GRAND GULF NUCLEAR STATION UNIT 1

CYCLE 6 RELOAD SUMMARY REPORT

DECEMBER 1991

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1.0 INTRODUCTION

This report is a supplementary document that summarizes the results of the analyses performed in support of GGNS Unit 1 Cycle 6 operation. The fresh fuel to be inserted in this cycle is an SNP 9x9-5 fuel type. It is similar to the 9x9-5 fuel inserted for Cycle 5 except for slightly increased pellet-to-clad gap size, increased prepressurization, and differences in enrichment and gadolinia loadings. This fuel has been shown to be compatible with the 8x8 and 9x9-5 fuel types that were inserted during previous reloads and will be resident in the core during Cycle 6 (Reference 1).

The SNP Cycle 6 Reload Analysis Report (Reference 1) and the Cycle 6 Plant Transient Analysis Report (Reference 2) serve as the basic framework for the reload analyses. Where appropriate, reference is made to these and other supporting documents for more detailed information and/or specifics of the applicable analyses. A list of references comprising both the generic and the GGNS-specific documents used in support of the Cycle 6 reload submittal is provided in Section 12.0 of this report.

2.0 CYCLE 6 RELOAD SCOPE

During the fifth refueling outage at GGNS Unit 1, depleted SNP 8x8 fuel assemblies will be replaced by SNP 9x9-5 fuel assemblies. Fuel related analyses of the limiting events were performed in support of Cycle 6. This included analyzing Cycle 6 for anticipated transients, the Fuel Misload Error Event, LOCA, and the Control Rod Drop Accident. These analyses were performed to support the safety and operating limits based on SNP methodology for both Two Loop and Single Loop Operation. Analyses for normal operation of the reactor consisted of fuel evaluations in the areas of mechanical, thermal-hydraulic, and nuclear design.

Based on SNP's design and safety analyses of the Cycle 6 reload core, the proposed changes to the GGNS Unit 1 Technical Specifications are as follows:

a. The MCPR Safety Limit values for Two Loop Operation and Single Loop Operation (SLO) are revised.

b. The MAPLHGR multiplier for Single Loop Operation is revised.

c. The flow-dependent MCPR limits are revised.

d. The power-dependent MCPR limits are revised.

e. The exposure-dependent MCPR limits are revised.

f. The LHGR limits for 8x8 SNP fuel types are revised for average planar exposures beyond 40,000 MWd/MTU.

g. The flow-dependent and power-dependent LHGR multipliers are revised and incorporate fuel type-specific multipliers.

3.0 CYCLE 5 OPERATING HISTORY

Cycle 5 core-follow operating data available at the time of the reload design analysis, together with projected plant operation through the end of Cycle 5, was used as a basis for the Cycle 6 core design and as input to the plant safety analyses. Cycle 5 has continued to operate as expected. No operating anomalies have occurred that would affect the licensing basis for Cycle 6. The Cycle 6 analyses were performed assuming a nominal Cycle 5 energy of 1698 GWd.

4.0 CYCLE 6 CORE DESCRIPTION

The Cycle 6 core will consist of 800 fuel assemblies. A breakdown by bundle type/bundle average enrichment is provided in the following table:

Cycle Inserted	Number of Bundles	Bundle Type
6	172	SNP 9x9/2.94 w/o U235
6	100	SNP 9x9/3.38 w/o U235
5	284	SNP 9x9/3.42 w/o U235
4	4	SNP 9x9/3.25 w/o U235
4	240	SNP 8x8/3.37 w/o U235

The anticipated Cycle 6 core configuration, together with additional bundle and core design details, is provided in Section 4.0 of the SNP Cycle 6 Reload Analysis Report (Reference 1). The Cycle 6 core is a conventional scatter load with the lowest reactivity bundles placed in the peripheral region of the core. The loading pattern was designed to maximize cycle energy and minimize power peaking factors. Cycle 6 is estimated to provide 1748 GWd of energy based on a Cycle 5 energy output of 1698 GWd.

5.0 FUEL MECHANICAL DESIGN

The mechanical design analyses for the SNP 8x8 and 9x9-5 fuel types are described in References 4, 5, and 10. The 8x8 fuel assembly design contains 62 prepressurized fuel rods and two water rods, one of which functions as a spacer capture rod. Seven spacers maintain fuel rod spacing. The 9x9-5 fuel assembly design contains 76 prepressurized fuel rods and five water rods, one of which serves as a spacer capture rod. Seven spacers maintain fuel rod spacing. The diametral pellet-to-clad gap on the 9X9-5 fuel rods is smaller on the interior high enrichment rods than on the peripheral rods in order to improve ECCS performance. The Cycle 6 reload batch uses increased prepressurization and gap sizes relative to the Cycle 5 reload batch. The additional analyses that were performed to support these design changes are described in Reference 21.

Mechanical design analyses were performed to evaluate cladding steady-state strain, transient stresses, fatigue damage, creep collapse, corrosion buildup, hydrogen absorption, fuel rod maximum internal pressure, differential fuel rod growth, creep bow, and grid spacer spring design. These analyses were performed to support peak assembly discharge burnups of 40 GWd/MTU for both the 8x8 and 9x9-5 fuel types. As shown in References 4, 5, and 10, all parameters meet their respective design limits; no fuel centerline melting will occur at 120% and 135% overpower conditions for 8x8 fuel and 9x9-5 fuel types, respectively. The effects of increased prepressurization and larger gap sizes do not impact the available design margins for the Cycle 6 reload batch (Reference 21). The Cycle 6 core design is bounded by the assumptions used in these analyses.

Fuel channels of a design similar to that used for the Cycle 5 reload batch will be used for reload Cycle 6 fuel. As was the case for Cycle 5, fuel channels manufactured by Carpenter Technology Corporation (CarTech) will be used for the Cycle 6 reload batch.

The mechanical responses of the 8X8 and 9x9-5 SNP assembly designs during seismic-LOCA events for Cycle 6 are essentially the same as for previous cycles because the physical properties and bundle natural frequencies are similar. Reference 7 presents the seismic-LOCA analysis for the 8x8 fuel and shows that the resultant loadings do not exceed the fuel design limits. Reference 23 presents the corresponding seismic-LOCA analysis for 9x9-5 fuel. The applicability of these analyses to the 8x8 and 9x9-5 fuel assemblies in the Grand Gulf Unit 1 core has been confirmed by SNP (Reference 1).

6.0 THERMAL HYDRAULIC DESIGN

XN-NF-80-19(A), Volume 4, Revision 1 (Reference 3) discusses the thermal-hydraulic design criteria that are used in the determination of the fuel cladding integrity safety limit and the bypass flow characteristics. SNP analyses were performed in accordance with XN-NF-80-19(A), Volume 3, Revision 2 (Reference 19) to determine the parameters that demonstrate compliance with these design criteria.

6.1 MCPR Safety Limits

The MCPR fuel cladding integrity safety limits are 1.06 and 1.07 for Two Loop Operation and Single Loop Operation (SLO), respectively. The methodology and generic uncertainties used in the Cycle 6 MCPR safety limit calculation, including the effects of channel bow, are provided in Reference 8.

6.2 Flow-dependent MCPR

The flow-dependent MCPR limits ($MCPR_f$) are revised for Cycle 6. The $MCPR_f$ limits are defined for only the Loop Manual mode of operation. The $MCPR_f$ limits are lower than the limits applicable to previous cycles as a result of a lower MCPR safety limit and smaller delta-CPRs for the slow flow runout event for the predominantly 9x9-5 Cycle 6 core. The $MCPR_f$ limits are defined over the same range of flows as for Cycle 5.

6.3 Power-dependent MCPR

The power-dependent MCPR limits ($MCPR_p$) are revised for Cycle 6. The most limiting events were used as the basis to confirm or establish the acceptability of the $MCPR_p$ limits for both Two Loop Operation and SLO during Cycle 6.

6.4 Exposure-dependent MCPR

The exposure-Dependent MCPR limits ($MCPR_e$) are revised for Cycle 6. The $MCPR_e$ limits were calculated for two exposure ranges instead of the three ranges used for Cycle 5. The limit is unchanged from the Cycle 5 value for the early part of the cycle and is lower than the Cycle 5 values for the latter part of the cycle.

The most limiting core-wide transients and local events were analyzed to confirm the acceptability of the $MCPR_e$ limits for use in Cycle 6.

These limits were established consistent with the Cycle 6 operating strategy.

6.5 Core Stability

The GGNS Unit 1 Technical Specifications implement the BWROG/GE Interim Recommendations for Stability Actions (IRSA). The IRSA boundaries, which were developed based on GE fuel experience, have been approved for application at GGNS Unit 1 containing SNP 8x8 fuel (Reference 25) and for application for Cycle 5 containing the first batch of SNP 9x9-5 fuel (Reference 17).

The Cycle 6 core will contain the second SNP 9x9-5 reload batch. Confirmatory analyses for Cycle 6 core stability consisted of a comparative evaluation of the stability characteristics for the Cycle 5 and Cycle 6 cores. In addition, a full 9x9-5 fueled core was analyzed. The results showed that the core decay ratios for the cycles analyzed are equivalent; the differences in stability performance are comparable to the variations observed for previous cycles. Therefore, the current GGNS-1 stability-related technical specifications are applicable for Cycle 6 operation as well as for a full 9x9-5 fueled core, provided that the core design and cycle operating strategy are not changed significantly.

7.0 NUCLEAR DESIGN

The neutronic methods used for the design and analysis of SNP reloads are described in SNP topical reports (References 9 and 24).

7.1 Fuel Bundle Nuclear Design

The Cycle 6 reload fuel utilizes SNP 9x9-5 fuel assemblies. Two basic bundle designs are used with different axially distributed U235 and burnable poison concentrations. For both designs, the top 12 inches and bottom 6 inches of each fuel rod contain natural uranium and the central 132 inch zone of each rod contains enriched uranium at one of eight different enrichments. The central zone of the 2.94 weight percent (w/o) bundle (identified in Section 4.0) has an average enrichment of 3.25 w/o U235, whereas the 3.38 w/o bundle has a central region with an average enrichment of 3.75 w/o U235. The neutronic design parameters and rod enrichment distribution are described in Section 4.0 of the Cycle 6 Reload Analysis Report (Reference 1).

7.2 Core Reactivity

The beginning of Cycle 6 (BOC6) cold core K_{eff} with the strongest worth control rod fully withdrawn at cold, 68 degrees F reactor conditions was calculated to be 0.98914. This corresponds to a shutdown margin (delta k/k) of 1.10%. BOC + 500 MWd/MTU and BOC + 7500 MWd/MTU were determined to be the most limiting conditions with a minimum shutdown margin of 1.03%. Therefore, the difference between the minimum shutdown margin in the cycle and the BOC shutdown margin, R, is 0.07%. The calculated shutdown margin is well in excess of the 0.38% delta k/k Technical Specification requirement (Section 3/4.1.1), and will be verified by testing at BOC6 to be greater than or equal to R + 0.38% delta k/k.

The Standby Liquid Control (SLC) system is designed to inject a quantity of boron that produces a concentration of no less than 660 ppm in the reactor core. Analyses were performed to show that the minimum shutdown margin is at least 3.0% delta k/k with the reactor in a cold, xenon-free state, at the most limiting cycle exposure, and with all control rods in their critical full power positions. This assures that the reactor can be brought from full power to a cold, xenon-free shutdown, assuming that none of the withdrawn control rods can be inserted, and confirms the basis of the Technical Specification requirement for the Cycle 6 reload core.

7.3 Spent Fuel Pool Criticality

The most reactive segment of the Cycle 6 fuel at its most reactive point in life is less reactive than analyzed in Reference 22. Therefore, the Reference 22 analysis is bounding for the Cycle 6 fuel.

8.0 CORE MONITORING SYSTEM

The POWERPLEX core monitoring system is and will continue to be utilized to monitor reactor parameters at GGNS. The core monitoring system is fully consistent with SNP's nuclear analysis methodology as described in References 9 and 24. In addition, the measured power distribution uncertainties are incorporated into the calculation of the MCPR Safety Limit as described in SNP's Nuclear Critical Power Methodology Report (Reference 8).

9.0 ANTICIPATED OPERATIONAL OCCURRENCES

In order to support the Cycle 6 operating limits, eight categories of system transients are considered, as described in SNP's Plant Transient Methodology Report (Reference 11). SNP has provided plant specific analysis results for the following system transients to determine the thermal margin requirements for operation during Cycle 6 (Reference 2):

1) Generator Load Rejection without Bypass (LRNB)
2) Feedwater Controller Failure (FWCF)
3) Loss of Feedwater Heating (LFWH)
4) Flow Excursion

Analyses performed for previous cycles have shown that the other system transients are non-limiting and, therefore, are bounded by one of the above. In addition, the Fuel Loading Error was analyzed in accordance with the methodology described in Reference 9. The Control Rod Withdrawal Error (CRWE) transient has been analyzed generically in Reference 18. Single Loop Operation is addressed in the Cycle 6 Transient Analysis Report (Reference 2).

9.1 Core-Wide Transients

The plant transient codes that were used to evaluate the LRNB and FWCF events are SNP's COTRANSA2 (Reference 26) and XCOBRA-T (Referance 20), which incorporate a one dimensional neutronics model to account for shifts in axial power shape and control rod effectiveness. Technical Specification scram times (Section 3/4.1.3) were used in the bounding analysis. The results of the LRNB and FWCF analyses are provided in the Cycle 6 Plant Transient Analysis Report (Reference 2) and a summary of results is provided in the Cycle 6 Reload Analysis Report (Reference 1). The LFWH event was analyzed by performing quasi-steady state analysis using the MICROBURN-B neutronics code (Reference 24). The LFWH event was analyzed consistent with the MEOD power/flow operating map for actual GGNS operating conditions during Cycles 1 through 5 and for various conditions anticipated during Cycle 6. A summary of this analysis is provided in Reference 2.

9.2 Local Transients

The Control Rod Withdrawal Error (CRWE) transient has been analyzed generically in Reference 18. The generic analysis provides a statistical evaluation of the consequences of the CRWE transient for BWR/6 plant configurations under conditions that cover the normal operating power/flow map, the extended load line region, and the increased core flow region. This analysis was reevaluated using the ANFB Critical Power Correlation (Reference 27) and the MICROBURN-B neutronics code (Reference 24). The evaluation demonstrated the continued applicability of the generic CRWE analysis results.

9.3 Reduced Flow and Power Operation

The off-rated thermal limits which were established for Cycle 1 MEOD operation (Reference 6), were revised appropriately for Cycle 6 operation. The power-dependent MCPR operating limits, which are based on the results of the Cycle 6 transient analyses and the CRWE generic analysis, were revised for Cycle 6. The flow-dependent MCPR limits are revised for Cycle 6 based on SNP's Cycle 6 analysis results. Flow rates used in the analysis are defined in Reference 12. The MCPR limits ensure that potential clad damage resulting from transition boiling is avoided.

Flow-dependent and power-dependent LHGR multipliers that are fuel type-specific were determined for Cycle 6. The flow and power ranges are unchanged from Cycle 5. For the power-dependent LHGR multipliers, bounding limit curves applicable to all core flows were established using SNP methodology. The LHGR limits ensure that the fuel mechanical design criteria are satisfied.

Flow-dependent MCPR limits and LHGR multipliers are determined for only the Loop Manual mode of operation because changes in plant configuration have been made to ensure that operation in the Non-Loop Manual mode is not possible (Reference 28)

9.4 ASME Overpressurization Analysis

In order to demonstrate compliance with the ASME Code overpressurization criterion of 110% of vessel design pressure, comparative evaluation of the peak vessel pressures calculated for previous cycles was performed for the limiting event, using an equivalent set of assumptions. The limiting event is the MSIV closure with failure of the MSIV position switch scram. Seven out of twenty safety/relief valves are assumed to be out of service. A conservative 6% tolerance is used for the safety valve setpoints. The results show that the maximum vessel pressure varies over a narrow range and is not sensitive to fuel and core design variations; sufficient margin is available to the transient pressure limit of 1375 psig (Reference 2).

10.0 POSTULATED ACCIDENTS

In support of Grand Gulf operation, SNP has analyzed the Loss-of-Coolant Accident (LOCA) for Two Loop Operation and for SLO to demonstrate that MAPLHGR limits for Cycle 6 reload fuel comply with 10CFR50.46 criteria. Methodology for the LOCA analysis is provided in References 13 through 15. The Rod Drop Accident (RDA) was analyzed for the Cycle 6 core to demonstrate compliance with the 280 cal/gm Design Limit. Methodology for the RDA analysis is described in XN-NF-80-19(A), Volume 1 (Reference 9). An SNP evaluation shows that the GE analysis of ATWS over-pressurization is applicable to SNP fuel and therefore remains valid for Cycle 6.

10.1 Loss-of-Coolant Accident (LOCA)

The generic BWR/6 LOCA break spectrum analysis (Reference 16) and the LOCA analysis performed in support of the Cycle 2 submittal (Reference 31) remain applicable for Cycle 6. A cycle-specific heatup analysis was performed for Cycle 6. The analysis confirms that the Peak Cladding Temperatures (PCTs) (1691 degrees F for 8X8 fuel and 1713 degrees F for 9X9-5 fuel) remain well below the 10CFR50.46 PCT limit of 2200 degrees F.

A detailed LOCA analysis was performed for Single Loop Operation (SLO) to determine an appropriate multiplier to be applied to the Two Loop Operation MAPLHGRs for the 8X8 and 9X9-5 fuel types (Reference 29). The multiplier was shown to be independent of the flow conditions in the idle loop (Reference 30). The multiplier was selected to ensure that the PCTs resulting from a LOCA during SLO are bounded by the corresponding PCTs for Two Loop Operation. The SLO LOCA analysis determines the highest PCTs over the range of exposures for the 8X8 and 9X9-5 fuel types. The calculated PCTs for the SLO LOCA (1631 degrees F for 8X8 fuel and 1609 degrees F for 9X9-5 fuel) are approximately 100 degrees F lower than the corresponding values for the Two Loop Operation LOCA (1738 degrees F for 8X8 fuel and 1713 degrees F for 9X9-5 fuel). The PCT calculated by the Two Loop Operation LOCA analysis for 8X8 fuel (1738 degrees F) is based on the higher (more conservative) MAPLHGR value used in the Reference 31 analysis; this PCT is higher than the Cycle 6 heatup analysis value (1691 degrees F), which is based on the Cycle 6 MAPLHGR limit for 8X8 fuel (Reference 1).

Confirmatory analyses were performed to show that the local Zr-H_2O reaction remains below 17% and that the core-wide metal-water reaction (CMWR) remains below 1% for the limiting LOCA event as required by 10CFR50.46. The results of these analyses are presented in Section 6.1 of Reference 1. As stated in the GGNS-1 UFSAR, the hydrogen recombiners have been sized to process the hydrogen released from 0.8% CMWR. Consistent with the Regulatory Guide 1.7 requirements for post-LOCA combustible gas control, the capability of the hydrogen recombiners to maintain post-LOCA combustible gas concentration below 4 volume percent has been confirmed.

10.2 Rod Drop Accident

SNP's methodology for analyzing the Rod Drop Accident (RDA) utilizes a generic parametric analysis that calculates the fuel enthalpy rise during the postulated RDA over a wide range of reactor operating conditions. For Cycle 6, Section 6.2 of Reference 1 shows a value of 166 cal/gm for the maximum deposited fuel rod enthalpy during the worst case postulated RDA. This value is well below the design limit of 280 cal/gm.

11.0 REFUELING OPERATIONS

As was done for Cycle 5, refueling operations will be addressed by a 10CFR50.59 Safety Evaluation.

12.0 REFERENCES

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2) EMF-91-168, "Grand Gulf Unit 1 Cycle 6 Plant Transient Analysis," Siemens Nuclear Power Corporation, October 1991.

3) XN-NF-80-19(P)(A), Volume 4, Revision 1, "Exxon Nuclear Methodology for Boiling Water Reactors: Application of the ENC Methodology to BWR Reloads," Exxon Nuclear Co., June 1986.

4) XN-NF-85-67(P)(A), Revision 1, "Generic Mechanical Design for Exxon Nuclear Jet Pump BWR Reload Fuel," Exxon Nuclear Co., September 1986.

5) ANF-88-152(P), Amendment 1, "Generic Mechanical Design for Advanced Nuclear Fuels 9x9-5 BWR Reload Fuel," Advanced Nuclear Fuels Corporation, September 1989.

6) "GGNS Maximum Extended Operating Domain Analysis," General Electric Company, March 1986.

7) XN-NF-81-51(A), "LOCA-Seismic Structural Response of an ENC BWR Jet Pump Fuel Assembly," Exxon Nuclear Co., May 1986.

8) ANF-524(P)(A), Revision 2, "Advanced Nuclear Fuels Corporation Critical Power Methodology for Boiling Water Reactors," including Supplements, Advanced Nuclear Fuels Corporation, April 1989.

9) XN-NF-80-19(A), Volume 1, Supplements 1 & 2, "Exxon Nuclear Methodology for Boiling Water Reactors: Neutronics Methods for Design and Analysis," Exxon Nuclear Co., March 1983.

10) ANF-88-183(P), "Grand Gulf Unit 1 Reload XN-1.3 Cycle 4 Mechanical Design," ANF Corporation, Supplement 1, August 1991.

11) XN-NF-79-71(P), Revision 2, "Exxon Nuclear Plant Transient Methodology for Boiling Water Reactors," Exxon Nuclear Co., November 1981.

12) NESDQ-88-003, Revision 0, "Grand Gulf Nuclear Station Unit 1 Revised Flow Dependent Thermal Limits," MSU System Services Inc., November 1988.

13) XN-NF-80-19(A), Volumes 2, 2A, 2B, & 2C, "Exxon Nuclear Methodology for Boiling Water Reactors: EXEM BWR ECCS Evaluation Model," Exxon Nuclear Co., September 1982.

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16) XN-NF-86-37(P), "Generic LOCA Break Spectrum Analysis for BWR/6 Plants," Exxon Nuclear Co., May 1986.

17) "Issuance of Amendment No.73 to Facility Operating License No. NPF-29, Grand Gulf Nuclear Station Unit 1, Regarding Fuel Cycle 5 Reload (TAC No. 76992), Letter from L. L. Kintner, USNRC, to W. T. Cottle, Entergy Operations Inc., dated November 15, 1990.

18) XN-NF-825(P)(A), Supplement 2, "BWR/6 Generic Rod Withdrawal Error Analysis, $MCPR_p$ for All Plant Operations Within the Extended Operating Domain," Exxon Nuclear Company, October 1986.

19) XN-NF-80-19(P)(A), Volume 3, Revision 2, "Exxon Nuclear Methodology for Boiling Water Reactors THERMEX: Thermal Limits Methodology Summary Description," Exxon Nuclear Co., January 1987.

20) XN-NF-84-105(P)(A), Volume 1, "XCOBRA-T: A Computer Code for BWR Transient Thermal Hydraulic Core Analysis," Exxon Nuclear Company, Inc., February 1987.

21) "Reload ANF-1.5 Principal Reload Fuel Design Parameters Document," MPEX-90/111, Letter from N. L. Garner, ANF Corporation, to J. B. Lee, Entergy Operations Inc., November 28, 1990.

22) ANF-90-060, "Criticality Safety Analysis for the Grand Gulf Spent Fuel Storage Racks with ANF 1.4 Fuel Assemblies," April 1990.

23) XN-NF-84-97(P)(A), "LOCA-Seismic Structural Response of an ENC 9x9 BWR Jet Pump Fuel Assembly," Exxon Nuclear Company Inc., August 1986.

24) XN-NF-80-19(P), Volume 1, Supplement 3, "ANF Methodology for BWRs: Benchmark Results for the CASMO-3G/MICROBURN-B Calculation Methodology," Advanced Nuclear Fuels Corporation, February 1989, as supplemented by ANF letter RAC:083:90 dated July 20, 1990.

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26) ANF-913, Volume 1, Supplements 1, 2, and 3, "COTRANSA2: A Computer Program for Boiling Water Reactor Transient Analysis," Advanced Nuclear Fuels Corporation, June 1989.

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29) EMF-91-172, "Grand Gulf Unit 1 LOCA Analysis for Single Loop Operation," Siemens Nuclear Power Corporation, October 1991.

30) GEXI-01391, "Allowed Valve Conditions in the Idle Loop for Grand Gulf Unit 1 Single Loop Operation," Letter, S. L. Leonard, SNP Corporation, to J. B. Lee, Entergy Operations Inc., November 7, 1991.

31) XN-NF-86-38, "Grand Gulf Unit 1 LOCA Analysis," Exxon Nuclear Co., June 1986.