ANF-88-152(NP)(A)

ANF-88-152(NP)(A)
AMENDMENT 1

ANF-88-152(NP)(A)
SUPPLEMENT 1

ADVANCED NUCLEAR FUELS CORPORATION

GENERIC MECHANICAL DESIGN FOR ADVANCED NUCLEAR FUELS 9X9-5 BWR RELOAD FUEL

NOVEMBER 1990

A Siemens Company

ADVANCED NUCLEAR FUELS CORPORATION

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ANF-88-152(NP)(A) Supplement 1	NRC CORRESPONDENCE

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

August 28, 1990

Mr. R. A. Copeland, Manager
Reload Licensing
Advanced Nuclear Fuels Corporation
P. O. Box 130
Richland, WA 99352

Dear Mr. Copeland:

SUBJECT: ACCEPTANCE FOR REFERENCING OF TOPICAL REPORT ANF-88-152(P), AMENDMENT 1, "GENERIC MECHANICAL DESIGN FOR ADVANCED NUCLEAR FUELS 9x9-5 BWR RELOAD FUEL" (TAC NO. 71826)

We have completed our review of the subject topical report dated September 1989 together with response to request for additional information dated December 15, 1989. Based on our review we conclude that ANF-88-152(P), Amendment 1 provides an acceptable basis for ANF 9x9-5 BWR fuel mechanical design. The enclosure to this letter provides our Safety Evaluation Report (SER) which details the basis and limitations of our approval. Our evaluation applies only to matters described in the topical report.

In accordance with procedures established in NUREG-0390, it is requested that the Advanced Nuclear Fuels Corporation publish accepted versions of this topical report, proprietary and non-proprietary, within three months of receipt of this letter. The accepted versions shall include an "A" (designating accepted) following the report identification symbol.

Should our criteria or regulations change so that our conclusions as to the acceptability of the report are invalidated, Advanced Nuclear Fuels Corporation 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,

A C Thadani

Ashok C. Thadani, Director
Division of Systems Technology
Office of Nuclear Reactor Regulation

Enclosure:
ANF-88-152(P), Amendment 1
Evaluation

SAFETY EVALUATION OF ADVANCED NUCLEAR FUELS CORPORATION

TOPICAL REPORT ANF-88-152(P), AMENDMENT 1

"GENERIC MECHANICAL DESIGN FOR

ADVANCED NUCLEAR FUELS 9x9-5 BWR RELOAD FUEL"

JULY 1990

OFFICE OF NUCLEAR REACTOR REGULATION

U.S. NUCLEAR REGUALTORY COMMISSION

WASHINGTON, D.C. 20555

In order to assure that the above stated objectives are met and follow the format of Section 4.2 of the SRP, this review covers the following three major categories: 1) Fuel System Damage Mechanisms, which are most applicable to normal operation and AOOs; 2) Fuel Rod Failure Mechanisms, which apply to normal operation, AOOs, and postulated accidents; and 3) Fuel Coolability, which are applied to postulated accidents. Specific fuel damage or failure mechanisms are identified under each of these categories in Section 4.2 of the SRP. The ANF design limits and analysis methods for boiling water reactors (BWRs), previously approved by NRC (References 3 and 4), will be briefly discussed in this report under each fuel damage or failure mechanism along with the results of the 9x9-5 mechanical analyses.

Provided in this report is the review of the 9x9-5 mechanical design analyses in order to ensure that those analyses demonstrate that all NRC-approved ANF design bases and criteria are met for each fuel damage or failure mechanism defined in the SRP and that those analyses are or will be performed using NRC-approved analysis methods. ANF has requested that both 9x9-5 designs be approved for application to a peak nodal (pellet) burnup level of 55 MWd/kgM (References 1 and 2).

Pacific Northwest Laboratory (PNL) has acted as a consultant to the NRC in this review. As a result of the NRC staffs and their PNL consultants review of the topical reports, a list of questions were sent by the NRC to ANF requesting clarification of specific analyses (Reference 10). In addition, ANF was requested to provide updated results from postirradiation fuel examinations of 9x9 and 9x9-5 fuel assemblies that were not complete at the time of the topical report submittal and a list of scheduled 9x9-5 fuel examinations for the future. ANF has provided responses to these questions in Reference 11.

The 9x9-5 design description is briefly discussed in the following section (Section 2.0). The fuel damage and failure mechanisms and ANF analyses of these mechanisms are addressed in Sections 3.0 and 4.0, respectively, while fuel coolability is addressed in Section 5.0.

2.0 FUEL SYSTEM DESIGN

The 9x9-5 fuel assembly is similar to the earlier 9x9 design defined in Reference 3 with these principal differences: 1) an increase in the number of water rods, 2) two different fuel rod diameters, and 3) a decrease in the fuel-to-cladding gap size of the fuel rods. Therefore, the results of the mechanical analyses for the 9x9-5 design are, in most cases, similar to those obtained earlier for the 9x9 design (References 3 and 4).

The only differences between the two 9x9-5 designs in References 1 and 2 are in the position and the design of the water rods in the assembly. The design changes have been in the number and position of the flow holes in the water rods, and changes in the tube size and thickness of the spacer capture rod.

As noted earlier, these changes have not altered the results of the mechanical analyses between References 1 and 2, but there may be differences in thermal margin of CPR between these two designs. Unless otherwise stated, the following evaluations apply to the analyses presented in both References 1 and 2 for both 9x9-5 designs.

3.0 FUEL SYSTEM DAMAGE

The design criteria presented in this section should not be exceeded during normal operation, including AOOs. Under each damage mechanism, there is an evaluation of the analysis methods and analyses used by ANF to demonstrate that the design criteria are not exceeded during normal operation, including AOOs, for both 9x9-5 designs in References 1 and 2.

(a) Stress

Bases/Criteria - In keeping with the GDC 10 SAFDLs, fuel damage criteria should ensure that fuel system dimensions remain within operational tolerances and that functional capabilities are not reduced below those assumed in the safety analysis. The ANF design criteria for BWR fuel cladding stresses are presented in Table 4.1 of Reference 12 and in Table 3.3 of References 1 and 2. These criteria are consistent with Section III of the ASME Boiler and Pressure Vessel Code (Reference 13) and the guidelines established in Section 4.2 of the SRP (Reference 6).

The ANF design criteria for BWR fuel assembly component stresses are provided in Section 2.1 of Reference 12 and Section 3.1.1 of References 1 and 2. The ANF criteria for fuel assembly stresses are consistent with Section III of the ASME code and the guidelines established in Section 4.2 of the SRP.

The ANF stress criteria for fuel rod cladding and assembly components have been approved by NRC for previous BWR designs up to extended-burnup levels (References 3 and 4) and are also acceptable for the 9x9-5 design.

Evaluation - The ANF methods of analysis for evaluating fuel rod cladding and assembly steady-state stresses are discussed and approved for application to BWR fuel designs in References 3 and 4. These methods are also acceptable for the 9x9-5 design.

The results of the 9x9-5 cladding stress analyses are presented in Table 3.3 of References 1 and 2. The assembly stresses are quoted as "essentially the same as those in the standard ANF 9x9 design" that are provided in Reference 3. These results demonstrate that the cladding and assembly steady-state stresses are below the ANF stress criteria for these components and, therefore, are acceptable for the 9x9-5 design.

(b) Strain

Bases/Criteria - The ANF design criteria for fuel rod cladding strain is that maximum uniform hoop strain (elastic plus plastic) shall not exceed 1%. This criteria is intended to preclude excessive cladding deformation from normal operation and AOOs. This is the same criterion for cladding strain that is used in Section 4.2 of the SRP (Reference 6).

The material property that could have a significant impact on the cladding strain criterion at extended burnup levels is cladding ductility. The strain criterion could be impacted if cladding ductility were decreased, as a result of extended burnup operations, to a level that would allow cladding failure without the 1% cladding strain criterion being exceeded in the ANF analyses.

Recent cladding ductility data from the Babcock & Wilcox Fuel Company (BWFC) (References 14 and 15) and other sources (References 16 and 17) have shown that cladding ductility has decreased significantly at local burnup levels between 50 to 63 MWd/kgM. These data demonstrate that uniform cladding ductility values are decreasing with increasing burnup. However, at local burnup levels less than 55 MWd/kgM, the cladding has shown adequate uniform strains (elastic plus plastic) of 1% or greater. Therefore, we conclude that the ANF cladding strain criterion for the 9x9-5 design is acceptable up to the burnup levels requested in this topical report.

Evaluation - ANF has used the NRC-approved RODEX2A code (Reference 18) to calculate steady-state cladding strains for the 9x9-5 design during normal operati. The combination of the RODEX2 (Reference 19) and RAMPEX (Reference 20) codes has been used by ANF to calculate cladding strains for the 9x9-5 design during transient operation. The results of these calculations have shown that steady-state and transient cladding strains for the 9x9-5 design are below the 1% uniform strain limit. Therefore, we conclude that cladding strain is acceptable for the 9x9-5 design up to the burnup levels requested in these topical reports.

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

Bases/Criteria - The ANF design criterion for strain fatigue limits the total cumulative damage factor (CDF) to a conservative value below 1.0, which accounts for a corrosive environment and other fatigue mechanisms. ANF has used a fatigue design curve from O'Donnell and Langer (Reference 21) that includes a safety factor of two on stress amplitudes, or a safety factor of twenty on the number of cycles, whichever is more conservative for this calculation. This analysis method is consistent with the SRP guidelines and accounts for the additional fuel duty experienced by extended burnup operation and, therefore, has been acceptable for previous ANF BWR designs. This strain fatigue design criterion is also acceptable for application to the 9x9-5 design.

As noted in the Cladding Strain section, the material property that could have a significant effect on cladding strain and, thus, strain fatigue at extended burnups, is cladding ductility. However, as discussed above, fuel rods at the extended burnup levels requested for the 9x9-5 design have shown adequate cladding ductility and performance. We conclude that the proposed extended burnup operation does not reduce the applicability of the fatigue limits and, thus, the ANF strain fatigue limit is found acceptable for application to the 9x9-5 design up to the requested extended burnup levels.

Evaluation - The ANF methodology for determining strain fatigue is based on the use of the RODEX2 code (Reference 19), RAMPEX code (Reference 20), and the O'Donnell and Langer fatigue curve (Reference 21). The RODEX2 code is used to provide initial steady-state conditions for ANF transient and accident analysis. Consequently, the RODEX2 code provides input to the RAMPEX code for each power change, and RAMPEX provides stress amplitudes for the various power cycles. This methodology has been found to account for daily load follow and, thus, additional fatigue load cycles that may result from extended burnup operation. This methodology has been acceptable for previous BWR designs (References 3, 4, and 5) and is also acceptable for the 9x9-5 design up to the requested extended burnup levels.

ANF has performed strain fatigue calculations for the 9x9-5 design using the above analysis methods and the duty cycles summarized in Table 3.4 of References 1 and 2 for transient operation. The RODEX2 power history input for steady-state operation is provided in Figures 3.3 and 3.4 of References 1 and 2. The allowable number of cycles for a particular cycle loading was determined from the fatigue design curve of O'Donnell and Langer. The calculated CDF for the 9x9-5 fuel design up to the burnup levels requested is well below the design limit (Section 3.4.5 of References 1 and 2). Therefore, we conclude that cladding strain fatigue is acceptable for the 9x9-5 design up to the extended burnup levels requested in these topical reports.

(d) Fretting

Bases/Criteria - The ANF design basis for fretting wear is that fuel rod failures due to fretting shall not occur. Since the SRP does not provide numerical limits for fretting wear, and since ANF has addressed fretting wear in the design analysis, we conclude that this response to the SRP guidelines is acceptable.

Evaluation - ANF has indicated (References 1 and 2) that fretting wear is insignificant for their fuel designs at extended burnups because of the lack of any significant dependence between fretting wear and exposure time from both in-reactor and out-of-reactor tests on ANF fuel assemblies. In order to support this, ANF has indicated that examination of ANF assemblies irradiated to extended burnup levels have shown no significant wear. The out-of-reactor tests have shown that the residual spacer spring holding force can be quite low without resulting in fretting damage to the cladding. The 9x9-5 design has utilized the same spacer spring design and holding forces as those for previous ANF designs; therefore, ANF concludes that this design will also have very little fretting wear. From this, we conclude that ANF has satisfactorily demonstrated that fretting wear will be acceptable for the 9x9-5 fuel design.

(e) External Corrosion and Crud Buildup

Bases/Criteria - The ANF fuel design basis for cladding corrosion and crud buildup is to prevent 1) significant degradation of cladding strength, and 2) unacceptable temperature increases. Because of the thermal resistance of corrosion and crud layers, formation of these products on the cladding result in an elevation of temperature within the fuel as well as the cladding. ANF uses a cladding outer surface temperature limit for corrosion that is specified in Reference 5 for BWR fuel.

The BWR temperature limit has been acceptable for previous ANF BWR designs at extended burnup levels and is also acceptable at the extended burnup levels requested for the 9x9-5 design.

Evaluation - The RODEX2A code is used to calculate outside cladding temperature and oxide thickness. This code appears to significantly underpredict cladding corrosion based on comparisons to recently collected 9x9 corrosion data (Reference 11). However, these recent corrosion data (Reference 11) demonstrate that cladding corrosion is within acceptable limits for the 9x9 designs.

ANF has performed cladding corrosion calculations up to the extended burnup levels for the 9x9-5 design; however, the significance of these predictions is negated because, as noted above, the code significantly underpredicts corrosion. It should also be noted that cladding corrosion at the extended burnup levels requested has not been a problem for ANF BWR designs in the past. We conclude that cladding corrosion for the 9x9-5 design is acceptable based on the acceptable corrosion data from 9x9 fuel rods (Reference 11) near the burnup levels requested in this topical report; acceptance is not based on AN's analytical model. ANF is urged to update their analytical model used for predicting BWR cladding corrosion.

(f) Rod Bowing

Bases/Criteria - Fuel and burnable poison rod bowing is a phenomenon that alters the design pitch dimensions between adjacent rods. Bowing affects local nuclear power peaking and the local heat transfer to t.. coolant. Rather than placing design limits on the amount of bowing that is permitted, the effects of bowing are included in the cladding overheating analysis, Section 4.0(c) of this report, by limiting fuel rod powers when bowing exceeds a predetermined amount. ANF has established a rod-to-rod clearance limit below which a penalty is imposed by a reduction in the minimum critical power ratio (MCPR) and above which no reduction in MCPR is recessary. This approach is consistent with Section 4.2 of the SRP, and the NRC has approved this for application to current ANF BWR designs up to extended burnup levels (Reference 4). We conclude that this approach is also acceptable for the 9x9-5 design up to the requested extended burnup levels.

Evaluation - ANF has described their analytical model for rod bowing for the 9x9 fuel design in Reference 4. The generic form of this analytical model appears to account for the differences in rod bowing for different ANF designs, and bounds the upper 95% of the 9x9 rod bowing data at a 95% confidence level (Reference 11). It should also be noted that rod bowing appears to saturate at high burnups, i.e., >30 MWd/kgM (Reference 22). Therefore, the linear dependence with burnup of the ANF rod bow model should become even more conservative than the 95/95 bounding condition at extended burnup levels. In addition, ANF has provided in their response to questions (Reference 11), higher burnup rod bowing data for the 9x9 design that demonstrates that the 9x9 rod bowing model is more conservative at higher burnups. The methods used by ANF to account for the effects of fuel rod bowing have been reviewed and approved for the 9x9 fuel design up to extended burnup levels (Reference 4).

We conclude that this analytical model is also acceptable for application to the 9x9-5 design at the requested burnup levels.

The ANF rod bowing calculations for the 9x9-5 design show that there is a small MCPR penalty for the large diameter rods at relatively high burnup levels. ANF has noted that this MCPR penalty will have no practical constraint on actual reactor operation because the 9x9-5 fuel assemblies at these high burnup levels will result in fissile material burnout which will limit the rod power for these rods below the MCPR limit. We conclude that the results of the ANF fuel rod bowing calculations for the 9x9-5 design are acceptable.

(g) Axial Growth

Bases/Criteria - The ANF design criteria for axial irradiation growth is that the fuel rods must be properly engaged in the fuel assembly structure and the fuel assembly must be compatible with the fuel channel and fuel assembly supports in the reactor during the design lifetime. The concern for BWR assemblies is to maintain engagement between the fuel rod end cap shank and the assembly tie plates, i.e., to prevent fuel rod disengagement from the tie plates. The change in BWR rod-to-tie plate engagement (and possible disengagement) is due to the growth rate of the tie rods that connect the bottom and top tie plates being greater than the growth rate of the fuel rods. The above design criteria for axial growth is consistent with the SRP guidelines and, therefore, is acceptable for application to the 9x9-5 design.

Evaluation - The ANF analysis method for evaluating rod-to-tie plate engagement is based on fuel rod and assembly growth measurements at Big Rock Point, Oyster Creek, and Barsebeck from 8x8 fuel assemblies. The ANF analysis of rod-to-tie plate engagement for the 9x9-5 assembly has shown in Table 3.1 of References 1 and 2 that the rods remain properly engaged in the assembly tie plates at the requested extended burnup levels.

Assembly and rod growth measurements have recently been made on 9x9 Lead Test Assemblies (LTAs) up to within 15% of the maximum exposures requested for the 9x9-5 design (Reference 11). These axial growth data from the 9x9 LTAs have shown a substantially lower growth rate than those predicted by the ANF axial growth model for the 9x9-5 design. For example, the measured 9x9 LTA rod engagements were a factor of 3 greater than those engagements predicted for the 9x9-5 design at equivalent burnup levels, i.e., the measurements show a factor of 3 more engagement of the fuel rods than those predicted in the ANF analyses. This demonstrates that there is considerable conservatisms in the ANF analyses of rod-to-tie plate engagement, and provides assurance that rod-to-tie plate engagement and compatibility with reactor internals will be maintained for the 9x9-5 assembly up to the requested extended burnup levels. Therefore, we conclude that axial growth is acceptable for the 9x9-5 design.

(h) Rod Internal Pressure

Bases/Criteria - Rod internal pressure is a driving force for, rather than a direct mechanism of, fuel system damage that could contribute to the loss of dimensional stability and cladding integrity. Section 4.2 of the SRP presents a rod pressure limit that is sufficient to preclude fuel damage in this regard, and it has been widely used by the industry; it states that rod internal gas pressure should remain below the nominal system pressure during normal operation, unless otherwise justified. ANF has elected to justify limits other than those provided in the SRP. A proprietary limit above system pressure for fuel designs has been justified in Reference 5. In addition, ANF has imposed a second limit (Reference 5) that requires the fuel-cladding gap to remain closed during constant and increasing rod power operation under normal reactor operating conditions, when internal rod pressure exceeds the system pressure. These ANF limits for BWR fuel designs are presented in Supplement 4 of XN-NF-82-06(P)(A), Revision 1 (Reference 5) and have been reviewed and accepted by the NRC. We conclude that these limits are also acceptable for the ANF 9x9-5 fuel design at extended burnup levels.

Evaluation - The RODEX2A fuel performance code, with conservat v power histories, has been used by ANF to show that the 9x9-5 fuel design is within the ANF design limits. The RODEX2A code has been reviewed and found acceptable by the NRC (Reference 18) for the calculation of BWR rod internal pressures at extended burnup levels provided that it is used to calculate rod internal pressures at peak pellet burnups of 50 MWd/kgM or greater. The ANF methodology for determining the BWR power histories used as input to this code have also been reviewed and found acceptable by the NRC (Reference 5) for extended burnup applications. We conclude that both the RODEX2A code with its limit on burnup application and the power history methodology are also acceptable for application to the ANF 9x9-5 design up to the extended burnup levels requested in this review.

ANF has presented calculations (Figures 3.11 and 3.12 in References 1 and 2), using the RODEX2A code, with conservative power histories, to show that internal rod pressures for the 9x9-5 design at the extended burnup levels requested (peak pellet burnup >50 MWd/kgM) are significantly lower than the ANF rod pressure limits. Consequently, we conclude that internal rod pressures are acceptable for the 9x9-5 design up to the extended burnup levels requested in this review.

(i) <u>Assembly Liftoff</u>

Bases/Criteria - The guidelines in Section 4.2 of the SRP to prevent assembly liftoff are that worst-case hydraulic loads operation and AOOs should not exceed the hold-down capability of the fuel assembly (which includes wet weight and hold-down spring forces). ANF has stated (Reference 3) that "the assembly hold-down spring must retain its ability to counteract the hydraulic force through life." We consider this to be consistent with the above SRP requirement and has been previously approved by the NRC (References 3, 4, and 5) for ANF BWR designs. We also consider this to be acceptable for the 9x9-5 design.

Evaluation - ANF has calculated (Section 3.3.3 in References 1 and 2) that the weight of the 9x9-5 fuel assembly and channel are well in excess of the total worst-case hydraulic loads, including buoyancy and, thus, 9x9-5 fuel assembly liftoff will not occur during normal operations and AOOs. Therefore, we conclude that the 9x9-5 design is acceptable with respect to assembly liftoff forces up to the extended burnup levels requested in this review.

4.0 FUEL ROD FAILURE

Fuel rod failure thresholds and methods for analyzing the failure mechanisms listed in the SRP are reviewed in the following. When the failure thresholds are applied to normal operation including AOOs, they are used as limits (and hence SAFDLs) since fuel failure under those conditions should not occur according to GDC 10 (Reference 7). When the thresholds are used for postulated accidents, fuel failures are permitted, but they must be accounted for in the dose calculations required by 10 CFR 100 (Reference 8). The basis or reason for establishing these failure thresholds is, thus, established by GDC 10 and 10 CFR 100. The threshold values, and the methods used to assure that they are met, are reviewed in the following.

(a) Hydriding

Bases/Criteria - The release of hydrogenous impurities inside the fuel rod can result in premature cladding failure due to the formation of hydride blisters and reduced ductility. Hydriding, as a cladding failure mechanism, is precluded by controlling the level of moisture and other hydrogenous impurities during fuel pellet fabrication. The ANF fabrication limit (Reference 5) for total hydrogen in fuel pellets is more stringent than the ASTM limit cited in the SRP and, thus, is acceptable for application to the 9x9-5 design up to the extended burnup levels requested in this review.

Evaluation - The moisture and hydrogenous impurity level of ANF fuel pellets is determined by taking a statistical sample of the fabricated pellets and measuring total hydrogen content to ensure that it is below the ANF limit. Cladding failures due to excessive moisture in the fuel typically occur early-in-life. Because ANF has not experienced any significant fuel failures due to hydriding in past ANF fuel designs, this method of testing the impurity level of ANF fuel pellets is found to be acceptable for the 9x9-5 fuel design. We conclude that ANF has provided reasonable assurance that hydriding, as a fuel failure mechanism, will not be significant for the 9x9-5 design up to the extended burnup requested in this review.

(b) Cladding Collapse

Bases/Criteria - If axial gaps in the fuel pellet column were to occur due to fuel 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. ANF's design criteria for preventing cladding collapse is to maintain a radial gap large enough to prevent pellet hang up and, therefore, axial gap formation. This design criteria has been reviewed and accepted by the NRC in the review of XN-NF-82-06, Revision 1 (Reference 5). We conclude that ANFs design criteria for cladding collapse is also applicable to the 9x9-5 design.

Evaluation - ANF uses the approved RODEX2A and COLAPX codes (References 18 and 23) to predict cladding creep collapse. The RODEX2A code is used to provide initial in-reactor fuel rod conditions to COLAPX, e.g., radial fuel-cladding gap size, fill gas pressure, and cladding temperatures. The COLAPX code calculates cladding ovality changes (flattening) and creep deformation of the cladding as a function of time. As shown in Table 3.1 of References 1 and 2, the 9x9-5 design does not exceed their design limit for cladding creep collapse. Therefore, we conclude that the 9x9-5 design is acceptable with respect to cladding collapse up to the burnup levels requested in this review.

(c) Overheating of Cladding

Bases/Criteria - As indicated in the SRP, Section 4.2.II.A.2(d), it has been traditional practice to assume that failures will occur if the thermal margin criterion is violated. 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 criterion for ANF BWR fuel to prevent cladding overheating, is that transition boiling shall be prevented and this is accomplished by ANF by specifying a minimum CPR (MCPR) value. This ANF criterion satisfies the intent of the CPR criterion in Section 4.2 of the SRP and, thus, is acceptable for the 9x9-5 design.

Evaluation - ANF has not presented their analysis methodology nor specific analyses of MCPR in References 1 and 2; however, ANF has obtained approval for a separate topical report (Reference 24) that presents their critical heat flux (CHF) correlations used in the evaluation of MCPR for the 9x9-5 design with the two different water rod designs. It is noted that there are two separate critical heat flux correlations for the two different water rod designs. We conclude that ANF has adequately addressed the analysis methods for cladding overheating in Reference 24 for the 9x9-5 design.

(d) Overheating of Fuel Pellets

Bases/Criteria - As a second criterion for avoiding cladding failure due to overheating, ANF precludes fuel centerline melting for normal operation and AOOs. This design limit is the same as given in the SRP, Section 4.2.II.A.2(e), and, therefore, is acceptable for the 9x9-5 design.

Evaluation - ANF utilizes a correlation for the fuel melting point that accounts for the effects of burnup and gadolinia content. This fuel melting limit has been reviewed and approved by NRC (Reference 5) with respect to application to fuel and gadolinia bearing fuel at extended burnup levels. We conclude that this correlation is also acceptable for application to the 9x9-5 design.

ANF uses the RODEX2A computer code (Reference 18) to calculate maximum possible fuel centerline temperatures for normal operation with the conservative power histories presented in Figures 3.3 and 3.4 of References 1 and 2 as input. As shown in Figures 3.13 and 3.14 of these same reports, the calculated centerline temperatures for the 9x9-5 design remain below the irradiated UO_2 melting point.

For AOOs, ANF uses the RODEX2 (Reference 19) and RAMPEX (Reference 20) codes to calculate maximum possible fuel centerline temperatures with an LHGR history at least 120% greater than the steady-state LHGR history used for normal operation. The rod powers versus burnup curve in Figure 3.2 of References 1 and 2 represents ANFs bounding rod powers for 9x9-5 power transients from 100% power. ANF has indicated (References 1 and 2) that the results of the RODEX2 and RAMPEX calculations, using the power histories in Figures 3.2, 3.3, and 3.4 of References 1 and 2, have shown that fuel centerline melting will not occur for the 9x9-5 design. Based on the above analyses, we conclude that there is reasonable assurance that fuel pellet centerline melting will not occur in the 9x9-5 fuel during normal operation and AOOs up to the burnup levels requested in this review.

(e) Excessive Fuel Enthalpy

Bases/Criteria - The SRP guidelines for a severe reactivity initiated accident (RIA) in a BWR state in Section 4.2.II.A.2(f) that "at zero or low power, fuel failure is assumed 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 is primarily intended to address cladding overheating effects, but it also indirectly addresses pellet/cladding interactions (PCI) of the type associated with severe RIAs. ANF utilizes this SRP guideline for evaluating fuel failure due to excessive fuel enthalpy and, therefore, this is acceptable for the 9x9-5 design.

Evaluation - ANF performs a detailed analysis of the BWR control rod drop accident using the methodology presented in the NRC approved report XN-NF-80-19(P)(A), Volume 1 (Reference 25). We conclude that the ANF analysis method for evaluating fuel failures due to excessive fuel enthalpy from a control rod drop accident for zero power core conditions is acceptable for the 9x9-5 design.

(f) Pellet/Cladding Interaction

Bases/Criteria - The design criteria in Section 4.2.II.A.2(g) of the SRP for mitigating PCI fuel failures are: 1) cladding uniform strain shall not exceed 1% during any AOO, and 2) the fuel centerline temperature must remain below the melting point of the fuel. Both of these criteria are utilized by ANF for their BWR designs [see Sections 3.0(b) and 4.0(d) of this report] and, therefore, are acceptable for application to the 9x9-5 design.

Evaluation - As noted earlier in Sections 3.0(b) and 4.0(d) of this report, the 9x9-5 cladding strains and fuel centerline temperatures up to the extended burnup levels requested are well within the cladding strain and fuel melting limits. In addition, it should be noted that fuel at extended burnup levels will experience a reduction in peak power capability due to fissile material burnout that should help mitigate the effects of PCI for the 9x9-5 design at extended burnup levels.

Based on these considerations, we conclude that ANF has adequately addressed the effects of PCI for the 9x9-5 fuel design up to the extended burnups requested in this review.

(g) Cladding Rupture

Bases/Criteria - 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 criteria in the SRP 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 emergency core cooling system (ECCS) analysis. ANF models the effects of cladding rupture as an integral part of their ECCS evaluation model, as discussed in Sections 5.0(a) and (c) of this report. Therefore, we conclude that ANF has addressed this issue for the 9x9-5 design.

Evaluation - The ANF cladding deformation and rupture models are described in XN-NF-82-07(P), Revision 1 (Reference 26). The NRC has reviewed XN-NF-82-07(P), Revision 1, and concluded that the models are acceptable for use in LOCA analyses. The link between cladding deformation and rupture models to the LOCA ECCS analysis is described in the NRC approved report XN-NF-80-19(P)A, Volumes 2, 2A, 2B, and 2C (Reference 27). Because the cladding in the 9x9-5 design is the same as that used in past BWR designs, we conclude that the previously approved cladding deformation and rupture models are also acceptable for application to the 9x9-5 design.

(h) Fuel Rod Mechanical Fracturing

Bases/Criteria - 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 induced by LOCA-seismic events. The design bases and criteria for mechanical fracturing of ANF BWR reload fuel are presented in XN-NF-81-51(P)(A) (Reference 28), which describes ANF's LOCA-seismic structural response analysis. The design basis is that the channeled fuel assemblies must withstand the external loads due to earthquakes and postulated pipe breaks without fracturing the fuel rod cladding. The design limit proposed by ANF is that the stresses, due to postulated accidents in combination with the normal steady-state fuel rod stresses, should not exceed the stress limits given in Table 3.1 of Reference 28. The stress allowables are derived from the ASME Boiler and Pressure Vessel Code, Section III, Appendix F for faulted conditions. This design limit for mechanical fracturing has been reviewed and approved in the NRC review of XN-NF-81-51(P)(A) and they remain acceptable for application to the 9x9-5 design.

Evaluation - The mechanical fracturing analysis is done as a part of the seismic-LOCA loading analysis. A discussion of the seismic-LOCA loading analysis is given in Section 5.0(d) of this report.

5.0 FUEL COOLABILITY

For major accidents in which severe damage might occur, cor[illegible]olability must be maintained, as required by Appendix A to 10 CFR 50 (e.g., GDC 27 and 35). The following paragraphs review the criteria and analyses methods that ensure coolability is maintained for 9x9-5 design for those severe damage mechanisms listed in Section 4.2 of the SRP.

(a) Fragmentation of Embrittled Cladding

Bases/Criteria - The ANF design criteria for ECCS evaluation meet the requirements of 10 CFR 50.46 as it relates to cladding embrittlement for a LOCA; i.e., the criteria of a peak cladding temperature limit of 2200°F and a 17% limit on maximum cladding oxidation. We conclude that these criteria or limits are also acceptable for application to the 9x9-5 design.

Evaluation - The principal cause of cladding embrittlement during severe accidents such as LOCA is the high cladding temperatures that result in severe cladding oxidation. The ANF methodology for evaluating cladding embrittlement is included in their approved report for LOCA-ECCS analysis (Reference 27). We conclude that this methodology is also acceptable for licensing applications for the 9x9-5 design.

(b) Violent Expulsion of Fuel

Bases/Criteria - In a severe RIA, such as a PWR control rod ejection, or a BWR control rod drop accident, large and rapid deposition of energy in the fuel could result in melting, fragmentation, and dispersal of fuel. The mechanical action associated with fuel dispersal might be sufficient to destroy fuel cladding and the rod-bundle geometry and to provide significant pressure pulses in the primary system. In order to limit the effects of an RIA event, Regulatory Guide 1.77 (Reference 29) recommends that the radially-average energy deposition at the hottest axial location be restricted to less than 280 cal/g. This regulatory guide was originally written for PWRs but the NRC requires the same fuel enthalpy limit for a rod drop in a BWR. ANF uses this limit in the topical report XN-NF-80-19(P)(A), Volume 1 (Reference 25), that presents ANF's rod-drop accident analysis methodology. This is consistent with the SRP guidelines and, therefore, is also acceptable for licensing application to the 9x9-5 design.

Evaluation - Using the NRC-approved analysis methodology presented in XN-NF-80-19(P), Volume 1, ANF calculates a maximum radially-averaged fuel enthalpy for the control rod accident for each cycle in which ANF fuel is present in order to assure that the calculated enthalpy is well below the 280 cal/g limit. We conclude that these analysis methods for fuel enthalpy are also acceptable for licensing application to the 9x9-5 design at the requested burnup levels.

(c) Cladding Ballooning

Criteria/Bases - Zircaloy cladding will balloon (swell) under certain combinations of temperature, heating rate, and stress during the LOCA. There are no specific design limits associated with cladding ballooning, other than 10 CFR 50 Appendix K requirement that the degree of swelling not be underestimated.

Evaluation - The ANF cladding ballooning model is an integral part of the cladding rupture temperature model for the LOCA ECCS analysis. The cladding ballooning and rupture model is addressed in the XN-NF-82-07, Revision 1 (Reference 26). Reference 26 has adopted the NUREG-0630 (Reference 30) data base and modeling, which specifies a method acceptable to the NRC for treating cladding swelling and rupture during a LOCA. These models have been approved by the NRC up to extended burnup levels (Reference 5).

There is evidence that cladding oxidation at extended burnup levels and LOCA temperatures may result in reduced cladding strains compared to those predicted by NUREG-0630. However, these data are not conclusive because these tests were not performed with an oxidizing atmosphere, nor under irradiation conditions. Irrespective of whether these data are applicable to a LOCA, reduced cladding strains would result in less flow blockage and, thus, the current analysis methods would be more conservative with respect to this criterion. In addition, the high cladding temperatures associated with the LOCA analysis will anneal irradiation damage effects on cladding properties.

The RODEX2 fuel performance code (Reference 19) is used to provide burnup dependent input to the LOCA analysis, e.g., stored energy and rod pressures, that are a function of initial steady-state operation. This initial steady-state fuel condition is also important to cladding ballooning. As noted earlier [see Section 5.0(a)], the RODEX2 code has been approved to provide initial conditions for the LOCA analysis of ANF designs beyond the extended burnups requested in this review. We conclude that the ANF methodology for calculating cladding ballooning during a LOCA is acceptable for licensing application to the 9x9-5 design up to the requested extended burnups in this review.

(d) Fuel Assembly Structural Damage From External Forces

Criteria/Bases - Earthquakes and postulated pipe breaks in the reactor coolant system would result in external forces on the fuel assembly. The SRP, Section 4.2 and associated Appendix A, states that fuel system coolability should be maintained and that damage should not be so severe as to prevent control rod (for PWRs) or control blade (for BWRs) insertion when required during these low probability accidents. The ANF design basis is that the fuel assembly will maintain a geometry that is capable of being cooled under the worst case accident Condition IV event and that system damage is never so severe as to prevent control rod or control blade insertions. This is consistent with the design basis presented in the SRP and, therefore, acceptable for the 9x9-5 design.

In order to assure that these design bases are met, ANF has proposed design limits on the stresses that can be experienced by the critical fuel assembly components. These design limits are based on unirradiated yield and ultimate tensile strengths, and have been approved in XN-NF-84-97(P) (Reference 31). Because yield and ultimate tensile strengths will only increase with increased irradiation and burnup, and because the decrease in cladding ductility at the burnups requested is not limiting [as noted in Section 3.0(b)], these limits were also approved for extended burnup levels of current ANF designs (Reference 5). These limits are also acceptable for application to the 9x9-5 design at extended burnup levels requested in this topical report.

Evaluation - ANF has analyzed seismic-LOCA on 9x9-5 fuel assemblies using the approved methodology described in the approved XN-NF-86-60, Appendix B (Reference 32). Previously ANF has concluded that seismic-LOCA performance between 8x8 and 9x9 fuel assemblies had little difference because the same channel box was used for both fuel types (Reference 32). Based on this conclusion, ANF has evaluated the seismic-LOCA loads for 9x9-5 fuel assemblies among a mixed core of ANF 8x8 and GE 8x8 fuel and concluded that the 9x9-5 fuel is bounded by the previously approved seismic-LOCA analyses for 9x9 fuel (Reference 32). Since the approved ANF methodology (Reference 32) was used for this evaluation, we conclude that the seismic-LOCA loads for 9x9-5 fuel are adequately addressed.

6.0 CONCLUSIONS

We have reviewed the ANF 9x9-5 fuel design and mechanical design analyses described in References 1 and 2 in accordance with the SRP, Section 4.2. We conclude that the 9x9-5 design as described in References 1 and 2 is acceptable for licensing application to BWRs up to a peak nodal (pellet) burnup of 55 MWd/kgM.

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ADVANCED NUCLEAR FUELS CORPORATION

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GENERIC MECHANICAL DESIGN FOR

ADVANCED NUCLEAR FUELS 9X9-5 BWR RELOAD FUEL

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1.0 INTRODUCTION AND SUMMARY

1.1 Introduction

This report provides a design description and summary of the supporting analyses, and test results applicable to the mechanical design of the ANF 9x9-5 BWR fuel.

This report is similar to XN-NF-85-67(P)(A), Revision 1[1], which contains the mechanical description and the results of mechanical analysis for ANF 8x8 and 9x9 fuel.

The design criteria, technical bases, and a description of mechanical fuel rod failure mechanisms are covered by a separate report, XN-NF-85-39[2]. Mechanical design requirements specified in XN-NF-85-39 are discussed in detail in the design evaluation section of this report. In addition, design bases requirements from the NRC Standard Review Plan[3] which are applicable to the evaluation of this report are identified according to section.

1.2 Summary

Mechanical design analyses have been performed to evaluate cladding steady state strain and stresses, transient strain and stresses, fatigue damage, creep collapse, corrosion, hydrogen absorption, fuel rod internal pressure, pellet and clad temperatures, differential fuel rod growth, rod bow, and grid spacer spring design of the 9x9-5 fuel. These analyses assume the following maximum discharge exposure:

- MWd/MTU assembly exposure
- MWd/MTU rod exposure
- MWd/MTU nodal exposure

The analyses demonstrate that the mechanical design criteria are not violated when the fuel is operated under the LHGR (linear heat generation rate) limits defined in this report.

All analyses herein have been performed with the RODEX2[4] and RODEX2A) computer codes using the same methodology already used in Reference 1.

All the analyses have been performed on a generic basis using conservative input data and the enveloping histories defined in Figures 3.3 and 3.4.

1.2.1 Design Description Summary

The ANF 9x9-5 fuel assembly design consists of a 9x9 matrix containing 76 fuel rods and 5 water rods supported by bi-metallic spacer grids. The fuel assembly design may contain natural uranium axial blankets at either end of the fuel rods to enhance neutron economy, and will incorporate gadolinia-bearing fuel rods to provide fuel management flexibility.

The assemblies are designed to allow handling in the same manner, to the same extent, and with the same equipment as that now being used for 8x8 and 9x9 fuel.

The mechanical design of the 9x9-5 fuel established from the experience with existing ANF fuel designs.

Detailed fuel design drawings in Appendix A provide dimensional details of the 9x9-5 fuel assemblies.

1.2.2 Mechanical Design Summary

The major analysis results are as follows:

- The maximum end of life (EOL) steady state cladding strain is calculated to be below the design limit.
- Cladding steady state stresses are calculated below the material strength limits.
- The cladding strain during anticipated operating occurrences (AOO's) does not exceed .
- The maximum fuel rod internal rod pressure remains below ANF's criteria limit.
- The fuel centerline temperature remains below the melting point during AOO's.
- The cladding fatigue usage factor is within the design limit.
- Structural members have adequate strength to support handling and hydraulic loads.
- The cladding diameter reduction
- Evaluations of assembly growth and differential fuel rod growths show that the design provides adequate clearances for compatibility with the reactor internals, fuel assembly channel, and fuel handling machine. Also, there is adequate engagement of the end caps in the upper tie plate and lower tie plate throughout the design life.
- The initial fuel rod design spacing is expected to be adequate to accommodate expected rod to rod gap closure for the fuel design life.
- The maximum EOL reduction in cladding thickness due to corrosion and the maximum concentration of hydrogen in the cladding are calculated to be well within the design limits.
- The fuel rod plenum spring and other miscellaneous components are shown to meet the respective design bases.
- The spacer springs meet all the design requirements and can accommodate the expected relaxation at the respective EOL exposures.

2.0 DESIGN DESCRIPTION

2.1 Fuel Assembly

The ANF 9x9-5 reload fuel assembly design is a 9x9 array with 76 enriched uranium fuel rods. The remainder are inert water rods. Eight of the fueled rods are also tie rods. Some of the rods contain gadolinia as a burnable absorber. Fuel rod pitch is maintained by seven spacers. The spacers are a The centrally located inert water rod captures the spacers to maintain the proper axial spacing.

All rods except for the tie rods have coil compression springs located between the top of the fuel rods and the bottom surface of the upper tie plate. These compression springs provide a force to aid in seating the fuel rods in the lower tie plate and react against the upper tie plate. The springs accommodate variations in rod lengths arising from manufacturing tolerances and permit axially non-uniform thermal and irradiation induced growth of the fuel rods.

The eight tie rods are structural members of the fuel assembly and establish the overall assembly length. These rods are threaded into the lower tie plate and latch into the upper tie plate. The tie rods carry the assembly weight during handling and provide the coil spring reaction support.

The assembly contains five water rods to improve uranium utilization.

For fuel rod removal, the upper tie plate must be depressed against the compression springs a short distance in order to allow the locking lugs to be rotated 90°. The upper tie plate is then free to be removed for fuel rod extraction or replacement.

The lower tie plate consists of a machined stainless steel casting with a grid plate for lower end cap engagement and a lower nozzle to distribute coolant to the assembly.

The upper tie plate is a cast and machined grid plate, with attached bail handle to provide for fuel assembly handling and orientation.

Assembly and component descriptions for the 9x9-5 fuel are presented in Table 2.1. Table 2.2 show the 9x9-5 geometric design parameters used for inputs to the analysis codes.

Detailed fuel design drawings in Appendix A provide dimensional details of fuel assemblies.

2.2 Rods

2.2.1 Fuel Rods

The fuel rods consist of UO_2 pellets in Zircaloy-2 tubing with Zircaloy-2 end cap plugs fusion welded on both ends of the tube.

The fuel rod cladding is

Each standard fuel rod contains a column of fuel pellets ranging from 144.0 to 150.0 inches in length (dependent upon application). The fuel column contains enriched UO_2 and may also contain natural uranium column segments on the top and/or bottom of the enriched column. The enriched column may be anywhere from about 132 to 150 inches in length.

The pellets are sintered to of theoretical density. The length to diameter ratio (L/D) of the enriched pellet is

The nominal pellet to cladding gap is

The fuel rod upper plenum contains an coil spring to prevent fuel column separation during fabrication, shipping, and early reactor operation.

The upper end cap plug is configured to allow remote under water handling and engagement into the upper tie plate. The lower end cap has a shaft and a tapered section which seats in the lower tie plate, and aids in fuel rod insertion into the fuel assembly during fabrication, inspection, and reconstruction. Both end caps are seal welded to the cladding in a helium atmosphere. The rod is pressurized with helium to atmospheres.

Fuel rod identification is maintained by a serial number on the lower end cap.

Fuel assembly component description for the 9x9-5 fuel is included in Table 2.1.

2.2.2 Tie Rods

The tie rod assembly serves to connect the upper and lower tie plates.

The tie rods are fueled and have upper and lower end caps designed for attachment to the tie plates. The lower end cap threads into the lower tie plate. The upper end cap is also threaded for the engagement of the tie plate locking hardware.

The description of the tie rod assemblies are provided in Table 2.1.

2.2.3 Water Rods

water rods are located in the fuel assembly. The water-filled rods are

The tubing and end fittings are made of

2.2.4 Spacer Capture Rod

The spacer capture rod (SCR) is a water rod. The lower end cap of the SCR is threaded and it connects to the lower tie plate.

The SCR has rings on the outside of the SCR at axial locations corresponding to the spacer locations.

2.3 Tie Plates

2.3.1 Upper Tie Plate

The upper tie plate serves as an easily removable structural member of the fuel assembly. The eight tie rods penetrate the upper tie plate to hold it in place.

A lifting bail is integral with the grid for fuel handling.

Also, there are posts on top of the grid which support and permit attachment of the channel.

The upper tie plate is made of cast stainless steel

2.3.2 Lower Tie Plate Assembly

The lower tie plate supports the fuel rods during handling and operation and distributes the coolant into the assembly. The eight tie rods thread into the lower tie plate forming a structural tie between the upper and lower tie plate.

seal springs
limit coolant bypass flow between the channel and tie plate.

The lower tie plate is made of cast stainless steel.

2.4 Spacer Grids

The spacer grids are an interlocking square array of strips producing a 9x9 array of cells. spring strips are mechanically secured within the structural strips. . strips which capture the springs are welded to each other at all intersections and to the side plates. The strips are dimpled and the springs are arranged such that each fuel rod is positioned by four support dimples and one spring. Backup lobes are provided on the spring to prevent excessive spring and cladding stresses which might occur under adverse handling conditions. Anti-hangup tabs are on the side plates of the spacers to prevent interference during channeling.

2.5 Miscellaneous Hardware

2.5.1 Compression Spring

The compression springs are located on the fuel rod and inert rod upper end cap shanks between the fuel rod end cap shoulder and the upper tie plate. The spring force supports the weight of the tie plate and channel, and aids in seating the rods into the lower tie plate. The spring dimensions are designed to account for manufacturing tolerances and differential rod growth.

was selected as the spring material.

2.5.2 Assembly Hardware

The retaining spring, locking sleeve, and the adjusting nut are assembled to the upper end cap of the tie rods. The purpose is to secure the upper tie plate to the tie rods.

The retaining spring maintains the correct location of the locking sleeve when the upper tie plate is depressed for installation or removal.

The locking sleeve rotates on the tie rod upper end cap to lock onto the upper tie plate. The adjusting nut threads onto the tie rod upper end cap to fasten the retaining spring and locking sleeve to the tie rod. The threads at the adjusting nut location are deformed to prevent further rotation of the nut after assembly.

The retaining spring is made of and the locking sleeve and adjusting nut are made of low-carbon stainless steel.

TABLE 2.1 FUEL ASSEMBLY COMPONENT DESCRIPTION

	Material	Characteristic 9x9-5
Fuel Assembly		
Array	....	9x9
Width, in.	....	5.149
Length, in.	....	176.014
No. of Spacers	....	7
Rod Pitch, in.	....	0.563 - 0.572
No. of Fuel Rods	....	75
No. of Water Rods	....	5
Fuel Rod Assembly		
Outside Diameter, in.		0.417
Plenum Length, in.		
Fuel Length, in.		145.24 - 150.00
Pressure, psig		30.0
Fuel Rod Assembly		
Outside Diameter, in.		0.443
Fuel Length, in.		150.00
Pressure, psig		30.0
Spacer Capture Rod		
Sleeves		
Outside Diameter, in.		0.546 (BWR-6)
Water Rod		
Outside Diameter, in.		0.546 (BWR-6)
Inside Diameter, in.		
Cladding		
Outside Diameter, in.		0.417
Cladding		
Outside Diameter, in.		0.443

TABLE 2.1 FUEL ASSEMBLY COMPONENT DESCRIPTION (CONTINUED)

	Material	Characteristic 9x9-5
Plenum Spring		
Coil Diameter, in.		
Wire Diameter, in.		
Free Length, in.		
Plenum Spring		
Coil Diameter, in.		
Wire Diameter, in.		
Free Length, in.		
End Caps		
Standard Upper		
Length, in.		
Shank Diameter, in.		
Standard Lower		
Length, in.		
Shank Diameter, in.		
Tie Rod Upper		
Length, in.		
Thread, mm		
Tie Rod Lower		
Length, in.		
Thread, mm		
SCR/WR Upper		
Length, in.		
Shank Diameter, in.		
SCR/WR Lower		
Length, in.		
Thread, mm		

TABLE 2.1 FUEL ASSEMBLY COMPONENT DESCRIPTION (CONTINUED)

	Material	Characteristic 9x9-5
Fuel Pellets		
UO_2	Sintered UO_2	
Diameter, in.		
Length, in.		
Density, %TD		
Dish, %		
Gadolinia	UO_2-Gd_2O_3	
Diameter, in.		
Length, in.		
Density, %TD		
Dish, %		
Natural	Sintered UO_2	
Diameter, in.		
Length, in.		
Density, %TD		
Dish, %		
Compression Spring		
Coil Diameter, in.		
Wire Diameter, in.		
Free Length, in.		
Upper Tie Plate		
Height		
Outside Dimension, in.		
Lower Tie Plate Assembly		
Outside Dimension, in.		
Lower Tie Plate Seal		
Height, in.		
Width, in.		

TABLE 2.1 FUEL ASSEMBLY COMPONENT DESCRIPTION (CONTINUED)

	Material	Characteristic 9x9-5
Locking Sleeve		
Height, in.		
Length, in.		
Width, in.		
Adjusting Nut		
Length, in.		
Diameter, in.		
Thread		
Retaining Spring		
Coil Diameter, in.		
Wire Diameter, in.		
Free Length, in.		

Pages 15-20 have been deleted.

3.0 DESIGN EVALUATION

ANF 9x9-5 reload fuel assembly components are designed to satisfy the performance and safety objectives described by XN-NF-85-39, Rev. 0, the criteria presented in this section, and the NRC Standard Review Plan[3], Section 4.2. The reactor fuel system 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 as to prevent control rod insertion when 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 internals to achieve acceptable flow distribution, including bypass flow, such that heat transfer requirements are met for all licensed modes of operation.

3.1 <u>Design Experience and Prototype Testing (Standard Review Plan Section 4.2 II.C)</u>

ANF has fabricated large quantities of JP-BWR 8x8 and 9x9 fuel.

Irradiation experience to date has shown that the fuel performs satisfactorily. Fuel fabrication and irradiation experience for both BWR and PWR designs is summarized in Table 3.6.

The 9x9-5 is an evolutionary design from the ANF 9x9 being operated by several U.S. reactors. The experience gained in the design, manufacturing, and irradiation of over BWR fuel assemblies by ANF has been applied to the 9x9-5 design. The features incorporated in the 9x9-5 do not represent a significant departure from other designs, consequently, this experience is applicable to the 9x9-5 design.

3.1.1 Fuel Assembly Structural Strength

Tie rods, upper tie plates and lower tie plates, and the upper tie plate locking hardware constitute the fuel assembly structural components during handling. In order to withstand expected handling loads, the assembly is designed to withstand a minimum axial load of the bundle weight with no permanent deformation. Also, the tie rod upper end caps are designed to withstand a loading of not less than Should the assembly be subjected to unexpected and excessive loads, failure is to occur at the tie rod end caps without breaching the cladding.

The tie rod upper end cap, upper and lower tie plates in the 9x9-5 design, are essentially the same as those in the standard ANF 9x9 design. The test results reported in Reference 1 are also applicable to the 9x9-5 design.

3.1.2 Hydraulic Performance Tests

Single phase hydraulic characteristics of the 9x9-5 fuel assemblies were experimentally determined[26,27] by hydraulic tests performed in ANF's Portable Hydraulic Test Facility. The testing was performed on full size components to find the loss coefficients of the lower tie plate (including the inlet hardware), spacers, rods, and the upper tie plates.

3.2 Fretting Wear (Standard Review Plan Section 4.2 IIA1(c))

Fretting wear of fuel rods under normal operating reactor conditions can be controlled by the use of design features that assure that the rods are positively supported at the grid spacers throughout the expected irradiation period. In the case of ANF fuel, this is accomplished by the spring arrangement of the spacer grid. The system in the grid spacer is designed such that the minimum rod contact forces throughout the design life are greater than the maximum fuel rod flow vibration forces, thus preventing rod fretting wear.

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

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

The effective spacer spring relaxation, based on this and other data, follow an asymptotic relationship with burnup. For typical ANF rod and spacer springs irradiated to MWd/MTU, the average spring force is approximately of the initial spring force. The spring force at the top and bottom grids is at least of the initial spring force. The residual spring force therefore has a substantial margin for the prevention of fretting wear during extended burnup.

ANF laboratory testing has shown that the residual spacer spring holding force can be quite low without resulting in fretting damage to the cladding. Extensive flow tests have been performed on ANF assemblies under various spacer spring load conditions. These tests have covered the range of no spring relaxation (i.e., reload corresponding to typically spring deflection) to total relaxation (zero preload). In testing to up to hours duration, no measurable fretting wear resulted from up to relaxation, provided there was contact between the spacer spring and the fuel rod. In these tests, fuel rod wear depths at spacer contact points has typically ranged from although wear of up to approximately in depth has been occasionally observed. Examination indicates that the wear is due primarily to fuel rod loading and unloading and not fuel rod motion during the testing.

The ANF assembly design methodology requires the consideration of flow vibration and irradiation effects on the spacer rod interaction. Flow tests and the fuel assembly irradiation have demonstrated the success of this approach. In addition, the methodology for the calculation of fuel rod stresses and strains requires the use of minimum clad wall thickness. This choice of input provides conservatism in the analysis which compensates for the minimal anticipated fuel rod wear.

3.3 Fuel Assembly Analysis

3.3.1 Differential Fuel Rod Growth and Assembly Growth (Standard Review Plan Section 4.2, IIA1(e))

Axial extension of Zircaloy fuel rods is primarily due to irradiation induced growth. Growth is also related to pellet-to-cladding interaction. Generally, higher growth is experienced by the tie rods. The tie rods undergo creep elongation in addition to irradiation growth because of tensile loads exerted between the upper and lower tie plate by the compression springs and fuel rods. The standard fuel rods must have sufficient engagement into the upper tie plate. The upper end cap shanks of the fuel rods must be long enough so disengagement from the tie plate does not occur througnout the assembly design life.

Fuel assembly growth is a direct result of the tie rod growth. The EOL assemblies growth are reported in Table 3.1. The values were calculated based on measurements of fuel rod engagement at Big Rock Point[23], on Oyster Creek fuel[20], and differential fuel rod/assembly length measurements at the Barsebeck reactor[24,28]. These growths do not result in any operational limitations and do not cause interferences with the surrounding reactor internals.

The maximum differential growth for the assemblies at EOL are shown in Table 3.1. Nominal engagement of the standard fuel rod end cap into the upper tie plate is also shown in Table 3.1.

Adding the maximum manufacturing tolerance to the measured differential growth ind cates that remains engaged in the throughout the expected lifetime.

3.3.2 Channel and Fuel Handling Compatibility

The differential growth between the channel and fuel assembly is important so the assembly and channel can be properly grappled. Growth between the channel and assembly has been evaluated at the BOL and EOL when fuel is at its extreme length. The fuel assembly length and is compatible with the channel throughout the life of the fuel.

3.3.3 Fuel Assembly Hydraulic Liftoff (Standard Review Plan Section 4.2 A1(g))

The hydraulic loads on a fuel assembly are calculated to be less than the weight of an assembly in water.

Table 3.1 summarizes the results of the hydraulic analysis performed to demonstrate the bundle's resistance to hydraulic liftoff during normal and anticipated operating modes. The maximum liftoff force includes buoyancy and pressure effects. The results show that the gravitational forces on the fuel are sufficient to prevent hydraulic liftoff under normal and anticipated operating conditions.

3.4 Fuel Rod Analysis

The following sections detail analyses performed to predict performance characteristics of the 9x9-5 fuel rods. The analysis have been performed with RAMPEX and approved versions of RODEX2, RODEX2A and COLAPX codes.

The fuel rod performance characteristics modeled by the RODEX2 and RODEX2A codes are:

- Gas release
- Radial thermal conduction 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

RODEX2 has been used to determine the initial conditions for fuel rod power ramping analysis (RAMPEX)[6]. RODEX2A has been used to determine the steady state strain, internal pressure, fuel cladding temperature, corrosion, hydrogen absorption fuel temperature, internal pressure, and the fuel rod internal pressures for cladding creep collapse (COLAPX)[7] analyses. Pellet density, swelling, densification, and fission gas release models, and cladding and pellet diameter are used in RODEX2 and RODEX2A to provide conservative subsequent calculation results.

The 9x9-5 geometric design parameters used for input conditions into RODEX2, RAMPEX, and COLAPX are presented in Table 2.2.

Table 3.2[4] provides the guidelines for selecting input variables from Table 2.2 to be used in specific applications of the RODEX2 code. The analyses performed with RODEX2A followed the same guidelines.

Gadolinia fuel rods are included in the evaluation. However, due to the designed lower enrichment of the gadolinia fuel, these are not the limiting rods in the fuel assembly.

3.4.1 Linear Heat Generation Rate Limits

Figure 3.1 is the LHGR limits used in the steady state fuel rod performance evaluation. These limits are for the steady state power maneuvering of ANF fuel.

Figure 3.2 is the limit to protect against power transients during anticipated operational occurrences (AOO's). This limit is considered in the transient mechanical analyses and applies to the 9x9-5 design. The first four figures are presented in terms of peak pellet power versus assembly planar exposure.

For evaluation of the LHGR limits, Figures 3.1 and 3.2 are conservatively translated into RODEX2 and RODEX2A inputs in the form of fuel rod nodal powers and time. Figures 3.3 and 3.4 show the respective power inputs versus exposure for a fuel rod. Neutronic analysis indicate that the small diameter rods will never operate to more than power of the most limiting rod in the assembly. Consequently, the input power history has been modified to reflect this result. These figures are in terms of pellet power versus pellet exposure.

This assumption leads to conservatively high rod average powers in evaluation of the LHGR limits. This assumption on axial peaking is identical to that assumed in the Reference 1 analysis.

The performance of the fuel rods have been analyzed to peak pellet exposures of for both fuel rod designs. This exposure is consistent with the maximum assembly exposure of

3.4.2 Steady State Strain, Hydrogen Absorption and Corrosion (Standard Review Plan Section 4.2, IIA1(d))

Calculations to determine cladding end of life strain, hydrogen absorption, and corrosion are performed using the RODEX2A computer code. The power histories are presented in Figures 3.3 and 3.4. The fuel rod design input parameters for this case are selected from Table 2.2 to provide the most conservative results with respect to strain, corrosion, and hydrogen absorption. Therefore, the tolerances used are those which correspond to

<u>Steady State EOL Strain</u>

Results of the analysis to determine cladding strain are obtained from the RODEX2A output. See Figures 3.5 and 3.6 for plots of the results. The cladding strain at end of life is well within the design criteria limit of 1.0%.

<u>Hydrogen Absorption</u>

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 440°F to 751°F) (227°C to 400°C) is:

$$H = H(O) + H(I) + H(C)$$

where:

- H = Net weight fraction of hydrogen in cladding (ppm)
- H(O) = Initial concentration of hydrogen in the cladding due to impurities introduced during cladding manufacturing and autoclaving
- H(I) = Concentration of hydrogen in the cladding due to internal sources such as the fuel (ppm)
- H(C) = Concentration of hydrogen in the cladding due to external sources such as absorption of hydrogen from the coolant (ppm

The primary consideration in determining H is the determination of H(C), which is evaluated using the methodology in Reference 19.

The internal source of hydrogen is from the fuel and is approximated to

The total concentration of hydrogen in the cladding is calculated from RODEX2A to be within the design criteria limits. See Figures 3.7 and 3.8 for the calculated hydrogen concentration versus time. The initial sources of hydrogen are also included as discussed above.

Cladding Corrosion

Cladding corrosion is calculated using the correlations from MATPRO[25]. The power histories in Figures 3.3 and 3.4 were used for the analysis.

The cladding temperature limit evaluated under Section 3.4.8 provides further assurance that the corrosion performance is acceptable.

The metal loss, calculated at the maximum exposure is always below the design criteria limit. See plotted results in Figures 3.9 and 3.10.

The conservative aspects of the calculation account for the additional adverse effects of crud deposition by defining a crud layer thickness in the input to the calculation. Crud deposition will increase the cladding surface temperature producing a higher corrosion rate.

3.4.3 Cladding Steady State Stress (Standard Review Plan Section 4.2, II.A.1(a))

The results of the steady state stress analysis and the appropriate stress limits are summarized in Table 3.3.

Each individual stress is calculated inside and outside the cladding and at both midspan and spacer level. The applicable stresses at each level are then combined to get the maximum stress intensities. The analysis is performed at BOL and EOL and at cold and hot conditions. The stress analysis assumes

Primary Stresses

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

Primary Bending Stresses

Bending stresses due to ovality are calculated with Timoshenko's equation[11].

Secondary Stresses

Cladding thermal gradient stress fuel rods operate with a temperature gradient across the cladding wall which may result in significant thermal stresses.
by(11):

Restrained Thermal Bow

Fuel rod bowing caused by a thermal gradient and restrained from bowing by the spacers is calculated

Restrained Mechanical Bow

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

Flow Induced Vibration

Vibrational stresses due to flow induced vibrations are calculated with

Contact Stress From Spacer Springs

The contact stresses at the spring locations are calculated using the finite element method. Calculations are performed with the ANSYS[15] general purpose finite element code. The circumferential and axial stresses induced by the contact load are incorporated in Table 3.3.

Fuel Rod End Cap

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 indicates the results are acceptable. The results of the analysis are presented in Table 3.4.

The ANSYS code[15], which allows thermal as well as stress analyses, is used to model the subject rod region. The problem is solved by a thermal pass and a stress pass, where the stress analysis uses the results of the thermal analysis as a part of the input.

3.4.4 Transient Occurrences (Standard Review Plan Section 4.2, IIA2(g))

RODEX2 and RAMPEX are used to evaluate the transient cladding strains. Per Section IIA2(g) of the NRC Standard Review Plan Section 4.2, the following is applicable, "The uniform strain of the cladding should not exceed 1%. In this context, uniform strain (elastic and inelastic) is defined as transient induced deformation". During a transient, the fuel centerline temperature must remain below the melting point of the fuel.

The RAMPEX[6] code is used for the determination of cladding transient strain. For input in each RAMPEX case, the RODEX2 output at a particular exposure is used. This RAMPEX input includes gas release, fuel densification, fuel swelling, and fuel relocation due to pellet cracking which all depends on the prior fuel operating history. A description of the ramp rates and powers are additional input to RAMPEX. The ramp rate considered is 125. kW/ft/hr in all of the analyses. Conservative assumptions were employed in selecting the initial conditions for ramping.

In all cases, the uniform cladding strain did not exceed nor was the reached at or below the LHGR curve in Figure 3.2.

3.4.5 <u>Cladding Cyclic Fatigue (Standard Review Plan Section 4.2, IIA1(b))</u>

Fuel shuffling and reactor power maneuvering will impose a repeated loading on the fuel rod cladding. In addition to the stress analysis for the maximum stress, a fatigue analysis is performed to account for the cyclic pattern of stresses. The RAMPEX[6] code is used to calculate the cyclic stresses.

Loading cycles assumed for the analysis are listed in Table 3.5. These duty cycles are expected to encompass the normal reactor operation over the design life of the fuel.

Input for the RAMPEX code is obtained from RODEX2 using the limiting power histories in Figures 3.3 and 3.4. The input to RAMPEX is selected to simulate each of the various duty cycles. The duty cycles are evaluated at different exposure points to account for burnup effects.

For each duty cycle, a maximum cyclic stress is selected from among the multiple RAMPEX cases run for the power histories. From this cyclic stress, an allowable number of cycles is determined from a S-N (stress versus number of cycles) curve. The cladding fatigue usage factor for each duty cycle is then calculated as the number of expected cycles divided by the number of allowable cycles. The total cladding usage factor is then figured as the sum of the individual usage factors for each duty cycle.

3.4.6 Cladding Creep Collapse (Standard Review Plan Section 4.2, IIA2(b))

The potential for cladding creep collapse is evaluated using RODEX2A and the COLAPX[8] codes.

The evaluation verifies that this criterion is met[17,18,19]. A bounding power history from that used for the steady state strain calculation was used in the analysis. The minimum initial cold gap and the cold gap at 6,000

3.4.7 Fuel Rod Internal Pressure (Standard Review Plan Section 4.2, IIA1(f))

Calculation of the fuel rod internal pressure is done with RODEX2A[5] on a generic basis. RODEX2A is an approved revision of the NRC-approved RODEX2[4] code. The revision has been prepared to more closely predict recently obtained data. In particular, RODEX2A has been benchmarked against fission gas release data from high burnup ramping programs.

In order to protect against fuel rod failure, the internal pressure acceptance criterion limits the fuel rod internal pressure to 800 psi above

3.4.8 Fuel and Cladding Temperatures (Standard Review Plan Section 4.2, IIA2(e), IIA2(d))

Fuel Temperature

The fuel centerline temperature is calculated as in the preceeding section on internal pressure. Figures 3.13 and 3.14 represent the maximum calculated temperatures for the 9x9-5 power history inputs.

Cladding Temperature

Cladding temperatures are calculated using the RODEX2A code. The input conditions are the same as described for the internal pressure calculations. The power history used are described in Figures 3.3 and 3.4. The results indicate that the design criteria for cladding temperature are met. See Figures 3.15 and 3.16 for the cladding temperatures versus expocures.

3.4.9 Fuel Rod Spacing and Rod Bow (Standard Review Plan Section 4.2, IIA1(e))

Rod-to-rod and rod-to-channel spacing must be maintained and rod bow must be limited so that no reduction in MCPR is incurred.

Measurements in gap spacing in irradiated fuel have been collected by ANF 7x7 and 8x8 fuel at burnups in excess of (assembly average)[20]. Also, 9x9 measurements have been obtained after one cycle of irradiation

The statistical derivation of rod bow from gap spacing data is given in Reference 21. The correlation in the data base has been modified in accordance to NRC requirements to include cold-to-hot and batch-to-batch variations.

The calculated minimum rod-to-rod spacing are listed in Table 3.1. These values are calculated at the design peak assembly exposure.

3.4.10 Fuel Rod Plenum Spring

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

The nominal spring force is shown in Table 3.1. This force is exerted by the spring on the fuel column. This load is greater than the fuel column weight 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 during initial fuel densification is expected to be less than 10%. The plenum spring design considers the relaxation effect of autoclaving the fuel rods.

3.4.11 Water Rod Design

The water rods for the 9x9-5 design are

The forces or stresses which act on the water rod are:

- Differential pressure across the tube wall
- Compressive axial force due to the compression spring
- Restrained mechanical bow
- Restrained thermal bow
- Flow-induced vibrations
- Radial temperature gradient
- Axial friction force with spacers
- Flow friction

For the SCR, the force due to the flow loss across the spacers and the axial force on the spacers from fuel rod thermal expansion and growth must also be considered.

The stresses due to restrained mechanical bow are assumed to occur when a water rod (or SCR), which is bowed within the specifications limits, is restrained from bowing when assembled into the bundle. The same assumptions are made for restrained thermal bow, except the tendency to bow is caused by a differential temperature across the diameter of the tube. These stresses and the stresses from flow induced vibrations are calculated in the same manner as described for the fuel rods in Section 3.4.2. The stresses from restrained bow and flow induced vibrations are low. This will be shown below in the calculation results.

The design incorporates a small gap between the spacer and the spacer capture rod connector to minimize stresses due to differentil thermal expansion at BOL. At EOL it is assumed that the relaxation of the spacer springs allows the spacer to slide axially and contact the spacer capture rod connector. Using an expected relaxation of 85.%, the total force on the spacers at the end of life is calculated to be 60. pounds.

Wall thinning o' 'he water rods and the SCR is calculated using the same corrosion model as i. RODEX2A.

The water rod connection to the lower tie plate has been analyzed for maximum stress. The loads applied to this connection result from pressure drop across the spacers and friction between the fuel rods and the spacer cell.

The analysis results indicate that the water rod, spacer capture rod, and spacer capture rod lower tie plate connector operate well within the elastic stress limits.

3.5 Fuel Assembly Component Evaluation

3.5.1 Grid Spacers

Load Deflection

The load deflection characteristics are determined from the Parts List requirements. Due to spacer cell and fuel rod diameter tolerance stackup, the spring deflection and BOL spring force are reported as ranges in Table 3.1.

Support Stiffness

The support stiffness required to force a node at a support level is generally considered to be five times the simple span stiffness. That is:

where:

This condition is easily met as the support dimples are very stiff.

The dimple stiffness, nominal spring rate, the resulting support stiffness, and the minimum required support stiffness (K(MIN)) are reported in Table 3.1. The estimated support stiffness is much greater than the required stiffness.

Acceptability of Minimum Spring Force

At BOL, the spring is required to counteract the flow induced vibration lateral acceleration forces and the forces due to mechanical thermal bowing. The minimum required spring force is:

F(MIN) =

where:

F(V,2) = The minimum spring force required to prevent

Table 3.1 summarizes the results of the evaluation of F(V,2), F(D,1) and the resulting F(MIN). The results demonstrate that the minimum BOL spring force exceeds the minimum required spring force with ample margin.

The Zircaloy fuel rods are expected to relax at a significantly greater rate than the springs, and complete relaxation of the fuel rods is expected by EOL. Therefore, at EOL, the necessary spring force is that required to overcome flow-induced vibration forces and prevent the fuel rod from lifting off from both dimples simultaneously. With a minimum initial load (as shown in Table 3.1) at 550°F, an irradiation-induced spring relaxation of would be permissible. Current irradiation data indicates a spring relaxation on the order of at EOL, as shown in Reference 2.

Acceptability of 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 (cold) at the beginning-of-life are calculated by finite element analysis. Calculated cladding stresses at the spacer contact points are incorporated into Table 3.3.

3.5.2 Miscellaneous Assembly Components

Compression Spring

The compression springs are located on the fuel rod and inert rod upper end cap shanks between the fuel rod end cap shoulder and the upper tie plate. The spring force must be sufficient to support the weight of the upper tie plate, secure the locking lugs, and aid in seating the rods against the lower tie plate. The compression spring must perform these functions throughout the assembly design life. The spring geometry was designed to account for manufacturing tolerances and differential fuel rod growth. was selected as the spring material because it has adequate corrosion resistance and is capable of maintaining a high strength under outlet coolant temperature conditions. The nominal spring constant and spring force are reported in Table 3.1.

Assembly Hardware

These components are assembled to the upper end cap of the tie rod. Their purpose is to secure the upper tie plate to the tie rods.

The retaining spring maintains the correct location of the locking sleeve in the unlocked position. The nominal spring force is reported in Table 3.1.

The upper tie plate grid is captured by the locking sleeves when the sleeves are in the "locked" position. The adjusting nuts thread to the tie rod upper end caps and fasten the locking sleeves and retaining springs to the tie rods.

was chosen as the material for the locking sleeve and adjusting nut. Structural adequacy of these components has been verified by testing as described in Section 3.1.

TABLE 3.1 SUMMARY OF DESIGN RESULTS

Assembly Growth at EOL, in.

Differential Fuel Rod/Assembly Growth

Maximum Differential Growth at EOL, in.
Nominal Upper End Cap/Upper Tie Plate
Engagement, in.
Nominal Lower End Cap/Lower Tie Plate
Engagement, in.
Minimum Calculated Rod-to-Rod Spacing, in.

Fuel Assembly Holddown

Maximum Liftoff Force, lbf
Minimum Downward Force, in.
Resulting Holddown Force, lbf

Creep Collapse

Minimum Initial Cold Gap, in.
Remaining Cold Gap at Rod Average
Exposure of 6,000 MWd/MTU, in.

Fuel Rod Plenum Spring

Nominal Plenum Spring Force, lbf

Grid Spacers

Spring Deflection Range, in.
Spring BOL Force Range, lbf
Single Dimple Level Stiffness, lbf/in.
Nominal Spring Stiffness, lbf/in.
Dimple Support Stiffness, lbf/in.
K(MIN), Dimple Support Stiffness Required
To Force a Node, lbf/in.
Spring Force Required to Prevent
Simultaneous Liftoff From Both Dimples
(EOL), lbf

TABLE 3.1 SUMMARY OF DESIGN RESULTS (CONTINUED)

Grid Spacers (Continued)	
F(V,2), Spring Force Required to Prevent Liftoff From One Dimple, lbf	
F(D,1), Spring Force Required to Counteract Fuel Rod Bowing, lbf	
F(MIN), Total Minimum Spring Force Required (BOL), lbf	
Compression Spring	
Spring Constant, lbf/in.	
Nominal Spring Force, lbf	
Retaining Spring	
Nominal Spring Force, lbf	

TABLE 3.2 RODEX2 AND RODEX2A VARIABLE INPUT GUIDELINES

Input Variable	Rod Internal Pressure	Steady State Strain	Transient Strain and Stress	Clad Collapse

TABLE 3.3 SUMMARY OF FUEL ROD CLADDING STEADY STATE STRESSES

9x9-5

		Stress Intensity (psi)	Design Limit (psi)	Ratio of Stress Intensity to Design Limit
1. Primary Membrane Stresses				
(Design limit is lower value of 2/3 Sy or 1/3 Su)	BOL Cold			
	EOL Hot			
2. Primary Membrane Plus Primary Bending				
(Design limit is lower of 1.0 Sy or 1/2 Su)	BOL Cold (Max. Ovality)			
(Included are membrane and ovality stresses)	BOL Hot (Max. Ovality)			
	EOL Hot (Max. Ovality)			
3. Primary Plus Secondary				
(Design limit is lower of 2.0 Sy or 1.0 Su)	BOL Cold			
	BOL Hot			
(Included are stresses from Item 2 above plus vibration, thermal gradient, mechanical and thermal bow, and spacer contact pressure)	EOL Hot			

TABLE 3.4 STRESS INTENSITIES AT LOWER END CAP

	Stress Intensity (psi)	Design Limits in Hot Conditions (psi)	Ratio of Stress Intensity to Design Limit
Weld Joint Primary Membrane Plus Primary Bending, Design Limit: 1/2 Su			
Weld Joint Primary Plus Secondary Design Limit: 1.0 Su			

TABLE 3.5 DESIGN DUTY CYCLES FOR CYCLIC FATIGUE

Duty Cycle Description	Total Number of Cycles 9x9-5
1. Startup following a refueling shutdown or major shutdown.	
2. Load follow - weekly reduction to 50% power.	
3. Load follow - daily reduction to 75% power.	
4. Control blade movements.	
5. Startup following a cold shutdown or minimum shutdown.	
6. Recovery following a scram.	
7. Loss of feedwater heaters.	
8. Turbine trip.	

TABLE 3.6 OPERATING EXPERIENCE WITH FUEL TYPES SUPPLIED BY ANF (SEPTEMBER 1988)

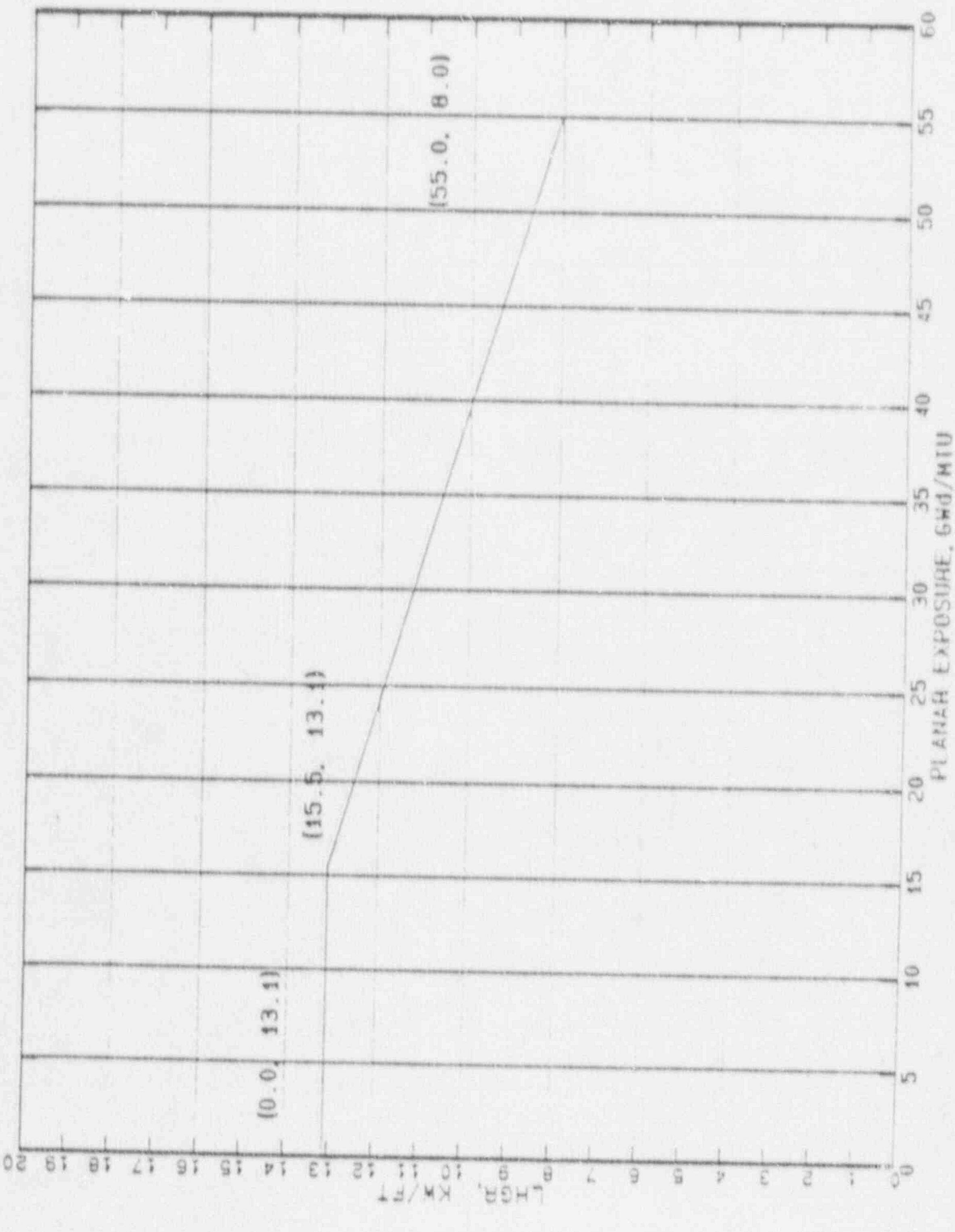

Figure 3.1 - LHGR LIMIT

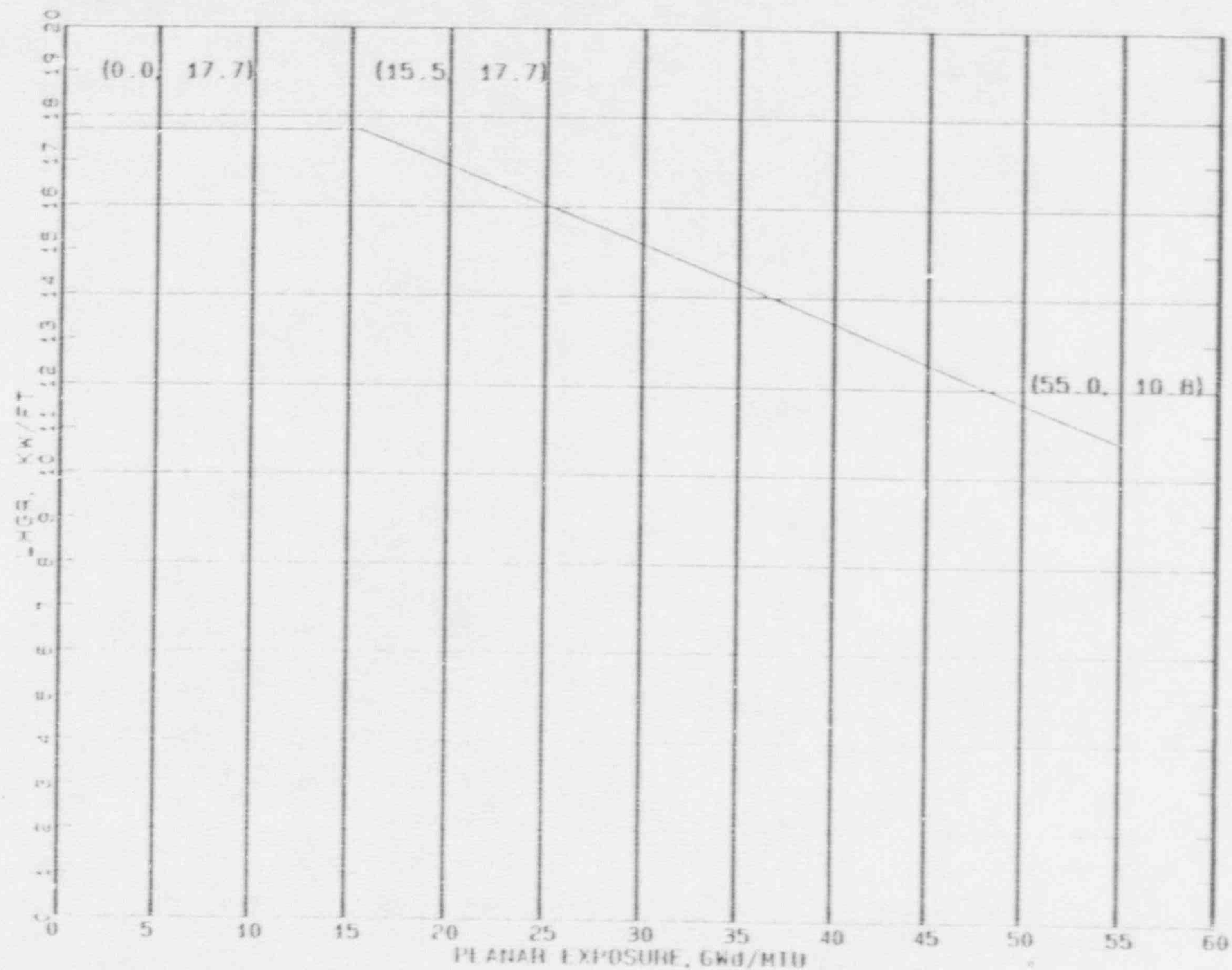

Figure 3.2 PROTECTION AGAINST POWER TRANSIENT

Pages 55-68 have been deleted.

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ADVANCED NUCLEAR FUELS CORPORATION

ANF-88-152(NP)(A)
Amendment 1
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GENERIC MECHANICAL DESIGN FOR ADVANCED NUCLEAR FUELS 9X9-5 BWR RELOAD FUEL

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September, 1989

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PREFACE TO AMENDMENT 1

Amendment 1 has been issued to incorporate

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1.0 INTRODUCTION AND SUMMARY

1.1 Introduction

This report provides a design description and summary of the supporting analyses, and test results applicable to the mechanical design of the ANF 9x9-5 BWR fuel.

This report is similar to XN-NF-85-67(P)(A), Revision 1[1], which contains the mechanical description and the results of mechanical analysis for ANF 8x8 and 9x9 fuel.

The design criteria, technical bases, and a description of mechanical fuel rod failure mechanisms are covered by a separate report, XN-NF-85-39[2]. Mechanical design requirements specified in XN-NF-85-39 are discussed in detail in the design evaluation section of this report. In addition, design bases requirements from the NRC Standard Review Plan[3] which are applicable to the evaluation of this report are identified according to section.

1.2 Summary

Mechanical design analyses have been performed to evaluate cladding steady state strain and stresses, transient strain and stresses, fatigue damage, creep collapse, corrosion, hydrogen absorption, fuel rod internal pressure, pellet and clad temperatures, differential fuel rod growth, rod bow, and grid spacer spring design of the 9x9-5 fuel. These analyses assume the following maximum discharge exposure:

- MWd/MTU assembly exposure
- MWd/MTU rod exposure
- MWd/MTU nodal exposure

The analyses demonstrate that the mechanical design criteria are not violated when the fuel is operated under the LHGR (linear heat generation rate) limits defined in this report.

All analyses herein have been performed with the RODEX2[4] and RODEX2A[5] computer codes using the same methodology already used in Reference 1.

All the analyses have been performed on a generic basis using conservative input data and the enveloping histories defined in Figures 3.3 and 3.4.

1.2.1 Design Description Summary

The ANF 9x9-5 fuel assembly design consists of a 9x9 matrix containing 76 fuel rods and 5 water rods supported by bi-metallic spacer grids. The fuel assembly design may contain natural uranium axial blankets at either end of the fuel rods to enhance neutron economy, and will incorporate gadolinia-bearing fuel rods to provide fuel management flexibility.

The assemblies are designed to allow handling in the same manner, to the same extent, and with the same equipment as that now being used for 8x8 and 9x9 fuel.

The mechanical design of the 9x9-5 fuel is established from the experience with existing ANF fuel designs.

Detailed fuel design drawings in Appendix A provide dimensional details of the 9x9-5 fuel assemblies.

1.2.2 Mechanical Design Summary

The major analysis results are as follows:

- The maximum end of life (EOL) steady state cladding strain is calculated to be below the design limit.
- Cladding steady state stresses are calculated below the material strength limits.
- The cladding strain during anticipated operating occurrences (AOO's) does not exceed
- The maximum fuel rod internal rod pressure remains below ANF's criteria limit.
- The fuel centerline temperature remains below the melting point during AOO's.
- The cladding fatigue usage factor is within the design limit.
- Structural members have adequate strength to support handling and hydraulic loads.
- The cladding diameter reduction
- Evaluations of assembly growth and differential fuel rod growths show that the design provides adequate clearances for compatibility with the reactor internals, fuel assembly channel, and fuel handling machine. Also, there is adequate engagement of the end caps in the upper tie plate and lower tie plate throughout the design life.
- The initial fuel rod design spacing is expected to be adequate to accommodate expected rod to rod gap closure for the fuel design life.
- The maximum EOL reduction in cladding thickness due to corrosion and the maximum concentration of hydrogen in the cladding are calculated to be well within the design limits.
- The fuel rod plenum spring and other miscellaneous components are shown to meet the respective design bases.
- The spacer springs meet all the design requirements and can accommodate the expected relaxation at the respective EOL exposures.

2.0 DESIGN DESCRIPTION

2.1 Fuel Assembly

The ANF 9x9-5 reload fuel assembly design is a 9x9 array with 76 enriched uranium fuel rods. The remainder are inert water rods. Eight of the fueled rods are also tie rods. Some of the rods contain gadolinia as a burnable absorber. Fuel rod pitch is maintained by seven spacers. The spacers are The centrally located inert water rod captures the spacers to maintain the proper axial spacing.

All rods except for the tie rods have coil compression springs located between the top of the fuel rods and the bottom surface of the upper tie plate. These compression springs provide a force to aid in seating the fuel rods in the lower tie plate and react against the upper tie plate. The springs accommodate variations in rod lengths arising from manufacturing tolerances and permit axially non-uniform thermal and irradiation induced growth of the fuel rods.

The eight tie rods are structural members of the fuel assembly and establish the overall assembly length. These rods are threaded into the lower tie plate and latch into the upper tie plate. The tie rods carry the assembly weight during handling and provide the coil spring reaction support.

The assembly contains five water rods to improve uranium utilization.

For fuel rod removal, the upper tie plate must be depressed against the compression springs a short distance in order to allow the locking lugs to be rotated 90°. The upper tie plate is then free to be removed for fuel rod extraction or replacement.

The lower tie plate consists of a machined stainless steel casting with a grid plate for lower end cap engagement and a lower nozzle to distribute coolant to the assembly.

The upper tie plate is a cast and machined grid plate, with attached bail handle to provide for fuel assembly handling and orientation.

Assembly and component descriptions for the 9x9-5 fuel are presented in Table 2.1. Table 2.2 show the 9x9-5 geometric design parameters used for inputs to the analysis codes.

Detailed fuel design drawings in Appendix A provide dimensional details of fuel assemblies.

2.2 <u>Rods</u>

2.2.1 <u>Fuel Rods</u>

The fuel rods consist of UO_2 pellets in Zircaloy-2 tubing with Zircaloy-2 end cap plugs fusion welded on both ends of the tube.

The fuel rod cladding is Each standard fuel rod contains a column of fuel pellets ranging from 144.0 to 150.0 inches in length (dependent upon application). The fuel column contains enriched UO_2

and may also contain natural uranium column segments on the top and/or bottom of the enriched column. The enriched column may be anywhere from about 132 to 150 inches in length.

The pellets are sintered to of theoretical density. The length to diameter ratio (L/D) of the enriched pellet is

The nominal pellet to cladding gap is 0.0065 in.

The fuel rod upper plenum contains an coil spring to prevent fuel column separation during fabrication, shipping, and early reactor operation.

The upper end cap plug is configured to allow remote under water handling and engagement into the upper tie plate. The lower end cap has a shaft and a tapered section which seats in the lower tie plate, and aids in fuel rod insertion into the fuel assembly during fabrication, inspection, and reconstruction. Both end caps are seal welded to the cladding in a helium atmosphere. The rod is pressurized with helium to atmospheres.

Fuel rod identification is maintained by a serial number on the lower end cap.

Fuel assembly component description for the 9x9-5 fuel is included in Table 2.1.

2.2.2 Tie Rods

The tie rod assembly serves to connect the upper and lower tie plates.

The tie rods are fueled and have upper and lower end caps designed for attachment to the tie plates. The lower end cap threads into the lower tie plate. The upper end cap is also threaded for the engagement of the tie plate locking hardware.

The description of the tie rod assemblies are provided in Table 2.1.

2.2.3 Water Rods

water rods are located in the fuel assembly arranged in a cross configuration. Pressure drop through the water

2.2.4 Spacer Capture Rod

The spacer capture rod (SCR) is a water rod. The lower end cap of the SCR is threaded and it connects to the lower tie plate.

The SCR has rings on the outside of the SCR at axial locations corresponding to the spacer locations.

2.3 Tie Plates

2.3.1 Upper Tie Plate

The upper tie plate serves as an easily removable structural member of the fuel assembly. The eight tie rods penetrate the upper tie plate to hold it in place.

A lifting bail is integral with the grid for fuel handling.

Also, there are posts on top of the grid which support and permit attachment of the channel.

The upper tie plate is made of cast stainless steel.

2.3.2 Lower Tie Plate Assembly

The lower tie plate supports the fuel rods during handling and operation and distributes the coolant into the assembly. The eight tie rods thread into the lower tie plate forming a structural tie between the upper and lower tie plate.

seal springs
limit coolant bypass flow between the channel and tie plate.

The lower tie plate is made of cast stainless steel.

2.4 Spacer Grids

The spacer grids are an interlocking square array of strips producing a 9x9 array of cells. spring strips are mechanically secured within the structural strips. The Zircaloy-4 strips which capture the springs are welded to each other at all intersections and to the side plates. The strips are dimpled and the springs are arranged such that each fuel rod is positioned by four support dimples and one spring. Backup lobes are provided on the spring to prevent excessive spring and cladding stresses which might occur under adverse handling conditions. Anti-hangup tabs are on the side plates of the spacers to prevent interference during channeling.

2.5 Miscellaneous Hardware

2.5.1 Compression Spring

The compression springs are located on the fuel rod and inert rod upper end cap shanks between the fuel rod end cap shoulder and the upper tie plate. The spring force supports the weight of the tie plate and channel, and aids in seating the rods into the lower tie plate. The spring dimensions are designed to account for manufacturing tolerances and differential rod growth.

was selected as the spring material.

2.5.2 Assembly Hardware

The retaining spring, locking sleeve, and the adjusting nut are assembled to the upper end cap of the tie rods. The purpose is to secure the upper tie plate to the tie rods.

The retaining spring maintains the correct location of the locking sleeve when the upper tie plate is depressed for installation or removal.

The locking sleeve rotates on the tie rod upper end cap to lock onto the upper tie plate. The adjusting nut threads onto the tie rod upper end cap to fasten the retaining spring and locking sleeve to the tie rod. The threads at the adjusting nut location are deformed to prevent further rotation of the nut after assembly.

The retaining spring is made of and the locking sleeve and adjusting nut are made of low-carbon stainless steel.

TABLE 2.1 FUEL ASSEMBLY COMPONENT DESCRIPTION

	Material	Characteristic 9x9-5
Fuel Assembly		
Array	----	9x9
Width, in.	----	5.251 (BWR-3,4,5)/ 5.149 (BWR-6)
Length, in.	----	171.29 (BWR-3)/ 176.014 (BWR-4,5,6)
No. of Spacers	----	7
Rod Pitch, in.	----	0.569 - 0.572 (BWR-3,4,5) 0.563 (BWR-6)
No. of Fuel Rods	----	76
No. of Water Rods	----	5
Fuel Rod Assembly		
Outside Diameter, in.		0.417
Plenum Length, in.		
Fuel Length, in.		145.24 (BWR-3) 150.00 (BWR-4,5,6)
Pressure, psig		
Fuel Rod Assembly		
Outside Diameter, in.		0.443
Fuel Length, in.		145.24 (BWR-3)/ 150.00 (BWR-4,5,6)
Spacer Capture Rod		
Sleeves		
Outside Diameter, in.		0.417
Water Rod		
Outside Diameter, in.		0.546
Cladding		
Outside Diameter, in.		0.417

TABLE 2.1 FUEL ASSEMBLY COMPONENT DESCRIPTION (CONTINUED)

	Material	Characteristic 9x9-5
Cladding		
Outside Diameter, in.		0.443
Plenum Spring		
Coil Diameter, in.		
Wire Diameter, in.		
Free Length, in.		
Plenum Spring		
Coil Diameter, in.		
Wire Diameter, in.		
Free Length, in.		
End Caps		
Standard Upper		
Length, in.		
Shank Diameter, in.		
Standard Lower		
Length, in.		
Shank Diameter, in.		
Tie Rod Upper		
Length, in.		
Thread, mm		
Tie Rod Lower		
Length, in.		
Thread, mm		
SCR/WR Upper		
Length, in.		
Shank Diameter, in.		
SCR/WR Lower		
Length, in.		
Thread, mm		

TABLE 2.1 FUEL ASSEMBLY COMPONENT DESCRIPTION (CONTINUED)

	Material	Characteristic 9x9-5
Fuel Pellets		
UO_2	Sintered UO_2	
Diameter, in.		
Length, in.		
Density, %TD		
Dish, %		
Gadolinia	UO_2-Gd_2O_3	
Diameter, in.		
Length, in.		
Density, %TD		
Dish, %		
Natural	Sintered UO_2	
Diameter, in.		
Length, in.		
Density, %TD		
Dish, %		
Compression Spring		
Coil Diameter, in.		
Wire Diameter, in.		
Free Length, in.		
Upper Tie Plate		
Height		
Outside Dimension, in.		
Lower Tie Plate Assembly		
Outside Dimension, in.		
Lower Tie Plate Seal		
Height, in.		
Width, in.		

TABLE 2.1 FUEL ASSEMBLY COMPONENT DESCRIPTION (CONTINUED)

	Material	Characteristic 9x9-5
Locking Sleeve		
Height, in.		
Length, in.		
Width, in.		
Adjusting Nut		
Length, in.		
Diameter, in.		
Thread		
Retaining Spring		
Coil Diameter, in.		
Wire Diameter, in.		
Free Length, in.		

Pages 15 - 20 have been deleted.

3.0 DESIGN EVALUATION

ANF 9x9-5 reload fuel assembly components are designed to satisfy the performance and safety objectives described by XN-NF-85-39, Rev. 0, the criteria presented in this section, and the NRC Standard Review Plan[3], Section 4.2. The reactor fuel system 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 as to prevent control rod insertion when 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 internals to achieve acceptable flow distribution, including bypass flow, such that heat transfer requirements are met for all licensed modes of operation.

3.1 <u>Design Experience and Prototype Testing (Standard Review Plan Section 4.2 IIC)</u>

ANF has fabricated large quantities of JP-BWR 8x8 and 9x9 fuel.

irradiation experience to date has shown that the fuel performs satisfactorily. Fuel fabrication and irradiation experience for both BWR and PWR designs is summarized Table 3.6.

The 9x9-5 is an evolutionary design from the ANF 9x9 being operated by several U.S. reactors. The experience gained in the design, manufacturing, and irradiation of over BWR fuel assemblies by ANF has been applied to the 9x9-5 design. The features incorporated in the 9x9-5 do not represent a significant departure from other designs, consequently, this experience is applicable to the 9x9-5 design.

3.1.1 Fuel Assembly Structural Strength

Tie rods, upper tie plates and lower tie plates, and the upper tie plate locking hardware constitute the fuel assembly structural components during handling. In order to withstand expected handling loads, the assembly is designed to withstand a minimum axial load of the bundle weight with no permanent deformation. Also, the tie rod upper end caps are designed to withstand a loading of not less than Should the assembly be subjected to unexpected and excessive loads, failure is to occur at the tie rod end caps without breaching the cladding.

The tie rod upper end cap, upper and lower tie plates in the 9x9-5 design, are essentially the same as those in the standard ANF 9x9 design. The test results reported in Reference 1 are also applicable to the 9x9-5 design.

3.1.2 Hydraulic Performance Tests

Single phase hydraulic characteristics of the 9x9-5 fuel assemblies were experimentally determined[26,27] by hydraulic tests performed in ANF's Portable Hydraulic Test Facility. The testing was performed on full size components to find the loss coefficients of the lower tie plate (including the inlet hardware), spacers, rods, and the upper tie plates.

3.2 Fretting Wear (Standard Review Plan Section 4.2 IIA1(c))

Fretting wear of fuel rods under normal operating reactor conditions can be controlled by the use of design features that assure that the rods are positively supported at the grid spacers throughout the expected irradiation period. In the case of ANF fuel, this is accomplished by the spring arrangement of the spacer grid. The system in the grid spacer is designed such that the minimum rod contact forces throughout the design life are greater than the maximum fuel rod flow vibration forces, thus preventing rod fretting wear.

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

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

The effective spacer spring relaxation, based on this and other data, follow an asymptotic relationship with burnup. For typical ANF rod and spacer springs irradiated to the average spring force is approximately of the initial spring force. The spring force at the top and bottom grids is at least of the initial spring force. The residual spring force therefore has a substantial margin for the prevention of fretting wear during extended burnup.

ANF laboratory testing has shown that the residual spacer spring holding force can be quite low without resulting in fretting damage to the cladding. Extensive flow tests have been performed on ANF assemblies under various spacer spring load conditions. These tests have covered the range of no spring relaxation (i.e., reload corresponding to typically 0.030 spring deflection) to total relaxation (zero preload). In testing to up to 1000 hours duration, no measurable fretting wear resulted from up to relaxation, provided there was contact between the spacer spring and the fuel rod. In these tests, fuel rod wear depths at spacer contact points has typically ranged from although wear of up to approximately in depth has been occasionally observed. Examination indicates that the wear is due primarily to fuel rod loading and unloading and not fuel rod motion during the testing.

The ANF assembly design methodology requires the consideration of flow vibration and irradiation effects on the spacer rod interaction. Flow tests and the fuel assembly irradiation have demonstrated the success of this approach. In addition, the methodology for the calculation of fuel rod stresses and strains requires the use of minimum clad wall thickness. This choice of input provides conservatism in the analysis which compensates for the minimal anticipated fuel rod wear.

3.3 Fuel Assembly Analysis

3.3.1 Differential Fuel Rod Growth and Assembly Growth (Standard Review Plan Section 4.2, IIA1(e))

Axial extension of Zircaloy fuel rods is primarily due to irradiation induced growth. Growth is also related to pellet-to-cladding interaction. Generally, higher growth is experienced by the tie rods. The tie rods undergo creep elongation in addition to irradiation growth because of tensile loads exerted between the upper and lower tie plate by the compression springs and fuel rods. The standard fuel rods must have sufficient engagement into the upper tie plate. The upper end cap shanks of the fuel rods must be long enough so disengagement from the tie plate does not occur throughout the assembly design life.

Fuel assembly growth is a direct result of the tie rod growth. The EOL assemblies growth are reported in Table 3.1.

The maximum differential growth for the assemblies at EOL are shown in Table 3.1. Nominal engagement of the standard fuel rod end cap into the upper tie plate is also shown in Table 3.1.

Adding the maximum manufacturing tolerance to the measured differential growth indicates that remains engaged in the throughout the expected lifetime.

3.3.2 Channel and Fuel Handling Compatibility

The differential growth between the channel and fuel assembly is important so the assembly and channel can be properly grappled. Growth between the channel and assembly has been evaluated at the BOL and EOL when fuel is at its extreme length. The fuel assembly length and is compatible with the channel throughout the life of the fuel.

3.3.3 Fuel Assembly Hydraulic Liftoff (Standard Review Plan Section 4.2 A1(g))

The hydraulic loads on a fuel assembly are calculated to be less than the weight of an assembly in water.

Table 3.1 summarizes the results of the hydraulic analysis performed to demonstrate the bundle's resistance to hydraulic liftoff during normal and anticipated operating modes. The maximum liftoff force includes buoyancy and pressure effects. The results show that the gravitational forces on the fuel are sufficient to prevent hydraulic liftoff under normal and anticipated operating conditions.

3.4 Fuel Rod Analysis

The following sections detail analyses performed to predict performance characteristics of the 9x9-5 fuel rods. The analysis have been performed with RAMPEX and approved versions of RODEX2, RODEX2A and COLAPX codes.

The fuel rod performance characteristics modeled by the RODEX2 and RODEX2A codes are:

- Gas release
- Radial thermal conduction 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

RODEX2 has been used to determine the initial conditions for fuel rod power ramping analysis (RAMPEX)[6]. RODEX2A has been used to determine the steady state strain, internal pressure, fuel cladding temperature, corrosion, hydrogen absorption fuel temperature, internal pressure, and the fuel rod internal pressures for cladding creep collapse (COLAPX)[7] analyses. Pellet density, swelling, densification, and fission gas release models, and cladding and pellet diameters are used in RODEX2 and RODEX2A to provide conservative subsequent calculation results.

The 9x9-5 geometric design parameters used for input conditions into RODEX2, RAMPEX, and COLAPX are presented in Table 2.2.

Table 3.2[4] provides the guidelines for selecting input variables from Table 2.2 to be used in specific applications of the RODEX2 code. The analyses performed with RODEX2A followed the same guidelines.

Gadolinia fuel rods are included in the evaluation. However, due to the designed lower enrichment of the gadolinia fuel, these are not the limiting rods in the fuel assembly.

3.4.1 Linear Heat Generation Rate Limits

Figure 3.1 is the LHGR limits used in the steady state fuel rod performance evaluation. These limits are for the steady state power maneuvering of ANF fuel.

Figure 3.2 is the limit to protect against power transients during anticipated operational occurrences (AOO's). This limit is considered in the transient mechanical analyses and applies to the 9x9-5 design. The first four figures are presented in terms of peak pellet power versus assembly planar exposure.

For evaluation of the LHGR limits, Figures 3.1 and 3.2 are conservatively translated into RODEX2 and RODEX2A inputs in the form of fuel rod nodal powers and time. Figures 3.3 and 3.4 show the respective power inputs versus exposure for a fuel rod. Neutronic analysis indicate that the small diameter rods will never operate to more than power of the most limiting rod in the assembly. Consequently, the input power history has been modified to reflect this result. These figures are in terms of pellet power versus pellet exposure.

This assumption leads to conservatively high rod average powers in evaluation of the LHGR limits. This assumption on axial peaking is identical to that assumed in the Reference 1 analysis.

The performance of the fuel rods have been analyzed to peak pellet exposures for both fuel rod designs. This exposure is consistent with the maximum assembly exposure

3.4.2 Steady State Strain, Hydrogen Absorption and Corrosion (Standard Review Plan Section 4.2, IIA1(d))

Calculations to determine cladding end of life strain, hydrogen absorption, and corrosion are performed using the RODEX2A computer code. The power histories are presented in Figures 3.3 and 3.4. The fuel rod design input parameters for this case are selected from Table 2.2 to provide the most conservative results with respect to strain, corrosion, and hydrogen absorption. Therefore, the tolerances used are those which correspond to

Steady State EOL Strain

Results of the analysis to determine cladding strain are obtained from the RODEX2A output. See Figures 3.5 and 3.6 for plots of the results. The cladding strain at end of life is well within the design criteria limit of

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 440°F to 751°F) (227°C to 400°C) is:

$$H = H(O) + H(I) + H(C)$$

where:

H = Net weight fraction of hydrogen in cladding (ppm)

H(O) = Initial concentration of hydrogen in the cladding due to impurities introduced during cladding manufacturing and autoclaving

H(I) = Concentration of hydrogen in the cladding due to internal sources such as the fuel (ppm)

H(C) = Concentration of hydrogen in the cladding due to external sources such as absorption of hydrogen from the coolant (ppm)

The primary consideration in determining H is the determination of H(C), which is evaluated using the methodology in Reference 19.

The internal source of hydrogen is from the fuel and is approximated to

The total concentration of hydrogen in the cladding is calculated from RODEX2A to be within the design criteria limits. See Figures 3.7 and 3.8 for the calculated hydrogen concentration versus time. The initial sources of hydrogen are also included as discussed above.

Cladding Corrosion

Cladding corrosion is calculated using the correlations from MATPRO[25]. The power histories in Figures 3.3 and 3.4 were used for the analysis.

The cladding temperature limit evaluated under Section 3.4.8 provides further assurance that the corrosion performance is acceptable.

The metal loss, calculated at the maximum exposure is always below the design criteria limit. See plotted results in Figures 3.9 and 3.10.

The conservative aspects of the calculation account for the additional adverse effects of crud deposition by defining a crud layer thickness in the input to the calculation. Crud deposition will increase the cladding surface temperature producing a higher corrosion rate.

3.4.3 Cladding Steady State Stress (Standard Review Plan Section 4.2. IIA1(a))

The results of the steady state stress analysis and the appropriate stress limits are summarized in Table 3.3.

Each individual stress is calculated inside and outside the cladding and at both midspan and spacer level. The applicable stresses at each level are then combined to get the maximum stress intensities. The analysis is performed at BOL and EOL and at cold and hot conditions. The stress analysis assumes

Primary Stresses

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

Primary Bending Stresses

Bending stresses due to ovality are calculated equation[9].

Secondary Stresses

Cladding thermal gradient stress fuel rods operate with a temperature gradient across the cladding wall which may result in significant thermal stresses.

Restrained Thermal Bow

Fuel rod bowing caused by a thermal gradient and restrained from bowing by the spacers is calculated

Restrained Mechanical Bow

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

Flow Induced Vibration

ational stresses due to flow induced vibrations are calculated

Contact Stress From Spacer Springs

The contact stresses at the spring locations are calculated using the finite element method. Calculations are performed with the ANSYS[(15)] general purpose finite element code. The circumferential and axial stresses induced by the contact load are incorporated in Table 3.3.

Fuel Rod End Cap

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 indicates the results are acceptable. The results of the analysis are presented in Table 3.4.

The ANSYS code[(15)], which allows thermal as well as stress analyses, is used to model the subject rod region. The problem is solved by a thermal pass and a stress pass, where the stress analysis uses the results of the thermal analysis as a part of the input.

3.4.4 Transient Occurrences (Standard Review Plan Section 4.2, IIA2(g))

RODEX2 and RAMPEX are used to evaluate the transient cladding strains. Per Section IIA2(g) of the NRC Standard Review Plan Section 4.2, the following is applicable, "The uniform strain of the cladding should not exceed 1%. In this context, uniform strain (elastic and inelastic) is defined as transient induced deformation". During a transient, the fuel centerline temperature must remain below the melting point of the fuel.

The RAMPEX[6] code is used for the determination of cladding transient strain. For input in each RAMPEX case, the RODEX2 output at a particular exposure is used. This RAMPEX input includes gas release, fuel densification, fuel swelling, and fuel relocation due to pellet cracking which all depends on the prior fuel operating history. A description of the ramp rates and powers are additional input to RAMPEX. The ramp rate considered is in all of the analyses. Conservative assumptions were employed in selecting the initial conditions for ramping.

In all cases, the uniform cladding strain did not exceed nor was the reached at or below the LHGR curve in Figure 3.2.

3.4.5 Cladding Cyclic Fatigue (Standard Review Plan Section 4.2, IIA1(b))

Fuel shuffling and reactor power maneuvering will impose a repeated loading on the fuel rod cladding. In addition to the stress analysis for the maximum stress, a fatigue analysis is performed to account for the cyclic pattern of stresses. The RAMPEX[6] code is used to calculate the cyclic stresses.

Loading cycles assumed for the analysis are listed in Table 3.5. These duty cycles are expected to encompass the normal reactor operation over the design life of the fuel.

Input for the RAMPEX code is obtained from RODEX2 using the limiting power histories in Figures 3.3 and 3.4. The input to RAMPEX is selected to simulate each of the various duty cycles. The duty cycles are evaluated at different exposure points to account for burnup effects.

For each duty cycle, a maximum cyclic stress is selected from among the multiple RAMPEX cases run for the power histories. From this cyclic stress, an allowable number of cycles is determined from a S-N (stress versus number of cycles) curve. The cladding fatigue usage factor for each duty cycle is then calculated as the number of expected cycles divided by the number of allowable cycles. The total cladding usage factor is then figured as the sum of the individual usage factors for each duty cycle. For the case of the 9x9-5 fuel these usage factors are well below the limit of cyclic failure.

3.4.6 Cladding Creep Collapse (Standard Review Plan Section 4.2, IIA2(b))

The potential for cladding creep collapse is evaluated using RODEX2A and the COLAPX[7] codes.

The evaluation verifies that this criterion is met(17,18,19). A bounding power history from that used for the steady state strain calculation was used in the analysis.

3.4.7 Fuel Rod Internal Pressure (Standard Review Plan Section 4.2, IIA1(f))

Calculation of the fuel rod internal pressure is done with RODEX2A(5) on a generic basis. RODEX2A is an approved revision of the NRC-approved RODEX2(4) code. The revision has been prepared to more closely predict recently obtained data. In particular, RODEX2A has been benchmarked against fission gas release data from high burnup ramping programs.

In order to protect against fuel rod failure, the internal pressure acceptance criterion limits the fuel rod internal pressure to

s

See Figures 3.5 and 3.6 and 3.11 and 3.12.

3.4.8 Fuel and Cladding Temperatures (Standard Review Plan Section 4.2, IIA2(e), IIA2(d))

Fuel Temperature

The fuel centerline temperature is calculated as in the preceeding section on internal pressure. Figures 3.13 and 3.14 represent the maximum calculated temperatures for the 9x9-5 power history inputs.

Cladding Temperature

Cladding temperatures are calculated using the RODEX2A code. The input conditions are the same as described for the internal pressure calculations. The power history used are described in Figures 3.3 and 3.4. The results indicate that the design criteria for cladding temperature are met. See Figures 3.15 and 3.16 for the cladding temperatures versus exposures.

3.4.9 Fuel Rod Spacing and Rod Bow (Standard Review Plan Section 4.2, IIA1(e))

Rod-to-rod and rod-to-channel spacing must be maintained and rod bow must be limited so that no reduction in MCPR is incurred.

Measurements in gap spacing in irradiated fuel have been collected by ANF 7x7 and 8x8 fuel at burnups in excess of (assembly average)[20]. Also, 9x9 measurements have been obtained after one cycle of irradiation

The statistical derivation of rod bow from gap spacing data is given in Reference 21. The correlation in the data base has been modified in accordance to NRC requirements to include cold-to-hot and batch-to-batch variations.

The calculated minimum rod-to-rod spacing are listed in Table 3.1. These values are calculated at the design peak assembly exposure.

3.4.10 <u>Fuel Rod Plenum Spring</u>

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

The nominal spring force is shown in Table 3.1. This force is exerted by the spring on the fuel column. This load is greater than the fuel column weight 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 during initial fuel densification is expected to be less than The plenum spring design considers the relaxation effect of autoclaving the fuel rods.

3.4.11 Water Rod Design

The water rods for the 9x9-5 design are very similar to designs currently under irradiation in several reactors both as 8x8 and 9x9 fuel assemblies.

The forces or stresses which act on the water rod are:

- Differential pressure across the tube wall
- Compressive axial force due to the compression spring
- Restrained mechanical bow
- Restrained thermal bow
- Flow-induced vibrations
- Radial temperature gradient
- Axial friction force with spacers
- Flow friction

For the SCR, the force due to the flow loss across the spacers and the axial force on the spacers from fuel rod thermal expansion and growth must also be considered.

The stresses due to restrained mechanical bow are assumed to occur when a water rod (or SCR), which is bowed within the specifications limits, is restrained from bowing when assembled into the bundle. The same assumptions are made for restrained thermal bow, except the tendency to bow is caused by a differential temperature across the diameter of the tube. These stresses and the stresses from flow induced vibrations are calculated in the same manner as described for the fuel rods in Section 3.4.2. The stresses from restrained bow and flow induced vibrations are low. This will be shown below in the calculation results.

corrosion model as in RODEX2A.

The water rod connection to the lower tie plate has been analyzed for maximum stress. The loads applied to this connection result from pressure drop across the spacers and friction between the fuel rods and the spacer cell. The friction may be originated due to differential thermal expansion or differential growth.

The analysis results indicate that the water rod, spacer capture rod, and spacer capture rod lower tie plate connector operate well within the elastic stress limits.

3.5 Fuel Assembly Component Evaluation

3.5.1 Grid Spacers

Load Deflection

The load deflection characteristics are determined from the Parts List requirements. Due to spacer cell and fuel rod diameter tolerance stackup, the spring deflection and BOL spring force are reported as ranges in Table 3.1.

Support Stiffness

The support stiffness required to force a node at a support level is generally considered to be five times the simple span stiffness. That is:

where:

This condition is easily met as the support dimples are very stiff.

The dimple stiffness, nominal spring rate, the resulting support stiffness, and the minimum required support stiffness (K(MIN)) are reported in

Table 3.1. The estimated support stiffness is much greater than the required stiffness.

Acceptability of Minimum Spring Force

At BOL, the spring is required to counteract the flow induced vibration lateral acceleration forces and the forces due to mechanical thermal bowing. The minimum required spring force is:

F(MIN) =

where:

F(V,2) = The minimum spring force required to prevent

Table 3.1 summarizes the results of the evaluation of F(V,2), F(D,1) and the resulting F(MIN). The results demonstrate that the minimum BOL spring force exceeds the minimum required spring force with ample margin.

The Zircaloy fuel rods are expected to relax at a significantly greater rate than the springs, and complete relaxation of the fuel rods is expected by EOL. Therefore, at EOL, the necessary spring force is that required to overcome flow-induced vibration forces and prevent the fuel rod from lifting off from both dimples simultaneously. With a minimum initial load (as shown in Table 3.1) at 550°F, an irradiation-induced spring relaxation of would be permissible. Current irradiation data indicates a spring relaxation on the order of at EOL, as shown in Reference 2.

Acceptability of 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 (cold) at the beginning-of-life are calculated by finite element analysis. Calculated cladding stresses at the spacer contact points are incorporated into Table 3.3.

3.5.2 Miscellaneous Assembly Components

Compression Spring

The compression springs are located on the fuel rod and inert rod upper end cap shanks between the fuel rod end cap shoulder and the upper tie plate. The spring force must be sufficient to support the weight of the upper tie plate, secure the locking lugs, and aid in seating the rods against the lower tie plate. The compression spring must perform these functions throughout the assembly design life. The spring geometry was designed to account for manufacturing tolerances and differential fuel rod growth. was selected as the spring material because it has adequate corrosion resistance and is capable of maintaining a high strength under outlet coolant temperature conditions. The nominal spring constant and spring force are reported in Table 3.1.

Assembly Hardware

These components are assembled to the upper end cap of the tie rod. Their purpose is to secure the upper tie plate to the tie rods.

The retaining spring maintains the correct location of the locking sleeve in the unlocked position. The nominal spring force is reported in Table 3.1.

The upper tie plate grid is captured by the locking sleeves when the sleeves are in the "locked" position. The adjusting nuts thread to the tie rod upper end caps and fasten the locking sleeves and retaining springs to the tie rods.

was chosen as the material for the locking sleeve and adjusting nut. Structural adequacy of these components has been verified by testing as described in Section 3.1.

TABLE 3.1 SUMMARY OF DESIGN RESULTS

Assembly Growth at EOL, in.

Differential Fuel Rod/Assembly Growth

- Maximum Differential Growth at EOL, in.
- Nominal Upper End Cap/Upper Tie Plate Engagement, in.
- Nominal Lower End Cap/Lower Tie Plate Engagement, in.
- Minimum Calculated Rod-to-Rod Spacing, in.

Fuel Assembly Holddown

- Maximum Liftoff Force, lbf
- Minimum Downward Force, in.
- Resulting Holddown Force, lbf

Creep Collapse

- Minimum Initial Cold Gap, in.
- Remaining Cold Gap at Rod Average Exposure of 6,000 MWd/MTU, in.

Fuel Rod Plenum Spring

- Nominal Plenum Spring Force, lbf

Grid Spacers

- Spring Deflection Range, in.
- Spring BOL Force Range, lbf
- Single Dimple Level Stiffness, lbf/in.
- Nominal Spring Stiffness, lbf/in.
- Dimple Support Stiffness, lbf/in.
- K(MIN), Dimple Support Stiffness Required To Force a Node, lbf/in.
- Spring Force Required to Prevent Simultaneous Liftoff From Both Dimples (EOL), lbf

TABLE 3.1 SUMMARY OF DESIGN RESULTS (CONTINUED)

Grid Spacers (Continued)

F(V,2), Spring Force Required to Prevent
Liftoff From One Dimple, lbf
F(D,1), Spring Force Required to Counteract
Fuel Rod Bowing, lbf
F(MIN), Total Minimum Spring Force Required
(BOL), lbf

Compression Spring

Spring Constant, lbf/in.
Nominal Spring Force, lbf

Retaining Spring

Nominal Spring Force, lbf

TABLE 3.2 RODEX2 AND RODEX2A VARIABLE INPUT GUIDELINES

Input Variable	Rod Internal Pressure	Steady State Strain	Transient Strain and Stress	Clad Collapse

TABLE 3.3 SUMMARY OF FUEL ROD CLADDING STEADY STATE STRESSES

9x9-5

		Stress Intensity (psi)	Design Limit (psi)	Ratio of Stress Intensity to Design Limit
1. Primary Membrane Stresses				
(Design limit is lower value of 2/3 Sy or 1/3 Su)	BOL Cold EOL Hot			
2. Primary Membrane Plus Primary Bending				
(Design limit is lower of 1.0 Sy or 1/2 Su)	BOL Cold (Max. Ovality)			
(Included are membrane and ovality stresses)	BOL Hot (Max. Ovality) EOL Hot (Max. Ovality)			
3. Primary Plus Secondary				
(Design limit is lower of 2.0 Sy or 1.0 Su)	BOL Cold BOL Hot			
(Included are stresses from Item 2 above plus vibration, thermal gradient, mechanical and thermal bow, and spacer contact pressure)	EOL Hot			

TABLE 3.4 STRESS INTENSITIES AT LOWER END CAP

	Stress Intensity (psi)	Design Limits in Hot Conditions (psi)	Ratio of Stress Intensity to Design Limit
Weld Joint Primary Membrane Plus Primary Bending, Design Limit: 1/2 Su			
Weld Joint Primary Plus Secondary Design Limit: 1.0 Su			

TABLE 3.5 DESIGN DUTY CYCLES FOR CYCLIC FATIGUE

Duty Cycle Description	Total Number of Cycles
1. Startup following a refueling shutdown or major shutdown.	
2. Load follow - weekly reduction to 50% power.	
3. Load follow - daily reduction to 75% power.	
4. Control blade movements.	
5. Startup following a cold shutdown or minimum shutdown.	
6. Recovery following a scram.	
7. Loss of feedwater heaters.	
8. Turbine trip.	

TABLE 3.6 OPERATING EXPERIENCE WITH FUEL TYPES SUPPLIED BY ANF (SEPTEMBER 1988)

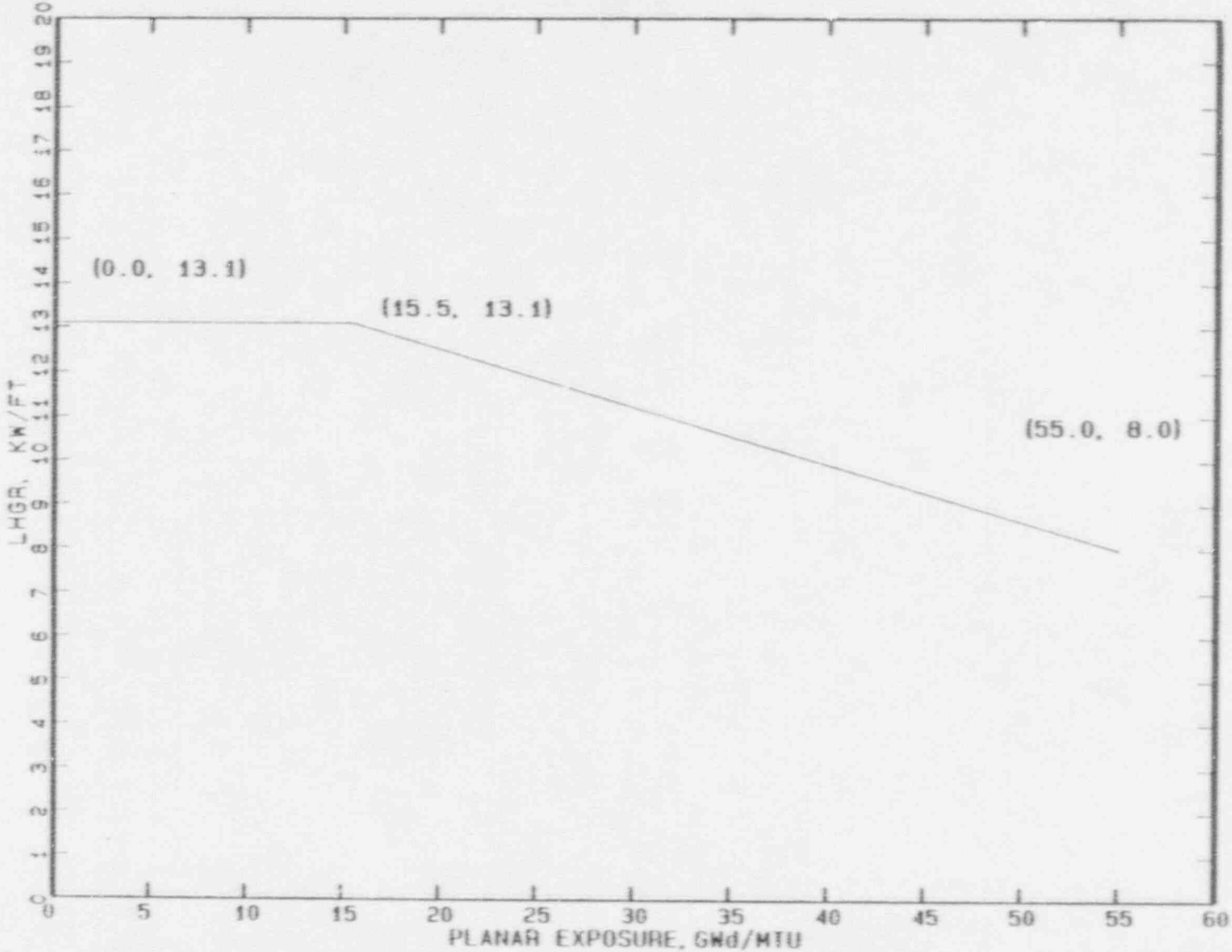

Figure 3.1 - LHGR LIMIT

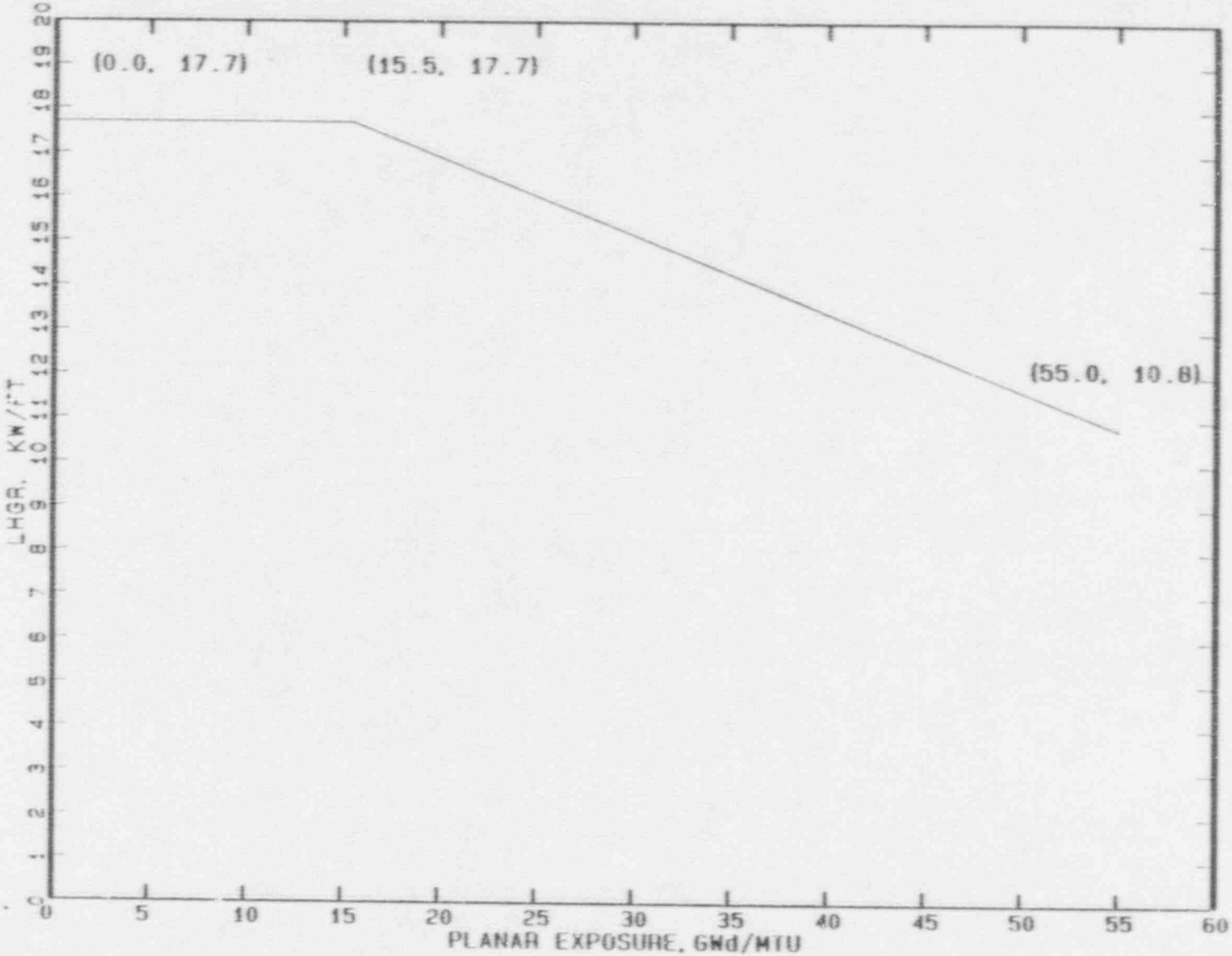

Figure 3.2 - PROTECTION AGAINST POWER TRANSIENT

Pages 55 - 68 have been deleted.

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Pages A-1 - A-7 have been deleted.

ADVANCED NUCLEAR FUELS CORPORATION

ANF-88-152(NP)(A)
Supplement 1

Issue Date: 11/30/90

NRC CORRESPONDENCE

ADVANCED NUCLEAR FUELS CORPORATION

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December 15, 1989
RAC:092:89

Mr. Robert C. Jones, Chief
Reactor Systems Branch
Division of Engineering and System Technology
Office of Nuclear Reactor Regulation
U. S. Nuclear Regulatory Commission
Washington, D.C. 20555

Subject: Responses to Additional Questions on ANF-88-152

Reference:
1. Letter, R. C. Jones (NRC) to R. A. Copeland (ANF), "Request For Additional Information On ANF-88-152", dated October 16, 1989.
2. ANF-88-152(P), Amendment 1, "Generic Mechanical Design for Advanced Nuclear Fuels 9x9-5 BWR Reload Fuel", Advanced Nuclear Fuels Corp., September 1989.
3. Letter, R. A. Copeland (ANF) to Director of NRR (USNRC), Transmittal of Generic Mechanical Design for Advanced Nuclear Fuels 9x9-5 BWR Reload Fuel", dated October 31, 1988. RAC:063:88.

Dear Mr. Jones:

Attached are the ANF responses to the additional information requested by the NRC in Reference 1 on the Mechanical Design of the ANF 9x9-5 fuel design for BWRs (Reference 2). These responses address questions and information regarding the review of the ANF mechanical design topical.

Please consider the information in these responses to be proprietary to ANF. The affidavit supplied with the original submittal provides the necessary information as required by 10 CFR 2.790(b) to support the withholding of the attachment from public disclosure (Reference 3). It is our intent to include these responses with any additional responses as a Supplement to the original report and bind these with the A version of the report when the review is completed.

If there are questions, or if I can be of further help, please contact me.

Sincerely,

Bob Copeland

R. A. Copeland
Manager, Reload Licensing

/sm

cc: Dr. Shih-Liang Wu (USNRC)
Mr. Carl Beyer (PNL)

ATTACHMENT

NRC COMMENTS AND ANF RESPONSES FOR ANF-88-152(P)

GENERIC MECHANICAL DESIGN FOR ADVANCED NUCLEAR FUELS

9X9-5 BWR RELOAD FUEL

Comment 1:

A summary of Advanced Nuclear Fuels (ANF) operating experience through September 1988 was provided in the subject document (see Table 3.6 of the subject document).

- Please provide an update of the 9x9 and 9x9-5 in-reactor operating experience. Also, it is noted that two rods have failed from the 9x9 fuel assemblies irradiated through September 1988 (Table 3.6); please provide information on the cause or mechanism for the failures, if a cause or mechanism is known or suspected.

- Since the issuance of XN-NF-82-06(P), Supplement 1, Revision 2 (January 1987) and the subject document, what additional data has been obtained from the 9x9 and 9x9-5 designs on waterside cladding corrosion, rod bow, and differential rod and assembly axial growth (used to determine engagement between the fuel rods and assembly tie plate)? Please provide a comparison of the 9x9 and 9x9-5 data obtained to date for cladding corrosion, rod bow, and rod-to-tie plate engagement with their respective correlations that are used for evaluating their effect on the 9x9-5 design. Those data from the 9x9 and 9x9-5 designs, respectively, should be identified separately. Have visual examinations for cladding fretting been performed for the 9x9 and 9x9-5 designs, and if so, at what assembly-average burnup levels? Also provide a list of future post-irradiation examination work that will be performed on the first three lead plants using 9x9-5 fuel assemblies.

Response 1:

There are currently 2354 ANF 9x9 assemblies under irradiation or already discharged. 134 of the 9x9 assemblies are of the 9x9-5 design. Eight ANF 9x9 assemblies have been discharged. Table 1.1 shows the irradiation experience of ANF 9x9 and 9x9-5 assemblies. Additional reloads of 9x9 and 9x9-5 assemblies have been contracted, and will be inserted into reactors in the US and Europe in the future.

Oxide Thickness Measurements

ANF 9x9 fuel rod corrosion data is presented in Figures 1.1 and 1.2. Oxide thickness data from irradiated ANF 9x9 fuel assembly measurements are presented in the Figures. Figure 1.1 shows the oxide thickness as a function of exposure, and Figure 1.2 shows the oxide thickness as a function of time at operating temperature.

In order for the MATPRO correlation to predict oxide thickness accurately, enhancement factors may be used; these factors are plant specific and account for different water chemistry conditions.

Rod-to-Rod Spacing Measurements

The results of the rod-to-rod spacing measurements in 9x9 lead assemblies, are presented in Figure 1.3. The figure shows the fractional closure on a 95% min gap bases as a function of assembly average burnup. The ANF rod bow correlation results are also shown in the figure. The rod to rod spacing shown in the figure are acceptable.

Fuel Assembly/Fuel Rod Growth and Differential Growth

For BWR fuel, the determination of fuel bundle growth and differential rod growth is important because of the possibility of fuel rod disengagement fro the upper tie plate, and to maintain the seal between the channel and the fuel bundle lower tie plate seal spring.

ANF designs its BWR fuel to allow for sufficient differential growth of the fuel rods so that all rods remain engaged in the upper tie plate at end-of-life.

The fuel channel material is fully annealed Zircaloy-4, and exhibits less growth than the cold worked, stress relieved tie rods. To ensure that the channel remains engaged with the lower tie plate, sufficient overlapping is provided to maintain the seal to end-of-life.

Fuel Inspection Plans for 9x9-5 BWR Fuel Types

Table 1.1 Exposure Summary of BWR Fuel (Oct. 1989)

	No. of Assemblies by Burnup Range (GWd/MTU)				
Fuel Type	0-10	10-20	20-30	30-40	≥ 40
9x9					
9x9-5					
Total					

Table 1.2 Poolside Measurements Since January 1987

Reactor	Report	Measurements**
Barsebeck-1	89-147(P)	OX,DIA,LEN,VIS
Dresden-2	87-140	VIS,OX,WITH,DIA
	87-83	VIS, ROD,OX,LEN,DIA,SCAN
	89-100	VIS,OX,SCAN,DIA
	89-101(P)	VIS,LEN
Dresden-3	89-006(P)	VIS,OX,LEN,ROD,SPRING
KRB-C	87-123(P)	VIS,LEN,DIA,OX,ROD
	88-140(P)	VIS,LEN,DIA,OX,ROD
	88-141(P)	LEN,DIA,ROD
	89-046(P)	ROD
	89-156(P)	VIS,LEN,DIA,OX,ROD
	89-157(P)	VIS,DIA,OX,SCAN
Hatch-2	(To be issued)*	VIS,OX,DIA,CRUD,LEN

**
VIS = visual
OX = oxide
LEN = rod or assembly length
WITH = spring withdrawal force
DIA = diameter measurements
ROD = rod-to-rod spacing
SCAN = gamma scan
SPRING = seal spring
CRUD = crud sampling

ANF

ANF 9X9 BWR CORROSION DATA

ANF

ANF 9X9 BWR CORROSION DATA

FIGURE 1.3 ANF 9X9 FUEL ASSEMBLY ROD-TO-ROD SPACING DATA

ANF

FIGURE 1.4 ANF 9X9 BWR ASSEMBLY AND ROD GROWTH

FIGURE 1.5 ANF 9X9 BWR MEASURED DIFFERENTIAL GROWTH

Comment 2:

What are the maximum manufacturing tolerances used to evaluate fuel rod upper end cap engagement in the upper tie plate of the assembly (Section 3.3.1 and Table 3.1)? In the response, describe how the manufacturing tolerances and rod and assembly axial growths are used to determine rod-to-tie plate engagement.

Response 2:

Upper End Cap Engagement determination of the (9x9-5) design fuel rod includes the nominal dimensions of the end cap, upper tie plate chamfer, and the distance from the upper end cap shoulder to the bottom of the upper tie plate, as shown below.

	Nominal	Drawing Tolerance
End Cap Length		
Plug Length		
End Cap Tip		
UTP Chamfer		
Installed Compression Spring Length		

The above dimensions and tolerances are obtained from the design drawing as required in the 9x9-5 Parts List. A quality control inspection insures that the parts are manufactured within the tolerance range specified on the drawing.

Comment 3:

Please provide the basis for the 9x9-5 minimum cold gap used in the creep collapse analysis (Table 3.1).

Response 3:

The reported number refers to radial gap. The minimum cold gap is determined by subtracting the maximum pellet outside diameter from the minimum cladding inside diameter and dividing by two.

XN-NF-85-67(P)(A), Rev. 1, "Generic Mechanical Design for Exxon Nuclear Jet Pump BWR Reload Fuel", reports diametral gap. The radial gap is shown below for comparison.

Rod Cold Gap:

Comment 4:

Please provide more information on how the Zircaloy cladding hydrogen concentration, attributed to absorption from the coolant, H(C), is determined as a function of time in reactor.

Response 4:

ANF uses the MATPRO Version 11, Revision 2 (NUREG/CR-0497) hydrogen absorption model to predict concentration of hydrogen in the cladding. Plant specific enhancement factors to the MATPRO correlation can be used to benchmark measured data and make more accurate predictions.

ANF-88-152(NP)(A)

ANF-88-152(NP)(A)
Amendment 1

ANF-88-152(NP)(A)
Supplement 1

Issue Date: 11/30/90

GENERIC MECHANICAL DESIGN FOR

ADVANCED NUCLEAR FUELS 9X9-5 BWR RELOAD FUEL

Distribution

RA Copeland/US NRC (15)