1.0 INTRODUCTION

The subject report, XN-NF-81-21, was prepared to present reload fuel design information and related safety analyses of the kind found in Section 4.2 of plant FSARs. Although some plants for which this fuel might be provided would not be required to meet all of the guidelines of the current Standard Review Plan, that would obviously be sufficient. Therefore, the NRC staff has reviewed XN-NF-81-21 in accordance with Section 4.2 of NUREG-0800, the latest version of the SRP.

2.0 FUEL SYSTEM DESIGN OBJECTIVES

The objectives of this fuel system safety review as described in Section 4.2 of the Standard Review Plan (SRP) 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. A "not damaged" fuel system 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. Objective (a) above implements General Design Criterion 10 (10 CFR 50, Appendix A), 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 100 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 channels to permit removal of 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 and 35). Specific coolability requirements for the loss-of-coolant accidents are given in 10 CFR 50 Section 50.46.

To assure that the above stated objectives are met, the following areas are 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, operating experience, prototype testing, and analytical predictions are compared with acceptance criteria for fuel system damage, fuel rod failure, and fuel coolability.

3.0 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 critieria for these design limits into three categories in the Standard Review Plan (SRP): (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 postulated accidents, and (c) fuel coolability criteria, which apply to postulated accidents.

The subsection designations below follow the organization of the SRP rather than XN-NF-81-21.

3.1 Fuel System Damage Criteria

In the following paragraphs we review the design bases and corresponding design limits for the damage mechanisms listed in the SRP. These design limits along with certain criteria that define failure (Section 3.2) constitute the Specified Acceptable Fuel Design Limits (SAFDLs) required by General Design Criterion (GDC) 10. The design limits in this section should not be exceeded during normal operation and AOOs.

(a) Design Stress

The design basis for fuel cladding stress is that the fuel system will not be damaged due to fuel cladding stresses. Design limits for cladding stress were derived by methods similar to those given in the ASME code, Section III. ENC specifies that the allowable primary membrane stresses should not exceed 2/3 of the yield strength or 1/3 of the ultimate tensile strength of Zircaloy-2 cladding in the unirradiated condition. Table 3.2 of XN-NF-81-21 shows other stress limits under various stress conditions.

As for other Zircaloy components, such as grid spacers, tie rods, spacer capture rods, methods similar to Section III of the ASME code were also used to derive the stress limits. Table 3.2 of XN-NF-81-21 shows the stress limits for these components also.

For stainless-steel components, such as upper and lower tie plates, the design basis is that loading should result in no significant plastic deformation during normal operation and AOOs. The ASME code, Section III was used as a general guide to deduce these stress limits, too.

ENC has used Section III of the ASME code as general guidance for developing design stress limits. This conforms with the SRP guidelines and is therefore acceptable.

(b) <u>Design Strain</u>

To prevent cladding failure due to plastic instability and localization of strain, ENC has used 1% total strain as a limit for steady-state conditions. This goes beyond the SRP guidelines and is thus acceptable. For transient conditions, which relate to PCI failures, the strain criterion is discussed in Section 3.2(f) of this SER.

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

The ENC design basis for strain fatigue is that the total cumulative damage factor (CDF) should not exceed [ ]* to account for a corrosive environment and other fatigue mechanisms. Exxon has used a fatigue design curve from [ ] that includes a safety factor of 2 on stress amplitudes or a safety factor of 20 on the number of cycles, whichever is more conservative. This is consistent with the SRP guidelines and is, thus, acceptable.

* Brackets indicate the deletion of proprietary information.

(d) Fretting Wear

The design basis for fretting wear is that fuel rod failures due to fretting shall not occur. Although no design limits were presented for fretting wear in XN-NF-81-21, the fuel rods and grid spacers were designed to prevent such wear. Since the SRP does not provide numerical acceptance criteria for fretting wear, and since fretting wear is addressed in the design analysis, we conclude that this deviation from the SRP is justified.

(e) External Corrosion and Crud Buildup

The JP-BWR fuel design bases for cladding oxidation and corrosion product buildup, respectively, are to prevent significant degradation of cladding strength and unacceptable temperature increases due to corros product buildup. Because of the increased thermal resistance of corrosion and crud layers, there is an increased potential for elevated temperature within the fuel as well as the cladding. Considering these effects, a cladding external temperature limit of [ ]F is proposed with an oxidation limit of [ ] inch. The [ ]F cladding external temperature limit is chosen on the grounds that corrosion rates are generally low below that temperature, while the external corrosion limit of [ ] inch is specified on the grounds that this degree of corrosion will not significantly affect design margins (i.e., increase cladding stresses above allowable levels). We agree with the rationale for these modest limits and conclude that they are acceptable.

(f) Rod Bowing

Fuel rod bowing is a phenomenon that alters the nominal spacing between adjacent fuel rods and between fuel rods and the surrounding channels. Bowing affects local heat transfer to the coolant and local nuclear power peaking. The ENC JP-BWR design basis for fuel rod bowing is that lateral displacement of the fuel rods shall not be of sufficient magnitude to impact safety margins. To accomplish this, ENC has established a design basis of a minimum gap spacing of [ ] mils.

In a recent letter (Chandler, April 27, 1982), ENC demonstrated that a gap spacing of [ ] mils would produce negligible effects on heat transfer. This conclusion was based on tests reported in the literature (Nixon et al., August 1975), which found negligible reductions in critical heat flux for rod-to-rod clearance as low as 60 mils.

The only neutronic effects of BWR fuel rod bowing is a potential for a small local peaking factor increase. There are no significant reactivity effects. The small local peaking factor increases on one rod would be accompanied by decreased peaking factors in other rods in the assembly and no net change in planar-average assembly power density would occur. Therefore, there would be no significant effect on LOCA temperature calculations, which are primarily dependent on the planar-average assembly power density in BWRs. The small increase in LHGR (linear heat generation rate) is not significant for other events since LHGR limits are not approached for other events, ,ch are limited by MCPR.

We, therefore, agree that the ENC JP-BWR gap spacing limit is appropriate.

(g) Axial Growth

Excessive axial differential growth of tie rods and fuel rods is a concern because it could lead to improper tolerances. The design basis for fuel rod and assembly tie rod growth is that upper and lower tie plate engagement shall be maintained for all fuel rods in the bundle during the design lifetime. This accomodation of axial growth is consistent with the guidelines of the SRP.

(h) Fuel Rod Pressure

Section 4.2 of the SRP identifies excessive fuel rod internal pressure as a potential fuel system damage mechanism. It calls for rod pressures to remain below nominal system pressure during normal operation unless otherwise justified. ENC has specified that internal gas pressure of fuel rods shall not exceed external coolant pressure, thus meeting the guidelines of the SRP.

(i) Assembly Liftoff

The SRP calls for the fuel assembly holdown capability (gravity and springs) to exceed worst-case hydraulic loads for normal operation and AOOs. Although ENC does not discuss such a criterion in XN-NF-81-21, it will be seen in Section 5.1(i) of this SER that ENC fuel design meets this SRP acceptance criterion.

3.2 Fuel Rod Failure Criteria

The NRC staff's evaluation of fuel rod failure thresholds of the failure mechanisms listed in the SRP is presented in the following paragraphs. When these failure thresholds are applied to normal or transient operation, 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, the number of fuel failures must be determined 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. ENC's design limit for hydrogen level in the as fabricated UO_2 fuel pellets is [ ] ppm total hydrogen. [ ] the ASTM specification, which is cited in the SRP and allows 2 g hydrogen per gram of uranium (i.e., 2 ppm).

(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 a gap (i.e., flattening). Because of the large local strains that would result from collapse, the cladding is assumed to fail. As indicated in XN-NF-81-21, it is ENC's design limit that the cladding shall not collapse. This JP-BWR fuel design limit agrees with the SRP and is acceptable.

(c) Overheating of Cladding

As indicated in the SRP Section 4.2.II.A.2(d), it has been traditional practice to assume that failures will not occur if the thermal margin criterion is satisfied. For BWR fuel, thermal margin is stated in terms of the minimum value of the critical power ratio (CPR) for the most limiting fuel assembly in the core. The design limit for ENC BWR fuel to prevent cladding overheating is that transition boiling shall be prevented. This satisfies the intent of MCPR criterion in the SRP and is thus acceptable. The review of thermal-hydraulic design methods is beyond the scope of this safety evaluation.

(d) Overheating of Fuel Pellets

As indicated in the SRP Section 4.2.II.A.2(e), it has been traditional practice to assume that failure will occur if fuel pellet centerline melting takes place. Thus, as a second method of avoiding cladding failure due to overheating, ENC avoids fuel pellet centerline melting during normal operation and AOOs as a design basis. The design limit corresponding to this design basis is that the peak linear heat generation rate during normal operation and AOOs does not result in fuel centerline melting, taking into account the effects of burnup and gadolinia content. We find this acceptable.

(e) Excessive Fuel Enthalpy

For a severe reactivity initiated accident (RIA) in a BWR at zero or low power, fuel failure is assumed in the SRP to occur if the radially averaged fuel rod enthalpy is greater than 170 cal/g at any axial location. The 170 cal/g enthalpy criterion, developed from SPERT tests (Grund et al., August 1969), is primarily intended to address cladding overheating effects, but it also indirectly addresses pellet/cladding interactions of the type associated with severe RIAs. Although ENC does not mention this criterion, we will show in Section 5.2(e) of this SER that the ENC fuel does not exceed this SRP criterion.

(f) Pellet/Cladding Interaction

As indicated in SRP Section 4.2, there are no generally applicable criteria for PCI failure. However, two acceptance criteria of limited application are presented in the SRP for PCI: (a) 1% transient-induced cladding strain, and (b) no centerline fuel melting. ENC proposes a transient strain limit of 1% at fluences less than [ ] n/cm^2, a transient strain limit of [ ] at fluences greater than [ ] n/cm^2, and a transient strain limit corresponding to a linear interpolation between [ ] and [ ] for intermediate fluences. Although ENC asserts that no power-range test failures have been observed at a mean circumferential plastic level below [ ] as calculated using ENC fuel performance codes such as RODEX2 (steady-state) and RAMPEX (transient), the NRC staff has not completed the RODEX2 review and review of RAMPEX is not planned. Moreover, some PCI failures have been observed in commercial reactor fuel with very low measured strain levels. Therefore, we are not sure that the [ ] cladding strain limit is sufficient as a generally applicable PCI failure criterion for all postulated reactor conditions, although it may be useful in some applications. However, the ENC design basis includes the 1% strain and centerline melt criteria of the SRP (and, in fact, goes beyond them), so we find this approach acceptable.

(g) Cladding Rupture

Zircaloy cladding will burst (rupture) under certain combinations of temperature, heating rate, and differential pressure -- conditions that occur during a LOCA. While there are no specific design limits associated with cladding rupture, the requirements of Appendix K to 10 CFR Part 50 must be met as those requirements relate to the incidence of rupture during a LOCA; therefore, a rupture temperature correlation must be used in the LOCA ECCS analysis. Cladding rupture is described analytically as a part of the ECCS evaluation model (XN-CC-33), and that analytical correlation will be evaluated in Section 5.2(g) of this SER.

(h) Fuel Rod 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 ENC design basis for JP-BWR fuel assembly mechanical fracturing is that the assemblies must withstand the external loads due to earthquakes and postulated pipe breaks without fracturing the fuel rod cladding. The design limit proposed by ENC is that the stresses due to postulated accidents in combination with the normal steady-state fuel rod stresses should not exceed the limit for normal cladding design stress given in Table 3.2 of XN-NF-81-21. This design limit for JP-BWR fuel mechanical fracturing exceeds the SRP guidelines and is, therefore, acceptable.

3.3 Fuel Coolability Criteria

For major accidents in which severe damage might occur, core coolability must be maintained as required by several GDCs (e.g., GDC 27 and 35). The following paragraphs discuss the staff's evaluation of limits that will assure that coolability is maintained for the severe damage mechanisms listed in Section 4.2 of the SRP.

(a) Fragmentation of Embrittled Cladding

To meet the requirements of 10 CFR 50.46 as it relates to cladding embrittlement for a LOCA, acceptance criteria of 2200°F on peak cladding temperature and 17 percent on maximum cladding oxidation must be met. These criteria are used by ENC (XN-NF-80-19(P), Volume 2).

(b) Violent Expulsion of Fuel

In a severe reactivity initiated accident (RIA) such as a BWR control rod drop, the large and rapid deposition of energy in the fuel can result in fuel melting, fragmentation, and violent dispersal of fuel droplets or fragments into the primary coolant. The mechanical action associated with such 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 the SRP as it relates to the prevention of widespread fragmentation and dispersal of fuel and the avoidance of pressure pulse generation within the reactor vessel, a radially averaged enthalpy limit of 280 cal/g should be observed. ENC uses this criterion in the generic topical report, Volume 1 of XN-NF-80-19(P), that presents their rod-drop accident analysis.

(c) Cladding Ballooning

Zircaloy cladding will balloon (swell) under certain combinations of temperature, heating rate, and stress during a LOCA. While Appendix K to 10 CFR 50 requires that the degree of swelling during a LOCA not be underestimated, there are no design limits required for cladding swelling. Cladding ballooning is described analytically as a part of the ECCS evaluation model (XN-CC-33), and that analytical correlation will be evaluated in Section 5.3(c) of this SER.

(d) Fuel Assembly Structural Damage from External Forces

Earthquakes and postulated pipe breaks in the reactor coolant system would result in external forces on the fuel assembly. SRP Section 4.2 and associated Appendix A state that fuel system coolability should be maintained and that damage should not be so severe as to prevent control rod insertion when required during these low probability accidents. These SRP design bases are used by ENC as described in Section 3.5.2 of XN-NF-81-21.

4.0 DESCRIPTION AND DESIGN DRAWINGS

The description and design drawings of major fuel assembly components, including fuel rods water rods, tie rods, upper and lower tie plates, spacer grids, compression springs, retaining springs, locking sleeves, and adjusting nuts are provided in Section 4.0 and Appendix A of XN-NF-81-21. In addition, design specifications are provided in Table 4.1 of Section 4.0. The material properties of fuel and cladding, which are essentially the same as those in RODEX2, are provided in Sections 5.2 and 5.3, respectively, of XN-NF-81-21. While each parameter listed in SRP Section 4.2.II.B is not provided in XN-NF-81-21, enough information is available in sufficient detail to furnish a reasonably accurate representation of fuel design, and this information therefore satisfies the intent of the SRP guidelines.

5.0 DESIGN EVALUATION

Design bases and limits were presented and discussed in SER Section 3. In this section we review ENC methods of demonstrating that the fuel design of XN-NF-81-21 meets the design criteria that have been established. This SER section will, therefore, correspond to Section 3 of the SER point by point. The methods of demonstrating that the design criteria have been met include operating experience, prototype testing, and analytical predictions.

5.1 Fuel System Damage Evaluation

The following paragraphs discuss the NRC staff's evaluation of the ability of the ENC fuel to meet the fuel system damage criteria described in Section 3.1. Those criteria apply only to normal operation and anticipated transients.

(a) Design Stress

As indicated in Section 5.4 of XN-NF-81-21, the primary membrane stresses are calculated using the Lamé equations recommended by Shariffi and Popov (Shariffi and Popov, December 1969). Primary bending stresses due to ovality are calculated with Timoshenko's equation (Timoshenko, 1956). The cladding thermal stress and thermal bow are calculated using standard equations described by Goodier (Goodier, March 1937) and Timoshenko and Gere (Timoshenko and Gere, 1961), respectively. Other secondary stresses, such as those caused by (a) mechanical bow between spacers, (b) flow induced vibration, and (c) contact from spacer dimples and springs, are also considered and calculated using conventional equations and equations described in the open literature (Roark, 1965; Paidoussis and Sharp, 1968; Paidoussis, March 1965). Table 5.1 of XN-NF-81-21 shows that the calculational results are well below the design limits for normal operation. Table 5.2 also shows stresses of rod end cap, based on ANSYS code calculations, satisfying design limits.

ENC has tested the assembly strength by having the tie plates subjected to axial tensile forces in excess of [ ] the fuel assembly dry weight. The result shows no evidence of plastic deformation. Further tests show the assembly would not fail until reaching a load of [ ] lbs. However, the failures occur as expected at the tie rod end caps with no detrimental effects on grid spacers or upper and lower tie plates.

On the basis of testing and analyses with standard engineering methods, we conclude that reasonable assurance has been provided that fuel assembly components including fuel rods, spacer grids, and upper and lower tie plates meet the stress design criteria.

(b) Design Strain

For cladding steady-state strain calculations, ENC uses the RODEX2 (XN-NF-81-58) 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 were performed for a peak discharged burnup fuel rod at a design maximum linear heat generation rate (LHGR), which was determined from MAPLHGR limits, fission gas release, and mechanical considerations (see Fig. 5-10 of XN-NF-81-21). The maximum calculated end-of-life (EOL) strain was found to be [ ], which is well within the design limit of 1% strain. Because the RODEX2 review has not yet been completed, the NRC staff will require that licensees using the JP-BWR fuel confirm or redo this strain analysis unless RODEX2 is approved without modification.

(c) Strain Fatigue

A cumulative damage factor (CDF) is used to evaluate the strain fatigue effect. The calculations are based on the duty cycles summarized in Table 5.3 of XN-NF-81-21, which conservatively envelope the expected plant operation. The stress amplitudes are enlarged by [ ] to account for other effects such as stress concentration due to fuel cracking and fretting wear. The allowable number of cycles is determined from the fatigue design curve of [ ] according to the enlarged stress amplitudes. The result given in Table 5.4 of XN-NF-81-21 shows a total CDF of [ ], which is well below design limit of [ ]. Therefore, we conclude that the JP-BWR fuel fatigue design limit has been met.

(d) Fretting Wear

ENC has run fretting-corrosion tests on nine prototype PWR assemblies and six prototype BWR assemblies. Fuel rod wear depths at spacer contact points typically range from [ ] to [ ] mils, although wear of [ ] mils in depth has been observed. There is no observable correlation between observed wear and test time. Examination indicates, in fact, that the wear is due primarily to fuel rod loading and unloading rather than fuel rod motion during the test. There has been little or no difference in the wear observed during prolonged tests. Examination of a large number of irradiated rods has not revealed wear significantly different from that observed after the loop tests. We thus conclude that the JP-BWR fuel will perform adequately with respect to fretting wear.

(e) External Corrosion and Crud Buildup

The fuel cladding is subjected to an external corrosive environment during irradiation and a corrosion layer on the cladding surface will impede the heat transfer and degrade the Zircaloy cladding ductility. ENC uses a two-stage corrosion rate model with modified correlations adapted from MATPRO-11 (NUREG/CR-0497) in RODEX2 to calculate temperature and oxide thickness. The calculated maximum cladding temperature for JP-BWR fuel applications is [ ], which is less than the design limit of [ ]. The calculated maximum thickness of the oxide layer at end-of-life is [ ] mils, which is within the design limit of [ ] mils. Because the RODEX2 review has not yet been completed, the NRC staff will require that licensees using the JP-BWR fuel confirm or redo this corrosion analysis unless RODEX2 is approved without modification.

As for crud buildup, ENC does not have a design limit because 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 the NRC staff believes 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 time in the reactor. We will, therefore, consider this issue on a generic basis as part of our ongoing study of the effects of extended burnup.

(f) Rod Bowing

ENC has reported (XN-NF-77-49) over 10,000 rod-to-rod measurements from inspection of irradiated ENC 7x7 and 8x8 BWR assemblies from the Oyster Creek reactor wherein burnups to 25,000 MWd/MTU were achieved. In the ENC report XN-NF-81-21, an empirical fuel rod bowing model is described that was constructed from these Oyster Creek data. The model does not conform to the recommended NRC guidance (Lear, June 7, 1978) inasmuch as a batch-to-batch variability factor of 1.5 was inadvertently omitted. However, even with the correct accounting for the batch-to-batch variability, we find that the 95/95 closure for the worst span in the hot condition will not exceed the ENC design gap spacing limit for burnups to at least 30,000 MWd/MTU.

(g) Axial Growth

Generally, a higher growth rate is experienced by the tie rods than by the fuel rods, and differential rod growth is predicted from fuel inspection data of similar ENC irradiated fuels. Although axial growth data for JP-BWR fuel rods are not available, a conservative extrapolation from other data predicts that the maximum differential growth of fuel rods will be [ ] in. Since the nominal engagement of the fuel rod end cap to the upper tie plate is [ ] in., we conclude that there is sufficient margin for the fuel rod growth.

Fuel assembly growth is a direct result of tie rod growth. Figure 5.13 of XN-NF-81-21 gives data on Zircaloy-2 axial growth versus fast neutron fluence from various sources of irradiation. The peak assembly average fast fluence at EOL is estimated to be 6×10^{21} n/cm^2. An extrapolation of Figure 5.13 in XN-NF-81-21 to this fluence gives a maximum assembly growth of [ ] in., which is very small compared with the nominal clearance of the assembly to the reactor internals (4 ft). We thus conclude that assembly axial growth will not be a problem.

(h) Rod Pressure

The fuel rod internal pressure is primarily a function of the initial pre-pressurization, fuel swelling, and fission gas release. ENC uses a physically based model in RODEX2 for rod internal pressure calculation, and they use the design power history in Figure 5.10 of XN-NF-81-21. This power history is considered to be a conservative upper bound for the peak power rod of JP-BWR fuel. The calculated EOL internal pressure does not exceed the system pressure. Because the RODEX2 review has not yet been completed, the NRC staff will require that licensees using the JP-BWR fuel confirm or redo this rod pressure analysis unless RODEX2 is approved without modifications.

(i) Assembly Liftoff

ENC has stated that JP-BWR fuel is designed to fit in existing channel boxes producing a total assembly weight of 660 lbs., which is about the same weight of a GE assembly. The calculated worst-case hydraulic load is about 4?0 lbs.

Thus a net holddown margin of more than 200 lbs. exists for JP-BWR fuel assemblies. We thus conclude that fuel assembly liftoff of the JP-BWR fuel design will not occur during normal operation.

5.2 Fuel Rod Failure Evaluation

The following paragraphs discuss the staff's evaluation of (a) the ability of the JP-BWR fuel to operate without failure during normal operation and anticipated transients, and (b) the accounting for fuel rod failures in the applicant's accident analysis. The fuel rod failure criteria described in Section 3.2 are used for this evaluation.

(a) Internal Hydriding

As indicated in Section 3.2(a), ENC limits hydrogen level to [ ] ppm in the manufacture of reactor fuel. ENC has not reported significant fuel failures due to hydriding. We, therefore, conclude that reasonable assurance has been provided that hydriding as a fuel failure mechanism will not be significant in the JP-BWR fuel.

(b) Cladding Collapse

ENC uses the approved COLAPX code (XN-72-23) to predict creep collapse. The COLAPX code calculates the geometry changes and creep deformation of the cladding as a function of time for the power history provided in Figure 5.10 of XN-NF-81-21. The most severe collapse condition is examined for the peak burnup fuel rod. The result shows that the cladding instantaneous collapse pressure remains greater than the differential pressure between primary coolant and fuel rod. Thus creep collapse is not expected to occur during the fuel rod lifetime.

(c) Overheating of Cladding

As stated in SRP Section 4.2, adequate cooling is assumed to exist when the thermal margin criterion (a critical power ratio, CPR) is satisfied. The method employed to meet the CPR design limit is normally reviewed as part of a thermal-hydraulic methods review and will not be discussed here. The ENC XN-3 critical power correlation is discussed in XN-NF-512(P).

(d) Overheating of Fuel Pellets

Section 3.4 of XN-NF-81-76 describes the fuel rod centerline temperature analysis at 120% overpower for the Dresden 3, Cycle 8 reload. Calculations were performed with RODEX2. The conditions that produced the smallest centerline temperature margin occurred at 21,200 MWd/MTU and a nominal power of 13.9 kW/ft (120% overpower of 16.7 kW/ft). The Dresden 3 analysis was, in fact, performed using the design peak pellet power limit shown in Figure 5.10 of XN-NF-81-21 so that the result is applicable generically to JP-BWR fuel applications that are bounded by this design curve. Calculated fuel centerline temperature was found to remain below the UO_2 melting temperature. Because the RODEX2 review has not yet been completed, the NRC staff will require that licensees using the JP-BWR fuel confirm or redo this fuel melting analysis in their reload safety analyses unless RODEX2 is approved without modifications.

(e) Excessive Fuel Enthalpy

A detailed analysis of the control rod drop accident is reported in the approved report, XN-NF-80-19(P), Volume 1. The analysis of the control rod drop accident for zero-power core conditions (XN-NF-81-76) shows a total enthalpy of 151 cal/g, which is below 170 cal/g and thus indicates no failures according to this SRP criterion. For the full-power RIAs in a BWR, ENC uses MCPR, as discussed in Section 3.2(c), to predict failures. Further discussion of various transients is documented in XN-NF-81-78. We thus conclude that the failure mecnanism of excessive fuel enthalpy for JP-BWR fuel has been properly addressed in various ENC documents.

(f) Pellet/Cladding Interaction

The only two PCI criteria currently used by the NRC in licensing (1% cladding strain and no fuel melting) are easily satisfied. ENC has a somewhat more restrictive transient strain limit that is a function of cladding fluence. The strains are calculated with the RAMPEX code (XN-NF-573) with input from RODEX2 (XN-NF-81-58). These codes may not precisely calculate cladding strain, but they are used by ENC to provide an engineering assessment of failure probability based upon comparison with available failure data. Inasmuch as the calculated transient strains were less than the ENC plastic strain design limit for the cases considered and because that limit and the lack of centerline melting meet or exceed the current Standard Review Plan requirements for PCI, we consider that the ENC JP-BWR fuel design meets the regulatory requirements related to PCI. Because the RODEX2 review has not yet been completed, the NRC staff will require licensees using JP-BWR fuel to confirm or redo the transient strain analysis unless RODEX2 is approved without modifications.

Exxon also addresses the type of PCI that is associated with stress corrosion cracking (SCC). ENC considers SCC to be the principal PCI failure mechanism encountered during changes in reactor operating conditions and addresses cladding texture, pellet design, and cladding internal surface roughness as important design features that can affect PCI resistance. While we believe that attention to such design features may help to reduce the PCI failure probability, 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, expecially during short-term transients. PCI, therefore, will continue to receive generic study.

(g) Cladding Rupture

The NRC staff has been generically evaluating three fuel cladding models that are used in ECCS analysis. Those models predict cladding rupture temperature, cladding burst strain (ballooning), and fuel assembly flow blockage (used only in PWR analysis). The staff has (a) discussed its generic evaluation of these models with vendors and other industry representatives (Denise, November 20, 1979), (b) published NUREG-0630, and (c) required supplemental calculations to confirm that their operating reactors would continue to be in conformance with the ECCS Acceptance Criteria of 10 CFR 50.46 if the NUREG-0630 correlations were used and certain other compensatory model changes were allowed (Eisenhut, November 9, 1979; Denton, November 26, 1979).

The requirement for supplemental ECCS calculations is the same as the present requirement for all operating license applications and all ECCS reanalyses for operating reactors. We are reviewing the ENC report XN-NF-82-07, which provides revised cladding swelling and rupture models, but until such time that the ENC evaluation model (XN-CC-33) is revised to incorporate XN-NF-82-07 , a supplemental calculation using the NUREG-0630 rupture temperature model will be required on a plant-specific basis each time a new ECCS analysis is performed. See paragraph (c) of the following Section 5.3 for a concurrent requirement on cladding ballooning model.

(h) Fuel Rod Mechanical Fracturing

The mechanical fracturing analysis is done as a part of the seismic-and-LOCA loading analysis, which is described in XN-NF-81-51. A discussion of the seismic-and-LOCA loading analysis is given in Section 5.3(d) of this SER.

5.3 Fuel Coolability Evaluation

The following paragraphs discuss the staff's evaluation of the ability of the JP-BWR fuel to meet the fuel coolability criteria described in Section 3.3. Those criteria apply to postulated accidents.

(a) Fragmentation of Embrittled Cladding

The primary degrading effect of a significant degree of cladding oxidation is embrittlement of the cladding. Such embrittled cladding will have a reduced ductility and resistance to fragmentation. The most severe occurrence of such embrittlement is during a LOCA. The overall effects of cladding embrittlement on the JP-BWR fuel design for the loss-of-coolant accident are analyzed in XN-CC-33 and are not reviewed further 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 to 10 CFR Part 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. ENC uses several fuel performance codes for this function and t this writing is about to make a transition from GAPEXX (XN-73-25) to RODEX2 (XN-NF-81-58). A licensee using the ENC JP-BWR fuel should make sure that the fuel performance code that is used has current NRC approval.

(b) Violent Expulsion of Fuel

ENC has evaluated the rod drop accident generically with the procedures described in the Volume 1 of XN-NF-80-19(P). ENC calculates a maximum radially averaged fuel enthalpy of 151 cal/g for the control rod drop accident, and this enthalpy value is well below the 280 cal/g limit. We, therefore, conclude that there is reasonable assurance that control rod drop should not be a problem with regard to violent expulsion of fuel for the JP-BWR fuel.

(c) Cladding Ballooning

As discussed in Section 5.2(g), a supplemental ECCS calculation may be required to show continued conformance with the ECCS Acceptance Criteria of 10 CFR 50.46 using NUREG-0630 correlations and certain other compensatory model changes.

The requirements for the supplemental ECCS calculation is the same as the present requirement for all operating license applications and all ECCS reanalyses for operating reactors. We are review : the ENC report XN-NF-82-07, which provides revised cladding swelling and rupture models, but until such time chat the ENC evaluation model (XN-CC-33) is revised to incorporate XN-NF-82-07, a supplemental calculation using the NUREG-0630 burst strain model will be required on a plant-specific basis each time a new ECCS analysis is performed. See paragraph (g) of Section 5.2 for a concurrent requirement on the cladding rupture model.

(d) Structural Damage from External Forces

For BWRs licensed before 1980, fuel assembly structural analyses were not reviewed in a manner similar to that described in Appendix A of SRP Section 4.2. And the NRC's "Unresolved Safety Issue" that backfitted this analysis to operating PWRs did not do so for BWRs (NUREG-0609). Therefore, for many BWRs for which ENC would supply fuel, there is no clear NRC requirement for this analysis. Nevertheless, ENC has performed generic analyses following the guidelines of SRP-4.2 Appendix A and comparing results with the acceptance criteria of Appendix A. This work has been documented in XN-NF-81-51, which is presently being reviewed by the NRC staff. Thus, for BWRs that were not originally reviewed in accordance with SRP-4.2 Appendix A, the staff's approval of the use of ENC JP-BWR fuel does not depend on the outcome of the staff's review of XN-NF-81-51.

6.0 TESTING, INSPECTION, AND SURVEILLANCE PLANS

6.1 Testing and Inspection of New Fuel

As described in SRP Section 4.2, testing and inspection plans for new fuel should include verification of significant fuel design parameters. 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.

A discussion of the ENC quality control program for the JP-BWR fuel is provided in XN-NF-1A, which addresses fuel system component parts, fuel pellets, rods and assemblies, and process control. Fuel system inspections vary for the different component parts and may include dimensions, visual appearance, audits of test reports, material certification, and non-destructive examinations. Pellet inspection is performed for dimensional characteristics such as diameter, density, length, and squareness of ends. Fuel rods, water rods, upper and lower tie plates, and spacer grid inspections consist of non-destructive examination techniques such as leak testing, weld inspection, and dimensional measurements. Process control precedures are described in detail. In addition, for any tests and inspections performed by other vendors on behalf of ENC, ENC reviews the quality control procedures and inspection plans to ensure that they are equivalent to those described in XN-NF-1A and are performed properly to meet all ENC requirements.

Based on the information provided in XN-NF-1A, we conclude that the new fuel testing and inspection program for the JP-BWR fuel is acceptable.

6.2 On-line Fuel System Monitoring

Routine on-line fuel rod failure monitoring is a matter that would be arranged with the licensees. It is not addressed in XN-NF-81-21.

6.3 Post-irradiation Surveillance

Routine poolside inspection of some discharged fuel assemblies is a matter that is normally arranged with the licensees. However, special surveillance related to the introduction of JP-BWR fuel has been arranged by ENC. ENC has made a commitment to implement such a post-irradiation surveillance program, as stated in Section 3.6 of XN-NF-81-21, in order to assess and monitor the performance of JP-BWR fuel. The surveillance program includes visual examination, such as underwater television and binocular scanning, and dimensional measurements of selected fuel assemblies. The removable upper tie plate feature will facilitate individual rod examination. In addition, ENC states that an extensive testing program has been conducted to verify the adequacy of the predicted fuel performance and the design bases in the Oyster Creek reactor.

This special post-irradiation surveillance program meets the SRP guidelines for introduction of a new fuel design and is, therefore, acceptable.

7.0 EVALUATION FINDINGS

The ENC jet-pump BWR fuel design described in XN-NF-81-21(P) has been reviewed in accordance with Section 4.2 of the Standard Review Plan (NUREG-0800). The staff concludes that, although most of the objectives of the fuel system safety review have been met, several issues must be addressed by a licensee proposing to use this fuel. They are listed in the following:

1. The licensee must confirm that the design power profile shown in Fig. 5.10 of XN-NF-81-21 bounds the power limits for the application in question.

2. Unless RODEX2 (presently under NRC review) is approved without modification, the licensee must confirm or redo the following analyses, which were reviewed on the basis of RODEX2 results.
   (a) Design Strain, SER Section 5.1(b).
   (b) External Corrosion, SER Section 5.1(e).
   (c) Rod Pressure, SER Section 5.1(h).
   (d) Overheating of Fuel Pellets, SER Section 5.2(d).
   (e) Pellet Cladding Interaction, SER Section 5.2(f).

3. Until such time that XN-NF-82-07 is approved and incorporated in the ENC ECCS evaluation model, a supplemental calculation using the NUREG-0630 cladding models must be provided on a plant-specific basis each time a new ECCS analysis is performed (see SER Sections 5.2(g) and 5.3(c)).

4. The licensee must make sure that the fuel performance code that is used to initialize Chapter 15 accident analyses has current NRC approval (see SER Section 5.3 (a)).

With the above provisions, the staff concludes that the JP-BWR fuel system has been designed such that (a) it will not be damaged as a result of normal operation and anticipated operational occurrences, (b) fuel damage during postulated accidents would not be severe enought to prevent control rod insertion when it is required, and (c) core coolability will always be maintained even after postulated accidents, thereby meeting the related requirements of the following regulations: 10 CFR 50.46; 10 CFR 50, Appendix A, General Design Criteria 10, 27, and 35; 10 CFR 50, Appendix K. This conclusion is based on two primary factors:

1. ENC has provided sufficient evidence that the design objectives will be met based on operating experience, prototype testing, and analytical predictions. Those analytical predictions dealing with fuel densification have been performed in accordance with methods that the staff has reviewed and found to be acceptable alternatives to NRC Regulatory Guide 1.126.

2. ENC has provided for testing and inspection of new fuel to ensure that it is within design tolerances at the time of core loading. On-line fuel failure detection and post-irradiation surveillance to detect anomalies or confirm that the fuel has performed as expected are the responsibility of the licensee.

The staff concludes that ENC has described methods of adequately predicting fuel rod failures during postulated accidents so that radioactivity releases are not underestimated, thereby meeting the related requirements of 10 CFR Part 100. In meeting those requirements, ENC has used the fission-product release assumptions of NRC Regulatory Guides 1.3 and 1.25.

On the basis of this review, we conclude that, with the above exceptions, all the requirements of the applicable regulations have been met and current regulatory positions have been followed for JP-BWR fuel design.

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