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ADVANCED NUCLEAR FUELS CORPORATION

MECHANICAL LICENSING REPORT FOR H.B. ROBINSON HIGH THERMAL PERFORMANCE FUEL ASSEMBLIES

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MECHANICAL LICENSING REPORT
FOR H.B. ROBINSON
HIGH THERMAL PERFORMANCE FUEL ASSEMBLIES

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October 1990

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1.0 INTRODUCTION

The purpose of this report is to provide the design description and evaluation of the Advanced Nuclear Fuels Corporation (ANF) 15x15 high thermal performance fuel assemblies for the H.B. Robinson reactor starting with Reload ANF-11.

The report describes the revised fuel assembly, which utilizes high thermal performance spacers (HTPs), intermediate flow mixers (IFMs) and a small hole debris resistant lower tie plate. The UO_2 and UO_2-Gd_2O_3 fuel rods are unchanged from previous reloads. The results of the mechanical analyses are provided to show compliance with the requirements of Section 4.2 of the Standard Review Plan (NUREG 800).

The fuel was evaluated for a peak assembly burnup value of 52.5 GWd/MTU and for an increased $F_{\Delta H}$ limit of 1.79 and a F_Q limit of 2.55. The evaluation of the rods and fuel assembly was performed in accordance with the USNRC approved methodology[1] supplemented with calculations in accordance with referenced documents ANF-88-133(P) and ANF-89-060(P)[2,3].

2.0 SUMMARY

The mechanical analysis shows the fuel assembly will meet the design criteria for the expected operating conditions and postulated accidents to the design assembly burnup of 52.5 GWd/MTU.

The key results of the fuel system damage evaluation are:

- The maximum steady-state cladding and assembly component stresses are within the ASME Boiler and Pressure Vessel Code limits.
- The maximum steady-state cladding strain is well below the 1% design limit.
- The cladding and assembly component fatigue usage factors are below the design limit.
- Fretting wear of the spacers and fuel rods is precluded.
- Corrosion of the fuel rod and the fuel assembly structural components is below the design limit.
- Fuel rod bowing will be limited so that it has no impact on thermal margins.
- Axial growth of the fuel rods and fuel assembly is accommodated within the design clearances.
- The fuel rod internal pressure remains below the reactor system pressure throughout life.
- Fuel assembly liftoff will not occur during normal operation.

The fuel rod failure evaluation shows that the fuel rod design will operate without failure during normal operation and anticipated transients, and the fuel performance is properly accounted for in the accident analyses.

The key results of the fuel rod failure evaluation are:

- Internal hydriding is prevented by controlling hydrogen in the manufacture of the fuel.
- Evaluation of the fuel rod behavior shows that cladding creep collapse will not occur.
- Adequate cooling exists to prevent overheating of the cladding.
- Fuel melting will not occur during normal operation and anticipated operational occurrences (AOOs).
- The transient circumferential strain is within the 1% design limit.
- The impact of cladding rupture is incorporated in the LOCA analysis.
- Fuel coolable geometry and the capability for control rod insertion is shown to be maintained during seismic and LOCA events.

3.0 DESIGN DESCRIPTION

The H.B. Robinson 15x15 fuel assemblies each contain 204 fuel rods, 1 instrument tube, and 20 guide tubes. The fuel assembly differs from previous reloads[4,5] in that six of the seven standard bi-metallic spacers are replaced by all Zircaloy high thermal performance (HTP) spacers and 3 intermediate flow mixers (IFMs) are added. The ..movable stainless steel upper tie plate with Inconel 718 leaf springs is unchanged from earlier reloads while the lower tie plate is replaced with a small hole debris resistant design.

3.1 Fuel Assembly Description

The high thermal performance (HTP) fuel assembly for H.B. Robinson incorporates the same fuel rod design as provided in previous reloads. The fuel assembly cage is revised, whereby, at six of the seven spacer locations, the bi-metallic grid spacer is replaced by an all Zircaloy HTP spacer. At three mid-span locations, in limiting DNB regions toward the top of the fuel assembly, intermediate flow mixers are added. The bi-metallic spacer at the bottom of the assembly is modified so that the fuel rod backup dimples cover the full spacer periphery, for increased resistance to handling damage. The fuel assembly outline is shown in Figure 3.1. The cage construction is shown in Figure 3.2. The HTP spacer is shown in Figure 3.3, the IFM in Figure 3.4.

There are several design differences that make the HTP and IFM grids an improvement over the ANF bi-metallic design. The change from an Inconel spring to a Zircaloy spring reduces the neutron capture cross-section of the spacers. The internal strips of both HTP and IFM grids contain slanted channels that create a swirling flow pattern in the coolant. This flow pattern increases coolant mixing and thus improves the heat transfer from the fuel rods. Also, the HTP spacer is structurally stronger than the bi-metallic spacer.

The stainless steel lower tie plate was revised by changing from large to small flow holes. This significantly increases the debris resistance of the fuel assembly by limiting the size and quantity of potentially damaging foreign objects which can pass through the lower tie plate and lodge in the fuel rods. The upper tie plate, the guide tube latching system, and the holddown spring are unchanged.

To accommodate differential guide tube to fuel rod growth to higher burnup, more space was provided between the upper and lower tie plates. This was achieved by removing 0.2 inch from the legs of the lower tie plate and adding 0.2 inch to the guide tube length. Table 3.1 summarizes the characteristics of the fuel assembly.

3.2 Fuel Rod Description

The fuel rod is unchanged from earlier H.B. Robinson reloads. The fuel rod cladding is cold worked and stress relieved Zircaloy-4 tubing. Each fuel rod contains a center column of either enriched UO_2 or enriched UO_2-Gd_2O_3 fuel pellets with an upper and lower column of natural UO_2 pellets that act as axial blankets. The pellets are cold pressed and sintered to 94% of the theoretical solid density. The upper plenum region contains an Inconel X-750 compression spring. The upper and lower ends of the rods are seal welded with Zircaloy-4 plug-type end caps. The fuel rods are pressurized with helium gas.

Figure 3.5 is a drawing of the UO_2 fuel rod assembly. Figure 3.6 is a drawing of the UO_2-Gd_2O_3 fuel rod assembly. The UO_2-Gd_2O_3 fuel rod assembly differs from the UO_2 rod in that the axial blankets account for 12 rather than 6 inches at the stack ends. Table 3.2 summarizes the characteristics of the fuel rods and pellets.

TABLE 3.1 FUEL ASSEMBLY CHARACTERISTICS

	Characteristic
Array	15x15
Number of Fuel Rods	204
Number of Guide Tubes	20
Instrument Tube	1
Fuel Rod Pitch (in)	0.563
Rod to Rod Spacing (in)	0.139
Rod to Guide Tube Spacing (in)	0.079
Number of Bi-Metallic Spacers	1
High Thermal Performance (HTP) Spacers	6
Intermediate Flow Mixers (IFMs)	3
Guide Tube OD (in)	0.544
Guide Tube ID (in)	0.511
Dashpot ID (in)	0.455
Spacer Nominal Envelope (in)	
Lower Tie Plate Envelope (in)	8.424
Upper Tie Plate Envelope (in)	8.406
Overall Length Less Spring (in)	159.710
Overall Length (in)	161.300

TABLE 3.2 FUEL ROD CHARACTERISTICS

Cladding	Value
O.D. (in)	.424
I.D. (in)	.3640
Length (in)	151.64
Fuel Column	
Pellet O.D. (in)	.3565
Percent of Theoretical Density (%)	94.0
Enriched Length (in)	
UO_2 Rods	132.0
UO_2-Gd_2O_3 Rods	120.0
Axial Blankets Length, Each End (in)	
UO_2 Rods	6.0
UO_2-Gd_2O_3 Rods	12.0
Total Active Length (in)	144.0
Rod Assembly	
Total Fuel Mass (kg)	
UO_2 Rods	2.403
UO_2 w/4% Gad	2.375
UO_2 w/6% Gad	2.361
Fill Gas Pressure (psia He)	305

Pages 8 - 16 have been deleted.

4.0 FUEL ASSEMBLY ENVIRONMENT

This section summarizes the reactor conditions used in the evaluation of the fuel rods and fuel assembly including the steady-state, transient, and cyclic conditions.

4.1 Reactor Conditions for Design

- Core thermal power	2300 MWth
- System pressure	2250.0 psia
- Number of assemblies	157
- Nominal measured core mass flow rate	109 x E6 lb/hr
- Maximum core flow for holddown analysis	113 x E6 lb/hr
- Flow fraction for heat transfer	.955
- Core inlet temperature	546.3°F
- Core outlet temperature	603.0°F
- Maximum overpower	18%
- Fraction of heat from fuel rods	.974
- Core average LHGR	5.98 kW/ft
- Maximum allowable pellet factor, F_Q	2.55
- Maximum allowable rod factor, $F_{\Delta H}$	1.79
- Peak assembly burnup	52.5 GWd/MTU

4.2 Steady-State Power Histories

Power histories were created consistent with the methodology of Reference 1. Four power histories were used in the mechanical design evaluation of the UO_2 fuel rods. These histories represent the projected high power rods in each of the three cycles, and the high power rod in the peak exposure fuel assembly. Three power histories were provided for use in the analyses of the fuel rods containing gadolinia. Again, these histories represent the projected gadolinia-bearing rods with the high power in each of the three cycles. Each history follows a rod throughout its three-cycle residence in the reactor. The histories are consistent with operation at limits of $F_{\Delta H}$ = 1.79 and F_Q = 2.55 and a peak assembly exposure of 52.5 GWd/MTU. Each of these design histories has been plotted and the following list specifies the figure where it can be found and the index used as its designator.

Rod History Description	Figure	Index
High First Cycle UO_2	4.1, 4.2	1
High Second Cycle, UO_2	4.1, 4.2	2
High Assembly Exposure, UO_2	4.1, 4.2	3
High Third Cycle, UO_2	4.1, 4.2	4
High First Cycle, NAF*	4.3, 4.4	1
High Second Cycle, NAF	4.3, 4.4	2
High Third Cycle, NAF	4.3, 4.4	3

These histories are analyzed with the RODEX2 code for the design criteria pertinent to steady-state operation, and are used to establish the initial conditions for power ramps.

4.3 Power Ramps

In order to determine the stress and strain of the cladding during transients, ramps to maximum LHGR conditions are added to the steady-state power histories at various times during the irradiation.

4.4 Duty Cycles

The fatigue usage factors are determined for the design cycling conditions using the stresses determined in the ramping evaluation.

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* Neutron Absorber Fuel (UO_2-Gd_2O_3)

TABLE 4.1 DUTY CYCLES

1. Current and Anticipated Practice

   a. Weekly valve operating test
      100% to 70%
      hold @ 70% for 2.5 hours
      70% to 100% @ PREMACX rate

   b. Twice/month Steam Generator leakage test
      100% to hot standby
      hold @ hot standby for one day
      hot standby to 30%
      hold @ 30% for 0.5 hour
      30% to 100% @ PREMACX rate

   c. Once/6 months Steam Generator inspection
      100% to 0 power
      cold for one week
      0 power to 30%
      hold @ 30% for 0.5 hour
      30% to 100% @ PREMACX RATE

2. Load Follow

   100% to 50% to 100% @ PREMACX rate - 2/day

3. Arbitrary Cycles

   a. 10 scrams/year

   b. Step load decrease of 95% - 2/year

   c. Step load increase from 0 power to 30% - 2/year

   d. Step load increase of 20% power - 1/week

   e. 30% to 100% @ PREMACX rate - 2/year

5.0 DESIGN EVALUATION

Fuel rod and assembly design analyses and tests were performed in accordance with the requirements of Chapter 4.2 of the Standard Review Plan (SRP) and the ANF design bases and methodology presented in References 1, 2 & 3.

5.1 Fuel System Damage Evaluation

The following paragraphs discuss the ability of the high thermal performance fuel assemblies to meet the fuel system damage criteria described in Section 3.1 of SRP 4.2. The criteria apply to normal operation and anticipated transients.

5.1.1 Steady-State Stress

a) Fuel Rod

The cladding steady-state stress analysis was performed considering primary and secondary membrane and bending stresses due to hydrostatic pressure, flow induced vibration, ovality, spacer contact, pellet clad interaction (PCI), thermal and mechanical bow, and thermal gradients. Stresses were calculated for various combinations of the following conditions and locations:

- BOL (beginning-of-life) and EOL (end-of-life)
- Cold and hot conditions
- At mid-span and at spacer locations
- At both the inner and outer surfaces of the cladding

The applicable stresses in each orthogonal direction were combined to calculate the maximum stress intensities which were then compared to the design criteria[6]. The results of the analysis indicate that all stress values are within acceptable design limits for all combinations of conditions and locations. A summary of the stress intensity results is shown in Table 5.1.

b) Guide Tubes

The control rod guide tubes tie the assembly structure together and provide channels for the insertion of the control rods. The bottom section of the guide tube consists of a reduced diameter to produce a dashpot action when the control rods approach the end of their travel during a reactor trip. The guide tube stresses due to differences in frictional loads between HTP and bi-metallic spacers and the fuel rods which result from differential thermal expansion between the fuel rods and the guide tubes are examined. The stresses due to control rod insertion are unchanged from the reference analysis[7].

Axial Stresses:

As the power level of the reactor is increased, differential thermal expansion between the Zircaloy guide tubes and the hotter Zircaloy clad fuel tends to put the guide tubes in tension. After a period at power, vibration loads tend to reduce or eliminate differential thermal expansion loads. With a reduction in power, differences in temperature between the guide tubes and fuel rods decrease and cause compression loading on the guide tubes.

The magnitude of guide tube loading depends on the differential expansion between the fuel rods and guide tubes and upon whether or not the fuel rods slip through the spacers. If the force developed by the differential thermal expansion times the guide tube stiffness exceeds the friction force, slippage occurs first at the outer spacers with the friction force being exerted on the inner spans. The spacer friction forces are accumulated toward the center of the assembly until the accumulated force equals the expansion force. The weight of the assembly and the holddown force are also considered in the calculations.

Buckling:

As the power level in the reactor is reduced, the difference in temperature between the guide tubes and fuel rods decreases causing compressive stresses resulting in a buckling type load in the guide tube spans. The differential thermal expansion was conservatively considered as an external load and not limited by differential strain.

c) Spacer

The spacer grid attachments must withstand the frictional forces due to the differential thermal expansion between the fuel rods and the guide tubes. The grids are attached to the guide tubes and instrument tubes by resistance spot welds. The HTP spacer represents the limiting case because its friction forces are greater than those in the IFM, and because the HTP spacers are attached to each guide tube in four places, whereas, the bi-metallic spacer is attached in eight places.

d) Tie Plate Strength

Structural tests have been made on both the upper and lower tie plates for the H.B. Robinson assemblies.

5.1.2 Steady-State Strain

The cladding steady-state strain was evaluated with the approved RODEX2 code[8]. The code considers the thermal-hydraulic environment at the cladding surfaces; the coolant and rod pressures; and the thermal, mechanical, and compositional state of the fuel and cladding.

The calculations are performed on a time incremental basis with conditions updated at each calculated increment so that the power history and path dependent processes can be modeled. The axial dependence of the power and burnup distributions are handled by dividing the fuel rod into a number of axial and radial regions. Power distributions can be changed at any desired time, and the coolant and cladding temperatures are readjusted in all the regions. All the performance models, e.g., the deformations of the fuel and cladding and gas release, are calculated at successive times during each period of assumed constant power generation.

5.1.3 Cyclic Fatigue

a) Fuel Rod

The stress results from the ramping analysis (Section 5.2.5) were used to evaluate the cladding fatigue damage through EOL due to the cyclic power variations defined in Table 4.1. For each of these repetitive operations, high and low reactor power levels were identified. Fatigue damage was evaluated for the specified number of repetitive operations as the reactor power is cycled back and forth between these two levels.

The transient stress results were evaluated to determine the fatigue usage for each cycle based on the O'Donnel and Langer[9] design curve shown in Figure 5.3. These results were accumulated to determine the total fatigue usage factor which is tabulated in Table 5.2.

b) Guide Tube

The variation in stresses due to differential expansion at beginning of life (BOL) are used in determining the fatigue usage throughout the life of the guide tubes.

c) Spacer

The grid attachment welds must withstand cyclic stresses. The operating duty cycles defined in Table 4.1 were used to estimate the number of cycles that would be applied to the joint.

5.1.4 Fretting Wear

a) Bi-Metallic Spacer

Each grid spacer is designed to support the fuel rod without fretting wear throughout the design life of the fuel. For the bi-metallic spacer, this is assured by maintaining a positive spring force greater than the flow induced lateral vibration forces.

Spring Relaxation:

During the irradiation of a fuel assembly there is a gradual reduction of the holding force exerted by the Inconel spacer spring in the bi-metallic spacer. The force reduction is due to irradiation induced spacer spring relaxation and to creepdown of the fuel rod cladding. The rate of change is more rapid early in burnup with a gradual decrease in the rate of change throughout irradiation. The clad creepdown eventually reaches a minimum diameter when firm pellet clad interaction is established.

Flow Vibration Force:

The spring force, F_v, required to counteract the maximum lateral acceleration forces due to flow induced vibrations to the extent of preventing lift-off of the rod from the dimples

dspan calculated from Paidoussis's equation[10]

EI = Bending stiffness of clad

L = Span length

End-of-Life Bi-Metallic Spacer Spring Force:

At the end-of-life, the spacer spring must overcome the force due to flow-induced vibration. The fuel rod cladding will have relaxed during irradiation so there are no forces due to thermal or mechanical bow.

b) HTP and IFM Spacers

The prevention of fretting corrosion in the HTP and IFM spacers is demonstrated in accordance with the methods of reference ANF-89-060[3] by a combination of analysis and fretting tests. The design analysis determines the projected maximum end-of-life gap considering spring relaxation, clad creepdown, minimum fuel rod diameter, and minimum initial spring deflection at the beginning-of-life. Flow test data are used to confirm that fretting corrosion will not occur for the largest possible projected gaps.

Interference Analysis:

The spring interference (or gap) throughout life is determined with an incremental calculation

Flow Test Results:

Flow tests of up to 1000 hours have been used to determine the fretting characteristics associated with the bi-metallic design for various combinations of spring/clad interferences (or gaps).

5.1.5 External Corrosion

a) Fuel Rod

The waterside corrosion and hydrogen pickup in the cladding were evaluated with RODEX2 for the seven power histories used in the steady-state strain analysis.

b) <u>Fuel Assembly Cage</u>

The spacer and guide tube components have been analyzed for oxidation and hydrogen pickup using the MATPRO corrosion prediction model in RODEX[2].

5.1.6 Rod Bowing

The impact of rod bowing on the high thermal performance fuel assembly MDNBR values was evaluated[10] using the ANF Rod Bowing Methodology[11].

5.1.7 Axial Growth

a) Fuel Rods

Axial cxtension of Zircaloy fuel rods in a reactor is primarily due to irradiation-induced growth. The assembly is designed to assure that an axial clearance will exist between the fuel rods and the upper and lower tie plates at the end-of-life.

b) Fuel Assembly

The minimum clearance between the tie plates and the core plates at beginning-of-life is 0.723 inches. Fuel assembly growth data are based on ANF measured data shown in Figure 5.14.

5.1.8 Fuel Rod Gas Pressure

Calculation of the gas pressure within a fuel rod is performed with the RODEX2 code. Power histories for the gas pressure analysis were developed from each of the corresponding power histories shown in Section 4.0 in accordance with the USNRC approved methodology[1].

5.1.9 Assembly Liftoff

Since the holddown spring characteristics and the overall fuel bundle length for the high thermal performance design is the same as the bi-metallic fuel design, and since the HTP assembly pressure drop is slightly below that of the bi-metallic design, the existing holddown system for the 15x15 bi-metallic assembly design will provide ample holddown for the high thermal performance fuel.

5.2 Fuel Rod Failure Evaluation

The following paragraphs discuss the ability of the high thermal fuel performance assemblies to operate without failure during normal operation and anticipated transients, and the accounting for fuel rod failures in accident analyses.

5.2.1 Internal Hydriding

Hydriding as a cladding failure mechanism is precluded by controlling the level of moisture and other hydrogenous impurities in the rod during fabrication. This is done by careful cleaning and drying of the clad, and by careful moisture control of the pellets during fuel fabrication.

5.2.2 Creep Collapse

Creep collapse calculations were performed with the RODEX2 and COLAPX[12] codes.

collapse will not occur.

5.2.3 Overheating of Cladding

As stated in SRP Section 4.2, adequate cooling is assumed to exist when the thermal margin criterion departure from nucleate boiling ratio (DNBR) is satisfied. The method employed to meet the DNBR design limit has been reviewed by the NRC as part of a thermal hydraulic codes and methods. The Cycle 14 reload analysis report gives the details of this analysis and results.

5.2.4 Overheating of Fuel Pellets

The prevention of fuel failure from overheating of the fuel pellets is accomplished by ensuring that the peak linear heat generation (LHGR) during normal operation and anticipated operational occurrences (AOOs) does not result in calculated fuel centerline melting. This analysis is performed as part of the plant transient analysis (Chapter 15 analyses).

5.2.5 Transient Stress and Strain

a) Normal Operation

In order to determine the stresses and strains of the cladding during transients, ramps to maximum LHGR conditions were applied to the steady-state histories periodically throughout the irradiation.

The ramp LHGR levels were determined by assuming that a pellet from a rod at the $F_{\Delta H}$ limit can be ramped to F_Q, where $F_{\Delta H}$ is the maximum allowable rod peaking factor in the core (1.79) and F_Q is the maximum allowable pellet peaking factor in the core (2.55)

5.2.6 Cladding Rupture

ANF's cladding rupture model for LOCA analysis is described in a generic report (XN-NF-82-07) entitled, "Exxon Nuclear Company ECCS Cladding Swelling and Rupture Model." This report adopted the NUREG-0630 data base and modeling. The NRC has reviewed this report and found it acceptable. The impact of rupture using this accepted model is implicit in the LOCA results reported in the reload analysis report for Cycle 14.

5.2.7 Fuel Rod and Assembly Damage from External Forces

The fuel assembly cage must assure that fuel assembly coolable geometry, and the ability to insert control rods, is maintained under anticipated accident conditions. Axial and lateral loading conditions due to seismic and LOCA events have previously been analyzed and reported in the XN-NF-76-47 (P)(A)[15] for the 15x15 fuel assembly with bi-metallic spacers.

Due to the incorporation of different fuel assembly components, namely HTP and IFM spacers and the debris-resistant lower tie plate, the seismic-LOCA evaluation reported in the reference has been readdressed. The differences between the reference and the revised fuel assembly design have been analyzed to show that the characteristics of the new fuel design are equivalent to or improved over the reference design.

a) Core Evaluation

b) HTP Spacer

The strength characteristics of the HTP spacer are compared with the data for the bi-metallic spacer and the simulated Westinghouse spacer considered in the reference analysis in Table 5.4. The comparative HTP and bi-metallic spacer load-deflection characteristics are shown in Figure 5.26.

The grid spacer strength was evaluated in accordance with SRP 4.2 Appendix A guidelines, whereby, the maximum allowable crushing load is the 95% confidence lower limit on the mean of the crush test measurements.

c) Guide Tubes

Lateral Bending Stresses:

Combined Stresses:

Buckling:

d) Lower Tie Plate

The purpose of the lower tie plate is to axially support the fuel assembly during normal and accident conditions.

e) Fuel Rod Stresses

The use of HTP spacer cages may have a slight effect on the resultant rod stresses due to increased beginning-of-life fuel rod-to-spacer friction and a potentially larger effect due to the increased strength of the HTP grid spacer. To determine the effect of these factors on the fuel rod seismic-LOCA stresses, each stress component was individually examined.

TABLE 5.1 SUMMARY OF LIMITING STEADY-STATE CLADDING STRESSES

			Primary Membrane		Net Primary		Primary and Secondary	
Location	Time	State	(ksi)	M.S.	(ksi)	M.S.	(ksi)	M.S.

Notes:

MS = Criteria/Stress Intensity - 1

Primary Membrane Stress
Criteria = lower value of 2/3 Y or 1/3 U
Y = yield strength (ksi)
U = ultimate strength (ksi)

Net Primary Stress = Primary Membrane + Primary Bending
Criteria = lower value of Y or 1/2 U

Primary + Secondary Stresses
Criteria = lower value of 2 Y or U

TABLE 5.2 MAXIMUM RAMP STRESS AND STRAIN SUMMARY

Power History	Maximum Hoop Stress (ksi)	Maximum Total Uniform Strain (%)	Fatigue Usage	No. of Cycles

* BOC ramp with a pellet chip

TABLE 5.3 CALCULATED VALUES FOR STEADY-STATE STRAIN, CORROSION, AND HYDROGEN ABSORPTION

Power History	Circumferential Outward Creep Strain (% O. Rad.)	Peak Local Oxide (microns)	Peak Hydrogen Absorption (ppm)

TABLE 5.4 SPACER CHARACTERISTICS AT OPERATING TEMPERATURE OF 650°F.

Dynamic Stiffness (lb/in)	Dynamic Strength (lbs)

TABLE 5.5 GUIDE TUBE COMBINED SEISMIC-LOCA STRESSES

Load Condition	Stress at lower end of outboard guide tube (psi)

* Stress criteria from ASME Boiler & Pressure Vessel Code, Section III, Division I, Appendix F, 1980.

TABLE 5.6 FUEL ROD COMBINED SEISMIC-LOCA STRESSES

Load Condition	Stress (psi) S_{hoop}	S_{axial}	S_{radial}

* Capability of HTP spacer.

** Radial stress is assumed 0 psi for conservative stress intensity calculation.

Pages 53 - 73 have been deleted.

6.0 TESTING, INSPECTION, AND SURVEILLANCE PLANS

6.1 Testing and Inspection of New Fuel

As described in SRP Section 4.2, testing and inspection plans for new fuel should include verification of significant fuel design parameters. ANF's high thermal performance fuel design is tested and inspected during manufacture to comply with the engineering and quality control requirements of ANF's reload parts list, drawing and specifications.

The requirements of ANF's quality control program are provided in ANF-1, which addresses design, process, quality, procurement and document control applicable to design and manufacture of fuel system component parts, fuel pellets, rods and assemblies. Fuel inspections vary for the different component parts and may include dimensions, visual appearance, audits of test reports, material certification, and non-destructive examinations. Pellet inspection is performed for dimensional characteristics such as diameter, density, length, and squareness of ends. Fuel rods, absorber rods, upper and lower plates, and spacer grid inspections consist of non-destructive examination techniques such as leak testing, weld inspection, and dimensional measurements.

Process control procedures are described in detail. In addition, for any tests and inspections performed by other vendors, ANF reviews the quality control procedures and inspection plans to ensure that they are equivalent to those described in ANF-1 and are performed properly. The requirements are changed from those for the existing H.B. Robinson fuel to the high thermal performance fuel only in the dimensional and fabrication details and inspection plans of the revised and new HTP and IFM components.

6.2 Post-Irradiation Surveillance

assemblies which are currently in their first cycle will be examined after irradiation to verify that the fuel assemblies are performing as anticipated.

7.0 REFERENCES

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FOR H.B. ROBINSON

HIGH THERMAL PERFORMANCE FUEL ASSEMBLIES

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