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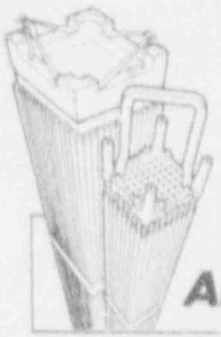

ADVANCED NUCLEAR FUELS CORPORATION

GRAND GULF 1 ANF-1.4 DESIGN REPORT MECHANICAL, THERMAL-HYDRAULIC AND NEUTRONIC DESIGN FOR ADVANCED NUCLEAR FUELS 9X9-5 FUEL ASSEMBLIES

APRIL 1991

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MECHANICAL, THERMAL-HYDRAULIC AND NEUTRONIC

DESIGN FOR ADVANCED NUCLEAR FUELS 9X9-5 FUEL ASSEMBLIES

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MECHANICAL, THERMAL-HYDRAULIC AND NEUTRONIC DESIGN FOR ADVANCED NUCLEAR FUELS 9X9-5 FUEL ASSEMBLIES

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DESIGN FOR ADVANCED NUCLEAR FUELS 9X9-5 FUEL ASSEMBLIES

APRIL 1991

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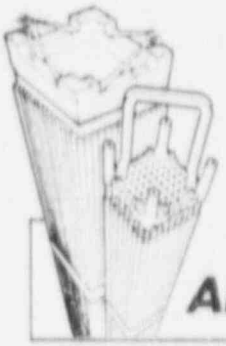

ADVANCED NUCLEAR FUELS CORPORATION

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1.0 INTRODUCTION AND SUMMARY

1.1 Introduction

This report provides a design description and a summary of the design criteria, technical bases, supporting analyses, and test results for the Advanced Nuclear Fuels (ANF) 9x9-5 Fuel Assemblies for the Grand Gulf Unit 1 Nuclear Power Reactor. The specific mechanical design analysis is reported in Reference 1.

1.2 Summary

The ANF 9x9-5 fuel design is shown to meet the Design Criteria and Technical Bases for Design. The fuel description, mechanical, thermal-hydraulic, and neutronic design are summarized below.

1.2.1 Design Description Summary

The 9x9-5 fuel assembly design uses 76 fuel rods and five water rods, one of 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 pressurized, clad with Zircaloy-2, contain either UO_2-Gd_2O_3 or UO_2 with a nominal density of 94.5% TD,

Natural

uranium axial fuel blankets are provided for greater neutron economy.

This design incorporates conventional water rods clustered in a cross configuration in the center of the bundle. The assemblies are designed to allow handling in the same manner, to the same extent, and with the same equipment as that now being used for 8x8 fuel.

1.2.2 Mechanical Design Summary

Mechanical design analyses were performed to evaluate cladding steady-state strain and stresses, transient stress, fatigue damage, creep collapse, corrosion and hydrogen absorption, fuel rod maximum pressure, differential fuel rod growth, creep bow, and grid spacer spring design.

The results of these analyses are presented in Reference 1.

1.2.3 Thermal-Hydraulic Summary

ANF 9x9-5 Reload Fuel Assemblies for Grand Gulf Unit 1 Cycle 5 have been evaluated to be hydraulically compatible with ANF 8x8 fuel resident in the reactor.

The ANF 9x9-5 fuel design is geometrically different from the resident ANF 8x8 design.

1.2.4 Neutronic Design Summary

The average enrichment and enrichment distribution have been selected to meet the Energy Services, Inc. energy requirements (Reference 8), while also meeting the requirements for shutdown margin and power coefficients defined in the Design Criteria and the operating license Technical Specifications.

2.0 DESIGN DESCRIPTION

Design description for the 9x9-5 fuel assemblies presented in Reference 1 are applicable to this report.

3.0 DESIGN EVALUATION

The design evaluation section presented in Section 3 of Reference 1 is applicable to this report.

4.0 THERMAL-HYDRAULIC ANALYSIS

The thermal-hydraulic analysis for the ANF fuel designs considers the hydraulic compatibility between the 9x9-5 fuel and the existing 8x8 fuel design currently in the Grand Gulf Unit 1 core. Further, the analysis assesses the thermal margin performance of the 9x9-5 fuel

This latter assessment is performed using the ANFB critical power correlation.

4.1 Design Criteria

Primary thermal-hydraulic design criteria for ANF BWR reload fuel applicable to Grand Gulf Unit 1 are as follows:

Hydraulic Compatibility

The hydraulic flow resistance of the reload fuel must be such that there are no significant unplanned impacts to total core flow or to the flow distribution among assemblies in the core.

Thermal Margin Performance

Fuel assembly geometry, including spacer design and rod-to-rod local power peaking, should minimize the likelihood of boiling transition during normal reactor operation as well as during anticipated operational transients. The fuel design should fall within the bounds of applicable empirically-based boiling transition correlations established and approved for ANF reload fuel. Within other applicable mechanical, neutronic, and fuel performance constraints the fuel design should achieve good thermal margin performance.

Bypass Flow

The fuel assembly design should provide sufficient flow in the bypass region between channels to provide adequate cooling for control rods and in-core detectors without unwarranted reductions in the active assembly flow.

4.2 Hydraulic Characterization

The basic geometric parameters for 8x8 and 9x9-5 fuel designs are summarized in Table 4.1.

Component loss coefficients for the 9x9-5 and 8x8 are based on tests documented in References 2, 3, and 4 and reported in Table 4.2.

The irreversible pressure loss coefficient for the orifice and the LTP is computed analytically using a model developed from test data reported in Reference 5.

The final aspect of the fuel hydraulic characterization is the development of the leakage flow resistances for the LTP flow holes for the LTP-to-channel seal. The method used to evaluate these resistances and their value has not changed since they were reported in Reference 6.

4.3 Hydraulic Compatibility and Thermal Performance

Hydraulic compatibility as it relates to thermal margin performance and the relative thermal margin performance of the 9x9-5 reload design and the 8x8 design have been determined. Detailed thermal-hydraulic analyses were

performed for the calculation of critical power ratios for both fuel types. The results are given for typical mixed core configuration (Grand Gulf Unit 1 Cycle 5) as well as for other configurations. The Cycle 5 configuration considers 284 9x9-5 reload fuel assemblies being co-resident with four (4) exposed 9x9-5 Lead Fuel Assemblies and 512 exposed 8x8 assemblies. Additionally, analyses for one other mixed core configuration as well as a full core of 9x9-5 loading and a full core of 8x8 have been performed.

The thermal-hydraulic analyses were performed in accordance with ANF's thermal-hydraulic methodology for BWR's which are implemented in ANF's XCOBRA code. Critical power ratios evaluated by XCOBRA are in accordance with ANF's ANFB Critical Power Correlation. This correlation is applicable to 8x8 and 9x9-5 fuel.

Table 4.3 summarizes the input conditions for the Cycle 5 analysis. These conditions reflect 100% design power and 100% design flow conditions for the Grand Gulf Unit 1 reactor

Table 4.3 also defines the core loading for the Cycle 5 core configuration.

Table 4.4 provides a summary of calculated thermal-hydraulic parameter results from the Grand Gulf 1 Cycle 5 mixed core configuration, and Table 4.5 provides a summary of results for all core configurations studied.

This confirms that the 9x9-5 fuel is hydraulically compatible with the reactor system.

4.5 Water Rod Design and Analysis

4.5.1 Design

The water rod design is based on the following criteria:

Analyses were performed to size the inlet holes for the water rods to satisfy the above criteria.

4.5.2 Analysis

The water rod design is analyzed to show adequate flow within the rod

Detailed hydraulic analyses of the water rods for the 9x9-5 reload fuel design for Cycle 5 are presented in Table 4.6.

TABLE 4.1 COMPARATIVE DESCRIPTION OF BWR/6
ANF 9x9-5 AND ANF 8x8 FUEL

Fuel Parameter	ANF 9x9-5	ANF 8x8

TABLE 4.2 HYDRAULIC CHARACTERIZATION COMPARISON
BETWEEN ANF 9X9-5 AND ANF 8X8 FUEL ASSEMBLIES*

TABLE 4.3 GRAND GULF UNIT 1 THERMAL-HYDRAULIC DESIGN CONDITIONS

Reactor Conditions		
Core Power Level (MWt) (100%)		3833
Core Exit Pressure (psia)		1050
Core Inlet Enthalpy (Btu/lbm)		527.9
Total Core Coolant Flow (Mlbm/hr)		112.5
Core Loading		
	Central Region	Peripheral Region
ANF 8x8	420	92
ANF 9x9-5 LTA	4	0
ANF 9x9-5 Reload	284	0
TOTAL	708	92
Core Power Distribution		
Axial Power Shape		Figure 4.1
Average Bundle Power(MWt)		4.79
Central Region		5.21
Peripheral Region		1.58
Average Bundle Power by Type (MWt)		
ANF 8x8 Central		4.60
ANF 9x9-5 LTA Central		5.82
ANF 9x9-5 Reload Central		6.10
ANF 8x8 Peripheral		1.58

TABLE 4.3 GRAND GULF UNIT 1 THERMAL-HYDRAULIC DESIGN CONDITIONS
(Continued)

Core Power Distribution (cont)	
Fuel Assembly Description	
Hydraulic Resistance Characteristics	Table 4.2
Number of Fuel Rods per Assembly	
ANF 8x8	62
ANF 9x9-5	76
Number of Spacers (all fuel types)	7
Active Fuel Length (ft)	12.5
Fuel Rod Length (tie plate-to-tie plate) (ft)	13.5

TABLE 4.4 GRAND GULF UNIT 1 CYCLE 5 - MIXED CORE
THERMAL-HYDRAULIC RESULTS

Core Average Results	
Exit Enthalpy (Btu/lbm)	644.2
Exit Quality (active region)	16.2%
Exit Void Fraction	0.683

TABLE 4.5 GRAND GULF UNIT 1 MIXED CORE THERMAL-HYDRAULIC RESULTS

Core Loading	Assembly
1/3 ANF9x9-5	ANF 9x9-5
	ANF 8x8
2/3 ANF9x9-5	ANF 9x9-5
All ANF9x9-5	ANF 9x9-5
All ANF8x8	ANF 8x8

Core Loading
1/3 9x9-5
2/3 9x9-5
All 9x9-5
All 8x8
All GE 8x8

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* As determined by the ANFB correlation

TABLE 4.6 ANF 9x9-5 WATER ROD FLOW AS A FUNCTION OF ACTIVE CHANNEL FLOW (AT CONSTANT POWER)

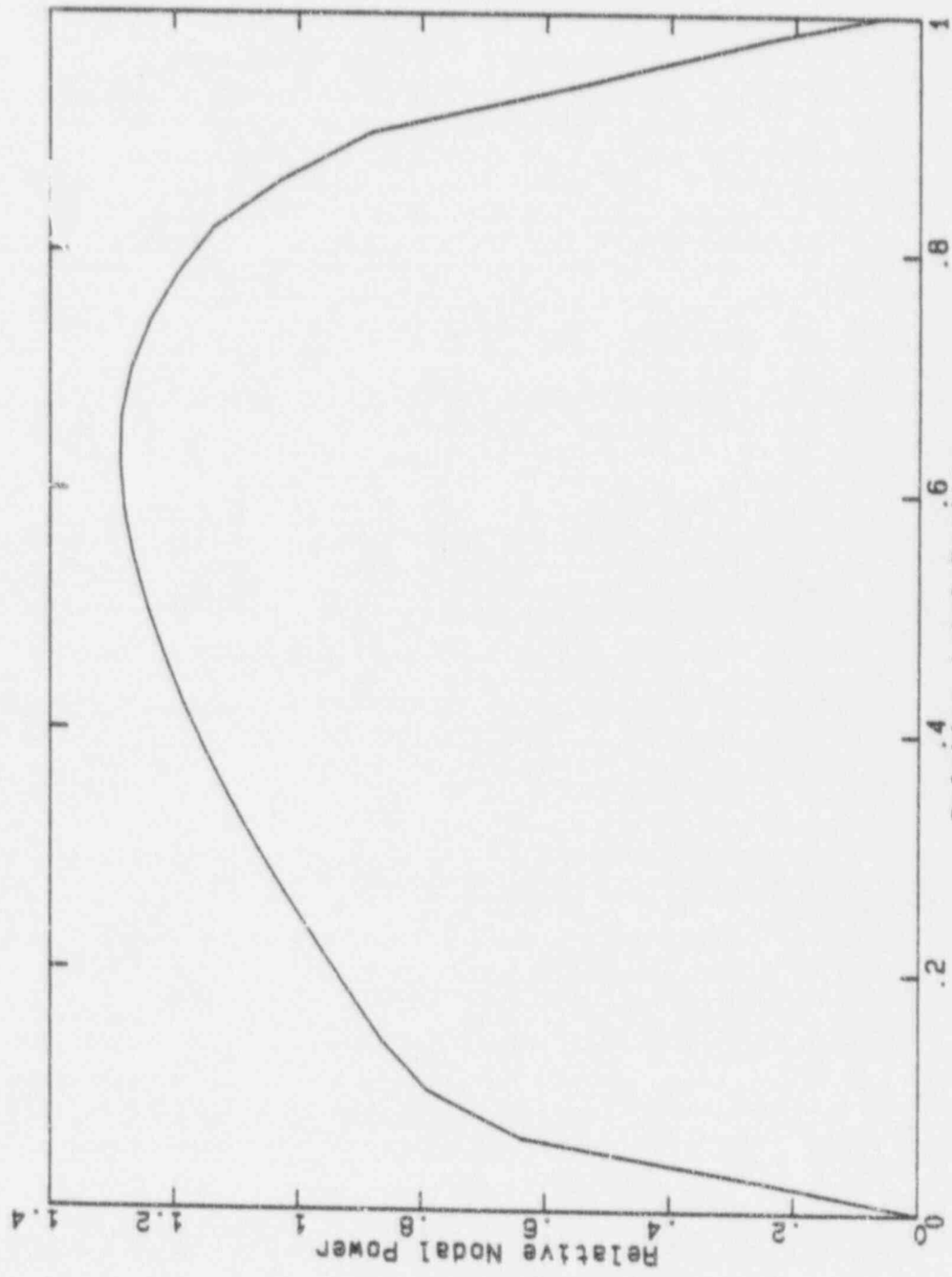

Figure 4.1 Grand Gulf Unit 1 EOC5
MICROBURN Axial Power Shape

Figure 4.2 Hydraulic Demand Curves For
ANF 9x9-5 and ANF 8x8 Fuel

5.0 NEUTRONIC DESIGN

The results of the neutronic design analysis for ANF's Type ANF-1.4 reload fuel assembly design for Grand Gulf Unit 1 are presented in this section.

The key nuclear design characteristics for the ANF Type ANF-1.4 fuel assembly are summarized below:

5.1 Neutronics Design Parameters

The key neutronic design parameters for the ANF Type ANF-1.4 fuel design are presented in Table 5.1.

5.2 Enrichment Level

The nominal enrichment level (average fissile content of the enriched lattice of the ANF-1.4 reload fuel assemblies is U-235.

The enrichment distribution of the ANF Type ANF-1.4 reload fuel designs were selected on the basis of maintaining a balance between the local power peaking factors, assembly reactivity, maximum average planar linear heat generation rate (MAPLHGR), and the minimum critical power ratio (MCPR) considerations. The enrichment distributions of the ANF Type ANF-1.4 reload design are shown in Figures 5.1 through 5.3. Figures 5.4 and 5.5 show the fuel pin design and placement for the ANF-1.4L and ANF-1.4H assembly types.

5.3 Beginning-of-Life Fuel Assembly Reactivity

The beginning-of-life (BOL) fuel assembly reactivity (k_∞) was calculated using the CASMO-3G[9] computer code for each of the enriched lattices.

The reactivity was computed at six specific moderator and fuel conditions expected during normal reactor operation.

At each condition, the controlled and uncontrolled values of k_∞ are determined using the CASMO-3G code.

The results of these calculations for each enriched lattice are presented in Table 5.2 along with the resulting control rod reactivity worth ($\Delta k_\infty/k_\infty$).

5.4 Exposure Dependent Fuel Lattice Reactivity and Isotopics

The exposure dependent fuel lattice reactivity (uncontrolled) was calculated with the CASMO-3G computer code. The fuel lattice reactivity (k_∞) values are shown in Tables 5.3 through 5.12 and are presented graphically in Figures 5.6 through 5.15.

The void and exposure dependent fuel lattice isotopics are also obtained from the CASMO-3G calculations. The isotopics are presented in Tables 5.13 through 5.42.

5.5 Local Power and F-Factor Distributions

5.6 Assembly Reactivity Defects at BOL

Lattice reactivity defects (Δk_∞) at BOL were evaluated using the results shown in Table 5.2. This information is provided for comparison purposes. The reactivity characteristics of the fuel assembly design are fully characterized in evaluating the transient performance of the fuel as part of the licensing phase.

The results of these calculations are tabulated in Tables 5.43 through 5.45 for uncontrolled lattices.

TABLE 5.1 GRAND GULF UNIT 1 RELOAD BATCH ANF-1.4
NEUTRONIC DESIGN VALUES

Fuel Pellets	
Fuel Material	UO_2 Sintered Pellets
Density	
g/cc	10.36
% of TD	94.5
Diameter, inches	
Reference Fuel Temperature, °F	1070.83
Fuel Rod	
Total Active Fuel Length, inches	150.0
Length of Axial Blanket, inches	
Top	12.0
Bottom	6.0
Cladding Material	Zircaloy-2
Clad I.D., inches	
Large	0.381
Small	0.359
Clad O.D., inches	
Large	0.443
Small	0.417

TABLE 5.1 GRAND GULF UNIT 1 RELOAD BATCH ANF-1.4
NEUTRONIC DESIGN VALUES (CONT.)

Fuel Assembly:	
Number of Rods Total	81
Fuel Rod Array; Fuel Assembly for Reload Batch ANF-1.4	Figures 5.1 Through 5.5
Core Data:	
Number of Fuel Assemblies	800
Rated Thermal Power Level, MWt	3833
Rated Core Flow, Mlb/hr	112.5
Core Inlet Subcooling, Btu/lbm	22.25
Core Pressure, psia	1050
Moderator Temperature, °F	551.00
Channel Dimensions:	
Thickness, inches	0.120
Internal Face-to-Face Dimension, inches	5.215
Corner Outside Radius, inches	0.500

TABLE 5.1 GRAND GULF UNIT 1 RELOAD BATCH ANF-1.4
NEUTRONIC DESIGN VALUES (CONT.)

Control Rod Data

Total Blade Span, inches	9.804
Total Blade Support Span, inches	1.550
Blade Thickness, inches	0.3280
Blade Face-to-Face Internal Dimension, inches	0.238
Number of B_4C Rods Per Blade	72
B_4C Rod O.D., inches	0.220
B_4C Rod I.D., inches	0.166
Percent of B_4C Theoretical Density	70

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(Tables 5.2 - 5.42)

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Section 5.0 Local Peaking and F-Factor Distribution

Figures 5.16 thru 5.318

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(Figures 5.16 - 5.318)

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MECHANICAL, THERMAL-HYDRAULIC AND NEUTRONIC

DESIGN FOR ADVANCED NUCLEAR FUELS 9X9-5 FUEL ASSEMBLIES

Distribution

GJ Busselman
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TABLE 5.43 GRAND GULF UNIT 1 RELOAD BATCH ANF-1.4
CALCULATED LATTICE REACTIVITY DEFECTS AT BOL,

Defects
Doppler Defect
Void Defect
Moderator Temperature Defect (Cold to Hot Standby)

TABLE 5.44 GRAND GULF UNIT 1 RELOAD BATCH ANF-1.4
CALCULATED LATTICE REACTIVITY DEFECTS AT BOL,

Defects
Doppler Defect
Void Defect
Moderator Temperature Defect (Cold to Hot Standby)

TABLE 5.45 GRAND GULF UNIT 1 RELOAD BATCH ANF-1.4
CALCULATED LATTICE REACTIVITY DEFECTS AT BOL,

Defects
Doppler Defect
Void Defect
Moderator Temperature Defect (Cold to Hot Standby)

Pages 76 - 90 have been deleted.
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