XN-NF-82-25(NP)(A)

GENERIC MECHANICAL DESIGN REPORT
EXXON 17x17 FUEL ASSEMBLY

JULY 1984

RICHLAND, WA 99352

EXXON NUCLEAR COMPANY, INC.

XN-NF-82-25 (NP)(A)

Issue Date: 7/17/84

GENERIC MECHANICAL DESIGN REPORT

EXXON 17 x17 FUEL ASSEMBLY

This is the NRC approved version of Document XN-NF-82-25(P), and has been prepared in accordance with NRC guidance. (This is a Non Proprietary verson.)

EXXON NUCLEAR COMPANY, Inc.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

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R. B. STOUT

Dr. Richard B. Stout, Manager
Licensing and Safety Engineering
Exxon Nuclear Company, Inc.
2101 Horn Rapids Road
P. O. Box 130
Richland, Washington 99352

Dear Dr. Stout:

Subject: Acceptance for Referencing of Licensing Topical Report
XN-NF-82-25(P)

The Nuclear Regulatory Commission (NRC) has completed its review of the Exxon Nuclear Company, Inc. (ENC) Licensing Topical Report XN-NF-82-25(P) entitled "Generic Mechanical Design Report Exxon 17x17 Fuel Assembly" dated April 1982 and the related response to NRC's request number 1 for additional information transmitted by letter R. B. Stout (ENC) to Dr. C. O. Thomas (NRC) dated November 24, 1982. This licensing topical report provides a generic summary of the design criteria, technical bases, supporting analysis and test results for the Exxon 17x17 reload fuel for Westinghouse reactors. A copy of our safety evaluation is enclosed.

Based on our review of the licensing topical report and the response to our request for additional information, we conclude there is reasonable assurance that the ENC 17x17 PWR reload fuel will perform acceptably under normal and postulated accident conditions.

As a result of our review, we conclude that the Exxon Nuclear Company, Inc. licensing topical report number XN-NF-82-25(P) entitled "Generic Mechanical Design Report Exxon 17x17 Fuel Assembly" dated April 1982 as augmented by the ENC response to NRC's request for additional information is acceptable for referencing in reload licensing applications to the extent specified and under the limitations stipulated in the licensing topical report and the enclosed evaluation. Because part of the ENC 17x17 PWR fuel design analysis, as described in XN-NF-82-25, was performed with the RODEX2 thermal analysis code, which is currently under review, any applicant desiring to use this type of fuel must confirm or redo the following analyses:

a. Design Strain (SER Section 4.2.3.1(b)).
b. External Corrosion (SER Section 4.2.3.1(e)).
c. Rod Pressure (SER Section 4.2.3.1(h)).
d. Overheating of Fuel Pellets (SER Section 4.2.3.2(d)).
e. Pellet Cladding Interaction (SER Section 4.2.3.2(e)).

With regard to thermal hydraulic design analysis, we have found the DNBR design criterion and the plant-specific thermal margin evaluation method acceptable. However, the correlation is still under review and will be addressed in an appropriate SER scheduled to be issued in early 1983. When this report is referenced, the reference must include both the proprietary and non-proprietary versions.

We do not intend to repeat our review of this topical report when it appears as a reference in a particular license application, except to assure that the material presented is applicable to the specific plant involved. Our acceptance applies only to the features described in the topical report and the response to our request for additional information.

In accordance with established procedures (NUREG-0390), it is requested that Exxon Nuclear Company, Inc., publish approved proprietary and non-proprietary versions of the topical report within three months of receipt of this letter. The accepted versions must include this letter and the enclosed evaluation following the title page and must appropriately incorporate the information in the initial paragraph above.

Should Nuclear Regulatory Commission criteria or regulations change, such that our conclusions as to the acceptability of the report are invalidated, Exxon Nuclear Company Inc., and/or the applicants referencing the topical report will be expected to revise and resubmit their respective documentation or submit justification for the continued effective applicability of the topical report without revision of their respective documentation.

Sincerely,

Cecil O. Thomas

Cecil O. Thomas, Chief
Standardization & Special
Projects Branch
Division of Licensing

Enclosure:
Evaluation

NON PROPRIETARY VERSION

EVALUATION

OF

LICENSING TOPICAL REPORT

XN-NF-82-25(P)

GENERIC MECHANICAL DESIGN REPORT EXXON 17x17 FUEL ASSEMBLY

APRIL 1982

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* The somewhat unorthodox numbering system for this SER is intended to facilitate comparison with the NRC Standard Review Plan.

1. Introduction

The Exxon Nuclear Company (ENC) 17x17 fuel assemblies are intended for use as reload assemblies in Westinghouse pressurized water reactors (PWRs). The 17x17 bundle array contains 264 fuel rods, 24 guide tubes, and 1 instrument tube and is similar to the 14x14 array (TOPROD) design (Ref. 1) except for an increased number of guide tubes (from 16 to 24) and grid spacers (from 7 to 8), which are meant to ensure adequate strength and stiffness.

The stated purpose of XN-NF-82-25 is to provide a design description and summary of the design criteria, technical bases, analyses and test results related to the design of ENC 17x17 reload fuel. The document is divided into eight major sections, as follows:

1. Introduction and Summary
2. Fuel System Design Objectives
3. Design Bases
4. Design Description
5. Design Evaluation
6. Thermal Hydraulic Design
7. Testing and Inspection Plan
8. References and Appendices

The topical report thus roughly parallels the format of the NRC Standard Review Plan (SRP) for the Fuel System Design (Ref.2) with respect to the mechanical design discussion, but the report structure is not identical to that part of the SRP. To facilitate comparison with the Standard Review Plan, therefore, most of our SER sections will be numbered like the SRP.

To render a stand-alone generic document for the ENC 17x17 reload fuel design, missing information was later supplied during the course of our review. That information will be incorporated into the approved revised report along with our safety evaluation.

4.2 Fuel System Design

The objectives of this fuel system safety review as described in Section 4.2 of the Standard Review Plan are to provide assurance that (a) the fuel system is not damaged as a result of normal operation and anticipated operational occurrences, (b) fuel system damage is never so severe as to prevent control rod insertion when it is required, (c) the number of fuel rod failures is not underestimated for postulated accidents, and (d) coolability is always maintained. "Not damaged" is defined as meaning that fuel rods do not fail, that fuel system dimensions remain within operational tolerances, and that functional capabilities are not reduced below those assumed in the safety analysis. This objective implements General Design Criterion 10 (Ref. 4), and the design limits that accomplish this are called Specified Acceptable Fuel Design Limits (SAFDLs). "Fuel rod failure" means that the fuel rod leaks and that the first fission product barrier (the cladding) has, therefore, been breached. Fuel rod failures must be accounted for in the dose analysis required by 10 CFR Part 100 (Ref. 5) for postulated accidents. "Coolability," which is sometimes termed "coolable geometry," means, in general, that the fuel assembly retains its rod-bundle geometrical configuration with adequate coolant channeling to permit removal or residual heat even after a severe accident. The general requirements to maintain control rod insertability and core coolability appear repeatedly in the General Design Criteria (e.g., GDC 27 (Ref. 6) and 35 (Ref. 7)). Specific coolability requirements for the loss-of-coolant accidents are given in 10 CFR Part 50.46 (Ref. 8).

To meet the above stated objectives of the fuel system review, the following specific areas are critically examined: (a) design bases, (b) description and design drawings, (c) design evaluation, and (d) testing, inspection, and surveillance plans. In assessing the adequacy of the design, several items involving operating experience, prototype testing, and analytical predictions are weighed in terms of specific acceptance criteria for fuel system damage, fuel rod failure, and fuel coolability. Exxon's fuel system design objectives, as presented in Section 2.0 of XN-NF-82-25, include the four review objectives presented above and, in addition, include two additional objectives that are of special interest to reload fuel; viz., that (a) the fuel assemblies are designed to withstand loads as a result of in-plant handling and shipping, and (b) the mechanical and hydraulic design of fuel assemblies will be compatible with coresident fuel and the reactor core internals to achieve acceptable flow distribution including bypass flow such that heat transfer requirements are met for all licensed modes of operation. These latter two design objectives are consistent with not only the review objectives of SRP Section 4.2 but also with the requirements of the "Standard Format" (Ref. 9) and SRP Section 4.4, Thermal and Hydraulic Design, respectively.

4.2.1 Design Bases

Design bases for the safety analysis address fuel system damage mechanisms and suggest limiting values for important parameters such that damage will be limited to acceptable levels. For convenience, we group acceptance criteria for these design limits into three categories in the Standard Review Plan: (a) fuel system damage criteria, which are most applicable to normal operation, including anticipated operational occurrences (AOOs), (b) fuel rod failure criteria, which apply to normal operation, AOOs, and accidents, and (c) fuel coolability criteria, which apply to accidents.

4.2.1.1 Fuel System Damage Criteria

In the following paragraphs we review the design bases and corresponding design limits for the damage mechanisms listed in the Standard Review Plan. These design limits along with certain criteria that define failure (see Section 4.2.1.2 of this SER) constitute the Specified Acceptable Fuel Design Limits (SAFDLs) required by General Design Criterion 10.

(a) Cladding Design Stress

The design basis for fuel rod cladding stress, as provided in XN-NF-82-25, is that the fuel system will not be damaged due to fuel cladding stresses exceeding material capability. The cladding steady-state primary and secondary stresses (provided in Table 3.1 of XN-NF-82-25) meet the 1977 ASME Boiler and Pressure Vessel Code, Section III (Ref. 11) requirements; for instance, the design limit for unirradiated general primary membrane stress is 2/3 yield strength or 1/3 ultimate strength. As indicated in SRP subsection 4.2.II.A.1, stress limits that are obtained by methods similar to those in Section III of the ASME Code are acceptable. The 2/3 yield strength and 1/3 ultimate strength tensile primary membrane stress limits are consistent with the ASME code and are traditional limits consistent with previous ENC design practice. These limits are, therefore, acceptable.

(b) Cladding Design Strain

The design basis for fuel rod steady-state cladding strain, as the basis is provided in Section 3.1.3 of the topical report, is to prevent cladding failure due to plastic instability or localization of strain. To satisfy that design basis, the total mean circumferential cladding strain for steady-state conditions is limited to 1% at end-of-life (EOL).

For transient conditions, and at fast fluences above a specified value, ENC proposed to use a reduced stress (not strain) limit to reduce the probability of stress-corrosion cracking (SCC)-induced pellet/cladding interaction (PCI). The stress limit is based upon a correlation with Studsvik ramp data that is reported to indicate that cladding failures will not occur below a particular stress value as calculated using ENC fuel performance codes. In effect, ENC is proposing a new PCI failure criterion based upon cladding stress. Inasmuch as the NRC is reviewing PCI generically and has at this time only one PCI-related strain criterion of limited application, viz., 1% cladding strain, we cannot comment on the proposed new ENC PCI failure criterion except to state that we understand that it corresponds to a calculated transient strain that is well below the 1% limit specified in the Standard Review Plan. On that basis, therefore, the Exxon SCC-type PCI stress criterion may be used while the issue of PCI receives continued generic study and other PCI criteria and models are considered.

(c) <u>Strain Fatigue</u>

The strain fatigue criteria provided in topical report Section 3.1.5 are the same as those described in SRP Section 4.2, viz., a safety factor of and are, therefore, acceptable.

(d) <u>Fretting Wear</u>

Although the Standard Review Plan does not provide numerical bounding value acceptance criteria for fretting wear, it does stipulate that the allowable fretting wear should be stated in in the safety analysis and that the stress and fatigue limits should presume the existence of this wear. Exxon's design basis for fretting corrosion and wear is that fuel rod failures due to fretting shall not occur. While Exxon does not use a specific numerical value for a fretting wear limit in the fuel rod stress and fatigue analysis, it is clear from the discussion in Section 3.1.6 of XN-NF-82-25 that the grid spacers are designed to prevent significant fretting wear. Therefore, since fretting wear is addressed in the design analysis, we conclude that the design method is acceptable.

(e) <u>External Corrosion and Crud Buildup</u>

Exxon's design basis for cladding oxidation and crud buildup is to prevent significant degradation of cladding strength and unacceptable temperature increases due to corrosion product buildup. With these considerations, Exxon specifies a maximum cladding external temperature to limit overall corrosion, while an external corrosion layer thickness is specified on the grounds that the degree of corrosion specified will not significantly affect design margins (i.e., increase cladding stresses above allowable levels). The Standard Review Plan does not provide numerical limits for cladding temperature or degree of oxidation for normal operation. However, Exxon's proposed limits appear conservative, and we thus conclude that they are acceptable and meet the intent of the SRP.

With respect to hydriding, the design basis stated in Section 5.1.2 of the topical report is that the as-fabricated and end-of-life cladding hydrogen levels are limited to prevent adverse effects on the mechanical behavior of the cladding due to hydriding. Exxon has established a hydrogen limit for the cladding to assure that the design basis is satisfied. Based on referenced data and operating experience , the hydrogen design limit is acceptable.

(f) Rod Bowing

Fuel rod bowing is a phenomemon that can alter the pitch between adjacent fuel rods and affect local nuclear power peaking and heat transfer. The ENC design basis for fuel rod bowing, expressed in Section 3.1.11 of XN-NF-82-25, is that lateral displacement of the fuel rods shall not be of sufficient magnitude to impact nuclear or thermal margins. ENC does not place design limits on the amount of bowing that is permitted, and the Standard Review Plan does not require set values. It is sufficient that ENC addresses the effects of bowing in the nuclear and thermal analysis.

(g) Axial Growth

Axial entension of the fuel rods results from both irradiation growth and pellet/cladding interaction. Excessive axial extension of fuel rods is a concern because it can interfere with the tie plates and result in excessive rod bowing or other damage. Moreover, axial extension of guide tubes could result in solid contact with the reactor core plates and possibly cause fuel assembly bowing.

The ENC design basis for 17x17 PWR fuel is that an assembly must have sufficient axial clearance between the tie plates and the fuel rods to preclude contact throughout the design life. ENC has established a beginning-of-life (BOL) cold clearance requirement, as a fraction of fuel column height, as a design limit to account for axial growth. The design basis and limit meet the guidelines of paragraph (e) of SRP Section 4.2.II.A.1 and are, therefore, acceptable.

(h) Fuel Rod Pressures

Section 4.2 of the SRP identifies excessive fuel rod internal pressure as a potential fuel system damage mechanism. In this sense, damage is defined as an increased potential for elevated temperatures within the rod as well as an increased potential for cladding failures. Because traditional analytical methods for fuel performance analysis do not adequately treat the effects of net outward stress on the cladding and because these effects (e.g., unstable high fuel temperatures and ballooning during DNB events) might be important, the Standard Review Plan calls for rod pressures to remain below nominal system pressure during normal operation unless otherwise justified. As indicated in Section 3.1.1.0 of XN-NF-82-25, the ENC 17x17 fuel rods are designed such that the internal gas pressure of the fuel rods does not exceed the coolant pressure, so the Standard Review Plan acceptance criterion is satisfied.

(i) Assembly Liftoff

It is specified in SRP Section 4.2.IIA.1(g) that worst-case hydraulic loads for normal operation, which includes anticipated operational occurrences, should not exceed the fuel assembly's holddown capability. The design basis for ENC 17x17 fuel assembly holddown, as provided in subsection 3.4.4 of XN-NF-82-25, is that the springs, when compressed by the upper core plate during reactor operation, will provide a net positive downward force during steady-state operation, based on the most adverse combination of component dimensional and material property tolerances. It is evident that the stated design basis is consistent with the Standard Review Plan and is, therefore, acceptable.

4.2.1.2 Fuel Rod Failure Criteria

The NRC staff's evaluation of fuel rod failure thresholds for the failure mechanisms listed in the SRP is presented in the following paragraphs. When these failure thresholds are applied to normal or transient operations, they are used as limits (and hence SAFDLs), since fuel failures under those conditions should not occur (according to the traditional conservative interpretation of GDC 10). When these thresholds are applied to accident analyses, they are used to determine the number of fuel failures for input to the radiological dose calculations required by 10 CFR 100. The basis or reason for establishing these failure thresholds is thus predetermined, and only the threshold values are reviewed below.

(a) Internal Hydriding

Hydriding as a cladding failure mechanism is precluded by controlling the level of moisture and other hydrogenous impurities during fabrication. As stated in the Standard Review Plan, the moisture level for Zircaloy-clad uranium oxide fuel should not exceed 20 ppm. The current industry standard (Ref. 12) for UO_2 fuel pellets, provided in terms of an equivalant hydrogen content, is 2 ppm (i.e., 2 mgH/gU). Exxon's fabrication limit for total hydrogen in the fuel pellets is less than the industry standard and SRP acceptance criterion and is, thus, acceptable. As noted in XN-NF-82-25, sufficent samples are taken to assure that this design limit is met with a probability of 95% at a confidence level of 95%.

In addition to the limit on fuel pellet moisture (hydrogen) content, Exxon utilizes a design limit for cladding hydrogen level. As noted in Section 4.2.1.1e of the SER, we find that limit acceptable, based on referenced data and operating experience.

(b) Cladding Collapse

If axial gaps in the fuel pellet column were to occur due to densification, the cladding would have the potential of collapsing into the gaps (i.e., flattening). Because of the large local strains associated with such collapse, the cladding is assumed to fail. As indicated in XN-NF-82-25, Exxon treats creep collapse as tantamount to failure. This approach is in agreement with the Standard Review plan and is, thus, acceptable.

(c) Overheating of Cladding

As stated in SRP Section 4.2.II.A.2, it has been traditional practice to assume that failures will not occur if the thermal margin criterion is satisfied. The design basis for Exxon 17x17 fuel rod cladding overheating, as provided in Section 3.1.12 of XN-NF-85-25, is that transition boiling shall be prevented. In Section 5.1 of the report, it is specified that avoidance of boiling transition for the limiting fuel rod in the core is at a 95% confidence level with at least a 95% probability. A minimum departure from nucleate boiling ratio (MDNBR) of using critical heat flux correlation, is said to satisfy the 95/95 statistical criterion. The cladding overheating design basis and limit are consistent with the thermal margin acceptance criterion of SRP Section 4.2 and are thus acceptable from the standpoint of fuel mechanical design. The review of thermal/hydraulics design methods (e.g., the critical heat flux correlation) is outside the scope of the fuel system design evaluation and is not addressed here. (See Section 4.4.)

(d) Overheating of Fuel Pellets

For radiological dose calculational purposes, it has been regulatory practice to assume that fuel rod failure will occur if fuel pellet centerline melting takes place. This conservative assumption provides assurance that axial or radial relocation of molten fuel will not occur and that contact of molten fuel with the cladding will thus be precluded. As a design basis, therefore, Exxon has established that the fuel centerline temperature should be below the melting point of the pellets during normal operation and anticipated operational transients.

The design limit corresponding to the above design basis is that the peak linear heat generation rate (LHGR) during normal operation and anticipated transients will not result in calculated centerline melting, taking into consideration burnup effects on the melting point of the fuel. The design limit is an acceptable representation of the design basis.

(e) Pellet/Cladding Interaction

Fuel rod failures due to pellet/cladding interaction tend to occur as the fuel pellets expand and exert stresses on the cladding during power increases. Although the exact mechanisms that contribute to PCI damage have not been established beyond doubt, operating experience indicates that irradiated Zircaloy does not always accommodate such stresses well, particularly when the Zircaloy has been exposed to certain embrittling (stress-corrosion) fission product species such as iodine or cadmium.

Although generally applicable regulatory criteria for PCI failure have not been established, two acceptance criteria of limited application are presented in SRP Section 4.2.II.A.2 for PCI: (a) 1% transient-induced cladding strain, and (b) no centerline melting. Since ENC utilizes the no centerline melting as a design basis for precluding fuel pellet overheating (see SER Sec-tion 4.2.1.2 (d)), the no melting PCI acceptance criterion is automatically satisfied. (See Section 4.2.1.1.b for a discussion of PCI-induced strain).

(f) Cladding Rupture (Bursting)

Zircaloy cladding will burst (rupture) under certain combinations of temperature, heating rate, and differential pressure. While there are no specific design limits associated with cladding rupture, the requirements of Appendix K to 10 CFR 50 (Ref. 13) must be met as those requirements relate to the incidence of rupture during a LOCA. The ECCS correlation used by Exxon is an approved model , and the objectives of paragraph (h) of SRP Section 4.2.II.A.2 are, thus, satisfied.

(g) Mechanical Fracturing

The term "mechanical fracture" refers to a fuel rod defect that is caused by an externally applied force, such as a hydraulic load or a load derived from core-plate motion. The Exxon design basis for PWR 17x17 fuel assembly mechanical fracturing is that the assemblies must withstand the external loads due to all events (earthquakes and postulated pipe breaks are the most limiting) without fracture of the cladding. The design limit applied by ENC is that the stresses due to postulated accidents in combination with the normal steady-state fuel rod stresses shall not exceed the normal cladding design stress limits as described in Section 4.2.1.1(a) of this SER. This is a conservative approach and is thus acceptable.

4.2.1.3 Fuel Coolability Criteria

For major accidents in which severe fuel damage might occur, core coolability must be maintained as required by several General Design Criteria (e.g. GDCs 27 and 35). In the following paragraphs we review limits that will assure that coolability is maintained for the severe damage mechanisms listed in Section 4.2 of the Standard Review Plan.

(a) Fragmentation of Embrittled Cladding

To meet the requirements of 10 CFR 50.46 (Ref. 8) as it relates to cladding embrittlement for a LOCA, acceptance criteria of 2200°F on peak cladding temperature and 17% on maximum local cladding oxidation must be met. As indicated in Exxon employs these criteria.

(b) Violent Expulsion of Fuel

In severe reactivity-initiated accidents such as a PWR control rod ejection, the large and rapid deposition of energy in the fuel can result in melting, fragmentation, and dispersal of fuel. The mechanical action associated with fuel dispersal can be sufficient to destroy the cladding and rod bundle geometry of the fuel and to produce pressure pulses in the primary system. To meet the guidelines of Regulatory Guide 1.77 (Ref. 15) as it relates to preventing widespread fragmentation and dispersal of the fuel and avoiding the generation of pressure pulses in the primary system of a PWR, a radially averaged enthalpy limit of 280 cal/g should be observed. As indicated in , ENC employs the 280 cal/g criterion.

(c) Cladding Burst Strain and Flow Blockage

To meet the requirements of Appendix K of 10 CFR 50 (Ref. 1) as it relates to swelling, the burst strain and flow blockage that result from cladding ballooning (swelling) must be taken into account in the analysis of cladding oxidation and peak cladding temperature. Burst strain and flow blockage models must be based on applicable data in such a way that the resultant degree of cladding swelling is not underestimated. There are no specific design limits associated with ballooning. The correlations used by Exxon are described in .

(d) Structural Damage from External Forces

Earthquakes and postulated pipe breaks in the reactor coolant system would result in external forces on the fuel assembly. The ENC 17x17 fuel design basis, provided in Section 3.4.2 of XN-NF-85-25, for earthquakes and postulated pipe breaks, is that the fuel assembly shall maintain coolable (rod-like) geometry and control rod insertability during the occurrence of a design basis seismic/LOCA event. This basis is consistent with the objective stated in the Standard Review Plan and is, therefore, acceptable.

4.2.2 Description and Design Drawings

The ENC 17x17 PWR fuel assembly design is described in Section 4.0 of XN-NF-82-25. Additional information is provided in . Some design features that differ from previous ENC PWR designs for Westinghouse reactors include the array (17x17 vs 14x14 and 15x15), number of grid spacers (8 vs 7), increased number of guide tubes (24 vs 16), and smaller diameter rods on a smaller pitch. Enough information is provided in sufficient detail in the XN-NF-82-25 report and supplemental references to provide a reasonably accurate and acceptable representation of the design.

4.2.3 Design Evaluation

Section 4.2.1 of this safety evaluation was used to present design bases and limits. In this section, we discuss Exxon's methods of demonstrating that the 17x17 fuel design meets the design acceptance criteria that have been established. This section will, therefore, parallel Section 4.2.1 of this safety evaluation report point-by-point. Methods of demonstrating that the acceptance criteria have been met include operating experience, prototype testing, and analytical predictions.

4.2.3.1 Fuel System Damage Evaluation

(a) Cladding Design Stress

As indicated in Section 3.4.1 of XN-NF-82-25, the steady-state primary membrane stresses (produced by the coolant pressure and fuel rod internal gas pressure) for the ENC 17x17 fuel rods are calculated by the equation recommended by . Primary bending stresses are calculated with an equation developed by . The cladding thermal stresses are calculated using standard equations described by and . Other stresses, such as those caused by mechanical bow between spacers and flow-induced vibration stresses are also considered and calculated using conventional models described in the open literature . Contact stresses at spacer spring locations are calculated using a commercially available general purpose finite element code . Inasmuch as standard analytical models were used and no steady-state stress limits were exceeded, we conclude that the design criteria for the ENC 17x17 fuel rod cladding stresses are satisfied for steady-state (normal operation) conditions. For transients, ramping stresses are discussed in the section dealing with PCI.

(b) Cladding Design Strain

For cladding steady-state strain calculations, Exxon uses the code , which is an interative calculational procedure that 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. Calculations were performed for what is believed to be the worst expected fuel rod power and fast flux history to determine cladding strain. With the minimum design pellet-to-cladding gap and the maximum fuel density, the calculated maximum end-of-life (EOL) steady-state strain was within the design limit of 1.0%. Based on the calculations, therefore, the ENC 17x17 fuel assembly strain design limit appears to be met.

The code is the latest of a series of thermal analysis and mechanical response codes develcped by Exxon. It is intended to replace , which has been available for important licensing calculations since 1978 and which was used to provide input to , which is an unreviewed precursor to . Because the review has not been completed, the NRC staff will require that licensees using the ENC 17x17 fuel confirm or redo the strain analysis using an approved model. is in an advanced stage of review , with completion of the review anticipated in the next few months.

(c) Strain Fatigue

In addition to the transient strain analyses discussed in SER Section 4.2.3.2e, a fatigue usage factor for the cladding was calculated. The calculations were based upon assumed duty cycles (summarized in Table 5.5 of XN-NF-82-25). Cladding stress amplitudes for the various power cycles were determined using an unreviewed code called , which calculates the pellet/cladding interaction during a power ramp. The power ramp rate was assumed to follow ENC's preconditioning recommendations. An assumed total strain concentration factor was applied to account for possible stress concentration in the cladding. The allowable number of cycles, determined from a fatigue design curve (based on a safety factor of that takes into account the maximum mean stress, indicated that the total usage factor was less than ENC's design acceptance criterion for the maximum cumulative usage factor. Although neither or are approved models or procedures, we do not believe their review is warranted at this time, and in light of the favorable results reported, we conclude that the ENC 17x17 fuel design criterion for cladding strain fatigue has been satisfied and that the fatigue analysis is acceptable.

(d) Fretting Wear

As indicated in Section 3.1.6 of XN-NF-82-25, a wide variety of ENC designs have been tested for fretting wear. Wear depths are reported to be typically less than , with the wear due primarily to fuel rod loading and reloading rather than fuel rod motion during the test. No correlation has been observed between wear and test time, and examination of a large number of irradiated rods has reportedly not revealed wear significantly different from that observed in the prototype tests described . We conclude, therefore, that the ENC 17x17 fuel rods will perform adequately with respect to fretting wear.

Fretting wear has also sometimes been observed on the inner surfaces of guide thimble tubes where the fully withdrawn control rods reside. Significant wear is limited to the relatively soft Zircaloy-4 guide thimble tubes because the Inconel or stainless steel control rod claddings are relatively wear resistant. The extent of the wear is both time-dependent and plant-dependent and has in some non-Westinghouse cases extended completely through the guide thimble tube wall. To a first approximation, however, the propensity for guide thimble tube wear in Exxon reload fuel should be equivalent to Westinghouse fuel in the same plant. Examinations on Exxon fuel that was discharged from H. B. Robinson Unit 2 in fact revealed no through-wall wear or major differences in the wear from that which was measured (Ref. 33) on Westinghouse fuel that had been discharged from Point Beach Units 1 and 2. Of the 100 guide thimble tubes examined by eddy current testing, only 11 had detectable wear. Therefore, as discussed in Ref. 34, we conclude that (a) the degree of guide tube wear measured by Exxon is acceptable, (b) the degree of wear in the Exxon fuel is similar to that in Westinghouse fuel, and (c) the issue of guide thimble wear in Exxon-fueled Westinghouse-NSSS plants has been adequately resolved.

(e) Oxidation, Hydriding, and Crud Buildup

The buildup of a corrosion film on the outer surface of a fuel rod during irradiation impedes heat transfer and results in higher temperatures throughout the fuel rod. In the ENC fuel rod thermal analysis, this corrosion film is comprised of two distinct components: (1) an inner component consisting of a zirconium oxide (ZrO_2) film, which is relatively thin and adherent, and (2) an outer component consisting of hydrated oxides and hydroxides of the structural materials in the primary coolant system. The effects of the ZrO_2 film thermal resistance are included in the by calculation of film conductivity as inversely proportional to the oxide thickness, which is also calculated by , and proportional to the oxide film's conductivity. Using the code in this manner, the maximum oxide layer thickness, resultant cladding temperature increase, and maximum cladding external temperature were well below the limits specified in Section 5.1.1 of XN-NF-82-25. Because the review has not been completed, the NRC staff will require that licensees using the 17x17 fuel confirm or redo this analysis using an approved model.

With regard to crud, which builds a film on the surface of the fuel and cladding, ENC considers the crud to be so loose, fluffy, and hydrated that little thermal resistance results and, therefore, the effects of crud are ignored . While we believe that the effects of crud on fuel rod overheating may be negligible early in life, we would expect the propensity for crud buildup to increase with service time in reactor. We will, therefore, consider this issue as part of our ongoing generic study of the effects of extended burnup on ENC fuel designs as reported in Exxon's topical report on extended burnup.

With regard to hydrogen absorption, ENC considers (a) the initial concentration of hydrogen in the as-fabricated cladding, (b) the concentration of hydrogen in the cladding due to internal sources such as the fuel, and (c) the concentration of hydrogen in the cladding due to external sources such as the coolant in determining the net weight of hydrogen in the cladding (in ppm). The primary consideration in determining the cladding hydrogen concentration is judged by Exxon to be the contribution from external sources. That contribution is treated as a function of the oxide film thickness on the external surface of the cladding (see report Section 5.2.5).

The net weight fraction of hydrogen in the cladding is predicted to be about a third of the design limit for the 17x17 fuel design. There is reasonable assurance, therefore, that hydriding of the fuel rod cladding will not be a problem with the ENC 17x17 fuel.

(f) Rod Bowing

ENC has a data base of several thousand rod-to-rod and rod-to-guide-tube spacing measurements on irradiated ENC PWR fuel from 3 PWRs and a somewhat smaller data base on BWR 7x7 and 8x8 fuel rods. ENC has used these measurements to establish an empirical model for predicting rod-to-rod gap closure as a function of burnup . The model, which is used to calculate thermal limits, has recently been reviewed and approved (Ref. 36b). We conclude, therefore, that Exxon's rod-to-rod gap closure model in acceptable but a plant-specific analysis must be performed to determine an appropriate DNBR penalty.

(g) Axial Growth

The BOL cold clearance requirement that Exxon uses to assume adequate axial clearance between tie plates and fuel rods (see discussion in SER Section 4.2.1.1.g) is based on a correlation and on growth measurements on irradiated ENC fuel rods. Exxon also asserts that, in the case of guide tubes, the metallurgical condition of the ENC Zircoloy-4 minimizes the irradiation growth.

While calculations based on alone would not provide sufficient assurance of the adequacy of the axial growth predictions for ENC 17x17 fuel (because, as acknowledged by Exxon, axial growth would be expected to be related to variable tubing parameters such as texture), the existence of good ENC measurements on irradiated tubing of similar metallurgical texture and characteristics supports the conclusion that the clearance requirement will be met). Therefore, we find the ENC analysis of 17x17 fuel assembly growth to be acceptable.

(h) Fuel Rod Pressure

To calculate fuel rod internal pressure for the 17x17 fuel design, ENC used the with an ENC-developed model for fission gas release. The calculated EOL internal pressure, reported in Section 5.10 of XN-NF-82-25, is psi, which is well below reactor system pressure. Because the review has not been completed, however, the NRC staff will require that licensees using ENC 17x17 fuel confirm or redo the rod pressure analysis with an approved code.

(i) Assembly Liftoff

In response to a staff question on assembly liftoff, ENC stated that a 20 percent overspeed transient might produce a 44 percent increase in hydraulic loading of the fuel assembly and could result in a temporary liftoff of some fuel assemblies. According to ENC analyses, the maximum liftoff height would be inches, which is a small fraction of the spacer and tie plate heights. Because the total deflection and load are within the elastic range of the spring system, a positive holddown force would obtain upon return to nominal flow and hydraulic load. We conclude, therefore, that fuel assembly liftoff has been adquately addressed in the ENC 17x17 fuel design.

4.2.3.2 Fuel Rod Failure Evaluation

(a) Internal Hydriding

As indicated in Section 3.1.8 of XN-NF 82-25, Exxon uses hydrogen control limits in the manufacture of reactor fuel. And, as indicated in Section 5.2.8 of that report, the EOC cladding hydrogen level is predicted to be about a third of the design limit, which, in turn, is based on data that showed that the combined effects of hydriding and irridation to not appear to be significant in the range of hydrogen concentrations approaching the ENC limit . We, therefore, conclude that reasonable assurance has been provided that hydriding as a fuel failure mechanism will not be significant in the ENC 17x17 PWR fuel.

(b) <u>Cladding Collapse</u>

The previously approved ENC cladding collapse procedure utilizes the , code to determine the cladding ovality in unsupported regions of the fuel pellet column to establish that the accumulation of creep ovality does not result in cladding collapse. Using that procedure, the calculated instantaneous cladding collapse pressure is usually shown to remain greater than the differential pressure between the reactor coolant system and the fuel rod.

The procedure does not, however, evaluate the likelihood or extent of pellet separation, and in the new ENC creep collapse calculational procedure Exxon contends that practical means of limiting pellet separation have been established, that cladding flattening cannot occur in the absence of large gaps, and that the conservatism of considering an infinite length of unsupported cladding may, therefore, be removed. The major means used to limit pellet separation consists of a plenum spring, which is placed above the fuel pellet column in each fuel rod; a primary purpose of the spring is to provide a positive compressive force on the fuel column throughout the densification phase of the fuel life.

In ENC's revised creep collapse calculational procedure, creep ovality is analyzed as usual with the , but uniform cladding creepdown is obtained using , and the two values are combined to provide the total fuel pellet-to-cladding gap closure as a function of burnup. If the combined creepdown does not exceed the initial minimal by the time the fuel achieves a given burnup, Exxon assumes that pellet hangups due to cladding creepdown will not occur (because densification of the UO_2 pellets will be essentially completed and the plenum springs will have closed any axial gaps).

The revised cladding collapse calculational procedure is described in an Exxon topical report on extended burnup , which is under review as part of a generic study. Since neither the review nor the extended burnup reviews have been completed, we will require each licensee using the ENC 17x17 fuel to provide, prior to the second cycle of operation, an analysis using approved methods that shows that creep collapse will not occur to the target burnup.

(c) Overheating of Cladding

As indicated in SRP Section 4.2.II.A.2, adequate cooling is assumed to exist when the thermal margin to limit the departure from nucleate boiling (DNB) in the core is satisfied. The analysis of margins to boiling transition, i.e., minimum departure from nucleate boiling ratio (MDNBR), is performed on a plant-specific basis and is, therefore, not discussed here. See Section 4.4 of this SER.

(d) Overheating of Fuel Pellets

According to information presented in Section 6.3 of XN-NF-82-25, the peak design linear heat generation rates (LHGR) calculated with the for ENC 17x17 fuel are expected to be about less than the values for other ENC PWR reload fuel and about one-third less than the steady-state LHGR required for centerline melting. The analyses assume the coincidence of maximum power peaking and the worst engineering tolerances that would maximize the resistance to heat transfer from the fuel rod to the coolant . The peak power was calculated to be well below the power level required for UO_2 centerline melting under control rod withdrawal or misoperation conditions (see SRP Sections 15.4.1, 15.4.2, and 15.4.3). The effect of gadolinia additions on UO_2 melting is described in Reference 41 (but ENC does not use gadolinia poison in the 17x17 fuel assembly design at this time). We conclude that the centerline melt criterion has been satisfied for the ENC 17x17 fuel design.

(e) Pellet/Cladding Interaction

There are two PCI criteria in current use in licensing of PWR fuel, viz., (1) no (centerline) UO_2 melting and (2) 1-percent cladding plastic strain. Fuel melting is addressed in SER subsection 4.2.3.2d, and as indicated, the no-fuel-melting criterion is not violated.

With regard to cladding transient-induced strain, it is indicated in Section 5.1.3 of XN-NF-82-25 that ENC utilizes a strain limit up to a given, relatively low, fast fluence level, and that above that fluence level, the strain limit is replaced with a stress limit. This design approach is a significant departure from that used for the ENC TOPROD design , and it does not satisfy the Standard Review Plan acceptance criterion for PCI transient strain.

In response to a staff question (Q490.19), however, Exxon stated that the SRP 1% transient strain limit would not be exceeded. It was pointed out that the proposed stress limit for higher fluences

Studsvik ramp data) corresponded to a strain level that was well below the 1% strain acceptance criterion. These stresses and strains are calculated with the with input from . Although these codes may not precisely calculate the actual stresses and strains in the cladding, they do provide ENC with an engineering assessment of the likelihood for PCI failure based upon comparison with available failure data. Because the review has not been completed, however, the NRC staff will require licensees using ENC 17x17 fuel to confirm the statement that 1% strain will not be exceeded using an approved code.

It is notable that Exxon considers stress corrosion cracking (SCC) to be the principal PCI failure mechanism "encountered during changes in reactor operation conditions" and addresses cladding design features (such as texture, thickness, and internal surface roughness) and pellet design features (such as L/D ratio, density, disk volume, and shoulder configuration) as factors that can affect PCI resistance. While we believe such features should help to lessen susceptability to PCI failure, we do not believe that there is sufficient evidence available to conclude that SCC is the predominant PCI failure mechanism or that other PCI mechanisms may not play a prominent role, especially during short-term transients. PCI will, therefore, continue to receive generic study.

(f) Cladding Rupture

Although the ENC cladding rupture temperature model described in XN-NF-82-25 was approved as an integral part of the ENC ECCS model), the NRC staff has concluded (Ref. 42) that the model is nonconservative over some regions of applicability. Because the requirements of Appendix K to 10 CFR 50 (Ref. 13) must be met as those requirements relate to the incidence of rupture during a LOCA, ENC elected to replace its previous approved ECCS cladding swelling and rupture model with that proposed in NUREG-0630 (Ref. 42) for fuel rod temperatures below 950°C. Above 950°C, ENC modified the NUREG-0630 model based upon additional data obtained after NUREG-0630 was issued. The NRC staff has recently completed review of the new ENC swelling and rupture model , and the model has been approved with modifications (Ref. 44). We conclude that cladding swelling and rupture has been adequately addressed for ENC 17x17 fuel.

(g) Mechanical Fracturing

The analysis for mechanical fracturing is usually done as part of the structural damage analysis. See Section 4.2.3.3(d) of this SER.

4.2.3.3 Fuel Coolability Evaluation

In the following paragraphs is discussed the staff's evaluation of the ability of Exxon's 17x17 fuel to meet the coolability criteria in Section 4.2.1.3.

(a) Fragmentation of Embrittled Cladding

The primary degrading effect of a significant degree of cladding oxidation is embrittlement. Such embrittled cladding will have a reduced ductility and resistance to fragmentation. The most severe manifestation of such embrittlement occurs during a LOCA. The overall effects of cladding embrittlement on the ENC 17x17 design for the loss-of-coolant accident are analyzed in   and are not reviewed here.

One of the most significant analytical methods that is used to provide input to the LOCA analysis is the steady-state fuel performance code, which is reviewed under Section 4.2 of the SRP. This code provides fuel pellet temperatures (stored energy) and fuel rod gas inventories for the ECCS evaluation model as prescribed by Appendix K (Ref. 13) to 10 CFR 50. The code accounts for fuel thermal conductivity, fuel densification, gap conductance, fuel swelling, cladding creep, and other phenomena that affect the initial stored energy. A licensee using the ENC 17x17 fuel must confirm that an NRC-approved fuel performance code was used to provide input for the plant ECCS analysis.

(b) Violent Expulsion of Fuel Material

Exxon has generically evaluated the rod ejection accident with the procedures described in the ENC Generic Rod Ejection Analysis Report . Using conservative assumptions, the pellet energy deposition for an ejected rod has been evaluated for standard ENC fuel for a typical PWR cycle) and was found to be well below the 280 cal/g limit hot full power at BOL). While is still under review and thus has not yet been approved for safety analyses related to licensing applications, the review has progressed to a point where it appears that no significant problems will be identified. Therefore, inasmuch as the maximum energy deposition was less than the Regulatory Guide (Ref. 15) value of 280 cal/g for coolability for a typical PWR standard fuel reload analysis performed with the ENC generic rod ejection model, we conclude that there is reasonable assurance that control rod ejection should not be a problem with the ENC 17x17 fuel design.

(c) Cladding Ballooning Strain and Flow Blockage

Although Exxon's cladding ballooning and assembly flow blockage models have been approved as integral parts of the ENC ECCS evaluation model, we concluded (Ref. 42) that both models were nonconservative over some regions of applicability. Consequently, Exxon modified and resubmitted its ballooning and blockage model based upon additional data. The staff has recently completed review of the new ENC ballooning and blockage model , and the model has been approved (Ref. 44) with some modifications. We conclude that cladding ballooning and flow blockage has been adequately addressed for ENC 17x17 fuel.

(d) Structural Damage from External Forces

Generic methods for performing this analysis are presented in and were approved by the NRC (Ref. 49). These methods are capable of analyzing cores of a mixed design such as would exist when a partial core of ENC 17x17 fuel is introduced. Since this fuel assembly analysis depends on plant-specific input motions, this analysis was not completed in a generic manner. Therefore, a licensee proposing to use the 17x17 fuel design must address the requirements of NUREG-0609 (or Appendix A of SRP Section 4.2 as appropriate) to show that the proposed cores containing the ENC 17x17 fuel will satisfy the acceptance criteria.

4.2.4 Testing, Inspection, and Surveillance Plans

4.2.4.1 Testing and Inspection of New Fuel

As required by SRP Section 4.2, testing and inspection plans for new fuel should include verification of significant fuel design parameters. While details of the manufacturer's testing and inspection programs should be documented in quality control reports, the programs for onsite inspection of new fuel and control assemblies after they have been delivered to the plant should also be described in the SAR.

Discussion of the Exxon quality control program is provided in Ref. 50 and addresses fuel systems component parts, pellets, rod inspection, assemblies, process control, etc. Fuel system component inspection depends on the component parts and includes dimensions, visual appearance, audits of test reports, material certification, and nondestructive examinations. Pellet inspections, for example, are performed for dimensional characteristics such as diameter, density, length, and squareness of ends. Fuel rod, control rod, burnable poison rod, and source rod inspections reportedly consist of nondestructive examination techniques such as leak testing, weld inspection, and dimensional measurements. Process control procedures are described in detail. In addition, Exxon stated in , that for any tests and inspections performed by others on behalf of Exxon, Exxon reviews the quality control procedures, inspection plan, etc., to ensure that they are equivalent to the description provided in Reference 50 and are performed properly to meet all Exxon requirements.

We conclude, based on the information provided in References and the commitment by Exxon to ensure the acceptablity of any tests and inspections performed by others on behalf of Exxon, that the new-fuel testing and inspection program for the ENC 17x17 fuel design is acceptable.

4.2.4.2 On-Line Fuel System Monitoring

Routine on-line fuel rod failure monitoring is a matter that would be arranged with the licensee. It is not addressed in the ENC topical report.

4.2.4.3 Postirradiation Surveillance

Routine poolside inspection of some discharged fuel assemblies is a matter that is normally arranged with the licensee. Guidance for the type of surveillance to be conducted is provided in SRP Section 4.2.II.D.3.

4.2.5 Mechanical Design Findings

Although we conclude that the ENC 17x17 fuel mechanical design is generally acceptable, the licensee proposing to use this fuel must make arrangements to provide the following:

1. Rod bowing penalties (see Paragraph 4.2.3.1(f)).

2. Cladding Collapse Analysis (see Paragraph 4.2.3.2(b)).

3. An analysis for mechanical fracturing (see Paragraph 4.2.3.2(g)) and structural damage from external forces (see Paragraph 4.2.3.3(d)).

4. Confirmation of the following analyses, which were reviewed on the basis of results.

   (a) Design Strain, SER Section 4.2.3.1(b).
   (b) External Corrosion, SER Section 4.2.3.1(e).
   (c) Rod Pressure, SER Section 4.2.3.1(h).
   (d) Overheating of Fuel Pellets, SER Section 4.2.3.2(d).
   (e) Pellet Cladding Interaction, SER Section 4.2.3.2(e).

5. Confirmation that an NRC-approved fuel performance code was used to provide input for the plant ECCS analysis.

With the above provisoes, we conclude that ENC 17x17 fuel has been designed so that (a) the fuel system will not be damaged as a result of normal operation and anticipated operational occurrences, (b) fuel damage during postulated accidents would not be severe enough to prevent control rod insertion when it is required, and (c) core coolability will always be maintained, even after severe postulated accidents, and thereby meets the related requirements of 10 CFR Part 50.46; 10 CFR Part 50, Appendix A, General Design Criteria 10, 27, and 25; and 10 CFR Part 50 Appendix K. This conclusion is based on the following:

1. Exxon has provided sufficient evidence that these design objectives will be met based on operating experience, prototype testing, and analytical predictions. Those analytical predictions dealing with structural response, control rod ejection, and fuel densification have been performed in accordance with (a) the guidelines of Regulatory Guide 1.77 (Ref. 15), and (b) methods that the staff has reviewed and found to be acceptable alternatives to Regulatory Guides 1.60 (Ref. 51) and 1.126 (Ref. 52).

2. Exxon has provided for testing and inspection of new fuel to ensure that it is within design tolerances at the time of core loading. The applicant or licensee will be required to make a commitment to perform on-line fuel failure monitoring and postirradiation surveillance to detect anomalies or confirm that the fuel has performed as expected.

The staff concludes that Exxon has described methods of adequately predicting fuel rod failures during postulated accidents so that radioactivity releases are not underestimated and thereby meets the related requirements of 10 CFR Part 100. In meeting these requirements, Exxon has done the following:

1. Used the fission produce release assumptions of Regulatory Guide 1.4 (Ref. 53), 1.25, (Ref. 54), and 1.77.

2. Performed the analysis for fuel rod failures for the rod ejection accident in accordance with the guidelines of Regulatory Guide 1.77.

On the basis of our review of the fuel system mechanical design, we conclude that the ENC 17x17 fuel assembly design has met all the requirements of the applicable regulations, regulatory guides, and current regulatory positions.

4.4 Thermal Hydraulic Design

Section 5.0, "Thermal-Hydraulic Design Analysis," of XN-NF-82-25 describes a thermal-hydraulic design criterion of using the minimum departure from nucleate boiling ratio (MDNBR) limit which provides 95 percent probability with 95 percent confidence of avoiding boiling transition. The critical heat flux (CHF) correlation will be used for CHF calculation. In addition, thermal margin to boiling transition will be evaluated on a plant-specific bases because each plant has its own full power operating conditions, core fuel type or types, and core response to anticipated operational occurrences.

The staff has found that both the design criterion based on MDNBR limit and the method of plant specific thermal margin evaluation are acceptable. The staff has also reviewed XN-NF-82-21, Revision 1 , which describes the thermal hydraulic design approach used by ENC in analyzing a core containing fuel assemblies having different thermal and hydraulic characteristics, and has found it a referable document. However, the validity of the correlation as well as the proposed MDNBR limit , described in XN-NF-621, Revision 1 , is still under staff review. Any limitations resulting from this review will be addressed in the appropriate safety evaluation report.

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GENERIC MECHANICAL DESIGN REPORT

EXXON 17 x 17 FUEL ASSEMBLY

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1.0 INTRODUCTION AND SUMMARY

1.1 INTRODUCTION

This report provides a design description and a summary of the design criteria, technical bases, supporting analyses, and test results for the Exxon Nuclear Company (ENC) 17x17 Pressurized Water Reactor (17x17 PWR) reload fuel for Westinghouse reactors. Design drawings of the fuel assembly and major components are included in Appendix A.

1.2 SUMMARY

The ENC 17x17 PWR fuel design is shown to meet the Design Criteria and Technical Bases for Design. The fuel description and mechanical design are summarized below.

1.2.1 Design Description Summary

As compared to previous ENC PWR designs for Westinghouse reactors (14x14 and 15x15), the ENC 17x17 design has an increased number of rods on a smaller pitch, and an increased number of guide tubes. The number of grid spacers has been increased from seven to eight. The grid spacers have been designed with structural members, and are overall for greater assembly rigidity. The expected effects of these changes to the fuel rod design are improvements in fuel reliability, performance and operating margins to safety limits.

The fuel assembly design for the 17x17 PWR reactors uses a design features for improved resistance to pellet-cladding interaction (PCI). The design has a quick-removable upper tie plate design to facilitate inspection and reconstitution of irradiated assemblies.

1.2.2 Mechanical Design Summary

Mechanical design analyses were performed to evaluate cladding steady-state stress and strain, power ramping stress and strain, fatigue damage, creep collapse, corrosion, hydrogen absorption, fuel rod internal pressure, differential fuel rod growth, creep bow, and grid spacer spring design. The analyses were performed to a peak rod burnup of .

- The maximum end-of-life (EOL) steady-state cladding strain, calculated with was , which is well below the design limit.
- The ramp stress, calculated with under different overpower conditions, does not exceed the design limit of .
- The cladding fatigue usage factor of is within the design limit.
- The cladding creepdown plus the reduction due to creep ovality is less than minimum initial gap up to the point of maximum fuel density.
- The fuel rod internal pressure was calculated to remain below typical reactor system operating pressures throughout the design lifetime of the fuel.
- An evaluation of the fuel assembly growth and the differential fuel rod growth indicates that the fuel assembly design provides adequate clearances at the design burnup.

o The maximum calculated EOL thickness of the oxide corrosion layer is ; and the maximum calculated concentration of hydrogen in the cladding is . These values are well within the design limits of , respectively.

o The spacer spring meets all the design requirements and can accommodate the maximum EOL expected relaxation while maintaining rod restraint.

1.2.3 Thermal Hydraulic Design Summary

o The MDNBR for the ENC fuel is determined to be at overpower using the critical heat flux correlation.

o Calculated temperatures are well below the centerline melting.

2.0 FUEL SYSTEM DESIGN OBJECTIVES

The 17x17 PWR fuel system design objectives provide that:

- The fuel system is not damaged as a result of normal operation and anticipated operational occurrences. "Not damaged" means that fuel rods do not fail, that fuel system dimensions remain within operational tolerances and that functional capabilities are not reduced below those assumed in the safety analysis.
- Fuel system damage is never so severe under any transient as to prevent control rod insertion when it is required.
- The number of fuel rod failures shall not be underestimated for postulated accidents.
- Coolability is always maintained.
- The fuel assemblies are designed to withstand loads as a result of in-plant handling and shipping.
- The mechanical and hydraulic design of fuel assemblies will be compatible with coresident fuel and the reactor core internals to achieve acceptable flow distribution including bypass flow such that heat transfer requirements are met for all licensed modes of operation.

3.0 DESIGN BASES

The fuel rod and fuel assembly design bases are established to satisfy the general performance and safety criteria presented in Section 2.0.

3.1 FUEL ROD

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 bases during 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.

3.1.1 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 of Zircaloy-4 are provided in Section 5.2.

3.1.2 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 (Table 3.1) are derived from the ASME Boiler Code, Section III,

3.1.3 Steady-State Cladding Strain

Tests on irradiated tubing indicate potential for failure at relatively low mean strains. These tests include tensile, burst and split ring tests, and the data indicate a ductility ranging between and 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 circumferential cladding strain for steady-state conditions is limited to at end-of-life. In addition, the cladding steady-state primary and secondary stresses must meet the design requirements defined in Table 3.1.

3.1.4 Cladding Tensile Strain Limits

Volatile fission products combined with high cladding stresses and transient strains is a potential cause of stress corrosion cracking failures. Stress corrosion cracking tests have shown that an iodine concentration greater than and tensile stresses are both needed to activate the stress corrosion cracking process at cladding inner surface temperatures between 300 and 400°C. At fast fluences below 10^{20} n/cm^2 there is insufficient fission product inventory to allow concentrations that would activate stress corrosion cracking. The strain limit at these conditions is therefore set at to prevent cladding failure due to plastic instability and localized strain. Power cycling at higher fluences may lead to transient releases of fission products. Where the fission gas composition begins to reach the range of susceptibility to stress corrosion cracking, lower limits on tensile strain are indicated. No power ramp test failures from the Studsvik ramp programs have been observed at a calculated peak circumferential stress level below . The design limits for transient strains are selected consistent with failure correlations used in the ENC fuel rod performance codes to minimize the potential for stress corrosion cracking failure.

3.1.5 Strain Fatigue

The number of cumulative strain fatigue cycles is limited to the design strain fatigue life.

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

3.1.6 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 depths at spacer contact points has typically ranged from

. Examination indicates that the wear is due primarily to fuel rod loading and unloading and not fuel rod motion during the test. There has been little or no difference between observed wear for hour, hour and hour tests. No active fretting corrosion has been observed despite spacer spring relaxation of up to 100% in several test assemblies. Examination of a large number of irradiated rods has substantiated the minimal wear observed after loop tests.

3.1.7 Corrosion

Corrosion reduces the material thickness and results in less load carrying capacity. At normal light water reactor (LWR) operating conditions, this mechanism is not limiting except under unusual conditions where high cladding temperatures greatly accelerate the corrosion rate.

3.1.8 Hydrogen Absorption

Hydrogen can be absorbed on either the outside or the inside of the cladding. The absorption of hydrogen can result in premature cladding failure due to reduced ductility and the formation of hydride platelets. Careful moisture control during fuel fabrication reduces the potential for hydrogen absorption on the inside. The fabrication limit for total hydrogen in the fuel pellets is less than the industry standard of 2 ppm. Sufficient samples are taken to assure that this criterion is met with a probability of 95% at a confidence level of 95%. Except under unusual conditions, significant absorption of hydrogen from the outside of the cladding is not expected.

3.1.9 Creep Collapse

The design basis for creep collapse of the cladding is 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.

Fuel densification may allow the formation of axial gaps in the pellet column. Evaluation of cladding creep stability under this condition considers the compressive load on the cladding due to the difference between primary system pressure and the fuel rod internal pressure. ENC fuel is designed to minimize the potential for the formation of axial gaps in the fuel; hence, creep collapse is not expected to occur.

3.1.10 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.

3.1.11 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 . The potential effect on thermal margins is negligible.

3.1.12 Overheating of Cladding

The design basis for fuel rod cladding overheat 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 limiting fuel rods during normal operation and anticipated operational occurrences. Operating limits are established according to the thermal limits methodology to assure an adequate degree of protection for the fuel.

3.1.13 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.

3.1.14 Mechanical Fracturing

The fuel assemblies are designed to withstand the external loads due to earthquakes and postulated pipe breaks without fracturing the fuel rod cladding. The design limit applied by ENC is that the stresses due to postulated accidents in combination with the normal steady state fuel rod stresses shall not exceed the normal cladding design stress limits.

3.2 FUEL PELLET

3.2.1 Pellet Physical and Mechanical Properties

The physical and mechanical properties of the uranium dioxide fuel is presented in Section 5.3.

3.2.2 Fuel Pellet Temperature

The center temperature of the hottest pellet shall be below the melting temperature of the UO_2. Fuel centerline temperature is calculated at overpower conditions to verify that fuel pellet overheating does not occur during normal operation and anticipated operational occurrences.

3.2.3 Fuel Pellet Density - The nominal design density of the fuel is , and along with conservative assumptions with regard to tolerances, this value is used in the analyses.

3.2.4 Densification and Swelling

The design bases for fuel densification and swelling are as established in Regulatory Guide 1.126 . Densification and swelling models are as described .

3.3 SPACER GRIDS

The spacer assembly is designed to withstand the thermal and irradiation induced differential expansion between the fuel rods and guide tubes and to withstand the design handling and accident loads discussed in Section 3.4.1.

The grids provide sufficient fuel rod support to limit fuel rod vibration and to prevent cladding fretting wear. The spring dimple system in the grid spacer is designed such that the minimum spring/dimple forces throughout the design life are greater than the maximum fuel rod flow vibration forces.

3.4 FUEL ASSEMBLY

3.4.1 Structural Design

The structural integrity of the fuel assemblies is assured by setting design limits on stresses and deformations due to various handling operational and accident loads. These limits are applied to the design and evaluation of upper and lower tie plates, grid spacers, guide tubes, holddown springs, and locking hardware.

The design bases for evaluating the structural integrity of the fuel assemblies are:

- Fuel Assembly Handling - Dynamic axial loads assembly weight.

- For all applied loads for normal operation and anticipated operational events - The fuel assembly component structural design criteria are established for the two primary material categories, austenitic stainless steels (tie plates) and Zircaloy (guide tubes, grids, spacer sleeves). The stress categories and strength theory for austenitic stainless steel presented in the ASME Boiler and Pressure Vessel Code Section III are used as a general guide. Zircaloy material properties are listed in Section 5.2.

Steady State Stress Design Limits are given in Table 3.1. Stress nomenclature is per the ASME Boiler and Pressure Vessel Code, Section III.

- Loads during postulated accidents - Deflection or failure of components shall not interfere with reactor shutdown or emergency cooling of the fuel rods.

The fuel assembly structural component stresses under faulted conditions are evaluated using primarily the methods outlined in Appendix F of the ASME Boiler and Pressure Vessel Code, Section III.

3.4.2 Coolability During Postulated Accidents

The fuel assembly design basis for earthquakes and postulated pipe breaks is that the fuel assembly shall maintain a coolable geometry and control rod insertability during the occurrence of the design basis seismic/LOCA event.

3.4.3 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.

3.4.4 Assembly Holddown

The design basis for fuel assembly holddown is that the springs, as compressed by the upper core plate during reactor operation, will provide a net positive downward force during steady-state operation, based on the most adverse combination of component dimensional and material property tolerances.

3.5 TESTING AND SURVEILLANCE

An extensive testing program has been conducted to verify the adequacy of the predicted fuel performance and the design bases

.

Post-irradiation examinations will continue to be performed to assess the performance of the 17x17 PWR fuel assembly and the predicted irradiation effects which were assumed in the design. Surveillance programs for the fuel design involve visual examination (e.g., televi ion and/or binocular scanning), and dimensional measurements of se ected fuel assemblies. The removable upper tie plate feature of the fuel assembly design simplifies fuel rod removal and facilitates individual rod examinations.

Table 3.1

Steady State Stress Design Limits*

	Stress Intensity Limits**	
	Yield Strength (σ_y)	Ultimate Tensile Strength (σ_u)
General Primary Membrane Stress		
Primary Membrane Plus Primary Bending Stress		
Primary Plus Secondary Stress		

* 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 due to 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

The ENC 17x17 array reload fuel assembly design for Westinghouse pressurized water reactors (PWR's) is an extension of the assembly design currently in production for reactors accommodating 14x14 and 15x15 array designs. The 17x17 array contains 264 fuel rods, 24 guide tubes and 1 instrument tube, a total of 289 positions. The increased number of guide tubes, and increased number of grid spacers (from 7 to 8), assures adequate strength and stiffness.

4.1 FUEL RODS

The fuel rods consist of short cylindrical UO_2 pellets in Zircaloy-4 tubular cladding. Zircaloy-4 end caps are welded to each end to give a hermetic seal.

The fuel rod cladding is Zircaloy-4

. Each standard fuel rod contains a column of enriched UO_2 fuel pellets. The pellets are pressed and sintered

and are dished on both ends.

.

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

4.2 SPACER GRIDS

The spacer grids are Zircaloy structures each of which provide a 17x17 array of 289 cells for maintaining separation of the 264 fuel rods, support for an instrument tube, and means for attachment to the 24 structural (guide) tubes.

The structure consists of interlocking specially formed Zircaloy-4 strips

Dimples, formed in the spacer strips, center the fuel rod within the cell. The dimples, along with springs, provide a positive compliant support for each rod, sufficient to prevent fretting due to vibration, yet still allow relative motion due to differential thermal expansion.

4.3 FUEL ASSEMBLY STRUCTURE

The fuel assembly structure consists of an upper tie plate, lower tie plate, guide tubes and spacer grids, which together provide the support for the fuel rods.

4.3.1 Lower Tie Plate

The lower tie plate is a heavy stainless steel member which provides the lower end support for the guide tubes, and engages pins installed in the reactor lower core support plate to provide positive positioning for the assembly within the reactor core. The Zircaloy guide tubes are to the lower tie plate.

4.3.2 Upper Tie Plate

The upper tie plate is a heavy stainless steel member which provides the upper end support for the guide tubes, and engages pins in the upper core plate to provide positive positioning. springs attached to the top of the upper tie plate, compressed by the upper core plate, provide a compliant loading to compensate for relative thermal expansion and for vertical growth of the assembly due to irradiation. The springs provide sufficient loading to prevent assembly lift-off due to hydraulic loads during normal operation.

4.3.3 Guide Tubes

Twenty-four (24) Zircaloy-4 guide tubes extend from the lower tie plate,

to the upper tie plate. At the upper tie plate, the guide tubes are mechanically positioned and locked. The locking mechanism is such that, while providing an absolute attachment of the tie plate to the guide tube in the locked mode, it can be readily unlocked using special tools. These features facilitate examination or reconstitution of assemblies by permitting instant removal and installation of the upper tie plate, providing access for removal and reinsertion of fuel rods. Zircaloy-4 sleeves are welded to the upper and lower end of the guide tubes to provide extra strength. The lower sleeve also serves as an attachment point for the bottom spacer grid, while the upper sleeve provides a means of transmitting axial loads between tie plate and guide tubes.

5.0 DESIGN EVALUATION

The fuel assemblies and fuel rods are designed to satisfy the performance and safety criteria of Section 2.0 and the mechanical design bases of Section 3.0. Effects of anticipated operational occurrences and postulated accidents on fuel integrity are determined in plant specific and generic analyses in the supporting topical reports. Material strength properties of major components are summarized in Table 5.1.

5.1 DESIGN CONSIDERATIONS AND EVALUATION FOR PRIMARY FUEL ROD FAILURE MECHANISMS

The individual failure mechanisms are discussed below along with the resulting design criteria and design features developed by ENC to prevent such failures.

5.1.1 External Cladding Corrosion

BWR and PWR cladding corrosion data (both in and out-of-reactor) have been reviewed and correlations developed to describe the in-reactor corrosion behavior of fuel rods . Cladding oxidation and corrosion product buildup are limited to prevent significant degradation of clad strength. A maximum PWR clad external temperature of is specified to limit overall corrosion. The specified external corrosion layer thickness limit will not significantly affect thermal and mechanical design margins.

Corrosion layer thicknesses are calculated with

Temperature increases as a consequence of postulated corrosion product buildup are also calculated in which is used for fuel performance and stored energy calculations.

5.1.2 Cladding Hydrogen Absorption

The as-fabricated and end-of-life cladding hydrogen levels are limited to prevent adverse effects on the mechanical behavior of the cladding due to hydriding . The effects of hydrogen on mechanical properties have been investigated at hydrogen concentrations to about . The most meaningful data, however, are in the range of about

The effect on strength and ductility depends on such factors as:

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

In tubing where the texture does not favor radially oriented hydrides, the combined effect of hydrides and irradiation do not appear to be significant at hydrogen concentrations in the

Based on data, cladding texture, and experience to date, the design limit for hydrogen in the cladding

5.1.3 Stress Corrosion Cracking

- Iodine Concentration

The combination of volatile fission products and high cladding stresses may lead to stress corrosion cracking. Quantitative data are available which indicates that the probability of failure is a function of fission product concentration at the inside cladding surface, local stress level, strain rate, and tubing texture.

Stress corrosion cracking of fuel rod cladding is considered the principal failure mechanism for the PCI failures that are encountered during changes in reactor operating conditions. Even though unanimous agreement has not been reached on which chemical species enhances failure, the iodine atmosphere is usually considered the primary attacking agent. The iodine concentration and cladding strain rate are significant in determining the ultimate ductility of the cladding; but if the stress level is low enough in the cladding stress corrosion cracking does not occur. Tests have been performed under EPRI support to evaluate the iodine stress threshold. Figure 5.1 shows typical data from this program and indicates that the time dependence of stress corrosion rupture is primarily controlled by two processes. The high stress process is represented by the steep slope portion of Figure 5.1 and is controlled by crack propagation. The lower stress process is represented by the shallow slope portion of Figure 5.1 and is controlled by time-dependent crack nucleation.

Stress corrosion cracking tests have shown that an iodine concentration greater than is needed to activate stress corrosion cracking process at normal inside cladding temperatures between 300 and 400°C. It is expected that these concentrations can never be reached under steady-state conditions due to recombination of free iodine. Reference (13) indicates that the highest sensitivity to low ductility stress corrosion failure is for strain rates between . Thus, stress corrosion cracking is anticipated to be active under transient reactor operating conditions.

- <u>Texture</u>

Stress corrosion cracks in metals preferentially initiate and propagate along specific crystallographic planes. The preferred crystallographic direction for stress corrosion cracks in zircaloy is along a plane at an angle of approximately 15° with the basal plane. Work has shown that grains with basal pole directions between 0° and 50° with the surface have a diminished tendency to crack in an iodine atmosphere.

Similarly, work carried out
has shown that zircaloy tubing with a higher frequency of basal poles in the radial direction of the tube has a higher iodine stress corrosion cracking threshold than tubing with more tangentially directed poles.

The crystallographic texture of zircaloy tubing in the radial direction is commonly characterized by the quantitative texture number f_r. A high f_r value is less susceptible to stress corrosion cracking than tubing with a low f_r value.

Measurement of the contractile strain ratio or R-value has recently been shown to be a method to determine the texture number f_r. The contractile strain ratio or R-value is defined as the ratio of the true plastic circumferential strain (ε_θ) to the true plastic wall thickness strain (ε_r) for a tube subject to axial plastic tensile strain

The relation between the texture number and the R-value is given by:

$$f_r = \frac{R}{R + 1}$$

For most zircaloy tubing, R can vary from approximately 1.00 to 1.85, which corresponds to a variation in f_r between 0.50 and 0.65. Measurement of the R-value can be a method to evaluate stress corrosion susceptibility. A high R-value indicates lower susceptibility and a low R-value indicates higher susceptibility to stress corrosion attack. A minimum R-value of is specified for high burnup fuel.

• <u>Stress and Strain Limits</u>

ENC has developed design criteria on a microscopic basis based on observed fuel rod performance during power ramp conditions.

At fast fluences below 10^{20} n/cm^2 (0.5 GWD/MTM), batch average burnup) there is insufficient fission product inventory to allow concentrations that would activate stress corrosion cracking. The strain limit at these conditions is therefore set at to prevent cladding failure due to plastic instability and localized strain.

At the higher fluence levels the stress limit is reduced to to reduce the probability of PCI failure. No power ramp test failures from the appropriate Studsvik ramp data (Figure 5.2) are observed at a peak circumferential stress level below as calculated using the ENC fuel performance codes . Evaluation of the Studsvik ramp data shown in Figure 5.2 indicates a reasonable correlation between measured total ramp stress and failure. Calculations of these cases with result in convervatively higher stresses in all cases.

* <u>Pellet Design</u>

To minimize PCI and steady-state and transient stress/strain levels, the ENC fuel rod design features pellets with an optimized pellet geometry,

. The effect of pellet geometry on clad strain is evaluated .

* <u>Cladding Internal Surface</u>

From ENC experience, a rough cladding inside surface finish significantly increases the loads required to insert a column of pellets and increases the probability of pellet cracking and chipping which may contribute to fuel failures.

Pellet-cladding interaction (PCI) that leads to fuel rod failure results primarily from stress corrosion cracking (SCC) of the cladding. Tests to evaluate SCC cracking in two batches of zircaloy tubing, have shown significant differences in susceptibility to SCC. When the internal surface of the more susceptible tubing was polished, the susceptibility decreased dramatically. Other research proposes that initiation of SCC is increased in cladding with inside surface flaws by one or more of the following mechanisms:

- Easier or more frequent breakdown of ID surface oxide film due to surface flaws.
- Locally increased stress or change of stress ratio.
- Ease of the crack initiation process through localized chemical differences.
- Removal of favorable surface texture.

The surface condition requirements selected are based upon SCC considerations and at the limit of tubing manufacturing capabilities.

5.1.4 Steady State Cladding Stress and Strain

Tests on irradiated tubing indicate failure at relatively low mean strains. The test results for tensile, burst, and split ring tests show a ductility between at normal reactor operating temperatures.

The presence of iodine or other fission products can cause the cladding to fail at lower strain levels. However, susceptibility to this type of failure (stress corrosion cracking) occurs only when the fission product concentration exceeds the threshold, the strain rate is between , and the stress is above a threshold value. As pointed out in Section 5.1.3 above, all of these conditions are unlikely under steady-state or near steady-state operation. Thus, creep and burst tests on irradiated cladding in a non-corrosive atmosphere can establish ductility limits since these 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 localized strain, ENC design criteria limit the steady-state cladding circumferential plastic strain to and the steady-state primary and secondary stresses to within the requirements defined in Table 3.1.

5.1.5. Fatigue Damage

Cyclic mechanical strains can cause cumulative damage and subsequent failure which may be predicted by fatigue analysis techniques. have developed a zircaloy fatigue analysis design curve which is presented in Figure 5.3. This curve is based on fatigue test data

.

The ENC design criterion limits the cumulative damage factor (C.D.F.) to account for a corrosive environment and other fatigue mechanisms. The cumulative damage factor is calculated as follows:

5.1.6. Creep Collapse

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 (a flattened area) in the region of a potential pellet column gap. During power changes such collapse could result in fuel failure.

Through proper design, the probability of creep collapse can be significantly reduced. Typical ENC pellets are stable dimensionally. Irradiation data for ENC fuel rods, in addition to resintering tests performed , show that densification is not likely to exceed

. This specification ensures stable pellets during irradiation.

An plenum spring is included in the ENC fuel rod design to prevent formation of gaps in the pellet column. This plenum spring provides a positive compressive force on the fuel column throughout the densification phase of the fuel life. No gaps larger than approximately inch have been observed during gamma scans of many irradiated fuel rods.

The fuel rods are helium prepressurized, which assists in the prevention of creep collapse if a pellet column gap were to develop. The design criterion is a free standing cladding until densification is complete.

5.1.7 Fuel Rod Internal Pressure

The fuel rod internal pressure is primarily a function of the initial fuel rod pressurization, fuel swelling, and fission gas release. The minimum fuel rod fill pressure is set at a level designed to assure acceptable thermal performance of the fuel, and to assure that the collapse criteria are met. Post-irradiation measurements have demonstrated that significant fission gas release can occur in LWR fuels when rod powers exceed a threshold level. This release can be magnified by a fission-gas release thermal feedback effect. Fission gas release can be reduced if the initial helium pressurization is high enough so that when fission gas release does occur no significant reduction in thermal conductivity across the pellet-to-cladding gap is incurred. As a result, fuel performance characteristics as well as margins to safety limits are not significantly degraded due to fission gas release effects. The maximum fill pressure is designed such that thermal performance is not limited at the beginning-of-life and the fuel rod end-of-life pressure does not exceed the reactor system pressure, as required by the internal fuel rod pressure design basis. Fission gas release in the fuel rod is calculated by , which accounts for the thermal feedback effect.

5.1.8 Creep Bow

Fuel rod bow is determined throughout the life of the fuel assembly so that reactor operating thermal limits can be established. These limits include the minimum critical power 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).

To-date, ENC has a data base of over fuel-rod-to-fuel rod and fuel-rod-to-guide tube spacing measurements from inspection of irradiated ENC fuel in to a maximum exposure of rod-to-rod measurements have been obtained from ENC 7X7 and 8X8 BWR assemblies reaching a burnup up to . These measurements have been used to establish an empirical model for determining rod bow as a function of burnup which is used to calculate thermal limits. See Section 6.6

Special features in the ENC fuel design significantly reduce the extent of fuel rod bow. These features include:

5.1.9 Fretting Corrosion

ENC incorporates a spacer grid with a zircaloy structure in 17x17 PWR fuel assembly designs. The spring dimple system in the grid spacer is designed such that the minimum spring/dimple forces throughout the design life are greater than the maximum fuel rod flow vibration forces.

Simulated flow tests at reactor flow, pressure, and temperature conditions have been performed on prototype assemblies for periods . Fretting tests for the 17x17 assembly are reported in reference . No active fretting corrosion has been observed even though spacer spring loads were purposely relaxed up to 100% in some assemblies. Examination of irradiated assemblies has not revealed wear significantly different from that observed in the prototype tests.

5.2 CLADDING MATERIAL CONSIDERATIONS

This section describes the physical properties of the Zircaloy-4 fuel rod cladding used in the mechanical design analyses.

5.2.1 Thermal Conductivity

Thermal conductivity for Zircaloy-4 is based on data published

5.2.2 Thermal Expansion

The mean coefficient of thermal expansion for Zircaloy-4 is taken from

5.2.3 Elastic Modulus and Poisson's Ratio

The temperature dependence of the modulus of elasticity, E, used in design calculations is based on

Poisson's ratios ( ) used in design calculations are as follows:

5.2.4 Effect of Temperature on Strength and Ductility

The effect of temperature on longitudinal yield and ultimate stresses for Zircaloy-4 cladding is shown in Figure 5.5. Based on ENC test data, the minimum yield strength as a function of temperature (over the range of interest for fuel rods) is described in the following equation:

For design calculations, transverse strengths are considered equivalent to longitudinal strength.

where:

R is defined as the ratio of the contractile strain in the radial direction to the contractile strain in the circumferential direction as determined in an uniaxial longitudinal tensile test. For isotropic material, R is equal to 1.0 and the percent of wall thinning is the same as the percent of diameter reduction. For a hexagonal lattice material such as Zircaloy with ony one slip system, isotropic behavior in tubing occurs only when the basal poles are oriented at 45° to the radial direction.

5.2.5 Ductility

Ductility in terms of total axial elongation measured on ENC tubing as a function of temperature is shown in Figure 5.6, 5.2.6 Effects of Irradiation on Strength and Ductility

Irradiation hardens the cladding up to the temperature where in-situ annealing occurs, as illustrated in Figure 5.7 . For design purposes, irradiation hardening is not taken into account, and the strength of the cladding at beginning-of-life (BOL) is assumed to be constant throughout its design lifetime.

Irradiation hardening reduces ductility as shown in Figure 5.8 . Total elongation at rapid strain rates reaches minimum values as low as , whereas uniform elongation reaches values less than at fast fluences greater than . At slow strain rates (such as in creep tests), uniform elongation is greater than .

5.2.7 Creep Rate Characteristics

Zircaloy creep rate used in fuel rod design is based on a general relationship

For steady state analyses, inelastic deformation is assumed to be a creep mechanism described by the equations above. These creep relationships are discussed in the report. For transient deformations, a linear strain hardening plastic model is utilized in the code.

5.2.8 Cladding Corrosion and Hydrogen Absorption

Based on available data and assumed control of coolant water chemistry (e.g., halides, hydrogen, and oxygen), the hydrogen absorption of zircaloy in the temperature range of

is:

where:

In these equations, the variable H_c has been converted from a fraction of the oxide thickness to units of average parts per million of weight in the cladding.

For typical ENC fuel rods where X_0, the initial oxide thickness,

For a typical 17x17 PWR operational history, it was found that the maximum thickness of the oxide layer at EOL was:

which is within the limit. The maximum weight fraction of hydrogen added to the cladding from the coolant was:

The concentration of hydrogen caused by internal sources such as fuel is , due to the controls on the fuel and the fuel rod assembly process.

Using for the initial concentration, the net weight fraction of hydrogen in cladding is:

This hydrogen concentration is less than the design limit.

5.3 FUEL MATERIAL CONSIDERATIONS

Physical and thermal characteristics of the fuel material considered in the mechanical design of the fuel rods, differential thermal expansion of fuel and cladding, fuel pellet swelling, fuel densification, and pellet cracking are provided in this section. These characteristics are incorporated into ENC's RODEX2 fuel performance code.

5.3.1 Thermal Conductivity

The thermal conductivity function for UO_2 is from

5.3.2 Thermal Expansion

The thermal expansion model for UO_2 is based on relationship, i.e.,

5.3.3 Melting Point

The value used for the UO_2 melting point is , the melting point is reduced with irradiation at the rate of

5.3.4 Swelling

Fission product swelling of UO_2 during reactor operation may be regarded as the sum of the contributions from solid and gaseous fission products. Solid fission products tend to accumulate inside the grains. Some of the solid species are volatile at temperatures readily exceeded during in-reactor operation, and may diffuse to the pellet boundaries, thus reducing solid swelling. Gaseous fission products consist mainly of the inert gases xenon and krypton. They tend to diffuse to and accumulate in the grain boundaries. The grain boundaries can accommodate gas to a certain thickness, which limits gas swelling to a saturation value. This saturation value consequently depends on the temperature and on the total boundary surface, i.e., the grain size. Since the boundary bubbles may be mechanically compressed, gaseous swelling also depends on external restraint from the cladding contact pressure.

The data which deal systematically with the effects of temperature, burnup and restraint, were used in establishing the swelling model incorporated in the code.

5.3.5 Densification

The physical process of densification is governed by the fission events which tend to annihilate the small pores. The correlation for densification based on burnup is utilized. This relation is:

5.3.6 Cracking (Pellet Relocation)

It is clear from experimental observations that an oxide fuel pellet will experience thermal stresses sufficient to cause significant cracking The most noticeable physical result of cracking is . A detailed investigation of approximately 80 irradiated fuel pellet cross-sections has shown that pellet cracking results This conclusion was based on observation of fuel with a broad range of physical parameters irradiated in nine reactors. These fuels exhibited .

Because measurements reported in . were based on the at room temperature, the change in the fuel pellet due to cracking is evaluated in on the basis of r, as follows:

Although the fuel pellet which results from pellet cracking is calculated on the basis of irradiation time, it is important to recognize that the effects of all variables which might affect , whether identified or not, are included in the basic data which developed the correlation; i.e., power cycling, crack healing, densification, and cladding restraint. Figure 5.12 shows an average fit of the data and the cracking curve from which the (pellet relocation) was calculated.

5.4 CLADDING STEADY-STATE STRESS

Each individual stress was calculated at both the inner and outer surfaces of the cladding. The applicable stresses at each orthogonal direction were then combined to get the maximum stress intensities. The analysis was performed at beginning-of-life (BOL) and end-of-life (EOL) at cold and hot conditions. The maximum stress intensities and the appropriate stress limits are reported in Table 5.2. The stress analysis assumes the most conservative conditions; for example, maximum fuel rod power, minimum fill gas pressure, and the most conservative fuel rod geometry.

5.4.1 Primary Stresses

- Primary Membrane Stresses

The primary membrane stresses are produced by the coolant pressure and fuel rod fill gas pressure. The stresses are calculated by

• Primary Bending Stresses

Bending stresses due to ovality are calculated with

5.4.2 Secondary Stresses

* Cladding Thermal Gradient Stresses

Fuel rods operate with a temperature gradient across the cladding wall which may result in significant thermal stresses. Assuming no stress relaxation, thermal stresses are calculated by :

• Restrained Thermal Bow

Stress due to circumferential gradients are conservatively estimated using relationships

• Restrained Mechanical Bow

Stress from mechanical bow between spacers, assuming maximum-as-built fuel rod bow is zero, is taken from

- Flow Induced Vibration Stresses

Vibrational stresses due to flow induced vibrations is calculated with the which assumes the following:

1) The structural stiffness of the fuel rod is due to the cladding only.
2) The sections of the fuel rod between spacers and/or tie plate supports are modelled structurally as a simple beam with pinned ends.

5.4.3 Contact Stress From Spacer Springs

The contact stresses at the spring locations are calculated using the finite element model shown in Figure 5.3. Calculations were performed with the ANSYS general purpose finite element code. The circumferential and axial stresses induced by the contact load are incorporated in Table 5.2.

5.5 FUEL ROD END CAP

Zircaloy end caps are seal welded to each end of the fuel rod cladding. The stress analysis is performed at the lower end cap since the maximum temperature gradients occur at this end.

The mechanical stress is caused by the pressure differential across the rod wall and by the axial load of the pellet stack weight and the plenum spring force. The thermal stress is caused by the temperature gradient between the end cap and the heat generating pellets. The stress analysis is for the standard ENC end cap design and envelopes both PWR and BWR applications. Therefore, the calculated stress intensity values are higher than what would be expected for the 17x17 PWR design, with the smaller rod diameter.

The ANSYS code which allows thermal as well as stress analyses, was used to model the subject rod region. The problem was solved by a thermal pass and a stress pass, where the stress analysis used the results of the thermal analysis as part of its input. The model is in axisymmetric geometry and was set-up such that the element system could be used in both analyses. The weld-joint region of the model is shown in Figure 5.14.

The calculation was made assuming direct contact between the pellet stack and the end cap. A bounding value ws taken for the end pellet LHGR. The results of the analysis are summarized in Table 5.3.

5.6 CLADDING STEADY-STATE STRAIN ANALYSIS

The cladding steady-state strain is evaluated with the code, which is an interactive calculational procedure that 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. Calculations are performed for the worst expected fuel rod power and fast flux history to determine a conservative history in terms of cladding strain.

In addition to evaluation of the fuel rod steady-state cladding strain, determines the initial conditions for fuel rod power ramping analyses and the fuel rod internal pressures for cladding creep analyses. Pellet density, swelling, densification, and fission gas release models, and cladding and pellet diameters are input to to provide the most conservative subsequent ramping or collapse calculations for the reference fuel rod design.

The fuel rod performance characteristics modelled by the code are:

- Gas Release
- Radial Thermal Condition and Gap Conductance

- Free Rod Volume and Gas Pressure Calculations
- Pellet-Cladding Interaction
- Fuel Swelling, Densification, Cracking and Crack Healing
- Cladding Creep Deformation and Irradiation Induced Growth

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 modelled. The axial dependence of the spatial power and burn-up distributions are handled by dividing the fuel rod into a number of fuel segments which are modelled as radially dependent regions whose axial deformations and gas release are summed. Power distributions can be changed at any desired time and the coolant and cladding temperatures are readjusted at all axial nodes. Deformations of the fuel and cladding and gas release are incrementally calculated during each period of assumed constant power generation. Gap conductance is calculated for each of these incremental calculations based on gas release throughout the rod and the accumulated deformation at the center of each axial region within the fueled region of the rod. These deformation calculations consider fuel densification, swelling and cracking, thermal expansion, cladding creepdown, irradiation induced growth, and fuel creep and crack healing.

The pellet-to-cladding interaction during reactor operation is dependent upon the power and flux history. The peak discharge burnup fuel rod was analyzed for maximum EOL cladding strain. The design power is summarized in Table 5.4. This history must be considered as being a conservative upper bound for the peak power rod since it leads to a maximum peak pellet discharge burnup of approximately , corresponding to a rod average discharge burnup of . With the minimum design pellet to cladding gap and the maximum fuel density, the maximum calculated EOL steady-state strain of is within the design criteria limit of .

5.7 CLADDING RAMP STRESS ANALYSIS

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

Analyses for power ramp conditions were performed for the fuel rod maximum power envelope summarized in Figure 5.15. The fuel rods were analyzed for pellet/cladding interaction pressure at the end of the third cycle of irradiation corresponding to a rod average exposure of

.

At this point in time, stresses were calculated with for a power ramp in which the power was escalated to a maximum linear heat generation rate (LHGR) of

The ramp rate was assumed to follow recommendations. Maximum hoop stress was determined to be . The recommended limit is .

5.8 CLADDING FATIGUE USAGE FACTOR

In addition to the transient strain analyses, a fatigue usage factor for the cladding was calculated. The calculations were based upon the duty cycles summarized in Table 5.5 which conservatively envelope the expected duty cycles of a typical PWR. As for the cladding ramp strain analysis, the power ramp rate was assumed to follow . Cladding stress amplitudes for the various power cycles were determined from analyses. The initial conditions were obtained from outputs and it was conservatively assumed that all the power changes occurred when a high ramp stress was calculated. To account for possible stress concentration in the cladding, an assumed total strain concentration factor was applied to the calculated cyclic cladding stresses. Table 5.6 summarizes the final cladding cyclic stresses for the reference case and the allowable number of cycles at each stress amplitude. The allowable cycles were determined from the fatigue design curve shown in Figure 5.3 which considers the effect of maximum mean stress. The total usage factor is less than the design criteria requirement of a maximum cumulative usage factor .

5.9 CREEP COLLAPSE

The approved ENC clad collapse procedure utilizes the to determine the clad ovality in unsupported regions of the fuel column to establish that the accumulation of creep ovality does not result in clad collapse. The procedure does not evaluate the likelihood or extent of pellet separation; thus, successful fuel performance will be observed in many cases where the code would predict collapse. In addition, as burnups and irradiation times increase, it becomes more likely that creep collapse could eventually occur in an unsupported tube. Since practical means of limiting pellet separation have been established, the conservatism of considering an infinite length of unsupported cladding may be removed once sufficient data on pellet separation for particular fuel are obtained and a criteria for pellet separation is accepted.

This section summarizes the existing evaluation procedure, the ENC fuel densification and gap formation experience to-date, the new evaluation procedure, a sensitivity analsis using the new procedure, and comparisons of the collapse evaluation with irradiated fuel data.

In order to guard against the unlikely event that sufficient densificat)n occurs to form pellet column gaps of sufficient size for clad flattening to occur the following evaluation is performed.

Creep ovality analysis is performed with the code using the existing creep collapse evaluation procedure. Cladding creep down is obtained from the corresponding analysis. The combination of cladding ovality increase and creep down are calculated, and at a rod average burnup of , the combined creep down shall not exceed the initial minimum gap. This will prevent pellet hangups due to cladding creep, allowing the plenum spring to close axial gaps until densification is substantially complete. The calculated value of creep ovality is . . The calculated value of cladding creepdown is . The sum is ., which is less than the minimum gap .

5.10 INTERNAL PRESSURE

The fuel rod internal pressure was calculated using the with an ENC-developed physically based model for fission gas release. The calculated EOL internal pressure of psi is well below the reactor system pressure.

5.11 FUEL ROD PLENUM SPRING

The plenum spring is spring which maintains a compact column of fuel pellets in the rods during handling, shipping, loading and initial fuel densification.

A nominal force is exerted by the spring on the fuel column. This load is greater which is sufficient to seat the fuel column through the expected conditions during handling, shipping and loading.

was selected as the spring material because it retains high strength properties at high temperatures. Irradiation induced relaxation of the plenum spring in the time period of the initial fuel densification is expected to be less than .

5.12 FUEL ASSEMBLY STRUCTURAL STRENGTH

Upper and lower tie plates, upper tie plate guide tubes, locking hardware, and guide tube sleeves constitute the fuel assembly structural components. In order to withstand expected handling loads, the assembly is designed to withstand axial tensile loads times the dry bundle weight and axial compressive loads times the dry bundle weight with no permanent deformation. Also, each guide tube to tie plate connection is designed to withstand a loading of not less than

5.12.1 Structural Testing

Structural testing was conducted to demonstrate compliance with the criteria for design. Testing included the tie plates, tie plate to guide tube connections, including the locking mechanism, axial loading of individual spacers at the outer edge, and lateral compressive loading of spacers. In all cases, measured strength greatly exceeds the strength criteria.

5.12.2 Guide Tube

The guide tubes in the fuel assembly provide the support for the grid spacers between the upper and lower tie plates. The guide tubes are fabricated from a single piece of Zircaloy-4 tubing drawn to two (2) different diameters. The larger diameter section at the top provides a relatively large annular area for rapid RCC insertion during a reactor trip and accommodates a small amount of upward cooling flow during normal operations. The small diameter section at the bottom, approximately 24 inches long, produces a dashpot action to decelerate a dropped control rod.

With the guide tubes, spacers, and tie plates assembled into a framework, the guide tubes and attachment hardware provide, throughout the design life of the fuel assembly, adequate strength to support the weight of the fuel assembly, support the holddown forces, and resist forces from fuel rod-guide tube differential thermal expansion.

Guide tubes are considered as restrained columns and are analyzed with appropriate load combinations. Column deflection is permissible within allowable bending stress constraints, displacement, and approach to column instability. The total stress allowed, primary plus bending, is equal to the yield strength of the material at the temperature of the load conditions.

The force transmitted to the fuel rods is estimated by:

Rod loading tests with similar prototype spacer assemblies confirmed this calculated value of F_R. The average load to push a rod through a spacer cell was as compared to assumed above.

The force applied to a guide tube during a temperature transient is:

At BOL, the assembly holddown springs exert a force less than . This results in an additional load per guide tube. The total load per guide tube would be

The column stability is defined by

The above critical load is significantly above the design load.

In actual practice, an initially bowed column bows an increasing amount as a compression load is applied rather than suddenly collapsing as the critical load is reached. As a result, the design load limit for a typical guide tube is more likely to be that which produces a bow unacceptable from a thermal hydraulic standpoint rather than the load which produces column instability.

The total guide tube bow may be determined by:

The resulting deflections are satisfactory from a thermal-hydraulic standpoint.

5.12.3 Irradiation Growth

In reactor, the Zircaloy-4 guide tubes and fuel rods increase in length as a function of exposure. To evaluate this growth, the correlation and data collected from examination of irradiated ENC fuel assemblies (Figure 5.16) are used. In the case of guide tubes, the design choice .

The clearance between core plates, the fuel assembly length, thermal expansion and the guide tube growth, along with conservative application of associated tolerances, are used to assure positive clearance throughout the design life.

To assure adequate clearance throughout the design life for the fuel rods between the constraints of the upper and lower tie plates, the BOL cold clearance requirement is set at of the fuel column height.

5.12.4 Assembly Holddown

The design of the holddown springs in the upper tie plate ensures that there is sufficient force to prevent fuel assembly lifting due to hydraulic pressure loads. Holddown springs are also designed to accommodate fuel assembly irradiation growth and differential thermal expansion between the assembly and the core support structure, including an accounting for the full range of component tolerances. The holddown spring must retain its ability to counteract the hydraulic lift force through life. In some designs, a small amount of spring relaxation might occur. This relaxation is compensated for by increased compression due to the bundle growth. This allows the holddown spring to continue to provide the design holddown forces throughout the fuel life.

For a typical 17x17 PWR design, the holddown spring constant is   , for a minimum   net holddown force during normal operation of   . The maximum applied force is less than   for the most extreme combination of tolerances.

5.12.5 Fuel Rod Creep Bow

Fuel rod bow is determined throughout the life of the fuel assembly so that reactor operating thermal limits can be established. These limits include the minimum critical power 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 postulated loss-of-coolant accident (LOCA).

Special features in the ENC fuel design significantly reduce the extent of fuel rod bow. These features include:

To-date, ENC has a data base of over fuel-rod-to-fuel rod and fuel-rod-to-guide tube spacing measurements from inspection of irradiated ENC fuel in to a maximum exposure of

. rod-to-rod measurements have been obtained from ENC 7x7 and 8x8 BWR assemblies reaching a burnup up to . These measurements have been used to establish an empirical model for determining rod bow as a function of burnup which is used to calculate thermal limits.

The rod bow data which is summarized in Figure 5.17 shows that the bow tends to stabilize at higher burnups. In addition, the fuel at higher burnups is not limiting from a thermal margin standpoint due to its lower power.

5.13 SPRING CHARACTERISTICS OF SPACER GRIDS

5.13.1 Spring Rate Evaluation

The support stiffness required to force a node at a support level is generally considered to be

This condition is easily met as the support dimples are very stiff. The support stiffness if given by:

The dimple stiffness was conservatively estimated from experimental mechanical tests on the 17x17 spacer strip design and determined in cold conditions. With a nominal spring rate in hot conditions , the support stiffness in hot conditions is:

5.13.2 Minimum Spring Force

The spring force F_{v1}, required to counteract the maximum flow vibration lateral acceleration forces to prevent the fuel rod from lifting off from both dimples simultaneously is given by:

The zircaloy fuel rods are expected to relax at a significantly greater rate

. Therefore, only lo^ding sufficient to overcome

. Current irradiation data indicates relaxation values on the order as shown in Figure 5.11, and a minimum cold, BOL, spring load would assure adequate loading for EOL hot conditions. The nominal design spring load is

.

, the spring force F_{V_2}, required to prevent

The minimum spring force from a bowing consideration is

.

also limits midspan bowing deflection. The spring force required dimple at each spacer level is estimated on the basis of the model shown in Figure 5.18. The bow is assumed to be symmetrical with respect to the center spacer. The minimum spring force, assuming uniform curvature, is defined as F_{D_1}.

The unrestrained fuel rod bow is due to manufacturing tolerances (mechanical bow) and diametral temperature differences during operation (thermal bow).

. Assuming a circular bow, :

a temperature difference T between diametrically opposed points in the cladding is:

The required spring force is:

The total minimum required spring force to maintain contact

A minimum BOL spring force of or greater meets this requirement with ample margin.

5.13.3 Load Deflection

Due to spacer cell and fuel rod diameter tolerance stack-up, spring deflection ranges from . The BOL spring force ranges from . The average spring rate over this range is

5.13.4 Maximum Spring Force

The maximum spring force is limited by the allowable stresses in the spring and in the cladding due to spring contact.

Spring deflection is limited by backup lobes on the leaf spring strip. The limit of deflection by the backup lobes allows the spring to operate in only the elastic range.

The clad stresses resulting from a maximum spring force of at the beginning-of-life are calculated as described in Paragraph 5.4.3. Calculated cladding stresses at the spacer contact points are within the limits summarized in Table 5.2.

Table 5.1

Summary of Fuel Component Mechanical Properties

Fuel Assembly Component	Material	Minimum Strength (psi)				
		Room Temperature		Elevated Temperature		
		Yield	Tensile	Temp(°F)	Yield	Tensile
Cladding	Zr-2 Tube					
Cladding	Zr-4 Tube					
End Caps and Connectors	Zr-2 or 4 Bar					
Guide Tubes	Zr-4 Tube					
Nuts and Cap Screws						
Spacer Structural Components	Zr-4					
Tie Plate Castings						
Coil Springs Including Plenum and Holddown Springs						
Spacer Tie Plate Seal or Tie Plate Leaf Springs						

* Elevated temperature value is not specified.

**Shear value is given since loading is in the shear mode.

Table 5.2

Summary of Limiting Stress Conditions

		Stress Intensity psi	Design Limit psi	Ratio of Stress Intensity to Design Limit
1. Primary Membrane Stresses				
(Design Limit is lower value of	BOL Cold			
	BOL Hot			
	EOL Cold			
	EOL Hot			
2. Primary Membrane Plus Primary Bending				
(Design limit is lower value of	BOL Cold			
(Stresses included in this category are the general membrane and ovality stresses.)	BOL Hot			
	FOL Cold			
	EOL Hot			
3. Primary Plus Secondary				
(Design limit is lower value of	BOL Cold			
	BOL Hot			
	BOL Cold			
(Stresses included in this category are the stresses from Item 2 above, plus vibration, thermal gradient, mechanical and thermal bow, and spacer contact pressure.)	BOL Hot			

Table 5.3

Stress Intensity at Lower End Cap

	Case 1 at Room Temperature		Case 2 End Pellet at		Design Limits	
	psi	(MPa)	psi	(MPa)	psi	(MPa)
Primary Membrane, Design Limit:						
Weld Joint Primary Membrane Plus Primary Bending, Design Limit:						
Weld Joint Primary Plus Secondary, Design Limit:						

Table 5.4

Power and Fast Flux History
(Pin with Maximum Discharge Exposure)

Time During Exposure (hrs)	Peak Pin Exposure (MWD/MT)	Peak Pin Power (kw/ft)	

Table 5.5

Duty Cycles

Table 5.6

Cyclic Stress Summary

Duty Cycle	Actual Cycles ni	Stress Amplitude (KSI)	Allowable Cycles Ni	
1				
2				
3				
4				
5				

Figure 5.1 Stress Threshold for Irradiated Zircaloy Cladding Tested in an Iodine Environment [68]

Figure 5.2 RIDGE HOOP STRESS VS. RESIDUAL GAP

Figure 5.3 Cyclic Fatigue Design Curve for Irradiated Zircaloy-2 or -4 Room Temperature to 600°F (316°C).

Figure 5.4 Effect of Spacer Cell Geometry
on Axial Restraint

Figure 5.5 Mechanical Strength of ENC Zircaloy-4 Tubing Versus Temperature

Figure 5.6 Effect of Temperature on Ductility of Zircaloy-2

*Room Temperature

Figure 5.7 Effect of Fast Neutrons on Yield Stress of Zircaloy-2

Figure 5.8 Effect of Fast Neutron Irradiation on the Mechanical Properties of Zircaloy-2

Figure 5.9 SHORT TERM CREEP

Figure 5.10 SHORT TERM CREEP. COMPARISON OF MODEL WITH DATA

Figure 5.11 UO_2 Thermal Conductivity (95% TD)

Figure 5.12 Closure of As-Fabricated Gap Versus Irradiation Time

Figure 5.13 Contact Stress Finite Element Model

Figure 5.14 View of Weld Joint Region Finite Element Model

Figure 5.15 Maximum Normalized Relative Pin Power as a Function of Fuel Pin Exposure

Figure 5.16 PWR Fuel Rod Growth

Figure 5.17 Gap Spacing Data

Figure 5.18 Illustration of Forces and Deflections of Bowed Fuel Rods

6.0 THERMAL HYDRAULIC DESIGN ANALYSIS

6.1 DESIGN BASIS AND CRITERIA

The primary thermal hydraulic design basis for Exxon Nuclear Company reload fuel is that fuel rod integrity should be maintained during normal operation and anticipated operational occurrences. Specific criteria are:

(1) Avoidance of boiling transition for the limiting fuel rod in the core with at least a 95% probability at a 95% confidence level.

(2) Fuel centerline temperatures should be below the melting point of the fuel pellets.

Observance of these criteria during anticipated operational transients is considered conservative relative to the requirement that anticipated operational transients not produce fuel rod failures or loss of functional capability.

The margin to boiling transition for 17x17 fuel is assessed with . With this correlation, a minimum departure from nucleate boiling ratio, MDNBR, of provides 95% probability against boiling transition with 95% confidence.

6.2 MDNBR EVALUATION

The evaluation of margins to boiling transition, i.e., MDNBR, is performed on a plant specific basis. This is necessary since,

- o normal full power operating conditions vary from plant to plant,
- o core response to anticipated operational occurrences and accidents is plant specific,
- o thermal margins will depend upon the amount and type of coresident fuel.

Thus, for each plant, power peaking limits are established to assure the thermal hydraulic criteria in respsect to DNBR are met.

6.3 FUEL CENTERLINE TEMPERATURE

Peak design linear heat generation rates (LHGR) for ENc 17x17 fuel corresponding to maximum allowed limits on total peaking, F_Q, are expected to be less than . This LHGR is about in other ENC PWR reload fuel, and is well below the steady-state LHGR required for centerline melt . Thus, penetration of the centerline melt criteria is not limiting for ENC 17x17 fuel.

7.0 TESTING AND INSPECTION PLAN

7.1 QUALITY ASSURANCE AND QUALITY CONTROL

Quality Assurance Programs and Quality Control Plans, concerning both ENC manufacturing and testing and vendors who perform tests and inspections on behalf of ENC are described in Topical Report XN-NF-1A, Revision 3, "Quality and Fabrication", August 1980. This report has been previously approved by the USNRC.

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GENERIC MECHANICAL DESIGN REPORT

EXXON 17 x 17 FUEL ASSEMBLY

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