ENCLOSURE

REVIEW OF TOPICAL REPORT NEDE-22148-P,

"EXTENDED BURNUP EVALUATION METHODOLOGY"

CONTENTS

1.0 INTRODUCTION

Economics and prudent utilization of resources have led utilities to seek more efficient use of current generation light water reactors (LWRs). Improved fuel utilization is one of the avenues being pursued for greater efficiency. One of the greater improvements in fuel utilization is to increase the fuel discharge exposure which is currently at batch average burnups of approximately 28 MWd/kgM for BWRs and approximately 33 MWd/kgM for PWRs to batch average burnups of approximately 40 MWd/kgM and 50 MWd/kgM or above, respectively. The higher discharge exposures result in a more complete consumption of the loaded U-235 and a better utilization of the plutonium produced in-reactor. The longer residence time in reactor also reduces spent fuel storage needs.

In response to this trend for extended burnup fuel operation, the Nuclear Regulatory Commission (NRC) has requested (Ref. 1) General Electric to prepare and submit a topical report for review and approval that covers extended burnup experience, methods and test data to provide a generic basis for operation at extended burnups.

General Electric Company (GE) has submitted such a report (Ref. 2) requesting generic licensing approval of their criteria and methods for application to licensing GE fuel at extended burnups. These criteria and methods have been submitted previously as either a part of Amendment 7 or Amendment 10 to NEDE-24011-P-A (Refs. 3 and 4, respectively). These submittals were approved in References 5, 6 and 7 for application of GESTAR-II to current burnup levels. In addition, GE has provided responses (Refs. 8 and 9) to NRC questions concerning the applicability of the criteria and methods submitted in References 2 and 3 to extended burnup operation.

Because the purposes of each vendor's high burnup topical report are slightly different, it is useful to quote GE's goal in preparing this report, as stated in Reference 2. "The purpose of this report is to review how the fuel burnup design parameter is explicitly considered in the General Electric fuel design and analyses process, thereby substantiating the ability to develop a safe and reliable extended burnup product. Approval of this report will, therefore, provide approval of the General Electric design and analyses process with respect to burnup considerations and additional review of burnup dependencies will not be required."

The General Electric Company has chosen not to associate any burnup limits with this report, but rather to associate a specific burnup limit with each specific fuel design.

The burnup limit is established in part to assure that the specified acceptable fuel design limits are satisfied. Also associated with a given fuel design, however, are the thermal-mechanical models and analytical methods used for its design and safety analysis. Since these methods are either developed

from or benchmarked to a data base which has a given burnup range, the validity of the models is also limited to a certain burnup. Since these thermal-mechanical models will be applied to given fuel design, their validity can also be assessed with respect to that design.

Therefore, this SER does not approve General Electric fuel designs and fuel design methods for a specific burnup. Rather, in a manner consistent with the goal of the GE report, it provides an approval of the manner in which GE considers high burnup in its design and safety processes.

This technical review and evaluation has been performed with the assistance of Battelle Pacific Northwest Laboratory (PNL) under contract (FIN B2533) with the United States NRC. The review has been based on References 2, 3, 4, 8 and 9, and Section 4.2 of the Standard Review Plan (SRP) (Ref. 10) and covers the fuel assembly and fuel rods but does not include the control blades or channel boxes for extended burnup operation.

This report follows the intent of Section 4.2 of the SRP, where appropriate for a generic review, to insure that all licensing requirements of the fuel system are reviewed with respect to extended burnup operation. The objective of Section 4.2 and this review are to provide assurance that as a result of extended burnup operation (a) the system is not damaged as a result of normal operation and anticipated operational occurrences, (b) fuel system damage is never so severe as to prevent control rod insertion when it is required, (c) the number of fuel rod failures is not underestimated for postulated accidents, and (d) coolability is always maintained. "Not damaged" is defined as meaning that fuel rods do not fail, that fuel system dimensions remain within operational tolerances, and that functional capabilities are not reduced below those assumed in the safety analysis. This objective implements General Design Criterion (GDC) 10 of 10 CFR Part 50, Appendix A ("General Design Criteria for Nuclear Power Plants") and the design limits that accomplish this are called Specified Acceptable Fuel Design Limits (SAFDLs). "Fuel rod failure" means that the first fission product barrier (the cladding) has been breached. Fuel rod failures must be accounted for in the dose analysis to demonstrate compliance with the offsite dose limits of 10 CFR Part 100 ("Reactor Site Criteria") for postulated accidents. "Coolability," which is sometimes termed "coolable geometry," means, in general, that the fuel assembly retains its rod-bundle geometrical configuration with adequate coolant channels to permit removal of residual heat after a severe accident. The general requirements to maintain control rod insertability and core coolability appear repeatedly in the General Design Criteria (e.g., GDC 27 and 35). Specific coolability requirements for the loss-of-coolant accidents are given in 10 CFR Part 50.46 ("Acceptance Criteria for Emergency Core Cooling Systems for Light Water Nuclear Power Reactors").

In addition to Section 4.2 of the SRP, we have reviewed those aspects of Section 4.3, "Nuclear Design" which are affected by high burnup and these are discussed in general terms in Section 5.0. A brief discussion of radiological consequences of operation with high burnup fuel is given in Section 6.0 and our conclusions and Regulatory Position are given in Section 7.0.

In order to meet the above stated objectives and follow the format of Section 4.2, this review covers the following three main categories: (1) Fuel

System Damage Mechanisms, which are most applicable to normal operation and anticipated operational occurrences, (2) Fuel Rod Failure Mechanisms, which apply to normal operation, anticipated operational occurrences and postulated accidents, and (3) Fuel Coolability, which applies to postulated accidents.

2.0 FUEL SYSTEM DAMAGE

The following review discusses the design bases/criteria and the evaluational methods used by GE to assure safe operation of their fuel designs.

The design criteria in this section should not be exceeded during normal operation including anticipated operational occurrences (AOOs). The evaluation portion of each damage mechanism demonstrates that the design criteria are not exceeded during normal operation including AOOs.

(a) Stress and Strain

Bases/Criteria - In keeping with the GDC 10 SAFDLs, fuel damage criteria should assure that fuel system dimensions remain within operational tolerances and that functional capabilities are not reduced below those assumed in the safety analysis. GE's design basis for the stress and strain of fuel assembly components is that the fuel will not fail due to stresses or strains exceeding the fuel assembly component mechanical capability. GE employs the concept of a design ratio that is defined as a ratio of effective stress or strain to a stress or strain limit. The stress and strain limits are conservative estimates of the unirradiated ultimate tensile stress and the corresponding strain. The effective stress or strain is calculated using von Mises criterion. The design ratio is limited to less than or equal to unity for design purposes. This design ratio is derived from ANSI/ANS-57.5-1981 (Ref. 11), which has some variations from ASME Code Section III which is referenced in the Standard Review Plan. For example, while ANSI/ANS-57.5-1981 uses a full ultimate tensile stress, the ASME Code Section III (Ref. 12) calls for only 70% of the same quantity.

GE has demonstrated, in response to our questions (Ref. 13) during the review of Amendment 7 (Ref. 3), that a conservative approach has been developed in calculating the design ratios. GE performs a Monte Carlo statistical analysis which results in a design ratio distribution for either stress or strain depending on whether yielding has occurred. In order to satisfy the design criterion, GE requires that the upper 95th percentile of the stress or strain distribution be less than unity. The NRC recently approved this new design criterion as a result of the Amendment 7 review (Ref. 5).

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 operation, to a level that would allow cladding failure without the strain ratio criterion being exceeded in the analyses. From examination of irradiated Zircaloy cladding ductility data, it has been concluded that ductility decreases with increasing fluence at low burnup levels, i.e., less than 8 MWd/kgM, but approaches a constant value or a small fluence dependence beyond these low burnups. Consequently, cladding ductility

has either little or no change for the increased burnup levels projected, i.e., from 28 MWd/kgM to approximately 40 MWd/kgM batch average burnups. In addition, GE has irradiated experimental and lead test assemblies with average burnups up to 45.6 MWd/kgM (this corresponds to peak pellet burnups of approximately 58 MWd/kgM) with no adverse effects in cladding or assembly ductility.

In view of the above, we find the GE approach to consideration of extended burnup in fuel stress and strain calculations to be acceptable. The validity of the criterion with respect to ductility to extended burnups must be addressed for specific fuel assembly designs.

Evaluation - The methods used by GE to calculate the design ratios for stress or strain consist of a Monte Carlo error analysis on the input parameters to the GESTR-MECHANICAL code (Ref. 14) resulting in a distribution of design ratios calculated by this code. The upper 95th percentiles of these design ratio distributions are required to be less than 1.0. This methodology has been found conservative and accepted by the NRC for current burnup levels (Refs. 5 and 6). The GESTR-MECHANICAL code is similar to the GESTR-LOCA code (Ref. 15) which was approved by the staff for application to extended burnup operation to a proprietary burnup limit.

Since several of these criteria are dependent on a data base, the continuing validity of these criteria at extended burnup depends on the validity of the data base to a particular application. In keeping with the goal of this review, we agree that the concept of using these criteria to determine whether an NRC review of changes to GESTR-MECHANICAL is necessary continues to be valid but the applicability of the criteria to a specific fuel design will be determined on a design specific basis. The intended burnup of that design will be one of the factors considered.

(b) Strain Fatigue

Bases/Criteria - GE's design basis for strain fatigue is that "the fuel assembly and the fuel rod cladding are evaluated to ensure that strain due to cyclic loadings will not exceed the fatigue capability." A fatigue usage limit of 1.0 is used to assure that the fatigue capability is not exceeded. The fatigue usage is defined as the ratio of the actual number of cycles at stress and the resulting strain to the allowable number of cycles at stress and strain. A fatigue curve in terms of strain amplitude and allowable number of cycles is given in Amendment 7 (Ref. 3). The use of this fatigue usage limit has been found acceptable by the NRC for current burnup levels (Refs. 5, 6 and 17).

As noted in the Stress and 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, extended burnup operation has shown little or no observable effects on cladding ductility and performance. From this, it is concluded that extended burnup

operation does not reduce the applicability of the fatigue limits and thus are found acceptable for use in extended burnup applications.

Evaluation - The GESTR-MECHANICAL code (Ref. 14) along with Monte Carlo/error propagation of the input parameters are used to predict a distribution of fatigue usage (similar to the distributions calculated for the stress and strain ratios). The upper 95th percentile of fatigue usage distribution is required to be less than 1.0. The use of this fatigue usage method has been found conservative and acceptable by the NRC (Refs. 5, 6, 7 and 17). This methodology accounts for internal and external rod pressures, cladding temperatures and pellet-cladding contact at extended burnups. Consequently, this methodology accounts for those cladding parameters that are important for determining strain fatigue at extended burnups and the GESTR-MECHANICAL code used for calculating the strain usage distribution is found to be acceptable for extended burnup application.

(c) Fretting Wear

Bases/Criteria - The Safety Evaluation Report (SER) on Amendment 7 to GESTAR II (Ref. 5) which dealt with GE fuel criteria stated that instead of providing a limit on fretting wear, GE considers the effect of fretting wear in design analysis based on testing and experience in reactor operations.

Because the SRP does not provide numerical acceptance criteria for fretting wear, and the fact that fretting wear is addressed in the design analysis for each bundle design, the NRC staff concluded in Reference 5 that the intent of the SRP has been adequately met. This is also found acceptable for extended burnup design analyses as long as past testing and experience covers the in-reactor residence time of the extended burnup fuel in question (see the evaluation below).

Evaluation - The current GE BWR designs hold individual rods in position in the fuel assembly by spacers located at intervals along the length of the fuel rod, and springs are provided in each spacer cell so that the fuel rod is restrained to avoid excessive vibration. Various in-pile and out-of-pile tests were described in Section 2.6.3 of NEDE-24011-P-A-6 (Ref. 18) along with the results of a continuing fuel surveillance program that has utilized nondestructive methods including eddy current measurements to locate discontinuities in the cladding and detailed visual examinations to characterize the nature of defects. It should be noted that fretting wear is a function of in-reactor residence time which is related to burnup. As stated in NEDE-24011-P-A-6, no significant fretting wear has been observed on GE fuel designs to current burnup levels and their corresponding in-reactor residence times.

In response to NRC questions, GE has indicated (Ref. 8) that no significant fretting wear has been observed on 8X8 fuel assemblies after seven operating cycles (over 8 years in-reactor residence time) with bundle average burnups to 45.6 MWd/kgM. As noted in the SER on Amendment 10 (Ref. 7), the grid and spring force spacer designs of these data are identical to all current GE 8X8 designs. From this information, it is concluded that GE has provided adequate design, testing and experience to show that fretting wear is precluded

on GE fuel designs for extended burnups with in-reactor residence times up to 8 years.

GE must justify on a design-specific basis, if necessary, that residence times longer than 8 years will not cause fuel damage due to fretting.

(d) External Corrosion and Crud Buildup

Bases/Criteria - The GE design bases for external cladding corrosion and crud buildup are to ensure that the cladding temperature increase and cladding thinning due to cladding oxidation and the cladding temperature increase due to crud, do not result in fuel rod failure due to reduced cladding strength. GE does not specify a limit for external corrosion or crud thickness; however, their effects are explicitly modeled in the thermal and mechanical analyses as a part of the GESTR-LOCA (Ref. 15) and GESTR-MECHANICAL (Ref. 14) codes. Because the SRP does not provide numerical limits on cladding oxidation and since GE includes cladding oxidation and crud effects at extended burnup levels in their analyses, it is concluded that GE's approach is consistent with the SRP guidelines and applicable to extended burnup operation.

Evaluation - It has been indicated that GE explicitly models the effects of cladding corrosion and crud in the GESTR-LOCA and GESTR-MECHANICAL codes, and thus, these effects are explicitly included in their thermal and mechanical analyses. The review of this methodology has been accepted in the SER (Ref. 5) of Amendment 7 for GE fuel designs to current burnup levels.

GE's method of measuring cladding corrosion measures the combined thickness of oxide and adherent crud and thus they are not separate entities in these analyses. GE has indicated in this review that the variability of the corrosion thickness data is considered statistically in the thermal and mechanical analyses to provide a 95% bounding condition. Upon examination of how these data are applied statistically, it has become apparent that the statistical variation is of the average oxide thickness on each measured rod between the 20-inch and 130-inch elevations of the rod length. An alternative and more conservative approach would be to treat the statistical variation of maximum oxide thicknesses between rods. The former (GE) method is less conservative, because it gives the statistical probability of the average thickness between the 20-inch and 130-inch elevations of a fuel rod at a particular burnup, while the latter results in a statistical probability of the maximum oxide thickness possible along the length of a fuel rod at a particular burnup. The difference in the upper bound corrosion thickness predicted by these two methods is on the order of only .0006 inches (0.6 mils) or less at extended burnup levels (approximately 60 MWd/kgM peak pellet burnup) and thus is judged to have an insignificant effect on the thermal and mechanical analyses. It should also be noted that the estimate of cladding thinning by GE is somewhat conservative for the mechanical analysis, because it includes the crud thickness which does not contribute to cladding thinning. Consequently, the method proposed by GE for estimating normal cladding oxide thickness is judged to be adequate for the thermal and mechanical analyses at extended burnup levels.

(e) Rod Bowing

Bases/Criteria - Fuel rod bowing is a phenomenon that alters the pitch of adjacent fuel rods and thus affects local nuclear power peaking and heat transfer to the coolant. GE's design basis for rod bowing is that the fuel rod is evaluated to ensure that rod bowing does not result in fuel failure due to boiling transition.

In the SER of GESTAR II Amendment 7, it is stated that this criterion is not acceptable because GE has not presented sufficient evidence that there would not be a reduction in the critical power ratio for significant amounts of bowing. However, it was also found that there was no reason to change the existing position on the effect of fuel rod bowing on the critical power ratio. This position is that no reduction in critical power ratio operating limits is required for GE fuel designs since the amount of rod bow observed in GE BWR fuel is small; however, any rod bow in excess of 50% gap closure should be reported. This position is also found to be applicable to extended burnup fuel.

Evaluation - GE has submitted a generic topical report on fuel rod bowing (Ref. 19). We have reviewed this report and performed an independent assessment of rod bow and concluded that significant rod bow is not expected in GE BWR designs for which data was then available (Ref. 20). In addition, the SER of Amendment 10 (Ref. 7) has concluded that the design changes associated with the GE8X8E and GE8X8EB designs are not expected to change rod bow characteristics when compared to earlier 8X8 designs.

In response to an NRC question on the effects of extended burnup on fuel rod bow, GE has indicated (Ref. 8) that rod bow is not expected to be significant, i.e., less than 50% gap closure, for extended burnup fuel. As verification, GE has responded that the examination of four fuel assemblies with bundle average burnups between 42 and 45 MWd/kgM has shown maximum gap closures significantly less than the 50% gap closure reporting requirement required by the NRC. This is a limited amount of data and thus is not judged to be conclusive; however, based on the data submitted by other fuel vendors fuel rod bow is not expected to be a significant problem for extended burnup fuel. Consequently, the current rod bow reporting requirement (greater than 50% gap closure) for GE fuel designs is found to be adequate for extended burnup fuel.

(f) Axial Growth

Bases/Criteria - The differential irradiation growth rates of fuel rods and assembly tie rods must be considered in GE assembly designs. An axial expansion space exists between the upper end plug shoulder of each fuel rod and the upper tie plate for GE assembly designs. Failure to maintain this expansion spacing can result in fuel rod bowing and possible rod failure. An expansion spring is positioned over the end plug shank and rests on the bottom of the upper tie plate. The function of the spring is to keep the rod seated in the lower tie plate and allow independent axial expansion (due to irradiation induced axial expansion of the fuel rod) of the end plug shank into the holes of the upper tie plate.

GE has not provided a design basis or limit for this expansion spacing in Amendment 7 (Ref. 3), Amendment 10 (Ref. 4), nor in the extended burnup report (Ref. 2); however, in response to an NRC question from the Amendment 10 review, GE has stated (Ref. 21) that the expansion spacing is sized to provide reasonable assurance that bottoming out of the expansion spring will not occur. This criterion is found to meet the intent of the SRP and is also found acceptable for extended burnup application.

Evaluation - Current irradiation experience with GE fuel indicates that gap closure is not a problem (Refs. 8 and 22). However, the data base used to determine rod-to-tie plate clearances does not extend to the fuel burnup range to which GE fuel assemblies are intended (Ref. 8). GE has stated that rod growth measurements from fuel bundles which have achieved bundle average burnups to 46 MWd/kgM are consistent with the rod-to-tie plate clearance data base.

Due to the lack of direct rod-to-tie plate clearance measurements over the full range of burnup proposed by GE and the diametral gap size changes in GE's new extended burnup design (Ref. 4), it is recommended that GE obtain confirmatory rod-to-tie plate measurements from their fuel surveillance program for their extended burnup fuel designs.

(g) Fuel Rod Pressures

Bases/Criteria - The SRP identifies fuel rod pressure as a potential fuel damage mechanism separate from the stress and strain criteria already discussed in this review.

In order to simplify the analysis of fuel system damage due to excessive rod internal pressure, the SRP states that rod internal pressure should remain below the nominal reactor coolant system (RCS) pressure during normal operation unless otherwise justified. GE has elected to justify limits other than those provided in the SRP. In the Amendment 7 submittal, GE has proposed that the rod pressure be limited so that the instantaneous cladding creepout rate due to internal rod pressure greater than RCS pressure is not expected to exceed the instantaneous fuel swelling rate, i.e., the fuel-to-cladding gap does not open. GE has shown that this new criterion is acceptable with respect to the applicability of analytical methods for modeling the cladding strain when the cladding is in tension, thermal feedback causing increased fission gas release, radiological dose estimates and fuel rod behavior in transients and accidents. This new criterion has been found acceptable in Reference 5. This criterion is not altered by extended burnup operation and thus is also found acceptable for extended burnup applications.

Evaluation - The Amendment 7 submittal has proposed that the GESTR-MECHANICAL code be used to determine that the above criterion is met. This has been found acceptable in Reference 5 for current burnup levels and, as discussed in Section 2.0(a) of this SER, is also acceptable for extended burnup up to the proprietary burnup limit of GESTR-MECHANICAL.

Another important parameter in the rod internal pressure calculation is the power history used as input to the GESTR-MECHANICAL code. The power history input is very important, because fission gas release and thus internal rod gas pressure is strongly dependent on the fuel thermal history. In response to an NRC question on power history, GE has indicated (Ref. 23) that the Maximum Linear Heat Generation Rate (MLHGR) limit is used along with a representative axial power profile for their rod pressure calculations. The MLHGR limit is determined by applying an appropriate peaking factor to the Maximum Average Planar Linear Heat Generation Rate (MAPLHGR) limit used in the LOCA analyses. The MLHGR limit represents the maximum bounding power possible for any rod in the core for a particular design and burnup. Therefore, it is concluded that this methodology is bounding with respect to rod pressure calculations and thus is acceptable for extended burnup applications.

(h) Fuel Assembly Liftoff

Bases/Criteria - The SRP calls for the fuel assembly holddown capability (net weight and spring forces) to exceed worst-case hydraulic loads for normal operation including AOOs. The Amendment 7 and 10 submittals have stated that the fuel assembly is evaluated to ensure that vertical liftoff forces are not sufficient to unseat the lower tieplate from the fuel support piece to such a degree that the resulting loss of lateral fuel bundle positioning would interfere with control blade insertion. These amendments reference an approved report (Ref. 24) that describes a liftoff limit to prevent control blade interference. This limit is not burnup dependent and thus is found to be applicable to extended burnup applications.

Evaluation - The GE assessment of fuel assembly holddown capability for normal operation including AOOs is addressed by a bounding calculation using worst case LOCA and seismic loading to assure that the above criterion is met. The methods used to assess the LOCA and seismic loadings will be addressed in Section 4.0(d) of this report.

(i) Control Material Leaching

This topic concerns control blades and is outside of the scope of this review. The issue has been satisfactorily resolved. See Reference 5 for a discussion of this topic.

3.0 FUEL ROD FAILURE

In the following paragraphs, GE fuel rod failure thresholds and analyses for the failure mechanisms listed in the SRP (Ref. 10), the GE extended burnup topical report (Ref. 2) and Amendment 7 (Ref. 3) are reviewed with respect to their application to extended burnups. When the failure thresholds are applied to normal operation including anticipated operational occurrences, they are used as limits (and hence SAFDLs) since fuel failure under those conditions should not occur according to the traditional conservative interpretation of General Design Criterion 10. When these thresholds are used for postulated accidents, fuel failures are permitted, but the resulting radiological doses must be within the limits specified in 10 CFR 100.

(a) Hydriding

Bases/Criteria - Internal hydriding as a cladding failure mechanism is a result of hydrogenous impurities introduced during the manufacturing process. This is an early-in-life failure mechanism which has been addressed adequately in Amendments 7 and 10 and will not be discussed further here.

External hydriding can result from cladding oxidation; however, this has not been a problem for light water reactor fuel exhibiting normal amounts of oxidation which has been addressed in Section 2.0(d) of this report and will not be discussed further.

Evaluation - These phenomena have been adequately addressed in Amendments 7 and 10 and Section 2.0(d), as noted above, and will not be discussed further.

(b) Cladding Collapse

Bases/Criteria - If axial gaps in the fuel pellet column were to occur due to densification, the cladding would have the potential of collapsing into a gap (i.e., flattening). Because of the large local strains that would result from collapse, collapsed cladding is assumed to be failed. In order to define a collapse criterion to reflect the maximum operational conditions of the reactor, GE has adopted (Ref. 25) a collapse criterion that is related to an assumed pressure increase during a turbine trip without bypass; that is, if the fuel rod can sustain, without collapse, an instantaneous increase in the hot system pressure of a given magnitude, it is considered safe against collapse during normal operation, including AOOs. The maximum ovality which precedes this collapse-safe transient is defined as the design limit ovality. The report (Ref. 25) that contains these limits and definitions has been reviewed and approved by NRC (Ref. 25). These limits are also found to apply to extended burnup fuel and thus are found acceptable to extended burnup applications.

Evaluation - GE has a cladding collapse model (Ref. 25) that has been approved for use on current burnup level fuel. The creep collapse analysis is dependent on elastic, plastic, and creep properties of the Zircaloy cladding along with cladding thickness, ovality and in-reactor residence time. As noted earlier, the burnup dependence due to irradiation effects of the material properties saturates at low burnups and thus does not change significantly at extended burnups. In addition, the GE model accounts for cladding thinning due to oxidation (see Section 2.0(e)) and the longer in-reactor residence times associated with extended burnup operation. Consequently, the GE methodology for evaluating cladding collapse has adequately modeled the effects of extended burnup and thus is acceptable for extended burnup applications.

(c) Overheating of Cladding

Bases/Criteria - As indicated in SRP Section 4.2.II.A.2, it has been traditional practice to assume that failures will not occur if the thermal margin criterion is satisfied. This is a conservative assumption for events that cause failures as a result of high cladding temperatures. For BWR fuel, the thermal margin is stated in terms of the minimum value of the critical power ratio (MCPR). As indicated in Section S.2.0 of Amendment 7, GE ensures

that adequate thermal margin is maintained by selecting an MCPR based on a statistical analysis as follows:

"Moderate frequency transients caused by a single operator error or equipment malfunction shall be limited such that, considering uncertainties in manufacturing and monitoring the core operating state, more than 99.9 percent of the fuel rods would be expected to avoid boiling transition."

Both the normal operation and transient thermal limits in terms of MCPR are derived from this approach, which is described fully in NEDE-10958-P-A (Ref. 26) and NEDO-10958-A. These design limits are consistent with the thermal margin guidelines of SRP Section 4.2.II.A.2 and thus have been found acceptable by the NRC (Ref. 26). These design limits are not reduced by extended burnup operation and thus are found to be applicable to extended burnup operation.

Evaluation - The analysis of cladding overheating is independent of burnup with the exception of the input of bundle power distribution. Section 5.0 of this SER concludes that the current GE methods used for this calculation are valid for extended burnup.

(d) Overheating of Fuel Pellets

Bases/Criteria - GE presented the bases and criteria for overheating of fuel pellets in Amendment 7 to GESTAR II (Ref. 3). The NRC staff approved these bases and criteria in the SER on Amendment 7 (Ref. 5). GE's design basis for fuel pellet overheating is that the fuel rod is evaluated to ensure that fuel rod failure due to fuel melting will not occur. To achieve the design basis, GE limits the fuel rod to (1) no fuel melting during normal steady-state operation and whole core anticipated operational occurrences, and (2) a small amount of fuel melting but not exceeding 1% cladding strain for local anticipated operational occurrences such as the rod withdrawal error. These criteria are also found acceptable for extended burnup applications.

Evaluation - The GESTR-MECHANICAL code is used to assure that these criteria are met. As discussed in Section 2.0(a) of this SER, GESTR-MECHANICAL is valid for application to extended burnup.

Gadolinia is mixed in with the UO2 in some fuel rods to act as a burnable poison for reactivity control. The gadolinia lowers the melting point and thermal conductivity of the fuel and GE currently limits the concentration of gadolinia to 6%. The NRC staff has agreed that this value may be increased up to 10% provided that GE carries out a successful test program to assure that the fuel rod analysis methods remain valid and that no unexpected phenomena occur for high gadolinia concentrations. GE has proposed and the staff has accepted (Ref. 27) a program to provide this verification.

The GESTR-MECHANICAL code is approved for application to evaluate incipient melting of extended burnup fuels with the above limitations on gadolinia fuel.

(e) Excessive Fuel Enthalpy

Bases/Criteria - For a severe reactivity initiated accident (RIA) in a BWR at zero or low power, fuel failure is assumed in the SRP to occur if the radially averaged fuel rod enthalpy is greater than 170 cal/gm at any axial location. The 170 cal/gm enthalpy criterion, developed from SPERT tests (Ref. 28), is primarily intended to address cladding overheating effects, but it also indirectly addresses pellet/cladding interactions associated with severe RIAs. As indicated in Section S.2.5.1.6 of Amendment 7 and in Reference 29, GE uses 170 cal/gm as a cladding failure threshold. This has been approved by the NRC (Ref. 5).

Evaluation - GE has stated that the models used in the analysis of the Rod Drop Accident are insensitive to extended burnups, because it is characterized almost exclusively by the control rod worth which is essentially constant in this burnup range. The NRC staff agrees with this position. Consequently, the analysis methods are found to be acceptable for extended burnup applications.

(f) Pellet Cladding Interaction

Bases/Criteria - As indicated in SRP Section 4.2.II.A.2.g, there are no specifically applicable NRC criteria for PCI failures. One design criterion used to limit failures is to restrict the cladding strain to less than 1%. GE has stated (Refs. 3 and 4) that they employ a 1% circumferential plastic strain limit for their fuel designs during AOOs and this has been found to be consistent with the SRP and thus acceptable for current burnup levels (Ref. 5).

As noted earlier, cladding ductility is not expected to be significantly affected at the peak pellet exposures of extended burnup operation of 60 MWd/kgM. Consequently, this strain limit is judged to be acceptable for extended burnup applications.

Past operating experience has shown that the 1% cladding strain criterion is not totally effective in preventing PCI because a limit on the average strain does not prevent highly localized strains that can result in fuel failure. As a result of developmental investigations and feedback from production fuel experience, operating restrictions known as Preconditioning Interim Operating Management Recommendations (PCIOMRs) were issued by GE to the BWR operators (Ref. 32). These restrictions have reduced the incidence of PCI failures and complement the 1% criterion. PCIOMRs have generally been effective in reducing PCI failures that result from operational power changes, but they are not intended to prevent PCI failures during unexpected transients and accidents. A further discussion of PCI failures as a result of off-normal events is provided in the Amendment 7 review (Ref.5).

Evaluation - The GESTR-MECHANICAL code is used by GE to determine that their fuel designs meet the above 1% cladding strain criterion. This code is approved for this analysis by the NRC for extended burnup as discussed in Section 2.0(a) of this SER.

In addition, it should be noted that extended burnup fuels will experience a reduction in power capability due to fissile material burnout that should help mitigate the effects of PCI at extended burnups. However, the new

extended burnup fuel designs submitted by GE in Amendment 10 (Ref. 4) has decreased the fuel-to-cladding gap size which may enhance PCI effects. Consequently, PCI should be monitored at extended burnups in GE's fuel surveillance program for this new design.

(g) Cladding Rupture

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

Evaluation - GE uses an empirical rupture-temperature correlation (Ref. 33) which has been updated to include the data and models from Reference 34 as a part of the LOCA emergency core cooling system (ECCS) analysis. The cladding deformation and rupture models are directly coupled to the models for cladding ballooning and flow blockage in the LOCA analysis. A more detailed discussion of these models and their relation to extended burnup operation is provided in the section on cladding ballooning and flow blockage, see Section 4.0(c). These models have been approved by the NRC (Ref. 35) for current burnup levels and for the reasons stated in Section 4.0(c) are also found acceptable for extended burnup applications.

Other parameters that are important to the LOCA analysis are those input to this analysis from steady-state (normal) operation. Those steady-state input parameters important for cladding rupture in the LOCA analysis and affected by extended burnup operation, such as stored energy and fission gas release, are explicitly modeled with the GESTR-LOCA code. As noted earlier, the NRC has approved this code for extended burnup application to the LOCA analysis up to a proprietary burnup limit.

Therefore, it is concluded that the GE methodology for determining cladding deformation and rupture during a LOCA are applicable to extended burnup fuel up to the proprietary burnup limit.

(h) Fuel Rod Mechanical Fracturing

Bases/Criteria - The term "mechanical fracture" refers to a cladding defect that is caused by an externally applied force such as a load derived from core-plate motion or a hydraulic load. These loads are bounded by the loads of a safe-shutdown earthquake (SSE) and LOCA, and the mechanical fracturing analysis is usually done as a part of the SSE-LOCA loads analysis (see Section 4.0(d) of this SER). The entire SSE-LOCA loads evaluation (including design limits) has been described by GE in a topical report that has been approved by the NRC (Ref. 24).

Evaluation - The discussion of the SSE-LOCA loading analysis is given in Section 4.0(d) of this SER.

4.0 FUEL COOLABILITY

For accidents in which severe fuel damage might occur, core coolability must be maintained as required by several General Design Criteria (e.g., GDC 27 and 35). In the following paragraphs, limits and methods to assure that coolability is maintained are reviewed for the severe damage mechanisms listed in the SRP.

(a) Fragmentation of Embrittled Cladding

Bases/Criteria - The most severe occurrence of cladding oxidation and possible fragmentation during an accident results from a LOCA. In order to limit the effects of cladding oxidation for a LOCA, GE uses (Refs. 18 and 33) acceptance criteria of 2200°F peak cladding temperature and 17% maximum cladding oxidation as prescribed by 10 CFR 50.46. These criteria are not affected by extended burnup operation.

Evaluation - The cladding oxidation models used to determine the amount of cladding fragmentation and embrittlement during the LOCA are not affected by extended burnup operation; however, some of the steady state operational input provided to the LOCA analysis is burnup dependent. Those burnup dependent input parameters important to the LOCA such as stored energy and fission gas release from steady-state operation are provided by the GESTR-LOCA steady-state fuel performance code. This code has been approved by the NRC for extended burnup application (Ref. 16). Consequently, the effects of extended burnup on these analysis methods have been adequately addressed by GE.

(b) Violent Expulsion of Fuel

Bases/Criteria - In a severe reactivity initiated accident (RIA) such as a BWR control rod drop, the large and rapid deposition of energy in the fuel can result in fuel melting, fragmentation, and violent dispersal of fuel droplets or fragments into the primary coolant. The mechanical action associated with such fuel dispersal can be sufficient to destroy the cladding and rod-bundle geometry of the fuel and to produce pressure pulses in the primary system. To meet the guidelines of the SRP as it relates to the prevention of widespread fragmentation and dispersal of fuel and the avoidance of pressure pulse generation within the reactor vessel, a radially averaged enthalpy limit of 280 cal/g should be observed. As indicated in References 16 and 29, GE employs this 280 cal/g criterion as a control rod drop accident design limit and thus is consistent with the SRP.

Evaluation - As discussed in Section 5 of this safety evaluation report, the methods used to calculate energy deposition as a result of reactivity insertion accidents (reactor physics codes) are also applicable to extended burnups. However, the extended burnup fuel is not expected to approach the 280 cal/gm criterion because fissile material burnout at extended burnups lowers the maximum possible fuel enthalpies when compared to maximum fuel enthalpies at lower burnups.

It should also be noted that a conservative approach is taken with respect to these calculations. A more realistic analysis would further decrease the calculated fuel enthalpies.

(c) Cladding Ballooning

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

Evaluation - The GE cladding ballooning model is directly coupled to the cladding rupture temperature model for the LOCA-ECCS analysis and these are addressed in the report NEDE-20566-P (Ref. 33). This report adopted the NUREG-0630 (Ref. 34) 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 (Ref. 35) for current burnup levels. The GE extended burnup topical report (Ref. 2) references this approved report as being applicable to the LOCA analysis of extended burnup fuel.

There is evidence that cladding oxidation at extended burnup levels and LOCA temperatures may result in reduced cladding strains (Ref. 36) from those predicted by NUREG-0630. 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 any irradiation damage effects on cladding properties.

The steady-state operational input that is provided to the LOCA analysis from the GESTR-LOCA fuel performance code is burnup dependent. As noted earlier (see Section 3.0(g)), the GESTR-LOCA code has been approved for extended burnup application.

From this evaluation, it is concluded that the GE methodology for calculating cladding ballooning during a LOCA adequately addresses extended burnup effects and this is acceptable for extended burnup applications.

(d) Fuel Assembly Structural Damage From External Forces

Bases/Criteria - Earthquakes and postulated pipe breaks in the reactor coolant system would result in external forces on the fuel assembly. SRP Section 4.2 and associated Appendix A state that fuel system coolability should be maintained and that damage should not be so severe as to prevent control rod insertion when required during these low probability accidents. The SRP recommends acceptance criteria to achieve these objectives.

The entire SSE-LOCA loads evaluation (including design limits) has been described by GE in an approved topical report (Ref. 24) to which the GE extended burnup report makes reference. The material property that could have an impact on these SSE-LOCA design limits at extended burnup levels is material ductility. These design limits could be impacted if cladding or assembly ductility were decreased, as a result of extended operations, to a level that would allow cladding or assembly failure not accounted for in the analyses. As

noted in Section 2.0(a), the decrease in material ductility is expected to be negligible for the increased exposure and burnup levels requested and no adverse effects have been observed for peak pellet burnups up to 58 MWd/kgM in GE commercial test assemblies.

From the above evaluation, we conclude that the SSE-LOCA design limits are applicable for extended burnup application.

Evaluation - Generic analysis methods for performing combined SSE-LOCA loading analyses have been described by GE in an approved topical report (Ref. 24). These analysis methods not only include the fuel assembly structural response but also fuel rod cladding loads and assembly lift-off forces for the combined SSE-LOCA events as prescribed by Appendix A to SRP 4.2. These loads and forces include those that are burnup dependent such as internal rod pressures. Therefore, these analyses are approved for extended burnup applications.

It should be noted that this analysis is plant specific, because it requires site specific input ground motions and thus cannot be completed in a generic manner. Therefore, an applicant for an operating license must perform site specific analyses using Reference 24 analysis methods in order to address the above criterion and Appendix A to SRP Section 4.2 guidelines.

5.0 NUCLEAR DESIGN

Typical extended fuel burnup and increased fuel cycle length core designs utilize higher fuel enrichments, low leakage patterns and/or axial blankets. Higher fuel enrichment is required to reduce the number of feed assemblies and offset the reactivity loss resulting from the higher fission product inventory. The core neutron economy is improved by reducing the radial leakage using low leakage loading patterns in which the high burnup fuel is located on the core periphery. Axial blankets are used to flatten the axial burnup distribution and improve fuel utilization.

These features affect the physics characteristics of high burnup core designs. The increased fuel depletion in high burnup cores results in an increase in the plutonium fission fraction and the fisson product inventory, the higher plutonium fission fraction in turn hardens the neutron spectrum and increases the neutron production per unit energy. The increased fission product inventory and use of burnable absorbers tends to increase absorption and also harden the neutron spectrum.

While the increased fuel burnup does affect the core physics characteristics, the changes are relatively small and the physics parameters are determined using standard calculational methods and procedures. The high burnup neutronic effects enter through the microscopic cross sections and fuel assembly lattice group constants. The present calculations of these parameters account for substantial levels of plutonium, fission products and burnable absorbers, and these methods are expected to adequately treat the neutronics changes associated with extended fuel burnup. The depletion methods used to

track the plutonium and fission product isotopics and various normalization procedures are also expected to be equally valid for high burnup fuel configurations.

The high burnup fuel physics characteristics and core configuration affect the core nuclear safety parameters. The major effect is to increase the power in the low burnup and/or centrally located fuel assemblies and to decrease the power in the high burnup and/or peripherally located fuel assemblies. The resulting increase in the number and power of the peak powered rods is typically controlled by use of burnable poison rods.

The increased fission product inventory and use of burnable absorbers increases thermal and epithermal absorption and hardens the core neutron spectrum. These factors combine to reduce the boron and control rod worth, prompt neutron life-time and Doppler coefficient. The moderator temperature coefficient may increase or decrease depending on the particular high burnup design, and is also controlled using burnable absorbers as in present core designs. The delayed neutron fraction is also reduced as a result of the increased plutonium fission fraction.

In addition to improving the neutron economy, the low leakage patterns reduce the pressure vessel damage fluence by shifting the power toward the center of the core and away from the vessel. This fluence reduction is partially offset, however, by the harder neutron spectrum and increased neutron production (per MeV) of the high burnup fuel.

The calculation of the high burnup core safety parameters is carried out using the same core and lattice methods and procedures used for present core designs. The changes in the core safety parameters resulting from the higher fuel burnup designs tend to be relatively small as a result of the low relative importance of the high burnup fuel and the tendency for the increase in plutonium fission rates and fission product inventory to saturate. These calculated safety parameters provide the core neutronics input to the required plant transient and accident analysis.

As the above discussion indicates, the effect of high burnup on the physics design is expected to result in relatively small changes in the predicted characteristics of the core, and also relatively small extensions in range of the methods used to calculate the characteristics. Because high burnup fuel is not subject to limiting duty and because of its low relative importance in determining the core characteristics, we conclude that present methods are adequate for high burnup designs. To provide added assurance that these methods are adequate, we recommend that General Electric pay special attention to comparisons of predicted and measured physics parameters (particularly power distributions) which are monitored during the reactor cycle. A systematic pattern of deviation between predictions and measurements would provide an indication of potential problems. We intend to take an active role in following these comparisons.

6.0 RADIOLOGICAL CONSIDERATIONS OF POSTULATED ACCIDENTS WITH EXTENDED BURNUP OPERATION

To ensure that accidents involving the movement of fuel do not constitute an offsite health and safety issue, design events are assessed. Analyses of fuel handling accidents assume release of the entire volatile radionuclide fuel assembly gap and plenum inventory under nominally 23 feet of water after the assembly has cooled substantially (usually at least 24 hours for BWR assemblies, 72 or 100 hours for PWR assemblies). For assemblies with burnup up to 38,000 MWd/t batch average at discharge, Regulatory Guide 1.25 assumptions are used. These stipulate an inventory of ten percent of the total fuel assembly iodines and noble gases (with the exception of 30 percent for ^{85}Kr) in the gap and plenum volumes released upon clad perforation. An iodine decontamination factor (DF) of 100 ("Evaluation of Fission Product Release and Transport for a Fuel Handling Accident," G. Burley, USAEC, Revised October 5, 1971) is assumed for 23 feet of water cover, and appropriate airborne radionuclide filtration/mixing, if any, is applied in the analysis before release to the atmosphere. The decontamination factor is based, in part, on an analysis of work presented in WCAP-7518-L, "Radiological Consequences of a Fuel Handling Accident," M. J. Bell, et al, June 1970, NES Proprietary Class 2.

For fuel handling accident offsite radiological consequence evaluations involving fuel assemblies with burnup > 38,000 MWd/t batch average at discharge (extended burnup assemblies), the analysis is presently performed using Regulatory Guide 1.25 assumptions, but with modified gap and plenum fractional volatile radionuclide inventories. The fractional inventories range from a few percent (less than the R. G. 1.25 ten percent recommendation) to as much as 40-50 percent for certain high burnups/radionuclide combinations. The gap and plenum fractional inventories for the highest-power assembly are computed as a function of at least burnup, and at most time, temperature, and burnup using the GAPCON-THERMAL-2 computer code in conjunction with the ANS 5.4 fission gas release standard (model) proposed by the American Nuclear Society in "Radioactive Gas Release from LWR Fuel," C. E. Beyer, draft NUREG CR-2715, April 1982. In generating these estimated fractional inventories, the conservative assumption of fuel assembly operation at a constant maximum-allowed peak linear heat generation rate (LHGR) for PWR's or MAPLHGR for BWR's is made. This assumption appears to be conservative within a factor of 2-3 for gap and plenum volatile inventories.

In addition to the conservative assumption regarding fuel assembly power operation noted above, there are two other significant sources of conservatism in the staff's analysis. The iodine decontamination factor (DF) assigned to the pool is taken to be a factor of 100. It can be inferred from the report upon which this factor is based (WCAP-7518-L) that this value is probably conservative by about a factor of three. Finally, plateout of volatile iodine released from the fuel into the gap and fuel rod plenum has been entirely neglected. Although not well quantified, a tentative estimate suggests that about 10 percent or less of the iodine released into the gap will remain volatile at the fairly low temperatures after the fuel has been allowed to cool for about a day or more.

Because of the significance of these conservatisms, the staff intends to study and quantify them in more detail and to use the results of such evaluations to appropriately revise the staff's Standard Review Plan (SRP), NUREG-0800. In the interim, the staff concludes that consideration of all three factors together noted above may permit a significant reduction of estimated thyroid doses compared to existing analyses. Adequate justification by licensees on a case-by-case basis, or by vendors on a generic basis, are likely to provide sufficient bases for departing from SRP criteria until such time as detailed changes can be made. A reduction by a factor of two is likely to be appropriate and conservative. Consequently, with regard to evaluation of thyroid doses for fuel-handling accidents involving extended-burnup fuel (>38,000 MWd/tonne), and pending SRP revision, it is likely that justification can be provided for lower estimates of thyroid doses from fuel handling accidents by a factor of two in departures from SRP review criteria.

7.0 REGULATORY POSITION

The review of General Electric Company's submittal, as described in NEDE-22148-P and responses to NRC questions in References 8 and 9, for application of their design criteria and analysis methods to extended burnups has been completed. This SER has not addressed capability of specific General Electric Company fuel designs to operate at extended burnup conditions. This will be addressed in staff SERs on these specific designs. Also, the validity of specific analytical methods for use at extended burnup will be addressed in SERs on these methods. As a result of our review, we conclude that GE should (1) continue to gather data on rod-to-tie plate clearance to confirm that design clearances are adequate for extended burnup operation (see Section 2.0(f) of this report). (2) Monitor comparisons of predicted and measured physics parameters (especially power distributions) which are monitored during the reactor cycle.

With the above provision, we conclude that the GE criteria and analysis methods, as described in the extended burnup topical report and Amendment 7 (Refs. 2 and 3), for extended burnup opertion are adequate such that (1) fuel damage is not expected to occur as a result of normal operation and anticipated operational occurrences, (2) fuel damage during postulated accidents would not be severe enough to prevent control rod insertion when it is required, and (3) core coolability will always be maintained even after postulated accidents. This conclusion is based on two primary factors:

(1) General Electric has provided sufficient evidence that the design criteria will allow for safe operation of GE design fuel at the proposed extended burnup levels; and

(2) these criteria will be met based on extended burnup operating experience, prototype testing, and analytical predictions.

8.0 REFERENCES

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2. Extended Burnup Evaluation Methodology, NEDE-22148 (Proprietary), General Electric Company, dated June 1982.

3. Letter from J. S. Charnley (GE) to F. J. Miraglia (NRC) "Proposed Revision to GE Licensing Topical Report NEDE 24011-P-A," dated February 25, 1983.

4. Letter from J. S. Charnley (GE) to C. O. Thomas (NRC), "Submittal of Proposed Amendment 10 to GE LTR NEDE-24011-P-A-6," dated November 30, 1984.

5. Letter, C. O. Thomas (NRC) to J. S. Charnley (GE), "Acceptance for Referencing of Licensing Topical Report NEDE-24011-P Amendment 7 to Revision 6, General Electric Standard Application for Reactor Fuel," dated March 1, 1985.

6. Letter from C. O. Thomas (NRC) to J. S. Charnley (GE), "Acceptance for Referencing of Licensing Topical Report NEDE-24011-P Amendment 7 to Revision 6, General Electric Standard Application for Reactor Fuel; SER Page Changes for Clarification," dated May 9, 1985.

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9. Letter from J. S. Charnley (GE) to C. O. Thomas (NRC), "Response to PNL Request for Additional Information on NEDE-22148-P," dated October 16, 1984.

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19. R. J. Williams, Assessment of Fuel-Rod Bowing in General Electric Boiling Water Reactors, NEDE-24284-P, General Electric Company, dated August 1980.

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21. Letter from J. S. Charnley (GE) to C. O. Thomas (NRC), "Responses to Request Number 1 for Additional Information on NEDE-24011-P-A-6, Amendment 10," dated March 11, 1985.

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