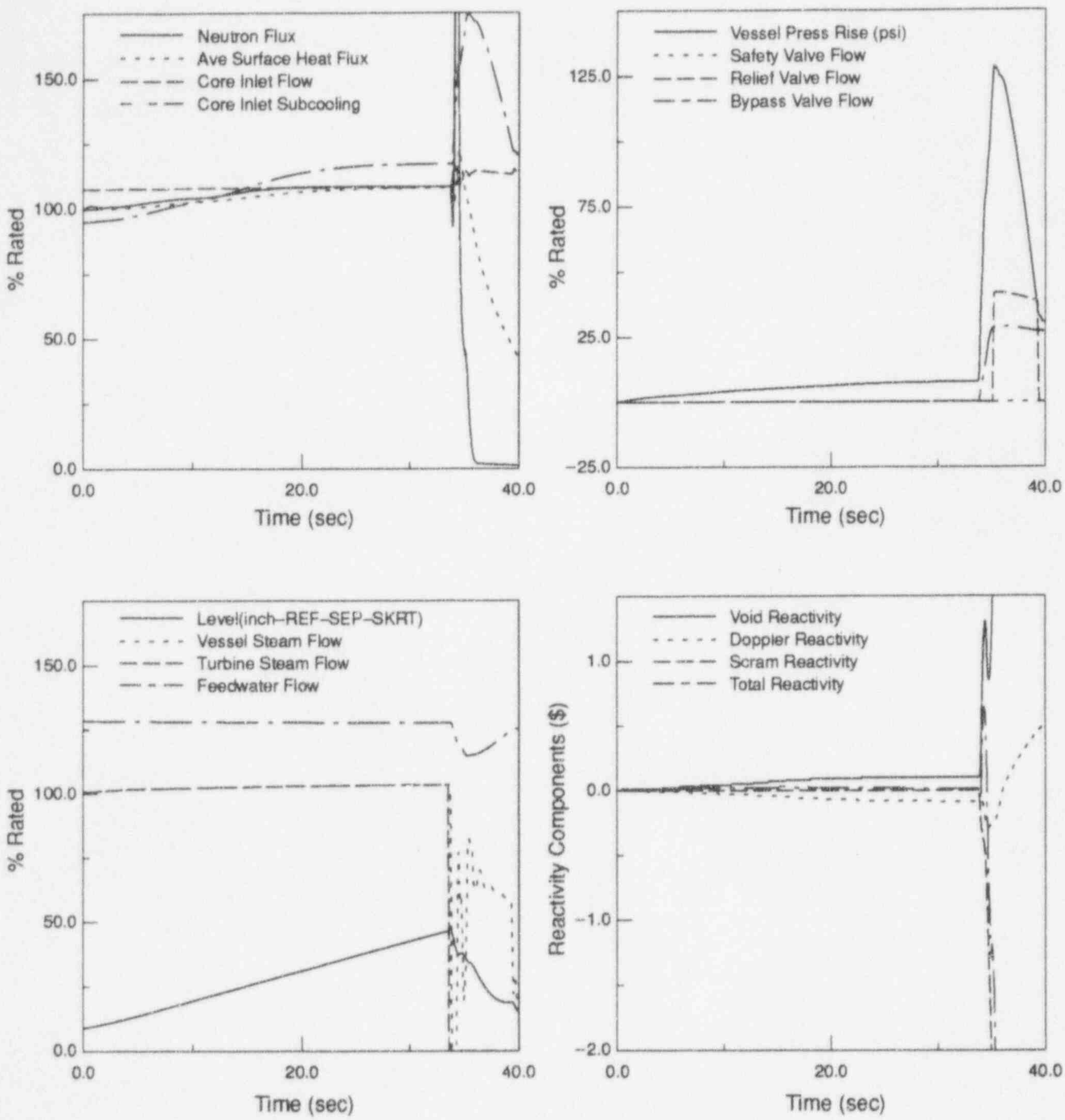



**Figure 4 Plant Response to FW Controller Failure (EOC11–5787 MWd/MT (5250 MWd/ST) to EOC11 ANALYZED AT 107.5% CORE FLOW)**

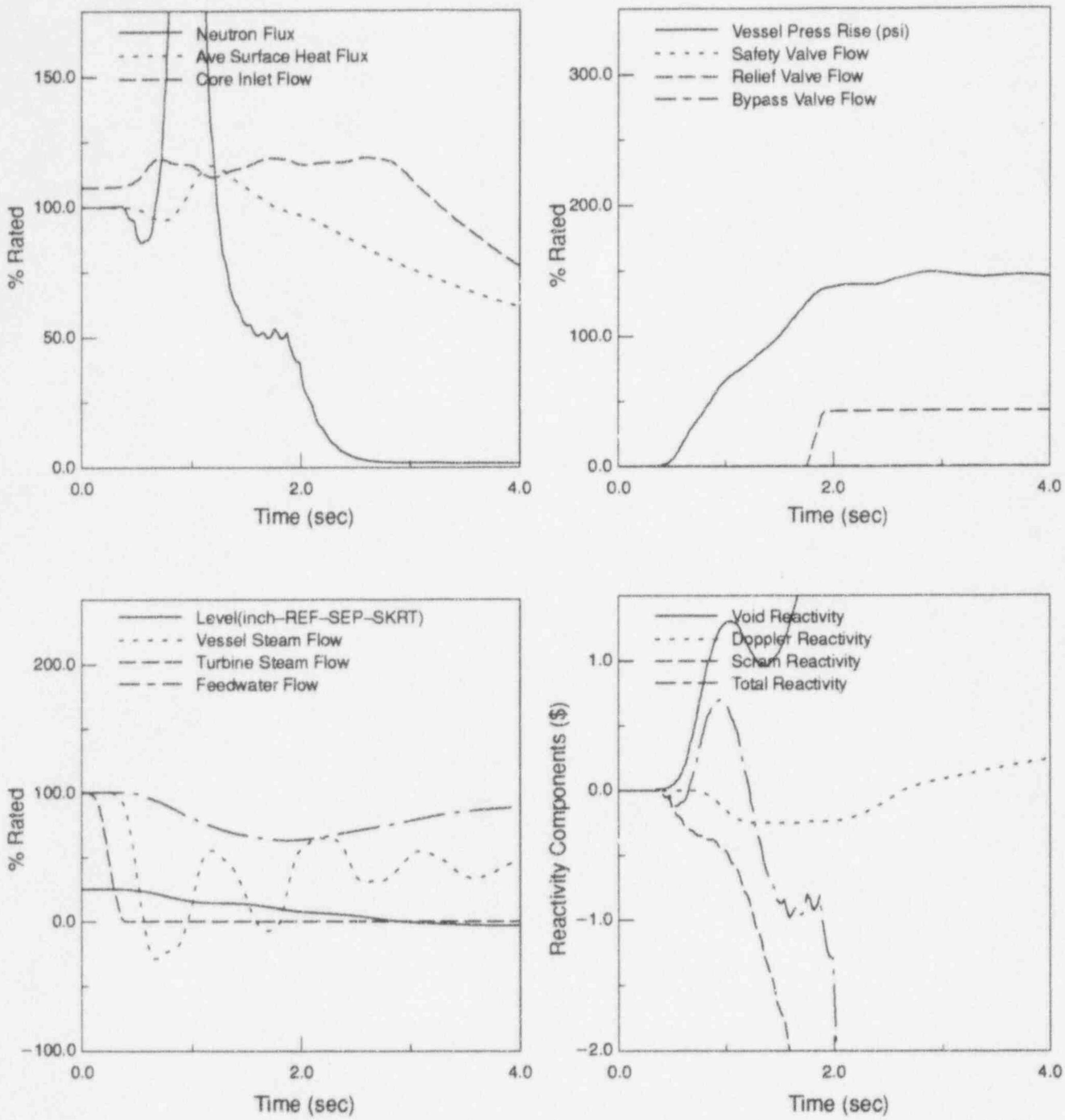



**Figure 5 Plant Response to Load Reject w/o Bypass (EOC11-5787 MWd/MT (5250 MWd/ST) to EOC11 ANALYZED AT 107.5% CORE FLOW)**

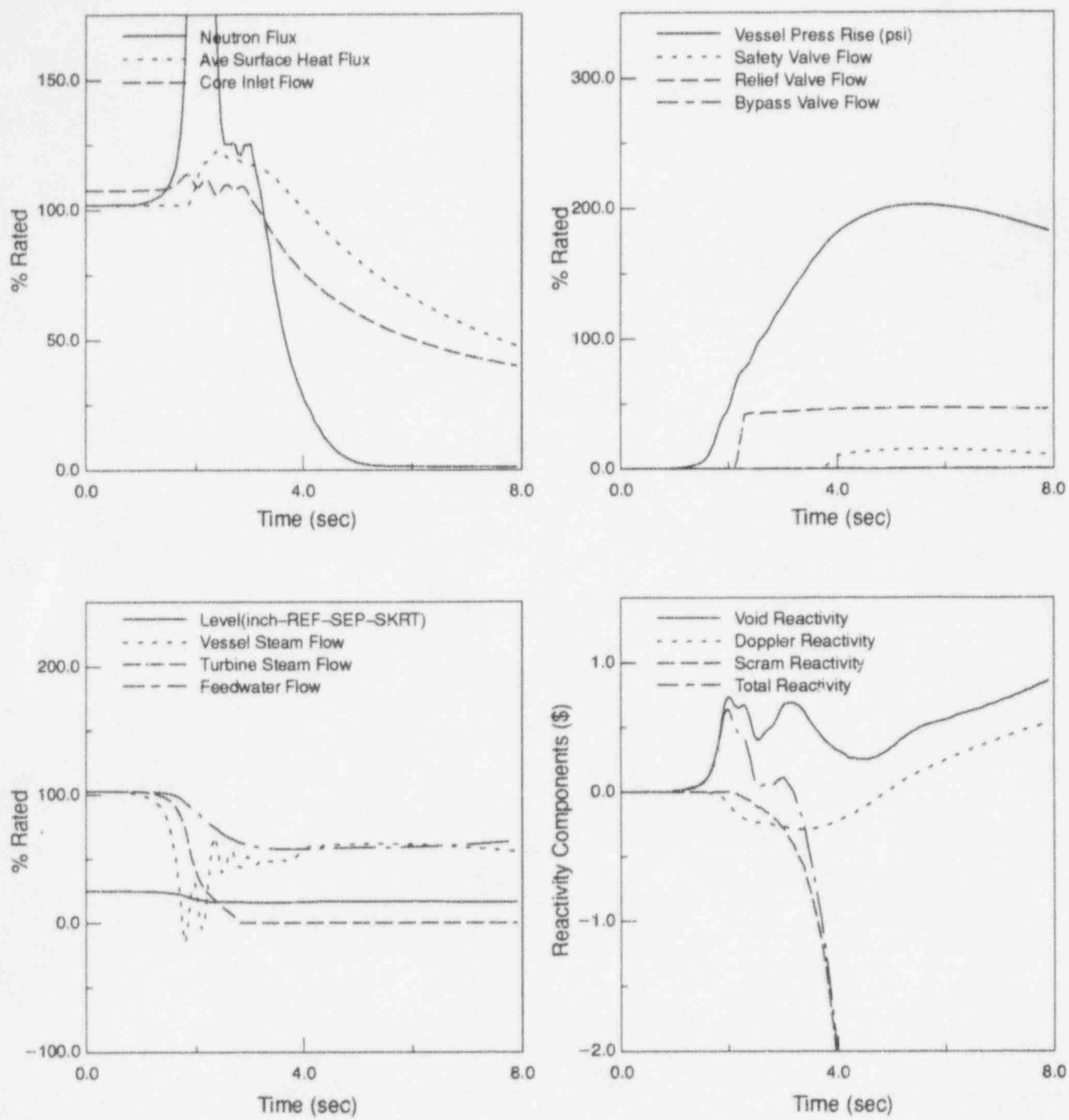



**Figure 6 Plant Response to MSIV Closure (Flux Scram)**

# Appendix A
# Analysis Conditions and Bases

To reflect actual plant parameters accurately, the values shown in Table A–1 were used this cycle. The use of the increased core flow for the analysis produces bounding results for the flow range down to 75% of rated core flow. Justification of operation at 100% power down to 75% power is provided in Reference A–1. The cycle 11 licensing analysis has verified the applicability of the MELLL flow range.

## Table A–1

| Parameter | 107.5% Flow Analysis Value |
|---|---|
| Thermal power, MWt | 1998.0 |
| Core flow, Mlb/hr | 74.2 |
| Reactor pressure, psia | 1066.5 |
| Inlet enthalpy, BTU/lb | 528.4 |
| Non–fuel power fraction | 0.038 |
| Steam flow analysis, Mlb/hr | 7.98 |
| Dome pressure, psig | 1035.8 |
| Turbine pressure, psig | 975.7 |
| No. of Safety/Relief Valves | 4 |
| No. of Single Spring Safety Valves | 2 |
| Relief mode lowest setpoint, psig | 1126.0 |
| Safety mode lowest setpoint, psig | 1253.0 |

For the overpressurization analysis, the MSIV closure (flux scram) case was analyzed at 102% licensed power and steamflow. Also, the maximum possible initial steam dome pressure of 1085 psig was used, which corresponds to the high pressure scram analytical limit. The most limiting end of cycle core conditions were utilized at 107.5% core flow, which produces a bounding result.

For the first introduction of GE11 fuel in Pilgrim, a plant specific evaluation was made of the GE11 fuel "Scram Speed Adjustment Factors" (SSAF) that adjust the option B MCPR limit to optain the Option A MCPR limit. This evaluation concluded that use of a 0.06 EOC scram speed adjustment factor is justified for the load rejection, turbine trip, and feedwater controller failure pressurization events. For Pilgrim GE11 fuel application, this supercedes the "generic" EOC value from the letter, J.F. Klapproth to USNRC, "GEMINI/ODYN Statistical Adders for GE11 fuel for BWR/2 and 3", September 23, 1992. The 0.08 generic mid-cycle adders are still applicable to GE11 fuel in Pilgrim.

A–1 . H.X. Hoang, *"Maximum Extended Load Line Limit Analyses for Pilgrim Nuclear Power Station Reload 9 Cycle 10"*, NEDC–32306P, March, 1994.

# Appendix B
# Increased Core Flow

The analyses performed for Cycle 11 included increased core flow throughout the cycle and after the all–rods–out condition is reached. There are no concerns regarding reactor internals pressure drop or flow–induced vibration as discussed in the increased core flow analysis document for the EOC–6 (NEDO–30242)..

# Appendix C
# Decrease in Core Coolant Temperature Events

The loss-of-feedwater heating (LFWH) and the high pressure coolant injection (HPCI) inadvertent start-up anticipated operational occurrences (AOO) are the only cold water injection events checked on a cycle-by-cycle basis. For both the LFWH and HPCI events, the delta CPR is not limiting when compared to the delta CPR of the liming pressurization AOO. This is based on the results of calculations performed with consideration of the cycle-to-cycle differences such as ARTS. Therefore, the LFWH and HPCI inadvertent start-up AOOs are not reported for Cycle 11.

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**Reload 10 Cycle 11**

Approved J.F. Klapproth

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Fuel and Facility Licensing

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Fuel Project Manager

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Appendix C: Decrease in Core Coolant Temperature Events

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| GE8B-P8DQB323-10GZ-80M-4WR-145-T (GE8x8EB) | 9 | 168 |
| GE10-P8HXB355-11GZ-100M-145-T (GE8x8NB-3) | 10 | 140 |
| New: | | |
| GE11-P9HUB378-15GZ-100T-141-T (GE11) | 11 | 136 |
| Total | | 580 |

## 3. Reference Core Loading Pattern

| | |
|---|---|
| Nominal previous cycle core average exposure at end of cycle: | 24245 MWd/MT ( 21994 MWd/ST) |
| Minimum previous cycle core average exposure at end of cycle from cold shutdown considerations: | 23804 MWd/MT ( 21594 MWd/ST) |
| Assumed reload cycle core average exposure at beginning of cycle: | 17222 MWd/MT ( 15623 MWd/ST) |
| Assumed reload cycle core average exposure at end of cycle: | 28520 MWd/MT ( 25873 MWd/ST) |
| Reference core loading pattern: | Figure 1 |

## 4. Calculated Core Effective Multiplication and Control System Worth – No Voids, 20°C

| | |
|---|---|
| Beginning of Cycle, $k_{effective}$ | |
| Uncontrolled | 1.103 |
| Fully controlled | 0.960 |
| Strongest control rod out | 0.986 |
| R, Maximum increase in cold core reactivity with exposure into cycle, $\Delta k$ | 0.003 |

## 5. •Standby Liquid Control System Shutdown Capability

| Boron (ppm) | Shutdown Margin (Δk) (20°C, Xenon Free) |
|---|---|
| 675 | 0.042 |

## 6. Reload Unique GETAB Anticipated Operational Occurrences (AOO) Analysis Initial Condition Parameters[1]

| Exposure: BOC11 to EOC11–5787 MWd/MT (5250 MWd/ST) ANALYZED AT 102% CORE FLOW | | | | | | | |
|---|---|---|---|---|---|---|---|
| | Peaking Factors | | | | | | |
| **Fuel Design** | **Local** | **Radial** | **Axial** | **R–Factor** | **Bundle Power (MWt)** | **Bundle Flow (1000 lb/hr)** | **Initial MCPR** |
| GE11 | 1.45 | 1.85 | 1.31 | 1.035 | 6.209 | 91.9 | 1.30 |

| Exposure: EOC11–5787 MWd/MT (5250 MWd/ST) to EOC11 ANALYZED AT 107.5% CORE FLOW | | | | | | | |
|---|---|---|---|---|---|---|---|
| | Peaking Factors | | | | | | |
| **Fuel Design** | **Local** | **Radial** | **Axial** | **R–Factor** | **Bundle Power (MWt)** | **Bundle Flow (1000 lb/hr)** | **Initial MCPR** |
| GE11 | 1.45 | 1.75 | 1.18 | 1.035 | 5.859 | 101.2 | 1.37 |

## 7. Selected Margin Improvement Options

| | |
|---|---|
| Recirculation pump trip: | No |
| Rod withdrawal limiter: | No |
| Thermal power monitor: | No |
| Improved scram time: | Yes (ODYN Option B) |
| Measured scram time: | No |
| Exposure dependent limits: | Yes |
| Exposure points analyzed: | 2 |

1. The delta CPR response for GE11 bounds all other fuel types in the core.

## 8. Operating Flexibility Options

| | |
|---|---|
| Single–loop operation: | Yes[2] |
| Load line limit: | Yes |
| Extended load line limit: | Yes |
| Maximum extended load line limit: | Yes[3] |
| Increased core flow throughout cycle: | Yes[4] |
| Increased core flow at EOC: | Yes |
| BOC to 5000 MWD/STU Flow point analyzed: | 102 % |
| 5000 MWD/STU to EOC Flow point analyzed: | 107.5 % |
| Feedwater temperature reduction throughout cycle: | No |
| Final feedwater temperature reduction: | No |
| ARTS Program: | Yes[5] |
| Moisture separator reheater OOS: | No |
| Turbine bypass system OOS: | No |
| Safety/relief valves OOS: | No |
| ADS OOS: | No |
| One Main steam isolation valve OOS: | Yes[6] |

2. "Pilgrim Nuclear Power Station Single–Loop Operation", NEDO–24268,June,1980

3. H.X.Hoang,"Maximum Extended Load Line Limit Analyses for Pilgrim Nuclear Power Station Reload 9 Cycle 10", NEDC–32306P,MARCH 1994.

4. "Safety Review of Pilgrim Nuclear Power Station Unit No. 1 at Core Flow Conditions Above Rated Flow Throughout Cycle 6", NEDO–30242, August, 1983.

5. "ARTS Improvement Program Analysis for Pilgrim Nuclear Power Station", NEDO–31312P,September, 1987.

6. MSIV Out of Service Report, NSE–82–0982, DRF B21–00238, September 1982.

## 9. Core–wide AOO Analysis Results[7][8]

**Methods used:** GEMINI; GEXL–PLUS

| **Exposure range: BOC11 to EOC11–5787 MWd/MT (5250 MWd/ST) ANALYZED AT 102% CORE FLOW** | | | | |
|---|---|---|---|---|
| | | | **Uncorrected ΔCPR** | |
| **Event** | **Flux (%NBR)** | **Q/A (%NBR)** | **GE11** | **Fig.** |
| FW Controller Failure | 225 | 115 | 0.23 | 2 |
| Load Reject w/o Bypass | 230 | 107 | 0.18 | 3 |

| **Exposure range: EOC11–5787 MWd/MT (5250 MWd/ST) to EOC11 ANALYZED AT 107.5% CORE FLOW** | | | | |
|---|---|---|---|---|
| | | | **Uncorrected ΔCPR** | |
| **Event** | **Flux (%NBR)** | **Q/A (%NBR)** | **GE11** | **Fig.** |
| FW Controller Failure | 304 | 123 | 0.30 | 4 |
| Load Reject w/o Bypass | 332 | 116 | 0.27 | 5 |

## 10. Local Rod Withdrawal Error (With Limiting Instrument Failure) AOO Summary[9]

Rod withdrawal error (RWE) is analyzed in General Electric BWR Licensing Report, *Average Power Range Monitor, Rod Block monitor and Technical Specification Improvement (ARTS) Program*, NEDC–30474–P, dated December 1983. A cycle–specific rod withdrawal analysis found the ΔMCPR is bounded by the generic RWE analysis reported in the referenced report. For a setpoint of up to 116%, the rated MCPR limit is 1.35.

7. The GE11 delta CPR response bounds all other fuel types in the core.

8. Analysis at 107.5% increased core flow was conservatively assumed for EOC11 analysis; 102% increased core flow was assumed as the BOC to EOC11–5250 MWd/ST early cycle analysis basis.

9. References: "ARTS Improvement Program Analysis for Pilgrim Nuclear Power Station", NEDC–31312–P, September, 1987 and H.X. Hoang, "ARTS Verification for Pilgrim Nuclear Power Station Reload 8 Cycle 9, GE–NE–187–11–0691, DRF A00–03980, June 1991. These documents were verified applicable to cycle 11.

## 11. Cycle MCPR Values[10 11 12]

Safety limit: 1.07

Single loop operation safety limit: 1.08

### Non-pressurization events:

| Exposure Range: BOC11 to EOC11 | |
|---|---|
| Rod Withdrawal Error (Setpoint can be selected up to 116%), All Fuels | **1.35** |
| Fuel Loading Error, GE11 Reload 10 Fuel | **1.12** |
| Fuel Loading Error, GE10 Reload 9 Fuel | **1.27** |

### Pressurization events:

| **Exposure range: BOC11 to EOC11-5787 MWd/MT (5250 MWd/ST) ANALYZED AT 102% CORE FLOW<br>Exposure point: EOC11-5787 MWd/MT (5250 MWd/ST)** | | |
|---|---|---|
| | **Option A** | **Option B** |
| | **GE11** | **GE11** |
| FW Controller Failure | 1.40 | 1.32 |
| Load Reject w/o Bypass | 1.35 | 1.27 |

| **Exposure range: EOC11-5787 MWd/MT (5250 MWd/ST) to EOC11 ANALYZED AT 107.5% CORE FLOW<br>Exposure point: EOC11** | | |
|---|---|---|
| | **Option A** | **Option B** |
| | **GE11** | **GE11** |
| FW Controller Failure | 1.45 | 1.39 |
| Load Reject w/o Bypass | 1.43 | 1.37 |

## 12. Overpressurization Analysis Summary

| Event | Psl (psig) | Pv (psig) | Plant Response |
|---|---|---|---|
| MSIV Closure (Flux Scram) | 1286 | 1302 | Figure 8 |

10. The minimum MCPR operating limit required by the SAFER/GESTR analysis is 1.20.

11. See Appendix C for discussion of decrease in core coolant temperature events.

12. For single-loop operation, the MCPR operating limit is not greater than the two-loop value.

## 13. Loading Error Results

From a misoriented bundle analysis with variable water gap, including a 0.02 penalty due to variable water gap R–factor uncertainty, the ΔMCPR for the fresh reload 10 GE11 fuel bundle is 0.05. The ΔMCPR for the reload 9 GE10 fuel bundle is 0.20.

## 14. Control Rod Drop Analysis Results

This is a banked position withdrawal sequence plant, therefore, the control rod drop accident analysis is not required. NRC approval is documented in NEDE–24011–P–A–US.

## 15. Stability Analysis Results

Pilgrim Nuclear Power Station is exempt from the current requirement to submit a cycle–specific stability analysis as documented in the letter, C. O. Thomas (NRC) to H. C. Pfefferlen (GE), *Acceptance for Referencing of Licensing Topical Report NEDE–24011 Rev. 6, Amendment 8, "Thermal Hydraulic Stability Amendment to GESTAR II,"* April 24, 1985.

Pilgrim Nuclear Power Station recognizes the issuance of NRC Bulletin No. 88–07, Supplement 1, *Power Oscillations in Boiling Water Reactors (BWRs)*, and will comply with the recommendations contained therein. Pilgrim Station is also complying with the NRC Bulletin No. 94–02, *Long–Term Solutions and Upgrade of Interim Operating Recommendations for Thermal–Hydraulic Instabilities in Boiling Water Reactors.* GE11 fuel has been demonstrated to have equivalent or better stability characteristics than BP8x8R fuel by the GESTAR Amendment 22 licensing analysis (Reference: NEDE 31917P, *GE11 Compliance with Amendment 22 of NEDE–24011–P–A (GESTRAR II), April 1991)*, and no unique or special actions are needed to comply with the above NRC Bulletins.

## 16. Loss–of–Coolant Accident Results

**LOCA method used:** SAFER/GESTR–LOCA

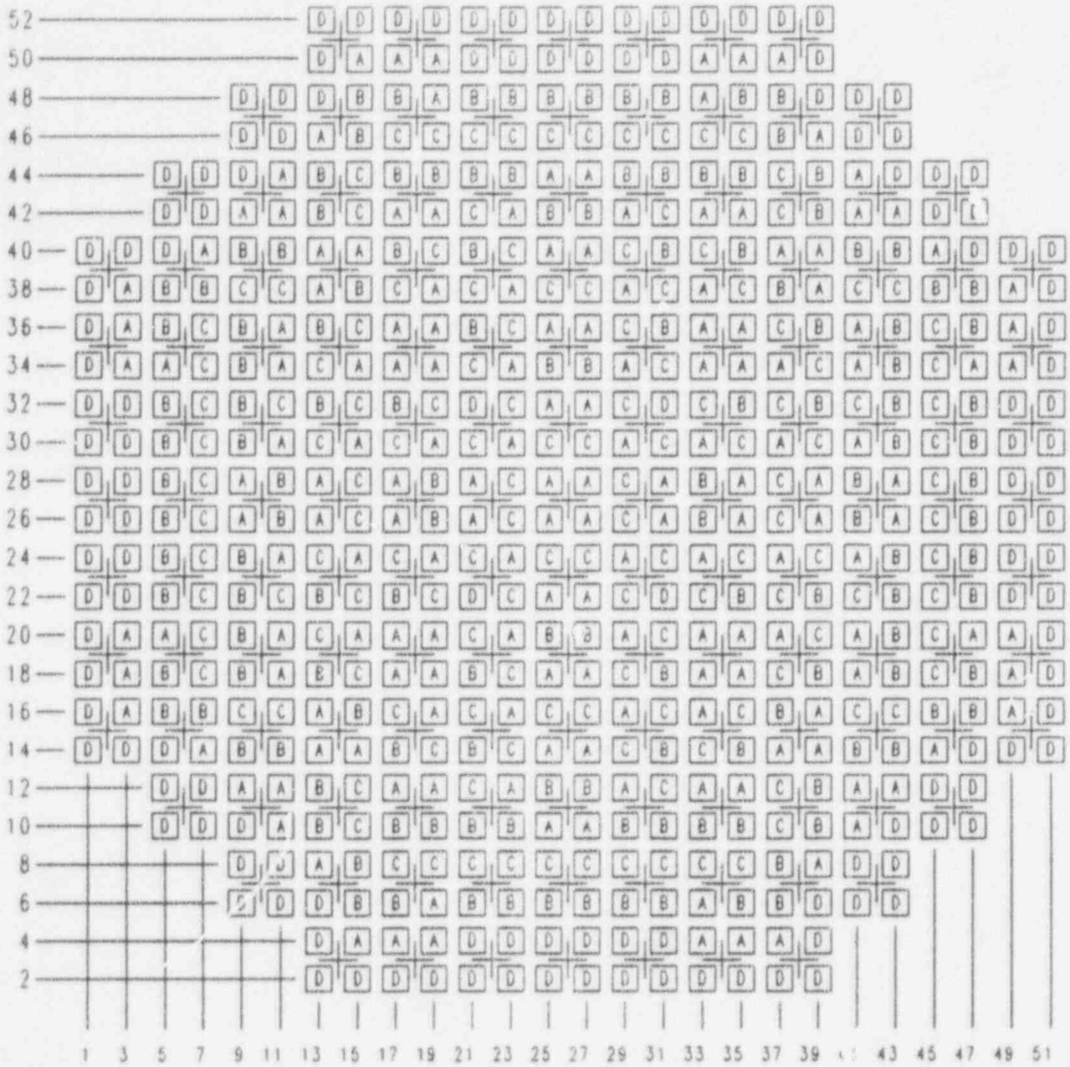
The LOCA analysis results are presented in Sections 5 and 6 of *Pilgrim Nuclear Power Station SAFER/GESTR–LOCA Loss–of–Coolant Accident Analysis*, NEDC–31852P, April, 1992 (Revision 1) as amended. The GE11 LOCA analysis for Pilgrim was performed using the same SAFER/GESTR analysis basis used for the previously analyzed BP/P8x8R and GE8x8EB/NB fuel types. Addition of the GE11 fuel will not significantly affect the overall system response of the plant for the various operating modes, and the GE11 analysis confirmed that the limiting break type and size and limiting ECCS failure (DBA recirculation suction line break with LPCIIV failure) do not change. The GE11 fuel analysis yielded a licensing basis peak PCT of 1815 °F and a peak local oxidation fraction of <0.3%, and all licensing basis criteria are met. The GE11 results are bounded by the 1825 °F licensing basis PCT for BP/P8x8R fuel and the overall licensing basis results reported in Table 6–1 of the Reference analysis.

The GE11 SAFER/GESTR results are applicable for a peak enriched lattice MAPLHGR of 12.16 kw/ft.,which bounds the MAPLHGRs for the reload 10 fuel. Therefore, the MAPLHGR limits reflect the thermal–mechanical limits for the reload fuel rather than LOCA/ECCS considerations. The most limiting and the least limiting MAPLHGRs for the new fuel are as follows:

**16. Loss–of–Coolant Accident Results (cont)**

Bundle Type: GE11–P9HUB378–15GZ–100T–141–T

| Average Planar Exposure | | MAPLHGR(kW/ft) | |
|---|---|---|---|
| (GWd/ST) | (GWd/MT) | Most Limiting | Least Limiting |
| 0.00 | 0.00 | 9.95 | 10.46 |
| 0.20 | 0.22 | 10.04 | 10.53 |
| 1.00 | 1.10 | 10.19 | 10.61 |
| 2.00 | 2.20 | 10.41 | 10.78 |
| 3.00 | 3.31 | 10.64 | 10.98 |
| 4.00 | 4.41 | 10.88 | 11.21 |
| 5.00 | 5.51 | 11.09 | 11.35 |
| 6.00 | 6.61 | 11.19 | 11.55 |
| 7.00 | 7.72 | 11.30 | 11.63 |
| 8.00 | 8.82 | 11.40 | 11.70 |
| 9.00 | 9.92 | 11.52 | 11.79 |
| 10.00 | 11.02 | 11.65 | 11.91 |
| 12.50 | 13.78 | 11.64 | 11.92 |
| 15.00 | 16.53 | 11.48 | 11.71 |
| 17.50 | 19.29 | 11.25 | 11.47 |
| 20.00 | 22.05 | 11.02 | 11.22 |
| 25.00 | 27.56 | 10.55 | 10.75 |
| 30.00 | 33.07 | 10.07 | 10.22 |
| 35.00 | 38.58 | 9.39 | 9.53 |
| 40.00 | 44.09 | 8.72 | 8.87 |
| 45.00 | 49.60 | 8.06 | 8.23 |
| 50.00 | 55.12 | 7.40 | 7.60 |
| 55.00 | 60.63 | 6.72 | 6.96 |
| 57.02 | 62.86 | 6.44 | 6.69 |
| 57.10 | 62.94 | — | 6.68 |
| 57.92 | 63.84 | — | 6.56 |
| 58.02 | 63.96 | — | 6.55 |



| Fuel Type | |
|---|---|
| A=GE8B–P8DQB323–10GZ–80M–4WR–145–T (Cycle 9)<br>B=GE10–P8HXB355–11GZ–100M–145–T (Cycle 10) | C=GE11–P9HUB378–15GZ–100T–141–T (Cycle 11)<br>D=BP8DRB300 (BP8x8R) (Cycle 8) |

**Figure 1 Reference Core Loading Pattern**

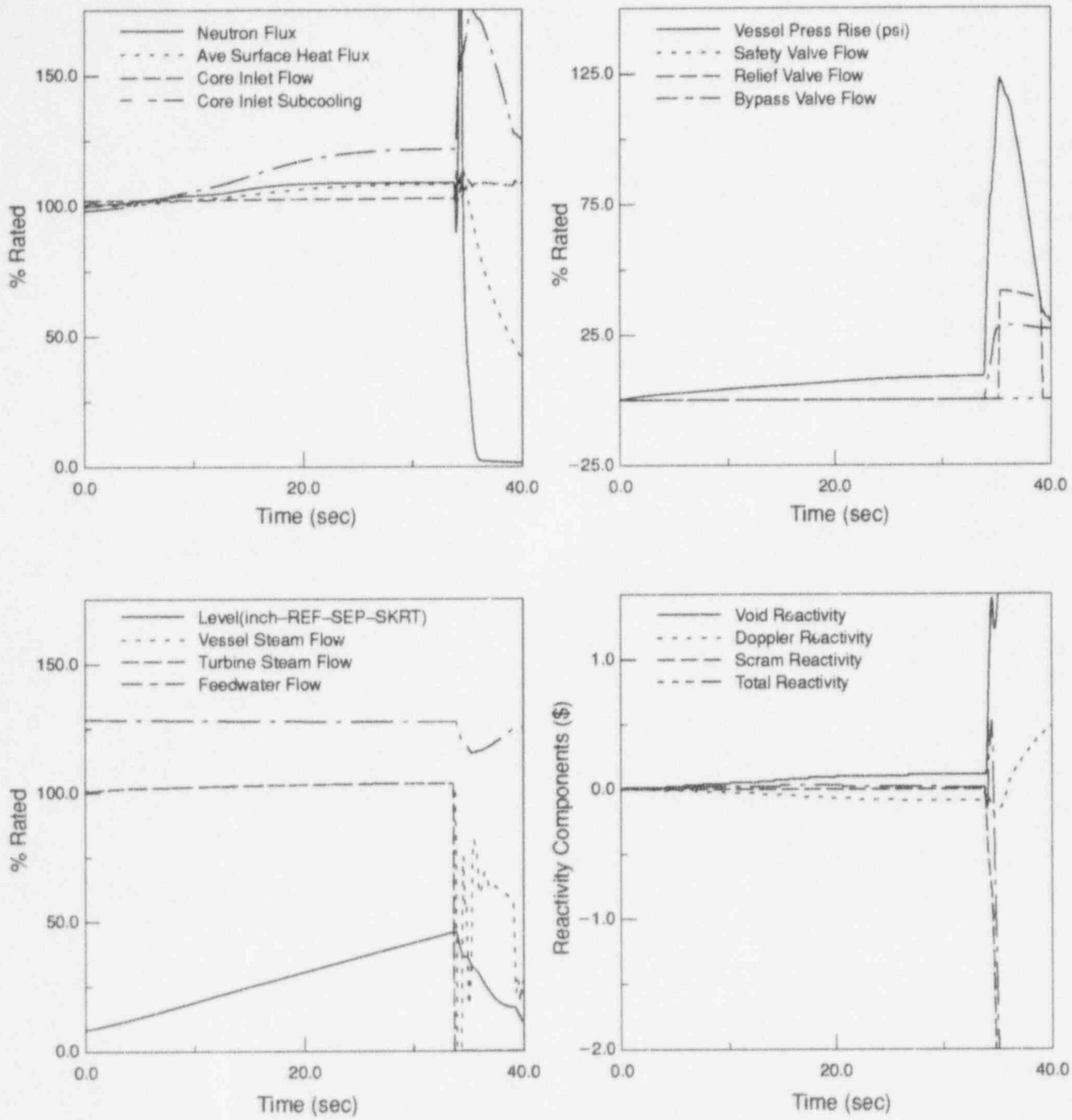



**Figure 2 Plant Response to FW Controller Failure (BOC11 to EOC11-5787 MWd/MT (5250 MWd/ST) ANALYZED AT 102% CORE FLOW)**

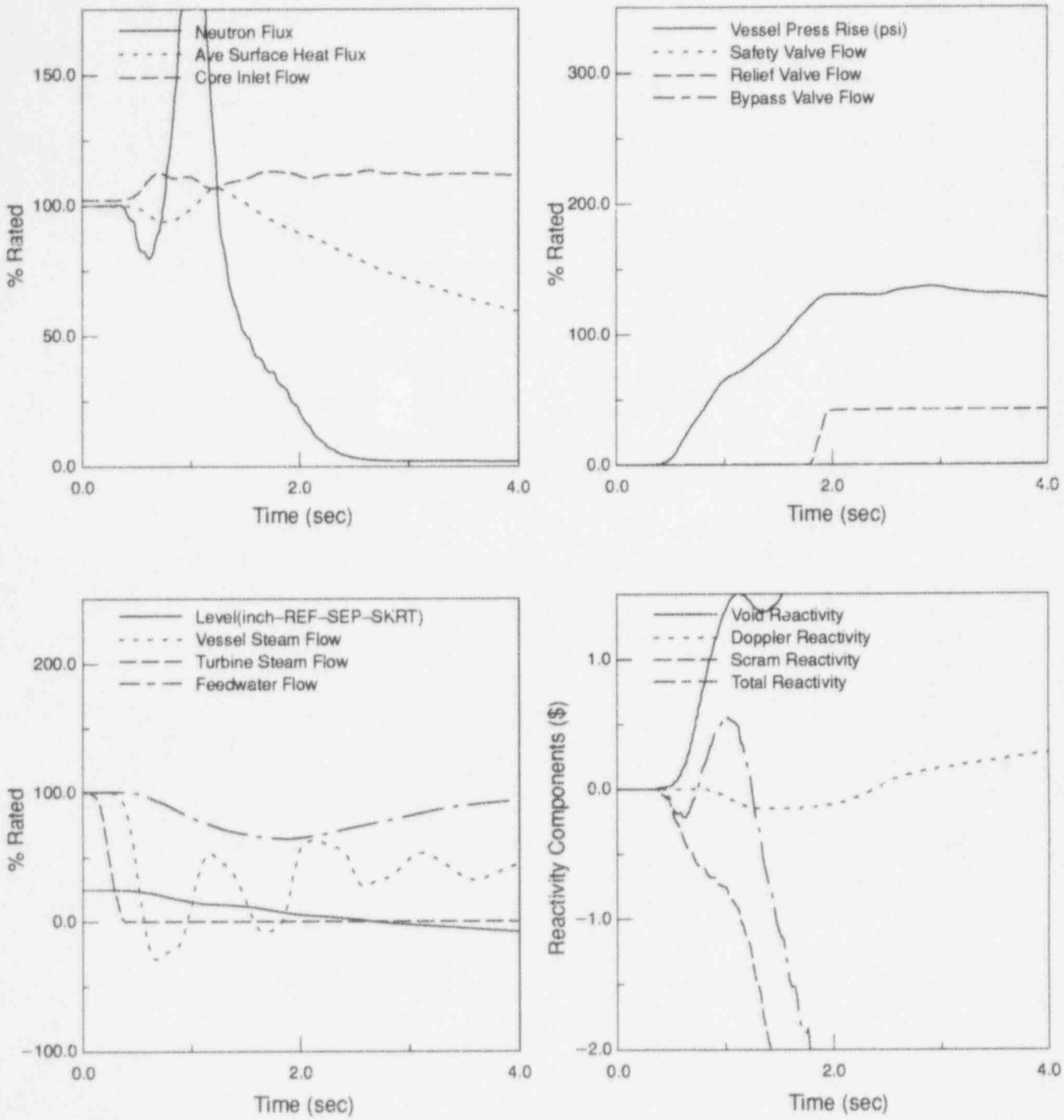



**Figure 3 Plant Response to Load Reject w/o Bypass (BOC11 to EOC11-5787 MWd/MT (5250 MWd/ST) ANALYZED AT 102% CORE FLOW)**

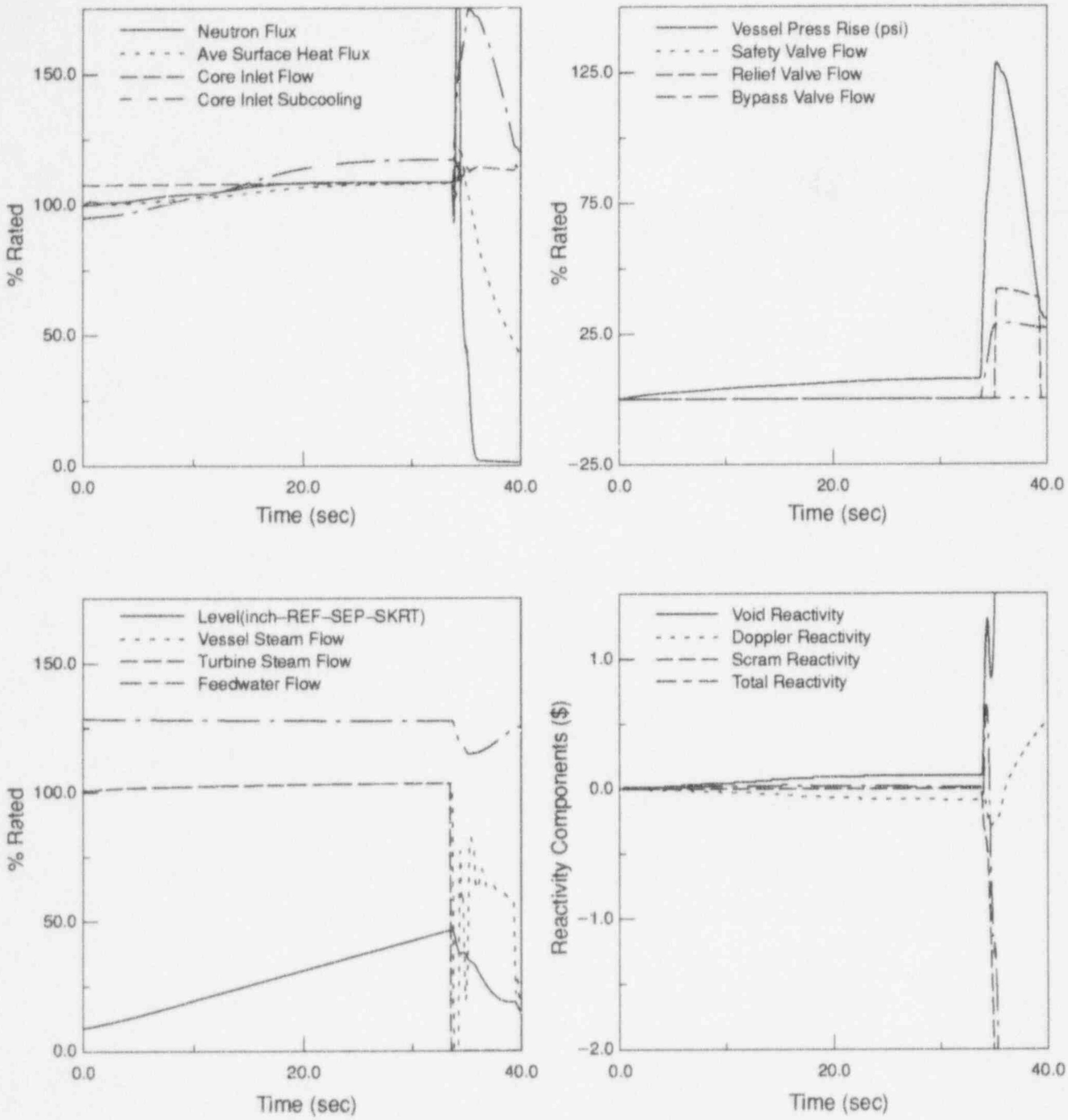



**Figure 4 Plant Response to FW Controller Failure (EOC11–5787 MWd/MT (5250 MWd/ST) to EOC11 ANALYZED AT 107.5% CORE FLOW)**

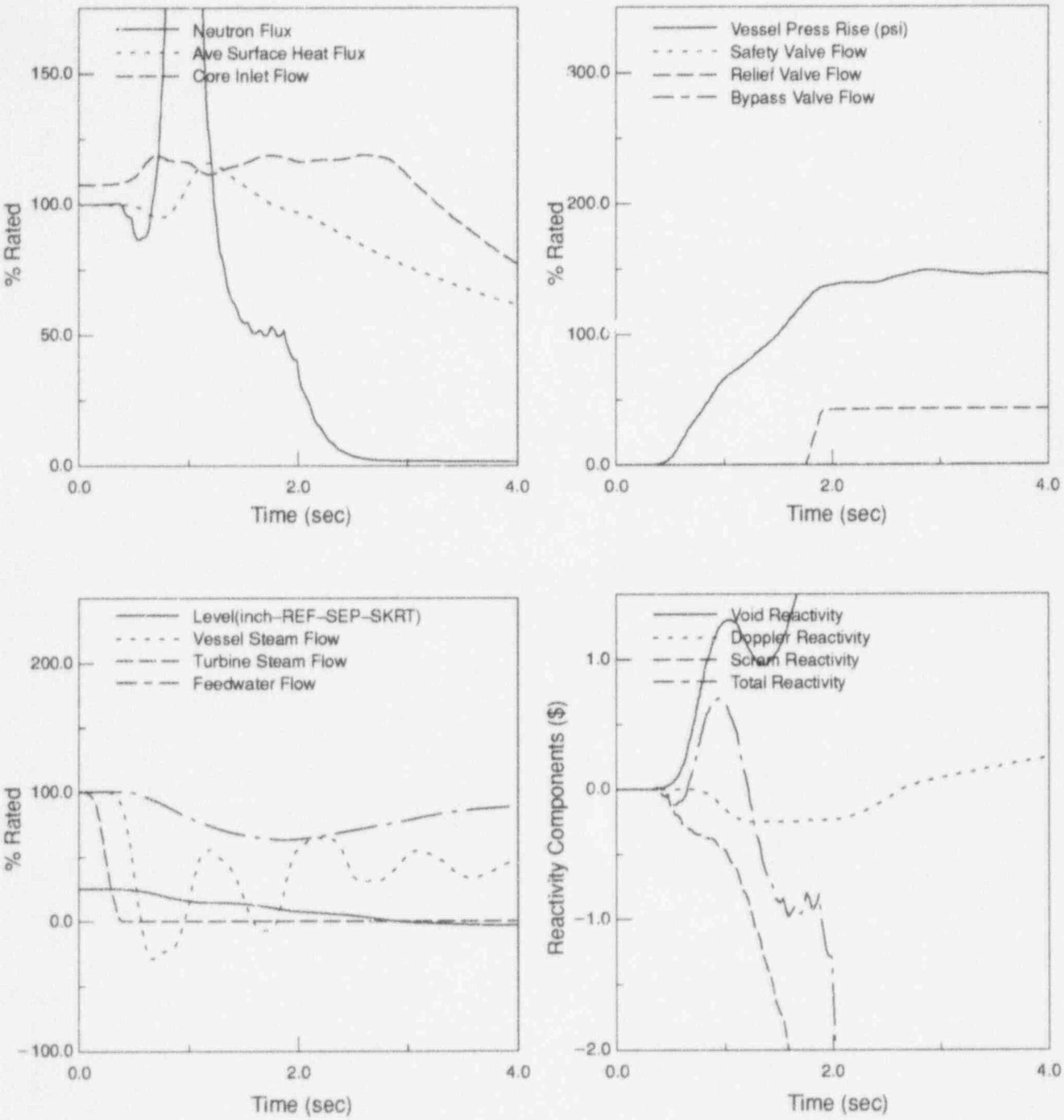



**Figure 5 Plant Response to Load Reject w/o Bypass (EOC11–5787 MWd/MT (5250 MWd/ST) to EOC11 ANALYZED AT 107.5% CORE FLOW)**

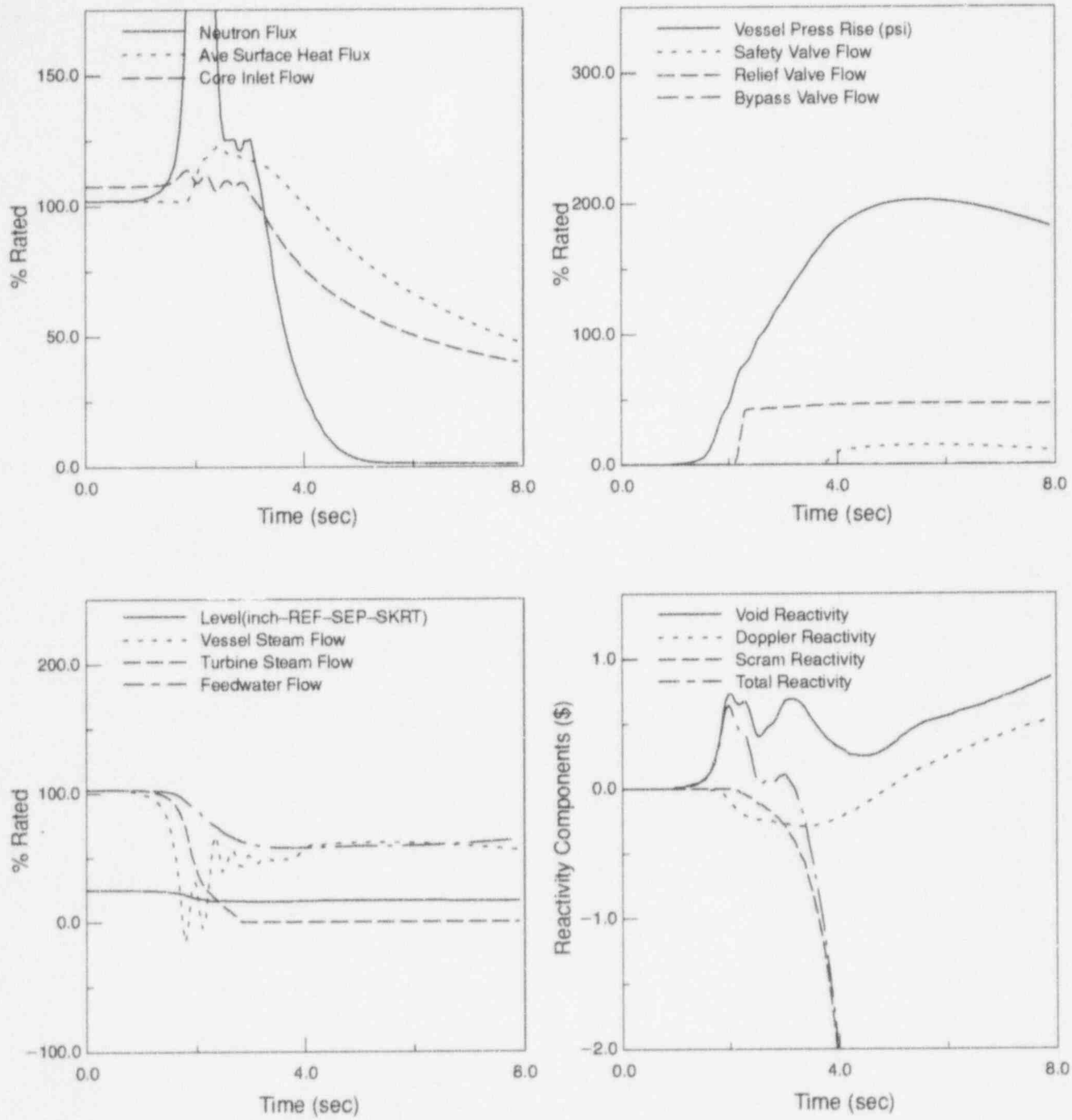



**Figure 6 Plant Response to MSIV Closure (Flux Scram)**

# Appendix A
# Analysis Conditions and Bases

To reflect actual plant parameters accurately, the values shown in Table A–1 were used this cycle. The use of the increased core flow for the analysis produces bounding results for the flow range down to 75% of rated core flow. Justification of operation at 100% power down to 75% power is provided in Reference A–1. The cycle 11 licensing analysis has verified the applicability of the MELLL flow range.

## Table A–1

| Parameter | 107.5% Flow Analysis Value |
|---|---|
| Thermal power, MWt | 1998.0 |
| Core flow, Mlb/hr | 74.2 |
| Reactor pressure, psia | 1066.5 |
| Inlet enthalpy, BTU/lb | 528.4 |
| Non–fuel power fraction | 0.038 |
| Steam flow analysis, Mlb/hr | 7.98 |
| Dome pressure, psig | 1035.8 |
| Turbine pressure, psig | 975.7 |
| No. of Safety/Relief Valves | 4 |
| No. of Single Spring Safety Valves | 2 |
| Relief mode lowest setpoint, psig | 1126.0 |
| Safety mode lowest setpoint, psig | 1253.0 |

For the overpressurization analysis, the MSIV closure (flux scram) case was analyzed at 102% licensed power and steamflow. Also, the maximum possible initial steam dome pressure of 1085 psig was used, which corresponds to the high pressure scram analytical limit. The most limiting end of cycle core conditions were utilized at 107.5% core flow, which produces a bounding result.

For the first introduction of GE11 fuel in Pilgrim, a plant specific evaluation was made of the GE11 fuel "Scram Speed Adjustment Factors" (SSAF) that adjust the option B MCPR limit to optain the Option A MCPR limit. This evaluation concluded that use of a 0.06 EOC scram speed adjustment factor is justified for the load rejection, turbine trip, and feedwater controller failure pressurization events. For Pilgrim GE11 fuel application, this supercedes the "generic" EOC value from the letter, J.F. Klapproth to USNRC, "GEMINI/ODYN Statistical Adders for GE11 fuel for BWR/2 and 3", September 23, 1992. The 0.08 generic mid-cycle adders are still applicable to GE11 fuel in Pilgrim.

A–1 . H.X. Hoang, *"Maximum Extended Load Line Limit Analyses for Pilgrim Nuclear Power Station Reload 9 Cycle 10"*, NEDC–32306P, March, 1994.

# Appendix B
# Increased Core Flow

The analyses performed for Cycle 11 included increased core flow throughout the cycle and after the all–rods–out condition is reached. There are no concerns regarding reactor internals pressure drop or flow–induced vibration as discussed in the increased core flow analysis document for the EOC–6 (NEDO–30242)..

# Appendix C
# Decrease in Core Coolant Temperature Events

The loss–of–feedwater heating (LFWH) and the high pressure coolant injection (HPCI) inadvertent start–up anticipated operational occurrences (AOO) are the only cold water injection events checked on a cycle–by–cycle basis. For both the LFWH and HPCI events, the delta CPR is not limiting when compared to the delta CPR of the liming pressurization AOO. This is based on the results of calculations performed with consideration of the cycle–to–cycle differences such as ARTS. Therefore, the LFWH and HPCI inadvertent start–up AOOs are not reported for Cycle 11.

Attachment 2

Memo to E.L. Heinlein from G. A. Watford dated February 3, 1992;
Subject: PNPS Technical Specification Scram Time
Requirements - DRF A12 - 00038 - 2

February 3, 1992

cc: J.S. Charnley
E.G. Thacker
D.C. Serell
DRF A12-00038-2

To: E.L. Heinlein

From: G.A. Watford

Subject: PNPS Technical Specification Scram Time Requirements

Reference:
1. Letter, R.V. Fairbank (BECo) to E.L. Heinlein (GE), same subject, 11/21/91.
2. Letter, R.V. Fairbank (BECo) to E.L. Heinlein (GE), same subject, 12/18/91.
3. Letter, G.A. Watford to E.L. Heinlein, same subject, 1/22/92.

This letter summarizes the information provided in Reference 3 and also provides additional information concerning the GEMINI scram times. The responses are also provided in the same format as the questions of References 1 and 2.

1) Average scram insertion time requirements for all operable control rods (TS 3.3.b.1) from deenergization of the scram pilot valve solenoids to dropout (DO) (reed switch opening) of Notches 04, 24, 34, and 44.

| Notch Position | Average Scram Time (seconds) |
|---|---|
| 44 DO | 0.504 |
| 34 DO | 1.249 |
| 24 DO | 2.013 |
| 04 DO | 3.575 |

2) Average scram insertion time requirements for the three fastest control rods in each group of four control rods in all two-by-two arrays (TS 3.3.c.2) from deenergization of the scram pilot valve solenoids to dropout of Notches 04, 24, 34, and 44.

| Notch Position | 3 out of 4 Scram Time (seconds) |
|---|---|
| 44 DO | 0.534 |
| 34 DO | 1.324 |
| 24 DO | 2.134 |
| 04 DO | 3.790 |

3) The $\mu$ and $\sigma$ values based on scram insertion times from deenergization of scram pilot valve solenoids to dropout of Notch 34 which are used to calculate $\tau_B$ (TS 4.11.C) consistent with GEMINI advanced physics methods.

$\mu$ = 0.937 seconds
$\sigma$ = 0.021 seconds

4) Correction factors required to account for measurement biases and uncertainties when demonstrating compliance with the scram insertion times requested in Items 1 and 2 above.

The limits specified in the responses to Items 1, 2, and 3, explicitly account for the uncertainties in the location of the position indication probes and for the uncertainty in the control rod position when pickup or dropout of the reed switch occurs. Any other measurement uncertainties and biases introduced by the BECo surveillance procedures and hardware configuration used in the measurements are specific to Pilgrim and are not included in the specified limits (e.g., determination of time zero, accuracy of measurement devices, etc.).

G.A. Watford
Systems Integration Engineering
M/C 740, Tel. 5-6136

verified by:

E.Y. Gibo, LSE
Control Rod Drive System
Reactor Design Engineering
M/C 771, Tel. 5-6783

Attachment 3
Memo to E.L. Heinlein from S. J. Peters dated September 3, 1993;
Subject: Time to Notch 34, 24, and 04 Dropout for Pilgrim - RNE93-260

Fuel Engineering
San Jose, California

General Electric Nuclear Energy
175 Curtner, San Jose, CA 95125

RNE93-260

September 3, 1993

TO: E.L. Heinlein

FROM: S.J. Peters

SUBJECT: Time to NOTCH 34, 24, and 04 DROPOUT for Pilgrim

REFERENCE: Letter, E.L. Heinlein to J.H. Paiscik, "Technical Specification Scram Time REquirements", February 5, 1993.

The referenced letter contains scram times to assure technical specification compliance for the fastest three rods in a clumped 2X2 control rod array at Pilgrim. At BECo request, the purpose of this letter is to update the time requirement for 10%, 30%, 50% and 90% insertion if it is determined by measuring from the NOTCH 44 DROPOUT, NOTCH 34 DROPOUT, NOTCH 24 DROPOUT and NOTCH 04 DROPOUT, respectively. The values are shown in the table below and they supersede the values reported in the referenced letter.

| | |
|---|---|
| NOTCH 44 DROPOUT | 0.538 seconds |
| NOTCH 34 DROPOUT | 1.327 seconds |
| NOTCH 24 DROPOUT | 2.137 seconds |
| NOTCH 04 DROPOUT | 3.793 seconds |

These values are based on removing the conservative assumption that the control reed switch is at the minimum tolerance, reasonable for averaging multiple control rod drives. All other effects discussed in the referenced letter remain conservatively included.

If you have any questions please call.

Steve Peters
S.J. Peters
Reload Nuclear Engineering 2
M/C 156, Ext. 51124

Verified by: JKasillas 9/8/93
J.K. Casillas
Reload Nuclear Engineering 1
M/C 171, Ext. 56910

cc. P.J. Savoia
E.G. Thacker II
DRF J11-02042

Attachment 4
S&SA Calculation 088 dated 6/28/95;
Subject: Scram times for Tech Spec 3.3.C.1