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MECHANICAL DESIGN REPORT SUPPLEMENT FOR KEWAUNEE HIGH BURNUP (49 GWd/MTU) FUEL ASSEMBLIES

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EXXON NUCLEAR COMPANY, INC.

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MECHANICAL DESIGN REPORT SUPPLEMENT FOR

KEWAUNEE HIGH BURNUP (49 GWd/MTU) FUEL ASSEMBLIES

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MECHANICAL DESIGN REPORT SUPPLEMENT FOR KEWAUNEE HIGH BURNUP (49 GWd/MTU) FUEL ASSEMBLIES

1.0 INTRODUCTION

The Kewaunee XN-1 through XN-4 Reload fuel was originally designed for an average fuel assembly burnup of 33 GWd/MTU, then reanalyzed for a peak fuel rod burnup of 43 GWd/MTU. This report describes the mechanical design analyses which show that the fuel from Reloads XN-1 to XN-4 can be irradiated to 49 GWd/MTU peak rod burnup, using as-built fuel dimensional characteristics. It also describes the analyses performed to qualify the XN-5 through XN-9 fuel Reloads, which are characterized by design improvements, for a peak rod burnup of 49 GWd/MTU.

2.0 SUMMARY

The fuel design for the Kewaunee plant has been modified starting with the XN-6 Reload to accommodate to higher burnup. The changes consist of tighter specifications for the cladding characteristics and the application of a fuel resinter density change limit. The as-built Reload XN-5 satisfied these new specifications.

The existing reload fuel designs and the modified design have been reanalyzed to support an increase in peak rod burnup up to 49 GWd/MTU. Issues not affected by the increased burnup are covered by the base Design Report.[1] Mechanical design analyses were performed to evaluate cladding steady-state strain, transient stress and strain, fatigue, creep collapse, corrosion, hydrogen absorption, fuel rod internal pressure, elongation, and fuel assembly growth.

Design criteria consistent with current ENC methodology were used in the analyses. Some of the design codes and techniques have been improved since the original mechanical design analysis was performed for Reloads XN-1 and XN-4 which justified up to a peak rod burnup of 43 GWd/MTU. The strain, pressure and collapse analyses have been performed using the RODEX2 code version approved by the NRC in 1983. The ramp stress/strain analysis has been evaluated against both the latest strain criteria and against stress criteria, namely to protect against failure by stress corrosion cracking.

The current analyses were performed to a peak rod burnup of 49 GWd/MTU, both for the reloads with the new specifications and for the earlier reloads. Bounding power histories have been used.

The results indicate that all the mechanical design criteria are satisfied.

o The maximum end-of-life (EOL) steady-state cladding strain meets the 1.0% design limit.

o The cladding stress and strain during power ramps, calculated using different overpower conditions, do not exceed the design stress corrosion cracking threshold or the 1.0% strain limit.

o The cladding fatigue usage factor is within the design limit.

o The end-of-life fuel rod internal pressure is less than the system pressure.

o The criterion for the prevention of creep collapse is satisfied.

o The maximum calculated EOL thickness of the oxide corrosion layer and the maximum calculated concentration of hydrogen in the cladding are within the design limits.

3.0 DESIGN BASES

The design considers effects and changes in physical properties of fuel assembly components which result from burnup.

The integrity of the fuel rods is ensured by analyzing the fuel to show that excessive fuel temperatures, excessive internal rod gas pressures, and excessive cladding stresses and strains do not occur. This end is achieved by showing the fuel rods to satisfy the design bases for normal operation and anticipated operational occurrences over the fuel lifetime. For each design basis, the performance of the most limiting fuel rod shall not exceed the specified limits.

The functional capability of the fuel assembly is ensured by analyzing the fuel assembly to show that the fuel system dimensions and properties remain within operational tolerances. This is achieved by showing that the fuel assemblies satisfy the design bases for normal operation and anticipated operational occurrences over the fuel lifetime.

3.1 CLADDING PHYSICAL AND MECHANICAL PROPERTIES

Zircaloy-4 combines a low neutron absorption cross section, high corrosion resistance, and high strength and ductility at operating temperatures. Principal physical and mechanical properties including irradiation effects on Zircaloy-4 are provided in Section 5.

3.2 CLADDING STRESS LIMITS

The design basis for the fuel cladding stress limits is that the fuel system will not be damaged due to fuel cladding stresses exceeding material capability. Conservative limits are derived from the ASME Boiler and Pressure Code, Section III, Article III-2000 (Reference 3).

The cladding may also be damaged by the combination of volatile fission products and high cladding tensile stresses which may lead to stress corrosion cracking.[4,5] Stress corrosion cracking of fuel rod cladding is considered the principal failure mechanism for PCI failures encountered during changes in reactor operating conditions.[6,7,8] Even though unanimous agreement has not been reached on which chemical species enhances failure, the iodine atmosphere is usually considered the primary attacking media in irradiated fuel. If the stress level is low enough in the cladding, then stress corrosion cracking does not occur. Tests have been done under EPRI support[9,10,11] to evaluate a stress threshold associated with stress corrosion cracking in an iodine atmosphere. Typical data from those programs show that the time dependence of stress corrosion rupture involves two processes. At lower stresses, time to failure is largely controlled by a time-dependent crack nucleation process. Thus, if stress levels remain low enough, a flaw or crack that would subsequently propagate will not be nucleated.

The concept used to avoid failures from the stress corrosion cracking failure mechanism from power ramps is to keep the fuel rods from operating above the stress threshold associated with the nucleation of a propagating stress corrosion crack. The modelling of the stress corrosion crack propagation process and methods for predicting the stress levels in fuel rods operating under prototype exposure histories, incorporate many assumptions. The design procedure used to evaluate ENC fuel rods uses a stress threshold determined from benchmarking studies using the RODEX2[12] and RAMPEX codes. The design criterion for the transient stress limit, resulting from a power ramp, is to keep the predicted stress levels below the stress threshold obtained in the benchmarking studies of test ramp cases.

The benchmarking test results were obtained from the Studsvik Inter-Ramp, Over-Ramp and Super Ramp test series. Conservatism in the design bases is obtained by using a safety factor on the code benchmarked failure stress threshold, by using conservative input values for the fuel rod dimensions in the design analyses, and by assuming worst case power histories and ramp powers for the analysis.

3.3 CLADDING STRAIN LIMITS

Tests[14,15] on irradiated tubing indicate potential for failure at relatively low mean strains. The data on tensile, burst and split ring tests indicate a ductility ranging between 1.2% and 5% at normal reactor operating temperatures. The failures are usually associated with unstable or localized regions of high deformation after some uniform deformation. To prevent cladding failure due to plastic instability and localization of strain, the total mean hoop cladding strain for steady-state conditions is limited to 1%, and the increment of the thermal creep during a transient is also limited to 1%.

3.4 STRAIN FATIGUE

Cyclic PCI loading, combined with other cyclic loading associated with relatively large changes in power, can cause cumulative damage which may eventually lead to fatigue failure. Cyclic loading limits are established to prevent fuel failures due to this mechanism. The design life is based on correlations which give a safety factor of 2 on stress amplitude or a safety factor of 20 on the number of cycles, whichever is more conservative.(16)

3.5 FRETTING CORROSION AND WEAR

The design basis for fretting corrosion and wear is that fuel rod failures due to fretting shall not occur. Since significant amounts of fretting wear can eventually lead to fuel rod failure, the grid spacer assemblies are designed to prevent such wear. The spring dimple system in the spacer grid is designed such that the minimum spring/dimple forces throughout the design life are greater than the maximum fuel rod flow vibration forces. Testing of a wide variety of ENC fuel designs shows fuel rod wear is due primarily to fuel rod loading and unloading, and not due to fuel rod motion during the test. There has been little or no difference between observed wear for 500 hour, 1000 hour and 1500 hour tests. No active fretting corrosion has been observed despite spacer spring relaxation in several test assemblies. Examination of a large number of irradiated rods has substantiated the minimal wear observed after loop tests. Numerous PWR reload batches with this typical ENC bimetallic spacer have operated in sixteen reactors with no adverse effects due to fretting corrosion or wear unrelated to baffle jetting.

3.6 CORROSION

Cladding oxidation and corrosion product buildup are limited in order to prevent significant degradation of clad strength. A PWR clad external temperature limit is chosen, as corrosion rates are very slow below this temperature, and therefore, overall corrosion is limited. An external corrosion layer limit is also specified, as this amount of corrosion will not significantly affect thermal and mechanical design margins. This decreasse in clad thickness does not increase clad stresses above allowable levels.

Corrosion product buildup, and the resulting temperature increases, are calculated directly in the RODEX2 code.

3.7 HYDROGEN ABSORPTION

The as-fabricated cladding hydrogen level and the fuel rod cladding hydrogen level during life are limited to prevent adverse effects on the mechanical behavior of the cladding due to hydriding. Hydrogen can be absorbed on either the outside or the inside of the cladding. Excessive absorption of hydrogen can result in premature cladding failure due to reduced ductility and the formation of hydride platelets.

The effects of hydrogen on mechanical properties have been investigated at hydrogen concentrations to about 1000 ppm. The effect on strength and ductility depends on such factors as:

- o The tube texture which tends to promote or minimize radially orientated hydrides.

- o Stress and temperature cycling which may promote reorientation of hydrides into radial directions. Tensile hoop stress tends to orient hydrides radially.[20]
- o Distribution of hydrides (hydride case layers on the I.D. or O.D. surface tend to promote brittle failures).
- o Ratio of cladding wall thickness to average length of hydride platelet.
- o The fineness and uniformity in dispersion of the second phase precipitate tend to improve corrosion resistance and decrease hydrogen absorption.

The calculation of hydrogen concentration due to pickup from the coolant is calculated in the RODEX2 code. Hydrogen absorption from inside the clad is minimized by careful moisture control during fuel fabrication.

3.8 CREEP COLLAPSE

The design basis for creep collapse of the cladding is that significant axial gaps due to fuel densification shall not occur, and therefore, that fuel failure due to creep collapse shall not occur. Creep collapse of the cladding can increase nuclear peaking, inhibit heat transfer, and cause failure due to localized strain.

If significant gaps form in the pellet column due to fuel densification, the pressure differential between the inside and outside of the cladding can act to increase cladding ovality. Ovality increase by clad creep to the point of plastic instability would result in collapse of the cladding. During power changes, such collapse could result in fuel failure.

Through proper design, the formation of axial gaps and the probability of creep collapse can be significantly reduced. Typical ENC pellets are stable dimensionally. For high burnup designs, the lot average resinter density change is limited by specification. This specification ensures stable pellets during irradiation, and limits the potential size of fuel column gaps.

An Inconel X-750 plenum spring is included in the ENC fuel rod design, and the rods are pressurized with helium to help prevent the formation of gaps in the pellet column. The plenum spring provides a compressive force on the fuel column throughout the densification phase of the fuel life, and the internal pressure prevents rapid clad creepdown as well as providing a good heat transfer medium for the fuel.

An analysis is performed in order to guard against the unlikely event that sufficient densification occurs to allow pellet column gaps of sufficient size for clad flattening to occur. With this method, creep ovality is calculated with the COLAPX code and cladding uniform creepdown is calculated with the RODEX2 code[12] utilizing conservative design conditions.

3.9 FUEL ROD INTERNAL PRESSURE

The internal gas pressure of the fuel rods shall not exceed the external coolant pressure. Significant outward circumferential creep which may cause an increase in pellet-to-cladding gap must be prevented, since it would lead to higher fuel temperature and higher fission gas release. Fuel rod internal pressure is calculated throughout life with the RODEX2 code.

3.10 CREEP BOW

Differential expansion between the fuel rods and lateral thermal and flux gradients can lead to lateral creep bow of the rods in the span between spacer grids. The design basis for fuel rod bowing is that lateral displacement of the fuel rods shall not be of sufficient magnitude to impact thermal margins. ENC fuel has been designed to minimize creep bow. Extensive post-irradiation examinations have confirmed that such rod bow has not reduced spacing between adjacent rods by more than 50%. The potential effect on thermal margins is negligible.

3.11 OVERHEATING OF CLADDING

The design basis for fuel rod cladding overheating is that transition boiling shall be prevented. Prevention of potential fuel failure from overheating of the cladding is accomplished by minimizing the probability that boiling transition occurs on the peak fuel rods during normal operation and anticipated operational occurrences. Margin to boiling transition is evaluated using applicable DNB correlations, with ENC's XCOBRA-IIIC based PWR thermal-hydraulic methodology.

3.12 OVERHEATING OF FUEL PELLETS

Prevention of fuel failure from overheating of the fuel pellets is accomplished by assuring that the peak linear heat generation rate (LHGR) during normal operation and anticipated operational occurrences does not result in fuel centerline melting. The melting point of the fuel is adjusted for burnup in the centerline temperature analysis.

3.13 FUEL ROD AND ASSEMBLY GROWTH

The design basis for fuel rod and assembly growth is that adequate clearance shall be provided to prevent any interference which might lead to buckling or damage. Cladding and guide tube growth measurements of ENC fuel are used in establishing the growth correlations used for calculations.

TABLE 3.1

STEADY STATE STRESS DESIGN LIMIT

Stress Category*	Stress Intensity Limits**	
	Yield Strength	Ultimate Tensile Strength
General Primary Membrane	2/3	1/3
Primary Membrane Plus Primary Bending	1.0	1/2
Primary Plus Secondary	2.0	1.0

*Characteristics of the stress categories are defined as follows:

a) Primary stress is a stress developed by the imposed loading which is necessary to satisfy the laws of equilibrium between external and internal forces and moments. The basic characteristic of a primary stress is that it is not self-limiting. If a primary stress exceeds the yield strength of the material through the entire thickness, the prevention of failure is entirely dependent on the strain-hardening properties of the material.

b) Secondary stress is a stress developed by the self-constraint of a structure. It must satisfy an imposed strain pattern rather than being in equilibrium with an external load. The basic characteristic of a secondary stress is that it is self-limiting. Local yielding and minor distortions can satisfy the discontinuity conditions of thermal expansions which cause the stress to occur.

** The stress intensity is defined as twice the maximum shear stress and is equal to the largest algebraic difference between any two of the three principal stresses.

4.0 DESIGN DESCRIPTION

4.1 FUEL ASSEMBLY

The 14x14 fuel assembly array includes 16 guide tubes, 179 fuel rods and one instrumentation tube. The grid spacers are of standard ENC bi-metallic design, and the fuel assembly tie plates are stainless steel castings with Inconel holddown springs. Fuel assembly characteristics are summarized in Table 4.1.

4.2 FUEL ROD

The fuel rods consist of cylindrical UO_2 pellets in Zircaloy-4 tubular cladding.

The Zircaloy-4 fuel rod cladding is cold-worked and lightly stress relieved. Zircaloy-4 plug type end caps are seal welded to each end. The upper end cap has external features to allow remote underwater fuel rod handling. The lower end cap has a truncated cone exterior to aid fuel rod reinsertion into the fuel assembly during inspection and/or reconstitution.

Each fuel rod contains a 144.0 inch column of enriched UO_2 fuel pellets.

The fuel rod upper plenum contains an Inconel X-750 compression spring to prevent fuel column separation during fabrication and shipping, and during in-core operation.

Fuel rods are pressurized with helium which provides a good heat transfer medium and assists in the prevention of clad creep collapse.

TABLE 4.1

FUEL ASSEMBLY DESIGN

FUEL PELLET	
Fuel Material	UO_2 Sintered Pellets
Pellet Diameter, (in.)	0.3565
CLADDING	
Clad Material	Zircaloy-4 Cold Worked and Stress Relieved
Clad ID, (in.)	0.364
Clad OD, (in.)	0.424
Clad Thickness, Nominal, (in.)	0.030
FUEL ROD	
Diametral Gap, Cold Nominal, (in.)	0.0075
Active Length, (in.)	144.0
Total Rod Length, (in.)	152.065
Fill Gas	Helium
SPACER	
Material	Zr-4 & Inconel 718
Rod Pitch	0.556
Envelope (in.)	7.763 square
GUIDE TUBE	
Material	Zr-4
ID/ID Above Dashpot (in.)	0.541/0.507
TIE PLATES	
Material	Stainless Steel
HOLDDOWN SPRINGS	
Material	Inconel

TABLE 4.1 (Continued)

CAP SCREWS	
Materials	Inconel and SS
FUEL ASSEMBLY	
Array	14x14
Assembly Pitch	7.803
No. Spacers	7
No. Fuel Rods	179
No. Guide Tubes	16
No. Instrumentation Tubes	1

5.0 FUEL ASSEMBLY MATERIAL PROPERTIES

The material properties used in the design evaluation are described in this section. The Zircaloy cladding properties and the UO_2 fuel properties utilized are as incorporated in the RODEX2 and RAMPEX fuel performance codes.

5.1 ZIRCALOY-4

5.1.1 Chemical Properties

Zircaloy-4 is used in three forms: (1) Coldworked and stress relieved cladding; (2) Recrystallized annealed tubing; and (3) Recrystallilzed annealed strip. The chemical properties are in accordance with Grade R60804 (RA-2).

5.1.2 Physical Properties

The Zircaloy cladding properties are as incorporated in the RODEX2 code.

5.2 FISSILE MATERIAL (URANIUM DIOXIDE)

5.2.1 Chemical Composition

a) Uranium Content

The uranium content shall be a minimum of 87.7% by weight of the uranium dioxide on a dry weight basis.

b) Stoichiometry

The oxygen-to-uranium ratio of the sintered fuel pellets shall be within the limits of 1.99 and 2.01.

5.2.2 Thermal Properties

The thermal properties utilized are as incorporated in the RODEX2 code.

5.2.3 Mechanical Properties

a) Mechanistic Fuel Swelling Model

The irradiation environment and fissioning events cause the fuel material to alter its volume and, consequently, its dimensions.

The details of the model are described in Appendix K of the RODEX2 report.

b) Fission Gas Release

The evaluation of fission gas release is done by the RODEX2 code. For design evaluations of end-of-life pressures, pellet-cladding interaction and general thermal mechanical conditions, a physically based two-stage release model is used. First stage fission gas release is to grain boundaries, and then the second stage release is from the grain boundaries to the interconnected free gas volume. This release model is described in detail in Appendix E of the RODEX2 report.

c) Melting Point

The value used for the UO_2 melting point (unirradiated) is 2805°C (5081°F). Based on measurements by Christensen, et al[45], the melting point is reduced linearly with irradiation at the rate of 12.2°C (22.0°F) per 10^{22} fiss/cm^3 or 32°C (57.6°F) per 10^4 MWd/MTU.

5.3 INCONEL SPRINGS

Coil springs are fabricated from Inconel X-750 wire or rod with an alloy composition in accordance with Table 5.4 (AMS 5699B).

6.0 CONDITIONS FOR FUEL ROD MECHANICAL DESIGN

6.1 REACTOR OPERATING CONDITIONS

Core power level (Nominal)	1650 MWt
Coolant operating pressure (Nominal)	2250 psia
Coolant flow rate (at nominal power)	
Total	68.2×10^6 lb/hr.
Active Core	65.2×10^6 lb/hr.
Heat generation in fuel	97.4%
Coolant inlet temperature (Nominal)	534°F
Number of assemblies in core	121
Maximum peak pellet LHGR	14.47 kW/ft.
Maximum peak rod burnup	49 GWd/MTU

The fuel shall be capable of load-follow operation between 40% and 100% of rated power for at least two months per year, and not preclude the transients set forth in the FSAR.

6.2 ROD DIMENSIONAL DATA

Some of the cladding and fuel pellet characteristics for each reload are listed in Table 6.1. The characteristics of Reloads XN-6 through XN-9 are based on specifications, while values for Reloads XN-1 to XN-4 are based on as-built measured data. The values for Reload XN-5 are covered by the specifications of Reloads XN-6 through XN-9.

6.3 EXPOSURE HISTORY

Multiple power histories were developed by Wisconsin Public Service and ENC.

Four bounding histories which simulate various fuel shuffling schemes were selected. Tables 6.2 to 6.3 give the power histories and the corresponding fast fluxes. The power histories used were:

Case A: Three high power cycles and a low power fourth cycle.

Case B: Medium-High power during four cycles.

Case C: Four cycles, three high power with low power during the second cycle.

Case D: Four cycles, three high power with lower power during the third cycle.

All power histories are such that the rod average burnup is 49 GWd/MTU.

6.4 DESIGN CRITERIA

The mechanical design criteria are:

1. The maximum steady-state primary and secondary stresses shall meet the ASME Boiler and Pressure Vessel Code, Section III requirements,(3) as defined in Table 3.1.

2. The maximum cladding hoop stress at pellet ends during power ramping is limited to avoid failure by stress corrosion cracking.

3. The cumulative usage factor for cyclic stresses shall not exceed 0.67.

4. The net cladding mean hoop strain shall not increase by more than 1% for steady-state operation. The increment of the cladding hoop thermal creep strain at pellet ends during a ramp is also limited to 1%.

5. Cladding creep collapse shall not occur.

6. The hydrogen absorption of the cladding and the thickness of the corrosion layer shall not exceed design limits.

7. The internal pressure in the fuel rod at end-of-life shall not exceed the system pressure.

8. The fuel elongation must be accommodated by the clearance between fuel rods and tie plates.

9. The fuel assembly growth must be accommodated by the clearance between the fuel assembly and the core plates.

10. Fuel rod creep bow throughout the design life of the assemblies shall be limited so as to maintain licensing and operational limit restraints.

11. The fuel rod plenum spring shall maintain a positive compression on the fuel column during shipping and during the fuel densification stage.

12. Cladding temperatures shall not exceed the design limits.

13. Pellet temperatures shall not exceed the melting temperature during normal operation and anticipated transients.

TABLE 6.1

FUEL ROD ATTRIBUTES

Reload Cladding	XN-6 Through XN-9	Units
Clad ID Avg.	0.3640	inch
OD Avg.	0.4240	inch
Pellet Density Avg.	94.0	%TD
Nominal Enrichment	3.2-3.4	%U-235

TABLE 6.2

POWER HISTORIES A AND B

BOUNDING CASE A - HIGH, HIGH, HIGH, LOW POWER

Cycle Burnup	Irradiation Time Hrs.	Peak Assembly Burnup MWd/MTU	Peak Rod Burnup MWd/MTU	Peak Rod Average LHGR kW/ft	Rod Fast Flux (>1MeV) $10^{13} n/cm^2 sec$
0	0	0	0	9.839	8.86
11000	7340	15500	17050	9.839	9.44
0	0	15500	17050	8.252	8.84
11000	7340	28500	31350	8.252	9.05
0	0	28500	31350	7.646	9.70
11000	7340	41000	44600	7.646	9.60
0	0	41000	44600	2.539	3.00
11000	7340	45000	49000	2.539	3.00

BOUNDING CASE B - MEDIUM POWER & HIGH BURNUP

Cycle Burnup	Irradiation Time Hrs.	Assembly Burnup MWd/MTU	Peak Rod Burnup MWd/MTU	Rod Average LHGR kW/ft	Rod Fast Flux (>1MeV) $10^{13} n/cm^2 sec$
0	0	0	0	8.079	7.28
11000	7340	12775	14000	8.079	7.75
0	0	12775	14000	6.925	8.70
11000	7340	23700	26000	6.925	8.90
0	0	23700	26000	6.925	9.00
11000	7340	35000	38000	6.925	9.15
0	0	35000	38000	6.348	8.70
11000	7340	45000	49000	6.348	8.80

BOUNDING CASE C - HIGH, LOW, HIGH, HIGH POWER

Cycle Burnup	Irradiation Time Hrs.	Assembly Burnup MWd/MTU	Peak Rod Burnup MWd/MTU	Rod Average LHGR kW/ft	Rod Fast Flux (>1MeV) 10^{13}n/cm^{2}sec
0	0	0	0	9.839	8.86
11000	7340	15500	17050	9.839	9.44
0	0	15500	17050	2.539	2.90
11000	7340	19500	21450	2.539	3.00
0	0	19500	21450	8.252	9.60
11000	7340	32500	35750	8.252	9.80
0	0	32500	35750	7.646	9.80
11000	7340	45000	49000	7.646	9.80

BOUNDING CASE D - HIGH, HIGH, LOW, HIGH POWER

Cycle Burnup	Irradiation Time Hrs.	Assembly Burnup MWd/MTU	Peak Rod Burnup MWd/MTU	Rod Average LHGR kW/ft	Rod Fast Flux (>1MeV) 10^{13}n/cm^{2}sec
0	0	0	0	9.839	8.86
11000	7340	15500	17050	9.839	9.44
0	0	15500	17050	8.252	8.84
11000	7340	28500	31350	8.252	9.05
0	0	28500	31350	2.539	2.95
11000	7340	32500	35750	2.539	3.00
0	0	32500	35750	7.646	9.80
11000	7340	45000	49000	7.646	9.80

7.0 FUEL ROD MECHANICAL DESIGN ANALYSIS

7.1 STEADY-STATE STRESS ANALYSIS

The stresses were calculated at BOL hot and cold conditions, and BOC4 hot and cold conditions when the pellet/clad mechanical interaction is a maximum (History D). The calculations were performed using the long term cladding behavior from the RODEX2[12] calculations done for the collapse determinations, and using spacer-induced stresses calculated by an ANSYS[46] calculation. The collapse calculation cladding conditions were chosen because the minimum cladding thickness is used; and therefore, the stresses would be higher. The ANSYS analysis was a finite element stress analysis done for 0.424 OD cladding. The maximum stress intensities have been determined using the same technique as in the original design report.[1]

The results indicate that the calculated stresses are well below the design limits.

7.2 STEADY-STATE STRAIN

The cladding steady-state strain was evaluated with the RODEX2[12] code, latest version, as approved by the NRC in 1983. The code calculates the thermal, mechanical and compositional state of the fuel, and cladding for a given duty history. Conservative input values were used in the strain analysis. Bounding dimension values covering all reloads were selected for the calculations. On the basis of previous experience, the calculations were performed for power history "D".

The criterion of 1% maximum at EOL is satisfied.

7.3 TRANSIENT CLADDING STRESSES AND STRAINS

The stresses and strains during operating transients were evaluated with the RAMPEX code, on the basis of long term fuel conditions evaluated with the RODEX2 code.(12)

The benchmarking of the 1981 versions of these codes has determined a failure threshold stress for ENC cladding. This threshold applies to the cladding hoop stress calculated at pellet ends assuming pellet hourglassing and a chip filling part of the pellet-to-clad gap in cold condition as a result of fuel handling. Using that chipping assumption, the stresses are evaluated at the beginning of each cycle, in order to obtain the maximum stress.

The RAMPEX code applies only to one axial location of the rod so that a complete analysis requires many RAMPEX runs. The power ramps are from 0% power to the BOL F_Q limit (14.47 kW/ft) for the highest power assembly, or to a level that would be consistent with the highest power assembly reaching the F_Q limit. The ramp rates have been selected on the basis of current plant ramp rates.

These results are within the design criteria limits, and by using the most conservative dimensional inputs, as in the steady-state strain analysis, they cover all the reloads.

7.4 CYCLING FATIGUE

The cladding stresses calculated with RAMPEX at the beginning of each cycle were used to determine a stress amplitude during each type of transient. The frequency of occurrence of power changes was the same as the duty cycles presented in Reference 1. The stress amplitudes have been taken as such, since they already include local effects and the RAMPEX code predicts conservative stresses. Moreover, the calculated stress at the beginning of a cycle has been taken as such for the rest of the cycle without taking into account the fuel rod conditioning, which would relax the stresses.

The damage for each type of transient is determined by dividing the expected number of occurrences in each cycle by the allowed frequency for the corresponding stress amplitude and by accumulating the damage for each cycle. The allowed frequency is determined by conservative relations deduced from the fatigue curves of O'Donnel and Langer.(16) There is substantial margin compared to the design limit of 0.67.

7.5 COLLAPSE

The collapse calculation is done using the procedure described in the extended burnup report, and since approved by the NRC.(52) RODEX2 is run first, assuming nominal pellet dimensions, nominal gap, minimum wall cladding, minimum fill gas pressure, fill gas absorption, and no gas release to determine the temperature and pressure conditions throughout the fuel rod lifetime, and to determine the clad creepdown. These conditions are used as input for COLAPX. The COLAPX code is run with a conservative flux history. The code then predicts the time dependent creep ovality deformations in an infinite length tube subjected to external pressure, internal pressure, and linearly varying temperature gradients through the thickness of the cylinder.

If significant gaps are not allowed to form, then tube ovality, as predicted by the COLAPX evaluation, cannot occur beyond the point of fuel support. The ENC fuel rod design uses an Inconel X-750 plenum spring to maintain an axial load on the pellet column well beyond the time when pellet densification is complete. This assists in the prevention of axial gaps. The pellet maximum resinter densification criteria also assures the presence of stable fuel so that the formation of significant gaps is prevented, and so that clad support is available during the life of the fuel.

In order to guard against the highly unlikely event that enough densification occurs to form pellet column gaps of sufficient size to allow clad flattening, an evaluation was performed. The cladding ovality increase was calculated with COLAPX, and the creepdown was calculated with RODEX2. The combined creepdown at the cladding minor axis was determined not to exceed the minimum level to allow the fuel column to relocate axially without the formation of axial gaps.

7.6 CORROSION AND HYDROGEN PICKUP

RODEX2 includes the MATPRO corrosion and hydrogen pickup model.(12) This model considers temperature, exposure time, irradiation enhancement, and original oxide film thickness as parameters. The most oxide thickness formed in any of the power histories is less than allowed.

The maximum hydrogen level in either cladding for all of the power histories is well below the limit for hydrogen content.

7.7 FUEL ROD ELONGATION

The fuel rod elongation must be less than the clearance between the tie plates. This was analyzed using the worst case tolerances for the fuel assembly and fuel rod, and using the maximum rod average fast fluence in the four power histories. The rod elongation model for fuel rods is based on ENC measured data for PWR rods. A design limit, which very conservatively bounds the data, is used.

The fuel assembly grows also from the fast irradiation damage to the guide tubes. However, because the guide tubes are annealed, the growth is not as rapid as the fuel rods. The MATPRO irradiation growth model[42] was used to compute tie plate separation growth. With conservative assumptions, the relative rod/assembly maximum elongation provides a minimum EOL clearance.

7.8 INTERNAL PRESSURE

A RODEX2 analysis was performed to evaluate the end-of-life (EOL) internal fuel rod pressure for extended burnup. To prevent cladding instability, the rod internal pressure cannot exceed the system pressure or else the cladding may creep away from the pellet, which increases the fuel rod pellet temperatures. Higher fuel temperatures result in increased fission gas release, and therefore, higher internal rod pressures. The results of this analysis show the EOL internal rod pressure does not exceed the system pressure of 2250 psia. The fuel rod will, therefore, remain stable throughout the expected power history.

7.9 ROD BOWING

Fuel rod bow is determined throughout the life of the fuel assembly so that the reactor operating thermal limits can be established. These limits include the minimum critical heat flux ratio associated with protection against boiling transition and the maximum fuel rod LHGR associated with protection of metal-water reaction, and peak cladding temperature limits for a postulated loss-of-coolant accident (LOCA).

The effect of rod bow on boiling transition and LHGR limits was evaluated to a peak average rod burnup of 49000 MWd/MTU. This evaluation was performed in accordance with the approved methodology. The evaluation of the impact on the most limiting DNB FSAR transient showed that, including the effects of rod bow, the DNBR is not reduced below the DNB limit.

The evaluation of impact on limiting LHGR showed that the effects of rod bow are conservatively bounded by the ECCS analysis and nuclear uncertainty factors. Therefore, no penalty for either effect due to rod bow need be included in operating limits for fuel operating at the 2.28 F^T_q and 1.55 $F_{\Delta H}$ limits to a burnup of 49000 MWd/MTM.

7.10 FUEL ROD PLENUM SPRING

The major functional requirements on the plenum spring are during shipments and during the densification phase of the fuel. Since both of these situations occur relatively early in the life of the fuel, no reanalysis is required for extended burnup. However, as the design of the rod end cap has been changed since Reload XN-5, the design of the spring has been reevaluated.

The spring force is proportional to the fuel stack weight for shipping, the spring never becomes solid, and it compresses the fuel column during the densification phase. The results, applicable to all reloads, show that the plenum spring serves its intended purpose.

A new spring has been designed for Reload XN-7. That spring also satisfies all the design criteria.

7.11 FUEL AND CLADDING TEMPERATURE

As the peak LHGR is reached during the first cycles of the fuel life, no reanalysis should be required for the fuel and cladding temperature at extended burnup. The fuel and cladding temperature were, however, reevaluated with RODEX2, and the results indicate that the design limits are not reached.

8.0 FUEL ASSEMBLY EVALUATION

8.1 GENERAL DESCRIPTION

The fuel assemblies consist of a 14x14 array occupied by 179 fuel rods, 16 guide tubes, and one instrument tube. Seven Zircaloy-4 spacers with Inconel springs are positioned along the length of the assembly to locate the fuel rods and tubes, and are attached to the guide tubes by resistance spot welds. The guide tubes are mechanically attached to the upper and lower tie plates to form the structural skeleton of the fuel assembly.

8.2 DESIGN CRITERIA

The mechanical design criteria for the fuel assembly are to provide for:

- o Dimensional Compatibility
- o Differential Thermal Expansion and Irradiation Growth
- o Fuel Rod Support
- o Fuel Assembly Holddown
- o Upper Tie Plate Removability
- o Handling and Storage

Since the design of the fuel assembly structure is unchanged, only the irradiation growth and the fuel rod support behavior are affected by the extended burnup.

Specifically, the criteria require the design to provide adequate clearance between the tie plates to accommodate fuel rod growth, and adequate clearance between the fuel assembly and core plates to accommodate fuel assembly growth. The criteria for fuel rod support is to provide sufficient spring force at EOL to minimize flow-induced vibrations and to prevent fretting corrosion at the spacer-fuel rod contact points, considering the effects of irradiation-induced spring force relaxation.

8.3 DESIGN ANALYSIS

8.3.1 Fuel Assembly Growth

The fuel assembly growth must be accommodated by the core plate separation in cold conditions. This is more restrictive because of the differential thermal expansion between the core barrel and the fuel assembly. The MATPRO growth model[42] is used to calculate the fuel assembly growth using the worst case dimensions, and conservatively using the peak rod average fast fluence.

The MATPRO correlation is compared with ENC PWR fuel assembly growth data for four designs with identical guide tube characteristics. The designs vary in guide tube length, number of tubes per assembly, and holddown force. Growth results for the lowest stressed design are greater than predicted by the MATPRO model. Growth of the three more highly loaded designs is less than the MATPRO curves with less growth occurring as the stress is increased. The nominal hot BOL holddown stress in the Kewaunee fuel is comparable to the stress of other fuel assemblies, which were measured to have less growth than the MATPRO model predicted.

Using MATPRO and conservative assumptions listed above, the maximum fuel assembly growth, at a burnup of 49 GWd/MTU, gives a fuel assembly/core plate clearance.

8.3.2 Spacer Spring Relaxation

The Inconel spacer springs are known to relax during irradiation and the fuel rod cladding tends to creepdown. Together, these two characteristics combine to reduce the spacer spring force on a fuel rod during its lifetime. These characteristics have been considered in the design of the spring to assure an adequate holding force when the assembly has completed its design operating life.

Spacer spring relaxation and rod creepdown characteristics have been monitored in relation to burnup on both BWR and PWR fuel rods by measuring the force required to pull a fuel rod through a spacer. Data have been obtained on fuel rods of several reactor types, including ENC 15x15 rods for Westinghouse reactors, which have attained an assembly burnup of 47700 MWd/MTU. Inspection of the 15x15 rods showed no evidence of significant fretting or wear damage at the contact points.

The spacer spring relaxation, based on this and other data, follow an asymptotic relationship with burnup. For the rod and spacer spring type incorporated in Kewaunee, the average spring force at 47700 MWd/MTU is adequate to prevent fretting wear.

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KEWAUNEE HIGH BURNUP (49 GWd/MTU) FUEL ASSEMBLIES

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