ADVANCED NUCLEAR FUELS *CORPORATION*

ANF-89-80(NP)
Revision 0

Issue Date: 5/23/91

DRESDEN UNIT 2 AND UNIT 3

8X8 EXTENDED BURNUP DESIGN REPORT

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1.0 INTRODUCTION

it also provides a summary discussion and references the detailed discussion of the design description, design criteria, technical bases, supporting analyses, and test results for the Advanced Nuclear Fuels Corporation Jet Pump Boiling Water Reactor fuel for the Dresden Unit 2 and Unit 3 Nuclear Power Reactors.

This document reports the results of design calculations performed to support a higher fuel assembly exposure than that reported previously.

The majority of the mechanical design related sections of this report are covered by specific references to the generic mechanical design report. Where applicable, the analysis has been extended to cover the increased burnup

2.0 SUMMARY

The ANF 8x8 fuel design for Dresden Unit 2 and Unit 3 has been evaluated for a peak assembly exposure The results of the evaluation indicate that the Design Criteria are met. The fuel description, mechanical, thermal/hydraulic, and neutronic design are summarized below.

2.1 Design Description Summary

The ANF 8x8 assembly design for Dresden 8x8 fuel uses 63 fuel rods and one centrally located water rod which functions as a spacer capture rod. Seven spacers maintain fuel rod spacing. The design uses a quick-removable upper tie plate design to facilitate fuel inspection and bundle reconstitution of irradiated assemblies.

The fuel rods are clad with zircaloy-2, The rods are pressurized, and contain either UO_2-Gd_2O_3 or UO_2 with a nominal density of 94.5% TD, and two diametral pellet-to-clad gap sizes (9.5 mils and 8.5 mils). Natural uranium axial fuel blanketing, at the top and the bottom of the fuel column, is provided for greater neutron economy.

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, differential fuel rod growth, creep bow, and spacer grid design. The analyses justify irradiation

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 below the material strength limits.

- The cladding strain during anticipated operating occurrences (AOO's) does not exceed 1.0%.
- The maximum fuel rod internal rod pressure remains below ANF's criteria limit.
- The fuel centerline temperature remains below the melting point during (AOO's).
- The cladding fatigue usage factor is within the design limit.
- Structural members have adequate strength to support handling and hydraulic loads.
- The cladding diameter reduction

  Compliance to this criterion prevents
  the possibility of creep collapse.
- Evaluations of assembly growth and differential fuel rod growths 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 the respective EOL exposures.

TABLE 2.1 FUEL ASSEMBLY COMPONENT DESCRIPTION

	Design Characteristics
Fuel Assembly	
Array	8x8
Width, inches	5.251
Length, inches	171.29
Rod Pitch, inches	.641
Number of Fuel Rods	63
Number of Water Rods	1
Fuel Rod	
Active Fuel Length, inches	145.24
Upper and Lower Natural UO_2, inches	6.00
Cladding Material	Zircaloy-2
Cladding OD, inches	.484
Cladding ID, inches	.414
Fuel Pellets	
Fuel Material	UO_2 Sintered Pellets
Density % of TD	94.5%

3.0 DESIGN DESCRIPTION

The design description for the Advanced Nuclear Fuels Corporation Jet Pump Boiling Water Reactor for Dresden 8x8 reload fuel is given in Table 2.1. More specific design data is provided in Reference 1.

4.0 DESIGN CRITERIA

The detailed design criteria for the Advanced Nuclear Fuels Corporation Jet Pump Boiling Water Reactor for Dresden 8x8 reload fuel is given in Reference 2, Section 2.0, and Reference 3.

5.0 MECHANICAL DESIGN

Analyses reported in the Generic Mechanical Design Report(1) were performed to justify irradiation

This document reports the results of design calculations performed to support higher fuel assembly exposure than that reported previously. Figure 5.1 is the LHGR limit used in the steady state fuel rod performance evaluation. Figure 5.2 is the limit to protect against fuel damage during anticipated operational occurrences (AOOs).

These values are consistent with the peaking factors used in the NRC approved Generic Mechanical Design Report and are conservative estimates of the maximum exposures to be reached with the Dresden 8x8 fuel.

5.1 Fuel Rod Analyses

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 These values are consistent with the peaking factors used in the NRC approved Generic Mechanical Design Report(1). The exposures assumed are conservative estimates of the maximum exposures to be reached with the Dresden 8x8 fuel. The design power history used in Reference 1 has been extended to the higher exposure values in Dresden. The analyses results reported here demonstrate compliance with the design criteria.

5.1.1 Maximum Cladding Strain During Steady State Operation

The maximum cladding strain during steady state operation is limited to <1% to avoid ductile cladding fracture.

The analyses have been performed with RODEX2A. The results indicate that the strain is below 1%.

5.1.2 Maximum Cladding Stress During Steady State Operation

Fuel rod cladding stresses during steady-state operation are calculated using linear elasticity theory. The design criteria is in accordance with the ASME pressure vessel code.

Each individual stress is calculated inside and outside the cladding and at both midspan and spacer level. The applicable stresses at each level are then combined to get the maximum stress intensities.

The assumptions made in the analyses reported in Reference 1 have been reviewed to determine if additional calculations were required.

Consequently, the analysis results reported in Table 3.3 of Reference 1 are applicable.

5.1.3 Anticipated Operational Occurrences Analysis

Two criteria are imposed on the fuel rod to avoid fuel failure during power changes caused by anticipated operational occurrences (AOO's). These are to limit the cladding strain to less than 1% and to maintain the maximum pellet temperature below melting. The AOO's are assumed to produce a maximum nodal power equal to those defined in Figure 3.4 of Reference 1. The analysis consists of calculating the cladding strain and fuel centerline temperature at the power levels defined in Figure 3.4 and verify that they remain below the design criteria.

The calculations performed in support of Reference 1 have been reviewed to determine if the higher exposure of the Dresden 8x8 fuel requires a reanalysis. It has been determined that the burnup at which the margin to the design criteria is the lowest is not at EOL, consequently, the analysis performed in support of Reference 1 are applicable to this design.

5.1.4 Fuel Rod Internal Pressure

The fuel rod internal pressure is limited

The analysis indicate that the maximum internal pressure is below the design criteria.

5.1.5 Fuel Pellet Centerline Temperature

The fuel pellet centerline temperature calculation performed in support of the results reported in Reference 1 has been reviewed. The review indicates that the minimum margin against fuel melting, accounting for the decrease in fuel melting temperature with burnup, takes place at BOL. Consequently, the analysis results in Figure 3.20 of Reference 1 are applicable.

5.1.6 Fuel Rod Cladding Fatigue

Fuel assembly shuffling, reactor power maneuvering, and anticipated operational occurrences (AOO's) will impose a cyclic loading on the cladding. To assure that the fuel rod does not fail due to stress cyclic fatigue, a fatigue analysis is performed.

A conservative calculation has been performed. The calculation assumed the number of cycles defined in Table 3.5 of Reference 1 was increased by to account for the additional residence time associated with the higher exposure The maximum cumulative damage was below the design criteria.

5.1.7 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 ANF 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 characteristic of the Dresden reactor compared to that in a PWR. Nevertheless, the fuel rods are analyzed to assure that fuel rod collapse will not occur.

The analysis results reported in Table 3.1 of Reference 1 are applicable.

5.1.8 Fuel Rod Spacing

Rod-to-rod and rod-to-channel spacing must not affect the assembly thermal performance. Thermal limits are not affected if the minimum rod-to-rod is greater The analysis performed in Reference 1 to calculate the maximum fuel rod bow has been evaluated for applicability at higher exposures.

. The maximum fuel rod channel closure at provides ample margin to the channel closure that could affect the assembly thermal performance.

5.1.9 Cladding Corrosion and Hydrogen Concentration

The current ANF design criteria is to maintain the metal loss due to corrosion to less than Hydrogen absorption is limited The analysis performed in Reference 1 have been evaluated and the effects of increasing the burnup have been obtained.

The evaluation indicates that at the revised exposure, the cladding corrosion and hydrogen absorption will remain well below the design criteria.

5.2 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.

5.2.1 Structural Strength

The structural strength of tie plates, locking mechanism, and tie rods is not decreased with exposure. The analysis and test results previously reported in Reference 1 are applicable.

5.2.2 Spacer Spring

ANF 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.

Increased exposures do not have a significant effect on spacer spring performance.

It is then concluded that the spacer spring design is acceptable

5.2.3 Assembly Growth

Assembly growth was determined by evaluating differential growth between standard fuel rods and tie rods.

Additionally, an evaluation of channel engagement with the lower tie plate seal as a function of irradiation exposure was performed. The calculations indicate that sufficient channel and end cap engagement is present to EOL.

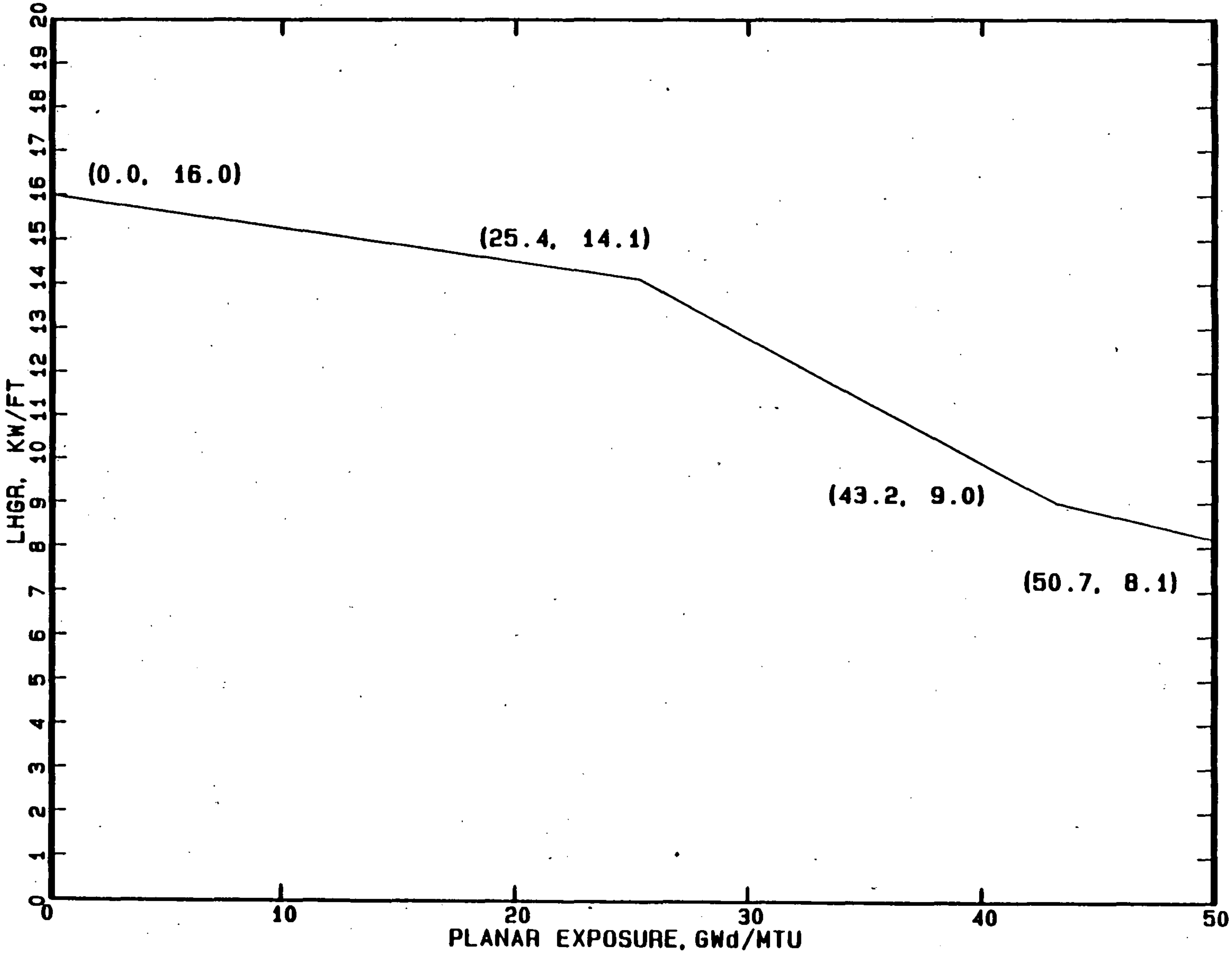

Figure 5.1 - LHGR LIMIT

GL Prime 147 910425.0719 PLOREX9A <FDSG>PEN147>CE>HBUG>UPL.A 36 617

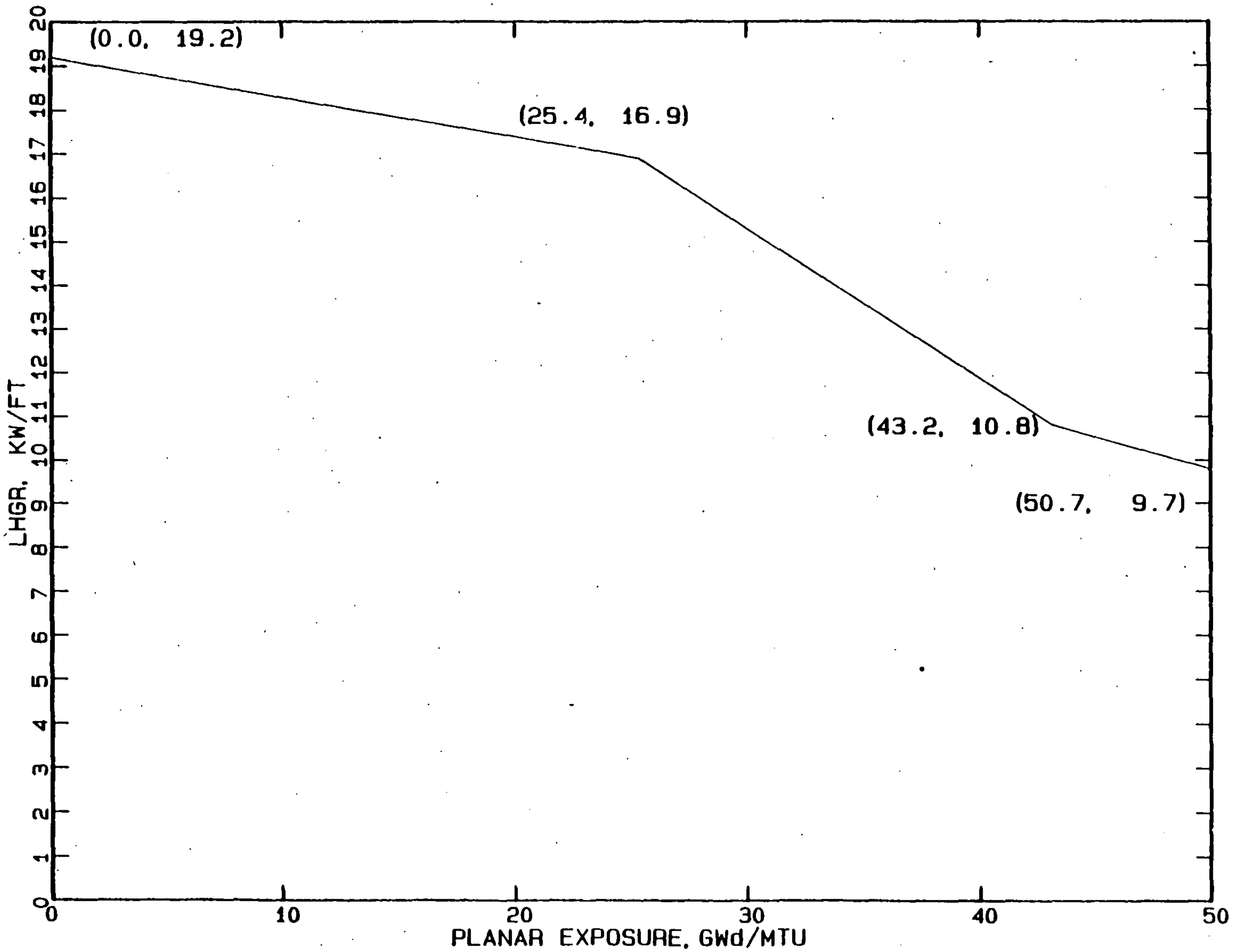

Figure 5.2 - PROTECTION AGAINST FUEL FAILURE LIMIT DURING AOO'S

6.0 REFERENCES

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8. XN-NF-82-06(P)(A), Supplement 1, Revision 2, "Qualification of Exxon Nuclear Fuel for Extended Burnup", May 1988.