SIEMENS

EMF-88-183(NP)
Supplement 1

Siemens Nuclear Power Corporation Grand Gulf Unit 1 Reload XN-1.3, Cycle 4 Mechanical Design Report Supplement 1

October 1991

Siemens Nuclear Power Corporation

Siemens Nuclear Power Corporation

EMF-88-183(NP)
Supplement 1

Issue date: 10/21/91

GRAND GULF UNIT 1

RELOAD XN-1.3, CYCLE 4

MECHANICAL DESIGN REPORT

SUPPLEMENT 1

Prepared by:

M. L. Crawford 10-18-91
M. L. Crawford
Project Engineer
Mechanical Design Engineering
Fuel Design

SNP-SR4121.012.1A (N/08/91)

CUSTOMER DISCLAIMER

IMPORTANT NOTICE REGARDING CONTENTS AND USE OF THIS DOCUMENT

PLEASE READ CAREFULLY

Siemens Nuclear Power Corporation's warranties and representations concerning the subject matter of this document are those set forth in the Agreement between Siemens Nuclear Power Corporation and the Customer pursuant to which this document is issued. Accordingly, except as otherwise expressly provided in such Agreement, neither Siemens Nuclear Power Corporation nor any person acting on its behalf makes any warranty or representation, expressed or implied, with respect to the accuracy, completeness, or usefulness of the information contained in this document, or that the use of any information, apparatus, method or process disclosed in this document will not infringe privately owned rights; or assumes any liabilities with respect to the use of any information, apparatus, method or process disclosed in this document.

The information contained herein is for the sole use of the Customer.

In order to avoid impairment of rights of Siemens Nuclear Power Corporation in patents or inventions which may be included in the information contained in this document, the recipient, by its acceptance of this document, agrees not to publish or make public use (in the patent use of the term) of such information until so authorized in writing by Siemens Nuclear Power Corporation or until after six (6) months following termination or expiration of the aforesaid Agreement and any extension thereof, unless expressly provided in the Agreement. No rights or licenses in or to any patents are implied by the furnishing of this document.

GRAND GULF UNIT 1

RELOAD XN-1.3, CYCLE 4

MECHANICAL DESIGN REPORT

SUPPLEMENT 1

TABLE OF CONTENTS

TABLE OF CONTENTS
(Continued)

TABLE OF FIGURES

GRAND GULF UNIT 1
RELOAD XN-1.3, CYCLE 4
MECHANICAL DESIGN REPORT
SUPPLEMENT 1

1.0 INTRODUCTION

This report provides a summary of an evaluation, applicable to the Siemens Nuclear Power Corporation (SNP) 8x8 fuel for the Grand Gulf Unit 1 Nuclear Power Reactor reload XN-1.3. The analyses support an extension of the assembly exposure and a modification of the mechanical LHGR limits

The mechanical design of Grand Gulf 1 XN-1.3 is essentially the same as the generic SNP Type 4/5/6 design; thus, several of the mechanical design related sections of this report are covered by specific references to generic mechanical design reports. Where applicable, the analyses have been extended, consistent with SNP's generically approved methodology, to cover the increased burnup and the revised mechanical LHGR limit.

2.0 SUMMARY

The SNP 8x8 fuel design for Grand Gulf 1 XN-1.3 has been evaluated to allow operation up to a peak assembly exposure with a mechanical LHGR limit The results of the evaluation indicate that all design criteria are met. The fuel mechanical design description is summarized below.

2.1 Design Description Summary

The SNP 8x8 assembly design for Grand Gulf 1 XN-1.3 reload consists of 62 fuel rods and two centrally located water rods, one of which functions as a spacer capture rod. Seven spacers maintain fuel rod spacing. The design also uses a quick-removable upper tie plate design to facilitate fuel inspection and bundle reconstitution of irradiated assemblies.

The fuel rods are Zircaloy-2 cladding, 35 mils thick. The rods are pressurized, and contain either UO_2-Gd_2O_3 or UO_2 fuel pellets with a nominal density of 94.5% of the theoretical density. The fuel rods contain two possible diametrical pellet-to-clad gap sizes 9.5 mils or 8.5 mils. Natural uranium axial fuel blankets, at the top and the bottom of the fuel column, are provided for greater neutron economy.

The SNP 8x8 fuel design for Grand Gulf 1 XN-1.3 reload includes two minor design modifications which allow higher fuel exposures. The shanks on the upper end caps and the spacer capture rod lower end cap have been lengthened to increase their engagement in the tie plates. The lower tie plate seal has also been redesigned to increase the length of engagement of the seal with the fuel channel. These modifications were incorporated to accommodate increased differential growth, between the fuel assembly and channel, and the fuel assembly and the fuel rods, which results from the higher assembly exposure.

2.2 Mechanical Design Summary

The mechanical design analyses were performed to evaluate cladding steady-state strain and stress, transient strain and stress, fatigue damage, creep collapse, corrosion, hydrogen absorption, fuel rod internal pressure, fuel temperature, differential fuel rod growth, creep bow, and spacer grid design. The analyses justify irradiation to

Major analysis results are:

- The maximum end-of-life (EOL) steady-state cladding strain is calculated to be below the 1.0% design limit.
- Cladding steady-state stresses are calculated to be below the material strength limits.
- The cladding strain during anticipated operational occurrences (AOOs) does not exceed 1.0%.
- The maximum fuel rod internal rod pressure remains below SNP's criteria limit.
- The fuel centerline temperature remains below the melting point during AOOs.
- The cladding fatigue usage factor is within the 0.67 design limit.
- Structural members have adequate strength to support handling and hydraulic loads.

Compliance with this criterion prevents the formation of fuel column gaps, and the possibility of creep collapse.

- Evaluations of assembly growth and differential fuel rod growth show that the design provides adequate clearances for compatibility with the fuel assembly channel. Also, there is adequate engagement of the end caps in the upper tie plate and lower tie plate throughout the design life.
- The initial fuel rod design spacing is expected to be adequate to accommodate expected rod-to-rod gap closure for the fuel design life.

- The maximum EOL reduction in cladding thickness due to corrosion and the maximum concentration of hydrogen in the cladding are calculated to be well within the design limits.

- The fuel rod plenum spring and other miscellaneous components are shown to meet the respective design bases.

- The spacer springs meet all the design requirements, and can accommodate the expected relaxation at EOL.

3.0 DESIGN CRITERIA

The detailed Siemens Nuclear Power Corporation design criteria for the Grand Gulf 1 XN-1.3 reload fuel is given in Reference 1.

4.0 MECHANICAL DESIGN

Three reports have previously been issued to document the mechanical design analyses for the Grand Gulf 1 SNP 8x8 fuel. These reports are:

- XN-NF-83-25, Revision 1, "Grand Gulf 1 XN-1 Design Report Mechanical, Thermal Hydraulic, and Neutronic Design for Exxon Nuclear JP BWR/6 Fuel Assemblies"[4], issued in August 1983;
- XN-NF-85-67 (P) (A), Revision 1, "Generic Mechanical Design For Exxon Nuclear Jet Pump BWR Reload Fuel"[1], issued in September 1986; and
- ANF-88-183 (P)," Grand Gulf Unit 1 Reload XN-1.3, Cycle 4 Mechanical Design Report"[7] issued November 1988".

The analyses in the first report were performed with the RODEX2 computer code and justified irradiation Fuel rod analyses reported in the generic mechanical design report were performed using the computer code RODEX2A and justified irradiation Both RODEX2 and RODEX2A codes have been approved for generic application by the NRC.[2,3] The generic mechanical design report was submitted and approved for generic use by the NRC in 1986.

The Grand Gulf Unit 1 Reload XN-1.3, Cycle 4 mechanical design report extended the fuel assembly burnup

This document reports the results of design calculations performed to support higher fuel assembly exposure and a slightly modified LHGR mechanical limit than has been reported previously. The fuel rod calculations in this report used the RODEX2A computer code.

The fuel assembly has been analyzed to a peak assembly exposure and peak rod and peak pellet exposures respectively. These values are conservative estimates of the maximum exposures to be reached with the Grand Gulf 1 XN-1.3 8x8 reload fuel. The analyses have been performed assuming a design power history identical to that used in Reference 1, at low to medium exposures and a slightly modified power history at higher exposures. This slightly modified power history was generated from the modified LHGR mechanical limit curve shown in Figures 4.1 and 4.2.

4.1 <u>Fuel Rod Analyses</u>

Fuel rod analyses, where required, have been performed to verify adequate performance of the fuel

The fuel rod and peak pellet exposures used in the analyses were

respectively. The design power history used in Reference 1 was modified at higher exposure values for Grand Gulf 1. The LHGR mechanical limits in Reference 1 have also been modified for Grand Gulf 1. The analyses results reported here demonstrate compliance with the design criteria.

4.1.1 <u>Maximum Cladding Strain During Steady State Operation</u>

The maximum cladding strain during steady state operation is limited to <1%. The analyses have been performed with RODEX2A. The results indicate that the strain is below 1%. In Figure 3.8 of Reference 1, the cladding strain is depicted up to a rod nodal exposure

strain is below the design limit.

4.1.2 <u>Maximum Cladding Stress During Steady State Operation</u>

Fuel rod cladding stresses during steady-state operation are calculated

elasticity theory. The design stress limits are in accordance with the ASME pressure vessel code.

The assumptions made in the stress analysis reported in Reference 1 have been reviewed. The review indicates that the original assumptions made in performing the analysis (e.g., internal rod pressure, clad thinning, etc.) still bound the conditions at the higher exposure and modified operating powers. Consequently, the analysis results reported in Table 3.3 of Reference 1 are applicable.

4.1.3 Anticipated Operational Occurrences Analysis

Two design criteria are imposed on the fuel rods to avoid fuel failure during power changes caused by anticipated operational occurrences (AOOs). The criteria are to limit the cladding strain to less than 1% and to insure that the maximum pellet centerline temperature remains below the pellet melting point. The AOOs are assumed to produce a maximum rod nodal power equal to those defined in Figure 4.2.

Using the methodology described in Reference 1, an analysis was performed with the mechanical LHGR limits in Figure 4.2

The results of the analysis verify that the cladding strain and fuel temperatures meet the design criteria for AOOs within the defined power limits.

4.1.4 Fuel Rod Internal Pressure

The fuel rod internal pressure is limited

Using the methodology described in Reference 1, an analysis was performed using the modified LHGR limits

The results of the analysis indicate that the maximum internal pressure remains below the design criteria

4.1.5 Fuel Pellet Centerline Temperature

The design criteria requires that fuel centerline temperature remain below the fuel melting point during operation. A fuel pellet centerline temperature analysis was performed using the methodology described in Reference 1 while applying the modified LHGR limit curve and higher exposure level. The results of the analysis indicated that the fuel pellet centerline temperature will remain below the fuel melting point. Therefore, the design criteria is met.

4.1.6 Fuel Rod Cladding Fatigue

Fuel assembly shuffling, reactor power maneuvering, and AOOs will impose a cyclic loading on the cladding. The design criteria requires that the calculated cyclic fatigue usage factor remain within 0.67 times the cladding fatigue life.

To assure that the fuel rod does not fail due to stress cyclic fatigue, a fatigue analysis was performed. The results of the analysis show that the cumulative fatigue damage remain below the 0.67 limit.

4.1.7 Cladding Collapse

Fuel failures due to cladding collapse have been observed in some PWR's in fuel rods designed and fabricated by other fuel vendors. No SNP fuel rod has ever failed due to this mechanism. The likelihood of a fuel rod failing due to cladding collapse in a BWR reactor is very small due to the lower operating coolant pressure of the Grand Gulf 1 reactor compared to that in a PWR. Nevertheless, the fuel rods are analyzed to assure that fuel rod collapse will not occur.

The pellet completes densification during early operation.

The analysis results reported in Table 3.1 of Reference 1 are still applicable for the extended exposure.

4.1.8 Fuel Rod Spacing

The design criteria states that changes in rod-to-rod and rod-to-channel gaps must be taken into account in establishing the thermal limits. Thermal limits are not affected if the minimum rod-to-rod gap is greater The analysis performed in Reference 1 to calculate the maximum fuel rod bow has been evaluated for applicability at higher exposures.

Therefore, a significant margin to the design criteria will remain.

4.1.9 Cladding Corrosion and Hydrogen Concentration

The design criteria for cladding corrosion limits metal loss due to corrosion Hydrogen absorption is limited Using the methodology in Reference 1, an analysis was performed The results of the analyses indicated that the cladding corrosion and hydrogen absorption will remain well below the design criteria.

4.2 Fuel Assembly Evaluation

The performance of the fuel assembly has been evaluated. The structural strength, spacer design, and assembly growth have been investigated. The results are as follows.

4.2.1 Structural Strength

The structural strength of tie plates, locking mechanism, and tie rods is not decreased with exposure or with minor modifications in the LGHR mechanical limits. The analysis and test results previously reported in Reference 1 are applicable.

4.2.2 Spacer Spring

SNP data indicates that spacer springs relax with irradiation. The observations indicate that the relaxation rate decreases with increased exposures and that it tends to saturate at higher exposures.

It is therefore concluded that the spacer spring design is acceptable

4.2.3 Assembly Growth

The design criteria requires that the rod end caps remain engaged in the tie plates and that the lower tie plate seal spring remain engaged in the fuel channel throughout the life of the fuel assembly. Since growth is exposure dependent, the minimum engagement will occur at the EOL.

Assembly growth was determined by evaluating differential growth between standard fuel rods and non-fuel rods and the tie rods. Additionally, an evaluation of channel engagement with the lower tie plate seal as a function of irradiation exposure was reviewed. The calculations indicate that sufficient channel and end cap engagement are present to EOL.

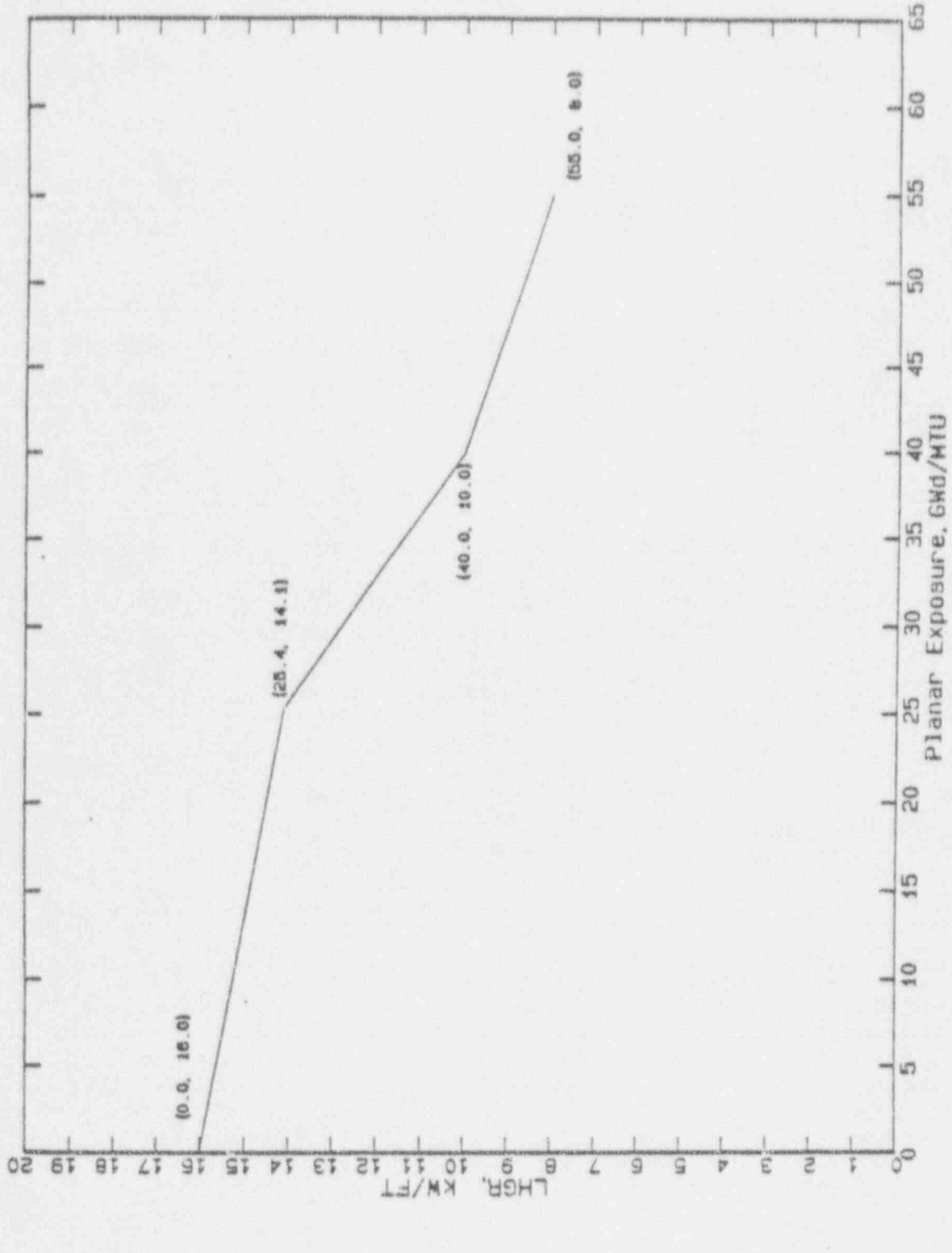

Figure 4.1 - LHGR LIMIT FOR 8X8 FUEL

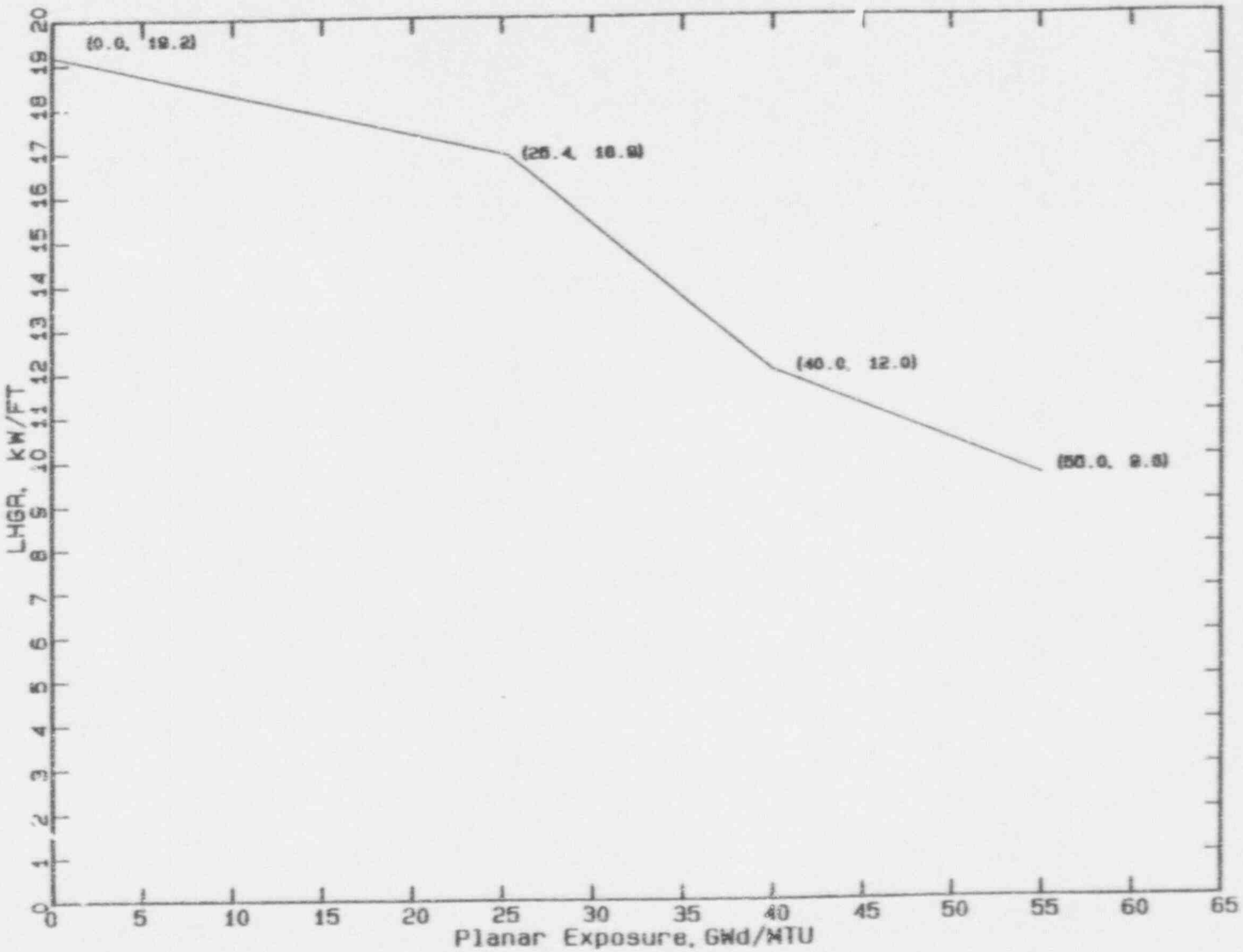

Figure 4.2 - PROTECTION AGAINST FUEL FAILURE LIMIT DURING AOO'S FOR 8X8 FUEL

5.0 REFERENCES

1. "Generic Mechanical Design for Exxon Nuclear Jet Pump BWR Reload Fuel", XN-NF-85-67(P)(A), Revision 1, September 1986.
2. "RODEX2 - Fuel Rod Thermal-Mechanical Response Evaluation Code", XN-NF-81-58(NP)(A), Supplement 1 & 2, Revision 2, March 1984.
3. "RODEX2A - Fuel Rod Thermal-Mechanical Evaluation Model", XN-NF-85-74(P)(A), August 1986.
4. "Grand Gulf 1 XN-1 Design Report, Mechanical, Thermal-Hydraulic and Neutronic Design for Exxon Nuclear JP BWR/6 Fuel Assemblies", XN-NF-83-25, Revision 1, August 1983.
5. "Qualification of Exxon Nuclear Fuel for Extended Burnup (BWR)", XN-NF-82-06(P)(A), Supplement 1, 4 and 5, November 1985.
6. XN-NF-82-06(P)(A), Supplement 1, Revision 2, "Qualification of Exxon Nuclear Fuel for Extended Burnup", May 1988.
7. "Grand Gulf Unit 1 Reload XN 1.3 Cycle 4 Mechanical Design Report", ANF-88-183(P), November 1988.

EMF-88-183(NP)
Supplement 1

Issue date: 10/21/91

GRAND GULF UNIT 1

RELOAD XN-1.3, CYCLE 4

MECHANICAL DESIGN REPORT

SUPPLEMENT 1

Distribution

S. L. Leonard (8)
Document Control

Attachment 10 to GNRO-91/00186