ANF-87-139(NP)

ADVANCED NUCLEAR FUELS CORPORATION

EXTENDED BURNUP REPORT FOR FORT CALHOUN RELOADS XN-4 AND XN-5 (BATCHES K AND L)

JANUARY 1988

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ADVANCED NUCLEAR FUELS CORPORATION

ANF-87-139(NP), Rev. 0
Issue Date: 1/27/88

EXTENDED BURNUP REPORT

FOR FORT CALHOUN

RELOADS XN-4 AND XN-5

(BATCHES K AND L)

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September 15, 1987

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1.0 INTRODUCTION AND SUMMARY

1.1 Introduction

This report describes the mechanical design analysis performed for the extended burnup of the Advanced Nuclear Fuels Corporation (ANF) XN-4 and XN-5 (Batches K and L) reload fuel assemblies for Fort Calhoun. It includes a design description, summary of the design criteria, and the results of the extended burnup design analyses. The analyses were performed in accordance with the current ANF methodology consistent with the USNRC approved "Qualification of ANF Fuel for Extended Burnup".

1.2 Summary - Batch K

The analysis has shown that the fuel for Batch K will meet the design criteria for irradiation to extended power histories of 49,000 MWd/MTU rod burnup.

The key analysis results are:

- The maximum steady-state cladding strain is well below the 1.0% design limit.
- The maximum steady-state cladding stresses meet the ASME Boiler and Pressure Vessel Code requirements with ample margin.
- The transient strain is within the 1% circumferential limit.
- The transient stress calculated during ramps up to the maximum allowable peaking factor is within the design limits.
- Cladding creep collapse is precluded.

- The fuel rod internal pressure at the end of life remains below the reactor system pressure.
- The maximum metal loss due to oxide corrosion and the maximum concentration of hydrogen in the cladding are below the design limits.
- The cladding fatigue usage factor is well below the design limit.
- There is space between the upper and lower tie plate to accommodate fuel rod growth.
- The fuel assembly growth is within the space available between the upper and lower core plates in the reactor core.

1.3 Summary - Batch L

The analysis has shown that the fuel for Batch L will meet the design criteria for irradiation to extended power histories of 51,600 MWd/MTU rod burnup.

The key analysis results are:

- The maximum steady-state cladding strain is well below the 1.0% design limit.
- The maximum steady-state cladding stresses meet the ASME Boiler and Pressure Vessel Code requirements with ample margin.
- The transient strain is within the 1% circumferential limit.
- The transient stress calculated during ramps up to the maximum allowable peaking factor is within the design limits.
- Cladding creep collapse is precluded.
- The fuel rod internal pressure at the end of life remains below the reactor system pressure.
- The maximum metal loss due to oxide corrosion and the maximum concentration of hydrogen in the cladding are below the design limits.
- The cladding fatigue usage factor is well below the design limit.
- There is space between the upper and lower tie plate to accommodate fuel rod growth.

- The fuel assembly growth is within the space available between the upper and lower core plates in the reactor core.

2.0 DESIGN DESCRIPTION

The Fort Calhoun 14x14 fuel assemblies contain 176 fuel rods, 5 guide tubes, and 9 bi-metallic Zircaloy 4/Inconel 718 grid spacers. Some of the assemblies contain neutron absorber rods.

The fuel rods consist of short cylindrical dished UO_2 pellets in Zircaloy-4 cold worked and lightly stress relieved tubular cladding. Zircaloy-4 end caps are welded to each end to give a hermetic seal.

The fuel rod upper plenum contains a compression spring to prevent fuel column separation during fabrication and shipping, and during in-core operation. The rods are pressurized with helium to improve heat transfer and reduce clad creep ovality.

3.0 DESIGN CRITERIA

The detailed fuel rod design establishes such parameters as pellet diameter and length, density, cladding-pellet diametral gap, fission gas plenum size, and rod prepressurization level. The design also considers effects and physical properties of fuel rod components which vary with burnup.

The integrity of the fuel rods is ensured by designing to prevent excessive fuel temperatures, excessive internal rod gas pressures, and excessive cladding stresses and strains. This end is achieved by designing the fuel rods to satisfy the design criteria during normal operation and anticipated operational occurrences over the fuel lifetime. For each design criteria, the performance of the most limiting fuel rod shall not exceed the specified limits.

Fuel rods are designed to function throughout the design life of the fuel based upon the reactor operating conditions designated below without loss of mechanical integrity, significant dimensional distortion, or release of fuel or fission products.

3.1 Reactor Operating Conditions For Design

The extended burnup analysis is based upon the following reactor operating conditions.

Fort Calhoun

Nominal Core Thermal Power	=	1500 MW
Nominal Coolant Pressure	=	2100 psia
Core Effective Flow	=	68.5×10^6 lbs/hr
Core Coolant Inlet Temperature at Nominal Power	=	545°F
Average Linear Power for Nominal Conditions	=	6.22 kW/ft
Maximum Rod Power for Nominal Operating Conditions	=	11.20 kW/ft ($F_r = 1.80$)
Maximum Linear Power for Nominal Operating Conditions	=	15.22 kW/ft ($F^T_Q = 2.447$)

3.2 Mechanical Design Criteria

The assemblies were designed for an assembly burnup of 40,000 MWd/MTU and subsequently analyzed for an assembly burnup of 43,000 MWd/MTU. This extended burnup analysis will allow a peak rod burnup of 49,000 MWd/MTU for Batch K corresponding to an assembly burnup of approximately 44,550 MWd/MTU and 51,600 MWd/MTU for Batch L corresponding to an assembly burnup of approximately 46,900 MWd/MTU. The analysis is conducted using the methods approved by the NRC.

3.2.1 Steady-State Stresses

The cladding steady-state primary and secondary stresses shall meet the 1977 ASME Boiler and Pressure Vessel Code Section III[1] requirements summarized below:

	Stress Intensity Limits: Yield Strength	Stress Intensity Limits: Ultimate Tensile Strength
General Primary Membrane	$\leq 2/3\ \sigma y$	$\leq 1/3\ \sigma u$
Local Primary	$\leq 1.0\ \sigma y$	$\leq 0.5\ \sigma u$
Primary Membrane Plus Primary Bending	$\leq 1.0\ \sigma y$	$\leq 0.5\ \sigma u$
Primary Plus Secondary	$\leq 2.0\ \sigma y$	$\leq 1.0\ \sigma u$

Primary stresses are 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.

Secondary stresses are 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 due to thermal expansions which cause the stress to occur.

3.2.2 Steady-State Strain

Cladding circumferential plastic strain shall not exceed 1.0% through end-of-life.

3.2.3 Transient Stresses

Cladding stress levels during transient occurrences shall be limited to a fraction of the threshold failure stress due to stress corrosion cracking. The total uniform strain, elastic and thermal creep shall not exceed 1% during a transient. The stress analysis shall be performed with the RODEX2-RAMPEX codes benchmarked to available power ramp test data, i.e., INTERRAMP, OVERRAMP and SUPERRAMP.

3.2.4 Creep Collapse

The fuel rod shall be designed such that at a rod average burnup when substantial axial consolidation has occurred, the total clad creep deformation shall not exceed the initial minimum diametral fuel cladding gap. This will prevent pellet hangups allowing the plenum spring to close axial gaps until densification is substantially complete, thus preventing the formation of pellet column gaps of sufficient size for clad flattening.

3.2.5 Rod Internal Pressure

The fuel rod pressure at end-of-life shall not exceed system pressure. This will assure that tensile stresses are below the stress level which could result in hydride platelet reorientation and reduction in cladding strength. This criteria is met by the Batch K and Batch L fuel. A review of DNBR limits for Condition III or IV postulated accidents events is required for fuel rods that exceed nominal system pressure. Analysis approved by the NRC has shown that the fuel rod gas pressure can safely exceed system pressure without causing any damage to the cladding.

3.2.6 Corrosion

Total cladding wall thinning due to generalized external and internal corrosion shall not exceed a value which will impair mechanical performance over the projected fuel rod design lifetime under the most adverse projected power conditions within coolant chemistry limits recommendations of Table 3.1. It will also assure that the metal/oxide interface temperature will remain well below the level where large increases in corrosion, due to the insulating effect of the oxide, would adversely affect the mechanical behavior of the cladding.

3.2.7 Hydrogen Absorption

For the projected fuel rod design lifetime and operating conditions, the hydrogen content of the cladding on a material weight basis shall not exceed the design limit under the most adverse projected power conditions.

3.2.8 Cycling Fatigue

The cumulative usage factor for cyclic stresses for all important cyclic loading conditions shall not exceed the design limit.

3.2.9 Fuel Rod Growth

The clearance between the upper and lower tie plate shall be able to accommodate the maximum differential fuel rod and fuel assembly growth to the extended burnup.

3.2.10 Fuel Assembly Growth

The fuel assembly growth shall not exceed the minimum space between the upper and lower core plates in the reactor cold condition (70°F). The reactor cold condition is limiting since the expansion coefficient of the stainless steel core barrel is greater than the coefficient of expansions of the zircaloy guide tubes.

4.0 POWER HISTORIES AND DUTY CYCLES

4.1 Steady State

The power histories used in the extended burnup analysis are based on the known behavior for the fuel assemblies in Cycles 9 and 10. Each history is assumed to follow a rod throughout its residence in the reactor. The power histories selected for the analysis cover the following scenarios.

	Assembly Burnup MWd/MTU	Rod Burnup MWd/MTU
XN-4 High Power First Cycle High Burnup	40,900	45,000
XN-4 High Power Second Cycle Low First	42,900	47,200
XN-4 High Power Third Cycle	44,550	49,000
XN-4 High Power Third Cycle Medium First	44,550	49,000
XN-5 High Power First Cycle High Burnup	46,900	51,600
XN-5 High Power Second Cycle Low First	43,860	48,250

All the power histories but one are based on average cycle lengths of 13,000, 12,000, 13,600, and 13,450 MWd/MTU for Cycles 9, 10, 11, and 12, respectively. The XN-4 high third medium first uses the actual cycle lengths of 13,123 and 12,144 for Cycles 9 and 10, respectively, which lowered the power by 1% to maintain the correct burnup. During Cycles 9 and 10, the reactor operated at 99% full power. This allowed the power history to be analyzed at 99% power and increased the time by 1%.

The Batch L power histories did not have a pre-specified third cycle. The highest Batch K third cycle, namely from the high third medium first, was chosen to represent the Batch L third cycle. This allowed the Batch L high second low first to also be analyzed as the Batch L high third. The power in the Batch L third cycle was then raised to reach a maximum burnup without exceeding the design criteria.

4.2 Transients

The power histories are pseudo steady-state histories which are analyzed with the RODEX2 code for all the design criteria pertinent to steady-state operation and to establish the initial conditions for power ramps. In order to determine the stress and strain of the cladding during transients, ramps to maximum LHGR conditions are added to these histories at various times during the irradiation.

The fuel rod behavior during these ramps is analyzed with the RAMPEX code which takes the RODEX2 fuel rod steady state characteristics as input for the starting conditions.

Ramps are analyzed for conditions up to F^T_Q or up to a value pro-rated by the ratio of the projected rod average LHGR to the limiting F_r LHGR.

4.3 Cyclic Fatigue

In this evaluation the ramp at refueling startup is assumed to go up to the F^T_Q or its pro-rated value. The cyclic stresses for the balance of the duty cycles are taken from the ramp analysis to above steady-state power level. This gives approximately the same maximum power as attained in the startup ramp analyses. The low power level in the cyclic range is also adjusted by a reduction of the target lower level to give a conservative cyclic power range.

The design cycling conditions were taken from the design report for Reloads Batch J to Batch L, and are described in Table 4.1. Ramps were evaluated for each of these conditions at a ramp rate of 5% per minute which bounds the rates in Table 4.1.

The fatigue analysis is based on the O'Donnel and Langer design curve shown in Figure 5.1. This curve is conservative in that it is based on fatigue test data with a factor safety of 2 on stress or 20 on the number of cycles, whichever is more conservative.

Table 4.1 Duty Cycles for Fort Calhoun

A. Cold Shutdown (4/year)
   1. 100% - 0% @30%/hr
   2. 0% - 112% @30%/hr

B. Hot Standby (10/year)
   1. 100% - 0% @30%/hr
   2. 0% - 112% @30%/hr

C. Scram (9/year)
   1. 100% - 0% (instantly)
   2. 0% - 112% @30%/hr

D. Fuel Shuffling (1/cycle)
   1. 0% - 1.87 x 100% @30%/hr

E. Step Load Decrease
   1. 100% to 70% (10/year)
   2. 100% to 30% (2/year)
   3. 100% to 5% (1/year)

F. Xenon Oscillations (2/Cycle @100% Power)
   1. ASI variation of -0.2 to +0.2*
   2. Rod will be used to damp oscillations

G. Load Follow (60/year)
   1. 100% for 12 hours
   2. 100% to 40% @30%/hr
   3. Hold at 40% for 6 hrs
   4. 40% to 100% @30%/hr

* Changed from ±.3 to ±.2 to be consistent with current Technical Specification limits.

5.0 DESIGN ANALYSES

Each design analysis was performed with the current ANF methodology which involves a well defined selection of appropriate data and parameters, and the latest approved versions of computer codes. This methodology, as required, has been submitted to the USNRC and approved.

Details specific to each analysis are presented in the following sections.

5.1 Cladding Steady-State Stress

5.1.1 Method

The cladding steady-state analysis was performed using the computer code STRESS4 which calculates zircaloy cladding stress considering primary and secondary membrane and bending stresses due to hydrostatic pressure, flow-induced vibration, ovality, spacer contact, PCI, thermal and mechanical bow and thermal gradients. Stresses were calculated for the various combinations of the following conditions:

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

The analysis was performed for the various sources of stress including pressure, thermal, spacer contact, PCI, and rod bow. The applicable stresses at each orthogonal direction were combined to calculate the maximum stress intensities which are compared to the ASME design criteria. Design margins at EOL were evaluated with irradiated properties, i.e., with yield and ultimate strengths higher than at BOL.

5.1.2 Results

The results of the analysis indicate that all stress values are within acceptable design limits for both beginning and end of life, hot and cold conditions. The end of life stresses show ample margin for both the hot and cold condition stresses.

5.2 Steady-State Strain Analyses

5.2.1 Method

The cladding steady-state strain is evaluated with the RODEX2 code, which has been approved by the NRC[2]. The code considers the thermal-hydraulic environment at the cladding surface, the pressure inside the cladding, and the thermal, mechanical and compositional state of the fuel and cladding.

Pellet density, swelling, densification, and fission gas release or absorption models, and cladding and pellet diameters are input to RODEX2 to provide the most conservative strain calculation or subsequent ramping or collapse calculations for the reference fuel rod design. The major fuel rod performance characteristics modeled by the RODEX2 code are:

- Radial Thermal Conduction and Gap Conductance
- Fuel Swelling, Densification, Cracking and Crack Healing
- Gas Release and Absorption
- Cladding Creep Deformation and Irradiation-Induced Growth
- Cladding Corrosion
- Pellet-Cladding Interaction
- Free Rod Volume and Gas Pressure

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., giving the deformations of the fuel and cladding and gas release, are calculated at successive times during each period of assumed constant power generation. The calculated cladding strain is reviewed throughout the life of the fuel and both the maximum circumferential strain and the maximum strain increment are compared with the design criteria.

5.2.2 Results

Steady-state strain calculations were performed using the six power histories that included the maximum LHGR for the first, second, and third cycles. The calculated strain did not exceed the strain limit.

Both the maximum strain and the positive strain increment above the maximum creepdown are below 1.0% positive strain.

5.3 Ramping Stress-Strain, and Fatigue Evaluation

5.3.1 Method

This section describes the ramping stress and strain evaluation, and the fatigue evaluation of the fuel rod. The ramps are assumed to occur anytime during the irradiation and may reach the maximum peaking factor allowed by the limits of operation. The ramps are analyzed either from cold shutdown or from a variety of hot powered starting conditions.

The approach to rated power at the beginning of each reactor cycle is performed to satisfy the ANF maneuvering and conditioning recommendations.

The power histories were analyzed for ramp stress and strain with ramps from cold shutdown at the beginning of each reactor cycle according to the conditions defined above.

The ramp cases from conditions other than from the beginning of cycle cold startup, were simulated as startups from hot shutdowns. The major difference of these ramps, compared to the cold startup ramps, is the ramp rate which is a conservative 5% of nominal core power per minute. The ramps are, therefore, performed from hot shutdowns to the terminal power in 20 minutes.

The clad response during ramping power changes is calculated with the RAMPEX code. This code calculates the pellet-cladding interaction during a power ramp for one axial node at a time. The initial conditions are obtained from RODEX2 output. The RAMPEX code considers the thermal condition of the rod in its flow channel, and the mechanical interactions that result from fuel and cladding creep at any desired axial section in the rod during the power ramp.

The RAMPEX code provides the hoop stress and the stress intensity. The RAMPEX code also calculates the total uniform strain, elastic plus plastic, which can be compared with the USNRC criteria of 1% maximum strain during any ramp.

The stress results of the ramping analysis are used to evaluate the cladding fatigue damage through life due to the cyclic power variations defined in Table 4.1

The fatigue analysis is based on the O'Donnel and Langer[4] design curve shown in Figure 5.1. The cyclic amplitudes of the maximum local stress intensity, as determined by RAMPEX over the power cycling range, are compared with this curve to determine the allowed cycles for each stress range. This result is combined with the projected number of duty cycles to determine a fatigue usage factor.

5.3.2 Results

The ramp stress and strains that occur during reactor power changes were analyzed using each of the six power histories.

All of the reactor cycle (startup) ramps were within the design limit.

The maximum total uniform strain ("elastic + plastic") was below the design limit of 1%.

The RAMPEX runs for the transient stress and strain analysis also provide the stress intensities needed for the fatigue analysis. The stress intensities at the appropriate power levels are taken from the RAMPEX results to determine the cyclic amplitudes for each type of duty cycle. Stress intensity amplitudes are determined for ramps throughout the life of the fuel. The total fatigue usage factor is determined by accumulating the usage for each cycle. All of the values are well below the design limit.

5.4 Creep Collapse

5.4.1 Method

Creep collapse calculations are performed with .ODEX2 and COLAPX codes. The RODEX2 code determines the cladding temperature and internal pressure history based on a model which accounts for changes in fuel rod volumes, fuel densification and swelling, and fill gas absorption. The reactor coolant, fuel rod internal temperature, and pressure histories generated by the RODEX2 analysis are input to the COLAPX code along with initial cladding ovality and the fast flux history. The COLAPX code calculates, by large deflection theory, the ovality of the cladding as a function of time while the uniform cladding creep down is obtained from the RODEX2 analysis.

The cladding ovality increase and creep down are summed, at a rod average burnup when substantial axial consolidation has occurred, to show that they remain less than the initial minimum pellet clad gap. Measurements of highly densifying irradiated fuel have demonstrated that pellet densification is essentially complete by the time the fuel has attained this burnup so that further creepdown after this phase will not result in significant pellet to pellet gaps.

5.4.2 Results

The combined radial creep down was shown to meet the design criteria. This will prevent pellet hangups due to cladding creep, allowing the plenum spring to close axial gaps until densification is substantially complete, and thus assures that clad collapse will not occur.

The pitch of the plenum spring is less than the spacing calculated for stiffening rings in a cylindrical shell under external pressure which will prevent clad collapse in the plenum area.

5.5 Fuel Rod Gas Pressure

5.5.1 Method

Calculation of the gas pressure within a fuel rod is performed with the RODEX2 code. The initial fill gas is found by calculating the initial free volumes and using the ideal gas law, along with input values for fill gas pressure and reference fill gas temperature. The free gaseous fission product yield is calculated for each axial region and the total yield obtained by summing the axial region contributions.

The six power histories are presented in Section 4.0. The Batch L high second high third power history was analyzed as both the high second and as the high third cycle histories. The powers of each history were multiplied for each cycle by a factor required for the highest projected rod power to reach the F_r limit.

The power histories used in the calculations were also modified to include a period of top or bottom peaking. The axial profiles thus created for the alternate top peak and bottom peak situations are intended to conservatively produce the effect of xenon transients in the gas release calculations. The axial shape index (ASI) for these profiles equals or conservatively exceeds the current Technical Specification limit of ±.20 allowed by the DNB/LCO[5].

5.5.2 Results

The calculations show that the rod internal gas pressure will remain below system pressure (2100 psi) for all histories. This is below the criteria approved by the NRC for use in extended burnup gas pressure analysis.

5.6 Cladding Corrosion And Hydrogen Absorption

5.6.1 Method

The waterside corrosion of fuel rods is evaluated with the MATPRO-11[3] correlation. The MATPRO-11 model is a two-stage corrosion rate model which is cubic in dependence on oxide thickness until a transition to a subsequent linear dependence occurs.

To calculate the rate changes as a function of both oxide thickness and the operating conditions of the fuel rod, the MATPRO model is incorporated into ANF's RODEX2 fuel performance code. The RODEX2 code determines the temperature increase of the water along the fuel rod assuming heat balance within a channel for the prescribed mass flow and inlet temperature. The radial temperature drops are evaluated successively between the water, the oxide surface, the metal/oxide interface, and the inside of the cladding using RODEX2 correlations and methods. To account for the change in corrosion rate due to the changing oxide layer and thermal conditions, the code includes an update in cladding temperature at every calculation step. This is an iterative process due to the continuously changing oxide thickness. Conditions are also revised at times where new power or flow conditions are prescribed.

5.6.2 Results

The waterside corrosion and the hydrogen pickup in the cladding were evaluated with RODEX2 for the six power histories used in the steady-state strain analysis. A conservative corrosion amplification factor was applied to the MATPRO model to bound the measured data on ANF cladding.

The maximum calculated metal loss due to waterside corrosion and the hydrogen pickup in the cladding were below the design limits.

5.7 Fuel Rod And Assembly Growth

5.7.1 Method

Fuel rod growth projected to occur during irradiation was based on measurements made on photographs showing the gap between the fuel rods and the upper and lower tie plates on ANF 14x14 assembly type fuel. The rod growth minus the assembly growth plus tolerances was compared with the clearance within the assembly for fuel rod growth.

The maximum assembly growth projection was based on a design curve for ANF fuel in other PWR reactors. The design curve envelopes all the data obtained from reactors with fuel assembly designs similar to Fort Calhoun.

5.7.2 Results

There is a space between the upper and lower tie plates to accommodate the maximum differential growth out to the maximum rod burnup for Batch K and for Batch L.

There is adequate space between the upper core plate and the assembly to accommodate fuel assembly growth.

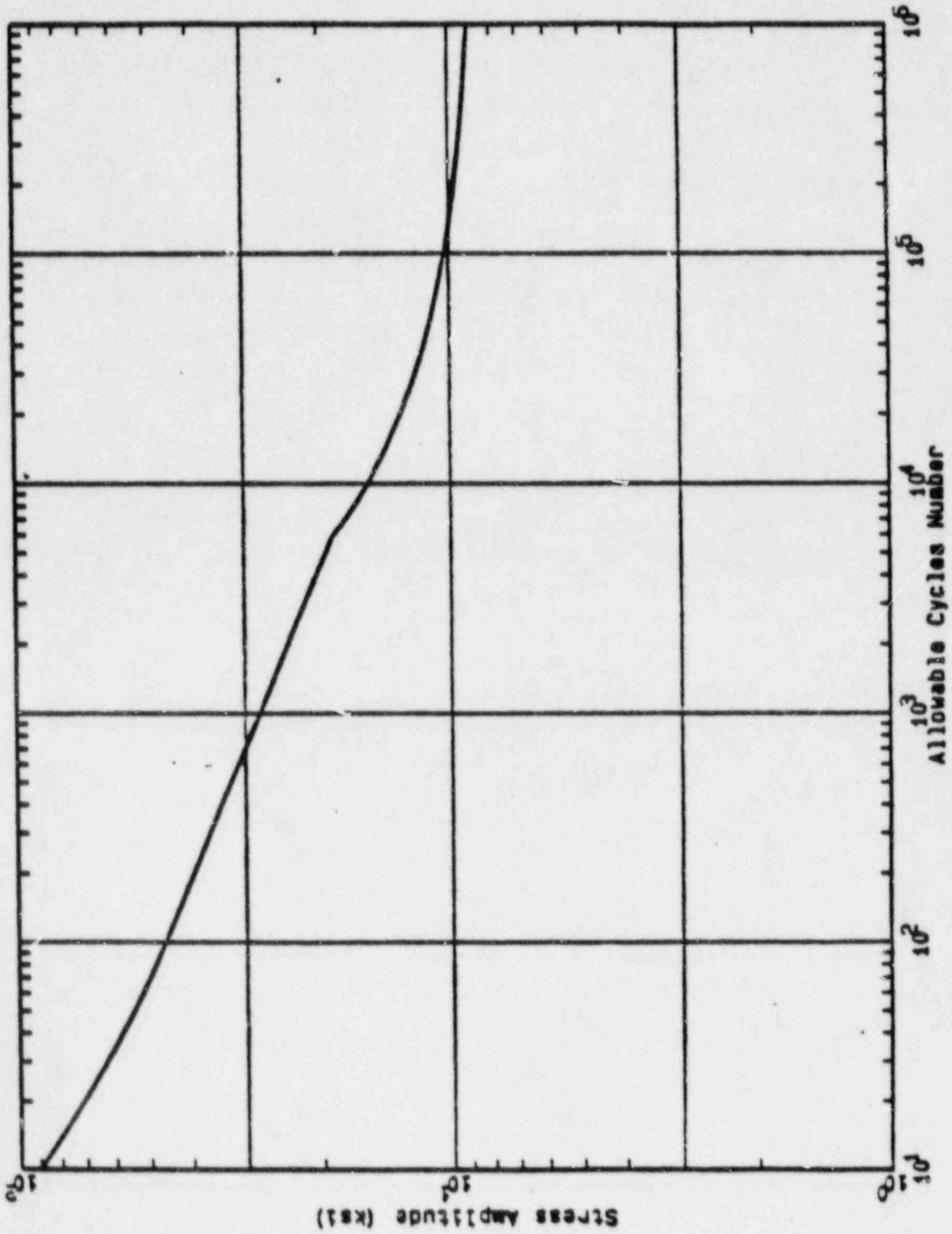

Figure 5.1 Fatigue Design Curve for Irradiated Zircaloy

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ANF-87-139(NP), Rev. 0
Issue Date: 1/27/88

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RELOADS XN-4 AND XN-5

(BATCH K AND L)

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