UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 99 TO FACILITY OPERATING LICENSE NO. NPF-29

ENTERGY OPERATIONS, INC.

GRAND GULF NUCLEAR STATION, UNIT 1

DOCKET NO. 50-416

1.0 INTRODUCTION

By letter dated December 5, 1991 (Ref. 1), the licensee (Entergy Operations, Inc.) submitted a request for revisions to the Grand Gulf Nuclear Station, Unit 1 (GGNS), Technical Specifications (TS). The revisions accommodate the core changes associated with Cycle 6 reload and operation.

The Cycle 6 reload will replace 272 SNP 8x8 fuel assemblies used in Cycle 5 with SNP 9x9-5 fuel assemblies. This is the second GGNS reload of this type; the GGNS Cycle 5 core consisted partly of SNP 9x9 fuel assemblies. In addition, a second batch of GE channels associated with the discharged 8x8 fuel will be replaced with CarTech channels. The core loading will retain 240 SNP 8x8 fuel assemblies and 4 lead test SNP 9x9-5 assemblies inserted in Cycle 4 and 284 SNP 9x9-5 fuel assemblies inserted in Cycle 5. Generally, the Cycle 6 reload is a normal reload with no unusual features other than the shift to a larger percentage of 9x9 fuel assemblies in the core. SNP 9x9 fuel has been used in other reactors; Susquehanna 2, for example, has been operating with an all SNP 9x9 fuel loading.

The Cycle 6 TS changes for GGNS are not extensive and are primarily related to the Minimum Critical Power Ratio (MCPR) safety limit, the Maximum Average Planar Linear Heat Generation Rate (MAPLHGR), the Linear Heat Generation Rate (LHGR), and associated factors for Cycle 6 core operation as calculated by SNP. SNP performed the Cycle 6 reload analyses using methodologies that have been used for previous reload submittals and have been reviewed and approved by the staff.

2.0 EVALUATION

2.1 Fuel Design

The GGNS Cycle 6 reload will include 272 new SNP 9x9-5 Fuel assemblies. These contain 76 prepressurized fuel rods and 5 water rods. The rod enrichment distribution is described in the Cycle 6 Reload Analysis Report (Ref. 2). The mechanical design analyses for the SNP 8x8 and 9x9-5 fuel types are described

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in References 6 and 13 and in Supplement 1 of the GGNS Reload XN-1.3, "Cycle 4 Mechanical Design Report" (Ref. 5). Sections 2.1.1 through 2.1.6 evaluate the mechanical design analyses provided by the licensee in Reference 5 to support peak assembly discharge burnups of 40 GWd/MTU for both the 8x8 and the 9x9-5 fuel types. The fuel mechanical design is similar to that approved for Cycle 5. The fuel channels that will be used for Cycle 6 are manufactured by Carpenter Technology Corporation (CARTECH) and are of design similar to those used for Cycle 5.

SNP has analyzed the response of the SNP 9x9-5 fuel assemblies during seismic-LOCA events and has concluded that the response is essentially the same as for previous cycles because of the similarities of the physical properties and bundle natural frequencies. The licensee has demonstrated that the resultant loadings do not exceed the fuel design limits for either the 8x8 or the 9x9-5 fuel.

2.1.1 Stress and strain

The licensee used the approved methodology described in XN-NF-85-67(P)(A), Revision 1 (Ref. 6), for stress analysis and the approved RODEX2A code for strain analysis. The stress analysis showed that the cladding stress remained below the ASME code limits. The strain analysis showed that the cladding strain remained below the 1 percent strain limit. We conclude that the licensee's stress and strain analyses, based on the approved methodology and RODEX2A code, are acceptable for GGNS.

2.1.2 Rod Internal Pressure

The licensee's rod internal pressure criterion is that the rod pressure will be limited to a value below that which would cause (1) an increase of the diametral gap due to outward cladding creep and (2) extensive departure from nucleate boiling (DNB) propagation. The licensee used the approved methodology, described in Reference 6, to analyze the rod pressure. The results showed that the peak rod pressure was slightly above the system pressure but still met the design criterion. We conclude that the licensee's rod pressure analysis, based on the approved methodology, is acceptable for GGNS.

2.1.3 Fuel Temperature

The licensee's fuel temperature design criterion is that the maximum fuel temperature shall be less than the melting temperature of UO_2. The licensee used the approved methodologies described in Reference 6 for fuel temperature analysis. The result showed that the maximum fuel temperature remained below the melting temperature. We conclude that the licensee's fuel temperature analysis, based on the approved methodology, is acceptable for GGNS.

2.1.4 Cladding Fatigue

The licensee used the approved methodology, described in Reference 6, for fatigue analysis. The results showed that the fatigue usage factor was less than the acceptance criterion. We conclude that the licensee's cladding fatigue analysis, based on the approved methodology, is acceptable for GGNS.

2.1.5 Cladding Collapse

The licensee's cladding collapse criterion is that the cladding failure due to collapse should not occur. The design criterion also requires that the pellet-to-cladding gap remain open during the pellet densification. This requirement assures that axial gaps will not fo m in the fuel column. The licensee used the approved methodology, described in Reference 6, for collapse analysis. The results showed that the cladding collapse does not occur during the lifetime. We conclude that the licensee's collapse analysis, based on the approved methodology, is acceptable for GGNS.

2.1.6 Cladding Corrosion

The licensee used the approved methodology, described in Reference 6, for corrosion analysis. The results showed that the hydrogen pickup and clad oxidation were below the acceptance criterion. We conclude that the licensee's corrosion analysis, based on the approved methodology, is acceptable for GGNS.

Based on our review of the information presented and the similarities to previously approved design and analyses, we find the mechanical design of the ANF 9x9-5 fuel for GGNS Cycle 6 to be acceptable.

2.2 Nuclear Design

The SNP nuclear design methodology is presented in References 7 and 8, which have been reviewed and approved by the staff.

The beginning of cycle (BOC) shutdown margin is calculated to be 1.10 percent delta-K, and BOC + 500 MWd/MTU and BOC + 7500 MWd/MTU were determined to be most limiting conditions with a shutdown margin of 1.03 percent delta-K. Thus the cycle minimum shutdown margin is well above the required 0.38 percent delta-K. The Standby Liquid Control System also fully meets shutdown requirements. The GGNS high density spent fuel storage racks were reviewed and approved by the staff for the Cycle 5 reload (Ref. 9). The most reactive segment of the Cycle 6 fuel at its most reactive point in life is less reactive than was analyzed for Cycle 5. Therefore, it was concluded that the Cycle 5 analysis is bounding for the Cycle 6 fuel and that the storage racks can safely accommodate the Cycle 6 fuel.

The GGNS Cycle 6 nuclear characteristics have been calculated with approved methodologies, the results meet applicable criteria, and the review concludes that the design is acceptable.

2.3 Thermal-Hydraulic Design

This is the second reload of this type at GGNS. That the SNP 9x9 fuel is thermal-hydraulically compatible with the retained SNP 8x8 fuel has been demonstrated by approved methodologies, by the use of a partial 9x9 core for GGNS Cycle 5, and by the use of SNP 9x9 fuel at other BWRs.

The thermal-hydraulic methodology and criteria used for GGNS Cycle 6 are the same as used for the previous reload and are described in References 10 and 11. These methodologies are acceptable for Cycle 6 analysis.

The MCPR safety limit has been determined to be 1.06 for two-loop operation (TLO) and 1.07 for single-loop operation (SLO). The methodology and generic uncertainties used by SNP to perform the MCPR safety limit calculation are provided in Reference 12. This calculation included an evaluation of the effects of channel bow. The flow-dependent MCPR, power-dependent MCPR, and the exposure dependent MCPR limits were all revised for Cycle 6. These calculations were performed using approved methods and the limits are acceptable.

GGNS is currently operating under the BWR Owner's Group/General Electric Interim Recommendations for Stability Actions (IRSA) with stability boundary TS that were approved by the staff for the Cycle 5 core consisting partly of SNP 9x9-5 fuel. A comparative evaluation of the stability characteristics of the Cycle 5 and Cycle 6 cores, as well as of a full 9x9-5 core, was performed by SNP. The results of the SNP evaluation showed that the core decay ratios for the cycles were equivalent. The staff review concludes that continued use of the current stability TS boundaries is acceptable.

2.4 Anticipated Operational Occurrences and Accident Analyses

To provide the basis for the TS values of the various operating limits (MCPR and LHGR), SNP has analyzed the system Anticipated Operational Occurrence (AOO) events that could provide the most limiting conditions. This approach is in accordance with the approved methodology for operating limit analysis. The AOO events include Load Rejection Without Bypass (LRNB), Feedwater Controller Failure (FWCF), Loss of Feedwater Heating (LFWH), Flow Excursion (FE), Control Rod Withdrawal Error (CRWE), and the Fuel Loading Error (FLE). Previous analyses have shown that other events are non-limiting. Plant initial conditions for the analyses covered the full range of Maximum Extended Operating Domain (MEOD) approved for GGNS. Analyses were performed for End-of-Cycle (EOC), EOC-30 EFPD (Effective Full Power Days), and EOC+30 EFPD to provide exposure-dependent MCPR limits. Results of these analyses were used to provide the TS MCPR and LHGR limits as functions of power, flow, and exposure.

The analysis of the AOO events and the establishment of limiting operating values for MCPR and LHGR used approved methods and considered required events and reactor conditions. The analysis and the results are therefore acceptable.

SLO was also analyzed by SNP. The pump seizure event was analyzed and the MCPR safety limit and the MAPLHGR multiplier were determined for single-loop operations. The SLO MCPR was calculated to be 1.07, and the MAPLHGR multiplier was found to be 0.86. The analyses were performed with approved methods, and the results are therefore acceptable.

Compliance with overpressure criteria was demonstrated by analysis of the main steam isolation valve (MSIV) closure event, assuming failure of the direct scram signal on MSIV position. The analysis used approved methods and resulted in a pressure within the required limits and is therefore acceptable.

Accident analyses were performed by SNP for the Loss-of-Coolant Accident (LOCA) and the Rod Drop Accident (RDA) to demonstrate that the required limits are met for GG1C6. This analysis was performed with approved methods and is therefore acceptable.

2.5 Technical Specification Changes

The following TS changes have been proposed for operation of Cycle 6.

(1) TS 2.1.2

The MCPR safety limit is decreased from 1.09 to 1.06 for TLO and from 1.09 to 1.07 for SLO.

(2) TS 3/4.2.1

The SLO MAPLHGR multiplier is changed from 0.8 to 0.86. This change was based on SNP's detailed LOCA analysis for SLO.

(3) TS 3/4.2.3 - Figure 3.2.3-1

Flow-dependent MCPR limits have been revised. The lower MCPR$_f$ limits result from the lower MCPR safety limit and smaller delta-CPR values due to the improved transient response of the 9x9-5 fuel.

(4) TS 3/4.2.3 - Figure 3.2.3-2

Power-dependent MCPR limits have been revised to provide common $MCPR_p$ limits for both TLO and SLO.

(5) TS 3/4.2.3 - Figure 3.2.3-3

Exposure-dependent MCPR limits have been revised to reflect the revision to the MCPR safety limit and the improved transient response of the Cycle 6 core.

(6) TS 3/4.2.4 - Figure 3.2.4-1

LHGR limits for 8x8 fuel types have been increased for average planar exposures greater than 40,000 MWd/MTU in order to bound the LHGR performance for the 8x8 fuel near the end of Cycle 6.

(7) TS 3/4.2.4 - Figures 3.2.4-2 and 3

Off-rated mechanical limits have been revised to reflect the predominantly 9x9-5 fueled core and the higher LHGR limit for SNP 9x9-5 fuel.

All of the above changes are based upon analyses performed with approved methods and yielding results within prescribed safety limits. They are therefore acceptable. There are also changes to the Bases associated with the above TS to reflect the changes to the specifications or minor administrative changes. The changes reflect the TS changes and are acceptable. These include Bases 2.1.1, 2.1.2, and 3/4.2.1.

The staff has reviewed the reports submitted for the Cycle 6 operation of GGNS and concludes that appropriate material was submitted and that the fuel design, the nuclear design, the thermal-hydraulic design, and the transient and accident analyses are acceptable. The TS changes submitted for this reload reflect the necessary modifications for operation in this cycle.

The staff has concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public.

3.0 STATE CONSULTATION

In accordance with the Commission's regulations, the Mississippi State official was notified of the proposed issuance of the amendment. The State official had no comments.

4.0 ENVIRONMENTAL CONSIDERATION

The amendment changes a requirement with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 and changes in surveillance requirements. The NRC staff has determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendment involves no significant hazards consideration, and there has been no public comment on such finding (57 FR 2593). Accordingly, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendment.

5.0 CONCLUSION

The Commission has concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

REFERENCES

1. Letter from W. T. Cottle, Entergy Operations, Inc., to NRC, "Cycle 6 Reload," December 5, 1991.

2. EMF-91-169, "Grand Gulf Cycle 6 Reload Analysis," Siemens Nuclear Power Corporation (SNP), October 1991.

3. EMF-91-168, "Grand Gulf Cycle 6 Transient Analysis," SNP, October 1991.

4. EMF-91-172, "LOCA Analysis for Single Loop Operation," SNP, October 1991.

5. ANF-88-183(P), "Reload XN-1.3, Cycle 4 Mechanical Design," Advanced Nuclear Fuels Corporation (ANF), Supplement 1, August 1991.

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

7. 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.

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

9. Letter to W. T. Cottle, Entergy Operations Inc., from L. Kintner, NRC, "Criticality Analysis for Cycle 5 Fuel in Spent Fuel Storage Racks," July 16, 1990.

10. 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.

11. 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.

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

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

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