ENCLOSURE 4

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Licensing Topical Report

General Electric Standard Application for Reactor Fuel

(GESTAR II)

(Supplement for United States)

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S.1 Introduction

This supplement to the GESTAR II base document (Reference S–1) provides the safety analyses methodology and information specific to the GE boiling water reactor plants in the United States. A list of these plants with their associated reactor power, total number of fuel bundles, active fuel length, power density and the lattice type used in each reactor is given in Reference S–2.

Cycle–specific information for each plant reload is provided to the utility using the format given in Appendix A. No other plant–unique information is provided unless a portion of the reload does not conform to the generic document. Any deviation from the generic document will be designated in the supplemental reload licensing report and detailed in an Appendix or in a separate, referenced report to the submittal. The supplemental reload licensing report documents the number and designation of new and irradiated bundles. Plant– and cycle–specific information for initial cores is provided in the plant–specific FSAR.

Limits on plant operation are established to assure that the plant can be operated safely and not pose any undue risk to the health and safety of the public. This is accomplished by demonstrating that radioactive release from plants for normal operation, anticipated operational occurrences (AOOs) and postulated accidents meet applicable regulations in which conservative limits are documented. This conservatism is augmented by using conservative evaluation models and observing limits that are more restrictive than those documented in the applicable regulations.

Those AOOs which result in a significant reduction in MCPR or a large increase in the local power and the limiting accidents are described in this supplement along with other U.S. specific requirements as summarized in the following sections.

S.1.1 Analysis of Anticipated Operational Occurrences (AOOs) and Accidents

The effects of various postulated AOOs and accident events are investigated for a variety of plant conditions in Section S.2. The events have been categorized into three groups according to frequency of occurrence:

(1) Incidents of moderate frequency (anticipated operational occurrences).

(2) Infrequent incidents (unexpected operational occurrences).

(3) Accidents (limiting faults).

Only those events in category (1) are required to meet the design requirements for AOOs specified in Sections 2 and 4 of the GESTAR II base document (Reference S–1). Details on each of the three categories are discussed further in Section S.2.1. Descriptions of each of the significant AOO and accident events are discussed in Section S.2.2. The initial conditions and inputs to the analysis models for calculating the AOO events are discussed in Section S.2.3.

S.1.2 Vessel Pressure ASME Code Compliance

The ASME Boiler and Pressure Vessel Code and other codes and standards require that the pressure relief system prevent excessive overpressurization of the primary system process barrier and the pressure vessel. The allowable pressure and prescribed evaluations are determined by these requirements. The analysis performed to demonstrate conformance to the requirements is documented in Section S.3.

S.1.3 Stability Analysis

Stability requirements are set forth in 10CFR50 Appendix A, General Design Criterion (GDC). GDC 10 states that the reactor core and associated coolant, control, and protection systems shall be designed with appropriate margin to assure that specified acceptable fuel design limits are not exceeded during any condition of normal operation, including the effects of anticipated occurrences. GDC 12 states that power oscillations that can result in conditions exceeding specified acceptable fuel design limits either should not be possible, or can be reliably and readily detected and suppressed.

All US BWRs including BWRs 2-6 and the ABWR have selected one of the NRC approved BWROG long–term stability solutions described in References S–79, S–80, and S-104 to meet the GDC criteria. Long–term solutions are of the prevention type (i.e., power oscillations are not possible), the detect and suppress type (i.e., power oscillations can be reliably and readily detected and suppressed), or are a combination of the two types. Stability compliance with GDC 10 and GDC 12 must be demonstrated on a plant and cycle–specific basis for each of the long–term solutions.

If the long–term solution is declared inoperable due to Part 21 issues or hardware failures, the Interim Corrective Action (ICA) as outlined in Reference S–91 or an equivalent solution (e.g., the Backup Stability Protection (BSP) as outlined in Reference S–90) can be used on an interim basis. These are described in Section S.4.2.

The plant and cycle–specific calculations required for each long–term stability solution are described in Section S.4.1.

S.1.4 Analysis Options

Several analysis options are available, on a commercial basis, to all owners of BWRs fueled by GE. As these options are selected by the BWR owners, plant–specific and/or generic–bounding analyses will be submitted for NRC approval. The first set of options provides MCPR margin improvement. The second set of options provides additional operating flexibility for BWRs. In some cases, these options are included only to describe their impact on the reload license, and separate approval must be obtained before they can be used on a specific plant. The currently available options are discussed in Section S.5.

S.2 AOO and Accident Analysis

AOOs and accident events are divided among eight individual categories in the FSARs as required by Reference S–4. The categories are as follows.

(1) **Decrease in Core Coolant Temperature**: Reactor vessel water (moderator) temperature reduction results in an increase in core reactivity. This could lead to fuel–cladding damage.

(2) **Increase in Reactor Pressure**: Nuclear system pressure increases threaten to rupture the reactor coolant pressure boundary (RCPB). Increasing pressure also collapses the voids in the core–moderator, thereby increasing core reactivity. This could lead to fuel cladding damage.

(3) **Decrease in Reactor Core Coolant Flow Rate**: A reduction in the core coolant flow rate threatens to overheat the cladding as the coolant becomes unable to adequately remove the heat generated by the fuel.

(4) **Reactivity and Power Distribution Anomalies**: AOO events included in this category are those which cause rapid increases in power that are due to increased core flow disturbance events. Increased core flow reduces the void content of the moderator, thereby increasing core reactivity and power level.

(5) **Increase in Reactor Coolant Inventory**: Increasing coolant inventory could result in excessive moisture carryover to the main turbine, feedwater turbines, etc.

(6) **Decrease in Reactor Coolant Inventory**: Reductions in coolant inventory could threaten the fuel as the coolant becomes less able to remove the heat generated in the core.

(7) **Radioactive Release from a Subsystem or Component**: Loss of integrity of a radioactive containment component is postulated.

(8) **Anticipated Transients Without Scram**: In order to determine the capability of plant design to accommodate an extremely low probability event, a multi–system mal–operation situation is postulated.

All of the AOO and accident descriptions and analyses for initial cores are given in the plant FSAR. The purpose of this section is to discuss the significant AOOs and accidents for both initial and reload cores and classify them according to expected frequency of occurrence.

S.2.1 Frequency Classification

Each of the significant accidents and AOOs is assigned to one of the frequency groups outlined below. The frequency of occurrence of each event is summarized based upon the nuclear safety operational analysis and currently available operating plant history. The frequency classifications are as follows:

(1) **Incidents of moderate frequency** – These are incidents that may occur with a frequency greater than once per 20 years for a particular plant. This event is referred to as an "anticipated (expected) operational occurrence."

(2) **Infrequent incidents** – These are incidents that may occur during the life of the particular plant (spanning once in 20 years to once in 100 years). This event is referred to as an “abnormal (unexpected) operational occurrence.”

(3) **Limiting faults** – These are incidents that are not expected to occur but are postulated because their consequences may result in the release of significant amounts of radioactive material. This event is referred to as a “design basis (postulated) accident.”

S.2.1.1 Unacceptable Results for Incidents of Moderate Frequency

The following are considered to be unacceptable safety results for core–wide incidents of moderate frequency (AOOs):

(1) a release of radioactive material to the environs that exceeds the limits of 10CFR20;

(2) a reactor operation induced fuel cladding failure;

(3) nuclear system stresses in excess of that allowed for the transient classification by applicable industry codes; and

(4) containment stresses in excess of that allowed for the AOO classification by applicable industry codes.

Compliance to the above related fuel criteria (1) and (2) is conservatively demonstrated by conformance to the fuel design limits specified in Section 2 of the base document and by maintaining the MCPR above the Fuel Cladding Integrity Safety Limit MCPR identified in Reference S–2.

S.2.1.2 Unacceptable Results for Infrequent Incidents (Unexpected Operational Occurrences)

The following are considered to be unacceptable safety results for infrequent incidents (unexpected operational occurrences).

(1) release of radioactivity which results in dose consequences that exceed a small fraction (10%) of 10CFR100 (or 10% of 10CFR50.67 for Alternate Source Term plants);

(2) failure of fuel cladding which could cause changes in core geometry such that core cooling would be inhibited;

(3) generation of a condition that results in consequential loss of function of the reactor coolant system;

(4) generation of a condition that results in a consequential loss of function of a necessary containment barrier; and

(5) nuclear system stresses in excess of those allowed for the accident classification by applicable industry codes.

S.2.1.3 Unacceptable Results for Limiting Fault (Design Basis Accidents)

The following are considered to be unacceptable safety results for limiting faults (design basis accidents):

(1) radioactive material release which results in dose consequences that exceed the guideline values of 10CFR100;

(2) failure of fuel cladding which could cause changes in core geometry such that core cooling would be inhibited;

(3) nuclear system stresses in excess of those allowed for the accident classification by applicable industry codes;

(4) containment stresses in excess of those allowed for the accident classification by applicable industry codes when containment is required; and

(5) radiation exposure to plant operations personnel in the main control room in excess of 5 Rem whole body, 30 Rem inhalation and 75 Rem skin.

S.2.2 Descriptions and Frequency Categorization of Significant AOOs, Infrequent Incidents, and Accidents

S.2.2.1 Anticipated Operational Occurrences (Moderate Frequency Events)

To determine the limiting AOO events, the relative dependency of CPR upon various thermal–hydraulic parameters was examined. A sensitivity study was performed to determine the effect of changes in bundle power, bundle flow, subcooling, R–factor and pressure on CPR for fuel designs.

Results of the study are given in Table S–1. As can be seen from this table, CPR is most dependent on the R–factor and bundle power. A slight sensitivity to pressure and flow changes and relative independence to changes in inlet subcooling was also shown. The R–factor is a function of bundle geometry and local power distribution and is assumed to be constant throughout a transient. Therefore, AOOs that would be limiting because of MCPR would primarily involve significant changes in power. Based on this, the AOOs most likely to limit operation because of MCPR considerations are:

(1) generator load rejection without bypass or turbine trip without bypass;

(2) loss of feedwater heating or inadvertent HPCI startup;

(3) control rod withdrawal error;

(4) feedwater controller failure (maximum demand); and

(5) pressure regulator downscale failure (BWR/6 only).

Subsequent AOO analyses verified the results of the above sensitivity study. Descriptions of the typical analyses performed for the above limiting events are given below. For reloads, the potentially limiting events are evaluated to determine the required operating limits. The analytical results for the limiting AOOs and the required operating limits are provided in the plant supplemental reload licensing report.

Two additional fuel loading error conditions, the mislocated bundle and the misoriented bundle event, are evaluated as infrequent incidents. If the applicability requirements in Section S.5.3 for treating the fuel loading error as an infrequent incident cannot be met, then it will be evaluated to meet the fuel cladding integrity safety limit MCPR. Descriptions of these events are given in S.2.2.2.1 for the Infrequent Incident, and S.2.2.1.8 and .9 for the AOO.

Some plant–unique analyses will differ in certain aspects from the typical calculational procedure. These differences arise because of utility–selected margin improvement options. A description of these options and their effect upon the AOO analysis is given in Section S.5. ATWS pump trip is assumed in the analysis of those plants listed in Table S–2.

The initial MCPR assumed for AOO analyses is usually greater than or equal to the GETAB operating limit. Figure 5.2–1 in Appendix B illustrates the effect of the initial MCPR on transient ΔCPR for a typical BWR core. This figure indicates that the change in ΔCPR is approximately 0.01 for a 0.05 change in initial MCPR. Therefore, nonlimiting GETAB AOO analyses may be initiated from an MCPR below the operating limit because the higher operating limit MCPR more than offsets the increase in ΔCPR for the event. This may also be applied to limiting AOOs if the difference between the operating limit and the initial MCPR is small (0.01 or 0.02).

S.2.2.1.1 Generator Load Rejection Without Bypass

Fast closure of the turbine control valves is initiated whenever electrical grid disturbances occur which result in significant loss of load on the generator. The turbine control valves are required to close as rapidly as possible to prevent overspeed of the turbine generator rotor. The closing causes a sudden reduction of steam flow, which results in a nuclear system pressure increase. The reactor is scrammed by the fast closure of the turbine control valves.

Starting Conditions and Assumptions. The following plant operating conditions and assumptions form the principal bases for which reactor behavior is analyzed during a load rejection:

(1) The reactor and turbine generator are initially operating at full power when the load rejection occurs.

(2) All of the plant control systems continue normal operation.

(3) Auxiliary power is continuously supplied at rated frequency.

(4) The reactor is operating in the manual flow control mode when load rejection occurs, although the results do not differ significantly for operation in the automatic flow control mode.

(5) The turbine bypass valve system is failed in the closed position.

Event Description. Complete loss of the generator load produces the following sequence of events:

(1) The power/load unbalance device steps the load reference signal to zero and closes the turbine control valves at the earliest possible time. The turbine accelerates at a maximum rate until the valves start to close. The turbine control valves on plants with electrical hydraulic turbine control (EHC) will close at a full stroke rate of approximately 0.150 sec. The turbine control valve on plants with a mechanical hydraulic turbine control (MHC) system will have a nonlinear closure signature that is a function of the MHC settings.

(2) Reactor scram is initiated upon sensing control valve fast closure.

(3) If the pressure rises to the pressure relief setpoint, part or all of the relief valves open, discharging steam to the suppression pool.

(4) On some plants, if the pressure rises above approximately 1135 psig, a trip of the recirculation pump drive motors occurs.

Identification of Operator Actions. No restart is assumed and the reactor is to be cooled down.

The operator should take the following actions:

(1) Control the reactor pressure.

(2) Ascertain that all control rods are in and that recirculation flow is at minimum.

(3) Put the reactor mode switch in the startup position before the reactor pressure decays to ≤ 850 psig.

(4) Secure the RCIC or emergency condenser if feedwater pumps are available.

(5) Check the necessity of starting the residual heat removal (RHR) system.

(6) Maintain turbine seals and steam jet air ejector (SJAE) operation.

(7) Check the turbine coastdown.

(8) When the reactor pressure decays to less than 300 psig, maintain the reactor water level using the condensate pump only, and continue steaming to the seals and SJAE until the shutdown cooling system is put into service.

(9) When the reactor is depressurized, close the main steam isolation valves (MSIVs) for maintenance on the bypass valves.

(10) Monitor torus temperature and take appropriate actions as described in the Technical Specifications.

Results and Consequences. For initial cores, the generator load rejection without bypass event is calculated and the ΔCPR results reported in the plant FSAR. For reload cores, an evaluation is performed to determine if this AOO could potentially alter the previous cycle MCPR operating limit. If it does, the results will be reported in the supplemental reload licensing report. Plant/cycle–specific results are determined using either the GENESIS, GEMINI or TRACG methods described in Subsection 4.3.1 of the GESTAR base document (Reference S–1) for reload cores.

S.2.2.1.2 Turbine Trip Without Bypass

A variety of turbine or nuclear system malfunctions will initiate a turbine trip. Some examples are: moisture separator and heater drain tank high levels, large vibrations, loss of control fluid pressure, low condenser vacuum and reactor high water level. The turbine stop valve closes, causing a sudden reduction in steam flow that results in a nuclear system pressure increase and the shutdown of the reactor.

Starting Conditions and Assumptions. The plant operating conditions and assumptions are identical to those of the generator load rejection.

Event Description. The sequence of events for a turbine trip is similar to those for a generator load rejection. Stop valve closure occurs over a typical period of 0.10 second.

Position switches at the stop valves sense the turbine trip and initiate reactor scram. If the pressure rises to the pressure relief setpoint, relief valves open, discharging steam to the suppression pool.

Identification of Operator Actions. Key operator actions required following the turbine trip without bypass are the same as required following a generator load rejection without bypass.

Results and Consequences. For initial cores, the turbine trip without bypass event is calculated and the ΔCPR results reported in the plant FSAR. For reload cores, an evaluation is performed to determine if this AOO could potentially alter the previous cycle MCPR operating limit. If it does, the results will be reported in the supplemental reload licensing report. Plant/cycle–specific results are determined using either the GENESIS, GEMINI or TRACG methods described in Subsection 4.3.1 of the GESTAR base document (Reference S–1).

S.2.2.1.3 Loss of Feedwater Heating

A loss of feedwater heating event results in a core power increase due to the increase in core inlet subcooling.

Starting Conditions and Assumptions. The following plant operating conditions and assumptions form the principal basis for which reactor behavior is analyzed during the loss of feedwater heating transient:

(1) The plant is operating at full power.

(2) The plant is operating in the manual flow control mode. The transient is moderated by the runback in core flow if operation is in the automatic flow control mode.

Event Description. Feedwater heating can be lost in at least two ways:

(1) Steam extraction line to heater is closed.

(2) Feedwater is bypassed around heater.

The first case produces a gradual cooling of the feedwater due to the stored heat capacity of the heater. In the second case, the feedwater bypasses the heater and the change in heating occurs during the stroke time of the bypass valve (about one minute, similar to the heater time constant). In either case, the reactor vessel receives cooler feedwater. The maximum number of feedwater heaters that can be tripped or bypassed by a single event represents the most severe transient for analysis considerations and the feedwater heaters are assumed to trip instantaneously. This event causes an increase in core inlet subcooling, which increases core power due to the negative void reactivity coefficient. In automatic recirculation flow control, some compensation of core power is realized by automatic reduction of core flow.

Identification of Operator Actions. For either case, power would increase at a very moderate rate. If power exceeded the normal power flow control line, the operator would be expected to reduce recirculation flow to return the power below its initial value, and subsequently insert control rods to return to operation within the normal power/flow range. If these steps were not done, the neutron flux could exceed the scram setpoint where a scram would occur.

Results and Consequences. For initial cores, the loss of feedwater heating event is calculated and the ΔCPR results reported in the plant FSAR. For reload cores, an evaluation is performed to determine if this AOO could potentially alter the previous cycle MCPR operating limit. If it does, the results will be reported in the supplemental reload licensing report. Plant/cycle–specific results are determined using the REDY, TRACG or 3–D Simulator model.

S.2.2.1.4 Inadvertent Start of HPCI Pump (Plants with HPCI only)

This AOO is similar to the loss of feedwater heater event. The high pressure coolant injection pump is inadvertently started and the cold water injection results in an increase in inlet

subcooling and a consequent increase in power. In most cases this event is bounded by the loss of feedwater heater event (Reference S–78).

Starting Conditions and Assumptions. The plant operating conditions and assumptions are identical to those of the loss of feedwater heater.

Event Description. The HPCI introduces cold water through the feedwater sparger. The normal feedwater flow is correspondingly reduced by the water level controls. The increase in inlet subcooling due to the inadvertent HPCI start is slightly less than that produced by the loss of feedwater heater event.

Identification of Operator Actions. The operator actions would be similar to those performed for the loss of feedwater heating event. In addition, the operator should determine the reason why the HPCI flow was initiated and follow proper procedures to shut off the pumps.

Results and Consequences. For initial cores, the inadvertent start of HPCI Pump event is calculated and the ΔCPR results reported in the plant FSAR. For reload cores, an evaluation is performed to determine if this AOO could potentially alter the previous cycle MCPR operating limit. If it does, the results will be reported in the supplemental reload licensing report. The REDY, ODYN system model or the TRACG system model may also be used to simulate this event. These models are described in Subsection 4.3.1 of the GESTAR base document (Reference S–1).

S.2.2.1.5 Rod Withdrawal Error

Starting Conditions and Assumptions. The reactor is operating at a power level above 75% of rated power at the time the control rod withdrawal error occurs. The reactor operator has followed procedures and up to the point of the withdrawal error is in a normal mode of operation (i.e., the control rod pattern, flow setpoints, etc. are all within normal operating limits). For these conditions, it is assumed that the withdrawal error occurs with the maximum worth control rod. Therefore, the maximum positive reactivity insertion will occur.

Event Description. While operating in the power range in a normal mode of operation, the reactor operator makes a procedural error and withdraws the maximum worth control rod to its rod block position. Due to the positive reactivity insertion, the core average power increases. More importantly, the local power in the vicinity of the withdrawn control rod increases and could potentially cause cladding damage due to overheating, which may accompany the occurrence of boiling transition, which is an assumed AOO failure threshold. The following list depicts the sequence of events for this AOO.

(1) Event begins, operator selects the maximum worth control rod, acknowledges any alarms and withdraws the rod at the maximum rod speed.

(2) Core average power and local power increase causing LPRM alarm.

(3) Event ends – rod block by RBM or RWL.

Identification of Operator Actions. Under most normal operating conditions, no operator action will be required, since the transient that occurs will be mild. If licensing limits are exceeded, the nearest local power range monitors (LPRMs) will detect this phenomenon and sound an alarm. The operator must acknowledge this alarm and take appropriate action to rectify the situation.

If the rod withdrawal error is severe enough, the rod block monitor (RBM) system will sound alarms, at which time the operator must acknowledge the alarms and take corrective action. Even for extremely severe conditions (i.e., for highly abnormal control rod patterns, operating conditions and assuming that the operator ignores all alarms and warnings and continues to withdraw the control rod), the RBM system will block further withdrawal of the control rod before the fuel cladding integrity safety limit is exceeded.

Results and Consequences. For BWR/3, 4 and 5 plants, the ΔCPR from a rod withdrawal error is reported for each fuel type. The value reported for a particular fuel type may be from either a plant/cycle–specific analysis or the generic bounding analysis. The rod withdrawal error has been analyzed generically for BWR/6's in Reference S–5 or may be analyzed on a plant specific basis. The applicability of these generic analyses to GE fuel designs is discussed in Reference S–2.

a. Plant/Cycle–Specific Analysis

The plant/cycle–specific analysis considers the continuous withdrawal of the maximum worth control rod at its maximum drive speed from the reactor, which is operating at rated power with a control rod pattern which results in the core being placed on thermal design limits. This condition is analyzed to ensure that the results obtained are conservative; this approach also serves to demonstrate the function of the RBM system.

Results for this worst–case condition for the reload will be given in the supplemental reload licensing report. Results for late in cycle reactivity limited control rod pattern based rod withdrawal error analyses may also be reported to provide appropriate late in cycle ΔCPRs.

b. Generic Bounding Analysis (BWR/3, 4 and 5 only)

Based on the large amount of data available from past reloads, a statistical analysis was performed to calculate generic bounding values of ΔCPR as a function of rod block monitor setpoint (Reference S–6). These values are listed in Table S–3. Interim approval of this method is provided in Reference S–7. When this basis is used, the ΔCPRs are conservative relative to the actual operating limit MCPR and are valid throughout the cycle. The applicability of the generic analysis to GE fuel designs is discussed in Reference S–2.

In cases where the generic bounding analysis results in a ΔCPR that is the limiting value for a particular fuel lattice type, a plant/cycle–specific analysis may be performed for that lattice type.

S.2.2.1.6 Feedwater Controller Failure – Maximum Demand

This event is postulated on the basis of a single failure of a control device; specifically, one that can directly cause an increase in coolant inventory by increasing the feedwater flow. The most severe applicable event is a feedwater controller failure during maximum flow demand. The feedwater controller is forced to its upper limit at the beginning of the event.

Starting Conditions and Assumptions. The starting conditions and assumptions considered in this analysis are as follows:

a. Feedwater controller fails during maximum flow demand.

b. Maximum feedwater pump runout is assumed.

c. The reactor is operating in a manual flow control mode, which provides for the most severe transient.

Event Description. A feedwater controller failure during maximum demand produces the following sequence of events:

a. The reactor vessel receives an excess of feedwater flow.

b. This excess flow results in an increase in core subcooling, which results in a rise in both core power and reactor vessel water level.

c. The rise in the reactor vessel water level eventually leads to high water level turbine trip, feedwater pump trip and reactor scram trip.

Identification of Operator Actions. Under most conditions, no operator action will be required. The reactor will scram on high water level and end the transient.

Results and Consequences. The influx of excess feedwater flow results in an increase in core subcooling that reduces the void fraction and thus induces an increase in reactor power. The excess feedwater flow also results in a rise in the reactor water level, which eventually leads to high water level, reactor scram, main turbine and feedwater turbine trip and turbine bypass valves being actuated. Reactor scram trip is actuated from the main stop valve position switches for plants without high water level trip. Relief valves open as steamline pressures reach relief valve setpoints.

For initial cores, the feedwater controller failure event is calculated and the ΔCPR results reported in the plant FSAR. For reload cores, an evaluation is performed to determine if this AOO could potentially alter the previous cycle MCPR operating limit. If it does, the results are reported in the supplemental reload licensing report. Plant/cycle–specific results are determined using either the GENESIS, GEMINI or TRACG methods described in Subsection 4.3.1 of the GESTAR base document (Reference S–1).

S.2.2.1.7 Pressure Regulator Downscale Failure (BWR/6 Plants Only)

Two identical pressure regulators are provided to maintain primary system pressure control. They independently sense pressure just upstream of the main turbine stop valves and compare it to two separate setpoints to create proportional error signals that produce each regulator output. The output of both regulators feeds into a high value gate. The regulator with the highest output controls the main turbine control valves. The lowest pressure setpoint gives the largest pressure error and, thereby, largest regulator output. The backup regulator is set 5 psi higher, giving a slightly smaller error and a slightly smaller effective output of the controller.

It is assumed, for the purpose of this AOO analysis, that a single failure occurs which causes a downscale failure of the pressure regulation demand to zero (e.g., high value gate downscale failure). Should this occur, it could cause full closure of turbine control valves, as well as inhibit steam bypass flow and thereby increase reactor power and pressure. When this occurs, reactor scram will be initiated when the high neutron flux scram setpoint is reached.

This AOO is not applicable to plants with the MEOD flexibility option (see Section S.5.2.7). The MEOD evaluation concluded that the single failure initiating this AOO was very remote and did not meet the probability requirements. The pressure control of each applicable plant is reviewed to insure that it is consistent with the MEOD basis.

Starting Conditions and Assumptions. The following plant operating conditions and assumptions form the principal bases for which reactor behavior is analyzed for this event:

(1) The reactor and turbine generator are initially operating at full power when downscale failure of the pressure regulator occurs.

(2) All of the plant control systems function normal.

(3) The reactor is operating in the manual flow control mode when load rejection occurs, although the results do not differ significantly for operation in the automatic flow control mode.

Event Description. Pressure regulation downscale failure produces the following sequence of events:

(1) A failure occurs such that the high value gate receives a zero demand signal, which initiates a turbine control valve closure.

(2) Neutron flux reaches high flux scram setpoint and initiates a reactor scram.

(3) Recirculation pump drive motors are tripped due to high dome pressure. Safety/relief valves also open due to high pressure.

(4) Vessel water level trip initiates main turbine and feedwater turbine trips.

(5) Group 1 safety/relief valves open again to relieve decay heat and then reclose.

Identification of Operator Action. The operator should:

(1) monitor that all control rods are inserted;

(2) monitor reactor water level and pressure;

(3) observe turbine coastdown and break vacuum before the loss of steam seals (check turbine auxiliaries);

(4) observe that the reactor pressure relief valves open at their setpoint; and

(5) monitor reactor water level and continue cooldown per the normal procedure.

Results and Consequences. For initial cores, the pressure regulator downscale failure event is calculated and the ΔCPR results reported in the plant FSAR. For reload cores, an evaluation is performed to determine if this AOO could potentially alter the previous cycle MCPR operating limit. If it does, the results will be reported in the supplemental reload licensing report. Plant/ cycle–specific results are determined using either the GENESIS, GEMINI or TRACG methods described in Subsection 4.3.1 of the GESTAR base document (Reference S–1).

S.2.2.1.8 Mislocated Bundle Event

If the mislocated bundle event cannot be evaluated as an Infrequent Incident per Section S.5.3, then the event is evaluated as an AOO for initial cores and reload cores where the resultant CPR response may establish the operating limit MCPR (OLMCPR). The evaluation of the mislocated bundle as an infrequent incident is discussed in Section S.2.2.2.1.

Starting Conditions and Assumptions. Proper location of the fuel assembly in the reactor core is monitored during fuel movements and verified by procedures during core loading. Verification procedures include inventory checks, current bundle location logs, serial number verifications and visual or photographic inspection of the loaded core. The verification procedures are designed to minimize the possibility of the occurrence of the mislocated bundle accident. Plant operation with a mislocated fuel bundle is a result of a failure in the core verification process following core fueling.

Event Description.

For Initial Cores. The initial core consists of bundle types with average enrichments in the high, medium or low range with correspondingly different gadolinia concentrations. The fuel bundle loading error involves interchanging a bundle of one enrichment with another bundle of a different enrichment. The following fuel loading errors are possible in an initial core:

(1) A high–enrichment bundle is misloaded into low–enrichment bundle location.

(2) A medium–enrichment bundle is misloaded into a low–enrichment bundle location.

(3) A low–enrichment bundle is misloaded into a high–enrichment bundle location.

(4) A low–enrichment bundle is misloaded into a medium–enrichment bundle location.

(5) A medium–enrichment bundle is misloaded into a high–enrichment bundle location.

(6) A high–enrichment bundle is misloaded into a medium–enrichment bundle location.

Because all low–enrichment bundles are located on the core periphery, the misloading of high– or medium–enrichment bundles into a low–enrichment bundle location [misloading errors (1) or (2)] is not significant. In these cases, the higher reactivity bundles are moved to a region of low reactivity and power resulting in an overall improvement in performance and no impact on thermal margin.

The third type of fuel loading error results in the largest enrichment mismatch. For initial cores using thermal traversing in–core probes (TIPs), this loading error does not result in an unacceptable operating consequence. Consider a fuel bundle loading error at beginning–of–cycle (BOC) and the low–enrichment bundle interchanged with a high–enrichment bundle located adjacent to the Local Power Range Monitor (LPRM) and predicted to be closest to technical specification limits. After the loading error has occurred and has gone undetected, assume, for purposes of conservatism, that the operator uses a control pattern that places the limiting bundle in the four–bundle array containing the misplaced bundle on thermal limits as recorded by the LPRM. As a result of loading the low–enrichment bundle in an improper location, the average power of the four bundles decreases. Normally, the reading of the LPRM will show a decrease in thermal flux due to the decreased power; however, in this case, an increase in the thermal flux occurs due to decreased neutron absorption in the low–enrichment bundle. The effect of the decreased thermal absorption is larger than the effect of power depression resulting in a net increase in the instrument reading. Thus, detected reductions in thermal margins during power operations will indicate a fuel loading error of this kind.

The fourth and fifth types of fuel loading errors are similar to the third type and also result in conservative operating errors.

The fuel bundle loading error with greatest impact on thermal margin is of the sixth type, which occurs when a high–enrichment bundle is interchanged with a medium–enrichment bundle located away from an LPRM. Since the medium– and high–enrichment bundles have corresponding medium and high gadolinia contents, the maximum reactivity difference occurs at the end of cycle (EOC) when the gadolinia has burned out.

For Reload Cores. The loading error involves the mislocation of at least two fuel bundles. One location is loaded with a bundle that would potentially operate at a lower critical power than it would otherwise. The other location would operate at a higher critical power. The low critical power location could have less margin to boiling transition than other bundles in the core; therefore, the MCPR operating limit is set to protect against this occurrence.

Identification of Operator Actions. There is a possibility that core monitoring will provide information that allows the operator or reactor engineer to recognize that an error exists and

determine appropriate mitigating actions. Where the high radial power mislocated bundle is adjacent to an instrument, the power adjustment in radially TIP or LPRM adapting monitoring systems will cause higher monitored bundle power. The reactor will be operated such that the most limiting of the bundles near the mislocation will be maintained below the operating limit MCPR. Where the mislocated bundle has a bundle between it and the instrument, the core monitoring may not recognize the mislocation.

If loading errors were made and have gone undetected, the operator would assume that the mislocated bundle would operate at the same power as the instrumented bundle in the mirror–image location and would operate the plant until EOC. For the purpose of conservatism, it is assumed that the mirror–image bundle is on thermal limits as recorded by the LPRM. As a result of placing the instrumented bundle on limits, the mislocated bundle may violate the Tech Spec operating limit MCPR.

Results and Consequences. Assuming the mislocated bundle is not identified, it is possible that the fuel bundle operates through the cycle close to or above the fuel thermal mechanical limit. Therefore, the MCPR operating limit is set to protect against this occurrence.

Further discussion on the analysis methods for the mislocated bundle accident is given in References S–45 and S–46.

S.2.2.1.9 Misoriented Bundle Event

If the misoriented bundle event cannot be evaluated as an Infrequent Incident per Section S.5.3, then the event is evaluated as an AOO for initial cores and reload cores where the resultant CPR response may establish the operating limit MCPR (OLMCPR). The evaluation of the misoriented bundle as an infrequent incident is discussed in Section S.2.2.2.1.

Starting Conditions and Assumptions. Proper orientation of the fuel assembly in the reactor core is monitored during fuel movements and verified by procedures during core loading. Five separate visual indications of proper fuel assembly orientation exist:

(1) The channel fastener assemblies, including the spring and guard used to maintain clearances between channels, are located at one corner of each fuel assembly adjacent to the center of the control rod.

(2) The identification boss on the fuel assembly handle points toward the adjacent control rod.

(3) The channel spacing buttons are adjacent to the control rod passage area.

(4) The assembly identification numbers that are located on the fuel assembly handles are all readable from the direction of the center of the cell.

(5) There is cell–to–cell replication.

Event Description. This fuel loading error involves the misorientation of a single fuel bundle. The power distribution in the misoriented bundle would be affected as well as its neighbors. The resulting power distribution could reduce the margin to boiling transition.

Identification of Operator Actions. There is a possibility that core monitoring will provide information that allows the operator or reactor engineer to recognize that an error exists and determine appropriate mitigating actions. If loading errors were made and have gone undetected, the plant would continue to operate until EOC.

Results and Consequences. Assuming the mislocated bundle is not identified, it is possible that the fuel bundle operates through the cycle close to or above the fuel thermal mechanical limit. Therefore, the MCPR operating limit is set to protect against this occurrence.

Analysis methods for the misoriented fuel assembly are discussed in detail in Reference S–46. Approval of these methods is given in Reference S–47 under the stipulation that a ΔCPR penalty of 0.02 be added for the tilted misoriented bundle. This 0.02 is added on to the calculated ΔCPR used in determining the operating limit when utilizing this method. GE applies the Fuel Cladding Integrity Safety Limit discussed in Section 4 of the base GESTAR II document (Reference S–1) and presented in Reference S–2 to the accident results reported in the plant FSAR or the supplemental reload licensing report. Individual utilities may elect to substitute an alternative approach as noted in Reference S–47 for this limit.

S.2.2.2 Unexpected Operational Occurrences (Infrequent Incidents)

S.2.2.2.1 Fuel Loading Error (Mislocated or Misoriented Bundle Event)

A generic bounding analysis of the fuel loading error (mislocated or misoriented bundle event) is provided in Reference S-99. The plant must meet the requirements of Section S.5.3 in order to apply this generic analysis. If the plant cannot meet the requirements of S.5.3, then the mislocated or misoriented bundle is evaluated as discussed in Sections S.2.2.1.8 and .9.

Starting Conditions and Assumptions. Proper location and orientation of the fuel assemblies in the reactor core is monitored during fuel movements and verified by procedures during core loading. Verification procedures address location, orientation, and seating through visual examinations of the loaded core. The verification procedures are designed to minimize the possibility of the occurrence of the mislocated or misoriented bundle event. Plant operation with a mislocated or misoriented fuel bundle is a result of a failure in the core verification process following core fueling.

Event Description. The description of the mislocated or misoriented fuel bundle event is the same as in Sections S.2.2.1.8 and .9 except that for the infrequent incident it is assumed that the event proceeds to cause fuel failures.

Identification of Operator Actions. The initial core power distribution indications and possible operator actions are the same as in Sections S.2.2.1.8 and .9. Should fuel failures occur, the offgas activity quickly increases. At that point, the operator would take steps to reduce power or scram the reactor to reduce or terminate the release.

Results and Consequences. Reference S-99 provides a bounding analysis based on a very conservative assumption of all of the fuel rods failing in five fuel bundles. Two scenarios for the fuel loading error were considered. The first assumed that the fission product activity is

airborne in the turbine and condenser following Main Steam Isolation Valve (MSIV) closure and leaks directly from the condenser to the atmosphere. In the second scenario, it was assumed that no automatic MSIV closure occurred and that the activity was transported to an augmented offgas system. Calculations of post–accident doses for the Exclusion Area Boundary (EAB) were performed for each scenario to compare radiological consequences with the applicable exposure limits. EAB doses were also calculated for both scenarios utilizing the alternate source term methodology.

The plant specific offgas system parameters and site atmospheric dispersion parameters are used to confirm the applicability of the EAB generic analysis. A conservative analysis for the control room dose was also established such that plant specific atmospheric dispersion parameters can also be used to confirm its applicability. Section S.5.3 defines the items that must be confirmed and documented with the reload design documentation to support application of the Infrequent Incident analysis option.

S.2.2.3 Design Basis Accidents (Limiting Faults)

In this category, evaluations of less frequent postulated events are made to assure an even greater depth of safety. Accidents are events that have a projected frequency of occurrence of less than once in every one hundred years for every operating BWR. The broad spectrum of postulated accidents is covered by five categories of design basis events. These events are the control rod drop, loss–of–coolant, main steam line break, one recirculation pump seizure, and refueling accident.

S.2.2.3.1 Control Rod Drop Accident Evaluation

There are many ways of inserting reactivity into a boiling water reactor; however, most of them result in a relatively slow rate of reactivity insertion and therefore pose no threat to the system. It is possible, however, that a rapid removal of a high worth control rod could result in a potentially significant excursion; therefore, the accident that has been chosen to encompass the consequences of a reactivity excursion is the control rod drop accident (CRDA). The dropping of the rod results in a high local reactivity in a small region of the core and for large, loosely coupled cores, significant shifts in the spatial power generation during the course of the excursion.

S.2.2.3.1.1 Sequence of Events

The sequence of events and approximate time of occurrence for this accident are described below:

Banked Position Withdrawal Sequence (BPWS) Plants — Event	Approximate Elapsed Time
(a) Reactor is at a control rod pattern corresponding to maximum increment rod worth.	–
(b) Rod pattern control systems (Rod Worth Minimizer, Rod Sequence Control System, or Rod Pattern Controller) or operators are functioning within constraints of BPWS. The control rod that will result in the maximum incremental reactivity worth addition at any time in core life under any operating condition while employing the BPWS becomes decoupled from the control rod drive.	–
(c) Operator selects and withdraws the drive of the decoupled rod along with the other required control rods assigned to the Banked–position group such that the proper core geometry for the maximum incremental rod worth exists.	–
(d) Decoupled control rod sticks in the fully inserted position.	–
(e) Control rod becomes unstuck and drops at the maximum velocity determined from experimental data (3.11 fps).	0
(f) Reactor goes on a positive period and initial power burst is terminated by the Doppler reactivity feedback.	≤ 1 sec.
(g) APRM 120% power signal scrams reactor (conservative; in startup mode APRM scram would be operative + IRM).	–
(h) Scram terminates accident.	≤ 5 sec.

(1) **Banked Position Withdrawal Sequence (BPWS) Plants**:

All plants except those listed in Table S–4 utilize the BPWS. Those Group Notch plants in Table S–4 that have modified their Rod Worth Minimizer (RWM) and have provided a separate submittal to the NRC, which enforces the BPWS as described in Reference S–9, are included in this group.

Those plants listed in Table S–4 that have implemented the modifications described in Reference S–10 are also included in this group. Plants that implement the modifications described in Reference S–10 must modify their technical specifications to assure high operability of their rod pattern control system, review procedures and quality control for second operator substitution, and provide a discussion of this review to the NRC.

To limit the worth of the rod that would be dropped in a BPWS plant, the rod pattern control systems are used below the plant–specific low power setpoint to enforce the rod withdrawal sequence. These systems are programmed to follow the

bank position withdrawal sequences (BPWS), which are generically defined in Reference S–11. Plants that have implemented the BPWS in accordance with Reference S-11 may also implement the Improved BPWS Control Rod Insertion Process as defined in Reference S-100.

Group Notch Plants — Event	Approximate Elapsed Time
(a) Reactor is at a control rod pattern corresponding to maximum increment rod worth.	–
(b) Rod worth minimizer is not functioning. Maximum worth control blade that can be developed at any time in core life under any operating conditions with the group notch RSCS operational becomes decoupled from the control rod drive.	–
(c) Operator selects and withdraws the control rod drive of the decoupled maximum worth rod along with the other required control rods assigned to its Rod Sequence Control System group such that the proper core geometry for the maximum incremental rod worth exists.	–
(d) Decoupled control rod sticks in the fully inserted position.	–
(e) Blade becomes unstuck and drops at the maximum velocity determined from experimental data (3.11 fps).	0
(f) Reactor goes prompt critical and initial power burst is terminated by the Doppler reactivity feedback.	≤ 1 sec.
(g) APRM 120% power signal scrams reactor (conservative; in startup mode APRM scram would be operative + IRM).	–
(h) Scram terminates accident.	≤ 5 sec.

(2) **Group Notch Plants**

Plants listed in Table S–4 are Group Notch plants. Those Group Notch plants that enforce the BPWS as described in Reference S–9, are not included in this group. In addition, those plants that have implemented the modifications of Reference S–10 are also not included in this group.

To limit the worth of the rod that could be dropped in a group notch plant that has not implemented the modifications of Reference S–10, a group notch rod sequence control system (RSCS) is installed to control the sequence of rod withdrawal. This system prevents the movement of an out–of–sequence rod before the 50% rod density configuration is achieved (except for plants operating in the BPWS mode described in Reference S–9), and prevents high–control rod worth beyond the 50% rod density configuration by enforcing a group notch mode of rod withdrawal. The 50% rod density configuration occurs during each reactor startup and corresponds to a “checkerboard” rod pattern in which 50% of the rods are fully inserted in the core and 50% are fully withdrawn. The rod drop accident design limit restricts peak

enthalpies in excess of 280 cal/gm for any possible plant operation or core exposure.

S.2.2.3.1.2 Analytical Methods.

Techniques and models used to analyze the control rod drop accident (CRDA) are documented in References S–12, S–13, S–14 and S–9. The information in these documents has been used for the development of design approaches to make the consequences of CRDA acceptable.

(1) Banked Position Withdrawal Sequence (BPWS) Plants

Control rod drop accident (CRDA) results from BPWS plants have been statistically analyzed and documented in Reference S–15. The results show that, in all cases, the peak fuel enthalpy in an RDA would be much less than the 280–cal/gm design limit even with a maximum incremental rod worth corresponding to 95% probability at the 95% confidence level. Based on these results, it was proposed to the NRC, and subsequently found acceptable, to delete the CRDA from the standard GE BWR reload package for the BPWS plants.

Because of the large margin available to CRDA design limits for BPWS plants, implementation of the advanced physics methods (Reference S–16) does not result in challenging the 280–cal/gm limit. Therefore, the impact of using the advanced physics methods of Reference S–16 as compared to the physics methods described in Reference S–17 on the generic BPWS analysis, is considered negligible. Applicability of the generic BPWS analysis to GE fuel designs is given in Reference S–2.

(2) Group Notch Plants

For group notch plants not operating in the BPWS mode described in Reference S–9 or that have not implemented the modifications described in Reference S–10, the highest control rod worth in the cold condition is determined for a series of rod drop states. Hot–standby cases are also run for any cold case that is not subcritical. The resultant peak fuel enthalpy for cold and, if applicable, hot–standby is then determined. This enthalpy value is then compared to the 280 cal/gm RDA design limit. The CRDA calculational procedures are independent of whether the physics models of either Reference S–17 or Reference S–16 are used.

Group notch plants operating in the BPWS mode described in Reference S–9 or those plants that have implemented the modifications of Reference S–10 can reference the statistical CRDA analysis documented in Reference S–15. This will allow these plants to delete the CRDA analysis from the standard GE–BWR reload package.

Results of the analysis for reload cores are supplied in the specific plant supplemental reload licensing report. For those group notch plants not operating in

the BPWS mode described in Reference S–9, these results include the resultant peak enthalpy in the cold and, if applicable, the hot–standby condition.

S.2.2.3.1.3 Effect of Fuel Densification

The effect of axial gap formation due to fuel densification on the rod drop accident results is discussed in Reference S–18. Based on this evaluation, it has been established that there is a 99% probability that increased local peaking in any fuel rod due to the formation of axial gaps will be less than 5%. This effect has been accommodated by adjusting the local peaking factor.

S.2.2.3.1.4 Results and Consequences

Results of radiological analyses for initial cores are reported in the FSAR. For reloads, based on a bounding analysis, it was conservatively determined that 850[1] fuel rods would reach a fuel enthalpy of 170 cal/gm, which is the enthalpy limit for eventual cladding perforation. Safety analysis reports written prior to the development of the model and techniques reported previously, and those used to predict the 850 failures, resulted in the failure of approximately 330 fuel rods for the 7x7 fuel. Based on these new models and assumptions, the resultant number of failures for a 7x7 core would be 660 fuel rods. If the conservative assumption is made that the fractional plenum activity for 8x8, 8x8R, P8x8R, and BP8x8R fuel is the same as for the 7x7 fuel, the resultant increase in activity released from the 8x8 fuel and the subsequent radiological exposures relative to 7x7 analysis for the failure of 330 rods is (850/330) (49/63) = 2 times the 7x7 analysis. As noted in the FSAR, even if the radiological exposures are increased by a factor of two, the effects are still orders of magnitude below those identified in 10CFR100. The radiological consequences of the CRDA, assuming a full core of more recent GE fuel designs, are discussed in Reference S–2.

Results of the enthalpy analysis for initial cores are reported in the FSAR.

Results of the analysis for reload cores are supplied in the specific plant supplemental reload licensing report. For group notch plants that are not operating in the BPWS mode described in Reference S–9 or that have not implemented the modifications of Reference S–10, these results include the resultant peak enthalpy in the cold and, if applicable, the hot–standby condition.

S.2.2.3.2 Loss–of–Coolant Accident

Two separate emergency core cooling system (ECCS) evaluation methodologies are available to determine the effects of the loss–of–coolant accident (LOCA) in accordance with the requirements of 10CFR50.46 and Appendix K. Either methodology can be used to calculate the LOCA results. The particular method used is the utility's option and depends upon economic and not safety considerations. The method used will be indicated in the FSAR for

[1] Includes a 10% allowance for uncertainties in the calculation.

initial cores or the supplemental reload licensing report for each cycle (see Appendix A of country–specific supplement).

The first methodology (SAFE/REFLOOD), identified in Sections S.2.2.3.2.1 and S.2.2.3.2.3 and discussed in detail in Reference S–19, utilizes conservative thermal–hydraulic/heat transfer correlations and conservative bounding values for key inputs. The resulting calculated peak cladding temperature (PCT) consists of compounded conservatisms and therefore is unrealistically high. However, as long as the resultant PCT is less than 2200°F (10CFR50.46 limit) and plant operation is not unduly restricted in order to remain under that limit, then this conservative method may satisfy utility needs.

The second methodology (SAFER/GESTR), identified in Sections S.2.2.3.2.4 and S.2.2.3.2.5, utilizes improved ECCS evaluation models (References S–20 and S–21) along with a realistic application approach (Reference S–22) to calculate a licensing PCT with margin substantiated by statistical considerations. Nominal values are used for most inputs, and Appendix K required inputs are utilized only for the limiting break in order to establish a licensing margin to 10CFR50.46 limits. This methodology was revised in Reference S–74 to extend the application to non–jet pump plants. Use of this improved methodology is optional and is dependent upon economic benefits and not safety concerns.

S.2.2.3.2.1 SAFE/REFLOOD LOCA Model Descriptions

Five different GE computer models are utilized to calculate LOCA analysis results for a BWR. Conservative values are used along with required Appendix K criteria as input to these models. The models are summarized below and discussed in detail in Reference S–19. NRC approval of this LOCA model and calculational procedure is given in Reference S–23. These models are applicable to prepressurized fuel and have been approved for prepressurized fuel in Reference S–24. Non–prepressurized fuel calculations result in conservative limits with respect to prepressurized fuel. The MAPLHGR values calculated by the codes described below are applicable to both nonbarrier and barrier fuel.

S.2.2.3.2.1.1 Short–Term Thermal–Hydraulic Model (LAMB)

The LAMB code is a model that is used to analyze the short–term thermodynamic and thermal–hydraulic behavior of the coolant in the vessel during a postulated loss–of–coolant accident. In particular, LAMB predicts the core flow, core inlet enthalpy and core pressure during the blowdown prior to the end of lower plenum flashing (~20 to 40 seconds depending on break size being evaluated). For a detailed description of the model and a discussion regarding sources of input to the model, refer to the LAMB Code Documentation portion of Reference S–19.

S.2.2.3.2.1.2 Transient Boiling Transition Model (SCAT)

The SCAT model is used to evaluate the short–term thermal–hydraulic response of the coolant in the core during a postulated loss–of–coolant accident. In particular, the convective heat transfer process in the thermally limiting fuel bundle is analyzed during the transient. For

a detailed description of the model and a discussion regarding sources of input to the model, refer to the SCAT Code Documentation portion of Reference S–19.

S.2.2.3.2.1.3 Long–Term Thermal–Hydraulic Model (SAFE/REFLOOD)

The SAFE model is used to analyze the long–term thermal–hydraulic behavior of the coolant in the vessel for all breaks. The SAFE and REFLOOD models calculate the uncovery and reflooding of the fuel and the duration of spray cooling. For a detailed description of the SAFE model and a discussion regarding sources of input to the model, refer to the SAFE Code Documentation portion of the Reference S–19.

Amendment 4, Saturated Counter–Current Flow Characteristics of a BWR Upper Tieplate, of Reference S–19 is a detailed description of the counter–current flow limiting (CCFL) of a BWR in the upper tieplate during saturated and subcooled water spray of the core. The CCFL phenomenon is modeled with a correlation based on experiments with electrically heated fuel bundles. Currently, no credit is taken for this ECCS model improvement. Not utilizing this model partially compensates for the non–conservative fission gas release correlation currently utilized with SAFE/REFLOOD (see Section S.2.2.3.2.3.4).

The REFLOOD model is used for all break sizes to calculate the system inventories after ECCS actuation when core reflooding occurs. REFLOOD accounts for the numerous bypass flow paths that exist in a BWR between the core and bypass regions. These bypass regions serve the important function of helping to refill the lower plenum and subsequently reflood the core region. For a detailed description of the REFLOOD model and a description regarding sources of input to the model, refer to the REFLOOD Code Documentation portion of Reference S–19.

S.2.2.3.2.1.4 Core Heatup Model (CHASTE)

The CHASTE model solves the transient heat transfer equations, for the highest power axial plane of the highest power assembly, for the entire LOCA transient. For a detailed description of the CHASTE model and a discussion regarding sources of input to the model, refer to the CHASTE Code Documentation section of Reference S–19.

The modified Bromley heat transfer correlation provides improved heat transfer credit for the time between departure from nucleate boiling (DNB) until the fuel rods become uncovered. This low flow film boiling period helps remove heat from the core and is described in detail in Amendment 1, Calculation of Low Flow Film Boiling Heat Transfer for BWR LOCA Analysis, of Reference S–19. As with the CCFL correlation (see Section S.2.2.3.2.1.3), no credit is taken for the Bromley model in ECCS analyses. This correlation, along with the CCFL correlation, compensates for the non–conservative fission gas release correlation currently utilized with SAFE/REFLOOD (see Section S.2.2.3.2.3.4).

The core heatup model used for the analysis is that described in Reference S–19. The model has been used to predict the results of a number of ECCS transient tests of a full–scale, stainless steel–clad heater rod bundle. These tests confirm the conservatism of the model as used for reload fuel.

The fuel rod cladding rupture temperature model, which describes the thermal–mechanical conditions that will result in fuel rod perforation, and the corresponding cladding strain model, which describes the extent of cladding deformation before and after perforation occurs, are discussed in Reference S–19. Further discussion of GE's cladding rupture and strain models, as related to NUREG–0630 requirements, is given in References S–26, S–27, S–28 and S–29. NRC approval of the rupture and strain models, as modified by these references, is given in a supplementary SER (Reference S–30).

S.2.2.3.2.2 Effect of Fuel Densification

Power spiking due to in–reactor fuel densification has not been explicitly considered in LOCA calculations submitted to the NRC. Approval of GE's analytical procedure to account for the effects of fuel densification power spiking is given in Reference S–31.

S.2.2.3.2.3 SAFE/REFLOOD LOCA Model Application Methodology

The previously described models and computer codes can be used to evaluate all plants. The LAMB Code calculates the short–term blowdown response and core flow, which are input to the SCAT code to calculate blowdown heat transfer coefficients. The SAFE code is used to determine longer–term system response and flows from the various ECC systems. Where appropriate, the output of SAFE is used in the REFLOOD code to calculate liquid levels. The results of these codes are used in the CHASTE code to calculate fuel rod cladding temperatures and maximum average planar linear heat generation rates (MAPLHGR) for each fuel type.

Most operating plants have been separated into groups for the purpose of LOCA analysis (Reference S–32). Within each plant group there will be a single lead plant analysis which provides the basis for the selection of the most limiting break size yielding the highest PCT. Also, the lead plant analysis provides an expanded documentation base to provide added insight into evaluation of the details of particular phenomena. The remainder of the plants in that group will have plant–specific analyses referenced to the lead plant analysis. The plant–specific LOCA analysis and the reference lead plant analysis for each plant is indicated in Table S–5.

Additional details of the analysis and justification for the choice of inputs for the reload analysis are given in Reference S–19. The difference in input parameters is not expected to result in significantly different results for the plants within a given group. Emergency core cooling system (ECCS) and geometric differences between plant groups may result in different responses for different groups but within any group the responses will be similar.

The LOCA analysis for each plant not specifically identified in Table S–5 is provided in the individual plant FSAR.

S.2.2.3.2.3.1 Lead Plant Selection

Lead plants are selected and analyzed in detail to permit a more comprehensive review and eliminate unnecessary calculations. This constitutes a generic analysis for each plant of that type which can be referenced in subsequent plant submittals.

Based on the criteria given in Reference S–32, the BWR/2s through BWR/4s have been divided into four groups. A lead plant was selected for each group whose LOCA response would be representative of the entire group. The four groups are identified as BWR/2, BWR/3, BWR/4 with loop selection logic (plants that have not incorporated the low pressure coolant injection (LPCI) system modification), and BWR/4 with LPCI modification.

For BWR/5 and BWR/6 plants, no lead plant was selected. Each of these plant analyses was performed on a plant–specific basis.

S.2.2.3.2.3.2 BWR/3 and BWR/4s

For BWR/3s and BWR/4s, the full complement of the LOCA codes (LAMB, SCAT, SAFE, REFLOOD, CHASTE) are used to evaluate the entire spectrum of break sizes as described in Reference S–19. These plants have been divided into three groups for the purpose of analysis: (1) BWR/3; (2) BWR/4 without LPCI modification; and (3) BWR/4 with LPCI modification. One BWR/3 is included in the second group due to similarities in bypass flow and reflooding characteristics.

Application of the LOCA analysis methods for partial and full core drilling of fuel bundles in the BWR/3s is covered in Reference S–33 and in BWR/4s in References S–34, S–35, S–36 and S–37. Approval for the LOCA analysis methods for BWR/3s is given in Reference S–38.

Application of the LOCA analysis methods in the evaluation of the effects of less than rated initial core flow is presented in Reference S–39. Approval of this evaluation is presented in Reference S–40.

S.2.2.3.2.3.3 Extension of ECCS Performance Limits

The effect of increased fission gas release from the fuel associated with higher exposures (greater than 33 GWd/MTU) on MAPLHGR has been evaluated (References S–41 and S–42). The evaluation shows that for BWR/3–6, PCT margins to the regulatory limit of 2200°F, when combined with PCT reductions due to ECCS model improvements (described in Amendments 1 and 4 of Reference S–19), will more than compensate for the PCT increase associated with increased fission gas release. Therefore, exposure–dependent fission gas release can be specifically accounted for without reducing current and proposed MAPLHGR technical specifications, provided no credit is taken for the ECCS model changes. NRC approval of this is given in Reference S–43. The impact of fission gas release will be analyzed on a case–by–case basis if the improved ECCS models are used in the ECCS performance analysis or if PCT margins are less than those specified in Reference S–42.

S.2.2.3.2.4 SAFER/GESTR LOCA Model Code Descriptions

Results of extensive LOCA experimental programs since 1974 have clearly demonstrated the large conservatisms that the SAFE/REFLOOD LOCA models (Section S.2.2.3.2.3) have with respect to modeling the vessel inventory, inventory distribution and core heat transfer. A new thermal–hydraulic model (SAFER) and a new fuel rod thermal–mechanical model (GESTR–LOCA) have been developed to provide more realistic calculations for LOCA analyses. The SAFER and GESTR–LOCA models are summarized below and discussed in detail in References S–20, S–21, S–74, S–92 (as reviewed by the NRC in the letter specified in Reference S–92) and S–93.

As with the SAFE/REFLOOD LOCA models (Section S.2.2.3.2.1), SAFER/GESTR–LOCA is also applicable to prepressurized fuel. Non–prepressurized fuel calculations result in conservative limits with respect to prepressurized fuel. The MAPLHGR values calculated by the codes are applicable to both nonbarrier and barrier fuel.

S.2.2.3.2.4.1 Realistic Thermal–Hydraulics Model (SAFER)

SAFER replaces the combination of the SAFE and REFLOOD ECCS performance evaluation models discussed in Section S.2.2.3.2.1.3.

The SAFER code employs a heatup model with a simplified radiation heat transfer correlation to calculate PCT and local maximum oxidation, which replaces the CHASTE heatup calculation (Section 2.2.3.2.1.4). The PCT and local maximum oxidation fraction from SAFER can be used directly.

S.2.2.3.2.4.2 Best Estimate Fuel Rod Thermal Mechanical Model (GESTR–LOCA)

The GESTR–LOCA model has been developed to provide best–estimate predictions of the thermal performance of GE nuclear fuel rods experiencing variable power histories. For ECCS analyses, the GESTR–LOCA model is used to initialize the fuel stored energy and fuel rod fission gas inventory at the onset of a postulated LOCA. Details of the GESTR–LOCA models are provided in Reference S–20.

S.2.2.3.2.4.3 Transient Boiling Transition Model (TASC)

TASC replaces the SCAT boiling transition model discussed in Section S.2.2.3.2.1.2.

The TASC model is used to evaluate the short–term thermal–hydraulic response of the coolant in the core during a postulated loss–of–coolant accident. In particular, the convective heat transfer response in the thermally limiting fuel bundle is analyzed during the transient. For a detailed description of the model and a discussion regarding sources of input to the model, refer to Reference S–94.

S.2.2.3.2.5 SAFER/GESTR–LOCA Model Application Methodology

Using the SAFER/GESTR–LOCA models, the LOCA events are analyzed with nominal values of inputs and correlations. A calculation is performed in conformance to Appendix K

and checked for consistency with generic statistical upper bound analyses that encompass modeling uncertainties in SAFER/GESTR–LOCA and uncertainties related to plant parameters.

As with the SAFE/REFLOOD LOCA models application methodology (Section S.2.2.3.2.2), the effects of power spiking due to in–reactor densification are considered negligible for SAFER/GESTR–LOCA analyses for similar reasons.

The details of the application methodology are summarized below and discussed in detail in References S–22, S–92 and S–93. The plant–specific LOCA analysis report for each plant is identified in Table S–5.

S.2.2.3.2.5.1 Appendix K Conformance

The SAFER/GESTR–LOCA Appendix K conformance calculation will be performed only for the limiting break of a nominally calculated break spectrum with a range of break flow multipliers between 0.6 and 1.0. The licensing PCT is obtained as described in Reference S–22.

S.2.2.3.2.5.2 BWR/2

BWR/2s have all been analyzed using SAFER/CORECOOL/GESTR–LOCA on a plant–specific basis. The analysis methodology is described in Reference S–74.

S.2.2.3.2.6 Total LOCA Analysis

The total LOCA analysis, based on the use of the SAFE/REFLOOD/CHASTE codes (Sections S.2.2.3.2.1 and S.2.2.3.2.3), is performed using the procedures outlined in Reference S–19. The total LOCA analysis based on the use of the SAFER/GESTR–LOCA codes (Sections S.2.2.3.2.4 and S.2.2.3.2.5), is performed using the procedures outlined in Reference S–22. The total LOCA analysis is generally provided for each plant independent of the supplemental reload licensing report. The supplemental reload licensing report will contain either the MAPLHGR and PCT as a function of exposure for fuel not previously licensed to operate in the specific reactor, or a reference to the analysis results. For multiple lattice fuel designs, each lattice has an associated MAPLHGR value. The MAPLHGR limit is determined by the LOCA analyses described in the preceding subsections. For each multiple lattice fuel bundle type, the supplemental reload licensing report will include a plot or table of the limiting value of MAPLHGR for the most limiting enriched lattice as a function of average planar exposure. Additional information is provided in Reference S–44.

S.2.2.3.3 Main Steam Line Break Accident Analysis

The analysis of the main steam line break accident depends on the operating thermal–hydraulic parameters of the overall reactor (such as pressure) and overall factors affecting the consequences (such as primary coolant activity). Results for initial cores are documented in the individual plant FSAR. Insertion of the reload fuel designs described in Reference S–2

and S–3 will not change any of these parameters; therefore, the previous reviewed results of this analysis will not change.

S.2.2.3.4 One Recirculation Pump Seizure Accident Analysis

This accident is assumed to occur as a consequence of an unspecified, instantaneous stoppage of one recirculation pump shaft while the reactor is operating at full power.

The pump seizure event is a very mild accident in relation to other accidents such as the LOCA. This is easily verified by consideration of the two events. In both accidents, the recirculation driving loop flow is lost extremely rapidly. In the case of the seizure, stoppage of the pump occurs; for the LOCA, the severance of the line has a similar, but more rapid and severe influence. Following a pump seizure event, flow continues, water level is maintained, the core remains submerged and this provides a continuous core cooling mechanism. However, for the LOCA, complete flow stoppage occurs and the water level decreases due to loss of coolant, resulting in uncovery of the reactor core and subsequent overheating of the fuel rod cladding. In addition, for the pump seizure accident, reactor pressure does not significantly decrease, whereas complete depressurization occurs for the LOCA. Clearly, the increased temperature of the cladding and reduced reactor pressure for the LOCA both combine to yield a much more severe stress and potential for cladding perforation for the LOCA than for the pump seizure. Therefore, it can be concluded that the potential effects of the hypothetical pump seizure accident are very conservatively bounded by the effects of a LOCA and specific analyses of the pump seizure accident are not required.

S.2.2.3.5 Refueling Accident Analysis

Identification of Causes. Accidents that result in the release of radioactive materials directly to the containment can occur when the drywell is open. A survey of the various conditions that could exist when the drywell is open reveals that the greatest potential for the release of radioactive material occurs when the drywell head and reactor vessel head have been removed. In this case, radioactive material released as a result of fuel failure is available for transport directly to the containment.

Various mechanisms for fuel failure under this condition have been investigated. With the current fuel design the refueling interlocks, which impose restrictions on the movement of refueling equipment and control rods, prevent an inadvertent criticality during refueling operations. In addition, the reactor protection system can initiate a reactor scram in time to prevent fuel damage for errors or malfunctions occurring during planned criticality tests with the reactor vessel head off. It is concluded that the only accident that could result in the release of significant quantities of fission products to the containment during this mode of operation is one resulting from the accidental dropping of a fuel bundle onto the top of the core.

This event occurs under non–operating conditions for the fuel. The key assumption of this postulated occurrence is the inadvertent mechanical damage to the fuel rod cladding as a

consequence of the fuel bundle being dropped on the core while in the cold condition. Therefore, fuel densification considerations do not enter into or affect the accident results.

Methods, Assumptions and Conditions. The assumptions and analyses applicable to this type of fuel handling accident are described below.

(1) GE is now manufacturing a new design of the refueling mast with grapple head (NF–500). The new design weighs more—619 pounds compared to 350 pounds. For plants not having employed the new NF–500 refueling mast, the following analysis is bounding.

(2) The number of fuel rods in a fuel bundle has gone from the initial 7x7 array, to the 8x8 array, and more recently to the 9x9 array and the 10x10 array with corresponding dimensional changes.

(3) During a refueling operation a fuel assembly is moved over the top of the core. While the fuel grapple is in the overhoist condition with the bottom of the assembly 34 feet above the top of the core (the maximum height allowed by the fuel handling equipment), a main hoist cable fails allowing the assembly, the fuel grapple mast and head to fall on top of the core impacting a group of four assemblies. The grapple head and mast are fixed vertically to the dropped assembly such that all the kinetic energy is transferred through the dropped assembly to the group of impacted assemblies. The dropped assembly impacts the core at a slight angle and the rods in this assembly are subjected to bending. After the assembly impacts the core, the assembly, grapple head and mast fall onto the core horizontally without contacting the side of the pressure vessel.

(4) The entire amount of potential energy, including the energy of the entire assemblage falling to its side from a vertical position (referenced to the top of the reactor core), is available for application to the fuel assemblies involved in the accident. This assumption neglects the dissipation of some of the mechanical energy of the falling fuel assembly in the water above the core and requires that the grapple cable break, allowing the grapple head and three sections of the telescoping mast to remain attached to the falling assembly.

(5) None of the energy associated with the dropped fuel assembly is absorbed by the fuel material (uranium dioxide).

(6) All fuel rods, including tie rods, were assumed to fail by 1% strain in compression, the same mode as ordinary fuel rods. For the fuel designs considered here, there is no propensity for preferential failure of tie rods.

The following analysis is provided for the GE11 (Reference S–76) and GE13 fuel bundles (the 9x9 array). The radiological consequences are provided for all fuel designs.

Analysis and Results. Because of the complex nature of the impact and the resulting damage to fuel assembly components, a rigorous prediction of the number of failed rods is not

possible. For this reason a simplified energy approach is taken and numerous conservative assumptions are made to assure a conservative estimate of the number of failed rods.

The number of failed fuel rods is determined by balancing the energy of the dropped assemblage against the energy required to fail a rod. The wet weight of the dropped bundle is 562 pounds for the 9x9 fuel rod array bundle (617 pounds for the 7x7 fuel rod array bundle) and wet weight of the grapple mast and head is 619 pounds. The drop distance is 34 feet. The total energy to be dissipated by the first impact is

$$E_1 = (562+619)\,(34) = 40{,}154\ ft\text{–}lb$$

One half of the energy is considered to be absorbed by the falling assembly and one half by the four impacted assemblies.

No energy is considered to be absorbed by the fuel pellets (i.e., the energy is absorbed entirely by the non–fuel components of the assemblies).

The energy available for clad deformation is considered to be proportional to the mass ratio:

$$\frac{mass\ of\ cladding}{(mass\ of\ assembly) - (mass\ of\ fuel\ pellets)}$$

and is equal to a maximum of 0.510 for the fuel designs considered here.

The energy absorbed by the cladding of the four impacted assemblies is:

$$(20{,}077\ ft\text{–}lb)\,(0.510) = 10{,}239\ ft\text{–}lb.$$

Each rod that fails is expected to absorb approximately 200 ft–lb before cladding failure, based on uniform 1% plastic deformation of the cladding.

The number of rods failed in the impacted assemblies is:

$$N_f = \frac{(10{,}239\ ft - lb)}{(200\ ft - lb)} = 51\ rods.$$

The dropped assembly is assumed to impact at a small angle from vertical, subjecting all the fuel rods in the dropped assembly to bending moments. The fuel rods are expected to absorb little energy prior to failure as a result of bending. For this reason it is assumed that all the rods in the dropped assembly fail. The total number of failed rods on initial impact is 74 + 51 = 125.

The assembly is assumed to tip over and impact horizontally on the top of the core from a height of one bundle length, approximately 160 inches. The remaining available energy is calculated by assuming a linear weight distribution in the assembly with a point load at the top of the assembly to represent the fuel grapple weight.

$$E_2 = W_G H_G + \int_0^{H_B} \frac{W_b}{H_B} y dy = W_G H_G + \frac{1}{2} W_B H_B$$

$$E_2 = (619\ lb)\left(\frac{160}{12}\right) + \frac{1}{2}(562)\left(\frac{160}{12}\right) = 12{,}000\ ft - lb.$$

As before, the energy is considered to be absorbed equally by the falling assembly and the impacted assemblies. The fraction available for clad deformation is 0.510. The energy available to deform the unfailed cladding in the impacted assemblies is one–half the energy resulting from the second impact:

$$E_c = (0.5)\ (12{,}000\ ft–lb)\ (0.510) = 3{,}060\ ft–lb$$

and the number of failures in the impacted assemblies is:

$$N_F = \frac{3{,}060\ ft - lb}{200\ ft - lb} = 15\ rods.$$

Since the rods in the dropped 9x9 assembly are considered to have failed in the initial impact, the total failed rods resulting from both impacts is 125 + 15 = 140.

The above analysis was completed using the GE12 and GE14 10x10 fuel rod arrays (References S–77 and S–95). The analysis resulted in 172 failed rods from both impacts.

This compares with 111 failed rods from the analysis for the 7x7 fuel rod array bundle presented in the individual plant FSAR.

Radiological Consequences Comparisons. For the purposes of this evaluation, it is conservatively assumed that the fractional plenum activity for any 9x9 rod will be 49/74, or 0.66 times the activity in a 7x7 rod. Based on the assumption that 140 9x9 rods fail compared to 111 for a 7x7 core, the relative amount of activity released for the 9x9 fuel is (140/111) (0.66) = 0.83 times the activity released for a 7x7 core. The activity released to the environment and the radiological exposures for all GE 9x9 fuel designs will therefore be less than 83% of those values presented in the FSAR for a 7x7 core. As identified in the FSAR, the radiological exposures for the 7x7 fuel are well below those guidelines set forth in 10CFR100; therefore, it can be concluded that the consequences of this accident with the new NF–500 mast and the 9x9 fuel will also be well below these guidelines.

A fuel bundle damage analysis and the resulting radiological consequences for the new NF–500 mast and the 8x8 fuel shows that the activity released to the environment and the radiological exposures will be less than 84% of those values presented in the FSAR for a 7x7 core. Similar to the above evaluation, the activity released to the environment and the radiological exposures for all GE 10x10 fuel designs will therefore be less than (172/111)(49/87.33) = 0.87 or 87% of those values presented in the FSAR for a 7x7 core.

S.2.3 Analysis Initial Conditions and Inputs

Inputs to the models utilized to analyze the AOO events discussed in Section S.2.2 are plant unique. The specific inputs related to the plant pressure relief systems (i.e., safety valves, safety/relief valves, etc.) are listed in the supplemental reload licensing report for each plant. Inputs such as thermal power, dome pressure, etc. are given in the individual plant supplemental reload licensing report. The initial conditions for the GETAB analysis are listed in the supplemental reload licensing report for each specific plant. Because the AOO model establishes operating conditions, only licensing basis values are given in the supplemental reload licensing report.

Cycle–dependent initial conditions for the GETAB analysis and the resulting reload parameters are given in the plant FSAR or the supplemental reload licensing report.

S.3 Vessel Pressure ASME Code Compliance Model

The pressure relief system was designed to prevent excessive overpressurization of the primary system process barrier and the pressure vessel and thereby precludes an uncontrolled release of fission products.

Prior to 1967, the design capacities of the safety valves for BWRs were determined according to the requirements of Section I, *Power Boilers*, of the ASME Boiler and Pressure Vessel Code. Under the provisions of this code, safety valve capacities were established to prevent either a vessel or pressure rise greater than 6% above the maximum allowable working pressure. At least one safety valve was to be set at or below the maximum allowable working pressure; the highest safety valve setting could not exceed 103% of the maximum allowable working pressure. No credit was allowed for reactor scram as a complementary pressure protection device. Thus, the required safety valve capacities were sized assuming essentially instantaneous isolation of the pressure vessel with no pressure relief other than that from the safety valves. Nine Mile Point–1 and Oyster Creek are the only plants that were designed to these criteria.

In 1991 Oyster Creek updated its overpressurization analysis (Reference S–88) to ASME Boiler and Pressure Code, Section III to be consistent with later BWRs and reducing the number of safety valves.

In 1995 Nine Mile Point Unit 1 updated its overpressurization analysis (reference S–89) to ASME Boiler and Pressure Code, Section III to be consistent with later BWRs and reduced the number of safety valves.

The vessel overpressure protection system for the other plants was designed to satisfy the requirements of Section III, *Nuclear Vessels*, of the ASME Boiler and Pressure Vessel Code. The ASME Boiler and Pressure Vessel Code, Section III, Class I, permits pressure transients up to 10% over design pressure, and requires that the lowest qualified valve setpoint be at or below the vessel design pressure and the highest setpoint is not greater than 105% of the vessel design pressure. Section III of the code allows credit to be taken for the scram protection system as a pressure protection device when determining the required safety valve

capacities for nuclear vessels. As required by the Code of Federal Regulations 10CFR50.55a, paragraph C1, applicable Section III code cases and addenda to which the above plants were designed vary from the 1963 edition, including addenda through summer 1964, to the 1965 edition including addenda through summer 1967. These editions and addenda to Section III of the code required the reactor pressure vessel to be designed to accommodate the normal operating loads and transient startup/shutdown and test cyclic loads expected during the 40–year life of the plant.

In 1968, GE went beyond the code requirements by establishing new design criteria in response to a NRC question. With these criteria, two categories of events (normal and accident) were analyzed for plants that had not received an operating license. The normal category of events included the design and operating loads as well as upset conditions previously analyzed. These loadings were required to meet the criteria documented in Section III of the code. The accident category included low probability of occurrence accidents or faulted conditions that were required to meet a set of limits developed by GE.

The Summer 1968 Addenda to the 1968 Edition of Section III to the ASME code revised the conditions to be considered when performing pressure vessel stress analyses. Loads were to be considered from four categories of conditions: (1) normal; (2) upset; (3) emergency; and (4) faulted.

The Addenda defines an upset condition as any deviation from normal operating conditions caused by any single error, malfunction or a transient which does not result in a forced outage. These events are anticipated to occur frequently enough that design should include the capability to withstand the upsets without operational impairment. Emergency conditions are stated as having ". . . a low probability of occurrence . . ." and require shutdown for correction but cause no gross damage to the system. Additionally, faulted conditions are " . . . those combinations of conditions associated with extremely low probability postulated events . . ." which may impair the integrity and operability of the nuclear system to the point where public safety is involved.

As documented in later FSARs and accepted by the NRC, GE has defined an upset event as one which has a 40–year encounter probability of occurrence of 10^{-1} through 1; an emergency event has a 40–year encounter probability of 10^{-3} though $<10^{-1}$; and a faulted event has a 40–year encounter probability of 10^{-6} through $<10^{-3}$. GE analyses have determined the probability of occurrence of MSIV closure is 1 event/plant–year (Reference S–48). Failure probability of the direct MSIV position switch scram failure such that scram occurs on neutron monitoring system signal is 1×10^{-3}/demand. Using the above probabilities, this event should be considered an "emergency" condition. Therefore, application of the "emergency" limit under these assumed failure conditions would be considered appropriate. However, in addition to conservatively assuming failure of the direct safety grade position scram signals in its licensing analysis, and conservatively relying upon indirectly derived signals (high neutron flux) from the Reactor Protection System, GE further conservatively applies the upset code requirements, and required pressure safety limits, rather than the more appropriate emergency limits. Application of the direct position scrams in the design basis could be used since they qualify as acceptable pressure protection devices when determining

the required safety/relief valve capacity of nuclear vessels under the provisions of the ASME safety code.

As described in the Summer 1968 Addenda of Section III, the following pressure limits are applied to the operating limit category:

(1) Under upset conditions, the code requires that reactor pressures are not to exceed 110% of design pressure (1.1 × 1250 = 1375 psig).

(2) For emergency conditions, it allows up to 120% of design pressure (1.2 × 1250 = 1500 psig).

(3) For faulted conditions, it allows up to 150% of design pressure (1.5 × 1250 = 1875 psig).

GE sensitivity studies (Reference S–49) show the effect of safety/relief valve failures on peak pressure for the MSIV closure event expectedly results in a peak pressure increase of less than 20 psi and depends on the plant total pressure relief capacity.

If an MSIV closure analysis which considers the failure of a safety/relief valve is performed, the following events are considered: (1) MSIV closure followed by indirect flux scram (estimated probability = 1×10^{-3}/demand), and (2) failure of one safety/relief valve. In addition, many conservatisms discussed previously would also be employed. According to the interpretation of the code, MSIV closure with indirect flux scram would be considered an emergency event. Therefore, the occurrence of failures in addition to the extremely low probability of this event constitutes emergency, if not faulted, conditions. Analysis of MSIV closure, flux scram and SRV failure under emergency conditions (1500 psi pressure limit) would be far less restrictive than the present analysis of MSIV closure followed by flux scram under upset conditions (1375 psi pressure limit), especially when considering the minimal effect of a failed SRV.

Overpressurization protection analysis is performed using the ODYN transient code (References S–50 and S–51). In accordance with Reference S–48, no addition of uncertainty to the calculations of pressure is needed. Results for this analysis are given in the FSAR or in the supplemental reload licensing report.

S.4 Stability Analysis Methods

Several types of stability analyses are performed to ensure continued acceptable plant–specific implementation of NRC approved long–term stability solutions:

- Core and channel decay ratio calculations are performed to ensure that the fuel is as stable as previously licensed GE fuel designs. If the fuel is not as stable as previously existing fuel designs, then the stability exclusion region must be revised to provide the same level of protection. This is a generic calculation that is applicable to all long-term stability solutions.

- CPR response calculations are performed to demonstrate the SLMCPR protection against a thermal hydraulic instability event using the detect and suppress methodology outlined in Reference S-85. The plant and cycle-specific core-wide mode DIVOM (Delta CPR over Initial CPR Vs. Oscillation Magnitude) data is required for Option I-D plant stability analysis and must be calculated in accordance with the BWROG plant-specific core-wide mode DIVOM procedure guideline specified in Reference S-102. The plant and cycle-specific regional mode DIVOM data is required for Option II and Option III plant stability analyses and must be calculated in accordance with the BWROG plant-specific regional mode DIVOM procedure guideline specified in Reference S-103. This is applicable to Option I-D, Option II and Option III.

- CPR margins calculations are performed to demonstrate the SLMCPR protection against a thermal hydraulic instability event using the integrated TRACG methodology outlined in Reference S-104. The demonstration of new fuel types beyond the currently approved fuel designs (up to GE14) and the BWR product line (up to BWR/6) must be demonstrated in accordance with the integrated TRACG methodology outlined in Section 6 of Reference S-104. This is applicable to DSS-CD.

The core and channel decay ratios are calculated with a NRC approved frequency domain model. The ODYSY code (References S-106 and S-96) is used in the frequency domain evaluations, even though the older FABLE code may be used in some legacy calculations (e.g., the E1A Standard Cycle). This calculation provides assurance that plants with prevention based long–term stability solutions will not have to unreasonably increase the size of their stability–based regions for the evaluated fuel design.

The continued applicability of the interim/backup stability solution is based on exclusion regions and reload validation of these exclusion regions is required to ensure full stability protection.

The applicability of the plant and cycle-specific DIVOM curve is demonstrated with a best–estimate coupled neutronic – thermal hydraulic model using TRACG. This is the same model that was used to generate the plant and cycle-specific DIVOM data. The DIVOM data is required for plants with a detect and suppress solution to demonstrate safety limit MCPR compliance.

The plant-specific CPR margins demonstration based on integrated TRACG model is required of all plants implementing the DSS-CD solution for the first time to ensure the safety limit MCPR compliance.

S.4.1 Long–Term Stability Solutions

S.4.1.1 Enhanced Option I–A

The BWROG Enhanced Option I–A (EIA) is a prevention solution. EIA was reviewed and approved by the USNRC as documented in References S–80 through S–84 and Reference S–

96 for operation up to and including the Maximum Extended Load Line Limit Analysis (MELLLA) domain. For plants implementing EIA, the prescribed reload validation (Reference S–80) is performed each cycle and the results documented in the supplemental reload licensing report. The validation confirms that the existing E1A stability regions provide adequate stability margin. If E1A reload validation criteria are not met, new E1A stability regions must be defined and implemented.

S.4.1.2 Option II

The BWROG Option II is a combination prevention and detect and suppress solution. Option II was reviewed and approved by the USNRC as documented in Reference S–79 for operation up to and including the MELLLA domain. Option II is only applicable to BWR 2 plants. A reload review criterion has been defined for Option II to ensure that the existing exclusion region is acceptable for each fuel cycle. If reload criteria are not met, the exclusion region must be recalculated. In addition, continued safety limit MCPR protection is demonstrated for each fuel cycle using the methodology documented in References S–85 and S-103 and in plant–specific Option II licensing topical reports. The results of the reload review and safety limit MCPR protection calculation are documented in the supplemental reload licensing report.

S.4.1.3 Option I–D

The BWROG Option I–D is a combination prevention and detect and suppress solution. Option I–D was reviewed and approved by the USNRC as documented in Reference S–79 for operation up to and including the MELLLA domain. Option I–D is only applicable to plants which can demonstrate that the core wide is the dominant oscillation mode for anticipated reactor instabilities. A reload review criterion has been defined for Option I–D to ensure that the existing exclusion region is acceptable and that the safety limit MCPR is protected for each fuel cycle. If reload criteria are not met, the exclusion region must be recalculated. In addition, continued safety limit MCPR protection is demonstrated for each fuel cycle using the methodology documented in References S–85 and S-102 and in plant–specific Option I-D licensing topical reports. The dominance of the core–wide mode of reactor oscillation is demonstrated at the most limiting power/flow point using the NRC–approved frequency stability code (e.g., Reference S–96 or S-106). The results of the reload review and safety limit MCPR protection calculation are documented in the supplemental reload licensing report.

S.4.1.4 Option III

The BWROG Option III is a detect and suppress solution. Option III was reviewed and approved by the USNRC as documented in Reference S–79 for operation up to and including the MELLLA domain. Continued safety limit MCPR protection is demonstrated for each fuel cycle using the methodology documented in References S–85 and S-103.

The results of the safety limit MCPR protection calculation are documented in the supplemental reload licensing report.

S.4.1.5 DSS-CD

The GEH Detect and Suppress Solution – Confirmation Density (DSS-CD) is a detect and suppress solution. DSS-CD was reviewed and approved by the USNRC as documented in Reference S–104 for operation up to and including the Maximum Extended Load Line Limit Analysis Plus (MELLLA+) domain. Continued safety limit MCPR protection is demonstrated for each fuel cycle using the methodology documented in Reference S-104.

The results of the safety limit MCPR protection calculation are documented in the supplemental reload licensing report.

S.4.2 Interim/Backup Stability Solution

S.4.2.1 Interim Corrective Action (ICA)

The ICA is an interim prevention solution based on exclusion regions for EIA, Option I-D, Option II and Option III. The currently used ICA regions were established in Reference S–91 based on original licensed thermal power, generally shorter fuel cycles, and more stable core designs. These regions are defined based on relative core flow and rod line points and not on specific stability criteria. New aggressive core design changes may have reduced stability margins. GE recommends that the impact of core design changes be included in plant/cycle–specific evaluations to assess the continued applicability of the ICA regions. The results of the ICA analysis are documented in the supplemental reload licensing report.

S.4.2.2 Backup Stability Protection (BSP) for Option III

The BSP for Option III is an alternative interim prevention solution based on exclusion regions. The currently used BSP regions were established in Reference S–90 based on revised ICA regions. These regions are defined based on relative core flow and rod line points and not on specific stability criteria. New aggressive core design changes may have reduced stability margins. GEH recommends that the impact of core design changes be included in plant/cycle–specific evaluations of the BSP regions.

The BSP for Option III is generated in accordance with Reference S-90. The BSP for Option III methodology is applied in the fuel cycle reload stability analysis. To calculate the BSP Scram and Controlled Entry Region boundaries, ODYSY decay ratio calculations are performed on the High Flow Control Line (HFCL) and on the Natural Circulation Line (NCL). Rated feedwater temperature and rated xenon concentrations are assumed for calculating the BSP Scram Region boundary endpoints. The two endpoints (A1 and B1 in Figure S-3), where the ODYSY acceptance criteria are met, are connected using either the Generic or Modified Shape Function to define the Scram Region boundary. The BSP Controlled Entry Region is calculated in a similar manner, also using the ODYSY acceptance criteria to define the two endpoints (A2 and B2), with either the Generic or Modified Shape Function as the connecting curve. Off-rated equilibrium feedwater temperature is assumed for calculating the Controlled Entry Region endpoints (A2 and B2). Equilibrium xenon condition is assumed for the HFCL endpoint (A2) while xenon-free condition is assumed for the NCL endpoint (B2).

If the MSF is used, each calculated BSP region boundary must be validated at a mid-point against the DR acceptance criterion, respectively. This mid-point MSF validation should be performed based on the calculated BSP boundaries. A mid-point can be defined as a state point that its flow is the average flow of the two corresponding bounding state points, i.e., A1 and B1 or A2 and B2 or another power/state point close to the mid-point on the MSF. The MSF validation point should be based corresponding to the HFCL conditions and its associated limiting exposure.

The typical calculated stability region boundaries are illustrated in Figure S-3 where the four calculated endpoints A1, A2, B1, B2 are based on a core decay ratio criterion of 0.80. Please note that the actual implemented BSP boundaries might be larger due to the Base BSP region definitions. The Base BSP regions for Option III are defined in Reference S-90.

According to Reference S-90, deliberate entry into the BSP Controlled Entry Region requires compliance with at least one of the stability controls outlined below:

1. Maintain core average Boiling Boundary (BB) ≥ 4.0 feet,

2. Maintain core average BB > Reference value (demonstrated to produce stable operation) and ≥ 3.0 feet,

   Maintain radial peaking factor ≤ Reference value (demonstrated to produce stable operation),

3. Maintain core decay ratio (DR) < 0.6 as calculated by an on-line core stability monitor,

4. The individual owner will determine appropriate limits for core DR (< 0.60) as calculated by a core stability monitor, or by pre-analysis of a reactor state trajectory through the Controlled Entry Region, or

5. Continuous dedicated monitoring of real time control room neutron monitoring instrumentation with manual scram required upon indication of a reactor instability induced power oscillation.

Usually, two sets of BSP regions may be generated for different rated feedwater temperature ranges. The results of the BSP for Option III analysis are documented in the supplemental reload licensing report.

S.4.2.3 Backup Stability Protection (BSP) for DSS-CD

The BSP for DSS-CD is a backup solution based on exclusion regions in case the DSS-CD solution is not operational.

The BSP for DSS-CD is generated in accordance with Reference S-103. The BSP for DSS-CD methodology is applied in the fuel cycle reload stability analysis. To calculate the BSP Scram and Controlled Entry Region boundaries, ODYSY decay ratio calculations are performed on the HFCL and on the NCL. Rated feedwater temperature and rated xenon

concentrations are assumed for calculating the BSP Scram Region boundary endpoints. The two endpoints (A1 and B1 in Figure S-4), where the ODYSY acceptance criteria are met, are connected using the Generic Shape Function to define the Scram Region boundary. The BSP Controlled Entry Region is calculated in a similar manner, also using the ODYSY acceptance criteria to define the two endpoints (A2 and B2), with the Generic Shape Function as the connecting curve. Off-rated equilibrium feedwater temperature is assumed for calculating the Controlled Entry Region endpoints (A2 and B2). Equilibrium xenon condition is assumed for the HFCL endpoint (A2) while xenon-free condition is assumed for the NCL endpoint (B2).

Please note that there are several differences between the BSP for DSS-CD and BSP for Option III:

a) The ODYSY core decay ratio (DR) acceptance criterion used for the Controlled Entry Region boundary intercept (A2) along the HFCL is different between the methodologies. The BSP for Option III uses a core DR acceptance criterion of 0.80 while the BSP for DSS-CD uses a core DR acceptance criterion of 0.60.

b) For BSP for Option III, the HFCL is defined as the highest licensed load line, up to the MELLLA boundary. For BSP for DSS-CD, the HFCL is defined as the MELLLA+ boundary.

c) The BSP for DSS-CD solution imposes the BSP Boundary restriction on the DSS-CD solution for short-term manual operation if the OPRM system is inoperable.

d) The Generic Shape Function is used in the BSP for DSS-CD.

The typical calculated stability region boundaries and the BSP Boundary are illustrated in Figure S-4; the three endpoints A1, B1 and B2 are calculated based on a core DR criterion of 0.80 while the A2 endpoint is calculated based on a core DR criterion of 0.60. The 0.60 criterion for the A2 endpoint (as illustrated in Figure S-4) provides additional stability margins for operation at off-rated conditions and for a two-pump trip to natural circulation flow.

Please note that the actual implemented BSP boundaries might be larger due to the Base BSP region definitions. The Base BSP regions for DSS-CD are defined in Reference S-103.

Deliberate entry into the Manual BSP Controlled Entry Region requires compliance with at least one of the stability controls outlined in Section 7.2.3.2 of Reference S-103.

Usually, two sets of BSP regions may be generated for different rated feedwater temperature ranges. Only the Automated BSP option is approved for use as an extended backup solution to DSS-CD. The results of the BSP for DSS-CD analysis are documented in the supplemental reload licensing report.

S.5 Analysis Options

Three groups of analysis options are presented in the following sections. The first group involves options that may be chosen to improve MCPR margin. The second group of improvements represents a collection of possible operating flexibility options. Also noted in the second group is the GE Licensing Topical Report, *Applicability of GE Methods to Expanded Operating Domains* (Reference S-101), which may be part of the licensing basis for EPU and MELLLA+ plants. The third group includes the requirements for applying the generic analysis in Reference S-99 for the Fuel Loading Error event. In some cases separate plant specific reports are submitted for approval before the option is available. Other options are supported by generic analyses that have been approved and only require that the plant choose to activate the option. In each case, the plant options are selected for each cycle and documented in the cycle design documentation and the plant supplemental reload licensing report (SRLR).

S.5.1 Available MCPR Margin Improvement Options

The following margin improvement options have been developed for operating BWRs:

(1) Recirculation Pump Trip

(2) Rod Withdrawal Limiter

(3) Thermal Power Monitor

(4) Exposure–Dependent Limits

(5) Improved Scram Times
   (a) Measured Scram Time
   (b) Generic Statistical Scram Time (ODYN Option B or TRACG Option B)

These margin improvement options will be made available, on a commercial basis, to all owners of operating BWRs.

As these options are selected by the BWR owners, plant–specific and/or generic bounding analyses will be submitted for approval. The plant supplemental reload licensing report will designate the options selected by that BWR owner.

S.5.1.1 Recirculation Pump Trip

For many of the plant operating cycles, the limiting AOOs are the turbine trip, generator load rejection, or other AOOs that result in a turbine trip. A significant improvement in thermal margin can be realized if the severity of these transients is reduced. The Recirculation Pump Trip (RPT) feature accomplishes this by cutting off power to the recirculation pump motors anytime that the turbine control valve or turbine stop valve fast closure occurs. This rapid reduction in recirculation flow increases the core void content during the AOO, thereby reducing the peak AOO power and heat flux.

Basically, the RPT consists of switches installed in both the turbine control valves and the turbine stop valves. When these valves close, breakers are tripped between the MG sets and the recirculation pump motors; this releases the recirculation pumps to coast down under their own inertia.

Recirculation pump trip is standard equipment in all later plants.

S.5.1.2 Rod Withdrawal Limiter System

The Rod Withdrawal Error (RWE) has become the limiting transient for some plants. A new Rod Withdrawal Limiter System (RWLS) concept has been developed. This new system will restrict control rod movement such that the Rod Withdrawal Error will be eliminated as a limiting AOO.

The RWLS functions by providing a rod withdrawal block as a function of rod distance traveled per rod selection. Core physics calculations performed for the RWE analysis, provide the decrease in CPR as a function of rod travel. After choosing an acceptable ΔCPR, an allowable rod movement is determined. This sets the RWLS trip point. Any attempt to withdraw the rod by more than the trip point results in a rod block. Thus, an upper bound is established on the CPR decrease that can result from any single rod withdrawal error.

This system is standard on all BWR/6 plants.

S.5.1.3 Thermal Power Monitor

The APRM simulated thermal power trip (APRM thermal power monitor) is a minor modification to the APRM system. The modified APRM system generates two upscale trips. On one, the APRM signal (which is proportional to the thermal neutron flux) is compared with a reference that is not dependent on flow rate.

During normal reactor operations, neutron flux spikes may occur due to conditions such as transients in the recirculation system, transients during large flow control load maneuvers, transients during turbine stop valve tests and transients in plants with equalizer lines when the recirculation equalizer lines are opened. The neutron flux leads the heat flux during transients because of the fuel time constant. And the neutron flux for these transients trips upscale before the heat flux increases significantly. (High heat flux is the precursor of fuel damage.) Thus, increased availability can be achieved without fuel jeopardy by adding a trip dependent on heat flux (thermal power).

For this trip, the APRM signal is passed through a low pass RC filter. It is compared with a recirculation flow rate dependent reference that decreases approximately parallel to the flow control lines.

In addition to increased availability, another benefit is that with the minor operational spikes filtered out, the heat flux trip setpoint is lower than the neutron flux trip setpoint. For long, slow AOOs such as the loss–of–feedwater heater, the heat flux and neutron flux are almost in

equilibrium. For these AOOs, the lower trip setpoint results in an earlier scram with a smaller increase in heat flux and a corresponding reduction in the consequences.

The APRM Simulated Thermal Power Trip is standard equipment in some BWR/4 plants and all BWR/5 and BWR/6 plants.

S.5.1.4 Exposure–Dependent Limits

The severity of any plant AOO pressurization event is worst at the End-Of-Rated (EOR) condition (rated core power, rated core flow, all-rods-out) because the EOR scram curve gives the worst possible scram response. It follows that some limits relief may be obtained by analyzing the AOOs at other interim points in the cycle and administering the resulting limits on an "exposure dependent" basis.

This technique is straightforward and consists merely of repeating certain elements of the AOO analyses for selected mid–cycle exposures. Because the scram reactivity function monotonically deteriorates with exposure (after the reactivity peak), the limit determined for an exposure E_1 is administered for all exposures in the interval $E_{i-1} < E \leq E_i$ where E_{i-1} is the next lower exposure point for which a limit was determined. This results in a table of MCPR limits to be applied through different exposure intervals of the cycle.

S.5.1.5 Improved Scram Times

S.5.1.5.1 Measured Scram Time

Control rod scram time data from two operating BWR/4 plants have been used to derive a more realistic scram insertion time specification to be used in plant AOO analyses. The total database exceeds 1600 rod scram times. The primary impact of measured scram time is in the plant pressure/power increase AOOs and feedwater controller failure. To use this option, a plant must show that the actual plant control rod insertion time (plus three standard deviations) is within the above more realistic specification or another derived scram time specification. Operating limits for plants taking credit for measured scram time are determined using either GENESIS, GEMINI or TRACG methods and procedures.

S.5.1.5.2 Generic Statistical Scram Time (ODYN Option B or TRACG Option B)

GE has developed a generic statistical scram time distribution for the purposes of generating the AOO ΔCPR adjustment factors required for ODYN Option B or TRACG Option B (see Section 4.0 of Reference S–1). Those plants operating under Option B MCPR operating limits will be taking advantage of the improved scram time benefits on the AOO performance, by demonstrating that actual scram speeds conform with the generic statistical scram times assumed. Operating limits for plants taking credit for the generic statistical scram time are determined using either GENESIS, GEMINI or TRACG methods and procedures.

S.5.2 Operating Flexibility Options

The following operating flexibility options have been developed for BWRs:

(1) Single–Loop Operation.

(2) Load Line Limit.

(3) Extended Load Line Limit.

(4) Increased Core Flow.

(5) Feedwater Temperature Reduction.

(6) ARTS Program (BWR/3–5).

(7) Maximum Extended Operating Domain for BWR/6 and Maximum Extended Load Line Limit Analysis for BWR/3–5.

(8) Turbine Bypass Out of Service.

(9) Safety/Relief Valves Out of Service.

(10) ADS Valve Out of Service.

(11) End–of–Cycle Recirculation Pump Trip Out of Service.

(12) Main Steam Isolation Valves Out of Service.

(13) Maximum Extended Load Line Limit Analysis Plus.

Figure S-5 provides a general illustration of the history of power-flow domain changes.

The supplemental reload licensing report indicates if an option has been chosen.

Some plants referencing GESTAR II as the applied reload methodology may include the GE Licensing Topical Report, Applicability of GE Methods to Expanded Operating Domains (Reference S-101), as part of their licensing basis. For such a plant, the limitations, conditions, and requirements of Reference S-101 are included in the analysis and licensing basis for the reload. The applicability of Reference S-101 has been expanded to include GNF2 fuel by Reference S-108. Reference S-101 has been updated to NEDC-33173P-A Revision 3 reflecting NRC approval of Supplement 2 (Reference S-109). This approval allows a reduction of the additional margin applied to the Safety Limit Minimum Critical Power Ratio (SLMCPR) in Revision 1. The limitations and conditions included in the NRC Safety Evaluation in Reference S-101 modify the SLMCPR margin to be applied to plants referencing NEDC-33173P, Applicability of GE Methods to Expanded Operating Domains (Reference S-101), as part of their licensing basis. The plan for the implementation of PRIME in downstream methods has been reviewed and approved by the NRC (Reference S-110).

S.5.2.1 Single–Loop Operation

Technical Specifications for a plant without a Single–Loop Operation (SLO) analysis do not allow operation beyond a relatively short period of time if an idle recirculation loop cannot be

returned to service. Typically, the plant shall not be operated for a period in excess of 24 hours with one recirculation loop out of service.

The capability of operating at reduced power with a single recirculation loop is highly desirable, from a plant availability/outage planning standpoint, in the event maintenance of a recirculation pump or other components renders one loop inoperative. The SLO analysis evaluates the plant for continuous operation at a maximum expected power output that is 20% to 30% below that which is attainable for two–pump operation.

To justify SLO, safety analyses have to be reviewed for one–pump operation. The MCPR fuel cladding integrity safety limit, AOO analyses, operating limit MCPR, and non–LOCA accidents are evaluated. Increased uncertainties in the total core flow and traversing in–core probe (TIP) readings result in a small increase in the fuel cladding integrity safety limit MCPR.

SLO can also result in changes to plant response during a LOCA. These changes are accommodated by the application of reduction factors to the two–loop operation MAPLHGRs if required. MAPLHGR reduction factors are evaluated on a plant–by–plant and fuel type dependent basis. In each subsequent reload, reduction factors are checked for validity and, if new fuel types are added, new reduction factors may be needed in order to maintain the validity of the SLO analysis.

S.5.2.2 Load Line Limit (BWR/2–4 Only)

For non–barrier fuel, fuel pellet–cladding interaction considerations inhibit withdrawal of control rods at high power levels. In order to attain rated power and not exceed rated core flow without control rod withdrawals at high power when using non–barrier fuel, operation above the rated load line is required during power ascension. Consequently, an analysis referred to as the Load Line Limit Analysis (LLLA) is performed to determine if the safety consequences of operation above the rated load line, but within a defined region of the power flow map, are bounded by the respective consequences of operation at the licensing basis conditions.

The region above the rated load line is known as the extended operating region and is defined by the locus of power/flow points bounded by:

(1) the rated load line;

(2) the APRM rod block line; and

(3) the rod block intercept line (BWR/2 and 3), or the rod block intercept line and the rated power line (BWR/4).

LLLA is performed on a plant/cycle–specific basis. However, after the LLLA is initially performed for a plant and cycle, on subsequent cycles only the following checks need to be made in addition to the standard reload analyses to support operation in the extended operation region:

(1) **LOCA** – The applicability of previous LOCA analyses to the extended operating region must be verified for each plant during each cycle.

(2) **AOOs** – The consequences of AOOs are evaluated to determine if operating limit adjustments are necessary for operation in the extended operating range.

BWR/5 and 6 are designed with expanded operating flexibility that supports plant operation in an extended region above the rated load line up to rated power. This expanded flexibility is validated whenever a fuel design with different transient response characteristics is introduced.

S.5.2.3 Extended Load Line Limit (BWR/2–6)

The Extended Load Line Limit Analysis (ELLLA) is similar to the LLLA described in Subsection S.5.2.2. However, the extended operating domain for ELLLA, instead, has an upper bound of the APRM rod block line to rated power for BWR/2–6.

Once ELLLA has been performed for a specific plant and cycle, it is reverified for applicability to subsequent cycles as described in Sub–section S.5.2.2. Because of the different extended operating regions for ELLLA and LLLA, the power/flow points chosen for analysis may be different.

Some plants have, in plant specific submittals, relaxed the APRM rod block setpoints. For these plants, the ELLLA region no longer corresponds to the APRM rod block line. The APRM setpoints and the analyzed operating domain are defined in the plant specific licensing documentation.

S.5.2.4 Increased Core Flow (ICF) Operation

Analyses are performed in order to justify operation at core flow rates in excess of the 100% rated flow condition. The analyses are done for application through the cycle or for application at the end of cycle only.

The limiting AOOs that are analyzed at rated flow as part of the supplemental reload licensing report are reanalyzed for increased core flow operation. In addition, the loss–of–coolant accident (LOCA), fuel loading error evaluated as an AOO only, rod drop accident, and rod withdrawal error are also re–evaluated for increased flow operation to assure that the higher flow and exposure capability does not significantly impact these analyses.

The effects of the increased pressure differences on the reactor internal components, fuel channels, and fuel bundles as a result of the increased flow are analyzed in order to ensure that the design limits will not be exceeded.

The thermal–hydraulic stability is re–evaluated for increased core flow operation, and the effects of flow–induced vibration are also evaluated to assure that the vibration criteria will not be exceeded.

S.5.2.5 Feedwater Temperature Reduction (FWTR)

Analyses are performed in order to justify operation at a reduced feedwater temperature at rated thermal power. Usually, the analyses are performed for end–of–cycle operation with the last–stage feedwater heaters valved out in order to increase the core rated power exposure capability. However, throughout cycle operation, some feedwater temperature reduction can be justified by analyses at the appropriate operating conditions for accommodating the potential of a feedwater heater being out of service.

The limiting AOOs are reanalyzed for operation at a reduced feedwater temperature. In addition, the loss–of–coolant accident (LOCA), fuel loading error evaluated as an AOO only, rod drop accident, and rod withdrawal error are also re–evaluated for operation at a reduced feedwater temperature to assure that the higher subcooling and exposure capability does not significantly impact these analyses.

The reactor core and thermal–hydraulic stability are re–evaluated, along with the increase in the feedwater nozzle fatigue usage factor, for operation at a reduced feedwater temperature throughout the cycle.

S.5.2.6 ARTS Program (BWR/3–5)

The ARTS program is a comprehensive project involving the Average Power Range Monitor (APRM), the Rod Block Monitor (RBM), and Technical Specification improvements.

Implementing the ARTS program provides for the following improvements that enhance the flexibility of the BWR during power level monitoring.

(1) The average power range monitor (APRM) trip setdown requirement is replaced by a power–dependent MCPR operating limit similar to that used in the BWR/6, and a flow–dependent MCPR operating limit to reduce the need for manual setpoint adjustments. In addition, another set of LHGR power– and flow–dependent limits will also be specified for more rigorous fuel thermal protection during postulated transients at off–rated conditions. These power– and flow–dependent limits are verified for plant–specific application during the initial ARTS licensing implementation and are applicable to subsequent cycles provided that there are no changes to the plant configuration as assumed in the licensing analyses. A plant may also include the power– and flow–dependent limits for MAPLHGR.

(2) The RBM system may be modified from flow–biased to power–dependent trips to allow the use of a new generic non–limiting analysis for the rod withdrawal error (RWE) and to improve response predictability to reduce the frequency of nonessential alarms. The applicability of the generic RWE analysis to GE fuel designs is discussed in Reference S–2.

The resulting improvements in the flexibility of the BWR provided by ARTS are designed to significantly minimize the time to achieve full power from startup conditions.

S.5.2.7 Maximum Extended Operating Domain for BWR/6 and Maximum Extended Load Line Limit Analysis for BWR/3–5

The modified operating envelope termed Maximum Extended Operating Domain (MEOD) for BWR/6 plants permits extension of operation into higher load line power/flow areas, provides improved power ascension capability to full power and additional flow range at rated power, and includes an increased flow region to compensate for reactivity reduction due to exposure during an operating cycle. Overall, MEOD can be utilized to increase operating flexibility and plant capacity factor. The higher load line aspect of MEOD is also applied to BWR/3–5 plants as a Maximum Extended Load Line Limit Analysis (MELLLA). The higher core flow aspect of MEOD is also applied to BWR/3–5 plants as an Increased Core Flow (ICF) Analysis (see Section S.5.2.4).

The extended load line region boundary of MEOD is typically limited to 75% core flow at 100% of the original plant licensed thermal power and the corresponding power/flow constant rod line. The increased–core–flow region is defined on a plant-specific basis (typically between 105 and 110% of rated core flow)and is limited by plant recirculation system capability, acceptable flow–induced vibration, fuel lift considerations, and force impact on the vessel internal components.

Evaluations performed for MEOD conditions include normal and AOOs, LOCA analysis, containment responses, stability, flow–induced vibration, and the effects of increased flow–induced loads on reactor internal components and fuel channels. The limiting AOOs applicable to each plant basis are evaluated for the normal range of operating power and flow conditions. The AOO analyses results are used to establish power and flow dependent MAPLHGR (or LHGR) limits to replace the APRM trip setdown requirement for protection at off–rated power and flow conditions. Also, the power and flow dependent MCPR limits are revised to incorporate the results of the AOO analyses. The MEOD power and flow dependent limits are evaluated for application to follow–on cycles.

S.5.2.8 Turbine Bypass Out of Service (TBOOS)

Some plant technical specifications require surveillance testing of the turbine bypass system response time. Operation of the turbine bypass system is assumed in the analysis of the feedwater controller failure–maximum demand event (see Section S.2.2.1.6). If this event is limiting or near limiting, the operating limit MCPR basis may be invalid if the bypass system cannot be demonstrated to meet response time requirements. Reload evaluations may incorporate a FWCF without credit for bypass operation calculation as a provision when required bypass surveillance cannot be performed, or other temporary factors render the system unavailable. Additionally, for extended operation with degraded bypass system operation, evaluations in support of this condition are augmented with the appropriate limiting events, such as the FWCF, for the applicable cycle.

S.5.2.9 Safety/Relief Valves Out of Service.

This option provides support to operate the plant with one or more safety and/or relief valves declared out of service and is normally included with the SRV setpoint tolerance increase

(References S–97 and S–98). The analysis shall include the vessel overpressure, fuel thermal limits, fuel performance during ECCS–LOCA events, high pressure systems performance (HPCS, RCIC, SLCS) and responses to Anticipated Transients Without Scram.

S.5.2.10 ADS Valve Out of Service.

This option provides justification for continuous operation with the automatic depressurization function of one automatic depressurization valve declared out of service. This contingency analysis shall allow flexibility when complying with the technical specification for continuous operation at full power with one ADS valve declared out of service.

S.5.2.11 End–of–Cycle Recirculation Pump Trip Out of Service.

In the event that the end–of–cycle recirculation pump trip becomes inoperable and is therefore not capable of performing its intended function (a recirculation pump trip during specific AOOs), operation can continue at full power when this option is included. Specific AOOs that are terminated by scram due to turbine control valve or turbine stop valve closure will be analyzed without credit to having the recirculation pumps trip system operable.

S.5.2.12 Main Steam Isolation Valves Out of Service

This option provides justification for continuous operation with a main steam isolation valve out of service when there is not compliance with the requirements of the technical specifications for the main steam isolation valves closure characteristics. The analyses include: fuel thermal limits analysis, vessel overpressure, fuel performance during events of ECCS–LOCA, and analysis of operational aspects, such as margin or adjustment to main steam high flow.

S.5.2.13 Maximum Extended Load Line Limit Analysis Plus

This option is based on the expanded operating range described in Reference S-107 as the Maximum Extended Load Line Limit Analysis Plus (MELLLA+). MELLLA+ extends the licensed operating ranges identified as Extended Load Line Limit Analysis (ELLLA), Maximum Extended Load Line Limit Analysis (MELLLA), and Maximum Extended Operating Domain (MEOD), which includes MELLLA with increased core flow (ICF). The MELLLA/MEOD operating range boundary is characterized by the statepoint of 100% OLTP at 75% of rated core flow. Uprated BWRs have restricted their operation consistent with the MELLLA boundary, which reduces the core flow range available for operation at uprated power. For plants that are uprated to 120% OLTP, the MELLLA boundary restricts the minimum core flow to 99% of rated core flow at full power operation. Figure S-5 provides a general illustration of the history of power-flow domain changes.

The MELLLA+ operating range expansion is applied as an incremental change to previously approved licensed power uprates. This option supports operation up to 120% OLTP with core flow as low as 80% of rated. For plants that have the MELLLA+ operating domain as part of

their licensing basis, the limitations, conditions, and requirements of Reference S-107 are included in the analysis and licensing basis for the reload.

S.5.3 Fuel Loading Error Analysis Requirements

Since 1978, the fuel loading error (FLE) has been analyzed as an AOO and, as such, the change in CPR for the event has been factored into the determination of the MCPR operating limit for each cycle. Section 6.3 of the GESTAR Rev 0 SER May 12, 1978 (Appendix C, Pg. US.C-4) describes the basis for this treatment of the FLE, which includes fuel-loading experience in that time period. In 1981, utilities began improving the procedures used for core verification following refueling. As shown in Reference S-99, the fuel loading error rate for the recent 25-year period and the trend for the most recent 10 years of refueling outages support the classification of the FLE event as an "Infrequent Incident." Section S.2.1 provides the basis for categorizing the FLE as an Infrequent Incident and the analysis limits.

The FLE will be analyzed as an Infrequent Incident provided that the plant confirms the requirements for application of the generic analysis. Should the plant be unable to confirm the requirements, the FLE will be evaluated to meet the fuel cladding integrity safety limit MCPR. Several items must be confirmed and documented through the reload design documentation. The first confirmation involves the core verification procedures applied following refueling, and the second involves the basis for the dose analyses and plant off-gas system bases used to perform the generic radiological analysis. The requirements apply for plants with either 10CFR100 or 10CFR50.67 radiological licensing bases.

Core Verification

The application of the Reference S-99 basis for the FLE requires that plant's core verification procedures must be consistent with those generally used during the recent historical period forming the basis for the Amendment 28 analysis of the event frequency. Therefore, the plant must confirm that their core verification procedures have the following characteristics:

1. During fuel movement, each move (location, orientation, and seating) is observed and checked at the time of completion by the operator and a spotter.
2. After completion of the core load, the core is verified by video recording the core using an underwater camera. The recording may involve two or more records made at different ranges to: provide clear resolution of the bundle serial number, illustrate the orientation in four bundle clusters, and illustrate the proper seating of the bundles. The core verification may take place during the recording process, by viewing after recording, or a combination.
3. Two independent reviewers perform the verification of the bundle serial number (location) and orientation. Each independent review records the bundle serial numbers on a core map, which is verified with the planned as loaded core.

Offsite Radiological Analysis

The plant Chi/Q values used in the applicability confirmation should represent limiting design basis accident Chi/Q values calculated using NRC guidance such as Regulatory Guide (RG)

1.145, "Atmospheric Dispersion Models for Potential Accident Consequence Assessments at Nuclear Power Plants, or other methods specifically approved by the NRC for offsite dose analysis at the plant site. The offsite radiological analysis depends on the plant configuration:

Scenario 1 - Plants that have a main steam line high radiation isolation trip.

For plants with a 10CFR100 radiological basis, the limiting 2-hour Chi/Q value at the exclusion area boundary (EAB) is 1.67×10^{-3} s/m^3. Therefore, the plant must confirm that the 2-hour Chi/Q value at the EAB is less than or equal to this value. Any dispersion coefficient less than this value will result in a dose less than the Thyroid 30 Rem limit.

For plants with a 10CFR50.67 radiological basis, the limiting 2-hour Chi/Q value at the exclusion area boundary (EAB) is 5.04×10^{-3} s/m^3. Therefore, the plant must confirm that the 2-hour Chi/Q value at the EAB is less than or equal to this value. Any dispersion coefficient less than this value will result in a dose less than the TEDE 2.5 Rem limit.

Scenario 2 - Plants that do not have a main steam line high radiation isolation trip.

Scenario 2 requires that the plant have an augmented offgas system with the capability to remove iodine indefinitely. The design capability of the augmented offgas system must be confirmed by Scenario 2 plants.

For plants with a 10CFR100 radiological basis, Figures S-6 and S-7 will be used to confirm the applicability of the generic analysis. Three parameters are needed to use these figures: the 2-hour Chi/Q value at the EAB and the hold-up time for krypton and xenon.

The following is an example of determining the dose to be compared to the limit:

Low temperature offgas systems supplied by GE provide minimum decay times of 46 hours for krypton and 42 days for xenon, at the design basis air in-leakage rate of 30 cubic feet per minute. For these decay times, the doses from Figures S-6 and S-7 for the 2-hour Chi/Q at the EAB value of 3×10^{-4} are approximately 1.6×10^{-3} and 7.9×10^{-3} for the krypton and xenon, respectively. Summing these results in an approximate total of 9.5×10^{-3} Rem, which is much less than the 2.5 Rem whole body dose limit. Using the plant specific parameters, the plant must confirm that the plant specific result is less than the 2.5 Rem whole body dose limit.

In a similar fashion, plants with a 10CFR50.67 radiological basis will use Figures S-8 and S-9 to confirm the applicability of the generic analysis. Using the plant specific parameters, the plant must confirm that the plant specific result is less than the 2.5 Rem TEDE dose limit.

Control Room Radiological Analysis

The control room Chi/Q values reported for use in the applicability confirmation should represent limiting design basis accident Chi/Q values calculated using NRC guidance such as Regulatory Guide 1.194, "Atmospheric Relative Concentrations for Control Room Radiological Habitability Assessments at Nuclear Power Plants," or other methods specifically approved by the NRC for control room dose analysis at the plant site.

For plants with a 10CFR100 radiological basis, the maximum allowable control room Chi/Q value is 1.81 x 10^{-3} s/m^3. Therefore, the plant must confirm that the maximum control room Chi/Q value is less than or equal to this value. Any dispersion coefficient less than this value will result in a dose less than the 30 Rem Thyroid limit.

For plants with a 10CFR50.67 radiological basis, the maximum allowable control room Chi/Q value is 1.25 x 10^{-2} s/m^3. Therefore, the plant must confirm that the maximum control room Chi/Q value is less than or equal to this value. Any dispersion coefficient less than this value will result in a dose less than the 5.0 Rem TEDE limit.

S.6 References

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Table S–1

Sensitivity of CPR to Various Thermal–Hydraulic Parameters

Parameter	Approximate Nominal Value	ΔCPR/Nominal CPR (ΔParameter/Nominal Parameter)
Bundle Power (or Relative Bundle Power)	6–6.7 MWt	0 to –1.0
Bundle Coolant Flow	G = 1.1 x 10^6 lbm/hr–ft^2	+0.2 (BWR/4)
Core Coolant Inlet Subcooling	20–27 Btu/lbm	+0.1
R–factor	1.04–1.10	–2.1
Core Pressure (with constant coolant subcooling)	1,035–1,055 psia	–0.6

Table S–2

Plants for which ATWS Pump Trip is Assumed in Transient Analyses

Duane Arnold	Cooper	Fitzpatrick	Hatch 1 & 2
Brunswick 1 & 2	Peach Bottom 2 & 3	Browns Ferry 1, 2, & 3	Vermont Yankee
Pilgrim	Millstone	Dresden 2 & 3	Quad Cities 1 & 2
Monticello	Fermi 2	Hope Creek 1 & 2	Limerick 1 & 2
Shoreham	Susquehanna 1 & 2	Hanford 2	LaSalle 1 & 2
Nine Mile Point 1 & 2	Clinton 1	Grand Gulf 1 & 2	Perry 1 & 2
River Bend 1	Oyster Creek		

Table S–3

ΔCPR as a Function of RBM Setpoint for Generic Rod Withdrawal Error Analysis

RBM Setpoint	ΔCPR
104	0.13
105	0.16
106	0.19
107	0.22
108	0.28
109	0.32
110	0.36

Table S–4

Group Notch Plants[2]

Browns Ferry 1, 2, & 3	Peach Bottom 2 & 3
Fitzpatrick	Cooper
Duane Arnold	Hatch 1 & 2
Brunswick 1 & 2	Fermi 2

[2] Plants that have implemented the requirements described in Reference S–9 or S–10 are no longer classified as Group Notch plants.

Table S–5
Specific Plant Analysis

Plant	Analysis Basis	Specific Plant LOCA Analysis Document	Reference Lead Plant LOCA Analysis Document
Nine Mile Point 1	SAFER/GESTR–LOCA	S–53	N/A
Nine Mile Point 2	SAFER/GESTR–LOCA	S–72	N/A
Dresden 2 and 3	SAFER/GESTR–LOCA	S–54	N/A
Quad Cities 1 and 2	SAFER/GESTR–LOCA	S–54	N/A
LaSalle 1 and 2	SAFER/GESTR–LOCA	S–73	N/A
Monticello	SAFER/GESTR–LOCA	S–55	N/A
Fermi 2	SAFER/GESTR–LOCA	S–56	N/A
Duane Arnold	SAFER/GESTR–LOCA	S–57	N/A
Pilgrim	SAFER/GESTR–LOCA	S–58	N/A
Browns Ferry 1, 2 and 3	SAFER/GESTR–LOCA	S–59	N/A
Hope Creek	SAFER/GESTR–LOCA	S–60	N/A
Fitzpatrick	SAFER/GESTR–LOCA	S–71	N/A
Cooper	SAFER/GESTR–LOCA	S–61	N/A
Hatch 1 and 2	SAFER/GESTR–LOCA	S–62	N/A
Brunswick 1 and 2	SAFER/GESTR–LOCA	S–63	N/A
Clinton	SAFER/GESTR–LOCA	S–64	N/A
Vermont Yankee	SAFER/GESTR–LOCA	S–65	N/A
River Bend	SAFER/GESTR–LOCA	S–66	N/A
Limerick 1 and 2	SAFER/GESTR–LOCA	S–67	N/A
Peach Bottom 2 and 3	SAFER/GESTR–LOCA	S–68	N/A
Perry	SAFER/GESTR–LOCA	S–69	N/A
Oyster Creek	SAFER/GESTR–LOCA	S–70	N/A
Susquehanna 1 and 2	SAFER/GESTR–LOCA	S–75	N/A
WNP–2	SAFER/GESTR–LOCA	S–86	N/A
Grand Gulf	SAFER/GESTR–LOCA	S–87	N/A

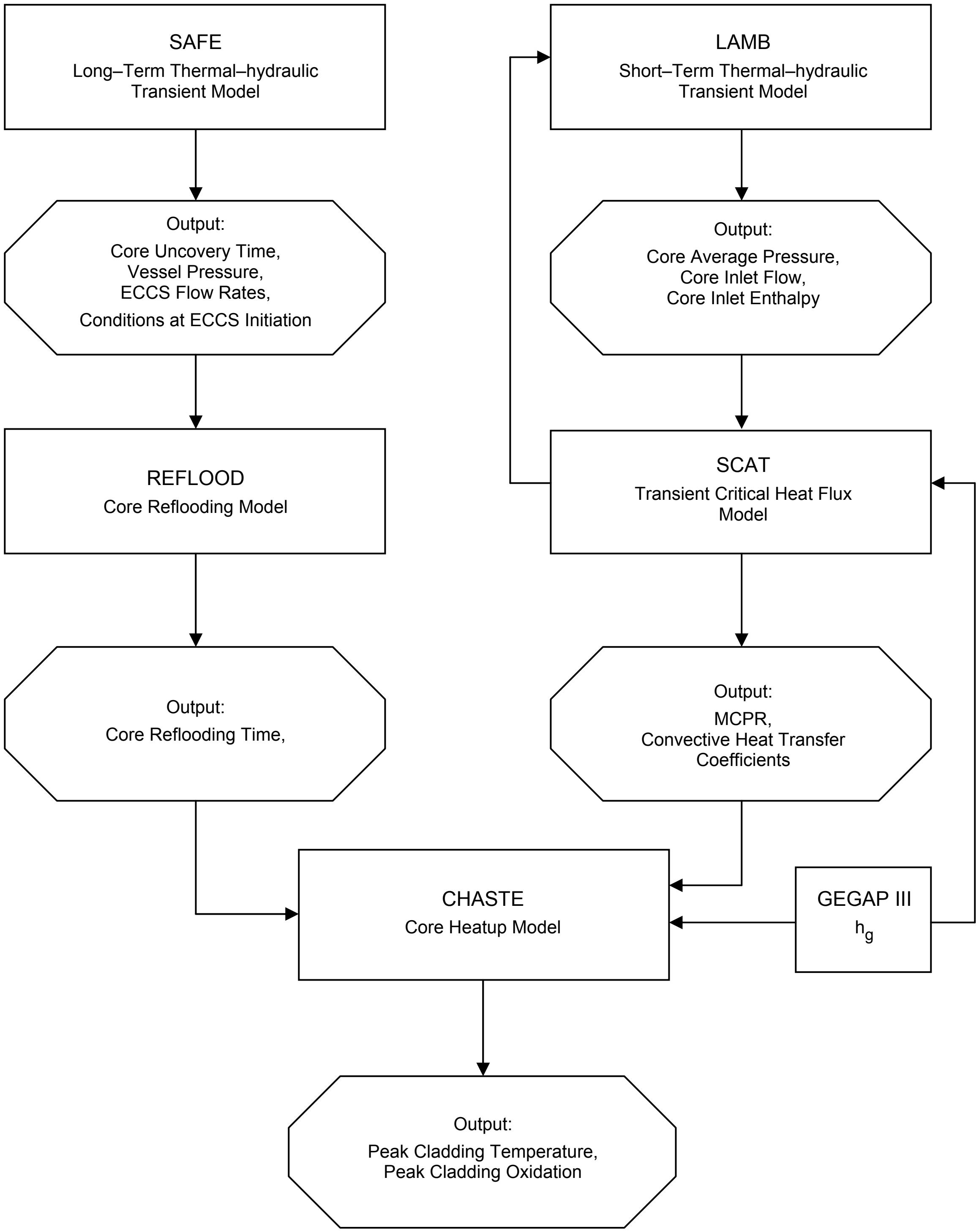


Figure S–1. Loss–of–Coolant Accident Evaluation Model (SAFE/REFLOOD Analysis Methods

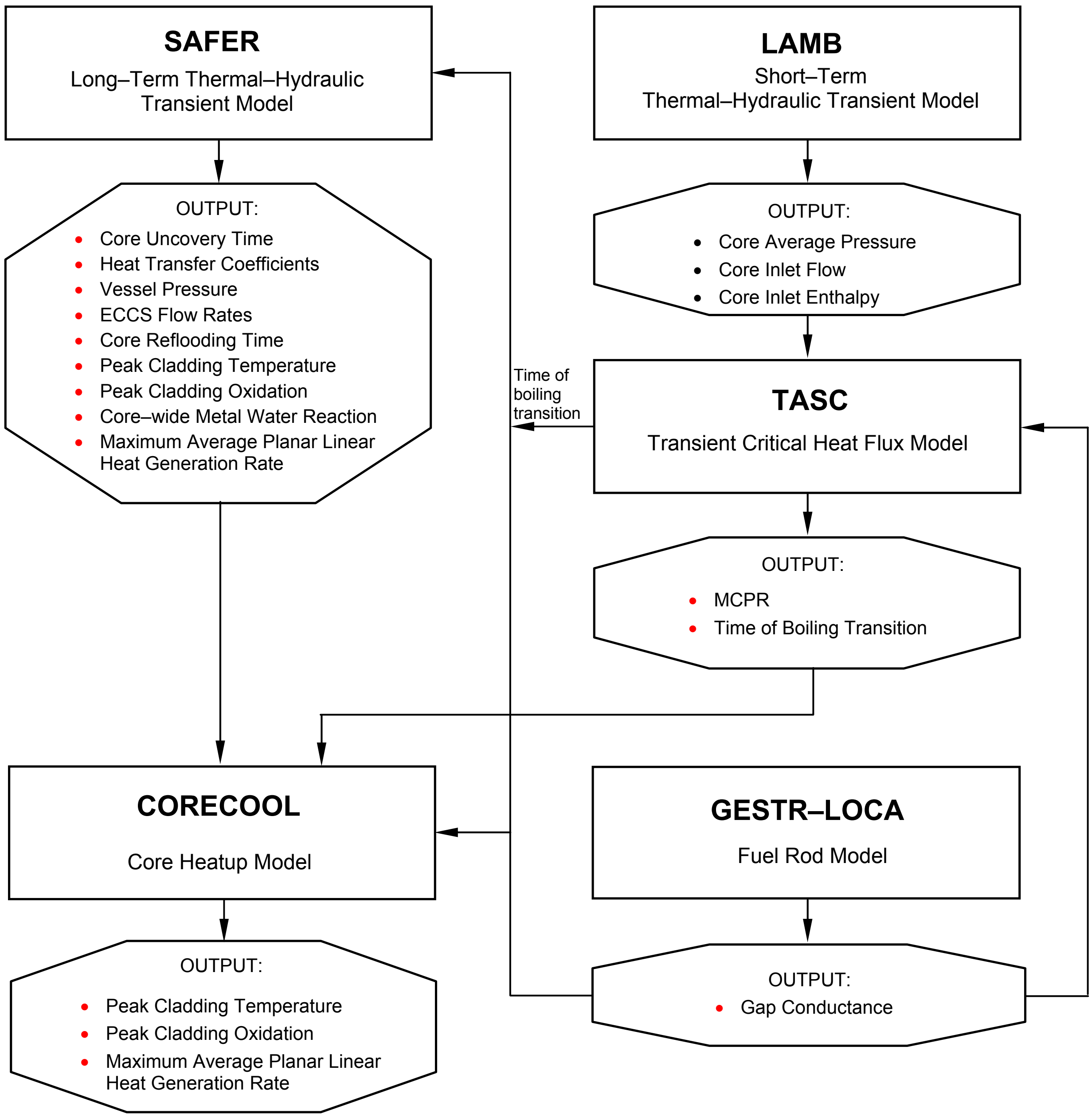


Figure S–2. Loss–of–Coolant Accident Evaluation Model (SAFER/GESTR Analysis Methods)

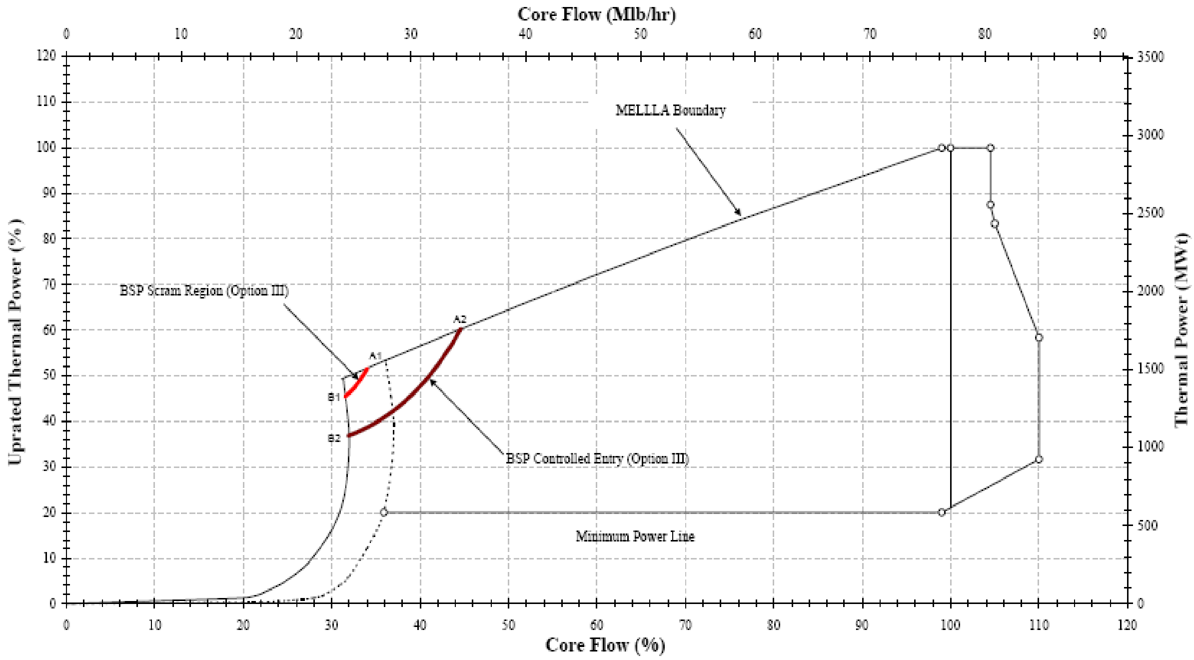


Figure S-3. Illustration of Scram and Controlled Entry Region Boundaries for BSP for Option III

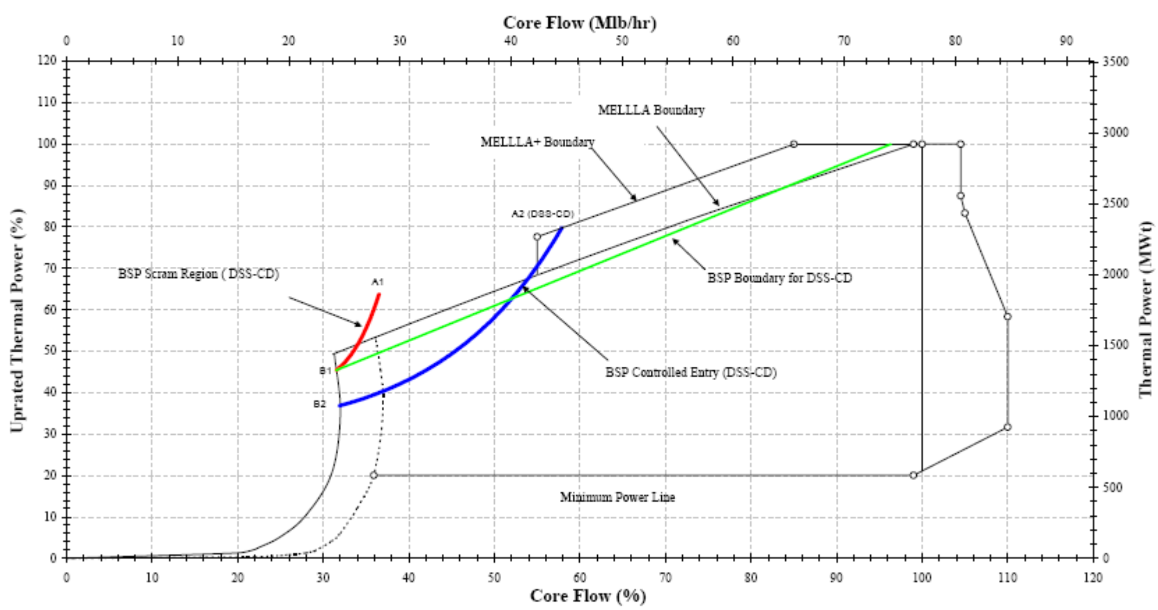
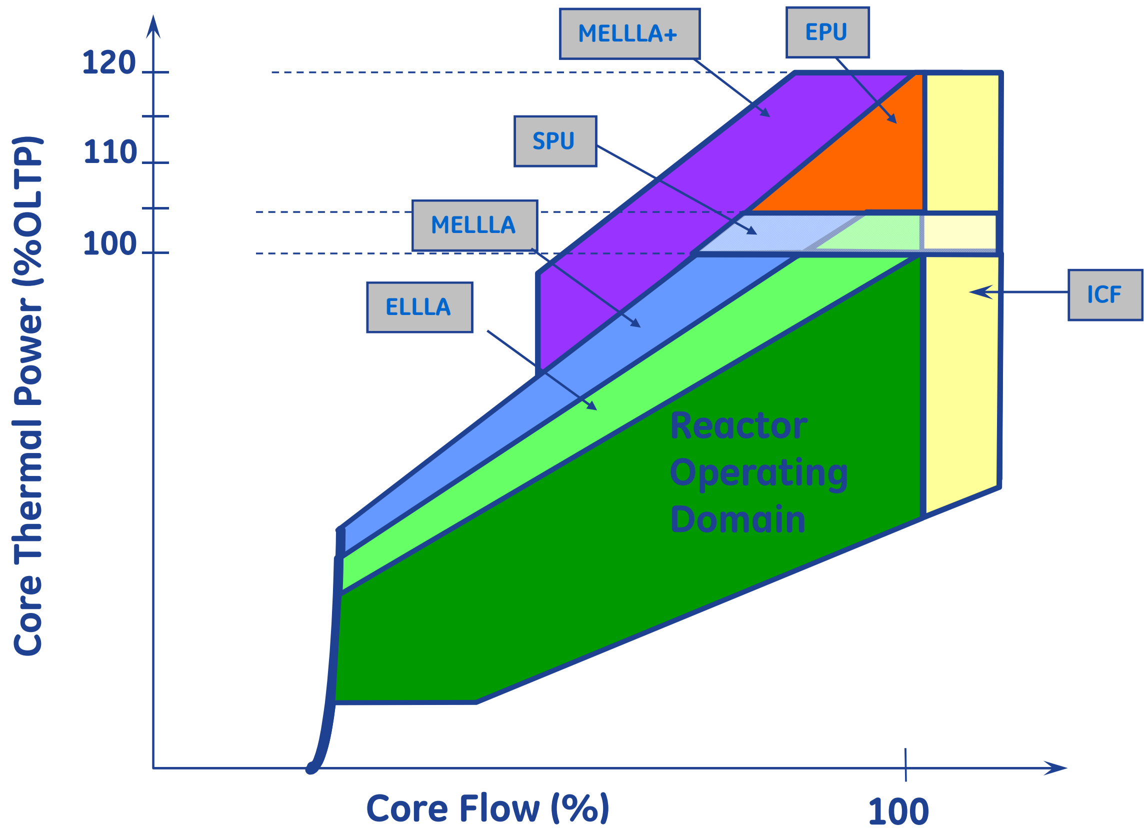


Figure S-4. Illustration of Scram and Controlled Entry Region Boundaries, and the BSP Boundary for BSP for DSS-CD



OLTP = Original Licensed Thermal Power

MELLLA = Maximum Extended Load Line Limit

SPU = Stretch Power Uprate (5% OLTP)

ELLLA = Extended Load Line Limit Analysis

ICF = Increased Core Flow

MELLLA+ = MELLLA Plus

EPU = Extended Power Uprate (20% OLTP)

Figure S-5. Power-Flow Operating Domain Illustration

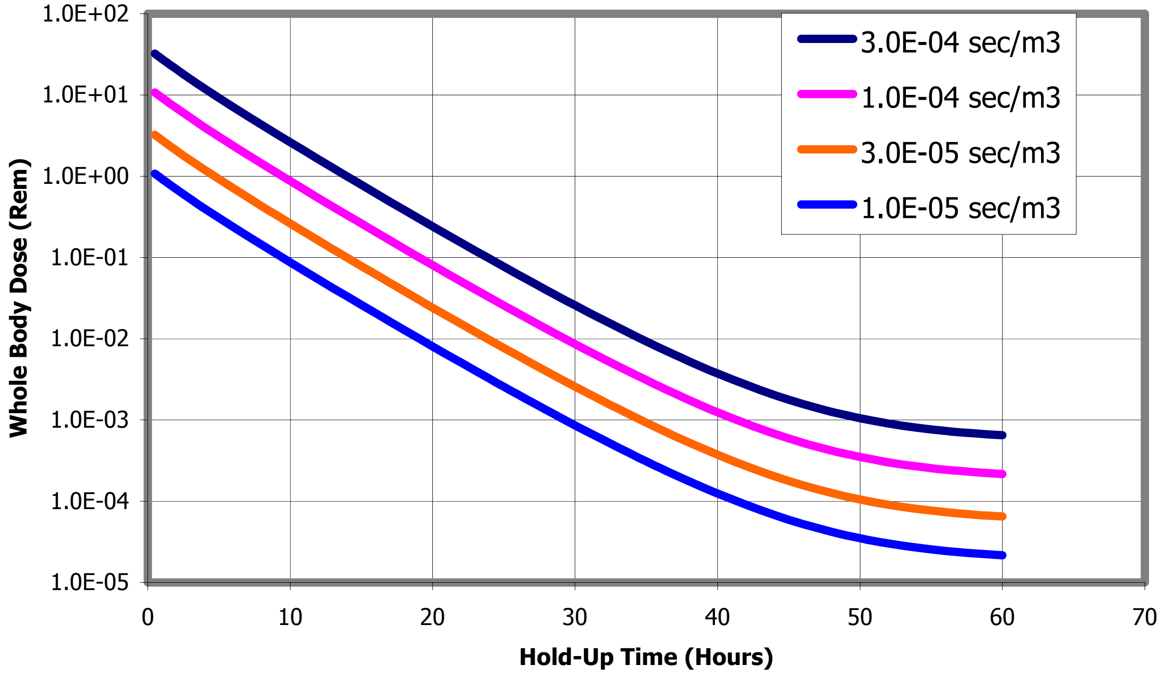


Figure S-6. Scenario 2 Krypton Whole Body Dose with Respect to Charcoal Hold Up

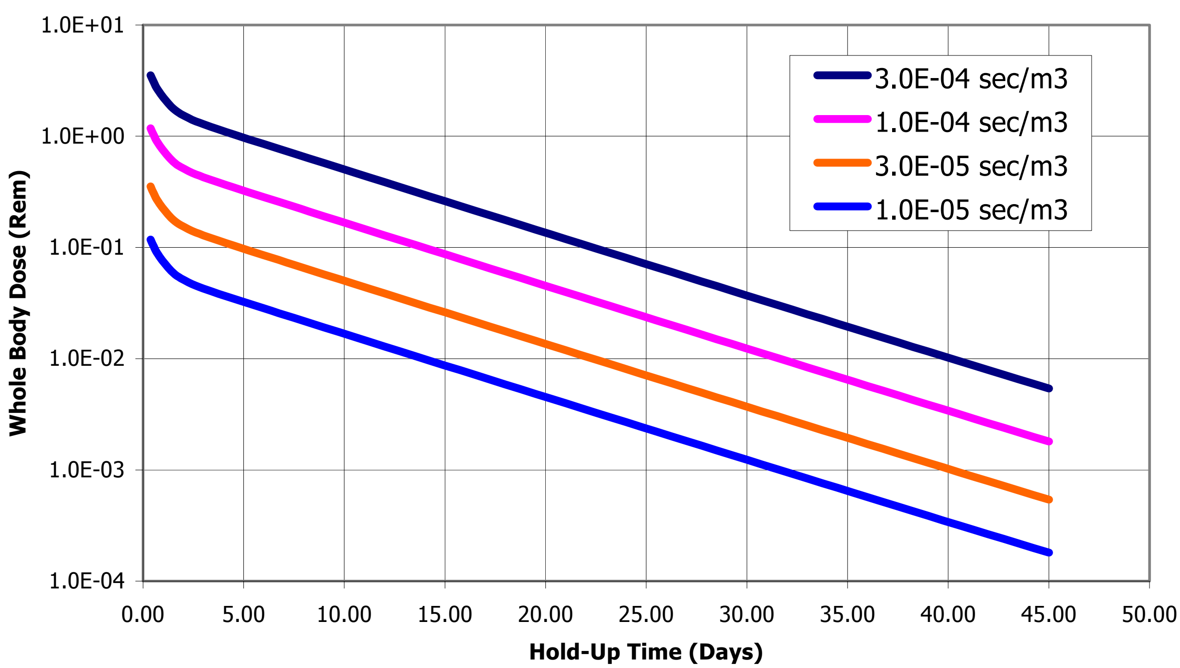


Figure S-7. Scenario 2 Xenon Whole Body Dose with Respect to Charcoal Hold Up

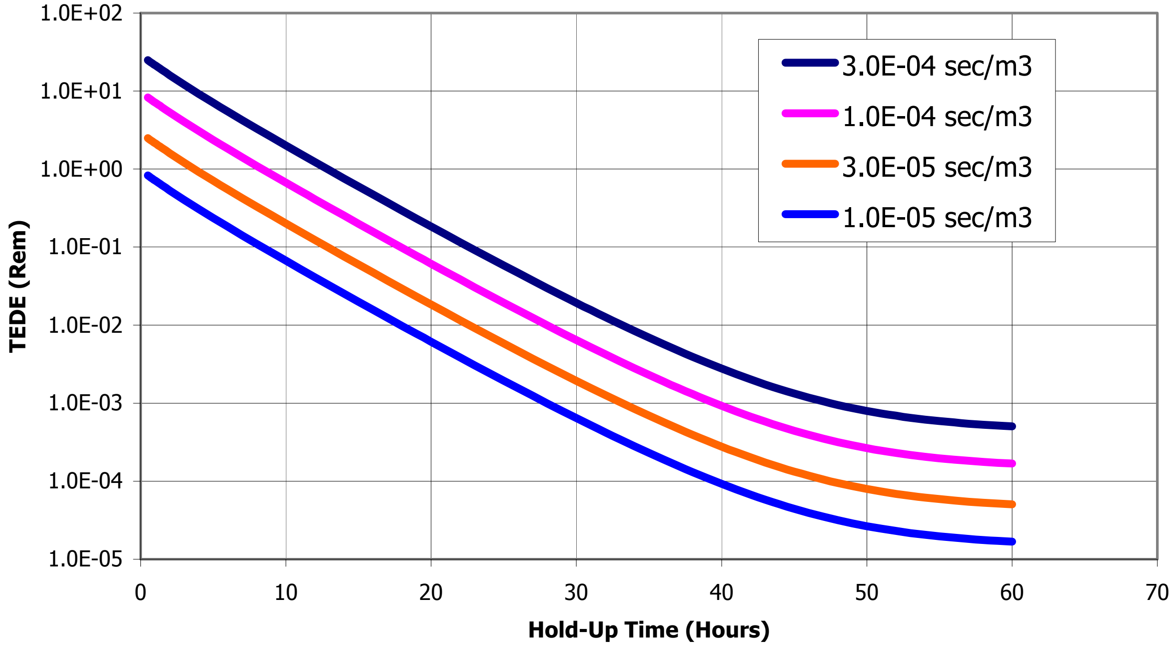


Figure S-8. Scenario 2 Krypton TEDE with Respect to Charcoal Hold Up Utilizing AST Methodology

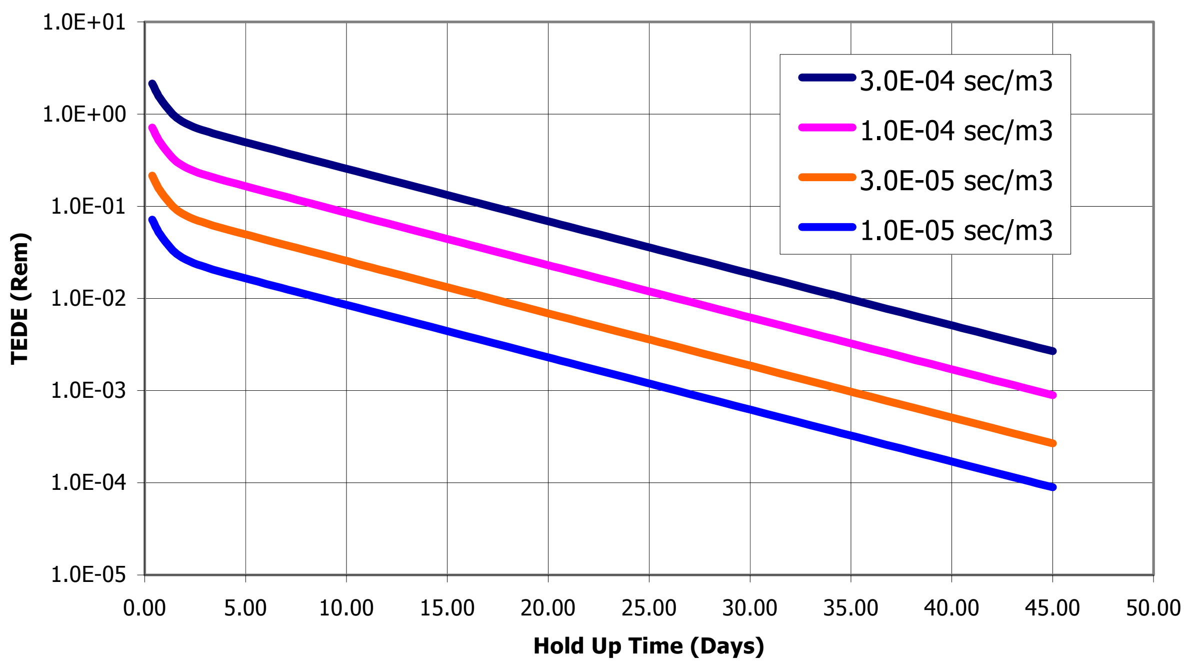


Figure S-9. Scenario 2 Xenon TEDE with Respect to Charcoal Hold Up Utilizing AST Methodology

Appendix A

Standard Supplemental Reload Licensing Report

APPENDIX A

STANDARD SUPPLEMENTAL RELOAD LICENSING REPORT

The following template provides the standard format to be used for an individual plant supplemental reload licensing report (SRLR) with end-of-cycle (EOC) limits reported. For plants that have chosen to use TRACG methods for analyzing pressurization transients, some adjustment of the information and format will be necessary. For plants that have met the requirements necessary to support the recatagorization of the fuel loading error, the ΔCPR results for the FLE events will not be provided, rather a statement regarding the recatagorization will be included.

Additional appendices and figures can be added as necessary to address plant and cycle specific issues. The following are typical lists of appendices and figures.

LIST OF APPENDICES

LIST OF FIGURES

The template includes symbols (denoted in blue) which represent plant/cycle specific information to be inserted at these locations. The following is the key to these symbols.

TEMPLATE SYMBOL KEYS:

[a] Insert plant/cycle specific wording
[n] Insert plant/cycle specific numbers
{ } Replace with plant/cycle applicable description
( ) Explanative description

Provided below is a tabulation of typical examples for each of the various types of plant/cycle applicable descriptions.

Description Category	*Example*
{Appropriate Operating Domain}	ICF
{Appropriate Exposure Range}	BOC to MOC
{Appropriate Fuel Design(s)}	GE14C
{Appropriate Extended Operating Domain Description}	Maximum Extended Load Line Limit Analysis
{Appropriate EOOS Condition Description}	Turbine Bypass Valve Out of Service
{Appropriate Transient Name}	Load Rejection w/o Bypass
{Appropriate Application Condition Name}	Equipment in Service

[nnnn] – [nnnn] - [nnnn] - SRLR
Revision [n]
Class I
{Issue Date}

Supplemental Reload Licensing Report

for

{Plant Name}

Reload [n] Cycle [n]

Important Notice Regarding Contents of This Report

Please Read Carefully

This report was prepared by Global Nuclear Fuel - Americas, LLC (GNF-A) solely for use by *{Utility Name}* ("Recipient") in support of the operating license for *{Plant Name}* (the "Nuclear Plant"). The information contained in this report (the "Information") is believed by GNF-A to be an accurate and true representation of the facts known by, obtained by or provided to GNF-A at the time this report was prepared.

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Acknowledgement

{Appropriate acknowledgement description}

The basis for this report is *General Electric Standard Application for Reactor Fuel*, NEDE-24011-P-A-*{Rev}*, *{Issue Date}*; and U.S. Supplement, NEDE-24011-P-A-*{Rev}*, *{Issue Date}*.

1. Plant-unique Items

Appendix A: Analysis Conditions
Appendix [a]: List of Acronyms

2. Reload Fuel Bundles

Fuel Type	Cycle Loaded	Number
Irradiated:		
{Appropriate Fuel Design(s)}	[n]	[nnn]
New:		
{Appropriate Fuel Design(s)}	[n]	[nnn]
Total:		**[nnn]**

3. Reference Core Loading Pattern

	Core Average Exposure	Cycle Exposure
Nominal previous end-of-cycle exposure:	[nnnnn] MWd/MT ([nnnnn] MWd/ST)	[nnnnn] MWd/MT ([nnnnn] MWd/ST)
Minimum previous end-of-cycle exposure (for cold shutdown considerations):	[nnnnn]MWd/MT ([nnnnn] MWd/ST)	[nnnnn] MWd/MT ([nnnnn] MWd/ST)
Assumed reload beginning-of-cycle exposure:	[nnnnn] MWd/MT ([nnnnn] MWd/ST)	0 MWd/MT (0 MWd/ST)
Assumed reload end-of-cycle exposure (rated conditions):	[nnnnn] MWd/MT ([nnnnn] MWd/ST)	[nnnnn] MWd/MT ([nnnnn] MWd/ST)
Reference core loading pattern:	Figure 1	

4. Calculated Core Effective Multiplication and Control System Worth - No Voids, 20°C

Beginning of Cycle, $k_{effective}$	
Uncontrolled	[n.nnn]
Fully controlled	[n.nnn]
Strongest control rod out	[n.nnn]
R, Maximum increase in strongest rod out reactivity during the cycle (Δk)	[n.nnn]
Cycle exposure at which R occurs	[nnnnn] MWd/MT ([nnnnn]MWd/ST)

5. Standby Liquid Control System Shutdown Capability

Boron (ppm) (at 20°C)	Shutdown Margin (Δk) (at 160°C, Xenon Free)	
	Analytical Requirement	**Achieved**
[nnn]	≥ [n.nnn]	[n.nnn]

6. Reload Unique GETAB Anticipated Operational Occurrences (AOO) Analysis Initial Condition Parameters [1]

Operating domain: *{Appropriate Operating Domain}*
Exposure range : *{Appropriate Exposure Range}* **(Application Condition:** *{Appropriate Application Condition}* **)**

	Peaking Factors						
Fuel Design	**Local**	**Radial**	**Axial**	**R-Factor**	**Bundle Power (MWt)**	**Bundle Flow (1000 lb/hr)**	**Initial MCPR**
{Appropriate Fuel Design(s)}	[n.nn]	[n.nn]	[n.nn]	[n.nnn]	[n.nnn]	[nnn.n]	[n.nn]

[1] Exposure range designation is defined in Table 7-1. Application condition number is defined in Section 11.

7. Selected Margin Improvement Options [2]

Recirculation pump trip:	[a]
Rod withdrawal limiter:	[a]
Thermal power monitor:	[a]
Improved scram time:	[a]
Measured scram time:	[a]
Exposure dependent limits:	[a]
Exposure points analyzed:	[a]

Table 7-1 Cycle Exposure Range Designation

Name	Exposure Range [3]
BOC to MOC	BOC[n] to EOR[n] - [nnnnn] MWd/MT ([nnnnn] MWd/ST)
MOC to EOC	EOR[n] - [nnnnn] MWd/MT ([nnnnn] MWd/ST) to EOC[n]
BOC to EOC	BOC[n] to EOC[n]

[2] Refer to the GESTAR basis document identified at the beginning of this report for the margin improvement options currently supported therein.
[3] End of Rated (EOR) is defined as the cycle exposure corresponding to all rods out, 100% power/100% flow, and normal feedwater temperature. For plants without mid-cycle OLMCPR points, EOR is not applicable.

8. Operating Flexibility Options [4]

The following information presents the operational domains and flexibility options which are supported by the reload licensing analysis.

Extended Operating Domain (EOD):	[a]
EOD type: *{Appropriate Extended Operating Domain Description}*	
Minimum core flow at rated power:	[nn.n] %
Increased Core Flow:	[a]
Flow point analyzed throughout cycle:	[nnn.n] %
Feedwater Temperature Reduction:	[a]
Feedwater temperature reduction during cycle:	[nnn.n] °F
Final feedwater temperature reduction:	[nnn.n] °F
ARTS Program:	[a]
Single Loop Operation:	[a]
Equipment Out of Service:	
{Appropriate EOOS Condition Description(s)}	[a]

9. Core-wide AOO Analysis Results [5]

Methods used: [a]

Operating domain: *{Appropriate Operating Domain}*
Exposure range : *{Appropriated Exposure Range}* **(Application Condition:** *{Appropriate Application Condition}* **)**

			Uncorrected ΔCPR	
Event	**Flux (% rated)**	**Q/A (% rated)**	*{Appropriate Fuel Design(s)}*	**Fig.**
{Appropriate Limiting Pressure and Power Increase Transient}	[nnn]	[nnn]	[n.nn]	[n]

[4] Refer to the GESTAR basis document identified at the beginning of this report for the operating flexibility options currently supported therein.
[5] Exposure range designation is defined in Table 7-1. Application condition number is defined in Section 11.

10. Local Rod Withdrawal Error (With Limiting Instrument Failure) AOO Summary

{Appropriate cycle-specific results discussion}

11. Cycle MCPR Values [6] [7]

Two loop operation safety limit: [n.nn]

Single loop operation safety limit: [n.nn]

Stability MCPR Design Basis: See Section 15

ECCS MCPR Design Basis: See Section 16 (Initial MCPR)

SLO Pump Seizure OLMCPR: See Pump Seizure Appendix (line included if applicable)

<u>Non-pressurization Events:</u>

Exposure range: BOC to EOC	
	{Appropriate Fuel Design(s)}
Control Rod Withdrawal Error (RBM setpoint at [nnn] %)	[n.nn]
Loss of Feedwater Heating (See Appendix [a])	[n.nn]
Fuel Loading Error (misoriented)	[n.nn]
Fuel Loading Error (mislocated)	[n.nn] (or, “Not Limiting”)

<u>Limiting Pressurization Events OLMCPR Summary Table:</u> [8]

Appl. Cond.	**Exposure Range**	**Option A**	**Option B**
		{Appropriate Fuel Design(s)}	*{Appropriate Fuel Design(s)}*
[n]	*{Appropriate Application Condition Name}*		
	{Applicable Exposure Range, e.g.“BOC to MOC”}	[n.nn]	[n.nn]
	{Applicable Exposure Range, e.g.“MOC to EOC”}	[n.nn]	[n.nn]

[6] Exposure range designation is defined in Table 7-1.
[7] For single loop operation, the MCPR operating limit is [n.nn] greater than the two loop value.
[8] Each application condition (Appl. Cond.) covers the entire range of licensed flow and feedwater temperature unless specified otherwise. The OLMCPR values presented apply to rated power operation based on the two loop operation safety limit MCPR.

Pressurization Events: [9]

<table>
<tr><td colspan="3">Operating domain: {Appropriate Operating Domain}
Exposure range : {Appropriate Exposure Range} (Application Condition: {Appropriate Application Condition})</td></tr>
<tr><td rowspan="2"></td><td>Option A</td><td>Option B</td></tr>
<tr><td>{Appropriate Fuel Design(s)}</td><td>{Appropriate Fuel Design(s)}</td></tr>
<tr><td>{Appropriate Transient Name}</td><td>[n.nn]</td><td>[n.nn]</td></tr>
</table>

12. Overpressurization Analysis Summary

Event	Psl (psig)	Pdome (psig)	Pv (psig)	Plant Response
MSIV Closure (Flux Scram) – *{Appropriate Operating Domain}*	[nnnn]	[nnnn]	[nnnn]	Figure [n]

13. Loading Error Results

Variable water gap misoriented bundle analysis: [a] [10]

Misoriented Fuel Bundle	ΔCPR
{Appropriate Bundle Design(s)}	[n.nn]

14. Control Rod Drop Analysis Results

{Appropriate Rod Drop Accident analysis description}

15. Stability Analysis Results

{Appropriate Stability results description}

[9] Application condition numbers shown for each of the following pressurization events represent the application conditions for which this event contributed in the determination of the limiting OLMCPR value.
[10] Includes a [n.nn] penalty due to variable water gap R-factor uncertainty.

16. Loss-of-Coolant Accident Results

16.1 10CFR50.46 Licensing Results

{Appropriate ECCS methodology and results description}

Table 16.1-1 Licensing Results

Fuel Type	Licensing Basis PCT (°F)	Local Oxidation (%)	Core-Wide Metal-Water Reaction (%)
{Appropriate Fuel Design(s) }	[nnnn]	< [n.nn]	< [n.nn]

The *{Appropriate methodology}* analysis results are documented in Reference [n] for *{Appropriate Fuel Design(s)}* in Section 16.4.

16.2 10CFR50.46 Error Evaluation

All reported errors have been corrected in the evaluation documented in Reference [n] for *{Appropriate Fuel Design(s)}* in Section 16.4.

OR (if reporting errors are applicable for this cycle)

The 10CFR50.46 errors applicable to the Licensing Basis PCT are show in the table below.

Table 16.2-1 Impact on Licensing Basis Peak Cladding Temperature for {Appropriate Fuel Design(s)}

10CFR50.46 Error Notifications		
Number	**Subject**	**PCT Impact (°F)**
[n]	*{Appropriate Error Description}*	[nnn]
	Total PCT Adder (°F)	[nnn]

The *{Appropriate Fuel Design(s)}* Licensing Basis PCT remains below the 10CFR50.46 limit of [nnn] °F.

16.3 ECCS-LOCA Operating Limits

The ECCS MAPLHGR operating limits for new fuel bundles in this cycle are shown in the tables below.

Table 16.3-1 MAPLHGR Limits

Bundle Type: *{Appropriate Bundle Design(s)}*

Average Planar Exposure		MAPLHGR Limit
GWd/MT	**GWd/ST**	**kW/ft**
0.00	0.00	[nn.nn]
[nn.nn]	[nn.nn]	[nn.nn]

The single-loop operation multiplier on LHGR and MAPLHGR, and the ECCS analytical initial MCPR values applicable to each fuel type in the new cycle core are shown in the table below.

Table 16.3-[n] Initial MCPR and Single Loop Operation LHGR and MAPLHGR Multiplier

Fuel Type	Initial MCPR	Single Loop Operation LHGR and MAPLHGR Multiplier
{Appropriate Fuel Design(s)}	[n.nnn]	[n.nn]

16.4 References

The SAFER/GESTR-LOCA analysis base report applicable to the new cycle core is listed below.

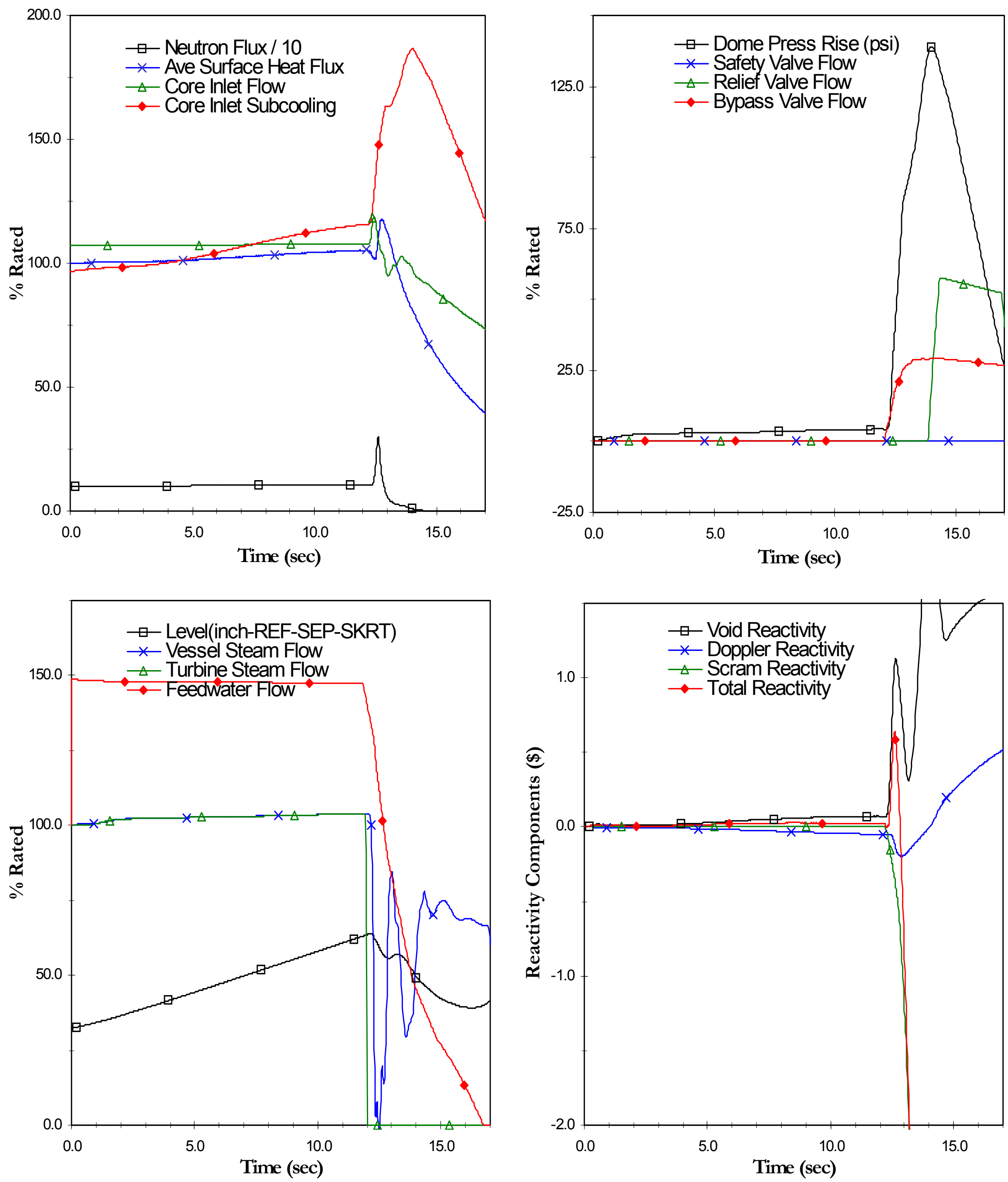
References for *{Appropriate Fuel Design(s)}*

1. *{Appropriate Reference(s) for this fuel design}*

{ Core Loading Map }

Fuel Type	
A= *{Appropriate Bundle Design(s)}* (*{Appropriate cycle}*) B= C= D=	E= F= G= H=

Figure 1 Reference Core Loading Pattern



Sample Figure [n] Plant Response to *{Appropriate Transient Analysis}*
(*{Appropriate Exposure Point and Operating Domain}* **)**

Appendix A
Analysis Conditions

The reactor operating conditions used in the reload licensing analysis for this plant and cycle are presented in Table A-1. The pressure relief and safety valve configuration for this plant are presented in Table A-2. Additionally, the operating flexibility options listed in Section 8 are supported by the reload licensing analysis.

Table A-1 Reactor Operating Conditions

Parameter	Analysis Value *{Appropriate Core Flow and Feedwater Temperature Condition(s)}*
Thermal power, MWt	[nnnn.n]
Core flow, Mlb/hr	[nnn.n]
Reactor pressure (core mid-plane), psia	[nnnn.n]
Inlet enthalpy, Btu/lb	[nnn.n]
Non-fuel power fraction	[n.nnn]
Steam flow, Mlb/hr	[nn.nn]
Dome pressure, psig	[nnnn.n]
Turbine pressure, psig	[nnn.n]

Table A-2 Pressure Relief and Safety Valve Configuration

Valve Type	Number of Valves	Lowest Setpoint (psig)
{Appropriate Valve Description}	[n]	[nnnn.n]

Appendix [a]
List of Acronyms

Acronym	Description
{Acronym}	{Acronym description}

Appendix B

Responses to NRC Questions

APPENDIX B

Table of Contents

REQUEST: Table of Contents

Complete the Table of Contents by adding all of the missing subsection headings (e.g., Subsection 2.4.1.2, **Safety Evaluation Results**). This will enable the user to more clearly understand the contents of the major sections and their interrelationships.

RESPONSE:

The Table of Contents is specifically limited to three digits so that it remains concise while directing the unfamiliar reader to the general area of interest. By adding detail, the Table of Contents would tend to become unwieldy to use and lose its effectiveness. Therefore, General Electric prefers to retain the current philosophy.

REQUEST: Section 1 – Introduction

The introduction is inadequate as presently written. It does not provide a clear presentation of the purpose, scope, philosophy, format, etc. of the Topical. The Topical will serve as a reference document for a broad range of reload reviewers. The introduction should be written to introduce the aforementioned aspects of the Topical. Specifically, it should briefly but clearly:

1. Discuss the Purpose of the Topical.

2. Discuss the Generic and Bounding analysis concepts.

3. Explain generic and bounding analysis Limitations and define those situations where Plant Specific analyses will be needed.

4. Discuss the relationship between the Topical and the Supplemental Reload Licensing Submittal.

5. Discuss the Scope (with regard to the plants involved).

6. Explain the purpose and applicability of the Thermal Margin Improvement Options Section relative to the other sections.

7. Explain the Applicability of the Physics Startup Test Section.

8. Explain when and why changes, addendums, revisions or supplements will be made and how they will be noted in the text.

RESPONSE:

The following information has been added to the applicable paragraphs in Section 1:

This report is intended to be a comprehensive reference document for plants adding General Electric reload fuel, and application of the report is limited to those plants.

The information contained in this report represents information that is independent of a plant specific reload application. Thus, efficiency is gained by not repeating information in individual reload submittals which are subjected to individual review.

The purpose and applicability of each section is contained in the initial paragraphs of that section.

Bounding analyses are performed where sufficient margins exist to allow analyses to be performed using parameters that are not expected to be exceeded in a group of reloads. Bounding analyses are not performed where they would result in a group of reloads. Bounding analyses are not performed where they would result in restricted operation. If any parameter in a given reload exceeds a parameter used in a bounding analysis, a new evaluation is required. This may take the form of a new analysis or sensitivity study which demonstrates that the amount that a parameter exceeds the bounding parameter is compensated for by another parameter being more conservative than the bounding value. Where a bounding analysis is not applicable, a plant specific analysis will be documented in the Plant Supplemental Submittal.

Until NRC approval, changes to this report will be indicated by brackets in the margins with the number of the amendment following the report number at the top of each page. When NRC approval is granted, the letter A will be added to the report identification number and a copy of the Safety Evaluation Report will be added as Appendix C. Appendix D will then be started and contain all requested changes noted in the above manner. As they are approved, they will be incorporated in the body of the approved report and the revised report distributed.

REQUEST: Subsection 2.1 Fuel Assembly Description

Discuss the safety significance of the changes made to the 8x8R fuel rod mechanical design as compared to the present 8x8 fuel rod mechanical design. That is, this section should include a qualitative discussion of the changes and the basis for the changes made to the radial and axial characteristics of the fuel rod. Comparison to the 7x7 characteristics should also be included. The effect of these changes on the fuel performance during normal (e.g., PCI, fuel duty) operation, anticipated transients (e.g., margin to 1% strain limit) and accident (e.g., LOCA PCT) should be briefly discussed. Include a discussion of the basis for the axial enrichment zoning of the UO_2 and UO_2 Gd_2O_3.

RESPONSE:

The 8x8R fuel design was introduced to realize an improvement in operating margins and uranium utilization. It should be recognized that 8x8R fuel represents only a very small change in fuel design (much less significant than previous changes). Both 8x8 and 8x8R fuel are analyzed using the same general models and methods and satisfy the same criteria. In each plant specific reload submittal, the safety analyses are performed using the same methods and satisfy the same criteria. Therefore, the introduction of 8x8R is not considered to have any significance with respect to plant safety. The effect of the fuel changes is described below.

The active fuel length is increased to 150 inches from 144 or 146 inches for BWR/4 reactors and to 145.2 inches from 144 inches for BWR/2 and 3 reactors. This is made possible by the lower fission gas release in the 8x8R design relative to the 7x7 fuel design, which allows a reduction in the fission gas plenum length originally sized for the 7x7 fuel. From this increase in active fuel length, the core

average power density is slightly reduced. Natural uranium is incorporated into the top or the top and bottom 6 inches of some fuel rods. This change reduces the core average enrichment for a fixed energy production.

The number of water tubes was increased from one to two and their outside diameter was enlarged from 0.493 to 0.591 inch. The larger water rods tend to reduce the maximum local power factor, decrease the amount of fissile inventory required to achieve a fixed energy production, and reduce the magnitude of the void coefficient of reactivity. The increase in the ratio of nonboiling to boiling water results in an additional reduction in the magnitude of the void coefficient. The changes to the void coefficient, due to the two water tubes, flatten the axial power distribution, which more than offsets the effect of the natural uranium on the axial power peaking.

The fuel rod diameter is reduced from 0.493 to 0.483 inches. The 10 mil reduction in fuel rod diameter was accomplished by reducing the pellet diameter by 6 mils and decreasing the cladding thickness by 2 mils. The maximum linear heat generation rate is preserved at 13.4 kW/ft as a design basis; therefore, the maximum fuel centerline temperature at full power remains very nearly the same.

Information on 7x7 fuel was provided in the individual plant initial core or reload submittal in which it was introduced.

REQUEST: Subsection 2.1.2 Water Rods

Briefly describe the safety basis for introducing a second water rod into the fuel assembly design. Briefly discuss the thermal hydraulic performance improvements (if any) expected from the two water rod design, as compared to a single water rod design.

RESPONSE

See Response to request on Subsection 2.1.

REQUEST: Subsection 2.1.3 – Other Fuel Assembly Components

This Section should contain subsections referring to the:

- Fuel Spacer Grid
- Finger Springs
- Upper and Lower Tieplates
- Channel Box Fastener
- Channel Box

Each subsection should contain a brief discussion of the design and purpose of the component.

RESPONSE:

Dividing Subsection 2.1.3 into smaller subsections does not serve to clarify the information within this subsection. A discussion of the design and purpose of the fuel spacers, finger springs, and upper and lower tieplates is already given in this subsection. As stated in Subsection 2.1, channels are addressed separately in Reference 2 1.

REQUEST: Subsection 2.2 – Functional Requirements

Provide a table which shows the normal, abnormal and accident conditions which were analyzed to demonstrate that the safety related functional requirements are met for each component listed in Subsection 2.1.3. Include in the table footnotes which reference the (sub)sections in the Topical which discuss, in detail, the analyses, tests or evaluations performed to show that the safety-related functional requirements are satisfied.

RESPONSE:

Capability to withstand normal, abnormal, and handling loads is addressed in the appropriate subsection of the mechanical evaluation Subsection 2.5. Handling loads are documented in Subsection 2.5.3. Seismic considerations are bounded by the evaluation of LOCA and seismic loads in Subsection 2.5.4. Individual accident analyses are documented, as indicated, in Subsection 5.5.

REQUEST: Subsection 2.3.1 – Cladding

Provide a reference for the plastic (modulus) behavior of Zircaloy 2. Table 2-2, used in the safety analyses, should reference the constitutive equations for burnup dependent material properties utilized in the thermal-mechanical evaluations (e.g., Zr 2 creep equation, irradiated growth equation). Table 2-2 indicates a total irradiation elongation of greater than 1%. Is this (>) a typographical error? Give the fluence corresponding to the percent total irradiated elongation. Sections 2.6 and 2.7 should also be briefly summarized and referenced. References 2-4 and 2-21 discuss the 7x7 fuel design. Provide a more recent reference to the 8x8 and 8x8R fuel performance in this section. The last sentence referring to UO_2 properties appears to have been incorrectly placed in this subsection. Equation 2-1 implies that the 1% plastic strain limit is for a uniaxial strain (stress) field. Describe or provide a reference as to how the 1% plastic strain criterion for the uniaxial stress is related to the experimental results from cylindrical rods, which are subject to a triaxial state of stress during irradiation.

RESPONSE:

A plastic modulus or behavior of the fuel cladding is not used in the analyses performed. With respect to interference between the pellet and the cladding, the interference is presumed to be accommodated solely by an increase in cladding strain. This is equivalent to specifying an infinite pellet modulus, in which case the actual cladding modulus (elastic or plastic) is unimportant with respect to determining the resultant cladding system.

Cladding creep and irradiation growth in the radial direction are not explicitly treated in fuel rod thermal or mechanical evaluations but are implicitly considered through data comparisons as a part of the integral fuel rod model verification. This model has been described in detail in Reference 1, and

model predictions compared to data in Reference 2-4. The relationship for cladding creep used in the cladding creep collapse analysis has been reported in Reference 2-16.

Credit is not taken for fuel rod irradiation growth in fuel rod analyses to determine fuel rod internal pressure; however, the differential axial growth among fuel rods is considered in the fuel assembly design. The relationships used for this axial irradiation growth plus plastic strain and creep of the fuel rod cladding have been added to Table 2-2. The equations are based on measurements of fuel rods operated in commercial BWR's and predict the upper and lower 95/95 tolerance limits of that data. More recent measurements on 8x8 fuel with pre irradiation measurements available, have indicated that these expressions are very conservative with respect to the 8x8 fuel design. The axial irradiation growth assumption used in densification analyses is defined in Reference 2-8.

The "total irradiated elongation" referred to in Table 2-2 was meant to describe the plastic strain capability of the cladding. The title used in Table 2-2 is being changed to avoid misunderstanding. The value provided is considered applicable over the range of peak pellet exposures expected during fuel lifetime (0–40,000 MWd/t).

A reference to Subsection 2.6 has been added to Subsection 2.3.1; Subsection 2.7 does not include material properties. Reference 2 4 was referred to in Subsection 2.3.1 because it provides supporting information in regards to the cladding plastic strain capability. Reference 2-21 did not provide any additional information with respect to cladding material properties. In Subsection 2.6.1, where fuel operating experience is discussed with respect to hydriding, a new reference (Reference 2) is being added which contains the most recent information on operating experience of BWR fuel from General Electric.

Equation 2-1 limits the diametral expansion of the fuel cladding by limiting the calculated plastic hoop strain to 1%. The calculated strain is based upon a one-dimensional (radial) model, which predicts only hoop strain. Reference 2-4 established a cladding damage limit of 1% plastic strain from data obtained from tests of irradiated zircaloy cladding. In addition to uniaxial tests, the test geometries included closed end burst specimens, which subject the tubing to a triaxial stress state. All of the burst specimens had a circumferential fracture elongation >1%, which supports a hoop strain limit of 1% in a multiaxial loading case.

REFERENCES:

1. "General Electric Standard Safety Analysis Report" Proprietary Supplement to Amendment 14, Docket No. STN-50-447, May 1974.

2. R. B. Elkins, "Experience with BWR Fuel through December 1976", NEDE-21660P (Proprietary) and NEDO-21660, July 1977.

REQUEST:

Subsection 2.3.2 – This section is missing. Please provide.

RESPONSE:

The subsection title was inadvertently omitted in the original submittal. This was corrected in Amendment 1.

REQUEST: Subsection 2.3.3 – Urania Gadolinia Fuel Pellets

Section 2.7.3 should be cross referenced for continuity.

RESPONSE:

Subsection 2.7.3 documents the manufacturing quality control of gadolinia and urania gadolinia fuel rods, while Subsection 2.3.3 provides the urania gadolinia fuel rod material properties. Thus, a cross reference is not provided.

REQUEST: Subsection 2.4 – Fuel Rod Thermal Analysis

State, in this paragraph, all the analyses performed under the heading “Fuel Rod Thermal Analysis” and which appear in the section (e.g., creep collapse, center melting). Provide a summary discussion of the distinction between the “Safety Evaluation” and the “Design Evaluations”. Summarize the plant specific reload parameter review discussed in Subsection 2.4.3. State the fuel types analyzed in the Safety Evaluation (include the 7x7 type, i.e., for kW/ft for 1% strain).

RESPONSE:

Given the character of Subsection 2.4, adding summaries for each thermal evaluation is redundant. However, this subsection has been revised in response to the requests on Subsections 2.4.1 and 2.4.2 to clarify the generic thermal analyses performed and their relationship to other analyses. Documentation of specific analyses and reload parameters remains in the appropriate subsection.

REQUEST: Subsection 2.4.1 – Safety Evaluation

Subsections 2.4.1.1 and 2.4.1.2 do not by themselves provide a complete “safety evaluation” of the fuel rod thermal performance. Subsection 2.4.1 simply establishes a correspondence between the 1% plastic strain limit and the local LHGR limit. The “safety evaluation results” and conclusions depend upon the localized anticipated transients described in Section 5 and reported in the reload supplement. Therefore, provide new headings for these sections to reflect the limits, analyses and results appearing in this section. Provide a discussion of the connection between the kW/ft values reported in the tables and the transient analyses described in Section 5. Discuss the application of these kW/ft limits to the transient analysis results that are to be reported in the plant specific reload supplements.

RESPONSE:

This subsection provides the basis and the results of a generic safety evaluation of linear heat generation rate (LHRG) associated with the 1% plastic strain safety limit. Results provided in the subsection are used in conjunction with results from analyses of plant transients which are described in Section 5.0. The peak LHGR calculated for normal and abnormal operating transients must be less than or equal to the LHGR at which 1% plastic strain is calculated to occur.

An introduction has been added to Subsection 2.4.1 which discusses the safety analysis and application of the results.

REQUEST: Subsection 2.4.1.1 – Basis for Safety Evaluation

See comments on Subsection 2.4.1. Provide a reference or summary description of the code used for this evaluation. Discuss the important assumptions incorporated in the analysis; e.g., consideration of tolerances (see comments on Subsection 2.7.1), fuel swelling rate, densification, etc. Discuss how local effects such as pellet hour glassing and radial cracking are considered in the evaluation.

RESPONSE:

The model used in the evaluation of the 1% plastic strain limit is described in detail in Reference 1. Dimensions used in conjunction with this evaluation are the most limiting combination of tolerances. A discussion of the model and associated assumptions has been added to Subsection 2.4.1.1.

REFERENCES:

1. "General Electric Standard Safety Analysis Report", Proprietary Supplement to Amendment 14, Docket No. STN-50-447, May 1974.

REQUEST: Subsection 2.4.1.2 Safety Evaluation Results

See comments on Subsection 2.4.1. Discuss the application of the tabular LHGR's vs Exposure to the transient analysis results appearing in the reload supplement to the applicable transients discussed in Section 5. Provide in Table 2-3 LHGR limits for the 7x7 and 8x8 fuel designs. Discuss the differences between the fuel types. Discuss the reasons for the burnup effect.

RESPONSE:

Differences between beginning of life and end of life are primarily due to changes in: [[

]]

The values calculated as resulting in 1% plastic strain in the cladding are used during specific plant evaluations of transients due to single operator error or equipment malfunctions to ensure that the safety limit is not exceeded. (See Section 5.)

This discussion has been added to Subsection 2.4.1.2. In Table 2-3, the title has been changed to indicate that the results are applicable to both the 8x8 and 8x8R fuel designs. Results for 7x7 fuel are given is noted in the response to the request on Subsection 2.1.

REQUEST: Subsection 2.4.2 – Design Evaluations

Discuss the purpose of the "Design Evaluations" as compared to the "Safety Evaluation" appearing in the previous section. Provide a substantially expanded summary description of the integral design model. Give a reference for the code. Provide a comparison of the models and assumptions used in the Safety Evaluation and the Design Evaluations. Discuss the conservatisms in the assumptions models and methods (e.g., operating conditions, consideration of tolerances, etc). Explain why the resulting design evaluations are considered generic (bounding for all plants appearing in Table 1-1). Discuss the effects of varying plant to plant operating conditions. List in tabular form which design analyses will be reviewed on a plant specific reload application and which will not. Give the basis for omitting any design evaluations on a plant specific review. Provide tables which list each of the key fuel and plant operating parameter values used in each of the generic design evaluations.

Provide justification for the values selected relative to plant specific conditions for the plants listed in Table 1-1. Provide a complete list of the parameters which will be reviewed for each specific reload to assure that the generic evaluations are applicable. State the acceptance criteria for each of the parameters. What procedure will be followed if a particular reload application is not bounded by the generic evaluation? State what information will be provided in the reload submittal in these circumstances. Provide a general comparison of the past and projected thermal performance improvement among the 7x7, 8x8 and 8x8R fuel designs during normal operation.

RESPONSE:

Safety evaluations are performed and measured against established safety criteria. The consequence of calculating values which exceed such criteria is that fuel failure must be assumed to occur. For plant normal and abnormal operation, this is not permissible. There are two such established safety criteria for the fuel: (1) the 1% plastic strain safety limit and (2) the fuel cladding integrity safety limit.

Design evaluations are also performed and measured against established criteria. These nonsafety related evaluations are included in the report in response to previous specific questions from the NRC. The criteria are not intended to predict failure of the fuel if they are exceeded, but are considered prudent design practice. These design evaluations are included in Subsection 2.5. Thermal analyses performed to provide input into these design analyses are given in Subsection 2.4.2.

The models used for fuel rod thermal analyses have been described in References 1 and 2. The latter model is used only for the evaluation of fuel rod initial conditions at the initiation of a loss of coolant accident.

Design and safety evaluations performed using the model described in Reference 1 assume the most limiting combination of tolerances for all critical dimensions. Assumptions used in conjunction with the model described in Reference 2 are stated in the reference.

The above information has been included in Subsection 2.4.

Conditions which vary from one design to another, or from plant to plant, were taken into account in design evaluations. All dimensional, enrichment, or gadolinia variations among the designs were either analyzed separately or the most limiting case was analyzed. Operating conditions which vary from plant to plant are parameters such as core pressure and the expected maximum power vs exposure for the peak duty fuel rod. In the generic analysis, the highest core operating pressure was used which results in the highest coolant temperatures. With respect to power vs exposure, the limiting

fuel rod is assumed to operate at its maximum permitted power over its entire lifetime, based on thermal mechanical limitations such as the stress limits presented in Table 2-6.

The power level and exposure range considered in the generic analyses, therefore, provides a basis for review of these same parameters on individual projects. For each reload fuel project, the following evaluations are performed to ensure the applicability of the generic analyses:

1. The performance of the reload fuel batch and other fuel already present in the core is projected for the specific plant.

2. The combinations of power, exposure, and residence time from this projected performance are compared with those used in the respective generic analyses. The application is acceptable only if the power levels and exposures analyzed in the generic analyses equal or exceed those protected for the project.

In addition, other plant dependent operating conditions such as core pressure are reviewed and compared with those used in the generic analyses to ensure that limiting conditions have not changed. Adherence to these procedures ensures the applicability of the generic design to individual reload fuel protects.

The above application procedure has been combined with the information in Subsection 2.4.3 and relocated in a new Subsection 2.7. The previous Subsection 2.7 now becomes Subsection 2.8.

A recent report on the performance of General Electric BWR fuel can be found in Reference 3. This reference has been included in Subsection 2.8.

REFERENCES:

1. “General Electric Standard Safety Analysis Report”, Proprietary Supplement to Amendment 14, Docket No. STN-50-447, May 1974.

2. “GEGAP III, A Model for the Prediction of Pellet Cladding Thermal Conductance in BWR Fuel Rods”, NEDC-20181, Revision 1, November 1973.

3. R.B. Elkins, “Experience with BWR Fuel through December 1976”, NEDE-21660P (Proprietary) and NEDO-21660, July 1977.

REQUEST: Subsection 2.4.2.1.1 – Power Spiking

Discuss the application of densification power spiking to Design and Safety Evaluation considerations (See Attachment 2).

RESPONSE:

See the response to the request on Attachment 2, Subsection 2.4.2.1.1.

REQUEST: Subsection 2.4.2.1.3 – Cladding Creep Collapse

It appears that the reference to Subsection 2.5.3.1.2 should be to Subsection 2.5.3.1.1. Make the correction. Reference 2-16 in (cross referenced) Subsection 2.5.3.1.1 refers to a generic pressure increase of 160 psi due to pressurization events, which is subsequently utilized for the collapse evaluation. Many turbine trip without bypass events previously analyzed for reloads report more than a 160 psi pressure increase. Justify the use of this magnitude of pressure increase for generic application to all reload cores. Provide a figure which gives the assumed DP and fast flux over the fuel lifetime for the collapse analysis. Discuss the treatment of the initial ovality, diameter and wall thickness tolerances used in the collapse analysis. Discuss the effects of a reactor vessel cold hydrostatic test on clad buckling potential. State the design criteria for fuel clad collapse during normal and abnormal operating conditions.

RESPONSE:

Analyses of the majority of abnormal operational transients will result in a pressure increase of 160 psi or less. There are a few infrequent events such as turbine trip without bypass or generator load rejection without bypass which may result in pressure increases greater than 160 psi on some operating plants. Studies indicate that the fuel rod is capable of withstanding transient differential pressure increases above rated conditions in excess of 250 psi without experiencing instantaneous collapse or subsequent creep collapse.

The model used for cladding creep collapse analysis utilizes maximum initial ovality, minimum wall thickness and nominal average internal diameter. The design criterion for fuel cladding collapse during normal and abnormal operating conditions is that collapse is not permitted to occur.

Cold hydrostatic test pressure is 1045 psia, which is below the 1065 psia design pressure.

REQUEST: Subsection 2.4.2.1.4 – Stored Energy

This section consists of one sentence and is inadequate. Provide a brief summary discussion in this section of the effect of densification on stored energy for LOCA analysis. (Subsection 5 5, relating to LOCA is Subsection 5.5.2 and had not yet been provided.) Provide or reference a discussion of the effect of the increase in stored energy on the consequences of a control rod drop accident from hot full power. Discuss the radial contraction model (for predicting stored energy) for the control rod drop accident form hot full power.

RESPONSE:

The effects of densification on stored energy are considered in the LOCA evaluation. Stored energy in the fuel pellet at the initiation of the LOCA is calculated using the model and assumptions described in Reference 1. Analysis of the LOCA is presented in Subsection 5.5.

Section 3.3 of Reference 5-17 discusses the control rod drop accident in the power range. Reference 1 in the response to the request on Subsection 5.5.1 describes the rod drop accident at rated power. Both of these analyses show the rod drop in the power range to be much less severe than the rod drop in the startup range.

REFERENCES:

1. “GEGAP-III, A Model for the Prediction of Pellet Cladding Thermal Conductance in BWR Fuel Rods”, NEDC-20181, Revision 1, November 1973.

REQUEST: Subsection 2.4.2.2 – Fuel Cladding Temperature

Provide the basis (e.g., crud and oxide resistance equations) for the results shown in the figures. State the design criteria for the cladding temperature during normal operation.

RESPONSE:

A direct limit on fuel cladding temperature is not used in design evaluations. However, the impact of high cladding temperature, such as decreased yield strength and reduced cladding thickness due to oxidation, is considered in the design evaluation. [[

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The model used to calculate the fuel cladding temperature is documented in Reference 1. This reference has been included into Subsection 2.4.2.2.

REFERENCES:

1. “General Electric Standard Safety Analysis Report”, Proprietary Supplement to Amendment 14, Docket No. STN-50-447, May 1974.

REQUEST: Subsection 2.4.2.3 – Fission Gas Release

State the design criteria for internal gas pressure. Discuss/reference the analysis procedure (models, methods, assumptions) used to calculate the internal gas temperature and pressure. Discuss the consideration of stable isotopes and high burnup dependence. State the limiting rod type (fuel length) considered in the calculation. Compare this to Hatch Reload 2. Discuss the peaking factors and axial

profile used in the analysis. Discuss the effect of radial pellet densification on calculated temperatures, gas release, volumes and pressures. State the maximum calculated gap pressure at end of life for normal and transient conditions.

RESPONSE:

No direct criteria are employed for the internal gas pressure. (See response to request on Subsection 2.4.) The internal pressure is used in conjunction with other loads on the fuel rod cladding when calculating cladding stresses and comparing these stresses to the design criteria. Details of this evaluation are described in Subsection 2.5.3.1.2.

The fuel rod internal pressure is calculated using the perfect gas law and the assumptions detailed in Subsection 2.4.2.3. [[

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REFERENCES:

1. *General Electric Standard Safety Analysis Report, Proprietary Supplement to Amendment 14*, Docket No. STN-50-447, May 1974.

REQUEST: Subsection 2.4.2.4 – Fuel and Cladding Expansion

What is meant by the “cladding average temperature” (radial, axial or volumetric)? What is meant by the average fuel temperature? Give the equation and reference for the fuel thermal expansion coefficient vs. temperature.

RESPONSE:

The cladding average temperature is calculated as the sum of inside and outside surface temperature divided by two at an axial elevation. Based on this cladding average temperature, the cladding axial thermal expansions for the separate axial nodes are summed to determine the total fuel rod cladding axial thermal expansion.

The average fuel temperature in this context is the volumetric average temperature of the fuel pellet cross section at an axial elevation. Again, this calculation is repeated for several axial nodes and the fuel thermal expansions are summed.

The above information has been included into Subsection 2.4.2.4.

The expression used for thermal expansion coefficient of the fuel is provided in Subsection 2.3.2.

REQUEST: Subsection 2.4.2.5 – Incipient Center Melting

Relate the significance of incipient center melting to the design evaluation. Explicitly state a design criteria (e.g., no center melting during normal operation). Relate the calculated results to the criteria. Discuss (or reference previous descriptions of) the code used to calculate the fuel temperatures. The reference to fuel material melting temperature should be to Subsections 2.3.2. and 2.3.3.

RESPONSE:

The model used to calculate fuel temperature is described in Reference l. The fuel is designed so that fuel melting is not expected to occur during normal steady state full power operation. The above information has been incorporated into Subsection 2.4.2.5.

REFERENCES:

1. “General Electric Standard Safety Analysis Report:, Proprietary Supplement to Amendment 14”, Docket No. STN-50-447, May 1974.

REQUEST: Subsection 2.4.2.6 – Pellet to Cladding Radial Differential Expansion

This section consists of a single sentence and is inadequate. State the purpose or criteria used in this section relative to thermal design (evaluation) considerations. Discuss the differences in the code modeling and analytical methods between the thermal (fuel temperature) evaluations and the mechanical (fuel stress and strain) evaluations. Relate the purpose of this evaluation to fuel rod thermal analyses and the fuel rod mechanical (stress/strain) analysis. Discuss how fuel rod tolerances (e.g., wall thickness) are considered in the design evaluations. Discuss how pellet hour glassing and radial cracking is accounted for in the fuel rod thermal analysis. Discuss the treatment of these

phenomena relative to the 1% plastic strain, fatigue and stress rupture limits. Discuss the handling of radial fuel densification effects. Discuss the effect of gap size changes, due to power/exposure effects, on the value of the gap conductance and resulting fuel temperature and pellet expansion.

RESPONSE:

The purpose of the evaluation described in this subsection is to provide inputs to the mechanical evaluation discussed in Subsection 2.5.3.1.2. No direct criteria are applied to the thermal evaluations. Indirectly, the resulting loads, combined with other loads, must not exceed stress criteria when the mechanical evaluations are performed (see also responses to requests on Subsections 2.4, 2.4.2.2, and 2.4.2.3).

A discussion of the models used in fuel rod thermal analyses has been added to Subsection 2.4 in response to request on Subsection 2.4.2. The same response also discusses assumptions made for the evaluations and the referenced model description discusses the details of the model. This same model and assumptions are used for analysis of 1% plastic strain and for inputs to fatigue analysis (see also responses to requests on Subsections 2.4.1.1 and 2.4.2. Since the present fuel rod analysis methods do not account explicitly for the effect of pellet hourglassing, pellet cracking, or creep relaxation of clad stresses, stress rupture is not explicitly addressed for the fuel rod. This approach is supported by experience in that large cladding plastic strains indicative of creep rupture have not been observed. The handling of radial fuel densification effects is described in responses to requests on Subsections 2.4.1.1, 2.4.2, and 2.4.2.1.4.

REQUEST: Subsection 2.4.3 – Plant Reload Parameters

See comments relating to Subsection 2.4.2.

RESPONSE:

See response to Request on Subsection 2.4.2.

REQUEST:

Subsection 2.5.1 – Analytical Criteria for Assurance of Mechanical Integrity

Identify the critical instability loads considered. State the references and/or test data which support them. Reference Topicals and/or tests used to support the adequacy of the 0.060 inch deflection limit. Your recent tests and references that indicate a lower limit should be provided. Provide the basis for the stress limits described in this section and presented in Table 2-6. Reference previous submittals on these specific limits and/or provide a direct comparison with the ASME Boiler and Pressure Vessel Code requirements. Where differences exist between these sources and the limits stated in the Topical, discuss the difference. Specify, if applicable, stress indices used for normal, upset, emergency and faulted conditions. An expansion of Tables 2-5 and 2-6 to the extent of the analogous diagrams appearing in the ASME Boiler Code Sections would be adequate for documenting the off normal reactor conditions. Define and clarify what deflection limits could result in more serious

consequences, as stated in paragraph 5 of Subsection 2.5.1. Describe these consequences and reference any supporting data.

Briefly describe and reference the scaling laws related to the dynamic similitude instability loads. Provide brief discussions and references of the comparison between analyses and loading test results. Discuss the "goodness-of-fit" between the results predicted by the codes used in the analysis and measured test data. Describe which components are subject to shakedown phenomena and how shakedown is handled in the stress criteria. Compare your methods of shakedown analysis with those of ASME Boiler and Pressure Vessel Code. Describe how the combined seismic/LOCA analysis is handled on a generic basis. If both are performed on a "plant specific" analysis, clearly specify this. If a bounding analysis, or "lead plant", approach is to be taken, provide and justify each of the conservatisms used in the analysis.

RESPONSE:

There are two instability (buckling) loads considered: cladding creep collapse and seismic. These loads are evaluated in Subsections 2.5.3.1.1 and 2.5.4, respectively. Both of these subsections reference topical reports for details of the analyses. The seismic LOCA evaluation given in Subsection 2.5.4 is considered to be a bounding evaluation (see the response to the request on Subsection 2.5.4).

A discussion of the tests performed to support the deflection limit is given in References 1 through 3. These references have been included in Subsection 2.5.1. The statement on deflection limits was meant to imply that the stated limits precluded serious consequences and has been clarified.

Stress limits were established as the material yield strength for normal and abnormal events and its ultimate strength for emergency and faulted events. The yield strength was used as the limit for normal and abnormal events so that the reactor may restart after experiencing these events. Use of the ultimate strength limit for emergency and faulted events ensures that the material will not fracture during the single application of loads resulting from these events. For additional information, see the response to the Request on Subsection 2.5.1.1.

With respect to shakedown, because unirradiated Zircaloy exhibits cyclic strain hardening and fully reversed load induced stress amplitudes are less than the monotonic unirradiated material yield strength, the strain response behavior of the material due to cyclic effects remains linear.

REFERENCES:

1. "BWR/6 Fuel Design Amendment No. 1," NEDE-20948-1P (Proprietary) and NEDO-20948-1, November 1976.

2. "BWR/4 and BWR/5 Fuel Design Amendment 1," NEDE-20944-1P (Proprietary) and NEDO-20944-1, January 1977.

3. Attachment to Letter MFN 114-77-050, G.G. Sherwood (GE) to Office of Nuclear Reactor Regulation (D.G. Eisenhut), "NRC Questions on Rod Bowing," March 29, l977

REQUEST: Subsection 2.5.1.1 – Stress Limits for Fuel Rod Analysis

See comments on Stress limits in Subsection 2.5.1. The reference to Subsection 2.5.3.1.3 should be to Subsection 2.5.3.1.2. Please correct.

RESPONSE:

GE fuel rods are designed to assure that the stress intensity limits of Table 2-6 are met for normal and abnormal operation. Fuel integrity during accidents is separately addressed in the evaluation of each accident. The use of the maximum shear stress theory for combined stresses and the stress intensity limits of Table 2-6 was adopted by General Electric during the mid 1960's and provides a very conservative design basis. Essentially all of the GE BWR fuel currently in operation was designed in compliance with these criteria. The resultant fuel rod designs have exhibited satisfactory dimensional stability, with no significant dimensional changes having been observed. Documentation of these limits was previously provided in References 1 through 3. The reference to 2.5.3.1.3 was corrected to 2.5.3.1.2 in Amendment 1.

REFERENCES:

1. "BWR/4 and BWR/5 Fuel Design", NEDE-20944 P (Proprietary) and NEDO-20944, October 1976.

2. "General Electric Boiling Water Reactor Generic Reload Application for 8x8 Fuel," NEDE-20360-1P, Revision 4 (Proprietary) and NEDO-20360, Revision 1, Supplement 4, March 1976 and November 1974.

3. "General Electric Standard Safety Analysis Report," NEDO-10741, April 1973.

REQUEST: Subsection 2.5.3 – Fuel Assembly Normalized and Transient Load Evaluations

The heading for this section is misleading relative to what is presented. Provide a concise table which indicates the loads or load combinations assumed to be applied to each of the fuel assembly components analyzed for each event category (i.e., normal operation, anticipated transients and accidents). Discuss the conditions considered, and cross reference these conditions with the functional requirements. Provide an additional table which indicates the limiting design parameters and each of the functional requirements.

RESPONSE:

This subsection evaluates the effect of normal and abnormal loads on the fuel assembly (functional requirements a and b). Loadings on the fuel assembly during handling are documented in Reference 2-14 of the report. Loadings applied to the fuel rods are given in Subsection 2.5.3.1.2. These loads result in the stresses given in Table 2.5.3-1. Stress combinations for each region of cladding and stress category are given in Table 2.5.3-2. This information has been included in Subsection 2.5.3.1.2. Loadings on other components are given in the appropriate subsection.

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REQUEST: Subsection 2.5.3.1.1 – Cladding Creep Collapse

See comments on Subsection 2.4.2.1.3. Consolidate this subsection with Subsection 2.4.2.1.3.

RESPONSE:

Cladding creep collapse is a densification consideration and is, as such included, briefly, in Subsection 2.4.2.1 with other densification considerations. The analysis itself, however, is a mechanical analysis which is properly presented in Subsection 2.5.3.1.1.

REQUEST: Subsection 2.5.3.1.2 – Stress Evaluation

Define "Stress Category" appearing in paragraph five.

RESPONSE:

The Stress categories are given in Table 2-6. This reference has been included in Subsection 2.5.3.1.2.

REQUEST: Subsection 2.5.3.1.4 – Fatigue Evaluation

The reference to Subsection 2.5.3.1.3 should be to Subsection 2.5.3.1.2. Compare the fatigue damage calculated for the 8x8R fuel design to the 8x8 and 7x7 fuel designs. Reference the model used for the analysis. Justify the K f(= 2) factor used for the fatigue cycling evaluation. Provide the source of the value selected. Relate the analyses performed in this section to PCI fuel failures. Relate the PCIOMR's utilized by a plant to any of the assumptions or inputs in the fatigue evaluation. Relate the mechanical design (analysis) evaluation to the PCIOMR's to be recommended for the 8x8R fuel design. Discuss the preconditioning program which will be recommended for the 8x8R fuel design. Compare these recommendations to those for the 7x7 and 8x8 fuel designs.

RESPONSE:

The intersection of a fuel rod tube and an end plug at the weld is a circumferential notch.
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The above information has been included in Subsection 2.5.3.1.4. Reference 2.5.3.1.3 was corrected to 2.5.3.1.2 in Amendment 1.

Preconditioning interim operating management recommendations (PCIOMR's) are recommendations made by General Electric (GE) which have been demonstrated to improve fuel performance during normal operation. Because they are operational recommendations (not requirements), no assumption

regarding their application is made in any design application. Thus, PCIOMR's are not related to any mechanical design evaluations.

With regards to pellet cladding interaction (PCI) failures, Reference 3 documents the bases for GE's conclusion that there is no significant safety concern relating to PCI failures. PCI failures are, therefore, considered an operational inconvenience to be avoided to the maximum extent practical, since they can result in plant limitations and increased maintenance problems. Thus, many plants follow PCIOMR's to reduce the risk of PCI failures.

REFERENCES:

1. D.H. Winne and B.M. Wundt, "Application of the Griffith Irwin Theory of Crack Propagation to the Bursting Behavior of Disks, Including Analytical and Experimental Studies", Transactions of the ASME, Vol. 80, November 1958.

2. W.J. O'Donnell and C.M. Purdy, "The Fatigue Strength of Members Containing Cracks", Journal of Engineering for Industry, May 1964.

3. Letter, G.G. Sherwood to Victor Stello, Jr., "Information Concerning Feedwater Nozzles and Pellet Clad Interaction", November 10, 1976.

REQUEST: Subsection 2.5.3.1.6 – Flow Induced Vibrations

Include in the discussion the impact of the new fuel design on the vibration of the instrument tubes and the resulting potential for channel box wear. Provide a general description of the problem and the "fix" taken relative to the fuel assembly lower tieplate flow holes. Reference and discuss operating experience which supports the adequacy of the fix. Cross reference Section 4.0 for analytical modeling and analysis of the fix.

RESPONSE:

Significant channel wear due to in core vibration was experienced in boiling water reactor (BWR's) which incorporated bypass flow holes in the core support plate. A description of the cause of the wear, the development of a plant modification to eliminate significant levels of in core vibration and the testing performed to demonstrate the efficacy of the modifications are given in Reference 1. The modifications consisted of eliminating the bypass flow path in the core support plate and creating an alternate flow path by introducing two holes in the fuel assembly lower tieplates. Data on plants operating with the modifications implemented are given in Reference 2.

All 8x8R fuel for BWR/4 plants incorporate finger springs in the channel to lower tieplate flow path and two alternate flow path holes in the lower tieplate. BWR/2 and 3 plants may incorporate finger springs and alternate flow path holes for 8x8R fuel. The hydraulic characteristics of 8x8R fuel are very similar to 8x8 fuel. They are within the testing and operation experience base in References 1 and 2. Some fuel assemblies may be fabricated with no finger spring or alternate flow path holes. Plant operation with this configuration and fuel similar to 8x8R is documented in Reference 1. Therefore, the introduction of 8x8R does not introduce any new features which could induce significant in core vibrations.

REFERENCES:

1. "Supplemental Information for Plant Modifications to Eliminate Significant In Core Vibrations" (NEDE-21156).

2. Letter, R.E. Engel (GE) to Robert L. Baer (NRC), "Results of Channel Wear Inspection", October 18, 1977.

REQUEST: Subsection 2.5.4 – Combined LOCA and Seismic Loads

Define, describe and give the magnitude of each of the loads applied to the 8x8R reload fuel assemblies. Discuss the conservatism of the generic load values of the reactors listed in Table 1-1. Provide a reference for the models methods and assumptions used to calculate the loads.

RESPONSE:

As discussed in Subsection 2.5.4, the BWR/6 LOCA and seismic evaluation documented in Reference 2-5 is conservative for operating plants. The magnitude of peak horizontal acceleration used in the BWR/6 analysis was 3.897 g. Maximum peak acceleration at operating plants listed in Table 1-1 is 3.0 g. The loads applied to the fuel assembly and models methods and assumptions are all documented in Reference 2-5.

REQUEST: Subsection 2.7.1 – Fuel Manufacturing

Discuss the general procedures and acceptance criteria used to evaluate fuel assemblies, subassemblies, components, parts and materials which deviate from the applicable manufacturing drawings or specifications. Discuss the consideration of geometrical tolerances within the fuel rod design and safety evaluations (e.g., kW/ft for 1% strain analyses).

RESPONSE:

The general procedures and acceptance criteria used to evaluate fuel assembly, subassembly, component parts and materials which deviate from applicable manufacturing drawings and specifications are defined in Reference 1. Methods defined in this document have previously been accepted by the NRC (Reference 2). This information has been incorporated in Subsection 2.8.1 (previously 2.7.1).

For a discussion of the consideration of geometrical tolerances within the fuel rod design and safety evaluations, see the responses to the requests on Subsections 2.4.1.1 and 2.4.2.

REFERENCES:

1. "Nuclear Energy Divisions BWR Quality Assurance Program Description", NEDO-11209-03A, pg. 69, November 1976.

2. Letter, C.J. Heltemes (NRC) to J.F. Quirk (GE), "NRC Acceptance of General Electric QA Topical Report", October 27, 1976.

REQUEST: Subsection 2.7.2 – Enrichment Control Program

Discuss or reference the analytical methods which are used to evaluate the effects of enrichment manufacturing tolerances on each of the power peaking factors. Tabulate these effects for each fuel type.

RESPONSE:

The effect of the enrichment tolerance on local power peaking factors has been investigated for several types of fuel bundles by intercomparison studies between calculated values and experimental measurements documented in Reference 3-4. The standard analytical method to evaluate the effect of enrichment tolerance on the local power peaking factor is the two dimensional lattice physics code (Reference 3-1). It has been found that the standard deviation of the manufactured fuel pellets is about 0.015 wt % U-235. The corresponding effect on the local power peaking factor is given in Reference 5-1 as 0.7%.

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This response has been incorporated into Subsection 3.2.2.

REQUEST: Subsection 2.7.4 – Surveillance Inspection and Testing of Individual Fuel Rods

Discuss the channel change out criteria and inspection procedures.

RESPONSE:

Channel inspection procedures and deformation limits are detailed in Reference 2-1. This reference, which provides all information on fuel channels, is given in Subsection 2.1.

REQUEST: Table 2-1

Include grid to grid spacing.

RESPONSE:

Spacer Pitch (Grid to grid) spacing for BWR/2,3 is 19.55 inches; for BWR/4, this spacing is 20.15 inches. This information has been included in Table 2-1.

REQUEST:

Table No. 2-8

Provide references for each of the formulas presented.

RESPONSE:

Stress	Reference
Membrane	Simple Force Balance
Vibration	1 and 2
Discontinuity Membrane	Simple Force Balance
Initial Ovality	3
Spacer Contact	4
Discontinuity Bending	5
Thermal Mismatch	5
Radial Thermal Gradient	2
Thermal Bow	2
End Plug Angularity	2
Pellet Cladding Interaction	5

These references have been included in Table 2-8.

REFERENCES:

1. E.P. Quinn, “Vibration of Fuel Rods in Parallel Flow”, GEAP-4059, July 1962.

2. R.J. Roark, “Formulas for Stress and Strain”, Fourth Edition, McGraw-Hill Book Company, 1965.

3. S. Timoshenko, “Strength of Materials, Part II”, Third Edition, D. Van Nostrand Company, 1956.

4. R.J. Roark, “Stresses and Deflections in Thin Shells and Curved Plates Due to Concentrated and Variously Distributed Loading,” NACA-TN-806, May 1941.

5. S. Timoshenko, and Woinowsky-Krieger, S., “Theory of Plates and Shell”, Second Edition, McGraw-Hill Book Company, 1959.

REQUEST: Table 2-10

Footnote the procedure used to obtain the design ratios.

RESPONSE:

The method of determining the design ratio is given in Subsection 2.5.3.1.2 in which Table 2-10 is referenced. A separate footnote for Table 2-10 is, therefore, redundant.

REQUEST: Table 2-11

Same as for Table 2-8.

RESPONSE:

Deflection	Reference
End Plug Angularity	1
Tube Induced Vibration	1 and 2
Thermal Bow	1
Tube Bow	Circular Arc
Axial Load Non tie Fuel Rods Tie Rods	 3 3

These references have been included in Table 2-11.

REFERENCES:

1. R.J. Roark, "Formulas for Stress and Strain", Fourth Edition, McGraw-Hill Book Company, 1965.

2. E.P. Quinn, "Vibration of Fuel Rods in Parallel Flow", GEAP-4059, July 1962.

3. S. Timoshenko, and Gere, J.M., "Theory of Elastic Stability", Second Edition, McGraw-Hill Book Company, 1961.

REQUEST: Table 2-13

Identify limiting component and location for this table. Provide the basis for $K_f = 2$ in the table (see Subsection 2.5.1).

RESPONSE:

See the response to the request on Subsection 2.5.3.1.4.

REQUEST: Figure 2-11

Provide the basis for correcting the maximum mean stress.

RESPONSE:

The basis for correcting the maximum mean stress is given in Reference 2-5 as documented in Subsection 2.3.1.

REQUEST: Figure 2-16

Discuss in the text the basis for the step change shown.

RESPONSE:

The temperature vs. time shown in the figure reflects the assumptions that at a location of the fuel rod, the fuel is operating at 13.4 kW ft/plus the calculated power spiking penalty, up to an exposure of 4000 MWd/t. At that point in time, a gap is assumed to form as a result of densification, and the cladding temperature therefore decreases.

This discussion has been added to Subsection 2.5.3.1.1.

REQUEST: Section 3 – Nuclear Evaluation

The nuclear evaluation section, as written, does not mention anywhere in the text, that there are significant differences between the 8x8 and 8x8R fuel designs. Except for one reference on page 3-3, there is nothing which states that 8x8, rather than 7x7, fuel is under discussion. Therefore, in order to reduce the general obscurity of Section 3, add a new subsection, after 3.1, which discusses the significant differences between the 8x8 and 8x8R fuel designs (water rods, natural uranium, rod diameter, and increase in length). The effects on average power density, local peaking, void coefficient, shutdown margin, axial power distribution, net energy production and all other important nuclear effects should be discussed.

RESPONSE:

This section discusses the methods used to perform nuclear evaluations. These methods are generic and not dependent upon fuel type. The generic analytical results presented are for 8x8 and 8x8R fuel. The introduction to this report (Section 1.0) limits the scope of the report to these two fuel types. As discussed in the response to the request on Subsection 2.1, the 8x8R fuel was not introduced because of safety considerations.

REQUEST: Subsection 3.1 – Introduction

For each reload cycle there will be a different set of four bundle arrays, with a Potential for wide differences in exposure among the bundles within an array.

(a) Is a new library of cross sections, lattice reactivities, and relative rod power generated for each cycle?

(b) Are differences in bundle exposure taken into account by performing four bundle GEBLA calculations or are single bundle calculations performed?

(c) Are the libraries of lattice parameters ever employed over a range of exposure such that extrapolations are required? What are the assurances that conservatism is attained?

RESPONSE:

The exposure used in the bundle design process covers the range of expected in core exposure.

The lattice nuclear libraries are generated by the two dimensional finite difference diffusion depletion lattice physics code (Reference 3-1). It is a single lattice infinite medium calculation. The purpose of generating exposure dependent nuclear libraries it to prepare input for the three dimensional BWR simulator code (Reference 3-2). It has been shown (References 3-3 and 3-4) that these lattice nuclear libraries yield good agreements with experimental measurements for different reactors at various cycles and exposures.

In the process of bundle design, the lattices are depleted ("burned") for the exposure range of 0–35 GWd/t. The nuclear libraries of cross sections, isotopic compositions, atom densities, lattice reactivities and relative rod powers are generated as a function of this exposure range. This exposure range is sufficient for most applications. In the few cases where nodal exposure of the bundle may exceed the nuclear library data points, a curve fitted extrapolation is made. These nodes would be low power, low reactivity nodes of the bundle.

Typically, a bundle would stay in core for four cycles. The accumulated bundle exposure would be in the range of 20–30 GWd/t. Accordingly, the lattice nuclear libraries cover the exposure range for all four of these cycles.

REQUEST: Subsection 3.2 – Bundle Nuclear Characteristics

Summarize the experimental data which justify application of the GEBLA lattice physics code to 8x8R fuel with two water rods.

(a) The discussion should include consideration of the data included in the document, Lattice Physics Methods Verification, and any new data which may be available.

(b) Comparisons between experiment and calculation should be provided for lattice reactivity, relative rod powers, and isotope buildup and burnout.

(c) Those instances where four bundle calculations are necessary to provide good agreement between calculations and experiment should be identified.

(d) RMS differences between experiment and calculation are not sufficient for this discussion. The maximum extent to which the calculations underpredict reactivity and rod powers should be included.

The discussions in Section 2, which are referenced here, state only that Gd_2O_3 is uniformly distributed in the UO_2 pellets. Provide more descriptive text on the Gd_2O_3 loading, and in particular, for each fuel type:

(a) Describe any axial variation in the Gd_2O_3 loading.

(b) Describe the pellet loading. Do all pellets contain gadolina or only selected pellets in the stack?

(c) Discuss the effect of granularity, failure to form a true homogeneous solution and inhomogeneities on the nuclear characteristics of the 8x8R fuel.

RESPONSE:

The qualification of the lattice physics code is based on thorough analyses of critical experiments and benchmark calculations using the Monte Carlo method. Listed below are descriptions of major experiments and benchmark calculations for this purpose:

1. **High Conversion Experiment** – This experiment allows for the examination of the lattice physics code reactivity calculation for a wide range of H/U ratios (0.95–4.165). Comparisons of eigenvalues between the design method and experiment are discussed in Reference 3-2.

2. **Gd Critical Experiment –** The assembly contained 8x8 bundles with Gd_2O_3 and B_4C rods of advanced BWR design. The gamma-scan rod fission rate data are compared with lattice physics code calculations in Reference 3-2.

3. **Thermal Critical Assembly** – The critical assembly consisted of 8x8 bundles under voided and unvoided conditions. The lattice physics code evaluated eigenvalue, U-238 Cd ratio, δ^{28} δ^{49} and Dy reactivity are compared to measurements. For detailed comparison, see Reference 3-2.

4. **Jersey Central Gamma Scan Experiment** – The assembly consisted of 16 7x7 array bundles. Experiments were performed with and without control curtains; this allowed testing of the control blade model in the code. In Reference 3-2, rod-to-rod gamma-scan data are compared with calculation.

5. **KRB Gamma Scan Experiment** – The gamma-scan measurement was made for an exposed bundle of the KRB reactor. The bundle was selected such that to minimize uncertainty in void and neutron leakage. Comparisons of rod-to-rod fission rates are discussed in Reference 3-2.

6. **Tsuruga Gamma Scan Experiment** – Rod-to-rod gamma-scan measurements were made for a bundle at different axial elevations, and lattice physics code calculations were performed for each level with varying void and control conditions. Reference 3-2 presents detailed comparisons.

7. **KKM Gamma Scan Experiment** – This experiment is similar to the Tsuruga experiment. Multi-bundle calculations were made to incorporate the radial leakage effect. For detailed comparisons, see Reference 3-2.

8. **Dresden-l Isotopic Measurement** – Isotopic measurement for the Dresden-1 reactor was analyzed to verify the burnup calculation of the lattice physics code. The analyses included burnup calculations for corner rods as well as interior rod. For details, see Reference 3-2.

9. **Yankee-Rowe Isotopic Measurement** – The Yankee-Rowe isotopic data for the first cycle was analyzed using the lattice physics code.

10. **High Temperature Critical Experiment** – Criticality and gamma-scan measurements were performed for 8x8 MO_2 reload fuels. The experiments included two substitution measurements by UO_2 bundle, which has two water rods. A selected number of experiments have been analyzed using the lattice physics code. Tables 3.2-1 and 3.2-2 compare rod-to-rod fission rates for bundles with two water rods. The leakage effect was small in these central test bundles and the agreement between the two data is generally good, including rods near the water rods. This fuel contained gadolinia rods which have axially gadolinia zoning. The fuel measured in these experiments has subsequently operated successfully for more than two years in an overseas plant.

11. **Monte Carlo Studies** – Monte Carlo calculations were performed frequently to qualify the code for specific problems. For example, Reference 3-2 contains comparisons between the design method and Monte Carlo calculations for 7x7 and 8x8 bundles. Additional comparisons are shown in Tables 3.2-3 and 3.2-4 for the 8x8 bundles with water rods. The agreement is shown to be generally good as before; there is no observable difference in rod power accuracy between the regular rod and the rod near the water rod.

For the 8x8 reload fuel bundles, there is no axial variation in the Gd_2O_3 loading (i.e., gadolinia distribution is axially uniform). For the 8x8R reload fuel bundles, there are some rods with 6 inches of natural U ends on top and on the bottom of part of these rods. These natural U ends contain no Gd_2O_3 (see description of the lattice 8DRL07l). Within the enriched fuel length (which is 138 in. long for a 150-in. active fuel length and 133.24-in. long for a 145.24-in. active fuel length), there are several Gd_2O_3 rods.

For a reload bundle, there may be three to seven gadolinia rods. In each gadolinia rod, all the pellets within the middle 138 in. or 133.2-in. section contain Gd_2O_3.

[[

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These manufacturing specifications were established based on neutronic and material considerations. The specifications are controlling for the manufacturing of the 8x8R reload fuel. Therefore, the reload

fuel bundle would have up-to-specification pellets. As a result, their neutronic and thermal behavior are acceptable.

Finally, as an auxiliary check of the core composed of numerous as-built fuel pellets, the local power peaking and reactivity have been monitored and compared with calculations by the three-dimensional BWR simulator. The calculations are based on nominal homogenized fuel rods. Intercomparisons between calculations and measurements have shown close agreements (Reference 3-4). These close agreements point to the conclusion that the manufacturing specifications provide an appropriate acceptance criteria.

**3.2-1. Percent Differences in Fission Density
High Temperature Critical A**

$\frac{\text{LATTICE PHYSICS CODE-EXP}}{\text{EXP}} \times 100$	1.061	-2.073		2.218	-1.820		-1.351	-0.285
		-5.534	2.018	2.056	2.146	2.670	-6.641	-0.986
			0.351	0.512	1.585	0.692	2.202	
					H_2O	0.679	2.540	-2.808
				H_2O	-0.541	0.280	1.026	-3.100
			-0.683			0.610	1.190	
							-3.629	-3.027
	0.096							-1.412

3.2-2. Percent Differences in Fission Density
High Temperature Critical B

$\frac{\text{LATTICE PHYSICS CODE-EXP}}{\text{EXP}} \times 100$	3.301	1.967		1.367	0.718		0.215	1.451
		-1.027	-0.212	-1.370	-1.073	0.638	-0.687	1.523
			-0.642	-2.577	-2.767	-0.832	0.0	
					H_2O	-2.301	-0.197	1.856
				H_2O	-2.373	-1.700	-.099	-1.098
		1.667				-1.695	0.967	
							-0.687	-0.851
	4.119							1.452

3.2-3. Percent Differences in Fission Density (8x8, 4Gd, 2 H_2O rods)
C Lattice

$\frac{\text{C-LATTICE PHYSICS CODE}}{\text{LATTICE PHYSICS CODE}} \times 100$

-3.33	0.54	-0.91	-2.33	-3.23	0.74	-1.62	-2.74
	2.98	0.65	2.22	0.84	-0.27	1.84	1.53
		-5.06 Gd	5.07	0.22	-4.45 Gd	-2.24	-1.38
			2.43	H_2O	4.50	0.93	-2.79
				0.82	-0.81	2.53	-4.14
					-5.63 Gd	0.45	-3.05
						1.95	3.25
							3.33

$$\% \text{ Re}\,activity\ Difference\left(\frac{Lattice\ Physics\ Code - MC}{MC}\right) = 0.2$$

3.2-4. Percent Differences in Fission Density
(8x8, 7Gd, 2 H_2O rods)
D Lattice

$\frac{\text{C-LATTICE PHYSICS CODE}}{\text{LATTICE PHYSICS CODE}} \times 100$

-0.43	1.38	-3.08	-2.81	3.10	3.63	2.78	1.84
	-5.38 Gd	-3.25	1.20	2.23	2.03	1.17	1.16
		-4.29 Gd	-2.74	-0.83	-0.35	-1.37 Gd	0.08
			-1.35	H_2O	0.31	-1.25	-1.02
				-3.43 Gd	-0.98	1.44	-2.12
					-4.87	1.66 Gd	-2.74
						-1.90	-2.08
							-4.24

$$\%\ \mathrm{Re}\,activity\ Difference\left(\frac{Lattice\ Physics\ Code - MC}{MC}\right) = 1.1$$

REFERENCES:

1. D.M. Rooney, "Ceramographic technique for revealing inhomogeneity in UO_2 specimens with small additions of selected oxides", NEDO-12024, General Electric Co., Pleasanton, CA (1969).

REQUEST: Subsection 3.2.1 – Reactivity

Provide explanatory text on the dependence of bundle reactivity and reactivity swing to average enrichment, gadolinia loading, void fraction, and exposure. The one sentence reference to Figures 3.1-1 through 3.1-24 is particularly inadequate and should be substantially expanded:

Explain the void history and why the curves cross.

Do the figures include equilibrium xenon? If so, at what flux were the curves calculated?

Some of the even numbered figures in the 3-1 series (e.g., Figures 3.1- 6 and 3.1-10) are not labeled with an exposure. Are all of these k_∞ vs. in channel void fraction plots done at zero exposure? Provide clarification.

RESPONSE:

The following information has been added to Subsection 3.2.1:

3.2.1.1 Factors Affecting Lattice Reactivity

The lattice reactivity is a function of lattice average enrichment, gadolinia loading, void fraction, hydrogen-to-U ratio and exposure. To delineate all these functional dependences, it is necessary to note the following.

For a given lattice, it is observed that:

(a) At zero exposure, the reactivity is highest at 0.0 void fraction (VF), followed by 0.4 VF and has least reactivity at 0.7 VF. This is because enriched U-235 fuel lattice is of higher reactivity for a softer neutron spectrum. The softer neutron spectrum is most abundant in the highly thermalized medium of 0.0 VF.

(b) The term "void history" refers to the fact that the k_∞ curves were obtained for the respective lattices based on a nuclear depletion (exposure) history in the water density of 0.0, 0.4 or 0.7 VF. In other words, assuming a 0.0, 0.4 or 0.7 VF water density for the whole exposure range, the illustrated k_∞ curves were obtained.

(c) [[ ]]

(d) The in situ production of the Pu nuclides increases with increasing void fraction. For example, 0.7 VF is better for Pu production than 0.4 and 0.0 VF. This is because the first has a harder (less thermalized) neutron flux spectrum than the last two. As the lattice attains higher exposures, the U-235 is increasingly depleted and the worth of the bred fissile Pu nuclides becomes more significant. At about 15 GWd/t and thereafter, the fractional fission of the bred Pu nuclides (Pu-239 and Pu-241) in the 0.7 VF case is a few percent greater than those of the 0.0 and 0.4 VF cases. The net result is that the 0.7 VF case is of higher reactivity than the low void fraction cases. This is the reason that the 0.7 k_∞ curves cross over and become of higher reactivity at about GWd/t than the lower void cases.

An exception to these k_∞ behavior and crossover phenomena at high exposure is the natural U lattice. Here, it is observed that, for all exposures, the higher the void fraction, the greater the reactivity. This is due to the fact that the natural U lattice, having no enriched U-235 and gadolinia, is primarily a U-238 system. It is most favored for Pu fissile nuclides generation under a hard neutron flux spectrum; namely, that of the 0.7 VF.

It is also observed that the maximum reactivity for the natural U lattice occurs earlier than the other lattices with enriched U and gadolinia. It peaks at about 2.0 GWd/t for the former and 6.0 GWd/t for the latter. This is because the enriched lattice is designed such that the gadolinia occurs around the end of the first cycle, usually about 6.0 GWd/t.

The primary driving force in fission for the natural U lattices is from the bred fissile Pu-nuclides. Since the Pu production is neutronically favored for a hard spectrum, the high void case always displays a higher reactivity. This reactivity difference increases at the higher exposure range with the increasing fractional fission by Pu-nuclides.

(e) [[

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REQUEST: Subsection 3.2.2 – Local Peaking Factors

This paragraph is very brief and extremely difficult to relate to actual technical specifications. Provide an expanded text which precisely defines all peaking factors (gross, local, radial, axial, total) and how the elevation is chosen, etc. Explain how these peaking factors are related to LHGR, APLHG, TPF, and MCPR, and how these quantities are limited by technical specifications. Describe how the peak local power migrates vertically and from pin to pin during exposure and how this causes “breaks” in Figures 3.2-1 through 3.2-12. Explain why infinite lattice local peaking factors are appropriate.

The maximum credible exposure which could be encountered for any assembly at any axial location within the assembly should be indicated. The possibility that peaking factors have not been calculated over a sufficiently wide range of exposure should be discussed.

The dependence of the maximum local peaking factor, P_L, on the local average void fraction should be included by either expanding Figures 3.2-1 through 3.2-12 to include void fractions ranging from zero to the maximum expected, or by justifying that P_L is not sensitive to void fraction.

Are the curves of local peaking factor in Figures 3.2-1 through 3.2-12 a worst-case (rather than a best estimate) curve as the word "maximum" in the legend implies? If not, what is the calculational uncertainty? Identify which pin is limiting on each portion of each curve.

Provide (explicitly or by reference) local power distribution for each fuel type at zero MWd/t and at a high exposure (e.g., 30,000 MWd/t).

RESPONSE:

The local peaking, gross radial, axial and total peaking factors are design parameters related to reload core analysis. Their respective definitions are shown in Table 3.2.2-1. These peaking factors determine, directly or indirectly, the thermal performance parameters such as maximum linear heat generation rate (MLHGR), maximum average planar linear heat generation rate (MAPLHGR), and minimum critical power ratio (MCPR). The relations between the various peaking factors and the core thermal performance parameters are detailed in Table 3.2.2-2. The peaking factor, by itself, does not constitute a limiting condition. The thermal performance parameters such as MLHGR, MAPLHGR, and MCPR do limit unacceptable combinations of these peaking factors.

For a given lattice at a given void fraction, the maximum local peaking factor will occur at different fuel rods as the exposure increases. This is due to the different depletion and generation rate of the various fissile nuclides in each fuel rod. Previous Figures 3.2-1 through 3.2-12 (now Figure 3-4) show the maximum local peaking factor for the respective lattices at different exposures. These curves are calculated at the typical void fraction of 0.4. [[

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It has been observed that local peaking factors from infinite lattice calculations exhibit close agreement with experimental results (Reference 3-2); hence, their use is appropriate.

The axial power shapes and axial peaking factors are dependent on the fuel bundle types and exposures, the in-core locations, the control rod pattern, the specific reload cycle and the plant. These axial power shapes and axial peaking factors are calculated by the three-dimensional BWR simulators which takes all of the effects into account. Therefore, while it is possible to provide curves showing the variation of the local peaking factor as a function of exposure, it is not possible to do so for the axial peaking factor because it depends on plant unique features and operating characteristics.

The local peaking factors are generated for the exposure range of 0–35 GWd/t. This covers the expected range of exposures for any fuel bundle at any axial location. If a particular location should exceed 35 GWd/t, the power at that location will be sufficiently low that there is no danger of exceeding any performance limits because of its low reactivity at high exposures.

The above information has replaced the documentation previously in Subsection 3.2.2.

The local peaking factor does vary with void fraction, and this dependence is taken into account in the calculations used to assign local peaking factors to each axial segment of the fuel. Figure 3-4 shows the volume at 0.40 voids, as this is the typical average bundle void fraction.

The curves of local peaking factor in Figures 3.2-1 through 3.2-12 are the nominally calculated values. The word “maximum” is used to denote the fact that the maximum pin power at each exposure is used to construct the curves. As stated earlier, these nominally calculated values are appropriate, as they agree well with experimental results.

Figures showing the local pin power distributions as a function of void fraction and exposure for typical fuel lattices are provided in Reference 1.

3.2.2-1. Definitions of Some Peaking Factors

Local Peaking Factor – Local peaking factor is the ratio of the heat flux in the highest powered rod at a given plane to that of the average rod in the plane.
Radial Peaking Factor (RPF) – The ratio of the fuel assembly power in a particular assembly to the power of the average fuel assembly.
Axial Peaking Factor (APF) – The ratio of the heat flux at the axial plane of interest to the heat flux averaged over the active length of the fuel (assembly or rod) of interest.
Gross Peaking Factor (GPF) – The product of the radial peaking factor for a fuel assembly and the maximum axial peaking factor for the same fuel assembly.
Total Peaking Factor (TPF) – The total peaking factor is that peaking factor which, when multiplied by the average linear heat generation rate of a specific bundle type, yields the technical specification limit on MLHGR for that bundle type. This definition of the total peaking factor is presented as an equation in the response to the request on Subsection 5.2.1.5. This response also discusses the relationship of the total peaking factor to the APRM rod block and scram setpoints.

3.2.2-2. Relationship of Technical Specifications and Peaking Factors

Maximum Linear Heat Generation Rate (MLHGR) The MLHGR is the maximum linear heat generation rate (expressed in kW/ft) in any fuel rod allowed by the technical specifications (tech specs) for a given fuel type. The MLHGR is attained when the product of the local, radial and axial peaking factors in an axial segment of a fuel bundle equals the total peaking factor for that fuel type.
Maximum Average Planar Linear Heat Generation Rate (MAPLHGR) The MAPLHGR is the maximum average linear heat generation rate (expressed in kW/ft) in any plane of a fuel bundle allowed by the tech specs for that fuel type. This parameter is obtained by averaging the LHGR over each fuel rod in the plane and its limiting value is selected such that: (a) the peak clad temperature during the design basis loss of coolant accident will not exceed 2200°F, and (b) the LHGR will not exceed the MLHGR in the plane of interest. The MAPLHGR is attained when the product of the gross peaking factor and the average rod peaking factor (1.0) equals the tech spec value.
Minimum Critical Power Ratio (MCPR) The MCPR is the minimum CPR allowed by the tech specs for a given bundle type. The CPR is a function of several parameters, the most important of which are bundle power, bundle flow, and bundle R factor. The R factor is dependent upon the local power distribution, but is only indirectly related to the local peaking factor. The limiting value of CPR is selected for each bundle type such that during the most limiting event of moderate frequency, the CPR in that bundle will not be less than the safety limit CPR. The MCPR is attained when the bundle power, R factor, flow, and other relevant parameters combine to yield the tech spec value. Therefore, MCPR is not directly related to any of the peaking factors described in Table 3.2.2 1.

REFERENCES:

1. "BWR/4 and BWR/5 Fuel Design," NEDE-20944-P (proprietary) and NEDO-20944, October 1976.

REQUEST: Subsection 3.2.3 – Doppler Reactivity

Provide estimates of the uncertainties in the nominal Doppler coefficients related to microscopic cross sections used and resonance capture calculation methods.

Include a discussion of the differences between the nominal or point kinetic doppler coefficient and the Doppler coefficient as used for a 3D transient or accident calculation. In particular, discuss the magnitude of and the basis for any "conservatism" factors which have been applied.

Explain how the Doppler coefficient becomes more negative as exposure and Pu-240 inventory increase and as void fraction increases. Explain further, preferably by including graphical illustrations, how the Doppler coefficient range stabilizes as the equilibrium cycle is approached.

Are the curves in Figures 3.3-1 through 3.3-24 best estimate or worst case values? That "design conservatism factor" is used?

Discuss the upturn in the void coefficient at 4000°F at 0.4 voids for 8DRO.711-G080M fuel at 200 MWd/t (Figure 3.3-9).

RESPONSE:

The subject of the Doppler reactivity coefficient model is described in detail in Reference 3-5 of the report and subsequent responses to NRC questions documented in Reference 3-7. Although no detailed uncertainty evaluation exists, Subsection 2.3.3 of Reference 3-5 discussed the comparison of the Doppler model to experimental data. These comparisons show good agreement between the calculated and experimental results.

Subsections 2.3.2, 2.3.5, and 4.l.2.2 of Reference 3 5 discuss the application of the point kinetic Doppler coefficient to 3D plant transient analyses. The response to Request 17 of Reference 3-7 discusses the application of the Doppler feedback to the continuous rod withdrawal error events and the control rod drop accident.

Subsection 2.3.1 of Reference 3-5 describes the methods used in calculating the Doppler reactivity including Pu-240. Subsequent NRC questions (e.g., Request 1, 2, 8c, etc. of Reference 3-5) and their response have previously addressed the subject of Pu-240 and fuel burnup effects on Doppler coefficients. The sensitivity of the Doppler coefficient to various lattice designs, voids exposure, and control are discussed in Subsection 2.3.4 of Reference 3-5.

Figure 2.3-3 of Reference 3-5 and subsequent figures demonstrate that the Doppler coefficient is not a parameter which varies significantly other than for moderator states and exposure. Therefore, the Doppler coefficient is stabilized and its range of variation will be essentially the same for all reload cores.

The above information has been incorporated into Subsection 3.2.3. The curves in Figures 3-5 and 3-6 are best estimate or nominal. Other than the "design conservatism factor" discussed in Reference 3-5, the nominal values for Doppler coefficients are used in design calculations.

The upturn in the Doppler coefficient in previous Figure 3.3-9 was due to a graphical error. This figure as been replaced.

REQUEST: Subsection 3.2.4 – Void Effect

Do Figures 3.4-1 through 3.4-12 refer to a complete bundle depleted at an average void fraction of 40% or do they refer to one axial height of a bundle, with a void fraction of exactly 40%? Please clarify.

Are Figures 3.4-1 through 3.4-12 best estimate or worst case? What "design conservatism factors" are used?

RESPONSE:

Figure 3-7 refers to the instantaneous void Δk_∞ of the various central lattices for the exposure range shown. Specifically, for a given lattice the change of infinite multiplication factor, Δk_∞, due to the change of the void content (from 40% to 0% and 70%) is plotted as a function of a hot operating BWR fuel bundle. The Δk_∞ shown was computed at hot operating conditions also. It is shown that, in all cases for enriched lattices, a loss of reactivity (i.e., a negative Δk_∞) occurs from increasing the void content from 40% to 70%. On the other hand, a gain of reactivity (a positive Δk_∞) occurs from decreasing the void content from 40% to 0%. This observation confirms the fact that BWR has a negative void coefficient.

The determination of the void coefficient and the design conservatism factors used in this application are discussed in References 3-5 and 3-7.

REQUEST: Subsection 3.3 – Reference Loading Pattern

This section does not discuss control rod patterns and worths. Provide a section which discusses how rod worths are restricted during startup, why the worths are reduced during power operation, and why no restriction on rod worths and pattern symmetry are needed during power operation.

A slight reversal of differential rod worth (i.e., a slight increase in core power following a notch insertion) when a control blade is almost completely withdrawn has been observed in BWRs. As part of the rod worth discussion requested above, provide a paragraph which discusses this phenomenon and the effect of the new fuel design on the power distribution and differential rod worth in the subcooled region at the bottom of the core.

Some parameters become limiting at the end of cycle. How is end of cycle defined (e.g., achievement of a particular exposure, achievement of ARO, coastdown by some percentage of rated power, etc.) and how are EOC limited parameters (e.g., delayed neutron fraction) assured to remain within bounds?

RESPONSE:

Rod worths during startups are restricted to meet the requirements imposed by the control rod drop accident. This is discussed in detail in Subsection 5.5.1 and in References 5-16 through 5-19. Rod worths as a function of moderator density and power level are discussed in the FSARs. Pattern symmetry during power operation is desirable but not necessary. Symmetry greatly simplifies the core limits calculations; however, the core instrumentation is adequate for limits monitoring during asymmetric operation.

The new fuel design is not expected to have a significant effect on reverse power response. The effect of any design change on differential rod worth in any region of the core is inherently accounted for in the design process.

End of cycle (EOC) is defined as the exposure at which the reactor no longer has the reactivity to sustain rated power at the all rods out (ARO) condition. In the case where the cycles include coastdown to a specific power level, EOC is defined as the exposure at which the reactor no longer has reactivity to sustain that specific power at the ARP condition. Operation beyond this exposure requires either: (a) verification that the validity of the operating limits will be maintained, or (b) calculation of a new set of operating limits.

REQUEST: Subsection 3.3.2.1.1 – Core Effective Multiplication and Control System Worth

What is the calculational uncertainty in the three eigenvalues given in the reload supplement table? How much of this uncertainty is due to possible deviation in the exposure at the end of the previous cycle? Do these values represent xenon free (and, for R, equilibrium Sm) conditions?

Describe how the core reactivity peaks as the burnable poison depletes and how the value of R quantifies this effect. Explain why the worth of the strongest control rod at BOC remains appropriate at the point of maximum reactivity.

RESPONSE:

These three eigenvalues (effective multiplication of the core, uncontrolled, fully controlled and with the strongest rod out) are calculated at an exposure corresponding to the minimum expected exposure of the previous cycle. The core is assumed to be in a xenon free condition. This procedure insures that the calculated values are conservative. The value of R includes equilibrium Sm. Further discussion of the uncertainty of these calculations is in Reference 3-4 and Reference l.

As exposure accumulates and burnable poison depletes in the least burned fuel bundles, an increase in core reactivity may occur. The nature of this increase depends on specifics of fuel loading and control state. For example, if one control cell is loaded with two fresh bundles, then the core reactivity calculated with that single control rod removed will probably increase as exposure accumulates. However, the core reactivity calculated with a different rod out may decrease with increased exposure.

Cold k_{eff} is calculated with the strongest control rod out at various exposures through the cycle. At each exposure point a search is made for the single strongest rod, the location of which may change with exposure. The value R is the difference of the strongest rod out k_{eff} at BOC and the maximum

calculated strongest rod out k_{eff} at any exposure point. The strongest rod out k_{eff} at any exposure point is equal to or less than the fully controlled k_{eff} at BOC plus the strong rod worth at BOC plus R.

REFERENCES:

"BWR 4/5/6 Standard Safety Analysis Report Rev. 2," Chapter 4, 1977.

REQUEST: Subsection 3.3.2.1.2 – Reactor Shutdown Margin

List any plant specific effects (e.g., operation with inverted absorber tubes) which may effect the reload evaluation of the shutdown margin.

This section addresses shutdown margin, but scram reactivity is not addressed. Explain and quantitatively justify why the scram curve (negative reactivity vs time) for all possible reloads will be bounded by the negative reactivity insertion rate assumed in the accident and transient analyses. Some reactors may be operated in "coastdown." The effect of operation beyond the ARO exposure point should be discussed.

RESPONSE:

Items which may affect the evaluation of reactor shutdown margin for a given cycle are:

1. an early termination or an extension of the previous cycle, and
2. the ability to insert a control rod to the fully inserted position.

The above information has been incorporated into Subsection 3.3.2.1.2. The topics of negative reactivity insertion rate during transient and accident conditions are addressed in Subsections 5.2 and 5.5, respectively, of the report. Also, as documented in Appendix A, the scram curve used for transient analyses is given for each reload. The scram curve for group notch plants is calculated and compared to the bounding curves for the control rod drop accident. Bank position withdrawal sequence plants are covered by Subsection 5.5.1.3 of Amendment 1.

The deviation of control rod worth due to inverted absorber tubes was examined by generic analyses. Acceptance criteria were established to limit the number of inverted tubes and the control worth deviation. These acceptance criteria have proven to be satisfactory in assuring that there always exist sufficient control rod worth.

Accordingly, the acceptance criteria of the control blade have preempted the inverted tube problem. As a result, a plant-by-plant inverted tube analysis is not necessary and therefore not part of the standard reload analyses.

Plants with shutdown margin adjustment for inverted B_4C tubes are:

Dresden 2

Dresden 3

Quad Cities l

Quad Cities 2

Millstone

Monticello

REQUEST: Subsection 3.3.2.1.3 – Standby Liquid Control System

According to the text, the shutdown margin for the SLCS (to be given in the reload supplement) is calculated at the "minimum control rod position," which is not necessarily all rods out. Explain how the exposure corresponding to the maximum net defect is chosen. If cases exist where this is EOC but not ARO, explain how exposure and/or control fraction will be limited to assure SLCS shutdown capability. Justify the use of "minimum control rod position" rather than the all rods out configuration.

What is the uncertainty in the ppm and shutdown margin figures on page A 3 of the reload supplement? Will the values given in an actual reload be best estimate or will conservatism factors be included?

RESPONSE:

The SLCS must have "the capability of bringing . . . from the minimum control rod position." However, the SLCS shutdown margin is calculated for a fully uncontrolled condition at the most reactive point in the cycle which can never be achieved by an operating reactor. This calculated value conservatively verifies that the system requirement is met.

The plant specific ppm is given in the Plant Supplemental Submittal and is the value in that plant's technical specification basis. The shutdown margin is calculated consistent with the technical specification ppm value.

REQUEST: Subsection 3.3.2.1.4 – Reactivity of Fuel in Storage

Verify that all operating BWR's are equipped with spent fuel storage racks with the two rack spacings given. If storage racks other than types A and B exist explain why this analysis is bounding.

Provide a third column in the table of page 3-7 which gives the exposure corresponding to the maximum k.

RESPONSE:

The two rack spacings apply for two storage racks designed by the General Electric Company. For other storage rack designs, specific information must be provided by the utilities.

The table in Subsection 3.3.2.1.4 has been revised to appear as follows:

Table

Bundle Type	Maximum k_∞	Exposure (GWd/t)
8D250	1.236	5.0
8D262	1.241	5.0
8D274L	1.238	5.0
8D274H	1.216	7.0
8D219L	1.159	0.0
8D219H	1.119	8.0
8DRL303	1.210	9.0
8DRL301	1.228	7.0
8DRL282L	1.239	5.0
8DRL282H	1.218	7.0
8DNL282L	1.226	8.0
8DNL282H	1.212	8.0
8DRL254	1.220	8.0

REQUEST: Subsection 3.3.2.1.5 – Reactivity Coefficients

Although the Doppler and void coefficients dominate reactor behavior during power operation, the moderator temperature coefficient is significant during reactor startup. Discuss the importance of the moderator temperature coefficient during plant heatup. Relate the moderator temperature coefficient(s) calculated in this section to the moderator temperature effect discussed in Section 3.5.3.

Equation 3.3-1 appears to have an incorrect "•" between k_∞^{uc} and (E_i , V). Please correct.

Provide estimates of the uncertainties in the nominal void coefficient related to microscopic cross sections used, resonance capture calculation methods, etc.

Discuss the differences between the nominal or point kinetics void coefficient and the void coefficient as used for a 3D transient or accident calculation. In particular, discuss the magnitude of and the basis for any “conservatism” factors which have been applied.

RESPONSE:

The moderator temperature coefficient is not a significant reactivity coefficient because its effect is limited to primarily the reactor startup range. Once the reactor reaches the power producing range, boiling begins and the moderator temperature remains essentially constant. As with the void coefficient, the moderator temperature coefficient is associated with a change in the moderating power of the water. The temperature coefficient is negative during power operation.

The range of values of moderator temperature coefficients encountered in current BWR lattices does not include any that are significant from the safety point of view. Typically, the temperature coefficient may range from +4 x 10^{-5} Δk/k°F to –14 x 10^{-5} Δk°F, depending on base temperature and core exposure. The small magnitude of this coefficient (relative to that associated with steam voids) and combined with the long time constant associated with transfer of heat from the fuel to the coolant, modes the reactivity contribution of moderator temperature of small importance.

Because of its relative insignificance, current core design criteria do not impose limits on the value of the temperature coefficient. A measure of design control over the temperature coefficient is exercised by applying a design limit to the void coefficient. This constraint implies control over the water to fuel ratio of the lattice; this, in turn, controls the temperature coefficient. Thus, imposing a quantitative limit on the void coefficient effectively limits the temperature coefficient.

The above information has been incorporated into a new Subsection 3.3.2.1.8.

The void coefficient model mused for plant transient analyses is discussed in detail in References 3-5 and 3-7. No detailed evaluation of uncertainties has been made; however, results of extensive sensitivity studies (Subsection 3.4), and comparisons of the point model void coefficients to more detailed three-dimensional void coefficients and one dimensional transient model are presented in Subsections 3.4.7, 3.4.8 and 3.5.2. The allowance for nuclear input uncertainties is discussed in Subsection 4.1.2 of the subject report.

Equation 3.3-1 was corrected in Amendment l.

REQUEST: Subsection 3.3.2.1.7 – Doppler Coefficient

Provide explicit definitions for “E” and “UH” in Equation 3.3 4.

RESPONSE:

Nomenclature for “E” and “UH” were provided in Amendment 1.

REQUEST: Subsection 3.4.2 – Acceptable Deviation From Reference

Include the results of the sensitivity studies performed to indicate how the validity of the licensing analysis is affected by differences between the reference design core and the actual design core.

RESPONSE:

The sensitivity studies identified in Subsection 3.4.2 are continuing. To date, these studies have identified items to be reviewed (Subsection 3.4.2) and the basis of any re examination (Subsection 3.4.3).

REQUEST: Subsection 3.4.2.5 (and 6) – Location of Reload (and Exposed) Bundles

What is meant by "regions of least importance"?

RESPONSE:

When a small change from the reference design core is necessary, the location of one or more fresh (or exposed) bundles may need to be modified also. Under such circumstances, the primary concern is to ensure that the core still meets reference licensing requirements such as cold shutdown margin and transient behavior. It should also approach a core incremental exposure close to the original reference exposure. To this end, the change of these bundle locations is made in "regions of least importance," which is usually in the periphery. This refers to the fact that these regions (bundle locations) do not contribute much in core exposure capabilities and are of low power. Therefore, the modifications of these loading regions do not result in power mismatch problems, shutdown degradations and/or adverse transient behavior.

REQUEST: Subsection 3.4.2.8 – Symmetry

Provide more text which explains how the process computer programs use assumptions of symmetry and how deviations from symmetry in fuel loading and rod patterns are allowed for. What portions of the safety analysis assume core quadrant symmetry?

RESPONSE:

Calculation of the safety limit MCPR by the GETAB analysis assumes core quadrant fuel bundle type symmetry. No such assumption is made in the other areas of safety analysis. It should be noted that the safety limit MCPR was derived for the most conservative power distribution and should also apply for the case of asymmetric reactor power. This is discussed further in Reference 1.

When the reactor core is being operated with a mirror or rotationally symmetric control rod pattern, the neutron flux at similarly symmetric narrow narrow gap locations in the four quadrants is considered to be equal. This fact is used to reflect the readings of the real strings into their symmetric counterpart locations where no real strings exist. This reflection is done prior to the commencement of the power distribution calculations.

In the few incidences where fuel bundles near the edge are quadrant loaded asymmetrically, the error induced by reflecting real readings is partially negated by the fuel type dependent correlations. Any remaining error is considered to be second order. Further, because such bundles are in low power regions, the chance that one of them is a limiting bundle is minimal.

In the rare case of the reactor being operated with an asymmetric control rod pattern, the reflection of real string readings is not utilized. In this instance, readings at locations without strings are inferred by interpolation of the real string values in the immediate vicinity.

The above information replaces the information previously given in Subsection 3.4.2.8. It should be noted that operating GETAB calculations do not require the process computer.

REFERENCES:

1. "Process Computer Performance Evaluation Accuracy Amendment l," NEDO-20340-l, December 1974.

REQUEST: Subsection 3.4.3

Discuss the reasons why the six listed parameters were chosen and why others (e.g., delayed neutron fraction, Doppler coefficient, SLCS effectiveness) can be excluded.

RESPONSE:

The listed parameters were chosen by one of the following two criteria:

1. It is a parameter whose magnitude or behavior is explicitly reported in the reload licensing analysis.

   Examples: Cold Shutdown Margin, Peak Fuel Enthalpy in Rod Drop Accident, Change in CPR Due to a Misloaded Assembly, and Rod Block Monitor Response.

2. It is a parameter important to the quantification of an operating limit.

   Examples: Scram Reactivity Insertion and Dynamic Void Coefficient Affect MCPR limit.

   Other parameters were excluded for the following reasons:

   (a) SLCS effectiveness with no significant change in cold shutdown margin, SLCS effectiveness will not change.

   (b) Doppler Coefficient and Delayed Neutron Fraction – These are slowly varying functions of exposure which do not change significantly over the expected range of exposure deviations.

The above information has been included in Subsection 3.4.3.

REQUEST: Subsection 3.5 – Startup Physics Test Program

The startup physics test program as outlined in this section lacks the necessary depth of discussion. The discussion will require a significant amount of additional detail in order to make clear the acceptability of the methods, procedures and acceptance criteria used for the various tests in the program. Specifically, the following comments are submitted on each subsection:

RESPONSE:

The startup physics program was originally included in the report in response to numerous NRC questions on recent reload submittals. This subsection was intended to represent the basis for a suitable restart test program. However, this program is the responsibility of the plant operator. Therefore, a detailed program is considered outside the scope of this report, and this subsection has been deleted.

REQUEST: Subsection 3.5.1 – Core Verification

Discuss or reference the procedures used to check the location and orientation of each fuel assembly during and after core loading. What changes are made to the plant process computer (settings, constants, coefficients, etc.) whenever the core must be reloaded differently from the reference core (e.g., due to excessive fuel failures) appearing in the reload submittal. Discuss how such differences may impact upon any of the other startup physics tests. Discuss the criteria used to select substitute fuel assemblies for failed or damaged assemblies.

RESPONSE:

See the response to the request on Subsection 3.5.

REQUEST: Subsection 3.5.2 – Cold Shutdown Margin

Discuss in detail the procedures used for the verification of cold shutdown margin with the highest worth control rod withdrawn. Provide the details of the measurement as well as the methods used to verify the Technical Specification requirements for shutdown margin. Discuss the acceptance criteria and the procedures followed if the acceptance criteria are exceeded.

RESPONSE:

See the response to the request on Subsection 3.5.

REQUEST: Subsection 3.5.3 – Moderator Temperature Coefficient

Describe in detail the procedures used for the moderator temperature coefficient measurement. This description should include the acceptance criteria and the procedures to be followed if the acceptance criteria is not met.

RESPONSE:

See the response to the request on Subsection 3.5.

REQUEST: Subsection 3.5.4 – SRM Performance

Describe in detail the procedures and methods used for the SRM Performance Test. This description should include the acceptance criteria and the procedures to be followed if the acceptance criteria is not met.

RESPONSE:

See the response to the request on Subsection 3.5.

REQUEST: Subsection 3.5.5 – Core Power Distribution

A power distribution comparison at a given control rod pattern and power level (>50% rated power with equilibrium xenon) must be done for each cycle. Discuss the details of this measurement and the methods used to compare the results with the predictions. The acceptance criteria and the procedures to be followed if the acceptance criteria are exceeded should also be discussed.

RESPONSE:

See the response to the request on Subsection 3.5.

REQUEST: Subsection 3.5.8 – Control Rod Drive Testing

Describe in detail the procedures used for the drive tests and the scram time tests. This description should include the acceptance criteria for these tests and the procedures to be followed if the acceptance criteria is not met.

RESPONSE:

See the response to the request on Subsection 3.5.

REQUEST: Additional Tests Required

Core Power Symmetry Test

A core power symmetry test above 75% rated power must be done with the other BOC physics tests. Discuss the details of this test and the methods used to analyze the results. The acceptance criteria and the procedures to be followed if the acceptance criteria are exceeded should be discussed.

Critical Eigenvalue Comparison

A critical eigenvalue comparison (nonvoided moderator condition) for a fixed control rod pattern must be done for each cycle. Discuss the details of the measurement and the methods used to compare the results with the prediction. The acceptance criteria and the procedures followed if the acceptance criteria are exceeded should also be discussed.

TIP Reproducibility Test

A TIP reproducibility test must be done at BOC at a power >75% rated power. Discuss the details of this test and how the results will be used to validate the reproducibility assumed in analysis.

RESPONSE:

See the response to the request on Subsection 3.5.

REQUEST: Subsection 3.X – Process Computer

Provide a section which addresses the changes made to the plant's process computer prior to each cycle of operation. The section should describe: (l) what elements of the process computer change from cycle to cycle; coefficients, constants, correlations, etc. and why they change; (2) what codes and methods are used to establish the new values; and (3) what quality assurance procedures (testing) are used at the site to verify that the process computer has been correctly reprogrammed. Discuss the impact of the new 8x8R fuel design relative to (1). Reference the topical report which discusses the process computer.

RESPONSE:

Before discussing the process computer update procedure, it should be emphasized that the process computer hardware is not a safety system and that the process computer software is not a part of the license basis. The process computer system is considered a tool which the reactor operator may use to more efficiently perform the limits monitoring function. The reload licensing process is valid irrespective of the availability of the process computer. Past experience has demonstrated that the BWR can operate for extended periods without the process computer.

During each refueling, the process computer is "updated" to reflect changes made to the core. The bundle identification (ID) array is rearranged in accordance with the shuffled core; ID's of discharged bundles are replaced with ID's of the reload bundles. Simultaneously, exposure and void history arrays are rearranged and the indices which assign bundle types to core locations are shuffled. If a new bundle is introduced, nuclear coefficients (power to TIP coefficients, local peaking factors, MAPLHGR limits, etc.) may be required. If a new mechanical design is introduced, such as 8x8R, new thermal hydraulic coefficients will be required. The "update" is accomplished by generating a data bank in San Jose which is transmitted to the utility. The generation of all data and of the data bank are performed in strict accordance to General Electric Quality Assurance Procedures. This includes verification and sign off of all data and independent checking of the accuracy of the data bank.

REQUEST: Section 4 – Steady State Hydraulic Analyses

Provide a subsection heading for the introductory paragraphs (to be consistent with the format of other sections). Clarify the modeling with a modeling diagram of the fuel assembly features which contribute significantly to the hydraulics within a single bundle at various axial elevations. Provide an additional modeling diagram of the model which will be used to determine the steady state hot (and average) bundle flows for each fuel type in a mixed core loading configuration containing up to three fuel types. Discuss the adequacy of the model in view of the various orificing bundle type combinations which can occur. Support the adequacy of the "coarseness" of the licensing hydraulics model with the results of model sensitivity studies which involved "finer" (additional channels) modeling. Discuss the anticipated fuel assembly type/orifice type patterns which will be employed on reloads. Provide a summary description (and reference for the code and solution methods for the hydraulic analyses. Describe modeling considerations related to other primary system components (e.g., steam separator, jet pump, etc.). Discuss the modeling of the bypass flow through the two water rods in the 8x8R fuel assembly. Describe the applicability of the tests, assumptions and modeling of References 4-4, 4-5 and 4-6 for this application.

RESPONSE:

Fuel Assembly Hydraulic Modeling

The fuel bundle and fuel support assembly consists of three basic regions as illustrated in Figure 4A. Figure 4-1 of the report is a schematic showing the features of the lower nonfuel region. During normal operation, this region is characterized by single phase flow. A pressure drop is calculated across the orifice and entrance to the lower tieplate, which includes friction, an irreversible loss due to area change, and a reversible pressure drop due to an area change. Flow through the lower tieplate holes and through the lower tieplate fuel support paths is calculated and subtracted from the flow through the fuel support orifice. The remaining flow passes through the lower tieplate grid, experiencing additional reversible and irreversible pressure changes. The flow through the channel lower tieplate path is calculated and subtracted from the lower tieplate grid flow. Flow through the water rods is also subtracted, as will be discussed separately.

The fueled region is divided into 24 axial segments or nodes over which the heat flux is assumed constant and coolant thermal properties are assumed to vary linearly. Fluid properties and pressure changes (using Equations 4-l, 4-2, 4-3, 4-7 and 4-8) are calculated across each node.

In the upper nonfueled region, friction and elevation pressure drops are calculated from the top of the active fuel to the upper tieplate. At this point, flow is re introduced from the water rods and the fluid properties are recalculated assuming thermal equilibrium. Reversible and nonreversible pressure changes across the upper tieplate into the nonrodded channel section are evaluated. Next, friction loss in this section is calculated, followed by acceleration into the plenum above the core.

“Hot” and Average Bundle Modeling

The “hot” and average bundle hydraulic modeling is as described above. Basically, a core is hydraulically modeled by as many as 12 different fuel types. A fuel type in this sense is described by orificing, fuel geometry (7x7, 8x8, or 8x8R), relative bundle power and leakage characteristics.

The “hot” bundle types are characterized by higher relative power. Because of power differences, “hot” and average central orifice region fuel types will have higher relative powers than corresponding peripheral region “hot” and average fuel types. A typical distribution for a reload core containing 7x7, 8x8, and 8x8R fuel is shown in Table 4A.

Sensitivity studies demonstrate that there is no need to divide the core power distribution into smaller increments. The core pressure drop, which provides the driving head for establishing the “hot” channel flows, is determined predominantly by the average channels. Thus, a core is adequately modeled by “hot” and average channels.

Orificing – Bundle Type Combinations

As discussed above, the various orifice bundle combinations are represented by the different fuel types used by the hydraulic model. This modeling is adequate to represent the various fuel assembly type/orificing patterns that occur in BWR’s.

Digital Computer Code

Steady state, core hydraulic analyses are performed using a digital computer program which hydraulically simulates BWR cores. The program user must specify the reactor core power level and distribution, inlet flow conditions, core operating pressure, and a hydraulic description of the reactor

fuel assemblies. The program will calculate either the required total flow and flow distribution for a specified core pressure drop or the flow distribution and core pressure drop for a specified total core flow rate.

The hydraulic model allows for up to 12 distinct parallel channels connected to common upper and lower plena. The power level and distribution, geometry, and hydraulic characteristics must be specified for each channel type and may differ in each channel type. A single parallel bypass region is also included in the hydraulic model. The hydraulic model is a constant pressure model, in the sense that all fluid properties are evaluated at a constant, user specified core operating pressure.

The solution procedure consists of a trial and error iteration of flow rates and pressure drops with concurrent calculation of enthalpy, quality and void distributions in individual channel types.

The iterations are usually initiated with physically derived initial guesses and generally involve either simple linear or quadratic interpolation techniques. User defined convergence criteria are employed.

Modeling of Other Primary System Components

Modeling considerations related to other primary system components (e.g., the steam separator, jet pump, etc) are discussed in NEDO-10802, "Analytical Methods of Plant Transient Evaluations for the General Electric Boiling Water Reactor", dated February 1973. In general, however, the other primary system components outside of the core shroud and above the reactor water level are not explicitly modeled in the steady state thermal hydraulic model.

Water Rod Modeling

The flow rate through each water rod in a given fuel assembly is iteratively calculated by balancing the lower to upper tieplate pressure drop in the external active flow region with the internal overall water rod pressure drop. Hydraulic modeling of the water rods includes entrance and exit losses, friction, elevation, and acceleration heating losses. The entrance loss modeling assumes a single phase liquid, while the exit loss formulation includes a two phase multiplier. The default value is the homogeneous multiplier, but it is possible to input other values in a linear with quality formulation, i.e.:

$$\phi^2_{TPL} = 1 + C_1 \cdot X$$

where X is the quality.

A similar two phase multiplier formulation is included in the nodal calculation of the water rod frictional losses.

Energy deposition rates in the water rod coolant are calculated nodally and consider the effects of gamma heating, neutron slowing down, and the combined convective conductive heat transfer rate from the external coolant. The nodal energy balance is used to determine the quality and void distributions required in the hydraulic analysis.

The water rod hydraulic modeling assumes that all of the water rods in a given fuel assembly are geometrically and hydraulically identical. The water rods may be either round or square in cross sectional shape and can be of different outer diameter and/or clad wall thickness than the fueled rods in the same assembly.

Applicability of References 4-4, 4-5 and 4-6

References 4-4, 4-5 and 4-6 provide the models for two phase friction and local pressure drops and void fraction.

The void fraction correlation used is a version of the Zuber Findlay model (Reference 11), where the concentration and void drift coefficient are based on comparison with a large quantity of world wide data (see attached references).

The General Electric Company has taken a significant amount of single- and two-phase pressure drop data in full scale geometries representative of 7x7, 8x8 and 8x8R fuel and correlated the two-phase multipliers on a best fit using the two-phase models reported in References 4-4 and 4-5 in conjunction with the void fraction model. The typical test conditions (Table 4B) are representative of BWR operating ranges. Figure 4A demonstrates the applicability of these models.

Void Fraction Models

1. Isbin, H.S., Rodrrguez, H.A., Larson, H.C., and Pattle, B.D., “Void Fractions in Two Phase Flow”, A.I. Ch.E. Journal, Volume 5, No. 4, 427-432, December 1959.
2. Isbin, H.S., Sher, N.C., Eddy, K.C., “Void Fractions in Two Phase Steam-Water Flow”, A.I. Ch.E. Journal, Volume 3, No. 1, 136-142, March 1957.
3. Marchaterre, J.F., “The Effect of Pressure on Boiling Density in Multiple Rectangular Channels”, ANL-5522, February 1956.
4. Janssen, E., and Kervinen, J.A., “Two Phase Pressure Drop in Staight Pipes and Channels; Water Steam Mixtures at 600 to 1400 psia”, GEAP-4616, May 1964.
5. Cook, W.H., “Boiling Density in Vertical Rectangular Multichannel Section with Natural Circulation”, ANL 5621, November 1956.
6. Mauer, G.W., “A Method of Predicting Steady State Boiling Vapor Fractions in Reactor Coolant Channels”, WAPD-BT-19, June 1960.
7. Rouhani, S.Z., “Void Measurements in the Region of Subcooled and Low Quality Boiling”, Symposium on Two-Phase Flow, University of Exeter, Devon, England, June 1965.
8. Firstenberg, A., and Neal, L.G., “Kinetic Studies of Heterogeneous Water Reactors”, STL 372-38, April 15, 1966.
9. Ferrel, J.K., “A Study of Convection Boiling Inside Channels”, North Carolina State University, Raleigh, N.C., September 30, 1964.
10. Rouhani, S.Z., “Void Measurements in the Region of Subcooled and Low Quality Boiling”, Part II, AE-RTL-788, Akticbolaget, Atomenergi, Studsvik, Sweden, April 1966.
11. Christensen. H., “Power to Void Transfer Functions”, ANL-6385, July 1961.
12. Egen, R.A., Dingee, D.A., Chastain, J.W., “Vapor Formation and Behavior in Boiling Heat Transfer”, BMI-1163, February 1957.

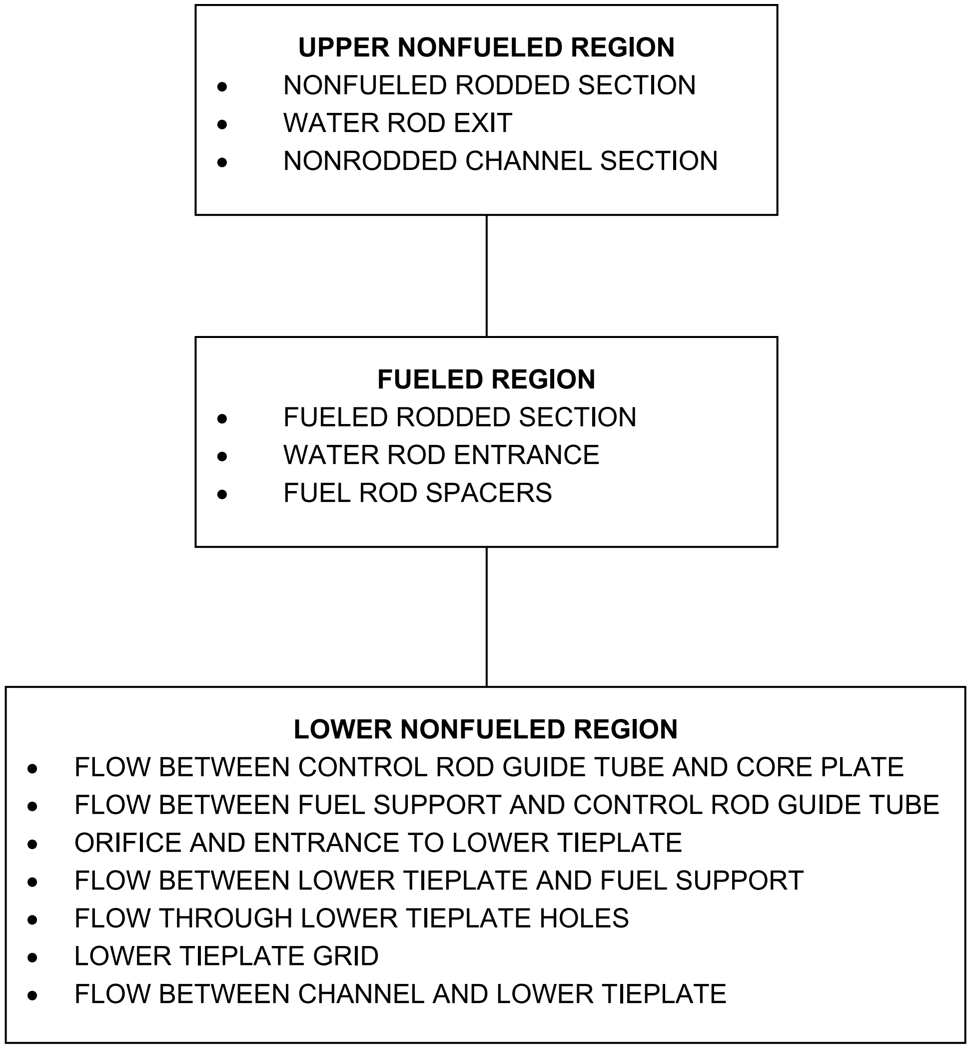

Figure 4A. Fuel Bundle – Support Hydraulic Modeling

4A. Description of a Typical Reload Core

Type	1	2	3	4	5	6	7	8	9
Identification	8x8R	8x8R	8x8R	8x8	8x8	8x8	7x7	7x7	7x7
Orifice Zone	Central	Central	Periph-eral	Central	Central	Periph-eral	Central	Central	Periph-eral
Number Fuel Assemblies	1	263	0	1	299	56	1	107	36
Relative Assembly Power	1.40	1.039	0.700	1.40	1.039	0.700	1.4	1.039	0.700

4B. Typical Range of Test Data

Measured Parameter	Test Conditions
Adiabatic Tests:	
Spacer single-phase loss coefficient	NRe* = 0.5 x 10^5 to 3.5 x 10^5
Lower tieplate + orifice single-phase loss coefficient	T = 100 to 500°F
Upper tieplate single-phase friction factor	P = 800 to 1400 psia
Spacer two-phase loss coefficient	G = 0.5 x 10^6 to 1.5 x 10^6 lb/h-ft^2
Two-phase friction multiplier	X = 0 to 40%
Diabetic Tests:	
Heated bundle pressure drop	P = 800 to 1400 psia
	G = 0.5 x 10^6 to 1.5 x 10^6 lb/h-ft^2
*Reynolds number	

REQUEST: Subsection 4.1

Provide or reference the friction factors used in modeling.

RESPONSE:

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REQUEST: Subsection 4.2

Provide or reference the tests discussed in the section.

RESPONSE:

Figure 4.2-1, attached, is a graphical presentation of the test results showing bundle pressure drop vs. bundle power with mass flux as a parameter. The solid line is GE's analytical prediction method of relating the three components of bundle power, flow, and pressure drop together. The tests were performed in this ATLAS test facility on a 8x8R assembly.

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Figure 4.2-1. Overall Pressure Drop vs. Bundle Power, BWR/2–5 (1,000 psia)

REQUEST: Subsection 4.3

A term of the acceleration of gravity is required for Equation 4-3, if r is in the conventional units of lbm/ft^3.

RESPONSE:

Using the conventional units of lbm/ft^3 for density in Equation 4-3, the acceleration due to gravity is unity and, therefore, was not included in the equation. Equation 4-3 has been modified to read:

$$\Delta P_E = \rho_g L.$$

REQUEST: Subsection 4.5 – Bypass Flow

Reference tests and analyses from which the empirical fits were established.

RESPONSE:

The experimental tests and analyses that have been performed to establish the flow coefficients for the bypass flow paths are described in Subsections 4.2, 5.2 and 5.3 in Reference l. This reference has been added to Subsection 4.5.

REFERENCES:

1. “Supplemental information for Plant Modification to Eliminate Significant In-Core Vibration,” NEDE-21156 (Proprietary), January 1976.

REQUEST: Section 5 – Operating Limits

Provide a section heading for the introductory paragraphs of this section to be consistent with the format of the other sections.

The design Linear Heat Generation Rate and Average Planar Linear Heat Generation Rate vs. exposure are important operating limits which should be addressed in this section for each fuel type. A brief discussion of their relation to reload transient and accident analysis should be included. Their relation to the 1% plastic cladding strain and 2200°F PCT for local transients (e.g., RWE, FLE) and LOCA, respectively, should be addressed. Additionally, gross core thermal power reductions toward EOC, from 100% power, may be required to maintain adequate margin between the lowest safety valve setpoint and peak pressure, from the most severe overpressure event. This potential gross core power (operating) limitation should be briefly discussed in this section.

The design requirement that the peak transient reactor vessel pressure not exceed 110% of the design pressure should also be included in the transient limits paragraph.

This section should also include a brief discussion of the operating procedures which BWR's follow during normal operation, to minimize PCI fuel failures. Reference to a more detailed discussion would also be appropriate.

RESPONSE:

The introductory paragraphs of this section develop the safety analysis philosophy which is used in subsequent sections to establish the limits for specific events. As such, specific limits and event descriptions are not contained.

The discussion of the limits for specific events is contained in the section which describes that event. If specific limits for a given event were described in a general section, they might be misinterpreted and applied to a different category of events. This could result in inappropriate limits being established.

As examples, the limit of 110% of design pressure is the General Electric interpretation of an ASME Code requirement. It is provided in the safety analysis as a demonstration that the Code requirement is satisfied for a given reload.

The event analyzed has been demonstrated to be more severe than any abnormal operational transient. Therefore, it is not included as a limit for transients.

As discussed in the response to the request on Subsection 2.5.3.1.4, General Electric has concluded that pellet cladding interaction failures do not represent a significant safety concern. Therefore, discussion of a recommendation to minimize an operational inconvenience is not appropriate for a section describing limitations placed on a reactor for safety considerations. This is also applicable for the recommended power reduction to maintain margin to the lowest safety valve setpoint.

REQUEST: Subsection 5.1.1 – Statistical Model

Reference and provide the GE report which describes the BWR process computer function.

The design power distribution (histogram), which is an important aspect of the statistical model used to establish the fuel cladding integrity safety limit, does not distinguish between 7x7 (49 rods) and 8x8 or 8x8R (64 rods) fuel types. For a mixed core reload, configurations there will, in general, be assembly wise power histograms for each fuel type. Provide either a description of the modification made to the statistical procedure for this situation or the justification for using a single design power distribution, for calculating the number of rods in the core in boiling transition. Discuss the procedures used to develop the histogram(s).

RESPONSE:

The GETAB fuel cladding integrity safety limit analysis is performed by a Monte Carlo controlled computer code which only represents the process computer functional. Therefore, the topical report on the process computer is not applicable to this calculational procedure. The reference for the calculational procedure and uncertainties is Appendix IV in Reference 5-1.

The current design procedure is to perform a bounding analysis establishing a single valued reload safety limit MCPR which applies to all the reload cycles for a given class of reload cores. As such, the

procedure to develop the design power distribution for the statistical analysis is, first, to search for a reload core loading configuration which would yield the worst CPR distribution in the core among the reload cycles, and, second, to search for the control rod pattern which yields as many fuel assemblies as possible at and near the MCPR operating limit at rated reactor power as per the procedure described in detail in Appendix IV, GETAB Licensing Topical Report NEDO-10958.

In generating the design power distribution, the 8x8 and the 8x8R reload cores are treated as different classes of reload cores because of differences in fuel type and core loading configuration. A 251/764 core 8x8R equilibrium cycle for the 8x8R reload core and a 251/764 core-7x7/8x8 first mixed cycle for the 8x8 reload core are selected for the statistical analysis. This selection was based on the results of evaluations which indicated that a least CPR mismatch between different fuel batches is expected in these cycles, thereby providing a worst CPR distribution among the reload cycles for their respective reload core class.

The term “worst” is used here describing the power distribution in the context of “relative” rather than “absolute” sense. Therefore, it is entirely possible to have some variation in high power flattening from the worst distribution for one case to the other (i.e., between the histograms used in establishing the safety limit MCPR for the 8x8 and 8x8R reload cores). What is important is to assure that the licensing basis histogram is conservative relative to the expected operating histograms for their respective reload core class.

The design basis CPR and relative bundle power histograms for the 8x8R reload core (Figure 1) are compared with two sets of operating histograms (Figures 2 and 3). The operating histograms include the histograms representing both severe and typical of those expected at the BWR/4 8x8R reload core when the reactor is operated at rated core power with the most limiting bundle close to the MCPR operating limit. As can be seen, the expected operating distributions are much more peaked than the ones assumed in the licensing basis, thus yielding an expected operating 99.9% statistical limit MCPR of approximately 1.01 to 1.03. This example illustrates the degree of conservatism included in the licensing basis safety limit MCPR.

In addition, an analysis was performed to evaluate the conservatism in the technical specification safety limit MCPR which was derived for the design basis power distribution. The result showed that the actual operating power distribution was much more peaked than the one assumed in design analysis, thus yielding the 99.9% statistical limit MCPR of 1.00, which is 5% lower than the technical specification safety limit MCPR of 1.05 (7x7 initial core). The following parameters were used in the analyses:

Operating Plant Data:	
Core Power:	3252 MW (99% rated)
Operating MCPR:	1.25
MCPR Operating Limit:	1.25
Radial Power Distribution:	Figure 5.1.1 4
The 99.9% Statistical Limit MCPR:	1.00
Design Basis:	
Technical Specification Safety Limit MCPR:	1.05
Design Basis Power Distribution:	Figure 5.1.1 5

The above results were discussed briefly in page 2 of Reference 1 in response to the NRC question on the effect of the TIP uncertainty on safety limit MCPR.

REFERENCES:

1. “Process Computer Performance Evaluation Accuracy Amendment 1”, NEDC-20340-1, December 1974.

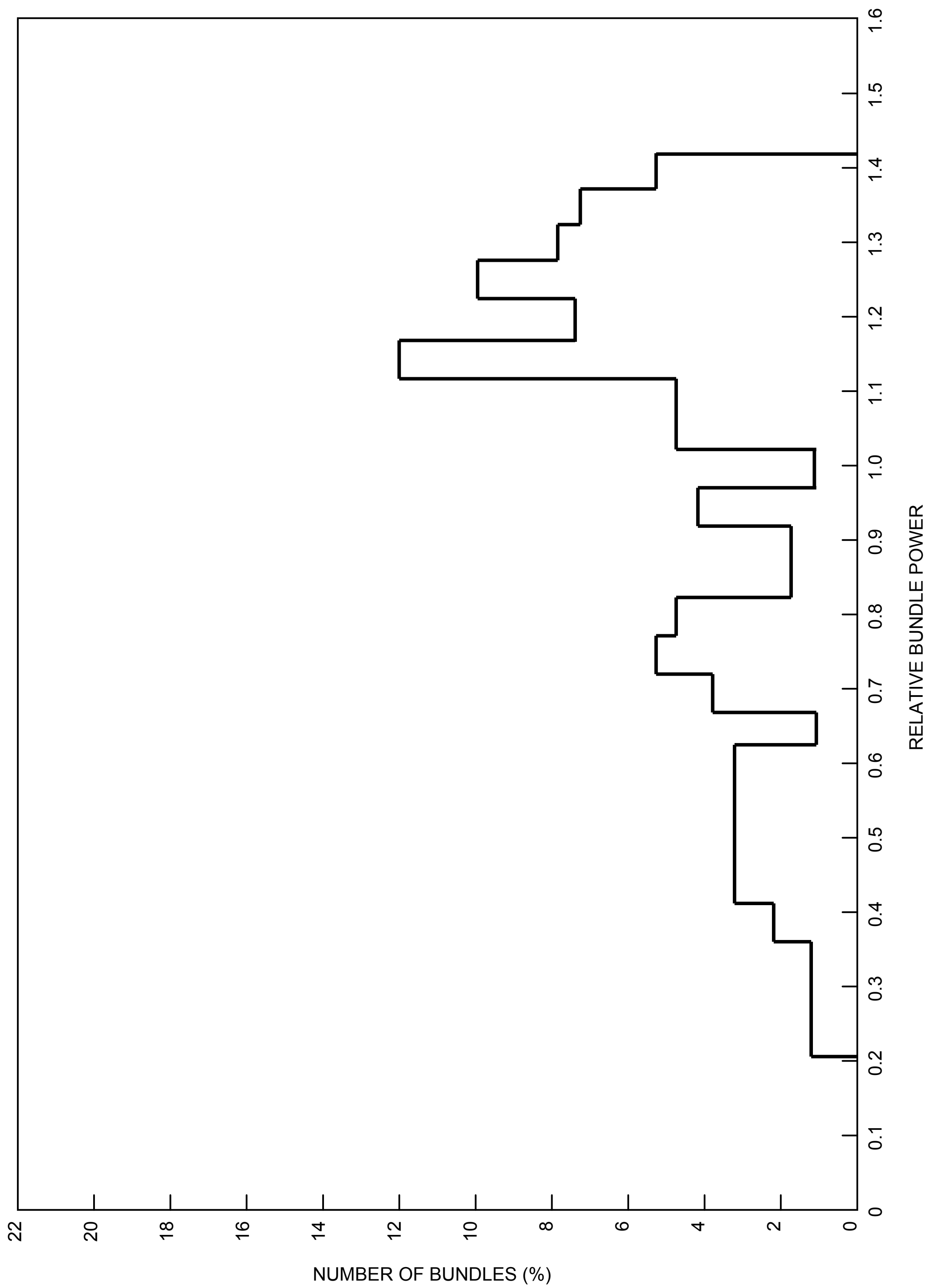

Figure 5.1.1-1a. Relative Bundle Power Histogram for Power Distribution Used in Statistical Analysis for BWR/4 8x8R Reload Core

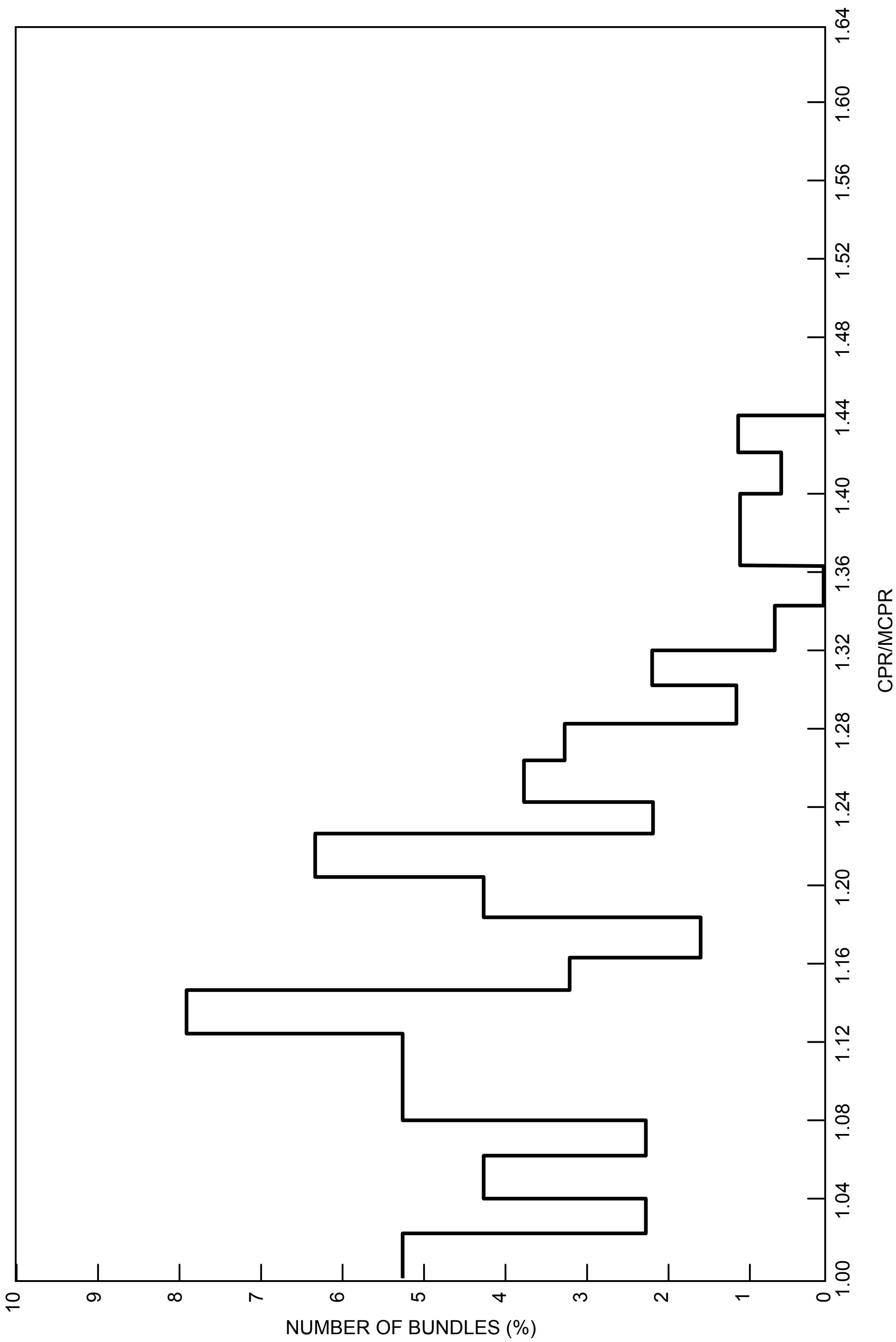

Figure 5.1.1-1b. Normalized CPR Histogram Used in Statistical Analysis for BWR/4 8x8R Reload Core

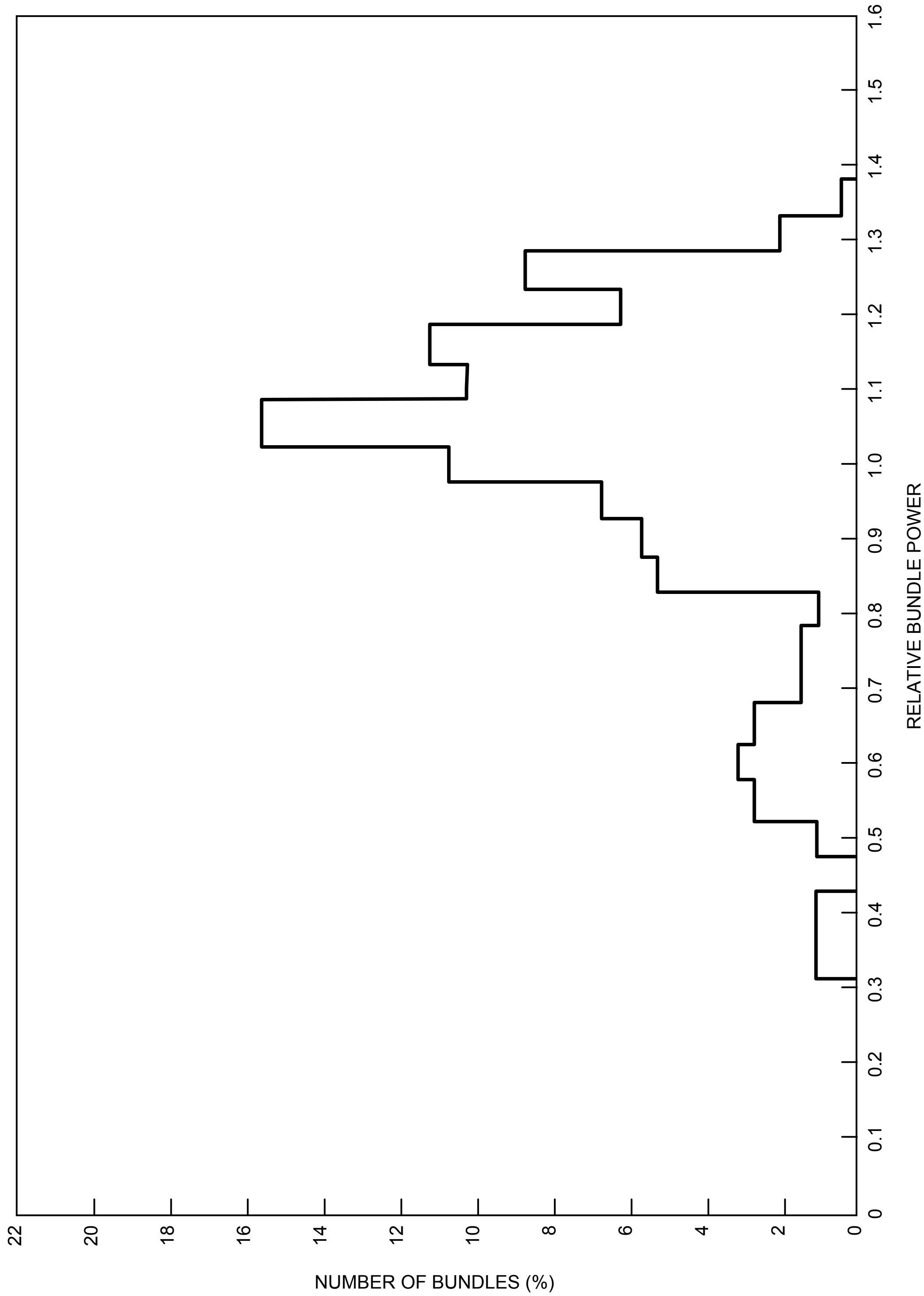

Figure 5.1.1-2a. Relative Bundle Histogram for a Typical 8x8R Reload Core Operating Power Distribution

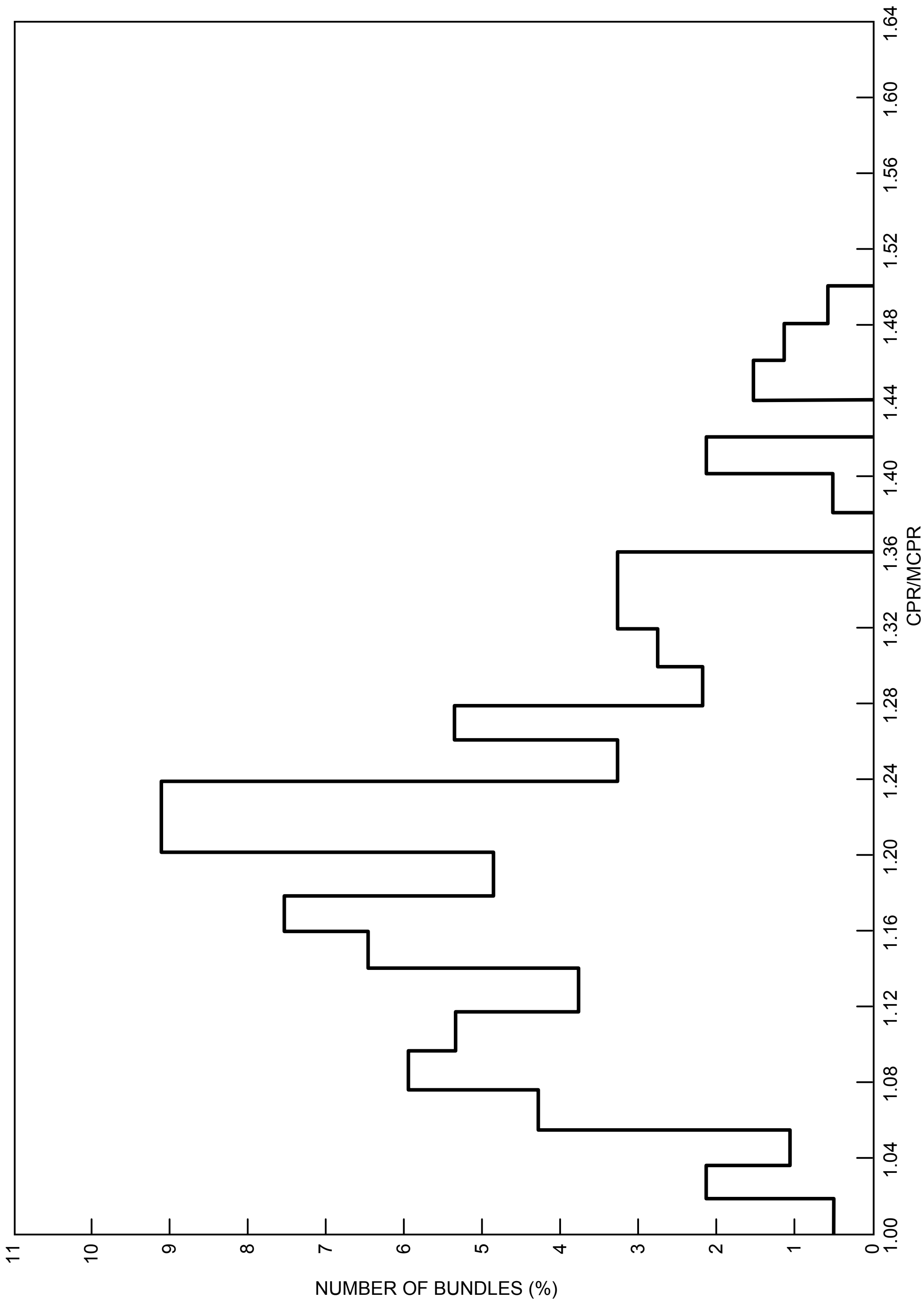

Figure 5.1.1-2b. Normalized CPR Histogram For a Typical 8x8R Reload Core Operating Power Distribution

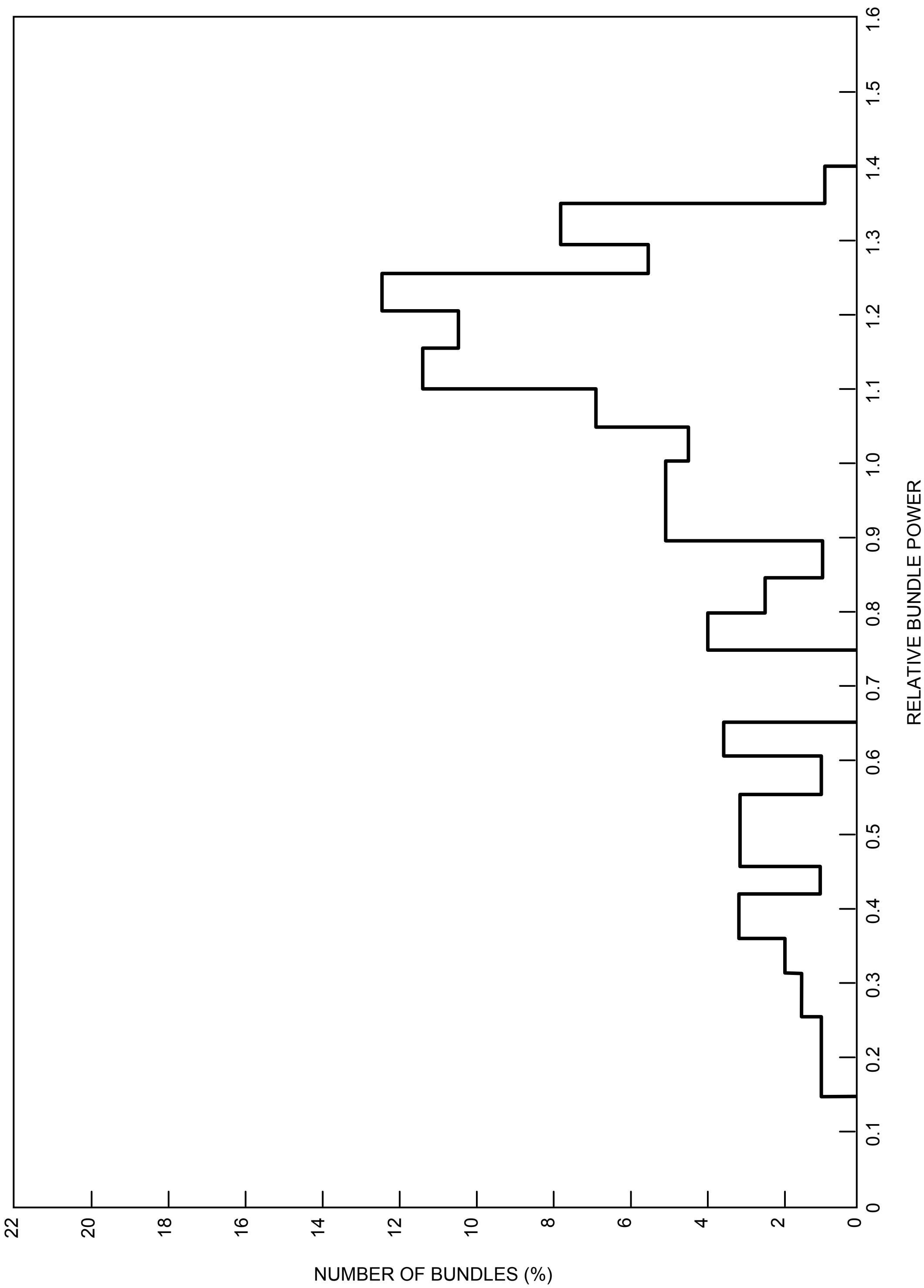

Figure 5.1.1-3a. Relative Bundle Histogram for Relatively Severe 8x8R Reload Core Operating Power Distribution

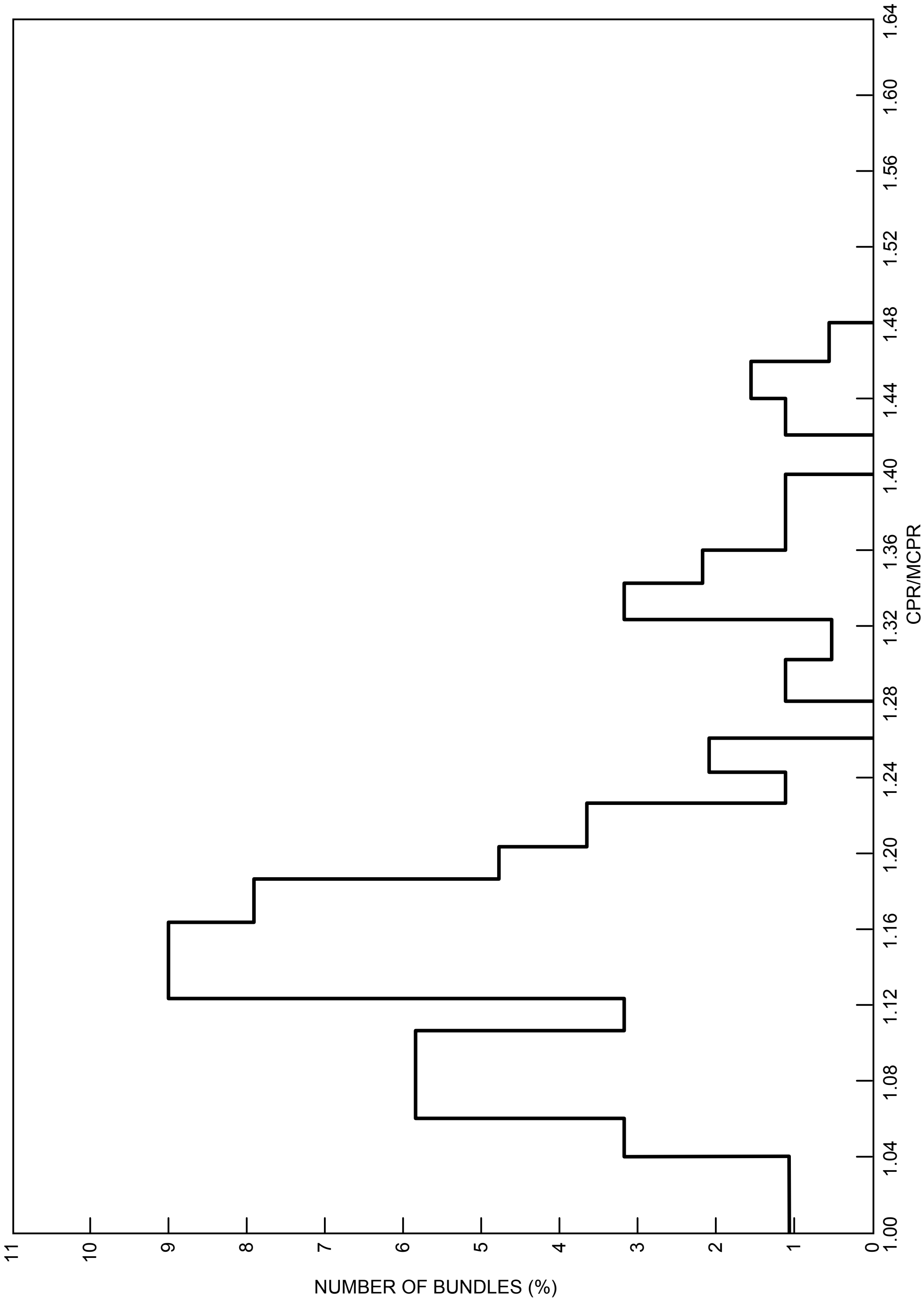

Figure 5.1.1-3b. Normalized CPR Histogram For Relatively Severe 8x8R Reload Core Operating Power Distribution

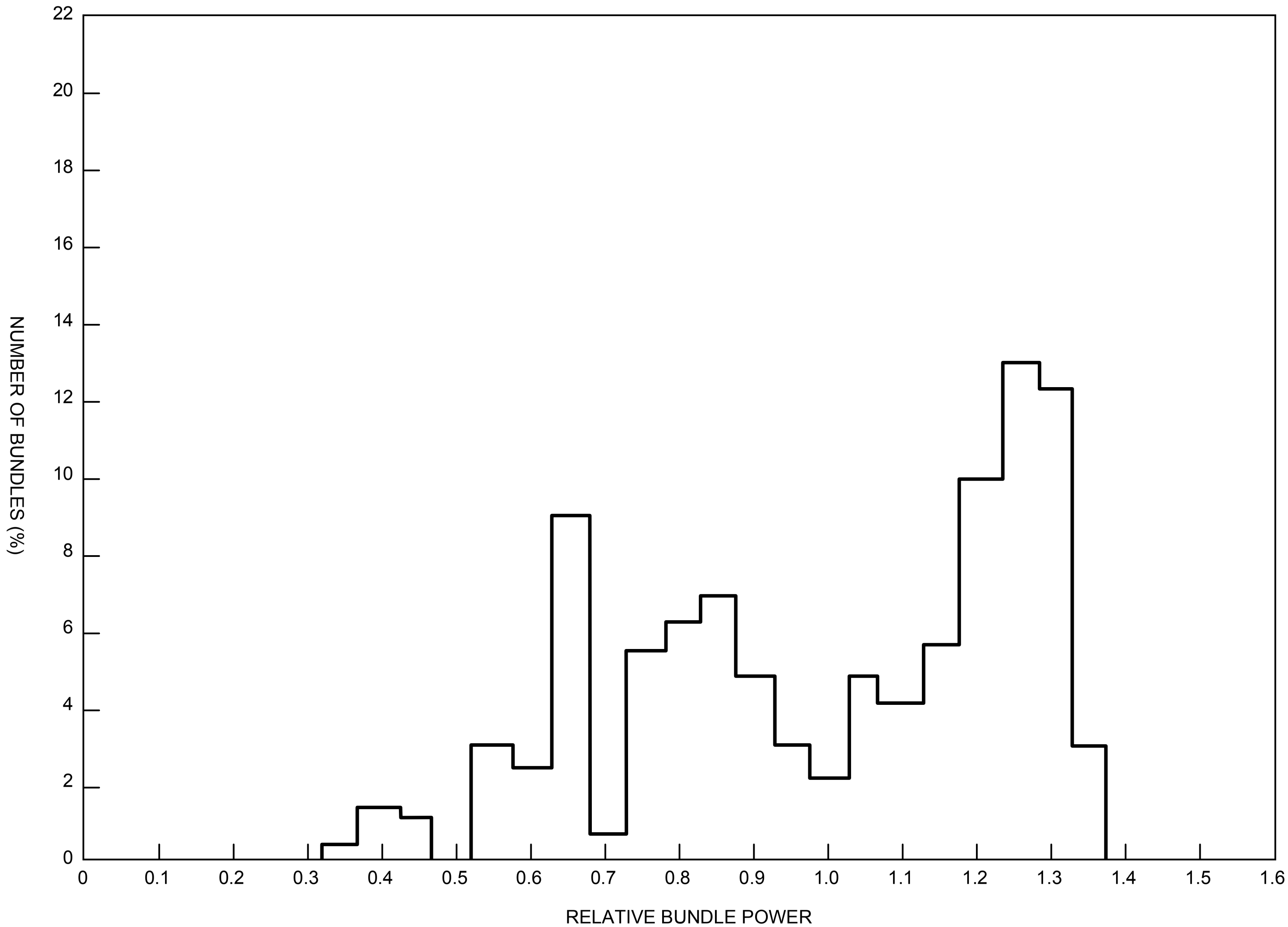

Figure 5.1.1-4. Relative Bundle Power Histogram for Actual Power Distribution

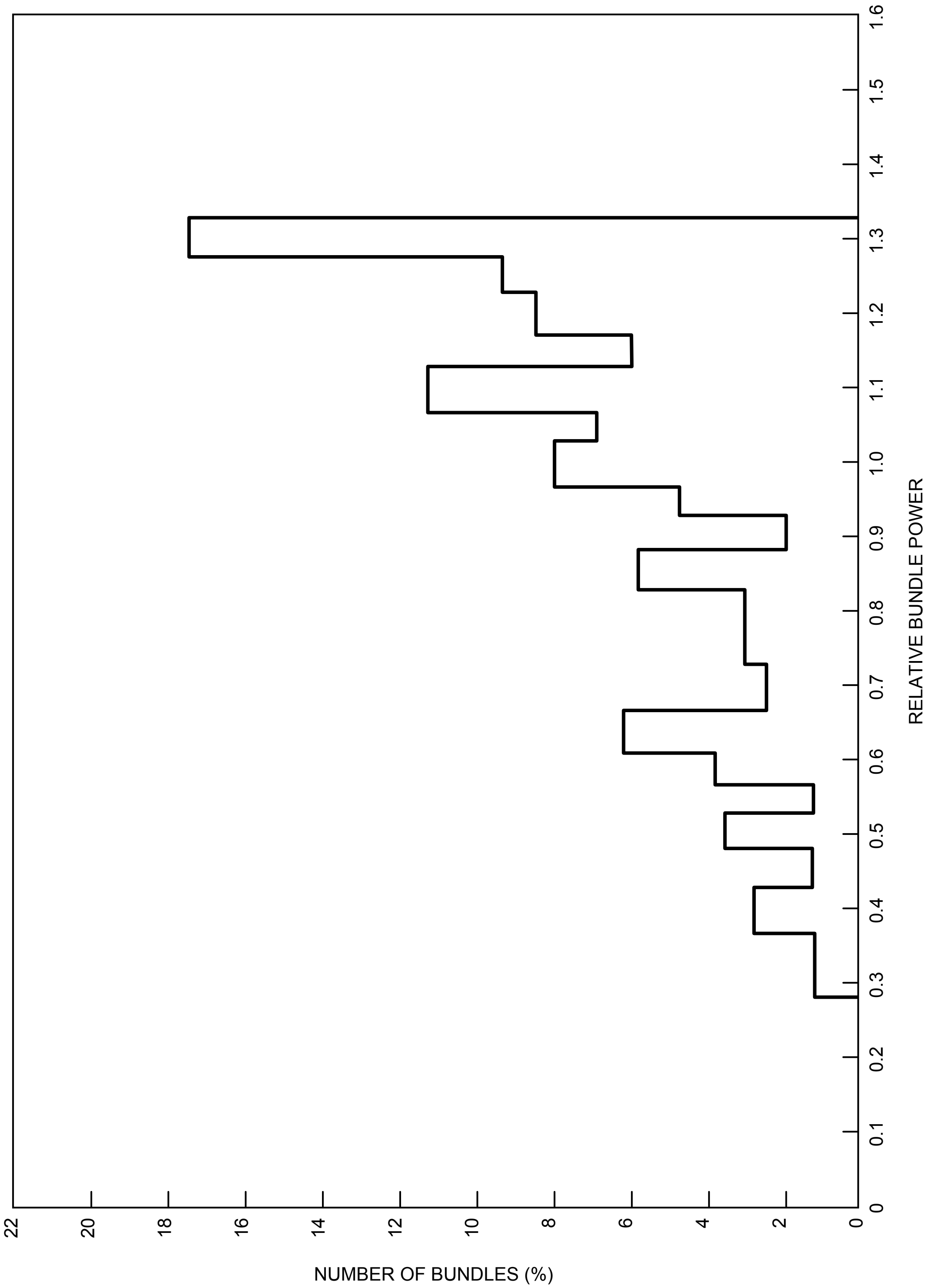

Figure 5.1.1-5. Relative Bundle Power Histogram for Power Distribution Used in Statistical Analysis

REQUEST: Subsection 5.1.2 – Bounding BWR Statistical Analysis

(a) Provide the results of the bounding statistical analysis for 8x8R reloads. Describe the reactor core selected, the relative bundle power histogram assumed, and the rod-by-rod R-factor distribution.

Provide a table of the rod-by-rod R-factor distribution used for the bounding statistical analysis for each fuel type (i.e., 7x7, 8x8, 8x8R) for each class of plant. Indicate fuel enrichments, if appropriate, as well as the cycle exposure.

(b) Using available test data (e.g., Figure 9-10 of Reference 1), quantitatively assess the effect of the nonconservative bias in the GEXL correlation for cells with axially varying R-Factors, in connection with the development of the 1.07 safety limit minimum critical power ratio for normal core operating conditions [i.e., evaluate the magnitude of the underprediction of the number of rods in boiling transition contributed from 8x8R assemblies with axially varying R-Factors (e.g., controlled cells) or conversely evaluate the extent of the underprediction of the SLMCPR (too low a SLMCPR) resulting from an underprediction of the number of rods in boiling transition in assemblies with axially varying R-Factors].

Using available test data, quantitatively assess the effect of the bias described above in connection with the development and applicability of the 1.07 SLMCPR relative to core operations at reduced flow and increased subcooling (e.g., during a TT w/o BP event with prompt recirculation pump trip), since the bias appears to increase for these conditions relative to normal operating conditions.

RESPONSE:

Results of the bounding statistical analysis for 8x8R reloads are presented in revised Subsection 5.1.2. A description of the reactor core selected, the relative bundle power histograms assumed, and the rod-by-rod R-factor distributions are also included. Because R-factor distributions are included in Subsection 5.1.2, the pin-to-pin power distributions which are used to determine these R-factor distributions have been deleted.

Verification that the R-factor uncertainty associated with uncertainties in calculating fuel rod peaking factors is not greater for 8x8R fuel type then for the 8x8 and 7x7 fuel types is as follows. In Reference 5-1, a conservative relation between σ_R and σ_P was shown to be:

$$\sigma_R = \frac{1}{2}\sigma_P$$

where

σ_R = standard deviation of the R factor

σ_P = standard deviation of the local peaking factor

From Equation (1), it can be observed that the standard deviation of the R-factor is directly proportional to that of the local power peaking. The evaluation of the uncertainty of the R-factor, therefore, depends on the determination of the uncertainties of the surrounding local power peaking factors. These local peaking factor uncertainties for the 8x8R fuel types are examined next.

The calculations of the local peaking factors for the different 8x8R, 8x8 and 7x7 fuel designs were done by the same lattice physics code. Since the same numerical scheme was employed, the nuclear uncertainties of the local peaking factors for these different fuel types should be nearly the same. Therefore, the R-factor uncertainties for the 8x8R fuel types are not greater than the 8x8 and 7x7 designs.

In addition, studies have been performed to evaluate the local peaking uncertainties for the 8x8R fuel design. They were done by using the Monte Carlo calculations and high temperature critical experiment data. These studies verified that the 8x8R local peaking uncertainty was about the same as the 8x8 and 7x7 fuel types. Details of these studies are discussed in Subsection 3.2.

The TIP detectors are located at the narrow-narrow water gap (i.e., the noncontrol blade side) of the four bundle assemblies. From an analytical viewpoint, the main source of TIP uncertainty is due to the correlation constants. The correlation constants for the 8x8R and the 8x8 fuel designs are obtained from data utilizing the same lattice physics code. These correlation constants are highly influenced by the localized effect of the narrow-narrow water gap and the surrounding thermal energy group nuclear interactions. The uncertainties of the correlation constants for the 8x8R and 8x8 fuel types are about the same. This is due to two factors:

1. the same numerical scheme is used for their generations, and

2. the extra water rod is sufficiently far away from the narrow narrow gap (several diffusion lengths) that its effect on the uncertainty of the correlation constants are negligible.

Therefore, the TIP measurement is essentially decoupled from small changes at the center of the fuel lattice. The extra water rod in the 8x8R fuel does not contribute additional TIP uncertainty.

In developing the safety limit MCPR of 1.07 for 8x8R reload cores, the partially controlled bundles (PCB) contributed only 0.3% of the total expected number of rods subject to boiling transition (ENRSBT) during a transient. If the GEXL critical power bias (based on the data shown in Figure 9-10 of Reference 1) is considered, which is ~0.015[1] at the operating condition of hot PCB (mass flux ~1.10 Mlb/hr ft^2, subcooling ~25 Btu/lb), the underprediction of the ENRSBT is only ~0.26, which can be translated into an underprediction of the safety limit MCPR by <0.0002, which is negligibly small.

However, it should be emphasized that the test data as shown in Figure 9-10 of Reference 1 fall well within the GEXL critical power predictability (with one sigma, 3.6%, as used in SLMCPR analysis), especially when considering that the deviation is only ~1.5% at typical rated operation condition. In other words, for the assemblies with axially varying R-factor, the GEXL prediction deviation is well covered by the bands of ±7% of the GEXL predictability for the constant R-factor assemblies. The bias shown in Figure 9-10 of Reference 1, a single set of test data, is not expected to be representative of all axially varying R factor cases.

When the core is at reduced flow and increased subcooling condition, the GEXL critical power bias shown in Figure 9-10 of Reference 1 is ~0.05 for an axially varying R-factor assembly, which is based

[1] Based on 14 datapoints (Figure 9-10 of Reference 1) with mass flux from 1.00 to 1.25 Mlb/hr ft^2 and subcooling from 18 to 50 Btu/lb.

on 8 data points with a mass flux of 0.75-1.00 Mlb/hr ft^2 and subcooling of 40-70 Btu/lb. For the development of the 1.07 safety limit MCPR for 8x8R reloads, the underprediction in ENRSBT is ~1.13 rods (from PCB) and it is equivalent to an underprediction in safety limit MCPR by ~0.0006. This small difference has no apparent effect on safety limit MCPR and can be neglected.

In conclusion, the GEXL bias shown in the available test data for the axially varying R-factor assembly has no impact at all in developing the safety limit MCPR of 1.07 for 8x8R reloads.

REFERENCES:

1. GE Letter (R.E. Engel) to NRC (D. Eisenhut), “Fuel Assembly Loading Error”, November 30, 1977.

REQUEST: Subsection 5.2 – MCPR Operating Limit

Discuss the plant transient code models (especially the multinoded thermal-hydraulic and heat transfer relationships) in References 5-2, 5-3 and 5-4, which derive their input parameter values from the fuel mechanical design (e.g., fuel rod diameters, stack lengths). Discuss which of these input parameters utilize a “weighted average” value of the various fuel types and which utilize a “limiting” fuel type value, for mixed reload core configurations (7x7/8x8/8x8R). Provide justification in each case. Describe how these fuel related plant transient code inputs are updated from plant to plant, from one cycle (reload) to the next (reload), as the core configuration changes; e.g., from a mixed core (7x7/8x8 8x8R) to a core having a single fuel type (8x8R). Justify the methods or assumptions used, by providing sensitivity studies where appropriate.

Several of the plant specific inputs appearing in Table 5-6 appear to be incorrect. Please make the corrections as noted in the attached Table A.

Several of the pressure relief system characteristics appearing in Table 5-4 appear to be incorrect. Please make the corrections as noted in the attached Table B.

The prompt reactor period to τ_o $(1/\beta)$ plays a significant role in the Reactor Kinetics model and the dynamic behavior of the core and plant system. The reload supplement must contain the value of to τ_o (or β and l*) used for each reload analysis for each plant.

Provide or reference a description of the bases (models, methods and assumptions) used to develop the nominal EOC and exposure-dependent scram reactivity function for the plant transient analysis.

Provide or reference additional discussion of the source of the uncertainties and biases in the derivation of the nuclear input data. State the approximate magnitudes (% of nominal value) of the uncertainties and biases for each parameter. Provide or reference the results of sensitivity studies, which show the influence of the DCF’s on the key transient output parameters (e.g., ΔCPR, peak pressure, fuel temperature) values for the limiting core wide events. Relative to Table 5-5, indicate which moderate frequency events will be analyzed as power increase transients and which will be analyzed as power decrease transients.

Discuss the extent to which plant-specific values are used for physical parameter inputs to the system transient code (e.g., steamline volumes, core volumes, plenum volumes, system masses) when performing plant specific reload licensing calculations.

Provide a detailed block diagram which illustrates the flow of the key inputs and outputs among the various computer codes (nuclear, system, hydraulic, thermal-hydraulic) used in the determination of the MCPR operating limit (i.e., ΔCPR) for a limiting local (e.g., RWE) event. Provide an additional block diagram for core wide events. Indicate where iterations may be required.

Justify the use of selected axial profile distribution given in Table 5-7 for the SCAT code analysis for all exposures and cycles of all of the plants (Table 1-1). Discuss why the selected profile, when used in connection with various radial and local peaking (R) factors during the cycle, is conservative. Discuss the sensitivity of transient ΔCPR to axial power profile.

The initial operating limit MCPR, assumed for CPR transient analysis, must equal or exceed the finally established operating limit in order to conservatively determine the ΔCPR. Provide a sensitivity plot of this effect and a statement of its consideration in reload thermal hydraulic analysis.

The brief paragraphs, which discuss the operating limit MCPR/low flow correction curves, K_f, for reduced core flow conditions, do not adequately describe the basis for the construction of the generic curves shown in Figure 5-7. A substantially more detailed discussion of the procedure employed is required. Describe the models, methods and assumptions used, which make the curves generic for BWR/2, BWR/3 and BWR/4 plants incorporating up to three different fuel designs. Either justify why the new fuel design (8x8R) is conservatively bounded by the analysis or provide an additional thermal hydraulic analysis to demonstrate that the indicated curves are bounding. Clearly state the criteria used for developing the curves for the automatic and manual flow control modes.

The pressure ordinate values in Figure 5-8 appear to be incorrect as shown. Please correct. The units and ordinate values for the peak surface heat flux, on the same figure, appear to be inconsistent. Please correct.

The use of an <u>example</u> of an event (RWE), which is more severe (larger ΔCPR at reduced power and flow, to show that the flow increase event and (K_f curves) is bounding at reduced flow is <u>not</u> adequate. Add the generic results to Table 5-8 of all other anticipated transients which become more severe at reduced flow and power; e.g., inadvertent startup of an idle loop, feedwater flow controller failure (increase). Explain the methods which were used to make the analysis of these events generic for all plants appearing in Table 1-1.

Describe the procedure used for establishing the slope of the linear approximation curve to the actual exponential power decay (coastdown) curve beyond end-of-cycle. Reference 5-8 does not discuss 8x8R fuel and does not present a generic analysis. Provide a generic analysis and discussion for an 8x8R reload core to support the conclusion that coastdown operation beyond full power operation is conservatively bounded by the EOC analysis.

RESPONSE:

The plant transient code models heat transfer with a single fuel element representing the entire core (Reference 5-2 in the report). Current procedures require this element to be the dominant fuel type (7x7, 8x8, or 8x8R), not a weighted average. As successive reloads move the core towards all 8x8R fuel, the inputs are changed from 7x7 and 8x8 towards 8x8R when that type becomes dominant (i.e.,

≥50% of the bundles). Fuel parameters input are: rod length, rod diameter, clad thickness, pellet diameter and the number of fueled rods per bundle.

Fuel-clad gap conductance, on the other hand, is a weighted average (rather than the expected value of a pellet with 1.0 peaking). It is calculated for a single fuel type core and is dependent on both fuel type and product line.

Sensitivity studies (Reference 5-2) have shown a weak effect of fuel time constant on the peak transient results. The trend is towards higher pressure and heat flux peaks on limiting pressurization events for a smaller fuel time constant. (Smaller rod size → smaller time constant.)

In addition, to determine the effect of using the dominant fuel type characteristics (i.e., fuel dimensions) to determine the core average transient response, as opposed to the fuel type with the fastest time constant, the generator load rejection without bypass transients were performed considering each of the three (7x7, 8x8 and 8x8 retrofit) fuel types to be dominant. The nuclear and thermal hydraulic inputs were based upon the following core makeup:

Fuel Type	No. Bundles in Core	Active Fuel Length (in.)
7x7	324 (42.4%)	144
8x8	187 (24.5%)	144
8x8 Retrofit	253 (33.1%)	150
Total	764	

The results of the core average transient response were:

Maximum	Fuel Type Assumed Dominant		
	8x8R	8x8	7x7
Steam line Pressure (psig)	1196	1196	1197
Vessel Pressure (psig)	1239	1239	1241
Neutron Flux (% initial)	289.7	292.1	305.9
Heat Flux (% initial)	112.4	112.4	111.9

These results indicate that varying the fuel type has only insignificant effects on the core average transient response.

The values of the plant inputs in Table 5-6 were corrected in Amendment 1. At present, nominal transient input values and allowable tolerances are given in Table 5-6 and GETAB initial conditions are documented in Table 5-8. Nominal transient input values were placed in Table 5-6 to preclude revisions with every reload. The values in Table 5-4 have been corrected. As indicated in the response to the request on Section 5 of Appendix A, the prompt reactor period should not be included in each reload submittal.

The core condition which serves as the starting point for the EOC scram reactivity calculation is determined by performing a Haling calculation.

At exposure points before EOC, the Haling power distribution is used to accumulate exposure from BOC to the exposure point of interest. The required control rod inventory for the before-EOC points are designed to achieve criticality, minimize scram reactivity response, and yield reasonable thermal performance.

Further descriptions of the models, methods, and assumptions for the calculation of the scram reactivity are given in Reference 1.

The results of sensitivity studies which show the influence of design conservative factors (DCF) on the key transient output parameters for the limiting events are found in Reference 5-2.

The moderate frequency events which are analyzed as power decrease transients are those events which result in a core coolant flow decrease. These events are: (1) recirculation flow controller failure-decreasing flow; (2) trip of one recirculation pump; and (3) trip of two recirculation MG set drive motors. Identification of power decrease and increase transients is given in Table 5-5.

Plant specific values are provided for physical parameter inputs to the system transient code as described in Reference 5-2. The parameters for each plant are checked with the plant owner and updated for each reload as necessary with current parameter data when performing plant specific reload licensing calculations.

Steady-state hydraulic calculations such as core bypass flow and core and channel pressure drops are initially performed. The results of these analyses are used as input for the nuclear evaluations (Section 3) to determine the nuclear transient input values. The hydraulic and nuclear evaluation results are then used as inputs to the core-wide transient analyses (Subsection 5.2). Operating MCPR limits (Subsection 5.2) for core-wide transients are determined from the results of the transient analyses. The nuclear models are used to determine the RWE transient results and MCPR as a function of rod position. Information relative to the models employed are documented in the above indicated section or subsection.

A description of the transient analysis procedures for abnormal operational transients is also given in Section 6.5.2 of Reference 5-1. Justification of the axial profile used in the transient analyses is given in Appendix V of Reference 5-l

The initial MCPR assumed for transient analyses is usually greater than or equal to the GETAB operating limit. The attached Figure 5.2-1 illustrates the effect of the initial MCPR on transient ΔCPR for a typical BWR core. This figure indicates that the change in DCPR is 0.01 for a 0.05 change in initial MCPR. Therefore, in some cases, nonlimiting GETAB transient analyses may be initiated from a MCPR below the operating limit because the higher operating limit MCPR more than offsets the increase in ΔCPR for the event. This may also be applied to limiting transients if the difference between the operating limit and the initial MCPR is small (0.01 or 0.02).

The method of development and justification of the K_f curves were previously provided to the NRC in response to questions on Quad Cities 2, Reload 1, in 1975. The purpose of the K_f factor is to define MCPR operating limit requirements at other than rated flow conditions. Specifically, the K_f factor provides the required thermal margin to protect against a slow flow increase transient (where a flux scram does not occur), which is the most limiting reduced core flow event

Development Criteria – Manual Flow Control Mode

The manual flow control mode K_f factors were calculated such that at the maximum flow state (as limited by the pump scoop tube setpoint) and the corresponding core power (along the reference flow control line) the relative power of the limiting bundle was adjusted (upward) until the MCPR was at the safety limit. Holding this relative power distribution constant, MCPR was calculated at several points along the flow control line.

The ratio of the MCPR's calculated at these points, divided by the operating limit MCPR, determines K_f at each point. This procedure was repeated for each maximum limit (scoop tube setting). The following is an example derivation of the 102.5% flow K_f curve for a BWR/4 core containing an essentially equal number of 7x7, 8x8, and 8x8R fuel bundles.

The power-flow relations used in this example are provided in Figure 5.2 2. Pertinent core parameters are:

Rated Core Power		3293 MW	
Rated Core Flow		102.5 Mlb/hr	
Total Number of Fuel Assemblies		764	
	7x7	**8x8**	**8x8R**
Number of Fuel Assemblies	254	255	255
R-Factor	1.100	1.100	1.051
Maximum Radial Power Factor	1.499	1.624	1.771

At 102.5% rated flow and 101.8% rated power, the MCPR is 1.06 for the maximum powered assemblies listed above. The MCPR for each of these fuel assemblies as a function of the core power and flow is tabulated in Table 5.2-1. Assuming a 1.20 operating limit MCPR for each of the fuel types, a K_f curve can be generated for the 102.5% maximum flow line using the relation of

$K_f = \frac{\text{MCPR (W)}}{1.20}$, where MCPR (W) is the MCPR at a given core flow fraction from Table 5.2-1.

Note that as a particular plant's operating limit MCPR increases above 1.20, the K_f curves (Figure 5-7) will become increasingly conservative (i.e., if the operating limit was to increase for a given reload 1.20 to 1.30, the curves would be overly conservative by the ratio 1.30/1.20, because the change in critical power ratio with core flow should not change from cycle to cycle). This "overconservatism" is presently being reviewed and future changes to the K_f approach are being considered.

Figure 5.2-3 compares the derived 102.5% core flow K_f curve with the standard curves (Figure 5-7). These results show that the 8x8R fuel closely follows the 102.5% line, but the 7x7 fuel is 3.3% nonconservative at 40% core flow. However, the K_f curves are conservative for all cores with 7x7 operating limits greater than 1.23.

Development Criteria – Automatic Flow Control Mode

For operation in the automatic flow control mode, the same procedure is employed except the initial power distribution is established such that the MCPR is equal to the operating limit MCPR at rated power and flow. Thus, it is assured that automatic power/flow increases will result in meeting the operating limit MCPR for any power/flow increase up to and including the rated power condition. However, an inadvertent flow increase in this mode will result in substantial margin to the safety limit MCPR.

The development of the generic K_f curves in the fuel reload licensing topical report employed a generic power-flow curve. The resulting critical power-flow relation is dominated by the GEXL correlation, not by plant size, power density, or slight variations in core inlet enthalpy and pressure.

The peak vessel dome pressure of 1180 psia has been changed to 1080 psia, and the ordinate values of l.0, 1.1 and 1.2 for the peak surface heat flux (% of initial) have been replaced with the values 100.0, 110.0 and 120.0, respectively.

The generic K_f curves were developed on the basis that the flow increase event was the most limiting at reduced flow. This basis was justified by studies performed on BWR/3 and 4 FSAR data (7x7 fuel). Specifically, inadvertent startup of an idle recirculation pump and feedwater flow controller failure (maximum demand) events were analyzed at reduced power and flow as summarized in Table 5.2-2, demonstrating the adequacy of K_f.

The most recent evaluations in this area were performed on a standard 764 bundle BWR/5 core with retrofit fuel, simulating equilibrium core conditions. The operating limit MCPR for this core was 1.27, and the safety limit MCPR was 1.07. The results of inadvertent cold loop startup and feedwater flow controller failure (increase) transients initiated from various power/flow states are presented in Table 5.2-3. In each case, the initial CPR was determined such that the minimum, CPR during the transient was l.07. Thus, as long as the product of K_f and the operating limit MCPR is greater than the initial CPR necessary to satisfy the safety limit MCPR, the K_f curves are conservative. This comparison is provided for the 102.5% maximum core flow K_f curve in Table 5.2 3 and demonstrates conservatism at all flows except 75% flow where this K_f curve is only slightly nonconservltive by 0.005.

The results of these analyses (Table 5.2 3) are considered generic because the data used bound the BWR/2 4 projected equilibrium conditions. While there may be variations between plants in their reduced power and flow transient response, a similar variation would also exist for the limiting rated power and flow transient, creating a sufficiently high operating limit MCPR that, when combined with the K_f curves, an appropriately high reduced flow operating limit is established.

The curve of power level decay versus exposure is determined by calculating the exposure capability at selected power levels with the BWR simulator code.

With respect to the pressurization transients discussed in Section 5.2, the primary reason for the decrease in transient severity during coastdown is the dynamic void coefficient, which becomes less negative as the power level is reduced. This is an inherent characteristic of the BWR, which does not change with the introduction of the 8x8 retrofit reload lattice.

REFERENCES:

1. Letter, G.L. Gorey (GE) to W.R. Butler (NRC), "1-Dimensional Methods for Calculating Scram Reactivity," 12 March 1976.

5.2-1 MCPR

Power	Flow	8x8R	8x8	7x7
101.8%	102.5%	1.060	1.060	1.060
100	100	1.075	1.075	1.073
93	90	1.129	1.131	1.128
86	80	1.182	1.186	1.184
78.9	70	1.234	1.241	1.242
71.5	60	1.287	1.298	1.304
63.9	50	1.339	1.353	1.365
56.0	40	1.389	1.407	1.428

5.2-2 Evaluation of BWR 3 & 4 Abnormal Operational Transients Initiated from Lower Power States SAR Data (7x7 Fuel)

Event	Power	Flow	Initial CPR	ΔCPR	Tech Spec K_f	Required K_f*
Idle Recirculation Loop Startup	60%	45%	1.64	0.08	1.13	1.0
Feedwater Control Failure (Max Demand)	65%	51.5%	1.60	0.06	1.105	1.0

*The required K_f is derived such that the MCPR safety limit is satisfied assuming an operating limit MCPR of 1.20.

5.2-3 Evaluation of BWR/5 Abnormal Operational Transients Initiated from Lower Power States (8x8R Fuel)

Event	Power	Flow	ΔCPR	Initial CPR[1]	K_f* 1.27 [2]
Feedwater Controller Failure (Increase)	105	100	0.179	1.249	1.270
Feedwater Controller Failure (increase)	100	91	0.185	1.255	1.270
Feedwater Controller Failure (increase)	97	85	0.199	1.269	1.270
Feedwater Controller Failure (increase)	91	75	0.205	1.275	1.270
Feedwater Controller Failure (increase)	85	61	0.223	1.293	1.353
Idle Recirculation Loop Startup	63.9	49.7	0.152	1.222	1.412
Idle Recirculation Loop Startup	56.2	40	0.361	1.431	1.463
Idle Recirculation Loop Startup	63	36	0.207	1.277	1.483

1 The initial CPR was determined by iteration on the bundle power such that the minimum CPR during the transient was 1.07.

2 K_f based on maximum core flow of 102.5%. The operating limit MCPR is 1.27.

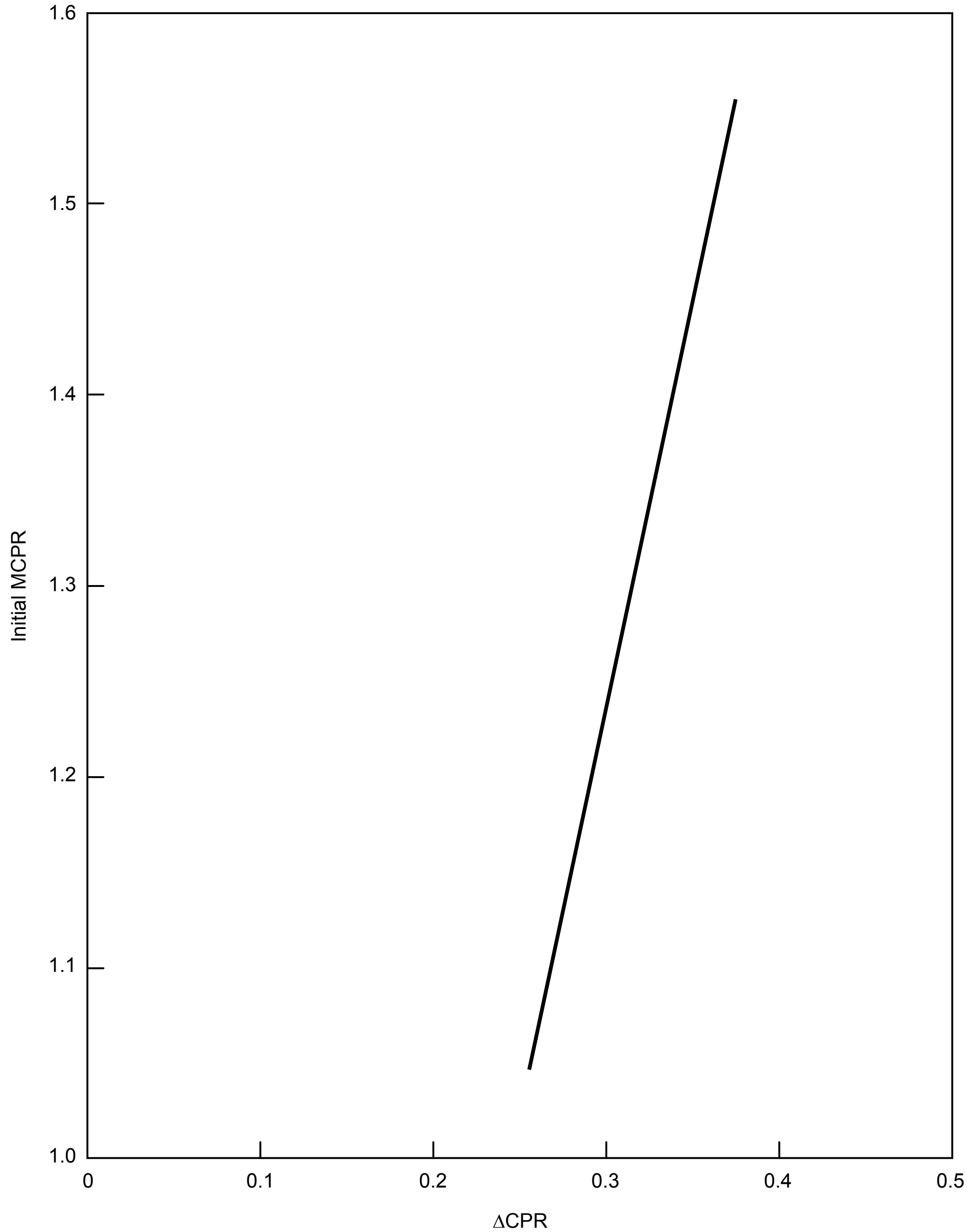

Figure 5.2-1. Effects of Initial MCPR on ΔCPR Typical BWR Core

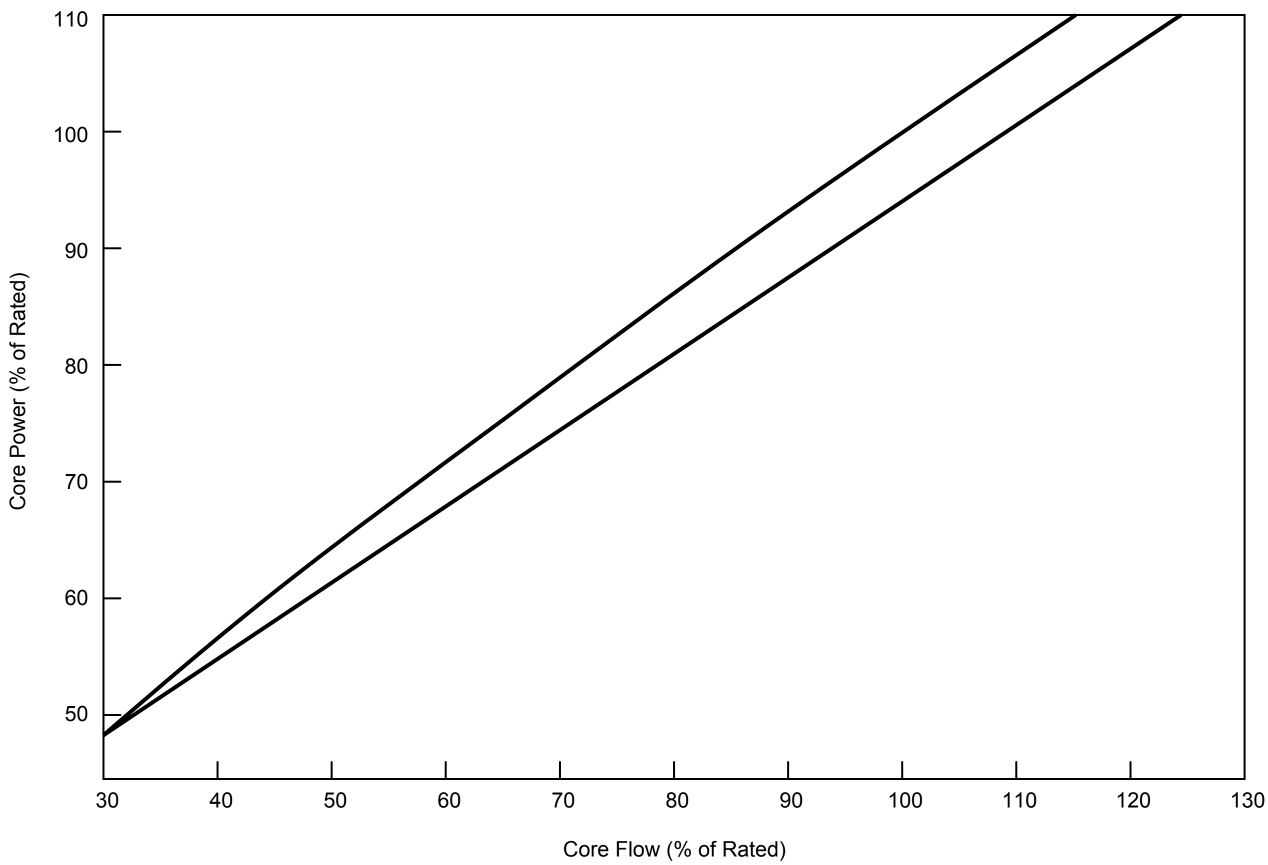

Figure 5.2-2. BWR/2–4 Power-Flow Map

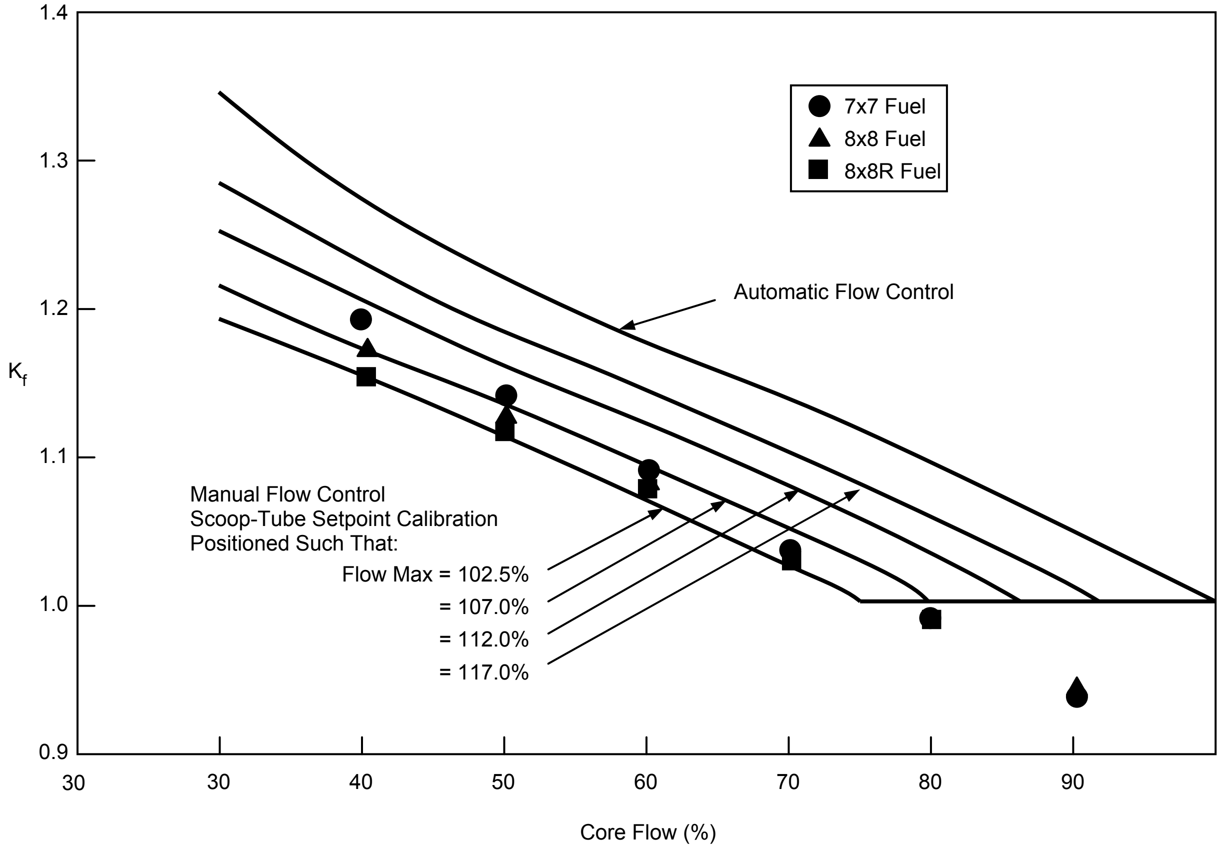

Figure 5.2-3. K_f Factor

REQUEST: Subsection 5.2.1 – Transient Descriptions

The CPR sensitivity study results presented in Section 5.2.1, as a justification for eliminating most of the FSAR transients from reload reanalysis, is not acceptable. It does not address the 8x8R fuel design. Furthermore, the CPR sensitivity, indicated by the GEXL correlation, is only one component of the total methodology used for determining the ΔCPR for an event. The magnitude of the changes in the various thermal-hydraulic parameters (i.e., REDY code output) during an anticipated transient is just as important. Therefore, provide the results of the SCAT code transient analyses, which show the change in critical power ratio for the most limiting event in each of the eight transient parameter (variation) categories on page 5-10. The analyses should model a representative BWR plant. Provide justification for the plant modeling, reactivity coefficients and scram curves selected. Compare these results to the results of events which are proposed for analysis on plant specific reload applications (e.g., TT w/o BP, LFWH, HPCI startup, FWCF, RWE). The results should be presented for 7x7, 8x8 and 8x8R fuel types in a representative mixed core loading situation. Provide an assessment of the

change, from cycle-to-cycle, in relative and absolute severity of the events analyzed above due to exposure effects. Provide conclusions relating to the adequacy of the limited group of events selected for reload reanalysis for all future fuel cycles.

For relatively slow transients (e.g., loss of feedwater heating), the assumption that the local (thermal) power distribution remains constant (and hence the R-factor remains constant) may not be valid. Provide an evaluation of the validity or conservatism of this assumption for relatively slow events. Evaluate this assumption for each of the various fuel designs (7x7, 8x8, 8x8R).

The following comments apply to all of the transient description subsections:

1. The first paragraph should be labeled as "Identification of Causes." This paragraph should appear for all events considered. Noteworthy examples should be included.

2. The second "Starting Conditions and Assumptions" paragraph should also indicate:

   (a) The percent of the licensed power level/steam flow which includes an allowance for the thermal power level uncertainty (i.e., ≥2%). An allowance is required for all transient analyses except for the 25 psi margin to safety valve setpoint analyses (as specified in Standard Review Plan Sections 5.22 and 15).

   (b) The initial state of the plant process variables.

   (c) The operation of the reactor protection systems (i.e., relief valves, scram). (See "Standard Format and Content of Safety Analysis Reports.")

3. The extent and depth of information provided in the "Event Description" paragraph is an inadequate qualitative description of the behavior of the major plant process variables (e.g., pressure, steam flow) and core variables (neutron flux, heat flux, voids) as well as the responses of the important reactor system equipment during the event. The time variations in the variables discussed should be qualitatively related to the time behavior of other key parameters and system equipment. The discussion should be representative of a typical plant. Any differences in the general transient behavior of the process variables due to plant-to-plant differences in equipment design (e.g., transient feedwater flow due to FW pump power source, ATWS RPT) should be noted on a plant-by-plant basis.

4. Provide a "Results" paragraph which states which results will be presented for each of the transients analyzed (e.g., pressure flux, steam flow histories, peak vessel pressure, peak kW/ft ΔCPR). State the section in the document which discusses the appropriate limits for these parameters. Where interpretation is the results is not obvious (e.g., RWE), provide a brief explanation. Provide a brief explanation of the REDY code parameters plotted as part of core-wide analyses (e.g., "WR Sensed Level").

Some plants reanalyze the Feedwater Control Failure-Increasing Flow for reloads. Furthermore, this event may be limiting for many plants in the event that transient (TT w/o BP) recategorization is approved. Therefore, provide a subsection which describes the Feedwater Control Failure—Increasing Flow event. Include in the section all the paragraph headings addressed in the other subsections of Subsection 5.2.1.

It is assumed that <u>all</u> of the transient events described in Subsection 5.2.1 will be analyzed by all of the plants listed in Table 1-l for the supplemental reload licensing submittal. If this is not the case, a

justification for eliminating one or more events from the reload analysis must be provided. A summary table, showing the plants, with the moderate frequency events analyzed, should be included in the Topical if appropriate.

RESPONSE:

The entire spectrum of transients covered in each of the eight transient parameter (variation) categories is given in the individual plant FSAR's for typical operating plants. A review of these transient results (e.g., pressure, power, flow) can be used to determine which categories and which transients in each category have the potential for being limiting. From this it is clearly established that the limiting transients will always be in the four groups of transients identified on page 5-11. These are the transients which involve significant effects on heat flux and reactor vessel pressure peaks. The fact that there are differences between the transient results when GETAB is used rather than the previous thermal analysis basis is recognized.

Other transients already analyzed in FSAR's have relatively less severe effects on heat flux performance and transient pressure peaks. However, the differences due to the GETAB analysis are not significant enough to warrant reanalysis. The reasons for this become obvious when the results of the study of the relative dependence of CPR upon various thermal-hydraulic parameters presented in Subsection 5.2.1 are reviewed. As stated, CPR is most sensitive to R-factor and bundle power. Because R-factor is a function of bundle geometry and local power distributions and does not change during a transient, CPR becomes primarily a function of changes in power. Further verification of the results can be found in GESSAR.

Therefore, General Electric has determined that the effect on CPR, as caused by the four groups of power increase transients, is so much more significant than the effects of any of all the other known transient possibilities, that reanalysis of the remainder of the transients is not warranted. However, to provide complete assurance that the most limiting transients are always identified and analyzed for each reload, all of the transients identified in Subsection 5.2.1 are analyzed for each plant (except inadvertent HPCI startup only for plants with HPCI). The Supplemental Reload Submittal will report the results of only the most limiting transient in each of the four groups.

The sensitivity of the plant response to transients due to exposure has been demonstrated by reload analyses for some operating plants for as many as eight operating cycles. Analyses of plants which have been operating for so many years show that, although the overall severity of transients is affected by degradation in scram until equilibrium is reached, the spectrum of limiting transients does not basically change. It has been found that when end-of-cycle conditions are significantly more severe than mid-cycle conditions, the operating limit determining transients may change. For example, if at end-of-cycle, the turbine trip without bypass is limiting, it may be found in a mid-cycle analysis that the TT w/o BP is so much less severe that the rod withdrawal error or loss of feedwater heating transient may be limiting. These predictable changes due to exposure are caused primarily by the sensitivity of the negative scram reactivity characteristic to exposure up to equilibrium cycle. The introduction of 8x8R fuel is not expected to change this since, as shown above, the plant response is basically no different than with 8x8 fuel.

The increase in inlet subcooling due to the loss of feedwater heating causes the core power to increase and moves the boiling boundary higher in the core. The result is that the axial power will shift lower in the core (~6 inches). This will have a slight affect on the local peaking distribution at the new boiling boundary. The effect on R-factor is minor because the bundle R-factor is the axial averaged rod R-

factor (see responses to NRC questions on fuel assembly loading error in Reference 1), and this 6-inch change in the peak power represents only l/24 of the total fuel length.

In analyzing this type of transient, the licensing basis assumes that the axial power shape is held constant (peaked at the middle of the fuel assembly). As noted above, these events result in an axial power shift lower in the bundle, which in the actual case will cause a smaller ΔCPR because bottom peaked power shapes yield higher critical power ratios than middle peaked, as was demonstrated in Section V of Reference 5-l. That is, the licensing basis calculation is conservative because credit is not taken for the shift in power shape.

Transients are precipitated by a single operator error or equipment malfunction. The initiating event for each transient has been included in either the first paragraph or event description portion of each transient description in Subsection 5.2.1. It should also be noted that this report was not written to conform to the Standard review plans which apply only for plants seeking construction and operating permits. Response to the recommendations for the “Starting Conditions and Assumptions” paragraph are as follows:

1. The percent of licensed power level/steam flow used in individual plant analyses is identified in the Supplemental Reload Submittal for each limiting transient. The 2% adder for power level uncertainty is not included as discussed in Reference 2. This letter indicates that the licensing basis, as defined in the bases statement on fuel cladding integrity in typical plant technical specifications, describes the conservatisms incorporated in transient analyses. Therefore, application of a 2% adder for uncertainties in power level is considered unnecessary.

2. The initial state of the plant process variables is identified in Tables 5-4, 5-6 and new Table 5-8. In addition, cycle-dependent data is provided in each Supplemental Reload Submittal.

3. The operation of the affected reactor protection systems is discussed in each transient description provided in Subsection 5.2.1. The individual plant FSAR provides a completely detailed, comprehensive description of the operation of the reactor protection system.

The intent of the “Event Description” paragraph is to provide a concise summary of events which assumes some prior familiarity with BWR plants. For a comprehensive discussion of the behavior of major process variables, core variables or the responses of reactor system equipment, see the individual plant FSAR and Reference 5-2 of the report.

All the “results” information requested are described in Appendix A, which shows the format and content of the Supplemental Reload Submittal. Concern for nonobvious interpretations is reduced by the new Appendix A format in Revision 1. An explanation of the REDY code parameters is provided in NEDO 10802.

The feedwater controller failure (maximum demand) event mentioned in this question was erroneously omitted in the original document. In description of the pressure regulator failure (open) event was included in its place. This error was corrected in Amendment 1.

The transient events on page 5-11 are analyzed each cycle for all plants with appropriate omissions for plants without HPCI. It should be noted, however, that only the limiting transient for each group is

reported in the Supplemental Reload Submittal. In general, for plants with the mechanical hydraulic control (MHC) feature in their turbines, the limiting transient will usually be turbine trip without bypass. Conversely, if the turbines are equipped with electrical hydraulic control (EHC), the limiting transient will usually be generator load rejection without bypass. The loss of feedwater heating transient is usually more severe than inadvertent HPCI startup.

REFERENCES:

1. Letter, R.E. Engel (GE) to D.G. Eisenhut (NRC), "Fuel Assembly Loading Error," November 30, 1977.

2. Letter, R.L. Gridley (GE) to D.G. Eisenhut (NRC), "MSIV Closure with Flux Scram-Sensitivity of Peak Vessel Pressure to Initial Power Level", September 12, 1977.

REQUEST: Subsection 5.2.1.1 – Generator Load Rejection Without Bypass

The "Starting Condition and Assumption" should include the assumed performance of the control, and relief valves. The closure time of the Turbine Controls valves should also be included in this paragraph. "Event Description" must be improved per item (c) above.

RESPONSE:

After the steamline pressure reaches the setpoint, the turbine control valves are assumed to operate in a fast closure mode, and the relief valves will open at the rate and delay time shown in Table 5-4. The closure time appears in the "Event Description" paragraph.

REQUEST: Subsection 5.2.1.2 – Turbine Trip Without Bypass

The "Starting Conditions and Assumptions" comments are the same as for 5.2.1.1.

RESPONSE:

The description in the "Starting Conditions and Assumptions" paragraph for this transient refers to the generator load rejection because the "Starting Conditions and Assumptions" for both transients are identical. Refer to the response to the request on Subsection 5.2.1.1.

REQUEST: Subsection 5.2.1.3 – Loss of Feedwater Heating

Provide a list of plants with an assumed 100°F LFWH capability and those with a documented 80°F LFWH capability. The "Event Description" paragraph must be improved per item (c) above.

RESPONSE:

	LFWH Capability (°F)
BWR/2	
Nine Mile Point 1	100
BWR/3	
Monticello	100
Millstone	100
Pilgrim	100
Quad Cities 1	145
Quad Cities 2	145
Dresden 2	145
Dresden 3	145
BWR/4	
Vermont Yankee	100
Duane Arnold	100
Cooper	100
Fitzpatrick	80
Hatch 1	100
Brunswick 1	100
Brunswick 2	100
Peach Bottom 2	100
Peach Bottom 3	100
Browns Ferry 1	100
Browns Ferry 2	100
Browns Ferry 3	100

REQUEST: Subsection 5.2.1.4 – Inadvertent Start of HPCI Pump

Provide an "Identification of Cause" paragraph. Provide a table which shows which plants have HPCI and which do not. The "Event Description" paragraph must be improved per item (c) above.

RESPONSE:

	HPCI (Yes/No)
BWR/2	
Nine Mile Point 1	No
BWR/3	
Monticello	Yes
Millstone	No
Pilgrim	Yes
Quad Cities 1	Yes
Quad Cities 2	Yes
Dresden 2	Yes
Dresden 3	Yes
BWR/4	
Vermont Yankee	Yes
Duane Arnold	Yes
Cooper	Yes
Fitzpatrick	Yes
Hatch 1	Yes
Brunswick 1	Yes
Brunswick 2	Yes
Peach Bottom 2	Yes
Peach Bottom 3	Yes
Browns Ferry 1	Yes
Browns Ferry 2	Yes
Browns Ferry 3	Yes

REQUEST: Subsection 5.2.1.5 – Rod Withdrawal Error

I. Provide an "Identification of Cause" paragraph. Discuss the basis for the initial power level assumption. Provide a discussion of the allowances made for various combinations of failed LPRM strings in the RWE analysis. Provide a discussion of the allowance made for the response loss in the local detectors due to excessive voiding in the bypass region. Provide a discussion of the plots to be presented on reloads and their interpretation. Discuss the LHGR limits for each type (and rod type). To what fuel design is the 17.5 kW/ft thermal design limit refer? Provide a statement that the resulting ΔkW/ft (above the respective design values) or peak kW/ft will be provided in the reload supplement for each fuel type.

For calculation of the rod withdrawal error using the 3D BWR Simulator:

1. How many of the 24 mesh point divisions which are available for use are actually used?

2. Is a full core calculation performed for an off-center rod or are quarter-or half-core symmetries sometimes assumed?

The multiplier to the APRM and RBM rod withdrawal block setpoints adjusts the setpoint downward in the event of plant operation with the operating maximum total peaking factor, MTPF, (for any fuel type) greater than the (respective fuel type) design maximum total peaking factor. The multiplier is of the form:

$$\text{Multiplier} = \text{Min}\left(\frac{\text{Design MTPF}}{\text{Operating MTPF}}\right),\ \text{i} = 1,\text{n (n = Fuel type)}$$

Provide a discussion in this section of the purpose of the multiplier. Provide a workable definition of the design Maximum Total Peaking Factor which can be used for a core reload configuration with up to three fuel types (7x7, 8x8, 8x8R) and is also consistent with the process computer programming. Discuss the algorithm used by the process computer to calculate the operating MTPF for each fuel type. Show that the definition is consistent with process computer function. Based on the design MTPF definition given, what adjustment to a proposed Technical Specification design MTPF will be required if the actual core loading (number of assemblies of each fuel type) differs from the reference core loading. Discuss the significance of cycle-to-cycle changes in the calculated design MTPF's as 8x8R reload assemblies replace assemblies of other fuel designs (8x8, 7x7). Show that these changes to the design MTPF values are consistent with the process computer program algorithm. Show that the MTPF definition and resulting multiplier give an adequate reduction in the RBM setpoints in view of the assumed initial (design) kW/ft of each type, the calculated ΔkW/ft for each fuel type as well as the kW/ft (corresponding to 1% strain) for each fuel type, for the most severe RWE.

Provide a similar discussion of the adequacy of this multiplier and the MTPF definition in connection with the adjustment to the High Flux Scram setpoint. Provide a statement that the required changes to these design MTPF's will be provided in the reload supplement.

II. The Control Rod Withdrawal Error (RWE) is a localized transient which can have a significant axial R-Factor variation when the rod block occurs to terminate the event. For a representative limiting RWE transient case provide the following information:

(a) R-Factor vs. axial location for 7x7, 8x8 and 8x8R fuel assembly types in a D-Lattice core for rod blocks occurring at 3.0, 5.0, 7.0 and 9.0 ft withdrawn.

(b) Axially average R-Factor, the minimum and maximum R-Factor for each fuel type and ft withdrawn case given in (a) above.

(c) For typical rated steady-state operating conditions using available test data, discuss the magnitude and trend in the GEXL correlation basis for rod blocks between 9.0 ft and 3.0 ft withdrawn in a C-Lattice core.

(d) Discuss the expected effects for a D-Lattice core relative to a C-Lattice core in connection with question (c) above. Address the R-Factor axial variation for the two lattice types for 8x8R fuel.

(e) Discuss any conservatisms not previously taken credit for in the GETAB methods which could serve to offset the under prediction of critical bundle power for the range of axial variation in R-Factor.

(f) Provide a table of the experimental data points and the corresponding GEXL predictions for the data given in Figure 9-10 of Reference 1.

RESPONSE:

I. A rod withdrawal error occurs when a reactor operator makes a procedural error and withdraws the maximum worth rod to its fully withdrawn position. This information was presented in the original submittal in the event description.

Rod Withdrawal Error calculations have been performed for a statistically significant number of normal operating conditions. These studies show that the effects of a rod withdrawal error are greatest at rated power and flow conditions. These studies also show that there is a lower bound for which a rod withdrawal error will not violate the safety limits. However, to provide conservatism in the analysis, all Rod Withdrawal Error analyses are assumed to start with an “on-limits” core at rated power and flow.

The RWE analysis is performed for 10 failure conditions representing all combinations of none, one and two failed strings. A composite response curve is formed which is the lowest response of each of the 10 failure conditions at each of rod withdrawal data points. The composite curve is used to determine the RBM setpoint. This insures that the worst responding failure is used throughout the rod withdrawal.

Under normal conditions, excessive bypass voiding does not occur. For plants which had plugged core plates, an additional analysis was performed to determine the response loss due to bypass voiding. This response loss was subtracted from the composite response curve before determining the RBM setpoint.

The standard tabulated data are provided in Section 10 of the Plant Supplemental Submittal, “Local Rod Withdrawal Error (With Limiting Instrument Failure) Transient Summary”. The Rod Block Monitor Reading at each rod position is presented, as is the change in CPR and the LHGR for each fuel type. The LHGR limits for each fuel type are given in the “Results and Consequences” as 13.4 kW/ft for the 8x8 fuel and 17.5/18.5 kW/ft for the 7x7 fuel. All 7x7 fuel operating in BWR/2,3 reactors listed in Table 1-1 have a MLHGR of 17.5 kW/ft and all 7x7 fuel operating in BWR/4 reactors listed in Table 1-l have a MLHGR of 18.5 kW/ft.

A statement has been added to the results and consequences paragraph to indicate that the maximum LHGR during this event will be reported in the Plant Supplemental Submittal.

A minimum of 24 axial mesh points are used for the neutronic and thermal hydraulics calculation. The LPRM readings used as input to the RBM channels are determined from the fluxes in the mesh points adjacent to the chamber. Therefore, eight are used for this part of the calculation. Full core calculations are performed for off-axis rods.

The purpose of the APRM flow-biased scram is to prevent fuel damage due to an abnormal operating transient from any point on the power flow map. The purpose of the multiplier is best illustrated by considering a transient from two different operating states. Assume a plant operating at rated power and flow with at least one fuel rod operating at its rated linear heat generation rate. By definition, the operating maximum total peaking factor is equal to the design maximum total peaking factor. If an abnormal operating transient with APRM flow-biased scram trip occurs from this state, the trip signal is generated when the core average neutron flux reaches 120% of rated and terminates the transient before the limiting fuel rod reaches its transient limit LHGR. Now, assume the same reactor is operating at 90% of rated thermal power, 100% of rated flow, and with at least one fuel rod operating at its rated linear heat generation rate. By definition, the operating maximum total peaking factor is equal to the design total peaking factor divided by 0.9. This requires that the APRM flow-biased scram trip be reduced from 120% to 108% of core average rated neutron flux. If the abnormal operating transient occurs from this state, the scram trip is again generated after a 20% increase in core average neutron flux and the margin to the transient limit LHGR is maintained. The multiplier ensures that a constant margin is maintained for all power distributions at constant flow.

The definition of the design maximum total peaking factor which is fuel type and plant dependent is:

$$\text{MTPF} = \frac{\text{MLHGR} \times \text{NBUN} \times \text{NRODS} \times \text{LF}}{\text{RP} \times \text{CHFF}}$$

where

MLHG	=	design linear heat generation rate limit (kW/ft);
NBUN	=	number of fuel bundles of all types in core;
NROD	=	number of active fuel rods per bundles of fuel being considered;
LF	=	active fuel length (ft);
RP	=	rated thermal power (kW); and
CHFF	=	cladding heat flux fraction.

The process computer is unaffected by the reload fuel design. Since the design total peaking factor is fuel type unique, it is unaffected by changes in loading pattern. Cycle-to-cycle changes require only the addition of data for new fuel types and deletion of data for types no longer in the core.

The APRM flow-biased rod block guards against exceeding the APRM scram trip point rather than guarding against the worst-case RWE. This is a function of the Rod Block Monitor, which takes its input signal from the LPRM strings surrounding the rod being withdrawn. Since the trip signals are taken from the local area where the error is occurring, no peaking factor multiplier is required.

The new design MTPF’s are provided in the application for changes to the Technical Specifications. However, General Electric is currently recommending a Technical Specification change which will define the APRM flow-biased scram and rod block setpoints in terms other than peaking factors. This will eliminate the need to redefine the peaking factor limit with every fuel change.

General Electric is recommending that the factor A/MTPF, which is used to adjust the APRM flow-biased trip settings, be replaced by F/MFLHGR. A is the design total peaking factor and may be different for each bundle type, MTPF is the maximum total peaking factor, F is the fraction of rated thermal power, and MFLHGR is the maximum fraction of limiting linear heat generation rate. The two

expressions are equivalent, but the former must be evaluated and checked for every bundle type; the latter requires only one calculation for the most limiting point in the core.

II. Response to this question was given in Reference 1.

REFERENCES:

1. Letter, Ronald E. Engel to Paul S. Check, "NEDE-24011-P-A: Axially Varying R-Factors", May 15, 1979.

REQUEST: Subsection 5.2.2 – Margin Improvement Options

(NOTE: The following requested *Margin Improvement Options* information does not represent a complete set of staff questions and concerns related to this subject, since their generic evaluation (except for Exposure Dependent Limits) is not within the scope of review of this Topical. Accordingly, acceptable responses to these questions should not be interpreted as acceptance, by the staff, of any or all of these options.)

Single recirculation loop operation during power operation is not yet approved for operating BWR's. When approved, it will have an impact on the reload reanalyses. If NEDE-24011-P is to provide a reference for the acceptability of reload cores, as they are affected by single loop operation, then a discussion of the effect of single-loop operation on normal operation, abnormal operational transients and accidents must be addressed. Include a discussion of the single-loop operation MAPLHGR reduction factor and the basis for the factor. Discuss the procedure used for correcting the ARPM and RBM flow-biased rod block settings and flow-biased scram setting, to account for backflow during single-loop operation. Reference should be made to other generic or lead plant evaluations where appropriate. Discuss single-loop operation as it relates to the reload stability analyses. Discuss the effect of single-loop operation on the total core flow relative to active recirculation flow and the flow uncertainty. Relate these effects to the generic calculations of the fuel cladding integrity safety limit MCPR.

RESPONSE:

Single-loop operation is not a MCPR margin improvement option. Although the reactor is assumed to be operated at some reduced flow and power (on or below the rated flow control line) during single-loop operation, the operating limit MCPR corresponding to the core flow is no different during single-loop operation than it is during two-loop operation. Change in safety limit MCPR is negligible (see Reference 1).

Single-loop operation will not have an effect on reload analysis and is therefore not addressed in the generic reload fuel application licensing topical report. Single-loop operation is addressed in licensing amendment submittals for individual BWR operating plants. For example, see the lead plant licensing submittal in Reference 1.

REFERENCES:

1. "Pilgrim Nuclear Power Station Unit 1 License Amendment for Single-Loop Operation with Bypass Holes Plugged," NEDO-20929, January 1976.

REQUEST: Subsection 5.2.2.4 – Thermal Power Monitor

Provide the functional form of the TPM Transfer Function. Discuss the manner in which the time constants and any other parameters entering into the formulation are determined for a given plant. Explain the physical significance of all terms and/or factors in the transfer function. Discuss the performance (response of the TPM) for normal and abnormal events. Provide the calculated flux trip level vs. flux frequency (e.g., Turbine Trip w/o BP to LFWH). Discuss the influence (if any) of reloads on the values of the parameters in the TPS transfer function.

Provide examples of sources of "spurious" and "momentary flux spikes" referred to in the second paragraph, which the system is designed to filter out. Give the frequency of these events.

RESPONSE:

The thermal power monitor (TPM) transfer function is that of a low pass RC filter, where R is resistance and C is capacitance. Resistance and capacitance values were selected which result in a 6-sec (±l sec) time constant. This RC filter is shown on APRM scram trip electrical diagrams for plant-specific installations. The RC filter time constant of 6 sec envelopes the fuel time constants for 7x7, 8x8 and 8x8R fuel pins, which are in the range of approximately 7 to 10 sec.

The performance of the APRM simulated thermal power trip (thermal power monitor) is discussed in a revision to Subsection 5.2.2,4. Examples of events which could cause spurious scrams due to momentary flux spikes are also given in this revision. The frequency of these events may be found in the operations logs of the BWR operating plants.

From the above information, it is seen that the operation of the TPM is independent of the fuel loading in the core. Also, no normal, abnormal, or accident events, other than the loss-of-feedwater heater transient, are affected by the operation of the TPM.

REQUEST: Subsection 5.2.2.5 – Exposure Dependent Limits

It is understood that most of the initial conditions and other input parameters for the GETAB transient analysis are exposure dependent, while others are conservatively assumed to be at their most adverse values during the cycle. Indicate which parameters, if any, are taken at their most limiting values during the cycle. Discuss where these assumptions may vary depending on the transient event or plant considered. Include the fuel loading error.

The licensing calculations for the n+l cycle are performed significantly before the end of cycle n. This results in the necessity of estimating the cycle (n) burnup in order to calculate the EOC-(n+l) exposure increment. Thus, the actual n+l cycle core incremental exposure at which the all-rods-out end-of-cycle conditions is attained will, in general, be different from the predicted exposure increment. Furthermore, model uncertainties will also contribute to a difference between the actual and the predicted EOC exposure for cycle n+l. Since the exposure-dependent limits are referenced from EOC, which is not precisely determined until it is actually achieved, provide, in sufficient detail, the manner in which the exposure-dependent operating limits will be conservatively implemented during plant operations. Discuss the procedures to be used by the reactor operator to make exposure corrections to determine the actual EOC exposure, in order to establish when in the cycle the operating MCPR limits should change.

RESPONSE:

The GETAB transient analysis initial condition parameters that are taken at their most limiting values during the reload cycle are the nonfuel power fraction, the core flow, the reactor pressure, and the coolant inlet enthalpy. The limiting values of these parameters are the same for all transient events analyzed, but are plant specific, as indicated in Table 5-6 (page 5-65).

The fuel loading error analysis uses the same GETAB transient analysis initial condition parameters that are used in the GETAB analysis of abnormal operational transients.

At end of cycle n, the details of the actual end-of-cycle exposure distribution are known. The exact BOC n+l core configuration is then input to the three-dimensional BWR simulator mode. Using expected full-power conditions for cycle n+l, the cycle n+l exposure capability, E_{n+l} is calculated by Haling depletion from this model of the actual BOC n+l core.

Early in cycle n+l, the calculated exposure capability, E_{n+l} is documented and is sent to the reactor operator. If a change in operating limits is scheduled to occur at an exposure of EOC-x, the reactor operator implements that change when cycle n+l core average exposure accumulation reaches E_{n+l}—x, as monitored by the process computer.

The exposure-dependent operating limits are conservative, as implemented, for two reasons: (1) the operating limits are determined by conservative analytical procedures, and (2) the method of implementation is conservative. Each of these statements will be expanded below.

When analyzing transient performance at exposures prior to the end-of-cycle (EOC), all-rods-out condition, it is necessary to consider the effect of the control rods on the transient parameters, because the scram reactivity and the dynamic void coefficient are sensitive to the control rod pattern. At any given exposure point, there are many control rod patterns which will render the core critical and within thermal limits. To ensure that conservative values of the important dynamic parameters are calculated, it is necessary to select special control patterns. Conservative values of both the scram reactivity and dynamic void coefficient result when “black-white” control patterns are used. A black-white control pattern is one in which control rods are either fully inserted (black), or fully withdrawn (white).

The scram reactivity is minimized with black-white patterns because:

(1) the fully inserted control rods provide no contribution to the scram reactivity,

(2) the fully withdrawn control rods begin their insertion in a region of zero power; thus, their impact during the early portion of the scram is minimized; and

(3) there are no partially inserted control rods, which generally provide a major contribution during the early portion of the scram.

The magnitude of the dynamic void coefficient is maximized because the core average axial power distribution is shifted toward the bottom of the core with black-white control rod patterns, and as a result the core average void fraction is increased. Since the dynamic void coefficient is directly proportional to the core average void fraction, its magnitude is therefore increased.

The black-white rod patterns are always used to generate the dynamic parameters at midcycle exposure points, unless it is not possible to meet thermal limits with these patterns. If thermal limits cannot be met (usually due to high axial peaking in the bottom of the core, which exceeds the

Technical Specification limit on linear heat generation rate), the black-white patterns are modified because the operating reactor would be prevented from operating with such patterns by means of its Technical Specification limits. Modifications to the black-white pattern are made by slight withdrawal of fully inserted rods and shallow insertion of withdrawn rods. Thus, a critical rod pattern which meets thermal limits is attained while minimizing the deviation from the black-white pattern.

The conservative nature of the limits, as implemented, can be shown most effectively by means of an illustration. In Figure 5.2.2.5-1, the MCPR at two intermediate exposures and at EOC have been plotted as a function of exposure. The solid circles represent the results derived from the black-white rod patterns, and the open circles represent limits derived from the nominally expected rod patterns. The MCPR limit varies smoothly between these points, as shown in the figure.

The Technical Specification limit on MCPR, however, is applied as a histogram, rather than as a smoothly varying function of exposure. The MCPR limit during any exposure interval is determined by the most limiting value at either end of that interval. For the example shown in Figure 5.2.2.5-1, the Technical Specification limit would be applied as shown by the heavy horizontal lines.

Now, if for some reason the projected EOC exposure differs from that which is actually attained, either of the following would result:

(a) The actual EOC exposure is greater than that projected. In this case, the step increases in MCPR limits shown in Figure 5.2.2.5-1 would occur sooner than actually required, and there is no potential violation of limits.

(b) The actual EOC exposure is less than that protected. In this case, the step increases in MCPR would occur later than required, and there is a potential violation of limits. This situation is illustrated in Figure 5.2.2.5-2. Note that there are small exposure intervals just prior to the step changes in limits for which the Technical Specification limit is slightly less conservative than that derived from the black-white patterns. However, the difference between these two MCPR limits is small. Also, note that the Technical Specification limit provides margin to the limit derived from the nominally expected rod patterns for reasonable deviations for end-of-cycle exposure.

Therefore, unless the projected and actual EOC exposures differ by a large amount, and differ in the non-conservative direction, there would be a negligible nonconservative difference between the Tech Spec limit and that required by the black-white rod patterns. Exposure differences of this magnitude are unlikely since the projected value used to determine MCPR limits is based upon the as-loaded core for the cycle of interest.

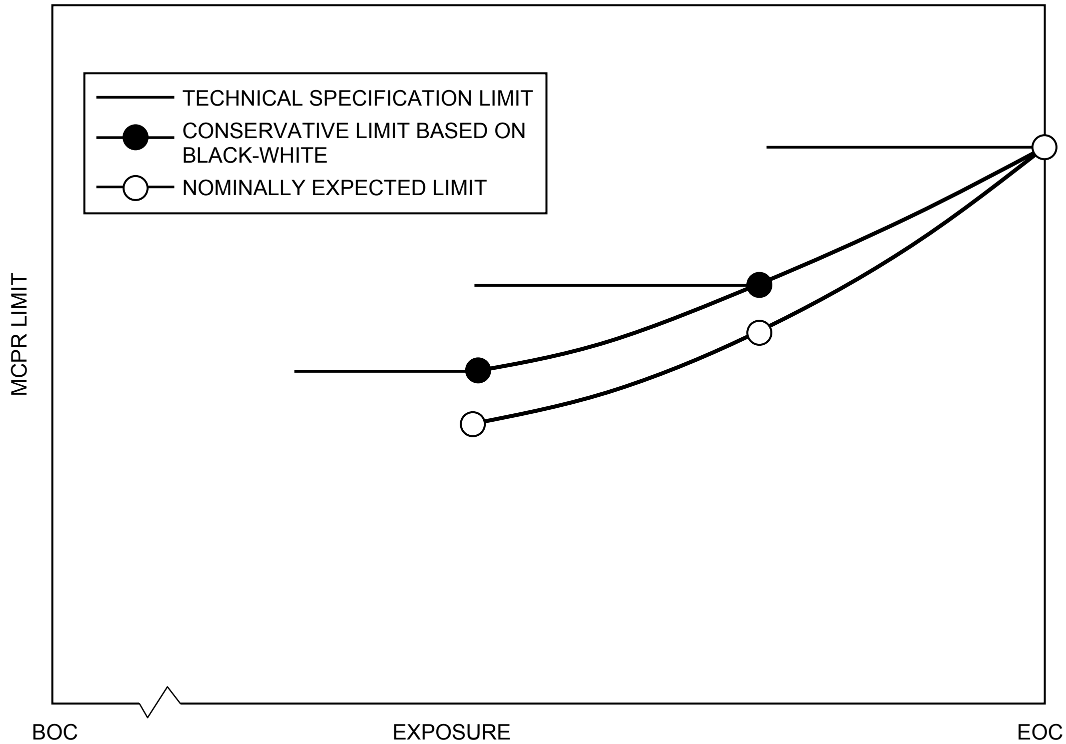

Figure 5.2.2.5-1. Typical Comparison of Exposure-Dependent MCPR Limit to Nominally Expected Limit

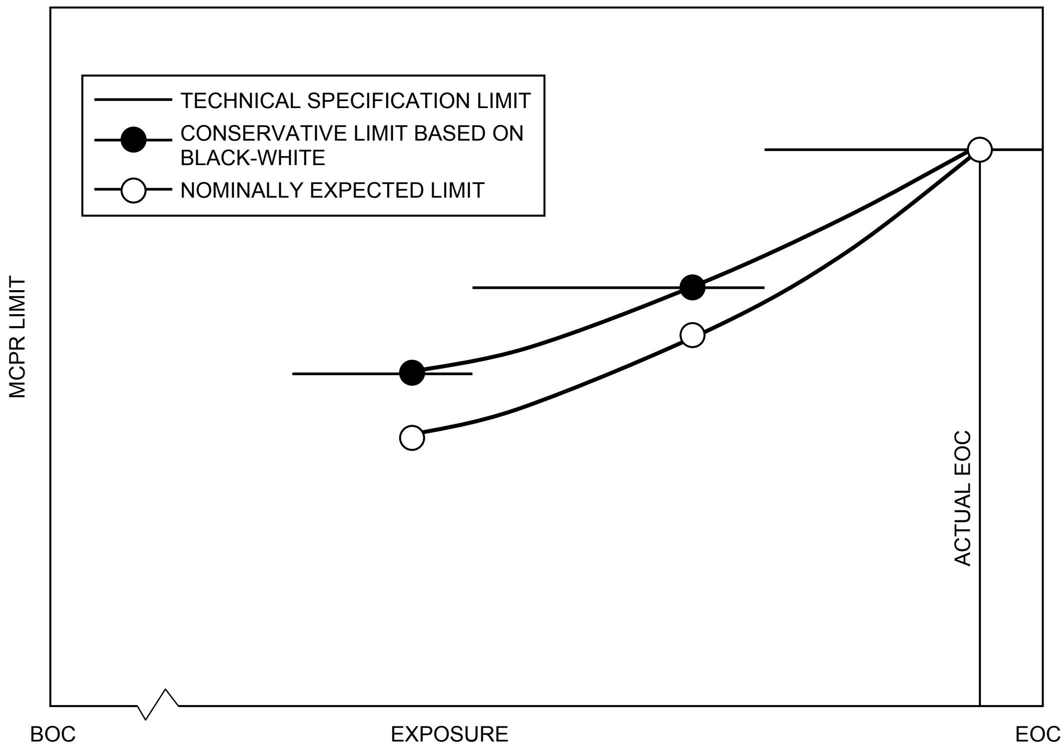

Figure 5.2.2.5-2. Typical Comparison of Calculated Exposure-Dependent Limit to Technical Specification Limit for EOC Exposure Less Than Projected

REQUEST: Subsection 5.2.3 – Effect (of Fuel Densification) on Local Power

A more explicit reference and detailed description of the applicability of the integral hypothesis is required. The same applies to the critical quality-boiling length to integral hypothesis. Identify reference more accurately (e.g., pg. 3-3 of Reference 5-1).

D.H. Lee in the British report AEEW-R-479, "An Experimental Investigation of Forced Convection Burnout in High Pressure Water, Part 4...," pages 11-16, extended the F factor correlation, such that C is a function of flux profile, pressure, hydraulic diameter, and mass flux. Provide a sensitivity study over the range of BWR conditions, with this extended F factor method, for an estimate of critical power reduction due to the presence of spikes.

RESPONSE:

There are three typographical errors on page 5-26. Line 1: Reference 5-13 should be Reference 5-11. Equation preceding Equation 5-2: There should be a coefficient C in front of the first integral. Line 2,

last paragraph: q''_o / a''_1 should be q''_o / q''_1. The latter error was corrected in Amendment l; the first two have been corrected with this amendment.

According to the Integral Concept as used in connection with boiling transition (BT) correlations and predictions for a given channel: the occurrence of BT at some point along the channel for a given pressure, mass flux, and inlet subcooling, depends in some manner upon the heat flux profile over the entire boiling length upstream of that point. (This is in contrast to the Local Conditions Hypothesis, according to which the occurrence of BT at some point depends upon the heat flux at that point.)

The Integral Concept is also known as the Integral Method, Integral Hypothesis, Upstream Memory Effect, or just Memory Effect. This concept is applied in the Tong F-factor scheme (Reference 5-11 of the report and References 1 through 3 below), the Critical Quality vs. Boiling Length scheme (e.g., GEXL)

(References 5-1 and 4 below), and an Equivalent Critical Boiling Length scheme due to Silvestri (References 3 and 4). It is probably best described in Reference 3, pages 99 ff.

In Subsection 5.2.3.1, the expression “Integral Hypothesis is reserved specifically for the critical Quality vs. Boiling Length scheme, and it will be so used here. The applicability of the integral hypothesis to BWR operating conditions is described in References 5-1 and 3. In Reference 5-1, page 3-3, B&W round tube data for various axial profiles are shown to correlate well in the X_C vs. L_B plane. The diameter (0.45 in.) and quality range (10% to 40%) correspond closely to BWR conditions. Similar results were obtained with GE Freon annulus data and British round tube data, always at conditions similar to BWR conditions.

In Reference 3, page 99, data are described which were reported earlier by DeBortoli, et al (Reference 5). These data are for two test sections, the first section having uniform axial profile, and the second the same except for the addition of a short hot patch at the exit end. The data demonstrate that the Local Conditions Hypothesis is only valid for boiling transition (BT) at high subcooling. Reference 3 further points out that the X_C vs. L_B scheme is valid, and equivalent to the Tong F-factor method, for BWR conditions.

In summary, the X_C vs. L_B scheme (i.e., the Integral Hypothesis) has been shown to be applicable to BWR fuel assemblies with various axial profiles because it correlates both simple channel data at BWR conditions and full-scale simulated fuel bundle data at BWR conditions.

A procedure for evaluating the constant C in the Tong F-factor scheme is given in Reference 6 (also reported in Reference 2); in terms of S, the axial flux gradient at the point near the exit end of the heated length for which local flux = average flux (normalized with respect to average flux and diameter to render it dimensionless); P/A, the ratio of peak-to-average flux; D, the channel (hydraulic) diameter; and K, a pressure dependent factor. Thus,

$$C = \frac{KS}{D\,(P/A)} \tag{5.2.3-1}$$

To assign a limiting value to C, the extreme BWR profile of Figure 5.2.3-1 has been selected for which S/(P/A) = 0.015 (no BWR profile is expected to yield a higher value of S/(P/A)). The other factors on the right of Equation 5.2.3-1 are K = 5.0 (corresponds to P = 1000 psia, see Reference 6), and D = 0.5 in. (typical for BWR’s as noted in the report).

$$C = \frac{5.0}{0.5} \times 0.015 = 0.15 \text{ in.}^{-1}$$

When this value of C is inserted in Equation 5-3 of the report, there results:

$$\frac{q_0''}{q_1''} = 1.00291$$

Following the reasoning given on page 5-27 of NEDE-24011-P relative to application of the Tong F-factor method, the critical power with flux spiking is reduced only 0.25% for C = 0.15 in.$^{-1}$. Obviously, the value suggested by Tong

$$C = \frac{0.135}{D} = 0.27 \text{ in.}^{-1}$$

and used in Subsection 5.2.3.1 is more conservative. According to this, the critical power is reduced 0.47% due to the presence of flux spiking.

REFERENCES:

1. L.S. Tong, Boiling Heat Transfer and Two-Phase Flow, reprinted 1975 by R.E. Krieger Publishing Co., Inc., New York (Chapter 6).

2. J.G. Collier, Convective Boiling and Condensation, McGraw-Hill Book Co. (UK) Ltd., Maidenhead, Berkshire England, 1972 (Chapter 9).

3. R.T. Lahey, Jr., and F.J. Moody, The Thermal Hydraulics of a Boiling Water Nuclear Reactor, American Nuclear Society, Hinsdale, Illinois, 1977 (Chapter 4, Sec. 4.3).

4. M. Silvestri, “On the Burnout Equation and on Location of Burnout Points, Energia Nucleare, V. 3, N. 9, September 1966.

5. RA. DeBertoli, S.J. Green, B.W. LeTourneau, M. Troy, and A. Weiss, “Forced-Convection Heat Transfer Burnout Studies for Water in Rectangular Channels and Round Tubes at Pressures Above 500 PSIA,” WAPD-188, Oct. 1958.

6. D.H. Lee, “An Experimental Investigation of Forced Convection Burnout in High Pressure Water; Part IV, Large Diameter Tubes at About 1600 P.S.I.” AEEW-R479, November 1966.

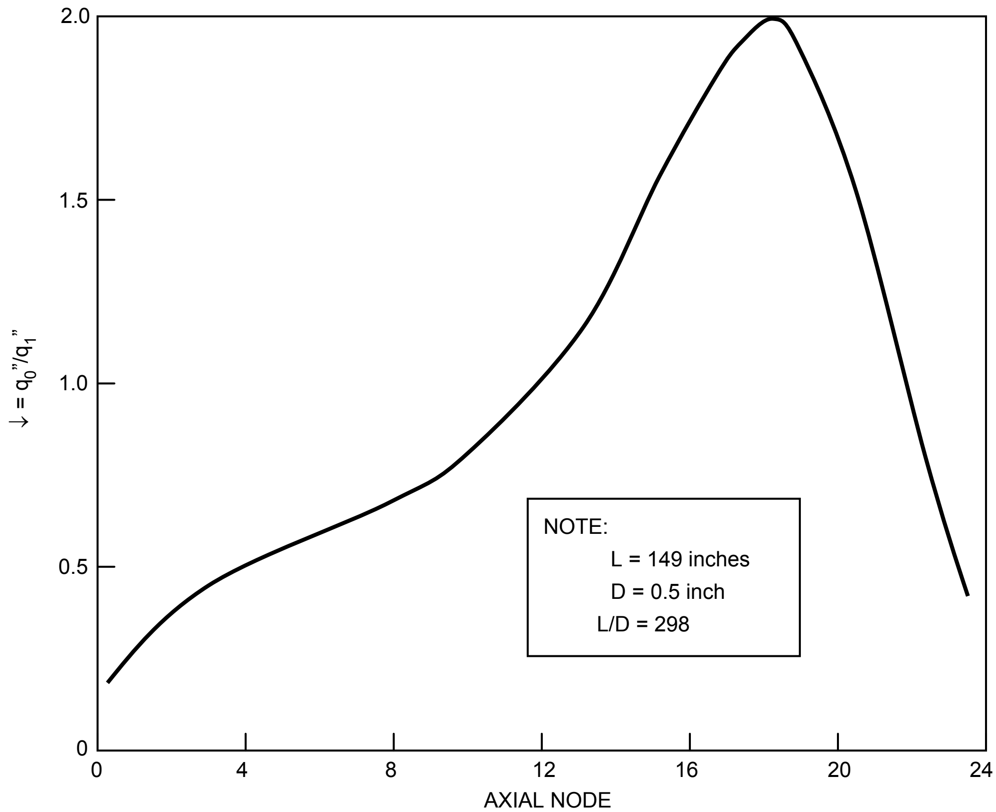

Figure 5.2.3-1. Extreme Profile

REQUEST: Subsection 5.3 – ASME Code Compliance for Vessel Pressure

Provide the assumed fast closure rate (position vs. time) of the MSIV for the event.

Indicate the reference where the MSIV fast closure event is demonstrated to be more severe than the Turbine Trip without Bypass when credit is not taken for the direct scram.

The initial core power levels indicated in Table 5-6 for the BWR/3 plants do not include an increment to account for the core thermal power measurement uncertainty. This is unacceptable for the analyses of this (MSIV closure) limiting pressurization event. Revise Table 5-6 so that the assumed initial steady-state core thermal power level is at least 102% of the licensed power level for this limiting event for all plants (in accordance with the requirements of Standard Review Plan 5.2.2).

Provide a statement regarding when in the cycle the event is most limiting.

Provide or reference the model used to calculate the peak pressure at the bottom of vessel from the transient model output.

Provide a statement as to the assumed performance of the control systems.

Discuss the methods used, assumptions and effects of ATWS, RPT for those plants which have it. List these plants in a table.

RESPONSE:

The MSIV closure event has been demonstrated in the past to be more severe than the turbine or generator trip without bypass when credit is not taken for direct scram. An illustration of the typical relationship between these events is shown in Figure 5.3-1 (attached). Limiting events which are affected by a scram trip are generally most limiting at the end of the cycle due to the dominating effect of the scram degradation in this portion of the cycle because the control blades are withdrawn from the core.

The model used in transient analyses is described in Reference 5-2 of the report. The flow control systems are considered for transient analyses in the manual mode unless otherwise specified. The manual mode presents the most severe conditions for events effected by core flow. Other control systems, such as the feedwater and pressure regulator, are considered in their normal controlling mode.

The power levels indicated in Table 5-6 are consistent with the power levels given in plant technical specifications and used as the basis for the safety analysis. The General Electric basis for code overpressure protection analysis has been documented in Reference 5-14. As noted in the response to the request on Subsection 5.2.1, GE specifically addressed the 2% adder to the initial power, to all transient analysis in Reference 1. Figure 5.3-2 shows the assumed closure rate used in this analysis.

The method of applying ATWS RPT, corresponding to the current design, is to trip the recirculation system after dome pressure has reached the specified setpoint. The assumptions considered are that: (1) trip occurs at the normal setpoint; (2) a 300-msec delay to account for breaker and logic systems delays exists; and (3) the minimum recirculation system inertias are used. The effects of ATWS RPT on transients analyzed for reload licensing generally result in an increase in peak vessel pressure and a reduction in peak neutron and average surface heat flux. No credit is taken for the mitigating effects of ATWS RPT system in the establishment of thermal limits. All BWR/4 plants, except Vermont Yankee, have installed the ATWS RPT. This information is documented in new Table 5-17.

REFERENCES:

1. Letter, R. L. Gridley (GE) to D.G. Eisenhut (NRC), MSIV Closure with Flux Scram Sensitivity of Peak Vessel Pressure to Initial Power Level, September 12, 1977.

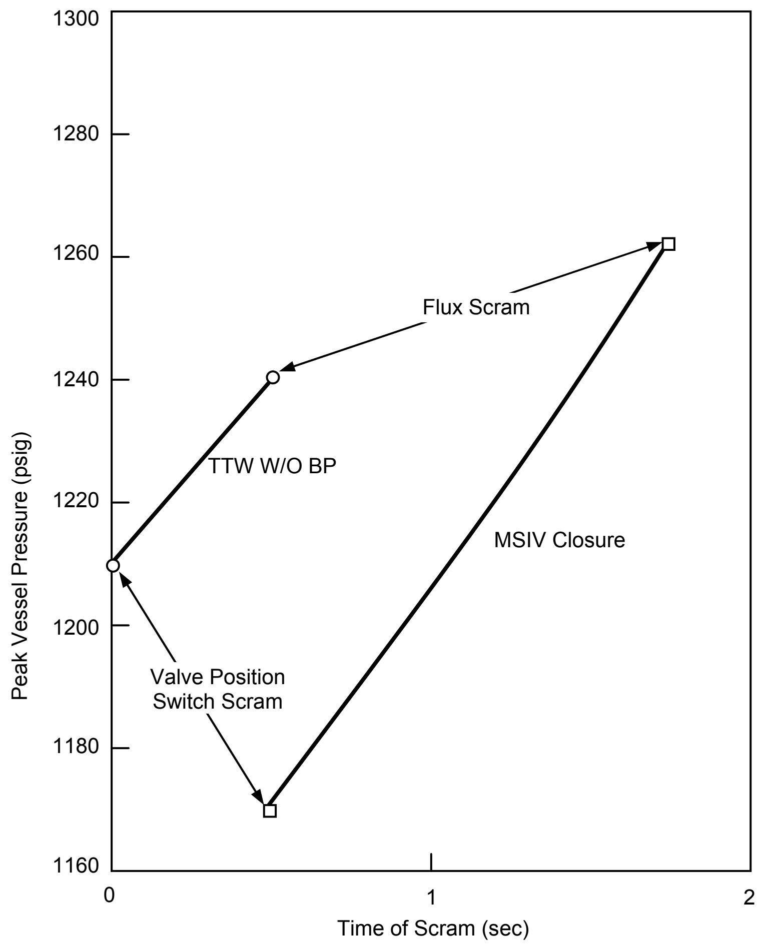

Figure 5.3-1. Effect of Scram Time on Peak Vessel Pressure

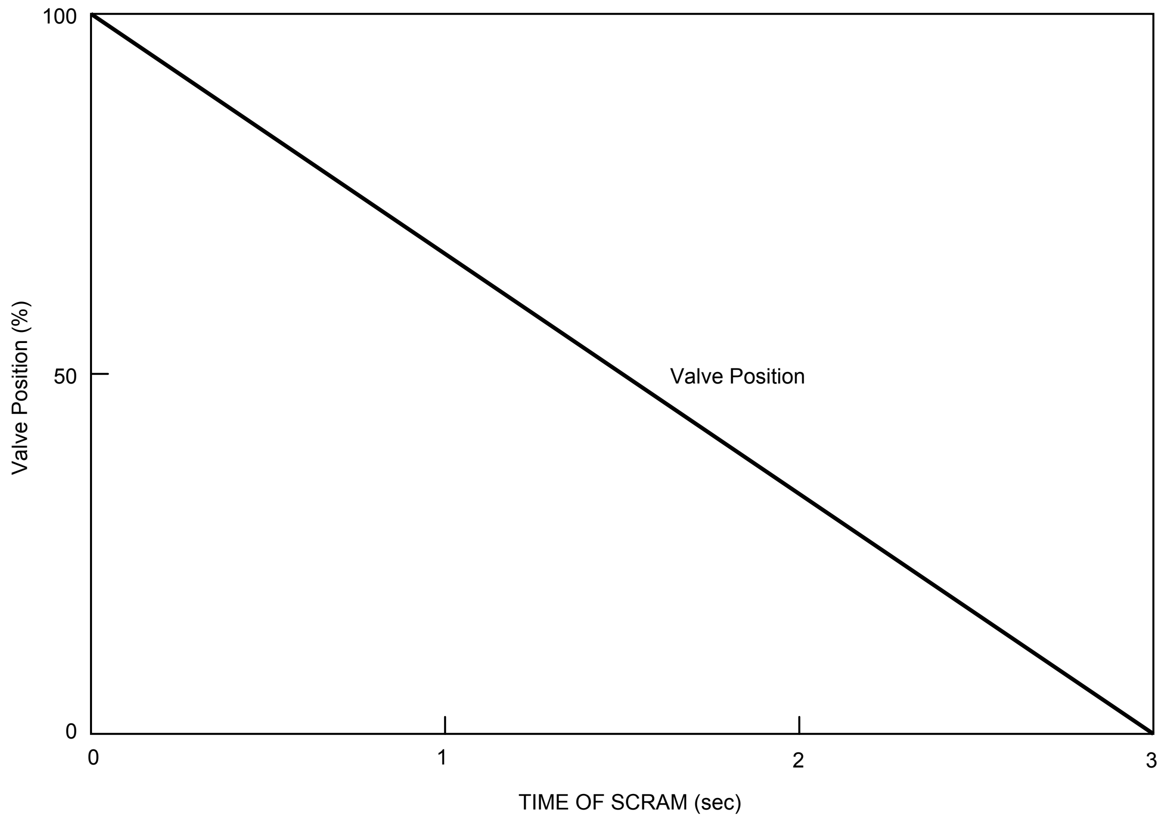

Figure 5.3-2. MSIV Closure

REQUEST: SubsectIon 5.4 – Stability Limits

Indicate by block diagram the important reactivity mechanisms that affect the thermal-hydraulic stability characteristics of a BWR.

Provide a comparison of the new two-water-rod 8x8R fuel design to the old single-water-rod 8x8 fuel design as it relates to total core and channel stability performance. Include the effects of the differing fuel length and diameter (time constant as well as the void and Doppler coefficients.

Discuss the methods used to determine the required nuclear parameter inputs for the stability codes.

Provide a discussion of the basis (in terms of fuel thermal-mechanical integrity) for the selection of a decay ratio of 1.0 as the “Acceptable Performance Limit.”

The discussion in Section 5.4 implies (by omission) that reload cores will no longer be evaluated from the standpoint of an “operational design guide decay ratio.” Either justify the elimination of this

stability criteria for reloads or include a discussion and basis of the operational decay ratio requirements.

Provide a general discussion of the location of the least stable state and why some plants may have it located on the rod block line, while, for other plants, it is located on the 105% rod line.

Provide a set of representative figures which quantitatively maps the limiting lines on the power flow map (natural circulation, minimum recirculation flow and rod block/105% rod lines) onto the decay ratio vs. percent power plot. Provide a general discussion of the mapping. Include the most limiting points and their respective limits.

RESPONSE:

The purpose of the stability analysis is to demonstrate analytically that the acceptable performance limit (i.e., the decay ratio is less than 1.0 or the damping coefficient is greater than 0) is satisfied. This limit was selected because it was considered prudent not to operate a plant in a potentially oscillatory mode. In Reference 1, it is demonstrated that there is considerable margin between plant operation with a limit cycle (decay ratio equal to 1.0 and previously established safety limits which are in themselves conservative.

The purpose of the reload stability analysis is to demonstrate the capability for safe plant operation. This is done by demonstrating for all fuel types that the acceptable performance limit is satisfied for each type of stability analyzed. Because all fuel types are analyzed for each reload, no comparison between 8x8 and 8x8R is considered necessary. Also, it is not considered necessary to provide an evaluation of the automatic flow control range where the operational design guide decay ratio is less than 0.25 because it is only a guide used for operation with automatic flow control.

For reload cores, the stability analysis with a linearized analytical model is described in Reference 2. A block diagram of the important reactivity mechanisms which affect thermal-hydraulic stability is shown in Figure 5-6 of Reference 2. The methods used to develop the nuclear feedback characteristics are given in Reference 3.

Stability analyses are performed along the rod line corresponding to the highest power level for which the safety analyses are performed. This power level is documented in the plant technical specifications. A power flow map (natural circulation, minimum circulation flow and rod block and safety analysis rod lines) is given in each plant final safety analysis report. A typical power flow map is shown on Figure l-5 of Reference 2. The location of the least stable point flow and the highest attainable power level is shown.

REFERENCES:

1. Letter, R.E. Engel to D.G. Eisenhut, “Boiling Water Reactor Stability Margins,” April 4, 1977.

2. “Stability and Dynamic Performance of the General Electric Boiling Water Reactor,” January 1977 (NEDO-21506).

3. “Generation of Void and Doppler Reactivity Feedback for Application to BWR Design,” December 1975 (NEDO-20964).

REQUEST: Subsection 5.5.1 – Control Rod Drop

According to Section 15.4.9 of the Standard Review Plan, one of the CRD fuel failure thresholds to be considered is fuel cladding dryout (MCPR <1.06) when the event occurs from an initial rated power condition. Provide a bounding assessment of the number of 8x8R failed fuel rods which would occur as a result of a CRD from rated conditions. Show that a CRD from the hot full-power operating condition gives less severe radiological consequences than one from the hot or cold shutdown conditions for all the plants listed in Table 1-1. Show that the calculated doses for this situation are also well within the 10CFR Part 100 exposure guidelines.

Provide a statement that the tabulated scram times used in the CRD analysis equal or exceed the technical specifications scram times for the plants considered.

Reference or provide a generic sensitivity study which identifies the key input parameters to the CRD accident (e.g., provide a table of all of the input parameters considered and their effects on the consequences of the CRD accident).

In Equation 5-14 the "+" should be replaced by an "=".

In the definition of BP, "ith" should be replaced with "jth".

Discuss the analyses which conclude that the closest approach of actual plants operating parameters to the 280 cal/gm is represented by Figure 2-25.

A maximum interassembly local peaking factor, P_F, of 1.3 forms the basis for the maximum allowable rod worth of 1.3% ΔK and is cited in the Technical Specifications of several operating BWRs. Describe the procedure (analysis and/or Tech Spec change) which will be used in connection with a reload submittal for one of these plants for the case in which a specific plant's accident reactivity, Doppler coefficient and scram reactivity conservatively fall within the bounding CRDA analysis, but the maximum interassembly local peaking factor, P_F, is greater than 1.3.

With regard to the interassembly local power peaking factor, P_F:

(a) Describe in more detail the methods used to calculate P_F as outlined in Equation 5-14.

(b) Is P_F to be calculated on a plant-cycle specific basis and provided with each reload submittal?

(c) If not, justify the statement that a P_F greater than 1.30 would not be expected to occur at any plant.

In order to base a reload license application on the bounding rather than a plant-specific control rod drop analysis, it is necessary that the plant-specific scram reactivity be within the bounding limit up to a total of 0.02 ΔK (pages 5-46 and 5-47). Do you propose that plant specific analyses are not necessary for cases where the above criteria is met but for which the plant specific scram is outside the bounding limit above ΔK = 0.02?

Justify the adequacy of global values for Doppler Coefficients and delayed neutron fractions in analyzing the control rod drop accident for reload cores containing highly exposed fuel bundles. Take into consideration the possibility that locally the delayed neutron fraction of Doppler coefficient may be smaller in magnitude than the global values.

Either justify that ß will never be less than the limiting values assumed for bounding analyses, or state clearly in the text that ß will be calculated and compared to the limiting value for each reload.

There appears to be all inconsistency between the wording of item (2)-(C) on page 5-38 and item (2)-(C) on page 5-39 with regard to the final position of the dropped rod. Please clarify this situation.

Provide a discussion of the bounding analysis methods which will be used in connection with plants using group notch withdrawal sequences.

Provide a discussion of the models methods and assumptions used to demonstrate that the maximum reactor pressure during any portion of the CRD is less than that which would cause the emergency condition stress limits (as defined in the ASME Boiler and Pressure Vessel Code, Section III) to be exceeded. Provide or reference a bounding or a plant-specific analysis (or a previously submitted plant analysis) which demonstrates that the criteria will be met by the reload fuel.

RESPONSE:

The study "Control Rod Drop Accident at Rated Power" by R.C. Stirn and C.J. Paone was submitted under cover of Reference 1. This study, utilizing the Hench-Levy Critical Heat Flux correlation, showed no fuel rods in transition boiling or reaching the 1% plastic strain linear heat generation rate.

The Technical Specifications are assumed to be an integral part of the licensing basis for each plant. The value(s) assumed for any parameter used in licensing calculations will be at least as conservative as the Technical Specifications value. The only exceptions are those cases in which the submittal is intended to justify a request for a technical Specification change.

The key input parameters to the control rod drop accident which influence the resultant peak fuel enthalpy are defined on pages 5-45 and 5-46. These parameters are the accident reactivity shape functions, total control rod worth, assembly local power peaking factor, delayed neutron fraction, Doppler reactivity feedback and rod drop velocity. The sensitivity of the control rod drop accident resultant peak fuel enthalpy is already documented in this report and References 5-17, 5-18 and 5-19.

Equation 5-14 was corrected in Amendment 1. The typographical error in the definition of BP was also corrected. Figure 5-25 shows only the reactivity shape functions for the bounding analyses and a typical not-bounding actual plant case. Calculations of the type used to generate Figures 5-22, 5-23, 5-24 and 5-25 are discussed in detail in References 5-17, 5-18, and 5-19.

Bounding curves for higher P_f were included in Amendment l. This amendment further stated that, because so many parameters are involved in the determination of the resultant peak fuel enthalpy due to a control rod drop accident, it is not realistic to set a specific value of maximum control rod worth that could be used in Technical Specifications. In the past, a local peaking factor of 1.3 was applied as the upper limit and, based on this local peaking value, a "maximum allowable" control rod worth of 1.3T Δk was set. In reality, some reload cores exceeded both the 1.3% Δk value for rod worth and the l.3 local peaking factor value, yet met all the boundary requirements. Therefore, no specific control rod worth requirement will be set other than those described above" (i.e., accident, reactivity, Doppler coefficient, and scram reactivity).

The bounding value for the Doppler reactivity coefficient given in Figure 5-26 was based on the beginning of life BOL or zero exposure values as stated on page 5-46. It is also stated on this page that this is conservative, since the Doppler coefficient will become more negative with increasing

exposure. By comparing Figure 5-26 with Figure 3-5, it will clearly be noted that the Doppler coefficient does become more negative with increasing exposure. The bounding values selected for Figure 5-26 were based on 7x7D initial core fuel. Figure 2.3-3 of Reference 5-5 clearly shows that this is the most limiting BWR lattice design with regards to Doppler reactivity. Even at the beginning of a fuel cycle, some of the four fuel bundles surrounding a control blade in a reload core design will be exposed; hence, the composite Doppler coefficient of the four bundles surrounding the blade will be more negative than the bounding values in Figure 5-26.

A bounding value of 0.005 was selected for the delayed neutron fraction. Because high worth control blades occur where at least one fresh fuel bundle has been loaded as one of the four fuel bundles surrounding the control blade, the delayed neutron fraction in a local region surrounding the blade will always be greater than the conservative bounding value of 0.005 even at end of cycle.

The bounding analysis methods are not dependent on the type of plant being analyzed. While the specifics may differ in such areas as rod patterns considered and distance a rod is allowed to drop, the basic calculations are identical. Inconsistencies in documentation in this subsection were corrected in Amendment 1.

Reference 2 discusses the maximum pressure occurring during a CRD relative to applicable stress limits. A copy will be transmitted as soon as it is available (first quarter 1978). Preliminary results indicate a maximum pressure increase of Less than 15 psi for this accident.

REFERENCES:

1. Letter, I.F. Stuart (GE) to V. Stello (NRC), "Analysis of Control Rod Drop Accident at Rated Power," December 7, 1973.

2. "Transient Pressure Rises Affecting Fracture Toughness Requirements for Boiling Water Reactors," T.E. Cooke, et al., to be issued.

REQUEST: Subsection 5.5.2 – Loss-of-Coolant Accident

Subsections 5.5.2.1 and 5.5.2.2 are missing; provide these subsections.

RESPONSE:

Documentation of the loss-of-coolant accident analytical methods can now be found in Subsection 5.5.2.

REQUEST: Subsection 5.5.4 – Fuel-Loading Error

The discussion of the fuel-loading error is inadequate. Provide a detailed discussion of the models methods and assumptions which are employed in the FLE analysis. Include a discussion of the procedures used to determine the most severe fuel loading situation vs. location in the reload core. Discuss how the fresh fuel reactivity effect is maximized. Discuss the assumption on the CPR value of the adjacent monitored bundles. Discuss which of the analysis inputs are plant specific and which incorporate conservative or bounding inputs. Discuss the assumptions used to determine the local

peaking factors and R-factor; e.g. fuel type (enrichment), water gap thickness. Discuss how fuel densification effects are incorporated into the evaluation of the peak LHGR. Discuss which analysis (misoriented or mislocation) is performed for each reload. Discuss whether separate analyses (where applicable) are performed for a postulated mislocation of a fresh (8x8R) assembly in a location intended for: (a) an exposed 7x7 location; (b) an exposed 8x8 location; and (c) an exposed 8x8R location.

RESPONSE:

The analytical models and assumptions used in the analysis of the fuel assembly loading error are described in References 1 and 2. These descriptions include the various alternative assumptions with suitable bases which can be used to reduce the undue conservatism in the most conservative methods. The above information has been incorporated into Subsection 5.5.4.

REFERENCES:

1. Letter, R.E. Engel (GE) to D.G. Eisenhut (NRC), "Fuel Assembly Loading Error", June 1, 1977.

2. Letter, R.E. Engel (GE) to D.G. Eisenhut (NRC), "Fuel Assembly Loading Error", November 30, 1977.

REQUEST: Appendix A – Supplemental Reload Licensing Submittal

The following additional information should be provided in Appendix A.

2. FUEL MECHANICAL DESIGN

Provide a brief statement to the effect that all of the generic thermal-mechanical design analyses were checked for applicability to the plant-specific reload application, in accordance with the provisions of Subsection 2.4.3. State the results and conclusions of this review in regard to the predicted reload fuel performance during life and the applicability of the generic calculations. Reference should also be made to generic tables appearing in the Topical (see comments relating to Subsection 2.4.2, particularly the last three sentences of the first paragraph.

3. NUCLEAR EVALUATION

Indicate the exposure corresponding to the reported value of "R". Provide the assumed core average exposure increment "window" for the previous cycle.

5. REACTOR LIMITS

This section should provide a summary paragraph giving the Operating Limits, to be followed by the paragraphs containing the Inputs and Results of the accident and transient evaluations. That is, replace (a) operating MCPR Limit, 5.1, by (a) Operating Limits. Include in paragraph (a):

1. Operating MCPR for each fuel type and exposure (if applicable).

2. Maximum Total Peaking Factor for each fuel type or equivalent.

3. MAPLHGR for the (8x8R) fuel type vs. exposure (when applicable).

4. Gross Core Thermal power limit vs. exposure (if applicable).

Include the prompt reactor period, τ_o, (or equivalently the prompt neutron lifetime l*, and delayed neutron friction β) in Table 1.

Additionally, the proposed new format for Appendix A results in a substantially reduced content of the plant-specific portion of the reload licensing submittal. Accordingly, we require that additional core wide transient analysis output parameters be provided in Appendix A. These parameters are: positive (void) reactivity vs. time, Doppler reactivity vs. time, scram reactivity vs. time and total reactivity vs. time. Finally, the plot of percent core power vs. percent core flow may be deleted, since it does not significantly contribute to the interpretation and/or evaluation of the transient results.

RESPONSE:

The goal of the Generic Reload Fuel Application is to reduce the documentation required in the individual plant supplemental submittal by incorporating generic information and bounding analyses in the generic reload licensing topical report. Much of the additional information requested is not required to demonstrate the safety of a plant-specific reload.

The thermal and mechanical fuel design analyses are implicitly generic. In addition, Subsection 2.7 states that the individual plant reload parameters are checked to ascertain that the generic analyses apply. Should these analyses not apply to a particular reload, an appendix documenting the plant specific analysis will be provided. This appendix will be referenced in Section 1 of the plant supplemental submittal.

The previous end-of-cycle core average exposure, both nominal and minimum from cold shutdown considerations, will bc reported in Section 3 of the supplemental submittal.

Note that Appendix A was revised in Amendment 1. Exposure dependent operating MCPR limit is given in Section 11, and MAPLHGR vs. exposure is given in Section 14. Local, radial, and axial peaking factors for each fuel type and exposure are given in Section 7. A discussion of the total peaking factor is given in the response to the request on Subsection 5.2.1.5. Gross core power relative to exposure for the limiting pressure and power increased transient is given in Section 9.

The prompt neutron lifetime, l*, becomes significant in transient analyses only if the reactor nears prompt criticality. Because the transient analyses presently do not approach prompt criticality, neither l* nor the prompt reactor period, to, is required. The delayed neutron fraction, β, is included in all of the reactivity inputs to the transient analyses.

General Electric believes that the Generic Reload Fuel Application results in a substantially increased amount of information for each plant reload. While the supplemental submittal is indeed smaller, the amount of generic information is significantly enhanced. However, the individual plant transient results will include void, Doppler, scram and net reactivity balances in plant supplemental submittals after 1 June 1978.

It should be noted that, thermally and hydraulically, 8x8 and 8x8R fuel are very nearly identical. There are some differences (i.e., active fuel length and R-factor). However, dynamically the thermal response of the two fuel designs are essentially the same. Thus, the ΔCPR differences due to transients

are almost indiscernible. Analyses performed on an operating plant for the load rejection w/o bypass event attest to this fact, as shown below:

Fuel Design	Initial CPR	ΔCPR
8x8	1.268	0.213
8x8R	1.267	0.212

Therefore, it has been concluded that both fuel designs will have the same operating limit MCPR and only the results of the retrofit fuel is reported.

REQUEST: Attachment 2-A – Maximum Spike–Subsection 2.4.2.1.1–Power Spiking

The 2.2% power spiking penalty referenced in this subsection has been previously used in connection with 8x8 fuel types having 144-inch and 146-inch fuel stack lengths. Since the 8x8R fuel assemblies for BWR/4's will have a pellet column 150 inches long (which will increase in the maximum gap size), an increase in the power spiking penalty can be expected. Provide an assessment of the maximum spike for the 8x8R design relative to the 8x8 design.

RESPONSE:

Power spiking analysis for the 8x8R fuel design was performed using a gap size consistent with the 150-inch active fuel length. The model used is described in Reference 1. The result, in the LTR, reports that the 2.2% power spiking penalty is applicable to both the 8x8 and the 8x8R designs. Since the time of calculating the power spiking penalty for the 8x8 fuel design, the calculational model used has been improved in order to provide greater stability and increased accuracy in calculated results. The improvements made consist of using continuous cumulative probability density functions rather than discrete stepwise distributions. Based on this model, the maximum calculated power spiking penalty at the top of the active fuel lengths for 8x8 and 8x8R, respectively, are 1.6 and 2.0%.

REFERENCES:

1. "Responses to AEC Questions—NEDM-10735," NEDM-10735 Supplement 1, April 1973.

REQUEST: Attachment 2-B – Normal Operation – Subsection 2.5.3.1.1 – Cladding Creep Collapse

Densification flux spiking is not mentioned in the cladding creep collapse evaluation. Explain why densification spiking does not significantly affect the collapse analysis results.

RESPONSE:

The calculated power spiking penalty is added to the MLHGR in the cladding creep collapse evaluation for the purpose of determining cladding temperature. Additional explanation of this assumption has been added to Subsection 2.5.3.1.1 in response to the request on Figure 2-16.

REQUEST:

Attachment 2-B. Normal Operation—Subsection 2.5.3.1.2 – Stress Evaluations

This section states that the thermal analysis inputs are given in Subsection 2.4. Included in 2.4 are the densification effects, which includes power spiking, as well as "decreased pellet-cladding thermal conductance resulting from increased radial gap size." Although the former consideration is conservative from a clad stress evaluation standpoint, the latter is nonconservative, since it will result in lower and delayed PCI stresses due to rod power changes. The anisotropic radial densification model was established to provide a conservative fuel rod thermal analysis for stored energy determination and is not considered to be appropriate for fuel rod mechanical design evaluations. Therefore, the stress evaluation should be based on power spiking but without including an increase in the pellet-to-clad gap due to densification. Therefore, provide a quantitative assessment of the effect of this new assumption on the limiting fuel rod stresses reported in Table 2-10.

RESPONSE:

An increase in power due to power spikes as a result of densification and axial gap formation is included in the thermal analysis performed and described in Subsection 2.4. No credit is taken for reduced pellet diameter as a result of densification in the thermal analysis used in conjunction with the stress analysis described in Subsection 2.5.3.1.2. Additional clarification regarding the models and assumptions used for the various thermal analyses has been added to Subsection 2.4 in response to Question 2.4.2. Since the thermal analysis in question was performed using the assumption stated, there is no change to the fuel rod stresses reported in Table 2-10.

REQUEST: Attachment 2-B. Normal Operation—Subsection 2.5.3.1.4 – Fatigue Evaluation

This section does not explicitly discuss densification effects in the cladding fatigue evaluation. Reference is made, however, to the application of the cladding stresses determined in Subsection 2.5.3.1.(2) in the fatigue analysis. Therefore, since the fatigue evaluation is based on the stress analysis, which in turn is affected by fuel densification spiking (and radial gap size change) effects, the above comments, relative to Subsection 2.5.3.1.2, apply equally to this subsection and should be addressed in your responses.

RESPONSE:

See the response to the request on Attachment 2, Subsection 2.5.3.1.2.

REQUEST: Attachment 2-C. Anticipated Transients – Subsection 2.4 – Fuel Rod Thermal Analysis

From the discussions in Subsections 2.4 and 2.4.1.1, it is unclear if densification spiking is explicitly considered in the 1% plastic strain evaluations; i.e., for local (RWE, FLE) transients. It is the staff's position that power spiking must be included in the analysis of such localized transients relative to the calculated initial kW/ft and kW/ft increase. Therefore, clearly describe how spiking is included; either in the calculated LHGRs corresponding to 1% strain, or in the peak transient LHGRs from the plant

specific analyses of the anticipated local events. Reference these descriptions in Subsections 5.2.1.5 and 5.5.4.

RESPONSE:

Based on the calculated exposure-dependent LHGR values in Subsection 2.4.1.2, it has been determined that the power required to produce 1% plastic strain in the cladding is equal to or greater than 175% of the design maximum steady-state power throughout life for all rod types in the assembly for the 8x8 fuel design and greater than 160% for the 8x8R fuel design. These ratios consider the presence of a calculated power spiking penalty being added to the MLHGR.

The calculated MLHGR during the RWE is compared to that associated with 1% plastic strain in the cladding to ensure that fuel damage is not expected during the event.

Relative to the loading error accident, the 1% cladding plastic strain is used to predict if fuel failures are expected to occur. If failures are predicted, a radiological evaluation of the consequences will be performed.

The above information has been incorporated into Subsections 2,4.1.2, 5.2.1.5, and 5.5.4, respectively.

REQUEST: Attachment 2-D. ACCIDENTS – 5.5.1 – Control Rod Drop Accident

The document referenced (5-20), for the effect of axial gap formation, due to fuel densification, on the control rod drop accident results (99% probability that local power peaking will be less than 5%) does not address the 8x8R fuel design with the longer (150 in. vs. 144 in.) fuel column length. Provide an evaluation of the maximum local peaking factor increase for the 150-in.-long fuel column associated with the BWR/4 8x8R fuel types. Clearly describe how the resulting power spike effect is accommodated by “adjusting” the local peaking. Indicate whether separate power spike penalties will be used for BWR/2’s and 3’s (144-in. fuel column) and BWR/4’s (150-in. fuel column) or whether a bounding BWR/4 spike will be used in the CRD evaluations.

RESPONSE:

An evaluation was performed for the 8x8R fuel regarding potential power spikes in the RDA condition. The maximum gap size associated with 150-in. active fuel length was used and the evaluation was performed for the top of the active fuel length. Results show that there is a 99% probability that the power spike will be less than 5% which is the same result reported for other fuel types in Reference 5-20. The 5% power spike effect is accommodated by simply increasing the maximum local peaking (P1) factor by 5% (i.e., Pe x 1.05). A bounding value (5%) is used for all CRD evaluations.

REQUEST: Attachment 2-D. Accidents – Subsection 5.5.2 – Loss-of-Coolant Accident

Provide a description of how densification spiking effects will be considered in the LOCA analysis.

RESPONSE:

Densification effects considered in the LOCA analyses is documented in Subsection 5.5.2 (see the response to the request on Subsection 5.5.2).

Responses To Questions On The Barrier Fuel Amendment (Amendment 6)

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QUESTION

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QUESTION

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REFERENCES:

1. Letter, J.S. Charnley (GE) to F.J. Miraglia (NRC), "Barrier Fuel Amendment to NEDE-24011-P-A-4," November 19, 1982.

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Responses To GESSAR II Questions Regarding SRP Section 4.2

QUESTION 4.3

GESTAR II (NEDE-24011), which contains the fuel system design safety analysis for GESSAR II, does not contain clearly identifiable design bases for most of the fuel damage, fuel failure, and coolability phenomena listed in Item II.A of Section 4.2 of the Standard Review Plan (SRP). Thus, except for cladding overheating (Item II.A.2(c)) and fuel pellet overheating [Item II.A.2(d)], we have not been able to identify design basis statements in the text of GESTAR-II or in the referenced documents, even with the aid of Appendix A in Amendment 5 to NEDE-24011. While it is possible in certain cases to infer the design bases, it is preferable to have them clearly stated. Therefore, for each of the fuel system phenomena discussed in Section 4.2 of the SRP, except the two cited above, provide a concise design basis statement which indicates the design objective related to that issue. In responding to this question, provide a cross-reference to Question 490.05.

RESPONSE 4.3

The design basis for each of the phenomena listed in Item II.A of SRP 4.2 was discussed in the presentation to the Core Performance Branch of the NRC on January 25, 1983.

QUESTION 4.4

Unless otherwise stated in Section 4.2 of the SRP, you should provide a design limit for each design basis. The design limit should be a numerical value of some parameter which provides assurance that the design basis (i.e., the objective or need) will be met. For all but the following phenomena, adequate design limits have been supplied or adequate explanations have been provided for the lack of design limits: (1) Fretting Wear [Item II.A.l(c)]; (2) External Corrosion and Crud Buildup [Item II.A.l(d)]; (3) Fuel and Burnable Poison Rod Pressures [Item II.A.l(f)]; (4) Fuel Assembly Liftoff [Item II.A.l(g)]. Accordingly, provide design limits for the above listed phenomena. Alternatively, discuss why no limits are required. Design limits for cladding rupture [Item II.A.2(g)], mechanical fracturing [Item II.A.2(h)], ballooning [Item II.A.3(c)] and fuel assembly structural damage [Item II.A.3(e)] are being addressed as part of separate generic reviews and need not be discussed in your FSAR now. When our generic review of these matters is completed, you should incorporate the appropriate resolutions in your FSAR.

RESPONSE 4.4

Each of the four phenomena listed above is discussed separately:

1. Fretting Wear – No design limit for fretting wear is required since testing and experience indicate that fretting wear has been eliminated as an active wear mechanism.
2. Corrosion – No separate design limit is required for corrosion and crud buildup because it is considered in the design analyses. Corrosion and crud buildup impact the calculated cladding temperature and material strength, and the ability of the clad to meet the stress limits prescribed in NEDE-24011-P-A-5. Thus, the amount of corrosion and crud buildup is limited by the stress limits on the clad.
3. Internal Rod Pressure – Limits for internal rod pressures are discussed in the response to Question 4.8.
4. Fuel Liftoff – The GE philosophy with respect to fuel lift is that under worst-case hydrodynamic loading conditions, vertical liftoff forces must not unseat the lower tie plate from the fuel support piece such that the resulting loss of lateral fuel bundle positioning could interfere with control blade insertion.

QUESTION 4.5

The fuel assembly description and drawing contained in GESTAR II are much less comprehensive than called for by Item II.B of Section 4.2 of the SRP. This particular item in the SRP contains a list of the information commensurate with an acceptable fuel system description. Accordingly, provide the information identified in Section 4.2 of the SRP.

RESPONSE 4.5

The GE BWR fuel assembly design is described in Section 2.1 of GESTAR II. Design specification limits are provided in Table 2-1 of that report.

The other information listed in Item II.8 of SRP 4.2 is not provided because it is either not used in the analyses or because enough detail is provided for the specific item to be derived. End plug dimensions are a function of U-235 enrichment, gadolinia concentration, and dependent on whether the end plug is in the upper or lower end of the fuel rod. The many sizes of end plugs possible make the amount of information available voluminous. Cladding inside roughness and pellet roughness data are available only as tolerances and, therefore, are not provided.

Tolerances are considered in the safety evaluation, which is performed and measured against approved safety criteria. The safety evaluation is performed assuming the most limiting combination of tolerances for all critical dimensions. This is discussed in Section 2.4.1 of GESTAR II.

In summary, GE believes that enough information is provided in sufficient detail to provide a reasonably accurate representation of the fuel design, thus satisfying the intent of the SRP.

QUESTION 4.6

In the recently submitted Appendix A to NEDE-24011-P-A-5, you state that the channel deflection analysis is provided in Section 5.3.2 of NEDE-21354-P. However, no such section exists in that topical report. Correct this reference. Furthermore, since the referenced channel box deflection report is relatively old (1976) and more data are available now regarding the magnitudes and rates of channel box deflection as a function of service, indicate whether: (1) the data verify the predictions of the deflection model in NEDE-21354-P; (2) your model adequately addresses channel bowing as well as bulging; and (3) you still recommend the periodic settling friction tests and measurements described in NEDE-21354-P, and if so, on what schedule. If you now recommend some other approach, or if NEDE-21354-P procedures have been revised, describe the changes and discuss their rationale.

RESPONSE 4.6

The reference in NEDE-24011-P-A-5 should be to Section 3.2 of NEDE-21354-P. This reference will be corrected in the next amendment to NEDE-24011-P-A-5.

The other parts of this question are addressed below:

1. Additional channel bulge data to higher exposures have been obtained since the publication of NEDE-21354-P. These data confirm the adequacy of the model used by General Electric to predict channel bulge.

2. The model in NEDE-21354-P addresses only channel bulge. GE has recommended to the utilities an approach to mitigate the effects of channel bowing.

3. The following general guidelines minimize the potential for and detect the onset of channel bowing:

   (a) Channels shall not reside in the outer row of the core for more than two operating cycles.

   (b) Channels that reside in the periphery (outer row) for more than one cycle shall be situated in a core location each successive peripheral cycle which rotates the channel so that a different side faces the core edge.

   (c) At the beginning of each fuel cycle, the combined outer row residence time for any two channels in any control rod cell shall not exceed four peripheral cycles.

After core alterations (i.e., reload) and before reaching 40% thermal power, a control rod drive friction test[2] is recommended for those cells exceeding the above general guidelines. After the technical specification scram speed surveillance test on each rod, as required by BWR/6 Standard Technical Specification 4.1.3.2.a, each control rod meeting the above conditions will be allowed to settle a total of two notches, one notch at a time, from the fully inserted position.

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[2] This control rod settling friction test provides an equivalent level of the tests described in NEDO-21354. This test provides adequate assurance of the scram function. The amount of friction detectable by this test is ~250 lb. Control rod drive tests indicate that the CRD will tolerate a relatively large increase in drive line friction (350 lb) while still remaining within technical specification limits. The control blade is in its most constrained, highest friction location when it is fully inserted. The ability of the blade to settle from this position demonstrates that the total drive line friction is less than the weight of the blade (~250 lb).

Total control rod drive friction is acceptable if the rod settles, under its own weight, to the next notch within approximately 10 seconds. If the rod settles too slowly, a rod block alarm will actuate, indicating possible impending channel box-control blade interference. The results of this test will be considered acceptable if no rod block alarm is received. This testing will give an early indication of this interference and will prompt an investigation into the source of the friction. If necessary, corrective action will be completed before startup after the next core alteration.

In lieu of friction testing, fuel channel deflection measurements may be used to identify the remaining amount of channel lifetime.

QUESTION 4.7

In GESTAR II (NEDE-24011), which is the primary support document for the fuel system for your proposed 238 Nuclear Island design, you have not provided a discussion of fuel assembly liftoff for normal operation and "abnormal transients" which are separate and distinct from our concerns regarding the seismic-and-LOCA-loads liftoff. As indicated in Item II.A.l(g) of Section 4.2 of the SRP, however, worst-case hydraulic loads for normal operation should not exceed the holddown capability of the fuel assembly. Although your letter from Gridley to Eisenhut, dated July 11, 1977, addresses this issue for plants and fuel designs of 1977 vintage, it is not evident that assembly liftoff will be precluded for normal operation, including anticipated operational occurrences or abnormal transients in your proposed 238 Nuclear Island. Accordingly, provide a discussion of your analysis of this issue. The design basis and limits aspects of this issue should be addressed as part of your response to Questions 490.0l and 490.02.

RESPONSE 4.7

Design changes which have occurred in the GE BWR fuel bundle since the July 11, 1977 letter from Gridley to Eisenhut have not changed the conclusions reached in that letter. Therefore, it is considered applicable to the fuel to be used in the 238 Nuclear Island design.

GE performs two analyses for the effect of hydrodynamic loads on our fuel bundles. The results of the analysis are provided in the plant-specific New Loads Report. The first analysis is for loads on the bundle resulting from upset conditions (i.e., loads from the combination of normal operation plus OBE plus scram plus SRV actuation). For these operating conditions, including anticipated operational occurrences, GE does not calculate any fuel bundle separation from the core support piece. The second analysis is for loads on the bundle resulting from faulted conditions (i.e., loads from the combination of normal operation, LOCA, SSE, SRV Actuation and Mark III containment hydrodynamic conditions). The method of analysis used for these faulted condition loads is provided in NEDE-21175-3-P, "BWR Fuel Assembly Evaluation of Combined SSE and LOCA Loadings". These faulted loads bound the loads calculated for upset conditions. NEDE-21175-3-P provides a discussion of the margin between the calculated fuel movement as a result of the faulted condition loadings and that movement required to disengage the bundle lower tie plate from the fuel support piece.

QUESTION 4.8

You state in Section A.4.2.1.1.6 of GESTAR II that there is no limit for internal gas pressure. The internal pressure is used in conjunction with other loads on the fuel rod cladding in calculating cladding stresses. The results of such calculations which are provided in Section 2.5.1 of NEDE-24011, show that the calculated cladding stresses can be accommodated. Although this analysis may satisfy our acceptance criteria for cladding stress (Item II.A.l.a of Section 4.2 of the SRP), it does not satisfy our acceptance criterion for rod internal pressure (refer to Item II.A.l.f of SRP Section 4.2 and Question 490.01) because this criterion involves more than stress limits on the cladding. The rod internal pressures used in your cladding stress calculations are well in excess of the nominal coolant system pressure. Accordingly, justify operation under these conditions and explain why the absence of an internal gas pressure limit does not appreciably decrease the margin of safety in calculating fuel system damage.

RESPONSE 4.8

Fuel and poison rod internal pressure increases with increasing burnup and at end-of-life the total internal pressure, due to the combined effects of the initial helium fillgas and the released fission gas, is at a maximum. This maximum internal pressure is used in conjunction with other loads on the fuel rod cladding to calculate cladding stresses.

While there are no limits on internal gas pressure stated in GESTAR II or elsewhere, the maximum internal gas pressure actually is limited by the stress limits in GESTAR II consistent with the following criterion: the stress in the cladding resulting from differential pressure will not exceed the stress limits specified in Section 2.5 of GESTAR II.

GE has performed evaluations for P8x8R and barrier P8x8R using the GESTR(M) mechanical model, in which the fuel rod cladding creepout rate was calculated and compared with the fuel pellet irradiation swelling rate, during late life operation when the fuel rod internal pressure is highest. The results of this evaluation demonstrate that the cladding creepout rate is less than the fuel pellet irradiation swelling rate. This indicates that the fuel cladding gap is not expected to increase under the maximum planned normal operating conditions.

GE has accumulated significant operating experience with all fuel types. This experience includes the operation of lead test assemblies to higher than design exposures. Based on numerous post-irradiation inspections performed, it is concluded that the fuel is adequately designed relative to fuel rod internal pressure since cladding creepout due to fuel rod internal pressure has not been observed.

GE will resolve the internal pressure issue to the satisfaction of the NRC staff before the reference of GESSAR II by the first Applicant.

Responses To Questions On
The Fuel Loading Error Analysis Basis Change
(Amendment 28)

By letter dated May 17, 2004, Global Nuclear Fuel (GNF) submitted proposed amendment 28 to the GESTAR II. The proposed amendment would revise the GESTAR II based on compliance with the Standard Review Plan, Section15.4.7, "Inadvertent Loading and Operation of a Fuel Assembly in an Improper Position".

The Nuclear Regulatory Commission (NRC) staff has reviewed the information GNF provided that supports proposed amendment 28 to the GESTAR II. In order for the staff to complete its evaluation, the following additional information is requested:

Transient Category

QUESTION

1) In a September 27, 1993 RAI response to the September 30, 1992 application GE stated that from 1981-1993 there were 26 reported cases of rotated fuel bundles and in addition a June 23, 1995 submittal reported that in a 16 plant historical data base evaluation, 133,787 bundles were inserted with 2 mislocations overlooked by the bridge crew. How many more reportable misloaded (for the purpose of the RAIs misloaded fuel will mean mislocated and misoriented fuel errors unless specified otherwise) fuel bundles have occurred in foreign and domestic plants using GE fuel since the 1993 RAIs for rotated fuel and the 1995 submittal for mislocated fuel? What verification process is used to find these loading errors? How many times have plants operated at power with misloaded fuel?

RESPONSE

1)(General): The quotes from the September 27, 1993 RAI response and June 23, 1995 submittal quoted above were incomplete from the subject letters. In order to be clear about the meaning of the stated numerical values, the complete statements are provided. The complete statement from the 1993 RAI response is, "Since 1981 when SIL-347 was first implemented, there have been 26 reported cases of misoriented bundles being discovered and corrected by the verification process." The complete statement from the 1995 submittal is, "Of the 16 plants responding to a survey on mislocated bundles, there have been 133,787 bundle insertions to date with 2 mislocated not identified by the refueling bridge crew. However, these mislocated bundles were identified during subsequent core verification prior to power operation."

1)(a): How many more reportable misloaded (for the purpose of the RAIs misloaded fuel will mean mislocated and misoriented fuel errors unless specified otherwise) fuel bundles have occurred in foreign and domestic plants using GE fuel since the 1993 RAIs for rotated fuel and the 1995 submittal for mislocated fuel?

For the purposes of counting occurrences of a bundle not being properly loaded in the core, the mislocated and misoriented bundles may be counted together. However, they are different errors. The misoriented bundle is placed in its proper location, but by a single error is not placed in its correct orientation. The cases reported in the 1993 and 1995 references, in every instance, are misoriented bundles. Mislocating a bundle requires that two bundles be placed in incorrect locations. Two bundles may be switched within the core-loading pattern, or a bundle that should be loaded in the core may be left in the pool and an incorrect bundle loaded in the core. The error rate history supports the assertion that the likelihood of a mislocated bundle is much less than a misoriented bundle.

A survey was performed in July through November 2004 and reconfirmed in February 2005 to cover late 2004 outages. Table 1 summarizes the results of the 1995-2/2005 operating plant history regarding mislocated and misoriented fuel bundles.

Table 1 Summary of Misloaded Fuel Bundle Events 1995-2/2005

Plant	Discovered in loading verification	Operated with	Comments
Browns Ferry 2	None	None	
Browns Ferry 3	None	None	
Brunswick 1	None	None	See 1)(c) Below
Brunswick 2	None	None	
Clinton	None	None	
Cooper	1 misoriented ~2001	None	
Dresden 2	None	None	
Dresden 3	None	None	
Duane Arnold	1 misoriented	None	
Fitzpatrick	3 misoriented – 2000	None	
Grand Gulf	1 misloaded (2 Assemblies) – 1996	None	
Hatch 1	None	None	
Hatch 2	None	None	
Hope Creek	Non-Specific response	None	See 1)(c) Below
LaSalle 1	None	None	
LaSalle 2	None	None	
Limerick 1	None	None	
Limerick 2	None	None	
Monticello	None	None	See 1)(c) Below
Nine Mile 1	None	None	
Nine Mile 2	None	None	
Oyster Creek	None	None	
Peach Bottom 2	None	None	
Peach Bottom 3	None	None	
Pilgrim	None	None	
Quad Cites 1	None	None	
Quad Cities 2	None	None	
River Bend	1 or more misoriented – 1995	None	
Vermont Yankee	1 misoriented – 2001	None	

For the 29 plants that responded to the survey, 9 misloading errors were identified during core verification and corrected. During the 10 years from 1995 to 2005, there have been no misloaded bundles that were undetected by the core loading verification process.

1)(b): What verification process is used to find these loading errors?

Each operating BWR has its own procedure for core verification following fuel and core component movements prior to startup. The procedures follow the recommendations of SIL-347. While the focus of SIL-347 is misoriented bundles, the utility procedures have generally applied the SIL-347 recommendation of “at least 2 independent reviewers” to the each of the three core verification elements: bundle location, orientation, and seating.

During fuel movement, each move (location and orientation) is observed and checked at the time of completion by the operator and a spotter. After completion of the core load, the core is verified by videotaping the core using an underwater camera. The taping may involve two or more videotape runs at different ranges to provide clear resolution of the bundle serial number, and to illustrate the orientation in four bundle clusters. The core verification may take place during the taping process, by viewing of the tapes after taping, or a combination. Verification of the bundle serial number (location) and orientation is performed by independent reviewers. Each independent team records the bundle serial numbers on a core map, which is verified with the design core-loading pattern.

1)(c): How many times have plants operated at power with misloaded fuel?

Respondents to the 2004 survey identified three misoriented bundles in years prior to 1995 that were not detected during verification. The refueling year when the error occurred and plants are as follows:

1979	Brunswick Unit 1,
1993	Monticello, and
1994	Hope Creek.

These misoriented fuel bundles were operated for a complete cycle with no failures (leakers) occurring in the affected or adjacent fuel bundles. Brunswick Unit 1 operated with a bundle rotated 180 degrees in 1979. The error was discovered after startup by viewing the core verification videotape. Continued operation was justified and NRC approved Brunswick Unit 1 operation with the rotated bundle. Hope Creek and Monticello identified the errors at fuel off-loading at the next outage.

There have been no confirmed mislocated fuel bundles that operated in the core for the entire BWR history.

2) The misloaded fuel bundle event is caused solely by operator error and is currently classified as an anticipated operational occurrence (AOO) and therefore is analyzed every fuel cycle. What has changed since GE's previous submittals that would prove that the probability of a misloaded fuel bundle event has significantly decreased and should be reclassified as an infrequent incident and should no longer be covered by General Design Criteria 10? Include in your answer how the changes in refueling schedules due to shorter outages and reliance on contractor personnel are included into the probability.

RESPONSE

Following is quoted from Section 6.3 of the GESTAR Rev 0 SER May 12, 1978

> The reload topical does not classify the fuel loading error (FLE) as an abnormal operational transient. Actual fuel loading experience at operating BWR's has lead the staff to conclude that the frequency of occurrence of a fuel loading error is an event which can occur approximately once per plant lifetime. A fuel loading error could also result in fuel degradation were it to occur, go undetected, and violate the safety limit MCPR. Accordingly, we have required that the fuel loading error be considered as a transient for reload safety analyses. Moreover, a fuel loading error can give rise to significant increases in bundle R-Factor and bundle power and hence a significant change in CPR.
>
> As discussed in Section 6.2.1, GE has proposed a safety limit of 1.0 for this event. This proposal is being reviewed separately by the staff. In the interim, we will continue to require that the fuel loading error be reanalyzed as a limiting local transient event which does not violate the safety limit MCPR.

> In summary, then, we conclude that the events appearing in Section 5.2.1 of Reference 3 together with the fuel loading error, provide a sufficient set of abnormal operational transients to be reanalyzed for each plant-specific BWR reload licensing application.

Therefore, although GE/GNF originally classified the FLE as an infrequent incident, since 1978 it has been analyzed as an AOO.

The data presented in Table 1 supports the classification of the FLE event as an "infrequent incident" per the NRC guidelines in Regulatory Guide 1.70 (RG 1.70), Revision 3, November 1978. Section 15.X.X (page 15-4) of RG 1.70 defines an Infrequent Incident as one that may occur once in a plant's lifetime. Because RG 1.70 was prepared in a time period when the lifetime was limited by the ASME vessel lifetime of 40 years, the plant lifetime used in the mathematics below is selected as 40 years. In order to compare this frequency specification with the observed number of errors reported in the 1)(c) Response, the number of expected errors for the 29 plants included in Table 1 is determined from the RG 1.70 definition as

1 FLE per Plant per Lifetime * 29 Plants / 40 year Lifetime = 0.725 FLE / year

For the 25-year period from 1980 to 2005, which spans the 3 reported uncorrected fuel loading errors, the RG 1.70 Infrequent Incident definition would allow 0.725 FLE/year*25 years or 18.125 errors. In other words, the 3 uncorrected errors that actually occurred are less than 20% of the threshold of the RG 1.70 definition for the Infrequent Incident.

Using the same numbers, the error rate experience for the 25-year period may be compared to the RG 1.70 definition as follows:

$$\frac{3\ \text{Errors}}{29\,\text{Plants} * 25\,\text{Years}} * \frac{40\,\text{Years}}{\text{Lifetime}} \text{ » } 0.17 \text{ FLE per Plant per Lifetime}$$

The FLE error rate experience of ≈ 0.17 FLE per Plant per Lifetime is compared to the RG 1.70 defined error rate of 1 FLE per Plant per Lifetime.

From refueling outage data for every operating domestic BWR from 1980 to date, the approximate number of fuel bundles moved in the 1980 to 1995 period, and from 1995 to 2005 were determined. The following table is a summary of number of refueling outages, approximate number of bundle movements, and error rate.

Time Period	Outages	Moves	Uncorrected Loading Errors
1980 to Jun 1995	241	148020	3
Jun 1995 to Jan 2005	179	118684	0

There are several conclusions that can be drawn from this data:

1. The number of fuel bundle moves before and after June 1995 are comparable and, therefore, can be used to compare the effectiveness of the core loading verification processes and procedures for these periods.

2. In the post 1995 period, the core loading verification processes and procedures are being more effectively applied as compared to the period prior to 1995.

3. The industry trend to shorter refueling outages and any change in the mix of employees versus contractors used during refueling outages has not created an adverse trend in fuel loading

errors. The industry trend toward longer cycles and fewer outages is a benefit to the overall probability of a FLE.

The error rate for the 25-year period and the trend for the most recent 10 years of refueling outages support the classification of the FLE as an "Infrequent Incident."

Thermal Limits

3) Your submittal states that fuel loading error consequence would be perforations in a small number rods in the misplaced bundle, and the resulting fission product will be detected by the offgas system. However, if a mislocated or misoriented fuel bundle remains in the core, despite the event categorization, any transient event (e.g. FWCF or TTWNBP) response could lead to higher delta CPR and exceeding the SLMCPR. This would result in violating GDC 10. Explain if the GE methodology uses a multiplier to the OLMCPR in order to incorporate the impact of operating with a misloaded bundle(s). If not, justify why the delta CPR due to fuel loading error should not be included in the calculations of the OLMCPR value based on other limiting transients.

RESPONSE

The responses to Transient Category Question 1 and 2 provide the justification for the classification of the FLE as an Infrequent Incident. With the re-classification of the FLE as an Infrequent Incident, the questions pertaining to the event as an AOO and the associated MPCR criteria are not applicable.

4) As an AOO, the fuel loading error (FLE) is analyzed every reload as one of the limiting transients. S.2.2.3.6, "Mislocated bundle Accident," in GESTAR II describes the possible mislocated bundle errors. However, in order to meet the current operating conditions, the core design strategies have changed since the establishments of the mislocated and misoriented fuel bundles analysis methodologies. The following questions assess the mislocated and misoriented fuel bundle analyses methodology as it relates to establishing the transient CPR.

a) Provide an evaluation that demonstrates if the mislocated fuel bundle scenarios used to establish the CPR represent limiting core configurations that are consistent with the current core design strategies. For example, does the current plant and cycle-specific mislocated fuel bundle transient event consider the potential for loading high enrichment bundle in core cells already containing two similarly hot bundles.

b) Identify the important parameters or conditions that would have the most impact on the FLE response. For example, what impact would the following conditions have on the FLE response: high enrichment, high GD loading, cycle exposure, exposure of the bundles in the control cell with the mislocated fuel bundle, the configuration and type of bundles in the surrounding cells.

c) What affects, if any, would a mislocated fuel bundle have on the CPR response for the core designs necessary for achieving the current operating strategies. State if the FLE transient could establish the limiting CPR response.

d) Similarly, discuss questions a, b, and c using the misoriented fuel bundle event.

RESPONSE

The responses to Transient Category Question 1 and 2 provide the justification for the classification of the FLE as an Infrequent Incident. With the re-classification of the FLE as an Infrequent Incident, the questions pertaining to the event as an AOO and the associated MPCR criteria are not applicable.

5) Justify why an event that could establish the limiting transient CPR response would be categorized as an accident as proposed in the amendment request.

RESPONSE

RG 1.70 categorizes events according to their frequency of occurrence. The definitions from RG 1.70 are as follows:

1. Incidents of Moderate Frequency - these are incidents, any one of which may occur during a calendar year for a particular plant.

2. Infrequent Incidents - these are incidents, any one of which may occur during the lifetime of a particular plant.

3. Limiting Faults - these are occurrences that are not expected to occur, but are postulated because their consequences would include the potential for the release of significant amounts of radioactive material.

The distinction between the Incidents of Moderate Frequency (AOOs) category and Infrequent Incidents category is purely one of frequency of occurrence.

The U.S. NRC Standard Review Plan 15.4.7, Inadvertent Loading And Operation Of A Fuel Assembly In An Improper Position, Draft Rev. 2, 1996 defines the FLE as an infrequent incident. This SRP also defines the acceptance criteria for the consequences of the event as a small fraction, which is characterized as 10%, of 10CFR100. The following quote is extracted from Section II, Acceptance Criteria, of SRP 15.4.7:

> 10 CFR Part 100 is applicable to SRP Section 15.4.7, because it specifies the methodology for calculating radiation exposures at the site boundary for postulated accidents or events that might be caused by a fuel-loading error. For events having a moderate-frequency of occurrence, any release of radioactive material must be such that the calculated doses at the site boundary are a small fraction of the 10 CFR Part 100 guidelines. A small fraction is interpreted to be less than 10% of the 10 CFR Part 100 reference values. For the purpose of this review, the radiological consequences of any fuel-loading error must include consideration of the containment, confinement, and filtering systems. The applicant's source terms and methodologies with respect to gap release fractions, iodine chemical form, and fission product release timing should reflect NRC-approved source terms and methodologies.

6) It is a historical staff position that the fuel loading error (FLE) has the potential to be a significant transient. For this reason the GESTAR II requires the FLE be analyzed as an AOO. The Staff's historical position as found in the safety evaluation report (SER) for GESTAR II, Rev.0, dated May 12, 1978, Section 6.3 "Transient Events Analyzed for Reloads" on page US.C-43 is as follows, "The reload topical does not classify the fuel loading error (FLE) as an abnormal operational transient. Actual fuel loading experience at operating BWR's has lead the staff to conclude that the frequency of occurrence of a fuel loading error is an event which can occur approximately once per plant lifetime. A fuel loading error could also result in fuel degradation were it to occur, go undetected, and violate

the safety limit MCPR. Accordingly, we have required that the fuel loading error be considered as a transient for reload safety analyses. Moreover, a fuel loading error can give rise to significant increases in bundle R-Factor and bundle power and hence a significant change in CPR." The staff strongly agrees with the explanation of the handling of the FLE in the1978 SER for the following reasons:

* Misloaded fuel bundles do occur. Undetected operation with a misloaded fuel bundle has occurred at Hope Creek. The event at Hope Creek was reported voluntarily. Since a FLE event is reported and docketed on a voluntary basis, the staff is unable to verify whether or not other FLE events have occurred. The industry trend towards minimizing outage times, to the shortest time possible, could place more pressure on the reloading personnel. This could lead to the probability of a FLE event increasing due to the potential for the misloaded bundle to go undetected. Thus it is a staff position that operation with a misloaded fuel bundle can occur.

* Core designs and operating conditions have changed such that the consequences of operating with a misloaded fuel bundle is becoming more severe in terms of the MCPR response during steady state operation. Extended power uprates and longer fuel cycles are causing the consequences of the FLE event to increase. In the core designs needed to fulfill the current operating strategies, the fuel enrichment, batch fraction and number of high powered bundles required every cycle have increased. Thus a FLE event could potentially place three hot bundles in a control rod cell. For these reasons the changes in fuel design are causing the CPR response for the FLE transient to become more limiting, thus more likely to establish the cycle specific OLMCPR. Therefore, it is a staff position that the MCPR response of the FLE transient needs to be calculated and analyzed every cycle since it has the potential to become the limiting transient.

* If a FLE goes undetected and the plant operates with a misloaded fuel bundle, the SLMCPR calculations and the SLMCPR value can become invalidated. If the limiting transient were to occur, a properly calculated OLMCPR value will offset the potential change in the SLMCPR calculations and this offset is what ensures the SLMCPR value will remain valid. If the calculation for the limiting transient was no longer performed, the OLMCPR offset for this transient would cease to exist and the SLMCPR calculation and SLMCPR value may not be protected for the transient. Therefore, it is a staff position that the cycle-specific CPR response for the limiting transient (which could be the FLE) must be used in the OLMCPR calculation so that a valid offset exists to ensure the SLMCPR calculations and the SLMCPR value remain valid if the limiting transient were to occur.

* A detailed PRA analyses has never been conducted to assess the probability of a FLE event based on the refueling practices that are currently performed and allowed by the current Technical Specifications.

* A preliminary FSAR survey indicated that a majority of the plants are required to analyze the FLE on a cycle specific bases to establish the CPR response.

After taking these concerns into consideration the staff is unable to find suitable safety findings to support the proposed changes. The staff finds that this amendment request did not provide sufficient bases to reverse the staff's historical position that the FLE should be analyzed on a cycle-specific basis as a limiting reactivity initiated event. In order for the staff to consider your proposed changes for acceptability, the following is needed:

1) Detailed information explaining the hardware, software, mechanical interlocks, etc (administrative procedures don't count) that prevent misloading errors from occurring.

2) Detailed analysis proving that the CPR value for the FLE transient is negligible now and how the value will remain negligible in the future. In your response provide typical CPR values that are used for establishing the OLMCPR.

3) Detailed PRA analysis that proves the FLE is an accident and not an AOO. Include a chart or table showing the historical data used in determining the event probability.

RESPONSE

The responses to Transient Category Question 1 and 2 provide the justification for the classification of the FLE as an Infrequent Incident. With the re-classification of the FLE as an Infrequent Incident, the questions pertaining to the event as an AOO and the associated MPCR criteria are not applicable. Responses to specific relevant questions follow.

* A detailed PRA analyses has never been conducted to assess the probability of a FLE event based on the refueling practices that are currently performed and allowed by the current Technical Specifications.

A detailed PRA model was submitted with the September 1992 submittal, MFN-183-92. In the June 1995 submittal, MFN-066-95, this model and the predicted probabilities were updated. However, the extensive period of refueling history as reflected in the responses to Transient Category Questions 1 and 2 makes a PRA model of limited value. In other words, there is no particular information provided by a model that is not reflected in the actual refueling data for the past 25 years.

1) Detailed information explaining the hardware, software, mechanical interlocks, etc (administrative procedures don't count) that prevent misloading errors from occurring.

The response to Transient Category Question 1 presented the loading and verification procedures. The overall error rate for the past 25 years, and in particular, the zero error rate for the past 10 years, demonstrates the effectiveness of the procedures.

The following 2 dose questions were provided by Letter from Andrew A. Lingenfelter, (GNF) to Document Control, (US NRC), Subject: Response to NRC Request for Additional Information Regarding Amendment 28 to GESTAR II (TAC NO. MC3559), FLN-2006-018, May 11, 2006.

NRC Question

1. For accident Scenario 1, GNF assumed that fission products from all fuel rods in five failed fuel assemblies were released to the turbine and condensers based on a power level of 5.75 per bundle. Please provide the fission product source term after applying a safety factor of 1.4 and a radial peaking factor of 2.5 as you proposed. State if you assumed the same source term for accident Scenario 2.

GE RESPONSE

The fission product source term (before applying any of the release fractions in Attachment B Table B-1) is provided below:

Table 1 – Scenario 1 and 2 Fission Product Source Term

Isotope	Source Term (Ci)

Isotope	Source Term (Ci)
I-128	3.26E+04
I-129	1.00E-01
I-130	8.61E+04
I-131	2.64E+06
I-132	3.81E+06
I-133	5.42E+06
I-134	5.96E+06
I-135	5.07E+06
Kr-83m	3.37E+05
Kr-85	3.34E+04
Kr-85m	7.18E+05
Kr-87	1.38E+06
Kr-88	1.95E+06
Kr-89	2.39E+06
Kr-90	2.36E+06
Kr-91	1.75E+06
Kr-92	8.47E+05
Xe-131m	2.94E+04
Xe-133	5.43E+06
Xe-133m	1.69E+05
Xe-135	1.84E+06
Xe-135m	1.05E+06
Xe-137	4.73E+06
Xe-138	4.50E+06
Xe-139	3.53E+06

The source term in the table above is the same source term that was used for Scenario 2.

NRC Question

2. For accident Scenario 2, please provide the amount of fission products released from the offgas system, resulting doses, and relevant dose calculations for several representative charcoal holdup times in Figures B-2, B-3, B-5, and B-6 of Attachment B.

GE RESPONSE

As discussed in Attachment B Section B.4.2.2, iodine is not considered in Scenario 2 due to the retention in the offgas charcoal beds. The remaining fission products available at the inlet to the offgas system are as follows:

Table 2 – Scenario 2 Fission Product Source Term at Offgas System Inlet

Isotope	Source Term (Ci)
Kr-83m	3.37E+04
Kr-85	3.34E+03
Kr-85m	7.18E+04

Isotope	Source Term (Ci)
Kr-87	1.38E+05
Kr-88	1.95E+05
Kr-89	2.39E+05
Kr-90	2.36E+05
Kr-91	1.75E+05
Kr-92	8.47E+04
Xe-131m	2.94E+03
Xe-133	5.43E+05
Xe-133m	1.69E+04
Xe-135	1.84E+05
Xe-135m	1.05E+05
Xe-137	4.73E+05
Xe-138	4.50E+05
Xe-139	3.53E+05

A sample calculation is provided to show how the curves in Attachment B Figures B-2 and B-3 were generated. The dose conversion factors that were used are as follows:

Table 3 – Scenario 2 Dose Conversion Factors (DCFs)

Isotope	Whole Body DCF (Rem-m^3/Ci-sec)	TEDE DCF (Sv-m^3/Bq-sec)
Kr-83m	6.44E-04	1.50E-18
Kr-85	5.58E-04	1.19E-16
Kr-85m	3.94E-02	7.48E-15
Kr-87	1.98E-01	4.12E-14
Kr-88	4.89E-01	1.02E-13
Kr-89	4.59E-01	N/A
Kr-90	3.18E-01	N/A
Kr-91	1.81E-01	N/A
Kr-92	1.88E-01	N/A
Xe-131m	5.03E-03	3.89E-16
Xe-133	1.13E-02	1.56E-15
Xe-133m	1.04E-02	1.37E-15
Xe-135	6.20E-02	1.19E-14
Xe-135m	1.08E-01	2.04E-14
Xe-137	4.69E-02	N/A

Isotope	Whole Body DCF (Rem-m^3/Ci-sec)	TEDE DCF (Sv-m^3/Bq-sec)
Xe-138	2.82E-01	5.77E-14
Xe-139	2.32E-01	N/A

The Attachment B Figure B-2 and B-3 whole body doses are calculated using the following formula:

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The following table provides a sample calculation for the krypton whole body dose, assuming a X/Q of 1.0E-04 sec/m^3 and an offgas holdup time of 40 hours:

Table 4 – Attachment B Figure B-2 Sample Calculation

Isotope	λ (sec^{-1})	Whole Body Dose (Rem)
Kr-83m	1.035E-04	7.284E-10
Kr-85	2.047E-09	1.865E-04
Kr-85m	4.298E-05	5.811E-04
Kr-87	1.520E-04	8.553E-10
Kr-88	6.876E-05	4.771E-04
Kr-89	3.656E-03	0.0
Kr-90	2.146E-02	0.0
Kr-91	7.967E-02	0.0
Kr-92	3.767E-01	0.0
	Total	**1.24E-03**

The resulting Kr whole body dose for a X/Q of 1.0E-04 sec/m^3 and an offgas holdup time of 40 hours is 1.24E-03 Rem, which is consistent with the applicable curve in Attachment B Figure B-2.

The following table provides a sample calculation for the xenon whole body dose, assuming a X/Q of 3.0E-04 sec/m^3 and an offgas holdup time of 40 days:

Table 5 – Attachment B Figure B-3 Sample Calculation

Isotope	λ (sec^{-1})	Whole Body Dose (Rem)
Xe-131m	6.691E-07	4.383E-04
Xe-133	1.517E-06	9.758E-03
Xe-133m	3.598E-06	2.088E-07
Xe-135	2.100E-05	1.044E-31
Xe-135m	7.551E-04	0.0
Xe-137	3.008E-03	0.0
Xe-138	8.136E-04	0.0
Xe-139	1.716E-02	0.0
	Total	**1.02E-02**

The resulting Xe whole body dose for a X/Q of 3.0E-04 sec/m^3 and an offgas holdup time of 40 days is 1.02E-02 Rem, which is consistent with the applicable curve in Attachment B Figure B-3.

The Attachment B Figure B-5 and B-6 TEDE doses are calculated using the following formula:

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The following table provides a sample calculation for the krypton TEDE dose, assuming a X/Q of 1.0E-04 sec/m^3 and an offgas holdup time of 40 hours:

Table 6 – Attachment B Figure B-5 Sample Calculation

Isotope	λ (sec^{-1})	TEDE Dose (Rem)
Kr-83m	1.035E-04	6.280E-12

Isotope	λ (sec^{-1})	TEDE Dose (Rem)
Kr-85	2.047E-09	1.472E-04
Kr-85m	4.298E-05	4.079E-04
Kr-87	1.520E-04	6.576E-10
Kr-88	6.876E-05	3.684E-04
Kr-89	3.656E-03	0.0
Kr-90	2.146E-02	0.0
Kr-91	7.967E-02	0.0
Kr-92	3.767E-01	0.0
	Total	**9.24E-04**

The resulting Kr TEDE dose for a X/Q of 1.0E-04 sec/m^3 and an offgas holdup time of 40 hours is 9.24E-04 Rem, which is consistent with the applicable curve in Attachment B Figure B-5.

The following table provides a sample calculation for the xenon TEDE dose, assuming a X/Q of 3.0E-04 sec/m^3 and an offgas holdup time of 35 days:

Table 7 – Attachment B Figure B-6 Sample Calculation

Isotope	λ (sec^{-1})	Whole Body Dose (Rem)
Xe-131m	6.691E-07	1.676E-04
Xe-133	1.517E-06	9.578E-03
Xe-133m	3.598E-06	4.832E-07
Xe-135	2.100E-05	6.448E-28
Xe-135m	7.551E-04	0.0
Xe-137	3.008E-03	0.0
Xe-138	8.136E-04	0.0
Xe-139	1.716E-02	0.0
	Total	**9.75E-03**

The resulting Xe TEDE dose for a X/Q of 3.0E-04 sec/m^3 and an offgas holdup time of 35 days is 9.75E-03 Rem, which is consistent with the applicable curve in Attachment B Figure B-6.

Responses To Questions On The Stability Analysis Change and The SRLR Template Update (Amendment 31)

QUESTION 1:

Please provide: (1) rationale to add Linear Heat Generation Rate (LHGR) in Sections 3.1.2 and 3.2.2, and Table 3.1; (2) description of inter-relationship among LHGR, Maximum Average Planar Linear Heat Generation Rate (MAPLHGR), Minimum Critical Power Ratio (MCPR) and Operating Limit MCPR (OLMCPR); and (3) assessment of LHGR impact on MAPLHGR, MCPR and OLMCPR during reactor operation for core design.

RESPONSE 1:

1. The change from MLHGR to LHGR operating limit in these locations is strictly a nomenclature clarification. The steady state operating limit on linear heat generation rate (LHGR) can be identified either as the maximum linear heat generation rate (MLHGR) or as the LHGR operating limit. Since the term MLHGR is sometimes used to denote the maximum value of the LHGRs obtained for a current operating state (as opposed to the operating limit) it is believed best to describe the limit as the LHGR operating limit.

2. The LHGR operating limit is established to ensure the steady-state LHGR value experienced by any fuel rod at any axial location will provide adequate protection to the fuel thermal-mechanical licensing acceptance criteria (as described in Section 1.1.2 of GESTAR II) should an Anticipated Operational Occurrence (AOO) occur. The MAPLHGR limit is a nodal average LHGR limit (as opposed to a peak rod), and addresses the fuel ECCS/LOCA licensing acceptance criteria (as described in Section 1.1.10 of GESTAR II). The LHGR and MAPLHGR operating values both derive from the same nodal power; however, their licensed limits are set by different criteria. The OLMCPR is an bundle power limit intended to assure that the most limiting fuel bundle will not experience boiling transition should an AOO occur (as described in Section 4.3.1 of GESTAR II). It is not dependent on the nodal average or peak rod LHGR values but on the bundle integrated power, flow, coolant properties and hot-rod factor (R-factor).

3. As noted in (2) LHGR, being a nodal quantity does not directly influence on MCPR. MCPR is determined in large part by the bundle integrated power, which depends on the contributions of all of the nodal powers. The LHGR does have a close relationship to the MAPLHGR margin however since both derive their values directly from the nodal power. The nodal power at any given location in the core divided by the linear feet of fuel rods at that location provides the axial planer average linear heat generation rate (APLHGR) for that location. This APLHGR value must remain below the MAPLHGR licensing limit. That same nodal APLHGR, combined with the pin-by-pin local peaking factors, determines the LHGR value for that location.

QUESTION 2:

Please define End-Of-Cycle (EOC) and End-Of-Rated (EOR) in Section 4.3.1.2.8 End-of-Cycle Coastdown Construction and S.5.13 Thermal Power Monitor, and describe the difference when EOC and EOR are applied to a reactor operation.

RESPONSE 2:

The term End-Of-Rated (EOR) refers to the cycle exposure at which the reactor has reached end-of-full-power reactivity at rated conditions (i.e., rated power, flow, pressure and feedwater temperature) with all control rods fully withdrawn. This EOR point is sometimes referred to as end of reactivity. The EOR point determines the highest exposure at which full power operation can be maintained. Continued operation after this point will result in reduced power (reactor coastdown). The AOO pressurization events are analyzed at this EOR condition. For plants with increased core flow and/or feedwater temperature reduction operating domains additional AOO events are analyzed at EOR incorporating these extended conditions. The term End-Of-Cycle (EOC) refers to the cycle exposure at which the reactor ceases operation for that cycle. The EOC condition can occur before, at or after EOR, although it most often occurs after EOR. In the situation where EOC occurs after EOR, the plant can extend full power operation by increasing core flow and/or by decreasing feedwater temperature (feedwater temperature reduction). The plant can further extend cycle operation (extend the EOC exposure point) via coastdown, whereby criticality is maintained by reducing reactor power (and void fraction) to compensate for the loss in reactivity due to extended fuel burnup.

QUESTION 3:

Based on Figure 5.2-1 in Appendix B on page US-6 under S.2.2.1, it appears that the change in ΔCPR versus initial MCPR may be 0.1 versus 0.5 not .01 versus 0.05. Please clarify that the statement is still valid when the change in ΔCPR is approximately 0.01 for 0.05 change in initial MCPR based on Figure 5.2-1 in Appendix B.

RESPONSE 3:

Figure 5.2-1 and the associated text in Appendix B of GESTAR II was generated to address an RAI on GESTAR II, Revision 5.

The specific response addressed the transient ΔCPR process and the initial MCPR assumption. Figure 5.2-1 and the discussion in GESTAR show the relative insensitivity of the ΔCPR to the initial MCPR assumption. This figure does indicate approximately a 0.01 change in ΔCPR for a 0.05 change in initial MCPR. The transient analysis process iterates the hot bundle power to achieve an initial MCPR such that the minimum CPR during the transient reaches the SLMCPR. In other words, the limiting transient could start at the OLMCPR and the minimum CPR would be the SLMCPR. Transients that are less severe have lower ΔCPRs, therefore, the initial MCPR in the transient needs to be lower for the transient minimum to reach the SLMCPR. The main point of the original response is to describe that starting non-limiting transients at a lower initial MCPR is acceptable because of the small impact to the ΔCPR. A non-limiting transient can begin the transient at a CPR less than the OLMCPR and have a minimum CPR above the SLMCPR.

QUESTION 4:

On page US-2 under S.1.3 Stability Analysis and on page US-37 under S.4.1.5 DSS-CD, please clarify that Reference S-103 is an approved version with an issued date.

RESPONSE 4:

S-103 will be completed with the published –A version as follows.

General Electric Boiling Water Reactor Detect and Suppress Solution – Confirmation Density, NEDC–33075P–A, Revision 6, January 2008.

QUESTION 5:

On page US-37 under S.4.1.3 Option I-D, please clarify which NRC-approved frequency stability code as Reference S-96 will be applied.

RESPONSE 5:

The ODYSY code will be the main NRC-approved frequency stability code, even though the older FABLE code may be used in some legacy applications (e.g., the E1A Standard Cycle evaluation).

In light of the new ODYSY Application LTR (NEDE-33213P-A), either Reference S-105 or Reference S-96 will be applied to Option I-D plants.

The acceptance version of NEDE-33213P will be added as Reference S-105.

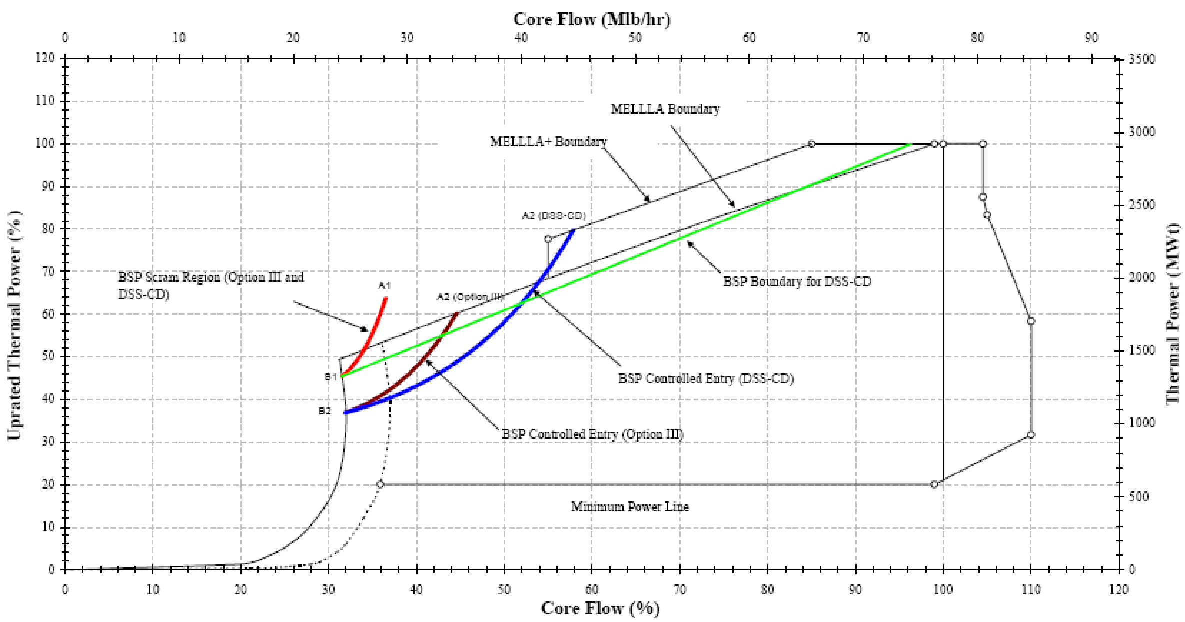
S–105 ODYSY Application for Stability Licensing Calculations Including Option I-D and II Long Term Solutions, NEDE–33213P–A, April 2009.

QUESTION 6:

Please provide a power/flow map to show the difference of the domain for Backup Stability Protection (BSP) for DSS-CD under S.4.2.3 and Backup Stability Protection (BSP) for Option III under S.4.2.2, and describe what role Reference S-103 plays to generate a defined exclusion regions.

RESPONSE 6:

The calculated stability region boundaries for BSP for DSS-CD and BSP for Option III are shown in the figure below. Please note that the actual implemented BSP boundaries might be larger due to the Base BSP region definitions.



These two BSP methodologies are very similar between the two long-term stability solutions of Option III and DSS-CD. However, two differences exist:

(1) The ODYSY core decay ratio (DR) acceptance criterion used for the Controlled Entry Region boundary intercept along the High Flow Control Line (HFCL) is different for the two solutions. The Option III BSP uses a DR acceptance criterion of 0.80 while the DSS-CD BSP uses a DR acceptance criterion of 0.60. For Option III BSP the HFCL is defined as the Maximum Extended Load Line Limit Analysis (MELLLA) boundary. For DSS-CD BSP the HFCL is defined as the MELLLA+ boundary.

(2) The other difference is the imposition of the BSP Boundary on the DSS-CD solution while in the manual mode. While such a boundary does not exist in the BSP for Option III, the Option III solution limits the upper boundary to the MELLLA boundary.

The BSP for DSS-CD will be generated in accordance with Reference S-103 (General Electric Boiling Water Reactor Detect and Suppress Solution – Confirmation Density, NEDC–33075P–A, Revision 6, January 2008.) while the BSP for Option III will be generated in accordance with Reference S-104.

The following Reference S-104 will be added.

S-104 GE to BWR Owners' Group Detect and Suppress II Committee: "Backup Stability Protection (BSP) for Inoperable Option III Solution," OG 02-0119-260, July 17, 2002.

The NRC also requested the content of this response be incorporated into the GESTAR revision, therefore a revised section was attached to this response as Attachment 1 to MFN 09-212 and is included as part of the revision of the US Supplement.

QUESTION 7:

It appears that references listed in Sections 3.6, 4.4, and S.6 consists of many methodologies without an approved version. Please describe the status of the References listed in Sections 3.6, 4.4, and S.6 References with respect to NRC approved version and necessary update.

RESPONSE 7:

There are a large number of references in GESTAR II, including those in Sections 3.6, 4.4, and S.6, which were submitted for information or reference and not for specific review and approval. The status of each reference in Sections 3.6, 4.4, and S.6 has been examined and corrections as follows have been noted. These reference updates will be included when Amendment 31 is implemented into GESTAR II as accepted.

(1) Reference S-11 is a 1977 Topical Report which was accepted, but the –A version was never published. It is recommended that the SE be included with the report in S-11, but no –A version published. Reference S-11 will be updated to include the SE as follows:

C. J. Paone, Banked Position Withdrawal Sequence, January 1977 (NEDO–21231) and Letter from OD Parr (USNRC) to GG Sherwood (GE), Subject: Topical Report - NEDO-2l23l, "Banked Position Withdrawal Sequence," January 18, 1978.

(2) Reference S-52 is no longer used and will be deleted.

(3) In the –A version of GESTARII for Amendment 31, Reference 4-16 will be updated to be the same as S-50. Reference 4-16 will be updated as follows:

Qualification of the One–Dimensional Core Transient Model (ODYN) for BWR's, NEDO–24154–A, Vol. 1 and 2: August 1986.

QUESTION 8:

In addition to the 7 RAI's transmitted by e-mail, the NRC also noted that the population of specific information in the SRLR template had changed compared to the current revision. The NRC requested that the changes be explained.

RESPONSE 8:

In several phone calls, the basis for reducing the level typical values included was discussed. In general, the changes were to eliminate unnecessary information in order to eliminate possible confusion and misuse. Subsequently, GNF proposed the following editorial additions in order to address the NRC staff questions about the elimination of detail:

Provided below is a tabulation of typical examples for each of the various types of plant/cycle applicable descriptions.

Description Category	*Example*
{Appropriate Operating Domain}	ICF
{Appropriate Exposure Range}	BOC to MOC
{Appropriate Fuel Design(s)}	GE14C
{Appropriate Extended Operating Domain Description}	Maximum Extended Load Line Limit Analysis
{Appropriate EOOS Condition Description}	Turbine Bypass Valve Out of Service
{Appropriate Transient Name}	Load Rejection w/o Bypass
{Appropriate Application Condition Name}	Equipment in Service

The agreed upon SRLR template was provided as Attachment 2 to MFN 09-512 and is included herein.

APPENDIX A

STANDARD SUPPLEMENTAL RELOAD LICENSING REPORT

Appendix A

STANDARD SUPPLEMENTAL RELOAD LICENSING REPORT

The following template provides the standard format to be used for an individual plant supplemental reload licensing report (SRLR) with end-of-cycle (EOC) limits reported. For plants that have chosen to use TRACG methods for analyzing pressurization transients, some adjustment of the information and format will be necessary. For plants that have met the requirements necessary to support the recatagorization of the fuel loading error, the ΔCPR results for the FLE events will not be provided, rather a statement regarding the recatagorization will be included.

Additional appendices and figures can be added as necessary to address plant and cycle specific issues. The following are typical lists of appendices and figures.

LIST OF APPENDICES

Analysis Conditions (will normally appear as the first appendix)
Decrease in Core Coolant Temperature Events
Pump Seizure
Partial Arc Condition
Thermal-Mechanical Compliance
Safety/Relief Valve Setpoint Tolerance Relaxation
Expanded Operating Domain Analyses
Equipment Out Of Service Analyses
Off-Rated Power and Flow Limits
List of Acronyms (will normally appear as the last appendix)

LIST OF FIGURES

Reference Core Loading Pattern
Plant response to Overpressurization Event (if required, multiple).
Plant response to Limiting Power and Pressure Increase Event (if required)

The template includes symbols (denoted in blue) which represent plant/cycle specific information to be inserted at these locations. The following is the key to these symbols.

TEMPLATE SYMBOL KEYS:

[a] Insert plant/cycle specific wording
[n] Insert plant/cycle specific numbers
{ } Replace with plant/cycle applicable description
( ) Explanative description

Provided below is a tabulation of typical examples for each of the various types of plant/cycle applicable descriptions.

Description Category	*Example*
{Appropriate Operating Domain}	ICF
{Appropriate Exposure Range}	BOC to MOC
{Appropriate Fuel Design(s)}	GE14C
{Appropriate Extended Operating Domain Description}	Maximum Extended Load Line Limit Analysis
{Appropriate EOOS Condition Description}	Turbine Bypass Valve Out of Service
{Appropriate Transient Name}	Load Rejection w/o Bypass
{Appropriate Application Condition Name}	Equipment in Service

[nnnn] – [nnnn] - [nnnn] - SRLR
Revision [n]
Class I
{Issue Date}

Supplemental Reload Licensing Report

for

{Plant Name}

Reload [n] Cycle [n]

Important Notice Regarding Contents of This Report

Please Read Carefully

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Acknowledgement

{Appropriate acknowledgement description}

The basis for this report is *General Electric Standard Application for Reactor Fuel*, NEDE-24011-P-A-{*Rev*}, {*Issue Date*}; and U.S. Supplement, NEDE-24011-P-A-{*Rev*}, {*Issue Date*}.

1. Plant-unique Items

Appendix A: Analysis Conditions
Appendix [a]: List of Acronyms

2. Reload Fuel Bundles

Fuel Type	Cycle Loaded	Number
Irradiated:		
{*Appropriate Fuel Design(s)*}	[n]	[nnn]
New:		
{*Appropriate Fuel Design(s)*}	[n]	[nnn]
Total:		**[nnn]**

3. Reference Core Loading Pattern

	Core Average Exposure	Cycle Exposure
Nominal previous end-of-cycle exposure:	[nnnnn] MWd/MT ([nnnnn] MWd/ST)	[nnnnn] MWd/MT ([nnnnn] MWd/ST)
Minimum previous end-of-cycle exposure (for cold shutdown considerations):	[nnnnn]MWd/MT ([nnnnn] MWd/ST)	[nnnnn] MWd/MT ([nnnnn] MWd/ST)
Assumed reload beginning-of-cycle exposure:	[nnnnn] MWd/MT ([nnnnn] MWd/ST)	0 MWd/MT (0 MWd/ST)
Assumed reload end-of-cycle exposure (rated conditions):	[nnnnn] MWd/MT ([nnnnn] MWd/ST)	[nnnnn] MWd/MT ([nnnnn] MWd/ST)
Reference core loading pattern:	Figure 1	

4. Calculated Core Effective Multiplication and Control System Worth - No Voids, 20°C

Beginning of Cycle, $k_{effective}$	
Uncontrolled	[n.nnn]
Fully controlled	[n.nnn]
Strongest control rod out	[n.nnn]
R, Maximum increase in strongest rod out reactivity during the cycle (Δk)	[n.nnn]
Cycle exposure at which R occurs	[nnnnn] MWd/MT ([nnnnn]MWd/ST)

5. Standby Liquid Control System Shutdown Capability

Boron (ppm) (at 20°C)	Shutdown Margin (Δk) (at 160°C, Xenon Free)	
	Analytical Requirement	**Achieved**
[nnn]	≥ [n.nnn]	[n.nnn]

6. Reload Unique GETAB Anticipated Operational Occurrences (AOO) Analysis Initial Condition Parameters [1]

Operating domain: *{Appropriate Operating Domain}*
Exposure range : *{Appropriate Exposure Range}* **(Application Condition:** *{Appropriate Application Condition}* **)**

	Peaking Factors						
Fuel Design	**Local**	**Radial**	**Axial**	**R-Factor**	**Bundle Power (MWt)**	**Bundle Flow (1000 lb/hr)**	**Initial MCPR**
{Appropriate Fuel Design(s)}	[n.nn]	[n.nn]	[n.nn]	[n.nnn]	[n.nnn]	[nnn.n]	[n.nn]

[1] Exposure range designation is defined in Table 7-1. Application condition number is defined in Section 11.

7. Selected Margin Improvement Options [2]

Recirculation pump trip: [a]

Rod withdrawal limiter: [a]

Thermal power monitor: [a]

Improved scram time: [a]

Measured scram time: [a]

Exposure dependent limits: [a]

Exposure points analyzed: [a]

Table 7-1 Cycle Exposure Range Designation

Name	Exposure Range [3]
BOC to MOC	BOC[n] to EOR[n] - [nnnnn] MWd/MT ([nnnnn] MWd/ST)
MOC to EOC	EOR[n] - [nnnnn] MWd/MT ([nnnnn] MWd/ST) to EOC[n]
BOC to EOC	BOC[n] to EOC[n]

[2] Refer to the GESTAR basis document identified at the beginning of this report for the margin improvement options currently supported therein.

[3] End of Rated (EOR) is defined as the cycle exposure corresponding to all rods out, 100% power/100% flow, and normal feedwater temperature. For plants without mid-cycle OLMCPR points, EOR is not applicable.

8. Operating Flexibility Options [4]

The following information presents the operational domains and flexibility options which are supported by the reload licensing analysis.

Extended Operating Domain (EOD):	[a]
EOD type: *{Appropriate Extended Operating Domain Description}*	
Minimum core flow at rated power:	[nn.n] %
Increased Core Flow:	[a]
Flow point analyzed throughout cycle:	[nnn.n] %
Feedwater Temperature Reduction:	[a]
Feedwater temperature reduction during cycle:	[nnn.n] °F
Final feedwater temperature reduction:	[nnn.n] °F
ARTS Program:	[a]
Single Loop Operation:	[a]
Equipment Out of Service:	
{Appropriate EOOS Condition Description(s)}	[a]

9. Core-wide AOO Analysis Results [5]

Methods used: [a]

Operating domain: *{Appropriate Operating Domain}*
Exposure range : *{Appropriated Exposure Range}* **(Application Condition:** *{Appropriate Application Condition}* **)**

			Uncorrected ΔCPR	
Event	**Flux (% rated)**	**Q/A (% rated)**	*{Appropriate Fuel Design(s)}*	**Fig.**
{Appropriate Limiting Pressure and Power Increase Transient}	[nnn]	[nnn]	[n.nn]	[n]

[4] Refer to the GESTAR basis document identified at the beginning of this report for the operating flexibility options currently supported therein.
[5] Exposure range designation is defined in Table 7-1. Application condition number is defined in Section 11.

10. Local Rod Withdrawal Error (With Limiting Instrument Failure) AOO Summary

{Appropriate cycle-specific results discussion}

11. Cycle MCPR Values [6] [7]

Two loop operation safety limit: [n.nn]

Single loop operation safety limit: [n.nn]

Stability MCPR Design Basis: See Section 15

ECCS MCPR Design Basis: See Section 16 (Initial MCPR)

SLO Pump Seizure OLMCPR: See Pump Seizure Appendix (line included if applicable)

Non-pressurization Events:

Exposure range: BOC to EOC	
	{Appropriate Fuel Design(s)}
Control Rod Withdrawal Error (RBM setpoint at [nnn] %)	[n.nn]
Loss of Feedwater Heating (See Appendix [a])	[n.nn]
Fuel Loading Error (misoriented)	[n.nn]
Fuel Loading Error (mislocated)	[n.nn] (or, "Not Limiting")

Limiting Pressurization Events OLMCPR Summary Table: [8]

Appl. Cond.	Exposure Range	Option A	Option B
		{Appropriate Fuel Design(s)}	*{Appropriate Fuel Design(s)}*
[n]	*{Appropriate Application Condition Name}*		
	{Applicable Exposure Range, e.g."BOC to MOC"}	[n.nn]	[n.nn]
	{Applicable Exposure Range, e.g."MOC to EOC"}	[n.nn]	[n.nn]

[6] Exposure range designation is defined in Table 7-1.
[7] For single loop operation, the MCPR operating limit is [n.nn] greater than the two loop value.
[8] Each application condition (Appl. Cond.) covers the entire range of licensed flow and feedwater temperature unless specified otherwise. The OLMCPR values presented apply to rated power operation based on the two loop operation safety limit MCPR.

Pressurization Events: [9]

<table>
<tr><td colspan="3">Operating domain: {Appropriate Operating Domain}
Exposure range : {Appropriate Exposure Range} (Application Condition: {Appropriate Application Condition})</td></tr>
<tr><td rowspan="2"></td><td>Option A</td><td>Option B</td></tr>
<tr><td>{Appropriate Fuel Design(s)}</td><td>{Appropriate Fuel Design(s)}</td></tr>
<tr><td>{Appropriate Transient Name}</td><td>[n.nn]</td><td>[n.nn]</td></tr>
</table>

12. Overpressurization Analysis Summary

Event	Psl (psig)	Pdome (psig)	Pv (psig)	Plant Response
MSIV Closure (Flux Scram) – *{Appropriate Operating Domain}*	[nnnn]	[nnnn]	[nnnn]	Figure [n]

13. Loading Error Results

Variable water gap misoriented bundle analysis: [a] [10]

Misoriented Fuel Bundle	ΔCPR
{Appropriate Bundle Design(s)}	[n.nn]

14. Control Rod Drop Analysis Results

{Appropriate Rod Drop Accident analysis description}

15. Stability Analysis Results

{Appropriate Stability results description}

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[9] Application condition numbers shown for each of the following pressurization events represent the application conditions for which this event contributed in the determination of the limiting OLMCPR value.
[10] Includes a [n.nn] penalty due to variable water gap R-factor uncertainty.

16. Loss-of-Coolant Accident Results

16.1 10CFR50.46 Licensing Results

{Appropriate ECCS methodology and results description}

Table 16.1-1 Licensing Results

Fuel Type	Licensing Basis PCT (°F)	Local Oxidation (%)	Core-Wide Metal-Water Reaction (%)
{Appropriate Fuel Design(s) }	[nnnn]	< [n.nn]	< [n.nn]

The *{Appropriate methodology}* analysis results are documented in Reference [n] for *{Appropriate Fuel Design(s)}* in Section 16.4.

16.2 10CFR50.46 Error Evaluation

All reported errors have been corrected in the evaluation documented in Reference [n] for *{Appropriate Fuel Design(s)}* in Section 16.4.

OR (if reporting errors are applicable for this cycle)

The 10CFR50.46 errors applicable to the Licensing Basis PCT are show in the table below.

Table 16.2-1 Impact on Licensing Basis Peak Cladding Temperature for {Appropriate Fuel Design(s)}

10CFR50.46 Error Notifications		
Number	**Subject**	**PCT Impact (°F)**
[n]	*{Appropriate Error Description}*	[nnn]
	Total PCT Adder (°F)	[nnn]

The *{Appropriate Fuel Design(s)}* Licensing Basis PCT remains below the 10CFR50.46 limit of [nnn] °F.

16.3 ECCS-LOCA Operating Limits

The ECCS MAPLHGR operating limits for new fuel bundles in this cycle are shown in the tables below.

Table 16.3-1 MAPLHGR Limits

Bundle Type: *{Appropriate Bundle Design(s)}*

Average Planar Exposure		MAPLHGR Limit
GWd/MT	**GWd/ST**	**kW/ft**
0.00	0.00	[nn.nn]
[nn.nn]	[nn.nn]	[nn.nn]

The single-loop operation multiplier on LHGR and MAPLHGR, and the ECCS analytical initial MCPR values applicable to each fuel type in the new cycle core are shown in the table below.

Table 16.3-[n] Initial MCPR and Single Loop Operation LHGR and MAPLHGR Multiplier

Fuel Type	Initial MCPR	Single Loop Operation LHGR and MAPLHGR Multiplier
{Appropriate Fuel Design(s)}	[n.nnn]	[n.nn]

16.4 References

The SAFER/GESTR-LOCA analysis base report applicable to the new cycle core is listed below.

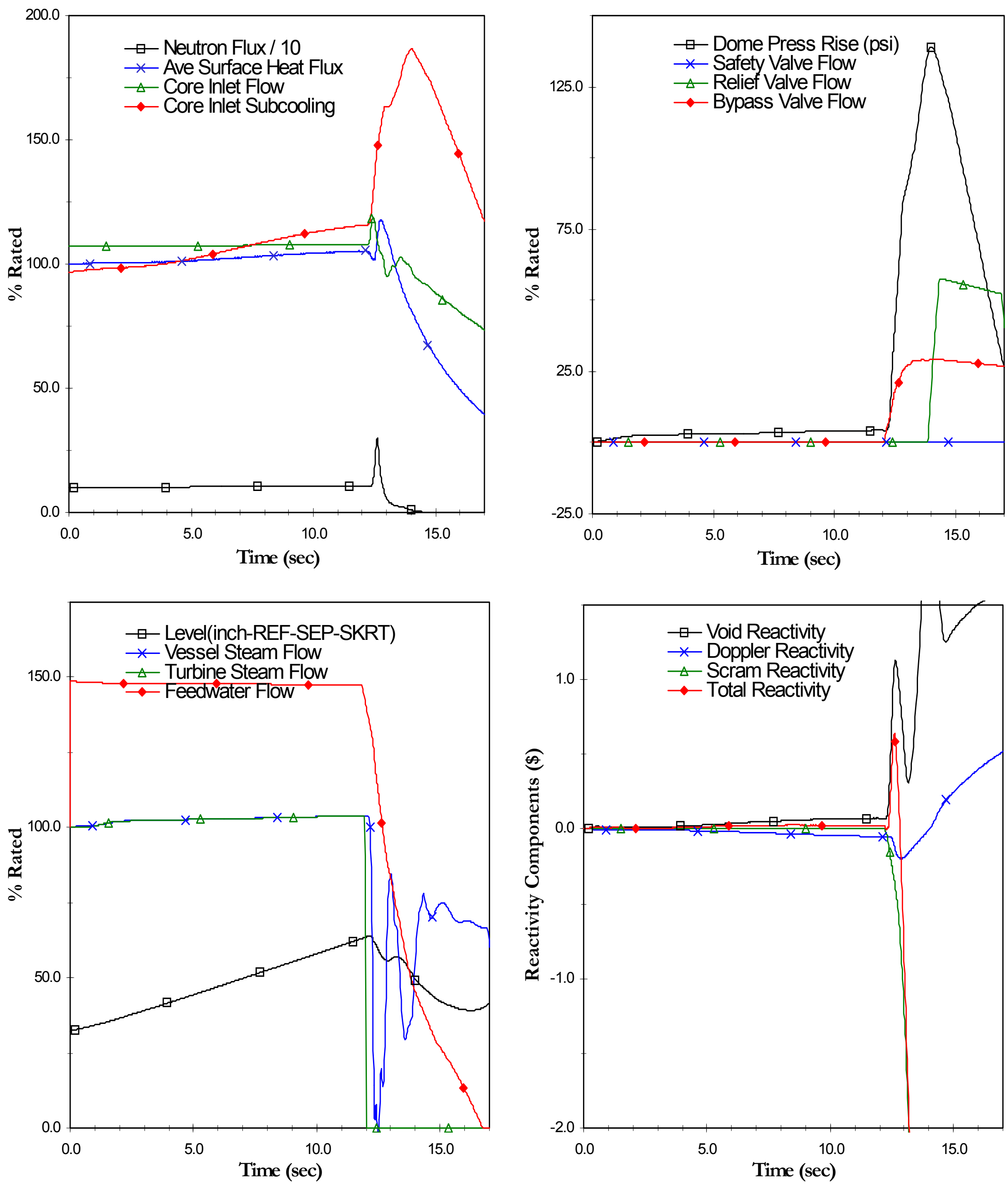
References for *{Appropriate Fuel Design(s)}*

1. *{Appropriate Reference(s) for this fuel design}*

{ Core Loading Map }

Fuel Type	
A= {*Appropriate Bundle Design(s)*} ({*Appropriate cycle*}) B= C= D=	E= F= G= H=

Figure 1 Reference Core Loading Pattern



Sample Figure [n] Plant Response to *{Appropriate Transient Analysis}*
(*{Appropriate Exposure Point and Operating Domain}* **)**

Appendix A
Analysis Conditions

The reactor operating conditions used in the reload licensing analysis for this plant and cycle are presented in Table A-1. The pressure relief and safety valve configuration for this plant are presented in Table A-2. Additionally, the operating flexibility options listed in Section 8 are supported by the reload licensing analysis.

Table A-1 Reactor Operating Conditions

Parameter	Analysis Value *{Appropriate Core Flow and Feedwater Temperature Condition(s)}*
Thermal power, MWt	[nnnn.n]
Core flow, Mlb/hr	[nnn.n]
Reactor pressure (core mid-plane), psia	[nnnn.n]
Inlet enthalpy, Btu/lb	[nnn.n]
Non-fuel power fraction	[n.nnn]
Steam flow, Mlb/hr	[nn.nn]
Dome pressure, psig	[nnnn.n]
Turbine pressure, psig	[nnn.n]

Table A-2 Pressure Relief and Safety Valve Configuration

Valve Type	Number of Valves	Lowest Setpoint (psig)
{Appropriate Valve Description}	[n]	[nnnn.n]

Appendix [a]
List of Acronyms

Acronym	Description
{Acronym}	{Acronym description}

RESPONSES TO QUESTIONS ON ADDITIONAL AMENDMENTS (References)

AMENDMENT 7

1. Letter from J.S. Charnley (GE) to C.O. Thomas (NRC), "Response to Request for Additional Information on Proposed Amendment to GE Licensing Topical Report NEDE-24011-P-A," MFN-230-83, December 19, 1983".

2. Letter from J.S. Charnley (GE) to C.O. Thomas (NRC), "Response to Request Number One for Additional Information on NEDE-20411, Revision 6, Amendment 7," MFN-050-84, April 23, 1984.

AMENDMENT 8

1. Letter from H.C. Pfefferlen (GE) to C.O. Thomas (NRC), "Response to Request Number One for Additional Information on NEDE-20411, Revision 6, Amendment 8," MFN-178-84/009-85, January 14, 1985.

AMENDMENT 9

No questions submitted.

AMENDMENT 10

1. Letter from J.S. Charnley (GE) to C.O. Thomas (NRC), "Response to Request Number One for Additional Information on NEDE-24011-P-A-6, Amendment 10," MFN-035-85, March 11, 1985.

2. Letter from J.S. Charnley (GE) to C.O. Thomas (NRC), "Response to Request Number Two for Additional Information on NEDE-24011-P-A-6, Amendment 10," MFN-053-85, April 26, 1985.

3. Letter from J.S. Charnley (GE) to C.O. Thomas (NRC), "Supplementary Information Regarding NEDE-24011-P-A-6, Amendment 10," MFN-060-85, May 2, 1985.

4. Letter from J.S. Charnley (GE) to C.O. Thomas (NRC), "Supplementary Information Regarding Use of SAFE/REFLOOD to High Exposures," MFN-132-85, October 31, 1985.

AMENDMENT 11

1. Letter from J.S. Charnley (GE) to C.O. Thomas (NRC), "Response to Request Number One for Additional Information on NEDE-24011, Revision 6, Amendment 11," MFN-094-85, July 18, 1985.

AMENDMENT 12

No questions submitted.

AMENDMENT 13

No questions submitted.

AMENDMENT 14

1. Letter from J.S. Charnley (GE) to G.C. Lainas (NRC), "Response to NRC Request for Additional Information for Amendment 14 to General Electric Licensing Topical Report NEDE-24011-P-A," MFN-032-86, May 7, 1986.

AMENDMENT 15

1. Letter from J.S. Charnley (GE) to G.C. Lainas (NRC), "Response to NRC Request for Additional Information for Amendment 15 to GE Licensing Topical Report NEDE-24011-P-A," MFN-057-086, dated July 11, 1986.

2. Letter from J.S. Charnley (GE) to G.C. Lainas (NRC), "Response to Second NRC Request for Additional Information for Amendment 15 to GE Licensing Topical Report NEDE-24011-P-A," MFN-008-087, dated January 16, 1987.

AMENDMENT 16

1. Letter from J.S. Charnley (GE) to M.W. Hodges (NRC), "Response to Reviewer Comments on GESTAR Amendment 16," MFN-111-087, dated November 11, 1987.

AMENDMENT 17

No questions submitted.

AMENDMENT 18

1. Letter from J.S. Charnley (GE) to M.W. Hodges (NRC), "Additional Information Pertaining to NRC Review of GESTAR Amendment 18," MFN-055-087, August 4, 1987.

2. Letter from J.S. Charnley (GE) to M.W. Hodges (NRC), "Requested Information for Amendment 18," MFN-115-87, December 10, 1987.

3. Letter (and enclosure) from J.S. Charnley (GE) to G. Lainas (NRC), "Additional Information Pertaining to Proposed Amendment 18 to NEDE-24011-P-A," MFN-003-087, dated January 29, 1987.

4. Letter (and attachment) from J.S. Charnley (GE) to M.W. Hodges (NRC), Pertaining to GESTAR Amendment 18 (GE8x8NB Fuel), MFN-065-087, dated October 14, 1987.

5. Letter from J.S. Charnley (GE) to M.W. Hodges (NRC), "Response to Questions on GESTAR-II Amendment 18," MFN-005-088, dated January 13, 1988.

6. Letter from J.S. Charnley (GE) to M.W. Hodges (NRC), "Additional Information on GESTAR-II Amendment 18," MFN-012-088, dated February 12, 1988.

AMENDMENT 19

1. Letter from J.S. Charnley (GE) to M.W. Hodges (NRC), "Supplementary Information Regarding GE Special Report MFN-106-85," MFN-085-86, dated September 9, 1986.

2. Letter from J.S. Charnley (GE) to M.W. Hodges (NRC), "Response to Request for Additional Information on GE Special Report MFN-106-85," MFN-004-087, dated January 14, 1987.

3. Letter from J.S. Charnley (GE) to M.W. Hodges (NRC), "Revised Response to Question 3 of Request for Additional Information on GE Special Report MFN-106-85," MFN-025-87, dated March 12, 1987.

AMENDMENT 31

1. Letter from A.A. Lingenfelter (GNF) to Document Control Desk, "Response to Request For Additional Information Relating To Amendment 31 to NEDE-24011P-A, "General Electric Standard Application For Reload Fuel (GESTAR II)," (TAC Number MD8425)," MFN 09-512, dated August 5, 2009.

2. Letter from A.A. Lingenfelter (GNF) to Document Control Desk, "Supplementary Response to Request For Additional Information Relating To Amendment 31 to NEDE-24011P-A, "General Electric Standard Application For Reload Fuel (GESTAR II)," (TAC Number MD8425)," MFN 09-512, Supplement 1, dated October 1, 2009.

3. Letter from A.A. Lingenfelter (GNF) to Document Control Desk, "Supplementary Response to Request For Additional Information Relating To Amendment 31 to NEDE-24011P-A, "General Electric Standard Application For Reload Fuel (GESTAR II)," (TAC Number MD8425)," MFN 09-512, Supplement 2, dated March 12, 2010.

Appendix C

NRC Safety Evaluation Reports

Appendix C

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Appendix C

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NRC Safety Evaluation Report Approving Revision 0 of NEDE-24011-P

UNITED STATES
NUCLEAR REGULATORY COMMISSION
Washington, D. C. 20555
May 12, 1978

Mr. Richard Gridley, Manager
Operating Reactor Licensing|
General Electric Company
175 Curtner Avenue
San Jose, CA 95125

Dear Mr. Gridley:

The staff has completed its review of the General Electric Company proprietary topical report NEDE-24011-P, "Generic Reload Fuel Application". Enclosed is the staff's evaluation of this report.

As a result of our review, we have determined that the above report, as amended, is acceptable for referencing in connection with reload licensing applications involving GE's retrofit and standard 8x8 fuel designs, when supplemented with the plant-specific information described in Appendix A to NEDE-24011-P and when conforming to the specific requirements and limitations stated in our safety evaluation report. When satisfying these specific requirements, (summarized in Section 11.0 of the enclosed SER) the information provided in the subject topical report, together with the plant-unique reload supplement also meets the technical information requirements contained in the present version of Branch Technical Position, DOR-1, "Guidance for Reload Submittals."

Except for the issues (lying outside the scope of review of the reload topical) identified during our review of NEDE-24011-P, the staff does not intend to repeat its review of the subject document when it appears as a reference in a particular reload license application.

Should regulatory criteria or regulations change such that any of our conclusions concerning NEDE-24011-P are invalidated, you will be notified and be given the opportunity to revise and resubmit your topical report for review should you desire.

In accordance with established procedure, it is requested that General Electric issue a revised version of NEDE-24011-P to include the NRC acceptance letter together with our evaluation. Since we have also determined that the non-proprietary version of the report (NEDO-24011) provided an adequate representation of the proprietary report, it is requested that GE follow this same procedure for the non-proprietary version. Both versions must be referenced in future license applications.

If you have any questions concerning our evaluation of NEDE-24011-P, please contact us.

Sincerely yours,

Darrell G. Eisenhut
Assistant Director for Systems and Projects,
Division of Operating Reactors

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Safety Evaluation for the General Electric Topical Report Generic Reload Fuel Application (NEDE-24011-P)

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April 1978

Generic Reload Fuel Application
Safety Evaluation
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1.0 INTRODUCTION AND CONCLUSIONS

By letter(1) dated May 27, 1977, The General Electric Company (GE) submitted a licensing topical report(2) entitled "Generic Reload Fuel Application" for staff review. The report, designated NEDE-24011-P, provides in a single comprehensive reference document, generic licensing information relative to the reload fuel and core design and analysis of operating BWR plants utilizing GE's new two water rod (retrofit) 8x8 fuel assembly for core reloads. The operating BWR plants to which this document applies are listed in Table 1-1 of subject topical report. By design, the information contained in the report is either independent of any particular plant reload application or does not vary from one reload cycle to the next for a specific plant.

Our review of the topical report addressed the generic mechanical, nuclear and thermal-hydraulic design aspects of the new two water rod 8x8 fuel assembly and refueled cores together with the abnormal operational transient and accident analysis information provided. Within each of these technical areas we reviewed the reload safety criteria employed, the codes and methods utilized, the fuel design description, the design-analysis conditions (applied loads) assumed, together with the generic analyses and/or test results, whenever presented. We also reviewed the adequacy of the supplemental information which will be provided as the plant-unique portion of a specific BWR reload application.

Based on our review and evaluation of the information provided by GE, the staff finds the GE licensing topic report, NEDE-24011-P, Revision 3,(3) to be acceptable for reference in BWR 8x8R and 8x8 reload licensing applications, when supplemented with the plant-specific information contained in Appendix A to NEDE-24011-P-3 and when subject to the requirements and limitations stated in this safety evaluation. Moreover, when meeting these additional requirements, it is our finding that NEDE-24011-P-3, coupled with the supplemental information contained in Appendix A, meets the information requirements of Branch Technical Position, DOR-1, "Guidance for Reload Submittals."

2.0 BACKGROUND

With the objective of improving the operating and safety margins, reliability, and neutron economy of initial and reload cores of modern Boiling Water Reactors, the General Electric Company has introduced new reactor fuel designs at various times during the past several years. Most recently, operating BWR's have been reloading with a standard 8x8 fuel design which was first approved for such purposes with the Pilgrim Plant in early 1974. The 8x8 fuel design replaced GE's previous 7x7 product line and incorporated such design improvements as lower linear power density, fuel rods bearing gadolinia as a burnable poison, and a central water rod to aid in reducing local power peaking and the dynamic void coefficient. These changes were intended to provide improved fuel performance characteristics during normal operation, anticipated transients, and accidents.

Based on several years of actual operating experience with the 8x8 fuel design in numerous BWR's together with neutron economy considerations, GE decided to modify the reference 8x8 fuel assembly design. Thus, GE proceeded to design, develop, and fabricate a new, improved 8x8 fuel design (designated 8x8R), incorporating such additional improvements as a second water rod, longer fuel pellet column, and naturally enriched fuel pellets at either end of the fuel stack. The primary purpose of the modifications made to the standard 8x8 design was to improve fuel utilization and fuel cycle costs via increased water-to-fuel ratio, improved axial power shaping, and reduced neutron leakage characteristics. An auxiliary objective of the new fuel design was to improve fuel safety via a decreased void coefficient and lower local bundle peaking factors.

Parallel with the design, development, and production of the retrofit 8x8R fuel for operating reactor reloads (and initial cores), GE initiated a program to standardize and simplify BWR reload licensing applications utilizing the new fuel. GE recognized that many aspects of the overall reload safety evaluation are either generic (or bounding) to all operating BWR's or are not expected to change from one reload cycle to the next for a particular BWR plant. In order to avoid the need for GE, the licensee, and the staff to review such "generic" aspects with each plant-specific reload application, GE consolidated them in a "Generic Reload Fuel Application" for a one-time-only staff review and evaluation.

Generic aspects included the safety criteria (or safety limits), the reload safety analysis codes and methods, the 8x8R reload fuel design description, the reload design-analysis conditions employed and in several instances, the generic safety analysis results (e.g., safety limit MCPR determination, clad collapse evaluation, fuel handling accident analysis).

In addition to the generic document, GE developed and evolved a supplemental core reload submittal that would address the plant (and reload) specific aspects which could not be addressed generically. Thus, the licensing standardization program, referred to as the GESTAR (General Electric Standardized Application for Reloads) Program, consisted of two parts: a generic reload topical report which addressed those subjects within the overall safety bases of the new fuel and reload core which are common to all operating BWR's, and a plant-unique (supplemental) application which would contain analyses performed to complete the reload application of a specific plant. Figure 2-1 schematically depicts the relationship between the two documents, which together make up the required technical bases for an 8x8R BWR reload application.

2-1. BWR Reload Application Technical Bases

	Technical Basis				
Technical Area	Safety Criteria Employed	Codes/ Methods/ Tests Utilized	Plant/Fuel/ Core Design Description	Design Conditions (applied Loads) Description	Analysis/Tests Results
Fuel Mechanical Design					
Fuel/Core Nuclear Design					
Fuel/Core Thermal Hydraulic Design					
Abnormal Operating Transients Analysis					
Accident Analysis					

Generic Reload Application (NEDE-24011 P

Reload Supplement (Plant Unique)

The initial version of the reload topical[2] was submitted by GE for staff review on May 27, 1977. Several revisions[3,4,5] to the original document were submitted by GE in response to several staff requests for additional information as well as to correct and delete certain information contained in the earlier versions. A chronology of our review of the generic reload topical is outlined in Table 2-2.

2-2. Review Chronology for Generic Reload Topical

Date	Description
5/27/77	GE submits initial version of NEDE-24011-P
6/16/77	GE/NRC meeting to discuss scope, content, review schedule, referencing, etc. of NEDE-24011-P
12/6/77	Staff requests further information
1/6/78	GE submits NEDE-24011-P, Rev. 1, to correct errors, revise fuel assembly designations, and accident description
1/27/78	ACRS presentation on NEDE-24011-P review status
1/31/78	Two day GE/NRC meeting to discuss draft responses to information requested by staff and to identify additional required information
2/3/78	GE submits NEDE-24011-P, Rev. 2, and Appendix B in response to further information requested by staff
3/17/78	GE submits NEDE-24011-P, Rev. 3, in response to identified additional information requested by staff

3.0 MECHANICAL DESIGN EVALUATION

The adequacy of the thermal-mechanical, structural and chemical design of the retrofit 8x8 fuel assembly has been evaluated by GE on a generic basis for the applicable BWR's appearing in Table 1-1 of Reference 3. The evaluation considered all modes of plant operation, including the effects of steady-state and normal operating transients, abnormal operating transients and postulated accident conditions. Our evaluation of the adequacy of the fuel bundle mechanical design is contained in the following subsections.

3.1 Fuel Mechanical Design Description

The retrofit fuel assembly design is a modified version of the General Electric 8x8 fuel assembly design currently in operation in 14 domestic BWR's. The generic fuel design is very nearly the same as that described in the BWR 6 Fuel Design and Hatch Unit No. 2 initial core fuel designs,(6) reviewed by the staff for first cycle operation.(7) The generic fuel design is also identical to the fuel design utilized in the second refueling of Hatch Unit No. 1,(8) recently approved(9) by the staff. For identification purposes, the new fuel design will be referred to as the "retrofit 8x8," "two water rod 8x8," or simply "8x8R," while the older 8x8 fuel design will be referred to as the "standard 8x8," "one water rod 8x8," or simply "8x8."

For comparison purposes, fuel assembly design parameters for the two fuel types (and the 7x7 design) are given in Table 3-1 herein. Except for the second water rod and the use of natural uranium at the fuel column ends, the design features of the retrofit 8x8 fuel assemblies are the same as those found in the standard 8x8 fuel assemblies currently operating in numerous BWR's. The 8x8 assemblies have exhibited satisfactory performance to-date.(10)

As seen in Table 3-1, the 8x8 fuel bundle contains 63 unpressurized fuel rods and one water rod whereas the 8x8R bundle utilizes 62 unpressurized fuel rods and two water rods. The two water rods in the 8x8R assembly have a slightly larger diameter than the single water rod used in the 8x8 assembly. The two larger water rods permit improved axial and local power flattening in the 8x8R fuel assembly, compared with both the 7x7 assembly and single water rod 8x8 assembly.

3-1. Comparison of Fuel Assembly Design Parameters

Design Parameter	Fuel Type		
	7x7	8x8	8x8R
Fueled Rods/Assembly	49	63	62
Active Fuel Length (in.)	144	144.145	145.24*/150*
Rod-to-Rod Pitch (in.)	0.738	0.640	0.640
Water/Fuel Ratio (cold)	2.53	2.60	2.75
Cladding O.D. (in.)	0.563	0.493	0.483
Cladding Thickness (in.)	0.037	0.034	0.032
Thickness/Diameter Ratio	0.0657	0.0689	0.0662
Fuel Pellet O.D. (in.)	0.477	0.416	0.410
Pellet/Clad Diametral Gap (mils)	12	9	9
Maximum Linear Heat Generation Rate (Kw/ft)	18.5/17.5	13.4	13.4
*Includes 6 inches of natural UO_2 at top and bottom of fuel column.			

The water rods are capped, hollow, Zircaloy tubes, with small flow holes at the top and bottom ends, to permit controlled coolant flow within the interior of the tubes. One of the water rods axially positions the even Zircaloy-4 fuel assembly spaces grids. The fuel column length of the 8x8R fuel assembly is specific to the BWR class. The retrofit fuel for BWR/2 and BWR/3 plants has a stack length of 145.24 inches while the retrofit fuel assembly for BWR/4 plants has a stack length of 150 inches. This represents an increase in length of 1.24 inches from the 144 inch reload 8x8 fuel assembly stack length previously used for the BWR/2's and BWR/3's and an increase of 4 inches (or 6 inches) from the 146 inch (or 144 inch) reload 8x8 fuel assembly stack length previously used for the BWR/4 class plants. Additionally, several U-235 enrichments are used within each reload fuel assembly to aid in reducing the local power peaking. Gadolinium, a burnable poison, is also used to supplement the rod-to-rod enrichment pattern in the fuel bundle. That is, selected interior fuel rods contain uniformly distributed gadolinium in the form of gadolinia-urania pellets for local power shaping early in life. Gadolinium-bearing fuel rods were first incorporated as a regular design feature of the initial core of Quad Cities Units No. 1 and 2, starting in 1971 and 1972, respectively. Moreover, since 1965, a substantial number of test and production gadolinia-urania rods have been successfully irradiated to substantial exposures.[11]

The combined effects of the additional water rod, longer fuel column, smaller fuel rod diameter, radial enrichment zoning and rods with gadolinia-bearing fuel pellets result in increased operating margins (in more of the fuel rods in the bundle) with respect to the linear power density design limit and maximum fuel temperatures.

The BWR/4 reload 8x8R fuel assemblies also incorporate finger springs, fastened to the lower tie plate to control coolant flow through the lower tie plate-to-channel bypass flow path. In addition, the BWR/4 retrofit reload assemblies will have two alternate path flow holes drilled in the Type 304 stainless steel lower tie plate orifice nozzle. BWR/2 and BWR/3 reload assemblies will also have these features on selected fuel assemblies for bypass flow control.

3.2 Materials Properties

The retrofit 8x8 fuel assembly components are fabricated with Zircaloy-2 and Zircaloy-4, Type 304 stainless steel, Inconel X and ceramic uranium dioxide and gadolinia. These materials are the same as those used for the design of the standard 8x8 and 7x7 fuel assemblies. A substantial number of reactor-years of operating experience has been accumulated with these materials under BWR core environmental conditions. This experience has shown these materials to be compatible with the BWR environment and to retain their functional capability during reactor operations during the design life of the fuel.

Reference 2 provides the materials properties used in the safety analyses associated with the mechanical design of the reload 8x8 fuel bundle. The various properties are the same as those used for the mechanical design of the standard 8x8 fuel assembly.

A 1% plastic strain limit is used as a safety limit for the Zircaloy-2 fuel rod cladding. Below this safety limit, perforation of the cladding, due to overstraining, is not expected to occur. The empirical basis for this strain limit is an estimate of the strain at which an internally pressurized tube reaches plastic instability. GE bases this limit on strain capability of irradiated Zircaloy cladding segments, of fuel rods operated in several BWR's(12). A 1% cladding plastic strain limit historically has been specified by GE as a fuel integrity safety limit for fuel consequences associated with abnormal operational transients.

We have reviewed the basis for the materials properties used in the mechanical design-analyses of the retrofit 8x8 fuel assembly and find them to be acceptable.

3.3 Fuel Rod Thermal-Mechanical Design

The generic thermal-mechanical evaluations of the retrofit 8x8 fuel rods are based on a maximum steady-state operating linear heat generation rate (LHGR) of 13.4 Kw/ft. The elastic stress limits for the fuel rod mechanical design, during normal and abnormal operating reactor conditions are based on the stress categories presented in the ASME Boiler and Pressure Vessel Code, Section III. The cladding is also designed to be free-standing during the fuel design lifetime. A fatigue analysis, using Miner's linear cumulative damage rule, was performed to assure that the cladding will not fail as a result of cumulative fatigue damage. In addition, for abnormal operational transients, a value of 1% plastic strain, discussed in Section 3.2, is established as a safety limit, below which damage due to cladding plastic deformation is not expected to occur. A fuel rod thermal-mechanical evaluation is performed to determine the equivalent local linear heat generation rate, which is established as the fuel cladding integrity safety limit for abnormal operational transients. Pellet cladding interaction, waterlogging, fretting-corrosion, hydriding, and lateral deflection have also been considered in the fuel rod mechanical design. We have reviewed the generic information provided by GE for the above thermal-mechanical design areas, together with the generic analysis applicability assessment which will be performed on a plant specific basis. Our evaluation is reported herein.

Cladding Stress Analysis

The elastic stress limits for the fuel rod cladding utilize the Tresca maximum shear stress theory to calculate stress intensities, which are then compared with the stress intensity limits given in Table 2-6 of the generic reload topical report. The maximum shear stress theory, as presented in the ASME Boiler and Pressure Vessel Code Section III for combined stress, as well as the stress limits, also served as a guide for GE for the design of the 7x7 and 8x8 fuel rods. This approach, based on material yield strength criteria, serves to assure elastic behavior from multidirectional loads such as pressure.

This criteria also minimizes the possibility of excessive deformation resulting from repetitive thermal-mechanical loadings. The results of the cladding stress analyses, using the stress models appearing in Table 2-8 of the reload topical, show that the calculated maximum stress intensities are all well within the applicable stress intensity limits during all normal and abnormal operating conditions. The analyses include load cycles derived from power changes such as those occurring during startup and shutdown, for both hot and cold conditions. Daily load changes and overpower conditions are also included in the stress evaluations. The stress evaluations incorporate the effects of fuel densification power spiking to substantiate the 13.4 Kw/ft design limit LHGR. On the basis of the information provided by GE, including actual BWR operating experience with 7x7, 8x8, and 8x8R fuel assemblies designed to the above stress intensity limits, the staff finds the generic fuel rod cladding stress analysis and results to be acceptable.

Cladding Collapse Analysis

Cladding collapse potential has been generically assessed as part of the overall thermal-mechanical design evaluation of the retrofit 8x8R fuel rods. A collapse analysis was performed using the generic methods described in the SAFE-CCLAPS Model.(13) This model has been previously reviewed and approved(14) by the staff. The limiting collapse criteria assumes an instantaneous increase of 160 psi from the hot full power condition due to a turbine trip with bypass failure. This event can occur at any time during the design life of the fuel assembly. The results of the analysis showed that clad collapse would not occur.

Based on our review of several BWR reload safety analyses, we did not consider a 160 psi pressure increase to be generically bounding for all operating BWR's. Accordingly, we requested GE to provide a generic clad collapse analysis which would be bounding for all operating BWR's for all future cycles. In response to this request, GE provided(15) a reassessment of cladding collapse potential based on differential pressure increases above 160 psi. The results of these studies show that the 8x8R fuel rod is capable of withstanding transient differential pressure increases above rated conditions in excess of 250 psi without experiencing instantaneous collapse or subsequent creep collapse. Additionally, these analyses include the effect of fuel densification power spiking on cladding temperature. Finally, cladding collapse has never been observed for fuel rods in operating BWR's. In view of the information provided by GE, the staff finds the cladding collapse analysis results for the 8x8R fuel assembly design to be generically acceptable for all reload cycles of the applicable BWR appearing in Table 1-1 of the reload topical report.

Fatigue Analysis

The fatigue analysis performed by GE uses Miner's linear cumulative damage rule(16). The fuel rod location GE considers subject to the greatest fatigue damage is the fuel rod clad tube-to-end plug weld juncture. The cyclic loads considered in the analysis are coolant pressure and thermal gradients as described in Tables 2-12 and 2-13 of Reference 3. The cyclic loads are reported by GE to be representative of a four-year residence time at maximum thermal gradients corresponding to beginning of life conditions. Additionally, incremental damage is based on the standard stress vs. number of cycles to failure curve developed by B.F. Langer and W. J. O'Donnell for Zircaloy-2 cladding(17).

The staff considers the fatigue damage limit, as described by GE, to be adequate.(18) Moreover, the results of the fatigue analysis, using the stress models appearing the Table 2-3 of Reference 3, show that the cumulative fatigue damage is well within the fatigue damage limit.

Fuel Cladding Integrity Safety Limit LHGR

In order to avoid fuel rod rupture, due to excessive cladding strain caused by rapid fuel pellet thermal expansion, GE has established a cladding plastic diametral strain limit of 1%. Using the previously accepted methods for calculating cladding strains, exposure-dependent linear heat generation rates (LHGR's), corresponding to 1% cladding plastic diametral strain were determined by General Electric. The corresponding LHGR's for UO_2 fuel rods are approximately 25, 23 and 20 Kw/ft at exposure levels of 0, 20,000 and 40,000 MWd/t, respectively.

Because urania-gadolinia fuel material has a lower thermal conductivity and melting temperature than urania fuel, the LHGR's corresponding to 1% plastic strain for the gadolinia bearing fuel roads in the 8x8R fuel assembly are lower than the above values. For a urania-gadolinia fuel rods in the retrofit 8x8 assembly, having the maximum gadolinia loading concentration, the calculated LHGR's corresponding to 1% plastic strain are not less than 22.0, 20.5 and 17.5 Kw/ft for 0, 20,000 and 40,000 Mwd/t, respectively. The above LHGR's, for the maximum gadolinia concentration fuel rods, are thus established as the generic exposure-dependent fuel cladding integrity safety limit LHGR's for both 8x8 and 8x8R fuel rods. Fuel rods with peak pellet LHGR's below the safety limit LHGR are not expected to exhibit cladding failure due to overstraining, during the most severe abnormal operational transient event.

The adverse effects of fuel densification power spiking have not, however, been directly considered in the establishment of the above LHGR's. Thus, the staff requires that the calculated maximum LHGR's (for each fuel type), for the most severe transient event, be augmented by an amount equal to the densification power spike penalty (equal to 2.2% at the top of the fuel column) before comparison with the above limits. On this basis, the above LHGR's are acceptable fuel cladding integrity safety limits for the consequences associated with abnormal operational transients. Transient evaluation methods are discussed in Section 6.0 herein.

Waterlogging

Another area of continuing generic review, which is addressed adequately in the reload topical, is the potential and consequences of operating with waterlogged fuel rods. We have reviewed the safety aspects of waterlogging failures that could result from pellet cladding interaction (PCI). A survey of the available information, which includes: (1) test results from SPEPT and NSRR in Japan and (2) observations of waterlogging failures in commercial reactors, indicates that rupture of waterlogged fuel rod should not result in failure propagation or significant fuel assembly damage that would affect coolability of the fuel rod assembly. Thus, we agree that the evaluation of waterlogging failures, as presented in the topical, is correct.

Fretting-Corrosion Wear

Fretting-corrosion wear, due to flow induced fuel rod vibration against the spacer contacts has been considered in the fuel assembly design. The fuel rod vibration and support characteristics of the retrofit 8x8 fuel design are very similar to the 7x7 and standard 8x8 fuel design. Moreover, the 8x8R fuel assembly will operate in the same core environment as the 7x7 and 8x8 assemblies. Fuel rod vibration experiments and years of actual reactor operating experience(19) has provided substantial confidence in the adequacy of the GE BWR fuel designs relative to fretting-corrosion wear behavior. Moreover, actual operating experience with lead 8x8R fuel assemblies(20) has shown the fuel to perform adequately relative to fretting wear. In view of the similarity of the 8x8R fuel design to the older GE BWR fuel designs together with the operating conditions to be associated with the 8x8R

reload assemblies, the staff finds that the fretting-corrosion wear potential of the retrofit 8x8 fuel assemblies to be acceptably low.

Lateral Deflection

Fuel rod lateral deflection, or bowing, has been investigated by GE and considered in the 8x8R fuel assembly design. The deflection limits on the magnitude of fuel rod bowing are based on: (1) cladding stress limits and (2) rod-to-rod and rod-to-channel clearance limits. Thermal-hydraulic tests[21] have demonstrated that a minimum clearance of .060 inches (design clearance is 0.157 inches) is sufficient to ensure a very low probability of local rod over-heating caused by a critical heat flux condition. In the GE fuel assembly surveillance programs, more than 2400 peripheral fuel rods have been examined by borescopic techniques. Additionally, GE studies[21,22,23] show: (1) no observable gross bowing in the standard 8x8 design, (2) very low frequency of minor bowing, (3) calculated deflections within the design limit, and (4) no significant DNB problem at small rod-to-rod and rod-to-channel clearances, based on thermal-hydraulic testing. In view of the above, the staff agrees that there is no significant safety concern relative to potential fuel rod lateral deflection associated with the 8x8R fuel assembly design.

Pellet Cladding Interaction

Pellet cladding interaction (PCI) is addressed in the generic reload topical. Since 1972, General Electric has made changes in the fuel assembly design and has recommended changes in the mode of reactor operation to reduce the incidence of PCI fuel failures. To minimize the potential for pellet ridging, a shorter chamfered pellet with no dishing will be used. The 8x8R design also includes a higher annealing temperature for the Zircaloy-2 cladding to achieve improved uniformity of mechanical properties. In addition to these design changes, General Electric continues to recommend specific operating procedures identified as Preconditioning Interim Operating Management Recommendations (PCIOMR's). Under these procedures, the 8x8R reload fuel will be preconditioned for subsequent full power operation and power cycling by first being maneuvered to full power at a slow ramp rate.

The staff is generically reviewing the fuel designs of all fuel suppliers regarding pellet cladding interactions. On the bases cited above and the thermal-mechanical stress and strain evaluations performed for the reload fuel rod design, the staff believes, pending completion of our generic review, that the 8x8R fuel rod design will exhibit adequate performance relative to PCI type fuel failure.

Fuel Densification

The reload topical[3] references the GE densification analysis[14] previously approved[25] by the staff. The effects of fuel densification on the fuel rod are to increase the stored energy, increase the linear thermal output and increase the probability of local power spikes from axial gaps.

The primary effects of densification on the fuel rod mechanical design are manifested in the calculation for fuel/cladding gap conductance, cladding collapse time and fuel duty (stress and fatigue evaluations). The approved analytical model incorporates time-dependent gap closure and cladding creepdown for the calculation of gap conductance. The cladding collapse time calculation also includes the effect of local gaps on cladding temperature. Finally, cladding collapse has never been observed in BWR fuels.

More recent densification analyses submitted by GE[26] and approved by the staff[27] have addressed the effects of increased densification in gadolinia-urania fuels. The stored energy effects of increased

densification in gadolinia-urania fuels are offset by the significantly lower LHGR in the gadolinia bearing fuel rods compared to the non-gadolinia bearing fuel rods in the bundle. With regard to densification power spiking effects, GE has shown that the offsetting effects of excess thermal expansion and axial heat transfer, not previously taken credit for, more than offsets the adverse spiking effects associated with gadolinia. Thus, the staff finds that fuel densification has been acceptably accounted for in the mechanical design of the retrofit 8x8 fuel assemblies. Fuel densification effects or transients and accident consequences are addressed separately in Sections 6.0 and 7.0 herein.

Fission Gas Release

The staff has questioned the validity of the fission gas release predictions in vendor thermal-mechanical performance codes, including GEGAP-III[28], at burnups in excess of 20,000 Kwd/t. By letter[29] dated January 18, 1978, the NRC requested that GE revise their fuel performance model to account for burnup enhanced gas release and submit the revised model for staff review within one year. The staff intends to request all licensees to provide a schedule for incorporating burnup enhanced fission gas release into their safety analysis. For the first cycle, (fresh) 8x8R reload fuel will not achieve burnups at which fission gas release enhancement occurs. Thus, the effect of enhanced fission gas release on safety analyses is not a concern for 8x8R fuel in the first cycle of operation. Our concern, relative to exposed 7x7, 8x8 and 8x8R fuel, is being handled generically, as described above.

Operating Experience

The evolution to GE's standard 8x8 and retrofit 8x8R fuel designs encompasses over 10 years of operating a fuel manufacturing experience. In addition to this general experience, GE has monitored the fuel behavior in commercial operating reactors by conducting fuel surveillance programs. Much of this previous experience with BWR fuel rods and assemblies is also considered applicable to the new 8x8R fuel assembly.

The standard 8x8 fuel design is currently in operation in 14 BWR's and a substantial number of fuel bundles (>250) are in their third and even fourth irradiation cycle. A detailed post-irradiation examination has been performed at Monticello on lead 8x8 developmental assemblies at the end of their first two cycles and indicates satisfactory performance.[30] Four lead demonstration assemblies of the 8x8R fuel design[34] began operation in Peach Bottom Unit No. 2, in March 1976. These four assemblies were extensively visually examined at the end of one cycle in mid-1977. The examination results have demonstrated that the 8x8R assemblies and channels are in excellent condition for continued operation.[20] These assemblies are presently operating satisfactorily in their second cycle of operation. In addition, one lead 8x8R assembly, containing several pressurized fuel rods, is presently in its first cycle of operation of Peach Bottom Unit No. 3.[31] Finally, the first batch reload application[8] of 8x8R assemblies began first cycle operation at E.I. Hatch Unit No. 1 during April 1978.

3.4 Fuel Assembly Structural Design

The reload 8x8 fuel assembly is designed to withstand the predicted thermal, pressure, and mechanical interaction loadings which can occur during handling, startup, normal operation, and abnormal operational transients without impairment of functional capability. The fuel assembly is designed to sustain predicted loadings from an operating basis earthquake. Also, the design-analysis of the fuel assembly shows that the functional capabilities will not be exceeded as a result of a safe shutdown earthquake. The ability of the 8x8R assembly and its components to meet these capabilities is

evaluated by (1) analyses based on classical methods and the ASME boiler and Pressure Vessel Code which are compared against acceptance criteria (design ratios) and (2) testing programs.

The adequacy of the fuel assembly structure during normal operations and normal operating transients is based principally on stress limits and stress formulations which are consistent with the requirements of the ASME Boiler and Pressure Vessel Code Section III. A detailed description of the stress categories is given in Table 2-19 of Reference 3. Based on our review of the generic analysis results provided by GE and actual reactor operating experience for the 7x7, 8x8 and lead 8x8R fuel assemblies, we find the retrofit 8x8R to be structurally adequate for the loads associated with normal operating conditions for the applicable operating BWR plants.

The adequacy of the fuel assembly structural design during abnormal operational transients principally relate to the fuel cladding integrity safety limit LHGR and cladding collapse potential. These are evaluated in Section 3.3 herein.

The adequacy of the 8x8R fuel assembly structure to withstand the peak loadings which occur during shipping and handling operations at room temperature has been documented[32] by GE on a generic basis. The analyses and test results reported in the referenced topical report is currently under separate review by the staff.

The generic combined seismic and LOCA load analysis[33] referenced in the reload topical is also currently under separate staff review. The use of the reference analysis as a bounding analysis for all of the plants appearing in Table 1-1 of the reload topical is not acceptable in that the reference analysis assumes a 100 mil wall thickness fuel channel, whereas many operating plants continue to utilize 80 mil channels. Thus, the reference analysis appears to be not applicable for such plants. GE has been informed of our concern. The staff will continue to review this issue on a generic basis.

The capability of the fuel assembly to withstand the control rod drop accident, pipe breaks inside and outside of containment, the fuel handling accident and the recirculation pump seizure accident is addressed separately in Section 7.0 herein.

3.5 Application of the Generic Mechanical Design Evaluation to Reloads

Design parameters which vary from one 8x8R fuel type (e.g., fuel stack length) to another, or from one operating BWR plant to another, have been considered by GE to the greatest extent possible in the generic mechanical design evaluation. GE states[3] that this evaluation assumed the most limiting combination of tolerances for all critical fuel dimensions together with plant design conditions (e.g., core operating pressure, maximum power vs. exposure for the peak duty fuel rod) which were expected to be bounding for all future reload cycle of the applicable operating BWR's. Accordingly, the need exists to validate the continued conservatism of the generically assumed design parameter values for each particular 8x8R reload application. The internal procedures used by GE for the validation process together with design parameters reviewed for each reload application are summarized in Reference 3 and have been discussed and reviewed in detail with GE directly as part of our evaluation of the reload topical. Based on our review of the information provided, we find the described generic analysis applicability review program to be acceptable.

4.0 NUCLEAR DESIGN EVALUATION

Our review of the nuclear design aspects consisted of an evaluation of the described and referenced nuclear analysis codes and methods for mixed core reload applications and the retrofit fuel design, the bundle nuclear characteristics associated with the new 8x8R fuel assembly, the general nuclear characteristics of the 8x8R reload core, fuel storage subcriticality, and the adequacy of the nuclear design information to be provided on a plant-specific reload basis.

4.1 Nuclear Design Methods

The staff has reviewed and evaluated the information presented(3) on the nuclear analysis methods. The basic calculational procedures used for generating neutron cross sections are part of General Electric's Lattice Physics Model.(35,36) In this model the many-group fast and resonance energy cross sections are computed by a GAM-type program. The fast groups are treated by integral collision probabilities to account for geometrical effects is fast fission. Resonance energy cross sections are calculated by using the intermediate resonance approximation, with energy and position-dependent Dancoff factors included. The thermal cross sections are computed by THERMOS-type program. The model accounts for the spatially varying thermal spectrum throughout the fuel bundle. These calculations are performed for an extensive combination of parameters including fuel enrichment and distribution, fuel and moderator temperature, burnup, voids, void history, the presence or absence of adjacent control rods and gadolinia concentration and distribution in the fuel rods. As part of the Lattice Physics Model, three-energy group two-dimensional XY diffusion theory calculations for one or four fuel bundles are performed. In this way, local fuel rod powers can be calculated as well as single bundle or four bundle (with or without a control rod present) average cross sections. The General Electric Company has submitted two licensing topical reports(35,36) which describe in detail and verify the adequacy of the procedures outlined above. The staff has previously reviewed these reports and has concluded(37) that the methods satisfy its requirements for core physics methods. These methods are also considered to be generically acceptable for the mixed reload cores incorporating up to three different fuel bundle designs including retrofit 8x8 fuel bundles.

The single or four bundle averaged neutron cross sections which are obtained from the Lattice Physics Model are used in either two or three-dimensional calculations. Two-dimensional XY diffusion theory calculations are usually performed with three energy groups at a given axial location to obtain gross power distribution, reactivities and average three-group neutron cross sections (for use in the one-dimensional axial calculations of scram reactivity). The three-dimensional nodal calculations use one energy group and can couple neutron and thermal-hydraulic phenomena. These three-dimensional calculations are performed using 24 axial nodes and 1 radial node per fuel bundle resulting in about 14,000 to 20,000 spatial nodes. This three-dimensional calculation provides power distributions, void distributions, control rod positions, reactivities, eigenvalues, and also radially collapsed average cross sections for use in the one-dimensional axial scram calculations. The three-dimensional calculations have been described and verified by the General Electric Company in two licensing topical reports.(38,39) The staff has previously reviewed these reports and has approved use of these methods in licensing applications.(37) These methods are also considered to be generically acceptable for mixed reload cores incorporating up to three different fuel bundle designs including the retrofit 8x8 fuel bundles.

The one-dimensional axial calculation referred to above is a space-time diffusion theory calculation which is coupled to a single channel thermal-hydraulic model. This axial calculation is used to generate the scram reactivity function for various core operating states. This one-dimensional space-time code has been compared by the General Electric Company with results obtained using the

industry standard code, WIGLE, (which is generally accepted as a reference 1-D space-time diffusion theory code) and shown to be conservative. Our consultant, Brockhaven National Laboratories, has performed an extensive study(40) of BWR scram reactivity behavior and has concluded that the end of cycle, all rods out configuration, represents the limiting condition for BWR scram system effectiveness. Thus, we conclude that the method and assumptions used by General Electric to obtain the scram reactivity curves are acceptable.

The Doppler, moderator void and moderator temperature reactivity coefficients are generated from data obtained from the Lattice Physics Model. The effective delayed neutron fraction and the prompt neutron lifetime are computed using the one dimensional space-time code. The power coefficient is obtainable by appropriately combining the moderator void, Doppler, and moderator-temperature reactivity coefficients.

The General Electric Company has submitted a licensing topical report(41) which describes the methods used for the generation of void and Doppler reactivity feedback for application to BWR design. The staff currently has this report under separate review. Until our review is complete, we will evaluate the adequacy of these methods on a plant-by-plant basis.

Comparisons between calculated and measured local and gross power distributions have been presented by the General Electric Company in two topical reports.(36,39) Predicted local (intra-bundle) power distributions were compared to data obtained from critical experiments and from gamma scans performed on operating plants. Gross radial and axial power distributions obtained from operating plants have been compared with values predicted by the BWR Simulator code. These comparisons have yielded values for calculational uncertainties to be applied to power distributions. Comparisons have also been made of calculated values of cold, xenon-free reactivity and hot operating reactivity of a number of operating reactors as a function of cycle exposure. These comparisons have been used to establish shutdown reactivity requirements. The staff has reviewed the two topical reports and found them to be acceptable(37) for reference in licensing actions.

4.2 Nuclear Design Characteristics

A particular reload core can generally contain a mixture of 8x8R bundles and bundles of the 7x7 and 8x8 designs. It is a general characteristic of a BWR core, however, that the individual fuel assemblies are loosely coupled; thus, the nuclear characteristics of an individual fuel bundle is not strongly affected by adjacent bundles. Therefore, it is possible to divide the analysis into a “lattice analysis,” the calculation of parameters of an infinite lattice of one bundle type and exposure, and a “core analysis,” the calculation of the parameters of an actual core, using the infinite-lattice parameters as input. The reload topical describes the lattice analysis for the various 8x8R and 8x8 bundles and also describes the analysis for the analysis methods for a reload core. The results for a specific reload will be submitted in the reload supplement.

As described earlier in Section 3.1, the reload 8x8R fuel assemblies have a longer active fuel length compared with the standard 8x8 reload fuel design and 7x7 fuel design. However, the top six inches and bottom six inches of the fuel column of the retrofit fuel assembly consists of fuel pellets with natural uranium enrichment. The central fuel region will contain pellets of varying enrichments depending on both the lateral coordinates within the bundle for the control of local peaking and the bundle average enrichment for reactivity requirements. Several lattice average enrichments are incorporated ranging from 0.711 wt.% (natural uranium) to 3.03 wt.%. Since these average enrichments include all rods and the six-inch end segments, the highest enrichment pellets may be as

high as 3.95 wt.%. The 8x8R fuel assembly also incorporates several fuel rods containing gadolinia as a burnable poison for local power shaping and reactivity control early in the cycle.

The retrofit 8x8R bundles incorporate two unfueled water rods symmetrically placed on either side of the lattice diagonal. This compares with a single water rod in the older 8x8 fuel bundle design. Each of the two water rods is also slightly larger than the water rod used in the 8x8 bundles. In addition, the fuel rod outside diameter (and pellet diameter) has been decreased by 10 (and 6) mils. The effect of the two larger water rods together with the smaller pellet diameter has resulted in a decrease in the fuel to water ratio.

Power Distribution

The limits on power distribution are determined by the specified acceptable fuel design limits (SAFDL) and by the accident and transient analyses. These limits are reflected in the technical specifications as limits on the linear heat generation rate (LHGR), average planar linear heat generation rate (APLHGR), critical power ratio (CPR), and (indirectly) on the total peaking factor (TPF). The criterion used for the review of power distributions is that these limits be assured during normal operation. For GE BWRs this criterion is met by monitoring the gross radial and axial power distributions and by pre-calculating the local power distributions.

Local power distributions and local peaking factors for the various 8x8R and 8x8 designs have been pre-calculated as a function of exposure and void fraction as part of the lattice analysis for uncontrolled and controlled configurations. The retrofit 8x8R bundles generally have lower peaking factors than the 8x8 and 7x7 bundles for the same assembly exposure. However, because an actual reload core will contain a mixture of fuel assembly exposures, and since local peaking varies with exposure, the new 8x8R bundles will not necessarily have the lowest local peaking factors at any given time during the reload cycle.

Exposures will be tracked by the plant process computer for each six-inch segment of each fuel bundle. Thus in principle, the local peaking factor can be calculated on-line anywhere in the core at any time. However, the approach taken for transient analysis purposes is to choose the peaking factor experienced by each fuel type at the end of a given exposure interval and to use this value in the safety analyses. The exposure interval is the entire cycle except in cases where the exposure-dependent limits are utilized (see Section 5.5). The bounding local peaking factors will be listed by fuel type in the reload supplement. The staff finds this to be acceptable.

GE has provided no local power distribution or local peaking factor information other than a locus enveloping the local peaking factor as a function of exposure for all 8x8R and 8x8 designs. This information is necessary for the staff to perform audit calculations. Furthermore, the staff will require further plant-specific information on local power distributions and local peaking factors for 8x8R core reloads involving fuel bundle types (designations) for which this information has not already been provided to the staff. Furthermore, reload applications which use end-of-cycle local peaking factors for transient analysis inputs which are less than the upper bounding envelope curve appearing in Figure 3-4 of the reload topical will be required to provide or reference additional local peaking factor information supporting the values selected.

Gross power distributions (radial and axial) will be monitored by periodic TIP scans, which will be updated between scans as necessary by means of the LPRM detectors. This basic method is unaffected either by a core reload or the new 8x8R fuel design. The 8x8R reload fuel will have a different void and axial power distribution than the 7x7 and 8x8 designs, due to the additional liquid water contained

in the two larger water rods and the presence of natural uranium at the fuel column ends. The calculational method used to transform detector signal to neutron flux and power, discussed in Reference 42, tracks 235U, 239Pu, and 240Pu inventories in six inch segments for each assembly. An iterative technique is used to obtain self-consistent axial power and void distributions. This calculational method provides the power distribution in individual bundles of the new 8x8R and previous fuel designs in an acceptable manner.

Although the gross, radial, and axial power distributions will change due to the change in void feedback (and radial self-flattening) this is not a major effect. Since incore methods are used to monitor APLHGR, LHGR, CPR, and TPF the assurance of operating limits is independent of changes in gross power distribution and, therefore, the changes in gross power are not of direct safety significance.

Reactivity Coefficients

Limits on reactivity coefficients are set by the transients, accidents, and thermal-hydraulic stability considerations, and General Design Criterion 11 which requires that in the power operating range the net effect of the prompt inherent nuclear feedback characteristics tend to compensate for rapid increases in reactivity. In a BWR core, the coolant is nearly isothermal at power operating conditions, and the only significant independent coefficients of reactivity are the Doppler coefficient and the void coefficient. During startup, there is also the effect of a moderator temperature coefficient.

Because the Doppler coefficient is least negative for unexposed (plutonium-free) fuel, the "worst case," least negative condition for the Doppler coefficient is at BOC. Also, since the new 8x8R fuel design has a different water to fuel ratio, void distribution, Dancoff factor, and pin self-shielding the behavior of the Doppler coefficient as a function of void and exposure is somewhat different than that of the standard 8x8 fuel design. These coefficients have been calculated(3) for the various 8x8R and 8x8 fuel bundle types as part of the lattice analyses. The overall value remains negative under all conditions, thus meeting the requirements of GDC 11. The EOC (or exposure interval) Doppler coefficient for a reload cycle will be reported in the reload supplement, and when multiplied by a 0.95 factor of conservatism will be used as input for the core wide transient analyses. Further discussion of the "design conservatism factors" is provided in Section 6.2.2 herein.

The void coefficient is the dominant reactivity feedback coefficient. Void coefficients have been calculated for the various 8x8R and 8x8 designs as part of the lattice analysis. They are always negative under all conditions encountered during reactor operation. The accident and transient analyses place lower as well as upper limits on the algebraic value of the void coefficient, depending on whether a power increase or a decrease event is being considered. The effect of the second water rod in the 8x8R fuel design is to reduce the absolute magnitude of the void coefficient. The EOC value for the cycle (or exposure interval) will be reported in the reload supplemental document, and when multiplied by a design conservatism factor of 1.25 will be used as an input for the core wide transient analyses.

The moderator-temperature coefficient will no longer be provided on a plant-specific basis in the reload supplement. This coefficient can become slightly positive during certain operating conditions. Its importance, however, is only during startup and shutdown, is very slow acting, and is overshadowed by the Doppler coefficient. Because of this effect, and since no credit is taken for the moderator-temperature coefficient in the safety analyses, this coefficient has no direct safety significance. Therefore, the staff finds it acceptable to exclude the moderator-temperature coefficient from the safety analyses.

GE has provided no quantitative information on Doppler or void coefficients other than a range covering all 8x8R and 8x8 designs. Again, the staff cannot perform audits without this information. Also, the dynamic coefficients just as transient analysis inputs cannot be related to the nuclear design information provided in the reload topical without knowing the delayed neutron fraction as a function of exposure. Therefore, this information will be required in a plant-specific basis for all reload applications involving 8x8R fuel types for which this information has not been submitted.

Shutdown Capability

Shutdown margin, reactivity control systems, and scram reactivity fall under General Design Criteria 20 through 29. When applied to BWR core reloads, these Criteria reduce to the following requirements.

- The control rods must be capable of rendering the core subcritical in a cold, xenon-free condition, at any time in the cycle, with the highest worth control rod stuck out of the core.

- The shutdown margin and scram reactivity curve must be consistent with the accident and transient analyses.

- The Standby Liquid Control System (SLCS) must be capable of rendering the core subcritical, in a cold, xenon-free condition, with the control rods at their minimum position, at any time in the cycle.

These requirements appear in each of the applicable BWR's Technical Specifications.

The retrofit 8x8 fuel bundles incorporate the use of small amounts of gadolinia as a burnable poison. With burnable poison in the reload core, fuel reactivity initially decreases, as samarium builds in, then increases to a peak as the burnable poison burns out, then finally decreases until EOC, as fissile nuclide depletion becomes dominant. Thus, the point of maximum core reactivity is generally not at BOC but occurs later in the cycle. Burnable poison depletion also causes some control cells to increase in worth, while others may decrease, thus causing the location of the strongest rod to change. The effective multiplication factor in a core configuration with the strongest control rod out, under a cold, xenon-free condition, will be calculated for each reload core. This calculation gives shutdown margin directly. The calculations will be performed for various exposures during the cycle, with a search for the strongest control rod at each exposure. To ensure conservatism, the minimum expected exposure for the previous cycle is assumed in the depletion calculations. The information in the reload supplement will include the BOC multiplication factors for all rods out and all rods in as well as the strongest rod out configurations, including the maximum reactivity increase with exposure. This information will adequately demonstrate that the technical specification shutdown margin requirements are met.

The plant-specific scram reactivity versus time (scram curve) will be calculated by GE for EOC (and at the end of an exposure interval) conditions. This nominal scram curve will be multiplied by an 0.80 factor for model uncertainty and error allowance, and will be used in the transient analyses. Control rod insertion times assumed in the calculation will correspond to the slowest scram speed allowed by the plant Technical Specifications. These conditions are conservative for earlier exposures because of the decrease in rod density as the EOC condition is approached. That is, at EOC there are fewer partially inserted rods to insert reactivity more quickly than the fully withdrawn rods. The power distribution used in the calculation at each exposure is based upon the Haling axial power and exposure distributions. Since actual EOC power distributions are generally more bottom-peaked than

the Haling distribution prediction, this calculation is considered conservative. Therefore, the calculation of the scram curve is acceptable, and the second of the three requirements will be demonstrated.

GE calculates the plant-specific multiplication factor and shutdown margin for a sodium pentaborate concentration in the coolant corresponding to the Technical Specifications basis for the SLCS. These calculations will be for an exposure corresponding to the maximum fuel reactivity with the core in a cold, xenon-free state. Additionally, all control rods are assumed to be out of the core. The results to be given in the reload supplemental are necessary to demonstrate that the SLCS Technical Specification requirements are satisfied. Thus, the third requirement will be demonstrated for the alternate shutdown system.

Control Rod Patterns and Reactivity Worths

The limits on control rod worths are necessary to limit the consequences of postulated accidents and transients. Additionally, the reactivity addition resulting from a single control rod notch shall not result in a reactor period which the operator cannot safety control. However, the maximum worth of one notch has never been excessive in the power range for an operating BWR.

The control rod drop accident analysis requires limits on the dropped rod worth during startup and the inadvertent rod withdrawal transient analysis requires limits on individual rod worth during power operation. During startup, the maximum dropped rod worth is assured by limiting the permissible control rod withdrawal sequences and patterns.

Modern GE BWRs are equipped with a “hard-wired” Rod Sequence Control System (RSCS). This system enforces the pattern restrictions during startup. Additionally, a process computer routine called the Rod Worth Minimizer (RWM) is intended to verify rod patterns during startup. The accident analysis, however, conservatively assumes that the RWM fails and only the RSCS is functioning. Plants equipped with a RSCS are called “Group Notch” plants because of the withdrawal sequence permitted by the RSCS. This method of pattern restriction is independent of fuel type and remains acceptable.

Older BWR plants (i.e., all BWR/2 and BWR/3 plants plus Vermont Yankee) are not equipped with a “hard-wired” (RSCS) system to enforce control rod withdrawal patterns during startup. These plants instead take credit for the Rod Worth Minimizer (RWM), a process computer routine which monitors startup rod patterns. Additionally, these plants now utilize a new, more sophisticated withdrawal sequence[43] programmed into the RWM to further reduce the worth of a dropped rod. This upgraded sequence is called the Banked Position Withdrawal Sequence (BPWS). This method of pattern control, like the Group Notch system, is independent of fuel type and remains acceptable.

During power operation, the voided condition of the moderator greatly reduces the worth of a dropped rod, and the control rod drop accident consequences become less severe. Therefore, above a specified power level (given in the plant’s Technical Specifications) the RSCS and/or RWM are automatically disengaged, and there is no safety-related system to control rod patterns. Further discussion may be found in the evaluation of the control rod drop accident analysis methods and results appearing in Section 7.3 herein.

The limits on rod worth resulting from the analysis of the rod withdrawal error transient are enforced by means of the Rod Block Monitor system (RBM). When a control rod is selected for movement, the nearest LPRM detectors are automatically monitored, and a rod block is effected when the local power

increase reaches the RBM setpoint. Thus, the RBM restricts the control rod worth through the local power coefficient, rather than via control rod patterns. This system is also independent of fuel type, and remains acceptable. Further discussion is provided in the evaluation of the rod withdrawal error transient methods appearing in Section 6.2.1 herein.

Actual vs. Reference Core Loading Patterns

The actual refueled core can be somewhat different from the reference core loading assumed in the plant-specific reload safety analyses. The staff is currently generically reviewing how extensive these variations may be and still have an acceptable safety analysis. In the interim, until our generic review is complete, deviations from the reference core will be examined on a plant-specific basis.

4.3 Reactivity of Fuel in Storage

The reload topical report addresses the reactivity of the retrofit 8x8 fuel in storage. The technical specification requirement for the storage of fuel in a spent fuel storage pool is that the effective multiplication factor, K_{eff}, of the fuel as stored in the fuel storage racks is less than or equal to 0.95 for all storage conditions.

The above requirement is met for both the Type A and Type B racks designed by GE if the uncontrolled infinite lattice multiplication factor, K_{∞} of the 8x8R fuel bundle in the reactor core configuration is less than or equal to 1.30. GE analyses show that all of fuel bundle (enrichment) types (shown in Table 1-2 of Reference 3) have a K_{∞} of less than 1.30 at the peak reactivity exposure point and 65°C. Thus, all plants utilizing the Type A or B racks will meet the fuel storage subcriticality requirement for any fuel enrichment appearing in Table 1-2 of the reload topical. The plants with either Type A or Type B racks on the date of issuance of this evaluation are Dresden Units 2 and 3, Quad Cities Units 1 and 2, Fitzpatrick, and Hatch Unit 1.

All of the other plants shown in Table 1-1 of Reference 3 have, or will shortly have, high density fuel storage racks and thus are not covered by the GE criticality analyses. The staff has evaluated the reactivity of fuel in storage for those plants in Table 1-1 which do not have Type A or Type B GE racks. Based on our evaluation, we have determined that all of these plants, except Cooper, can meet the aforementioned subcriticality requirements for all of the fuel assembly enrichments listed in Table 1-2. Cooper can also meet these requirements but only for reload 8x8R fuel assemblies with a lattice average enrichment of not more than 2.72 wt.% U-235.

In summary, therefore, it is the staff's finding that all of the plants appearing in Table 1-1 of Reference 3, except for Cooper, may currently receive and store any 8x8R reload fuel assembly. Cooper may store any 8x8R reload assembly having a lattice enrichment of the 2.72%. Additionally, if 8x8R assemblies with more than 3.03 wt.% lattice enrichment are to be designed and fabricated by GE for storage, then the staff will reevaluate these conclusions. Finally, licensees of BWR plants currently having Type A or Type B GE racks which elect to install replacement high density racks are required to submit adequate design information related to the new racks for our review.

4.4 Plant Specific Nuclear Design Information

The nuclear design information which will be provided on a plant-specific basis appears in Appendix A to Reference 3. However, most nuclear design parameters are either generic or are given as inputs to the transient and accident analyses. Therefore, the nuclear design information which is plant-specific

consists only of the shutdown capability (including SLCS) reanalysis results. Based upon our review, we find the Appendix A information to be sufficient, with the requirements noted in Section 4.3 herein on local power distributions and reactivity coefficients.

5.0 THERMAL AND HYDRAULIC DESIGN EVALUATION

Our review of the thermal-hydraulic design aspects addressed the applicability and adequacy of the referenced and described thermal and hydraulic models and methods[3] for mixed cores and the new retrofit fuel design; the compatibility of the hydraulic characteristics of the 8x8R fuel design compared to the 7x7 and standard 8x8 fuel types fuel cladding integrity safety limit MCPR determination, and the adequacy of the supplemental thermal-hydraulic design-analysis information to be provided on a plant-specific basis. Our evaluation of these thermal-hydraulic aspects is contained in the following subsections.

5.1 Thermal and Hydraulic Methods

5.1.1 Steady-State Hydraulic Methods

The core steady-state hydraulic analysis is performed to establish flow, pressure, enthalpy, void, and quality distributions within the core. This analysis also establishes initial reactor coolant conditions for reactor physics calculations and the analysis of anticipated operational transients. The hydraulic model of the reactor core includes descriptions of the orifices, lower tie plates, fuel rods, fuel assembly spacers, upper tie plates, fuel channels and core bypass flow paths. The core steady-state hydraulic model is composed of separate effects models, which simulate various pressure loss characteristics, and composite models, which simulate the channel-by-channel and core bypass flow paths.

The separate effects hydraulic models of the core and channel components consider frictional, local, elevation, and acceleration hydraulic pressure loss characteristics. The frictional characteristics of the core components are modeled by use of the single phase frictional pressure drop equation with a two-phase multiplier, Φ_{TPF}. GE has correlated these multipliers, on a best-estimate basis, to a significant amount of multi-rod geometry date[44], that are representative of modern BWR fuel bundles. The largest collection of this data was acquired from the ATLAS loop during development testing for the GEXL correlation. The data for these correlations cover the range of BWR conditions. On this basis, the use of these correlations is appropriate.

The local pressure drop characteristics have been established in a manner similar to the formulation used for the frictional pressure drop characteristics. It differs to the extent that a local pressure loss coefficient is substituted for the product of friction factor and characteristic length-to-diameter ratio. This is a common hydraulic analysis procedure. This modeling has also been verified[15] experimentally throughout the range of conditions by the ATLAS loop tests for the GEXL correlation[44]. This modeling technique is used to simulate the pressure losses of the orifice, lower tie plate, spacer, upper tie plate, and lower tie plate bypass flow holes.

The acceleration pressure drop has two components, i.e., area change and density variation. The area change is modeled similar to the local pressure drop. Since an area change is generally treated in this manner, this modeling approach is acceptable. The density variation uses the same formulation as the elevation pressure drop characteristic, except that it accounts for density variations along the fluid channel. This is also a standard hydraulic analysis practice, and is acceptable.

These separate effects hydraulic characteristics are utilized to simulate the hydraulic conditions through the orifices, lower tie plates, fuel rods, water rods, fuel rod spacers, upper tie plate and fuel channel. The core bypass flow paths have been modeled from experimental[45] results and verified by analytical techniques. These tests were previously reviewed and were found to be acceptable for this use.[46]

The above separate effects hydraulic models, which simulate reactor core component pressure losses and flow paths, permit a composite model of a single fuel channel to be simulated. The fuel channel is then categorized into a fuel “channel type” in order to reduce the number of nodes in the analysis. The fuel channels are grouped by “channel type” and modeled as a single typical channel of that type. Thus, the flow distribution of a particular fuel channel is assumed to be the same as the typical channel for that fuel channel type.

A channel type is classified by five characteristics: (1) orificing type (central or peripheral), (2) fuel geometry (7x7, 8x8, or 8x8R), (3) relative bundle power (high power or average), (4) lower tie plate type (drilled or undrilled), and (5) bypass type (finger springs or no finger springs).

With regard to the core relative bundle power distribution, sensitivity studies show(15) that classification by high power and average power density channels adequately models the core flow distribution. This is due to the fact that average channel characteristics are dominant in establishing the core pressure drop. Therefore, categorization as a function of channel power density need not be broken down into additional sub-channels. The other characteristics completely cover the range of channel type possibilities.

In order to perform channel type categorizations, each fuel channel must have the same pressure drop across its length. This is a major assumption of the steady-state hydraulic analysis. This has been shown to be valid by flow distribution and pressure drop measurements in several operating BWR’s(47,48,49). These tests further show that the pressure drop across any fuel channel or bypass flow path in the core is the same as for any other fuel channel or bypass flow path in the core. The above referenced documents have been previously accepted(50) for justification of this assumption.

The steady-state hydraulic analysis uses a digital computer code to calculate the hydraulic characteristics of the core. The code utilizes a trial and error iteration for flow rate, pressure drop, enthalpy, quality, and void distribution for each channel type. It equates the total plenum-to-plenum differential pressure across each flow path, and matches the sum of the flows to the total core flow. Comparison(15) of analytical predictions to tests performed in the ATLAS test facility as a function of pressure drop, mass flux, and bundle power show reasonably good agreement, i.e., <6% error for the range of interest. This generally qualifies the calculational technique and modeling for the steady-state hydraulic analysis methods.

In view of the above the staff finds the steady-state hydraulic analysis methods to be adequate for reload core licensing applications incorporating up to three different fuel types, including the 8x8R fuel design.

5.1.2 Thermal-Hydraulic Stability Methods

Thermal-hydraulic stability analyses for BWR core reload applications consider two aspects: (1) channel hydrodynamic performance and (2) reactor core (reactivity) performance. General Electric evaluates the channel hydrodynamic performance with a revised version or the FABLE code.(51) For the channel hydro-dynamic stability analysis, transfer functions are calculated for each fuel type. The fuel dynamic response is established based on linear feedback models of the channel continuity, momentum, and energy equations. The channels are then coupled hydrodynamically with the core neutronics to evaluate reactor care (reactivity) stability. The core model calculates the linearized reactivity response of the reactor system with density-dependent feedback. This analysis is performed using the FABLE code for given reactor flow rate, core power, and power distribution. A third type of stability, total reactor system stability, is not evaluated for reload submittals since only the core and

channel hydrodynamic performance vary from one reload cycle to the next. Thus, the BWR control system response need not be reevaluated for core reloads.

The above stability aspects are restricted by a criteria or limit on the decay ratio. The decay ratio is the ratio of the magnitude of the second overshoot to the first overshoot resulting from a step perturbation to the core. For a BWR, the decay ratio is generally a measure of neutron flux response to a step change in reactivity. GE has proposed that the acceptance criteria for BWR stability correspond to the analytical threshold of instability. That is, GE considers a plant to be stable with adequate margin when the predicted decay ratio, at the most adverse time in the cycle and the most adverse operating state, is less than 1.0.

The staff has expressed generic concerns regarding reactor core thermal-hydraulic stability at the least stable reactor condition. This condition could be reached during an operational transient from high power if the plant were to sustain a trip of both recirculation pumps without a reactor trip. The concerns are motivated by observed increasing predicted decay ratios as the equilibrium fuel cycle is approached and as the reload fuel design changes. The staff concerns relate to both the consequences of operating at a decay ratio of 1.0 and the capability of the analytical methods to conservatively predict decay ratios.

The General Electric Company is addressing these staff concerns through meetings, topical reports and a stability test program. The participants of the on-going stability test program include GE and the licensee of a large BWR/4. Although a final test report has not yet been received by the staff, it is expected that the test results will aid considerably in resolving the staff concerns.

As an interim measure, the staff has imposed a Technical Specification requirement on operating BWR's which restricts planned operation in the natural circulation mode, whenever the predicted decay ratio is greater than 0.5 at the least stable reactor operating state. The staff believes that this requirement provides adequate stability margin, based on our review of GE's analytical capabilities and a comparison of predicted to measured data. Accordingly, we will continue the above interim measure for 8x8R reload cores until the generic stability issues are resolved.

Finally, GE has proposed to delete the plant-specific reload presentation of the predicted stability operational design guide decay ratios on the basis that they are simply a "guide," and have no safety significance. The staff considers GE's proposal to be acceptable. The staff agrees that the design guide is simply a guide and currently has no safety significance. This finding is based on current BWR operating procedures and technical specification requirements. If the modes or requirements of operation changes, (e.g., remote load control) the significance of the stability characteristics at the operational design guide for automatic flow control will be reevaluated.

5.2 Fuel Cladding Integrity Safety Limit MCPR

General Design Criterion 10 requires that the reactor core be designed with appropriate margin, to assure that specified acceptable fuel design limits (SAFDL) are not exceeded during any condition of normal operation, including the effects of abnormal operational transients. In order to avoid fuel damage caused by overheating of the cladding, transient consequences are limited such that more than 99.9% of the fuel rods would be expected to avoid boiling transition during a transient event. This design basis has been previously accepted[52] for initial and reload core applications in connection with the staff's review of the General Electric Thermal Analysis Basis (GETAB) method.[53]

The above design basis can also be stated in terms of a statistically determined Minimum Critical Power Ratio (MCPR) safety limit. The GETAB statistical analysis procedure, including codes, correlations and analytical procedures has also been previously reviewed and approved by the staff in connection with MCPR safety limits established for initial core and reload core applications. Our review, therefore, centered upon evaluation of the adequacy of the described statistical analysis procedures for 8x8R reload cores, which can contain up to three different fuel types (7x7, 8x8 and 8x8R) as well as a review of the adequacy of the key generic inputs of the statistical analysis.

The generic nominal values of the plant process variable (e.g., core flow, dome pressure) used in the GETAB statistical analysis, are shown in Table 5-2 of Reference 3. The values shown in the table correspond to the same previously approved generic 251/764 core selected for the GETAB statistical analysis, for operating BWR's which have previously reloaded with the standard 8x8 fuel assemblies. Substitution of the retrofit 8x8 reload fuel assemblies in the statistical analysis does not alter our previous conclusion on the acceptability of the generic core process variable parameter values selected.

The generic statistical analysis uncertainties associated with the core process variables, bundle power determination, CHF correlation, and fuel assembly manufacturing tolerances appear in Table 5-1 of Reference 3. The uncertainties are the same or more conservative than those shown in the GETAB report. The only uncertainties in the table which are potentially reload core or fuel design dependent are for TIP Readings, R-Factor, GEXL Correlation, and Channel Flow Area uncertainties. The standard deviation for the TIP Readings uncertainty is 8.7% whereas the GETAB report uses a 6.3% uncertainty. The latter uncertainty is appropriate for an initial core. The increase in TIP uncertainty to 8.7% is a consequence of the increase in the uncertainty in the bundle power measurement of a reload (exposed) core. This uncertainty is also considered to be generically adequate for retrofit 8x8 fuel assembly reload cores. Table 5-1 indicates an R-Factor uncertainty of 1.6%, which is also the same as that used for 8x8 reloads. The R-Factor uncertainty is derived from the uncertainty in the local power peaking distribution calculation. The addition of a second water rod in the retrofit fuel design is not expected to increase the uncertainty in the power distribution calculation, based on the approved neutronics methods. The 3.0% Channel Flow Area uncertainty, shown in Table 5-1, accounts for manufacturing and operationally induced variations in the free flow area within the assembly. Although the effective channel flow area for the 8x8R assembly is slightly different than for the 8x8 assembly, the manufacturing tolerances are the same. Thus, a channel flow area uncertainty of 3.0% (which is the same as the 8x8 assembly) is acceptable.

A value of 1.038 was selected for the nominal value of the retrofit 8x8 R-Factor. This compares with 1.098 and 1.100, for the 7x7 and 8x8 assembly R-Factors, previously used in connection with standard 8x8 core reloads. The generic design core-wide bundle histogram used in the new GETAB statistical analysis for 8x8R reloads appears in Figure 5-1a of Reference 3. The CPR histogram is also different from the histogram previously used in the GETAB statistical analyses of BWR 2/3/4 D-Lattice 8x8 reload cores. The revised histogram indicates fewer bundles at and near the MCPR safety limit. General Electric was requested to provide additional justification to support the new retrofit 8x8 R-Factor and CPR histogram which were used in the new statistical analysis.

The additional information[15] submitted by GE states that an all equilibrium 8x8R 251/764 reload core is expected to yield the least CPR mismatch between different reload batches, thus providing the flattest CPR distribution among reload cycles. To arrive at this conclusion, GE first surveyed various 8x8R reload core loading configurations, to establish which would yield the worst CPR distribution and then searched various control rod patterns, to determine which pattern would yield the most fuel assemblies at and near the MCPR operating limit at rated reactor power. The staff agrees that an

equilibrium 8x8R (all 8x8R) reload core will provide the flattest CPR histogram because the 8x8R fuel bundle R-Factor is relatively low and does not vary significantly with exposure compared with 7x7 or standard 8x8 assemblies. Since an equilibrium 8x8R reload cycle is limiting, the CPR histogram and bundle R-Factor used in the generic statistical analysis reflects a single (8x8R) bundle type. This is in contrast with the generic reload cycle selected for establishing the safety limit MCPR for standard 8x8 reloads which included both 7x7 and 8x8 fuel types. Thus, two different R-Factors were used in the generic determination of the standard 8x8 reload core safety limit MCPR. The flatter core power (and CPR) distribution in the new GETAB statistical analysis gives rise to a more adverse bundle-to-bundle critical heat flux (CHF) probability distribution when compared with an actual or expected reload core CPR histogram. The design histogram, therefore, results in a conservative prediction of the total number of rods in boiling transition when the limiting bundle is placed on the thermal (MCPR safety) limit.

The 8x8R generic R-Factor selected for the statistical analysis is equal to 1.038 and corresponds to beginning of cycle conditions. This compares with 1.100 and 1.098 for the 8x8 and 7x7 R-Factors previously used for the GETAB statistical analysis for standard 8x8 reloads. The additional information[15] submitted by GE states that the lower bundle R-Factor associated with the 8x8R fuel design compared to the 7x7 and standard 8x8 fuel designs stems from the significantly flatter intra-assembly power distribution of the two water rod fuel design. The flatter local power distribution (and lower R-Factor), therefore, also gives rise to a more adverse rod-by-rod CHF accounting when the limiting bundle is placed on the thermal limit.

To assess the degree of conservation afforded by the design basis CPR histogram (and R-Factor) a statistical GETAB analysis was performed[15] using both a severe and typical large BWR 8x8R reload core CPR histogram. The statistical analysis results show that a .04 to .06 MCPR margin is representative of the conservatism available with the design basis histogram.

In view of the information provided, the staff concludes that when using the previously approved GETAB statistical method, the new CPR histogram and R-Factor, together with the continued applicability of the generic core and GETAB uncertainties inputs selected for the statistical analysis, provides an adequately conservative basis for generically determining the safety limit MCPR for BWR reload cores incorporating the retrofit 8x8 fuel design. Accordingly, the staff finds the calculated 1.07 safety limit MCPR to be generically acceptable as a transient limit for 8x8R reload licensing applications associated with the plants listed in Table 1-1 of the reload topical report.

5.3 Plant Specific Thermal Hydraulic Design Information

The thermal and hydraulic design information, which will be provided on a plant-specific basis, appears in Appendix A to Reference 3. Based on our review of Appendix A, we find the proposed thermal-hydraulic information to be adequate for our review of the plant-specific thermal-hydraulic aspects of BWR core reloads when including the core reactivity stability decay ratio as a function of core power along the minimum pump speed line for those plants which are precluded from planned operation in the natural circulation condition.

6.0 ABNORMAL OPERATIONAL TRANSIENTS EVALUATION

In connection with abnormal operational transients (i.e., "transients") our generic review addressed the following: safety criteria and related safety limits for the fuel and reactor coolant pressure boundary; analytical codes and methods employed for simulating fuel, core and plant system transient performance; generic features and design parameters associated with the new 8x8R fuel assembly design which could affect transient behavior; reload core safety-analysis conditions, including transient events analyzed, initial conditions, protection and control equipment performance assumptions, and generic transient analysis results where presented. We have also reviewed the exposure dependent limits margin improvement option and the adequacy of the supplemental transient analysis information to be provided on a plant-specific basis. Our evaluation of these transient aspects is contained in the following subsections.

6.1 Transient Criteria-Safety Limits

Although there are numerous General Design Criteria (GDC) which relate to reactor performance and protection during abnormal operational transients, only two need be considered with regard to reactor safety reviews associated with BWR core reloads. These are GDC-10 and 15. GDC-10 requires that specified acceptable fuel design limits shall not be exceeded during a transient, while GDC-15 requires that design conditions of the reactor coolant pressure boundary shall not be exceeded during the most severe pressurization transient.

In connection with the above requirements, the following specific criteria have been considered generically applicable for transients associated with GE BWR core reloads:

A. The calculated critical power ratio of any fuel assembly in the core shall not violate the safety limit minimum critical power ratio, which is established such that 99.9% of the fuel rods in the core would not be expected to experience boiling transition.

B. The calculated local linear heat generation rate along any fuel rod in the core shall not violate the safety limit linear heat generation rate, which is established such that 1% plastic cladding strain is not exceeded.

C. The calculated peak pressure associated with the reactor coolant pressure boundary (RCPB) shall not exceed 110% of the reactor coolant pressure boundary design pressure.

D. The calculated release of radioactive material to the environs shall not exceed the radiological limits discussed in 10CFRPart 20 and Appendix I to 10CFR50.

In connection with Criterion A above, GE has performed a generic statistical calculation to determine the minimum critical power ratio (MCPR), such that 99.9% of the fuel rods in an 8x8R reload core would avoid boiling transition during transients affecting the entire core. Our evaluation of generic application of a 1.07 safety limit MCPR to the BWRs listed in Table 1-1 of the reload topical is described in Section 5.2 herein. In addition, the staff continues to require that the safety limit MCPR not be violated during transients affecting the core locally.

With regard to Criterion B, GE has performed generic thermal-mechanical calculations to determine the local liner heat generation rate (LHGR), which would result in 1% cladding inelastic diametral strain in a fresh or exposed fuel rod over life. The resulting LHGR safety limits for the retrofit 8x8, standard 8x8, and 7x7 fuel rod types are evaluated in Section 3.3 herein.

With regard to Criterion C, the ASME Boiler and Pressure Vessel Code, Section III, permits a peak transient nuclear steam supply system pressure equal to 10% above the vessel design pressure. Since the design pressure for the applicable plants is 1250 psig, the ASME Code would permit a transient pressure up to 1375 psig. Thus, the staff considers a reactor coolant pressure boundary pressure safety limit of 1375 psig to be appropriate for transients. Additionally, it should be noted that several of the BWR plants appearing in Table 1-1 of the reload topical incorporate safety valves with discharge lines that are not piped directly to the containment suppression pool. For these plants, the opening of one or more safety valves during a severe pressurization transient would cause high energy steam to be discharged directly to the primary containment atmosphere. This would result in an undesirable pressure loading of the primary containment and exposure of qualified safety system equipment within the containment to a high energy steam environment. Additionally, lifting safety valves could lead to the possibility that one of the valves could fail to reseat. Such an occurrence would result in an uncontrolled blowdown of the primary system. To preclude this undesirable sequence of events, GE currently recommends that (for BWR plants equipped with safety valves with discharge lines not piped to the suppression pool) a minimum 25 psi margin be available at all times during the operating cycle between the predicted worst-case transient steam line pressure and the lowest safety valve setpoint. For “transients” (as distinct from the limiting overpressurization event discussed in Section 8.0) this internal GE requirements has the effect of superseding the 1375 psig ASME code allowable pressure limit for the affected plants. Implementation of this requirement can have the effect of limiting reactor thermal power toward the end of cycle. The staff considers the internal GE recommendation to be acceptable.

Finally, if Criteria A, B, and C are satisfied for each abnormal operational transient event analyzed for a core reload application, it can be concluded that the radiological consequences will also necessarily be within the acceptable doses calculated for the FSAR. Thus, Criterion D will be automatically satisfied.

6.2 Transient Analysis Methods

GE utilizes a number of interrelated computer codes, correlations and analytical procedures to simulate the fuel, core, and composite plants system transient performance of BWR’s for a variety of abnormal operating transients. These codes, correlations, and procedures are used to provide a conservative prediction of the maximum changes in the safety-related parameters (i.e., LHGR, CPR, and reactor coolant system pressure) that can occur during the most limiting transients.

All of the basic transient codes and methods described and/or referenced in the reload topical have been previously utilized generically by GE in connection with BWR safety analyses performed for both initial core and reload core licensing. In general, the staff has previously generically reviewed and approved the codes, correlations, and analytical procedures (e.g., SCAT Code, 3D BWR Simulator Code, GEXL correlation, and GETAB method) for licensing applications. However, in some cases, a specific code (e.g., REDY Code), although it has been used by GE for initial and core licensing applications, and has considerable recognition by the staff that it is acceptable for such purposes, it may not have been formally reviewed and approved by the staff. Accordingly, it should be mentioned at this juncture that the staff, in the review and evaluation of the reload topical, has not endeavored to completely and generically resolve the outstanding concerns and issues associated with the ongoing reviews of the reference transient codes and methods. This would go beyond the scope of review of the topical. We have, instead, limited our review to an evaluation of the applicability of transient codes and correlations as well as the adequacy of the reload transient analysis procedures in connection with (a) the new water rod 8x8R fuel design and (b) mixed BWR reload cores containing up to three different fuel types (7x7, 8x8, 8x8R). Our generic approvals where stated in this evaluation, therefore,

should be interpreted as only broadening or extending the recognized state of acceptance of the base codes and methods. That is, our approvals herein do not attempt to resolve any open issues associated with the base codes and methods.

6.2.1 Transient Analysis Methods for Local Events

Rod Withdrawal Error

The rod withdrawal error (RWE) is an abnormal operational transient which affects only a limited number of fuel assemblies in the core. The local and radial peaking factors can increase substantially in the fuel assemblies in the immediate vicinity of the withdrawn control rod. Thus, this transient is of safety concern with regard to potential fuel rod overheating (MCPR) and clad overstraining (1% plastic strain). Since the rate and magnitude of the gross core power increase for this event is low, the reactor pressure increase is not large enough to be of concern relative to the RCPB pressure safety limit.

The method used by GE to calculate the consequences of this transient involves a series of steady-state BWR Simulator Code[38] calculations. The simulated reload core is assumed to be at its most reactive exposure with no xenon or samarium present. The rod pattern selected is such that the maximum worth control rod is fully inserted and the laterally adjacent or diagonally adjacent bundles are at their thermal operating limits. A series of steady-state calculations are then performed for succeeding positions of the worst-case control rod, using the BWR Simulator Code. The code also calculates the response actions of the Rod Block Monitor Subsystem assuming the most adverse detector failure allowed by the plant technical specifications. The results are then used to select a setpoint for the Rod Block Monitor such that neither fuel cladding integrity safety limit is violated.

Using a series of steady-state calculations to approximate the transient behavior of the core is the standard method for all GE BWR reloads. Past analyses and reviews have shown that even at the maximum control rod drive withdrawal speed and rod worth the rate of power increase is small and, thus, a quasi-static approximation (in the power range) is valid. Because the new 8x8R fuel rod has a faster thermal time constant than the 7x7 or 8x8 rod types, and because the codes assume homogenized bundles, both the RWE quasi-static procedure and the codes remain generically acceptable for calculating bundle power and local rod powers for 8x8R reload applications.

The rod withdrawal error event can result in the thermally limiting fuel assembly having a significant R-Factor axial variation due to the partially withdrawn control rod when the rod block occurs. A comparison of critical bundle power predictions, using the GEXL correlation, with available test date[54] having axially varying R-Factors, shows that the GEXL correlation predictions result in a systematic nonconservative bias for such conditions. Accordingly, the staff will continue to generically review GE’s overall RWE methodology for predicting bundle CPR’s while evaluating the acceptability of the RWE analysis results for core reloads on a plant-specific basis.

Fuel Loading Error

The fuel loading error (FLE), like the control rod withdrawal error, is an abnormal operational transient which affects only a limited number of fuel assemblies in the core. The local and radial peaking factors increase sufficiently in the misloaded bundle to be of safety concern relative to fuel rod overheating (MCPR) and clad overstraining (1% plastic strain). Since there is no attendant increase in the gross core power, the RCS pressure is unaffected by a FLE. Thus, there is no concern relative to the RCPB safety limit.

The fuel bundle loading error analysis performed by GE for core reloads considers both misoriented and mislocated fuel bundle cases. Our evaluation of the methods used for each type of loading error condition follows.

For the misoriented fuel loading error, an infinite lattice diffusion-depletion calculation is first performed. From this calculation, rod-by-rod local power peaking, K_∞, and R-Factor are established for both a 90° and 180° misorientation of a fresh reload 8x8R assembly. The maximum R-Factor and bundle power increase, due to the rotation, are then computed versus exposure for the entire cycle. A slight adjustment to these values is made to account for differences between the infinite lattice assumption and the discrete four bundle configuration. Finally, using the adjusted maximum R-Factor and maximum power change, the misoriented bundle critical power ratio is computed, using the 3-D BWR Simulator Code.[38]

By letter[55] dated June 1, 1977, GE proposed a revision to the above FLE analysis method which involved a change in the assumed size of the water gap surrounding the misoriented bundle in a manner which would more accurately model the misoriented bundle geometrical tilt. This is in contrast with the present method wherein a constant water gap is assumed. The gap size assumed in the old procedure is the same as for a correctly oriented assembly. This conservatively maximized the bundle power increase for the misloaded bundle, since the rods with the highest enrichment are assumed adjacent to an unrealistically large water gap. In actuality, the misoriented fuel bundle will lean toward the control rod water gap. This will reduce the gap size toward the top of the bundle, thus lowering power peaking toward the top of the assembly in the higher enrichment rods. A detailed description of the new analysis methods for the misoriented bundle is contained in a GE letter[54] dated November 30, 1977.

For the case of the mislocated bundle loading error, a revised calculational procedure has also been proposed by GE as outlined below:

- All symmetrically unique four fuel bundle cells in the core are identified.
- Cells which would be expected to yield the largest decrease in CPR are then identified for bundle mislocation analysis consideration. The cells selected generally have relatively low reactivity, since placement of a fresh bundle of high reactivity in a low reactivity cell results in a larger cell reactivity increase. Thus, the bundle power increase is larger, yielding a larger ΔCPR.
- The above loading of a high reactivity bundle into a low reactivity cell (for each cell selected) is then analyzed with the 3-D BWR Simulator Code. The CPR response is calculated for the anticipated control rod patterns throughout the cycle.
- The CPR of each bundle in the core is multiplied by the factor necessary to reduce the CPR of the limiting bundle in the simulated core to the operating limit MCPR.
- After the lead bundle is placed on the thermal limit, the maximum change in CPR is calculated for each of the cells containing the assumed mislocated bundles. The maximum change in CPR is identified.
- With the maximum ΔCPR identified, some fuel cells are eliminated from subsequent consideration, since even if these cells had a fuel loading error, their minimum CPR during the cycle would be acceptable.

- For the cells which violate the safety limit, the process is repeated, starting from the second step, until the limiting mislocation error is determined.

A detailed description of the above procedure is also contained in GE's November 30, 1977 letter to the staff.

Additionally, GE has proposed [55] a second procedure for the mislocation fuel loading error analysis. This alternate procedure differs from the first only in the method of predicting the CPR in the monitored bundles. A statistical comparison of actual process computer CPR data with Haling power distribution fuel bundle CPR predictions was performed. Using the operating data, it is possible to perform a Haling power distribution calculation for determining the fuel bundle CPRs and then to correct them to achieve an improved prediction of the actual CPR in each fuel bundle. A detailed description of this alternative procedure is presented in Reference 54.

The two proposed procedures for the fuel mislocation error differ from the current method in that iterative calculations are not performed. Additionally, rather than adjusting the core CPR distribution, the mislocated bundle location is assumed to be at the operating limit MCPR using the current method. The proposed analysis methods remove some of the excessive conservatism of the current method, and thus, models the bundle mislocation error more realistically.

In the evaluation of these methods, we have considered the probability and consequences of possible fuel loading errors, the measure of conservatism inherent in the analysis methods together with the ability of the BWR's monitoring systems, to detect fuel loading errors. GE has proposed[55] that a safety limit of 1.0 be applied for bundle mislocation and misorientation errors. The staff has this proposal under review. In the interim, we will continue to require that the most severe fuel loading error shall not result in a violation of the safety limit MCPR or the safety limit LHGR. Furthermore, with regard to the proposed new methods for the mislocation loading error, we have found[56] these methods to be acceptable as proposed. Additionally, the staff finds the proposed methods for predicting MCPR's for the misoriented FLE to be acceptable when adequately adjusted to compensate for identified non-conservatism in the CPR predictions for conditions involving axially varying R-Factors. We require that a 0.02 ΔCPR penalty be applied to the ΔCPR predictions when using the new method, to account for the systematic non-conservative bias in the CPR predictions compared to test data with axially varying R-Factors. On these bases, we find the current and proposed FLE methods to be acceptable.

6.2.2 Transient Methods for Core Wide Events

Abnormal operational transients which affect the entire core are of safety concern only with regard to fuel rod overheating (CPR) and RCPB overpressurization considerations. Local (intra-assembly) and radial peaking factors during core wide transients remain essentially unchanged from their relatively low normal operating values. Thus, even though gross core power may increase significantly, local LHGR's do not closely approach the safety limit LHGR during such occurrences and are not a safety concern for initial or reload cores.

GE uses a framework of codes for predicting the hot bundle transient critical power ratio during core wide transient events. This framework has been consistently used by GE for initial and reload core licensing applications.

GETAB Transient Analysis

The central code in the GETAB transient analysis is the SCAT code(57), which incorporates the GEXL correlation(53) for predicting the change in bundle critical power ratio (CPR) during the transient. The SCAT code has been previously reviewed and approved by the staff in connection with transient CPR calculations of 7x7 and 8x8 bundles for ECCS Appendix K analyses. The code is also considered to be equally acceptable for transient analysis applications. The two water rod 8x8R bundle geometry input for the SCAT code analysis does not represent a significant variation from fuel designs previously approved for analyses with the code (i.e., 7x7 or single water rod 8x8). The longer heated length (e.g., 150 inches vs. 144 inches) and fuel rod diameter changes do not represent calculational difficulty. Thus, the 8x8R fuel element design is considered to be well within the intended analysis capability of the code.

The critical bundle power correlation used in the SCAT code analysis is the GEXL correlation. The GEXL correlation, employing the previously approved R-Factor formulation(52), results in non-conservative predictions of experimental CPR data for certain 8x8R local peaking factor distributions. However, these distributions are not expected to occur during the first operating cycle of the retrofit 8x8 assemblies. Thus, the continued use of the GEXL correlation in the first operating cycle of the 8x8R assemblies is considered acceptable. Additional data should be submitted by GE to the staff for review, to justify the conservatism of the GEXL correlation for the second and subsequent cycles of operation of the retrofit 8x8 bundles, when local peaking factors may increase sufficiently to cause non-conservative CPR calculations.

Geometrical differences between the 8x8 and 8x8R fuel designs which can affect the bundle critical power calculation include the heated length, L, and thermal diameter, D_0. GE was requested to provide additional information which would justify the acceptability of a single GETAB transient analysis for the two 8x8 fuel designs for a given core-wide transient event. The sensitivity results presented(15) show that there is a ΔCPR difference of approximately 0.001 between the two fuel geometries. Thus, we find it acceptable to perform a single GETAB transient analysis for both 8x8 fuel types for a particular core-wide event.

The effect of fuel densification on SCAT bundle critical power calculations has been considered. GE has presented analyses of the effect of densification power spikes on bundle critical power. These analyses utilized an “Integral Concept.”(58) The integral concept is widely used and considered to be an acceptable method for quantification of boiling transition correlations. The integral concept also requires an empirical base. This base has been found to conservatively represent BWR conditions by comparison with an independently established procedure.(59) GE has additionally demonstrated the effect of densification on R-Factor and has concluded that the effect is insignificant. Based on the analyses presented, we find that the effects of fuel densification are appropriately considered in the bundle CPR calculations.

GE develops the SCAT code initial conditions and transient history inputs from the nuclear analysis, core hydraulic analysis and plant system transient analysis. The plant specific inputs which do not vary from cycle to cycle appear in Table 5-6 and 5-7 of Reference 3. The remaining GETAB transient inputs are calculated on a plant and reload-specific basis for each fuel type. The initial hot bundle flow for each fuel type is determined by the models and methods described in Section 4 of References 3 and 15. These methods are evaluated in Section 5.1.1 herein. The initial integral bundle power and local pin powers are determined by the GE BWR Simulator Code and Lattice Physics Methods, respectively. These codes and methods are evaluated in Section 4.1 herein.

Plant System Transient Analysis

GE develops the balance of the required input data for the GETAB transient (SCAT) code analysis from the output of the plant system transient (REDY)[60,61,62] code analysis. The plant system transient results required for each transient event analyzed by the SCAT code consist of normalized core flow vs. time, reactor core pressure vs. time and core (hot bundle) nuclear power vs. time. The REDY code results are input into the GETAB transient analysis without modification (no conservatism factors applied to the output). Since safety analysis consequences (i.e., ΔCPR, RCS pressure increase) must be conservatively calculated, this would be an acceptable procedure provided the unmodified REDY code output is already sufficiently conservative, or provides for an overall adequately conservative CPR methodology. In this regard, the REDY code and related methods are currently under staff review to evaluate the conservatism afforded by its transient predictions.

The REDY code by design is a best-estimate code. GE believes that adequate conservatism exists in the code predictions of plant system transient performance via the factors of conservatism applied to key nuclear (core) transient inputs. As seen in Table 5-5 of Reference 3, GE applies "design conservatism factors" (DCF's) of 0.95, 1.25 and 0.80 to the nominal Doppler, void and scram reactivities predicted by the nuclear analysis for the point kinetics core model in the REDY code. These factors contribute to the currently used licensing basis analysis methods, and are intended to account for the non-conservatisms and uncertainties associated with the calculation of the nuclear input parameters and the plant transient analysis models and methods.

Staff concern for the adequacy of GE's plant system transient methods has been raised by the apparently non-conservative predications of transient tests recently conducted at a large BWR/4 reactor. The tests involved three end-of-cycle manual turbine trips, initiated from intermediate power levels with the direct (turbine stop valve position switch) reactor trip intentionally disabled. This required the reactor to trip on the indirect (high neutron flux) scram. Several key transient test parameters were underpredicted, even when the present licensing basis plant transient methods (REDY code and DCF's) were employed.

GE has evaluated[63] the differences between the turbine trip test conditions and the licensing basis pressurization transient event (turbine/generator trip without bypass with a direct reactor scram) using a normalized REDY code model, as well as an improved transient code model. The GE evaluation indicates that a degree of conservatism is available when using the licensing basis methods to predict the consequences of the licensing basis pressurization events. The staff agrees with this conclusion. The staff, in the interim, while reviewing the ODYN plant system transient code proposed[63,64,65] by GE, has concluded that the present plant transient methods adequately predict the consequences of the limiting (licensing basis) core wide events.

Several of the plant system transient code models derive their input values from the fuel mechanical design. For example, the multi-noded thermal-hydraulic and heat transfer relationships utilize the fuel rod (fuel and clad) diameters and fuel column length as inputs. These parameters can, therefore, affect the dynamic behavior of the core via fuel thermal time constant and axial void sweep effects. When a substantial fraction of the core is composed of a mixture of fuel designs, (i.e., 7x7, 8x8, and 8x8R) the proper selection of the input values for fuel modeling must be considered. The REDY code models heat transfer with a single fuel element representing the entire core.

GE's current reload analysis procedure (except for fuel-clad conductance) requires the single fuel element to be the dominant fuel type rather than a weighted average. For some BWR core reloads, this would result in the modeling of a 7x7 fuel element, since the 7x7 bundle can be the dominant fuel type during the transition cycles from 7x7 to 8x8R fuel. Since the 7x7 fuel element has a somewhat slower

fuel time constant compared with the 8x8 or 8x8R fuel element, this procedure could result in a small non-conservatism in the predicted fuel consequences (e.g., peak heat flux). Fuel type sensitivity studies(15) performed by GE show that the current selection procedure results in an insignificant non-conservatism compared to an 8x8 or 8x8R dominant fuel type modeling. Thus, the staff finds the fuel type selection procedures for mixed reload cores to be acceptable.

GE also uses the REDY code predictions for evaluating conformance with the criteria relating to overpressurization of the reactor coolant system. REDY code simulations of the aforementioned transient tests (using the licensing basis DCFs) demonstrate that the peak transient pressure is consistently overpredicted by the code. The staff has considered the differences between the nature of the turbine trip tests and licensing basis pressurization events (i.e., turbine/generator trip without bypass with direct scram and main steam isolation valve closure with indirect high flux scram) and concludes that the code can be expected to also overpredict the peak transient pressure due to the licensing basis events. The use of the REDY code is, therefore, also considered acceptable for RCS overpressurization evaluations discussed in this section and Section 8.0 herein.

The plant specific REDY code input data, relating to pressure relief system characteristics which do not vary from cycle to cycle, are listed on a plant by plant basis in Table 5-4 of Reference 2. Except for Browns Ferry Units 2 and 3, these characteristics with some corrections(3), are considered to be acceptable. The Browns Ferry Units 2 and 3 pressure relief valve setpoints in the table involve an as yet unapproved technical specification change which is currently being reviewed separately on a plant-specific basis.

6.2.3 Transient Initial Conditions and Assumptions

As part of our evaluation of the Generic Reload Fuel Application, we reviewed the initial conditions and significant assumptions which will be incorporated in the plant-unique transient analyses. The conservatism associated with the initial conditions and assumptions generally conforms to the conditions and assumptions appearing in the associated FSAR analyses, NRC Standard Review Plans, and Regulatory Guides. Plant-system transient mitigating systems characteristics (e.g., pressure-relief valve setpoints, time delays, and control rod scram speeds) are taken at their most adverse technical specification values; core dynamic nuclear parameters (with conservatism factors) are representative of the most adverse time in cycle (e.g., end-of-cycle scram reactivity function, Doppler and void reactivity coefficients); initiating disturbances are assumed to occur with a conservative magnitude and rate (e.g., all turbine control valves close at the most rapid credible closure speed). With regard to the initial core thermal power level assumption, Table 5-6 of Reference 3 shows that the BWR/4 plants include an increment sufficient to account for power measurement uncertainties (calorimetric error). Thus, the power levels to be assumed for transient analysis of the BWR/4 plants are considered acceptable. However, Table 5-6 lists the assumed initial power levels for the BWR/2 and BWR/3 plants which contain no increment to account for such uncertainties. This is not consistent with those Chapter 15 Standard Review Plans which address transient analysis requirements. The staff will continue to generically review separately the initial power level assumption for the BWR/2 and BWR/3 plants.

We have also reviewed the plant-specific GETAB transient analysis initial conditions which appear in Tables 5-7 and 5-8 of Reference 2 which do not vary from cycle to cycle. We find these conditions to be acceptable as corrected.(3) For conservatism, since the CPR increases with increasing initial CPR, the initial MCPR assumed in the plant-specific GETAB transient analyses should equal or exceed the established MCPR operating limit for the plant. The remaining GETAB initial conditions, which are input into the SCAT code on a plant-specific and fuel type dependent basis, are also acceptable.

6.3 Transient Events Analyzed for Core Reloads

BWR plant system disturbances caused by a single operator error or a single equipment malfunction (i.e., transients) can be assigned to one of eight separate categories. These categories are reactor coolant system (1) pressure increase, (2) temperature (moderator) decrease, (3) positive reactivity insertion, (4) inventory decrease, (5) flow decrease, (6) flow increase, (7) temperature (moderator) increase, and (8) inventory increase.

In order to address all of the credible transient events in these eight categories, the initial operating licenses for the BWR plants listed in Table 1-1 of Reference 3 were based on the analysis of a spectrum of approximately 20 to 25 FSAR events, assignable to one of the above categories. In this manner, the most severe transient events relative to LHGR, CPR, and RCS pressure were identified. The relative and absolute severity of the consequences of the events are generally plant-specific and often cycle-specific as well. It has also been GE's experience that regardless of the BWR plant considered, most of the events result in fairly mild plant disturbances. Thus, only a few events are severe enough to be potentially limiting. Furthermore, although the most limiting event may differ from plant-to-plant and reload-to-reload, it is GE's experience that the most limiting transient always can be expected to come from the same selected group of transient events. Thus, GE asserts in the reload topical that most of the events analyzed as part of the FSAR need not be reanalyzed or reassessed for plant-specific reload core licensing applications. Accordingly, GE generically proposes that only the limiting events be reanalyzed on a plant-specific basis with each core reload. The proposed selected group of generically limiting events for all operating BWR's consists of: Turbine Trip (or Generator Trip) with Bypass Failure, Loss of Feedwater Heating (or Inadvertent High Pressure Coolant Injection for those plants with HPCI), Feedwater Controller Failure, and Control Rod Withdrawal Error. The staff agrees in principle to the logic for eliminating many of the FSAR transient events from reload analysis. In order to assure that an adequate set of events will be reanalyzed for each reload application, we have reviewed the proposed spectrum of events relative to each criterion in Section 6.1.

In connection with Criterion A, GE performed a sensitivity study, using the GEXL correlation, to determine which events can cause the largest reduction in bundle CPR. The sensitivity study established the change in bundle CPR relative to changes in bundle power, bundle flow, inlet subcooling, R-Factor and core pressure. The results of this study are provided in Table 5-10 of the reload topical. Table 5-10 shows that changes in bundle CPR is most strongly affected by bundle R-Factor, with bundle power having the next most important influence. However, since the local peaking factors, and hence, R-Factors remain relatively constant during all transient (except for local events), GE concludes that events which primarily involve significant changes in power will be limiting from an operating MCPR standpoint. The sensitivity study also shows a strong, although lesser, sensitivity of bundle CPR to core pressure changes. However, because of the negative BWR core void coefficient, a severe pressurization transient will also give rise to a significant bundle power increase. Thus, the two influencing parameters (core pressure and bundle power) are not independent. Based on the sensitivity study, the staff agrees that events which result in a large bundle power increase will also cause a large decrease in bundle MCPR.

The sensitivity study provided by GE only addressed the relative influence of important bundle thermal and hydraulic (GEXL) parameters on bundle CPR. However, the change in bundle CPR is a function of both the aforementioned sensitivities and the magnitude of the changes in the important GEXL thermal and hydraulic parameters which occurs during a transient. Thus, before a generic conclusion can be reached, the staff believes it is necessary to also evaluate the relative change in the thermal-hydraulic parameters which can occur during a transient event. This relative change in the thermal-hydraulic parameters can, in turn, vary from plant-to-plant and cycle-to-cycle. In regard to

these effects, we have reviewed plant-to-plant design differences which can affect transient thermal and hydraulic performance (e.g., relief valve capacities, setpoints, time delays, scram speeds, core power densities, auxiliary equipment performance characteristics). We have also reviewed FSAR plant transient analyses and reload transient analyses to determine the relative changes in the key thermal and hydraulic parameters for the various transient events. Based on these reviews, we have concluded that plant-to-plant transient behavior will be very similar, despite differences in operating BWR designs. Furthermore, the most severe transients for any particular plant will be expected to fall into the same selected group of events. We have also evaluated the effects of changes to cycle-to-cycle core characteristics (e.g., reactivity coefficients, peaking factors, fuel design, scram characteristics) on thermal-hydraulic transient behavior.

Based on our review of numerous BWR safety analyses, we also conclude that cycle-to-cycle changes in core characteristics will have a secondary effect on the relative and absolute severity of individual transients. Accordingly, events which are most severe, relative to thermal and hydraulic transient behavior, are not expected to change from cycle-to-cycle. Thus, based on our review of the CPR sensitivity study provided by GE and our evaluation of plant system design differences and transient analyses performed for the initial and reload core licensing, we agree that the events appearing in Section 5.2.1 of the reload topical are among the most limiting transients relative to changes in critical power ratio.

The reload topical does not classify the fuel loading error (FLE) as an abnormal operational transient. Actual fuel loading experience at operating BWR's has lead the staff to conclude that the frequency of occurrence of a fuel loading error is an event which can occur approximately once per plant lifetime. A fuel loading error could also result in fuel degradation were it to occur, go undetected, and violate the safety limit MCPR. Accordingly, we have required that the fuel loading error be considered as a transient for reload safety analyses. Moreover, a fuel loading error can give rise to significant increases in bundle R-Factor and bundle power and hence a significant change in CPR.

As discussed in Section 6.2.1, GE has proposed a safety limit of 1.0 for this event. This proposal is being reviewed separately by the staff. In the interim, we will continue to require that the fuel loading error be reanalyzed as a limiting local transient event which does not violate the safety limit MCPR.

In summary, then, we conclude that the events appearing in Section 5.2.1 of Reference 3 together with the fuel loading error, provide a sufficient set of abnormal operational transients to be reanalyzed for each plant-specific BWR reload licensing application.

Criterion B requires that the maximum local linear heat generation rate of any fuel rod in the core shall not exceed the safety limit LHGR. In this regard, local events are the most limiting because of the high transient peaking factors. The staff agrees that the control rod withdrawal error is a local event which can result in a high local LHGR's, thus, this event must be reanalyzed with respect to the safety limit LHGR for each plant specific reload application. Again, the generic reload topical does not classify the fuel loading error (FLE) as an abnormal operational transient. The topical categorizes this event as an accident.

For the reasons discussed above, we require that the FLE be reanalyzed as a local transient event with the maximum local LHGR's reported in the reload supplement. The predicted peak local LHGR's in the misloaded fuel assembly shall not violate the safety limit LHGR for this event. The staff, therefore, considers the rod withdrawal error and the fuel loading error to be an adequate set of transient events for reload reanalysis, relative to the LHGR safety limit.

Criterion C must be satisfied for the most severe abnormal operational transient. The evaluation takes credit for the direct (first) reactor scram signal and is intended to show that adequate relief valve capacity will be available during the reload cycle. Included in the selected group of transient events is Turbine Trip Without Bypass (or Generator Trip Without Bypass) and Feedwater Control Failure - to maximum demand. The staff agrees that these events have been and will continue to be the most severe pressurization transients relative to Criterion C. Thus, we find the selected group of transients to be adequate for evaluating relief valve adequacy for core reloads.

6.4 MCPR Operating Limits for Less than Rated Flow - K_f Factors

The analysis of abnormal operational transients for BWR core reloads is generally performed assuming rated core flow conditions. The purpose of the K_f factor is to adjust the MCPR operating limits when the plant is operating at less than rated core flow. This is necessary since some transient events are more severe (larger DCPR) when initiated at reduced core flow. An adjustment to the MCPR operating limit also is required to protect against the possibility of an inadvertent flow increase transient since this will cause an increase in core and bundle power. At less than full core flow, the required MCPR operating limit is the product of the full flow operating limit and the appropriate K_f factor.

For reactor operation in the automatic flow control mode, the K_f factor assures that the operating limit MCPR will not be violated should the most limiting (flow increase) transient occur from less than rated flow. In the manual flow control mode, the K_f factors assure that the safety limit MCPR will not be violated should the most limiting (flow increase) transient occur. A plot of the generic K_f factors is shown in Figure 5-7 of the reload topical.

GE indicated (Reference 2) that at low power (and flow) conditions events such as: inadvertent startup of an idle recirculation pump, recirculation flow controller failure (increasing flow), feedwater flow controller failure (maximum demand), and rod withdrawal error become more severe (larger ΔCPR) at reduced flow than at design flow conditions. However, GE states that the most limiting event with respect to reduced flow conditions is the inadvertent flow increase transient. According to GE's analysis, the NCPR decrease (ΔMCPR) caused by an inadvertent flow increase starting at reduced flow is larger than the change (increased) in ΔCPR which results when the other transients occur at reduced flow conditions.

The information provided in the topical[2] was insufficient for the staff to evaluate the generic acceptability of the K_f curves to all reload core fuel types (including the 8x8R fuel assembly) for the applicable BWR's. The staff requested GE to provide additional information to justify the generic applicability and conservative construction of the K_f curves appearing in the figure. The additional information requested was subsequently provided by GE in Reference 15.

For plant operation in the manual flow control mode, K_f factors for a given scoop tube setting were calculated as follows: At the maximum flow operating state (as limited by the pump scoop tube setpoint) and corresponding core power (along the reference flow control line), the limiting bundle of each type was placed at the thermal (MCPR safety) limit. The MCPR at several intermediate points along the flow control line was then calculated. The appropriate K_f value at each of the selected core flow points was then determined by dividing the operating limit MCPR assumed in the analysis into the MCPRs at each of the operating points. The generic analyses were performed assuming an operating limit MCPR of 1.20. Thus, the manual flow control K_f curves become non-conservative for

8x8R fuel when the plant specific 8x8R operating limit MCPR falls below 1.20. The K_f curves are also considered to be adequately conservative for 7x7 fuel whenever the plant-specific 7x7 MCPR operating limit is equal to or greater than 1.23. In the event that the calculated MCPR operating limit for a reload application falls below the above values, the staff will require that an appropriate correction be made to the manual flow control K_f curves appearing in Figure 5-7 of the reload topical or to the plant-specific MCPR operating limits.

For the case of plant operation in the automatic flow control mode, the K_f factors were calculated using the same procedure used for the manual flow control mode, except that the limiting bundle of each type was placed at the operating limit MCPR. This assures that an automatic power/flow increase, up to and including the rated power condition, will not violate the operating limit MCPR. Furthermore, GE provided sensitivity studies[15] at reduced core flows to demonstrate that a flow increase event is the most limiting transient (largest reduction in CPR) which can occur at reduced core flow.

The staff considers the criteria and methods used to develop the Kf curves to be generically acceptable. Furthermore, GE states that the input data to the transient analyses, performed to construct the K_f curves, conservatively bounds actual and projected BWR 2/3/4 D-lattice plant-specific inputs, up to and including equilibrium cycle conditions. Thus, we find the proposed K_f curves to be generically applicable and adequately conservative for BWR/2/3/4 D-lattice 8x8R core reloads, with the limitations noted, relative to the operating limit MCPRs used for the construction of the curves.

6.5 Thermal Margin Improvement Options

GE describes several proposed thermal "Margin Improvement Options" in Section 5.2.2 of the topical. The options are (1) Transient Recategorization, (2) Recirculation Pump Trip, (3) Rod Withdrawal Limiter, (4) Thermal Power Monitor, and (5) Exposure-Dependent Limits. Since all of the improvement options, except for exposure-dependent limits, involve relatively broad generic issues which could not be adequately evaluated within the scope and schedular constraints of the topical review, only the exposure-dependent limits option is generically evaluated herein.

Exposure-Dependent Operating Limits

The severity of certain abnormal operational transients is worst at the end of the cycle, primarily because the EOC, all rods out scram gives the least effective scram reactivity response. Accordingly, operating limits (i.e., DCPR) relief may be obtained during much of the cycle by analyzing transients at additional intermediate points in the cycle (when the scram reactivity is more favorable) and administrating the resulting limits on an "exposure-dependent" basis.

The procedure used by GE for developing exposure-dependent operating limits consists of analyzing transient events (which rely on reactor scram for protection) at selected mid-cycle exposures, in addition to the end-of-cycle "worst-case." Because the scram reactivity function monotonically degrades with cycle burnup towards end-of-cycle, the operating limit determined for a given exposure Ei, can be conservatively administrated in the exposure interval E, where E_{i-1}<E <E_i, and E_{i-1} is the next earlier exposure point analyzed.

Conservatism in the predicted consequences relative to the EOC analysis is achieved through the use of "black-white" control rod and patterns for calculating both the scram reactivity function and dynamic void coefficient. A black-white control rod pattern consists of rods either fully inserted or

fully withdrawn with the reactor critical at full power. This is considered to be more adverse than the expected rod pattern, based on a Haling calculation, which will include partially inserted rods at the exposure point of interest. Thus with the black-white pattern, fully inserted rods provide no contribution to the scram reactivity, the contribution of the fully withdrawn rods is minimized since the rods begin their insertion in a region of relatively low power, and finally, no partially inserted control rods are available for rapidly contributing negative reactivity during the early portion of the scram. Similarly, the magnitude of the void coefficient is maximized by the black-white pattern because power is shifted toward the bottom of the core, which in turn, increases the core average void fraction. The use of black-white control rod patterns adds a measure of conservatism beyond the end of cycle analysis which simply accurately reflects the actual all-rods-out configuration. However, since the codes and correlations used for determining exposure-dependent operating limits are the same as those used by GE for determining EOC analysis limits, our concerns are the same as those stated in Section 6.2.2 herein.

Additionally, during our review the staff raised questions regarding conservative implementation of the calculated limits which would appear in plant technical specifications. The exposure-dependent limits are referenced from the end-of-cycle, e.g., EOC minus 1000 MWd/t. The "true" cycle (exposure) length is not precisely predictable, and hence, is not exactly known until the actual end-of-cycle, all-rods-out condition is actually attained. This is due to uncertainties which exist in the burnup of the previous cycle, actual vs. projected (idealized) rod patterns and nuclear model uncertainties. Thus, the actual end of cycle exposure would not be precisely known during the cycle, nor would an exposure corresponding to EOC minus 1000 MWd/t, for example.

GE has informed the staff that a reanalysis of the projected cycle length is performed early into the operational cycle. This procedure permits previous cycle exposure history uncertainties to be eliminated from the earlier cycle length predictions. Using this procedure and the standard nuclear methods evaluated in Section 4.1 herein, gives rise to errors in these predictions which are relatively small in comparison to actual measured cycle lengths.

A small error (delay) in the implementation of a new intermediate exposure point operating limit would give rise to an insignificant non-conservatism in the predicted consequences (i.e., MCPR), even for the transient event having the greatest sensitivity (ΔCPR) to time in cycle. Moreover, such delay would be adequately compensated by the additional methods conservatism afforded by the use of idealized black-white control patterns compared with actual Haling rod patterns.

In view of the analysis methods and procedures used by GE for calculating cycle exposure and exposure-dependent limits, the staff considers the exposure-dependent operating limits methods to be generically acceptable for BWR core reload applications and that the resulting technical specifications requirements can be adequately implemented.

6.6 Plant-Specific Transient Information

The proposed revised standard format and content of the supplemental transient analysis information (provided on a plant-specific basis) appears in Appendix A to NEDE-24011-P. In general, the staff considered the proposed new format and content to be significantly abbreviated when compared with the supplemental information currently being provided. Accordingly, the staff requested GE to include certain additional information in the supplement, in order to permit the staff to perform an adequate review of the plant-specific aspects. We have reviewed GE's responses[15] to this request and find that the additional information to be provided, when coupled with the originally proposed information, and

requirements stated in this section, forms an adequate basis for the staff to review the plant-specific transient analysis aspects of BWR core reload applications.

7.0 ACCIDENT ANALYSIS EVALUATION

7.1 Loss of Coolant Accident

ECCS Appendix K Model Applicability

The Loss of Coolant Accident analysis models and methods described in the topical consist of the previously reviewed and approved GE ECCS-LOCA models. However, because of the physical differences between the standard 8x8 (and 7x7) and the retrofit 8x8 fuel designs described in Section 3.1, we reviewed the acceptability of continued application of the previously approved, unchanged ECCS-LOCA models to the new fuel. Our review of GE's responses(66) to our request for justification of such continued application is as follows:

The staff agrees with the following assertions made by GE:

- All parameters of the new 8x8R fuel, such as hydraulic diameter, pressure, flow, power, and temperature are within the range of data used in developing the GEXL correlation which is used in developing the GEXL correlation which is used in the ECCS-LOCA models to determine time-to-departure from nucleate boiling for the retrofit fuel. Also, the R-Factors used in this (LOCA) application of GEXL result from a conservative and therefore acceptable initialization procedure.

- Slightly higher peak cladding temperatures (PCT's) are calculated for the new fuel (compared to the standard 8x8 fuel at the same MAPLHGR). This is due to the small change in fuel dimensions (resulting in reduced surface area) and a shift in local power peaking toward the center of the bundle. These effects are properly included in the models, so continued application of the models in the new PCT-MAPLHGR range is acceptable.

- GE has previously stated that substantial changes in rod dimensions, spacing, linear heat generation rate, and lattice design do not significantly affect spray cooling heat transfer coefficients. We agree with GE that the changes from the standard 8x8 fuel design to the two water rod retrofit fuel design will not affect the overall conservatism and acceptability of the spray cooling coefficients assumed for the new fuel in the ECCS model inputs.

- The radiative heat transfer model used in CHASTE code was written to handle calculations with various size rods, including rods of unequal radii. Hence, it is capable of calculating radiative heat transfer for the new fuel design, and its application for that purpose is acceptable.

- The data base used to develop the swelling and rupture model covered the range of internal pressures and temperatures expected for the new retrofit 8x8 fuel. The swelling and rupture model is, therefore, equally acceptable to both the standard and retrofit 8x8 fuel designs.

- The data base used to develop the gap conductivity model included the range of temperature, internal pressure, and gap size applicability to the retrofit fuel design. Application of the gap conductivity model to the new fuel is, therefore, acceptable.

- It has been known by the staff that GE's method of initializing gap conductivity (as a function of assumed fuel rod linear power level) can, under certain conditions, be slightly non-conservative (less than 20°F in PCT). GE has shown that this initialization method, when applied to the retrofit 8x8 fuel, is slightly less non-conservative than when it is applied to the standard 8x8

fuel design, where its application has previously been accepted. The initialization method, therefore, does not introduce unacceptable non-conservatisms in the PCT results for either fuel design.

- The retrofit 8x8 fuel has a more uniform axial power profile and for some BWR/4s, as much as a six-inch longer active fuel length. These factors make it possible that the plan of maximum PCT could shift to a higher elevation (power is lower above the core midplane, but loss-of-nucleate boiling occurs earlier at the higher elevation). However, the application of the generic model includes a calculation to demonstrate that such a shift has not occurred.(67) Continued use of that calculation provision will ensure that the application of the model to the new fuel will be at the axial plane producing the highest PCT, and will therefore be acceptable.

For the reasons stated above, we conclude the continued application of the present GE ECCS-LOCA ("Appendix K") models to the 8x8 retrofit reload fuel is generically acceptable.

7.2 Steam Line Break Accident

The radiological consequences of a postulated steam line break outside of the primary containment are dependent on the amount of primary coolant lost during the accident of the concentration of the radioactivity in the coolant. The amount of coolant lost is primarily a function of plant system parameters, which would be insignificantly changed by introduction of the 8x8 fuel assemblies into a BWR core. The concentration of radioactivity in the coolant is limited by the plant technical specifications and will not, in general, be changed for a particular reload. Therefore, the previously calculated radiological consequences of a postulated steam link break accident are unaffected by the use of 8x8R fuel assemblies.

7.3 Control Rod Drop Accident

The postulated control rod drop accident assumes that a control rod has been fully inserted and becomes stuck in this position. The control rod drive is assumed to be uncoupled and withdrawn. The rod subsequently becomes free and rapidly falls out of the core onto the withdrawn drive coupling. The amount of positive reactivity introduced into the reactor core is at a rate consistent with the maximum control rod drop velocity.

There are two criteria which must be satisfied in the analysis of the control rod drop accident:

- Reactivity excursions must not result in a fuel enthalpy greater than 280 cal/g at any axial pellet location in any fuel rod. This limit assures that dispersal of fuel into the reactor coolant will not occur.

- The maximum reactor pressure during any portion of the accident must be less than the value that will cause reactor system stresses to exceed the emergency condition stress limits defined in the ASME code.

The analysis of the control rod drop accident was performed by General Electric on a generic (bounding analysis) basis and presented in Reference for both plants with Bank Position Withdrawal Sequence and Group Notch. The bundle cross sections, developed by the lattice calculations (discussed in Section 4.1) for the rod drop excursion model, are homogenized. As a result, the rod drop excursion model can readily accommodate the difference between 7x7, 8x8 or 8x8R fuel. Therefore, the calculational model used in the generic analysis remains acceptable for the new fuel

design. The evaluation of the control rod drop accident thus consists of ensuring that the appropriate parameters of the new reloaded core are bounded by the input parameter values used in the generic analysis.

The generic analysis assumes the slowest scram allowed by the technical specifications (and that the dropped rod does not scram), the most rapid credible rod drop velocity, and the smallest (i.e., high exposure) value for delayed neutron fraction. The remaining parameters of interest include the Doppler feedback, the scram reactivity, and the accident reactivity characteristics. The bounding characteristics also include the effects of densification power spiking on the peak pellet enthalphy.

We have reviewed the bounding calculations presented in References 3 and 15 with regard to the 280 cal/g limit and find them to be acceptable for reference, provided the key input parameters for a plant-specific reload application conservatively fall within the assumed bounding analysis values.

General Electric was also requested to provide a generic overpressurization analysis which demonstrates that the maximum RCPB pressure occurring during a control rod drop accident would not cause applicable ASME stress limits to be exceeded. The results of this analysis was not available prior to the completion of our review. Preliminary results[15] indicate that the pressure transient is relatively mild, i.e., ~15 psi. The staff will continue to review this aspect of the control rod drop accident on a generic basis.

7.4 Fuel Handling Accident

The refueling accident has been generically reanalyzed[3] by GE to determine the radiological consequences for the 8x8R fuel assembly. The analysis assumes (1) the fuel assembly is dropped from the maximum height (maximum potential energy) allowed by the fuel handling equipment, (2) none of the kinetic energy is viscously dissipated as the assembly falls through the water covering the core and (3) none of the kinetic energy is absorbed by the fuel material (UO_2) in the assembly. Using energy methods to predict cladding failures, it is shown that a total of 125 8x8R fuel rods fail during the accident. This compares with 111 rods for a 7x7 core. There would be no difference in failed rods for an 8x8 core. The evaluation also conservatively assumes that the fractional plenum activity in the 8x8R rod is the same as for a 7x7 rod. In actuality, an 8x8R rod would have substantially lower fission gas release from the pellets and hence gap activity as compared to a 7x7 rod. This is as a result of the significantly lower linear heat generation rate (fuel temperatures) applicable to the new fuel bundle design.

Comparing the average gap activity per 8x8R fuel rod to the average gap activity per 7x7 rod together with the number of failed rods for each bundle type (125 vs. 111), it is shown by GE that the 8x8R fuel bundle results in a relative activity release of only 88% of the activity released for a 7x7 core. Thus, of the total activity available for release above the core, the fission product activity component attributable to the fuel is less for the 8x8R fuel than for the 7x7 fuel.

The plant-specific FSAR analyses for the initial cores of the plants listed in Table 1-1 of Reference 3 showed fuel handling accident dose consequences which were appropriately well within the guidelines set forth in 10CFR100. Thus, we conclude that the dose consequences of the fuel handling accident, associated with the 8x8R fuel assembly will also be acceptably well within 10CFR100 guidelines.

7.5 Recirculation Pump Seizure Accident

GE states[69] that analyses of the pump seizure event have shown that although core coolant flow drops rapidly for this event, thermal margin (MCPR) does not decrease significantly before fuel heat flux begins dropping sufficiently to restore greater thermal margins. Additionally, the water level swell in the reactor vessel from this event produces a trip of the main turbine and feedwater turbine. This results in a stop valve closure position switch reactor scram. After the minimum critical power ratio occurs, heat flux decreases much more rapidly than the rate at which heat is removed by the coolant. GE predicts no clad perforations from this event. Additionally, the bypass valves and momentary opening of some of the pressure relieving valves limit the RCPB pressure to well within the RCPB pressure limit allowed by the ASME code. These results are acceptable to the staff.

7.6 Plant-Specific Accident Analysis Information

Loss of Coolant Accident

With regard to LOCA analyses, we will continue to require that plant specific analyses be provided for each fuel type in the core for each reactor cycle. The requirements for specific analyses to be performed and documented are given in detail below. These requirements have been negotiated with and accepted by GE and are further documented in Reference 68, and are equally applicable to the retrofit 8x8 fuel. Where previously performed analyses are equally applicable for a new cycle (e.g., the same fuel or fuel type at exposures previously analyzed where all parameters previously assumed are conservative for the new cycle conditions), such analyses can be justified as being applicable and can be referenced without the necessity of performing new calculations.

Lead Plant Analysis

For jet pump BWRs, the reload application must provide or reference one of three “lead plant” analyses (BWR/3, BWR/4 with and without LPCI modification). Each lead plant analysis must include four small-break-model analyses (in certain cases, three are sufficient) which must satisfy each of the requirements below. The small-break-model analyses required for each lead plant must include: (1) the worst recirculation suction line break size assuming HPCI failure; (2) the worst recirculation line break location and small break size assuming LPCI injection valves failure; (3) the worst recirculation discharge line break size assuming HPCI failure (for Brunswick-2 only, the LPCI-modified lead plant); (4) the transition break size with the most limiting single failure; and (5) the same as (4), but with the same single failure assumed with the large-break-model analysis of the transition break (necessary only if the small break and the large break analyses’ worst single failures are different for the transition break).

For each “lead plant” analysis, the following must also be provided: (a) justification that the worst size break has been selected for each of the above conditions (this does not necessarily require quantitative demonstration that the worst size has been found: qualitative arguments based on other lead plant analyses for similar plants with similar or identical ECCS-LOCA models, with appropriate sensitivity arguments, will be considered); (b) justification for selection of a particular break size for the transition break; and (c) documentation that the worst small break size, location, and single failure combination is included in the analyses provided.

The large break model must be used to analyze the transition break and the largest break. In addition, demonstration must be provided that the largest break produces the highest peak clad temperature. This may be done by presenting REFLOOD code results which show that core uncovery time increases with increasing break area. If the longest core uncovery time is not predicted for the largest

break, then the REFLOOD code results must be used to determine the break size that will produce the highest PCT. That break size must then be fully analyzed in a conservative manner and documented to the same extent as required for the transition break and the largest break.

For LPCI modified plants, one additional break must be analyzed. This is the largest break, at the second most limiting large break location, with the worst single failure for that break size and location (i.e., the largest suction or discharge line, whichever analysis was not provided as the DBA analysis).

Non-Lead Plants

Analysis of only the design basis accident (DBA) is acceptable for non-lead plants, except for LPCI modified plants where it is also required that analysis be provided for the largest break at the second most limiting location (as explained above for "lead plants"). Also, thorough justification must be provided for each non-lead plant concerning its similarity to the referenced lead plant with respect to LOCA break spectrum, single failure analysis, ECCS equipment, and fuel type. The analyses provided must state and justify that the worst break size-break location - single failure combination possible for the plant has been determined and analyzed (i.e., that this worst combination is the DBA).

Documentation

Each of the analyses specified above must be documented in a report formally submitted for staff review. The minimum documentation requirements stated below apply to lead and to non-lead plant analyses unless otherwise stated.

For each break analyzed, including DBA and non-DBA breaks, water level, pressure, PCT, and heat transfer coefficients vs. time must be provided. In addition, for analyses performed with the large break model, core average inlet flow and MCPR vs. time must also be provided.

For each DBA, in addition to the above, MAPLHGR vs. exposure, core wide metal water-reaction, local oxidation fraction vs. exposure, and PCT vs. time of the highest power plane experiencing DNB before jet-pump uncovery must be provided. Normalized power vs. time must be provided for each lead plant and be provided or referenced by each non-lead plant. In addition, for each lead plant and for each type of fuel in the lead plant (at two representative exposures), the hot rod's location and the location and rupture temperature of all rods that ruptured must be provided.

For each plant analysis, the significant input parameters used in the analysis must be provided, including power, steam flow, and dome pressure for the plant. Also, peak technical specification LHGR, design axial peaking factor, and initial MCPR assumed in the analyses must be provided for each fuel type. And finally, a table of single failures considered and the valve failure analysis must be provided. (Both may be provided by referencing previous documented analyses, if they are still applicable.)

Control Rod Drop Accident

The generic (bounding) control rod drop accident analysis, described in Section 7.3 herein, may be referenced, provided the key input parameters for a plant-specific reload application conservatively fall within the assumed bounding analyses values. The key parameters are Doppler coefficient, scram reactivity function, and accident reactivity function and are shown in Figures 5-22 through 5-28 of Reference 3 for both hot and cold shutdown conditions. If any of the key control rod drop accident inputs for a plant-specific reload application do not fall within the bounding analysis inputs presented in these figures, then the accident analysis must be reanalyzed on a plant specific basis. In such a case,

the plant specific analyses will be performed using an actual hot and/or cold Doppler coefficient of reactivity corresponding to the beginning of cycle, which is most limiting for this accident since the Doppler coefficient is least negative at BOC. The other key parameters will also be at their worst case plant-specific values.

The results of the combined generic and/or plant-specific control rod drop accident analyses must show that the positive reactivity insertion rate of dropped rod is compensated sufficiently by the negative Doppler and scram reactivity effects to limit the peak pellet enthalpy to a maximum of 280 cal/g.

8.0 OVERPRESSURIZATION ANALYSIS EVALUATION

The adequacy of the reactor protection system for mitigating the consequences of the most severe reactor isolation (overpressurization) event must be reevaluated with each reload core. This is necessary due to cycle-to-cycle changes in the core dynamic behavior (e.g., scram reactivity) as well as due to possible changes to pressure relieving equipment characteristics (e.g., safety valve setpoint). GE has generically presented in Section 5.3 of the reload topical the safety criteria, models, and analytical procedures together with the important assumptions which will be used in the plant specific overpressurization analyses. Our evaluation of these methods follows.

GE states that for "upset" conditions, the ASME Boiler and Pressure Vessel Code, Section III, Class I, permits pressure transients of up to 110% of the design pressure. Since the design pressure of the reactor vessel for the applicable plants is 1250 psig, the ASME code will permit up to 1375 psig pressure during overpressurization transients. The referenced 1375 psig pressure limit is also consistent with the current pressure safety limit appearing in the applicable plants' technical specifications, and thus has been previously accepted by the staff as an appropriate maximum pressure limit for the reactor coolant pressure boundary (RCPB).

GE states that the most limiting event with regard to demonstrating the adequacy of the safety valves is closure of all main steam isolation valves (MSIV's). This is based on the results of sensitivity studies(15) performed by GE which show that when credit is taken for the indirect (high flux) scram and ASME qualified safety and/or safety relief valves, the MSIV closure event is slightly more severe than the turbine trip with bypass failure. This sensitivity study was performed using the current version of the plant transient (REDY) code. However, because of the simplified steam line dynamics model associated with the REDY code (see Section 6.2.2), the staff requires that a reevaluation of the limiting pressurization event be performed when the new plant transient (ODYN) code is approved by the staff. In the interim, we believe that the demonstrated conservatisms in the pressure transient predictions associated with the REDY code (see Section 6.2.2) are adequate for the staff to allow continued evaluation of MSIV closure with indirect scram as the limiting overpressurization event.

The characteristics of the plant-specific pressure relieving equipment taken credit for the overpressurization analysis are listed in Table 5-4 of Reference 3. These characteristics, with some corrections,(3) have been found to be acceptable.

The plant-specific initial conditions of the plant process parameters (e.g., initial core power, steam flow) are listed in Table 5-6 of Reference 3. Sensitivity studies,(70) relating peak transient vessel pressure to initial operating core pressure, indicates that an increase in the initial core pressure by 25 psi above nominal pressure will result in an increase in the peak transient pressure by less than 10 psi. This relative sensitivity, when coupled with the fact that higher initial pressure will result in an earlier (high pressure) indirect scram, not taken credit for in the analysis, leads the staff to consider reload overpressurization analyses performed on the basis of nominal full power initial core pressure to be acceptable. Except for the initial core power level assumption for the BWR/2 and BWR/3 plants appearing in the table, these parameter values, with some corrections,(3) have also been found to be acceptable. Table 5-6 shows assumed initial power levels for BWR/2 and BWR/3 reactors which do not include an allowance for power measurement uncertainty. The assumed initial power level for these plants are thus not consistent with Standard Review Plan 5.2.2, which requires as increment sufficient to account for such uncertainties. The staff will continue to review this requirement for the affected plants.

Plant-specific, cycle-to-cycle, core dynamic REDY code input characteristics for the overpressurization analysis based on the end-of-cycle all rods out condition with the design conservatism factors appearing in Table 5-5 of Reference 3 applied. As discussed in Sections 4.1 and 6.2.2 herein, the staff considers this to be an adequately conservative procedure for representing the most adverse core dynamic properties for a pressurization event which can exist during a reload cycle.

Most BWR/4 plants have installed an ATWS recirculation pump trip (RPT). This reactor protection feature consists of an automatic trip of both recirculation pumps initiated after the dome pressure has reached the specified high pressure setpoint. The RCS pressure achieved during the postulated overpressurization event is sufficiently high to case both pumps to trip. Analyses have shown that pump trips during rapid pressurization transients tend to increase the peak transient pressure. Accordingly, the staff has informed GE that plants having ATWS RPT must analyze the overpressurization event with pump trip modeling included.

Reference 15 describes how the ATWS RPT will be modeled for the affected BWR/4 plants listed in Table 5-17 of Reference 3. The described modeling is acceptable to the staff.

GE states that the ASME code "emergency" limit (120% of design pressure) is properly applicable to a low probability event such as MSIV closure with coincident failure of the (direct) valve position switch scram. However, GE instead applies the ASME code "upset" limit, corresponding to 110% of design pressure, as the required pressure safety limit. Accordingly GE believes that sufficient conservatism is applied to the event limit so as to preclude the need to assume failure of any ASME qualified safety or safety-relief valves. However, the staff has taken the position on operating plants that the overpressurization analysis include the effect of failure of one (lowest setpoint) safety valve. GE has provided a sensitivity study[71] which shows the effect of safety valve failures on peak pressure for the MSIV closure event. The results show that peak pressure increase would be less than 20 psi and depends on the plant total pressure relief (safety valve) capacity. The staff finds that the GE overpressurization analysis, performed for reloads on the basis of no failed safety valves, is acceptable, provided the calculated peak pressure, when increased by an increment attributable to one failed safety valve, as given in Reference 71, shows adequate margin to the RCPB pressure safety limit.

9.0 STARTUP PHYSICS TESTING

During the course of our review of the generic reload topical, General Electric elected to remove the section included in the initial version of the document, relating to startup physics testing. Such testing information continues to be required for staff approval of core reload applications. Accordingly, information related to startup physics testing must be submitted on a plant-specific basis.

The following plant-specific startup physics test program information must be included with each BWR reload application:

- Control rod drive tests and scram times - for both hot and cold conditions
- Shutdown margin verification - with the highest worth control rod withdrawn
- Critical eigenvalue comparison - for a fixed control rod pattern and unvoided moderator
- Power distribution comparison - for a given control rod pattern and power level
- TIP reproducibility test - above 75% power
- TIP uncertainty test - above 75% power
- For each of the above tests the following information must be provided:
- A brief description of the test
- The methods used for the test
- The methods used to compare the measured and predicted values
- The acceptance criteria for the test
- The actions to be taken of the acceptance criteria are not met

10.0 TECHNICAL SPECIFICATION CHANGES

Operating BWR's which reload with 8x8R fuel, in general, will require technical specification changes to the (1) MCPR Safety Limit, (2) MCPR operating limits, and (3) MAPLHGR vs. average planar exposure curves (tables). The complete basis for the required technical specification change to the MCPR safety limit has been documented by GE in the reload topical on a generic basis, for the operating BWR's appearing in Table 1-1 of Reference 3. Completion of the bases for the changes to the MCPR operating limits and MAPLHGR vs. average planar exposure technical specifications is contained in the plant-specific analysis results provided in the reload supplement.

In accordance with the discussion in Section 5.2 of this evaluation, the MCPR safety limit must be increased from 1.06 to 1.07 for the affected plants shown in Table 1-1 of Reference 3. This increase is necessary to accommodate the combined adverse effects of a flatter intra-assembly power peaking distribution which is associated with the retrofit 8x8 reload assembly and a revised core relative bundle power histogram (distribution) for an 8x8R reload cycle approaching equilibrium cycle conditions. Based on the information submitted by GE, the staff finds the proposed 1.07 safety limit MCPR to be generically acceptable for the applicable operating BWR's which reconstitute their cores with 8x8R reload fuel assemblies. Furthermore, for these BWR's to which the GETAB uncertainties discussed in Section 5.2 continue to apply, we require that the revised technical specifications include a change in the MCPR safety limit from 1.06 to 1.07.

Finally, since changes to the MCPR operating limits and MAPLHGR vs. average planar exposure curves require submittal of plant-specific analysis information, our evaluation of proposed technical specification changes to these limiting conditions of operation will be performed on a plant-by-plant basis.

11.0 SUMMARY OF BWR RELOAD RELATED ISSUES, REQUIREMENTS AND LIMITATIONS

The following is a summary listing of the reload related issues, requirements and limitations identified during our review of the Generic Reload Fuel Application, together with the subsection where the item is addressed.

Issue	Subsection
Fission Gas Release	3.3
Shipping and Handling Loads	3.4
Combined Seismic and LOCA Loads	3.4
Fuel Channel Deflection	3.4
Void and Doppler Feedback Coefficients	4.1
Acceptable Deviation from the Reference Core	4.2
Thermal-Hydraulic Stability Methods and Limits	5.1.2
GEXL Predictions for Axially Varying R-Factors	6.2.1
GEXL Predictions for 8x8R Peaking Factors	6.2.2
REDY Code Qualification	6.2.2
Power Measurement Uncertainty Allowance	6.2.2/8.0
Control Rod Drop RCS Pressure Increase	7.3
Limiting Pressurization Event	8.0
Requirement	
Local Power Distributions and Peaking Factors	4.2
Void and Doppler Coefficients	4.2
Stability Decay Ratios for Minimum Pump Speed	5.3
LOCA Analysis Documentation	7.6
Control Rod Drop Accident Reanalysis	7.6
Physics Startup Test Program	9.0
Limitation	
Applicable BWR's	1.0
LHGR Densification Spiking Allowance	3.3
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NRC
Safety Evaluation Report
Approving Revision 1
of
NEDE-24011-P

UNITED STATES
NUCLEAR REGULATORY COMMISSION
Washington, D. C. 20555
April 16, 1976

MFN 117-79

Mr. Richard Gridley, Manager
Operating Reactor Licensing
General Electric Company
175 Curtner Avenue
San Jose, California 95125

Dear Mr. Gridley:

The NRC staff has completed its review of a proposed Amendment (dated August 1978) to the previously approved General Electric Topical Report entitled, "Generic Reload Fuel Application," (NEDE-24011-P-A). The proposed Amendment, which appears in Appendix D of NEDE-24011-P-A, relates to the use of General Electric's pressurized retrofit 8x8R (P8x8R) fuel design for operating BWR reloads.

Based on our evaluation of the information provided by GE, we have determined that the proposed Amendment, together with a summary description of related changes to be made to the topical, are acceptable. Moreover, we have determined that the proposed and planned revisions may be incorporated into the main body of the previously approved reload topical report and referenced in connection with P8x8R BWR reload licensing applications. It should be noted, however, that the attached safety evaluation supplement simply expands our approval relative to the scope of GE fuel types covered by the generic reload fuel application and in no way alters or resolves, respectively, the conclusions or issues stated in our original safety evaluation.

If you have any questions concerning our evaluation of NEDE-24011-P-A, please contact us.

Sincerely,

Thomas A. Ippolito, Chief
Operating Reactors Branch #3
Division of Operating Reactors

Enclosure:

Safety Evaluation

Safety Evaluation Supplement For An Amendment Dated August 1978 To The General Electric Topical Report Generic Reload Fuel Application

1.0 INTRODUCTION AND CONCLUSIONS

By letter[(1)] dated May 27, 1977, the General Electric Company (GE) submitted for staff review a licensing topical report[(2)] entitled "Generic Reload Fuel Application." The report, designated NEDE-24011-P, after being amended[(3,4,5)] several times by GE, was approved[(6)] by the staff for reference in connection with BWR reload licensing applications which would utilize GE's standard (one water rod) 8x8 or retrofit (two water rod) 8x8 fuel designs. Both fuel types, as described in the generic reload topical report, incorporate fuel rods which are evacuated and back filled with helium to one atmosphere pressure.

After our approval of the reload topical report and several applications of the retrofit fuel design to operating reactor reloads, GE decided to slightly modify the reference retrofit 8x8 fuel design. The modification involves increasing the fuel rod helium prepressurization level from one atmosphere to three atmospheres. It is GE's intent, therefore, to change from an "unpressurized" retrofit 8x8 fuel assembly to a "prepressurized" retrofit 8x8 fuel assembly for future BWR reload (and initial core) applications. Accordingly, by letters[(7,15)] dated September 11, 1978 and February 27, 1979, GE requested that the staff supplement its original evaluation,[(6)] by reviewing and approving a proposed amendment to the approved version[(13)] of the reload topical report, which addresses the use of prepressurized retrofit 8x8 fuel in BWR reload applications.

Most of the technical and safety bases in the proposed amendment for the prepressurization design change is provided via reference to a separate GE topical report[(8)] on prepressurized 8x8R fuel. The supporting topical report generically describes the effects of prepressurization in connection with fuel thermal-mechanical performance, as well as abnormal operational transient and postulated accident consequences, including the Loss of Coolant Accident. The significance of prepressurization relative to thermal-hydraulic stability is also considered. The subject topical report on prepressurization, together with other supplemental[(9,10,11)] information provided by GE, has been reviewed and approved[(12)] for reference in connection with licensing applications. Our evaluation herein addresses the use of prepressurized 8x8R fuel for BWR reload licensing applications which may also incorporate a mixture of GE fuel types, e.g., 8x8, 8x8R, in addition to the P8x8R fuel in the refueled core. Numbers appearing in parenthesis following each section heading herein refer to the section numbers used in our safety evaluation[(6)] of the generic reload topical.

Based on our review and evaluation of the information provided by GE, we find the proposed amendment, dated August 1978, appearing in Appendix D to NEDE-24011-P-A (relating to prepressurized retrofit 8x8R fuel), together with a summary description[(15)] of related changes to be made to the topical, to be acceptable. Moreover, these proposed revisions to NEDE-24011-P-A may be incorporated into the body of the previously approved topical and referenced in connection with P8x8R BWR reload licensing applications.

2.1 Fuel Safety Limits (6.1)

As noted in our evaluation[6] of NEDE-24011-P, GE utilizes two transient criteria in connection with fuel performance during abnormal operational transients. These criteria, or safety limits (SAFDL, of GDC 10, 10CFR50, Appendix A), are intended to protect against either overstraining or overheating of the cladding during transient events. To preclude fuel rod failure from excessive strain during transients, GE has established a 1.0 percent cladding plastic strain limit. To preclude fuel rod overheating, GE has established the requirement that 99.9% of the fuel rods in the core avoid boiling transition during transients.

Using previously accepted methods, GE calculated exposure-dependent linear heat generation rates (LHGRs), which would result in one percent cladding plastic strain for the unpressurized standard 8x8 and unpressurized retrofit 8x8 fuel types. These calculated safety limit LHGRs, which appear in Table 2-3 of Reference 13, were found to be acceptable in connection with our evaluation of the generic reload topical report. One of the principle effects of prepressurization with helium to three atmospheres is to increase the fuel-to-cladding gap conductance. Thus, for the same local linear heat generation rate, prepressurized 8x8R fuel temperatures, and hence fuel thermal expansion strains, will be less than for unpressurized 8x8R fuel. Put another way, prepressurized 8x8R fuel could attain a somewhat higher LHGR at which one percent cladding strain occurs. However, GE has referenced the safety limit LHGRs previously calculated for unpressurized 8x8 and unpressurized 8x8R as applicable to the P8x8R fuel. This is acceptable to the staff and effectively results in additional conservatism in the calculated safety limit LHGRs. Should GE, at some future date, wish to take credit for this available margin however, we will require that they submit a revised thermal-mechanical analysis for our review.

With regard to clad overheating, GE had statistically determined[6] a safety limit minimum critical power ratio for an 8x8R reload core such that 99.9 percent of the fuel rods would not be expected to experience boiling transition. The resulting 1.07 safety limit MCPR, which was calculated using the GETAB statistical procedure, was reviewed and approved by the staff in connection with unpressurized 8x8R reloads. The subject GETAB statistical analysis for the retrofit fuel incorporated the local R-Factor distribution which appears in Table 5-2B of Reference 6. The R-Factors shown in the table were calculated using a local peaking factor distribution applicable to the unpressurized 8x8R fuel. The use of prepressurized rods will have the effect of slightly reducing fuel temperatures during power operation which will result in a small reduction in the local Doppler feedback effect on local (pinwise) power peaking. GE states[9,10] that the resulting difference between unpressurized 8x8R and prepressurized 8x8R local power peaking is insignificant. Moreover, higher peaking in the P8x8R assemblies would tend to reduce the flatness of intrabundle peaking. Since decreased peaking (flatter power distribution) results in more rods in boiling transition in the GETAB statistical analysis, the use of the 8x8R R-Factor distribution for P8x8R reloads is considered conservative. Thus, the staff finds the statistical safety limit, originally derived for 8x8R BWR reloads, to be equally acceptable for P8x8R BWR reloads.

2.2 Abnormal Operational Transients (6.2.2)

The improvement in fuel rod gap conductance and attendant reduction in fuel time constant which stem from fuel rod prepressurization can affect fuel, core and plant system transient performance principally in two competing ways. Both effects are caused by the more rapid transfer of thermal energy from the fuel rod to the moderator/coolant. First, the more rapid transfer of energy to the moderator will cause voids to generate more rapidly, which has the beneficial effect of terminating the gross core nuclear power transient more quickly. However, more rapid void generation will have the

adverse effect of more rapidly increasing flow quality, thereby resulting in a more adverse channel thermal-hydraulic condition (lower critical bundle power).

The degree to which the above two competing effects offset each other (as determined by GE's transient analysis methods) is dependent on the relative number of P8x8R assemblies in the specific reload core. This arises from the gap conductance value used in GE's plant transient (REDY or ODYN) calculation for mixed reload cores which is determined by the dominant fuel type. Thus, for initial P8x8R reloads, where there are still relatively few prepressurized fuel assemblies, the dominant fuel type will not be the P8x8R but rather either a 7x7, 8x8 or 8x8R fuel type. Thus, for these initial reloads there will be no credit given for the smaller P8x8R fuel time constant more rapidly terminating the nuclear transient on a core-wide basis. However, the adverse effect of higher flow quality (void fraction) within the P8x8R fuel assembly channels, which results from the same reduction in fuel time constant, will still be present whenever P8x8R assemblies are in the core. Thus, the transient critical bundle power in the prepressurized fuel assemblies will still be decreased more relative to the unpressurized 8x8R and unpressurized 8x8 assemblies. GE sensitivity studies[10] indicate that for core-wide events the P8x8R assemblies will have a slightly larger transient ΔCPR (~0.01) than the unpressurized 8x8 and retrofit unpressurized 8x8R fuel types. Thus, the P8x8R assemblies will require a correspondingly higher operating limit MCPR than the 8x8R/8x8 assemblies whenever the limiting transient is a rapid pressurization transient (i.e., when fuel time constant effects are important).

Therefore, in view of the above discussion, whenever operating MCPR limits for mixed (P8x8R, 8x8R and 8x8) reload cores are established based on rapid core-wide transient events, we find it acceptable to either: (1) perform separate GETAB transient analyses (separate operating limits) for the prepressurized and unpressurized fuel assemblies or, (2) perform a single GETAB transient analysis (a single operating limit) which conservatively incorporates the fuel rod thermal characteristics of the P8x8R fuel assembly.

2.3 Accidents (7.0)

Our generic evaluation of the applicability of GE's accident analysis models and methods to prepressurized 8x8R fuel as well as our evaluation of the effects of prepressurization on previously reviewed BWR accident analysis results is contained in Reference 12. Events considered by GE included the Control Rod Drop, Fuel Loading Error, Refueling and Loss of Coolant accidents.

In connection with the Loss of Coolant Accident (LOCA) we concluded that the existing approved LOCA-ECCS models and methods remain valid for 8x8R fuel prepressurized with helium to three atmospheres. In addition, based on sensitivity studies performed by GE, we also concluded that compared to unpressurized 8x8R fuel prepressurizing to three atmospheres results in lower calculated peak cladding temperature for all BWR classes. Alternatively, margin for improvement (increasing) in the maximum average planar linear heat generation rates (MAPLHGRs) associated with the unpressurized 8x8R fuel type would be available for P8x8R fuel. Accordingly, we conclude that LOCA analyses previously performed and accepted for unpressurized 8x8R fuel are conservatively bounding for prepressurized 8x8R fuel. Thus, BWR reloads utilizing GE's pressurized 8x8R fuel may reference such prior analyses and resulting MAPLHGR vs. exposure technical specification limits in connection with showing compliance with the requirements of 10CFR50.46. Additionally, P8x8R core reloads for which there is no existing approved 8x8R LOCA-ECCS analysis and related technical specifications (no unpressurized 8x8R fuel in the core), we find it acceptable to include the beneficial effects of prepressurization to three atmospheres when performing LOCA-ECCS analyses to show compliance with 10CFR50.46.

Finally, relative to the other accidents, based on our review of the information provided by GE, we agree that the methods and results for the Control Rod Drop Accident, Refueling Accident and Fuel Loading Error, contained in Reference 13, remain valid and acceptable for prepressurized 8x8R fuel.

2.4 Thermal-Hydraulic Stability

Because prepressurization decreases the fuel thermal time constant, reactor core thermal-hydraulic stability will also be affected since it involves coupled neutronic thermal-hydraulic dynamic behavior. Sensitivity studies(14) performed with GE's licensing basis stability methods indicate that the core stability decay ratio monotonically increases with increasing fuel rod gap conductance. Thus, it is to be expected that actual core stability at the least stable operating state will decrease somewhat (increased decay ratio) during the transition from unpressurized to prepressurized fuel. Additional stability studies(10) have been performed by GE more recently, utilizing their licensing basis stability code and gap conductance input from their approved GEGAP-III computer code. These studies indicate that prepressurizing 8x8R fuel to three atmospheres will cause the actual core stability decay ratio to increase by approximately 0.08 for operating BWR/2&3s and approximately 0.10 for BWR/4s. However, GE has historically utilized a constant gap conductance value of 1000 Btu/hr-ft^2-°F for licensing calculations. This conservatively bounds the gap conductance values predicted by GEGAP-III for both unpressurized and prepressurized 8x8R fuel designs. Moreover, GE states(10) that a significant decrease in calculated decay ratios (0.2 to 0.3) would be realized if GEGAP gap conductance values were used instead of a constant value of 1000 Btu/hr-ft^2-°F. Thus, although no change in decay ratios will be predicted on a licensing basis for core reloads with prepressurized 8x8R fuel compared to core reloads with unpressurized 8x8R fuel, GE believes that adequate conservatism will be retained in P8x8R core stability calculations.

As discussed in our evaluation(6) of Reference 13, the staff has expressed concerns relating to the capacity of GE's overall stability methods (including input assumptions; e.g., gap conductance, models, analytical procedures) to conservatively predict decay ratios.

As an interim measure until these issues are resolved, the staff has imposed a Technical Specification requirement on operating BWRs which restricts planned reactor operation in the natural circulation mode whenever the predicted decay ratio is greater than 0.5 at the least stable reactor operating state. We believe that this requirement will continue to provide adequate stability margin for P8x8R reloads until the generic stability issues are resolved.

2.5 Overpressurization Analysis (8.0)

The faster fuel time constant of prepressurized fuel results in more (thermal) energy being deposited in the fuel channel (within the reactor coolant pressure boundary) in a shorter period of time when compared with unpressurized fuel. However, GE sensitivity studies show that this more rapid energy transfer has a negligible effect on the peak system pressure associated with pressurization type transients. Nevertheless, current GE BWR system transient methods for mixed reload cores will account for this small effect via the dominant fuel type selection procedure discussed in Reference 13. Thus, the staff finds that the effects of fuel prepressurization will be adequately accounted for in vessel overpressurization analyses.

3.0 TECHNICAL SPECIFICATIONS

We have reviewed those fuel related BWR Technical Specifications which are either generic to all operating BWRs or specific to a particular BWR.

Regarding generic fuel related Technical Specifications, we have determined that the 1.07 safety limit minimum critical power ratio referenced in connection with 8x8R reloads are also acceptable for reference in connection with P8x8R reloads. Additionally, the safety limit linear heat generation rates (appearing in Table 2-3 of Reference 13) applicable to 8x8R reloads are also acceptable for reference for BWRs which reload with prepressurized 8x8R. GE has also proposed no change to the 8x8 and 8x8R 13.4 KW/ft design linear heat generation rate for the prepressurized 8x8R fuel. This is also acceptable.

In connection with plant-unique fuel related Technical Specifications, we have reviewed both the operating limit minimum critical power ratio (OLMCPR) and the maximum average planar linear heat generation rate (MAPLHGR) requirements for the P8x8R fuel assemblies in the reload core. As discussed in Section 2.2 herein, whenever rapid pressurization corewide events are limiting for the 8x8R fuel types, it is acceptable to either (1) determine one OLMCPR on the basis of a single GETAB transient analysis which models the P8x8R gap conductance characteristics or, (2) perform separate GETAB transient analyses for the P8x8R and 8x8R fuel types, to establish separate OLMCPRs for each. Finally, as discussed in Section 2.3 herein, for those operating BWRs which already have MAPLHGR vs. Exposure curves for unpressurized 8x8R fuel in the Technical Specifications, we find it acceptable to reference these curves to show compliance with the requirements of 10CFR50.46 for the reload P8x8R fuel. Plants which do not have such already existing Technical Specifications for 8x8R fuel may develop MAPLHGR vs. Exposure curves for the P8x8R reload fuel assemblies from LOCA-ECCS analyses which take credit for the beneficial effects of prepressurization to three atmospheres.

4.0 REFERENCES

1. General Electric letter (R. Engle) to NRC (R. Baer) dated May 27, 1977.

2. "Generic Reload Fuel Application," General Electric Report NEDE-24011-P dated May 1977.

3. "Generic Reload Fuel Application - Amendment l," General Electric Report NEDE-24011-P-1 dated January 1978.

4. "Generic Reload Fuel Application - Amendment 2," General Electric Report NEDE-24011-P-2 dated February 1978.

5. "Generic Reload Fuel Application - Amendment 3," General Electric Report NEDE-24011-P-3 dated March 1978.

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7. General Electric letter (J. Quirk) to NRC (O. Parr) dated September 11, 1978.

8. R. B. Elkins, "Fuel Rod Prepressurization - Amendment 1," General Electric Report NEDE-23786 dated May 1978.

9. General Electric letter (E. Fuller) to NRC (O. Parr) dated June 8, 1978.

10. General Electric letter (E. Fuller) to NRC (O. Parr) dated August 14, 1978.

11. General Electric letter (H. Pfefferten) to NRC (R. Tedesco) dated October 27, 1978.

12. NRC letter (O. Parr) to General Electric (G. Sherwood) dated November 21, 1978.

13. "Generic Reload Fuel Application," General Electric Report NEDE-24011-P-A dated May 1977.

14. "Stability and Dynamic Performance of the General Electric Boiling Water Reactor," General Electric Report NEDO-21506 dated January 1977.

15. General Electric letter (J. Quirk) to NRC (O. Parr) dated February 27, 1979.

NRC Safety Evaluation Report Approving Revision 2 of NEDE-24011-P

UNITED STATES
NUCLEAR REGULATORY COMMISSION
Washington, D. C. 20555
November 7, 1980

MFN 195-80

General Electric Company
Attn: Mr. R.E. Engel, Manager
Reload Fuel Licensing
Safety and Licensing Operation
175 Curtner Avenue
San Jose, California 95125

Dear Mr. Engel:

SUBJECT: **Acceptance For Licensing Reference of Changes to Topical Report Number NEDE-24011-P-A-L, "Generic Reload Fuel Application" Dated August 1979.**

The Nuclear Regulatory staff has completed its consideration of the changes to Topical Report NEDE-24011-P-A-l proposed in R.E. Engel letter to T.A. Ippolito, "Proposed Revision of Fuel Design Analysis Input Parameters" July 21, 1980. In this letter General Electric proposed the changes as follows:

(a) Approval to use nominal dimensions rather than worst-tolerance dimensions as input to the TEXACO and CLAM codes for the calculation of "non-safety related" fuel design parameters.

(b) Approval to increase the present peak-pellet exposure limit from 44,000 MWd/t to 50,000 MWd/t.

The NRC staff consideration is as follows:

Nominal Input Values for TEXACO and CLAM

General Electric uses several analytical codes for fuel design and analysis. GEGAP-III is used for most of the temperature-related analyses including stored energy for LOCA analysis, and GEGAP-III has been reviewed carefully and approved by the NRC. TEXACO and CLAM are used for some stress analyses discussed in GESTAR. General Electric refers to the TEXACO and CLAM analyses as non-safety related although we consider them safety related because they address requirements of the Standard Review Plan. Nevertheless, we have considered these analyses to be less important than the GEGAP-III analyses and we have, therefore, never reviewed TEXACO, CLAM or similar codes by the other vendors. It thus seems inappropriate for us to specify the input to such a code as long as its developer/user asserts that it is being used conservatively. GE has made that assertion in Reference 2 and they have supported the claim with comparative calculations with a new, improved code using conservative input. Furthermore, GE's discussion of TEXACO or CLAM input in GESTAR was not a requirement in the first place—they could have simply omitted it. We, therefore, approve this change with the understanding that GE will maintain full responsibility for the proper application of TEXACO and CLAM.

Higher Peak Pellet Exposures

General Electric has a self-imposed peak-pellet exposure limit that they would like to increase. Although extended-burnup operation is expected in the future, this GE request is related to a change from four shorter cycles to three 18-month cycles and, as such, is not a request for extended-burnup operation. It is found, however, with the three 18-month cycles that the distribution of burnup within a reload batch is spread out so that the peak-pellet burnup is higher although the average burnup is essentially unchanged.

Several methods of characterizing fuel burnup are in use: peak pellet burnup, planar averaged burnup in the peak-power elevation of the peak bundle, average burnup of the peak bundle, average burnup of the discharged fuel bundles, etc. Since (a) there are no precipitous changes in fuel behavior in the transition range leading to high burnups, (b) potential releases to the environment are related more directly to whole-core properties, and (c) total energy production is directly related to average burnup, we believe that average burnup of the discharged fuel bundles is the best way to characterize the burnup level of a core for high-burnup considerations. Therefore, we see no reason to enforce an arbitrary limit of peak pellet exposure when the average discharge burnup will not increase significantly. Accordingly, we approve the GE request for an increase in peak-pellet exposure.

Per your commitment in the request for change letter, these changes must be reflected in a revision of NEDE-24011-P-A-l in accordance with established requirements.

Sincerely,

Robert L. Tedesco, Assistant Director for Licensing
Division of Licensing

NRC
Safety Evaluation Report
Approving Revision 3
of
NEDE-24011-P

(GE letter obtains NRC approval through NRC lack of response.
See attached letter.)

General Electric Company
175 Curtner Avenue, San Jose, California 95125

NUCLEAR POWER SYSTEMS DIVISION

October 5, 1981

MFN 181-81
REE-079-81

U.S. Nuclear Regulatory Commission
Division of Project Management
Office of Systems Integration
Washington, DC 20555

Attention: Robert L. Tedesco, Assistant Director
Licensing Management

Gentlemen:

SUBJECT: **Administrative Amendments To GE Licensing Topical Report, NEDE-24011-P-A-l, "GENERIC FUEL APPLICATION"**

References: 1) Letter, R.E. Engel (GE) to R.L. Tedesco (NRC), "GE/NRC Meeting on Licensing Topical Report Update", 4/9/81

2) NUREG-0390, "Topical Report Review Status"
3) Letter, J.F. Quirk (GE) to Olan D. Parr (NRC), "General Electric Licensing Topical Report NEDE-24011-P-A, Generic Reload Fuel Application", 9/11/81

This letter submits amendments to the subject document that have been classified as "administrative" by a GE review committee. It is requested that the NRC respond in the manner summarized in Reference 1 by November 20, 1981, so that this document can be maintained as a current reload fuel document.

Four amendments are included in this package:

1) New Bundle Enrichments;

2) Improved Scram Times Clarification of text for plants operating under ODYN Option B;

3) Power/Core Flow-MCPR Operating Limit: Clarification in reporting power/core flow inputs used in calculating the MCPR operating limit: and

4) Control Rod Worth Calculations: Performance of a prompt criticality check to determine conditions at which the maximum control rod worth can be achieved, and calculation of maximum control rod worth at these conditions.

The GE review committee classified these amendments as administrative based on the criteria established in l0CFR50.59. These criteria are delineated for each amendment in the enclosed Classification Justification Review Sheets. These sheets also include a brief description of each change

and a list of the text pages affected by the change. The change is signified by "10/81" in the margin of the text.

As stated in Reference l, the NRC may agree, disagree, or not comment on a revision classification within forty-five days of receipt of this letter. If the NRC disagrees with the classification, the revision will be submitted for NRC review in accordance with the procedures identified in Reference 2. Should the NRC either agree or not comment within 45 days, the proposed revision will be implemented on the designated date. The page changes enclosed with the notification letter will then be incorporated into the LTR and the LTR reissued.

Some of the pages in this revision contain information which the General Electric Company customarily maintains in confidence and withholds from public disclosure. The information has been handled and classified as proprietary to the General Electric Company as indicated in the affidavit attached to the initial submittal given in Reference 3. We hereby request that this information be withheld from public disclosure in accordance with the provisions of l0CFR2.790.

Very truly yours,

Ronald E. Engel, Manager
Reload Fuel Licensing
Safety and Licensing Operation

REE:pes/555-6 92381

NRC
Safety Evaluation Report
Approving Revision 4
of
NEDE-24011-P

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

MFN 062-82

General Electric Company
Attn: Mr. R.E. Engel
Reload Fuel
Licensing, Nuclear Safety and Licensing Operation
175 Curtner Avenue
San Jose, California 97125

Dear Mr. Engel:

Reference: Letter, R.E. Engel (GE) to R.L. Tedesco (NRC), "Administrative Amendments to GE Licensing Topical Report NEDE-24011-P-A-3," dated 1/29/82

In accordance with our previous agreement concerning administrative amendments, we are providing comments on your recent submittal. We have discussed these comments with your staff and approve your amendment subject to the following:

Title Change

The title has been changed from General Electric Generic Reload Fuel Application (also called GESTAR) to General Electric Standard Application for Reactor Fuel (or GESTAR II). We note that NEDE-24011 was reviewed some years ago as a reload document, and as such it was not reviewed according to current Standard Review Plan guidelines. Consequently, we will accept the referencing of NEDE-24011 in new plant applications as a source of information, but we will not recognize it as a sufficient body of information that meets current requirements.

Effects of Gadolinia

The discussion of the effect of gadolinia additions in report Section 2.4.1.1 no longer indicates that the gadolinia-urania rods are designed to provide margins similar to standard UO2 rods. That change would be substantive. Therefore, the original wording must be reinstated.

Conservative Dimensional Tolerances

We have previously approved a change from a conservative combination of tolerances to a nominal combination. The reference to our approval must be indicated in Section 2.4.2.3.

Sincerely,

Robert L. Tedesco, Assistant Director for Licensing
Division of Licensing

NRC
Safety Evaluation Report
Approving Revision 5
of
NEDE-24011-P

UNITED STATES
NUCLEAR REGULATORY COMMISSION
Washington, D. C. 20555
November 29, 1982

MFN 185-82

General Electric Company
Attn: Dr. G.G. Sherwood, Manager
Safety and Licensing
175 Curtner Avenue
San Jose, California 95114

Dear Dr. Sherwood:

SUBJECT: Admistrative Amendments To GE Licensing Topical Reports NEDE-24011-P-A-4 and NEDE-24011-A-4-US—Partial Rejection

References:
1. Letter, J.S. Charnley (GE) to R.L. Tedesco (NRC), Administrative Amendments to "GE Licensing Topical Reports NEDE-24011-P-A-4 and NEDE-24011-P-A-4-US dated August 31, 1982".
2. NUREG-0390 "Topical Report Review Status".

General Electric Company submitted by Reference 1 their proposal of Administrative Amendments to their licensing topical reports NEDE-24011-P-A-4 and NEDE-24011-P-A-4-US. The staff has examined the changes made within the 8 categories of amendments submitted. We agree that, in general, the changes are administrative. An important exception, however, is the category called Fuel Design Options, which for the most part concerns barrier fuel. We consider barrier fuel to be a design change of the kind normally reviewed by the staff. We have discussed this matter with General Electric by telephone and have described the scope of information that should be submitted for review. That information should then be submitted in accordance with NUREG-0390, reference 3.

With regard to the eighth category of administrative changes, viz., "Typos/Clarification/ Editorial Changes," we note that GE has included some information deletions in this category. For instance, in Tables 2.7a, and 2.7b, the minimum cold plenum length, maximum end plug angularity, and an item called "displacement of hardware inside rod" have been deleted. We will accept these three deletions because they involve design features that either are relatively unimportant or are compensated for elsewhere. However, we do not believe that deletions of technical information, whether in figures, tables or text, should, as a general rule, be defined as editorial or as clarification. Likewise, changes in existing figures or insertions of new design curves that are based on new data should not be treated as administrative.

We understand that the fatigue curve (Fig. 2-11b) was part of the original fatigue analysis that was reviewed by the former Division of Operating Reactors, and that the curve is being added for completeness of the documentation. This, we agree, is administrative. We want to emphasize, however, that any change to that curve (perhaps involving new data) should be reviewed and not treated as administrative.

The category of changes labeled Typos/Clarification/Editorial Changes constitutes the second largest of the eight categories of changes (about 90 pages of altered text, tables, and figures). Yet there seem to be several typographical and editorial errors in these revisions. Presumably, these errors will require

further administrative amendments to correct the documentation. We believe that unless GE can do a better job in quality assuring these proposed amendments, a major benefit of the administrative amendment process, viz., a reduction in staff review, will, in fact, not be achieved.

On page D-US, 2-31a Section S.2.2.3.1 the “16 hours” should be “12 hours”.

General Electric is free to issue the proposed Revision 5 to the licensing topical with the exception of the Fuel Design Options change.

Sincerely,

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

NRC
Safety Evaluation Report
Approving Revision 6
of
NEDE-24011-P

UNITED STATES
NUCLEAR REGULATORY COMMISSION
Washington, D. C. 20555
April 13, 1983

MFN 075-83

Ms. J.S. Charnley, Manager
General Electric Company
175 Curtner Avenue
San Jose, California 95125

Dear Ms. Charnley:

Subject: Acceptance For Referencing Of Licensing Topical Report NEDE-24011-P-A-4, "Barrier Fuel Amendment To General Electric Standard Application For Reactor Fuel"

We have completed our review of the subject topical report submitted November 19, 1982 by General Electric Company letter MFN 177-82. We find this report is acceptable for referencing in license applications for Boiling Water Reactor plants to the extent specified and under the limitations delineated in the report and the associated (NRC) evaluation which is enclosed. The evaluation defines the basis for acceptance of the report.

We do not intend to repeat our review of the matters described in the report and found acceptable when the report appears as a reference in license applications except to assure that the material presented is applicable to the specific plant involved. Our acceptance applies only to the matters described in the report.

In accordance with established procedures (NUREG-0390), it is requested that GE publish accepted versions of this report, proprietary and nonproprietary, within three months of receipt of this letter. The accepted versions should incorporate this letter and the enclosed evaluation between the title page and the abstract. The accepted versions shall include an –A (designating accepted) following the report identification symbol.

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

Sincerely,

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

Enclosure:
As stated

Topical Report Evaluation

Report Number:	NEDE-24011-P-A-5
Report Title:	Barrier Fuel Amendment to NEDE-24011-P-A-4 (GESTAR-II)
Report Date:	Unspecified (submitted via letter dated November 19, 1982).
Originating Organization:	General Electric Company
Reviewed by:	Reactor Fuels Section, Core Performance Branch Division of Systems Integration

SUMMARY OF TOPICAL REPORT

By letter dated August 31, 1982 (Ref. 1), General Electric submitted several amendments to the approved versions (NEDE-24011-P-A-4 and NEDE-24011-P-A-US) of the GE licensing topical report, GESTAR-II (General Electric Standard Application for Reactor Fuel). In accordance with an earlier agreement regarding administrative amendments, the staff requested (Ref. 2) that General Electric submit (Ref. 3) an amendment concerning barrier fuel* for review in accordance with the procedures identified in Reference 4. This topical report evaluation, therefore, addresses the NRC staff's review of the barrier fuel amendment to GESTAR-II.†

The GESTAR-II report presents generic information relative to General Electric fuel designs and analyses. The barrier fuel amendment thus consists of a collection of the pages in the GESTAR-II report that contain some reference to barrier fuel. In many instances the only changes contained in the barrier fuel amendment consisted of the words "barrier fuel" or "BP8x8R fuel" (the designation chosen for the barrier fuel design), which were simply added to the report text or to a table in the report, while much of the substantive information about barrier fuel was contained in other reports (Refs. 5-7).

REGULATORY EVALUATION

Because GESTAR-II had been subjected to considerable scrutiny as part of previous reviews of the fuel designs for several BWRs, such as WNP-2, GESSAR-II, Perry, Clinton, etc., it was not necessary to perform another complete Standard Review Plan-type review of the barrier fuel design. Instead, our review focused on several of the effects of those design features that were unique or particularly important to barrier fuel with respect to potential fuel damage mechanisms and fuel behavior parameters. Questions or staff positions were issued on each of those subjects (Ref. 8). Our inquiries and staff positions and GE's responses (Ref. 9) will be incorporated into GESTAR-II by amendment. The reviewed subjects will be discussed in the order in which they were addressed in Reference 8.

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* The term "barrier fuel" stems from the use of a 0.003-inch thick, high purity zirconium liner, i.e., barrier, which is metallurgically bonded to the inner surface of the Zircaloy-2 portion of the fuel rod cladding. The overall dimensions of the fuel rods are the same as for the GE8x8 prepressurized retrofit bundle.

† Unless otherwise specified, in this report evaluation "GESTAR-II" refers to the latest approved version of NEDE-24011.

Waterside Corrosion

In recent cases that have involved large-scale waterside corrosion, failures have been associated with certain common features, including a “minor anomaly” in the Zircaloy cladding metallurgy and gadolinia poison rods (Refs. 10-13). Because the barrier fuel amendment did not state that the barrier cladding would be used with gadolinia poisons, a question (Q 490.1) was raised concerning such potential application. Also, inasmuch as the barrier cladding is fabricated by special “co-reduction” methods that will produce a metallurgical bonded 2-layer tubing, it was important to know whether the barrier cladding had been altered metallurgical in a way that would affect waterside corrosion.

In response, GE indicated (Ref. 9) that barrier cladding is to be used on both gadolinia rods and rods without gadolinia. GE also stated that “...there will be no difference in the relative behavior of barrier fuel and standard fuel with regard to waterside corrosion” because the barrier cladding and standard cladding have identical outside surfaces and because both claddings are produced “using the same fundamental manufacturing processes.” Furthermore GE has stated (Ref. 14) that a change has been made in the cladding manufacturing process to assure that newly-manufactured fuel rods will not be susceptible to waterside corrosion failures. We conclude, therefore, that there is reasonable assurance that the barrier cladding will not be susceptible to waterside corrosion.

Rod Internal Pressure

The maximum internal rod pressures for barrier fuel as well as standard 8x8R and P8x8R designs are listed in Section 2.4.2.3 of GESTAR-II to be about twice the coolant system pressure even for normal operation. This is inconsistent with the Standard Review Plan, which calls for rod pressures to remain below nominal system pressure during normal operation unless otherwise justified. GE has chosen to provide a basis for acceptance other than suggested in the SRP and we are pursuing this subject generically as part of the review of GESSAR-II (the latest BWR plant under FSAR review). We will resolve this issue on the GESSAR-II docket.

Effect of Barrier Cladding Properties on Design Margins

Inasmuch as about 10% of the barrier cladding thickness is taken up by the zirconium liner, which is weaker and more ductile than Zircaloy-2, the barrier fuel could have reduced margins for stress/strain, creep collapse, and fatigue. While such reductions in margin might be relatively small, we believed that they should be reviewed and in staff question 490.3 (Ref. 8), we asked for quantitative comparison of the relative effects of these phenomena on the barrier and standard cladding fuel designs, including the effects of burnup or exposure where such effects are pertinent.

In response to our request, General Electric replied (Ref. 9) that the barrier and non-barrier fuel designs both meet the same approved criteria, but GE did not provide (Ref. 15) numerical design limits or calculational values that would show relative margins. However, during a meeting in San Jose in March 1983, GE’s fuel design model was discussed, and a comparison of design input parameter assumptions and design margins for the barrier and non-barrier fuel types was presented. It was shown that for cladding stress GE’s analytical objective is to demonstrate that no design ratio exceeds unity.*

* The “design ratio” is defined as the calculated stress or (stress intensity) divided by the appropriate stress (or stress intensity) limit.

The comparison of parametric inputs and calculational results for P8x8R (non-barrier) and BP8x8R (barrier) fuel indicated that, while the stress design objective was met for both fuel types (i.e., no design ratio exceeded unity), it was necessary to assign a nominal initial helium internal rod pressure for the barrier fuel to avoid calculating design ratios greater than one. For non-barrier fuel, margins were sufficiently large to permit the assumption of zero initial rod pressure. Even if design ratios were to equal unity, however, GE believes that there would still be substantial real margin because of conservatisms built into the design model. On the basis of the information presented at the March 1983 meeting, we conclude that the barrier fuel stress design limits are adequately met, even though in some cases the barrier fuel design margins are smaller than for non-barrier fuel.

A similar conclusion applies to the creep collapse analysis. In the fatigue analysis, however, the margin is actually greater for barrier fuel than non-barrier fuel because the stress concentration factor is lower (i.e., the zirconium layer is more ductile than Zircaloy).

With regard to the LHGR for 1% cladding strain, we had noted that the same LHGR values were listed for barrier and non-barrier fuel in GESTAR-II Table 2-3a. Inasmuch as 10% of the barrier fuel cladding wall thickness is comprised of high purity zirconium, which is weaker and more ductile than Zircaloy-2, it follows that the LHGR for 1% strain of barrier fuel cladding would be slightly less than that for standard, all Zircaloy-2 cladding. In the March 1983 meeting, however, we learned that GE calculates cladding strain solely on the basis of the calculated thermal expansion of the U02 pellets. For this GE calculational method, therefore, the cladding material properties have no bearing on the result. Inasmuch as the 1% cladding strain criterion has only a loose association with observed failure propensity, and since the LHGR values for fuel in operation are well below those calculated for 1% cladding strain, we conclude that this matter has been adequately addressed.

Stress Analysis Assumptions and Results

In our review of the barrier fuel amendment we noted that the stress analyses for the barrier fuel (BP8x8R) and the standard prepressurized fuel (P8x8R) designs were performed differently (see Section 2.5.3.1.2) even though both types of fuel are prepressurized. We also noted that, in the Fatigue Evaluation, Section 2.5.3.1.4, a minimum cladding thickness had been used for the unpressurized fuel, whereas nominal thickness were used for pressurized barrier and non-barrier fuel. We raised a question (Q490.4) to obtain the reason for the different input assumption in these calculations and to obtain a comparative stress analysis for each of the fuel types, using a consistent set of input assumptions, to show the relative margins to failure.

As indicated in the preceding discussion on design margins, it was necessary to assume a nominal prepressurization level (internal rod pressure) for the barrier fuel to avoid exceeding the design ratio criterion. Inasmuch as no design limits are exceeded and since the prepressurization level is not excessive, that input assumption is appropriate and acceptable.

With regard to the use of nominal dimensions instead of minimum dimensions in the fatigue analysis, it was pointed out by GE in response (Ref. 9) to Q490.4 that both assumptions have NRC approval (Ref. 16) and that GE had simply not bothered to redo the earlier calculations. Inasmuch as (a) the fatigue analysis was performed with an approved, conservative code (TEXICO), (b) the barrier cladding has a larger fatigue margin than non-barrier fuel (due to having a lower stress concentration factor), and (c) the margins were very substantial in either case, this is an acceptable response.

Effect of Precursor Material

In GEAP-25163-6 (Ref. 7), it is indicated that three barrier fuel liner types are being tested: (1) liners prepared from low oxygen sponge (approx. 440 ppm oxygen) zirconium, (2) liners prepared from crystal bar (approx. 90 ppm oxygen) zirconium, and (3) liners prepared from reactor grade (approx. 1000 ppm oxygen) zirconium. It is also noted in the report that there appears to be an impurity effect on Zr-liner performance, and that power ramp failures have been observed in tests at Studsvik. This raised a question (Q490.5) regarding the potential effect of liner precursor material on subsequent fuel rod cladding performance.

GE replied (Ref. 9) that the GE commercial product is the zirconium-liner cladding with the low oxygen sponge precursor material. We thus expect to be informed of any plan to change the precursor material and in such a case to be provided with data to show that there are no adverse effects of such changes on fuel performance. It should be noted, therefore, that this safety evaluation addresses the use of barrier fuel produced from crystal bar or low oxygen sponge zirconium only.

Lead Test Assembly (LTA) Irradiations

There are currently four barrier fuel LTAs in the core of Unit 1 of the Quad Cities Nuclear Power Station (Ref. 7). Only one of these LTAs is fully prototypic of the GE commercial product; two of the LTAs have a copper lining, and one LTA has a crystal bar Zr liner. Performance of the two Zr-liner LTAs has reportedly been excellent (Ref. 9) through two cycles (B.U. approx. 22MWd/kgU) of irradiation.

The Quad Cites 1 barrier fuel LTAs are undergoing continued irradiation to higher burnups. According to Ref. 6, the two Zr-liner LTAs were disassembled after their first cycle of irradiation and ten selected rods (5 from each assembly) were given detailed inspections, including length measurements, profilometry, ultrasonic and eddy current examinations, and visual inspections. We expect these examinations to continue and to continue being reported as part of a joint cooperative program between Commonwealth Edison, General Electric, and the Department of Energy, and we view these examinations as providing needed support for barrier fuel design and licensing.

Quad Cities 2 Demonstration Irradiation

The on-going demonstration irradiation of pellet/cladding interaction (PCI)-resistant BWR fuel involves a large-scale (144 bundles) irradiation in Unit 2 of the Quad Cities Nuclear Power Station. It is proposed that about half [64] of the bundles would be power ramped in groups of 16; i.e., one group of 16 could be ramped at the end of each of four successive reactor cycles. The program began with the current cycle of operation (Cycle 6), which is nearing its end. Because the demonstration irradiation in QC-2 provides the operating experience leg of the licensing triad (prototype testing and analytical predictions are the other two legs—see SRP Section 4.2.II.C) for demonstrating the acceptability of a new fuel design, we raised a question (490.7) concerning the status and results (e.g., off-gas activity, number of failed rods measured or inferred) obtained to date on the demonstration irradiation. We also asked for a description of the ramp tests and the post-irradiation and interim examinations that would be conducted on the demonstration bundles.

In response (Ref. 9), GE stated that off-gas activity has been steady during QC-2 Cycle 6 operation and that it is inferred from isotopic data that there are no failed bundles in the core at this time. With regard to the ramp test description, GE referred to a letter from L.O. DelGeorge (Comm. Ed.) to D.G. Eisenhut, dated December 3, 1981 (Ref. 17). In that letter Commonwealth Edison promised to provide

the demonstration irradiation operating plan when more refined predictions became available (anticipated by June 1982). The requested details were actually provided in November 1982 (Ref. 18).

With regard to interim examinations of the demonstration bundles, it was agreed (Ref. 18) that Commonwealth Edison would notify NRC headquarters and the region office should off-gas activity increase during the EOC6 ramp tests. It was further agreed that Commonwealth Edison would sip the test cell assemblies and any buffer region assemblies that were scheduled for reinsertion for Cycle 7, to confirm that the cladding was sound, even if no failure indications were evident from offgas and coolant monitoring. We believe that there is reasonable assurance that the demonstration irradiation is proceeding well and according to plan and that the interim examination program is adequate. Post-irradiation examinations, as apart from interim examinations, will be discussed later in this report evaluation.

Hydriding Behavior

As indicated in GEAP-25163-6 (Ref. 7), hydrogen charging laboratory experiments indicate that preferential hydriding of the Zr-liner occurs. It was also stated, however, that in-reactor performance of barrier fuel was not expected to be affected. In response to a staff question (Q490.8) concerning the reason why barrier fuel performance would not be affected, GE replied (Ref. 9) that only the cladding over the fuel rod plenum had a higher concentration of hydrides in the liner, and that the usual hydrogen distribution was expected in cladding adjacent to the fuel pellets. We believe that this observation provides significant assurance that hydriding will not be a problem in barrier fuel. We expect that additional confirmation will be provided by the post-irradiation examinations (discussed in the next section of this safety evaluation) to be conducted as part of PHASE 3 of the QC-2 demonstration irradiation cooperative program.

Surveillance

Section 4.2 of the Standard Review Plan indicates that, for a fuel design that introduces new features, a surveillance program commensurate with the nature of the changes should be performed. From our recent discussions with GE and Commonwealth Edison, we understand that a post-irradiation examination (PIE) program is intended for the Quad Cities Unit 2 demonstration bundles. This is consistent with our understanding from an earlier review of that demonstration program (see NRC's SER in Ref. 5). Because we believe that an adequate surveillance program will be performed at Quad Cities, we consider the surveillance issue to be resolved.

Fuel Rod Bowing Measurements

Because the barrier fuel rod cladding consists of two different zirconium based metals that are metallurgically bonded by a co-reduction process, it is possible that the barrier-clad fuel rods may be more susceptible to bowing than standard Zircaloy-2 rods. Hence, we directed a question (Q490.10) on barrier fuel rod bowing measurements. General Electric's response (Ref. 9) referred to visual examinations on the LTAs in QC-1 that indicated that no noticeable rod bowing has occurred. GE further stated, however, that visual examinations will continue to be performed, including rod bowing measurements as required, on the barrier LTAs.

With regard to the barrier LTAs, it is our expectation that such inspections will continue at each refueling outage at QC-l as long as the LTAs continue to be irradiation tested. Since the LTAs are undergoing such detailed inspections, including disassembly, we believe that actual rod-to-rod gap spacing measurements could be made to provide hard data on the magnitude of rod bowing in barrier fuel at a relatively low cost.

With regard to the demonstration assemblies in QC-2, a representative number of the barrier demonstration assemblies could be examined visually for rod bowing upon discharge. We suggest, therefore, that such measurements be considered in the planning of the PHASE 3 portion of the QC-2 barrier fuel demonstration.

SUMMARY

We have completed our review of the barrier fuel amendment to NEDE-24011. In our evaluation we focused on those aspects of the barrier cladding that are unique or particularly relevant to barrier fuel performance. Based on our evaluation of the information provided in (a) the amendment, (b) in other reports dealing with the lead test assemblies and demonstration irradiation program, and (c) in responses to requests for additional information, we conclude that there will be a reasonable assurance that the substitution or first-core use of barrier fuel will not result in unacceptable hazards to the public. Therefore, the barrier fuel amendment is approved as a generic reference and may be included in NEDE-24011-P-A.

REFERENCES

1. J. S. Charnley (GE), letter to R. L. Tedesdo (NRC), “Administrative Amendments to GE Licensing Topical Reports NEDE-24011-P-A-4 and NEDE24011-P-A-4-US,” August 31, 1982.

2. Carl H. Berlinger (NRC), memorandum to J.S. Berggren, Administrative Amendments to GE Licensing Topical Reports NEDE-24011-P-A-4 and NEDE-24011P-A-4-US,” October 19, 1982.

3. J. S. Charnley (GE), letter to Frank J. Miraglia (NRC), “Barrier Fuel Amendment to NEDE-24011-P-A-4,” November 19, 1982.

4. NUREG-0390, “Topical Report Review Status.”

5. “Generic Information for Barrier Fuel Demonstration Bundle Licensing,” NEDO-L24259-A, February 1981.

6. J. R. Thompson, “Barrier Fuel Lead Test Assemblies—Pre-irradiation Characterization and First Cycle Post-irradiation Examination,” GEAP-25447-1, November 1981.

7.
    a. Demonstration of Fuel Resistant to Pellet-Cladding Interaction, PHASE-2 Seventh Semiannual Report, January-June, 1982, complied by H.S. Rosenbaum, GEAP-25163-7, September 1982.

    b. Demonstration of Fuel Resistant to Pellet-Cladding Interaction, PHASE-2, Fifth Semiannual Report, July-December 1981, compiled H.S. Rosenbaum, GEAP-25163-6, March 1982.

    c. Demonstration of Fuel Resistant to Pellet-Cladding Interaction, PHASE-2, Fifth Semiannual Report, January-June 1981, complied by H.S. Rosenbaum, GEAP-25163-5, September 1981.

    d. Demonstration of Fuel Resistant to Pellet-Cladding Interaction, PHASE-2, Fourth Semiannual Report, July-December 1980, compiled by H.S. Rosenbaum, GEAP-25163-4, March 1981.

e. Demonstration of Fuel Resistant to Pellet-Cladding Interaction, PHASE-2, Third Semiannual Report, January-June 1980, compiled by H.S. Rosenbaum, GEAP-25163-3, September 1980.

f. Demonstration of Fuel Resistant to Pellet-Cladding Interaction, PHASE-2, Second Semiannual Report, July-December 1979, compiled by H.S. Rosenbaum, GEAP-25163-2, March 1980.

g. Demonstration of Fuel Resistant to Pellet-Cladding Interaction, PHASE-2, First Semiannual Report, January-June 1979, compiled by H.S. Rosenbaum, GEAP-25163-1, August 1979.

h. Demonstration of Fuel Resistant to Pellet-Cladding Interaction, PHASE-1, Final Report, compiled by H.S. Rosenbaum, GEAP-23773-2, March 1979.

i. Demonstration of Fuel Resistant to Pellet-Cladding Interaction, Second Semiannual Report, January-June 1978, compiled by H.S. Rosenbaum, GEAP-23773-1, September 1978.

j. Demonstration of Fuel Resistant to Pellet-Cladding Interaction, First Semiannual Report, July-December, 1977, compiled by H.S. Rosenbaum, GEAP-23773, February 1978.

8. C.H. Berlinger (NRC), memorandum for J.S. Berggren, “Questions on Barrier Fuel Amendment to GE Licensing Topical Report NEDE-24011-P-A-4,” January 13, 1983.

9. J.S. Charnley (GE), letter to C.O. Thomas (NRC), Subject: “Response to Questions on Barrier Fuel Amendment to NEDE-24011-P-A-4, February 22, 1983.

10. J.S. Charnley (GE), letter to F.D. Coffman (NRC), “Presentation Slides—December 11, 1979 meeting on Vermont Yankee Fuel,” December 11, 1979.

11. R. Engel (GE), letter to M. Tokar (NRC), “Corrosion Product Control,” October 2, 1980.

12. R.L. Smith (VYNPC), letter to USNRC, NRR, November 21, 1980.

13. L.O. DelGeorge (COMM.ED.), letter to B.J. Youngblood (NRC), February 9, 1981, with LRG working paper response dated December 2, 1980.

14. M. Tokar (NRC), memorandum to C.H. Berlinger, “GE Proprietary Presentation on Waterside Corrosion of Gadolinia Fuel,” October 13, 1982.

15. J.S. Charnley (GE), letter to C.H. Berlinger (NRC), “Presentation Made During NRC-GE Meeting,” January 25, 1983.

16. R.L. Tedesco (NRC), letter to R.E. Engel (GE), “Acceptance for Licensing Reference of Changes to Topical Report Number NEDE-24011-P-A-l, ‘Generic Reload Application’ dated August 1979,” November 7, 1980.

17. L.O. DelGeorge (Comm. Ed.), letter to D.G. Eisenhut (NRC), December 3, 1981.

18. L.O. DelGeorge (Comm. Ed.), letter to D.G. Eisenhut (NRC), Subject: “Quad Cities Station Unit 2 Barrier Fuel Ramp Test Information,” November 1, 1982.

19. Lester S. Rubenstein (NRC), memorandum to Robert L. Tedesco, “Quad Cities Unit 2 Cycle 6 Barrier Fuel Demonstration,” December 17, 1981.

NRC
Safety Evaluation Report
Approving Section 4.2
of
GESTAR II BWR/6 Nuclear Island Design

Section 4.2 of the Safety Evaluation Report related to the final design approval of the GESSAR II BWR/6 Nuclear Island Design

NUREG-0979

Docket No. 50-447

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GESSAR II - SECTION 4.2 SER

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4 REACTOR

4.1 General

The safety analysis for the GESSAR II fuel system design, as originally presented in Section 4.2 of the GESSAR II safety analysis report, was supplanted by information provided in GESTAR II (General Electric's generic topical report for reactor fuel—NEDE-24011) and in other GE generic topical reports. Because the format of the GESTAR II report does not coincide with the SRP for the fuel system design (Section 4.2), GE provided a "roadmap" (Appendix A to NEDE-24011-P-A-5) that indicated where the required information could be found in the GESTAR II report. Although GESTAR II is a reviewed and approved document, it was originally submitted as a reload document only, and as such was not reviewed in accordance with the SRP (NUREG-0800). It was necessary, therefore, to perform an SRP-type review of NEDE-24011* with special attention to those portions of the document that appeared not to be in conformance with the SRP and other NRC guidelines or regulations. The following sections address the staff's review of GESTAR II and other documents and information submitted in support of the GESSAR II fuel system design.

4.2 Fuel System Design

The objectives of this fuel system safety review as described in Section 4.2 of the SRP are to provide assurance that (1) the fuel system is not damaged as a result of normal operation and anticipated operational occurrences, (2) fuel system damage is never so severe as to prevent control rod insertion when it is required, (3) the number of fuel rod failures is not underestimated for postulated accidents, and (4) coolability is always maintained. A "not-damaged" fuel system is defined as one whose fuel rods do not fail, fuel system dimensions remain within operational tolerances, and functional capabilities are not reduced below those assumed in the safety analysis. Objective (1) above implements GDC 10 (10CFR50, Appendix A), and the design limits that accomplish this are called specified acceptable fuel design limits (SAFDLs). "Fuel rod failure" means that the fuel rod leaks and that the first fission product barrier (the cladding) has, therefore, been breached. Fuel rod failures must be accounted for in the dose analysis required by 10CFR100 for postulated accidents. "Coolability," which is sometimes called "coolable geometry," means, in general, that the fuel assembly retains its rod-bundle geometrical configuration with adequate coolant channeling to permit removal of residual heat even after a severe accident. The general requirements to maintain control rod insertability and core coolability appear repeatedly in the general design criteria (e.g., GDC 27 and 35). Specific coolability requirements for the loss-of-coolant accidents are given in 10CFR50.46.

To ensure that the above stated objectives are met, the following areas are examined: (1) design bases, (2) description and design drawings, (3) design evaluation, and (4) testing, inspection, and surveillance plans. In assessing the adequacy of the design, operating experience, prototype testing, and analytical predictions are compared with acceptance criteria for fuel system damage, fuel rod failure, and fuel coolability.

* Unless otherwise specified, reference to NEDE-24011 is intended to apply to the latest approved version of the report, NEDE-24011-P-A-4.

The staff's technical review focused primarily on the need to clarify GE's design bases and limits for the GESSAR II/GESTAR II fuel, using as an aid in conducting the review the GE "roadmap" (letter from J.S. Charnley (GE), Sept. 8, 1982) that was provided to cross-reference the GESTAR II report with SRP Section 4.2. With the aid of the roadmap, it became clear that some questions regarding GE's design bases and limits remained, and so a meeting was held between GE and NRC staff in Bethesda, Maryland, at which GE presented (letter from J.S. Charnley (GE), Jan. 25, 1983) their design bases and limits and responded to other staff questions that had been issued earlier (memorandum from C.H. Berlinger, Oct. 12, 1982).

4.2.1 Design Bases

Design bases for the safety analysis address fuel system damage mechanisms and suggest limiting values for important parameters so that damage will be limited to acceptable levels. For convenience, the acceptance criteria for these design limits are grouped into three categories in the SRP: (1) fuel system damage criteria, which are most applicable to normal operation, including anticipated operational occurrences (AOOs), (2) fuel rod failure criteria, which apply to normal operation, AOOs, and accidents, and (3) fuel coolability criteria, which apply to accidents. For each of these three categories (or levels) of fuel damage, damage mechanisms that are believed to be pertinent to that degree of damage are listed in the SRP, and a design-basis statement, indicating GE's design objective relative to the damage mechanism, is called for along with a related design limit—often a numerical value of a design parameter (e.g., 1% cladding plastic strain).

In response to staff questions (letter from J.S. Charnley (GE), Jan. 25, 1983), GE provided design-basis and design-limit statements which were not in accordance with the typical SRP statements applicable to the fuel design review and which in some cases were not acceptable. This required that the staff develop alternate bases for completing its review to compensate for these deviations. Because the SRP is a guide and does not contain enforceable requirements, the staff has proceeded to evaluate the "alternate" proposals. It, therefore, has listed in the following sections GE's design-basis and design-limit statements for the fuel damage considerations identified in SRP Section 4.2 and has provided its interpretation of those statements, consistent with the intent of the SRP and GDC 10.

With regard to conditions of normal operation, including abnormal operating occurrences (AOOs), the relationship of the design bases and limits to GDC 10's requirement for the preservation of SAFDLs is of crucial importance. Traditionally, the staff has interpreted GDC 10 to mean that fuel should be designed to preclude (all but a few stochastic) fuel failures during normal operation, including AOOs. It will be demonstrated in the following sections that the staff continues to hold GE's GESSAR II/GESTAR II fuel design to that standard.

4.2.1.1 Fuel System Damage Criteria

The following paragraphs discuss the NRC staff's evaluation of the design bases and corresponding design limits for the damage mechanisms listed in the SRP. These design limits along with certain criteria that define failure (see Section 4.2.1.2 of this SER) constitute the SAFDLs required by GDC 10. The design limits in this section should not be exceeded during normal operation including AOOs.

4.2.1.1.1 Cladding Design Stress

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 analyses. GE's design basis for cladding stress is that "the fuel rod cladding is evaluated to ensure that the fuel will not fail due to fuel cladding stresses exceeding the cladding

mechanical capability." To satisfy that design basis, GE uses the strength theory, terminology, and stress categories presented in ASME Code, Section III, as a guide for determining fuel rod cladding stress limits. Because, as indicated in SRP Section 4.2.II.A.l, stress limits that are obtained by methods similar to those given in Section III to the ASME Code are acceptable, the GE cladding stress limits are acceptable.

4.2.1.1.2 Strain Fatigue

GE's design basis for cladding fatigue is that "the fuel rod cladding is evaluated to ensure that the fuel will not fail due to cyclic fuel rod loadings exceeding the cladding fatigue capability"; that is, the fuel is designed not to fail because of fatigue. Strain fatigue calculational results listed in Tables 2-13 and 2-13a of NEDE-24011 indicate that criteria described in SRP Section 4.2—a safety factor of 2 on stress amplitude or a safety factor of 20 on the number of cycles—are applied. In addition, where stress rupture and fatigue cycling are both significant, NEDE-24011 indicates that GE applies the following limiting condition:

$$\frac{\text{Actual time at stress}}{\text{allowable time at stress}} + \frac{\text{actual number of cycles at stress}}{\text{allowable cycles at stress}} \leq 1.0$$

This design limit is more conservative than that in the SRP; the staff has, therefore, on previous applications referencing NEDE-24011 found the design limit and analysis of strain fatigue to be acceptable.

In a recent presentation (letter from J.S. Charnley (GE), Jan. 25, 1983), however, GE provided a new fatigue design limit. The new fatigue limit is the lower bound curve of a larger data set than that used by O'Donnell and Langer, whose work is referenced in the SRP. The NRC staff has been briefed by GE on this new information (memo from D.A. Powers (NRC), Mar. 1983) and has compared this new curve (Figure 2-lla of NEDE-24011) with the O'Donnell and Langer curve after adjusting it for a safety factor of 2 on stress amplitude and/or 20 on the number of cycles. The staff found this comparison to be quite favorable and thus concludes that the new GE fatigue curve is conservative and acceptable.

4.2.1.1.3 Fretting Wear

Although the SRP does not provide numerical bounding-value acceptance criteria for fretting wear, it does stipulate that the allowable fretting wear should be stated in the safety analysis and that the stress and fatigue limits should presume the existence of this wear. GE's design-basis statement (letter from J.S. Charnley, Jan. 25, 1983) is that "the fuel assembly is evaluated to ensure that the fuel will not fail due to fretting wear of the assembly components." The staff accepts that statement and the discussion presented in presented in Section 2.6.3 of NEDE-24011 as providing sufficient evidence that fretting wear is considered in the design analysis. Further confirmation of this is provided in a GE letter (letter from R.E. Engel, Aug. 11, 1981), which also indicates that analytical criteria for the fuel assembly mechanical design are provided in Sections 2.5.1 and 2.5.1.1 of NEDE-24011. Although those criteria do not directly address fretting wear, assurance is provided that the general requirement (discussed in SRP Section 4.2) concerning reduction of functional capability is met. Because the SRP does not provide numerical acceptance criteria for fretting wear, and because fretting wear is addressed in the design analysis, the NRC staff concludes that the intent of the SRP has been adequately met.

4.2.1.1.4 External Corrosion and Crud Buildup

With respect to external corrosion and crud buildup, GE's design basis (letter from J.S. Charnley, Jan. 25, 1983) is that "the fuel rod is evaluated to ensure that the cladding temperature increase and cladding metal thinning due to cladding corrosion, and the cladding temperature increase due to the buildup of corrosion products, do no result in fuel rod failure due to reduced cladding strength." GE does not specify a maximum cladding external temperature to limit corrosion or an external corrosion or crud layer thickness that would significantly affect design margins. However, the SRP does not provide numerical limits for cladding temperature or degree of oxidation for normal operation. Inasmuch as GE has demonstrated that corrosion and crud buildup are accounted for in the design analysis, the staff concludes that GE's approach is consistent with the SRP guidelines.

4.2.1.1.5 Dimensional Changes

Fuel assembly components such as the fuel rods and channel boxes may undergo various types of dimensional changes such as fuel rod bowing, axial growth, and channel box deflection. Such phenomena are related to neutron fluence, fuel burnup, and assembly core residence time and must be accounted for in the fuel design analyses to establish operational tolerances and to ensure that all effects are accommodated in the thermal and mechanical design.

Fuel rod bowing is a phenomenon that alters the pitch dimensions between adjacent fuel rods and thus affects local nuclear power peaking and heat transfer to the coolant. GE's design basis for rod bowing (letter from J.S. Charnley (GE), Jan. 25, 1983) is that "the fuel rod is evaluated to ensure that rod bowing does not result in fuel failure due to boiling transition." To ensure that that design objective is satisfied, GE has established (NEDE-24011) a "deflection limit" corresponding to about half the nominal spacing to ensure that the rod-to-rod and rod-to-channel clearances are sufficient to allow free passage of coolant water to all heat transfer surfaces. Although the SRP does not suggest specific numerical limits for rod bowing, the proposed deflection limit is consistent with the current industry treatment of fuel rod bowing and is, therefore, acceptable.

With regard to channel box deflection, GE has issued a generic report on channel design and deflection (NEDE-21354-P) and has responded to NRC questions in two supplements. The GE report documents (1) the fuel channel description, (2) the design bases and analyses, and (3) the creep deflection phenomenon. In the design-basis section of the generic report, the design basis for channel structural adequacy is discussed in terms of material properties such as yield strength, stress-rupture strength, and operational duty including normal maneuvers, transients, and accidents. The discussion meets the guidance of the SRP with regard to design bases and is, therefore, acceptable in that respect.

With regard to irradiation-induced axial growth of the fuel assemblies, GE's design basis (letter from J.S. Charnley (GE), Jan. 25, 1983) is that "the fuel assembly is evaluated to ensure that irradiation-induced axial growth does not result in fuel failure due to insufficient axial expansion space." This design basis is consistent with the intent of the SRP and is, therefore, acceptable.

4.2.1.1.6 Fuel and Poison Rod Pressures

The applicant's design-basis statement for fuel rod internal pressure (letter from J.S. Charnley (GE), Jan. 25, 1983) is that "the fuel rod is evaluated to ensure that the effects of fuel rod internal pressure will not result in fuel failure due to excessive cladding pressure loading," and GE contends that a rod internal pressure limit (less than or equal to the coolant system pressure) is not necessary. The applicant has indicated that rod internal pressure is used in conjunction with other loads on the fuel rod cladding in the calculation of cladding stresses. The results of such calculations, as provided in NEDE-

24011, show that the calculated cladding stresses can be accommodated. The staff has examined these calculations and agrees that the acceptance criterion for cladding stress (see SRP Section 4.2.II.A.1.a) has been met, but that criterion is not the same as the one recommended by the NRC for rod internal pressure (see SRP Section 4.2.II.A.1.f).

The SRP (NUREG-0800) identifies fuel and burnable poison rod pressure as a potential fuel system damage mechanism separate from that identified under the cladding stress criterion. This is because the criterion for fuel rod internal pressure involves more than the cladding mechanical limits. There are a number of reasons for this distinction:

1. Outward (tensile) cladding stresses may force analytical methods into a mode for which they were not designed or where greater uncertainties exist.

2. The higher releases of fission gas associated with higher rod pressures may lead to a level of undesirable positive thermal feedback.

3. The higher releases of fission gas associated with higher rod pressures may lead to underestimating the radiological consequences of accidents using release assumptions of RGs 1.25 and 1.77.

4. Net positive pressures would lead to ballooning during non-LOCA departure from nucleate boiling events, and such behavior is not analyzed in the accident analysis.

In past reviews of GE BWRs, the staff has ignored the absence of a design limit on rod internal pressure because it believed that the rod pressure was nevertheless remaining below system pressure. Although the staff notes that the rod internal pressures reported in the GE cladding stress analysis are well in excess of the nominal coolant system pressure, this has been the case for a number of license applications which refer to the generic analyses in NEDE-24411-P-A. Those calculations rely on conservative assumptions which are made to simplify the analysis and, as a result, do not provide an accurate estimate of rod internal pressure (the calculated pressures are conservatively high). To assess the impact of fuel rod internal pressure on the safety analysis of other plants, the staff has, in the past, relied on other, more representative (but still conservative) analyses from GE (letter from G. Sherwood, Dec. 22, 1976). These calculations show that fuel rod internal pressures remain below the nominal system pressure for planar average burnups below 44 MWd/kg U. This conclusion remained unchanged for the newer prepressurized fuel design as well (NEDO-23786-1). On the basis of these calculations, the staff concluded in the past that the rod internal pressure criterion, although not explicitly used by GE, had nevertheless been met.

More recent information from GE (letter from J.F. Quirk, Jan. 21, 1982) indicates that fuel rod internal pressures may exceed system pressure even on the basis of the more representative calculations. The staff attributes this change in calculated pressure to the higher burnup levels now being analyzed and to the advent of more appropriate fission gas release data and methods of analysis. The change has thus prompted the staff to reexamine the matter of conformance with Section 4.2.II.A.1.f of the SRP.

Although NUREG-0800 proposes the nominal coolant system pressure as an acceptable limit for fuel rod internal gas pressure, alternative criteria may also be considered if justified. In general, such alternatives must show that fuel rod internal pressures in excess of system pressure do not (1) lead to fuel system damage during normal operation and AOOs, (2) prevent control rod insertion when required, (3) result in an underestimate of the number of fuel failures in, or radiological consequences of, postulated accidents, or (4) lead to loss of coolable geometry.

GE has proposed (letter from J.S. Charnley, Jan. 25, 1983) an alternative criterion which relates cladding creepout to fuel swelling rate. This criterion, which allows rod internal pressures to exceed system pressure, appears to preclude the situation in which the fuel-to-cladding gap could open because of the high internal gas pressure. If so, such a criterion may indeed meet the intent of objective (1) given above. However, it does not address the remaining objectives of its review with regard to rod internal pressure. The applicant has not provided a basis from which the staff could conclude that the remaining objectives of the rod pressure criterion have been met. At this time, the staff regards the issue as unresolved and will continue to address the rod internal pressure criterion in a supplement to this SER.

4.2.1.1.7 Fuel Assembly Liftoff

The SRP calls for the fuel assembly holddown capability (gravity and springs) to exceed worst-case hydraulic loads for normal operation, which includes AOOs. The GE design basis for assembly liftoff (letter from J.S. Charnley, Jan. 25, 1983) is that "the fuel assembly is evaluated to ensure that interference sufficient to prevent control blade insertion will not occur." Thus GE's design basis would not preclude assembly liftoff. However, in a generic meeting on this issue held in Bethesda, Maryland, on January 26, 1983, GE stated that a vertical liftoff greater than 0.52 in. would be required to permit sufficient lateral displacement to result in control blade interference. Therefore, although a specific numerical limit for liftoff has not been provided on the GESSAR II docket, the staff will accept the generic analysis limit of 0.52 in., which has been proposed as a physical bounding value for all GE fuel designs and plants, as the design limit for GESSAR II.

4.2.1.1.8 Control Material Leaching

The SRP and GDC require that control rod reactivity be maintained. Control rod reactivity can sometimes be lost by leaching of certain poison materials if the control rod cladding is breached, as indeed has been observed (letter from D.G. Eisenhut (NRC), Oct. 22, 1979). GE has postulated (NEDE-24226-P) that breaching by cracking is caused by stress corrosion resulting from solidification (sintering) of the boron carbide (B_4C) particles in the rods followed by swelling (caused by helium and lithium) of the sintered B_4C. The stress in the tubes caused by the B_4C is believed to accelerate the intergranular corrosion that proceeds from the outside surface of the rod cladding.

Before the discovery of the B_4C leaching loss mechanism, the previously defined life-limiting parameter for the control blades was loss of boron by depletion. The current generic criterion defining end of control blade life is a loss of total reactivity equal to 10% of initial control blade worth. If total boron-10 loss by depletion and leaching is considered, GE has determined (NEDE-24226-P) that a 10% reduction in worth occurs when the total boron-10 reduction, averaged over the top quarter of the blade, reached 34%. Based on its review of the available data and discussions with GE personnel, the staff has determined that the 10% reduction-in-worth criterion is appropriate.

GE's design basis for control rod reactivity (letter from J.S. Charnley, Jan. 25, 1983) is that "the fuel system is evaluated to ensure that the reactivity control required to bring the reactor to cold shutdown is maintained." That statement is consistent with the SRP guidelines and GDC requirements and the design-life criterion defined above. The GESSAR II design basis for control rod reactivity is, therefore, acceptable.

4.2.1.2 Fuel Rod Failure Criteria

The NRC staff's evaluation of fuel-rod-failure thresholds for the failure mechanisms listed in the SRP is presented in the following paragraphs. When these failure thresholds are applied to normal or transient operation, they are used as limits (and hence SAFDLs) because fuel failure under those conditions should not occur (according to the traditional conservative interpretation of GDC 10). When these thresholds are applied to accident analyses, the number of fuel failures must be determined for input to the radiological dose calculations required by 10CFR100. The basis or reason for establishing these failure thresholds is thus predetermined, and only the threshold values are reviewed below.

4.2.1.2.1 Internal Hydriding

To ensure that failure will not occur because of internal cladding hydriding, GE specifies a fabrication limit on moisture content. Based on the reported hydride formation threshold value, the H_2O specification limit for 8x8 fuel (set at a value less than half the hydride formation threshold value) was established on the basis of the previous limit for 7x7 fuel and information obtained from Joon (1972). To include the total fuel moisture content, GE converts the total fuel rod moisture limit to a total fuel column hydrogen limit. The GE limits for moisture and hydrogen are well below those specified in SRP Section 4.2, and the current American Society for Testing and Materials specification (C776-76, Part 45) for a UO_2 pellet-equivalent limit of 2 ppm of hydrogen from all sources is also satisfied. The GE limits for fuel moisture and hydrogen content to preclude hydriding failures are, therefore, acceptable.

4.2.1.2.2 Cladding Collapse

If axial gaps in the fuel pellet column were to occur because of 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. To define a collapse criterion to reflect the operational conditions of the reactor, GE (NEDE-20606-P-A) has adopted 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. NEDE-20606-P-A, which contains these limits and definitions, has been reviewed and approved (letter from W.R. Butler (NRC), Apr. 1975). The NRC staff, therefore, concludes that the cladding collapse design limit has been adequately addressed for the GESSAR II fuel design.

4.2.1.2.3 Overheating of Cladding

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 by high temperature cladding mechanisms. For BWR fuel, thermal margin is stated in terms of the minimum value of the critical power ratio (MCPR), which corresponds to the most limiting fuel assembly in the core. As indicated in Section 5 of NEDE-24011-P-A, 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 and NEDO-10958-A. These design limits are consistent with the thermal margin guidelines of SRP Section 4.2.II.A.2 and are thus acceptable from the standpoint of the fuel mechanical design. The review of thermal/hydraulic design methods is described in Section 4.4 of this SER.

4.2.1.2.4 Overheating of Fuel Pellets

Although it is stated in SRP Section 4.2.II.A.2.e that it has been traditional practice to assume that failure will occur if fuel pellet centerline melting takes place, that conservative assumption has been strictly applied only to PWR fuel. For BWRs, a limited amount of calculated UO_2 melting has been permitted for certain events such as rod withdrawal as long as the melting is not sufficient to cause 1% cladding plastic strain. See Section 4.2.1.2.6 for a discussion of the 1% strain limit. It should be noted that fuel melting is not expected to occur for normal steady-state full-power operation (see Section 2.4.2.5 of NEDE-24011-P-A).

GE's design-basis/limit statement for fuel pellet overheating (letter from J.S. Charnley, Jan. 25, 1983), namely, that "the fuel rod is evaluated to ensure that fuel rod failure due to excessive fuel melting will not occur during steady-state operation" is, however, somewhat vague and is not totally consistent with the NEDE-24011 statement and the long-standing NRC acceptance criterion for the avoidance of fuel melting during steady-state operation. This design limit implies that some fuel melting during steady-state operation is acceptable as long as it is not "excessive." That interpretation is clearly not in keeping with regulatory practice, and the current wording of the GESSAR II design-basis statement for fuel pellet overheating is, therefore, unacceptable. On the other hand, it is also evident that GE's intent is to avoid fuel failures resulting from fuel melting. The objective is consistent with regulatory practice as documented in SRP Section 4.2.II.A.2.e. The staff, thus, concludes that although GE's statement for fuel pellet overheating is unacceptable because it appears not to preclude fuel pellet melting during steady-state operation, it is clear from other documents (NEDE-24011) that GE really intends, as a design objective, to avoid such overheating, and thus the issue is closed.

4.2.1.2.5 Excessive Fuel Enthalpy

For a severe reactivity-initiated accident (RIA) in a BWR at zero or low power, fuel failure is assumed in the SRP to occur if the radially averaged fuel rod enthalpy is greater than 170 cal/g at any axial location. The 170 cal/g enthalpy criterion, developed from SPERT tests (Grund et al., Aug. 1969), is primarily intended to address cladding overheating effects, but it also indirectly addresses pellet/cladding interactions of the type associated with severe RIAs. As indicated in the letter from R.E. Engel of August 11, 1981 and in NEDO-10527, GE uses 170 cal/g as a cladding failure threshold and thus satisfies the SRP.

4.2.1.2.6 Pellet/Cladding Interaction

Fuel failures resulting from pellet/cladding interaction (PCI) have been encountered in operating BWR fuel (NEDE-24343-P). PCI generally occurs during a power increase as the fuel pellet expands and exerts stresses on the cladding. Although the exact mechanisms that contribute to PCI damage have not been established beyond a doubt, operating experience indicates that irradiated Zircaloy cannot readily accommodate stresses or strains of this kind, particularly when the Zircaloy has been exposed to certain embrittling (stress-corrosion) fission product species such as iodine or cadmium.

As indicated in SRP Section 4.2.II.A.2.g, there are no generally applicable criteria for PCI failure, but two related criteria are given: (1) 1% cladding strain and (2) centerline melting. The centerline melting

criterion was discussed in Section 4.2.1.2.4. GE uses a 1% plastic strain "safety" limit (i.e., design limit) for cladding strain based on data from fuel rods operated in BWRs (NEDO-10505). As noted in NEDE-24011-P-A, none of the data reported fell below the 1% plastic strain value, but a statistical distribution fit to the available data indicated that the 1% plastic strain value was approximately the 95% point in the total population. The distribution implied, therefore, that there was a small (less than 5%) probability that some cladding segments may have plastic elongations less than 1% at failure. Although a 1% plastic strain limit will not necessarily preclude all types of PCI failures (Tokar (NRC), Nov. 14, 1979), that strain criterion coincides with the other licensing criteria for PCI in the SRP and is, therefore, acceptable.

In a letter dated January 25, 1983 from J.S. Charnley, GE stated that "the fuel rods are evaluated to ensure fuel rod failure due to pellet-clad mechanical interaction excluding internal environmental effects will not occur during anticipated operational occurrences." That statement appears to be inappropriate for two reasons. First, the phrase "excluding internal environmental effects" implies that only PCI caused by cladding creepdown is considered in the design analysis, and that PCI caused by pellet thermal expansion (with or without UO_2 melting during an event such as control rod withdrawal) is ignored. The staff knows that that interpretation is incorrect because as indicated earlier the 1% cladding strain limit for rod withdrawals explicitly addresses the thermal expansion contribution resulting from UO_2 centerline melting. Second, the phrase "during anticipated operational transients" implies that the potential for PCI failures during normal operation power changes was not addressed as a part of GE's design analysis. That would be a direct violation of GDC 10. Although PCI failures can, and do, occur in relatively small numbers during normal BWR operation, the staff is confident that GE does not deliberately design fuel to have such failures. Therefore, although the staff believes that GE's recent statement for PCI is worded inappropriately, GE's continued use of the acceptance criterion of 1% cladding strain is in conformance with the SRP.

4.2.1.2.7 Cladding Rupture

Zircaloy cladding will rupture (burst) under certain combinations of temperature, heating rate, and stress during a LOCA. Although Appendix K to 10CFR50 requires that the incidence of rupture during a LOCA not be underestimated, there are no design limits required for cladding rupture. rupture-temperature correlation (NEDO-20566) is used in the LOCA emergency core cooling system analysis, and that correlation is evaluated in Section 4.2.3.2 of this SER.

4.2.1.2.8 Fuel Rod Mechanical Fracturing

The term "mechanical fracture" refers to a cladding defect that is caused by an externally applied force such as a hydraulic load or a load derived from core-plate motion. These loads are bounded by the loads of a LOCA and safe shutdown earthquake (SSE), and the mechanical fracturing analysis is usually done as a part of the seismic-and-LOCA loads analysis (see Section 4.2.3.3.4 of this SER). The entire seismic-and-LOCA loads evaluation (including design limits) has been described by GE in a topical report (NEDE-21175-3) to which GESSAR II makes reference. Because the staff's review of that report has not been completed, it is not clear what design limit will be used for the mechanical fracturing analysis. The NRC staff will report on this issue in a supplement to this SER.

4.2.1.3 Fuel Coolability Criteria

During major accidents in which severe fuel damage might occur, core coolability must be maintained as required by several GDCs (e.g., GDC 27 and 35). The following paragraphs discuss the staff's

evaluation of limits that will ensure that coolability is maintained for the severe damage mechanisms listed in Section 4.2 of the SRP.

4.2.1.3.1 Fragmentation of Embrittled Cladding

To meet the requirements of 10CFR50.46 as it relates to cladding embrittlement for a LOCA, acceptance criteria of 2200°F for peak cladding temperature and 17% for maximum cladding oxidation must be met. These criteria are used by the applicant (see NEDE-20566-A).

4.2.1.3.2 Violent Expulsion of Fuel

In a severe reactivity initiated accident 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 NEDO-10527 and Section 5.5 of NEDE-24011-P-A, GE uses this 280 cal/g criterion as a control rod drop accident design limit.

4.2.1.3.3 Cladding Ballooning

Zircaloy cladding will balloon (swell) under certain combinations of temperature, heating rate, and stress during a LOCA. Although Appendix K to 10CFR50 requires that the degree of swelling during a LOCA not be underestimated, there are no design limits required for cladding swelling. A burst-strain correlation (NEDO-20566) is used in the LOCA emergency core cooling system analysis, and that correlation is evaluated in Section 4.2.3.3 of this SER.

4.2.1.3.4 Fuel Assembly Structural Damage From External Forces

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

The entire seismic-and-LOCA loads evaluation (including design limits) has been described by GE in a topical report (NEDE-21175-3) to which GESSAR II makes reference. Because the staff's review of that report has not been completed, it is not clear what the design limits will be. The final resolution of this issue will be reported in a supplement to this SER.

4.2.2 Description and Design Drawings

The GESSAR II fuel assembly design is described in Section 2.1 of NEDE-24011. Design specification details are provided in Table 2-1 (NEDE-24011) and in the BWR/4 and BWR/5 fuel design topical reports (NEDE-20944-P and NEDE-20944-lP). Although each parameter listed in SRP Section 4.2.2 is not provided in the GE topical reports, enough information is given in sufficient detail to provide a reasonably accurate representation of the GESSAR II fuel design and thus satisfy the intent of the SRP. GE has also stated (letter from J.S. Charnley, Jan. 25, 1983) that there is sufficient descriptive information in GESTAR II (NEDE-24011) to permit audit calculations to be made. The staff has verified this statement and concludes that the GESSAR II fuel description is adequate.

4.2.3 Design Evaluation

Design bases and limits were presented and discussed in Section 4.2.1 of this SER. In this section the staff reviews GE methods of demonstrating that the GESSAR II fuel design meets the design criteria that have been established. This SER section will, therefore, correspond to Section 4.2.1 of this SER point by point. The methods of demonstrating that the design criteria have been met include operating experience, prototype testing, and analytical predictions.

4.2.3.1 Fuel System Damage Evaluation

The following paragraphs discuss the NRC staff's evaluation of the ability of the GESSAR II fuel to meet the fuel system damage criteria described in Section 4.2.1.1. Those criteria apply only to normal operation and anticipated transients.

4.2.3.1.1 Cladding Design Stress

A description of the fuel rod stress analysis is provided in Section 2.5.3.1.2 of NEDE-24011-P-A. Detailed evaluations of fuel assembly component structural integrity are presented in NEDE-23542-P along with descriptions of the methods used. As indicated in Section 2.3.1 of NEDE-20944-P, additional information regarding the fuel thermal and mechanical analyses is presented in NEDO-20360. The results of these analyses show that the GESSAR II fuel assembly design meets the stress criteria established by GE. The NRC staff has not audited these results because problems with design stress during normal operation are not expected based on general LWR fuel operating experience.

4.2.3.1.2 Strain Fatigue

According to information provided in Section 2.5.3.1.4 of NEDE-24011-P-A, GE's fuel design fatigue analysis utilizes the linear cumulative damage rule, that is, Miner's hypothesis (Miner, 1945). In the analysis, it is assumed that the area most subject to fatigue damage is the intersection of a fuel rod tube and an end plug at the weld. That intersection is a circumferential notch. Alternating stresses in the cladding wall that result from flow-induced vibration are substantially below those that correspond to Zircaloy's endurance limit. Following Peterson's fatigue theory (O'Donnell and Purdy, May 1964), fatigue failure at a sharp notch is assumed to be caused by the action of stress at a finite distance below the material surface. The method to calculate the stress at a small distance below the surface of a notch is provided in a reference (Winnie and Wundt, Nov. 1958). Using that method, and taking the depth below the notch from a reference (O'Donnell and Langer, 1964), GE has calculated (NEDE-24011) that the cumulative fatigue damage is less than the allowable damage. These are well-established standard theories and methods of performing a fatigue analysis and are, therefore, acceptable methods for a safety analysis. Although the staff has not been given a basis for GE's selection of the frequency and magnitude of power cycles (listed in Tables 2-13 and 2-13a of NEDE-24011-P-A), it believes these inputs to be fatigue analysis appear reasonable. Inasmuch as the maximum fatigue damage factors calculated are significantly below 1.0 and GE has not reported any fuel failures resulting from fatigue, the staff finds the fatigue analysis to be acceptable.

4.2.3.1.3 Fretting Wear

In the GE BWR P8 x 8R fuel design, individual rods in the fuel assembly are held in position 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 are described in Section 2.6.3 of NEDE-24011 along with the results of a continuing fuel surveillance program that has used nondestructive methods including eddy currents to locate discontinuities in the cladding and detailed visual examinations to characterize the nature of defects. As reported in NEDE-

24011, no significant fretting wear has been observed. GE concludes (NEDE-24011) that significant fretting wear is avoided through the use of an active spring force to eliminate any clearance that would otherwise exist between the spacer structure and the fuel rods. Based on the results of these fuel test and surveillance programs, the staff concludes that reasonable assurance has been provided that GESSAR II fuel rods and spacers will perform adequately with respect to fretting wear.

In addition to the fuel rods and spacers, there is another fuel system component whose functionality must be ensured as an objective of the review of BWR fuel system fretting concerns, namely, the fuel assembly channel box. The fuel "channel" that encloses the fuel bundle performs three functions:

1. The channel provides a barrier to separate two parallel flow paths (one to cool the fuel bundle and the other to cool the bypass region between channels).

2. The channel guides the control blade and provides a bearing surface for it.

3. The channel provides rigidity for the fuel bundle.

Thus, the potential for cracks or holes in a "channel" or channel "box" is of concern because it would allow part of the cooling water that normally flows through a fuel bundle to flow out of the cracks or holes and bypass the fuel rods. Such a change in flow pattern would reduce the safety margin for fuel thermal performance and could lead to fuel overheating and damage in the event of some anticipated operating transients and postulated accidents. Significant channel box cracking and wear could also adversely affect mechanical strength.

In the mid-1970s, channel box wear and cracking was observed. The wear was located adjacent to in-core neutron monitor and startup source locations. It was postulated (NEDO-21084), and later confirmed by out-of-reactor testing, that the wear was caused by vibration of nearby in-core tubes resulting from primarily a high-velocity jet of water flowing through the bypass flow holes in the lower core plate. To eliminate significant vibration of instrument and source tubes and the resultant wear on channel box corners, GE will incorporate modifications to GESSAR II similar to those for BWRs currently in operation (described in NEDE-21156). Those modifications involve the elimination of the bypass holes in the lower core plate and addition of two holes in the lower tie plate of each assembly to provide an alternate flow path. This design modification has been determined to have negligible adverse effects on the mechanical, thermal, and nuclear performance of the channel boxes, as is discussed in the staff's generic safety evaluation on this subject (memo from D.G. Eisenhut, Mar. 2, 1976). Because channel box wear has been observed (letter from R.E. Engel (GE), Oct. 15, 1977) to have been significantly reduced in operating BWRs following the design modification, the staff concludes that there is reasonable assurance that channel box wear and cracking will not be a problem in GESSAR II.

4.2.3.1.4 External Corrosion and Crud Buildup

Waterside Corrosion

Corrosion problems associated with stainless steel cladding in BWRs, together with a desire to improve neutron economy, led to a change some years ago from stainless steel to Zircaloy cladding material, which has good resistance to the hot water and steam environment encountered under typical BWR operating conditions (Garzarolli et al., 1978). In several recent cases, however, cladding failures have been associated with external "waterside" corrosion (e.g., see GE Projects Division memorandum, Apr. 6, 1979, and Maury, Nov. 10, 1981). Although the details of the most recent occurrences are still under investigation, the occurrences have been characterized by GE as "crud-

induced local corrosion failure" (NEDE-24343-P). The corrosion is reportedly associated with a variably high copper concentration in the core coolant water and a minor anomaly in the Zircaloy cladding metallurgy (letter from J.S. Charnley (GE), Dec. 11, 1979; letter from R.E. Engel (GE), Oct. 3, 1980; letter from R. L. Smith (W NPC), Nov. 21, 1980; letter from R.L. Smith, February 5, 1981; and letter from L. DelGeorge (Comm. Ed.), Dec. 2, 1980). The source of the copper contamination in the affected plants appears to be the copper-bearing main condenser tubes (letter from L. DelGeorge (Comm. Ed.), Feb. 9, 1981). All the affected plants have copper alloy condenser tubing. It is notable also that virtually all the RWR waterside-corrosion failures have involved gadolinia burnable poison rods.

The NRC staff has been following this issue generically, and in a recent meeting with the staff, GE identified (memo from M. Tokar (NRC), Oct. 13, 1982) what were believed to be some factors responsible for the corrosion failures and stated that a change had been made in the manufacturing process to assure that such failures will not occur in newly manufactured fuel bundles. GE's presentation provided an explanation of the problem, and their solution to it appeared plausible (memo from M. Tokar (NRC), Oct. 13, 1982). Further documentation on this subject has been provided in a recent letter from Washington Public Power Supply System (letter from G.O. Bouchey (WPPSS), Dec. 23, 1982). The staff, therefore, concludes that the issue should be considered resolved generically.

Crud Buildup

The buildup of a corrosion film and a crud layer on the outer surface of a fuel rod during irradiation causes gradual flow reductions and impedes heat transfer to the coolant. The effects of crud buildup on flow are discussed in Section 4.4.5 of this SER. As indicated in Section 2.4.2.2 of NEDE-24011, GE calculates the cladding surface temperature using the cladding surface heat flux at a given axial position of a fuel element in conjunction with an overall cladding-to-coolant film coefficient that is taken to represent the combined effects of crud and oxide resistances and a liquid film resistance based on the Jens-Lottes (Jens and Lottes, May 1981) wall superheat equation. The impact of high cladding temperature, such as decreased yield strength and reduced cladding thickness due to oxidation, was considered in GE's design evaluation (NEDE-24011). GE's methods for analyzing the effects of oxidation and crud on fuel cladding temperatures were reviewed and approved in connection with NEDE-24011 and the NRC staff, therefore, finds that approach acceptable for GESSAR II.

4.2.3.1.5 Dimensional Changes

Fuel Rod Bowinq

GE has stated (NEDE-21660-P) that BWR fuel operating experience, testing, and analysis indicate that there is no significant problem with rod bowing even at small rod-to-rod and rod-to-channel clearances. GE noted that (1) no gross bowing has been observed (excluding the rod bowing-related failures in an early design), (2) a very low frequency of minor bowing has been observed, (3) mechanical analysis indicates deflections within design bases, and (4) thermal-hydraulic testing has shown that small rod-to-rod and rod-to-channel clearances pose no significant problem. The staff has completed its review of a GE generic topical report (NEDE-24289-P) that is intended to update the GE rod bowing experience and to document the overall GE rod bowing safety analyses. The NRC staff's review has concluded, based on GE's submittals and an independent staff assessment which supplemented a contractor's technical evaluation report, that there is no significant rod bowing problem in GE BWR fuel rods (memo from L.S. Rubenstein (NRC), Mar. 8, 1983).

Channel Box Deflection

BWR fuel channels provide structural stiffness for the fuel assemblies and distribute the coolant flow between the assemblies and channel bypass regions. The channels are subject to time-dependent, permanent dimensional changes (i.e., deflections) that result from irradiation, creep, and stress-relaxation effects. The resultant bulge (resulting from long-term creep) or bow (resulting from differential irradiation-induced axial growth) reduces the size of the gap available for control blade insertion. Channel box deflection is thus a phenomenon that can limit channel life because of the potential adverse effects on the ability of the control blades to move freely.

In a generic topical report (NEDE-21354-P), GE describes a channel lifetime prediction method and a backup recommendation for periodic channel deflection measurements that consist of settling friction tests. Upon consideration of the factors involved, the NRC staff concluded that the settling friction tests or an acceptable alternative (such as channel dimensional deflection measurements) should be performed, and in a memorandum (memo from L.S. Rubenstein (NRC), Sept. 18, 1981) the staff outlined a method that could be used to resolve the channel box deflection issue for several near-term BWR operating license applications. Basically, the staff advocated a multistep procedure that had been proposed by the Zimmer applicant. The key ingredient of the Zimmer plan was a commitment to (1) perform some control rod settling friction tests, which would provide an exact profile of control rod drive friction versus position at refueling outages, or (2) make some actual channel dimensional measurements. Several plants agreed to the Zimmer proposal. Subsequently, the BWR Licensing Review Group (LRG II) submitted a position paper (letter from D. C. Holtzscher (Illinois Power Company), May 17, 1982) on channel box deflection that incorporated several of the same features as the Zimmer proposal, although the settling friction test was simplified. As part of the GESSAR II fuels review, GE was asked (memo from C.H. Berlinger (NRC), Oct. 12, 1982) a question (Q 490.12) on channel box deflection, and in separate responses (letter from G. Sherwood (GE), Dec. 21, 1982 and letter from G. Sherwood, Feb. 2, 1983), GE reiterated the LRG II position.

Both the LRG II position statement and the more recent GESSAR Il submittal indicate that the simplified settling friction test provides an equivalent level of test to that described in NEDE-21354-P. To be precise, however, the test is equivalent to a so-called “screening-type” test (described in NEDE-21354-P) to identify any control rod drive where the force required to insert is greater than approximately 250 lb (equal to the settling pressure times the piston area). Whereas in NEDE-21354-P it is indicated that rods failing the screening test would be given another settling friction test to obtain an exact friction-versus-position profile, the latter type of test is not mentioned by LRG II or GESSAR II. The position taken by the LRG II and GESSAR II is that failure of the proposed settling time test would “prompt an investigation,” which, if necessary, would lead to corrective action.

Although the staff believes that a commitment to perform an exact settling friction profile test (or actual dimensional measurement) is preferable (because it would provide an estimate of the margin and physical state of the system in an unambiguous way), the LRG II and GE in NEDE-21354-P have stated that the control rod drives will tolerate a relatively large increase in driveline friction (~350 lb) while still remaining within Technical Specification limits. The screening-type test proposed would, thus, provide assurance of the scram function. Therefore, the staff has accepted (memo from L.S. Rubenstein (NRC), Aug. 19, 1982) the LRG II position that the proposed actions will preclude excessive channel bowing in the LRG II plants (i.e., River Bend and Perry), and on the same basis, staff accepts GESSAR II’s endorsement of that position.

Some very recently available data have shed some new light on the deflection phenomenon. The new data (from Commonwealth Edison measurements) are described in a recent report (EPRI NP-2483) on a rather exhaustive study funded by the Electric Power Research Institute (EPRI). As indicated in

Section 4 of the EPRI report, the Commonwealth Edison data indicate that prime candidates for channel bowing are manufactured from mismatched halves, that is, channels manufactured from two pieces of stock material not from the same original batch. The GE fabrication process does not preclude use of unmatched halves. In response (letter from G. Sherwood (GE), Feb. 2, 1983) to the NRC staff question (Q 490.04) on channel deflection, however, GE stated that an approach to “mitigate the effects of” (i.e., reduce) channel bowing has been recommended to utilities.

4.2.3.1.6 Fuel and Poison Rod Pressure

Because the staff has not yet agreed that the design limit proposed by GE for rod pressure is acceptable (see 4.2.1.1.6 of this SER), it cannot comment on the acceptability of the fuel design to meet the appropriate design limit. The staff will report on resolution of this issue in a supplement to this SER.

4.2.3.1.7 Fuel Assembly Liftoff

The potential for BWR fuel assembly liftoff is addressed in a GE letter (letter from R.L. Gridley (GE), July 11, 1977), which covers accident conditions as well as normal operation. Section 4.2.II.A.1.g of the SRP deals with normal operation only, however, and so in this SER normal operation liftoff is separated from the combined seismic-and-LOCA liftoff concern, which is addressed in Section 4.2.3.3.4 of this SER.

The Gridley July 1977 letter and the response (letter from G. Sherwood (GE), Feb. 2, 1983) to a staff question indicate that for loads resulting from normal operation, including AOOs, GE does not calculate any bundle separation from the core support piece. The staff concludes, therefore, that there is reasonable assurance that fuel liftoff will not occur under such conditions.

4.2.3.1.8 Control Material Leaching

The loss of boron carbide (B_4C) by leaching from cracked control blade tubing is addressed in NRC IE Bulletin 79-26, Revision 1, which requires operating BWRs to perform various actions including, but not limited to, shutdown-margin tests. Subsequent to the issuance of IE Bulletin 79-26, GE performed analyses and postirradiation examinations (NEDE-24226-P) and developed a boron depletion model (NEDE-24325-P). This model supports GE’s claim that the amount of boron loss can be accurately determined analytically and that potential control blade degradation resulting from this mechanism will not significantly affect plant operation. Further generic review of this matter may thus lead to a suspension or revision of IE Bulletin 79-26 requirements. In this connection, National Nuclear Corporation has recently reported (EPRI NP-2730) the development, under an EPRI-funded program, of a nondestructive method to determine the loss of B_4C powder from BWR control blades. Inasmuch as costly hot-cell examinations have heretofore been the only procedure available to monitor the performance of BWR control blades, the new nondestructive technique may render moot the need for adherence to analytical calculation methods and limits. Because, until new mitigation approaches are adopted, the GESSAR II plants will be subject to the IE Bulletin when a plant starts operation, assurance will be maintained that control blade reactivity will not be significantly degraded by B_4C leaching.

4.2.3.2 Fuel Rod Failure Evaluation

The following paragraphs discuss the staff’s evaluation of (1) the ability of the GESSAR II fuel to operate without failure during normal operation and anticipated transients and (2) the accounting for

fuel rod failures in the applicant's accident analysis. The fuel rod failure criteria described in Section 4.2.1.2 of this SER were used for this evaluation.

4.2.3.2.1 Internal Hydriding

GE uses a hot vacuum outgassing (drying) method for removal of the moisture contamination of loaded fuel rods to ensure that the level of moisture is well below the limits discussed in Section 4.2.1.2.1 of this SER. And, to prevent the introduction of other hydrogenous impurities such as oils and plastics, various procedural controls are used during manufacturing (NEDE-24011). If even in the face of those manufacturing controls, moisture or hydrogenous impurities are still inadvertently introduced into a fuel rod during manufacture, further assurance against chemical attack is provided by the use of a hydrogen getter material placed in the upper plenum of the fuel rod. The incorporation of a hydrogen getter in GE fuel rods is described in NEDO-20922. GE calculations, based on an assumed initial rod moisture content equal to the specification limit, indicate that no cladding damage will occur during the active hydrogen/moisture gettering period (NEDE-24011-P-A). Field experience (NEDE-24343-P) has confirmed that hydriding is not an active failure mechanism for fuel manufactured since mid-1972. On the basis of the information presented, we conclude that adequate assurance has been provided that fuel and burnable poison rod hydriding will not be a problem at GESSAR II.

4.2.3.2.2 Cladding Collapse

As stated in Section 2.5.3.1.1 of NEDE-24011-P-A, GE has performed a cladding (long-term creep) collapse analysis. Based on the results of that analysis, which was performed with an approved model (NEDE-20606-P-A), cladding collapse was not calculated to occur for either 8x8 or 8x8R fuel rods. P8x8R (prepressurized) fuel will have an even greater margin to collapse. The NRC staff, therefore, concludes that there is reasonable assurance that long-term creep collapse will not occur in GESSAR II during normal operation.

4.2.3.2.3 Overheating of Cladding

The methods used to show that the minimum critical power ratio (MCPR) design basis will be met are reviewed as part of a thermal-hydraulics review in Section 4.4 of this SER. Analyses that show that the MCPR design basis is met for normal operation and anticipated transients are discussed in Chapter 15 of the GESSAR II FSAR. Other sections of FSAR Chapter 15 discuss accidents where the MCPR criterion is used to define failure in the accident analyses. All of the MCPR-related analyses are discussed in Section 15 of this SER.

4.2.3.2.4 Overheating of Fuel Pellets

Fuel melting temperature is discussed in Section 2.4.2.5 of NEDE-24011 as a function of exposure (burnup) and gadolinia content (of burnable poison rods). GE states in that report that fuel melting is not expected to occur during normal operation, and that statement is based on fuel temperature calculations performed with a model described in the proprietary supplement to Amendment 14 of GESSAR II. Although limited melting during certain events such as an uncontrolled control rod withdrawal is permissible, such melting is not predicted to occur.

The staff has reviewed the UO2 properties (thermal conductivity and melting point) that are important in reaching this conclusion and agree that UO_2 melting will not be a problem for GESSAR II fuel during normal operation and anticipated transients. In addition, GE submitted the topical report, NEDE-20943-P, which contained gadolinia fuel property information (including the effects of

gadolinia concentration on thermal conductivity and melting temperature). Midway through the staff review, GE withdrew this report. On a revised schedule the staff proceeded to review and approve selected GE gadolinia fuel properties as submitted in an appendix to the GESTR-LOCA report (Appendix B to NEDE-23758-1-P).

However, the thermal conductivity equation used in the GESSAR II fuel centerline melting analysis described in NEDE-24011 (results of analysis contained in Table 2-4) and Amendment 14 to GESSAR II was not the same equation submitted and approved in Appendix B to NEDE-23758-1-P. GESSAR II references NEDE-20943-P (which was withdrawn), which provided a different thermal conductivity equation. The staff requires that GE revise GESSAR II and its reference documents to reflect the use of approved gadolinia fuel properties and to eliminate reference to the withdrawn report. This confirmatory issue will be addressed in a supplement to this SER.

4.2.3.2.5 Excessive Fuel Enthalpy

Large fuel enthalpies occur only in the postulated control rod drop accident. GE's analysis of the control rod drop accident is described in Section 5.5.1 of NEDE-24011 and in NEDO-10527. The analysis of this event is reviewed in Section 15.4.9 of this SER.

4.2.3.2.6 Pellet/Cladding Interaction

The two PCI criteria (limited fuel melting and 1% cladding plastic strain) in current use in licensing of BWR fuel are easily satisfied for normal operation and anticipated transients. Fuel melting is addressed in Section 4.2.3.2.4 and, except for uncertainties with regard to gadolinia fuel, the fuel-melting criterion is met. The model used by GE in the evaluation of the 1% plastic strain limit is described in detail in the proprietary supplement to Amendment 4 of GESSAR II. As stated in NEDE-24011, dimensions used in conjunction with the model for that evaluation are the most limiting combination of tolerances. These results show that the GESSAR II fuel meets the 1% strain limit. The NRC staff has not audited these results because the traditional 1% strain limit is not believed to be a very effective limit.

Based on results 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 (NEDS-10456-PC). PCIOMRs have generally been effective in reducing PCI failures that result from operational power changes, but they would not prevent PCI failures during unexpected transients and accidents.

The staff is continuing to assess the potential for PCI failure during power-increasing transients and accidents, and new techniques are being developed to analyze the potential for PCI failures (NUREG/CR-1163) and determine whether or not further staff actions are required.

In conclusion, (1) the applicant has met NRC's licensing criteria for normal operation and anticipated transients, (2) the applicant will impose operating restrictions to reduce the potential for PCI, and (3) the NRC staff is studying the need for new licensing requirements in this area. There are currently no other PCI licensing requirements that must be met for GESSAR II.

4.2.3.2.7 Cladding Rupture (Bursting)

The staff has been generically evaluating three fuel materials models that are used in the emergency core cooling system (ECCS) analysis. Those models predict cladding rupture temperature, cladding burst strain (ballooning), and fuel assembly flow blockage (used only in PWR analyses). The staff has

(1) discussed its evaluation with vendors and other industry representatives (memo from R.P. Denise (NRC), Nov. 20, 1979), (2) published NUREG-0630, "Cladding Swelling and Rupture Models for LOCA Analysis," and (3) required licensees to confirm that their operating reactors would continue to be in conformance with 10CFR50.46 if the NUREG-0630 models were substituted for the present materials models in their ECCS evaluations and certain other compensatory model changes were allowed (letter from D.G. Eisenhut (NRC), Nov. 9, 1979, and memo from H.R. Denton (NRC), Nov. 26, 1979). This latter requirement was made a condition of approval to the GE ECCS evaluation model (letter from R.L. Tedesco (NRC), Feb. 4, 1981).

A generic sensitivity study of fuel rod cladding ballooning and rupture phenomena during a LOCA was submitted (letter from R.H. Buchholz (GE), May 15, 1981) in response to this condition of approval. As reported in the generic study, GE assessed the BWR ECCS sensitivity to rupture temperature by using three rupture temperature models: (1) the GE CHASTE model, (2) the NUREG-0630 model, and (3) a proposed GE model termed the adjusted model. Later submittals provided clarification and additional information. The staff has completed (memo from L.S. Rubenstein (NRC), Apr. 5, 1982) its generic review of this issue. The resolution of this issue is described in Section 4.2.3.3.3 of this SER.

4.2.3.2.8 Fuel Rod Mechanical Fracturing

The mechanical fracturing analysis is usually done as a part of the seismic-and-LOCA loads analysis (see Section 4.2.3.3.4 of this SER). A report containing an analysis has been recently submitted by GE (NEDE-21175-3) and is currently under review by staff consultants at the Idaho National Engineering Laboratory. The staff will report on this issue in a supplement to this SER.

4.2.3.3 Fuel Coolability Evaluation

The following paragraphs discuss the staff's evaluation of the ability of the GESSAR II fuel to meet the fuel coolability criteria described in Section 4.2.1.3. Those criteria apply to postulated accidents.

4.2.3.3.1 Fragmentation of Embrittled Cladding

The primary degrading effect of cladding oxidation is embrittlement. Such embrittled cladding will have reduced ductility and resistance to fragmentation. The most severe manifestation of such embrittlement occurs during the postulated loss-of-coolant accident (LOCA). The overall effects of a LOCA on peak cladding temperature, oxidation, and embrittlement are generically analyzed in NEDE-20566 and are reviewed in Section 15.6.3 of this SER.

One of the more significant analytical methods that is used to provide input to the analysis in Section 15.6.5 of this SER is the steady-state fuel performance code, which is reviewed in Section 4.2. This code provides fuel pellet temperatures (stored energy) and fuel rod internal pressures for the ECCS evaluation model as prescribed by Appendix K to 10CFR50.46. Initial conditions for the LOCA have been calculated for the GESSAR II safety analysis with an approved version of the GE fuel performance code, GEGAP-III (NEDO-20181). This code incorporates the effects of cladding creepdown as well as fuel densification, swelling and relocation in the calculation of fuel temperatures and fission gas release.

In 1976, the NRC staff questioned (letter from D.E. Ross (NRC), Nov. 23, 1976) the validity of fission gas release calculations in most fuel performance codes including GEGAP-III for local burnups greater than 20 MWd/kg U. GE was informed of this concern and was provided with a method (NUREG-0418) of correcting fission gas release calculations for burnups greater than 20 MWd/kg U.

Although a reanalysis was not performed specifically for the GESSAR II design, the GE reanalysis (letter from G. Sherwood (GE), Dec. 22, 1976) included early reflooding plants and reportedly bounds the GESSAR II case. Although fuel rod internal pressures were shown to remain below system pressure for planar average exposures below 44 MWd/kg U, the generic reanalysis did result in higher initial stored energy and rupture pressure in the LOCA analysis. Under LOCA conditions, the higher fission gas release resulted in an increase of 85°F in calculated peak cladding temperature (PCT) at 33 MWd/kg U planar average exposure. This increase in PCT has not been explicitly accounted for in the GESSAR II safety analysis. However, GE requested (letter from M. Pifferetti (NRC), Aug. 21, 1981) that credit for calculated PCT margin as well as credit for recently approved but unapp1ied ECCS evaluation model changes be used to offset any operating penalties resulting from high burnup fission gas release. This proposal was found acceptable (memo from C.S. Rubenstein (NRC), Oct. 22, 1981) for planar average burnups as high as 50 MWd/kg U provided that the generic analysis was found applicable to the plant and that no additional credit was taken for the ECCS evaluation model changes. In response to this concern, the applicant has stated (FSAR Amendment 12) that the GE proposal does apply to GESSAR II. The staff concludes that the effects of enhanced fission gas release at high burnup have been adequately considered in the GESSAR II safety analysis with regard to the potential fragmentation of embrittled cladding.

4.2.3.3.2 Violent Expulsion of Fuel

The analysis, which demonstrates that the acceptance criteria are met, is evaluated in Section 15.4.9 of this SER.

4.2.3.3.3 Cladding Ballooning

As discussed previously in Section 4.2.3.2.7 of this SER, GE has submitted a generic sensitivity study (letter from R.H. Buchholz (GE), May 15, 1981) in response to a condition in the approval of the GE ECCS evaluation model. The staff has completed its generic review of this issue (memo from L.S. Rubenstein (NRC), Apr. 5, 1982). The staff review concluded the following:

1. The proposed "adjusted" cladding rupture temperature model is acceptable because it does not underestimate the incidence of rupture based on applicable data including those reported in NUREG-0630 and more recent data.

2. The existing cladding models for prerupture strain, postrupture strain, and strain used for oxidation calculations are acceptable because GE has shown that the calculated values of peak cladding temperature and percent oxidation are sufficiently insensitive to variations in cladding strain.

These conclusions generically resolve the issues related to NUREG-0630 and remove the condition in the staff approval of the ECCS evaluation model. Because these cladding models will be used in the plant-specific GESSAR II ECCS analyses, the issue of cladding ballooning and rupture is resolved.

4.2.3.3.4 Fuel Assembly Structural Damage From External Forces

An analysis has been provided by GE (NEDE-21175-3) to show that the GESSAR II fuel meets structural requirements (including liftoff) similar to those of Appendix A of Section 4.2 of the SRP (NUREG-0800). That analysis is currently under review. Because previous generic analytical methods presented in earlier versions of NEDE-21175 have been approved by the NRC staff (letter from O.P. Parr (NRC), May 17, 1979) and because favorable sample results were also presented in

Amendment 2 of NEDE-21175, the staff considers this issue to be confirmatory for GESSAR II. It will report on this issue in a supplement to this SER.

4.2.4 Testing, Inspection, and Surveillance Plans

4.2.4.1 Testing and Inspection

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

The GE generic reload report (NEDE-24011-P-A) and the BWR 4/5 topical report (NEDE-20944) contain descriptions of (1) the type of quality control inspections performed during manufacturing and (2) the plan for onsite inspection and testing of new fuel assemblies. The staff concludes, on the basis of the information provided in these referenced reports, that the new-fuel testing and inspection program for the GESSAR II fuel design is acceptable.

4.2.4.2 On-Line Fuel System Monitoring

BWRs such as GESSAR II have two independent radiation detection systems that are directly capable of sensing fission product releases from failed fuel rods. The main steam line radiation (MSLR) monitors are used to detect high radiation levels resulting from gross fuel failure. In GESSAR II four redundant MSLR monitors will be used, each of which provides a signal to trip the turbine and initiate reactor scram independent of operator action. Because the MSLR monitors are located relatively close to the reactor core, they are capable of sensing gross fission product releases relatively quickly (i.e., in a few seconds) while the off-gas system radiation (OGSR) monitors are capable of detecting low-level emissions of noble gases, which could indicate the occurrence of minor fuel damage, in 2 to 3 min, the time required for the activity to travel from the core to the detectors. The OGSR monitors are set to sound alarms that would initiate operator action (see Chapter 11 of the GESSAR II FSAR for details). The NRC staff has reviewed the BWR activity monitors (NUREG-0401) and concludes that the combination of (1) both radiation detection systems and (2) implementation of applicable technical specifications on limiting safety system instrumentation setpoints and specific activity (Technical Specification Section 2.2 and Technical Specification 3/4.4.5, respectively) provide reasonable assurance of the adequacy of the on-line fuel system monitoring for GESSAR II.

4.2.4.3 Postirradiation Surveillance

As indicated in SRP Section 4.2.II.D.3, a routine fuel inspection program to provide information on irradiated and discharged fuel should be provided. A typical program involves visual examination of selected assemblies (commonly 5 to 10% of the discharged fuel), concentrating on the lead bundles. Visual examinations normally include, but are not necessarily limited to, crud buildup, rod bowing, and missing components. Additional inspections should be performed depending on the results of operational monitoring including coolant activity and the visual inspections. A commitment to provide a minimal postirradiation surveillance program, consistent with the above guidelines, before reactor startup will be made a license condition for GESSAR II that will be agreed to during license review.

4.2.5 Mechanical Design Evaluation Findings

The GESSAR II fuel system design has been reviewed in accordance with SRP Section 4.2. The staff concludes that although most of the objectives of the fuel system safety review have been met, two confirmatory issues and one open issue have to be resolved before the review can be completed. The two confirmatory issues are (1) fuel assembly seismic-and-LOCA loads evaluation (SER Sections 4.2.1.2.8, 4.2.3.2.8, 4.2.1.3.4, and 4 2.3.3.4) and (2) overheating of gadolinia fuel pellets (SER Section 4.2.3.2.4). The open issue is fuel rod internal pressure (SER Sections 4.2.1.1.6 and 4.2.3.1.6).

When the confirmatory and open issues are resolved, the staff will be able to conclude that the GESSAR II fuel system has been designed so that (1) it will not be damaged 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, thereby meeting the related requirements of the following regulations: 10CFR50.46; GDC 10, 27, and 35; and 10CFR50, Appendix K. This conclusion is based on two primary factors:

(1) With the exception of the remaining confirmatory and open issues, the applicant has provided sufficient evidence that the design objectives will be met based on operating experience, prototype testing, and analytical predictions. Those analytical predictions dealing with control rod drop and fuel densification have been performed in accordance with RG 1.77 and methods that the staff has reviewed and found to be acceptable alternatives to RG 1.126.

(2) The applicant has provided for testing and inspection of new fuel to ensure that it is within design tolerances at the time of core loading. On-line fuel failure detection equipment will be in place to provide warning of cladding perforations during plant operation. Licensees using the GESSAR II fuel will be required to make a commitment to perform postirradiation surveillance to detect anomalies or confirm that the fuel has performed as expected.

The staff concludes that GE has described methods of adequately predicting fuel rod failures during postulated accidents so that radioactivity releases are not underestimated, thereby meeting the related requirements of 10CFR100. In meeting those requirements, GE has (1) used the fission-product release assumptions of RGs 1.3 and 1.25, and (2) performed the analysis for fuel rod failures for the rod drop accident in accordance with the guidelines of RG 1.77.

On the basis of this review, the staff concludes that, with the above exceptions, all the requirements of the applicable regulations have been met and guidelines of the applicable regulatory guides and current regulatory positions have been followed for the GESSAR II fuel design.

NRC
Safety Evaluation Report
Approving Amendment 7
to
NEDE-24011-P

UNITED STATES
NUCLEAR REGULATORY COMMISSION
Washington, D. C. 20555
March 1, 1985

MFN 036-85

Ms. J.S. Charnley, Manager
Nuclear Fuel Licensing
General Electric Company
175 Curtner Avenue
San Jose, California 92125

Dear Ms. Charnley:

SUBJECT: Acceptance For Referencing of Licensing Topical Report NEDE-24011-P Amendment 7 To Revision 6, "General Electric Standard Application For Reactor Fuel"

We have completed our review of the subject topical report submitted by General Electric Company (GE). We find the report to be acceptable for referencing in license applications to the extent specified and under the limitations delineated in the report and the associated NRC evaluation, which is enclosed. The evaluation defines the basis for acceptance of the report.

We do not intend to repeat our review of the matters described in the report and found acceptable when the report appears as a reference in license applications, except to assure that the material presented is applicable to the specific plant involved. Our acceptance applies only to the matters described in the report.

In accordance with procedures established in NUREG-0390, it is requested that GE publish accpeted versions of this report, proprietary and non-proprietary, within three months of receipt of this letter. The accepted versions shall incorporate this letter and the enclosed evaluation between the title page and the abstract. The accepted versions shall include an -A (designating accepted) following the report identification symbol.

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

Sincerely,

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

Enclosure:
As stated

Enclosure

Safety Evaluation Report On Amendment 7 to GESTAR II

1.0 INTRODUCTION

With the submittal of Amendment 7 to the licensing topical report NEDE-24011-P, General Electric intends to upgrade this document to the format of Standard Review Plan Section 4.2. Although there are some variations from the SRP Section 4.2 design criteria and limits, most of the GE objectives are achieved and open questions are resolved by this amendment. It is expected that our future reviews involving GE fuel designs can be streamlined and expedited if GE references this document.

General Electric uses the fuel/thermal-hydraulic performance code, GESTR-MECHANICAL (MFN 170-84-0), to demonstrate conformance with some of the fuel performance/design criteria listed in the SRP Section 4.2. The GESTR-MECHANICAL code is derived from the GE computer code GESTR-LOCA (NEDE-23785-1-P), which was previously approved by the staff (Rubenstein, September 26, 1983). Rather than performing fuel rod bounding analysis for a LOCA, GESTR-MECHANICAL is intended as a best-estimate and realistic fuel rod design code. A discussion of GESTR-MECHANICAL is presented in Section 8.0 of this evaluation. GE has proposed an implementation schedule for GESTR-MECHANICAL and Amendment 7, which also can be found in Section 8.0.

The latest approved version of GESTAR II is NEDE-24011-P-A-6, which applied to the 8x8, 8x8R, P8x8R, and BP8x8R fuel designs. With the approval of Amendment 7, GE intends to apply GESTAR II Amendment 7 to P8x8R, BP8x8R, and a new fuel design, 8x8E (Charnley, November 10, 1984) which has not yet been approved by the staff. Therefore, our evaluation of the acceptability of the proposed criteria of Amendment 7 discussed in this SER applies only to P8x8R and BP8x8R fuel.

The equivalent portion of Section 2 of NEDE-24011-P-A-6 will become Appendix C to GESTAR II, and will apply to the old fuel designs of 8x8 and 8x8R only. Since the 8x8 and 8x8R fuel designs are not currently in production, we were informed by GE of the possibility of eventually removing Appendix C from GESTAR II in the future. The material properties for the P8x8R and BP8x8R designs will be given in the special report on GESTAR-MECHANICAL (MFN 170-84-0) and partially in GESTAR II Amendment 7. This is discussed in more detail in Section 8.0 of this SER.

2.0 FUEL SYSTEM DESIGN

The objectives of this fuel system safety review as described in Section 4.2 of the SRP are to provide assurance that (a) the fuel system is not damaged as a result of normal operation and anticipated operational occurrences, (b) fuel system damage is never so severe as to prevent control rod insertion when it is required, (c) the number of fuel rod failures is not underestimated for postulated accidents, and (d) coolability is always maintained. A “not damaged” fuel system is defined as meaning that fuel rods do not fail, that fuel system dimensions remain within operational tolerances, and that functional capabilities are not reduced below those assumed in the safety analysis. Objective (a) above implements General Design Criterion (GDC) 10 (10CFR50, Appendix A), and the design limits that accomplish this are called Specified Acceptable Fuel Design Limits (SAFDLs). “Fuel rod failure” means that the fuel rod leaks and that the first fission product barrier (the cladding) has, therefore,

been breached. Fuel rod failures must be accounted for in the dose analyses 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 flow paths to permit removal of residual heat even after a severe accident. The general requirements to maintain control rod insertability and core coolability appear repeatedly in the General Design Criteria (e.g., GDC 27 and 35). Specific coolability requirements for the loss-of-coolant accident are given in Section 50.46 of the Code of Federal Regulations.

To assure that the above stated objectives are met, the following areas are examined: (a) design bases, (b) description and design drawings, (c) design evaluation, and (d) testing, inspection, and surveillance plans. In assessing the adequacy of the design, operating experience, prototype testing, and analytical predictions are compared with acceptance criteria for fuel system damage, fuel rod failure, and fuel coolability.

3.0 DESIGN BASES

Design bases for the safety analysis address fuel system damage mechanisms and suggest limiting values for important parameters such that damage will be limited to acceptable levels. For convenience, the acceptance criteria for these design limits are grouped into three categories in the SRP: (a) fuel system damage criteria, which are most applicable to normal operation, including anticipated operational occurrences (AOOs), (b) fuel rod failure criteria, which apply to normal operation, AOOs, and accidents, and (c) fuel coolability criteria, which apply to accidents. For each of these three categories (or levels) of fuel damage, damage mechanisms that are believed to be pertinent to that degree of damage are listed in the SRP, and a design basis statement, indicating the applicant’s design objective relative to the damage mechanism, is called for along with a related design limit—often a numerical value of a design parameter (e.g., l-percent cladding plastic strain).

With regard to conditions of normal operation, including AOOs, the relationship of the design bases and limits of GDC-10’s requirement that specified acceptable fuel design limits (SAFDLs) not be exceeded is of crucial importance. Traditionally, we have interpreted GDC 10 to mean that fuel should be designed to preclude (all but a few stochastic) fuel failures during normal operation, including AOOs. It will be demonstrated in the sections to follow that we continue to hold the fuel designs discussed in GESTAR II to that standard.

3.1 Fuel System Damage Criteria

The following paragraphs discuss the NRC staff's evaluation of the design bases and corresponding design limits proposed by GE for the damage mechanisms listed in the SRP. These design limits along with certain criteria that define failure constitute the SAFDLs required by GDC 10. The design limits in this section should not be exceeded during normal operation including AOOs.

1. Stress and Strain

   In keeping with the GDC 10, 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.

To satisfy the design basis in the past, GE used the strength theory, terminology, and stress categories presented in the ASME Code, Section III, as a guide for determining fuel rod cladding stress limits.

However, in the GESTAR Amendment 7 submittal, GE employs a concept of a design ratio, which is defined as a ratio of effective stress or strain to a stress or strain limit. The stress and strain limits refer to ultimate tensile stress and the corresponding strain. The effective stress and stain are calculated using von Mises's criterion. The design ratio is limited to be less than or equal to unity for design purposes.

This design ratio is derived from ANSI/ANS-57.5-1981, which has some variations from the acceptable ASME Code Section III. For example, while ANSI/ANS-57.5-1981 uses a full ultimate tensile stress, the ASME Code Section III calls for only 70% of the same quantity.

GE has demonstrated, in response to our questions (Charnley, April 23, 1984) during the review of Amendment 7, that a conservative approach has been developed in calculating the design ratios. GE performs a Monte Carlo statistical analysis which results in two design ratio distributions for stress and strain. In order to satisfy the design criterion, GE requires that the upper 95 percentile of both distributions be less than unity.

We consider that the GE approach to this design criterion is an acceptable alternative to the ASME Section III approach specified in the SRP because appropriate conservatisms are incorporated into the analysis including the use of an upper 95% tolerance limit of the design ratio distributions and a bounding power history. Therefore, we conclude that the GE design ratio criterion is acceptable.

2. Strain Fatigue

GE's design basis for strain fatigue is that the fuel assembly and the fuel rod cladding are evaluated to ensure that failure due to cyclic loadings will not exceed the fatigue capability. GE uses a fatigue design limit called fatigue usage, which is defined as a ratio of actual number of cycles at stress or strain to allowable cycles at stress or strain. This ratio must be less than unity.

In the past, we have approved the use of fatigue usage as a viable alternative to the SRP approach because the associated information, for example, Figures 2-11a and 2-11b in NEDE-24011-P-A-5, demonstrated that adequate conservatism existed, i.e., a safety factor of 2 on stress amplitude or a safety factor of 20 on the number of cycles. GE has provided an identical curve to Figure 2-11b in Figure 4-2 of the Amendment 7 submittal. We, therefore, conclude that the GE fatigue usage design criterion is acceptable.

3. Fretting Wear

Although the SRP does not provide numerical bounding-value acceptance criteria for fretting wear, it does stipulate that the allowable fretting wear, should be stated in the safety analysis and that the stress and fatigue limits should presume the existence of the wear. GE's design basis is that the fuel assembly is evaluated to ensure that the fuel will not fail due to fretting wear of the assembly components. 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.

Since the SRP does not provide numerical acceptance criteria for fretting wear, and since fretting wear is addressed in the design analysis for each bundle design, the NRC staff concludes that the intent of the SRP has been adequately met.

4. External Corrosion and Crud Buildup

With respect to external corrosion and crud buildup, GE's design basis is that the fuel rod is evaluated to ensure that the cladding temperature increase and cladding metal thinning due to cladding oxidation and the cladding temperature increase due to the buildup of corrosion products do not result in fuel rod failure due to reduced cladding strength. GE does not specify a maximum cladding external temperature to limit corrosion or an external corrosion or crud layer thickness that would significantly affect design margins. However, the SRP does not provide numerical limits for cladding temperature or degree of oxidation for normal operation. Inasmuch as GE has demonstrated that corrosion and crud buildup are accounted for in the design analysis, we conclude that GE's approach is consistent with the SRP guidelines.

5. Dimensional Changes

Fuel assembly components such as the fuel rods and channel boxes may undergo various types of dimensional changes such as fuel rod bowing, axial growth, and channel box deflection. Such phenomena are related to neutron fluence, fuel burnup, and assembly core residence time and must be accounted for in the fuel design analyses to establish operational tolerances and to assure that all effects are accommodated and the thermal and mechanical design.

Rod Bowing

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 our SER dated March 8, 1983, we have approved the GE topical report NEDE-24284-P. "Assessment of Fuel-Rod Bowing in General Electric Boiling Water Reactors." This topical report describes GE's position on fuel rod bowing, which is that there is no effect on boiling transition due to rod bowing. However, the maximum gap closure due to fuel rod bowing reported by GE was approximately 50%, and while we can agree with the GE conclusion for bowing of this magnitude, it is not clear that a greater amount of bowing will not result in a reduction in the critical power ratio.

Therefore, while the present basis for rod bowing of GE fuel is acceptable, we cannot agree with the GE criterion that rod bowing, even to contact, will have no effect on the critical power ratio.

A further discussion of the details of the thermal hydraulic tests which lead to this conclusion and recommendations for resolving this issue are given in Section 5.1(5).

Channel Box Deflection

With regard to channel box deflection, no design basis is provided in the Amendment 7 submittal. However, GE has issued a generic report on channel design and deflection

(NEDE-21354-P) and has responded to NRC questions in two supplements to this report. The GE report documents (a) the fuel channel description, (b) the design bases and analyses, and (c) the creep deflection phenomenon. In the design section of the generic report, the design basis for channel structural adequacy is discussed in terms of material properties such as yield strength, stress-rupture strength, and operational duty including normal maneuvers, transients, and accidents. The discussion meets the guidance of the SRP with regard to design bases and is, therefore, acceptable in that respect

Irradiation Axial Growth

GE does not provide a design basis or limit for fuel rod irradiation axial growth in the Amendment 7 submittal. However, during discussions with GE in the course of this review, GE stated that irradiation growth is considered in the fuel design; an expansion spring is located between the upper end plug shank of each fuel rod and the upper tieplate. The function of this spring is twofold: (1) it keeps the rod seated in the lower tieplate, and (2) it allows independent axial expansion by sliding within the holes of the upper tieplate. Although there is an irradiation growth correlation in the GESTR-MECHANICAL code (MFN 170-84-0), GE informed us that the correlation was not intended for use in axial shoulder gap design; it is merely used in the fuel rod thermal-mechanical analysis to reflect an overall response during irradiation. SRP Section 4.2 states that limits need not be provided for dimensional changes such as irradiation growth. However, irradiation growth must be considered in the fuel design. We, therefore, conclude that the GE method of considering irradiation axial growth is consistent with the Standard Review Plan and is therefore acceptable.

6. Fuel and Poison Rod Pressures

The SRP (NUREG-0800) identifies fuel and burnable poison rod pressure as a potential fuel system damage mechanism separate from that identified under the cladding stress criterion. This is because the criterion for fuel rod internal pressure involves more than the cladding mechanical limits. There are a number of reasons for this distinction: (a) outward (tensile) cladding stresses may force analytical methods into a mode for which they were not designed or where greater uncertainties exist, (b) the higher releases of fission gas associated with higher rod pressures may lead to a level of undesirable positive thermal feedback, (c) the higher releases of fission gas associated with higher rod pressures may lead to underestimating the radiological consequences of accidents using release assumptions of Regulatory Guides 1.25 and 1.77, and (d) net positive pressures would lead to ballooning during non-LOCA DNB events and such behavior is not analyzed in the accident analysis.

Although NUREG-0800 proposes the nominal coolant system pressure as an acceptable limit for fuel rod internal gas pressure, alternative criteria may also be considered if justified. In general, such alternatives must show that fuel rod internal pressures in excess of system pressure do not (a) lead to fuel system damage during normal operation and AOOs, (b) prevent control rod insertion when required, (c) result in an underestimate of the number of fuel failures in, or radiological consequences of, postulated accidents, or (d) lead to loss of coolable geometry.

GE describes a design basis in the GESTAR Amendment 7 submittal for rod pressure in which the effects of fuel rod internal pressure during normal steady-state operation will not result in fuel failure due to excessive cladding pressure loading. GE contends that a rod internal pressure limit of less than or equal to the RCS pressure is not necessary. Instead, GE

proposes that the rod pressure be limited so that the instantaneous cladding creepout rate due to internal pressure greater than RCS pressure is not expected to exceed the instantaneous fuel swelling rate.

To demonstrate that this proposed criterion is acceptable in terms of items (a) through (d) above, GE demonstrates that for the design basis transients and accidents of interest in a BWR, either the cladding does not heat up significantly or the existing fuel damage criteria used are still applicable when the initial fuel rod internal pressure exceeds the initial RCS pressure.

In the case where the cladding does not heat up significantly, that is the safety limit MCPR is not exceeded, there is no significant change in the fuel rod geometry so that control blade insertion and bundle coolability will be maintained.

For those events in which the cladding does heat up significantly above its normal temperature, GE has demonstrated that there are other criteria which assure that conditions (a) through (d) will not occur. For example, the LOCA event is governed by the criteria set forth in 10CFR50.46 that the cladding temperature will not exceed 2200°F, the maximum amount of local oxidation on any fuel rod will not exceed 17% and that a coolable geometry will be maintained. These criteria are independent of the initial internal pressure of the fuel rod. However, the internal pressure of the fuel rod is taken into account explicitly in determining the stored energy and in calculating the amount of fuel rod swelling and rupturing. In addition, the number of failed fuel rods assumed for radiological calculations is 100% of those in the core. Therefore, a rod internal pressure greater than the RCS pressure will not result in under-estimating the radiological consequences of a LOCA. Therefore, a fuel rod internal pressure greater than RCS pressure is acceptable for LOCA.

Similarly GE has evaluated the rod drop accident and has demonstrated, in response to a staff question, that the criterion for fuel failure in a rod drop accident is still applicable (Charnley, April 23, 1984).

We have reviewed all the transients and accidents postulated for a BWR and conclude that there is no case in which a fuel rod internal pressure greater than the RCS pressure results in an existing SAFDL becoming invalid.

Therefore the GE design criterion for rod internal pressure is acceptable.

7. Fuel Assembly Liftoff

The SRP calls for the fuel assembly holddown capability (gravity and springs) to exceed worst-case hydraulic loads for normal operation, which includes AOOs. The GE design basis for assembly liftoff is that “the fuel assembly is evaluated to ensure that interference sufficient to prevent control blade insertion will not occur.”

GE 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. The approved report NEDE-21175-3-P describes a liftoff limit to prevent such interference with control blade insertion. GESTAR Amendment 7 references this topical report. We

therefore conclude that the fuel assembly liftoff design criterion has been adequately addressed in GESTAR II.

8. Control Material Leaching

   The SRP states that control reactivity must be maintained. Control rod reactivity can sometimes be lost by leaching of certain poison materials if the control rod cladding is breached, as indeed has been observed (Eisenhut, October 22, 1979). GE has postulated (NEDE-24226-P) the breaching by cracking is caused by stress corrosion resulting from solidification (sintering) of the boron carbide (B_4C) particles in the rods followed by swelling (due to helium and lithium) of the sintered B_4C. The stress in the tubes caused by the B_4C is believed to accelerate the intergranular corrosion that proceeds from the outside surface of the rod cladding.

   Prior to the discovery of the B_4C leaching loss mechanism, the previously defined life-limiting parameter for the control blades was loss of boron by depletion. The current generic criterion defining end of control blade life is a loss of total reactivity equal to 10 percent of initial control blade worth. Based on our review of the available data and discussions with GE, we have determined that the 10 percent reduction-in-worth criterion is appropriate.

   Section A.4.3.1.1.1 of GESTAR II states that the core shall be capable of being rendered subcritical at any time or at any core conditions with the highest worth control rod fully withdrawn which is merely a statement of GDC 27. We interpret the statement as an indication that GE will maintain control reactivity in all circumstances including boron carbide (B_4C) leaching mentioned above. We, therefore, conclude that the GESTAR II design basis for control rod reactivity is acceptable.

3.2 Fuel Rod Failure Criteria

The NRC staff's evaluation of fuel rod failure thresholds proposed by GE for the failure mechanisms listed in the SRP is presented in the following paragraphs. Then these failure thresholds are applied to normal or transient operation, they are used as limits (and hence SAFDLs), since fuel failure under those conditions should not occur (according to the traditional conservative interpretation of GDC 10). When these thresholds are applied to accident analyses, the number of fuel failures must be determined for input to the radiological dose calculations required by 10CFR100. The basis or reason for establishing these failure thresholds is thus predetermined, and only the threshold values are reviewed below.

1. Internal Hydriding

   Hydriding as a cladding failure mechanism is precluded by controlling the level of moisture and other hydrogenous impurities during fabrication. To ensure that failure will not occur due to internal cladding hydriding, GE specifies a fabrication limit on hydrogen content, which is less than or equal to the limit stated in the SRP Section 4.2.II.A.2 (a), i.e., 2 ppm hydrogen for a UO_2 pellet.

   Since GE uses a hydrogen limit identical to that stated in the SRP we conclude that the design limit for hydriding is acceptable.

2. Cladding Collapse

   If axial gaps in the fuel pellet-column were to occur due to densification, the cladding would have the potential of collapsing into a gap (i.e., flattening). Because of the large local strains that would result from collapse, collapsed cladding is assumed to be failed. In order to define a collapse criterion to reflect the operational conditions of the reactor, GE has (NEDE-20606-P-A) adopted 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. NEDE-20606-P-A, which contains these limits and definitions, has been reviewed and approved (Butler, April 1975). The NRC staff, therefore, concludes that the cladding collapse design limit has been adequately addressed for GESTAR II.

3. Overheating of Cladding

   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 by high temperature cladding mechanisms. For BWR fuel, thermal margin is stated in terms of the minimum value of the critical power ratio (MCPR), which corresponds to the most limiting fuel assembly in the core. As indicated in Section S.2.0 of NEDE-24011, 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 and NEDO-10958-A. These design limits are consistent with the thermal margin guidelines of SRP Section 4.2.II.A.2 and are thus acceptable from the standpoint of the fuel mechanical design.

4. Overheating of Fuel Pellets

   Although it is stated in SRP Section 4.2.II.A.2.e that it has been traditional practice to assume that failure will occur if fuel pellet centerline melting takes place, that conservative assumption has been strictly applied only to PWR fuel. For BWRs, a limited amount of calculated $U0_2$ melting has been permitted for certain events such as the rod withdrawal error as long as the melting is not sufficient to cause l-percent cladding plastic strain. See paragraph 2.1.2(6) below for a discussion of the l-percent strain limit.

   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.

The GE design basis and limits appear to be consistent with the guidelines of the SRP. We thus conclude that the GE design criterion of fuel pellet overheating is acceptable.

5. Excessive Fuel Enthalpy

   For a severe reactivity initiated accident (RIA) in a BWR at zero or low power, fuel failure is assumed in the SRP to occur if the radially averaged fuel rod enthalpy is greater than 170 cal/g at any axial location. The 170 cal/g enthalpy criterion, developed from SPERT tests (Grund, et al., August l969), is primarily intended to address cladding overheating effects, but is also indirectly addresses pellet/cladding interactions of the type associated with severe RIAs. As indicated in GESTAR II Section S.2.5.1.6 and in NEDE-10527, GE uses 170 cal/g as a cladding failure threshold and thus satisfies the SRP.

6. Pellet/Cladding Interaction

   Fuel failures due to pellet/cladding interaction (PCI) have been encountered in operating BWR fuel (NEDE-24343-P). PCI generally occurs during a power increase as the fuel pellet expands and exerts stresses on the cladding. Although the exact mechanisms that contribute to PCI damage have not been established beyond a doubt, operating experience indicates that irradiated Zircaloy cannot readily accommodate stresses or strains of this kind, particularly when the Zircaloy has been exposed to certain embrittling (stress-corrosion) fission product species such as iodine or cadmium.

   As indicated in SRP 4.2 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 employs a 1% plastic strain "safety" limit (design limit) for cladding strain based on data from fuel rods operated in BWRs (NEDO-10505). This is consistent with the SRP. We therefore conclude that the PCI design criterion and limit are acceptable for GESTAR II.

7. Cladding Rupture

   Zircaloy cladding will rupture (burst) under certain combinations of temperature, heating rate, and stress during a LOCA. While Appendix K to 10CFR50 requires that the incidence of rupture during a LOCA not be underestimated, there are no design limits required for cladding rupture. A rupture-temperature correlation (NEDE-20566-P) is used in the LOCA emergency core cooling system (ECCS) analysis as stated in Section S.2.5.2.14 of GESTAR II, and that correlation is evaluated in Section 5.2 of this SER.

8. Fuel Rod Mechanical Fracturing

   The term "mechanical fracture" refers to a cladding defect that is caused by an externally applied force such as an hydraulic load or a load derived from core-plate motion. These loads are bounded by the loads of a LOCA and safe-shutdown earthquake (SSE), and the mechanical fracturing analysis is usually done as a part of the seismic-and-LOCA loads analysis (see Section 5.3(4) of this SER). The entire seismic-and-LOCA loads evaluation (including design limits) has been described by GE in the approved topical report (NEDE-21175-3) to which GESTAR II makes reference. We thus conclude that fuel rod mechanical fracturing criteria in NEDE-21175-3 are acceptable for GESTAR II.

3.3 Fuel Coolability Criteria

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

1. Fragmentation of Embrittled Cladding

   To meet the requirements of 10CFR50.46 as it relates to cladding embrittlement for a LOCA, acceptance criteria of 2200°F on peak cladding temperature and 17 percent on maximum cladding oxidation must be met. These criteria are described in GESTAR II Section S.2.5.2 and NEDE-20566-P.

2. Violent Expulsion of Fuel

   In a severe reactivity initiated accident (RIA) such as a BWR control rod drop, the large and rapid deposition of energy in the fuel can result in fuel melting, fragmentation, and violent dispersal of fuel droplets or fragments into the primary coolant. The mechanical action associated with such fuel dispersal can be sufficient to destroy the cladding and rod-bundle geometry of the fuel and to produce pressure pulses in the primary system. To meet the guidelines of the SRP as it relates to the prevention of widespread fragmentation and dispersal of fuel and the avoidance of pressure pulse generation within the reactor vessel, a radially averaged enthalpy limit of 280 cal/g should be observed. As indicated in NEDO-10527 and Section S.2.5.1 of GESTAR II, GE employs this 280 cal/g criterion as a control rod drop accident design limit.

3. Cladding Ballooning

   Zircaloy cladding will balloon (swell) under certain combinations of temperature, heating rate, and stress during a LOCA. While Appendix K to 10CFR50 requires that the degree of swelling during a LOCA not be underestimated, there are no design limits required for cladding swelling. A burst-strain correlation (NEDE-20566-P) is used in the LOCA emergency core cooling system (ECCS) analysis, and that correlation as evaluated in Section 5.3 of this SER.

4. Fuel Assembly Structural Damage From External Forces

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

   The entire seismic-and-LOCA loads evaluation (including design limits) has been described by GE in the approved topical report NEDE-21175-3 to which GESTAR II makes reference. We conclude that the criteria for fuel assembly structural damage from external forces in NEDE-21175-3 are acceptable for GESTAR II.

4.0 DESCRIPTION AND DESIGN DRAWINGS

The fuel assembly design description and drawings are provided in Section 2.1, Table 2-1, and Figures 2-1 and 2-2 of the Amendment 7 submittal. As compared to the previously approved version of GESTAR II, NEDE-24011-P-A-6, for the fuel assembly design description and drawings, the information (including figures and tables) in the current submittal is considered minimal. To remedy this, GE provides most necessary information including material properties in the fuel performance code GESTR-MECHANICAL report (MFN 170-84-0). We consider this an acceptable alternative. Therefore, we conclude that the GESTAR II Amendment 7 fuel description and design drawings are acceptable.

5.0 DESIGN EVALUATION

Design bases and limits were presented and discussed in SER Section 3.0. In this section we review GE methods of demonstrating that GESTAR II meets the design criteria that have been established. This SER section will, therefore, correspond to Section 3.0 of the SER point by point. The methods of demonstrating that the design criteria have been met include operating experience, prototype testing, and analytical predictions.

5.1 Fuel System Damage Evaluation

The following paragraphs discuss the NRC staff's evaluation of the conformance of GESTAR II with the fuel system damage criteria described in Section 3.1. Those criteria apply only to normal operation and anticipated operational transients.

1. Stress and Strain

   In response to NRC questions (Thomas, April 4, 1984), GE describes a method and procedures to calculate design ratios for stress and strain, including the use of the GESTR-MECHANICAL (MFN 170-84-0) code for error propagation analysis and the Monte Carlo statistical analysis, in a step-by-step fashion (Charnley, April 23, 1984).

   The input parameters to the GESTR-MECHANICAL code include uncertainties determined from the tolerances and specifications of manufacturing data. The Monte Carlo calculations use results from GESTR-MECHANICAL and result in a distribution of design ratios. GE states that a conservative limit of the upper 95 percentile of the design ratio distribution is required to be less than 1.0. GE has determined the design ratio distributions for the P8x8R and BP8x8R fuel designs, and has confirmed that the design criterion is met.

   Based on the facts that (1) the GESTR-MECHANICAL code report (MFN 170-84-0) provides most of the material properties and stress/strain analyses, (2) that GESTAR II Amendment 7 provides the conservative ultimate tensile strength for Zircaloy 2 cladding, and (3) the design ratio distribution approach is conservative, we conclude that GE has satisfactorily demonstrated that the P8x8R and BP8x8R fuel designs will meet the stress/strain limits in GESTAR II.

2. Strain Fatigue

   GE analyzed strain fatigue using GESTR-MECHANICAL and the standard error propagation statistical method as already described in Section 5.1(1). The fatigue curve in terms of strain amplitude and allowable cycles is given in the Amendment 7. The fatigue

evaluation accounts for internal and external pressures, cladding temperature, and pellet-cladding contact. The results show that the upper 95th percent of this fatigue usage as less than 1.0.

We, therefore, conclude that there is reasonable assurance that the fuel P8x8R and BP8x8R designs will conservatively meet the strain fatigue criterion in GESTAR II.

3. Fretting Wear

In the GE BWR P8x8R and BP8x8R fuel designs individual rods in the fuel assembly are held in position 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 along with the results of a continuing fuel surveillance program that had utilized non-destructive methods including eddy currents to locate discontinuities in the cladding and detailed visual examinations to characterize the nature of defects. As stated in NEDE-24011-P-A-6, no significant fretting wear had been observed. GE concluded that significant fretting wear was avoided through the use of an active spring force to eliminate any clearance that would otherwise exist between the spacer structure and the fuel rods. GE states, in Amendment 7, that this conclusion applies to the P8x8R and the BP8x8R fuel designs. We will continue to examine fretting as a potential fuel damage mechanism for any new fuel designs.

In addition to the fuel rods and spacers, there is another fuel system component whose functionality must be assured as an objective of the review of BWR fuel system fretting concerns; viz., the fuel assembly channel box. The fuel “channel” that encloses the fuel bundle performs three functions: (1) the channel provides a barrier to separate two parallel flow paths (one to cool the fuel bundle and the other to cool the bypass region between channels); (2) the channel guides the control blade and provides a bearing surface for it; and (3) the channel provides rigidity for the fuel bundle. Thus, the potential for cracks or holes in a “channel” or channel “box” is of concern since it would allow part of the cooling water that normally flows through a fuel bundle to flow out of the cracks or holes and bypass the fuel rods. Such a change in flow pattern would reduce the safety margin for fuel thermal performance and could lead to fuel overheating and damage in the event of some anticipated operating transients and postulated accidents. Significant channel box cracking and wear could also adversely affect mechanical strength.

In the mid-1970s, channel box wear and cracking was observed in a number of BWR-4s. The wear was located adjacent to incore neutron monitor and startup source locations. It was postulated (NEDO-21084), and later confirmed by out-of-reactor testing, that the wear was caused by vibration of nearby incore instrument and source tubes due primarily to a high-velocity jet of water flowing through the bypass flow holes in the lower core plate. To eliminate significant vibration of instrument and source tubes and the resultant wear on channel box corners, GE incorporates modifications for BWR-4s similar to those described in NEDE-21156 for BWR fuel currently in operation including those designs covered by GESTAR II Amendment 7. Those modifications involve the elimination of the bypass holes in the lower core plate and the addition of two holes in the lower tieplate of each assembly to provide an alternate flow path. This design modification has been determined to have negligible adverse effects on the mechanical, thermal, and nuclear performance of the channel boxes, as is discussed in our generic safety evaluation on this subject (Eisenhut,

March 2, 1976). Because channel box wear has been observed (Engel, October 15, 1977) to have been significantly reduced in operating BWR-4s following the design modification, we conclude that there is reasonable assurance that channel box wear and cracking is not a problem.

4. External Corrosion and Crud Buildup

Waterside Corrosion

Corrosion problems associated with stainless steel cladding in BWRs, together with a desire to improve neutron economy, led to a change some years ago from stainless steel to Zircaloy cladding material, which has good resistance to the hot water and steam environment encountered under typical BWR operating conditions (Garzarolli, et al., 1978). In several recent cases, however, cladding failures have been associated with external "waterside" corrosion (e.g., GE Protects Division Memorandum, April 6, 1979, and Maury, November 10, 1981). While the details of the most recent occurrences are still under investigation by GE, the occurrences have been characterized by GE as "crud-induced local corrosion failure" (NEDE-24343-P). The corrosion is reportedly associated with a variably high copper concentration in the core coolant water and a minor anomaly in the Zircaloy cladding metallurgy (Charnley, December 11, 1979; Engel, October 3, 1980; Smith, November 21, 1980; Smith, February 5, 1981; and DelGeorge, December 2, 1981). All the affected plants have copper alloy condenser tubing. It is notable also that virtually all the BWR waterside corrosion failures have involved gadolinia burnable poison rods.

The NRC staff has been following this issue generically, and in a recent meeting with the staff, GE identified (Tokar, October 13, 1982) what were believed to be some factors responsible for the corrosion failures and stated that a change had been made in the manufacturing process to assure that such failures will not occur in newly manufactured fuel bundles. Further documentation on this subject has been provided in a letter from Washington Public Power Supply System (Bouchey, December 23, 1982).

GE also considers corrosion in the overall fuel rod thermal-mechanical analysis, for example, stress and strain, rod pressure, cladding collapse, and PCI. Inasmuch as GE has proposed design changes to eliminate the waterside corrosion concern and considers the effects of corrosion in fuel rod thermal mechanical analysis, we conclude that corrosion is adequately addressed in GESTAR II Amendment 7.

Crud Buildup

The buildup of a corrosion film and a crud layer on the outer surface of a fuel rod during irradiation causes gradual flow reductions and impedes heat transfer to the coolant. The SRP does not specify a crud buildup limit. GE considers the effect of crud buildup as an integral part of the GESTR-MECHANICAL fuel rod analysis. This effect will reflect on the analyses of stress and strains, rod bowing, gas pressure, cladding collapse, overheating of pellets, and PCI (by increasing the temperature of the cladding). We consider this approach acceptable for design evaluation.

The effects of corrosion crud buildup on BWR fuel intended for high burnup applications has been addressed by GE in NEDE-22148-P. This report is currently under review by the staff. We will address the effects of corrosion and crud buildup during burnup in our SER on this report.

5. Dimensional Changes

Rod Bowing

The effect of fuel rod bowing on critical power ratio of a boiling water reactor is described in NEDE-24284-P, "Assessment of Fuel-Rod Bowing in General Electric Boiling Water Reactors". As stated in the safety evaluation report dated March 8, 1983, the staff review of this topical report had concluded that significant fuel rod bowing in the GE BWR fuel is not anticipated and therefore no operational penalties are required. However, in order to assure that this assumption remains valid, GE is required to notify NRC of any observation of fuel rod bow with a gap closure of greater than 50% with GE supplied fuel.

GE maintains that a CPR penalty is not necessary regardless of the magnitude of fuel rod bowing because (1) the GE test data for BWR fuel show little effect of fuel rod bowing on CPR, and (2) the CHF test results simulating PWR fuel rods bowed to contact (zero gap) show insignificant reduction of critical heat flux at low pressure. The staff has performed a further review in light of the GE proposal to see if any change in our position is warranted.

The GE test data presented in NEDE-24284-P consist of 9-rod, 16-rod and 64-rod experimental results. The 9-rod test bundle represents a rod bow of 78% gap closure and had a maximum CPR reduction of 9 percent. GE in a letter dated June 12, 1984 indicates that the 9-rod test bundle is not prototypical of the BWR fuel and therefore should be dismissed. The 16-rod and 64-rod test bundles represent gap closures of 55 percent and 62 percent, respectively. The test results of these bundles show a maximum CPR reduction of about 4 percent. GE maintains that the CPR reduction is minimal because 4 percent as within the experimental uncertainty for data reproducibility. Though these test results may show small CPR reduction, the test geometries represent a moderate rod bow configuration which has been shown by PWR fuel rod bow tests (Hill, ASME 75-WA/HT-77; Markowski, ASME 77-HT-91) to have an insignificant effect on CHF. Therefore, a conclusion can not be drawn from the GE test results to preclude CPR reduction for more severe rod bowing such as bow-to-contact.

On the other hand, the PWR fuel rod bow-to-contact test results have shown no adverse effect on CHF at low pressure. However, these tests were performed at a pressure greater than 1600 psia, well above the BWR operation pressure of about 1000 psia. The staff is not aware of any data available to extend the PWR test results to the BWR operating pressure. GE maintains that the PWR low pressure data represent the same flow regime (annular flow) expected in the BWR. While we agree with this, the physical phenomenon of the effect of rod bow on CHF is not fully understood; there is little basis therefore to confirm that the PWR low pressure test results can be extrapolated to BWRs solely due to the same flow regime.

Based on the above observations, the staff cannot make a finding that the CPR penalty can be eliminated regardless of the magnitude of the fuel rod bowing.

Our conclusion that no reduction in CPR is necessary because of the limited amount of fuel rod bowing in GE fuel will remain unchanged.

Channel Box Deflection

BWR fuel channels provide structural stiffness for the fuel assembles and distribute the coolant flow between the assembles and channel bypass regions. The regions are subject to

time-dependent, permanent dimensional changes (i.e., deflections) that results from irradiation, creep and stress-relaxation effects. The resultant bulge (resulting from long-term creep) or bow (resulting from differential irradiation-induced axial growth) reduces the size of the gap available for control blade insertion. Channel box deflection is thus a phenomenon that can limit channel life because of the potential adverse effects on the ability of the control blades to move freely.

In a generic topical report (NEDE-21354-P), GE describes a channel lifetime prediction method and a backup recommendation for periodic channel deflection measurements that consist of settling friction tests. Upon consideration of the factors involved, the NRC staff concluded that the settling friction tests or an acceptable alternative (such as channel dimensional deflection measurements) should be performed, and in a memorandum (Rubenstein, September 18, 1981) the staff outlines a method that could be used to resolve the channel box deflection issue for several near-term BWR operating license applications. Basically, the staff advocated a multistep procedure that had been proposed by the Zimmer applicant. The key ingredient of the Zimmer plan was a commitment to (a) perform some control rod settling friction tests, which would provide an exact profile of control rod drive friction versus position at refueling outages, or (b) make some actual channel dimensional measurements. Several other applicants agreed to the Zimmer proposal. Subsequently, the BWR Licensing Review Group (LRG-II) submitted a position paper (Holtzscher, May 17, 1982) on channel box deflection that incorporated several of the same features as the Zimmer proposal, although the settling friction test was simplified. As part of the GESSAR-II fuels review, GE was asked (Berlinger, October 12, 1982) a question on channel box deflection, and in separate responses (Sherwood, December 21, 1982 and Sherwood, February 2, 1983), GE reiterated the LRG-II position.

Both the LRG-II position statement and the later GESSAR-II submittal indicate that the simplified settling friction test provides an equivalent level of test to that described in NEDE-21354-P. To be precise, however, the test is equivalent to a so-called “screening-type” test (described in NEDE-21354-P) to identify any control rod drive where the force required to insert is greater than a critical valve. Whereas in NEDE-21354-P it is indicated that rods failing the screening test would be given another settling friction test to obtain in exact friction-versus-position profile, the latter type of test is not mentioned by LRG-II or GESSAR-II. The position taken by the LRG-II and GESSAR-II applicants is that failure of the proposed settling time test would “prompt an investigation,” which, if necessary, would lead to corrective action.

While we believe that a commitment to perform in exact settling friction profile test (or actual dimensional measurement) is preferable (because it would provide an estimate of the margin and physical state of the system in an unambiguous way), the LRG-II and General Electric Company in NEDE-21354-P have stated that the control rod drives will tolerate a relatively large increase in driveline friction while still remaining within Technical Specification limits. The screening-type test proposed would, thus, provide assurance of the scram function. Therefore, we have accepted (Rubenstein, August 19, 1982) the LRG-II position that the proposed actions will preclude excessive channel bowing in the LRG-II plants (i.e., River Bend and Perry), and by the same token, we accept GESSAR-II's endorsement of that position.

In a recent letter (Charnley, July 18, 1984), GE states that the recommendation in NEDE-21354-P and the GESSAR-II position will be implemented in GESTAR II. We, therefore, conclude that channel box deflection issue is resolved in GESTAR II.

Irradiation Axial Growth

GE relies on operational experience in determining the adequacy of expansion spring design. According to the GE results, the expansion spring experiences little or no axial compression during irradiation exposure. This suggests that fuel rod irradiation growth is small. Therefore, no interference between the fuel rods and the upper tieplate would be expected.

Based on the staff's independent analysis, the irradiation axial growth is indeed rather small for BWRs. We, therefore, conclude that irradiation axial growth is not a problem for those fuel designs covered by GESTAR II.

6. Fuel and Poison Rod Pressures

In order to calculate whether the criterion of cladding creepout rate less than fuel swelling rate as met, GE first uses the fuel performance code GESTR-MECHANICAL (MFN 170-84-0) to get fuel and cladding irradiation conditions including temperature, pressure, fluence, etc. Then a bounding peak rod power is chosen for the analysis.

Using these conditions in the cladding creepout and fuel swelling equations, a comparison between cladding creepout rate and fuel swelling rate is made. GE states that confirmation has been made for the current fuel designs of P8x8R and BP8x8R that the cladding creepout rate does not exceed the fuel swelling rate throughout the lifetime of the fuel rod. If for any reason this new fuel design criterion were to be violated, GE could modify such physical parameters as fuel rod pressure or peak rod power limit to correct the problem. Since GE uses cladding creepout and fuel swelling equations from the GESTR-MECHANICAL code, we consider that the analysis method is acceptable.

7. Fuel Assembly Liftoff

The potential for BWR fuel assembly liftoff is addressed in a GE letter (Gridley, July 11, 1977), which covers accident conditions as well as normal operation. Since 4.2.II.A.1.g of the SRP deals with normal operation only, however, and so in this SER normal operation liftoff is separated from the combined seismic-and-LOCA liftoff concern, which is addressed in SER Section 5.3(4).

The Gridley, July 1977 letter and the response (Sherwood, February 2, 1983) to a staff question on GESSAR-II indicated that for loads resulting from normal operation, including AOOs, GE does not calculate any bundle separation from the core support plate. Furthermore, this steady-state analysis is bounded by the combined seismic-and-LOCA analysis in the approved report NEDE-21175-3, which shows no significant liftoff. We conclude, therefore, that there is reasonable assurance that fuel liftoff will not occur for the fuel and core designs discussed in GESTAR II Amendment 7.

8. Control Material Leaching

The loss of boron carbide (B_4C) by leaching from cracked control blade tubing is addressed in NRC IE Bulletin No. 79-26, Revision 1, which requires operating BWRs to perform

various actions including, but not limited to, shutdown-margin tests. Subsequent to the issuance of IE Bulletin 79-26, GE performed analyses and post-irradiation examinations (NEDE-24226-P) and developed a boron depletion model (NEDE-24325-P). This model supports GE's claim that the amount of boron loss can be accurately determined analytically and that potential control blade degradation due to this mechanism will not significantly affect plant operation. Further generic review of this matter may thus lead to a suspension or revision of IE Bulletin 79-26 requirements. In this connection, National Nuclear Corporation has recently reported (EPRI NP-2730) the development, under an EPRI-funded program, of a nondestructive method to determine the loss of B_4C powder from BWR control blades.

Inasmuch as costly hot cell examinations have heretofore been the only procedure available to monitor the performance of BWR control blades, the new non-destructive technique may render moot the need for adherence to analytical calculation methods and limits. Since, until new mitigation approaches are adopted, GESTAR II will be subject to the IE Bulletin, assurance will be maintained that control blade reactivity will not be significantly degraded by boron carbide leaching. We, therefore, conclude that control material leaching problem is satisfactorily resolved in GESTAR II.

5.2 Fuel Rod Failure Evaluation

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

1. Internal Hydriding

   GE employs a hot vacuum outgassing (drying) method for removal of the moisture contamination of loaded fuel rods to assure that the level of moisture is well below the limit discussed in SER Section 3.2 (1). To prevent the introduction of other hydrogenous impurities such as oils, plastics, etc., various procedural controls are utilized during manufacturing (NEDE-24011-P-A-6). If even in the face of those manufacturing controls, moisture or hydrogenous impurities are still inadvertently introduced into a fuel rod during manufacture, further assurance against chemical attack is provided by the use of a hydrogen getter material placed in the upper plenum of the fuel rod. The incorporation of a hydrogen getter in GE fuel rods is described in NEDO-20922. GE calculations, based on an assumed initial rod moisture content equal to the specification limit, indicate that no cladding damage will occur during the active hydrogen/moisture gettering period (NEDE-24011-P-A-6). Field experience (NEDE-24343-P) has confirmed hydriding is not an active failure mechanism for fuel manufactured since mid-1972. On the basis of the information presented, we conclude that adequate assurance has been provided that fuel and burnable poison rod hydriding will not be a problem for the fuel rod designs covered in GESTAR II Amendment 7

2. Cladding Collapse

   GE has performed a cladding (long-term creep) collapse analysis. Based on the results of that analysis, which was performed with an approved model (NEDE-20606-P-A), cladding collapse was not calculated to occur for either P8x8R or BP8x8R fuel rods. The NRC staff, therefore, concludes that there is reasonable assurance that long-term creep collapse will not occur in the fuel designs covered by GESTAR II Amendment 7.

3. Overheating of Cladding

   Since Amendment 7 references the current approved analysis in Section 5.2.0 of NEDE-24011-P-A-6, we conclude that overheating of cladding as addressed adequately in GESTAR II.

4. Overheating of Fuel Pellets

   GE has used GESTR-MECHANICAL with a bounding power history to determine the maximum expected fuel centerline temperature during normal operation and AOOs. The results show that no fuel melting is expected to occur for P8x8R and BP8x8R fuel designs for whole core anticipated operational occurrences but may occur for local anticipated operational occurrences such as the rod withdrawal error. In the latter case, GE states that the 1% cladding strain limit will be met. Based on the use of the GESTR-MECHANICAL method, and the fact that the appropriate criteria are satisfied, we conclude that the UO_2 fuel melting analysis is acceptable for GESTR II.

   We have also approved the properties of fuel containing gadolinia described in Appendix B to NEDE-23785-1-P. GESTR-MECHANICAL has all the major gadolinia fuel properties from Appendix B except a few small changes. In a letter dated February 2, 1984, GE stated (Charnley, February 2, 1984) that the GESTR-MECHANICAL code with gadolinia fuel properties was used for new plants including GESSAR-II, and the results indicated that gadolinia fuel melting is not expected to occur during normal operation and whole core AOOs. Based on the use of GESTR-MECHANICAL and the fact that the appropriate criteria are satisfied, we conclude that the UO_2-Gd_2O_3 fuel melting analysis is acceptable for the fuel designs included in GESTAR II Amendment 7.

5. Excessive Fuel Enthalpy

   Large fuel enthalpies occur only in the postulated control rod drop accident. Amendment 7 references the approved analysis of the control rod drop accident in Section S.2.5.1 of NEDE-24011-P-A-6 and in NEDO-10527. We thus conclude that the excessive fuel enthalpy issue is adequately addressed in GESTAR II.

6. Pellet Cladding Interaction

   As mentioned earlier, GE employs a l% plastic strain "safety" limit (design limit) for cladding strain based on data from fuel rods operated in BWRs (NEDO-10505). As noted in NEDE-24011, none of the data reported which represented failed fuel rods fell below the 1% plastic strain value, but a statistical distribution fit to the available data indicated that 1% plastic strain value was approximately the 95% point in the total population. The distribution implied, therefore, that there was a small (less than 5%) probability that some cladding segments may have plastic elongations less than 1% at failure.

   Earlier operating experience has also shown that PCI failures occurred at total strains significantly less than l%. It therefore appears that high local strains are also important to PCI failure and may occur even while the average strain criterion of 1% is met.

   Subsequently, based on results 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 (NEDS-10456-PC). These restrictions reduce 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 would not prevent PCI failures during anticipated operational occurrences and accidents.

More recently GE has also introduced a new fuel design which specifically addresses PCI and resulting operating restrictions (NEDE-24011-P-A-5). The staff has approved this design but has not specifically addressed the extent of the difference in behavior of this fuel to other GE fuel types during anticipated operational occurrences and accidents.

As discussed above, the use of PCIOMRs and the introduction of a new fuel specifically designed to prevent PCI failures has been generally effective in reducing PCI failures during normal operation. In order to assess whether PCI failures should be explicitly considered in the evaluation of the potential radiological consequences of off-normal (i.e., transient and accident) conditions, an NRC task force was formed to study this question. This task force has produced a draft report for comment (Van Houten, May 1984). The report addresses both BWR and PWR fuel.

The findings of this draft report state that PCI does not appear to be a concern for core wide, promptly scrammed BWR anticipated transients; however, these conclusions are based on data which are not altogether representative of current BWR operation. The number of rods tested was limited, the rods operated at relatively low powers prior to the tests and only “modest” burnup rods were used. In addition, the task force report stated that localized transients, such as those associated with the movement of a single control rod, such as the postulated “Rod Withdrawal Error From Subcritical Conditions” could lead to fuel failure from PCI. GE has also reached this same conclusion (Sherwood, November 10, 1976).

The staff as currently evaluating the comments received on this task force report. Pending completion of our evaluation, we believe that, although the predicted number of fuel rod failures may not correctly account for those rods which fail from PCI, there is reasonable assurance that the level of protection in terms of offsite dose is acceptable based on the following factors which are not taken into account explicitly in licensing analyses of postulated transients and accidents:

(1) For some postulated transients and accidents, fuel rods which are postulated to fall from other mechanisms (e.g., boiling transition, peak pellet enthalpy) might be the same fuel rods which would also fail from PCI,

(2) PCI is fundamentally different than boiling transition in that boiling transition indicates a loss of the ability to remove heat from the fuel which can lead to a loss of coolable geometry and fuel failure propagation if the event were severe enough. PCI has no such potential.

(3) More realistic analyses show (as expected) that peak power, ramp rate and hold times are less severe than those calculated using licensing assumptions. Thus, the realistic calculations show that the potential for PCI causing significant number of fuel failures is low.

4. Cladding Rupture (Bursting)

   The GE cladding rupture model for ECCS analysis is described in the approved report NEDE-20566-P. An amendment to this report adopted the NUREG-0630 data base and modeling. The Amendment 7 submittal refers to Section S.2.5.2.1.4 in NEDE-24011-P-A-6 which references this report. We therefore conclude that cladding rupture is properly addressed in GESTAR II.

5. Fuel Rod Mechanical Fracturing

   The mechanical fracturing analysis is usually done as a part of the seismic-and-LOCA loads analysis, which is described in the approved report NEDE-21175-3. A discussion of the seismic-and-LOCA loading analysis is given in Section 5.3(4) of this SER.

5.3 Fuel Coolability Evaluation

The following paragraphs discuss the staff's evaluation of the ability of the fuel to meet the fuel coolab111ty criteria described in Section 3.3. Those criteria apply to postulated accidents.

1. Fragmentation of Embrittled Cladding

   The primary degrading effect of cladding oxidation is embrittlement. Such embrittled cladding will have reduced ductility and resistance to fragmentation. The most severe manifestation of such embrittlement occurs during the postulated loss-of-coolant accident (LOCA). Amendment 7 references NEDE-20566-P and Section S.2.5.2 of GESTAR II for a discussion of the overall effects of a LOCA on peak cladding temperature, oxidation and embrittlement. Both of these documents have been previously approved by the staff. The effects of fission gas release on stored energy and subsequently on peak cladding temperature, oxidation and embrittlement is considered in Section S.2.5.2.5 of GESTAR II. We thus conclude that the issue of fragmentation of embrittled cladding as adequately addressed in GESTAR II Amendment 7.

2. Violent Expulsion of Fuel

   Amendment 7 refers to the current approved results in Section S.2.5.1 of NEDE-24011-P-A-6 and NEDO-10527. We thus conclude that the violent expulsion of fuel issue is adequately addressed in GESTAR II.

3. Cladding Ballooning

   The GE cladding ballooning model for ECCS analysis is addressed in the approved report NEDE-20566-P. This report adopted the NUREG-0630 data base and modeling, which specifies a method acceptable to the NRC for treating cladding swelling and rupture during a LOCA. Amendment 7 refers to Section S.2.5.2.1.4 in NEDE-20411-P-A-6 which references this report. We thus conclude that cladding ballooning is adequately addressed for GESTAR II.

4. Fuel Assembly Structural Damage from External Forces

Generic methods for performing combined seismic-and-LOCA loading analysis are presented in the approved topical report NEDE-21175-3. This analysis includes not only the fuel assembly structural responses but also fuel assembly liftoff during the seismic-and-LOCA events to conform to the requirements described in Appendix A to SRP 4.2.

Since this analysis depends on plant-specific input ground motions, this analysis is not completed in a generic manner. Therefore, an applicant proposing to reference GESTAR II must address the guidelines of Appendix A to SRP Section 4.2, to show that the fuel design will satisfy the structural acceptance criteria.

6.0 TESTING, INSPECTION, AND SURVEILLANCE PLANS

6.1 Testing and Inspection of New Fuel

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

Section 2.3.1 in the Amendment 7 submittal and NEDO-11209-04A contain descriptions of (1) the type of quality control inspections performed during manufacturing and (2) the plan for on-site inspection and testing of new fuel assemblies. We conclude, on the basis of the information provided in these referenced reports, that the new fuel testing and inspection program for GESTAR II is acceptable.

6.2 On-Line Fuel System Monitoring

The method of on-line fuel system monitoring as proposed by the applicant and is subject to approval by the staff. This topic is not addressed in GESTAR II.

6.3 Post-irradiation Surveillance

As indicated in the SRP, a routine fuel inspection program to provide information on irradiated and discharged fuel should be provided. In a letter dated November 23, 1983, from J.S. Charnley (GE) to C.H. Berlinger (NRC), GE proposed a generic fuel vendor surveillance program, which would satisfy the intent of SRP Section 4.2.II.D.3 that each licensee/applicant performs post-irradiation fuel surveillance on fuel irradiated in the licensee/applicant’s reactor. The program proposed by GE would allow GE to assume the responsibility for post-irradiation fuel surveillance of GE designed and manufactured fuel. This program was approved by the staff in a letter dated June 27, 1984. Therefore, we conclude that post-irradiation surveillance is adequately addressed in GESTAR II as long as licensees/applicants referencing GESTAR II will endorse the GE fuel surveillance program or an acceptable alternative.

7.0 EVALUATION FINDINGS

The fuel system design in GESTAR II has been reviewed in accordance with SRP Section 4.2. The staff concludes that, although most of the objectives of the fuel system safety review have been met,

several issues must be addressed by a licensee/applicant proposing to use this fuel design in GESTAR II.

These issues are:

1. The licensee/applicant must provide a plant-specific analysis of combined seismic-and-LOCA loading using the approved method in NEDE-21175-3 or another acceptable method to demonstrate conformance to the structural acceptance requirements described in Appendix A to SRP 4.2. (SER Section 5.3(4))

2. The licensee/applicant must provide an acceptable post-irradiation surveillance program or endorse the approved GE fuel surveillance program. (SER Section 6.3)

   With the above provisions, the staff concludes that the fuel designs covered in GESTAR II Amendment 7 have been designed such that (1) it will not be damaged 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 thereby meeting the related requirements of the following regulations: 10CFR50.46; GDC 10 and 27; and 10CFR50, Appendix K. This conclusion is based on two primary factors:

   (1) General Electric has provided sufficient evidence that the design objectives will be met based on operating experience, prototype testing, and analytical predictions.

   (2) General Electric has provided for testing and inspection of new fuel to ensure that it is within design tolerances at the time of core loading. The licensee/applicant will perform on-line fuel failure monitoring and post-irradiation surveillance to detect anomalies or confirm that the fuel has performed as expected.

   The staff concludes that General Electric has described methods of adequately predicting fuel rod failures during postulated accidents so that radioactivity releases are not underestimated.

   On the basis of this review, we conclude that, with the above exceptions all the requirements of the applicable regulations have been met and guidelines of the applicable Regulatory Guides and current regulatory positions have been followed for GESTAR II.

8.0 IMPLEMENTATION OF THE GESTR-MECHANICAL CODE

Because Amendment 7 and GESTR-MECHAHICAL are so fundamental to GE's fuel analysis work, a transition will be needed from the Amendment 6 - TEXICO approach to the Amendment 7 - GESTR-MECHANICAL approach. GE may begin to use the Amendment 7 - GESTR-MECHANICAL methods and criteria as approved with issuance of this SER. Within three weeks of issuance of this SER, GE should submit an implementation schedule which should include licensing applications using the Amendment 6 - TEXICO approach. All other licensing actions should be done with the Amendment 7 - GESTR-MECHANICAL approach.

There were some early applications of GESTR-MECHANICAL in 1984 as indicated in an April 24, l984 letter from J.S. Charnley (GE) to R. Lobel (NRC). These included analyses of proposed future fuel designs and lead test assemblies scheduled for insertion in the fall of 1984.

As discussed in the Introduction, GESTR-MECHANICAL (MFN 170-84-0) is derived from the GE computer code GESTR-LOCA (NEDE-23785-l-P) which was previously approved by the staff (Rubenstein, September 26, l983). While the purpose of GESTR-LOCA is to conservatively bound expected behavior of fuel rods during a Loss of Coolant Accident, GESTR-MECHANICAL is used by GE as a design code and therefore 1t must be capable of accurately predicting fuel behavior rather than conservatively bounding such behavior. This requires, according to GE, more frequent updating of the models and material properties in GESTR-MECHANICAL.

In order to make GE's design requirement for frequent updating compatible with the NRC's responsibility for assuring that these changes are acceptable for safety calculations to assure compliance with General Design Criterion 10, GE proposed that not every change should require review by the staff, but only those changes which exceeded a significance test.

The significance test consists of two parts, a sensitivity of the code to the changes made and how these changes affect the capability of the code to predict data. The final form of the significance test agreed to with GE consist of five criteria. These are discussed in detail in Attachment 1 to a letter from J. S. Charnley (GE) to C.O. Thomas (NRC) dated December 14, 1984.

The first criterion limits the increase in centerline temperature of a GE fuel rod. If the change in centerline temperature of a GE fuel rod calculated with both the old and new versions of GESTR-MECHANICAL is greater than the criterion, then an NRC review is required.

The next three criteria limit the change in the predictive capability of the code for fission gas release, centerline temperature and fuel rod mechanical deformation to less than an agreed-to limit. If this limit is exceeded, an NRC review of the changes to GESTR-MECHANICAL is required.

The final criterion states that the introduction of a new fuel performance phenomenon, previously not explicitly analytically treated by the fuel performance code, requires NRC approval prior to application of the revised code.

These criteria are expected to either (1) demonstrate that the predictive capability of the code and its impact on GE fuel rods remains the same within acceptable limits or (2) explicitly demonstrate that an NRC review of the changes to the code is necessary.

It should also be pointed out that these criteria are only for the purpose of deciding whether an NRC review is necessary and will not influence the outcome of that review one way or another.

As part of our review of GESTAR II Amendment 7, GE proposed a number of changes to the fuel and cladding properties and several calculational models for GESTR-LOCA (NEDE-23785-1-P) which the staff has previously reviewed and approved.

GESTR-LOCA with these proposed changes will constitute the first NRC-recognized version of GESTR-MECHANICAL. These changes are described in Attachment 1 to a letter from J. S. Charnley, GE to R. Lobel, NRC, dated December 14, 1984.

GE used the criteria discussed above and demonstrated that the changes to the properties and models in GESTR-MECHANICAL were within the bounds of these criteria. In addition, a helium generation and release model was added. This was considered a new fuel performance phenomenon and was reviewed by the staff. We find this model to be acceptable based on the following:

1. Predictions of fuel rod internal pressure made after incorporating this model show adequate agreement with data, and,

2. the helium gas release model seems reasonable and is in agreement with other data available to the staff.

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NRC
Safety Evaluation Report
Approving Amendment 8
to
NEDE-24011-P

UNITED STATES
NUCLEAR REGULATORY COMMISSION
Washington, D. C. 20555
April 24, 1985

MFN 061-85

Dr. H.C. Pfefferlen, Manager
BWR Licensing Programs
Nuclear Safety & Licensing Operation
General Electric Company
175 Curtner Avenue
San Jose, California 92125

Dear Dr. Pfefferlen:

SUBJECT: Acceptance for Referencing of Licensing Topical Report NEDE-24011, Rev. 6, Amendment 8, "Thermal Hydraulic Stability Amendment to GESTAR II"

We have completed our review of the subject topical report submitted by the General Electric Company (GE) by letter dated September 30, 1983. We find the report to be acceptable for referencing in license applications to the extent specified and under the limitations delineated in the report and the associated NRC evaluation, which is enclosed. The evaluation defines the basis for acceptance of the report.

We do not intend to repeat our review of the matters described in the report and found acceptable when the report appears as a reference in license applications, except to assure that the material presented is applicable to the specific plant involved. Our acceptance applies only to the matters described in the report.

In accordance with procedures established in NUREG-0390, it is requested that GE publish accepted versions of this report, proprietary and non-proprietary, within three months of receipt of this letter. The accepted versions shall incorporate this letter and the enclosed evaluation between the title page and the abstract. The accepted versions shall include an -A (designating accepted) following the report identification symbol.

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

Sincerely,

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

Enclosure:
As stated

Safety Evaluation
of the
General Electric
Topical Report NEDE-24011 Amendment 8
General Electric Standard
Application For Reload Fuel,
Amendment #8

March, 1985
CORE PERFORMANCE BRANCH

ENCLOSURE

1.0 INTRODUCTION

This SER evaluates the thermal-hydraulic stability licensing criteria proposed by General Electric in NEDE-24011 Amendment 8. The GE report NEDE-22277-P-1, "Compliance of the General Electric Boiling Water Reactor Fuel Designs to Stability Licensing Criteria" (Reference 14), is the principal document submitted in support of Amendment #8 to GESTAR. This evaluation has been supported by review and audit calculations performed by Oak Ridge National Laboratory under contracts FIN B0777 (TER-reference 8) and FIN B0794 (TER-reference 9). The results obtained by ORNL in their audit calculations and comparisons to plant data and experiments have been used by the staff to set the uncertainty value of GE's methodology and to determine the acceptability of GE's proposed licensing criteria.

2.0 DESCRIPTION OF GE'S THERMAL-HYDRALIC STABILITY METHODS AND PROPOSED LICENSING CRITERIA

2.1 Thermal-Hydraulic Stability Analysis Methods

To investigate the stability of the large nonlinear dynamic BWR system the stability of individual components is evaluated before analyzing their inter?action with the total system. For the BWR, these individual components are the channel and reactor core. The hydrodynamic stability of individual channels is analyzed and then the channels are coupled hydraulically and combined with neutronics and heat transfer to study the stability of the core. A linearized, small-perturbation frequency domain model, FABLE (1) is used to perform these calculations. Linear, small-perturbation theory is a special case of the general theory of nonlinear systems analysis. The interaction of the reactor core with the physical control systems associated with the nuclear steam supply and, hence, the total system stability, is investigated with the nonlinear plant transient simulator digital model, REDY [(1)].

Qualification of the analytical models is demonstrated by comparisons with operating plant tests. Control rod oscillator tests at several plants are used to provide open loop and closed loop response characteristics of the BWR subjected to reactivity perturbations. In addition, pressure setpoint oscillation tests provide system response characteristics for the neutron flux/core-exit-pressure transfer function. These test conditions are simulated using the REDY and FABLE models and the results are compared to test data. Qualification of the FABLE channel hydrodynamics model is performed by comparisons to electrically-heated channel experiments and data from operating reactor tests.

The output from the GE analysis is a limiting best estimate decay ratio.* This decay ratio is found in the low flow/high power portion of the power flow map at the intersection of the power flow curve and the rod block line under natural circulation conditions.

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* For an oscillatory response, the decay ratio is defined as the ratio of two subsequent peaks which are both on one side (i.e., above or below) of the average value of the oscillatory parameter. Decay ration is used as a measure of a system's stability. For decay ratio <1.0, the system is damped and the oscillatory response decays, for decay ratio >1.0, the system is undamped and the oscillations increase in magnitude. For the special case of

2.2 Stability Tests

The GE methods have been benchmarked against various operating plant test data. The principal data come from the tests performed at Peach Bottom[3] (1977, 1978), Vermont Yankee[5] (1981) and a recent test at an overseas BWR plant.

The possibility of instability in a BWR has been investigated since the startup of early BWRs. These early tests oscillated a control rod within one notch position (6 inches) and measured the response of the reactor (core-exit-pressure and APRM signal). For modern higher-power density reactors, control rod oscillator tests are not desirable because of high cost and poor signal-to-noise ratios in large reactor cores. A technique using pressure perturbations was developed and stability tests were performed at the end of Cycle 2 and during Cycle 3 at Peach Bottom 2 in 1977 and 1978. These stability tests were performed at low core flows (near minimum pump speed) and at varying core powers (up to the design reference condition). During Cycle 3, the tests were performed at various cycle exposures to evaluate the effects of fuel exposure on stability.

The test results verified that the small pressure perturbation technique provides a simple method for determining BWR reactor core stability margins. In addition, stability data were obtained at decay ratio conditions higher than those achieved in earlier control rod oscillator tests. Stability characteristics above the rated rod line at minimum pump speed were demonstrated with adequate margin to stability at all test conditions (maximum decay ratio = 0.5). Detailed descriptions of the Peach Bottom-2 stability tests during Cycles 2 and 3 can be found in References 3 and 4.

Success of the pressure perturbation technique used at Peach Bottom 2 and the desire for data close to the stability threshold led to stability tests at Vermont Yankee Nuclear Power Station in March 1981. The tests were performed before and after the first rod sequence exchange of fuel cycle 8. The stability tests were conducted at natural circulation flow, single-recirculation pump operation, at minimum pump speed, and two-pump operation at minimum pump speed. The core power was varied to points extending above the rated rod line.

Limit cycle oscillations of average neutron flux as measured by the Average Power Range Monitor (APRM) Subsystem were achieved at the intersection natural circulation and rated rod line without external pressure perturbations. Visual inspection of the control room APRM strip chart recordings showed that the amplitude of the APRM limit cycle oscillation could be distinguished from the normal APRM noise level. Thus, during this test occurrence of APRM limit cycle oscillations as the system stability approached limit cycle operation was observable in the control room through the regular instrumentation.

The APRMs and Local Power Range Monitors (LPRMs) oscillated in phase with a slight phase shift due to the time lag associated with fluid mass transport in the axial direction. No secondary effects of the limit cycle operation were noted and the oscillations remained bounded. The average operating conditions did not change, except for a slight power drift resulting from xenon burnout. The limit cycle oscillations were suppressed when a few control rods were inserted slightly. All other test conditions were stable including two points above the rated rod line at minimum recirculation pump speed. Reference 5 contains a detailed description of the tests and results.

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decay ratio = 1.0, limit cycle response is achieved, where the oscillations remain at a constant magnitude. Limit cycles are the characteristic response of nonlinear systems as they approach the stability threshold.

Recent stability tests at an overseas BWR plant have also demonstrated the occurrence of limit cycle neutron flux oscillations at natural circulation and several percent above the rated rod line. The oscillations were again observable on the APRMs and LPRMs and were suppressed by minimal control rod insertion. It was predicted that limit cycle oscillations would occur at the operating state tested; however, the characteristics of the observed oscillations were different from those previously observed in other stability tests. Examination of the detailed test data of these more recent tests showed that some LPRMs oscillated out of phase with the APRM, signal and at higher amplitudes than the core average. Although the regional oscillations were larger than the core average (6 to 7), margins to safety limits were maintained and the oscillations were detected and suppressed by control rod insertion.

2.3 GE Proposed Licensing Criteria

The final GE Proposed Licensing Criteria(12) were submitted in response to staff questions(11) and are as follow:

The compliance of the General Electric Company's Boiling Water Reactor (BWR) Systems, exclusively using GE BWR fuel designs, to the stability criteria set forth in GDC-12 has been demonstrated. The bounding fuel thermal-mechanical analyses cover all licensed GE BWR fuel designs including those contained in GESTAR through Amendment 10. Future GE BWR fuel designs will also be in compliance provided that the following stability compliance criteria calculated using approved methods are satisfied.

1. Neutron flux limit cycles, which oscillate up to the 120% APRM high neutron flux scram setpoint or up to the LPRM upscale alarm trip (without initiating scram) prior to operator mitigating action shall not result in exceeding specified acceptable fuel design limits (Safety Limit Minimum Critical Power Ratio and l% Cladding Plastic Strain).

2. The individual channels shall be designed and operated to be hydrodynamically stable or more stable than the reactor core for all expected operating conditions (analytically demonstrated).

These criteria will be evaluated on a generic fuel type basis for future fuel designs as they are added to GESTAR.

Because the stability compliance criteria are independent of plant specific characteristics, cycle-by-cycle decay ratios will not be evaluated for specific plants. However, the operational effects of introducing new fuel designs or special operating modes, will still be evaluated on a generic basis for representative NSSS product lines and fuel designs. The new fuel designs and extended operating modes will be evaluated using approved methods to determine their stability characteristics relative to current fuel designs (described above) which have demonstrated acceptable operational characteristics.

Based on the operating experience with current fuel designs, operator recommendations have been developed (SIL-380, Revision 1) (Reference 7) for high power density plants (e.g., BWR/4/5/6) which define a region of the operating map where operation is not recommended. In addition, a second region is defined in which increased monitoring of potential neutron flux oscillations is appropriate. If the stability performance of the new design is bounded by that of the current fuel designs then the plant performance is consistent with the basis for SIL-380, Revision 1 and these recommendations still apply. If the stability performance of the new design is not consistent with the current designs, then SIL-380 Revision 1 will be modified for that design such that the stability margins calculated at the boundaries of the monitored region will be maintained consistent with SIL-380 Revision 1.

3.0 STAFF EVALUATION

The staff has evaluated the GE proposed licensing criteria. This evaluation which is based on the input from two ORNL evaluation reports (References 8,9) and on numerous discussions with GE staff has resulted in the staff position stated in Section 4.0 of the SER. A summary of the ORNL TERs follows:

3.1 Review of General Electric Thermal-Hydraulic Stability Methodology (December 31, 1983)

In Reference 8, ORNL presents an evaluation of General Electric's methodology for calculating the stability of boiling water reactors for fuel reload licensing purposes. This evaluation is primarily based on comparative analysis of stability tests performed at Peach Bottom and Vermont Yankee versus results of GE's calculations for these tests.

ORNL compares decay ratios presented in a fuel reload submittal document with decay ratios both measured and recalculated at the end of cycle for that same fuel load. They also look at the impact that fitting procedures used by GE have on the numerical value determined from experimental data for the so-called measured decay ratio.

In this review ORNL concludes that a criterion specifying that the decay ratio (DR) shall be less than 0.8 should be set for GE's decay ratio calculations in fuel reload licensing submittals. If the 0.8 criterion is not met, a non-conformance region in the power-flow operating map must be defined; the reactor operator would be required to take a series of precautions to control the reactor within this region.

3.2 Evaluation of the Thermal-Hydraulic Stability Methodology Proposed by the General Electric Company, Part II (September 30, 1984)

Reference 9 contains ORNL's evaluation of the thermal-hydraulic stability methodology proposed by the General Electric (GE) Company to license reload fuel. The results of this evaluation complement the ones contained in the Reference 8 (Section 3.1) in which the capability of the General Electric Company to predict the stability of reload cores was evaluated.

The results of ORNL's initial review showed that calculated decay ratios are affected by two sources of error. One is input related because of the imprecision involved in calculating the operating conditions for which the stability will be a minimum during a fuel cycle. The other is related to core modeling, since it was shown that different decay ratios have been calculated for reactor core operating conditions which yielded equal experimental decay ratios. Based on the magnitude of the errors found in that review, ORNL proposed an acceptance criterion of decay ratio less than 0.8 for fuel reload calculations.

In NEDE-22277-P-1 (Reference 14) GE proposes two different approaches to demonstrate compliance with stability criteria for reload calculations:

Approach 1

Demonstrate that the calculated core and channel hydrodynamic decay ratio are less than 1.0 for all expected operating conditions.

Approach 2

This approach involves two steps:

(a) Demonstrate that each generic BWR fuel design satisfies the following compliance criteria:

(i) Neutron-flux limit cycles, which oscillate up to the 120% APRM high-neutron-flux scram setpoint (without initiating scram) shall not exceed specified acceptable fuel design limits

(ii) The individual channels shall be designed and operated to be hydrodynamically stable (decay ratio 1.0) or more stable than the reactor core for all expected operating conditions.

(b) Establish operator guidelines to terminate limit cycle oscillations.

The first approach was covered in ORNL's initial report (3.1), where ORNL recommends the threshold of 0.8 for decay ratio calculations to account for calculational uncertainties in predicting the 1.0 threshold proposed by GE. Reference 9 is related to the second approach.

The main points of this new GE proposal which need to be proven are whether:

(a) Neutron-flux limit cycles due to core-wide instabilities and oscillating up to 120% of rated average core power do not exceed current fuel design limits.

(b) The effects of limit cycles on fuel integrity can be calculated for generic fuel designs. This type of calculation is not necessary for every fuel reload.

(c) Local channel instability oscillations are not possible because the channels are designed and operated to be more stable than the core.

(d) If limit cycle oscillations occur the operator is capable of identifying and terminating them following the recommendations in SIL-380 Revision 1.

The results of the ORNL evaluation are:

(a) Core-wide limit cycles with the average power oscillating at frequencies greater than 0.25 Hz and up to 120% rated power are not likely to produce boiling transition and, thus, fuel integrity is likely to be maintained.

(b) The above result is applicable to generic fuel designs because these calculations depend mainly on the fuel geometry, and not on its neutronic characteristics.

(c) Local instabilities due to flow oscillations have been observed in recent experiments, and therefore, they are a possible phenomenon in BWR operation. In those experiments, the ratio of local to average power oscillations was a factor of five (i.e., the local power oscillated 60% while the average power oscillated only 12%) and the frequency of oscillation was close to 0.4 Hz. Assuming that this ratio and frequency remain approximately constant, our calculations show that boiling transition is not likely to occur even if the average power oscillates up to 120% of rated (i.e., the local power oscillates up

to 600% of rated).* Therefore, local instabilities can be considered by the same standard as the reactivity instability [result (a)].

(d) The operator recommendations contained in SIL-380, if properly implemented, are considered to be sufficient to identify and terminate limit cycle oscillations.

Based on these results, the following recommendations were proposed:

(a) Stability calculations must be performed for each fuel reload.

(b) If the calculations show that the decay ratio is less than 0.8 for all expected operating conditions during that cycle, the stability licensing criterion is met.

(c) If for some expected operating condition the decay ratio is greater than 0.8, then:

   (i) A nonconformance region should be determined in the power-flow operating map.

   (ii) A procedure should be established to make the operator aware of the possibility of oscillations in that operating region.

   (iii) Special operator instructions should be established to identify and terminate abnormal power oscillations should they occur.

   (iv) Calculations should be performed showing that limit cycle oscillations up to the 120% APRM-high-neutron-flux scram point plus anticipated transients (such as generator load rejection with bypass failure) do not reduce the critical power ratio (CPR) below the safety limit CPR for the particular fuel design. (Note: this calculation might be performed for a generic fuel type and plant design).

4.0 STAFF POSITION - ACCEPTANCE CRITERIA FOR GE BWR FUEL DESIGNS FOR THERMAL-HYDRAULIC STABILITY

The staff finds the GE fuel reloads bounded by the conditions in Table 1 meet the stability criteria set forth in General Design Criteria 10 and 12 provided that the BWR being reloaded has in place operating procedures and Technical Specifications which assure detection and suppression of global and local instabilities. Such detection and suppression should cover all modes of operation with particular emphasis on natural circulation and single loop operation. Fuel reloads meeting these requirements need not perform cycle specific stability calculations. Technical Specifications which enforce the recommendations of GE SIL-380 would meet these requirements.

* Staff comment — There is no proof or certainty that local/avg ratio is not higher than 6 to 1—in fact it has been observed to be as high as 7 to 1 in recent tests. Therefore, monitoring of local oscillations is a very important ingredient in proper stability monitoring procedures.

4.1 Exception to Acceptance Criteria for Plants Which Have Not Yet Implemented Improved Stability Technical Specifications

For GE reloads using Table 1 fuels in plants which have not yet implemented improved stability monitoring Technical Specifications the current practice of using the methods of NEDE-22277-P-1 to calculate a cycle specific decay ratio must be continued. This reload will be considered acceptable if the decay ratio is shown to be less than 0.80 for all possible operating conditions. BWR 2/3 type reactors using only the approved GE fuel types described in Table 1 have been shown to have adequate stability margins and therefore are acceptable and their reload cycles are exempted from the current requirement to submit a cycle specific stability analysis to the NRC.

4.2 New Fuel Designs

Should GE develop fuel designs in the future which exceed the bounds of Table 1 the prementioned acceptance criteria and exceptions may still be applied to such fuel if any of the following procedures are followed.

1. Show that the generic calculations presented in NEDE-22277-P-1 are applicable to the new fuel.

   or

2. Redo the generic calculations presented in NEDE-22277-P-1 in order to expand the approved bounds of Table 1 to include new fuel.

   or

3. Perform cycle specific calculations using the methods of NEDE-22277-P-1 and show the decay ratio to be less than 0.8.

1. Acceptable Fuel Types and Operating Conditions

Acceptable Fuel Types

All licensed GE BWR fuel designs contained in GESTAR (NEDE-24011-P-A-6 through Amendment 10).

e.g.

7x7

8x8

P8x8R

BP8x8P

GE8x8E

GE8x8EB

Acceptable Operating Condition

All licensed modes of operation in GESTAR (NEDE-24011-P-A-6 through Amendment 10).

e.g.

1. Standard power/flow map in FSAR
2. Operating Flexibility Options in GESTAR
   a. Load Line Limit (LLLA)
   b. Extended Load Line (ELLLA)
   c. Increased Core Flow (ICF)
   d. Single Loop Operation (SLO)
   e. Feedwater Temperature Reduction (FWTR)

Acceptable Exposure Range

Initial cycle to equilibrium cycle exposure for limits approved in GESTAR (NEDE-24011-P-A-6 through Amendment 10).

5.0 REFERENCES

1. “Stability and Dynamic Performance of the General Electric Boiling Water Reactor,” General Electric Company, Licensing Topical Report, January 1977 (NEDO-21506).

2. “General Electric Standard Application for Reactor Fuel,” General Electric Company Proprietary, April 1983 (NEDE-24011-P-A-6 and NEDE-24011-P-A Country Supplements).

3. L.A. Carmichael and R.O. Niemi, “Transient and Stability Tests at Peach Bottom Atomic Power Station Unit 2 End of Cycle 2,” Electric Power Research Institute, 1978 (EPRI NP-564).

4. F.B. Woffinden and R.O. Niemi, “Low Flow Stability Tests at Peach Bottom Atomic Power Station Unit 2 During Cycle 3,” Electric Power Research Institute, 1981 (EPRI NP-972).

5. S.F. Chen and R.O. Niemi, "Vermont Yankee Cycle 8 Stability and Recirculation Pump Trip Test Report," General Electric Company, March 1982 (NEDE-25445).

6. R.B. Linford, "Analytical Methods of Plant Transient Evaluations for the General Electric Boiling Water Reactor," General Electric Company, Licensing Topical Report, February 1973 (NEDO-10802).

7. "BWR Core Thermal-Hydraulic Stability," General Electric Company, February 1984 (Service Information Letter 380, Revision 1).

8. J.M. Leuba & P.J. Otaduy, "Review of General Electric Thermal-Hydraulic Stability Methodology", ORNL, December 31, 1983.

9. J.M. Leuba & P.J. Otaduy, "Evaluation of the Thermal-Hydraulic Stability Methodology Proposed by the General Electric Company", ORNL, September 30, 1984.

10. Letter, J.F. Quirk to C.O. Thomas, "Submittal of Proprietary Report on Compliance of GE BWR Fuel Designs to Stability Licensing Criteria" (NEDE-22277-P-1), dated November 6, 1984.

11. Letter, C.O. Thomas to H.C. Pfefferlen, "Request Number One for Additional Information on NEDE-24011," Rev. 6, Amendment 8, December 26, 1984.

12. Letter, H.C. Pfefferlen to C.O. Thomas, "Response to Request Number One for Additional Information on NEDE-24011," Rev. 6, Amendment 8, dated January 14, 1985.

13. R.B. Linford, "Analytical Methods of Plant Transient Evaluations for the General Electric Boiling Water Reactor," February 1973 (NEDO-10802).

14. G.A. Watford, "Compliance of the General Electric Boiling Water Reactor Fuel Designs to Stability Licensing Criteria", October 1984. (NEDE-22277-P-1).

NRC
Safety Evaluation Report
Approving Amendment 9
to
NEDE-24011-P

UNITED STATES
NUCLEAR REGULATORY COMMISSION
Washington, D. C. 20555
January 25, 1985

MFN 018-85

Ms. J.S. Charnley
Fuel Licensing Manager
Nuclear Safety and Licensing Operation
General Electric Company
175 Curtner Avenue
San Jose, California 95125

Dear Ms. Charnley:

SUBJECT: Acceptance for Referencing of Licensing Topical Report Amendment 9 to NEDE-24011, Revision 6, "GESTAR-II General Electric Standard Application For Reactor Fuel"

We have completed our review of the subject topical report submitted January 25, 1985, by General Electric (GE) letter. We find the report to be acceptable for referencing in license applications to the extent specified and under the limitations delineated in the report and the associated NRC evaluation, which is enclosed. The evaluation defines the basis for acceptance of the report.

We do not intend to repeat our review of the matters described in the report and found acceptable when the report appears as a reference in license applications, except to assure that the material presented is applicable to the specific plant involved. Our acceptance applies only to the matters described in the report.

In accordance with procedures established in NUREG-0390, it is requested that GE publish accepted versions of this report, proprietary and non-proprietary, within three months of receipt of this letter. The accepted versions shall incorporate this letter and the enclosed evaluation between the title page and the abstract. The accepted versions shall include an -A (designating accepted) following the report identification symbol.

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

Sincerely,

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

Enclosure:
As stated

ENCLOSURE

EVALUATION

Our evaluation of General Electric's proposed Amendment 9 to GESTAR II (NEDE-24011) is given below. A summary of each change, as stated by GE, is given first followed by our evaluation.

ITEM 9A: FUEL BUNDLE INFORMATION

These changes were made to NEDE-24011-P-A to update the fuel bundle information. This includes new fuel enrichments as well as correction of typos. Barrier bundles inadvertently left out have been included.

GE routinely reports details of bundle axial and radial enrichment loadings in GESTAR-II. These details are included in the individual core power distribution analyses and safety analyses which are performed as part of any reload or first core design. These figures are used by the staff during these reviews as an aid to qualitatively understand the cycle-to-cycle nuclear parameter variations that are encountered and the effects on the steady-state core physics and transient analyses and also to enable the staff to perform independent analyses when required. Since the trend of the GE fuel designs is to more complex bundle loading patterns, this information becomes more important.

We have reviewed the fuel bundle information submitted with the amendment and find it acceptable since the fuel designs have been generically approved and the modifications described in this amendment are reasonable variations of this design.

ITEM 9B: SPENT FUEL STORAGE CRITERIA

This change is incorporated into NEDE-24011-P-A to reflect that the reported fuel maximum exposure dependent K_∞'s are for the 20°C to 100°C temperature range.

The method used by GE, to generate the values of K_∞ for the table on pages 3-13, 3-14 and 3-15 of Amendment 9 has not been previously documented. This method was discussed with the staff during the course of this review and we find it to be acceptable. The approach consists basically of adding on a constant value to the K_∞ value calculated at 20°C in order to determine the most conservative K_∞ at different temperatures. GE should provide a detailed description of this method in the approved version of this amendment.

ITEM 9C: LATTICE K DATA

The lattice K_∞ data in NEDE-24011-P-A were updated to report the maximum exposure dependent lattice reactivity between 20°C and 100°C. This table was also updated to include all limiting lattices and to correct existing errors.

Since this change is a consequence of item 9B which is acceptable, it, too, is acceptable.

ITEM 9D: PLANT SPECIFIC FUEL INFORMATION

Appendix A of NEDE-24011-P-A was updated to include fuel information which is documented in the plant FSAR. The applicable GESTAR section is referenced eliminating the need to report these values in the FSAR. Tables S.1.1, S.2-3b, S.2-6, S.2-6a, and S.2-8 were updated to include current plant specific fuel information. Projects which have been canceled were eliminated from these tables.

These tables contain information necessary to perform various safety calculations. We have performed an audit review of the values in these tables and find that they agree with those in the plant FSARs. This change is acceptable.

ITEM 9E: CONTROL ROD DROP ACCIDENT ANALYSIS

These changes were made to NEDE-24011-P-A to reflect the current CRDA analysis. The cycle-specific CRDA analysis has been discontinued for BPWS plants based on the fact that in all cases the peak fuel enthalpy from a CRDA would be much less than the 280 cal/gm limit. This change has been internally approved by the NRC. Currently, the analysis performed for the group notch plants consists of two parts: a bounding analysis and, if needed, a plant-specific analysis. To simplify procedures, the bounding analysis has been eliminated and the plant-specific analysis will be performed each cycle.

As stated above, we have agreed with GE that a Rod Drop Accident analysis is not necessary for BPWS plants. This position is set forth in our SER (Reference 1).

Eliminating the bounding analysis for the group notch plants is acceptable since the plant specific will still be done.

ITEM 9F: MISLOCATED BUNDLE ACCIDENT ANALYSIS

This change was incorporated into NEDE-24011-P-A to provide additional information on the mislocated bundle accident analysis which is now being provided on initial core plants.

This change adds to GESTAR II the standard description of the mislocated bundle accident analysis which appears in FSARs for BWRs and is therefore acceptable.

ITEM 9G: PRESSURE RELIEF SYSTEM INFORMATION

These changes were made in Table S.2-4.1 of NEDE-24011-P-A to incorporate the Hatch 2 pressure relied system information and to update the Zimmer data. The projects which have been canceled were also removed from this table. The valve information given in Table S.2-4.2 for valves C and K has been updated to reflect current information.

GE states that these changes to information on valve flow rates and actuation pressures do not affect safety analyses; rather, the changes are incorporating values used in the safety analyses. We have conducted an audit review of these changes and agree with GE’s conclusion. Therefore, the change is acceptable.

ITEM 9H: BARRIER FUEL MAPLHGRs

The method for reporting barrier fuel MAPLHGRs is clarified.

GE states that no change has been made to the calculation of MAPLHGRs and that the change merely states that MAPLHGR curves for barrier and non-barrier fuel are calculated with the same codes. We have reviewed the possible differences between barrier and non-barrier fuel with respect to stored energy and LOCA analyses and find that GE's statement is acceptable. In particular, we examined the behavior of the barrier with respect to zirconium-water reaction following rupture of the cladding during a postulated LOCA and the temperature distribution in the fuel rod before and during a LOCA.

ITEM 9I: STABILITY 0F 8X8R, P8X8R, BP8X8R BUNDLES

This change was made to NEDE-24011-P-A to relate the stability to the 8X8R, P8X8R, and PB8X8R bundles.

GE's justification for this change is as follows:

There is no change in the stability analysis in 8X8R, P8X8R, BP8X8R bundles. This change merely states these bundles are equivalent in this analysis. This is consistent with all previous cycle-specific reload submittals, in which only one set of stability results are reported for 8X8R, P8X8R, and BP8X8R fuel.

We find this change to NEDE-24011 to be acceptable. The GE stability analysis is done in a conservative manner so that consideration of the variation in fuel design between these three fuel bundle types is not required.

ITEM 9J: ROD BOW DATA

This change was incorporated into NEDE-24011-P-A to reflect the NRC acceptance of the rod bow LTR (Reference 2).

GE states that the information incorporated into NEDE-24011 has been previously accepted by the staff (Reference 3). We have reviewed the proposed material and agree that no new information is presented.

ITEM 9K: RESPONSES TO NRC BARRIER QUESTIONS

This change incorporates the responses to NRC questions on the barrier fuel amendment into Appendix B of NEDE-24011-P-A.

GE states that this change incorporates the responses raised on the barrier fuel amendment, inadvertently left out of Appendix B.

Since the change is purely editorial, we find it acceptable.

ITEM 9L: RESPONSES TO NRC QUESTIONS ON GESSAR-II, SRP SECTION 4.2

These responses were incorporated into Appendix B of NEDE-24011-P-A to provide additional approved fuel information which has come up on plant dockets. They are referenced in the FSAR roadmap given in Appendix A. The actual SER on SRP Section 4.2 is incorporated into Appendix C. Incorporation of this approved information will facilitate NRC reviews of initial core FSARs which reference GESTAR II.

Since this information is entirely editorial, we find its addition to NEDE-24011 to be acceptable.

ITEM 9M: NRC LETTER ON NEDE-24011-P-A-4

This change was made to NEDE-24011-P-A to incorporate into Appendix C the NRC letter accepting Amendment 4.

GE states that the reason for this change is to incorporate the NRC letter accepting Amendment 4 of NEDE-24011-P-A, which was inadvertently left out of Appendix C.

Since this change is editorial, we find it to be acceptable.

ITEM 9N: EDITORIAL CHANGES

These proposed changes have been reviewed and we agree that they are only editorial and their addition to NEDE-24011 is acceptable.

REFERENCES

1. Letter from O. Parr, NRC, to G.G. Sherwood, General Electric Company, "Topical Report-NEDO-21251, Banked Position Withdrawal Sequence", January 18, 1978.

2. NEDE-24284, "Assessment of Fuel Rod Bowing in General Electric Boiling Water Reactors", August 1980.

3. Letter, C.O. Thomas (NRC) to J.F. Quirk (GE), "Acceptance for Referencing of Licensing Topical Report NEDE-24284-P, NEDO-24284," May 31, 1983.

NRC
Safety Evaluation Report
Approving Amendment 10
to
NEDE-24011-P

UNITED STATES
NUCLEAR REGULATORY COMMISSION
Washington, D. C. 20555
May 28, 1985

MFN 082-85

Ms. J.S. Charnley
Fuel Licensing Manager
General Electric Company
175 Curtner Avenue
San Jose, California 95125

Dear Ms. Charnley:

SUBJECT: Acceptance for Referencing of Licensing Topical Report NEDE-24011-P-A-6, Amendment 10, "General Electric Standard Application For Reactor Fuel"

We have completed our review of the subject topical report submitted by the General Electric Company (GE) letter dated November 30, 1984. We find the report to be acceptable for referencing in license applications to the extent specified and under the limitations delineated in the report and the associated NRC evaluation, which is enclosed. The evaluation defines the basis for acceptance of the report.

We do not intend to repeat our review of the matters described in the report and found acceptable when the report appears as a reference in license applications, except to assure that the material presented is applicable to the specific plant involved. Our acceptance applies only to the matters described in the report.

In accordance with procedures established in NUREG-0390, it is requested that GE publish accepted versions of this report, proprietary and non-proprietary, within three months of receipt of this letter. The accepted versions shall incorporate this letter and the enclosed evaluation between the title page and the abstract. The accepted versions shall include an -A (designating accepted) following the report identification symbol.

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

Sincerely,

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

Enclosure:
As stated

ENCLOSURE

SAFETY EVALUATION ON AMENDMENT 10 TO GESTAR II (NEDE-24011-P)

1.0 INTRODUCTION

By letter dated November 30, 1984 (Reference 1) General Electric Company submitted Amendment 10 to GESTAR II. The submittal of Amendment 10 by General Electric introduces two new fuel designs designated GE8x8E and GE8x8EB. Both designs are intended for irradiation to extended burnup levels with the latter incorporating a thin zirconium barrier (liner) to the inner diameter of the cladding. This submittal relies on the criteria and methods submitted in Amendment 7 (Reference 2) and the GESTR-LOCA documentation (Reference 3) to provide assurance that these two new designs are safe for operation in boiling water reactors. Because the review of GE's extended burnup topical report (Reference 4) is not complete many of the design bases/criteria and analysis methods reviewed in this submittal have not been approved for application to extended burnup levels. The exception to this is the GESTR-LOCA code (Reference 3) which has been approved for extended burnup. Consequently, the approval of the GE8x8E and GE8x8EB designs for operation to extended burnup levels is contingent on NRC approval of GE's extended burnup topical report (Reference 4). It should be noted, however, that the review of Reference 4 is nearly complete and there are no outstanding issues at this time. Our review, documented in this safety evaluation report, covers the GE8x8E and GE8x8EB fuel designs to those burnups to which BWRs with GE fuel are presently operating.

This review and safety evaluation will follow the Standard Review Plan (SRP) (Reference 5) to ensure that all licensing requirements of the fuel system are addressed with respect to these new fuel designs. Amendment 7 to NEDE-24011-P-A (which was approved by the staff in References 6 and 7) has updated GESTAR II to follow the format of Section 4.2 of the Standard Review Plan and was used extensively in this review.

Those portions of the review which address Section 4.2 of the Standard Review Plan are given in this Safety Evaluation Report in Sections 2.0 through 7.0. Section 8.0 addresses the nuclear design of these new fuel bundles. Section 9.0 addresses the thermal hydraulic design of these fuel bundles. Our conclusions are given in Section 10.0.

2.0 FUEL SYSTEM DESIGN

The GE8x8E and GE8x8EB fuel designs are modifications of the GE P8x8R and BP8x8R designs. The modifications are to aspects of the fuel rod design to enable the fuel bundles to attain higher burnups and also to the design of the bundle upper tie plate. The new extended burnup design features described in Amendment 10 include changes to the peak linear heat generation rate, plenum volume, helium fill gas pressures, fuel density, fuel/cladding gap, different enrichments, additional water rods and gadolinium rod changes. Some of these changes are reflected in Table 2-1c and 2-1d and Figure 2-2 of Amendment 10.

The objectives of this fuel system safety review, as described in Section 4.2 of the SRP, are to provide assurance that as a result of the new design features (a) the fuel system is not damaged as a result of normal operation and anticipated operational occurrences (AOOs), (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. These objectives implement General Design Criterion (GDC) 10 of 10CFRPart 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 fuel rod leaks and that the first fission product barrier (the cladding) has, therefore, been breached. Fuel rod failure must be accounted for in the dose analysis to show compliance with the offsite dose limits of 10CFRPart 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 an accident. The general requirements to maintain control rod insertability and core coolability appear in the General Design Criteria (e.g., GDC 27 and 35). Specific coolability requirements for the loss-of-coolant accidents are given in 10CFRPart 50.46, “Acceptance Criteria for Emergency Core Cooling Systems for Light Water Nuclear Power Reactors”.

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.

Under the heading for each SAFDL, there is a Bases/Criteria section and an Evaluation section. The criteria sections address the limiting values that have been submitted by General Electric in Amendment 7 under the three major categories of failure mechanisms listed above. It is the purpose of this review to determine if these criteria are applicable and acceptable for the GE8x8E and GE8x8EB designs submitted in Amendment 10. These criteria along with certain definitions for fuel failure constitute the SAFDLS required by GDC 10.

The evaluation sections review the methods that General Electric used to demonstrate that the design criteria have been met for these designs. These methods may include operational experience, prototype testing and analysis models. In addition, the new features of this design are reviewed with respect to each damage mechanism to determine if they could have an adverse impact on fuel system performance.

3.0 FUEL SYSTEM DAMAGE

The design criteria in this section should not be exceeded during normal operation including anticipated operational occurrences (AOOs). The evaluation portion for 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 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 (Reference 8), 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 (Reference 9) calls for only 70% of the same quantity.

GE has demonstrated, in response to our questions (Reference 10) during the review of Amendment 7, 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 (Reference 6). GE has indicated that the same materials and material fabrication procedures are used for the GE8x8E and GE8x8EB assemblies as those used for past 8x8 designs. Consequently, this criterion is also found to be acceptable for the GE8x8E and GE8x8EB designs.

The effects of extended burnup operation on cladding ductility and how this may affect the above criterion will be addressed in the review of Reference 4.

Evaluation

GE has determined the design ratio distributions for the GE8x8E and GE8x8EB designs, and has stated (Reference 1) that the design criterion is met. Therefore, the new fuel designs are acceptable with respect to stress and strain limits.

(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 (Reference 2). The GESTR-MECHANICAL code (Reference 11) 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 the distribution of fatigue usage is required to be less than 1.0. This methodology accounts for internal and external pressures, cladding temperatures and pellet-cladding contact. The use of this fatigue usage limit and method has been found to be conservative and acceptable by the NRC (References 6, 7, and 12). We also find the limit and method to be applicable to the GE8x8E and GE8x8EB designs.

Evaluation

GE has determined (Reference 1) that the 95% upper tolerance limit of the fatigue usage distributions for the GE8x8E and GE8x8EB designs meet the above criterion and thus these fuel designs are acceptable.

(c) Fretting Wear

Bases/Criteria

In our Safety Evaluation Report on Amendment 7 to GESTAR II (Reference 6) which dealt with fuel criteria, we 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.

Since the SRP does not provide numerical acceptance criteria for fretting wear, and since fretting wear is addressed in the design analysis for each bundle design, the NRC staff concluded in Reference 6 that the intent of the SRP has been adequately met.

Evaluation

In the current GE BWR P8x8R and BP8x8R fuel design individual rods in the fuel assembly are held in position 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. The same design is employed by GE in the GE8x8E and GE8x8EB fuel designs. Various in-pile and out-of-pile tests were described in Section 2.6.3 of NEDE-24011-P-A-6 along with the results of a continuing fuel surveillance program that had utilized nondestructive methods including eddy current measurements to locate discontinuities in the cladding and detailed visual examinations to characterize the nature of defects. As stated in NEDE-24011-P-A-6, no significant fretting wear had been observed. GE concluded that significant fretting wear was avoided through the use of an active spring force to eliminate any clearance that would otherwise exist between the spacer structure and the fuel rods. Based on the previous good experience of BWR fuel with respect to fretting wear, and the fact that the Amendment 10 fuel designs are identical with respect to those parameters important to fretting wear (such as spring force and cladding and grid dimensions) we find the GE8x8E and GE8x8EB designs acceptable with respect to fretting wear up to current BWR burnups. An evaluation of fretting wear at high burnup will be addressed in our safety evaluation on Reference 4.

In addition to the fuel rods and spacers, there is another fuel system component whose functionality must be assured as an objective of the review of BWR fuel system fretting concerns, viz., the fuel assembly channel box (References 13 and 14). Since the channel box design will not change with use of these new fuel designs, our evaluation, given in Reference 15, applies to the use of current channel box designs with this new fuel.

(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 metal thinning due to cladding oxidation and the cladding temperature increase due to the buildup of corrosion products, 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 (Reference 3) and GESTR-MECHANICAL (Reference 11) codes. Because the SRP

does not provide numerical limits on cladding oxidation and since GE includes cladding oxidation and crud effects in their analyses, it is concluded that GE's approach is consistent with the SRP guidelines and applicable to the GE8x8E and GE8x8EB fuel designs.

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 safety evaluation (Reference 6) of Amendment 7 and the application of the methodology to high burnup will be addressed further in the safety evaluation of GE's extended burnup topical report (Reference 4). We find the current GE methods applicable to the new fuel designs for the burnup range of current BWR fuel.

(e) Rod Bow and Channel Box Deflection

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 our SER on GESTAR II Amendment 7 we stated that we could not find this criterion acceptable since GE had not presented sufficient evidence that there would not be a reduction in the critical power ratio for significant amounts of bowing. However, we also found no reason to change the existing position on the effect of fuel rod bowing on critical power ratio. This position is that no reduction in critical power ratio operating limits is required 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 applicable to the GE8x8E and GE8x8EB fuel designs.

The design bases for channel box deflection are described in the GE topical report NEDE-21354-P. We found these bases acceptable in Reference 6. GE has not proposed any change to channel box design in Amendment 10.

Evaluation

GE has submitted a generic topical report on fuel rod bowing (Reference 16). The NRC reviewed this report and performed an independent assessment of rod bow and concluded that significant rod bow is not expected in GE BWR designs (Reference 17).

During our review of Amendment 10, we requested GE to describe the possible effects that the higher enrichments and heat rating proposed for these two new fuel designs may have on fuel rod bowing and channel box deflections. GE's response (Reference 18) has indicated that no change in rod bowing or channel box deflection characteristics is expected. This is due to the fact that both axial and transverse fast flux/fluence distributions are not expected to change in the new fuel designs. In addition, GE has calculated that the difference in maximum cladding average temperature is less than 5°C between the new and previous designs. The effect of extended burnup on rod bowing will be discussed further in the safety evaluation of Reference 4; however, it is anticipated that no further restrictions with respect to GE rod bow will be required for extended burnup operation. Consequently, the current methodology and requirements including the 50% rod bowing reporting requirement for GE rod bow stated above

are found to be applicable to the GE8x8E and GE8x8EB designs. These designs are therefore acceptable with respect to fuel rod bowing.

(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 either Amendment 7 or Amendment 10; however, in response to Question 1.3 of this review GE has stated (Reference 18) that the expansion spacing is sized to provide a reasonable assurance that bottoming out of the expansion spring will not occur. This criterion is found to meet the intent of the SRP.

Evaluation

Current irradiation experience with General Electric fuel indicates that axial gap closure is not a problem (Reference 30). GE has also indicated that no metallurgical changes to the fuel or tie rod cladding material have been made so that the data are applicable to the GE8x8E and GE8x8EB designs. This methodology is found to be acceptable for these new designs. The applicability of these data and methods for extended burnups will be addressed in the review of Reference 4.

(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. This is because the criterion for fuel rod internal pressure involves more than the cladding mechanical limits. There are a number of reasons for this distinction: (a) outward (tensile) cladding stresses may force analytical methods into a mode for which they were not designed or where greater uncertainties exist, (b) the higher releases of intended fission gas from the fuel associated with higher rod pressures may lead to a level of undesirable positive thermal feedback, (c) the higher releases of fission gas associated with higher rod pressures may lead to underestimating the radiological consequences of accidents using release assumptions of Regulatory Guides 1.25 and 1.77, and (d) net positive pressures could lead to ballooning during non-LOCA boiling transition events and such behavior is not considered in the accident analyses.

In order to simplify the analysis of fuel system damage due to excessive rod internal pressure, the SRP states that rod internal pressures 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 items (a) through (d)

described above and this has been found acceptable in Reference 6. This criterion is also found acceptable for the GE8x8E and GE8x8EB designs.

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 to be acceptable in Reference 6 and is also found to be acceptable for the GE8x8E and GE8x8EB designs.

The power history used as input for the rod internal pressure calculations is being addressed in the safety evaluation of the extended burnup topical report (Reference 4). In response to a question from the review of Reference 4 GE has indicated that the Maximum Linear Heat Generation Rate (MLHGR) limit is used for the fuel design analysis in question. This MLHGR limit represents the bounding power possible for a particular design and burnup. Although this review is not complete this power history is bounding and thus conservative, and is acceptable in the range of burnups for current BWR fuel. GE has presented analysis results (Reference 19) based on calculations with the GESTR-MECHANICAL code and the MLHGR power history that show the GE8x8E and GE8x8EB designs are significantly below the above criterion.

(h) Fuel Assembly Liftoff

Bases/Criteria

The SRP calls for the fuel assembly holddown capability (wet 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. This amendment references an approved report (Reference 20) that describes a liftoff limit to prevent control blade interference. This limit is found to be applicable to the GE8x8E and GE8x8EB fuel designs and it is therefore concluded that this design criterion is acceptable.

Evaluation

Amendment 10 references an evaluation (Reference 21) that GE states is applicable to the GE8x8E and GE8x8EB for BWR/2, 3 and 4 plants which concludes that the fuel assemblies can withstand worst case LOCA and seismic loadings without assembly lift as well as normal operation. For BWR/5 and 6 designs, these evaluations are plant specific. The LOCA and seismic loads are found to bound possible loads from normal operation and AOOs and thus these evaluations are found to be acceptable for the GE8x8E and GE8x8EB designs.

(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 6 for a discussion of this topic.

4.0 FUEL ROD FAILURE

In the following paragraphs, GE fuel rod failure thresholds and analyses for the failure mechanisms listed in the Standard Review Plan and Amendment 7 are reviewed with respect to the GE8x8E and GE8x8E3 designs submitted in Amendment 10. 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 required by 10CFR100.

(a) Hydriding

Bases/Criteria

Internal hydriding as a cladding failure mechanism is precluded by controlling the level of hydrogen impurities during fabrication. To ensure that failure will not occur due to internal cladding hydriding, GE specifies a fabrication limit on hydrogen content, which is less than or equal to the limit stated in SRP Section 4.2.II.A.2(a), i.e., 2 ppm hydrogen for a UO2 pellet.

Since GE uses a hydrogen limit less than or equal to that stated in the SRP we have concluded in Reference 6 that the design limit for hydriding is acceptable and this is also found to be acceptable for the GE8x8E and GE8x8EB designs.

Evaluation

In addition to the manufacturing controls on moisture and hydrogen impurities GE has introduced a hydrogen getter in their fuel designs as an additional assurance against internal hydriding. GE in-reactor experience has shown that internal hydriding has not been an active failure mechanism for fuel manufactured since mid-1972. Based on this information we have concluded (Reference 6) that internal hydriding is not a problem for GE fuel and this is also found applicable to the GE8x8E and GE8x8EB designs since the same design and manufacturing controls will apply. Internal hydriding with respect to extended burnup operation will be discussed further in the safety evaluation of Reference 4.

(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 operational conditions of the reactor, GE has adopted (Reference 22) 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 (Reference 22) that contains these limits and definitions has been reviewed and approved by NRC (Reference 22). These limits are also applicable to the GE8x8E and GE8x8EB designs.

Evaluation

GE has indicated (Reference 1) that cladding collapse is not calculated to occur for the GE8x8E and GE8x8EB designs using the approved models for cladding collapse (Reference 22).

(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), which corresponds to the most limiting fuel assembly in the reactor core. 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 (Reference 23) 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 (Reference 6). These design limits are also found to be applicable and acceptable to the GE8x8E and GE8x8EB designs.

Evaluation

See Section 9.0 for a discussion of the applicability of GE's thermal margin calculations to GE8x8E and GE8x8EB fuel.

(d) Overheating of Fuel Pellets

Bases/Criteria

GE presented the bases and criteria for overheating of fuel pellets in Amendment 7 to GESTAR II (Reference 2). We approved these bases and criteria in our SER on Amendment 7 (Reference 6). 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.

Evaluation

GE has stated that the above bases and criteria are met for the GE8x8E and GE8x8EB fuel. The GESTR-MECHANICAL code is used to perform the analyses. We find this acceptable. Further discussion of overheating of pellets at extended burnup will be discussed in our review of Reference 4.

Gadolinia is mixed in with the UO2 in some fuel rods to act as a burnable poison for reactivity control. This gadolinia lowers the melting point and thermal conductivity of the fuel.

GE currently limits the concentration of gadolinia to 6%. We have 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 higher gadolinia

concentrations. GE has proposed and the staff has accepted (Reference 24) a program to provide this verification.

(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 ~ the radially averaged fuel rod enthalpy is greater than 170 cal/g at any axial location. The 170 cal/g enthalpy criterion, developed from SPERT tests (Reference 25), 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 26, GE uses 170 cal/g as a cladding failure threshold. This has been approved by the NRC (Reference 6). The applicability of this fuel enthalpy limit to extended burnup fuel will be discussed in the review of Reference 4. We find this criterion to be applicable to the GE8x8E and GE8x8EB fuel designs.

Evaluation

Bounding analyses of the Rod Drop Accident are reported in GESTAR II. If these analyses are found not to be bounding for operation during a certain cycle, a cycle-specific analysis is performed. The above criterion is applicable to the GE8x8E and GE8x8EB fuel designs.

(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 (References 1 and 2) 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 by the NRC (Reference 6). This strain limit is also judged to be acceptable for the GE8x8E and GE8x8EB designs.

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 (Reference 27). 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 (Reference 6).

Evaluation

The GESTR-MECHANICAL code is used by GE to determine that their fuel designs meet the above 1% cladding strain criterion. This code has been approved for this analysis by the NRC (Reference 6). GE has stated (Reference 1) that the calculated plastic strains for the GE8x8E and GE8x8EB designs are less than the 1% strain criterion using the GESTR-MECHANICAL code at the maximum power and exposure conditions expected as a result of AOOs.

(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 10CFR50 Appendix K requirement that the degree of swelling not be underestimated.

Evaluation

GE uses an empirical rupture-temperature correlation (Reference 28) which has been updated to include the data and models from Reference 29 as a part of the LOCA emergency core cooling system (ECCS) analysis. This correlation has been reviewed and approved in Reference 38.

The design changes made as a result of the GE8x8E and GE8x8EB designs have not changed the applicability of this rupture-temperature correlation and thus the correlation is approved for application with respect to these designs. The applicability of this correlation and other LOCA models to extended burnup fuel will be addressed in the review of Reference 4.

(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 hydraulic load or a load derived from core-plate motion. These loads are bounded by the loads of a LOCA and safe-shutdown earthquake (SSE), and the mechanical fracturing analysis is usually done as a part of the seismic-and-LOCA loads analysis (see Section 5.0(d) of this SER). The entire seismic-and-LOCA loads evaluation (including design limits) has been described by GE in a topical report that has been approved by the NRC (Reference 20).

Evaluation

The discussion of the seismic-and-LOCA loading analysis is given in Section 5.0(d) of this SER.

5.0 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 for the severe damage mechanisms listed in the Standard Review Plan are reviewed.

(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 (References 28 and 31) acceptance criteria of 2200°F on peak cladding temperature and 17% on maximum cladding oxidation as prescribed by 10CFR50.46. These criteria will be applied to the GE8x8E and GE8x8EB designs.

Evaluation

Since General Electric's methods explicitly account for design changes in the fuel, and the GE8x8E and GE8x8EB fuel designs are within the ranges of all the empirical relationships of the GE LOCA calculational methods, we find the GE LOCA methods acceptable for use with these new fuel designs. Results of these analyses are reported on a plant-specific basis.

(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 26 and 31, GE employs this 280 cal/g criterion as a control rod drop accident design limit and thus is consistent with the SRP. The applicability of this limit to extended burnup fuel will be addressed in the review of Reference 4.

Evaluation

Bounding analyses of the Rod Drop Accident are reported in GESTAR II. These analyses employ methods which are applicable to the GE8x8E and GE8x8EB fuel designs. If these analyses are not found to be bounding for operation during a certain cycle, a cycle specific analysis is performed. The above criterion is used in all cases.

(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 10CFR50 Appendix K requirement that the degree of swelling not be underestimated.

Evaluation

The GE cladding ballooning model for ECCS analysis is addressed in the approved report NEDE-20566-P (Reference 28). This report adopted the NUREG-0630 data base and modeling, which specifies a method acceptable to the NRC for treating cladding swelling and rupture during a LOCA. Amendments 7 and 10 refer to Section S.2.5.2.1.4 in NEDE-24011-P-A-6 (Reference 31) which references this report.

The design changes made as a result of the GE8x8E and GE8x8EB designs have not reduced the applicability of these methods and thus are approved for application with respect to these designs. The application of the methods and models used in the LOCA analysis for extended burnup fuel will be addressed in the review of Reference 4.

(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 seismic-and-LOCA loads evaluation (including design limits) has been described by GE in an approved topical report (Reference 20) to which Amendments 7 and 10 make reference. These design limits are found to be applicable to the GE8x8E and GE8x8EB designs.

Evaluation

Generic analysis methods for performing combined seismic-and-LOCA loading analyses have been described by GE in an approved topical report (Reference 20). 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 seismic-and-LOCA events as prescribed by Appendix A to SRP 4.2.

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 proposing to reference the subject document (NEDE-24011-P, Amendment 10) must perform site specific analyses using Reference 20 analysis methods in order to address the above criteria and Appendix A to SRP Section 4.2 guidelines.

6.0 DESCRIPTION and DESIGN DRAWINGS

The fuel assembly design description and drawings are provided in Section 2.1, Tables 2-1a through d, and Figures 2-1a and 2-2 of the Amendment 10 submittal. This design description is considered to be minimal; however, through presentations made by GE (Reference 19) and responses (Reference 18) to NRC questions the Amendment 10 descriptions of the GE8x8E and GE8x8EB designs are found to be acceptable.

7.0 TESTING, INSPECTION, AND SURVEILLANCE PLANS

7.1 Testing and Inspection of New Fuel

Section 2.3.1 in Amendment 7 to GESTAR II and NEDO-11209-04A contain descriptions of (1) the type of quality control inspections performed during manufacturing and (2) the plan for on-site inspection and testing of new fuel assemblies. We concluded in Reference 6 that the new fuel testing and inspection program for GESTAR II is acceptable, and we find it to be applicable to the GE8x8E and GE8x8EB fuel designs.

7.2 On-Line Fuel System Monitoring

This topic is independent of the fuel design and was not addressed during this review.

7.3 Post-irradiation Surveillance

In a June 27, 1984 letter (Reference 32) the staff accepted a GE proposal for post irradiation surveillance which, in part, pertained to new fuel designs. This letter stated, in part: “For new fuel designs, GE will conduct a general visual examination of the exterior surfaces of a statistically meaningful number of fuel bundles upon discharge. This examination will be conducted at two applications of the new design from the first year in which the design is introduced.”

GE has confirmed that this surveillance program will be used for GE8x8E and GE8x8EB fuel designs.

8.0 Nuclear Design

GESTAR II discusses the analytical methods used by General Electric Company to perform nuclear design and safety calculations. Since the new fuel bundles do not involve extending the ranges of the various nuclear parameters involved in core nuclear design and safety analysis, we find the use of GE’s nuclear methods acceptable for the GE8x8E and GE8x8EB fuel designs to the range of burnups currently used with GE fuel. The use of nuclear analysis methods for extended burnup will be discussed in the staff SER on Reference 4.

9.0 Thermal-Hydraulic Evaluation

9.1 Applicability of the GEXL Correlation for GE 8x8E/GE 8x8EB Fuel Design

In Section 5.2.1.1 of Amendment 10 (Reference 1), GE states that the GEXL correlation (Reference 23) and its uncertainty limits as previously approved by the NRC for the 8x8 (with one water rod) and 8x8R (with two water rods) fuel designs will be used to predict the critical power for the GE8x8E and GE8x8EB fuel designs. Since the GE8x8E and GE8x8EB fuel bundles will contain several features which could alter the bundle power distribution from that of the previous GE bundle designs for which GEXL was approved, the staff requested GE to provide justification for the continued use of GEXL for the new bundle designs.

In response, GE stated in Reference 18 that the major thermal-hydraulic differences between the 8x8R and the new GE fuel designs are (1) installation of two to six fuel rod size water rods replacing fuel rods and (2) use of an improved upper tie plate design. GE also indicated that only additional water rods may affect the use of the GEXL correlation since the upper tie plate design is not included in the formulation of GEXL. We agree with this conclusion. To confirm the validity of the continued use of GEXL for the GE8x8E and GE8x8EB fuel designs, GE provided comparisons of GEXL predictions with data for 8x8 fuel bundles with four and sixteen water rods.

The staff reviewed the results of these data comparisons and agreed with GE’s conclusion that use of GEXL is appropriate for the fuel bundle configurations for which GE presented data (four water rods) since the measured values of critical power are adequately predicted by GEXL for the data presented. However, no data comparisons for the fuel bundles with other numbers of water rods were presented. Since GE intends to apply GEXL to fuel bundles with up to eight water rods, the staff requested GE to justify that the effect of various numbers of water rods on the critical power calculation can be adequately predicted by the GEXL correlation. Since the effect of the number of water rods would be included in the GEXL R factor which accounts for fuel bundle local power distributions, we requested GE to submit data showing that, for different local power distributions but the same R factor, the same critical power would be obtained within an acceptable uncertainty band.

In response, GE showed in Reference 33 that (1) the experimental critical power data are within the measurement error bound for two and four water rod test bundles with approximately equal R factors, (2) GEXL compares well with an 8 x 8 fuel bundle configuration with 16 water rods, and that (3) the GEXL prediction is generally conservative as compared with this experimental critical power data (although conservatism in GEXL is not required).

In response to a staff question regarding the applicability of the uncertainties included in GETAB (Reference 18) for the GE8x8E and GE8x8EB fuel designs, GE stated that use of the uncertainties in GETAB will be directly applicable or conservative for the GE new fuel designs since (1) the methodology used to calculate the bundle pressure drop, which affects the bundle flow, remains unchanged, (2) the use of more accurate lattice physics methods and more accurate core instruments will result in smaller uncertainties for the R factor and Traversing-Incore-Probe measurements, respectively, and (3) the higher enrichment of the GE8x8E and GE8x8EB fuel will result in a more peaked power distribution, resulting in less fuel bundles being near limits and use of the more peaked power distribution will therefore result in a larger margin to the safety limit.

While these arguments appear plausible, GE has not presented any data or calculations to quantitatively support these positions. However, we agree with the GE conclusion that, based on the above arguments, the GE8x8E and GE8x8EB fuel bundles should not have higher uncertainties than the present GE fuel bundles. Therefore, the use of these arguments to justify that present GETAB uncertainties for the GE8x8E and GE8x8EB fuel designs is valid is acceptable but no quantitative credit may be taken for these differences.

The staff has reviewed the test data base (References 18 and 33) provided by GE to support the use of GEXL for determination of critical power for BWR cores incorporating the GE8x8E and GE8x8EB fuel designs. The staff finds that (1) the GEXL correlation adequately predicts critical power for the GE8x8E and GE8x8EB fuel designs, and (2) the use of uncertainties considered in GETAB is appropriate for the GE8x8E and GE8x8EB fuel designs. Therefore, we conclude that the GEXL correlation and uncertainties in GETAB are acceptable for the GE8x8E and GE8x8EB fuel designs.

9.2 LOCA Analysis of GE8x8E and GE8x8EB Fuel Designs

In Attachment 1 of Amendment 10 to NEDE-24011, GE states that the loss-of-coolant-accident (LOCA) analysis of GE8x8E/GE8x8EB will be performed using either the approved SAFER/GESTR code (Reference 3) or the current SAFE/REFLOOD procedure (Reference 28).

As for the use of the SAFE/REFLOOD procedure, the methodology used is generically applicable for the MAPLHGR limit determination, but the staff was concerned that the effects of enhanced fission gas release at higher burnup (i.e., greater than 20 MWD/kgU) were not adequately considered in the fuel performance model. In response to this concern, GE requested (References 34 and 35) the credit for approved, but unapplied, ECCS evaluation model changes and calculated peak cladding temperature margin to avoid MAPLHGR penalties at higher burnups. This request was approved (Reference 36) for the present GE fuel designs. The staff finds that this staff position is also applicable to the GE8x8E and GE8x8EB fuel designs since (1) a plant specific analysis for determination of MAPLHGR limits will be performed with approved methods for the specific cycle of operation and (2) GE (Reference 19) shows that the maximum fuel temperature for the GE8x8E and GE8x8EB fuel decreases with increasing exposures (for burnups greater than 30 MWD/kgU) and is less than that for the P8x8R and BP8x8R fuel designs.

9.3 Thermal Hydraulic Stability

The staff has completed the generic review related to the thermal hydraulic stability for BWR cores. In the evaluation report (Reference 37), the staff concludes that GE fuel reloads (including those with GE8x8E and GE8x8EB fuel) meet the stability criteria set forth in General Design Criteria 10 and 12 provided that the BWR has in place operating procedures and Technical Specifications which are consistent with the recommendations of GE SIL-380, to assure detection and suppression of global and local instabilities. For the reload core without the appropriate Technical Specifications to monitor core stability, the current procedure of using the methods of NEDE 22277-P-1 to calculate a cycle specific decay ratio should be continued. The reload will be considered acceptable if the decay ratio is shown to be less than 0.80 for all possible operating conditions. BWR 2/3 type reactors using the approved GE fuel types (including GE8x8E and GE8x8EB) have been shown to have adequate stability margins and, therefore, are acceptable and their reload cycles are exempted from the current requirement to submit a cycle specific stability analysis to the NRC.

9.4 Upper Tie Plate

The GE8x8E and GE8x8EB fuel designs are provided with a new upper tie plate with different flow characteristics than the upper tie plates for the P8x8R and BP8x8R designs. This will affect thermal margin and stability of the bundles, but since the difference in flow characteristics can be treated explicitly in calculations showing compliance with the pertinent SAFDL's; we find this change to be acceptable.

10.0 Conclusions

The GE8x8E and GE8x8EB fuel system designs as described in Amendment 10 of NEDE-24011-P have been reviewed in accordance with the SRP Section 4.2. As a result of our review, we conclude that the use of the GE8x8E and GE8x8EB fuel designs is acceptable for all BWRs.

The issues to be addressed by applicants are: (1) BWR/5 and 6 plant-specific analysis of combined seismic-and-LOCA loading using the GE methods (Reference 20) approved by the NRC or another acceptable method to demonstrate conformance to the structural acceptance guidelines described in Appendix A to SRP 4.2 (see Sections 4.0(h) and 5.0(d) of this report) and (2) an acceptable post-irradiation surveillance must be provided or an endorsement of the approved GE fuel surveillance program (see Section 7.0 of this report).

With the above provisions, we conclude that the GE8x8E and GE8x8EB fuel systems described in Amendment 10 have been designed such that (1) they will not be damaged 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:

General Electric has provided sufficient evidence that the design objectives will be met based on operating experience, prototype testing, and analytical predictions.

General Electric has provided for testing and inspection of new fuel to ensure that it is within design tolerances at the time of core loading. The licensee will perform on-line fuel failure monitoring and post-irradiation surveillance to detect anomalies or confirm that the fuel has performed as expected.

As stated in various parts of this safety evaluation, our approval for extended burnup must await completion of our review of NEDE-22148-P, "Extended Burnup Methodology" (Reference 4) which

we expect to finish shortly. We find the GE8x8E and GE8x8EB acceptable for irradiation to present GE BWR burnup levels.

11.0 REFERENCES

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

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

3. S.O. Akerlund, et al., The GESTR-LOCA and SAFER Models for The Evaluation of the Loss-of-Coolant Accident, Volume I: GESTAR-LOCA A Model for the Prediction of Fuel Rod Thermal Performance, NEDE-23785-1-P (Proprietary), General Electric Company, December 1981.

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5. Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants—LWR Edition, NUREG/0800, Section 4.2, “Fuel System Design,” Rev. 2, July 1981.

6. 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.

7. 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; SER Page Changes for Clarification”, May 9, 1985.

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12. Letter from Thomas A. Ippolito, NRC, to Richard Gridley, GE, April 16, 1979.

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

19. Letter from J.S. Charnley, GE, to R. Lobel, NRC, “Presentation on GE8x8E and GE8x8EB Fuel Designs,” November 14, 1984. (Proprietary)

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37. Letter from L.S. Rubenstein to D. Crutchfield, "Safety Evaluation of GE Topical Report NEDE-24011 (GESTAR) Amendment 8", dated April 17, 1985.

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
Washington, D. C. 20555
December 3, 1985

MFN 148-85

Ms. J.S. Charnley, Manager
Fuel Licensing
General Electric Company
175 Curtner Avenue
San Jose, California 95125

Dear Ms. Charnley:

SUBJECT: Acceptance for Approval of Fuel Designs Described In Licensing Topical Report NEDE-24011-P-A-6, Amendment 10 For Extended Burnup Operation

In a letter dated May 28, 1985 from C. O. Thomas, NRC, to J. S. Charnley, GE, the staff transmitted the results of our review of the GE8x8E and GE8x8EB fuel designs described in NEDE-24011-P-A-6, Amendment 10. In our safety evaluation we stated that the approval of these fuel designs for extended burnup must await completion of our review of NEDE-22148-P, "Etended Burnup Methodology" and found the GE8x8E and the GE8x8EB fuel designs acceptable for irradiation to present GE burnup levels. Our review of NEDE-22148-P is for irradiation to present GE burnup levels. Our review of NEDE-22148-P is now completed. In the enclosed safety evaluation report we conclude that the GE8x8E and GE8x8EB designs are acceptable for operation to extended burnup up to the burnup requested by GE in Amendment 10. This is based on the acceptabiity of the design for extended burnup and the approval of the GE design and analysis methods for considering extended burnup as described in the NEDE-22148-P.

We do not intend to repeat our review of the matters described in the report and found acceptable when the report appears as a reference in license applications, except to assure that the material presented is applicable to the specific plant involved. Our acceptance applies only to the matters described in the report.

In accordance with procedures established in NUREG-0390, it is requested that GE publish accepted versions of this report, proprietary and non-proprietary, within three months of receipt of this letter. The accepted versions shall incorporate this letter and the enclosed evaluation between the title page and the abstract. The accepted versions shall include an -A (designating accepted) following the report identification symbol.

Should our critiera or regulations change such that our conclusions as to the acceptability of the report are invalidated, GE 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,

Herbert N. Berkow, Director Standardization and Special Projects Directorate
Division of PWR Licensing-B
Enclosure:
As stated

ENCLOSURE
SAFETY EVALUATION ON AMENDMENT 10 TO GESTAR II (NEDE-24011-P) FOR EXTENDED BURNUP

1.0 INTRODUCTION

By letter dated November 30, 1984 (Reference 1) General Electric Company submitted Amendment 10 to GESTAR II. This submittal introduced the GE8x8E and the GE8x8EB fuel designs. Both designs are intended for irradiation to extended burnups.

The GE8x8E and GE8x8EB fuel designs are modifications of the GE P8x8R and BP8x8R designs. The modifications are to aspects of the fuel rod design which enable the fuel bundles to attain higher burnups and also to the design of the bundle upper tie plate. The new extended burnup design features described in Amendment 10 include changes to the peak linear heat generation rate plenum volume, helium fill gas pressures, fuel density, fuel/cladding gap, different enrichments, additional water rods and gadolinium rod changes. Some of these changes are reflected in Table 2-1c and 2-1d and Figure 2-2 of Amendment 10.

In a safety evaluation report dated May 28, 1985 (Ref. 2) we approved the use of these two fuel designs "to present GE burnups levels" and stated that approval of these fuel designs for extended burnup must await completion of our review of NEDE-22148-P, "Extended Burnup Methodology" (Ref. 3) which describes the General Electric design and analyses process with respect to extended burnup. This review is now completed. The results of our review have been transmitted to GE by letter dated August 13, 1985 (Ref. 4). We have concluded that the GE Extended Burnup Methodology is acceptable. In reaching this conclusion, we examined the effects of burnup on all Specified Acceptable Fuel Design Limits listed in Standard Review Plan, Section 4.2. Our SER on NEDE-22148-P did not include any burnup limits. General Electric chose not to associate any burnup limits with this report, but rather chooses 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, as we noted in the SER on NEDE-22148-P, associated with a given fuel design are the 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.

General Electric has chosen to justify a proprietary burnup limit for the GE8x8E and GE8x8EB fuel assemblies which is higher than that for the P8x8R and BP8x8R. We have reviewed the information submitted by GE to support this burnup limit (Refs. 1, 3, 5, and 7). In particular, GE has provided calculations of fuel rod temperatures and pressures to the higher burnup and a description of the designs of the GE8x8E and GE8x8EB in Reference 7. In addition, we have reviewed those areas which were identified in the original Amendment 10 SER (Ref. 2) as requiring more justification for high burnup operation. This was done by generically approving the method by which burnup is considered in the design and analytical processes (as described in Reference 3 and approved in Reference 4) and assuring that the design of the GE8x8E and GE8x8EB fuel assemblies was consistent with these design and analytical processes. In particular, the following areas which were identified in our SER on Amendment 10 were reviewed:

(a) The effects of cladding ductility on the stress/strain criteria at extended burnup

(b) Fretting wear at extended burnup

(c) Corrosion and crud buildup at extended burnup

(d) Effect of fuel rod bowing at extended burnup

(e) Effect of axial growth at extended burnup

(f) Effect of fuel rod internal pressure

(g) Effect of internal hydriding at extended burnup

(h) Overheating of fuel pellets at extended burnup

(i) Excess fuel enthalpy criterion at extended burnup

(j) Cladding rupture models at extended burnup

(k) Violent expulsion of fuel at extended burnup

(l) Cladding ballooning at extended burnup

(m) Each of these topics is discussed in the following section.

In addition, we raised a question during this review of the acceptability of the SAFE/REFLOOD LOCA codes at high burnup since the GEGAP III code was not specifically approved for high burnup application. At the time of approval of GEGAP III, (April, 1974) high burnup was not an issue.

This will be discussed in Item (m) in the next Section.

2.0 EVALUATION

(a) The effects of cladding ductility on the stress/strain criteria at extended burnup.

In our SER on Amendment 10 (Ref. 2) we stated that the effects of cladding ductility on the stress/strain criteria would be addressed in the review of NEDE-22148. During our review of NEDE-22148 GE provided data to demonstrate that cladding ductility decreases at a low burnup and then remains constant. Based on our review of cladding data at high burnup, we concur that this conclusion is reasonable. Thus, the criteria for stress and strain remain valid at high burnup. Therefore, application of this criterion to the GE8x8E and GE8x8EB fuel designs is acceptable.

GE has stated in Section 2.2.1.1.3 of Amendment 10 (Ref. 1) that these criteria are met for the GE8x8E and GE8x8EB fuel assemblies.

(b) Fretting Wear at Extended Burnup

In our SER on Amendment 10 we stated that an evaluation of fretting wear would be included in our safety evaluation report on NEDE-22148. During our review of this report, GE stated that no significant fretting wear had been observed on 8x8 fuel assemblies after 8 years residence time with bundle average burnups to 45.6 MWD/kgM. As discussed in Reference 2, our SER on NEDE-24011 Amendment 10 which approved the GE8x8E and GE8x8EB fuel designs to existing BWR burnups, the grid and spring force spacer designs are identical to those on which these fretting data were

obtained. In an SER on NEDE-22148-P we stated 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.

(c) Corrosion and Crud Buildup at Extended Burnup

In our SER on Amendment 10 we stated that because the SRP does not provide numerical limits on cladding oxidation and since GE includes cladding oxidation and crud effects in their analyses, it is concluded that GE's approach is consistent with the SRP guidelines and applicable to the GE8x8E and GE8x8EB fuel designs. The review of this methodology was accepted in the safety evaluation of Amendment 7 to GESTAR II (Refs. 8 and 9) and the application of the methodology to high burnup was to be addressed further in the safety evaluation of GE's extended burnup topical report NEDE-22148 (Ref. 3). This evaluation stated that, based on the available data base and the conservative way in which corrosion and crud buildup are treated, 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.

Since these data and analyses are applicable to the GE8x8E and GE8x8EB, and since GE has stated that they were used in the applicable analyses of the GE8x8E and GE8x8EB, we find these fuel designs acceptable with respect to corrosion and crud buildup to the burnup proposed by GE.

(d) Effect of Fuel Rod Bowing at Extended Burnup

In our previous SER on Amendment 10 we stated that 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. We stated that we could not find this criterion acceptable since GE had not presented sufficient evidence that there would not be a reduction in the critical power ratio for significant amounts of bowing. However, we also found no reason to change the existing position on the effect of fuel rod bowing on critical power ratio. This position is that no reduction in critical power ratio operating limits is required 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 to the staff. This position was also found applicable to the GE8x8E and GE8x8EB fuel designs.

We further stated that the effect of extended burnup on rod bowing will be discussed further in the safety evaluation of NEDE-22148. This evaluation has concluded that, based on the limited amount of data presented by GE for rod bowing at bundle average burnups between 42 and 45 MWd/kgM, GE has shown maximum gap closures significantly less than the 50% gap closure reporting requirement required by the NRC. Consequently, the current rod bow reporting requirement (greater than 50% gap closure) for GE fuel designs is found to be adequate for the GE8x8E and GE8x8EB fuel designs.

(e) Effect of Axial Growth at Extended Burnup

In our SER on NEDE-22148, we stated that current irradiation experience with GE indicates that gap closure is not a problem (Refs. 10, and 11). 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. 10). 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 diametrical gap size changes in the GE8x8E and GE8x8EB fuel designs (Ref. 1), we recommended that GE obtain confirmatory rod-to-tie plate measurements from their fuel

surveillance program for these designs. Experience has shown that, with adequate surveillance, sufficient time exists to take corrective actions before any serious problems can develop.

(f) Effect of Fuel Rod Internal Pressure

In our previous SER on Amendment 10 we stated that the power history used as input for the rod internal pressure calculations is being addressed in the safety evaluation of the extended burnup topical report NEDE-22148-P. That review concluded that the GE methodology is bounding with respect to rod pressure calculations and thus is acceptable for extended burnup applications. Using this methodology, GE has demonstrated (Ref. 7) that the fuel rod internal pressure criterion is satisfied for the GE8x8E and GE8x8EB fuel designs to the target burnup. These calculations were done at the higher linear heat generation rate proposed by GE for these new fuel designs.

(g) Effect of Internal Hydriding at Extended Burnup

In our previous SER on Amendment 10 we stated that since GE uses a hydrogen limit less than or equal to that stated in the SRP we have concluded in Reference 8 that the design limit for hydriding is acceptable and this is also found to be acceptable for the GE8x8E and GE8x8EB designs. Furthermore internal hydriding is basically an early-in-life failure mechanism and is not a significant consideration at extended burnups.

(h) Overheating of Fuel Pellets at Extended Burnup

The GE bases and criteria for overheating of fuel pellets were presented in Amendment 7 to GESTAR II (Ref. 12). We approved these bases and criteria in our SER on Amendment 7 (Refs. 8 and 9). 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. In order to calculate fuel melt?ing and cladding strain, GE uses the GESTR-MECHANICAL code. In our SER on NEDE-22148-P we approved the use of this code to the proprietary extended burnup limit proposed by GE. Using GESTR-MECHANICAL, GE has calculated that the above criteria are satisfied for the GE8x8E and GE8x8EB to their proprietary burnup limit.

(i) Excess Fuel Enthalpy Criterion at Extended Burnup

In our previous SER on Amendment 10 we stated that the applicability of this fuel enthalpy limit to extended burnup fuel will be discussed in the review of NEDE-22148-P. During our review of NEDE-22148-P GE stated that the models used in the analysis of the Rod Drop Accident are insensitive to extended burnup, because this accident is characterized almost exclusively by the control rod worth which is essentially constant in this burnup range. We agreed with this position. Consequently, the analysis methods were found to be acceptable for extended burnup applications. These methods will be used in analyses to demon?strate that the 170 cal/gm enthalpy criterion is met for the GE8x8E and GE8x8EB fuel designs for specific plant applications.

(j) Cladding Rupture Models at Extended Burnup

In our previous SER on Amendment 10 we stated that the designs of the GE8x8E and GE8x8EB fuel assemblies have not changed the applicability of this rupture-temperature correlation and thus the correlation is approved for application with respect to these designs. We further stated that the

applicability of this correlation and other LOCA models to extended burnup fuel will be addressed in the review of NEDE-22148-P. In our SER on NEDE-22148-P we concluded that the GE methodology to determine cladding deformation and rupture during a LOCA is applicable to extended burnup fuel up to the proprietary burnup limit. GE will use this methodology in the analysis of the GE8x8E and the GE8x8EB fuel designs.

(k) Violent Expulsion of Fuel at Extended Burnup

GE employs a 280 cal/g criterion as a control rod drop accident design limit and thus is consistent with the SRP. The applicability of this limit to extended burnup fuel was addressed in the review of NEDE-22148-P. We stated that the methods used to calculate energy deposition as a result of reactivity insertion accidents (reactor physics codes) are also applicable to extended burnups. In addition, 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. Thus, the 280 cal/gm criterion is acceptable for application to the GE8x8E and GE8x8EB fuel designs at extended burnup.

(l) Cladding Ballooning at Extended Burnup

In our previous SER on Amendment 10 we stated that the design changes made as a result of the GE8x8E and GE8x8EB designs have not reduced the applicability of these methods and thus are approved for application with respect to these designs. The application of the methods and models used in the LOCA analysis for extended burnup fuel was addressed in the review of NEDE-22148-P. In our SER on NEDE-22148-P we 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. Therefore, these methods are acceptable for the GE8x8E and GE8x8EB fuel designs at extended burnup and will be used in LOCA analyses for these two fuel designs.

(m) Applicability of SAFE/REFLOOD at Higher Burnups

During the review of the GE8xE and GE8x8EB fuel designs, because of the higher target burnup, we raised a question about the suitability of using GE's older LOCA analysis methods, denoted SAFE/REFLOOD, for high burnups since these methods, particularly the stored energy model, GEGAP III, were not specifically approved for high burnup. In particular, a model was added to GEGAP III to account for enhanced fission gas release at high burnup. To offset the effects of this enhanced fission gas release model, GE requested in References 1 and 2, that approved, but unapplied, ECCS evaluation model improvements be used to offset MAPLHGR penalties due to high burnup fission gas release. In Reference 3, we approved this request, provided that predicted PCT increases due to high burnup fission gas release did not exceed the reduction in PCT associated with the improved models. For cases which could not meet this provision, GE, with our approval, has reduced MAPLHGR limits at high exposures to account for the increased fission gas.

In a letter dated October 31, 1985 (Reference 17), GE stated that "GE will continue to comply with the provisions" of Reference 15. GE further stated in the October 31, 1985 letter that: "We have demonstrated that adequate margin exists for BWR/3, 4, 5 and 6 plants (Reference 14), and the NRC has accepted this (Reference 16)."

"For BWR/2's MAPLHGR penalties for increased fission gas will be applied when appropriate. Fission gas release at planar exposures greater than 50,000 MWD/MT will be determined using the GESTR-LOCA fuel performance model. At planar exposures less than 50,000 MWD/MT, the results of current evaluations (using GEGAP) will still be applied."

We find this acceptable since it satisfies our concern that, based on the results given in Reference 13, peak cladding temperatures calculated with GEGAP III with enhanced fission gas release could be nonconservative with respect to the offsetting credit from evaluation model approvements for BWR-2's above 50,000 MWd/MT.

3.0 REGULATORY POSITION

Based on our findings as discussed above, we conclude that the GE8x8E and GE8x8EB fuel assemblies are acceptable for use to the target burnup limit requested by GE in Reference 1.

REFERENCES

1. 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.

2. Letter from C. O. Thomas (NRC) to J. S. Charnley (GE) "Acceptance for Referencing of Licensing Topical Report NEDE-24011-P-A-6, Amendment 10, 'General Electric Standard Application for Reactor Fuels'," dated May 28, 1985.

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

4. Letter from C. O. Thomas, (NRC) to J. S. Charnley (GE), "Acceptance for Referencing of Licensing Topical Report NEDE-22148-P, 'Extended Burnup Evaluation Methodology'", August 13, 1985.

5. Letter, J. S. Charnley, (GE) to C. O. Thomas (NRC), "Supplementary Information Regarding NEDE-24011-P-A, Amendment 10," May 2, 1985.

6. Memorandum from L. S. Rubenstein (NRC) to T. Novak (NRC), "Extension of General Electric Emergency Core Cooling System Performance Limits," dated June 25, 1981.

7. Letter from J. S. Charnley (GE) to R. Lobel (NRC), "Presentation on GE8x8E and GE8x8EB Fuel Designs," November 14, 1984. (Proprietary)

8. 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.

9. 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.

10. Letter from J. S. Charnley (GE) to L. S. Rubenstein (NRC), "1983 Fuel Experience Report," (Proprietary), dated October 12, 1983.

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

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

13. Letter, R. E. Engel (GE) to T. A. Ippolito (NRC), "Extension of ECCS Performance Limits," May 6, 1981.

14. Letter, R. E. Engel (GE) to T. A. Ippolito (NRC), "Extension of ECCS Performance Limits," May 28, 1981.

15. Letter, D. G. Eisenhut (NRC) to all Operating BWR's, "High Burnup MAPLHGR Limits (Generic Letter 82-03)," March 31, 1982.

16. Letter, C. O. Thomas (NRC) to J. S. Charnley (GE), "Acceptance for Referencing of Licensing Topical Report NEDE-24011-P-A-6, Amendment 10, 'General Electric Standard Application for Reactor Fuel"', May 28, 1985.

17. Letter, J. S. Charnley, (GE), to R. Lobel, (NRC), "Supplementary Information Regarding Use of SAFE/REFLOOD to High Exposures", October 31, 1985.

NRC
Safety Evaluation Report
Approving Amendment 11
to
NEDE-24011-P

UNITED STATES
NUCLEAR REGULATORY COMMISSION
Washington, D. C. 20555
November 5, 1985

MFN 141-85

Ms. J.S. Charnley, Manager
Fuel Licensing
General Electric Company
175 Curtner Avenue
San Jose, California 95125

Dear Ms. Charnley:

SUBJECT: Acceptance for Referencing of Licensing Topical Report NEDE-24011-P-A Rev. 6, Amendment 11, "General Electric Standard Application For Reactor Fuel" (GESTAR II)

We have completed our review of the subject topical report submitted by the General Electric Company (GE) letter dated February 27, 1985. We find the report to be acceptable for referencing in license applications to the extent specified and under the limitations delineated in the report and the associated NRC evaluation pertaining to treatment of uncertainties in the calculation of Operating Limit MCPR values, which is enclosed. The evaluation defines the basis for acceptance of the report.

We do not intend to repeat our review of the matters described in the report and found acceptable when the report appears as a reference in license applications, except to assure that the material presented is applicable to the specific plant involved. Our acceptance applies only to the matters described in the report.

In accordance with procedures established in NUREG-0390, it is requested that GE publish accepted versions of this report, proprietary and non-proprietary, within three months of receipt of this letter. The accepted versions shall incorporate this letter and the enclosed evaluation between the title page and the abstract. The accepted versions shall include an -A (designating accepted) following the report identification symbol.

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

Sincerely,

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

Enclosure:
As stated

ENCLOSURE
EVALUATION OF AMENDMENT 11 TO NEDE-24011-P-A

By letter dated February 28, 1985 (Reference 1) General Electric Company (GE) submitted Amendment 11 to the GE Licensing Topical Report NEDE-24011-P-A, "General Electric Standard Application for Reactor Fuel" (GESTAR II). Additional information was submitted in a meeting on June 21, 1985 and subsequently in Reference 2 in response to a request from the staff (Reference 3). The Core Performance Branch and the Reactor Systems Branch have reviewed the information submitted and prepared the following evaluation.

Amendment 11 to GESTAR II alters the document to include an updated version of the ODYN code among the calculational techniques used for plant transient analyses and alters the manner in which calculational uncertainties are treated in obtaining core operating limits. A description and justification of the code revisions were included in References 1 and 2. The changes to the ODYN calculational model include:

1. Improved Neutronics Methods

   These methods are described in Reference 4, which has been reviewed and approved by the staff (Reference 5).

2. Inclusion of GESTR-M Fuel Performance Model

   This model has been approved by the staff as part of the approval of Amendment 7 to GESTAR-II (Reference 6).

3. Improved Bulkwater Model

   Improvements include more detailed nodalization, use of a drift flux rather than a homogeneous formulation in the void correlation and use of a void profile and feedwater quenching. Reference 2 presents comparisons of both the new and current void correlations with experiments and demonstrates the superiority of the new correlation.

4. Improved Upper Plenum Model

   The improved model uses a drift flux rather than a homogeneous model and an improved calculation of the mass holdup.

5. Improved Separator Mass Storage Model

   The improved model uses a transient, homogeneous mass balance rather that a quasi-steady-state mass balance.

Data were provided in Reference 2 on the results of comparisons of the old and new ODYN calculations to the Peach Bottom turbine trip tests. These data showed that the new ODYN results provided generally better agreement with the test data than did the old ODYN calculations. Breakdown of the calculations to separate out the effects of the various improvements showed that most of the improvement occurred from the inclusion of the previously approved methods in the

calculations. Based on improved agreement with experiments and the refinement of the calculational models as described above we conclude that the improvement to the ODYN code are acceptable.

In addition to implementing the new model, GE intends to continue use of the current model for appropriate non-limiting calculations. We find this acceptable.

Amendment 11 also revised the manner in which code uncertainties are handled in obtaining the Option A and Option B MCPR operating limits. However insufficient justification has been provided for this change and we conclude that the currently used treatment of uncertainties must continue to be used. This has been discussed with GE and they concur in this condition to the staff approval of this amendment.

REFERENCES

1. Letter, J.S. Charnley (GE) to C.O. Thomas (NRC), “Amendment 11 to GE LTR NEDE-24011-P-A”, February 27, 1985.

2. Letter, J.S. Charnley (GE) to C.O. Thomas, (NRC), “Response to Request No. 1 for Additional Information on NEDE-24011 Rev. 6, Amendment 11,” July 18, 1985.

3. Letter, C.O. Thomas, (NRC) to J.S. Charnley (GE), “Request Number 1 for Additional Information on NEDE-24011 Rev. 6, Amendment 11”, May 9, 1985.

4. NEDE-30130-A, “Steady State Nuclear Methods”, May, 1985.

5. Letter, C.O. Thomas (NRC) to J.S. Charnley (GE), “Acceptance for Referencing of Licensing Topical Report NEDE-30130, ‘Steady-State Nuclear Methods’ December 22, 1983”. (See also page following the title page of the approved report).

6. Approval letter, C.O. Thomas (NRC) to J.S. Charnley (GE), dated March 1, 1985.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
Washington, D. C. 20555
March 22, 1986

MFN 029-086

Ms. J.S. Charnley
Fuel Licensing Manager
General Electric Company
175 Curtner Avenue
San Jose, California 95125

Dear Ms. Charnley:

SUBJECT: Acceptance For Referencing Of Licensing Topical Report NEDE-24011-P-A, "GE Generic Licensing Reload Report," Supplement To Amendment 11

We have completed our review of the subject topical report submitted by General Electric Company by letter dated October 9, 1985.

We find the report to be acceptable for referencing in license applications to the extent specified and under the limitations delineated in the report and the associated NRC evaluation, which is enclosed. The evaluation defines the basis for acceptance of the report.

We do not intend to repeat our review of the matters described in the report and found acceptable when the report appears as a reference in license applications, except to assure that the material presented is applicable to the specific plant involved. Our acceptance applies only to the matters described in the report.

In accordance with procedures established in NUREG-0390, it is requested that GE publish accepted versions of this report, proprietary and non-proprietary, within three months of receipt of this letter. The accepted versions shall incorporate this letter and the enclosed evaluation between the title page and the abstract. The accepted versions shall include an -A (designating accepted) following the report identification symbol.

Should our criteria or regulations change such that our conclusions as to the acceptability of the report are invalidated, GE 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,

Gus C. Lainas, Assistant Director for BWR
Division of BWR Licensing
Office of Nucelar Reactor Regulation

Enclosure:
As stated

SAFETY EVALUATION REPORT FOR AMENDMENT 11 TO NEDE-24011

By letter from Cecil O. Thomas (NRC) to Ms. J.S. Charnley (GE) dated November 5, 1985, the Nuclear Regulatory Commission found Amendment 11 to NEDE-24011 acceptable for referencing in license applications with qualifications. In particular, the evaluation noted:

> Amendment 11 also revised the manner in which code uncertainties are handled in obtaining the Option A and Option B MCPR operating limits. However insufficient justification has been provided for this change and we conclude that the currently used treatment of uncertainties must continue to be used. This has been discussed with GE and they concur in this condition to the staff approval of this amendment.

General Electric provided supplementary information which supports the proposed method of treating uncertainties in letter JSC-065-85 from J.S. Charnley to C.O. Thomas dated October 9, 1985. The October 9 letter was revised in JSC-005-86 from J.S. Charnley to H.N. Berkow dated January 16, 1986. The new application methodology is for the GEMINI/ODYN transient analysis methodology. The proposed approach is similar to the previously approved GENESIS/ODYN methodology. The revised amendment 11 describes the derivation of statistical adjustment factors to be applied to GEMINI/ODYN results to determine plant operating limits for BWR/4 and BWR/5 plants operating without recirculation pump trip. Factors for other types of plants will be provided in future reports.

Background (GENESIS/ODYN Application)

The current licensing basis approved with the GENESIS/ODYN models for calculating the ýCPR for pressurization events is performed in accordance with either or both of two methods known as Option A and Option B. These currently used options are summarized below:

Option A

This approach is comprised of the two-step calculation which follows:

1. The pressurization transient is analyzed using the GENESIS/ODYN models to obtain the change in the critical power ratio (DCPR) for the core. The initial CPR ($ICPR_c$) is then determined such that $ICPR_c$ minus DCPR equals the MCPR Fuel Cladding Integrity Safety Limit. Conservative input parameters are used in the analysis, e.g. the scram speed is per technical specifications, and a conservative Haling power shape, as well as other maximum equipment specifications, is used.

2. The licensing basis ICPR is given as:

$$\text{ICPR Licensing Basis} = 1.044\ ICPR_c$$

where

$ICPR_c$ = value of ICPR calculated using GENESIS/ODYN models,

i.e. ICPR = ΔCPR + MCPR Safety Limit

Option B

This procedure provides for a statistical determination of the pressurization transient ΔCPR/ICPR such that there is a 95% probability with 95% confidence (95/95) that the event will not cause the critical power ratio to fall below the MCPR Fuel Cladding Integrity Safety Limit. This approach can be satisfied in one of two ways:

1. A plant-specific statistical analysis can be performed per the approved statistical methodology procedures to determine the 95/95 DCPR/ICPR; or

2. Generic ΔCPR/ICPR statistical adjustment factors (SAF) for groupings of similar type plants can be applied to plant-specific calculations to derive the 95/95 DCPR/ICPR value. This procedure is characterized by the following expression:

$$\left(\frac{\Delta CPR}{ICPR}\right)_{95/95} = \left(\frac{\Delta CPR}{ICPR}\right)_c + SAF$$

where $\left(\frac{\Delta \mathrm{CPR}}{\mathrm{ICPR}}\right)_c$ = ΔCPR/ICPR

calculated for the pressurization event per the assumptions of Step 1 in Option A

By substituting into the above expression the relationship:

$$\Delta \mathrm{CPR}_{95/95} = \mathrm{ICPR}_{95/95}\ \mathrm{MCPR\ Safety\ Limit},$$

it follows that the $(\mathrm{ICPR})_{95/95}$ is determined from the following expression:

$$\mathrm{ICPR}_{95/95} = \frac{\mathrm{MCPR\ Safety\ Limit}}{1-\left[\left(\frac{\Delta \mathrm{CPR}}{\mathrm{ICPR}}\right)_c + \mathrm{SAF}\right]}$$

This statistical Option B uses a GENESIS/ODYN model uncertainty of 37% of ΔCPR/ICPR at the 2s level to determine the 95/95 ICPR. This uncertainty was determined by the staff when the GENESIS/ODYN model predictions were compared to the full scale turbine trip qualification data.

Utilities using Option B must demonstrate that their plant's scram speed distribution is consistent with that used in the statistical analysis. This is accomplished through an approved technical specification which consists of testing at the 5% significance level and allows adjustment of the operating limit MCPR if the scram speed is outside the assumed distribution.

GEMINI/ODYN Application

The GEMINI/ODYN set of methods has been compared against actual test data. The results of the comparison indicate an improvement in prediction accuracy with GEMINI/ODYN. The true 95/95 ΔCPR/ICPR will be determined using the same fundamental approach established for the current GENESIS/ODYN Option B and accounting for the improvement in prediction accuracy. The resulting procedure, which will be used with the GEMINI/ODYN models, simplifies the current two option

approach (Option A and Option B) into one. The GEMINI/ODYN licensing bases are discussed in the following sections.

Licensing Analysis

Licensing analyses accomplished with the GEMINI/ODYN models will permit plants to operate under a single set of MCPR limits if scram speed compliance procedures identical to those in current plant Technical Specifications are followed. If scram speed compliance is not demonstrated, more conservative MCPR operating limits must be met. The licensing analysis with the GEMINI/ODYN models will be calculated in a similar manner as with the GENESIS/ODYN models. The statistical determination of the transient ΔCPR/ICPR adjustment factor for the pressurization event will continue to assure a 95% probability with 95% confidence that the critical power will not fall below the MCPR Fuel Cladding Integrity Safety Limit. The ODYN model uncertainty used in statistical analysis will be revised from the 37% ΔCPR/ICPR used for the GENESIS/ODYN methods to a value to reflect the improved accuracy of the GEMINI/ODYN set of methods demonstrated by comparison to data.

GEMINI/ODYN Technical Specification Limits

The technical specification limit will be determined from the following general equation:

$$\mathrm{OLMCPR}_{\mathrm{Tech\ Spec}} = \mathrm{OLMCPR}_{95/95} + \frac{\tau_{ave} - \tau_B}{\tau_A - \tau_B}(\Delta\mathrm{OLMCPR})$$

where ΔOLMCPR = factors derived by the new methodology and

$\mathrm{OLMCPR}_{95/95} = \Delta\mathrm{CPR}_{95/9} + \mathrm{MCPR\ Safety\ Limit}$.

The definitions of τ_A, τ_B and τ_{ave} remain the same as those currently appearing in Technical Specifications. For plants that demonstrate scram speed compliance (i.e. $\tau_{ave} \le \tau_B$) using the NRC-approved procedures, the Technical Specification limit becomes:

$$\mathrm{OLMCPR}_{\mathrm{Tech\ Spec}} = \mathrm{OLMCPR}_{95/95} \text{ (for } \tau_{ave} \le \tau_B\text{.)}$$

If scram speed compliance is not demonstrated by a plant (i.e. $\tau_{ave} > \tau_B$) or if a plant chooses not to perform the scram speed compliance procedures (i.e. $\tau_{ave} = \tau_A$), then a more conservative operating limit must be used.

The cycle-specific reload submittal will contain results for both the case where scram speed compliance is demonstrated ($\tau_{ave} \le \tau_B$, $\mathrm{OLMCPR}_{\mathrm{Tech\ Spec}} = \mathrm{OLMCPR}_{95/95}$), and for the case

where $\tau_{ave} = \tau_A$ ($\mathrm{OLMCPR}_{\mathrm{Tech\ Spec}} = \mathrm{OLMCPR}_{95/95} + \Delta\mathrm{OLMCPR}$).

As is the case today, the actual operating limit will be a straight-line interpolation between these two values dependent on the results of scram speed testing. We have reviewed the new methodology to be used with the GEMINI/ODYN codes and find it to be acceptable.

General Electric also requested approval for use of the statistical methodology for GEMINI/ODYN with GENESIS/ODYN. Because the GENESIS/ODYN methodology is more conservative than the

GEMINI/ODYN methodology, we find this application to also be acceptable. In addition, continued use of the GENESIS/ODYN methodology and its statistical factors is also acceptable.

NRC
Safety Evaluation Report
Approving Amendment 12
to
NEDE-24011-P

UNITED STATES
NUCLEAR REGULATORY COMMISSION
Washington, D. C. 20555
October 11, 1985

MFN 127-85

Ms. J.S. Charnley, Manager
Fuel Licensing
Nuclear Safety and Licensing
General Electric Company
175 Curtner Avenue
San Jose, California 95125

Dear Ms. Charnley:

SUBJECT: Acceptance for Referencing of Licensing Topical Report NEDE-24011-P-A, "General Electric Standard Application For Reactor Fuel", Revision 6, Amendment 12

We have completed our review of the subject topical report submitted by the General Electric Company (GE) by letter dated May 17, 1985. We find the report to be acceptable for referencing in license applications to the extent specified and under the limitations delineated in the report and the associated NRC evaluation, which is enclosed. The evaluation defines the basis for acceptance of the report.

We do not intend to repeat our review of the matters described in the report and found acceptable when the report appears as a reference in license applications, except to assure that the material presented is applicable to the specific plant involved. Our acceptance applies only to the matters described in the report.

In accordance with procedures established in NUREG-0390, it is requested that GE publish accepted versions of this report, proprietary and non-proprietary, within three months of receipt of this letter. The accepted versions shall incorporate this letter and the enclosed evaluation between the title page and the abstract. The accepted versions shall include an -A (designating accepted) following the report identification symbol.

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

Sincerely,

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

Enclosure:
As stated

ENCLOSURE
Review of the Proposed Amendment 12 to Revision 6 of NEDE-24011-P-A "General Electric Standard Application for Reactor Fuel" (TACS 57908)

INTRODUCTION

In a letter of May 17, 1985 (Reference 1) General Electric proposed an Amendment (12) to NEDE-24011-P-A (Revision 6) incorporating into GESTAR the use of, and credit for Banked Position Withdrawal Sequences (BPWS) for those plants which utilize the Group Notch Rod Sequence Control System (GNRSCS). This change was presented in an attachment to Reference 1 and was discussed in a letter of May 10, 1985 (Reference 2). The following is a review of material relevant to that proposal.

REVIEW

In going critical, heating up and going to power, BWR control rods are pulled in patterns which are devised to minimize rod reactivity worths associated with the Control Rod Drop Accident (CRDA), and which are supervised by pattern control systems, such as the Rod Worth Minimizer (RWM) or the Rod Sequence Control System (RSCS). There are two classes of pattern operation and surveillance of interest to this review.

1. GNRSCS plants (some of the BWR-4s) operate with a hard wired RSCS plus computer controlled RWM backup. For the first 50 percent of the control rods to be withdrawn each rod is (individually) fully withdrawn in a defined sequence. For rods beyond 50 percent the rods are (effectively) withdrawn in defined banks one notch at a time.

2. BPWS plants (BWR-2 and 3s plus Vermont Yankee, a BWR-4) operate with the RWM alone and for the first 50 percent of the rods the effective withdrawal is in the form of (stepped) defined bank patterns. Beyond 50 percent it is also in the form of stepped bank withdrawal.

For the first 50 percent the BPWS bank positions result in lower peak rod reactivity worths for the CRDA than the RSCS (non-banked) patterns. Beyond 50 percent the Group Notch patterns provide somewhat lower worth than the corresponding BPWS patterns.

For the GNRSCS plants the reload analysis procedures require determination of CRDA parameters. The higher rod worths of the non-banked mode frequently results in exceeding standard parameter limits which are used to define conditions such that reactor specific analyses are not needed. For these cases new analyses are required. Sometimes it is necessary to rearrange the core loading pattern to get within CRDA required limits.

For the BPWS reactors the rod worths are sufficiently low, both in the first 50 percent and beyond, that it has been possible to perform a generic statistical study of the maximum rod worths of the class

of BPWS reactors and conclude that, with a high degree of confidence, the CRDA will not exceed required limits. Thus reload, reactor specific, analyses are not required. This methodology has been reviewed and approved (see Reference 3).

If the GNRSCS plants could use BPWS for the first 50 percent (and since beyond 50 percent rod worths are lower for GNRSCS than BPWS) they too would have the lower maximum rod worth of the BPWS plants, and it would then be appropriate to apply the approved BPWS statistical analysis to them and eliminate the need for reload analysis and review.

General Electric has proposed that for the GNRSCS plants which elect to implement the change, the RWM be reprogrammed for the BPWS patterns in the first 50 percent withdrawal and become the primary pattern control system for this part of the withdrawal rather than the RSCS. The withdrawal control beyond 50 percent would remain the same (RSCS controlled Group Notch). Thus the more conservative (lower maximum worth) pattern would be supervised by the RWM computer system rather than a hard wired system (the RSCS would not enforce the first 50 percent bank positions). This would, however, be the same as has been approved for the BPWS-RWM plants.

Thus the primary decision of the review is whether an improved (lower peak notch) pattern, monitored by the RWM is better than, or at least as acceptable as, the less desirable (higher rod worth) pattern, monitored by a hard wired system.

EVALUATION

The BPWS produces lower reactivity worths both for the CRDA and for the normal withdrawal single notch worth problem. For the latter the bigger notch worth of the non BPWS reactors have, on occasion, produced IRM scrams which have been a source of some contention in the past. The BPWS, particularly if the improved form of the "Reduced Notch Worth Procedure" is used, can do a lot to eliminate that problem (see Reference 6).

With the BPWS, because of the approved statistical analysis approach, there will be no need to determine rod worths and other parameters or to do CRDA calculations for a reload and no need to review such matters. It will also tend to minimize fuel pattern changes from design optimums which might be needed only to lower CRDA calculated results.

It is noted, furthermore, that staff analyses of the CRDA event (via our BNL consultants), using more accurate three dimensional and hydraulic feedback methods, indicate CRDA consequences to be significantly below those of standard General Electric methods (see References 4, 5, and 6). Thus the General Electric results of analyses for the non BPWS patterns (as well as all other patterns) are artificially high. This has been discussed in our review and approval of the statistical analysis for the BPWS plants (Reference 3). That review approved the statistical analysis approach, including the deletion of CRDA analysis for reloads for RWM (only) BPWS plants.

Furthermore, after hard wired systems were first required for new reactors the staff subsequently examined the probabilities of a CRDA event exceeding limits and decided that they were sufficiently low that backfitting of a hard wired system to older plants (i.e., those currently operating with RWM only) was not necessary (Reference 4). That analysis is applicable to the GNRSCS reactors also and provides additional assurance that operation in the BPWS mode under RWM control for withdrawals below 50 percent is acceptable.

Thus our review has concluded that it is preferable for the GNRSCS plants to have the improved pattern control of the BPWS as monitored by the RWM for the first 50 percent of withdrawal. The GNRSCS group notch mode should be retained beyond 50 percent. Plants making the change will be able to take credit for the statistical analysis of the CRDA and will not have to analyze the event for reloads.

The changes made as Amendment 12 to NEDE-24011-P-A (as provided in the attachment to Reference 2) provide a satisfactory indication of the change and the distinction between GNRSCS plants which have and have not made the change to BPWS. Plants which elect to change will have to provide a submittal to the NRC indicating that BPWS patterns will be enforced and change Technical Specifications as required for this mode of operation.

CONCLUSIONS

General Electric has proposed to amend NEDE-24011-P-A such that Group Notch plants which elect to change to BPWS supervised by the RWM for the first 50 percent of withdrawal may take credit for the CRDA statistical analysis and conclusions which have been approved by the staff for BPWS plants. This would result in these plants being able to delete CRDA analysis from reload analysis procedures. Our review has concluded that such a change, retaining Group Notch withdrawal beyond 50 percent withdrawal, is acceptable. Plants electing to change must provide a submittal to NRC indicating that the PBWS patterns will be thus enforced and changing any related Technical Specification as required to so indicate.

REFERENCES

1. Letter from J.S. Charnley, GE, to C.O. Thomas, NRC, “Proposed Amendment 12 to GE Licensing Topical Report NEDE-24011-P-A”, May 17, 1985.

2. Letter from J.S. Charnley, GE, to C.O. Thomas, NRC, “Licensing Credit for Banked Position Withdrawal Sequences on Group Notch Plants”, May 10, 1985.

3. Memorandum (and enclosure) from L.S. Rubenstein, NRC, to G.C. Lainas, NRC, “Change in GE Analysis of the Control Rod Drop Accident for Reloads”, February 15, 1983.

4. Letter (and enclosure) from B.C. Rusche, NRC, to R. Fraley, ACRS, “Generic Item II A-2 Control Rod Drop Accident (BWRs)”, June 1, 1976.

5. BNL-NUREG-28009, “Thermal-Hydraulic Effects on Center Rod Drop Accidents in a Boiling Water Reactor”, July, 1980, by H. Cheng and D. Diamond (BNL).

6. Memorandum from H. Richings, NRC, to W. Johnston, NRC, “A Review of the ACRS BWR RDS LER Concern”, October 8, 1980.

NRC
Safety Evaluation Report
Approving Amendment 13
to
NEDE-24011-P

UNITED STATES
NUCLEAR REGULATORY COMMISSION
Washington, D. C. 20555
March 26, 1986

MFN 028-086

Ms. J.S. Charnley
Fuel Licensing Manager
General Electric Company
175 Curtner Avenue
San Jose, CA 95125

Dear Ms. Charnley:

SUBJECT: Acceptance For Referencing Of Licensing Topical Report NEDE-24011-P-A Amendment 13, Rev. 6 "General Electric Standard Application For Reactor Fuel"

We have completed our review of the subject topical report submitted by the General Electric Company (GE) by letter dated September 24, 1985. We find the report to be acceptable for referencing in license applications to the extent specified and under the limitations delineated in the report and the associated NRC evaluation, which is enclosed. The evaluation defines the basis for acceptance of the report.

We do not intend to repeat our review of the matters described in the report and found acceptable when the report appears as a reference in license applications, except to assure that the material presented is applicable to the specific plant involved. Our acceptance applies only to the matters described in the report.

In accordance with procedures established in NUREG-0390, it is requested that GE publish accepted versions of this report, proprietary and non-proprietary, within three months of receipt of this letter. The accepted versions shall incorporate this letter and the enclosed evaluation between the title page and the abstract. The accepted versions shall include an -A (designating accepted) following the report identification symbol.

Should our criteria or regulations change such that our conclusions as to the acceptability of the report are invalidated, GE 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,

Gus C. Lainas, Assistant Director for BWR
Division of Licensing
Office of Nuclear Reactor Regulation

Enclosure:
As stated

ENCLOSURE
SAFETY EVALUATION REPORT ON AMENDMENT 13 TO GESTAR II

In a letter dated September 24, 1985, (Reference 2) General Electric (GE) submitted a proposed amendment to NEDE-24011, "General Electric Standard Application for Reactor Fuel" (GESTAR). This is designated as an "Administrative Amendment," and as Amendment 13 to Revision 6 of NEDE-24011. Additional information and some changes to the original submittal were provided in an addendum dated January 21, 1986 (Reference 8).

An administrative Amendment does not present new technical information to be reviewed, but rather incorporates approved material and such changes as are needed to improve the form, consistency or content of this report. The purpose of the present review is to verify that the changes introduced in Amendment 13 are of an administrative nature, and that they are acceptable.

The GESTAR report (Reference 1) describes generic information related to the fuel design and analyses of General Electric Boiling Water Reactor plants for which General Electric provides fuel. The report outlines the fuel design, fuel thermal-mechanical analysis, nuclear and hydraulic analysis methods. The report provides information that is independent of a plant-specific application. Plant-specific information and methods used to determine reactor limits are given in the U.S. Supplement accompanying the base document.

The generic information presented in the base document is supplemented by plant-cycle-unique information and analytical results. The cycle-unique information includes a listing of the fuel to be loaded in the core and safety analysis results. This information is documented in the plant FSAR for initial core loadings and in a separate plant-unique cycle-dependent submittal for each reload, when required. The format for the reload submittal is given in the U.S. Supplement to this document.

Both the base document and the U.S. Supplement are periodically updated via amendments to specific sections of these documents as new or additional data related to the fuel design, thermal-mechanical analysis or nuclear and hydraulic methods are developed. All previous amendments through Amendment 12 have been approved. (Amendment 11 is in final processing.)

Amendment 13 consists of nine sections, each applicable to a specific technical area. The evaluation of the proposed amendment is outlined in the following sections. The order of each section follows the order in which the nine sections (13A through 13J) are presented in the amendment.

A. GEMINI PHYSICS INFORMATION

This section of the amendment consists of the changes which have been introduced in both the main document and the U.S. Supplement in order to incorporate the GEMINI physics models (Ref. 3) into GESTAR. The GEMINI physics methods have been approved (Ref. 4). The impact of GEMINI models on several events has been incorporated.

GE initially proposed that figures 3-1 through 3-7 showing calculated ranges of important neutronic parameters such as infinite multiplication factors as a function of exposure and void fraction, Doppler Coefficient plotted as a function of temperature and void reactivity effects be deleted. GE considers this information to be unnecessary to meet NRC requirements. The retention of these curves was discussed with GE and it was decided that this information would be submitted in a separate letter from GE and this letter would be referenced in GESTAR. It would continue to be addressed as

"typical". It would represent reasonable extremes for current fuel types and would be updated (via a letter) when fuel changes were sufficient to produce a significant change in the curves. This is acceptable.

The fuel storage infinite multiplication factor, koo, for GE supplied high density fuel storage racks has been changed from 1.35 to 1.33. This is a conservative change, incorporating directly a 2 sigma uncertainty factor in the analysis rather than using nominal values, and represents the criteria that GE is currently using. This is an acceptable change.

The changes in this section have been found to be administrative.

B. SAFER/GESTR-LOCA

This section contains the pages which have been changed in the U.S. Supplement to incorporate the improved ECCS evaluation methodology SAFER/GESTR-LOCA (Refs. 5-7). This methodology has been approved.

This is a completely rewritten section and discusses both the more conservative SAFE/REFLOOD method and the new and improved SAFER/GESTR method. The latter is presented as an option. Both methods satisfy the requirements of 10CFR50.46 and Appendix K. This section also contains an updated discussion of the fuel rod cladding rupture model in the CHASTE code.

The changes in this section have been found to be administrative.

C. NEW BUNDLES

A number of new bundle descriptions have been added to Appendices C and D of the base document. Corrections have been made to several figures in both Appendices. Exposure values in all tables have been converted to metric notation (MWd/MT). Pages in this section of the Amendment have been designated U.S.DC-n and U.S.DD-n with n being the page number and the prefix U.S. indicating U.S. Supplement. However, General Electric has informed us that the correct page designation of this section should be simply C-n and D-n, thus indicating that this section (13C) is part of Appendices C and D of the main document not of the U.S. Supplement as originally indicated. No GE 8x8E bundles have been included in this section.

The changes in this section have been found to be administrative.

D. APPENDIX U.S. A UPDATE

Appendix A of the U.S. Supplement (Supplement Reload Licensing Submittal form) has been updated to incorporate within the format for reload submittals the SAFER/GESTR option, the new stability criteria, the maximum extended operating domain (MEOD) and the APRM RBM technical specifications (ARTS) pperating flexibility options as well as other changes. This does not imply generic approval for MEOD or ARTS (see Section G).

The "number drilled" column in item 2 of the Appendix A of the U.S. Supplemental Reload Licensing Submittal has been eliminated since all GE reload fuel now utilizes lower tie plates. The "REDY Events Only" heading in item 6 has been changed to "Cold Water Injection Events Only."

The initial submittal proposed elimination of the R-factor for each fuel design shown in item 7 and the "redundant heat flux values" presented in item 10 of the Supplemental Reload Licensing Submittal. Following staff questions, the Addendum (Reference 8) indicated that these would be retained.

The statement "Includes Power Spiking Penalty" has been deleted from items 11 and 14. This is acceptable since this penalty is included in all fuel rod thermal-mechanical design analyses.

The initial report indicated that stability information would only be given for BWR 4/5/6 plants which do not implement monitoring procedures. The addendum (Reference 8), however, says that the existing stability section will be maintained for all plants.

Item 17 (Loss of Coolant Accident Results) has been changed to be compatible with the analysis methods.

The changes in this section have been found to be administrative.

E. LOSS OF FEEDWATER HEATING ANALYSIS

GE has incorporated a generic loss of feedwater heating bounding analysis into the U.S. Supplement of GESTAR. However, it is the staff conclusion at this time that the material submitted for review has been insufficient for a complete review and that the analysis has not received a generic approval. Thus, inclusion of the generic Loss of Feedwater Heating analysis is not considered to be an appropriate administrative change at this time and should not be a part of Amendment 13. GE has agreed to this change.

F. APPENDIX U.S.-C UPDATE

Appendix C to the U.S. Supplement containing safety evaluation reports (SERs) has been rearranged. The U.S.-C Update has added title pages and an index of the SERs. The latest SERs have been added while superseded SERs have been eliminated. These changes are considered to be administrative.

G. NEW OPERATING FLEXIBILITY OPTIONS

Two new sections (S.2.2.3.6 and 5.2.2.3.7) have been added to the U.S. Supplement to incorporate the ARTS and MEOD operating flexibility options. These sections were included in order to describe the impact of these options on the reload license. Separate approval must be obtained before these options can be used on a specific plant. This is an administrative change.

H. AS-LOADED CORE CRITERIA

The initial report deleted a sentence in paragraph 3.4.2.5. After discussion with GE it was agreed that the sentence would be returned. The only other change in this section is of a minor administrative nature.

I. MISCELLANEOUS CHANGES

This section of the amendment contains changes which were made to both the base document and U.S. Supplement. The GETAB and Transient Operating Parameter Tables in the U.S. Supplement were updated with the latest data. To make the base document applicable to foreign supplements, references

in the base document to specific sections in the U.S. Supplement were removed. The section describing the calculation of the cold effective multiplication factor, k_{eff}, with the strongest control rod withdrawn, was re-written to provide a clearer description. A number of typographical errors and corrections made to miscellaneous tables and figures were also introduced in this section of the amendment.

These changes are considered to be administrative.

CONCLUSIONS

It is concluded that, with the exception of the proposed incorporation of the generic Loss of Feedwater Heating analysis given in Section E, the changes to GESTAR proposed for this amendment are of an administrative nature and that no new and unreviewed technical area has been introduced. The changes that have been made are acceptable.

REFERENCES

1. “General Electric Standard Application for Reactor Fuel,” NEDE-24011 P-A-6, April, 1985.

2. Letter from J. Charnley, GE, to C. Thomas, NRC, dated September 24, 1985, “Proposed Administrative Amendment to GE Licensing Topical Report NEDE-24011-P-A.”

3. “Steady State Nuclear Methods,” April 1985 (NEDE-30130-P-A and NEDO-30130-A).

4. Letter from C.O. Thomas (NRC) to J.S. Charnley (GE), “Acceptance for Referencing of Licensing Topical Report NEDE-30130, Steady State Nuclear Methods,” December 22, 1983.

5. S.O. Akerlund et al., “The GESTR-LOCA and SAFER Models for the Evaluation of the Loss-of-Coolant Accident: Volume 1—GESTR-LOCA—A Model for the Prediction of Fuel and Thermal Performance,” February 1985 (NEDE-23785-1-PA and NEDO-23785-1-A.)

6. K.C. Chan et al., “The GESTR-LOCA and SAFER Models for the Evaluation of the Loss-of-Coolant Accident: Volume II—SAFER—Long Term Inventory Model for BWR Loss-of-Coolant Analysis,” February 1985 (NEDE-23785-1-PA and NEDO-23885-1-A).

7. B.S. Shiralkar et al., “The GESTR-LOCA and SAFER Models for the Evaluation of the Loss-of-Coolant Accident: Volume II—SAFER/GESTR—Application Methodology,” (NEDO-23785-1-A).

8. Letter from J. Charnley, (GE) to H. Berkow, NRC, dated January 21, 1986, “Addendum to Proposed Administrative Amendment to GE Licensing Topical Report NEDE-24011-P-A.”

NRC
Safety Evaluation Report
Approving Amendment 14
to
NEDE-24011-P

UNITED STATES
NUCLEAR REGULATORY COMMISSION
Washington, D. C. 20555
December 27, 1987

Ms. J.S. Charnley, Manager
Fuel Licensing
General Electric Company
175 Curtner Avenue
San Jose, California 95125

Dear Ms. Charnley:

SUBJECT: Acceptance For Referencing Of Amendment 14 To General Electric Licensing Topical Report NEDE-24011-P-A, "General Electric Standard Application For Reactor Fuel" (TAC No. 60113)

The staff has completed its review of the subject amendment submitted by the General Electric Company by letter dated October 2, 1985. This Amendment 14 provides the description of the licensing design basis analysis to support the proposed Upgraded Minimum Critical Power Ratio Fuel Cladding Integrity Safety Limit to reflect the loading of current fuel bundle designs.

We find the Amendment 14 to be acceptable for referencing in license applications to the extent specified and under the limitations delineated in the Amendment 14 and the associated NRC technical evaluation. The evaluation defines the basis for acceptance of the amendment.

We do not intend to repeat our review of the matters described in Amendment 14 and found acceptable when the report appears as a reference in license applications, except to assure that the material presented is applicable to the specific plant involved. Our acceptance applies only to the matters described in the Amendment 14.

In accordance with procedures established in NUREG-0390, it is requested that GE publish accepted versions of this amendment, proprietary and non-proprietary, within three months of receipt of this letter. The accepted versions shall incorporate this letter and the enclosed evaluation between the title page and the abstract. The accepted versions shall include an -A (designating accepted) following the report identification symbol.

Should our criteria or regulations change such that our conclusions as to the acceptability of the report are invalidated, GE 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,

Ashok C. Thadani, Assistant Director for Systems
Division of Engineering & Systems Technology

Enclosure:
Amendment 14 to NEDE-240l1-P-A Evaluation

ENCLOSURE
SAFETY EVALUATION REPORT FOR AMENDMENT 14 TO GENERAL ELECTRIC LICENSING TOPICAL REPORT NEDE-24011-P-A

1.0 INTRODUCTION

By letter from J.S. Charnley (GE) to C.O. Thomas (NRC) dated October 2, 1985, GE submitted Amendment 14 to NEDE-24011-P-A, "General Electric Standard Application for Reactor Fuel" (GESTAR II), for NRC review. The purpose of this amendment is to propose an upgraded Minimum Critical Power Ratio (MCPR) Fuel Cladding Integrity Safety Limit that reflects the loading of current fuel bundle designs. The current MCPR Fuel Cladding Integrity Safety Limit of 1.07 for reload cores (Ref. 1) was established in 1978 (Ref. 2) based on fuel design characteristics typical of those utilized at the time. The generic 1.04 MCPR Fuel Cladding Integrity Safety Limit for D-lattice plants is proposed to be applied to the second successive reload core of P8x8R, BP8x8R, GE8x8E, or GE8x8EB fuel types with high bundle R-factor (≥ 1.04).

2.0 EVALUATION

Amendment 14 (Ref. 3) to GESTAR II and responses (Refs. 4, 4a, and 4b) to the NRC request (Ref. 5) for additional information on the subject report have provided the basic supporting information for the staff review.

The rationale used by General Electric (GE) to derive an upgraded MCPR Fuel Integrity Safety Limit for high bundle R-factor GE fuel is based on two considerations: (l) utility's demand for higher energy cycles and more efficient fuel utilization; and (2) the loading of bundles with high R-factor resulting in increasing conservatisms with respect to the current MCPR safety limit of 1.07.

The analysis to derive the MCPR Fuel Cladding integrity Safety Limit has been performed using the previously approved correlation and methodology (Refs. 1, 2, 6, 7, and 8). All of the non-fuel related uncertainties used in the previously approved analysis given in Reference 1, Table S.2-l, remain the same. The only changes from the current analysis are in those areas which are impacted by the use of fuel with higher R-factor. The statistical analysis includes a design basis, a conservative operating case, and a typical operating case.

The relative bundle power histogram, the CPR/MCPR core map, the bundle R-factor distribution, and the CPR histogram associated with the design basis and the conservative operating case are provided in Sections 2.0 and 3.0 of Attachment 1 to Amendment 14, which also includes the CPR/MCPR core map associated with a typical operating state case. A new R-factor uncertainty is obtained using the latest approved lattice physics models (Refs. 9 and l0). Monte Carlo trials (Ref. 7) were performed to determine the MCPR Fuel Cladding Integrity Safety Limit using the inputs and updated uncertainties documented in Section 2.0 of Attachment 1 to Amendment 14.

The primary basis to assure a bounding value for the calculated safety limit MCPR is to select a core operating configuration which maximizes the number of fuel bundles at or near the MCPR operating limit. This is accomplished through the assumption of a large, equilibrium core and by deliberately

selecting control rod patterns which results in the highest number of fuel bundles at or near the MCPR operating limit.

The design basis case allows adjustment of all control rods in the core to produce the most severe power distribution. After a large number of trials, the power distribution with the highest number of fuel bundles operating at or near the MCPR operating limit was selected as the design basis case. This case had core fuel bundles among all cases near the MCPR operating limit as shown in Figure 2-2 (Ref. 3). The calculated safety limit MCPR for this case was 1.04.

The conservative case allows adjustment of only those control rods which are scheduled to be used during normal operation. Again, after a large number of trials, the power distribution with the highest number of fuel bundles operating at or near the MCPR operating limit was selected as the conservative case. This case had fewer fuel bundles than that of the design basis case near the MCPR operating limit as shown in Figure 3-2 (Ref. 3). The calculated safety limit MCPR for this case was 1.01. Although only scheduled rods were used in selecting the power distribution for this case, it was not typical of actual plant operation because the number of fuel bundles near the MCPR operating limit were deliberately maximized.

The typical case presented in Figure 3-4 (Ref. 3) is representative of conditions expected to occur during actual operation at times when the core is near its operating limit. This case had the least number of bundles near the MCPR operating limit. The safety limit MCPR was not calculated for this case but it is expected that it would be less than 1.00.

The analytical results indicate that the design basis calculation yields a conservative MCPR safety limit relative to a more realistic operating state since the design basis analysis conservatively bounds the actual plant conditions. This updated MCPR safety limit of 1.04 may be generically applied to D-lattice plants with a core which is operated with second successive reloads of high bundle initial R-factor (≥ 1.04) GE fuel (i.e., P8x8R, BP8x8R, GE8x8E or GE8x8EB). Under this applicable criterion the bundles nearest the MCPR operating limit during the operating cycle will be either the fresh bundles or those bundles exposed for one cycle only. The remaining exposed fuel (i.e., the fuel entering the third cycle of operation) could not be operated near their MCPR limits without driving the high bundle R-factor fuel to powers in excess of their Technical Specification limits.

Based on results of this review, the staff has found that GE has performed its statistical analyses including a design basis case using the previously approved methodology/code and has also established the criterion for application of the proposed upgraded MCPR safety limit to D-lattice plants, which are acceptable. Therefore, the staff concludes that Amendment 14 to the GE Licensing Topical Report NEDE-24011-P-A, "General Electric Standard Application for Reactor Fuel" is acceptable.

3.0 CONCLUSION

Based on the review discussed in Section 2.0 of this SER, the staff concludes that Attachment 1 to Amendment 14 to the GE Licensing Topical Report NEDE-24011-P-A provides an acceptable basis to incorporate an upgraded MCPR Fuel Cladding Integrity Safety Limit that reflects the loading of current fuel bundle designs.

The staff has concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and (2) such activities will be conducted in compliance with the Commission's regulations,

and the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

4.0 REFERENCES

1. “General Electric Standard Application for Reactor Fuel,” NEDE-24011-P-A-6, April 1983.

2. “Basis for 8x8 Retrofit Fuel Thermal Analysis Application,” NEDE-24131, September 1978.

3. Letter, J S. Charnley (GE) to C.O. Thomas (NRC), “Amendment 14 to General Electric Licensing Topical Report NEDE-24011-P-A,” October 2, 1985.

4. Letter, J.S. Charnley (GE) to G.C. Lainas (NRC), “Response to NRC Request for Additional Information for Amendment 14 to General Electric Licensing Topical Report NEDE-24011-P-A,” May 7, 1986.

5. Letter. J.S. Charnley (GE) to G.C. Lainas (NRC), “Resubmittal of Amendment 14 to GE Licensing Topical Report NEDE-24011-P-A,” January 26, 1987.

6. Letter, J.S. Charnley (GE) to M.W. Hodges (NRC), “Implementation of Reduced MCPR Safety Limit,” May 4, 1987.

7. Letter, M.W. Hodges (NRC) to J S. Charnley (GE), “Request for Additional Information for Amendment 14 to General Electric Licensing Topical Report NEDE-24011-PA,” March 7, 1986.

8. “General Electric BWR Thermal Analysis Basis (GETAB): Data, Correlation and Design Application,” NEDE-10958-P-A, January l977.

9. “General Electric BWR Thermal Analysis Basis (GETAB): Data, Correlation and Design Application,” NED0-10958-P-A, January 1977.

10. “Basis for BWR/6 8x8 Fuel Thermal Analysis Application,” NEDE-24196, June 1979.

11. Licensing Topical Report NEDE-30130-A, “Steady State Nuclear Methods,” May l985.

12. Letter, C.O. Thonas (NRG) to J.S. Charnley (GE, “Acceptance for Referencing of Licensing Topical Report, NEDE-30130, Steady State Nuclear Methods,” December 22, 1983.

NRC
Safety Evaluation Report
Approving Amendment 15
to
NEDE-24011-P

UNITED STATES
NUCLEAR REGULATORY COMMISSION
Washington, D. C. 20555
March 14, 1988

MFN 32-88

Ms. J.S. Charnley, Manager
Fuel Licensing
General Electric Company
175 Curtner Avenue
San Jose, California 95125

Dear Ms. Charnley:

SUBJECT: **Acceptance For Referencing Of Amendment 15 To General Electric Licensing Topical Report NEDE-24011-P-A, General Electric Standard Application For Reactor Fuel" (TAC No. 60903)**

The staff has completed its review of the subject amendment submitted by the General Electric Company by letter dated January 23, 1986. This Amendment 15 provides the description of the expanded database from full scale 8x8 fuel assemblies used for a refinement of the previously approved GEXL correlation. The refinement is intended to improve the accuracy of the boiling transition prediction in a form described as the GEXL-Plus correlation.

We find the Amendment 15 to be acceptable for referencing in license applications to the extent specified and under the limitations delineated in the Amendment 15 and the associated NRC technical evaluation. The evaluation defines the basis for acceptance of the amendment.

We do not intend to repeat our review of the matters described in Amendment 15 and found acceptable when the report appears as a reference in license applications, except to assure that the material presented is applicable to the specific plant involved. Our acceptance applies only to the matters described in Amendment 15.

In accordance with procedures established in NUREG-0390, lt is requested that GE publish accepted versions of this amendment, proprietary and non-proprietary, within three months of receipt of this letter. The accepted versions shall incorporate this letter and the enclosed evaluation between the title page and the abstract. The accepted versions shall include an -A (designating accepted) following the report identification symbol.

Should our criteria or regulations change such that our conclusions as to the acceptability of the report are Invalidated, GE 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,

Ashok C. Thadani, Assistant Director for Systems Division of Engineering & Systems Technology
Office of Nuclear Reactor Regulation
Enclosure:
Amendment 15 to NEDE-74011-P-A Evaluation

ENCLOSURE
SAFETY EVALUATION REPORT FOR AMENDMENT 15 TO GENERAL ELECTRIC LICENSING TOPICAL REPORT NEDE-24011-P-A

1.0 INTRODUCTION

By letter from J.S. Charnley (GE) to C.O. Thomas (NRC) dated January 23, 1986, General Electric (GE) suhmitted Amendment 15 (Reference 1) to NEDE-2401l-P-A, "General Electric Standard Application for Reactor Fuel" (GESTAR II), for NRC review. The purpose of this amendment is to incorporate the GEXL-PLUS boiling transition thermal limits correlation. The current GE critical quality-boiling length correlation (GEXL) (References 2a and 2b) was developed in 1970's as a means of predicting the occurrence of boiling transition in boiling water reactor (BWR) fuel. Since then, the GE fuel design has evolved through several product lines. Testing of these new fuel designs at the ATLAS Facility has expanded the existing boiling transition database for 8x8 fuel assemblies. A comparison of the current GEXL boiling transition prediction to this expanded database has shown a refinement is necessary to improve the accuracy of the boiling transition prediction for more accurate application to currently utilized fuel designs. The result of this refinement is a modified form of GEXL which is referred to as the GEXL-PLUS correlation (Reference 1).

2.0 EVALUATION

Amendment 15 (Reference 1) to General Electric Licensing Topical Report NEDE-2401l-P-A, "General Electric Standard Application for Reactor Fuel" (GESTAR II), (Reference 3) and responses (References 4 and 5) to the NPC requests (References 6 and 7) for additional information on the subject report have provided the basic supporting information for the staff review. The staff has coordinated this subiect review through our consultant, Pacific Northwest Laboratory (PNL). The detailed evaluation of the subject report was described in the PNL Technical Evaluation Report, FATA-87-108 (Reference 8).

Tables 3-2 and 3-3 of Reference 2a provide a listing of the data used to develop the GEXL correlation and used for checking the GEXL correlation, respectively. This database for the GEXL correlation, as currently incorporated in the GE thermal analysis basis (References 2a and 2b), is extremely broad containing those test data from the 7x7 and 8x8 assemblies that constituted GE's fuel products available in the 1970's. As GE's fuel products evolved and improved, it became apparent that a refinement of the previously approved GEXL correlation was necessary to improve the accuracy of boiling transition predictions for new fuel designs. To achieve this improvement, additional bundle testing was performed at the ATLAS Test Facility to expand the full-scale database for GE's new 8x8 fuel designs. The data generated by these assemblies comprise the expanded 8x8 database (Table 2-l of Reference 1).

GEXL-PLUS is based exclusively on expanded data generated from full-scale, 64-rod bundle tests of prototypical 8x8 fuel assemblies. The form of the GEXL-PLUS correlation is similar to that of GEXL with the difference being two additional terms in GEXL-PLUS to account for the effect of a specific flow regime on boiling transition. This modification of GEXL correlation is made to improve the accuracy of boiling transition predictions for new GE fuel design.

PNL has performed an evaluation on the GEXL-PLUS Correlation from both thermal-hydraulic and statistical perspectives (Reference 8). The statistical perspective considers the fitting, evaluation, and

validation of the GEXL-PLUS model form using GE's expanded database. The thermal-hydraulic perspective considers the mathematical form of the GEXL-PLUS model and the variables it is based on. The thermal-hydraulic portion of the review also compares GEXL-PLUS to its predecessor, GEXL. The GEXL-PLUS model is in general a better fit to the 8x8 database than GEXL, but the standard deviation of the mean CPR over the database (0.0268 as calculated by GE, 0.0272 in PNL's calculations, see Table 3.1 of Reference 8) is not an adequate measure of the uncertainty of the correlation. The uncertainty in a GEXL-PLUS prediction of critical power depends on the conditions for which the prediction is made, and is not constant over the claimed region of validity. The MCPR limit on the correlation must be specified such that the limit encompasses the regions where the correlation is nonconservative (for G_0.5, and Whsub_70 Btu/lbm), as well as the regions where it is conservative.

Based on our review in conjunction with PNL's recommendation (Reference 8, the staff has concluded the following:

1. The GEXL-PLUS correlation is in general a better fit to the 8x8 database than GEXL correlation.

2. The GEXL-PLUS correlation is applicable to CPR calculations over the operating range given below:

Pressure	800 to 1400 Psia
Mass Flux	0.1 to 1.50 M lbm/hr-ft^2
inlet subcooling	0 to 100 Btu/lbm
axial power profile	cosine, inlet peaked, outlet peaked
radial power peaking	1.03 to 1.27
fuel geometry	8x8 and 8x8 R

and subject to the conditions stated in Items 3 to 6.

3. The actual mean ECPR values should be used in MCPR design limit calculations in the regions of G 0.50 M lbm/hr-ft^2 and hsub 70 Btu/lbm where the GEXL-PLUS correlation appears to be nonconservative (i.e., mean ECPR 1.0).

4. A value of 0.037 should be used as the ECPR standard deviation of the GEXL-PLUS correlation for the entire range of application as stated in Item 1.

5. The tolerance interval multiplier used in calculating the MCPR limits should be chosen taking into account the number of ECPR data points used in computing the mean in each nonconservative region (i.e., regions where G_0.50 and Whsub_70). In regions where the mean ECPR is essentially unbiased or conservative (i.e., the mean ECPR _1.0), the limit can be determined as a correction on an assumed mean of 1.0 if desired.

6. An alternative to the use of an ECPR standard deviation of 0.037 as stated in Item 4 is for GE to partition the region of validity (as defined by the ranges on the thermal-hydraulic gnd

qeometric parameters) into homogeneous groups for the ECPRs (i.e., common populations), and compute MCPR using a mean ECPR and standard deviation for each population.

3.0 CONCLUSION

Based on the review discussed in Section 2.0 of this SEP, the staff concludes that Attachment 1 to Amendment 15 to the GE Licensing Topical Report NEDE-24011-P-A provides an acceptable basis to incorporate the GEXL-PLUS boiling transition thermal 1imits correlation.

The staff has concluded, based on the considerations discussed above, that: (l) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and (2) such activities will be conducted in compliance with the Commission’s regulations, and the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

4.0 REFERENCES

1. Letter, J.S. Charnley (GE) to C.O. Thomas (NRC), “Amendment 15 to General Electric Licensing Topical Report NEDE-24011-P-A,” January 23, 1986.

2. “General Electric BWR Thermal Analysis Basis (GETAB): Data, Correlation and Design Application,” NEDE-10958-PA, January 1977.

   a. “General Electric BWR Thermal Analysis Basis (GETAB): Data, Correlation and Design Application,” NEDO-10958-A, January 1977.

3. “General Electric Standard Application for Reactor Fuel,” NEDE-24011-P-A-7, Auqust 1985.

4. Letter, J.S. Charnley (GE) to G.C. Lainas (NRC), “Response to NRC Request for Additional Information for Amendment 15 to GE Licensing Topical Report NEDE-24011-P-A,” July 11, 1986.

5. Letter, J.S. Charnley (GE) to G.C. Lainas (NRC) “Response to Second NRC Request for Additional Information for Amendment 15 to GE Licensing Topical Report NEDE-24011-P-A,” January 16, 1987.

6. Letter, M.W. Hodges (NRC) to J.S. Charnley (GE), “Request for Additional Information for Amendment 15 to General Electric Licensing Topical Report NEDE-24011-P-A,” May 12, 1986.

7. Letter, M.W. Hodges (NPC) to J.S. Charnley (GE), “Second Round Request for Additional Information for Amendment 15 to General Electric Licensing Topical Report NEDE-24011-P-A,” November 6, 1986.

8. J.M. Cuta and G.F. Piepel, Review of General Electric Company Topical Report NEDE-24011-P-A Amendment 15 “GEXL-PLUS Boiling Transition Thermal Limits Correlation.,” FATA-87-108 (Proprietary), September 1987.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
Washington, D. C. 20555
May 5, 1988

Ms. J.S. Charnley, Manager
Fuel Licensing
General Electric Company
175 Curtner Avenue
San Jose, California 95125

Dear Ms. Charnley:

SUBJECT: Acceptance For Referencing Of Application Of Amendment 15 To General Electric Licensing Topical Report NEDE-24011-P-A, "General Electric Standard Application For Reactor Fuel" (TAC No. 60903)

The staff has completed its review of the subject Application of GESTAR-II Amendment 15 submitted by the General Electric Company by letter dated March 22, 1988. This subject submittal describes the application of the GEXL-PLUS correlation approved previously by USNRC.

We find the Application of GESTAR-II Amendment 15 to be acceptable for referencing in license applications to the extent specified and under the limitations delineated in the Amendment 15 and the associated NRC technical evaluation. The evaluation defines the basis for acceptance of the amendment.

We do not intend to repeat our review of the matters described in the Application of GESTAR-II Amendment 15 and found acceptable when the report appears as a reference in license applications, except to assure that the material presented is applicable to the specific plant involved. Our acceptance applies only to the matters described in the Application of GESTAR-II Amendment 15.

In accordance with procedures established in NUREG-0390, it is requested that GE publish accepted versions of this amendment, proprietary and non-proprietary, within three months of receipt of this letter. The accepted versions shall incorporate this letter and the enclosed evaluation between the title page and the abstract. The accepted versions shall include an -A (designating accepted) following the report identification symbol.

Should our criteria or regulations change such that our conclusions as to the acceptability of the report are invalidated, GE 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,

Ashok C. Thadani, Assistant Director for Systems
Division of Engineering & Systems Technology
Office of Nuclear Reactor Regulation

Enclosure:
Application of Amendment 15 to NEDE-24011-P-A Evaluation

ENCLOSURE
SAFETY EVALUATION REPORT FOR APPLICATION OF AMENDMENT 15 TO GENERAL ELECTRIC LICENSING TOPICAL REPORT NEDE-24011-P-A

1.0 INTRODUCTION

By letter from J.S. Charnley (GE) to M.W. Hodges (NRC) dated March 22, 1988, General Electric (GE) submitted its proposal for Application of GESTAR-II Amendment 15, with two Attachments (Reference 1) for NRC review. The purpose of this submittal is to address application of the GEXL-PLUS correlation (Reference 2) to derivation of Safety Limit Minimum Critical Power Ratio (SLMCPR) and Operating Limit Minimum Critical Power Ratio (OLMCPR).

2.0 EVALUATION

The staff's SER (Reference 3) incorporating our consultant's detailed evaluation (Reference 4) for GESTAR-II Amendment 15 (Reference 2) has identified the applicable ranges of the GEXL-PLUS correlation and has defined an over-all ECPR standard deviation of 0.037 to bound even the nonconservative regions of $G < 0.5M$ lbm/hr-ft^2 and Δh sub > 70 Btu/lbm.

To facilitate application of the GEXL-PLUS correlation, GE proposes three alternative adjustments to account for the nonconservative region identified in References 3 and 4 (one for high inlet subcooling, one for low mass flux and another for treatment of standard deviation).

GE proposes adjustment for the nonconservative behavior of GEXL-PLUS at inlet subcooling values above 70 Btu/lbm by applying a 3% bias to any evaluations where such values occur. The 3% value is acceptable based on the finding of Reference 1.

GE proposes adjustment for the nonconservative behavior of GEXL-PLUS at mass flux values below 0.5 Mlbm/hr-ft^2 by adjusting the mean ECPR according to the formula

$$ECPR = 1.0 + 0.32\ (0.4 - CF) \tag{1}$$

where CF is the fraction of rated core flow. This is an acceptable way to characterize the behavior of the mean ECPR in the region below $G = 0.5$ Mlbm/hr-ft^2, provided that CF always produces the same or conservative results when substituted for G, hot bundle flow in Mlbm/hr-ft^2. Therefore, plant specific applications should verify the validity of the substitution of CF for hot bundle flow.

GE proposes using their originally proposed ECPR standard deviation of 0.0268 in computing the SLMCPR and OLMCPR values as opposed to the value of 0.037 discussed in References 3 and 4. GE provides the following bases:

1. Conditions outside of the ranges 1000-1265 psig on pressure, 0.5-1.4 Mlbm/hr-ft^2 on mass flux, and 18-65 Btu/lbm on inlet subcooling are unlikely in the event that a limiting transient should occur.

2. Even though certain subregions within the restricted GEXL-PLUS region of applicability have standard deviation above 0.0268, the GEXL-PLUS behavior in these subregions is sufficiently conservative so as to offset the effects of larger standard deviations.

GE uses the equation

$$SLMCPR = 1.0\ (M-1.00) + 0.7\ (\sigma - 0.0268) \quad (2)$$

to evaluate the effects on SLMCPR of standard deviations greater than 0.0268 and means different than 1.00. Equation (2) was derived using a previously approved methodology (99.9% safety limit approach).

Based on our evaluation (Ref. 5), our conclusions on GE's proposals (Ref. 1) are given below:

1. The relation [1.0 + 0.32(0.4) – CF] for the mean ECPR for conditions with hot bundle flow less than 0.5 Mlbm/hr-ft^2 is acceptable, provided that plant specific applications verify that the fraction of core flow for conditions at or below 40% of rated flow is the same as or slightly less than the hot bundle flow in Mlbm/hr-ft^2.

2. The proposed 3% adjustment is adequate for offsetting the nonconservative behavior of GEXL-PLUS at inlet subcooling >70 Btu/lbm and is acceptable.

3. There are subregions of the GEXL-PLUS region of applicability for which the ECPR standard deviation is larger than the 0.0268 value proposed for use by GE. However, in these subregions GEXL-PLUS is either conservative enough, or the proposed adjustments for high inlet subcooling and low mass flux are conservative enough, so as to offset the effect of using 0.0268 instead of the actual standard deviations of the subregions. Therefore, we conclude that with the proposed adjustments to the mean ECPR, the use of the standard deviation of 0.0268 in GEXL-PLUS everywhere in its region of applicability is acceptable. The resulting safety and operating limits are at least as large as what would be obtatned using actual means and standard deviations.

3.0 CONCLUSION

Based on the review discussed in Section 2.0 of this SER, the staff concludes that the Application of GESTAR-II Amendment 15 with plant specific verification discussed above provides an acceptable basis for licensing applications.

The staff has concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and (2) such activities will be conducted in compliance with the Commission's regulations, and the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

4.0 REFERENCES

1. Letter, J.S. Charnley (GE) to M.W. Hodges (NRC), "Application of GESTAR-II Amendment 15," dated March 22, 1988.

2. Letter, J.S. Charnley (GE) to C.O. Thomas (USNFC), "Amendment 15 to General Electric Licensing Topical Report NEDE-24011-P-A," dated January 23, 1986.

3. Letter, A.C. Thadani (USNRC) to J.S. Charnley (GE), "Acceptance for Referencing of Amendment 15 to General Electric Licensing Topical Report NEDE-24011-P-A, "General Electric Standard Application for Reactor Fuel," dated March 14, 1988.

4. J.M. Cuta and G.F. Piepel, "Review of General Electric Company Topical Report NEDE-24011-P-A Amendment 15," FTS-87-108 Pacific Northwest Laboratory, Pichland, Washington, September 1987.

5. Letter, S.H. Bian (PNL) to T. Huang (USNRC), "Technical Evaluation Report on GEXL-PLUS application, FIN 2996," dated April 7, 1988.

NRC
Safety Evaluation Report
Approving Amendment 16
to
NEDE-24011-P

UNITED STATES
NUCLEAR REGULATORY COMMISSION
Washington, D. C. 20555
April 20, 1988

Ms. J.S. Charnley, Manager
Fuel Licensing
General Electric Company
175 Curtner Avenue
San Jose, California 95125

Dear Ms. Charnley:

SUBJECT: Acceptance for Referencing of Amendment 16 to General Electric Licensing Topical Report NEDE-24011-P-A, "General Electric Standard Application For Reactor Fuel"

The staff has completed its review of the subject amendment submitted by the General Electric Company by letter dated August 8, 1986. This Amendment 16 proposes changes to the GESTAR II base document and the U.S. Supplement. These changes are of an administrative nature primarily intended to follow more closely NRC standard format guidance.

We find the Amendment 16 to be acceptable for referencing in license applications to the extent specified and under the limitations delineated in the Amendment 16 and the associated NRC technical evaluation. The evaluation defines the basis for acceptance of the amendment.

We do not intend to repeat our review of the matters described in Amendment 16 and found acceptable when the report appears as a reference in license applications, except to assure that the material presented is applicable to the specific plant involved. Our acceptance applies only to the matters described in the Amendment 16.

In accordance with procedures established in NUREG-0390, it is requested that GE publish accepted versions of this amendment, proprietary and non-proprietary, within three months of receipt of this letter. The accepted versions shall incorporate this letter and the enclosed evaluation between the title page and the abstract. The accepted versions shall include an -A (designating accepted) following the report identification symbol.

Should our criteria or regulations change such that our conclusions as to the acceptability of the report are invalidated, GE 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,

Ashok C. Thadani, Assistant Director for Systems
Division of Engineering & Systems Technology
Enclosure:
Amendment 16 to NEDE-24011-P-A Evaluation

ENCLOSURE
SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION RELATING TO AMENDMENT 16 GENERAL ELECTRIC TOPICAL REPORT NEDE-24011-P "GENERAL ELECTRIC STANDARD APPLICATION FOR REACTOR FUEL"

1.0 INTRODUCTION

In a letter dated August 8, 1986, (Ref. 1) General Electric (GE) submitted a proposed anendment to NEDE-24011, "General Electric Standard Application for Reactor Fuel (GESTAR II) (Ref. 2). This amendment is considered an '"administrative amendment," and is designated Amendment 16 to Revision 7 of NEDE-24011. (Effectively the changes are to Revision 8, which includes Amendments through 13.) Additional information and some changes to the original submittal were provided in response to questions in an addendum dated November 11, 1987 (Ref. 3).

An administrative amendment does not present new technical information to be reviewed, but rather incorporates approved material and such changes as are needed to improve the form, consistency or content of this report. The purpose of the present review is to verify that the changes introduced Amendment 16 are of such an administrative nature, and that they are acceptable. The staff review of this amendment was performed with the assistance of consultants from Brookhaven National Laboratory (BNL).

The GESTAR II report provides generic information related to the fuel design and analyses of General Electric Boiling Water Reactor plants for which General Electric provides fuel. The report outlines the fuel design, fuel thermal-mechanical analysis, nuclear and hydraulic analysis methods used in reload analyses. The report provides information that is independent of a plant-specific application. Plant-specific information and methods used to determine reactor limits are given in the U.S. Supplement accompanying the base document. The generic information presented in GESTAR II is supplemented for reload submittals by plant-cycle-unique information and analytical results.

Both the base document and the U.S. Supplement are periodically updated via amendments to specific sections of these documents as new or additional data related to the fuel design, thermal-mechanical analysis or nuclear and hydraulic methods are developed.

2.0 EVALUATION

Amendment 16 proposes substantial rewrites of both the GESTAR II base document and the U.S. Supplement. The documents were rewritten not to change technical content but rather to more closely follow the NRC Standard Review Plan (NUREG-0800) and Regulatory Guide 1.70 (Refs. 4 and 5) formats for the fuel sections of a standard FSAR and, in the process, to eliminate information not necessary for NRC review of BWR fuel reloads. Most sections of both the base document and supplement and their appendices have undergone some (proposed) changes, ranging from minor editorial changes through major rewrites to complete elimination or transfer to separate documents. As a brief summary of these changes (as given in Ref. 1), Sections 1 through 4 as well as Appendix A of the base document have been amended, and appendices B, C and D have been eliminated, (Appendix B is considered to be redundant to Appendix A and Appendices C and D have been transferred to documents to be published), Section S and Appendix A of the U.S Supplement have been amended for

compatibility with the rest of the GESTAR II document and Appendices B and C have not undergone any changes.

The primary purpose of the review of Amendment 16, is to determine that presently existing (through Amendment 13) GESTAR II technical information, necessary for the review of individual utility reloads and other related submittals, is not significantly reduced and that existing NRC approved data, methodology and criteria are not altered by the transformations of the amendment.

The review by BNL of the initial submittal (Ref. 1) concluded that the changes were, for the most part, generally acceptable administrative changes. However, in a number of areas, existing material necessary for reload review was apparently deleted in the process of change. These concerns with missing material were discussed with 6E and resulted in a supplemental submittal by GE (Ref. 3) indicating revisions to the amendment to supply the missing material or explaining the deletions.

The following listing briefly describes the changes to each section, including the resolution of the deletions indicated in the initial BNL review, and the acceptability of the changes.

Base Document

Section 1.0

Introduction (and items such as Table of Contents): Only minor editorial changes reflecting the changes throughout the volume were made. These are acceptable administrative changes.

Section 2.0

Fuel Mechanical Design: The general structure of this section remains the same and only editorial changes are made. Since appendices C and D of this volume are to be removed and made into separate documents, references in this section are now modified to include these documents. The review has concluded that the changes are administrative and acceptable.

Section 3.0

Nuclear Design: This section has undergone an extensive rewrite and format change in order to conform to the format of Regulatory Guide 1.70. The BNL review had indicated that the technical information has not changed except for the deletion of the sections on local power peaking and on the standby liquid control system (SLCS). In response to questions GE, in Reference 3, has indicated that the section on the SLCS will be added back into GESTAR II, and that the discussion on how local peaking factors are used in limit analysis is not a part of Regulatory Guide 1.70 specifications and is thus not needed in GESTAR II. These are reasonable responses and the review has concluded that the changes to this section are appropriate and acceptable administrative changes.

Section 4.0

Thermal Hydraulic Design: As with Section 3.0, this section is rewritten to conform to the format of Regulatory Guide 1.70. In addition discussions of the operating and the safety limit MCPRs (Minimum Critical Power Ratio) have been relocated into this section from the U.S. Supplement. The review had indicted that there are no technical changes in this reformatting and the result is an acceptable administrative change. The only deletion in the original Section 4.0 material is the discussion of bypass flow determination, which has been replaced (see Ref. 3) with a reference to the GE report NEDE-21156, from which the GESTAR II discussion was taken. Since all GE fuel bundles

are now equipped with finger spring seals that discussion in the bypass flow section has been removed. These changes are acceptable. In the relocated U.S. Supplement material the figures giving bundle power and CPR histograms have been deleted and replaced by references to the GE report NEDE-10958 which is the source of this information. This change is also acceptable.

Appendix A

Safety Analysis Report Roadmaps: The format of this section is largely unchanged, but some material has been rewritten to make it compatible with the previously discussed format changes in the base document. The review has found these changes generally acceptable, with the exception of some apparently missing material, primarily related to referencing. This material is discussed in the GE response to questions (Ref. 3). The material is either in other sections of the amendment, is indicated as being supplied by applicants, is no longer applicable (e.g., current analysis with GEMINI methods eliminates need for the discussion of maximum operating conditions) or is to be corrected. These explanations and corrections are reasonable and acceptable.

Appendix B

FSAR Example: This appendix is basically redundant to Appendix A and has been eliminated. This is reasonable and acceptable.

Appendix C

Thermal Mechanical Bases, and

Appendix D

Bundle-Specific Design Data: These appendices (which describe fuel bundles and discuss design models) have been removed entirely and it is proposed (by GE) that they be published as separate documents. They would become NEDE-31151-P and NEDE-31152-P and are referenced in GESTAR II. These reports have been submitted as part of the GESTAR II Amendment 18 material and the specifics of the acceptance of this change will be addressed in the review of that amendment. However, the relocation of this material is acceptable.

U.S. Supplement

Section S

Transient and Accident Limits and Analyses: This section has been rewritten and reorganized to follow the frequency categories defined in Chapter 15 of Regulatory Guide 1.70 for the significant events of Final Safety Analysis Report (FSAR) Chapter 15 events. This discussions of ASME vessel pressure code compliance, stability analysis and reload analysis options have also been reformatted but contain the same information. As previously discussed, some information on safety limits has been moved to the base document. Also discussion (and related figures) of the effects of fuel densification (particularly gaps) on the operating limit MCPR has been deleted since the issue has been resolved and is not needed in GESTAR II. Two deleted figures on the interaction of LOCA analyses methods will be returned (Reference 3). The review has found that no significant technical changes have been made and that the reformatted presentation is a reasonable and acceptable administrative change.

Appendix US-A

Standard Supplemental Reload Submittal: This section is a format example for reload submittals. Minor editorial changes have been made which are consistent with the changes discussed above for this amendment. These changes are acceptable.

Appendices US-B and US-C

No changes have been made to these sections.

3.0 CONCLUSION

It is concluded, based on the above review, that, with the incorporation of the minor changes indicated by GE in Reference 3, the changes to GESTAR II proposed by Amendment 16 are of an administrative nature and that no new and unreviewed technical area has been introduced. The changes that have been made are acceptable.

4.0 REFERENCES

1. Letter (and attachments) from J.S. Charnley, GE, to H.N. Berkow, NRC, dated August 8, 1986, "Proposed Amendment 16 to GE Licensing Topical Report NEDE-240l1-P-A."

2. GE Topical Report NEDE-24011-P-A, "General Electric Standard Application for Reactor Fuel."

3. Letter (and attachments) from J.S. Charnley, GE, to M.W. Hodges, NRC, dated November 11, 1987, "Response to Reviewer Comments on GESTAR Amendment 16."

4. NUREG-0800, "U.S. Nuclear Regulatory Commission Standard Review Plan."

5. Regulatory Guide 1.70, "Standard Format and Content of Safety Analyses Reports for Nuclear Power Plants."

NRC
Safety Evaluation Report
Approving Amendment 17
to
NEDE-24011-P

UNITED STATES
NUCLEAR REGULATORY COMMISSION
Washington, D. C. 20555
December 27, 1987

Ms. J.S. Charnley, Manager
Fuel Licensing
General Electric Company
175 Curtner Avenue
San Jose, California 95125

Dear Ms. Charnley:

SUBJECT: ACCEPTANCE FOR REFERENCING OF LICENSING TOPICAL REPORT NEDE-24011-P-A, "GENERAL ELECTRIC STANDARD APPLICATION FOR REACTOR FUEL", REVISION 8, AMENDMENT 17

We have completed our review of the subject topical report submitted by the BWR Owners Group (BWROG) by letter dated August 15, 1986. The report contains proposed changes to NEDE-24011 and an attached report by BWROG and General Electric (GE) justifying those changes. We find the report to be acceptable for referencing in license applications to the extent specified and under the limitations delineated in the report and the associated NRC evaluation, which is enclosed. The evaluation defines the basis for acceptance of the report.

We do not intend to repeat our review of the matters described in Amendment 16 and found acceptable when the report appears as a reference in license applications, except to assure that the material presented is applicable to the specific plant involved. Our acceptance applies only to the matters described in the report.

In accordance with procedures established in NUREG-0390, it is requested that GE publish accepted versions of this amendment, proprietary and non-proprietary, within three months of receipt of this letter. The accepted versions shall incorporate this letter and the enclosed evaluation between the title page and the abstract. The accepted versions shall include an -A (designating accepted) following the report identification symbol.

Should our criteria or regulations change such that our conclusions as to the acceptability of the report are invalidated, GE 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,

Ashok C. Thadani, Assistant Director for Systems
Division of Engineering & Systems Technology

Enclosure:
As stated

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION RELATING TO AMENDMENT 17 GENERAL ELECTRIC TOPICAL REPORT NEDE-24011-P "GENERAL ELECTRIC STANDARD APPLICATION FOR REACTOR FUEL"

1.0 INTRODUCTION

In a letter of August 15, 1986 from T.A. Pickens, Chairman of the BWR Owners Group (BWROG), an Amendment 17 to the General Electric (GE) Topical Report NEDE-24011-P (GESTAR II) was proposed (Ref. 1). Attached is a BWROG and GE report, "Modification to the Requirements for Control Rod Drop Accident Mitigating Systems", along with proposed changes to GESTAR II and sample changes to Technical Specifications relating to the mitigating systems.

This submittal is a request to (1) eliminate the required use of the rod sequence control system (RSCS) on those reactors having such a system, while retaining the rod worth minimizer (RWM) to provide backup to the operator for control rod pattern control, and (2) lower the setpoint for turn off of the RWM (or RSCS) to 10 percent of full power from its current 20 percent level.

The report indicates that the primary justifications for these proposals are that (1) the RSCS is redundant to the RWM and is therefore not needed to integrate the control rod drop accident (RDA), and (2) existing calculations demonstrate that the RDA is not of significant concern above 10 percent power and therefore a mitigation system is not needed for higher power level operation.

The proposal also indicates that additional justifications or incentives for the RSCS removal are provided by: (a) Existing NRC sponsored RDA analysts methodology improvements show significantly less severe peak fuel enthalpy for a given dropped rod reactivity worth; (b) an existing NRC probability study demonstrates an extremely low probability for an event exceeding fuel damage criteria: and (c) RSCS elimination will reduce operation complexity and startup-shutdown times and will permit quicker power reduction in emergency conditions such as ATWS, particularly for Group Notch RSCS (GNRSCS) reactors.

2.0 BACKGROUND

The RSCS is a hard wired (as opposed to a computer controlled) system designed to monitor and block when necessary operator control rod selection, withdrawal and insertion actions, and thus assist in preventing significant control rod pattern errors which could lead to a control rod with a high reactivity worth (if dropped). A significant pattern error is one of several abnormal events all of which must occur to have a RDA which might exceed fuel energy density limit criteria for the event. It is designed only for possible mitigation of the RDA and is active only during low power operation (currently generally less than 20 percent power) when a RDA might be of significance. It provides rod blocks on detection of a significant pattern error. It does not prevent a RDA. A similar pattern control function is also performed by the RWM, a computer controlled system. All reactors having a RSCS also have a RWM.

BWRs beginning with Oyster Creek have all had a RWM for RDA rod pattern control. (BWR6s do not.) The RSCS is the result of NRC staff and consultant reviews in the early 1970S of GE RDA analyses methodology. These reviews concluded that the dropped rod and scram reactivity insertion rates being used were not suitably conservative. The corrected and improved, but still very

conservative, consultant and GE calculations immediately following these reviews indicated that previously calculated margins to the acceptance criteria for peak fuel enthalpy in a licensing RDA event analysis (assuming a maximum non-error rod worth) had been significantly reduced. They also indicated that, unlike previous conclusions, single errors in rod patterns could, in some cases, lead to rod worths for which a RDA could exceed limits. This was particularly likely to be the ease for the part length gadolinium controlled cores which were then being introduced, and for cores in which axial burnup is accounted for in the analysis.

These changes in analysts led to a NRC stiff interest in improved prevention of high reactivity worth error rods. There was at the time also staff dissatisfaction with the adequacy of the RWM and its use (to be discussed in Section 3). Interaction between the staff and GE led to the development by GE of the RSCS and its required installation for reactors coming on line. (There was also a tightening of mechanical Specifications (TS) for the RWM on operating reactors.) The near term BWR4 reactors (except Vermont Yankee), a total of 13 reactors, received the Group Notch version of the RSCS (GNRSCS), and later BWR4 and 5 reactors, a total of 9, an improved RSCS not requiring the group notch restriction, but rather using a Banked Position Withdrawal Sequence (BPWS) described in Reference 4. (Group notch operation requires rod group bank steps of only a single notch for control rod densities less than 50 percent, and this requires considerable time for that segment of startup or shutdown operations, and it can not enforce the improved BPWS patterns for rod densities above 50 percent. BPWS operation uses considerably fewer group steps and is much less complex.) All of the RSCS reactors retain a RWM as a parallel system. BWR6s have a pattern control system not involving the RSCS or RWM. All BWR 2 and 3 reactors plus Vermont Yankee, 10 reactors, have only a RWM.

Subsequent to the adoption of the RSCS for new reactors, the question of backfit of a RSCS for operating reactors arose. To answer this an extensive probability study of the likelihood of exceeding peak enthalpy limits as a result of a RDA was carried out by the staff. It concluded that there is no need for backfit of a RSCS (and none was done, Reference 5) and no evident need for it on new reactors (but the requirement was none-the-less continued). This probability study is presented in an appendix in Reference 2 which is a general discussion by the staff for the ACRS of RDA problems and solutions. (This reference may also be used as general background material.) The ACRS reviewed the probability study, along with other RDA methodology, during its examination of BWR RDA problems under the ACRS Generic Item II.A.2 (later designation II.A.1), and indicated it considered the problems to be resolved in its 1977 status report (Reference 7). The probability study will be further discussed in Section 3.

Also, subsequent to the RDA analysis methods studies and changes which led to the RSCS requirements, the methodologies have been further investigated and improved by staff consultants at Brookhaven National Laboratory (BNL) in areas of three dimensional and thermal feedback modeling (see for example Reference 3). The results of calculations with these methods indicate significantly lower peak fuel enthalpy values for a given rod worth compared to GE calculations using the standard, more approximate and conservative methodology in use during the RSCS development time frame (and used ever since). While there is some uncertainty with the improved methods about some aspects of the thermal-hydraulic feedback (see Section 3.2) even a conservative interpretation of the calculations indicates that maximum single error RDA events are not likely to exceed criteria.

3.0 EVALUATION

As indicated in the background discussion, there were at the time (about 1973) RSCS requirements were promulgated by the staff a number of ”perceived” problems or unknown factors relating to the

RDA which were of significance to the decision. The perceived situation leading to the RSCS nay be summarized as follows:

1. With the GE methodology and the (then) newly required reactivity insertion and scram rates, rod worths achievable without pattern error produced RDA analyses results approaching enthalpy limits. Even minor pattern errors, therefore, could lead to exceeding limits, and therefore any pattern error should be (redundantly) prevented.

2. The RWM was not effectively required by TS and was poorly maintained or improved and frequently bypassed and thus provided no significant protection. Furthermore, computer related protection systems were (then) inherently distrusted. Second operator substitution for the RWM was used routinely and is apparently providing minimal protection since procedures and quality control were frequently poor.

3. There existed no (trusted) study of the probability of exceeding enthalpy 1imits is a result of a RDA and improved calculational methods providing none realistic modeling of an event were only in early stages of development and use. There was therefore nothing to alleviate concerns arising from the above perceptions.

Since that time information about these areas has been significantly expanded or modified, leading to a revised perception of the problem area.

3.1 Probability Study

A primary factor in the evaluation of the present proposal is the staff probability study which was performed to provide a basis for examining RSCS backfit requirements on then operating reactors. That study (Ref. 2) carried out (in 1975) an independent analysis of the probabilities, individual and combined, of the multiple events that must occur in order that a RDA exceed the staff acceptance criteria of 280 cal/gm in the peak fuel pellet.

To have the possibility of exceeding limits a RDA must occur, and must occur with reactor conditions and related parameters within a narrow range of a much broader set of possible conditions. The event must involve all of the following: (l) a drive-blade disconnect, (2) which is not discovered before rod drop occurs, (3) the blade must stick, (4) and not be discovered, (5) the sticking must occur in upper 1/6 of core, (6) the drive must be lowered at least 2-3 feet, (7) an incorrect rod pattern must have been selected and pulled and, (8) the error not detected, (9) the error must directly involve the dropped rod and, (10) the error must provide an unusually high worth for that rod, (11) the rod blade must unstick and drop, (12) the drop must occur at low power (less than 10%), (13) it must occur when the relevant overall rod pattern is such as to enhance the rod worth (a small fraction of pattern development time).

The study determined conservative probabilities for these occurrences and their combination. It concluded that a reasonable (and quite possibly conservative) estimate of the probability of having a RDA exceeding 280 cal/gm is about 10^{-12} per reactor year. This is a large margin to an acceptance criterion of 10^{-7} per reactor year, and allows for considerable uncertainty in the input information or unforeseen interactions among elements of the analysis."

It may be noted that about 10 times the number of reactor years included in that study have accumulated since the probabilities were developed. There has been no occurrence of a rod drop or even a combination of any two of the necessary initiating events listed above (e.g., 1, 3, 7). The

increased statistical data indicate that the individual probabilities used in the study have remained about the same or decreased, and the combination would be significantly smaller.

This study was done and its conclusions were reached assuming that neither the RSCS or RWM were in use. It thus concluded that backfit of a RSCS on an operating reactor was not needed. It also concluded that the study did not appear to demand the extra probability decrease which could be achieved from use of the RWM. It nevertheless recommended that it be retained and that TS be such is to assure a reasonable degree of operability.

3.2 RDA Analysis Methodology

The probability study was performed assuming results from the GE RDA calculation methodology. In the years since, BNL has continued studies of RDA methodology and results have indicated a substantial reduction in enthalpy for a given rod worth as a result of better geometrical and moderator reactivity feedback modeling. The results from Reference 3, for example, for ”zero” power RDA events indicate very low peak enthalpies (less than about 130 cal/gm for maximum rod worths (about 1.5% delta k) assuming no pattern errors, and less than 200 cal/gm for maximum rod worths (about 2.5% delta k) which might exist assuming maximum single error patterns. These enthalpies are for a large degree of moderator subcooling. The calculations give much lower enthalpies for lower subcooling margin conditions usually found in relevant BWR operating conditions. However, these very low values are currently uncertain because of uncertainties about the degree of superheating rather than voiding which might occur in the moderator under the very rapid transient heating conditions. However, a conservative conclusion from these results (using moderate and large subcooling to mock up delayed voiding) is that there is a large likelihood that error patterns would not lead to a rod worth which could exceed limits in a “zero” power RDA. The study also clearly showed that in the 10 percent power range (and above) peak enthalpies would always be well below limits.

The probability level of exceeding the 280 cal/gm fuel enthalpy limit as determined in the 1975 study could be further reduced by giving credit for current improved analysis methods and results and for the use of BPWS rod patterns which reduce the expected maximum rod worths.

3.3 RWM Operation

At the time the RSCS requirements were being developed the staff also was changing the TS requirements for the RWM in the operating reactors without an RSCS. Whereas previously the TS had generally permitted RWM operation to be replaced by a second operator check off system without further restrictions, the TS were altered to require more mandatory use of the RWM. Various forms of these changes were developed by the staff project managers, but they were equivalent in requiring at least partial active use of the RWM for low power operation throughout the reactor cycle. This appears to have resulted in greatly improving the RWM availability and use, and reducing the routine use of the second operator. It demonstrated that the RWM could be a reliable system for providing any necessary Limiting Condition of Operation (LCO) surveillance. (Note that the RSCS is a LCO surveillance system.) Furthermore in the intervening time, computer controlled LCO surveillance and even protection systems have become recognized as acceptable by the staff.

Examples are the Combustion Engineering COLSS and CPC surveillance and protection systems, and the GE process computer system which is a recognized acceptable system for power density LCO monitoring in BWRs. A more direct example of RWM acceptance can be noted in the approval (Ref. 6) of BPWS generic RDA statistical analysis credit for GNRSCS reactors electing BPWS for 50 to

100 percent rod density, a pattern which can only be monitored by the RWM, not the RSCS. Thus the RWM can currently be considered an acceptable system for rod pattern control.

The proposal for GESTAR II changes and TS change examples for reactors electing to remove the RSCS do not propose any minimal RWM use or restrictions on the substitution of second operators for the RWM. The staff review of this proposal and the past effectiveness of restrictions on RWM bypass has concluded that the TS accompanying a removal of the RSCS should provide for initializing operation without the active use of the RWM.

The specifications should provide strong incentives for RWM maintenance and use without engendering excessive operational restrictions. Furthermore, the review concludes that the occasional necessary use of a second operator RWM replacement should be strengthened by a utility review of relevant procedures, related forms and quality control to assure that the second operator provides in effective and truly independent monitoring process. A discussion of this review should accompany the request for RSCS removal.

3.4 RSCS Requirements

It can be concluded from the above discussion that the “perceived” problems listed at the beginning of Section 3.0 have been satisfactorily addressed since the RSCS was required on new reactors. The probability study, improved calculations and RWM acceptability have alleviated the concerns expressed at that time. The changes in the three listed areas may be briefly summarized as follows.

1. Improved methodology studies (and to a lesser extent improved rod patterns) indicate that a RDA involving no errors in the rod pattern would result in peak enthalpy far below NRC limits, and even with maximum single error (and most multiple error) patterns would not exceed limits. Furthermore the probability level of exceeding enthalpy limits is very low even without consideration of these greater margins.

2. RWM TS improvements have improved RWM reliability and use, and computer controlled systems are considered acceptable.

3. Probability and methods studies have advanced sufficiently to modify initial perceptions.

The review thus concludes that the RSCS is not needed and operation without it is acceptable. However, the review also concludes, as previously indicated, that suitable provisions should be made for RWM use and operator backup, and rod patterns should be in accord with BPWS concepts.

The proposal indicates that in addition to the lack of need for the RSCS there is further incentive for its removal based on the added system and operation complexity and time consumption produced by the RSCS. This is particularly true for the GNRSCS reactors for which many hours of tedious operator actions are added to low power operations. The review concludes that these are appropriate incentives for RSCS removal, given the conclusion that operation is acceptable without the RSCS.

3.5 RWM Cutoff Power Level

In addition to the removal of the RSCS the proposal requests a lowering of the cutoff power level for required RWM (and RSCS where still applicable) operation from 20 to 10 percent power. This is based on existing calculations which demonstrate that no significant RDA can occur above 10 percent power. This is acceptable since both the old GE calculations and, as previously discussed, the BNL

calculations indicate that by 10 percent power peak fuel enthalpy is reduced well below required limits, even for significant error patterns. This is because of the reduced available rod worth and more effective action of feedback mechanism (even when assuming no direct moderator heating). The 20 percent limit was required as an extreme bound because of then existing uncertainties in the analysis. Based on current analyses the 10 percent level is acceptable.

3.6 TS and GESTAR II Changes

The removal of the RSCS and lowering of the RWM turnoff setpoint requires some TS changes. These would vary in detail for various reactors depending on existing TS format. The proposal presents sample TS changes which could be followed or used as a model. These examples deal with only RSCS removal (removing references to the RSCS) and lowering of the setpoint. These are satisfactory models for that purpose, but are not sufficient. As previously discussed, the staff review has concluded that RWM TS should require provisions for minimizing operations without RWM use as was previously done for operating reactors. Thus a RSCS removal request should include such TS changes. In addition, as also discussed, the request should present a discussion of the review of second operator procedures. Finally it is recommended that a BPWS pattern or its equivalent (or an improved version such is the Reduced Notch Worth Procedure) be used in order to reduce potential maximum rod worths.

The submittal included proposed changes to GESTAR II (as Amendment 17 to NEDE-24011-P) necessary to recognize the removal of the RSCS and the change in the setpoint in that document. These are straight forward, editorial type changes and are acceptable. However, there should also be in GESTAR II some direct statement, beyond the proposed references to the BWROG letter and associated report (Reference 1), of the above requirements for TS change and second operator procedures review.

4.0 CONCLUSIONS

At the time the RSCS and 20 percent power rod pattern control cutoff were required for new reactors for RDA mitigation there were numerous perceived problems associated with the RDA, In the intervening time these problems have been resolved and the need for the RSCS (and the RWM above 10 percent power) is no longer apparent. The probability study, independent improved RDA calculation results, BPWS rod patterns, and improved RWM operability have contributed to this resolution. Furthermore, there is sufficient incentive in the reduction of unneeded operational complexity and in some cases wasted low power operating time and emergency power reduction to justify removal of the system. The review has thus concluded that it is acceptable to remove the TS requirements for the RSCS for those BWR 4 and 5 reactors which have either the GN or BPWS RSCS, and to lower the turnoff setpoint for the (required) RWM to 10 percent power.

The proposed Amendment 17 changes to GESTAR II indicating these changes are acceptable is are the proposed example TS changes. However, it is required that the TS for the RWM for these reactors be altered to require use of the RWM to an extent which would minimize substitution of a second operator and thus provide a strong incentive to maintain and improve that system. It is also required that utilities requesting RSCS removal review the procedures, independence and quality control for second operator substitution and provide a discussion of that review. Finally, it is recommended that rod patterns used for these reactors be at least equivalent to BPWS patterns. These requirements should be directly stated in GESTAR II.

5.0 REFERENCES

1. Letter from T.A. Pickens, BWROG, to G.C. Lainas, NRC, dated August 15, 1986, "Amendment 17 to General Electric Licensing Topical Report NEDE-24011-P-A."

2. Letter and enclosure from B.C. Rusche, NRR, to R. Fraley, ACRS, dated June 1, 1976, "Generic Item IIA-2 Control Rod Drop Accident (BWRs)."

3. BNL-NUREG 28109, "Thermal-Hydraulic Effects on Center Rod Drop Accidents in a Boiling Water Reactor", H. Cheng and D. Diamond, October 1980.

4. NEDO-2131, "Banked Position Withdrawal Sequence", C. Paone, January 1977.

5. Memorandum from V. Stello, NRR, to K. Goller, NRR dated July 21, 1975, "DTR Determination on the Need to Backfit Certain BWRs with Additional RDA Protection."

6. Letter and attachment from C. Thomas, NRC, to J. Charnley, GE, dated October 11, 1985, "... NEDE-24011 ... Revision 6, Amendment 12."

7. Letter to J. Hendrie from M. Bender, "Status of Generic Items Relating to Light-Water Reactors: Report No. 6", November 15, 1977.

NRC
Safety Evaluation Report
Approving Amendment 18
to
NEDE-24011-P

UNITED STATES
NUCLEAR REGULATORY COMMISSION
Washington, D. C. 20555
May 12, 1988

Ms. J.S. Charnley, Manager
Fuel Licensing
General Electric Company
175 Curtner Avenue
San Jose, California 95125

Dear Ms. Charnley:

SUBJECT: ACCEPTANCE FOR REFERENCING OF AMENDMENT 18 TO GENERAL ELECTRIC LICENSING TOPICAL REPORT NEDE-24011-P-A, “GENERAL ELECTRIC STANDARD APPLICATION FOR REACTOR FUEL”

The staff has completed its review of the subject amendment submitted by the General Electric Company by letter dated October 31, 1986. This Amendment 18 primarily presents information for the incorporation of the GE8x8NB fuel bundle design into NEDE-24011-P-A. Also included in the amendment are some editorial changes, and the former Appendices C and D of the base document have been replaced with separate reports.

We find the Amendment 18 to be acceptable for referencing in license applications to the extent specified and under the limitations delineated in the Amendment 18 and the associated NRC technical evaluation. The evaluation defines the basis for acceptance of the amendment.

We do not intend to repeat our review of the matters described in Amendment 18 and found acceptable when the report appears as a reference in license applications, except to assure that the material presented is applicable to the specific plant involved. Our acceptance applies only to the matters described in the Amendment 18.

In accordance with procedures established in NUREG-0390, it is requested that GE publish accepted versions of this amendment, proprietary and non-proprietary, within three months of receipt of this letter. The accepted versions shall incorporate this letter and the enclosed evaluation between the title page and the abstract. The accepted versions shall include an -A (designating accepted) following the report identification symbol.

Should our criteria or regulations change such that our conclusions as to the acceptability of the report are invalidated, GE 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,

Ashok C. Thadani, Assistant Director
for Systems
Division of Engineering & Systems Technology

Enclosure:
Amendment 18 to NEDE-24011-P-A Evaluation

ENCLOSURE
SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION RELATING TO AMENDMENT 18 TO GENERAL ELECTRIC TOPICAL REPORT NEDE-24011-P-A "GENERAL ELECTRIC STANDARD APPLICATION FOR REACTOR FUEL"

1.0 INTRODUCTION

In a letter dated October 31, 1986 (Ref. 1), General Electric (GE) submitted a proposed Amendment 18 to NEDE-24011, "General Electric Standard Application for Reactor Fuel" (GESTAR II) (Ref. 14). The primary purpose of this amendment is to incorporate the GE8x8NB fuel design into the GESTAR II document. The original submittal was supplemented with an additional request for changes (Ref. 2), and was supplemented with additional information submitted with the letters of References 7, 8, 13, 15, 16, and 17. These information submittals either expanded on the initial information or answered questions raised by the staff or its consultants. The staff review of the amendment, including the supplemental information, was performed with the assistance of consultants from Brookhaven National Laboratory.

Amendment 18 presents proposed changes to the GESTAR II base document and to the U.S. Supplement, primarily information in support of the GE8x8NB fuel design. In addition, several editorial changes have been introduced in this amendment for the purpose of improving the format and accuracy of NEDE-24011-P-A. These editorial changes make no technical changes and have been found acceptable. As indicated in Amendment 16 and further discussed in this amendment, appendices C and D have been removed from the base document and are being issued as separate documents (Refs. 3 & 4).

The GE8x8NB design is very similar to the NRC approved GE8x8EB design (Refs. 5 & 6). The major changes between these two fuel types are (a) a large central water rod (LCWR), replacing the individual water rods, and (b) a U^{235} axial enrichment distribution. The new fuel design has been analyzed with approved methods. New data has been added for the new fuel for the GEXL correlation and a revised correlation for the new fuel, GEXL-PLUS/GE8x8NB, is presented in the amendment.

2.0 EVALUATION

2.1 Fuel Thermal-Mechanical and Hydraulic Design

The thermal-mechanical fuel rod design of the GE8x8NB fuel is the same as that of the approved GE8x8EB fuel. The GE8x8NB fuel design limits for extended burnup operation are the same as those for the GE8x8EB fuel. The water rod is not subjected to significant axial stresses and no significant bowing deflection is expected (Response 11, Reference 10).

The major geometrical changes included in the GE8x8NB fuel are a bundle design with 60 fuel rods and one large central water rod, and a high performance spacer. In order to optimize the pin power distribution, the U^{235} enrichment varies radially and axially within the fuel bundle. Within each reload

bundle selected fuel rods contain various amounts of gadolinia (Gd_2O_3). This burnable absorber is distributed both radiallv and axially within the fuel assembly.

The modifications to the bundle geometry incorporated in the GE8x8NB fuel design result in changes to the bundle flow area, hydraulic diameter, heated perimeter and spacer loss coefficients. GE has performed special tests to determine the loss coefficients for the upper and lower tie plates, spacer and water rod over the expected range of operating conditions. Under normal operating conditions voiding is not expected to occur in the LCWR in the centrally orificed fuel assemblies. GE expects slight voiding to occur (3%) in the LCWR in the high powered peripherally orificed fuel assemblies (radial peaking factor of about 0.7). For most of the peripherally orificed fuel assemblies the peaking factor is expected to be 0.3 to 0.4 with no voiding expected to occur in the LCWR's of these assemblies.

The analysis of the GE8x8NB fuel thermal-mechanical and hydraulic design has been carried out with approved codes and methods and the analysts and results are acceptable.

2.2 GEXL-PLUS Application to GE8x8NB Fuel

Amendment 15 to GESTAR II (Ref. 9) incorporates the new GEXL-PLUS boiling transition thermal limits correlation. The GEXL-PLUS correlation was developed to incorporate the ATLAS test data for new GE fuel designs in the GE thermal limits correlation. Additional ATLAS tests have recently been performed for the GE8x8NB fuel and indicate that the form of the GEXL-PLUS correlation is applicable to the GE8x8NB fuel, however, the numerical values of the coefficients must be modified in order to provide accurate predictions. The data base and verification of the correlation with these new coefficients. GEXL-PLUS/GE8x8NB, is included in Attachment 4 to Reference 2.

The ATLAS tests were performed using twenty-five GE8x8NB prototype fuel bundles. Eighteen of these bundles were used to determine the GEXL-PLUS/GE8x8NB coefficients and the remaining seven were used to test the derived correlation. Except for the axial power shape, the measurement data spans the region of applicability of the correlation. The applicable regions are:

Pressure:	700 to 1400 psia
Mass Flux:	0.10 to 1.5×10^6 lbm/hr-ft^2
Inlet Subcooling:	0 to 100 BTU/lbm
Axial Shape:	Inlet-peaked, cosine, outlet-peaked
R-Factor:	1.10 for limiting bundles

The measurements were made for both an inlet-peaked ($F_z = 1.6$) and cosine power shape but, as in the earlier GEXL and GEXL-PLUS measurements, no top-peaked axial power shape measurements were made. The measurement techniques and procedures used were the same as those used in the development of the GEXL-PLUS data base.

The sixteen GEXL-PLUS/GE8x8NB correlation coefficients were determined using the same numerical procedure as was used in determining the GEXL-PLUS coefficients. Certain coefficients were found to have only a negligible effect on the predictions and were taken to be zero. Terms for

which there was a potential for correlation with existing terms were taken to have the same values as in the GEXL-PLUS correlation (Response 4, Reference 10). The fitting process was tested, in part, by calculating the correlation and demonstrating that the sixteen term function was smooth.

The GEXL-PLUS correlation with the GE8x8NB coefficients was verified by comparing the predictions and measurements of bundle critical power. A comparison of the measurement and correlation predictions is a function of the independent variables indicates that the correlation systematically overpredicts the critical power for low flows, G 0.50×10^6 lbm/hr-ft^2 and/or for high inlet subcoolings, delta-h_{sub} 70 BTU/lbm. In response to question 1 of Reference 10. GE has indicated that the case of high subcoolings is very unlikely during operation. The staff agrees. These regions of non-conservatism are the same as those identified for the GEXL-PLUS correlation.

In order to ensure a conservative determination of CPR it is concluded that:

1. The actual mean Experimental Critical Power Ratio (ECPR) values should be used in minimum CPR (MCPR) design limit calculations in the regions of G 0.50×10^6 lbm/hr-ft^2 and delta-h_{sub} 70 BTU/lbm where the GEXL-PLUS/GE8x8NB correlation appears to be nonconservative (i.e., mean ECPR 1).

2. In determining the ECPR standard deviation, the tolerance interval multiplier used in calculating the ECPR limits should be chosen taking into account the number of ECPR data points used in computing the mean in each nonconservative region. In regions where the mean ECPR is essentially unbiased or conservative (i.e., mean ECPR 1), the limit can be determined using a correction to an assumed mean of l.0 if desired.

In addition to the dependence of ECPR on the region of applicability of the GEXL-PLUS/GE8x8NB correlation, the responses to questions 1 and 2 of Reference 10 indicate that the ECPR standard deviation is also a function of the region of applicability. The ECPR standard deviation is significantly larger than the 2.68% determined by averaging overall regions, in the case of large R-factors (1.08), increased pressures (1200 psia) and inlet-peaked power distributions. In order to account for the region dependence of the GEXL-PLUs/GE8x8NB standard deviation either (1) a bounding value of 3.6% should be used over the entire range of application or (2) the range of application should be partitioned into homogenous groups for the ECPRs (i.e., common populations), and the MCPR should be computed using a mean and standard deviation for each population.

Comparison of the ECPR standard deviation for the inlet-peaked and cosine power shapes (Tables 1–2 of the response to question 1 of Reference 10) suggests that the standard deviation depends on the axial power shape. While no data is available for outlet-peaked bundles, the review has concluded that the use of an ECPR standard deviation that bounds the measured standard deviation for both the inlet-peaked and cosine shapes is acceptable. This conclusion is based on (1) the substantial conservatism in the SLMCPR calculation (0.4 for the C-Lattice and 0.3 for the D-Lattice) resulting from the use of an extremely conservative design basis radial power distribution rather than a conservative (but not extreme) radial power shape and (2) the fact that the limiting MCPR location generally does not occur in an outlet-peaked bundle (Ref. 11).

2.3 Safety Limit MCPR

The determination of the safety limit MCPR (SLMCPR) for the GE8x8NB fuel is described in Attachment 3 to Reference 3. The methodology used to determine the GE8x8NB SLMCPR is the

same as that used in previous GE submittals that have been reviewed and approved by the staff. The differences in the GE8x8NB SLMCPR calculation are:

1. the GEXL-PLUS/GE8x8NB correlation uncertainty,

2. the core radial power distribution,

3. the bundle R-factor distribution, and

4. The R-factor uncertainty.

The GEXL-PLUS/GE8x8NB correlation uncertainty has been determined using all measurement data to be 2.68%. As discussed in the previous section, either a bounding value of 3.6% or a region dependent value should be used in the determination of the SLMCPR. Conservative design basis core radial power distributions for the C and D lattice cores, which maximize the number of bundles at or near MCPR limits, were used in the Monte Carlo safety limits analyses. R-factor pin-wise distributions characteristic of the GE8x8NB design for both the C and D lattices were input as a function of exposure. GE has indicated in response to Question 10 of Reference 10 that the plant and fuel specific uncertainties used for the GE8x8EB fuel are equally applicable to the GE8x8NB fuel. This is reasonable and acceptable.

The SLMCPR was determine using the approved Monte Carlo methodology which combines the GEXL-PLUS/GE8x8NB correlation uncertainty and plant measurement uncertainties to determine the SLMCPR which ensures that 99.9% of the fuel rods are expected to avoid boiling transition. The SLMCPR values determined for the C and D lattice are 1.07 and 1.06, respectively. These are acceptable values.

In order to estimate the conservatism in these limits an additional SLMCPR calculation was performed using a conservative (but not as extreme as the design basis shape) radial power distribution. The SLMCPR determined by this calculation was 1.03 for both the C and D lattices.

2.4 Neutronic Analysis

GE has employed approved methods (Ref. 12) in the lattice and full core neutronic analyses of the GE8x8NB fuel design. The flatter power distribution of this new fuel design is the result of a flatter U235 enrichment distribution and the replacement of the individual water rods by a large central water rod. The presence of the LCWR causes a slight reduction in the active channel flow area and in the magnitude of the negative void reactivity coefficient.

In the calculations of the GE8x8NB assembly GE divides the water in the lattice into two separate regions: (a) the in-channel region which consists of the active flow region surrounding the fuel rods and (b) the ex-channel region which represents the water in the gaps between adjacent fuel assemblies as well as the water inside the LCWR. A radially uniform void distribution is assumed in the in-channel region while full density water is assumed in the in-channel region. Cross section sets are generated with approved lattice codes at 0, 40, and 70% in-channel voids for use in the three-dimensional simulator code.

The lattice methods used for the calculation of the GE8x8NB fuel have been qualified using Monte Carlo simulation (Ref. 7). Lattice k-infinity, void coefficients and rod-by-rod power distributions obtained from GE standard nuclear methods calculations were compared with Monte Carlo results.

The agreement in k-infinity, was with in about 0.5 percent. The void coefficients agreed to within about 8 percent while the pin-by-pin comparisons indicated a difference of about 3 percent.

These results demonstrate satisfactory agreements of the GE lattice methods for application to bundles of the GE8x8NB design. The application of the GE lattice methods (Ref. 12) to the new fuel design is therefore acceptable.

2.4.1 Void Reactivity Coefficient

The void coefficients for the GE8x8NB and P8x8R/BP8x8R designs are very similar at low power operation. At cold zero-void conditions the void coefficients for the two designs are essentially the same since both designs have almost the same water-to-fuel ratios. During normal operation at rated power, however, the negative void coefficient has a smaller magnitude due to the smaller voided region in the NB design due to the presence of the LCWR. GE has calculated the void coefficient as a function of channel voids for a typical BWR/4 equilibrium cycle core consisting entirely of GE8x8NB fuel. Similar full core calculations were also performed for a BWR/4 core loaded entirely with P8x8/BP8x8R assemblies. Comparison of the void coefficients of the two cores indicated that with the exception of the low void region (10% voids), the GE8x8NB fueled core has a slightly less negative coefficient than the core loaded with P8x8R/BP8x8R fuel. The difference in the coefficients increases with increasing voids and is 10 percent for high void concentrations (65% voids). The changes in the void coefficients are generally acceptable. The effect of the smaller void coefficient on the stability performance of the new design is discussed later in this section.

2.4.2 Moderator Temperature Coefficient

A GE calculation of the moderator temperature coefficient (MTC) indicates that the MTC for the GE8x8NB fuel is very similar to that of the approved GE8x8EB fuel (Refs. 7 & 8). Comparisons of the MTC for the two fuel designs in the temperature range from 20° to 160°C indicate that the GE8x8NB fuel is slightly less negative than the GE8x8EB fuel. Calculations of the moderator temperature coefficients for a typical equilibrium cycle BWR/4 core consisting entirely of GE8x8NB fuel have shown that the coefficient is always negative at the beginning-of-cycle (BOC) but that at end-of-cycle (EOC), for temperature below 55°C, the coefficient becomes slightly positive. This behavior is very similar to the behavior of the MTC of the approved GE8x8EB fuel in the same temperature range. The MTC of the GE8x8EB fuel attains a small positive value at a slightly lower temperature.

GE has estimated that in the worst case in which at EOC the moderator temperature undergoes an instantaneous drop to 20°C, the corresponding reactor period would be about 125 sec. This period is longer than the period resulting from standard control rod movements during the normal reactor startup. The power increase corresponding to a 125 seconds period is slow and thus easily controlled. On the other hand, with the exception of startup, it is highly unlikely that the moderator temperature of a BWR would be below 100°C since beyond BOC reactors are, in general, brought down to “cold” conditions not much below 100°C. At this temperature, the MTC is negative. Possible positive MTC thus presents no control problem with this fuel.

2.5 Stability

The General Electric thermal-hydraulic stability licensing criteria set forth in Amendment 8 to the GESTAR licensing topical report (Ref. 13) allow for the application of the generic bounding stability approach to new fuel designs provided that the stability performance of the new design is bounded by that of current fuel designs. A summary of the thermal-hydraulic stability performance of the

GE8x8NB fuel design is presented in Reference 7. Each of the factors affecting stability performance of this fuel design, vis., pressure drop, void coefficient and the fuel rod thermal time constant, have been evaluated. Using full-scale bundle components, two-phase pressure drops were measured for varying bundle power and bundle mass flux for the new GE8x8NB fuel design. These results (Ref. 7) have shown that the total pressure drop for the GE8x8NB design is slightly lower than that for the current design. The effect on the decay ratio of this slight decrease in the two-phase pressure drop improves stability relative to the current designs.

The effect of a decrease in magnitude of the negative void coefficient results in a slight improvement in the decay ratio. The fuel time constant for the GE8x8NB fuel is identical to the time constant for the GE8x8EB fuel. Thus, the overall stability performance of the new fuel is slightly improved relative to that of the current GE8x8EB fuel design.

Based on these considerations it is expected that stability calculations in the operating regions of a reactor loaded with this fuel will generally have a decay ratio less than 1.00 and it will become an approved fuel is indicated in the NRC Generic Letter No. 86-02 on Thermal Hydraulic Stability. If a ratio greater than 0.80 is calculated for a reactor (thus indicating the possibility of in actual ratio greater than 1.00 as indicated in the generic letter), the regions to be avoided in operation and surveillance regions will have to be considered.

2.6 LOCA

In a loss of coolant accident (LOCA) the large central water rod of the GE8x8NB fuel provides an additional unheated surface which serves as a heat sink. In non-jet pump BWRs, early quenching of the LCWR will provide an additional radiation heat sink and will reduce the peak cladding temperature (PCT) significantly. GE estimates a PCT reduction by as much as 150°F, depending on the core spray flow rate. The LOCA impact of the LCWR on jet pump plants is negligible since the cooling in these plants is dominated by the holdup of the coolant in the fuel bundle.

LOCA calculations are carried out with approved methodologies and codes. The LCWR is modeled in the CORECOOL and CHASTE codes by four smaller water rods which preserve the total perimeter of the LCWR. This modeling, however, results in a distortion of the view factors between the LCWR and its surrounding rods, causing a slight underprediction of the radiative heat transfer to the surrounding rods. For PCT exceeding 1800°F, this underprediction leads to a small underprediction of PCT. GE indicates that conservatism in water rod wetting correlations are larger than the small PCT due to the use of approximate view factors (Ref. 8).

System depressurization resulting from a LOCA can cause voiding in the LCWR. In evaluating the effect of LCWR voiding on LOCA radiative heat transfer, the heat sink potential of the LCWR is assumed to include only the heat capacity of the Zircaloy material of the LCWR. This approach is conservative.

Based on the LOCA evaluations the GE8x8NB fuel is found acceptable.

GE has requested (Ref. 17) the approval of the use of the current GEXL correlation as well as GEXL-PLUS/GE8x8NB for LOCA calculations. The GEXL correlation is conservative when applied to the GE8x8NB fuel design and is conservative or has no impact in LOCA calculations. Its use is thus conservative and acceptable.

2.7 Transients

The effect of GE8x8NB fuel on the applicability of standard, approved methodologies used for required reload transient and accident analyses is discussed in Reference 2. The methods used to analyze these events have approved nuclear models to represent necessary fuel characteristics. The qualification of these methods to model GE8x8NB fuel is discussed in Reference 7. The qualification is straightforward and reasonable and the methods are acceptable for analyses involving GE8x8 fuel. This includes ODYN standard adjustment factors which are independent of fuel design. Thus required reload analyses which are reactor-cycle specific are acceptable. This includes over pressure analyses, plant option analyses and fuel loading error. The fuel handling accident is bounded by existing analyses.

Several analyses are generic, however, and are based on statistical analysis of previous event analyses with other fuel types. This includes the Rod withdrawal Error (RWE) for BWR 2-5 for which the generic approach is statistically based. Since the base information does not yet exist for GE8x8NB fuel this analysis must be reactor-cycle specific until the data base for a statistical analysis is developed and approved. The ARTS and BWR-6 generic analyses are insensitive to fuel parameters, however, and are acceptable for GE8x8NB fuel. (The BWR-6 generic analysis is limited to average bundle fuel enrichment not greater than 3.25 percent.)

The control Rod Drop event is plant specific for non-Bank Position Withdrawal Sequence (BPWS) plants and the analysis methodology is acceptable for GE8x8NB fuel. For BPWS plants the generic analysis is statistically based. However, the generic results have such a large margin to required limits (and expected insensitivity to GE8x8NB fuel) that it is acceptable for GE8x8NB fuel without further analysis.

2.8 Appendices C and D

As stated in Amendment 16, Appendices C and D of GESTAR II have been removed. It was proposed in Amendment 16 to publish the contents of these appendices as separate GE reports. These reports have been submitted in connection with Amendment 18 (see References 1 and 15). These submittals (Refs. 3 & 4) are acceptable. In response to alternative suggestions by GE on the presentation details and upkeep of the Appendix D information, the staff has decided that the reports should continue to exist, although they would not receive any formal review process, and that (1) the information in the Reference 4 report should be updated periodically to include new fuel assembly patterns, and (2) axial distribution information for GE8x8EB and GE8x8NB fuel zone descriptions should be presented in individual reactor cycle reload submittals along with the MAPLHGR information, preferably in connection with the LOCA analysis submittal. GE has agreed.

3.0 CONCLUSION

Amendment 18 to Revision 8 of NEDE-24011-P-A and the supporting documentation have been reviewed and found acceptable with the following provisions. In order to ensure a conservative determination of CPR it is required that:

1. The actual mean ECPR values should be used in MCPR design limit calculations in the regions of G 0.50×10^6 lbm/hr-ft^2 and delta-h$_{sub}$ 70 BTU/lbm where the GEXL-PLUS/GE8x8NB correlation appears to be nonconservative (i.e., mean ECPR 1).

2. In determining the ECPR standard deviation, the tolerance interval multiplier used in calculating the MCPR limits should be chosen taking into account the number of ECPR data

points used in computing the mean in each nonconservative region. In regions where the mean ECPR is essentially unbiased or conservative (i.e., mean ECPR 1), the limit can be determined using a correction to an assumed mean of l.0 if desired.

3. In order to account for the region dependence of the GEXL-PLUS/GE8x8NB standard deviation either (1) a bounding value of 3.6% should be used over the entire range of application or (2) the range of application should be partitioned into homogeneous groups for the ECPRs (i.e., common populations) and the MCPR should be computed using a mean and standard deviation for each population.

With these restrictions, the use of GE8x8NB fuel is described in Amendment l8 and supplements is acceptable.

The editorial changes presented in the amendment and the transfer of Appendices C and D to separate reports are also acceptable.

4.0 REFERENCES

1. Letter from J.S. Charnley (GE) to G.C. Lainas (NRC), “Proposed Amendment 18 to GE Licensing Topical Report NEDE-24011-P-A,” (GE 8x8NB Fuel), October 31, 1986.

2. Letter from J.S. Charnley (GE) to M.W. Hodges (NRC), “Changes to the U.S. Supplement of GESTAR (Pertaining to Amendment 18 of GESTAR),” August 7, 1987.

3. “General Electric Fuel Bundle Designs Evaluated with Texico/Clam Analyses Bases,” NEDE-31151P.

4. “General Electric Fuel Bundle Designs Evaluated with GESTAR-MECHANICAL Analyses Bases,” NEDE-31151P.

5. Letter from C.O. Thomas (NRC) to J.S. Charnley (GE), Acceptance for Referencing of LTR NEDE-24011-P-A-6, Amendment 10, GE Standard Application for Reactor Fuel,” May 28, 1985.

6. Letter from H.N. Berkow (NRC) to J.S. Charnley (GE), “Acceptance for Approval of Fuel Designs Described in LTR NEDE-24011-P-A-6, Amendment 10, for Extended Burnup Operation,” December 3, 1985.

7. Letter from J.S. Charnley (GE) to M.W. Hodges (NRC), “Additional Information Pertaining to NRC Review of GESTAR Amendment 18,” August 4, l987.

8. Letter from J.S. Charnley (GE) to M.W. Hodges (NRC), “Requested Information for Amendment 18,” December l0, 1987.

9. Letter, J.S. Charnley (GE) to C.O. Thomas (NRC), “Amendment 15 to General Electric Licensing Topical Report NEDE-24011-P-A,” January 23, 1986.

10. Letter. J.S. Charnley (GE) to M.W. Hodges (NRC), “Response to Questions on GESTAR-II Amendment 18,” January 13, 1988.

11. Letter, J.S. Charnley (GE) to M.W Hodges (NRC), "Additional Information on GESTAR-II Amendment 18," February 12, 1988.

12. Letter, C.O. Thomas (NRC) to J.S. Charnley (GE), "Acceptance for Referencing of Licensing Topical Report NEDE-30130, 'Steady-State Nuclear Methods'," December 22, 1983.

13. Safety Evaluation of the General Electric Topical Report NEDE-24011 Amendment 8, March 1985, contained in Reference 14, U.S. Supplement, Appendix C.

14. GE Topical Report NEDE-24011-P, "General Electric Standard Application for Reactor Fuel" (GESTAR II).

15. Letter (and enclosure) from J. Charnley (GE) to G. Lainas (NRC), dated January 29, 1987, "Additional Information Pertaining to Proposed Amendment 18 to NEDE-24011-P-A."

16. Letter (and attachment) from J. Charnley (GE) to M.W. Hodges (NRC), dated October 14, 1987, Pertaining to GESTAR Amendment 18 (GE8x8NB Fuel).

17. Letter from J. Charnley (GE) to M.W. Hodges (NRC), dated December 11, 1987, "Continued Use of GEXL Correlation for LOCP Analysis of GE8x8NB.

NRC
Safety Evaluation Report
Approving Amendment 19
to
NEDE-24011-P

UNITED STATES
NUCLEAR REGULATORY COMMISSION
Washington, D. C. 20555
November 17, 1987

MFN 112-87

Ms. J. S. Charnley, Manager
Fuel Licensing
Nuclear Safety and Licensing
General Electric Company
175 Curtner Avenue
San Jose, California 95125

Dear Ms. Charnley:

SUBJECT: ACCEPTANCE FOR REFERENCING OF AMENDMENT 19 TO GENERAL ELECTRIC LICENSING TOPICAL REPORT NEDE-24011-P-A (GESTAR-II), "GENERAL ELECTRIC STANDARD APPLICATION FOR REACTOR FUEL," APRIL 7, 1987

The staff has completed its review of Amendment 19 to the subject topical report submitted by the General Electric Company by letter dated April 7, 1987.

We find Amendment 19 to be acceptable for referencing in license applications to the extent specified and under the limitations delineated in the report and the associated NRC evaluation, which is enclosed. The evaluation defines the basis for acceptance of Amendment 19.

We do not intend to repeat our review of the matters described in Amendment 19 and found acceptable when the report appears as a reference in license applications, except to assure that the material presented is applicable to the specific plant involved. Our acceptance applies only to the matters described in Amendment 19.

Should our criteria or regulations change such that our conclusions as to the acceptability of Amendment 19 are invalidated GE and/or the applicants referencing the amendment will be expected to revise and resubmit their respective documentation, or submit justification for the continued effective applicability of the amendment without revision of their respective documentation.

Sincerely,

Ashok C. Thadani, Assistant Director for Systems
Division of Engineering & Systems Technology
Office of Nuclear Reactor Regulation

Enclosure:
Amendment 19 to NEDE-24011-P-A Evaluation

ENCLOSURE
SAFETY EVALUATION OF AMENDMENT 19 TO GENERAL ELECTRIC LICENSING TOPICAL REPORT NEDE-24011-P-A (GESTAR-II), "GENERAL ELECTRIC STANDARD APPLICATION FOR REACTOR FUEL"

1.0 INTRODUCTION

In a letter dated August 15, 1985 (Ref. 1), the General Electric Company (GE) proposed changes to the Technical Specifications of BWRs with GE fuel. These changes affect power distribution limits and are provided, as an example using BWR/4 Standard Technical Specifications, in Attachment 2 of the letter. The proposed changes would remove redundant limits from the Technical Specifications and, thereby, simplify the Technical Specifications. These redundant limits include the limits on the linear heat generation rate (LHGR) and associated total peaking factor (TPF). Upon NRC concurrence that the proposed Technical Specification changes are acceptable, individual utilities could incorporate similar changes to their plant's Technical Specifications with both minimum review and assurance of acceptance.

In its review of the GE proposal the NRC staff had a number of telephone discussions with GE technical and licensing staff members. The staff expressed a number of concerns with regard to the GE proposal. Subsequently, GE resubmitted its proposal (Ref. 2) incorporating changes to resolve the staff's concerns. In response to other staff concerns, additional information was submitted by GE with Reference 3. On November 19, 1986, the staff transmitted to GE a request for additional information (Ref. 4). GE provided responses to this request for additional information with References 5 and 6.

On April 7, 1987, General Electric resubmitted (Ref. 10) all the relevant documentation (Refs. 2, 3, 5, 6, 8 and 9) on these proposed changes to the Technical Specifications on power distribution limits as part of proposed Amendment 19 to GE Licensing Topical Report NEDE-24011-P-A (GESTAR-II). This evaluation constitutes, therefore an evaluation of proposed Amendment to GESTAR-II.

The staff's evaluation of the proposed Technical Specification changes follows.

2.0 EVALUATION

A. MAPLHGR and LHGR

Two of the core thermal performance parameters for a BWR are the maximum linear heat generation rate (MLHGR) and the maximum average planar linear heat generation rate (MAPLHGR). Both of these quantities are expressed in terms of kW/ft. The MLHGR is determined from the mechanical design analysis limits for a particular fuel bundle type. GE obtains MAPLHGR by averaging the LHGR over each fuel rod in a given plane of a particular fuel bundle and selecting a limiting value as a function of fuel burnup. The following two conditions must be met (see Table 3-3 of Ref. 7) when selecting the MAPLHGR limit:

1. The peak cladding temperatures during the design basis loss-of-coolant accident (LOCA) analysis must not exceed 2200°F, and

2. The LHGR must not exceed the MLHGR

in the plane of interest.

For some GE fuel bundle designs MAPLHGR depends only on bundle type and burnup. Other GE fuel bundles (e.g., P8x8R/BP8x8R and GE8x8E/GE8x8EB), however, have MAPLHGRs that vary axially depending upon the specific combination of enriched uranium and gadolinia that comprises a fuel bundle cross section at a particular axial node. Each particular combination of enriched uranium and gadolinia, for these fuel bundle types, is called a lattice type by GE. Thus, these particular fuel bundle types have MAPLHGRs that vary by lattice type (axially) as well as with fuel burnup.

The more limiting of either LOCA or LHGR kW/ft determines MAPLHGR at a given fuel burnup for a particular fuel bundle type and, if required, for a given lattice type. Consequently, the maximum LHGR allowed during operation for a given fuel bundle/lattice type (i.e., MAPLHGR times the local peaking factor [LPF]) will, in practice, not exceed the MLHGR mechanical fuel design limit for any fuel bundle burnup at a high probability and confidence level. This is based on the mechanical fuel design MAPLHGR limit being derived from the LHGR limit using an LPF for an uncontrolled fuel bundle at a nominal (proprietary) void condition. An analysis provided by GE (Ref. 3) demonstrates that, for the particular BWR considered, the MAPLHGR calculated using LPFs for the nominal void condition may exceed the MAPLHGR calculated using LPFs at the true void condition by no more than a small amount (proprietary) for 95% of the fuel lattices with 95% confidence. However, this potential non-conservatism in the calculated LHGRs is more than offset by a conservative allowance (proprietary) used in the fuel rod mechanical design analysis. A review of Reference 3 suggests that the results obtained should be typical of results that would be obtained for other large, modern BWRs. Therefore, when considering the neutronic similarities, we conclude that the LHGR fuel mechanical design limit will not be exceeded with a high degree of confidence. In addition, in the event that the MAPLHGR limit times the associated LPF exceeds the LHGR limit for any node, process computer monitoring will alert the operator to the need for corrective action.

General Electric will provide documentation for MLHGRs, MAPLHGRs, and LPFs as follows. The MLHGR limit will be transmitted to the NRC by letter which will be incorporated as a reference in Reference 7 by a future amendment (Ref. 5, response to Question 1). The process computer software supplied by GE will continue to provide licensees with the capability to monitor MLHGR (Ref. 5, response to Question 2).

In a revised response (Refs. 6 and 8) to Question 3 of Reference 4, GE states the following:

1. For any bundle types included in a reload which are not included in GESTAR-II (Ref. 7), or one of its approved amendments, the plant owner will submit a bundle description providing the same type of information found in Reference 7. This bundle description will be proprietary to GE.

2. Each non-proprietary reload submittal will include a table of the most limiting and least limiting MAPLHGR for each multiple lattice bundle type. This document will state that applicable ECCS performance criteria of 10CFR50.46 are met.

3. For each multiple lattice fuel bundle type, the Technical Specifications will include a plot of the limiting value of APLHGR for the most limiting lattice (excluding natural uranium) as a function of planar exposure. The APLHGR of any lattice (excluding natural uranium) in this bundle type shall not exceed this most limiting lattice plot when conformance to the operating limit APLHGR is performed by hand calculation.

4. The licensee, in support of a reload license application, will submit to the NRC, on a proprietary basis, GE proprietary information which will also be available to the operator in the plant control room, the following information for each bundle type:

   a. The axial location of each lattice in the bundle.

   b. The composite MAPLHGR, considering both thermal-mechanical and ECCS requirements, as a function of average exposure for each lattice in the bundle.

GE states (Ref. 6, response to Question 3 of Ref. 4) that LPFs as a function of exposure will be placed in NEDE-24011-P A (Ref. 7) in a future amendment. However, GE has proposed placing fuel-bundle specific information in a new and separate document designated NEDE-31152P. Upon NRC approval, LPFs will be placed in this document (NEDE-31152P).

Based on the above considerations, we conclude, therefore, that BWR Technical Specification limits, based on a MAPLHGR and MLHGR as defined in Table 3-3 of Reference 7 and on power peaking factors as defined in Table 3-4 Reference 7, may be revised to remove the redundant MLHGR limit from the Technical Specifications.

The following related Technical Specification changes, as indicated in Attachment 2 of Reference 2 and Reference 9, are, therefore, acceptable:

1. Definition: Average Planar Linear Heat Generation Rate (APLHGR)

   Minor wording changes will be made. The revised definition is:

   The Average Planar Linear Heat Generation Rate (APLHGR) shall be applicable to a specific planar height and is equal to the sum of the heat generation rate per unit length of fuel rod for all fuel rods in the specified bundle at the specified height, divided by the number of fuel rods in the fuel bundle at that height.

2. Definition: Linear Heat Generation Rate (LHGR)

   This definition will be deleted.

3. Definition: Total Peaking Factor (TPF)

   This definition will be deleted since it is a function of LHGRs and will no longer be needed when LHGRs are removed from the Technical Specifications.

4. Definition: Maximum Total Peaking Factor (MTPF)

   This definition will be deleted since it is the maximum TPF for a given type of fuel and since TPFs will be deleted from the Technical Specifications.

5. Definition: Limiting Control Rod Pattern

   This definition will be changed to delete reference to the LHGR and to define a Limiting Control Rod Pattern in terms of MCPR and MAPRAT. For a given fuel bundle the quantity MAPRAT is the largest value of the ratio of the APLHGR (at a specific height in the bundle) divided by the exposure dependent MAPLHGR limit (for that specific height). The revised definition is:

A Limiting Control Rod Pattern shall be a pattern which results in the core being at a limiting value for MAPRAT or MCPR.

6. Limiting Condition For Operation/Surveillance Requirements/Bases: LINEAR HEAT GENERATION RATE

This Technical Specification along with its associated Surveillance Requirements and Bases will be deleted from the Technical Specifications.

7. Limiting Condition For Operation/Surveillance Requirements/Bases: AVERAGE PLANAR LINEAR HEAT GENERATION RATE

For each multiple lattice fuel bundle type, the Technical Specifications will include a plot of the limiting value of APLHGR for the most limiting lattice (excluding natural uranium) as a function of planar exposure. The APLHGR of any lattice (excluding natural uranium) in this bundle type shall not exceed this most limiting lattice plot when conformance to the operating limit APLHGR is performed by hand calculation.

LIMITING CONDITION FOR OPERATION

3.2.1 During power operation, the APLHGR, i.e., the LCO, for each type of fuel as a function of axial location and average planar exposure shall not exceed limits based on applicable APLHGR limit values that have been approved for the respective fuel and lattice type determined by the approved methodology described in GESTAR-II. (This approval is based on and limited to the GESTAR-II methodology.) When hand calculations are required, the APLHGR for each type of fuel as a function of average planar exposure shall not exceed the limiting value for the most limiting lattice (excluding natural uranium) shown in the applicable figures for X, Y, and Z fuel types during two recirculation loop operation.

(X, Y and Z can be, for example, BP/P8x8R, GE8x8EB, and LTA 310 fuel types. The Limiting Condition for Operation continues with requirements for single recirculation loop operation.)

APPLICABILITY:

Operational Condition 1, when Thermal Power is greater than or equal to (25%) of Rated Thermal Power.

ACTION:

If at any time during operation it is determined by normal surveillance that the limiting value of APLHGR is being exceeded, action shall be initiated within * _________to restore APLHGR to within prescribed limits. If the APLHGR is not returned to within prescribed limits within *_________hours, reactor power shall be decreased at a rate which would bring the reactor to the cold shutdown condition within 36 hours unless APLHGR is returned to within limits during this period. Surveillance and corresponding action shall continue until reactor operation is within the prescribed limits.

* The times specified in a licensee's current action statement are to be used here.

SURVEILLANCE REQUIREMENTS

4.2.1 All APLHGRs shall be verified to be equal to or less than the limits of Specification 3.2.1.

a. At least once per 24 hours,

b. Within 12 hours after completion of a Thermal Power increase of at least 15% of Rated Thermal Power, and

c. Initially and at least once per 12 hours when the reactor is operating with a Limiting Control Rod Pattern for MAPRAT.

BASES

3/4.2.1 This specification assures that the peak cladding temperature (PCT) following the postulated design basis Loss-of-Coolant Accident (LOCA) will not exceed the limits specified in 10CFR50.46 and that the fuel design analysis limits specified in NEDE-24011-P-A (Reference 5) will not be exceeded. Mechanical Design Analysis: NRC approved methods (specified in Reference 5) are used to demonstrate that all fuel rods in a lattice, operating at the bounding power history, meet the fuel design limits specified in Reference 5. This bounding power history is used as basis for the fuel design analysis MAPLHGR limit.

LOCA Analysis: A LOCA analysis is performed in accordance with 10CFR50 Appendix K to demonstrate that the permissible planar power (MAPLHGR) limits comply with the ECCS limits specified in 10CFR50.46. The analysis is performed for the most limiting break size, break location, and single failure combination for the plant.

The limiting values for APLHGR are given in Figures __________ through __________. Approved limiting values of APLHGR as a function of fuel type are given in __________ for Reload and fuel. Approved limiting values of APLHGR as a function of fuel and lattice types are given in for Reload fuel.

(The Basis statement continues with information for single recirculation loop operation.)

8. Bases Section 3/4.2 on Power Distribution Limits

A reference needs to be added. This reference is:

5. General Electric Standard Application for Reactor Fuel, NEDE-24011-P-A.

B. APRM Flow-Biased Scram and Rod Block Setpoint Section

Deletion of the Limiting Condition for Operation (LCO) on LHGR and associated references to the TPF requires adjustment of the setdown factor T in the APRM flow-biased rod block and trip function setpoint equations. This setdown factor T is required to prevent thermal-hydraulic limits from being exceeded and the assumptions used in plant transient analyses from being violated at any operating power level when the power density (peaking factor) becomes too large in any portion of the reactor core. GE proposed to define T in terms of APLHGR. We have evaluated the proposed setdown factor with the current setdown factor. Our evaluation proceeds as follows:

Setdown Factor T (Current Definition)

FLPD — fraction of limiting power density

$$= \frac{\text{LHGR (at a given location)}}{\text{LHGR (limit for that bundle type)}}$$

CMFLPD — highest value of FLPD which exists in the core

= max. FLPD, (for the core)

FRTP — fraction of rated thermal power

$$= \frac{\text{measured thermal power}}{\text{rated thermal power}}$$

TPF — total peaking factor

$$= \frac{\text{LHGR (at a given location)}}{\text{LHGR (core average for the bundle type)}}$$

MTPF — maximum TPF which exists in the core for a given class of fuel at a given operating state

$$T = \frac{\text{FRTP}}{\text{CMPFLPD}} \text{ or equivalently}$$

$$T = \frac{\text{design TPF (= 2.43 for some 8x8 fuel)}}{\text{MTPF}}$$

Note that, if the design value of TPF varies for different types of fuel, the value of T (based on TPF) would be the minimum value of the function evaluated over all types of fuel.

The current Technical Specifications provided in Attachment 2 of Reference 2 gave both of the above forms for T in Specification 3.2.2. The first part states that T is the lowest value of the ratio of FRTP divided by CMFLPD. However, this is redundant since, by definition, FRTP is a single value for the reactor at a given operating state and CMFLPD is the maximum value determined for the core. The second part gives the second of the above two forms of T evaluated over all types of fuel bundles and here, presumably, the lowest value obtained is used. In any event, the proper use of either of the two definitions of T would yield the same value. In the APRM flow-biased rod block and trip setpoint equations, T is applied only if it is less than (or equal to) unity.

Setdown Factor T (Proposed Definition)

MAPRAT - maximum ratio of the APLHGR at a specific height in the fuel bundle divided by the exposure dependent APLHGR limit (MAPLHGR) for that specific height.

CMAPRAT - max. MAPRAT, (for the core)

$$T = \frac{\text{FRTP}}{\text{CMAPRAT}}$$

By definition T will be the lowest possible value of the above ratio in the core. The setdown factor T is applied in the setpoint equation only if it is less than unity.

Our evaluation is based on our numerical results obtained over a range of parameters typical of an 8x8 GE fuel bundle. The results indicate that both values of the setdown factor T determined by either one of the two current definitions given, as expected, the same value for T. When APLHGR for the fuel mechanical design analysis is less than APLHGP for LOCA requirements, the proposed definition of T gives the same value as the two previous definitions. When APLHGR for LOCA requirements is less than APLHGR for the fuel mechanical design analysis, then the proposed definition gives a somewhat lower (conservative) value for T than the two current definitions. We conclude, therefore, that the proposed definition of T is acceptable in the proposed Technical Specification change because it results in either the same or more conservative values for the setdown factor than when it is obtained with the previous definitions of T.

Based on the preceding discussion, the following related Technical Specification revisions on the determination of the setdown factor T, as indicated in Attachment 2 of Reference 2, are acceptable:

1. Definition: Fraction of Limiting Power Density

   This definition will be deleted as it will no longer be required to compute the setdown factor T.

2. Definition: Core Maximum Fraction of Limiting Power Density

   This definition will be deleted as it will no longer be required to compute the setdown factor T.

3. Definition: Maximum Average Planar Linear Heat Generation Rate Ratio

   This definition will be added to define a quantity needed to compute the proposed setdown factor T. The definition is:

   The Maximum APLHGR Ratio (MAPRAT) for a fuel bundle shall be the largest value in the bundle of the ratio of the APLHGR (at a specific height in the fuel bundle) divided by the exposure dependent MAPLHGR limit (for that specific height).

4. Definition: Core Maximum Average Planar Linear Heat Generation Ratio

   This definition will be added to define a quantity needed to compute the proposed setdown factor T. The definition is :

   The Core Maximum APLHGR Ratio (CMAPRAT) shall be the highest value of the MAPRAT that exists in the core.

5. Limiting Safety System Settings: Reactor Protection System Instrumentation Setpoints

   A modification is required to Bases 2.2.1. The last paragraph of item 3 concerning the Average Power Range Monitor needs to be changed. This paragraph should read as follows:

   The APRM setpoints were selected to provide adequate margin for the Safety Limits and yet allow operating margin that reduces the possibility of unnecessary shutdown. The flow

referenced trip setpoint must be adjusted by the specified formula in Specification 3.2.2 in order to maintain these margins when CMAPRAT is greater than FRTP.

6. Limiting Condition For Operation/Surveillance Requirements/Bases: APRM Setpoints

The proposed evaluation for the setdown factor T requires a number of changes to Specification 3/4.2.2 as follows:

The definition of T in the Limiting Condition For Operation will be changed to the following:

> T = ratio of FRTP divided by CMAPRAT. T is applied only if less than 1.0.

The footnote to the action statement will be correspondingly changed and will now read:

> With CMAPRAT greater than the FRTP during power ascension up to 90% of Rated Thermal Power, rather than adjusting the APRM setpoints, the APRM gain may be adjusted such that the APRM readings are greater than or equal to 100% times CMAPRAT, provided that the adjusted APRM reading does not exceed 100% of Rated Thermal Power and a notice of adjustment is posted on the reactor control panel.

The Surveillance Requirement will be changed to replace reference to CMAFLPD, MTPF and TPF with reference to CMAPRAT. It will now read as follows:

> 4.2.2 The FRTP and CMAPRAT shall be determined, the value of T calculated, and the most recent actual APRM flow biased simulated thermal power-upscale scram and flow biased neutron flux-upscale control rod block trip setpoints verified to be within the above limits or adjusted, as required:
>
> a. At least once per 24 hours,
>
> b. Within 12 hours after completion of a Thermal Power increase of at least 15% of Rated Thermal Power and,
>
> c. Initially and at least once per 12 hours when the reactor is operating with CMAPRAT greater than or equal to FRTP.

The bases for this Technical Specification will be changed to incorporate the LHGR and APRM setdown factor T changes discussed in this evaluation. The changed basis will read:

> 3/4.2.2 APRM Setpoints
>
> The flow biased simulated thermal power-upscale scram setting and flow biased neutron flux-upscale control rod block functions of the APRM instruments are adjusted to ensure that fuel design and safety limits are not exceeded. The scram settings and rod block settings are adjusted in accordance with the formula in this Specification when the combination of Thermal Power and CMAPRAT indicates a highly peaked power distribution.

3.0 CONCLUSIONS

Based on the results of our evaluation, we conclude that the Technical Specification changes proposed by GE, which delete the LHGR (Linear Heat Generation Rate) Technical Specification, are acceptable. This conclusion is based on all applicable APLGHRs (average planar LHGRs) including (as a function of fuel burnup and, where necessary, fuel bundle lattice type) the more limiting of either APLHGR based on ECCS analysis requirements or APLHGR based on fuel mechanical design analysis requirements. We also conclude that the proposed Technical Specifications will, in practice, result in the same operating power distribution limits and safety margins as the current Technical Specifications. In addition, the proposed Technical Specifications will reduce Technical Specification complexity by removing Specifications which are in effect redundant, and will preclude the need for inclusion of numerous lattice specific MAPLHGR curves. We conclude, therefore, that proposed Amendment 19 to NEDE-24011-P-A (GESTAR-II) is acceptable.

This evaluation may be referenced by licensees with GE fuel in support of submittals for similar changes to their Technical Specifications as discussed in this evaluation and Attachment 2 of Reference 2 and Reference 9. Licensees will be required to maintain procedural controls on LHGR limits using process computer monitoring and to provide documentation on MAPLHGRs and LPFs, as discussed in this evaluation, with such Technical Specification changes.

4.0 REFERENCES

1. Letter from J.S. Charnley (GE) to R. Lobel (NRC), “Proposed Changes to Technical Specifications for Power Distribution Limits,” General Electric Company Special Report MFN 106-85, August 15, 1985.
2. Letter from J.S. Charnley (GE) to C.O. Thomas (NRC), “Revision 1 of GE Special Report MFN 106-85,” letter number MFN 004-86, January 16, 1986.
3. Letter from J.S. Charnley (GE) to M.W. Hodges (NRC), “Supplementary Information Regarding GE Special Report MFN 106-85,” letter number MFN 085-086, September 9, 1986.
4. Letter from M.W. Hodges (NRC) to J.S. Charnley (GE), dated November 19, 1986.
5. Letter from J.S. Charnley (GE) to M.W. Hodges (NRC), “Response to Request for Additional Information on GE Special Report MFN 106-85,” letter number MFN 004-087, January 14, 1987.
6. Letter from J.S. Charnley (GE) to M.W. Hodges (NRC), “Revised Response to Question 3 of Request for Additional Information on GE Special Report MFN 106-85,” letter number MFN 025-087, March 12, 1987.
7. “General Electric Standard Application For Reactor Fuel,” NEDE-24011-P-A (as revised through the Amendment 10 approval letter of May 28, 1985 from C.O. Thomas (NRC) to J.S. Charnley (GE)).
8. Letter from J.S. Charnley (GE) to M.W. Hodges (NRC), “Recommended MAPLHGR Technical Specifications for Multiple Lattice Fuel Designs,” letter number MFN 023-087, March 9, 1987.
9. Letter from J.S, Charnley (GE) to M.W. Hodges (NRC), “Recommended MAPLHGR Technical Specifications for Multiple Lattice Fuel Designs,” letter number MFN 021-087, March 4, 1987.

10. Letter from J.S. Charnley (GE) to M.W. Hodges (NRC), "Proposed Amendment 19 to GE LTR NEDE-24011-P-A (Power Distribution Limits)," letter number MFN 031-087, April 7, 1987.

NRC
Safety Evaluation Report
Approving Amendment 21
to NEDE-24011-P

UNITED STATES
NUCLEAR REGULATORY COMMISSION
Washington, D. C. 20555
March 17, 1989

Ms. J. S. Charnley, Manager
Fuel Licensing
General Electric Company
175 Curtner Avenue
San Jose, California 95125

Dear Ms. Charnley:

SUBJECT: ACCEPTANCE FOR REFERENCING OF AMENDMENT 21 TO GENERAL ELECTRIC LICENSING TOPICAL REPORT NEDE-24011-P-A, "GENERAL ELECTRIC STANDARD APPLICATION FOR REACTOR FUEL"

The staff has completed its review of the subject amendment submitted by the General Electric Company by letter dated July 25, 1988. This Amendment 21 primarily presents information for the incorporation of the three GE8x8NB fuel bundle design options into NEDE-24011-P-A.

We find the Amendment 21 to be acceptable for referencing in license applications. The safety evaluation defines the basis for acceptance of the amendment.

We do not intend to repeat our review of the matters described in Amendment 21 and found acceptable when the report appears as a reference in license applications, except to assure that the material presented is applicable to the specific plant involved. Our acceptance applies only to the matters described in the Amendment 21.

In accordance with procedures established in NUREG-0390, it is requested that GE publish accepted versions of this amendment, proprietary and non-proprietary, within three months of receipt of this letter. The accepted version shall incorporate this letter and the enclosed evaluation between the title page and the abstract. The accepted versions shall include an -A (designating accepted) following the report identification symbol.

Should our criteria or regulations change such that our conclusions as to the acceptability of the report are invalidated, GE 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,

Ashok C. Thadani, Assistant Director for Systems
Division of Engineering & Systems Technology
Office of Nuclear Reactor Regulation

Enclosure:
Amendment 21 to NEDE-24011-P-A Evaluation

ENCLOSURE

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION RELATING TO AMENDMENT 21 TO GENERAL ELECTRIC TOPICAL REPORT NEDE-24011-P-A "GENERAL ELECTRIC STANDARD APPLICATION FOR REACTOR FUEL"

1.0 INTRODUCTION

In a letter dated July 25, 1988, General Electric submitted a proposed Amendment 21 to NEDE-24011-P-A, "General Electric Standard Application for Reactor Fuel" (GESTAR II). The primary purpose of this amendment is to incorporate the GE8x8NB-1, GE8x8NB-2, and GE8x8NB-3 fuel bundle designs into the GESTAR II document.

Amendment 21 presents proposed changes to the GESTAR II base document and to the U.S. Supplement to describe the GE8x8NB-1, GE8x8NB-2, and GE8x8NB-3 fuel designs. In addition, several typographical errors are corrected.

The GE8x8NB-1, GE8x8NB-2, and GE8x8NB-3 designs are evolved from the NRC approved GE8x8NB design using two unique features: interactive channel design and lower tie plate with a 40-mil offset toward the control blade. The GE8x8NB-1 design is the GE8x8NB fuel design plus the interactive channel, the GE8x8NB-2 is the GE8x8NB fuel design plus the offset lower tie plate, and the GE8x8NB-3 is the GE8x8NB fuel design plus the combination of interactive channel and offset lower tie plate. The interactive channel design improves CPR performance by using flow trippers inside the channel wall to increase turbulent mixing. The offset lower tie plate reduces the magnitude of the difference between the wide and narrow water gaps around the D-lattice channels of BWR/2, 3, and 4 designs. This modification makes the D-lattice channel design of BWR/2, 3, and 4 plants more agreeable with the C-lattice channel design of BWR/5 plants in analyses. All three new fuel design options have been analyzed with approved methods.

2.0 EVALUATION

2.1 Fuel Mechanical Design

The interactive channel used in GE8x8NB-1 and GE8x8NB-3 designs has stiffer corners, but thinner side walls than a standard channel with uniform side walls. The interactive channel for BWR/2, 3, 4, and 5 plants may also include axial grooves on the outside channel walls, while the interactive channel for BWR/6 plants has no grooves. GE analyzed the channel design in terms of its functional capability, stress loading, dimensional clearance, and seismic/LOCA dynamic capability. GE concluded that the interactive channel has made no major deviation from the previously approved results of the standard channel with uniform side walls. We therefore conclude that the interactive channel mechanical design is acceptable for use in the GE8x8NB-1 and GE8x8NB-3 designs.

GE stated that the 40-mil offset in the lower tie plate of the GE8x8NB-2 and GE8x8NB-3 designs had no major impact on the mechanical capability compared to the approved GE8x8NB design. We thus conclude that the offset lower tie plate is acceptable for use in the GE8x8NB-2 and GE8x8NB-3 designs.

2.2 Fuel Nuclear Design

GE stated that there were no major changes in neutronic calculations from the previously approved methodology described in GESTAR II except to account for the geometry changes of the offset lower tie plate in GE8x8NB-2 and GE8x8NB-3 designs. GE pointed out that there will be a mixed core situation of the standard lower tie plates and offset lower tie plates for a few cycles during transition from standard to offset lower tie plates. Detailed calculations to account for this mixed core situation will be reported in reload submittals. We consider this acceptable.

2.3 Safety Limit MCPR

For GE8x8NB-1 design with the interactive channel option, GE applies the approved GE8x8NB safety limit MCPR for safety calculations. GE stated that this is a conservative approach because the interactive channel with flow trippers tends to improve the CPR performance by increasing turbulent mixing. Thus we conclude that the GE8x8NB safety limit MCPR is acceptable for GE8x8NB-1 fuel design.

As mentioned earlier, the offset lower tie plate option in GE8x8NB-2 and GE8x8NB-3 designs makes the D-lattice channels more agreeable with the C-lattice channels in analyses. GE applies the approved GE8x8NB C-lattice safety limit MCPR for safety calculations of GE8x8NB-2 and GE8x8NB-3. The C-lattice design has higher safety MCPR limit than the D-lattice design. Since the D-lattice design with offset lower tie plate in GE8x8NB-2 and GE8x8NB-3 designs has a geometry between C-lattice and D-lattice designs, GE stated that the use of C-lattice MCPR for the offset lower tie plate option is a conservative approach. We therefore conclude that the GE8x8NB C-lattice safety limit MCPR is acceptable for GE8x8NB-2 and GE8x8NB-3 fuel designs.

2.4 GEXL-PLUS Applications

For critical power correlation calculations, GE proposed to use the approved GEXL-PLUS correlation for GE8x8NB-1 and GE8x8NB-3 designs with interactive channel. This is a conservative approach because the interactive channel design tends to improve thermal-hydraulic performance by increasing turbulent mixing while the GEXL-PLUS correlation can only deal with standard channel. GE will improve the GEXL-PLUS correlation to include the interactive channel feature when more performance data are available. We therefore conclude that the use of GEXL-PLUS correlation for GE8x8NB-1 and GE8x8NB-3 designs is acceptable.

GE also applied the approved GEXL-PLUS correlation for GE8x8NB-2 design with offset lower tie plate and concluded that the offset lower tie plate had little effect on critical power calculations. We thus conclude that the use of GEXL-PLUS correlation for GE8x8NB-2 design is acceptable.

2.5 Stability

The three GE8x8NB options do not significantly change the stability performance of the GE8x8NB design. A summary of the thermal-hydraulic stability performance of the GE8x8NB fuel design is

presented in the approved Amendment 18 to GESTAR II. Each of the factors affecting stability performance of this fuel design, viz., pressure drop, void coefficient and the fuel rod thermal time constant, have been evaluated. Using full-scale bundle components, two-phase pressure drops were measured for varying bundle power and bundle mass flux for the new GE8x8NB fuel design. These results have shown that the total pressure drop for the GE8x8NB design is slightly lower than that for the current GE8x8EB design. The effect on the decay ratio of this slight decrease in the two-phase pressure drop improves stability relative to the current designs. The effect of a decrease in magnitude of the negative void coefficient results in a slight improvement in the decay ratio. The fuel time constant for the GE8x8NB fuel is identical to the time constant for the GE8x8EB fuel. Thus, the overall stability performance of the GE8x8NB fuel is slightly improved relative to that of the current GE8x8EB fuel design. This conclusion can also apply to the three GE8x8NB design options.

Therefore the previously approved bounding stability approach for GE8x8NB fuel is applicable for all three GE8x8NB options. We consider this approach acceptable.

2.6 Transient

GE stated that the plant-specific or cycle-specific analyses will be performed for plant transients such as vessel overpressurization, control rod drop accident, LOCA, and fuel loading error. We consider that this approach is acceptable for all three GE8x8NB options.

3.0 CONCLUSION

We have reviewed the three design options of GE8x8NB fuel and conclude that these three options are acceptable for licensing applications and Amendment 21 is approved to incorporate into the GESTAR II document.

NRC
Safety Evaluation Report
Approving Amendment 22
to NEDE-24011-P

UNITED STATES
NUCLEAR REGULATORY COMMISSION
Washington, D. C. 20555
July 23, 1990

MFN 100-90

Ms. J. S. Charnley, Manager
Fuel Licensing
General Electric Company
175 Curtner Avenue
San Jose, California 95125

Dear Ms. Charnley:

SUBJECT: ACCEPTANCE FOR REFERENCING OF AMENDMENT 22 TO GENERAL ELECTRIC LICENSING TOPICAL REPORT NEDE-24011-P-A "GENERAL ELECTRIC STANDARD APPLICATION FOR REACTOR FUEL" (TAC NO. 71444)

The staff has completed its review of the subject amendment submitted by the General Electric Company by letter dated July 26, 1989. This Amendment 22 presents a set of licensing acceptance criteria applicable to GE fuel designs. Future fuel designs meeting these criteria and methods will not require prior NRC approval.

We find the Amendment 22 to be acceptable for referencing in license applications. The safety evaluation defines the basis for acceptance of the amendment

he do not intend to repeat our review of the matters described in Amendment 22 and found acceptable when the report appears as a reference in license applications, except to ensure that the material presented is applicable to the specific plant involved. Our acceptance applies only to the matters described in the Amendment 22.

In accordance with procedures established in NUREG-0390, it is requested that GE publish accepted versions of this amendment, proprietary and non-proprietary, within three months of receipt of this letter. The accepted version shall incorporate this letter and the enclosed evaluation between the title page and the abstract. 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, GE 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,

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

Enclosure:
Safety Evaluation

ENCLOSURE 1

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION RELATING TO AMENDMENT 22 TO GENERAL ELECTRIC TOPICAL REPORT NEDE-24011-P-A "GENERAL ELECTRIC STANDARD APPLICATION FOR REACTOR FUEL"

1.0 INTRODUCTION

In order to support customer needs and remain competitive, the fuel vendors are continually improving their fuel designs. Generally, the changes in design are made with approved methodologies. The regulatory procedures to qualify and approve the new designs are standard. However, the review and approval of these new designs place a burden on the staff resources.

Recently, the staff proposed that a set of acceptance criteria, to be satisfied by new fuel designs, be established for each fuel vendor. Once the acceptance criteria are approved, the fuel designs or changes satisfying the criteria would not require explicit staff review. Satisfaction of the acceptance criteria would be sufficient for approval by reference to the acceptance criteria. Also, the staff requires that the acceptance criteria be entirely non-proprietary so that any interested party will have access to the acceptance criteria. The objective of this approach is to expedite the review process and reduce the staff and industry resources needed for review of new fuel designs.

In response to the staff proposal, General Electric submitted a topical report, "Proposed Amendment 22 to GE Licensing Topical Report NEDE-24011-P-A," dated September 9, 1988, proposing the new fuel licensing acceptance criteria for staff review. The proposed acceptance criteria consider fuel thermal-mechanical, nuclear, and thermal-hydraulic aspects of design analyses. If a fuel design complies with the fuel acceptance criteria, it is acceptable for licensing applications without the explicit staff review. In a letter dated May 10, 1989 from J.S. Charnley (GE) to M. W. Hodges (USNRC), GE further stated that all the fuel design information in GESTAR II will be transferred to GE Fuel Bundle Designs Information Report (NEDE-3!152P) and will be provided to the NRC. The staff's evaluation of the proposed criteria follows.

2.0 EVALUATION

2.1 General Criteria

GE has proposed five general criteria to deal with generic problems:

1. "NRC-approved analytical models and analysis procedures will be applied."

   This statement is consistent with the current and past practices for new fuel designs. Therefore, this is acceptable.

2. "New design features will be included in lead test assemblies."

   The staff requires that significant new design features be tested before full implementation. GE philosophy apparently is consistent with the staff approach. Therefore, this is acceptable.

3. "The generic post-irradiation fuel examination program approved by the NRC will be maintained."

   We have generically approved the GE post-irradiation examination program in a letter from L. S. Rubenstein (NRC) to R L. Gridley (GE), dated June 27, 1984. GE will continue to use this approved program for new fuel designs. This is acceptable.

4. "New-fuel-related licensing issues identified by the NRC will be evaluated to determine if the current criteria properly address the concern; if necessary, new criteria will be proposed to the NRC for approval."

   GE, in discussions with the staff, agrees that they also have a responsibility to identify new licensing issues. If, for a future new-fuel design, the existing fuel licensing acceptance criteria cannot adequately address all concerns, GE will submit new criteria for the staff review. This approach is acceptable.

5. "If any of the criteria in subsection 1.1 (of Amendment 22) are not met for a new fuel design, that aspect will be submitted for review by the NRC separately."

   If for future new fuel design, any fuel licensing acceptance criteria are not satisfied, GE will submit that specific area for the staff review. However, the related parts or the whole parts associated with this specific area in the fuel design may have to be submitted in order to assist the staff review. This approach is acceptable.

   For future new fuel designs which satisfy the licensing acceptance criteria, we require that GE notify NRC of the first application of each new fuel design based on the approved fuel licensing criteria, and that the fuel bundle design information report be submitted to NRC prior to loading of the new fuel into a reactor.

2.2 Thermal-Mechanical Fuel Licensing Acceptance Criteria

1. "The fuel rod and fuel assembly component stresses, strains, and fatigue life usage shall not exceed the material ultimate stress or strain and the material fatigue capability."

   GE uses a design ratio concept, that is, a ratio of effective stress/strain to ultimate stress/strain, to determine the adequacy of the stress and strain loading on fuel rod and fuel assembly. The design ratio must be less than unity. The design ratio concept was approved in Amendment 7 to GESTAR II. The staff safety evaluation approving Amendment 7 addresses the level of conservatism in calculating the design ratio which was concluded to be acceptable. Therefore, the staff considers that the stress and strain criteria are acceptable. For strain fatigue analysis, GE requires that fatigue life usage, that is, the ratio of actual number of cycles to stress or strain to allowable cycles at stress or strain, must be less than unity. This fatigue criterion was also approved in Amendment 7 to GESTAR II because the associated information indicated that adequate conservatism existed in the methods. Therefore, this criterion is acceptable.

2. “Mechanical testing will be performed to ensure that loss of fuel rod and assembly component mechanical integrity will not occur due to fretting wear when operating in an environment free of foreign material.”

   The Standard Review Plan (SRP) (NUREG-0800) states that allowable fretting wear should be considered in the overall safety analyses. GE considers the effect of fretting wear in design analyses based on testing and experience in reactor operations. It is assumed that foreign material existing as debris or loose parts are not present in the core. The proposed fretting wear criterion was approved in Amendment 7 to GESTAR II; thus, this criterion is acceptable.

3. “The fuel rod and assembly component evaluations include consideration of metal thinning and any associated temperature increase due to oxidation and the buildup of corrosion products to the extent that these effects influence the material properties and structural strength of the components.

   GE considers the effect of external corrosion and buildup of crud on the cladding surface in the design analysis. This approach is consistent with the GE past experience and was approved in Amendment 7 to GESTAR II: thus, this criterion is acceptable.

4. “The fuel rod internal hydrogen content is controlled during manufacture of the fuel rod consistent with ASTM standards C776-83 and C934-85 to assure that loss of fuel rod mechanical integrity will not occur due to internal cladding hydriding.”

   Hydriding as a cladding failure mechanism is precluded by controlling the level of moisture and other hydrogenous impurities during fabrication. GE employed ASTM [American Society for Testing and Materials] standards to control the hydrogen content. Since the use of ASTM standards is consistent with the SRP, this criterion is acceptable.

5. “The fuel rod is evaluated to ensure that fuel rod or channel bowing does not result in loss of fuel rod mechanical integrity due to boiling transition.”

   Fuel assembly components such as fuel rods and channel boxes may undergo various types of dimensional changes such as rod bowing, irradiation growth, and channel box deflection. The result of irradiation growth is rod bowing or channel box deflection. Such phenomena are related to neutron fluence, fuel burnup, and core residence time. Rod bowing can affect local nuclear power peak and heat transfer to the coolant. Channel box deflection can also affect fuel assembly performance with respect to boiling transition. GE-proposed acceptance criteria of rod bowing and channel box deflection are consistent with the previously approved design criteria; thus, this criterion is acceptable.

6. “Loss of fuel rod mechanical integrity will not occur due to excessive cladding pressure loading.”

   In the SRP, the staff stated that fuel and burnable poison rod internal gas pressures should remain below the nominal system pressure during normal operation, unless otherwise justified. GE proposed an alternative higher rod internal pressure criterion; the justification is discussed in the NRC safety evaluation of Amendment 7 to GESTAR II. The staff approved the GE proposal of rod pressure greater than system pressure; therefore, this GE-proposed criterion is acceptable.

7. "The fuel assembly (including channel box) control rod and control rod drive are evaluated to assure control rods can be inserted when required."

   The control rod insertability is required during combined seismic and loss-of-coolant accident (LOCA) loading. GE described these analyses in the approved report NEDE-21175-3-P; which also dealt with the assembly vertical liftoff analyses as part of control rod insertability requirement. This control rod insertability criterion is acceptable.

8. "Loss of fuel rod mechanical integrity will not occur due to cladding collapse into a fuel column axial gap."

   If axial gaps in the fuel column were to occur as a result of densification, the cladding would have the potential of collapsing into a gap. The GE-proposed cladding structural design criterion to preclude collapse was approved in Amendment 7 to GESTAR II; thus, this criterion is acceptable.

9. "Loss of fuel rod mechanical integrity will not occur due to fuel melting."

   The GE 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 during normal operation and corewide anticipated operational occurrences (AOOs). This criterion was approved in Amendment 7 to GESTAR II; thus, it is acceptable.

10. "Loss of fuel rod mechanical integrity will not occur due to pellet-cladding mechanical interaction."

    Fuel failures due to pellet-cladding interaction have been encountered in operating boiling-water-reactor (BWR) fuel. The SRP stated that to preclude pellet-cladding-interaction (PCI) failures, two criteria should be met, although they may not be sufficient: (a) the cladding uniform strain should not exceed 1 percent and (b) fuel melting should be avoided. In the safety evaluation of Amendment 7 to GESTAR II, the staff concluded that the GE design criteria are consistent with the SRP; thus, the PCI criterion is acceptable.

2.3 Nuclear Licensing Acceptance Criteria

1. "A negative Doppler reactivity coefficient shall be maintained for any operating conditions."

   A negative Doppler coefficient guarantees instantaneous negative reactivity feedback to any rise in fuel temperature, thus providing an inherent self-control feature of BWR fuel. This criterion, was previously approved in GESTAR II; thus, it is an acceptable fuel design limitation.

2. "A negative core moderator void reactivity coefficient resulting from boiling in the active flow channels shall be maintained for any operating conditions."

   A negative core moderator void coefficient in the active flow channels flattens the radial power distribution and provides ease of reactor control due to negative void feedback. This criterion was previously approved in GESTAR II; thus, it is acceptable.

3. "A negative moderator temperature coefficient shall be maintained above hot standby."

   The moderator temperature coefficient is associated with the change in the water moderating capability. A negative moderator temperature coefficient during power operation provides inherent protection against power excursion. Since this criterion is consistent with the SRP, it is acceptable.

4. "For a super prompt critical reactivity insertion accident (e.g., control rod drop accident) originating from any operating condition, the net prompt reactivity feedback due to prompt heating of the moderator and fuel shall be negative."

   To mitigate the effects of a superprompt critical reactivity insertion accident such as a control rod drop accident, the mechanical and nuclear fuel design shall be such that the prompt reactivity feedback provides an automatic shutdown mechanism. The negative prompt reactivity feedback criterion is consistent with the SRP requirement and is, thus, acceptable.

5. "A negative power coefficient, as determined by calculating the reactivity change, due to an incremental power change from a steady-state base power level, shall be maintained for all operating power levels above hot standby."

   A power coefficient like the Doppler coefficient or moderator temperature coefficient is maintained negative for reactivity control. Since this power coefficient criterion is consistent with the SRP requirement, it is acceptable.

6. "The plant shall be calculated to meet the cold shutdown margin requirement for each plant cycle specific analysis."

   The core must be designed to remain subcritical with adequate margin with the most reactive control rod in its fully out position and all other rods fully inserted. Since this criterion satisfies the SRP requirement, it is acceptable.

7. "The effective multiplication factor for new fuel designs stored under normal and abnormal conditions shall be shown to meet fuel storage limits by demonstrating that the peak uncontrolled lattice k-infinity calculated in a normal reactor core configuration meets the limits provided in Section 3 (of GESTAR-II) for GE-designed regular or high density storage racks."

   For fuel storage racks, the design criterion of k-infinity is 0.90 for regular racks, and 0.95 for high-density racks, and has been previously approved in GESTAR II. Since this acceptance criterion is consistent with the GESTAR II requirements and the SRP, it is acceptable.

2.4 New Fuel Design Licensing Evaluation

Licensing evaluations of new fuel designs will include generic analyses of a large BWR/4 or BWR/5 plant at limiting points of the cycle for an equilibrium loading of the new fuel design to assure that nuclear design criteria are satisfied and safety limit MCPR values are correct. In addition, Chapter 15 safety analyses are performed for each reload application on a cycle specific basis for limiting anticipated operational occurrences and bounding accidents. The cycle specific plant operating limit

MCPR is determined and the effect of the new fuel design on previously evaluated accidents must be reconfirmed or reanalyzed.

The GE approach is consistent with the applicable regulations and is therefore acceptable.

2.5 Thermal-Hydraulic Licensing Evaluation

"Flow pressure drop characteristics shall be included in plant cycle specific analyses for calculation of the Operating Limit MCPR."

GE stated that a coupled thermal-hydraulic-nuclear analyses will be performed to determine fuel bundle flow and power distribution using the various bundle pressure loss coefficients applicable to a plant specific cycle analyses. The margin to the thermal limits of each fuel bundle is determined using the consistent set of calculated bundle flow and power. GE will explicitly model these pressure drop characteristics in the analysis. The staff finds this evaluation approach to be consistent with the SRP and, therefore, acceptable.

2.6 Safety Limit MCPR Licensing Evaluation

1. "Safety Limit MCPR shall be recalculated following steps in Subsection 1.2.5.B (of Amendment 22) or reconfirmed when a new fuel design or new critical power correlation is introduced."

   The safety limit MCPR is sensitive to bundle design and critical power correlations, for example, GEXL or GEXL-PLUS. The bundle design depends on rod diameter, thermal time constant, spacer, and bundle R-factor. Any change in fuel design or critical power correlation will affect the safety limit MCPR. Therefore, recalculation or reconfirmation of MCPR is necessary. This evaluation commitment is acceptable.

2. GE has established six conditions to be assumed when calculating the safety limit MCPR. These conditions are consistent with the current procedures described in the approved GESTAR II. Therefore, these six conditions for performing MCPR analysis are acceptable.

2.7 Operating Limit MCPR Licensing Evaluation

1. "Plant Operating Limit MCPR is established by considering the limiting anticipated operational occurrences for each operating cycle."

   The operating limit MCPR is determined by adding the change in the MCPR for the limiting AOO to the safety limit MCPR. The limiting AOO events are described in the approved GESTAR II. GE stated that the operating limit MCPR is cycle dependent, and is calculated prior to the cycle operation. This procedure is consistent with the approved GESTAR II approach and is, therefore, acceptable.

2. "For each new fuel design the applicability of generic MCPR analyses described in Section 4 (of GESTAR II) or in the country-specific supplement to this base document shall be confirmed for each operating cycle or a plant-specific analysis will be performed."

   The applicability of new fuel design to the generic MCPR analyses needs to be examined. GE will document its applicability to the rod withdrawal error in the fuel design information

report. This acceptance criterion is consistent with the approved approaches in GESTAR II; thus, it is acceptable.

2.8 Critical Power Correlation Licensing Evaluation

1. "The currently approved critical power correlations will be confirmed or a new correlation will be established when there is a change in wetted parameters of the flow geometry; this specifically includes fuel and water rod diameter, channel sizing and spacer design."

   The coefficients for the critical power correlation are dependent on many fuel assembly design parameters. Changes in fuel design that affect the critical power correlation are not limited to those identified in the proposed criterion. GE has indicated in discussions with the staff that they intend to confirm the applicability of the existing approved critical power correlation or develop a new correlation for any fuel design change that results in parameters impacting the correlation, e.g., rod-to-rod peaking factors, which are significantly outside of the range tested. The staff finds the proposed criterion acceptable with the condition that the critical power correlation for a new fuel design shall be evaluated by testing if features of the new design are atypical of fuel designs previously tested and may impact the applicability of the existing CPR correlation.

2. "A new correlation may be established if significant new data exist for a fuel design."

   When significant new data have been generated for a fuel design, a better fit to the data may be achieved by adjusting the coefficients in the critical power correlation. This acceptance criterion is consistent with the staff position; thus, it is acceptable.

3. GE has established seven conditions to be assumed when calculating a new critical power correlation. These conditions are consistent with the current procedures described in the approved GESTAR II. Therefore, these seven conditions for establishing critical power correlation are acceptable.

2.9 Stability Licensing Acceptance Criteria

The new fuel design must meet either of the two criteria described below.

1. "The stability behavior, as indicated by core and limiting channel decay ratios, must be equal to or better than a previously approved GE BWR fuel design."

   The GE thermal-hydraulic stability licensing criteria set forth in Amendment 8 to GESTAR II allow for the application of the generic bounding stability approach to new fuel designs provided that the stability performance of the new design is bounded by that of current fuel design. GE describes six steps for evaluating the new fuel stability performance against the currently approved fuel design. The staff has approved the GE stability licensing criteria; thus, this stability criterion is acceptable.

2. "If the core and limiting channel decay ratios are not equal to or better than a previously approved GE fuel design, it must be demonstrated that there is no change to the exclusion zone."

If the new fuel design cannot meet the preceding stability acceptance criterion, then GE will demonstrate that there is no change to the exclusion zone as an alternate method for stability acceptance evaluation. The GE proposed criterion is consistent with the staff position; thus, it is acceptable.

2.10 Overpressure Protection Analysis Licensing Evaluation

1. "Adherence to the ASME overpressure protection criteria shall be demonstrated on plant cycle specific analysis."

   GE will demonstrate the adequacy of the plant overpressure protection system on cycle-specific analysis based on core loading pattern. This approach is consistent with the current procedures described in the approved GESTAR II; therefore, this approach is acceptable.

2.11 Loss-of-Coolant Accident Analysis Methods Licensing Evaluation

1. "The criteria in 10CFR50.46 shall be met on plant-specific or bounding analyses."

   The emergency core cooling system (ECCS) criteria in 10CFR50.46 are met by the exposure-dependent maximum average planar linear heat-generation rate (MAPLHGR) limit in plant-specific or bounding analyses. GE will continue to evaluate these ECCS criteria for any new fuel design; thus, this approach is acceptable.

2. "Plant MAPLHGR adjustment factors must be confirmed when a new fuel design is introduced."

   Plant MAPLHGR limit is sometimes adjusted for a special operational configuration or region. GE will confirm the revised MAPLHGR limit for new fuel design before each cycle operation; thus, this approach is acceptable.

2.12 Rod Drop Accident Analysis Licensing Evaluation

1. "Plant cycle specific analysis results shall not exceed the licensing limit in GESTAR-II."

   The current licensing limit of control rod drop accident analysis is 280 cal/gm. GE will perform the rod drop analysis each cycle to ensure compliance with the 280 cal/gm licensing limit. The GE acceptance criterion is consistent with the licensing criterion; therefore, it is acceptable.

2. "Applicability of the bounding BPWS analysis must be confirmed."

   The bounding rod drop accident analysis for plants with bank position withdrawal sequence (BPWS) procedure is dependent on the fuel design and should be confirmed for each new fuel design. The staff agrees with the GE assessment; therefore, this approach is acceptable.

2.13 Refueling Accident Analysis Licensing Evaluation

"The consequence of a refueling accident as presented in the country-specific supplement GESTAR-II or the plant FSAR shall be confirmed as bounding or a new analysis shall be performed."

The consequence of refueling accident is mainly dependent on the amount of fuel rods in a bundle. If there is a change in the number of fuel rods or a new fuel design is proposed, the effect on the refueling accident must be reconfirmed or reanalyzed; therefore, this approach is acceptable.

2.14 Anticipated Transient Without Scram Licensing Acceptance Criteria

The new fuel must meet either of the acceptance criteria described below:

1. "A negative core moderator void reactivity coefficient, consistent with the analyzed range of void coefficients, shall be maintained for any operating conditions above the startup critical condition."

   For core response to an anticipated-transient-without-scram (ATWS) event, the core moderator void reactivity coefficient is the key parameter. If the coefficient remains within the range of void coefficients used in the ATWS point kinetics analyses, the conclusion of BWR mitigation of an ATWS event is still valid for new fuel designs. The GE-proposed ATWS criterion is consistent with the staff guidelines; therefore it is acceptable.

2. "If the preceding criterion is not satisfied, the limiting events will be evaluated to demonstrate that the plant response is within the ATWS criterion."

   For new fuel designs that have core moderator void reactivity coefficients outside the range of point model void coefficients, a specific evaluation of ATWS will be performed for limiting cases to comply with the ATWS acceptance criteria. The GE-proposed criterion is consistent with the staff guidelines; therefore, it is acceptable.

3.0 CONCLUSION

The staff has reviewed the GE submittal, Amendment 22 to GESTAR II, and concludes that the submittal describing a set of licensing acceptance criteria and methods for new fuel design is acceptable for future licensing applications. As a condition of the acceptance, the critical power correlation for a new fuel design shall be evaluated by testing if features of the new design are atypical of fuel designs previously tested and may impact the applicability of the existing CPR correlation. For future reload application, we require that GE notify NRC of the first application of a new fuel design based on the approved fuel licensing criteria, and that the fuel bundle design information report be submitted to NRC prior to loading of the new fuel into a reactor. However, should NRC criteria or regulations change so that staff conclusions as to the acceptability of this submittal are invalidated, or should circumstances arise causing some criteria to be invalid, GE will be expected to revise or resubmit its documentation for further review by the staff.

NRC
Safety Evaluation Report
Approving MFN 089-89
Channel Bow Effects On Thermal Margin

UNITED STATES
NUCLEAR REGULATORY COMMISSION
Washington, D. C. 20555
January 11, 1991

Ms. J.S. Charnley, Manager
Fuel Licensing
General Electric Nuclear Energy
175 Curtner Avenue
San Jose, California 95125

Dear Ms. Charnley:

SUBJECT: ACCEPTANCE FOR REFERENCING OF TOPICAL REPORT TITLED "GE-NUCLEAR ENERGY REPORT MFN086-89"

The staff has completed its review of the Topical Report, "GE-Nuclear Energy Report MFN086-89," submitted by the General Electric Company (GE) by letter of November 15, 1989. GE submitted additional information by letter of September 26, 1990.

This topical report describes GE's assessment of the effect of fuel channel box bowing on thermal margins for plants with GE fuel. We find that the application of the fuel channel box bow assessment is acceptable for use for boiling water reactor (BWR) fuels under the limitations delineated in the enclosed safety evaluation report that was prepared by the U.S. Nuclear Regulatory Commission (NRC). The evaluation defines the basis for acceptance of this topical report.

We will not repeat our review of the matters found acceptable as described in this topical report when the report appears as a reference in license applications, except to ensure that the material presented is applicable to the specific plant involved. Our acceptance applies only to the matters described in the application of this topical report.

In accordance with procedures established in NUREG-0390, we request that GE publish an accepted version of this topical report within 3 months of receipt of this letter. The accepted version 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, GE 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,

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

Enclosure:
MFN-086-89 Evaluation

ENCLOSURE

SAFETY EVALUATION FOR THE TOPICAL REPORT "FUEL CHANNEL BOW ASSESSMENT"

1.0 INTRODUCTION

By letter to the NRC staff (Ref. 1) the General Electric Company (GE) has submitted its assessment of the impact of fuel channel bowing on thermal margins for plants with GE fuel. The submittal includes a determination of the channel deflection due to bowing, the resulting effect on local power peaking, and a determination of the increase in operating limit minimum critical power ratio (MCPR) required to insure adequate thermal margins. The assessment concludes that the plant specific reduction in thermal margin varies from negligible to a few percent. The submittal includes a method for accounting for the effects of channel bow which provides a best estimate MCPR prediction as well as an uncertainty estimate. The estimated uncertainty is found to be much smaller than the existing MCPR uncertainty allowance and is neglected.

GE plans to account for the reduction in MCPR at operating plants by adjusting the R-factor values in the process computer data bank. The MCPR adjustment will be based on a prediction of core average bow using an empirical irradiation growth model that has been qualified by comparison to channel bow measurements, Section 3.5 of this evaluation. It is also assumed that the fuel channels are not reused and the methodology is only applicable to the first fuel channel lifetime

Brookhaven National Laboratory (BNL) has been the staff consultant in this review under FIN No. A-3868.

2.0 SUMMARY OF THE FUEL CHANNEL BOW ASSESSMENT

The GE channel bow assessment includes a quantitative evaluation of the impact of increased channel-to-channel spacing together with a method for accounting for the reduction in thermal margin. The principal elements in this assessment are the determination of: (1) the expected mean and standard deviation of the fuel channel displacements, (2) the sensitivity of the local pin powers to increased channel-to-channel spacing, and (3) the process computer bundle R-factor adjustments required to insure adequate thermal margin. The GE evaluation of each of these factors is summarized in the following.

2.1 Determination of Fuel Channel Bowing

The GE evaluation attributes the fuel channel bowing to three sources: (1) initial bow -B_I, (2) stress relaxation bow with irradiation - delta B_{SR}, and (3) bowing due to irradiation growth of zircaloy - delta B_{IR}. The stress relaxation bow is assumed to be random and averages to zero over the core, while the radial channel bowing due to irradiation growth delta B_{IR} is taken to be a linear function of the strain

difference between the channel walls defining the water gap. The core average bow, which is used to determine the bundle R-factor adjustment, is determined by a full core Monte Carlo procedure in which the bow for each channel is varied randomly about its predicted value by an amount determined by the uncertainty in the initial, stress relaxation and irradiation growth bow components. This calculation results in a substantial core average channel bow for the D-lattice (with the wider water gaps) than that for the C-lattice. The Monte Carlo calculation also calculates a standard deviation in the core average bow due to the input uncertainties. The core-average bow is assumed to have a sinusoidal axial shape with a zero deflection at the top and bottom of the core.

2.2 Calculation of the Increase in Local Power Peaking

The effect of the increased channel-to-channel water gaps that result from channel bowing is calculated with the GE lattice physics code. The primary assumption in these calculations is that the increase in power peaking is determined by the average bow $\overline{B}$ of the four bundles surrounding the control blade. This assumption was validated by three calculations in which the cell average bow $\overline{B}$ was fixed: (1) a single bowed bundle, (2) two diagonal bowed bundles, and (3) two adjacent bowed bundles. While the bundle displacement geometry varied in these calculations, the pin power increases were very similar supporting the sole dependence on $\overline{B}$. The GE analysis then assumes that the power peaking calculations can be performed using a symmetric perturbation in which all four bundles are displaced by the same amount. GE has performed sensitivity calculations that indicate that (to a good approximation) the percent rod power increase is a linear function of the water gap thickness. In addition, the GE calculations indicate that the sensitivity to bowing is a function of bundle exposure and void fraction and is independent of lattice design and enrichment. The calculations presented indicate an increase in corner rod power for a completely symmetric standard deviation in the water gaps.

2.3 Impact of Fuel Rod Bowing on MCPR

The impact of channel bowing on the BWR thermal margin calculation is incorporated as a bundle adjustment applied directly to the R-factor used in the GEXL correlation. The fractional adjustment depends on bundle exposure and void fraction and is linear in the core-average bow $\overline{B}$. The adjustment depends on the rod location in the bundle but is independent of the bundle fuel design. The adjustment is used to convert the core-average bow $\overline{B}$ into a bundle dependent R-factor increase and also to determine the R-factor uncertainty introduced by the spatial/model variation of the bowing $-\sigma_B$. The GE assessment concludes that the R-factor uncertainty is very small compared to existing R-factor uncertainty allowance and may be neglected.

3.0 TECHNICAL EVALUATION

The GE assessment of the effect of channel bowing on the determination of the critical power ratio and the monitoring of the core power distribution is given in Reference 1. The initial review of this assessment resulted in a series of questions which were transmitted to GE in References 2 and 3. The evaluation of the GE channel bowing assessment of Reference 1 and the responses of References 4 and 5 is summarized in the following.

3.1 Determination of Channel Bow

As the fast neutron fluence exposure to the zircaloy fuel channels increases with fuel burnup, the channels undergo irradiation growth and are deflected from their nominal core positions. The increased growth of the channel walls adjacent to the narrow water gaps (regions of relatively high fast flux) tends to deflect the fuel channels diagonally toward the narrow-narrow gap and away from the wide-wide gap. Fuel channels in fast flux gradients (e.g., on the core periphery) also experience channel deflection, however, this bowing may be either diagonal or parallel to the channel faces. The GE bowing methodology assumes that the channel displacement is diagonal, away from the control rod gaps. In order to account for this simplification an additional R-factor uncertainty allowance is included in the MCPR evaluation (Response 19, Ref. 4).

The distribution of channel bow (mean and standard deviation) depends on the cycle core loading. This dependence results from (1) the dependence of the channel strain on the bundle fluence and initial bow and (2) the geometrical dependence of the bowing on the arrangement of the fuel bundles in the core. In the GE methodology the effect of channel bowing is evaluated statistically in terms of the core average bow and the standard deviation about this average $-\sigma_B$. These statistical parameters are determined using a Monte Carlo technique. In this procedure the fuel bundle channel bow is varied randomly based on (1) allowable fabrication tolerances (which determine the initial bow and the zircaloy growth via the channel texture factor) and (2) the uncertainty in the bowing model prediction. In addition, using a cycle-specific reload batch fraction and reload batch average discharge exposure, the bundle loading in each four bundle cell is varied randomly based on cell loading probabilities. The bundle-specific prior operating history is also varied based on design and operating practice. Separate Monte Carlo analyses are performed for the D-lattice plants which have larger control rod water gaps and increased bowing.

The use of generic (1) bundle loading probabilities and (2) bundle operating histories to determine the core-average bow $\overline{B}$ and standard deviation σ_B for a specific reactor cycle introduces an additional uncertainty into the calculation of the effects of channel bowing. In Response-2 of Reference 5 GE has indicated that for Monte Carlo trials, drawn from the same generic probabilities, the plant-to-plant variation in core average bow $\overline{B}$ and its associated uncertainties was found to be negligible compared to plant specific variations.

The added uncertainty in addressing a typical rather than a plant specific core loading is negligible compared to the uncertainty allowance already associated with calculating the MCPR safety limit. That is, channel box bow uncertainties factored into the statistical calculation of the R-factor uncertainties plus MCPR uncertainties, make “typical” channel box bow uncertainties seem negligible in comparison. Consequently, the GE methodology for determining the core average channel box bow and the standard deviation accounts for all significant effects and provides an acceptable estimate of the true calculational uncertainty.

3.2 Effect of Fuel Channel Bow on Local Power Peaking

The effect of channel bowing, and the resulting increased water gaps, on rod power peaking is calculated with the GE lattice physics code. The calculation models the fuel bundles surrounding the control channel water gaps. The boundary conditions imposed on the outer boundary of this cell affect the sensitivity of the power peaking to changes in the water gap thickness. The GE method determines these sensitivities using reflecting boundary conditions. The comparison of reflecting and periodic boundary conditions provided in Response-3 (Ref. 4) indicates that the peaking sensitivities are in approximate agreement, except for the single bundle displacement which is underpredicted, using

reflective boundary conditions. An additional uncertainty allowance is included in the MCPR methodology to account for the effect of these assumed boundary conditions (Response-19, Ref. 4).

In the bowing analysis the four fuel bundles included in the calculational cell are not the actual neighboring bundles that occur in the cycle core loading. However, the power peaking sensitivity to bowing is believed to be independent of the enrichment and exposure distributions of the three neighboring fuel bundles (Response-19, Ref. 4).

The determination of the power sensitivity to channel box bowing is made in a conservative manner, using approved GE methods, and includes an adequate allowance for known uncertainties and is therefore acceptable.

3.3 Effect of Fuel Channel Bowing on the CPR

The decrease in critical power ratio due to channel bowing is determined by an exposure and void dependent correlation which is linear in the cell average channel bow $\overline{B}$. In Figures 20.1 and 20.2 of Response-20, (Ref. 4) GE has provided the dependence of the individual rod powers and increase in R-factor on channel bowing for typical bundle designs. The dependence of both the rod powers and the R-factor is, to a good approximation, linear for bowing up to 60 mils, which is greater than the channel offset expected during normal BWR operation. The slope of the R-factor curves is used to determine the linear sensitivity A_j for each rod in the bundle as a function of exposure and void fraction. A fuel design independent generic value for A_j (E, V) is determined for each rod location in the bundle. The uncertainty in R-factor resulting from the use of the generic, rather than a fuel dependent, sensitivity is less than 0.25 percent. An allowance for this simplification is included in the R-factor uncertainty determination (Response-19, Ref. 4).

It is important to note that the simple linear dependence of the local power peaking and R-factor on the cell-average bow $\overline{B}$ assumes that the limiting MCPR location does not occur in the bundle (in the four-bundle cell) having the largest bow (Response-12, Ref. 4). This could occur in the case of a fresh fuel rod bundle contained in a reused second-lifetime channel with large bow. However, in Response-23 (Ref. 4) GE has indicated that the proposed channel bowing MCPR evaluation methodology is not applicable to second-lifetime fuel channels.

In most cases the fuel rod located in the corner of the fuel bundle experiences the largest increase in power. However, since this rod is not necessarily limiting the R-factor sensitivity A_j (E, V) is calculated for all rods in the bundle.

The core fuel loadings for certain plants include fuel from multiple fuel vendors. In this case, the determination of the core MCPR requires the calculation of the bowing penalty for non-GE fuel. GE has indicated in Response-1 (Ref. 5) that (1) all rods bundles in the core will be GE fuel designs and (2) in each application the utility will confirm that the nominal (unbowed) dimensions of the GE and non-GE fuel channels are identical. Consequently, the calculated R-factor sensitivities A_j (E, V) are applicable to all fuel bundles in the core.

The channel bowing displacement for the non-GE fuel channels is required for the determination of the core average channel bow. GE has indicated in Response-1 (Ref. 5) that in the absence of bowing data for non-GE channels, it will assume the performance of the GE and non-GE channels to be equivalent and will apply the GE bowing correlations to all channels. The acceptability of this

procedure will depend on the design and performance of the specific non-GE channels, and justification and appropriate uncertainty allowance should be provided in each reload application.

It is concluded that, with the limitation to first lifetime channels and the provision for non-GE fuel channels, the calculation of the reduction in CPR due to channel bowing is acceptable.

3.4 Fuel Misloading/Misorientation

Both the average channel bow $\overline{B}$ and the sensitivity of the R-factor to channel bowing A_j may be affected by a fuel misloading or misorientation. The sensitivity A_j, however, is to a good approximation determined by the location of the fuel rod in the bundle and is independent of the fuel bundle orientation. In addition, the fuel channels are generally bowed so that the control rod water gap is increased and, consequently, a misorientation will result in a reduced water gap adjacent to the rotated high enriched rods and an increase in CPR.

In the fuel misloading the maximum delta CPR penalty occurs when a highly exposed low-powered fuel bundle is replaced by a fresh high-powered bundle. Assuming the misloaded bundle is not contained in a second-lifetime channel, the new fuel bundle will have minimum bow and the associated CPR bowing penalty will be less than the value determined using the standard bowing analysis (Response 3, Ref. 5).

The analysis of fuel misloading/misorientation is therefore considered acceptable.

3.5 Evaluation of GE Channel BOC Bow Data

General Electric (GE) provided the NRC staff with channel box bow data as a function of burnup. This data was compared with other channel box bow data obtained from such sources as ANF, EPRI, KWU, and the Swedish Regulatory Authority. The evaluation of the GE data with other inhouse data is given below.

The bow data provided to NRC by GE was data with burnup, ranging from 0 to 50 GWd/MTu. Also, it was assumed in this evaluation, that the statistically evaluated correlation utilized by GE included the addition of two times the standard deviation, i.e.,

$$\text{Bow (mils)} = /\text{Bow min}/ + 2\,\sigma$$

where /Bow min/ refers to the mean value, i.e., average bow of the channel away from the control blade, and sigma is the standard deviation.

To aid the NRC staff in evaluating this data, the staff correlated all of the bow data available to it from fuel vendors and licensees. This bow data was usually in the form of scatter plots of channel bow as a function of burnup. The staff developed limiting curves for each data source. The data from all of the sources were plotted as limiting curves. This provided the NRC staff with a visual representative of all the limiting bow data as a function of exposure and in a graphical manner. The results were very favorable. The plots indicate that there is very good agreement among all the data especially within the constraint of the single bundle lifetime, which is taken to be below approximately (40–50) GWd/MTu. That is, all of the data were consistent in magnitude and trends.

In fact, from 0 to approximately 50 GWD/MTu, channel bowing can be taken to be approximately linear. Beyond 50 GWD/MTu, data from all sources is sparce, consequently, viable data comparison is very difficult. As a result, based on the small number of data points available, one can only point to a trend. As such, beyond 50 GWD/MTu, channel bowing as a function of exposure is not well characterized. However, the rate of bowing with exposure does appear to have increased.

Based on the evaluation discussed above, we conclude that the GE data used in the analysis of channel bow as a function of exposure for single bundle lifetime channel boxes are acceptable.

4.0 TECHNICAL POSITION

The General Electric assessment of the effects of channel bowing, including the determination of critical power and power distribution monitoring, as described in References 1, 3, and 5 has been reviewed in detail. It is concluded that the proposed methodology is acceptable for fuel channel bowing analyses and for referencing in reload licensing applications with the following conditions:

1. The methodology for determining the effect of channel bowing is not applicable to second-lifetime fuel channels (Section 3.3 and Section 3.4).

2. Additional justification and appropriate uncertainty allowances should be provided for each application of the GE procedures and correlations to the determination of the channel bowing of non-GE fuel channels (Section 3.3).

5.0 REFERENCES

1. Letter, J.S. Charnley (GE) to R.C. Jones, Jr. (NRC), “Fuel Channel Bow Assessment,” November 15, 1989.

2. Letter, Robert C. Jones (NRC) to J.S. Charnley (GE), “Request for Additional Information Regarding Fuel Channel Bow Assessment,” March 30, 1990.

3. Letter, J.S. Charnley (GE) to R.C. Jones, Jr. (NRC), “Responses to Channel Bow Questions,” May 3, 1990.

4. Letter, Robert C. Jones (NRC) to J.S. Charnley (GE), “Request for Additional Information Regarding the Letter MFN086-99, ‘Fuel Channel Bow Assessment’,” June 6, 1990.

5. Letter, J.S. Charnley (GE) to R.C. Jones, Jr. (NRC), “Responses to Channel Bow Questions,” September 26, 1990.

NRC
Safety Evaluation Report
Approving Amendment 25 and
Two LTRs on Uncertainties for
Safety Limit MCPR Evaluations

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001
March 11, 1999

MFN-003-99

Mr. Glen A. Watford, Manager
General Electric Company
P.0. Box 780
Wilmington, NC 28402

SUBJECT: ACCEPTANCE FOR REFERENCING OF LICENSING TOPICAL REPORTS NEDC-32601P, METHODOLOGY AND UNCERTAINTIES FOR SAFETY LIMIT MCPR EVALUATIONS; NEDC-32694P, POWER DISTRIBUTION UNCERTAINTIES FOR SAFETY LIMIT MCPR EVALUATION; AND AMENDMENT 25 TO NEDE-24011-P-A ON CYCLE-SPECIFIC SAFETY LIMIT MCPR (TAC NOS. M97490, M99069 AND M97491)

Dear Mr. Watford:

The staff has reviewed the subject reports submitted by GE Nuclear Energy (GENE) by letters dated December 13, 1996, for NEDC-32601P; June 10, 1997, for NEDC-32694P; and December 13, 1996, for Amendment 25 to NEDE-24011P. These submittals provide (1) the description of the procedures used to account for the reload-specific core design and operation in determining the cycle-specific safety limit minimum critical power ratio (SLMCPR) in NEDC32601P; (2) the power distribution uncertainty for the new GE 3D-MONICORE core surveillance system in NEDC-32694P; and (3) the methodology and uncertainties required for the implementation of cycle-specific SLMCPR in Amendment 25 to NEDE-24011-P-A. The staff has found the subject reports to be acceptable for referencing in license applications to the extent specified and under the limitations stated in the GENE letter dated March 1, 1999, the enclosed report, and the U. S. Nuclear Regulatory Commission (NRC) technical evaluation. The evaluation defines the basis for acceptance of the report.

The staff will not repeat its review of the matters described in the GENE Topical Reports NEDC-32601P, NEDC-32694, and Amendment 25 to NEDE-24011-P-A and found acceptable when this letter request appears as a reference in license applications, except to ensure that the material presented applies to the specific plant involved. NRC acceptance applies only to the matters described in the GENE Topical Reports NEDC-32601P, NEDC-32694P, and Amendment 25 to NEDE-24011-P-A. In accordance with procedures established in

NUREG-0390, the NRC requests that GE publish accepted versions of the submittals, proprietary and non-proprietary, within 3 months of receipt of this letter. The accepted versions shall incorporate this letter and the enclosed evaluation between the title page and the abstract and an -A (designating accepted) following the report identification symbol.

If the NRC's criteria or regulations change so that its conclusions that the submittal is acceptable are invalidated, GE and/or the applicant referencing the submittal will be expected to revise and resubmit

its respective documentation, or submit justification for the continued applicability of the submittal without revision of the respective documentation.

Sincerely,

Frank Akstulewicz, Acting Chief
Generic Issues and Environmental Project Branch
Division of Reactor Program Management
Office of Nuclear Reactor Regulation

Enclosures:
NEDC-32601P, NEDC-32694P, and Amendment 25 to NEDE-24022-P-A Evaluation

ENCLOSURE 1

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION RELATING TO GENERAL ELECTRIC LICENSING TOPICAL REPORTS NEDC-32601P, NEDC-32694P, AND AMENDMENT 15 to NEDE-24011-P-A

1. INTRODUCTION

By letters dated December 13, 1996, June 10, 1997, and December 13, 1996, from R. J. Reda (GE) to USNRC, General Electric Nuclear Energy (GENE) submitted licensing topical reports: NEDC-32601P, 'Methodology and Uncertainties for Safety Limit MCPR Evaluation"

(Reference 1); NEDC-32694P, "Power Distribution Uncertainties for Safety Limit MCPR Evaluation," (Reference 2); and Amendment 25 to NEDE-24011-P-A, Proposed Amendment 25 to GE Licensing Topical Report NEDE-24011-P-A (GESTAR II) on Cycle-Specific Safety Limit MCPR," (Reference 3), respectively. The purpose of the submittal is (1) for NEDC32601P to update values of the CPR correlation uncertainties contained in NEDE-10958-P-A (GETAB, Reference 4) based on the most recent analysis of available data; (2) for NEDC32694P to update values of the power distribution uncertainties contained in NEDE-31152P, Revision 5 based on the most recent analysis of available data, and (3) for Amendment 25 to NEDE-24011-P-A to provide for cycle-specific Safety Limit Minimum Critical Power Ratios (MCPRs).

The NRC staff was assisted in this review by its consultant, Brookhaven National Laboratory (BNL). The NRC staff's evaluation includes those three topical reports and the responses to staff's Request for Additional Information (RAI) dated January 8,1998 (GA W-98-002, MFN-004-98, Reference 5), January 9,1998 (GAW-98-003, MFN-005-98, Reference 6), January 28, 1998 (GAW-98-005, MFN-008-98, Reference 7), April 17, 1998 (GAW-98-009, Reference 8), and July 29, 1998 (GAW-98-012, MFN-017-98, Reference 9). The staff adopted the findings recommended in our consultant's Technical Evaluation Report (Enclosure 2).

2 EVALUATION

This review includes three topical reports involving the Safety Limit Minimum Critical Power Ratio (SLMC PR) methodology and input uncertainties described in NEDC-32601 P, the methodology for constructing the bounding statepoint power distribution described in NEDC-32694P, and the overall procedures for determining the cycle-specific SLMCPR described in Amendment 25 to GESTAR II. The details of the evaluation are provided in Enclosure 2.

2.1 Methodology and Uncertainties for Safety Limit MCPR Evaluation (NEDC-32601P)

The topical report, NEDC-32601 P, provides an update to the Safety Limit MCPR methodology and inputs to be used in the evaluation of the Safety Limit MCPR for BWRs (GETAB, Reference 4) including plant surveillance measurement uncertainties and local R-Factor uncertainties. The plant surveillance component uncertainties include the reactor pressure, feedwater temperature and flow, core inlet temperature and flow, and channel flow area and friction factors. The plant surveillance uncertainty revisions are based on current BWR practice, and are generally evaluated using the error methodology (Reference 10). The R-Factor provides the critical power dependence on the local pin power distribution (References 4 and 11) in the GEXL correlation. The R-factor uncertainty analysis

includes an allowance for power peaking modeling uncertainty, manufacturing uncertainty and channel bow uncertainty.

Based on the review of the NEDC-32601 P topical report and the responses to the staff's request for additional information (RAI) (References 5, 8, and 9), we find the SLMCPR methodology and associated uncertainties to be acceptable, however, actions should be taken as follows:

(1) The TGBLA fuel rod power calculational uncertainty should be verified when applied to fuel designs not included in the benchmark comparisons of Table 3.1 of Reference 1, since changes in fuel design can have a significant effect on calculation accuracy.

(2) The effect of the correlation of rod power calculation uncertainties should be reevaluated to insure the accuracy of the R-Factor uncertainty when the methodology is applied to a new fuel lattice.

(3) In view of the importance of MIP criterion and its potential sensitivity to changes in fuel bundle designs, core loading and operating strategies, the MIP criterion should be reviewed periodically as part of the procedural review process to insure that the specific value recommended in NEDC-32601p is applicable to future designs and operating strategies.

2.2 Power Distribution Uncertainties for Safety Limit MCPR Evaluations (NEDC-32694P)

The power distribution uncertainty topical report NEDC-32694P provides a description of the 3D-MONICORE core surveillance system and the determination of the associated bundle power uncertainty for use in SLMCPR calculation. The 3D-MONCORE system uses three-dimensional coarse-mesh diffusion theory methods, together with models for interfacing with the incore TIP and LPRM instrumentation, to determine the detailed core statepoint. The uncertainty in the 3D-MONICORE prediction of bundle power was determined by comparisons of measured and calculated TIP integrals and gamma scanned bundle powers.

Based on the review of Reference 2 and the responses to the staff's RAI (References 6 and 8) we have found that the 3D-MONICORE power distribution uncertainties are acceptable for determining the SLMCPR, MAPLHGR and LHGR core limits, however, the 3D-MONICORE bundle power calculational uncertainty should be verified when applied to fuel and core designs not included in the benchmark comparisons of Tables 3.1 and 3.2 of Reference 2.

2.3 Amendment 25 to GE Licensing Topical Report NEDE-24011-P-A (GESTAR II) on Cycle-Specific Safety Limit MCPR

Amendment 25 to GESTAR II provides the methodology and uncertainties required for the implementation of cycle-specific Safety Limit MCPRs that replace the former generic, bounding SLMCPR. General procedures are given describing the analysis to be used in determining the cycle-specific SLMCPR. These procedures require that the analysis be performed for the specific fuel bundle design and core loading used in the cycle reload design.

Based on the review of References 3 and 7, we have found that the proposed methodology to be acceptable for performing cycle-specific SLMCPR analyses.

3 CONCLUSION

Based on our review of Topical Reports NEDC-32601P, NEDC-32694P, and Amendment 25 to NEDE-2401 1-P-A (GESTAR II), the staff concludes that the input plant system uncertainties, the power distribution uncertainties associated with the application of 3D-MON ICORE, and the proposed cycle-specific determination of the SLMCPR are acceptable. In letter dated PM TO SUPPLY, GENE has stated that they will take the following actions whenever a new fuel design is introduced.

(1) The TGBLA fuel rod power calculational uncertainty should be verified when applied to fuel designs not included in the benchmark comparisons of Table 3.1 of NEDC-32601P, since changes in fuel design can have a significant effect on calculation accuracy.

(2) The effect of the correlation of rod power calculation uncertainties should be reevaluated to insure the accuracy of R-Factor uncertainty when the methodology is applied to a new fuel lattice.

(3) In view of the importance of MIP criterion and its potential sensitivity to changes in fuel bundle designs, core loading and operating strategies, the MIP criterion should be reviewed periodically as part of the procedural review process to insure that the specific value recommended in NEDC-32601 P is applicable to future designs and operating strategies.

(4) The 3D-MONICORE bundle power calculational uncertainty should be verified when applied to fuel and core designs not included in the benchmark comparisons in Tables 3.1 and 3.2 of NEDC-32694P.

4 REFERENCES

1. GE Letter RJR-96-139 MFN-185-96 dated December 13, 1996 from R. J. Reda to USNRC transmitting a topical report, NEDC-32601 P, “Methodology and Uncertainties for Safety Limit MCPR Evaluations,” December 1996.

2. GE Letter RJR-97-074 MFN-022-97 dated June 10, 1997 form R. J. Reda to USNRC transmitting a topical report, NEDC-32694P, “Power Distribution Uncertainties for Safety Limit MCPR Evaluations,” January 1997.

3. GE Letter RJR-96-133 MFN-179-96 from R. J. Reda to USNRC, “Proposed Amendment 25 to GE Licensing Topical Report NEDE-2401 1-P-A (GESTAR II) on Cycle Specific Safety Limit MCPR,” December 13, 1996.

4. General Electric BWR Thermal Analysis Basis (GETAB): Data, Correlation and Design Application, NEDE-10958-PA, January 1977.

5. GE Letter GAW-98-002 MFN-004-98 from Glen A. Watford to USNRC, Responses to Request for Additional Information for GE Topical Report NEDC-32601 P, Methodology and Uncertainties for Safety Limit MCPR Evaluations, January 8,1998.

6. GE Letter GAW-98-003 MFN-005-98 from Glen A. Watford to USNRC, Responses to Request for Additional Information for GE Topical Report NEDC-32694P, Power Distribution Uncertainties for Safety Limit MCPR Evaluations, January 9,1998.

7. GE Letter GAW-98-005 MFN-008-98 from G. A. Waterford to USNRC, Responses to Request for Additional Information for Amendment 25 to GE Topical Report NEDE-24011-*P-A* (GESTAR II) on Cycle-Specific Safety Limit MCPR (TAC No. M97491), January 28, 1998.

8. GE Letter GAW-98-009 MFN-014-98 from Glen A. Watford to USNRC, Responses to NRC Request for Additional Information associated with SLMCPR Methodology and Uncertainty Topical Reports NEDC-32601P and NEDC-32694P, April 17, 1998.

9. GE Letter GAW-98-012 MFN-017-98 from Glen A. Watford to USNRC, Additional Information Associated with SLMCPR Methodology and Uncertainty Topical Report NEDC-32601P, July 29, 1998.

10. "Recommended Practice - Setpoint Methodologies," Part II, ISA-RP 67.04, Instrument Society of America, September 1994.

11. NEDC-32505, Revision 1, "R-Factor Calculation Method for GEII, GEI2, and GEI3 Fuel," June 1997.

ENCLOSURE 2

TECHNICAL EVALUATION REPORT

Report Titles:	1)	Power Distribution Uncertainties for Safety Limit MCPR Evaluations
	2)	Methodology and Uncertainties for Safety Limit MCPR
	3)	Amendment 25 to GE Licensing Topical Report NEDE-24011-P-A (GESTAR) on Cycle-Specific Safety Limit MCPR
Report Numbers:	1)	NEDC-32694P
	2)	NEDC-32601P
	3)	NEDE-24011-P-A
Report Dates:	1)	January 1997
	2)	December 1996
	3)	December 1996

Originating Organization: General Electric Company

1.0 INTRODUCTION

In Reference-1, the General Electric Company (GE) has submitted the proposed GESTAR modifications for including the cycle-specific Safety Limit MCPR (SLMCPR), replacing the generic bounding SLMCPR methodology included in GESTAR, for NRC review and approval. These modifications provide the licensing methods to be used in determining the cycle-specific SLMCPR for each plant reload. In support of these modifications, GE has submitted the two additional licensing topical reports: (1) NEDC-32601P (Reference-2), “Methodology and Uncertainties for Safety Limit MCPR,” and (2) NEDC-32694P (Reference-3), “Power Distribution Uncertainties for Safety Limit MCPR Evaluations.” The NEDC-32601P Topical Report describes the procedures used to account for the reload-specific core design and operation in determining the cycle-specific SLMCPR. In this topical report, the values of the plant monitoring uncertainties and local R-Factor uncertainty used in the SLMCPR determination are also reviewed and updated to reflect current recommended practices.

The NEDC-32694P Topical Report provides the power distribution uncertainty for the new GE 3D-MONICORE core surveillance system. The 3D-MONICORE power distribution uncertainties are determined based on an uncertainty propagation analysis and on comparisons with benchmark measurements. The resulting 3D-MONICORE uncertainties are used in the determination of the SLMCPR for the plants employing the 3D-MONJCORE system.

The review of the GE core monitoring and SLMCPR analysis was included in the NRC vendor inspections (Nos. 99900003/95-01 and 99900003/96-01) at the General Electric Nuclear Energy Facility in Wilmington, NC during the weeks of August 14 through September 1, 1995 and May 6

through May 10, 1996. Several important concerns were identified during these reviews including: (1) the level of conservatism in the operating state assumed in the cycle-specific determination of the SLMCPR and (2) the effect of the 3D-MONICORE uncertainties on the SLMCPR uncertainty analysis. These concerns are addressed in the safety limit methodology and uncertainty analysis Topical Report NEDC-32694P and the power distribution uncertainty Topical Report NEDC32694P, respectively.

The purpose of this review was to evaluate these methodology modifications and updates to insure that the changes in the monitoring uncertainties are acceptable and that adequate margin is included in the determination of the SLMCPR. The methodology changes are summarized in Section 2, and the evaluation of the important technical issues raised during this review is presented in Section 3. The Technical Position is given in Section 4.

2.0 SUMMARY OF THE REVISED SLMCPR METHODOLOGY

2.1 Power Distribution Uncertainties for Safety Limit MCPR Evaluations (NEDC-32694P)

The power distribution uncertainty Topical Report NEDC-32694P provides (1) a description of the 3D-MONICORE core surveillance system and (2) the determination of the associated bundle power uncertainty for use in SLMCPR calculations. The 3D-MONICORE system uses three-dimensional coarse-mesh diffusion theory methods, together with models for interfacing with the incore TIP and LPRM instrumentation, to determine the detailed core statepoint. The physics methods used in 3D-MONICORE are identical to those used in BWR fuel design calculations and core management evaluations. 3D-MONICORE solves a modified diffusion theory equation in order to allow the local normalization of the power distribution to the TIP and LPRM incore measurements. However, prior to this normalization, the TIP/LPRM measurements are compared to the instrument responses predicted by 3D-MONICORE. If these comparisons indicate that certain measurements are suspect, this data is rejected and the normalization is performed with the remaining reliable TIP/LPRM measurements.

The uncertainty in the 3D-MONICORE prediction of bundle power was determined by comparisons of measured and calculated TIP integrals and gamma scanned bundle powers. These comparisons included a wide range of fuel enrichments, poison loadings and operating conditions. The increased uncertainty between TIP measurements was determined by comparing LPRM-updated TIPs and TIP measurements taken immediately following the LPRM update. The uncertainty analysis also accounts for TIP and LPRM failures (i.e., measurement rejection). The NEDC-32694P uncertainty analysis indicates that the 3D-MONICORE power distribution uncertainty is less than the value presently used in the GETAB SLMCPR determination.

2.2 Methodology and Uncertainties for Safety Limit MCPR (NEDC-32601P)

The NEDC-32601P Topical Report documents the latest updates to the GETAB (Reference-4) (1) plant surveillance measurement uncertainties, (2) local R-Factor uncertainties and (3) SLMCPR methodology. The plant surveillance component uncertainties include the reactor pressure, feedwater temperature and flow, core inlet temperature and flow, and channel flow area and friction factors. The plant surveillance uncertainty revisions are based on current BWR practice, and are generally evaluated using the error methodology of Reference-5. The uncertainty analysis accounts for the overall instrument channel accuracy, drift, calibration, process uncertainty, and plant environmental effects. In most cases, a simple sum-of-the-squares combination of the contributing uncertainties is employed, however, the uncertainty in the inlet subcooling (i.e., core inlet temperature) is determined using the process computer heat balance to propagate the uncertainties.

In most cases, the reevaluation of the plant surveillance uncertainties concluded that the presently accepted GETAB uncertainty values are conservative. A detailed analysis is provided to support the revised values in the cases where the reevaluation results in a reduction in the component uncertainty.

In the GEXL correlation, the R-Factor provides the critical power dependence on the local pin power distribution (References 4 and 6). The R-Factor uncertainty analysis includes an allowance for power peaking modeling uncertainty, manufacturing uncertainty and channel bow uncertainty. The TGBLA (Reference-7) power peaking modeling uncertainty is determined by comparisons of TGBLA with MCNP (Reference-8) and quarter-core benchmark calculations for a range of BWR fuel bundle and core designs. The power peaking uncertainty determined by this analysis was confirmed with gamma scan measurements taken following Cycle-8 of the Duane Arnold Plant (Reference-9).

The uncertainty in the power peaking resulting from channel bow is determined using the procedures of Reference-10, and the uncertainty introduced by the manufacturing process is based on estimated fuel enrichment and density measurements. The R-Factor uncertainty is determined by propagating the resulting local power peaking uncertainty using the R-Factor dependence on peaking factor.

The revised methodology includes updates to the calculation process used to determine the SLMCPR. The operating core statepoint is determined using the PANACEA (Reference-7) 3D-simulator program. The statepoint information used in the SLMCPR calculation includes the channel flows, bundle powers, local void fraction and the TIP detector responses. In addition, the bundle and exposure dependent R-Factors are obtained from the PANACEA statepoint data and used to determine the critical power. The SLMCPR is determined by randomizing the statepoint surveillance input and correlation data to determine the MCPR margin required to insure that 99.9 % of the rods avoid boiling transition.

The SLMCPR is sensitive to the assumed statepoint radial power distribution. In the cycle-specific methodology, the power distribution is selected to provide a reasonable bound on the number of rods expected to experience boiling transition. This selection is made subject to the condition that the core is critical and within thermal limits. For current BWR reload designs the limiting radial power distribution includes a centrally located high powered region which is either circular or annular in shape. Control rod patterns which provide these limiting power distributions are described and recommended. In order to quantify the severity of power distributions with respect to the number of rods in boiling transition a core weighting parameter is defined. The frequency distribution of this parameter is used to compare and select the limiting power distribution.

The determination of the SLMCPR using the revised methodology and input uncertainties is compared to the presently accepted GETAB methodology for several plants. For the cases evaluated, the effect of the changes in methodology and uncertainties is small ~.01 ΔSLMCPR.

2.3 GESTAR II Amendment 25 on Cycle-Specific Safety Limit MCPR (NEDE-24011-P-A)

Amendment 25 to GESTAR II provides the methodology and uncertainties required for the implementation of the cycle-specific Safety Limit MCPR. A set of general procedures are given describing the analysis to be used in determining the cycle-specific SLMCPR. These procedures require that the analysis be performed for the specific fuel bundle design and core loading used in the cycle reload design. The core radial power distribution must represent a reasonable bound on the number of fuel bundles at or near thermal limits, and the fuel assembly local power distribution must be based on the actual bundle design. The cycle-specific analysis is performed at multiple exposure points throughout the cycle, and either the most limiting or an exposure-dependent SLMCPR is used

in determining the Operating Limit MCPR (OLMCPR). The cycle-specific procedures require that the SLMCPR be recalculated or' reconfirmed for each plant operating cycle.

In the reload process, the final core loading plan is evaluated relative to the reference design criteria including the OLMCPR. If the cycle-specific determination results in an increased SLMCPR, the final core loading plan may fail to satisfy the specified acceptance criteria. In this case, calculations of the sensitivity of the OLMCPR to changes in the SLMCPR are used to determine the acceptability of the calculated cycle-specific SLMCPR.

While Amendment 25 provides the overall procedures for determining the cycle-specific SLMCPR, the detailed SLMCPR methodology and input uncertainties are described in NEDC-32601P and the methodology for constructing the bounding statepoint power distribution is described in NEDC-32694P.

3.0 SUMMARY OF THE TECHNICAL EVALUATION

The GE Topical Reports NEDC-32694P, NEDC-32601P and Amendment 25 to NEDE-24011-P-l (GESTAR II) provide the basis for the cycle-specific determination of the SLMCPR, input plant system uncertainties and the power distribution uncertainties associated with the application of 3D-MONICORE. The review of the GE methodology focused on: (1) the assumptions made in the cycle-specific SLMCPR methodology and the changes relative to the presently approved generic SLMCPR approach and (2) the basis for the changes in the SLMCPR uncertainty values. As a result of the review of the methodology, several important technical issues were identified which required additional information and clarification from GE. This information was requested in References-11 and 12 and was provided in the GE responses included in References 13–16. This evaluation is based on the material presented in the topical reports (References 1-3) and in References 13–19. The evaluation of the major issues raised during this review are summarized in the following.

3.1 Power Distribution Uncertainties for Safety Limit MCPR Evaluations (NEDC-32694P)

The 3D-MONICORE system is used to perform the steady-state on-line core performance evaluation. The 3D-MONLCORE models are based on accepted BWR calculational methods. The neutronics model is essentially the same as that described in Reference-7 and the thermal-hydraulics model is the same as presently used in the P-1 Process Computer Analysis (Reference-13, Response 11.4).

The 3D-MONICORE power distribution uncertainties are required for determining the SLMCPR, LHGR and MAPLHGR limits. The (axially integrated) bundle power uncertainty is required for the SLMCPR and the nodal power uncertainty is required for determining MAPLHGR and LHGR. The radial bundle power uncertainty is considered to be a statistical combination of: (1) the uncertainty in the four-bundle power associated with the TIP location and (2) the uncertainty in the allocation of the four-bundle power to the individual bundles. The four-bundle power uncertainty is determined by a comparison of the predicted and measured TIP responses, and the uncertainty in the power allocation is determined by comparisons of calculated and measured (gamma-scanned) bundle powers.

While the calculated bundle powers were determined with the "core tracking" system, rather than with 3D-MONICORE, GE has indicated (in Reference- 13, Responses 1.2 and 1.6) that the difference in these codes has no effect on the uncertainty estimates. The TIP comparisons include cores with both part length fuel rods and axially zoned gadolinium, and all current fuel product lines except for GE13. However, in view of the similarity of the GE13, GE11, and GE12 fuel designs, this is considered acceptable (Reference-13, Response-II.4). In addition, GE has indicated that the core follow calculations employed the same methods to process and accumulate the void-history and fuel exposure

as used in the on-line core surveillance (Reference-l 3, Response-II.8). However, it is concluded that since changes in the fuel and core design can have a significant effect on the calculation accuracy, the 3D-MONICORE bundle power calculational uncertainty should be verified when applied to fuel and core designs not included in the benchmark comparisons of Tables-3.l and 3.2 of NEDC-32694P.

The review of the calculation-to-measurement (C/M) comparisons indicated an increased uncertainty at end-of-cycle. However, the cycle-average four-bundle power uncertainty is considered acceptable since the uncertainty estimate does not take credit for the uncertainty increase due to TIP measurement uncertainty. The nodal power uncertainty is determined by a statistical combination of the 3D-MONICORE bundle power uncertainty and an accepted TIP axial power uncertainty. The TIP uncertainty is measured once per cycle to ensure that it satisfies the specified acceptance criteria.

The 3D-MONICORE system allows rejection of the TIP measurement data based on a specified acceptance criteria. During the review it was noted that the 3D-MONICORE acceptance criteria will, under certain conditions, reject good TIP measurement data. However, in Responses-I.7 and I.10 (Reference- 13), GE has indicated that the rejection of TIP data is very rare. In addition, in most cases TIP rejection is due to poor agreement between measured and calculated data and, when the acceptance criteria results in rejection of measurements which are in good agreement with the calculations, the effect on the core power distribution uncertainty is negligible.

The uncertainty methodology determines the effect of TIP and LPRM rejection and the LPRM update of the power distribution using comparisons of calculations and measurements. In these comparisons the recommended value for the rejection criteria parameter is used. It is noted that after ten years of operation, no correlation has been observed between the rejected TIP locations and the core locations that are difficult to calculate, such as the peripheral core locations, part-length fuel bundles and partially controlled fuel bundles. It is concluded that the TIP rejections are generally a result of erroneous measurement data rather than miscalculation. It is also noted that the TIP rejection only affects the 3D-MONICORE system and the other BWR surveillance systems use the measured TIP/LPRM data.

The process computer monitors kw/ft and LHGR as well as the SLMCPR. The uncertainty analysis for the 3D-MONICORE LHGR evaluation is provided in Response-II.5 (Reference- 14) and accounts for the effect of both the TIP and LPRM update uncertainties on the nodal power calculation.

Based on the review of the NEDC-32694P topical report and supporting documentation provided in References 13 and 14, it is concluded that the 3D-MONICORE power distribution uncertainties are acceptable for determining the SLMCPR, MAPLLIGR and LHGR core limits subject to the condition identified above (in the third paragraph of this section).

3.2 Methodology and Uncertainties for Safety Limit MCPR (NEDC-32601P)

3.2.1 Process Computer Uncertainties

The reevaluation of the process computer uncertainties provided in the NEDC-32601 P topical report were reviewed in detail. The topical report provides a description of both the instrumentation and modeling uncertainties that are required for the SLMCPR analysis. The evaluation of the core inlet subcooling uncertainty employs the heat balance used by the process computer to relate the inlet subcooling to the available instrumentation signals. Using this relation, the inlet subcooling variance is determined by the individual component variances (e.g., feedwater flow and temperature, core flow, steam carry under fraction). While the coefficients that weight the individual uncertainty components depend on the reactor statepoint, the analysis neglects this dependence and assumes constant

weighting coefficients. In Response-I. 1 (Reference-15), GE has shown using a Monte Carlo procedure that these constant weighting coefficients are conservative.

The calculation of the bundle critical power is sensitive to the channel flow area and friction factors. The two-phase friction factor is based on measurements made at the full scale ATLAS test facility covering a range of power and flow The uncertainty in the two-phase friction factor is based on the comparisons with test data. The uncertainty in the single-phase friction factor is determined by comparison of the calculations to total pressure drop measurements made at the ATLAS facility. In Response-II.6 (Reference-15), it is noted that, since the total pressure drop measurement includes both the single-phase and two-phase losses, the inferred single-phase loss coefficient is conservative.

The channel flow area is subjected to random variations due to non-uniform channel bulge and crud/corrosion buildup which result in channel-to-channel variations in flow. The SLMCPR uncertainty analysis accounts for the effect of these variations by increasing the uncertainty in the channel-to-channel friction factor multiplier (Response-I.2, Reference-15).

3.2.2 R-Factor Uncertainties

The fuel rod power calculational uncertainty determines the GEXL R-Factor uncertainty and is separated into three components; modeling, manufacturing and bowing. The modeling uncertainty is determined by comparison of the TGBLA calculation to MCNP benchmark lattice calculations. The Table-3.l TGBLAIMCNP rod power comparisons include all GE fuel designs which are currently loaded in operating BWRs (Response-II.1, Reference-iS). A range of gadolinium rods is included in the comparisons in order to simulate the effects of depleted fuel rods (Response-II.2, Reference-15). The fuel rod power peaking uncertainty is determined by weighting the variance for each fuel design by the number of rods in the lattice (Response-II.3, Reference-iS). However it is concluded that since changes in the fuel lattice design can have a significant effect on the calculation accuracy, the TGBLA fuel rod power calculational uncertainty should be verified when applied to fuel designs not included in the benchmark comparisons of Table-3.1 of NEDC-32601P.

In addition to the TGBLA!MCNP comparisons, GE has evaluated the effect of void fraction uncertainty on the fuel rod power calculation (Response-II.10, Reference-15). Estimates of the lattice-average void fraction uncertainty were determined by comparison with measurement. The local void fraction uncertainty was determined by comparison with detailed subchannel calculations. The effect of the lattice-average and local void fraction uncertainties on the fuel rod power calculation was determined by sensitivity calculations and found to be negligible.

The fuel rod manufacturing uncertainty includes the effects of fuel enrichment, density and rod position uncertainty. The uncertainty in fuel enrichment and density was determined from measurements on a large number of fuel rods performed as part of manufacturing studies. The fuel rod position uncertainty was determined from a series of rod spacing measurements performed on a high burnup fuel bundle. In Responses 11.4, 11.5, and 11.9 of References 14 and *15,* GE has shown that the effects of these uncertainties are conservatively included in the R-Factor analysis. In Response-II.10 (Reference- 14), the effect of local fuel bundle exposure uncertainty on rod power is shown to be negligible. It is important to note that the power peaking uncertainty is determined using a components of uncertainty approach and then independently confirmed by a comparison with gamma scan measurements.

In the approved GETAB methodology of Reference-4, the power peaking calculation errors in neighboring fuel rods are assumed to be correlated so that each of the fuel rods has exactly the same

calculational error. In the proposed methodology, the modeling errors in neighboring fuel rods are assumed to be uncorrelated. As a result, the uncertainty in the R-Factor is reduced significantly in the proposed methodology. In Response-II.13 of Reference 14 and in References 17-19, GE has evaluated this effect for the 8x8, 9x9 and 10x10 lattices and has indicated that the R-Factor uncertainty will be increased (relative to the presently approved value of Reference-4) to account for the correlation of rod power uncertainties. However, in References-18 and 19 (Table-1), it is noted that the effect of the rod-to-rod correlation has a significant dependence on the fuel lattice (e.g., 9x9 versus 10x10). Therefore, in order to insure the adequacy of the R-Factor uncertainty, the effect of the correlation of rod power calculation uncertainties should be reevaluated when the NEDC-32601P methodology is applied to a new fuel lattice.

3.2.3 SLMCPR Evaluation Methodology

The SLMCPR is sensitive to the "flatness" of the bundle power distribution of the initial reactor statepoint. GE has defined a MCPR Importance Parameter (MIP) to allow a quantitative assessment of the flatness of the power distribution and identify limiting statepoints for SLMCPR analysis. In Response-III.2 (Reference- 15), the expression for determining MIP is derived and shown to provide a quantification of the effect of the bundle power distribution on the SLMCPR. In Reference-15 GE provides the specific MIP criterion (Response-III.5) and the thermal limits and reactivity constraints (Response-III.6) for selecting the bundle power distribution to be used in the SLMCPR analysis.

The determination of the selected MIP criterion is based on an extensive evaluation of operating reactor statepoints. In view of the importance of this MIP criterion and its potential sensitivity to changes in fuel bundle designs, core loadings and operating strategies, there is a need to insure that the specific value recommended in NEDC-32601P is applicable to future designs and operating strategies. In response to this concern, GE has indicated that the MIP criterion will be reviewed periodically as part of the procedural review process (Response III.6, Reference-15).

In the presently approved GETAB methodology (Reference-4), the bundle power calculation error is assigned to the four bundles surrounding the TIP in a correlated manner so that each of the four bundles is perturbed simultaneously by the same amount. In the proposed methodology, the calculational error is assumed to be uncorrelated and the individual bundle powers are varied independently in the Monte Carlo uncertainty propagation. The increased variability in the proposed methodology results in a (non conservative) reduction in the SLMCPR. In Response-III.1 of Reference-14, GE has revised the NEDC-32601P methodology to allow for the correlation of the bundle power calculation modeling errors.

Based on the review of the NEDC-32601P topical report and supporting documentation provided in References 14 and 15, we find the SLMCPR methodology and associated uncertainties to be acceptable subject to the conditions identified in Sections-3.2.2 and 3.2.3.

3.3 GESTAR II Amendment 25 on Cycle-Specific Safety Limit MCPR (NEDE-24011-P-A)

Amendment 25 to GESTAR II provides the modifications required for performing the cycle-specific SLMCPR analysis. In the cycle-specific analysis, a search is performed to determine the initial reactor statepoint for use in the Monte Carlo statistical analysis. The purpose of this search is to determine a reactor statepoint that satisfies both (1) the operations criteria required for operating statepoints and (2) the MIP flatness criterion to insure that the statepoint provides a bounding SLMCPR. In the information provided in support of Amendment 25 (Reference-1, Corrective Action-4, Item-3), it is noted that this search may be terminated before all criteria are satisfied. However, in Responses 2 and

3 (Reference- 16), GE has indicated that if the MIP criterion is not initially satisfied the search will be expanded, by relaxing the operations criteria, to insure that the MIP criterion is satisfied.

In the presently approved GETAB methodology, the limiting power shape is assumed to include a centrally located annular ring of high-powered fuel bundles. While the proposed cycle-specific methodology does not require the power distribution to include this high-powered annular zone, it is indicated in Response-4 (Reference-16) that the control rods are selected so that this power shape is not precluded from the search for the bounding statepoint.

Based on the review of Amendment 25 and the supporting information provided in Reference-16, we find the proposed methodology to be acceptable for performing cycle-specific SLMCPR analyses.

4.0 TECHNICAL POSITION

The Topical Reports NEDC-32694P, NEDC-32601P and Amendment 25 to NEDE-24011-P-A (GESTAR II) and supporting documentation provided in References 13–16 have been reviewed in detail. Based on this review, it is concluded that the proposed cycle-specific determination of the SLMCPR, the input plant system uncertainties, and the power distribution uncertainties associated with the application of 3D-MONICORE are acceptable subject to the conditions stated in Section 3 of this evaluation and summarized in the following.

1) Since changes in the fuel and core design can have a significant effect on the calculation accuracy, the 3D-MONICORE bundle power calculational uncertainty should be verified when applied to fuel and core designs not included in the benchmark comparisons of Tables-3.l and 3.2 of NEDC-32694P (Section 3.1).

2) Since changes in fuel design can have a significant effect on the calculation accuracy, the TGBLA fuel rod power calculational uncertainty should be verified when applied to fuel designs not included in the benchmark comparisons of Table-3.l of NEDC-32601P (Section-3.2.2).

3) In order to insure the adequacy of the R-Factor uncertainty, the effect of the correlation of rod power calculation uncertainties should be reevaluated when the NEDC-32601P methodology is applied to a new fuel lattice (Section-3.2.2).

4) In view of the importance of the MIP criterion and its potential sensitivity to changes in fuel bundle designs, core loadings and operating strategies, the MIP criterion should be reviewed periodically as part of the procedural review process to insure that the specific value recommended in NEDC-32601P is applicable to future designs and operating strategies (Section-3.2.3).

5.0 REFERENCES

1. "Proposed Amendment 25 to GE Licensing Topical Report NEDE-24011-P-A (GES TAR II) on Cycle-Specific Safety Limit MCPR," RJR-96-133, Letter R. J. Reda (GE) to U.S. NRC, dated December 13, 1996.

2. "GE Licensing Topical Report, Methodology and Uncertainties for Safety Limit MCPR Evaluations," RJR-96-139, Letter, R. J. Reda (GE) to U.S. NRC, dated December 13, 1996.

3. "GE Licensing Topical Report, Power Distribution Uncertainties for Safety Limit MCPR Evaluations," RJR-97-074, Letter, R. J. Reda (GE) to U.S. NRC, dated June 10, 1997.

4. “General Electric BWR Thermal Analysis Basis (GETAB) Data, Correlation and Design Application,” NEDO-10958-A, January 1977.

5. “Recommended Practice - Setpoint Methodologies,” Part II, ISA-PP 67.04, Instrument Society of America, September 1994.

6. “R-Factor Calculation Method for GE11, GE12, and GE13 Fuel,” NEDC-32505P, November 1995.

7. “Steady State Nuclear Methods,” NEDE-30130-P-A, April 1985.

8. “MCNP - A General Monte Carlo N-Particle Transport Code, Version 4a,” LA-12625, J. F. Breismeister, Ed., Los Alamos National Laboratory (1993).

9. “Gamma Scan Measurements of the Lead Test Assembly at the Duane Arnold Energy Center Following Cycle-8,” NEDC-3 1569-P. April 1988.

10. “Fuel Channel Bow Assessment,” GENE Report MFN086-89, Letter, J. S. Charnley (GE) to R. C. Jones (NRC), dated November 15, 1989.

11. “Request for Additional Information for GE Topical Reports NEDC-32601 P and NEDC-32694P,” Letter, J. H. Wilson (NRC) to R. J. Reda (GE), dated August 20, 1998.

12. “Request for Additional Information for Amendment 25 to GE Topical Report NEDE-24011-P-A (GESTAR II) on Cycle-Specific Safety Limit MCPR (TAC No. M97491),” Letter, J. H. Wilson (NRC) to R. J. Reda (GE), dated October 21, 1998.

13. “Responses to Request for Additional Information for GE Topical Report NEDC-32694P,” “Power Distribution Uncertainties for Safety Limit MCPR Evaluations,” GAW-98-003, Letter, G. A. Watford (GE) to U.S. NRC, dated January 9, 1998.

14. “Responses to NRC Request for Additional Information Associated with SLMCPR Methodology and Uncertainty Topical Reports NEDC-32601P and NEDC-32694P,” GAW98-009, Letter, G. A. Watford (GE) to U.S. NRC, dated April 17, 1998.

15. “Responses to Request for Additional Information for GE Topical Report NEDC-32601P, Methodology and Uncertainties for Safety Limit MCPR Evaluations,” GAW-98-002, Letter, G. A. Wafford (GE) to U.S. NRC, dated January 8, 1998.

16. “Responses to NRC Request for Additional Information for Amendment *25* to GE Topical Report NEDE-2401 1-P-A (GESTAR II) on Cycle-Specific Safety Limit MCPR (TAC No. M97491),” GAW-98-005, Letter, G. A. Watford (GE) to U.S. NRC, dated January 28, 1998.

17. “Additional Information Associated with SLMCPR Methodology and Uncertainty Topical Report NEDC-32601P,” GAW-98-012, Letter, G. A. Watford (GE) to U.S. NRC, dated June 12, 1998.

18. “Additional Information Associated with SLMCPR Methodology and Uncertainty Topical Report NEDC-32601P,” GAW-98-012, Letter, G. A. Watford (GE) to U.S. NRC, dated July 29, 1998.

19. "Additional Information Associated with SLMCPR Methodology and Uncertainty Topical Report NEDC-32601P," GAW-98-017, Letter, G. A. Watford (GE) to U.S. NRC, dated September 9, 1998.

NRC
Safety Evaluation Report
Approving Amendment 26 –
Implementing Improved GE Steady–State Methods

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

November 10, 1999

Mr. Glen A. Watford, Manager
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SUBJECT: AMENDMENT 26 TO GE LICENSING TOPICAL REPORT NEDE-2401 1-P-A, "GESTAR II" IMPLEMENTING IMPROVED GE STEADY-STATE METHODS (TAC NO. MA6481)

Dear Mr. Watford:

By letter dated October 22, 1999, General Electric Nuclear Energy (GENE), submitted a request for review of Amendment 26 to GESTAR as a standard amendment. GESTAR Amendment 26 was submitted by letter dated August 13, 1999, as an administrative change, and involved three areas. These areas were: (1) clarifying classification of BWR 6 pressure regulator failure downscale event; (2) implementing improved General Electric (GE) steady-state methods; and (3) incorporation of Boiling Water Reactor Owners Group (BWOG) approved stability options. This submittal was later supplemented by letters of September 17 and October 14, 1999. The NRC staff has reviewed Item 2 of the submittal, in relation to the GE improved steady-state methods, and has prepared the enclosed safety evaluation. We find Item 2 of the proposed Amendment 26 acceptable, as discussed in the safety evaluation. Items 1 and 3 will be addressed under a separate cover letter.

Sincerely,

Stuart A. Richards, Director
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Project No. 691

Enclosure: Safety Evaluation

cc w/encl: See next page

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SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

AMENDMENT 26 TO GENERAL ELECTRIC COMPANY (GE)

TOPICAL REPORT NEDE-2401 1-P-A, "GESTAR II"

ITEM 2: IMPLEMENTING IMPROVED GE STEADY-STATE METHODS

INTRODUCTION

By letter dated October 22, 1999 (Reference 1), General Electric Company Nuclear Energy group (GENE), submitted a request for review of Amendment 26 to GESTAR as a standard amendment. GESTAR Amendment 26 had been submitted by letter dated August 13, 1999 (Reference 2), as an administrative change involving three areas. These areas were for (1) clarifying classification of BWR 6 pressure regulator failure downscale event; (2) implementing improved GE steady-state methods; and (3) incorporation of Boiling Water Reactor Owners Group (BWROG) approved stability options. This submittal was later supplemented by letters of September 17, 1999 (Reference 3) and October 14, 1999 (Reference 4).

BACKGROUND and DISCUSSION

From August 14, 1995, through September 1, 1995, an NRC inspection team conducted a performance-based evaluation of the GENE reload core design, safety analysis and licensing processes, core component and fuel assembly fabrication, and fuel related inspection services activities at the Nuclear Energy Production (NEP) facilities in Wilmington, North Carolina. Along with the strengths noted in Inspection Report No. 99900003/95-01 (Reference 5), the team observed weaknesses in certain activities that affect quality. The most significant concern was the accuracy of reload design methods used to ensure that the reactor can safely be operated. The team concluded that errors in calculated hot and cold core reactivity for recent reloads with newer fuel designs and longer cycle lengths were due to a weakness of the NEP reload design process. However, the team noted that NEP was addressing this weakness by implementing improvements in its currently approved steady-state nuclear methods (Reference 9). These improvements included a formal review process, increased bench marking of certain calculations for an extensive number of bundle design conditions, implementing revised computer codes, and developing a new lattice code. The team noted that the proposed design improvements should be thoroughly documented, peer reviewed, and monitored over a period of time to ensure that the new design methods meet the requirements placed on them. The team concluded that the near-term use of the revised computer codes in combination with the eigenvalue selection process should help reduce uncertainties in the cold critical and shutdown margins. The team also concluded that assurance of adequate shutdown margin can be strengthened by joint GENE/licensee actions consistent with the plant startup safety analysis. The team concluded, given that the introduction of the new nuclear models and codes

constitutes a major upgrade of GENE'S nuclear design methods, the use of the new models as design tools after the first quarter of 1996 was not an unreasonable schedule.

As part of the response to the inspection observations of weaknesses, as discussed above, GENE began to inform the staff of progress in the qualification and bench marking of the improved steady-state nuclear methods. In December 1995 (Reference 6), GENE provided an update to the NRC staff on the status of the improved methods. In July 1996 (Reference 7), GENE provided the background for the methods and a roadmap for implementation of the revised lattice physics (TGBLA) and three-dimensional (3D) simulator (PANACEA) codes. The qualification of these methods addressed the observed weakness in previous qualification databases by significantly expanding the scope of benchmarks versus both higher-order analytical standards and measured operating plant data. GENE also analyzed the effect of the model change-over on the plant dynamic response calculated for reload licensing analyses.

GENE outlined their implementation plan, following the completion of the code revisions, testing and release, characterized as completion of three areas followed by an implementation review, summarized as follows:

- address all technical issues related to implementation,
- modify all interface codes and related databases for compatibility,
- revise technical documentation design procedures and design bases for consistency, and
- complete application design review for all process areas.

In January 1998 (Reference 8), GENE informed the staff of the completion of a Level 2 design review of the improved methods, consistent with the GE quality assurance process for a qualified engineering computer program which has been loaded on the program library and is accessible to users. GE also stated that 27 additional plant cycles have been analyzed and the results were consistent with the benchmarking discussed in Reference 7. GE also provided additional results from test cases, using the improved methods to generate input for the approved stability code ODYSY, demonstrating that the differences in calculated decay ratios are within the uncertainty of the stability methodology.

3.0 EVALUATION

In August 1999 (Reference 2), GE submitted Amendment 26 to GESTAR II, and in October 1999 (Reference 1), requested NRC review as a standard amendment. The NRC staff has reviewed Item 2 of the submittal, in relation to the GE improved steady-state methods.

Since the 1995 inspection, GENE has updated the staff periodically on the status of the improved steady-state methods, as noted above. The staff has reviewed the findings of the

1995 vendor inspection (Reference 5), along with the description of the improved methods relative to the previously approved steady-state Licensing Topical Report (Reference 9) and

the benchmark and qualification data and the impact on transient analyses provided in the proprietary attachment to Reference 7. The staff also notes that a Level 2 design review through the GE Quality Assurance process has been completed for the improved methods and that continuing qualification checks have been performed (Reference 8). Based on the staff's review the inclusion of the improved GE steady-state methods into Amendment 26 of GESTAR II is acceptable.

4.0 CONCLUSIONS

The staff finds that Item 2, Implementing Improved GE Steady-State Methods, is acceptable and appropriate for inclusion in Amendment 26 to the GE Licensing Topical Report NEDE 24011-P-A (GESTAR II), as discussed in the above evaluation.

5.0 REFERENCES

1 Letter from G. A. Watford (GE) to USNRC, "Review of Amendment 26 to GESTAR II," October 22, 1999 (MFN-034-99).

2. Letter from G. A. Watford (GE) to USNRC, "Amendment 26 to GE Licensing Topical Report NEDE-2401 1-P-A (GESTAR II) for (1) Clarifying Classification of BWR 6 Pressure Regulator Failure Downscale Event, (2) Implementing Improved GE Steady-State Methods, and (3) Incorporation of BWROG Approved Stability Options," August 13, 1999 (MFN-008-99).

3. Letter from G. A. Watford to USNRC, "Additional Information Regarding Amendment 26 to NEDE-24011-P-A (GESTAR II)," September 17, 1999 (MFN-032-99).

4. Letter from G. A. Watford (GE) to USNRC, "Technology Update Meeting - Proposed Content," October 14, 1999 (MFN-033-99).

5. Letter from R. M. Gallo (NRC) to C. P. Kipp (GENE), "NRC Inspection Report No. 99900003/95-01," March 5, 1996.

6. Letter from R. J. Reda (GE) to R. C. Jones (NRC), "GE Fuel Technology Update," December 12, 1995.

7. Letter from R. J. Reda (GE) to R. C. Jones (NRC), "Implementation of Improved GE Steady-State Methods," July 2, 1996.

8. Letter from G. A. Watford (GE) to E. D. Kendrick (NRC), "Implementation of Improved GE Steady-State Nuclear Methods," January 8, 1998.

9. "Steady-State Nuclear Methods," NEDE-30130-P-A (proprietary) and NEDO-30130-A, April 1985.

Principal Contributor: E. Kendrick

Date: November 10, 1999

NRC
Safety Evaluation Report
Approving Amendment 26 –
Clarifying Classification of BWR–6
Pressure Regulator Failure Downscale Event

FLN-2000-003

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

March 29, 2000

Mr. Glen A. Watford, Manager
Nuclear Fuel Engineering
GE Nuclear Energy
P.0. Box 780
Wilmington, NC 28402

SUBJECT: AMENDMENT 26 TO GE NUCLEAR ENERGY LICENSING TOPICAL REPORT NEDE-24011-P-A (GESTAR II) - CLARIFYING CLASSIFICATION BWR-6 PRESSURE REGULATOR FAILURE DOWNSCALE EVENT (TAC NO. MA6481)

Dear Mr. Watford:

By letter dated October 22, 1999, General Electric (GE) Nuclear Energy, submitted a request for review of Amendment 26 to GESTAR as a standard amendment. GESTAR Amendment 26 was submitted by letter dated August 13, 1999, as an administrative change and involved three areas. These areas were for: 1) clarifying classification of BWR-6 pressure regulator failure downscale event, 2) implementing improved GE Nuclear Energy steady-state methods, and 3) incorporation of BWROG-approved stability options. The August 13, 1999, submittal was further supported by a meeting with the staff on March 17, 1998, and supplemented by a letter dated October 14, 1998. The staff has reviewed Item 1 of the submittal, in relation to clarifying the classification of the pressure regulator failure downscale event for the BWR-6, and has prepared the enclosed evaluation. We find that this area (Item 1) of the proposed Amendment 26 is acceptable as discussed in the enclosure. Item 2 was found acceptable in the staff's letter dated November 10, 1999, and Item 3 has been determined to be an administrative change and no further review is required. This completes the review of Amendment 26 to GESTAR and closes TAC No. MA6481.

Sincerely,

Stuart A. Richards, Director
Project Directorate IV and Decommissioning
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Enclosure: Safety Evaluation

Project No. 691

cc w/encl: See next page

GE Nuclear Energy Project No. 691

cc:

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION REGARDING AMENDMENT 26 TO GENERAL ELECTRIC NUCLEAR ENERGY TOPICAL REPORT NEDE-24011-P-A, ITEM 1: CLARIFYING CLASSIFICATION OF BWR-6 PRESSURE REGULATOR FAILURE DOWNSCALE EVENT

1.0 INTRODUCTION

By letter dated October 22, 1999 (Reference 1), the General Electric (GE) Nuclear Energy group submitted a request for review of Amendment 26 to GESTAR as a standard amendment. GESTAR Amendment 26 had been submitted by letter dated August 13, 1999 (Reference 2), as an administrative change involving three areas. These areas were for: 1) clarifying classification of BWR-6 pressure regulator failure downscale event, 2) implementing improved GE Nuclear Energy steady-state methods, and 3) incorporation of SWROG-approved stability options. The August 13, 1999, submittal was further supported by a meeting with the staff on March 17, 1998 (Reference 3), and supplemented by a letter dated October 14, 1998 (Reference 4).

2.0 BACKGROUND AND DISCUSSION

Two identical pressure regulators are provided to maintain primary system pressure control. They independently sense pressure just upstream of the main turbine stop valves and provide input to control the turbine control valves and turbine bypass valves to maintain a constant turbine inlet pressure. Failure of one pressure regulator is a mild event; the backup regulator will maintain control of the system pressure. Failure of both regulators is a more limiting event. A downscale failure of the pressure regulation demand to zero could cause full closure of the turbine control valves, as well as inhibit steam bypass flow and thereby increase reactor power and pressure. If this occurs, a reactor scram would be initiated when the high neutron flux scram setpoint is reached.

3.0 EVALUATION

Pressure Regulator Downscale Failure (BWR-6 plants)

The original BWR-6 licensing basis defined the pressure regulator downscale failure (PRDF) as an Anticipated Operational Occurrence (AOO) or moderate frequency event. The basis of this classification was the identification of a single initiating failure which lead to failure of both the primary and the backup regulators. This event did not affect any operating limits at rated conditions and at other than rated power and flow, protection was provided by the

peaking factor setdown imposed on the average power range monitor (APRM) high flux scram.

The operating flexibility option, *Maximum Extended Operating Domain (MEOD),* a commercial offering of General Electric (GE) for the BWR-6 plants to expand the power/flow map provided, among other improvements, the basis for removing the total peaking factor setdown of the flux scram and replaced it with a set of power and flow dependent operating limits. At this time, the classification of the PRDF as an AOO was also reevaluated and it was concluded that the expected frequency of occurrence of the single initiating failure was below the moderate frequency event definition, and was an infrequent event. As BWR-6 plants implemented the MEOD option, by NRC review and approval of a plant specific MEOD implementation report or as part of the Operating License application, the PRDF was considered to be outside the AOO range and, therefore, was no longer considered in establishing the operating limits which protect the fuel cladding during A00s. This was not explicitly documented in GESTAR II at this time. Following a thorough historical basis review, GE confirmed that the PRDF was not included in the MEOD evaluations because of the low probability of occurrence. GE also performed a formal analysis of the expected probability of occurrence for the BWR-6 plants with GE pressure control systems, as supported by IEEE 500-1984. This was discussed with the staff in a meeting on March 17, 1998, and GE provided documentation of their evaluation to the NRC as Reference 4.

Based on a review of the historical basis and the review and approval of the MEOD flexibility options for BWR-6 plants, the staff agrees that the PRDF is not an AOO. Therefore, the MEOD plants with GE pressure control systems are not required to evaluate the PRDF event, other than to assure that the pressure control of each plant is consistent with the approved MEOD basis. Thus, the staff agrees with the proposed wording for Amendment 26 for Item 1, "Clarifying Classification of BWR-6 Pressure Regulator Downscale Event."

4.0 CONCLUSIONS

We find that this area (item 1, Clarifying Classification of BWR-6 Pressure Regulator Downscale Event) is acceptable and appropriate for inclusion in Amendment 26 to the GE Licensing Topical Report NEDE-24011-P-A (GESTAR II), as discussed in the above evaluation,

5.0 REFERENCES

1. Letter from G. A. Watford (GE) to USNRC, “Review of Amendment 26 to GESTAR II,” October 22, 1999 (MFN-034-99).

2. Letter from G. A. Watford (GE) to USNRC, “Amendment 26 to GE Licensing Topical Report NEDE-24011-P-A (GESTAR II) for (1) Clarifying Classification of BWR 6 Pressure Regulator Failure Downscale Event, (2) Implementing Improved GE Steady-State Methods, and (3) Incorporation of BWROG Approved Stability Options,” August 13, 1999 (MFN-008-99).

3. Memorandum to T. H. Essig, Generic Issues and Environmental Projects Branch (NRR) from M. J. Davis, Project Manager, "Summary of Technical Information Exchange Meeting held on March 17, 1998, with GE Nuclear Energy," dated April 14, 1998.

4. Letter from G. A. Watford (GE) to J. H. Wilson (NRC), "Proposed Revision of GESTAR II to Clarify Classification of Pressure Regulator Downscale Failure Event," October 14, 1998 (MFN-039-98).

Principal Contributor: E. Kendrick

Date: March 29, 2000

NRC
Safety Evaluation Reports
Approving Amendment 27 –
to
NEDE-24011-P

July 16, 2004

Mrs. Margaret Harding, Manager
Nuclear Fuel Engineering
Global Nuclear Fuel
P. O. Box 780
Wilmington, NC 28402

SUBJECT: FINAL SAFETY EVALUATION FOR GLOBAL NUCLEAR FUEL (GNF) LICENSING TOPICAL REPORT NEDE-24011P, "GESTAR II, AMENDMENT 27" (TAC NO. MC0347)

Dear Mrs. Harding:

By letter dated August 6, 2003, as supplemented by letter dated March 15, 2004, Global Nuclear Fuel (GNF) submitted Amendment 27 to Licensing Topical Report (LTR) NEDE-24011P, "GESTAR II," to the staff for review. On April 6, 2004, an NRC draft safety evaluation (SE) regarding our approval of NEDE-24011P, was provided for your review and comments. By letter dated April 29, 2004, GNF stated they had no comments on the draft SE.

In a meeting with GNF on December 2, 2003, GNF identified those changes in Amendment 27 which it felt were administrative in nature. By letter dated March 15, 2004, GNF requested that the administrative changes in Amendment 27 be reviewed first to allow plants with Spring outages to take advantage of later revisions and newly approved codes. The remainder of the Amendment 27 changes will be approved consistent with the schedule provided in our October 1, 2003, acceptance letter. The staff has found that NEDE-24011P, Amendment 27, is acceptable for referencing in licensing applications for General Electric designed boiling water reactors to the extent specified and under the limitations delineated in the LTR and in the enclosed SE. The SE defines the basis for acceptance of the LTR.

Our acceptance applies only to material provided in the subject LTR. We do not intend to repeat our review of the acceptable material described in the LTR. When the LTR appears as a reference in license applications, our review will ensure that the material presented applies to the specific plant involved. License amendment requests that deviate from this LTR will be subject to a plant-specific review in accordance with applicable review standards.

In accordance with the guidance provided on the NRC website, we request that GNF publish accepted proprietary and non-proprietary versions of this LTR within three months of receipt of this letter. The accepted versions shall incorporate this letter and the enclosed SE between the title page and the abstract. It must be well indexed such that information is readily located. Also, it must contain historical review information, such as questions and accepted responses, draft SE comments, and original LTR pages that were replaced. The accepted version shall include a "-A" (designating accepted) following the LTR identification symbol.

If future changes to the NRC's regulatory requirements affect the acceptability of this LTR, GNF and/or licensees referencing it will be expected to revise the LTR appropriately, or justify its continued applicability for subsequent referencing.

Sincerely,

/RA by Robert A. Gramm for/

Herbert N. Berkow, Director
Project Directorate IV
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Project No. 712

Enclosure: Safety Evaluation

cc w/encl: See next page

Global Nuclear Fuel Project No. 712

cc:
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SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

AMENDMENT 27 TO TOPICAL REPORT NEDE-24011-P-A, "GESTAR II"

GLOBAL NUCLEAR FUEL

PROJECT NO. 712

1.0 INTRODUCTION

By letter dated August 6, 2003, Global Nuclear Fuel (GNF) submitted a request for review of Amendment 27 to GESTAR II as a standard amendment. The August 6, 2003, submittal was supported by a meeting with the staff on December 2, 2003, and supplemented by a letter dated March 15, 2004. The meeting on December 2, 2003, and the March 15, 2004, letter supplemented the letter dated August 6, 2003, to identify the administrative changes involved in Amendment 27 to GESTAR II. These changes included updated revision references, editorial corrections, and addition of new approved codes. This safety evaluation (SE) addresses the administrative changes noted in the March 15, 2004, letter.

2.0 BACKGROUND AND DISCUSSION

In a meeting with GNF on December 2, 2003, GNF identified those changes in Amendment 27 which it felt were administrative in nature. By letter dated March 15, 2004, GNF requested that the administrative changes in Amendment 27 be reviewed first to allow plants with Spring outages to take advantage of later revisions and newly approved codes. The remainder of the Amendment 27 changes will be approved consistent with the schedule provided in our October 1, 2003, acceptance letter.

3.0 EVALUATION

The staff has evaluated each of the proposed changes below. A table with a summary of the changes and descriptions is attached.

Update of Reference NEDE-31152-P

This change updates the various sections of GESTAR II to reflect the latest revision of NEDE-31152-P, "General Electric Fuel Bundle Design," to Revision 8 from Revision 7. The staff agrees that these changes are administrative changes and are therefore acceptable.

ODYSY Licensing Topical Report

This change updates the various sections of GESTAR II to reflect the NRC approval of NEDC-32992P-A, "ODYSY, Application for Stability Licensing Calculations," July 2001. Prior to approval of ODYSY the core and channel decay ratios for fuel designs were calculated using identical operating state conditions for power, flow, inlet subcooling, axial and radial core power shaped, and core pressure. In accordance with the ODYSY procedure, the axial and radial core power shapes will now correspond to the actual operating conditions at these steady state points. The staff has previously reviewed and approved the use of this licensing topical report (LTR). The addition of references to ODYSY are administrative changes and are therefore acceptable to the staff.

Correction of Maximum Linear Heat Generation Rate (MLHGR) to Linear Heat Generation Rate (LHGR)

This change corrects an error made in nomenclature. The correct term should have been LHGR limit and not MLHGR. The staff agrees that the correct term is LHGR limit. The change is an administrative change and therefore is acceptable to the staff.

Update LTRs to Correct Revision and Date Not Previously Corrected

This change updates reference sections 2.4, 3.6, and 4.4 to reflect the latest revisions used for various codes. These changes are for revisions that were made in previous amendments to GESTAR II, but the references were not updated. The staff agrees that these changes are administrative changes and therefore are acceptable to the staff.

Add Three References to Section 3.2.2.2, "Power Distribution Accuracy"

This change adds three references to Section 3.2.2.2. One of the references added currently exists and two new references were added. The two new references are NEDC-32601P-A, "Methodology and Uncertainties for Safety Limit MCPR Evaluation" and NEDC-32694P-A, "Power Distribution Uncertainties for Safety Limit MCPR Evaluations." These LTRs have been reviewed and approved by the staff. The staff agrees that adding these references is an administrative change and therefore is acceptable to the staff.

Delete Dopler/Void Coefficient Requirements from Section 3.2.3.1

This change removes the requirement for Doppler and void coefficients, which are used in the REDY code, to be provided in the Final Safety Analysis Report (FSAR) or in the Supplemental Reload Licensing Report (SRLR). The REDY code has been replaced by the NRC-approved ODYN code (and TRACG), in which these two coefficients are no longer directly inputted. Therefore, these values no longer need to be provided in the FSAR or the SRLR. Therefore, the removal of this requirement is administrative as these coefficients are not used. The staff agrees that the removal of this requirement is an administrative change and therefore is acceptable to the staff.

MLHGR Versus LHGR With PANAC11

In Table 3-1, "Definition of Fuel Design Limits," item (b) "all fuel design limits" has been revised to "all fuel rod thermal mechanical design limits" and "MLHGR" was revised to "exposure-dependent MLHGR." These changes were made to clarify these two options identified in Amendment 19 to GESTAR II. There has been no change to the content. The staff agrees that these changes clarify the existing options and does not change their intent and therefore are acceptable to the staff.

Adds TRACG, ODYN, and TASC References to Section 4.3.1.2

This change adds NEDE-32906P-A, Revision 1, "TRACG Application for Anticipated Operational Occurrences Transient Analysis," as an approved transient analysis method for application to anticipated operational occurrence (AOO) transients. TRACG for AOOs was previously reviewed and approved by the staff in NEDE-32906P-A, Revision 1 and has been incorporated into Amendment 27 of GESTAR II (Sections 4.3.1.2.1, 2, 5 and 6, 4.4, S2.2.2.1.1, 2, 3, 4, 6, and 7, S.5.1, S.5.1.5 and Appendix B [page 36]). As with any approved methodology, the use of TRACG is subject to the conditions and limitations outlined in its safety evaluation. It also adds the latest approval of NEDC-24154P-A, "Qualification of the One-Dimensional Core Transient Model (ODYN) for Boiling Water Reactors" (Supplement 1 - Volume 4), and NEDC-32084P-A, Revision 2, "TASC-03A-A Computer Program for Transient Analysis of a Single Channel." These LTRs have been previously reviewed and approved. The staff agrees that adding these references is an administrative change and therefore is acceptable to the staff.

Add ODYN to Section 4.3.1.2.2

This change allows ODYN to be used in AOO slow events. This is the same as the change described above for Section 4.3.1.2. This LTR has been previously reviewed and approved. The staff agrees that adding this reference is an administrative change and therefore is acceptable to the staff.

Add TASC-03A to Section 4.3.1.2.5

This change allows TASC to be used in AOO rapid events. This is the same as the change described above for Section 4.3.1.2. This topical report has been previously reviewed and approved. The staff agrees that adding this reference is an administrative change and therefore is acceptable to the staff.

Delete Tables S-6, S-7, and S-8

This change deletes three tables that are no longer being used. These tables were marked as not being used. Since these tables are not being used, the staff agrees that they can be deleted. The staff agrees that deleting these tables is an administrative change and therefore is acceptable to the staff.

Add References to SAFER/GESTR and Upper Bound PCT to Sections S2.2.3.2.4 and S.2.2.3.2.4.1

This change adds NEDC-32950P, "Compilation of Improvements to GENE's SAFER ECCS-LOCA Evaluation Model" and NEDC-23785P-A, Supplement 1, Revision 1, "GESTAR-LOCA and SAFER Models for Evaluation of Loss-of -Coolant Accident Volume III, Supplement 1, Additional Information for Upper Bound PCT Calculation," to Sections S2.2.3.2.4 and S2.2.3.2.4.1. These LTRs have been previously reviewed and approved. The staff agrees that adding these references is an administrative change and therefore is acceptable to the staff.

Add GE14 for Fuel Handling Accident, Section 2.2.3.5

This section requires that the fuel handling accident be analyzed for GE fuel. This analysis is performed using the methods described in Revision 10 to GESTAR. For GE14 fuel, this analysis is documented in Reference S-95 (NEDC-32868P, Revision 1, "GE14 Compliance with Amendment 22 of NEDE-24011-P-A (GESTAR II)"). This change adds this reference for the analysis of the fuel handling event for GE14 fuel. As the fuel handling analysis methodology has been previously approved, the staff finds this change acceptable. In addition, the title was corrected to NEDC-32868P from NEDE-32868P.

Table S-5, "Specific Plant Analysis" Update

These changes update Table S-5. This table is for information only and has been updated to reflect the latest information available. The staff agrees that these changes are administrative changes and therefore are acceptable to the staff.

Correct Error in Figure S-2, "Loss-of-Coolant Accident Evaluation Model (SAFER/GESTR Analysis Methods)"

This figure has been updated to reflect the use of CORECOOL code instead of CHASTE for the core heatup model. The use of CORECOOL was approved in NEDC-30996P-A. This LTR has been previously reviewed and approved. The staff agrees that changing this reference in Figure S-2 is an administrative change and therefore is acceptable to the staff.

Correct Typographical Errors

The word lattice was misspelled as lattic. The staff agrees that this change is editorial and therefore is acceptable to the staff.

4.0 CONCLUSIONS

Based on the above, the staff finds that proposed changes are administrative or editorial changes and therefore are acceptable and appropriate for inclusion in Amendment 27 to LTR NEDE 24011-P-A, "GESTAR II." As noted in Section 2.0, this SE only applies to the administrative changes in Amendment 27. The remainder of the changes are still under staff review.

Attachment: Table

Principal Contributor: A. Wang

Date: July 16, 2004

Amendment 27 Administrative Changes	Sections Affected by Change									
Description of Change	1.1.7, 1.2.7	1.2.8	1.4	2.2	2.4	3.2.2.2	3.2.3	3.2.4.3	3.6	T3-1
Update Ref NEDE-31151P			X		X				X	
ODYSY Licensing Topical Report		X	X							
Correct MLHGR to LHGR limit				X						X
Update LTRs to correct revision, date. Failed to correct in previous revisions to GESTAR.					X				X	
Add 2 references to dist. accuracy, from Amend 26						X			X	
Delete Doppler/void coeff							X			
MAPLHGR vs LHGR with P11										X
Correct equation terms										
TRACG										
ODYN										
TASC-03A										
Delete three tables not in use										
ECCS LOCA										
Add GE 14 for fuel handling accident										
Latest LOCA-ECCS reports are added or changed. For information.										
Correct errors in figure										
Correct typo										

Amendment 27 Administrative Changes	Sections Affected by Change							
Description of Change	4.1	4.2.3	4.2.4.1	4.2.4.2	4.2.4.3	4.2.4.4	4.3.1.2.1	4.3.1.2.2
Update Ref NEDE-31151P								
ODYSY Licensing Topical Report								
Correct MLHGR to LHGR limit	X							
Update LTRs to correct revision, date. Failed to correct in previous revisions to GESTAR.								
Add 2 references to dist. accuracy, from Amend 26								
Delete Doppler/void coeff								
MAPLHGR vs LHGR with P11								
Correct equation terms			X	X	X	X		
TRACG							X	X
ODYN								X
TASC-03A								
Delete three tables not in use								
ECCS LOCA								
Add GE 14 for fuel handling accident								
Latest LOCA-ECCS reports are added or changed. For information.								
Correct errors in figure								
Correct typo								

Amendment 27 Administrative Changes	Sections Affected by Change							
Description of Changes	4.3.1.2.5	4.3.1.2.6	4.3.1.2.7	4.3.1.2.8	4.4	TA. 15.0-2	A.16	Sup TofC
Update Ref NEDE-31151P					X		X	
ODYSY Licensing Topical Report								
Correct MLHGR to LHGR limit				X				
Update LTRs to correct revision, date. Failed to correct in previous revisions to GESTAR.					X			
Add 2 references to dist. accuracy, from Amend 26								
Delete Doppler/void coeff								
MAPLHGR vs LHGR with P11								
Correct equation terms								
TRACG	X	X			X			
ODYN					X			
TASC-03A	X				X			
Delete three tables not in use								X
ECCS LOCA								
Add GE 14 for fuel handling accident								
Latest LOCA-ECCS reports are added or changed. For information.								
Correct errors in figure								
Correct typo								

Amendment 27 Administrative Changes	Sections Affected by Change						
Description of Change	S.1.3	S.2.2.1.1	S.2.2.1.2	S.2.2.1.3	S.2.2.1.4	S.2.2.1.6	S.2.2.1.7
Update Ref NEDE-31151P							
ODYSY Licensing Topical Report							
Correct MLHGR to LHGR limit							
Update LTRs to correct revision, date. Failed to correct in previous revisions to GESTAR.							
Add 2 references to dist. accuracy, from Amend 26							
Delete Doppler/void coeff							
MAPLHGR vs LHGR with P11							
Correct equation terms							
TRACG		X	X	X	X	X	X
ODYN					X		
TASC-03A							
Delete three tables not in use							
ECCS LOCA							
Add GE 14 for fuel handling accident							
Latest LOCA-ECCS reports are added or changed. For information.							
Correct errors in figure							
Correct typo							

Amendment 27 Administrative Changes	Sections Affected by Change						
Description of Change	S.2.2.3.2.4	S.2.2.3.2.5	S.2.2.3.2.6	S.2.2.3.5	S.4	S.4.1.1	S.4.1.2
Update Ref NEDE-31151P							
ODYSY Licensing Topical Report						X	
Correct MLHGR to LHGR limit							
Update LTRs to correct revision, date. Failed to correct in previous revisions to GESTAR.							
Add 2 references to dist. accuracy, from Amend 26							
Delete Doppler/void coeff							
MAPLHGR vs LHGR with P11			X				
Correct equation terms							
TRACG							
ODYN							
TASC-03A	X						
Delete three tables not in use							
ECCS LOCA	X		X				
Add GE 14 for fuel handling accident				X			
Latest LOCA-ECCS reports are added or changed. For information.							
Correct errors in figure							
Correct typo							

Amendment 27 Administrative Changes	Sections Affected by Change							
Description of Change	S.4.1.3	S.4.2	S.5.1	S.5.1.1	S.5.1.5	S.5.2	S.5.2.2-S.5.212	S.6
Update Ref NEDE-31151P								x
ODYSY Licensing Topical Report	X							x
Correct MLHGR to LHGR limit								
Update LTRs to correct revision, date. Failed to correct in previous revisions to GESTAR.								x
Add 2 references to dist. accuracy, from Amend 26								
Delete Doppler/void coeff								
MAPLHGR vs LHGR with P11								
Correct equation terms								
TRACG			X		X			
ODYN								
TASC-03A								x
Delete three tables not in use								
ECCS LOCA								x
Add GE 14 for fuel handling accident								
Latest LOCA-ECCS reports are added or changed. For information.								
Correct errors in figure								
Correct typo								

Amendment 27 **Administrative Changes**	**Sections Affected by Change**				
Description of Change	T S-5	F S-2	App A, 8	App A, 9	App B p.36
Update Ref NEDE-31151P					
ODYSY Licensing Topical Report					
Correct MLHGR to LHGR limit					
Update LTRs to correct revision, date. Failed to correct in previous revisions to GESTAR.					
Add 2 references to dist. accuracy, from Amend 26					
Delete Doppler/void coeff					
MAPLHGR vs LHGR with P11					
Correct equation terms					
TRACG				X	
ODYN					
TASC-03A					
Delete three tables not in use					
ECCS LOCA					
Add GE 14 for fuel handling accident					
Latest LOCA-ECCS reports are added or changed. For information.	X				
Correct errors in figure		X			
Correct typo					X

March 17, 2005

Mrs. Margaret Harding, Manager
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Global Nuclear Fuel
P. O. Box 780
Wilmington, NC 28402

SUBJECT: FINAL SAFETY EVALUATION FOR GLOBAL NUCLEAR FUEL (GNF) AMENDMENT 27 TO LICENSING TOPICAL REPORT NEDE-24011-P-A, "GESTAR II" (TAC NO. MC0347)

Dear Mrs. Harding:

By letter dated August 6, 2003, as supplemented by letters dated March 15 and December 8, 2004, Global Nuclear Fuel (GNF) submitted Amendment 27 to Licensing Topical Report (LTR) NEDE-24011-P-A, "GESTAR II," to the staff for review. In its March 15 letter, GNF identified those changes in Amendment 27 which it felt were administrative in nature and requested that the administrative changes in Amendment 27 be reviewed first to allow plants with spring outages to take advantage of later revisions and newly approved codes. By letter dated July 16, 2004, the staff issued its final safety evaluation (SE) approving the requested administrative changes. This letter transmits the staff's final SE approving the remaining, more technical, changes requested by GNF in Amendment 27. A draft version of this SE (ML050110461) was shared with GNF for a proprietary information and factual error review. By e-mail dated February 3, 2005 (ML050600166), GNF replied that there were no issues with the technical content and that there was no proprietary information in the draft SE.

The staff has found that NEDE-24011-P-A, Amendment 27, is acceptable for referencing in licensing applications for General Electric designed boiling water reactors to the extent specified and under the limitations delineated in the LTR and in the enclosed SE. The SE defines the basis for acceptance of the LTR.

Our acceptance applies only to material provided in the subject LTR. We do not intend to repeat our review of the acceptable material described in the LTR. When the LTR appears as a reference in license applications, our review will ensure that the material presented applies to the specific plant involved. License amendment requests that deviate from this LTR will be subject to a plant-specific review in accordance with applicable review standards.

In accordance with the guidance provided on the NRC website, we request that GNF publish the accepted proprietary and non-proprietary versions of this LTR within three months of receipt of this letter. The accepted versions shall incorporate this letter and the enclosed SE after the title page. Also, they must contain historical review information, including NRC requests for additional information and responses. The accepted version shall include a "-A" (designating accepted) following the LTR identification symbol.

If future changes to the NRC's regulatory requirements affect the acceptability of this LTR, GNF and/or licensees referencing it will be expected to revise the LTR appropriately, or justify its continued applicability for subsequent referencing.

Sincerely,

/RA/
Herbert N. Berkow, Director
Project Directorate IV
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Project No. 712

Enclosure: SE

cc w/encl: See next page

Global Nuclear Fuel Project No. 712

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SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

AMENDMENT 27 TO TOPICAL REPORT NEDE-24011-P-A, "GESTAR II"

GLOBAL NUCLEAR FUEL

PROJECT NO. 712

1.0 INTRODUCTION

By letter dated August 6, 2003, as supplemented by letters dated March 15 and December 8, 2004, Global Nuclear Fuel (GNF) submitted Amendment 27 to Licensing Topical Report (LTR) NEDE-24011-P-A, "GESTAR II," to the staff for review. By letter dated March 15, 2004, GNF identified those changes in Amendment 27 which it felt were administrative in nature and requested that the administrative changes in Amendment 27 be reviewed first to allow plants with spring outages to take advantage of later revisions and newly approved codes. By letter dated July 16, 2004, the staff issued its final safety evaluation (SE) approving the requested administrative changes. This SE addresses the remaining, more technical, changes requested by GNF in Amendment 27.

A draft version of this SE (ML050110461) was shared with GNF for a proprietary information and factual error review. By e-mail dated February 3, 2005 (ML050600166), GNF replied that there were no issues with the technical content and that there was no proprietary information in the draft SE.

2.0 REGULATORY EVALUATION

The staff has evaluated each of the proposed changes below. The regulatory guidance used in this review is contained in NUREG-0800 Standard Review Plan (SRP) Section 4.2, "Fuel System Design," Draft Revision 3, April 1996 (available at the NRC external website). This guidance document provides acceptable methods for the review of fuel system designs and fuel analysis methodology adherence to applicable General Design Criteria (GDC).

3.0 TECHNICAL EVALUATION

Update to Sections 1.1.7.C.vii and 1.2.7.C.vii
This change corrects the critical power correlation uncertainty equation to a standard form. The staff agrees with the correction to this equation.

Update to Section 3.2.4.3
This change reworded the paragraph to state that the standby liquid control system (SLCS) is analyzed to bring the core to a shutdown condition with all rods out. This change is consistent with a previous response to an NRC question contained in Appendix B of the GESTAR II LTR, page US.B-43, and is therefore acceptable.

Update to Section 3.4.2.2
This change reworded the paragraph to clarify the evaluations to be performed on axial exposure distributions. In discussions with the staff, GNF clarified that the proposed changes to this section do not affect the time varying axial power shape (TVAPS) methodology. GNF stated that the TVAPS description does not apply to this change nor any other proposed change in Amendment 27. The change to Section 3.4.2.2 clarifies the currently approved methodology. The staff agrees that these changes are administrative and therefore acceptable.

Update to Section 3.4.3
This change requires that the SLCS shutdown margin be re-examined if the final loading pattern does not meet requirements. The staff agrees that the SLCS shutdown margin needs to be evaluated for this condition and therefore this change is acceptable.

Update to Section 4.2.3
This change allows the use of 1967 steam tables or later editions. The staff accepts the use of more recent international standard steam tables. Therefore, this change is acceptable.

Update to Section 4.2.4
This change clarifies the equations and variable definitions. This change does not alter the algorithm but only clarifies its usage. Therefore, this change is acceptable.

Update to Section 4.3.1.2.7
This change clarifies the text for easier understanding of off-rated limits. This clarification does not alter the underlying meaning of the text and is therefore acceptable.

Update to Table A.15.0-2
This change updates Table A.15.0-2 to reflect the final resolution on anticipated transients without scram (ATWS). The ATWS criteria for new fuel designs is contained in GESTAR Section 1.1.13 and 1.2.13. NRC review and approval of each new fuel bundle design must include an evaluation against these ATWS criteria. In response to the staff's request, GNF agreed to revise Table A.15.0-2 to cite GESTAR Section 1.1.13 and 1.2.13 for the ATWS acceptance criteria as opposed to NEDE-31152P. GNF provided the revised table in its letter dated December 8, 2004. This final change does not alter the current licensing basis for ATWS nor the requirement for future staff review of new fuel designs and is therefore acceptable.

Update to Section S.1.3
This change adds GDC 10 requirements and stipulates the actions that can be used on an interim basis if the long-term stability analysis solution is incapable of performing its intended safety function due to Part 21 issues or hardware failures. This change is consistent with currently approved methods and is therefore acceptable.

Update to Section S.2.2.3.2.6
The minimum critical power ratio (MCPR), maximum average planar linear heat generation rate (MAPLHGR), and linear heat generation rate (LHGR) represent the three thermal limits typically monitored by BWRs. Amendment 19 to GESTAR provided the capabilities to fold the LHGR into the MAPLHGR. Most plants discontinued specifying LHGR because it was included within MAPLHGR. This change updates the text to show that MAPLHGR need not include the

LHGR limit as the LHGR limit may be specified separately. This change does not alter the previously approved thermal margin monitoring requirements and is therefore acceptable.

Update to Section S.4
This change clarifies stability solutions to those currently approved and being applied in refueling analyses. This change is consistent with currently approved positions and is therefore acceptable.

Update to Section S.4.1.2
This change adds a reference to plant-specific Option II LTRs. Referencing plant-specific LTRs is acceptable. Therefore, this change is acceptable.

Update to Section S.4.2.1
This change adds text recommending that the impact of core design changes be evaluated with respect to stability interim corrective actions. This guidance helps to ensure the continued applicability of stability interim corrective actions and is therefore acceptable.

Update to Section S.5.1.1
This change updates the text to indicate the availability of recirculation pump trips (RPTs). This clarification does not change the RPT features nor whether it may be credited. Therefore, this change is acceptable.

Updates to Sections S.5.2, S.5.2.9, S.5.2.10, S.5.2.11, and S.5.2.12
This change adds operating flexibility options for safety/relief valves out of service, ADS valve out of service, end-of-cycle recirculation pump trip out of service, and main steam isolation valves out of service. These options have already been analyzed by GNF and have been implemented on a plant-specific basis for operating BWRs. In the future, licensees planning to implement any of these options will need to (1) ensure that implementation meets the provisions under 10 CFR 50.59 or (2) seek NRC approval prior to implementation. Since this change adds flexibility options already being implemented in operating plants and does not change the requirement for NRC review and approval of each option on a plant-specific basis, it is acceptable.

Update to Section S.5.2.2
At the staff's request, GNF has rescinded this change since the validation of the previous cycle loss-of-coolant accident (LOCA) analyses applies universally and not just to SAFE/REFLOOD LOCA methodology (Reference 3).

Update to Section S.5.2.3
This change corrects the extended load line limit (ELLLA) operating range. Specifically, the change removes the 100 percent power/75 percent flow point which is applicable to the maximum extended load line limit analysis (MELLLA), not ELLLA. This correction does not alter the currently approved application of ELLLA and is therefore acceptable.

Update to Section S.5.2.4
This change adds text clarifying the need to evaluate the effects of increased core flow on limiting transients. This change does not change the intent or magnitude of evaluations required to support increased core flow operation and is therefore acceptable.

Update to Section S.5.2.5
This change adds text clarifying the need to evaluate the effects of reduced feedwater temperature on limiting transients. This change does not change the intent or magnitude of evaluations required to support reduced feedwater temperature operation and is therefore acceptable.

Update to Section S.5.2.6
This change adds text clarifying the need to apply reduced operating limits for low power and low flow conditions. The use of MAPLHGR and LHGR limits is currently allowed and will be adjusted to provide protection during postulated transients at off-rated conditions. This clarification is acceptable.

Update to Section S.5.2.7
This change adds text clarifying the high load line aspects of maximum extended operation domain (MEOD) and the application of these limits to BWR 3/5 plants as MELLLA. This change does not alter the currently approved applications of MEOD or MELLLA and is therefore acceptable.

Update to Section S.5.2.8
This change adds text clarifying the response time requirements of the turbine bypass system. This clarification of response time does not change the intent of Section S.5.2.8 and is therefore acceptable.

Update to Section US.A.8
This change updates the operating flexibility options to be consistent with the changes noted above and is acceptable.

4.0 CONCLUSION

As discussed in the technical evaluation, the staff has reviewed the proposed changes and found each of them to be acceptable and appropriate for inclusion in Amendment 27 to NEDE-24011-P-A, "GESTAR II."

The staff finds Amendment 27 to be acceptable for referencing in license applications and does not intend to repeat our review of the matters described in Amendment 27 when the report appears as a reference in licensing applications, except to ensure that the material presented is applicable to the specific plant involved. Our acceptance applies only to the matters described in Amendment 27.

5.0 CONDITIONS AND LIMITATIONS

None.

6.0 REFERENCES

1. Letter from M. E. Harding (GNF) to U.S. Nuclear Regulatory Commission, “Review of Amendment 27 to GESTAR II, NEDE-24011-P-A," FLN-2003-009, August 6, 2003.

2. Letter from M. E. Harding (GNF) to U.S. Nuclear Regulatory Commission, "Review of Amendment 27 to GESTAR II, NEDE-24011-P-A, March 15, 2003.

3. Letter from M. E. Harding (GNF) to U.S. Nuclear Regulatory Commission, "Amendment 27 to GESTAR II, NEDE-24011-P-A," FLN-2004-034, December 8, 2004.

4. Letter from U.S. Nuclear Regulatory Commission to M. E. Harding (GNF), "Final Safety Evaluation for Global Nuclear Fuel (GNF) Licensing Topical Report NEDE-24011P, 'GESTAR II, Amendment 27'," July 16, 2004, ADAMS ML042010353.

Principal Contributor: P. Clifford

Date: March 17, 2005

NRC
Safety Evaluation Report
Approving Amendment 28 –
Misloaded Fuel Bundle Event
Licensing Basis Change
to
Comply with Standard Review Plan 15.4.7

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

March 8, 2007

Andrew A. Lingenfelter, Manager
GNF Engineering
Global Nuclear Fuels - Americas, LLC
P.O. Box 780, M/C F12
Wilmington, NC 28402

SUBJECT: FINAL SAFETY EVALUATION FOR GLOBAL NUCLEAR FUEL (GNF) TOPICAL REPORT (TR) AMENDMENT 28, "MISLOADED FUEL BUNDLE EVENT LICENSING BASIS CHANGE TO COMPLY WITH STANDARD REVIEW PLAN 15.4.7," TO GENERAL ELECTRIC STANDARD APPLICATION FOR RELOAD FUEL (GESTAR II) (TAC NO. MC3559)

Dear Mr. Lingenfelter:

By letter dated May 17, 2004, as supplemented by letters dated August 23, 2004, May 11 and June 2, 2006, GNF submitted Amendment 28, "Misloaded Fuel Bundle Event Licensing Basis Change to Comply with Standard Review Plan 15.4.7," to General Electric Standard Application for Reload Fuel (GESTAR II) to the U.S. Nuclear Regulatory Commission (NRC) staff. By letter dated December 6, 2006, an NRC draft safety evaluation (SE) regarding our approval of Amendment 28 to GESTAR II was provided for your review and comments. By letter dated January 30, 2007, GNF commented on the draft SE. The NRC staff's disposition of GNF's comments on the draft SE are discussed in the attachment to the final SE enclosed with this letter.

The NRC staff has found that Amendment 28 to GESTAR II is acceptable for referencing in licensing applications for GE-designed boiling water reactors to the extent specified and under the limitations delineated in the TR and in the enclosed final SE. The final SE defines the basis for acceptance of the TR.

Our acceptance applies only to material provided in the subject TR. We do not intend to repeat our review of the acceptable material described in the TR. When the TR appears as a reference in license applications, our review will ensure that the material presented applies to the specific plant involved. License amendment requests that deviate from this TR will be subject to a plant-specific review in accordance with applicable review standards.

In accordance with the guidance provided on the NRC website, we request that GNF publish accepted proprietary and non-proprietary versions of this TR within three months of receipt of this letter. The accepted versions shall incorporate this letter and the enclosed final SE after the title page. Also, they must contain historical review information, including NRC requests for additional information and your responses. The accepted versions shall include a "-A" (designating accepted) following the TR identification symbol.

If future changes to the NRC's regulatory requirements affect the acceptability of this TR, GNF and/or licensees referencing it will be expected to revise the TR appropriately, or justify its continued applicability for subsequent referencing.

Sincerely,

Ho K. Nieh, Deputy Director
Division of Policy and Rulemaking
Office of Nuclear Reactor Regulation

Project No. 712

Enclosure: Final SE

cc w/encl: See next page

Global Nuclear Fuel Project No. 712

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

FINAL SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

TOPICAL REPORT GESTAR II AMENDMENT 28

"MISLOADED FUEL BUNDLE EVENT LICENSING BASIS CHANGE TO COMPLY WITH STANDARD REVIEW PLAN 15.4.7"

GLOBAL NUCLEAR FUEL

PROJECT NO. 712

1.0 INTRODUCTION AND BACKGROUND

By letter dated May 17, 2004, as supplemented by letters dated August 23, 2004, May 11 and June 2, 2006 (References 1 through 4, respectively) Global Nuclear Fuel (GNF) submitted Amendment 28, "Misloaded Fuel Bundle Event Licensing Basis Change to Comply with Standard Review Plan (SRP) 15.4.7," to General Electric Standard Application for Reload Fuel (GESTAR II). This amendment proposes to make changes to GESTAR II to reclassify the misloaded fuel bundle event from "incident of moderate frequency" to "infrequent incident." Regulatory Guide (RG) 1.70, "Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants (LWR [Light Water Reactor] Edition)," Revision 3 (Reference 5), defines incidents of moderate frequency as incidents that may occur during a calendar year and infrequent incidents as events that may occur during the lifetime of a plant.

Historically, General Electric (GE) and GNF have considered two potential types of bundle loading errors: the misoriented bundle and the mislocated bundle. In the mislocated bundle event, GE assumed a more reactive higher power bundle can inadvertently be switched with a depleted, lower power bundle. Analyses showed that the consequences of the mislocated bundle were not expected to be severe, and normal plant operating limits provide sufficient protection to meet the licensing basis for this event. In the misoriented bundle event, GE assumed the bundle to be rotated 90 degrees or 180 degrees out of normal position. In the D-lattice reactors where the water gaps are non-uniform around the bundle, rotation can cause increases in local rod power through increased moderation. The consequences of mislocated or misoriented fuel loading errors (FLEs) are typically bounded by other events.

The proposal would change the way that the analysis is performed of the misloaded fuel bundle event, also termed as FLE, from that of an "incident of moderate frequency" category to that of an "infrequent incident" category. With this change, the event would be subject to NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants," (Reference 6) Section 15.4.7, "Inadvertent Loading and Operation of a Fuel Assembly in an Improper Position." The FLE would then be evaluated at less demanding radiological consequence dose acceptance limits (a small fraction of Part 100 of Title 10 of *Code of Federal Regulations* (10 CFR) limits rather than 10 CFR Part 20 limits).

2.0 REGULATORY EVALUATION

There is no specific guidance in SRP Section 15.4.7 as to acceptable methods for the radiological consequence analysis for the misloaded fuel bundle event other than specifying the acceptable dose limit as a small fraction of 10 CFR Part 100 limits. Also, this event is not addressed in RG 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors," (Reference 7) as a design-basis accident (DBA). Therefore, GNF proposed: (1) use of radiological consequence analysis guidance provided in SRP Section 15.4.9, Appendix A, "Radiological Consequences of Control Rod Drop Accident (BWR)" for the reactor plants whose DBAs are analyzed using the source term provided in Technical Information Document (TID)-14844, and (2) use of radiological consequence analysis guidance provided in RG 1.183, Appendix C, "Assumptions for Evaluating the Radiological Consequences of a BWR Rod Drop Accident," for the reactor plants whose DBAs are analyzed using the alternative source term (AST).

GNF evaluated the radiological consequences of the misloaded fuel bundle event against a small fraction of the 10 CFR Part 100 limits (30 rem to the thyroid and 2.5 rem to the whole body) for the TID-14844 source term, and a small fraction of the 10 CFR 50.67 limit (2.5 rem total effective dose equivalent (TEDE)) for the AST. These dose acceptance criteria are more restrictive than those specified in SRP Section 15.4.9 (75 rem to the thyroid and 6.3 rem to the whole body) for the TID-14844 source term or those specified in SRP Section 15.0.1, "Radiological Consequence Analysis Using Alternative Source Terms," (6.3 rem TEDE) for the AST.

The regulations in 10 CFR Part 100 specify how the exclusion area, low population zone (LPZ), and population center distance should be determined. Radiation criteria stipulated in 10 CFR Part 100 provide reference values to be used in the site suitability determination based on postulated fission product releases associated with accidents.

The regulations in 10 CFR Part 100 also specify the methodology for calculating radiation exposures at the site boundary for postulated accidents or events that might be caused by an FLE. For infrequent incidents, any releases of radioactive material must be such that the calculated doses at the site boundary are a small fraction of the 10 CFR Part 100 guidelines. As specified in SRP Section 15.4.7, a small fraction is interpreted to be less than 10 percent of 10 CFR Part 100 reference values. Meeting the requirements of 10 CFR Part 100 provides assurance that, in the event of an undetected FLE, radiation exposure at the site boundary will not exceed a small fraction of the reference values specified in 10 CFR Part 100.

Appendix A to 10 CFR Part 50, General Design Criterion (GDC) 13, "Instrumentation and control," states that "Instrumentation shall be provided to monitor variables and systems over their anticipated ranges for normal operation, for anticipated operational occurrences [(AOOs)], and for accident conditions as appropriate to assure adequate safety...." An FLE could adversely affect the fission process (power distribution), the integrity of the reactor core, and the reactor coolant pressure boundary. Meeting the requirements of GDC 13 provides assurance that an FLE will be detected before it can affect power distribution, core integrity, or could produce unacceptable stress on the reactor coolant pressure boundary.

SRP Section 15.4.7, gives the criteria found acceptable by the U.S. Nuclear Regulatory Commission (NRC) staff for meeting GDC 13 and 10 CFR Part 100 requirements. SRP Section 15.4.7 also provides the accident dose guidelines (a small fraction of 10 CFR Part 100 limits) for the exclusion area boundary (EAB) and LPZ. Appendix A to 10 CFR Part 50, GDC 19, "Control room," provides the control room dose assessment limits (5 rem TEDE).

3.0 TECHNICAL EVALUATION

The FLE event is the improper loading of a fuel bundle and subsequent operation of the core. Two types of FLEs are possible, the mislocation of a fuel assembly and the misorientation of a fuel assembly. GNF evaluated two scenarios for the misloaded fuel bundle event. The first scenario (Scenario 1) assumes the release of fission products from the core to the environment via the turbine and condensers following main steam isolation valve (MSIV) closure for those plants having a main steam line high radiation isolation trip capability. In the second scenario (Scenario 2), GNF assumed that no automatic MSIV closure occurred in that fission products were transported to an augmented offgas system for those plants having no main steam line high radiation isolation trip capability. Results show that offsite doses will not exceed 10 percent of the 10 CFR Part 100 limits which is the acceptance criteria for the FLE analyzed as an "infrequent incident."

3.1 RADIOLOGICAL CONSEQUENCES

GNF assumed that no fuel melt occurs as a result of this event and that this event will result in failure of the equivalent of five fuel assemblies (primary and four adjacent). GNF stated that the adverse consequences from an FLE would be the failure of one or more fuel rods in a single fuel assembly that is operating in a higher-than-normal power range. The incident would be similar to a fuel assembly operating with one or more leaking fuel rods. However, the radiological consequences would be difficult to assess for each fuel bundle in the core for each operating cycle. Therefore, in order to bound the consequences for this event, GNF conservatively assumed that all of the fuel rods in five failed fuel assemblies will experience instantaneous failure.

GNF used a conservative fuel bundle radial peaking factor of 2.5 instead of a radial peaking factor of 1.5 as specified in SRP Section 15.4.9, Appendix A to ensure that the peak bundle power to bundle average cycle power ratio was bounded. In addition, GNF used a safety factor of 1.4 to address the variation in fission product inventory over the cycle of the operating fuel.

3.1.1 Scenario 1

This scenario assumes the release of fission products from all of the fuel rods in five failed fuel assemblies in the reactor core. The release to the environment is modeled as a ground level release via the turbine and condensers following MSIV closure.

Consistent with the guidelines provided in SRP Section 15.4.9, Appendix A, and RG 1.183, Appendix C, GNF assumed that:

- 10 percent of the core inventory of noble gases and iodine and 20 percent of alkali metals (instead of 12 percent for alkali metals as specified in RG 1.183) were released to the coolant,

- 100 percent of noble gases, 10 percent of iodine, and 1 percent of the remaining nuclides released from the failed fuel assembly to the coolant reach the turbine and condensers before MSIV closure,

- of those fission products which reach the turbine and condensers, 100 percent of noble gases, 10 percent of iodine, and 1 percent of the remaining nuclides are available for release to the environment from the turbine and condensers, and

- the turbine and condensers leak to the environment at a rate of 1 percent per day as a ground-level release for a period of 24 hours, at which time the leakage is assumed to terminate.

GNF proposed no deviation or departure from the guidelines provided in SRP Section 15.4.9 or RG 1.183. However, as noted above GNF uses a more conservative alkali metal release percentage, than that specified in RG 1.183.

GNF back-calculated the following bounding atmospheric dispersion factors (χ/Q values) from the radiological consequence dose criteria (a small fraction of 10 CFR 50.67 for the AST and a small fraction of 10 CFR Part 100 for the TID-14844 source term) assuming these χ/Q values represent the limiting χ/Q values for a ground level release from the condensers to the EAB and LPZ.

Source Terms	Dose Criteria	EAB/LPZ χ/Q Value (s/m^3)
TID	30 rem thyroid	1.67E-3
AST	2.5 rem TEDE	5.04E-3

GNF labeled these χ/Q values as 2-hour χ/Q values but applied them for the entire 24-hour release.

The NRC staff performed an independent confirmatory dose calculation to verify GNF's results. A χ/Q value at the EAB and LPZ less than 1.67E-3 s/m^3 will result in a thyroid dose at or below the 30 rem limit for the TID-14844 source term and a χ/Q value at the EAB and LPZ less than 5.04E-3 s/m^3 will result in a TEDE at or below the 2.5 rem limit for the AST. The relationships between calculated doses and χ/Q values for the TID-14844 source term and AST are shown in Figures B-1 and B-4, respectively, in Attachment B, "Fuel Loading Error Event Radiological Analysis for Offsite and Control Room Dose," of Reference 4.

GNF assumed that the control room is not isolated during this event and neither the emergency filtration system nor control room air recirculation system is assumed to be operational. GNF selected the following ranges of two control room variables:

Control room volumes:	1.0E+3 to 1.0E+6 ft^3
Control room air flow rates:	1.0E+2 to 1.0E+5 cfm

GNF back-calculated the following bounding control room χ/Q values that result in meeting the respective radiological consequence dose acceptance criteria:

Source Terms	Dose Criteria	Control Room χ/Q Value (s/m^3)
TID	30 rem thyroid	1.81E-3
AST	5 rem TEDE	1.25E-2

The highest radiological doses occurred with the highest control room air flow rate due to the inhalation dose and largest control room volume due to the gamma immersion dose. Control room doses as a function of the control room χ/Q values are shown in Figures B-7 and B-8 of Amendment 28 for the TID-14844 source term and AST, respectively.

3.1.2 Scenario 2

This scenario assumes the release of fission products from all fuel rods in five failed fuel assemblies in the reactor core to the environment via the plant stack as an elevated release through the offgas system. In this scenario, it was assumed that the MSIVs did not close immediately after initiation of the event and that steam flow continued for a period of time. The main steam line radiation monitor (MSLRM) and the steam jet air ejector radiation monitor would alarm almost immediately. These monitors are required by the BWR technical specifications and will activate an alarm in the main control room.

There is no specific guidance in SRP Section 15.4.7, SRP Section 15.4.9, or RG 1.183 regarding acceptable methods for the radiological consequence analysis for this scenario. However, in May 1991, the NRC staff reviewed and accepted the methodology proposed in the BWR Owners' Group Licensing Topical Report, NEDO-31400, "Safety Evaluation for Eliminating the Boiling Water Reactor Main Steam Line Isolation Valve Closure Function and Scram Function of Main Steam Line Radiation Monitor." In its safety evaluation (SE) (Reference 9), the NRC staff concluded that the removal of the MSLRM trips, that automatically shut down the reactor and close the MSIVs, is acceptable. The NRC staff further concluded that the removal of the automatic reactor shutdown and MSIV closure trips from the MSLRM does not change the radiological consequences evaluated in the Final Safety Analysis Reports for meeting the dose acceptance criteria specified in SRP Section 15.4.7.

The augmented offgas system designed and supplied by GE and currently in use at operating BWRs typically contains, catalytic recombiners, a series of charcoal adsorber delay tanks, and high efficiency particulate air filters to achieve adequate decay of noble gases and removal of iodine prior to release to the environment from the plant stack. The system is designed to process non-condensible and volatile fission products received from the condenser evacuation system to meet 10 CFR Part 20 and Appendix I to 10 CFR Part 50 limits prior to release from the plant stack. The system typically provides minimum decay times of 46 hours for krypton and 42 days for xenon for the offgas received from the main condenser evacuation system. The delay time in the charcoal adsorber delay tanks is proportional to the mass of charcoal and to the dynamic adsorption coefficients for the gas. These are, in turn, functions of the operational temperature and humidity conditions in the charcoal. GNF stated, and the NRC staff agrees, that any iodine releases from the offgas system are negligible, because the iodine is retained in the charcoal adsorber delay tanks for decay.

The offgas system effluent is continuously monitored by the offgas system post-treatment radiation monitor and by the stack effluent monitor. The monitor trip outputs are used to initiate closure of the offgas system discharge, and the trip setpoint is set so that valve closure is initiated prior to exceeding the offsite and control room doses. These monitors are also equipped with a trip circuit that actuates corresponding main control room annunciators. Therefore, the NRC staff finds that the radiological consequence resulting from Scenario 2 is bounded by that of Scenario 1.

3.2 FUEL LOADING

In the GNF responses to the NRC staff requests for additional information (RAIs), GNF utilized the error rate for the past 25 years, in particular, the zero error rate for operation with an FLE over the past 10 years to demonstrate the effectiveness of core verification procedures. Since 1995 there have not been any cases of a plant operating with a misoriented or mislocated fuel bundle. Although no hardware, software, or mechanical interlocks, etc. are in place, each operating BWR has its own core verification procedures that follow the recommendations of Service Information Letter (SIL)-347 (Reference 10). Details of SIL-347 are provided below. The recommendations outlined in SIL-347 contributed to the prevention of operation with an FLE in the past 10 years.

GNF stated that the extensive period of refueling history as reflected in its responses makes a probabilistic risk assessment (PRA) model of limited value. GNF also states, "...There is no particular information provided by a model that is not reflected in the actual refueling data for the past 25 years." In its submittal, GNF provided a table summarizing FLEs that occurred between 1995 and February 2005, although no plants have operated with an FLE. From this table, GNF calculated the probability of an FLE as 0.19 FLE per plant per lifetime. This number is less than the value used in defining infrequent incidents (1 FLE per plant per lifetime).

3.2.1 SIL-347 Background

During the 1980's, four plants had reported operation with misoriented fuel bundles. GE issued a SIL-347 highlighting the importance of preventing misoriented bundles and provided recommended guidelines for developing procedures for core loading verification to help eliminate their recurrence. The action of refueling a LWR, be it a BWR or PWR, requires the movement of fuel assemblies from one location to another within the core, and the retrieving and loading of new and burned fuel assemblies from the spent fuel pool. Each movement of the assemblies, location and orientation, is monitored, observed, and checked at the time of completion by the fuel movement operator and spotters.

Since 1978, the FLE has been analyzed as an AOO and, as such, the change in critical power ratio (CPR) for the event has been factored into the determination of the minimum CPR (MCPR) operating limit for each cycle. Section 6.3 of the NRC SE for NEDE-24011P-US, Revision 0, "Generic Reload Fuel Application," dated May 12, 1978 (Reference 11), describes the basis for this treatment of the FLE, which includes fuel loading experience in that time period.

In response to SIL-347, utilities began in 1981 to improve the procedures used for core verification following refueling. The typical procedure of core verification at the plants which

experienced misoriented bundles was to scan the core with an underwater television camera at a distance close enough to distinguish the bundle serial numbers on top of the lifting bails. In the scan, and in a subsequent verification of the videotapes, one person was responsible for reading the serial number and verifying orientation. The conclusion reached by GE after studying the affected sites was that the close-up picture needed for serial number verification did not permit easy recognition of proper bundle orientation, and the verifier's attention was primarily focused on the difficult task of reading the serial numbers. As a result of its findings, GE proposed all BWR owner/operators ensure that their reload procedures provide for a separate scan of the core during final core verification to verify bundle orientation. GE also recommended guidelines for developing a procedure for bundle orientation verification. This was provided as Attachment A to the SIL-347.

The FLE rate for the recent 25-year period and the trend for the most recent 10 years of refueling outages support the classification of the FLE event as an "infrequent incident." Section S.2.1, supplemental to GESTAR, provides the basis for categorizing and analyzing the FLE as an infrequent incident and the associated analysis limits. Upon approval of the proposed Amendment 28, licensees may choose to analyze the FLE as an infrequent incident, or to continue analyzing the event as an AOO. In order to apply the infrequent incident option, several items must be confirmed and documented with the reload design procedures. The first group of these involves the core verification procedures applied following refueling, and the second involves the input parameters and plant offgas system bases used to perform the generic radiological analysis. The requirements apply for licensees with either 10 CFR Part 100 or 10 CFR 50.67 radiological licensing bases.

3.2.2 Core Verification

To select the infrequent incident option, the licensee's core verification procedures must be consistent with those generally used during the recent historical period forming the basis for the Amendment 28 analysis of the event frequency. Therefore, the licensee must certify that its core verification procedures have the following characteristics:

1. During fuel movement, each move (location, orientation, and seating) is observed and checked at the time of completion by the operator and a spotter.

2. After completion of the core load, the core is verified by a video recording of the core using an underwater camera. The recording may involve two or more records made at different ranges to provide clear resolution of the bundle serial number, and to illustrate the orientation in four bundle clusters. The core verification may take place during the recording process, by viewing after recording, or a combination.

3. Two independent reviewers perform the verification of the bundle serial number location, orientation, and seating. Each independent team records the bundle serial numbers on a core map, which is verified with the design core-loading pattern.

3.2.3 Staff Requirements

In RAI response 1(b) of Reference 2, GNF stated that each operating BWR licensee has its own procedure for core verification following fuel loading and core component movements prior

to startup. The procedures follow the recommendations of SIL-347. However, GNF also pointed out that while the emphasis of SIL-347 was on the misoriented bundle, the utilities generalized its procedures to include the recommendations provided in SIL-347, namely the requirement of "at least 2 independent reviewers of core assemble configuration" and applied them to each of the three core verification elements: bundle location, bundle orientation, and bundle seating.

Therefore, the NRC staff concludes that the recommendations of SIL-347 as expanded by the BWR licensees has reduced the likelihood of a FLE. The NRC staff finds that there is enough information present to conclude that the FLE can be reclassified as an infrequent incident on a plant-specific basis. Because the approval requires certain plant-specific verifications, the documentation must be reconfirmed every refueling outage.

The NRC staff conclusion is based on information provided by GNF which supports the classification of the FLE as an "infrequent incident," based on the FLE error rate for the period since 1980 and plant data from refueling outages since SIL-347 recommendations have been implemented. Although there are no hardware, software, mechanical interlocks, etc. that prevent an FLE from occurring, operating BWRs have procedures for core verification following fuel and core component movements prior to startup, which follow the recommendations of SIL-347. Since 1995, the use of these procedures has prevented core operation with a mislocated or misoriented fuel bundle.

4.0 LIMITATIONS AND CONDITIONS

NRC staff requires that users of GESTAR II, Amendment 28, generate EAB and LPZ χ/Q values in a manner consistent with RG 1.145, "Atmospheric Dispersion Models for Potential Accident Consequence Assessments at Nuclear Power Plants," (Reference 12). All comparisons with χ/Q values in GESTAR II, Amendment 28, should use the limiting 5 or 0.5 percentile plant-specific 2-hour EAB and 8-hour LPZ χ/Q values unless the user provides a plant-specific analysis for NRC review that justifies use of other χ/Q values. Users of GESTAR II, Amendment 28, should generate control room χ/Q values in a manner consistent with RG 1.194, "Atmospheric Relative Concentrations for Control Room Radiological Habitability Assessments at Nuclear Power Plants," (Reference 13). All comparisons with χ/Q values in GESTAR II, Amendment 28, should use the limiting 5 percentile plant-specific 2-hour control room χ/Q value unless the user provides a plant-specific analysis for NRC review that justifies use of other χ/Q values. Thus, hold-up of effluent in a tank with delayed release to the environment as postulated in Scenario 2 does not justify use of a lower plant-specific χ/Q value representative of a later time interval (e.g., one day hold-up does not justify comparison of the GESTAR II, Amendment 28, χ/Q value with a plant-specific 1 to 4-day χ/Q value) without further review by the NRC staff.

The FLE can now be analyzed as an infrequent incident provided that the licensee confirms the requirements for application of the generic analysis in Amendment 28. Licensees seeking to apply the infrequent incident basis must confirm that their core refueling verification procedures meet the requirements defined in Section 5.3, "Fuel Loading Error Analysis Requirements," of the GESTAR US Supplement and Section 3.2.2 of this SE. This confirmation will be documented every refueling outage through the plant-specific reload design documentation and the analysis basis stated in the Supplemental Reload Licensing Report (SRLR). Additionally,

the input parameters and plant offgas system bases used to perform the generic radiological analysis must be confirmed and documented with the reload design procedures.

Should a bundle mislocation, misorientation, and seating occur and go undetected, the plant-specific acceptance of this submittal will be revoked, and the classification of this event will revert from "infrequent incident" classification back to an "incident of moderate frequency" classification immediately for that plant. The classification of the event back to an "incident of moderate frequency" for that plant is permanent, unless NRC approves a plant-specific amendment request at a future date. This TR approval may not be used as the justification for the plant-specific amendment request.

5.0 CONCLUSION

The NRC staff has reviewed GESTAR II, Amendment 28, to assess the acceptability of the justifications therein for changing the way that the analysis of an FLE is performed from that of an "incident of moderate frequency" category to that of an "infrequent incident" category. The NRC staff finds GESTAR II, Amendment 28 acceptable for referencing given the limitations and conditions of Section 4.0 of this SE.

The NRC staff finds that GNF has provided an acceptable method for determining the radiological consequences resulting from a misloaded fuel bundle event. The radiological consequence of the two scenarios, as discussed in Section 3.1 of this SE, would meet the dose acceptance criteria provided in SRP Section 15.4.7, RG 1.183, Appendix C, and SRP Section 15.4.9 when using bounding χ/Q values at the EAB, LPZ, and control room. Therefore, the NRC staff concludes that the changes requested to reclassify the misloaded fuel bundle event as an "infrequent incident" from an "incident of moderate frequency" are acceptable with respect to the radiological consequences resulting from a misloaded fuel bundle event.

Additionally, the NRC staff concludes that GNF has provided a sufficient basis for approval of the reclassification, because the necessary action to prevent such events are plant-specific. Therefore, the NRC staff finds that there is sufficient basis to support a reclassification of the FLE on a plant-specific basis as described above. The NRC staff approval applies only to licensees implementing GESTAR.

6.0 REFERENCES

1. Letter FLN-2004-009 from M. E. Harding (GNF) to USNRC, "GESTAR II Amendment 28, Misloaded Fuel Bundle Event Licensing Basis Change to Comply with Standard Review Plan 15.4.7," May 17, 2004 (ADAMS Accession No. ML062860291).

2. Letter FLN-2004-026 from M. E. Harding (GNF) to USNRC, "GESTAR II Amendment 28, Revision 1, Misloaded Fuel Bundle Event Licensing Bases Change to Comply with Standard Review Plan 15.4.7," August 23, 2004 (ADAMS Accession No. ML062860295).

3. Letter FLN-2006-018, from A. A. Lingenfelter (GNF) to USNRC, "Response to NRC Request for Information Regarding Amendment 28 to GESTAR II (TAC NO. MC3559)," May 11, 2006 (ADAMS Accession No. ML061350416).

4. Letter FLN-2006-020 from A. A. Lingenfelter (GNF) to USNRC, "Transmittal of Updated Attachments Supporting GESTAR II Amendment 28 and Associated GESTAR II Sections (TAC NO. MC3559)," June 2, 2006 (ADAMS Accession No. ML061580106).

5. Regulatory Guide 1.70, "Standard Format and Contents of Safety Analysis Reports for Nuclear Power Plants (LWR Edition)," Revision 3, November 1978 (ADAMS Accession No. ML011340122).

6. NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants."

7. Regulatory Guide 1.183, "Inservice Inspection of Pressurized Water Reactor Steam Generator Tubes, Revision 1, July 1975 (ADAMS Accession No. ML003740256).

8. Technical Information Document-14844, "Calculation of Distance Factors for Power and Test Reactor Sites," March 23, 1962 (ADAMS Accession No. ML051770099).

9. Letter from A. C. Thadani (USNRC) to G. J. Beck (BWROG), NRC Safety Evaluation RE: Licensing Topical Report, NEDO-31400, "Safety Evaluation for Eliminating the Boiling Water Reactor Main Steam Line Isolation Valve Closure Function and Scram Function of Main Steam Line Radiation Monitor," May 15, 1991 (ADAMS Legacy Library Accession No. 9105230048).

10. GE Nuclear Services Information Letter No. 347, "Misoriented Fuel Bundles," December 1980 (ADAMS Accession No. ML062860435).

11. Letter from D. G. Eisenhut (USNRC) to R. L. Gridley (GE), NRC Safety Evaluation RE: Generic Reload Fuel Application, (GESTAR II, Revision 0) May 12, 1978 (ADAMS Accession No. ML062860426).

12. Regulatory Guide 1.145, "Atmospheric Dispersion Models for Potential Accident Consequence Assessments at Nuclear Power Plants," Revision 1 (ADAMS Accession No. ML003740205).

13. Regulatory Guide 1.194, "Atmospheric Relative Concentrations for Control Room Radiological Habitability Assessments at Nuclear Power Plants" (ADAMS Accession No. ML031530505)

Attachment: Resolution of Comments

Principle Contributors: A. Attard
J. Lee

Date: March 8, 2007

RESOLUTION OF COMMENTS ON DRAFT SAFETY EVALUATION FOR GLOBAL NUCLEAR FUELS (GNF) TOPICAL REPORT (TR) AMENDMENT 28, "MISLOADED FUEL BUNDLE EVENT LICENSING BASIS CHANGE TO COMPLY WITH STANDARD REVIEW PLAN 15.4.7," TO GENERAL ELECTRIC STANDARD APPLICATION FOR RELOAD FUEL (GESTAR II)

By letter dated January 30, 2007 (Agencywide Document Access and Management System Accession No. ML070360512), GNF provided comments on the draft safety evaluation (SE) for Amendment 28 to GESTAR II. The following is the NRC staff resolution of those comments.

1. GNF Comment: Page 3, Section 3.0, 1st Paragraph, Last Sentence is not correct. The dose limit is the acceptance criteria for the event, not the basis for the categorization.

   Resolution: Re-worded.

2. GNF Comment: Page 5, Section 3.1.1, Last Paragraph, Last Sentence, the Figure numbers should be B-7 and B-8.

   Resolution: Comment incorporated.

3. GNF Comment: Page 6, Section 3.2.1, 2nd Paragraph, Last Sentence, the proper SE to be referenced is the SE for Revision 0 of GESTAR.

   Resolution: Comment incorporated.

4. GNF Comment: Page 9, Section 4.0, Last Paragraph, the duration of the revocation of the infrequent event categorization and the steps that a licensee needs to take to re-apply the infrequent event classification should be clarified in the limitation.

   Resolution: Limitation re-worded.

5. GNF Comment: Page 9, Section 5.0, 3rd Paragraph, Last Sentence, the survey, which is the basis for the event statistics, includes responses from both GE and mixed fuel vendor plants. Also, the core verification procedures are not vendor specific. Therefore, the restriction to plants containing only GE fuel is not appropriate and the last sentence should be deleted.

   Resolution: Comment incorporated.

6. GNF Comment: Page 9, Section 6.0, References 3 and 4, "AA White" should be "AA Lingenfelter."

   Resolution: Comment incorporated.

7. GNF Comment: Page 10, Section 6.0, Reference 7, should be Regulatory Guide 1.183, “Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Plants,” July 2000.

   Resolution: Comment incorporated.

8. GNF Comment: Page 10, Section 6.0, Reference 11, the text "NEDE-24011-A-15-US" does not belong in this reference.

   Resolution: Text deleted.

NRC
Safety Evaluation Report
Approving Amendment 29 –
to
NEDE-24011-P

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

RECEIVED
OCT - 2 2006
GENERAL ELECTRIC COMPANY

September 28, 2006

Andrew A. Lingenfelter, Manager
GNF Engineering
Global Nuclear Fuels - Americas, LLC
P.O. Box 780, M/C F12
Wilmington, NC 28402

SUBJECT: FINAL SAFETY EVALUATION FOR GLOBAL NUCLEAR FUEL (GNF) AMENDMENT 29 TO TOPICAL REPORT NEDE-24011P-A/NEDO-24011-A, "GENERAL ELECTRIC STANDARD APPLICATION FOR RELOAD FUEL (GESTAR II)" (TAC NO. MD1497)

Dear Mr. Lingenfelter:

By letter dated April 25, 2006, GNF submitted Amendment 29 to the Topical Report (TR) NEDE-24011P-A/NEDO-24011-A, entitled "General Electric Standard Application for Reload Fuel (GESTAR II)," to the U.S. Nuclear Regulatory Commission (NRC) staff for review.

The NRC staff has found that Amendment 29 to GESTAR II is acceptable for referencing in licensing applications for General Electric designed boiling water reactors to the extent specified and under the limitations delineated in the TR and in the enclosed final SE. The final SE defines the basis for acceptance of the TR.

Our acceptance applies only to material provided in the subject TR. We do not intend to repeat our review of the acceptable material described in the TR. When the TR appears as a reference in license applications, our review will ensure that the material presented applies to the specific plant involved. License amendment requests that deviate from this TR will be subject to a plant-specific review in accordance with applicable review standards.

In accordance with the guidance provided on the NRC website, we request that GNF publish accepted proprietary and non-proprietary versions of this TR within three months of receipt of this letter. The accepted versions shall incorporate this letter and the enclosed final SE after the title page. Also, they must contain historical review information, including NRC requests for additional information and your responses. The accepted versions shall include a "-A" (designating accepted) following the TR identification symbol.

If future changes to the NRC's regulatory requirements affect the acceptability of this TR, GNF and/or licensees referencing it will be expected to revise the TR appropriately, or justify its continued applicability for subsequent referencing.

Sincerely,

John W. Lubinski, Deputy Director
Division of Policy and Rulemaking
Office of Nuclear Reactor Regulation

Project No. 712

Enclosure: Final SE

cc w/encl: See next page

Global Nuclear Fuel

Project No. 712

cc:

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12/21/05

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

AMENDMENT 29 TO TOPICAL REPORT NEDE-24011-P-A/NEDO-24011-A

"GENERAL ELECTRIC STANDARD APPLICATION FOR RELOAD FUEL (GESTAR II),"

GLOBAL NUCLEAR FUEL

PROJECT NO. 712

1.0 INTRODUCTION

By letter dated April 25, 2006, GNF submitted Amendment 29 to the Topical Report (TR) NEDE-24011P-A/NEDO-24011-A, entitled "General Electric Standard Application for Reload Fuel (GESTAR II)," to the U.S. Nuclear Regulatory Commission (NRC) staff for review (Agencywide Documents Access and Management System Accession No. ML0611800250). The changes in Amendment 29 are administrative in nature and include updated revision references and addition of new approved codes.

2.0 EVALUATION

The NRC staff has evaluated each of the proposed changes below.

Add Reference to Creep Collapse methods to Sections 2.2.2.2.3 and 2.4

This change revises Section 2.2.2.2.3 to reflect approval of NEDC-33139P-A, "Cladding Creep Collapse Licensing Topical Report" for fuel designs newer than GE14 and adds the TR as Reference 2-18 in Section 2.4 of GESTAR II. This TR has been previously reviewed and approved. The NRC staff agrees that adding this reference is an administrative change and, therefore, is acceptable.

Add Reference to Supplement 2-A of NEDE-32906P-A to Section 4.4

This change modifies Reference 4-40 to include the approved Supplement 2-A to NEDE-32906P-A, "TRACG Application for Anticipated Operational Occurrences Transient Analyses," in Section 4.4 of GESTAR II. This TR has been previously reviewed and approved. The NRC staff agrees that adding this reference is an administrative change and, therefore, is acceptable.

Add Reference to Improved Bank Position Withdraw Sequences (BPWS) to Sections S.2.2.3.1.1 and S.6

This change revises Section S.2.2.3.1.1 to reflect approval of NEDO-33091-A, "Improved BPWS Control Rod Insertion Process" as an option for plants that have implemented the BPWS in accordance with Reference S-11 of GESTAR II and adds the TR as Reference S-99 in Section S.6 of the GESTAR II. This TR has been previously reviewed and approved. The NRC staff agrees that adding this reference is an administrative change and, therefore, is acceptable.

3.0 CONCLUSIONS

Based on the above, the NRC staff finds that proposed changes are administrative and therefore, are acceptable and appropriate for inclusion in Amendment 29 to TR NEDE 24011-P-A, "GESTAR II."

Principal Contributor: S. Wu
M. Honcharik

Date: September 28, 2006

NRC
Safety Evaluation Report
Approving Amendment 30
to
NEDE-24011-P

February 11, 2008

Andrew A. Lingenfelter, Manager
GNF Engineering
Global Nuclear Fuels - Americas, LLC
P.O. Box 780, M/C F12
Wilmington, NC 28402

SUBJECT: FINAL SAFETY EVALUATION FOR GLOBAL NUCLEAR FUEL (GNF) AMENDMENT 30 TO TOPICAL REPORT (TR) NEDE-24011P-A/ NEDO-24011-A, "GENERAL ELECTRIC STANDARD APPLICATION FOR RELOAD FUEL (GESTAR II)" (TAC NO. MD5742)

Dear Mr. Lingenfelter:

By letter dated May 11, 2007, GNF submitted Amendment 30 to TR NEDE-24011P-A/ NEDO-24011-A, entitled "General Electric Standard Application for Reload Fuel (GESTAR II)" to the U.S. Nuclear Regulatory Commission (NRC) staff for review.

The NRC staff has found that Amendment 30 to GESTAR II is acceptable for referencing in licensing applications for General Electric-designed boiling water reactors to the extent specified in the enclosed final SE. The final SE defines the basis for acceptance of the TR.

Our acceptance applies only to material provided in the subject TR. We do not intend to repeat our review of the acceptable material described in the TR. When the TR appears as a reference in license applications, our review will ensure that the material presented applies to the specific plant involved. License amendment requests that deviate from this TR will be subject to a plant-specific review in accordance with applicable review standards.

In accordance with the guidance provided on the NRC website, we request that GNF publish accepted proprietary and non-proprietary versions of this TR within three months of receipt of this letter. The accepted versions shall incorporate this letter and the enclosed final SE after the title page. Also, they must contain historical review information, including NRC requests for additional information and your responses. The accepted versions shall include a "-A" (designating accepted) following the TR identification symbol.

If future changes to the NRC's regulatory requirements affect the acceptability of this TR, GNF and/or licensees referencing it will be expected to revise the TR appropriately, or justify its continued applicability for subsequent referencing.

Sincerely,

/RA/

Ho K. Nieh, Deputy Director
Division of Policy and Rulemaking
Office of Nuclear Reactor Regulation

Project No. 712

Enclosure: Final SE

cc w/encl: See next page

If future changes to the NRC's regulatory requirements affect the acceptability of this TR, GNF and/or licensees referencing it will be expected to revise the TR appropriately, or justify its continued applicability for subsequent referencing.

Sincerely,

/RA/

Ho K. Nieh, Deputy Director
Division of Policy and Rulemaking
Office of Nuclear Reactor Regulation

Project No. 712

Enclosure: Final SE

cc w/encl: See next page

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SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

AMENDMENT 30 TO TOPICAL REPORT NEDE-24011P-A/NEDO-24011-A

"GENERAL ELECTRIC STANDARD APPLICATION FOR RELOAD FUEL (GESTAR II)"

GLOBAL NUCLEAR FUEL (GNF)

PROJECT NO. 712

1.0 INTRODUCTION AND BACKGROUND

By letter dated May 11, 2007, GNF submitted Amendment 30 to Topical Report (TR) NEDE-24011P-A/NEDO-24011-A, entitled "General Electric Standard Application for Reload Fuel (GESTAR II)" to the U.S. Nuclear Regulatory Commission (NRC) staff for review (Agencywide Documents Access and Management System (ADAMS) Accession No. ML071370095). The changes in Amendment 30 are administrative in nature and include updated revision references and addition of new approved codes.

2.0 EVALUATION

The NRC staff has evaluated each of the proposed changes below.

Add Reference to TR NEDC-33173P to Sections S.5 and S.5.2

This change revises Sections S.5 and S.5.2 to reflect approval of NEDC-33173P, "Applicability of GE Methods to Expanded Operating Domains," and adds the TR as Reference S-102 in Section S.6 of GESTAR II. This TR has been previously reviewed and approved (ADAMS Package Accession No. ML073340231). The NRC staff agrees that adding this reference is an administrative change and, therefore, is acceptable.

Hope Creek Loss-of Coolant Accident (LOCA) Analysis Reference Updated

This change revises Reference S-60 in Section S.6 and Table S-5 to correct the Hope Creek LOCA analysis, which was out-of-date. The NRC staff agrees that change is administrative and, therefore, is acceptable.

3.0 CONCLUSION

Based on the above, the NRC staff finds that the proposed changes are administrative and therefore are acceptable and appropriate for inclusion in Amendment 30 to TR NEDE 24011-P-A.

Principle Contributor: M. Honcharik

Date:

ENCLOSURE

Global Nuclear Fuel Project No. 712

cc:

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08/08/07

NRC
Safety Evaluation Report
Approving Amendment 31
to
NEDE-24011-P

September 10, 2010

Mr. Andrew A. Lingenfelter
Vice President, Fuel Engineering
Global Nuclear Fuel–Americas, LLC
P.O. Box 780, M/C A-55
Wilmington, NC 28401

SUBJECT: FINAL SAFETY EVALUATION FOR AMENDMENT 31 TO GLOBAL NUCLEAR FUEL TOPICAL REPORT NEDE-24011-P, "GENERAL ELECTRIC STANDARD APPLICATION FOR REACTOR FUEL (GESTAR II)" (TAC NO. MD8425)

Dear Mr. Lingenfelter:

By letter dated December 7, 2007, (Agencywide Documents Access and Management System Accession No. ML073470843), Global Nuclear Fuel – Americas, LLC (GNF) submitted Amendment 31 to topical report (TR) NEDE-24011-P, "General Electric Standard Application for Reactor Fuel (GESTAR II)" to the U.S. Nuclear Regulatory Commission (NRC) staff. By letter dated June 30, 2010, an NRC draft safety evaluation (SE) regarding our approval of Amendment 31 to NEDE-24011-P was provided for your review and comment. By letter dated July 30, 2010, GNF commented on the draft SE. The NRC staff's disposition of GNF's comments on the draft SE are discussed in the attachment to the final SE enclosed with this letter.

The NRC staff has found that Amendment 31 to NEDE-24011-P is acceptable for referencing in licensing applications for GNF-designed fuel for boiling water reactors to the extent specified and under the limitations delineated in the TR and in the enclosed final SE. The final SE defines the basis for our acceptance of the TR.

Our acceptance applies only to material provided in the subject TR. We do not intend to repeat our review of the acceptable material described in the TR. When the TR appears as a reference in license applications, our review will ensure that the material presented applies to the specific plant involved. License amendment requests that deviate from this TR will be subject to a plant-specific review in accordance with applicable review standards.

In accordance with the guidance provided on the NRC website, we request that GNF publish accepted proprietary and non-proprietary versions of this TR within three months of receipt of this letter. The accepted versions shall incorporate this letter and the enclosed final SE after the title page. Also, they must contain historical review information, including NRC requests for additional information and your responses. The accepted versions shall include an "A" (designating accepted) following the TR identification symbol.

As an alternative to including the RAIs and RAI responses behind the title page, if changes to the TR were provided to the NRC staff to support the resolution of RAI responses, and the NRC staff reviewed and approved those changes as described in the RAI responses, there are two ways that the accepted version can capture the RAIs:

1. The RAIs and RAI responses can be included as an Appendix to the accepted version.
2. The RAIs and RAI responses can be captured in the form of a table (inserted after the final SE) which summarizes the changes as shown in the approved version of the TR. The table should reference the specific RAIs and RAI responses which resulted in any changes, as shown in the accepted version of the TR.

If future changes to the NRC's regulatory requirements affect the acceptability of this TR, GNF and/or licensees referencing it will be expected to revise the TR appropriately, or justify its continued applicability for subsequent referencing.

Sincerely,

/RA/

Thomas B. Blount, Deputy Director
Division of Policy and Rulemaking
Office of Nuclear Reactor Regulation

Project No. 712

Enclosure:
Final SE

cc w/encl: See next page

A. Lingenfelter - 2 -

As an alternative to including the RAIs and RAI responses behind the title page, if changes to the TR were provided to the NRC staff to support the resolution of RAI responses, and the NRC staff reviewed and approved those changes as described in the RAI responses, there are two ways that the accepted version can capture the RAIs:

1. The RAIs and RAI responses can be included as an Appendix to the accepted version.
2. The RAIs and RAI responses can be captured in the form of a table (inserted after the final SE) which summarizes the changes as shown in the approved version of the TR. The table should reference the specific RAIs and RAI responses which resulted in any changes, as shown in the accepted version of the TR.

If future changes to the NRC's regulatory requirements affect the acceptability of this TR, GNF and/or licensees referencing it will be expected to revise the TR appropriately, or justify its continued applicability for subsequent referencing.

Sincerely,

/RA/
Thomas B. Blount, Deputy Director
Division of Policy and Rulemaking
Office of Nuclear Reactor Regulation

Project No. 712

Enclosure:
Final SE

cc w/encl: See next page

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Global Nuclear Fuel Project No. 712

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FINAL SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

AMENDMENT 31 TO TOPICAL REPORT NEDE-24011-P-A/NEDO-24011-A

"GENERAL ELECTRIC STANDARD APPLICATION FOR REACTOR FUEL (GESTAR II)"

GLOBAL NUCLEAR FUEL – AMERICAS, LLC

PROJECT NO. 712

1.0 INTRODUCTION

By letter dated December 7, 2007, Global Nuclear Fuel – Americas, LLC (GNF) submitted Amendment 31 to topical report (TR) NEDE-24011-P-A, "General Electric Standard Application for Reactor Fuel (GESTAR II)" to the U.S. Nuclear Regulatory Commission (NRC) staff for review and approval (Reference 1). GESTAR II provides a fuel design and core reload process used extensively by licensees with GNF or GE Hitachi Nuclear Energy Americas, LLC (GEH) fuel designs. This amendment proposes to make changes to GESTAR II and its U.S. Supplement to incorporate updates to the Stability Analysis sections and the Supplemental Reload Licensing Report (SRLR) template, as well as several administrative changes.

The NRC staff's safety evaluation (SE) was based on review of the GESTAR II Amendment 31 submittal, GNF's responses (References 2, 3, and 4) to the NRC staff's Request for Additional Information (RAI, Reference 5), and several conference calls for the clarification of the draft RAI responses.

2.0 REGULATORY EVALUATION

TR NEDE-24011-P-A/NEDO-24011-A, provides an NRC-approved fuel design and core reload process. The approved methodology and acceptance criteria detailed within TR NEDE-24011 are cited within many boiling water reactor (BWR) technical specifications as references in the core operating limits report.

NRC-approved methodologies are used to establish setpoints and demonstrate the adequacy of the protection systems to prevent violation of the critical power ratio (CPR) safety limits in compliance with Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50, Appendix A, General Design Criteria (GDC) for Nuclear Power Plants, GDC-10 "Reactor Design" and GDC-12, "Suppression of Reactor Power Oscillations."

ENCLOSURE

Criterion 10 - Reactor design. The reactor core and associated coolant, control, and protection systems shall be designed with appropriate margin to assure that specified acceptable fuel design limits are not exceeded during any condition of normal operation including the effects of anticipated operational occurrences (AOOs).

Criterion 12 - Suppression of reactor power oscillations. The reactor core and associated coolant, control, and protection systems shall be designed to assure that power oscillations which can result in conditions exceeding specified acceptable fuel design limits are not possible or can be reliably and readily detected and suppressed.

To assure compliance with GDC 10 and 12, the NRC staff confirms that the thermal and hydraulic design of the core and the reactor coolant system (RCS) has been accomplished using acceptable analytical methods, provides acceptable safety margins from conditions that could lead to fuel damage during normal reactor operation and AOOs, and is not susceptible to thermal-hydraulic instability. Regulatory guidance for the review of the thermal and hydraulic design and the suppression of reactor power oscillations is provided in NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants" (SRP), Section 4.4, "Thermal and Hydraulic Design," and SRP Section 15.9, "BWR Core Stability." SRP Section 4.4 describes the normal review of thermal and hydraulic design and requires that additional independent audit analyses be performed for new CPR correlations. SRP Section 15.9 describes the possibility of thermal-hydraulic instability in boiling water reactors (BWRs), analytical methods and codes to predict the stability characteristics of BWRs, and long-term stability solutions.

3.0 TECHNICAL EVALUATION

The NRC staff's technical review of GESTAR II Amendment 31 included review of the proposed changes and updates to the following sections of GESTAR II in accordance with the provisions in Chapter 15.0 of NUREG-0800.

3.1 Sections 3.1.2, 3.2.2, and Table 3.1 - Modified "Maximum Linear Heat Generation Rate (MLHGR)" to "LHGR Operating Limit" for consistency with terminology used in Section 2.

The MLHGR is one of the important technical specification (TS) limits used to determine that the fuel will not exceed required licensing limits during AOOs or accidents. It is cited in Section 3.1.2 - "Overpower Bases," Section 3.2.2 - "Power Distribution," and Table 3.1 - "Definition of Fuel Design Limits." The linear heat generation rate (LHGR) operating limit is established to ensure that the steady-state LHGR value experienced by any fuel rod at any axial location will not violate the fuel thermal-hydraulic licensing acceptance criteria should an AOO occur. The steady-state operating limit on LHGR can be identified either as the "MLHGR" or as the "LHGR operating limit." The change from "MLHGR" to "LHGR operating limit" in these sections is strictly a nomenclature clarification. Therefore, the proposed change is acceptable.

3.2 Section 3.4.2.10 - Added a new criteria section for the stability analysis.

This new Section 3.4.2.10, "Stability" states that the stability analysis for the reference core is applicable to the actual core if the core loading remains within the allowable criteria of GESTAR II Section 4.2, "Description of Thermal-Hydraulic Design of the Reactor Core" and the exposure remains within the specified window. The NRC staff reviewed this added section for stability analysis and found it acceptable because the reference core will use the same inputs applied to actual core specified in GESTAR II Section 4.2.

3.3 Section 4.3.1.2.8 - Modified the text used in AOO transient power/flow conditions to use the End-Of-Rated (EOR) terminology instead of End-Of-Cycle (EOC) to be more consistent with current terminology.

AOO analyses are performed at the rated core power, rated core flow, all-rods-out conditions, referred to as EOR. In other words, the term EOR refers to the cycle exposure at which the reactor has reached end-of-full-power reactivity at rated conditions (i.e., rated power, flow, pressure, and feedwater temperature) with all control rods fully withdrawn. This EOR point is sometimes referred to as end of reactivity. The EOR point determines the highest exposure at which full power operation can be maintained. The term EOC refers to the cycle exposure at which the reactor ceases operation for that cycle. The EOC condition can occur before, at, or after EOR, although it most often occurs after EOR. The change reflects a consistency with the current terminology used in AOO transient analyses, and therefore, is acceptable.

3.4 Section S.4 - Modified the text to reflect that the stability analysis methods are now performed or confirmed on a cycle-specific basis.

CPR response calculations are performed to demonstrate the safety limit minimum CPR (SLMCPR) protection against a thermal-hydraulic instability event using the detect and suppress methodology. The proposed change to the plant- and cycle-specific core-wide mode DIVOM (Delta CPR over Initial MCPR Versus Oscillation Magnitude) data for Option I-D plant stability analysis and the plant- and cycle-specific regional mode DIVOM data for Option II and Option III plant stability analyses is acceptable because it reflects a plant application of the BWROG position on plant- and cycle-specific core-wide mode and the regional mode DIVOM procedure guideline.

3.5 Section S.4.1.1 though S.4.1.4 - Clarified that Enhanced Option I-A, Option II, Option I-D, and Option III were reviewed and approved by the NRC staff for operation up to the Maximum Extended Load Line Limit Analysis (MELLLA) domain.

BWROG Long-Term Stability Solutions described in Sections S.4.1.1 - "Enhanced Option I-A," S.4.1.2 - "Option II," S.4.1.3 - "Option I-D," and S.4.1.4 - "Option III" were reviewed and approved by the NRC staff for operation up to and including the MELLLA domain. Therefore, the proposed clarification is acceptable.

3.6 Section S.4.1.5 - Added a new section to include the reviewed and approved Detect and Suppress Solution - Confirmation Density (DSS-CD) stability method.

The proposed new section, which includes the GEH DSS-CD stability solution and a reference to the results of the SLMCPR protection calculation in the SRLR, is acceptable because the DSS-CD was reviewed and approved by the NRC staff for operation up to and including the MELLLA Plus (MELLLA+) domain.

3.7 Section S.4.2.2 and S.4.2.3 - Added two new sections on backup stability protection (BSP) for Option III and DSS-CD.

These two BSP methodologies (i.e., S.4.2.2 BSP for Option III, and S.4.2.3 BSP for DSS-CD) are very similar. However, two differences exist:

(1) The ODYSY core decay ratio (DR) acceptance criterion used for the Controlled Entry Region boundary intercept along the High Flow Control Line (HFCL) is different for the two solutions. The Option III BSP uses a DR acceptance criterion of 0.8, while the DSS-CD BSP uses a DR acceptance criterion of 0.6. For Option III BSP, the HFCL is defined as the MELLLA boundary. For DSS-CD BSP, the HFCL is defined as the MELLLA+ boundary.

(2) The other difference is the imposition of the BSP boundary on the DSS-CD solution while in the manual mode. While such a boundary does not exist in the BSP for Option III, the Option III solution limits the upper boundary to the MELLLA boundary. The Base BSP regions for Option III are defined in a GE letter to the BWR Owners' Group (BWROG), "BSP for Inoperable Option III Solution," dated July 17, 2002 (Reference 6), which provides the requirement to meet at least one of five stability controls if there is deliberate entry into the BSP Controlled Entry Region. Only the automated BSP option is approved for use as an extended backup solution to DSS-CD.

A licensee submitting a license amendment request to implement Option III will be required to demonstrate its capability or procedure for meeting at least one of the five stability controls specified in Reference 6 as BSP for inoperable Option III.

The BSP is required for an alternative interim prevention solution in case the Option III or DSS-CD solution is not operational. Therefore, the proposed change to add two new sections on BSP for Option III and DSS-CD with illustrations of the power/flow maps in Figure S-7 and Figure S-8, respectively, is acceptable.

3.8 Section S.5.1.3 - Clarified that the Average Power Range Monitor (APRM) Simulated Thermal Power Trip is standard equipment in some BWR/4s, and all BWR/5s and /6s.

The proposed change to add "in some BWR/4 plants and all BWR/5 and BWR/6 plants" provides a clarification that the APRM Simulated Thermal Power Trip is standard equipment in some specific BWR plants. Therefore, it is acceptable.

3.9 Section S.5.1.4 - Modified the text used in AOO transient power/flow conditions to use the EOR terminology instead of EOC to be consistent with current terminology.

The proposed change from EOC to EOR is acceptable since it is consistent with current terminology as described in Section 3.3 of this SE.

3.10 Section S.6 - Added four new references.

- S-101 GE-NE-0000-0031-6498-RO, Plant-Specific Core-wide Mode DIVOM Procedure Guideline, June 2, 2005.
- S-102 GE-NE-0000-0028-9714-R1, Plant-Specific Regional Mode DIVOM Procedure Guideline, June 2, 2005.
- S-103 General Electric Boiling Water Reactor Detect and Suppress Solution - Confirmation Density, NEDC-33075P-A, Rev. 6, January 2008.
- S-104 ODYSY Application for Stability Licensing Calculations Including Option I-D and II Long Term Solutions, NEDE-33213P-A, April 2009.

The NRC staff reviewed the proposed addition of four new references and found them acceptable. S-101 and S-102 are the BWROG positions that deal with plant-specific guidelines for core-wide and regional mode DIVOM calculations. S-103 and S-104 are approved methodologies used for DSS-CD and for Option I-D and II, respectively.

When implementing NRC-approved methodologies, licensees must be in compliance with NRC Generic Letter 88-16 (Reference 7) guidance, which addresses the appropriate modifications to the Administrative Controls section of a facility's TS that are necessary to implement and use a Core Operating Limits Report (COLR). In particular, (1) identification of the individual specifications that address the core operating limits may be included, if desired, in the Reporting Requirements of the plant TS; (2) the supporting TRs by number and title shall be provided in the Reporting Requirements of the plant TS; and (3) specification of the TRs by number, title, revision level, and date of the approved TR shall be provided in the COLR.

3.11 U.S. Supp. - App. A - Replaced SRLR template with current revision.

The NRC staff participated in several phone calls with GNF to clarify the need for plant-specific numbers or specific events as examples to add to the SRLR template, which consists of sixteen sections. Subsequently, GNF addressed the issue in its RAI 8 response (References 3, 4, and 5) by providing a tabulation of typical examples for each of the various types of plant/cycle applicable descriptions as a part of the Template Symbol Keys in the beginning of Appendix A, "Standard Supplemental Reload Licensing Report." The NRC staff found the proposed addition to the SRLR template is acceptable because each applicable section of the SRLR template has a footnote indicating the appropriate information from GESTAR II or providing a specific explanation.

The NRC staff recommends that more specific information for the safety limiting values and fuel loading pattern should be added in future amendments as examples to make them more comprehensive and understandable for the users.

4.0 CONCLUSION

In its review, the NRC staff found that the proposed changes submitted in GESTAR II Amendment 31 are administrative in nature for terminology, or incorporate updates for currently approved BWR stability methodologies and a simplified SRLR template. Based on this review, the NRC staff concludes that the proposed changes to GESTAR II and its U.S. Supplement to incorporate updates to the stability analysis section, the SRLR template update, and several administrative changes in GESTAR II Amendment 31 are acceptable.

5.0 REFERENCES

1. Letter from GNF to USNRC, FLN-2007-036, “Proposed GESTAR II Amendment 31, Stability Analysis and SRLR Template Update,” dated December 7, 2007. (ADAMS Package Accession Number ML073470832)

2. Letter from GNF to USNRC, MFN 09-512, “Response to Request for Additional Information Relating to Amendment 31 to NEDE-24011P-A, ‘General Electric Standard Application for Reload Fuel (GESTAR II),’ (TAC Number MD8425),” dated August 5, 2009. (ADAMS Package Accession Number ML092220262)

3. Letter from GNF to USNRC, MFN 09-512, Supplement 1, “Supplementary Response to Request for Additional Information Relating to Amendment 31 to NEDE-24011P-A, ‘General Electric Standard Application for Reload Fuel (GESTAR II),’ (TAC Number MD8425),” dated October 1, 2009. (ADAMS Package Accession Number ML092780409)

4. Letter from GNF to USNRC, MFN 09-512, Supplement 2, “Supplementary Response to Request for Additional Information Relating to Amendment 31 to NEDE-24011P-A, ‘General Electric Standard Application for Reload Fuel (GESTAR II),’ (TAC Number MD8425),” dated March 12, 2010. (ADAMS Package Accession Number ML101100588)

5. Letter from USNRC to GNF, “Request for Additional Information Re: Amendment 31, ‘Stability Analysis and SRLR Template Update,’ to the Topical Report (TR) NEDE-24011P-A, ‘General Electric Standard Application for Reload Fuel (GESTAR II),’ (TAC Number MD8425),” dated August 5, 2008. (ADAMS Accession Number ML082060097)

6. Letter from GE to BWR Owners’ Group Detect and Suppress II Committee, OG 02-0119-260, “Backup Stability Protection (BSP) for Inoperable Option III Solution,” dated July 17, 2002.

7. NRC Generic Letter 88-16, "Removal of Cycle-Specific Parameter Limits from Technical Specifications," dated October 4, 1988. (ADAMS Legacy Accession Number 8810050058)

Principal Contributor: Tai Huang, SRXB/DSS

Date: September 10, 2010

#	Location in Draft SE	Draft SE Text	GNF Comment and Basis	NRC Staff Resolution
1	Page 1 Section 2.0 Line 36 -37	The NRC staff will ensure that approved methodologies are used to establish setpoints and demonstrate the adequacy of the protection systems to prevent violation of the critical power ratio (CPR) safety limits in compliance with Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50, Appendix A, General Design Criteria (GDC) for Nuclear Power Plants, GDC-10 "Reactor Design" and GDC-12, "Suppression of Reactor Power Oscillations."	The sentence suggests some un-described action on the part of the NRC. GNF uses approved methodologies to perform analyses to establish protection system setpoints. As stated, GNF interprets this as being a general statement and the actions associated with the statement to be fully the NRC's. The NRC actions to be taken should be delineated or the sentence deleted.	Section revised to: TR NEDE-24011-P-A/NEDO-24011-A, provides an NRC-approved fuel design and core reload process. The approved methodology and acceptance criteria detailed within TR NEDE-24011 are cited within many boiling water reactor (BWR) technical specifications as references in the core operating limits report. NRC-approved methodologies are used to establish setpoints and demonstrate the adequacy of the protection systems to prevent violation of the critical power ratio (CPR) safety limits in compliance with Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50, Appendix A, General Design Criteria (GDC) for Nuclear Power Plants, GDC-10 "Reactor Design" and GDC-12, "Suppression of Reactor Power Oscillations."
2	Page 4 Section 3.7 Line 11	For DSS-CD BSP, the HFLC is defined as the MELLLA+ boundary.	HFLC should be HFCL	Corrected.

ATTACHMENT

#	Location in Draft SE	Draft SE Text	GNF Comment and Basis	NRC Staff Resolution
3	Page 4 Section 3.7 Line 23-25	The NRC staff will require a detailed analysis for an Option III plant-specific application to confirm that BSP for inoperable Option III meets at least one of the five stability controls specified in Reference 6.	GNF offers several comments regarding the highlighted statement from Section 3.7: 1. Plants do not use GESTAR II as the licensing process to make a plant specific License Amendment Request for the application of the Option III stability solution. Therefore, GESTAR II is not an appropriate place to make such a declaration. 2. The five stability controls from Reference 6 were placed in GESTAR at NRC's request for information purposes regarding Option III. These generic requirements are a Backup Stability Protection (BSP) defense in depth feature for plant operation, and not directly tied to reload safety analysis.	Sentence revised as: A licensee submitting a license amendment request to implement Option III will be required to demonstrate its capability or procedure for meeting at least one of the five stability controls specified in Reference 6 as BSP for inoperable Option III.

#	Location in Draft SE	Draft SE Text	GNF Comment and Basis	NRC Staff Resolution
4	Page 5 Section 3.10 Line 10-11	S-103 and S-104 are approved methodologies used for DSS-CD and for Option I-D and II, respectively.	There are elements of S-104 that are not limited to Option I-D and II. See the portion of the S-104 SE reproduced below. 5.0 CONCLUSION.......... While the NRC has previously reviewed the ODYSY code and found it applicable to operating BWR designs (Reference 4), the current approval is limited only to those BWRs implementing either Option I-D or II LTS solution, with the exception of the shared elements with other solutions as specified in the subject LTR. GNF suggests something like: S-103 and S-104 are approved methodologies used for DSS-CD and for Option I-D and II, respectively. S-104 also includes approved elements common to other stability solutions.	Comment not accepted. The additional sentence is not needed. The draft SE statement does not exclude the applicability of S-104 to other stability solutions.

#	Location in Draft SE	Draft SE Text	GNF Comment and Basis	NRC Staff Resolution
5	Page 5 Section 3.10 Line 13-21	The NRC staff will require that future reload applications of the approved methodologies be in compliance with NRC Generic Letter 88-16 (Reference 7) guidance, which addresses the appropriate modifications to the Administrative Controls section of a facility's TS that are necessary to implement and use a Core Operating Limits Report (COLR). In particular, (1) identification of the individual specifications that address the core operating limits may be included, if desired, in the Reporting Requirements of the plant TS; (2) the supporting TRs by number and title shall be provided in the Reporting Requirements of the plant TS; and (3) specification of the TRs by number, title, revision level, and date of the approved TR shall be provided in the COLR.	This highlighted statement from the SE involves activities associated with the plant's TS and COLR, which are controlled and maintained by the plant. The TS are approved by the NRC and the content of the COLR, when selected and licensed, for each plant was likewise approved by the NRC. The function of GESTAR is to define the reload safety analysis work scope and reporting structure through the SRLR, not to delineate the detailed referencing requirements that the NRC finds satisfactory for the COLR. GNF does not produce the COLR and is rarely consulted regarding the content.	Comment partially accepted. Section revised as: When implementing NRC-approved methodologies, licensees must be in compliance with NRC Generic Letter 88-16 (Reference 7) guidance, which addresses the appropriate modifications to the Administrative Controls section of a facility's TS that are necessary to implement and use a Core Operating Limits Report (COLR). In particular, (1) identification of the individual specifications that address the core operating limits may be included, if desired, in the Reporting Requirements of the plant TS; (2) the supporting TRs by number and title shall be provided in the Reporting Requirements of the plant TS; and (3) specification of the TRs by number, title, revision level, and date of the approved TR shall be provided in the COLR.

NRC
Safety Evaluation Report
Approving Amendment 32
to
NEDE-24011-P

July 30, 2009

Andrew A. Lingenfelter, Manager
GNF Engineering
Global Nuclear Fuel - Americas, LLC
P.O. Box 780, M/C F12
Wilmington, NC 28402

SUBJECT: FINAL SAFETY EVALUATION FOR AMENDMENT 32 TO GLOBAL NUCLEAR FUEL TOPICAL REPORT NEDE-24011-P GENERAL ELECTRIC STANDARD APPLICATION FOR RELOAD (TAC NO. MD9939)

Dear Mr. Lingenfelter:

By letter dated October 15, 2008, Global Nuclear Fuel - Americas, LLC (GNF) submitted Amendment 32 to Topical Report (TR) NEDE-24011-P, "General Electric Standard Application for Reload (GESTAR II)" to the U. S. Nuclear Regulatory Commission (NRC) staff. By letter dated January 15, 2009, an NRC draft safety evaluation (SE) regarding our approval of Amendment 32 to NEDE-24011-P, was provided for your review and comment. By letter dated June 10, 2009, GNF commented on the draft SE. The NRC staff's disposition of GNF's comments on the draft SE are discussed in the attachment to the final SE enclosed with this letter.

The NRC staff has found that Amendment 32 to NEDE-24011-P is acceptable for referencing in licensing applications for General Electric-designed boiling water reactors to the extent specified and under the limitations delineated in the TR and in the enclosed final SE. The final SE defines the basis for acceptance of the TR.

Our acceptance applies only to material provided in the subject TR. We do not intend to repeat our review of the acceptable material described in the TR. When the TR appears as a reference in license applications, our review will ensure that the material presented applies to the specific plant involved. License amendment requests that deviate from this TR will be subject to a plant-specific review in accordance with applicable review standards.

In accordance with the guidance provided on the NRC website, we request that GNF publish accepted proprietary and non-proprietary versions of this TR within three months of receipt of this letter. The accepted versions shall incorporate this letter and the enclosed final SE after the title page. Also, they must contain historical review information, including NRC requests for additional information and your responses. The accepted versions shall include a "-A" (designating accepted) following the TR identification symbol.

If future changes to the NRC's regulatory requirements affect the acceptability of this TR, GNF and/or licensees referencing it will be expected to revise the TR appropriately, or justify its continued applicability for subsequent referencing.

Sincerely,

/RA/

Thomas B. Blount, Deputy Director
Division of Policy and Rulemaking
Office of Nuclear Reactor Regulation

Project No. 712

Enclosures:
1. Non-proprietary version of the Final SE
2. Proprietary version of the Final SE

cc w/encl 1 only: See next page

A. Lingenfelter - 2 -

If future changes to the NRC's regulatory requirements affect the acceptability of this TR, GNF and/or licensees referencing it will be expected to revise the TR appropriately, or justify its continued applicability for subsequent referencing.

Sincerely,

/RA/

Thomas B. Blount, Deputy Director
Division of Policy and Rulemaking
Office of Nuclear Reactor Regulation

Project No. 712

Enclosures:
1. Non-proprietary version of the Final SE
2. Proprietary version of the Final SE

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Package: ML091680754
Cover letter: ML091680405
Non-Prop Final SE: ML091750066
Prop Final SE: ML091680403

***No major changes to SE input. NRR-043**

OFFICE	PSPB/PM	PSPB/LA	SNPB/BC*	PSPB/BC	DPR/DD
NAME	MHoncharik	DBaxley (CHawes for)	AMendiola	SRosenberg	TBlount
DATE	7/20/09	7/14/09	11/13/08	7/22/09	7/30/09

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FINAL SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

TOPICAL REPORT NEDE-24011-P-A/NEDO-24011-A

"GENERAL ELECTRIC STANDARD APPLICATION FOR RELOAD (GESTAR II)"

GLOBAL NUCLEAR FUEL

PROJECT NO. 712

1.0 INTRODUCTION AND BACKGROUND

By letter dated October 15, 2008, Global Nuclear Fuel (GNF) submitted Amendment 32 to topical report (TR) NEDE-24011-P, entitled "General Electric Standard Application for Reload Fuel (GESTAR II)" (Reference 1). GESTAR II provides a fuel design and core reload process used extensively by licensees with GNF or GE Hitachi Nuclear Energy Americas, LLC (GEH) fuel designs. This U. S. Nuclear Regulatory Commission (NRC)-approved process allows GNF to modify fuel assembly designs without undergoing a formal NRC submittal and review. As part of this process, GNF provides written notification outlining the new design and acknowledging compliance with the requirements of the NRC-approved GESTAR process. Upon notification, the NRC staff may conduct an audit of the engineering calculations supporting the new fuel design. Amendment 32 to GESTAR II was necessitated by an NRC staff audit of the GNF2 fuel design compliance report.

By letter dated March 14, 2007, GNF submitted a GESTAR II Compliance Report for the advanced fuel assembly design referred to as GNF2 (Reference 2). A subsequent NRC staff audit of the GESTAR II Compliance Reports (Reference 3) yielded several NRC staff findings which need to be addressed prior to batch implementation of GNF2 fuel. One of the findings involved the use of General Electric Stress and Thermal Analysis of Reactor Rods - Mechanical (GESTR-M) fuel thermal-mechanical methodology for GNF2 fuel above a rod power of 13.4 KW/ft. Issues associated with GSTR-M had been the focus of recent GEH notifications pursuant to Title 10 of the *Code of Federal Regulations* (10 CFR) Part 21 (References 4 and 5). In response to the audit finding, GEH supplemented the Part 21 Notification (Reference 6) to expand its assessment of the adequacy of GSTR-M to GNF2 fuel design at rod powers in excess of 13.4 KW/ft. As part of its finding that the application of GSTR-M does not constitute a reportable condition under 10 CFR Part 21, GEH included a qualification (i.e., condition) which imposed a limit of applicability for [ ]. A subsequent revision to the GNF2 GESTAR II Compliance Report (Reference 7) captured this qualification.

In response to NRC staff concerns regarding the application of a fuel rod nodal exposure limit, which is more restrictive than the NRC staff's current approval of GESTAR II (including GSTR-M methods), GNF submitted Amendment 32 to GESTAR II in order to capture this interim GNF2 exposure limit.

ENCLOSURE 1

2.0 REGULATORY EVALUATION

TR NEDE-24011-P-A/NEDO-24011-A, provides an NRC-approved fuel design and core reload process. The approved methodology and acceptance criteria detailed within TR NEDE-24011 are cited within many boiling water reactor (BWR) technical specifications as references in the core operating limits report.

Regulatory guidance for the review of fuel rod cladding materials and fuel system designs and adherence to 10 CFR Part 50, Appendix A, General Design Criteria (GDC) 10, 27, and 35 is provided in NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants" (SRP), Section 4.2, "Fuel System Design". In accordance with SRP Section 4.2, the objectives of the fuel system safety review are to provide assurance that:

- The fuel system is not damaged as a result of normal operation and anticipated operational occurrences (AOOs),
- Fuel system damage is never so severe as to prevent control rod insertion when it is required,
- The number of fuel rod failures is not underestimated for postulated accidents, and
- Coolability is always maintained.

GESTAR II provides a licensed reload methodology, including an NRC-approved fuel thermal-mechanical design methodology utilized to demonstrate compliance with the fuel design criteria in SRP Section 4.2. The fuel thermal-mechanical design methodology and design criteria in GESTAR II are based upon the NRC-approved GSTR-M methodology. The GSTR-M methodology is approved up to a peak pellet exposure of [ ]. In response to NRC staff concerns regarding the conservatism of GSTR-M calculations, GEH performed several Part 21 Notification evaluations (References 4, 5, and 6). The GEH basis for concluding that the application of GSTR-M for GNF2 fuel does not constitute a reportable condition under 10 CFR Part 21 is based upon limiting this application to [ ]. Specifically, the Part 21 Notification (Reference 6) states:

> More specifically, this evaluation demonstrates that the GESTR-M code and associated application methodology is adequate for GNF2 fuel [ ].

A subsequent revision to the GNF2 GESTAR II Compliance Report (Reference 7) captured this qualification, stating:

> The GNF2 peak pellet exposure based on the GESTR-Mechanical model is limited [ ] consistent with Reference 58 [Part 21, Supplement 2].

The GEH justification (as to why the GSTR-M application to GNF2 does not constitute a reportable condition under 10 CFR Part 21) does not address operation [ ]. GEH has not demonstrated (1) an adequate level of conservatism within the GSTR-M methodology nor (2) an acceptable GNF2 fuel performance over the entire range of the NRC staff's original review and approval [ ]. Furthermore, as will be discussed in Section 3.0 of this safety evaluation (SE), independent calculations performed by the NRC staff

reveal that the GNF2 fuel rod design violates design requirements prior to the approved end-of-life (EOL).
In response to NRC staff concerns regarding the application of a fuel rod nodal exposure limit, which is more restrictive than the NRC staff's current approval of GESTAR II (including GSTR-M methods), GNF submitted Amendment 32 to GESTAR II.

3.0 TECHNICAL EVALUATION

The latest approved version of GESTAR II is Amendment 30 to NEDE-24011-P (Reference 8). Amendment 31 updates the Stability Analysis and is currently undergoing staff review. Amendment 32 addresses staff audit findings and proposes a more restrictive fuel exposure limit for application of GESTAR II to GNF2 fuel.

Up to a peak pellet exposure [ ], GEH has addressed (1) the adequacy of GSTR-M for application to GNF2 (Reference 6) and (2) GNF2 fuel design's compliance with GESTAR II approved design methodology and design criteria (Reference 7). Amendment 32 (Reference 1) builds upon these previous evaluations and specifically addresses each audit finding.

The staff's assessment of GEH's response to each audit finding (located in Table 1, Enclosure 1, Reference 1) is provided below.

Audit Finding #1:

Based on limited lead use assembly (LUA) operating history and the lack of a post-irradiation examination to validate in-reactor performance up to EOL exposure, GEH has neither met the intent of the GESTAR II LUA requirement, nor satisfied established regulatory practice.

In its response, GEH states that the LUA program for GNF2 "…is completely consistent with GESTAR II and with the long history of LUAs applied under GESTAR II." Further, GEH states that the "…evolutionary changes from an experience base of over 26,000 GE14 and GE12 bundles, not warranting more extensive LUA exposure or examinations." The NRC staff does not accept this position and expects further in-reactor experience and inspection prior to batch application.

GEH acknowledges that continued inspections at interim exposures are planned and will reveal any unanticipated behavior well before GNF2 reload bundles reach similar exposures. In addition, the exposure of GNF2 LUAs will always lead the reloads by a substantial margin.

Approval of the application of GESTAR II and use of GNF2 fuel for the [ ] is supported by the limited LUA operating experience documented in the compliance report and in Amendment 32. Extension [ ] requires further LUA operating experience and inspection along with NRC review and approval.

Audit Findings #2, #6, #7, #8, and #9:

All findings are related to adequacy of GSTR-M methods [ ] and GNF2 No Clad Liftoff (NCLO) rod internal pressure design calculations.

In addition to the detailed information provided by GEH, the NRC staff has performed independent calculations using the FRAPCON-3 fuel thermal-mechanical model. The NRC staff's calculations are documented in Table 3 of the audit report (Reference 3). The FRAPCON-3 calculations confirm that the GNF2 fuel rod design satisfies all thermal-mechanical design criteria except NCLO rod internal pressure criteria. Independent calculations reveal that the NCLO criteria (cladding creep outward, fuel pellet/cladding gap opening) are violated prior to the approved EOL [ ].

The FRAPCON-3 calculations confirm earlier concerns regarding the adequacy of GSTR-M at higher burnup. Specifically, GSTR-M calculations under predict UO_2 fuel temperature, which results in an under prediction of fission gas release and rod internal pressure. Hence, GSTR-M calculations do not predict clad liftoff (CLO) for the GNF2 fuel rod design.

In Item 2 of Table 1 (Reference 1), GNF states that GE11 fuel rods were licensed at [ ]. Based upon the concerns discussed above, this would bring into question the adequacy of the GSTR-M calculations for these higher power fuel rods. During a past audit, the NRC staff discussed the impact of the GSTR-M 10 CFR Part 21 concerns on the GE11/13 rod designs. Crediting the larger fuel rod plenum region of the GE11/13 (relative to the GE14 design), GEH provided sample rod internal pressure calculations, which demonstrate significant pressure margin to the NCLO criteria. GEH stated that millions of GE11/13 rods have operated to design exposures with no indications of problems due to high internal rod pressure. This design is now being phased out in BWR/3-6 reactors, but is still being supplied to BWR/2 reactors. However, its application to BWR/2 reactors is limited to a peak linear heat generation rate [ ] due to the loss-of-coolant accident response characteristics for these reactors. The NRC staff accepts the disposition of this issue for GE11/13 fuel rods designs.

The NRC staff's independent calculations predict CLO of the GNF2 fuel rod design, but at an exposure [ ]. Based upon the GEH thermal-mechanical analyses and the NRC staff's independent calculations, the NRC staff finds the application of GSTR-M to GNF2 fuel acceptable up to a peak pellet exposure of [ ]. Extension [ ] requires further justification. This may involve using an approved PRIME methodology and/or a modified GNF2 fuel rod design. NRC review and approval is required to [ ].

<u>Audit Finding #3</u>:

The GNF2 design continues to use the [ ] strain design criteria. While this approach is consistent with GESTAR II, it does not address issues identified by the NRC staff during the economic simplified BWR (ESBWR) review of GE14E fuel design. Note that GEH plans to revise the fuel rod cladding strain design criteria for the ESBWR fuel design (GE14E). GEH needs to demonstrate, via empirical data, that the GNF2 fuel rod cladding is capable of achieving the [ ] at EOL conditions or revisit the criterion.

This item is discussed in Supplement 2 of the Part 21 Notification (Reference 6). GEH states that the exposure-dependent strain limits proposed for ESBWR are consistent with the analyzed strain criteria [ ]. The NRC staff agrees with this statement. However, the GNF2 Alloy X-750 grid spaces provide an additional source of hydrogen pickup for the fuel rod

cladding which must be considered when setting the exposure-dependent breakpoint. Extension [ ] requires further justification for the exposure-dependent strain limits for GNF2 and NRC review and approval.

Audit Finding #4:

The GNF2 fuel rod design does not include limits for cladding corrosion. While this approach is consistent with GESTAR II, corrosion limits are required to ensure that key assumptions related to fuel performance analyses remain applicable. Specifically, an upper limit on local cladding oxidation (corresponding to oxide spallation) and an upper limit on local cladding hydrogen content (corresponding to the strain limit) need to be provided.

This item is being addressed for the GE14E fuel assembly design in the ESBWR design review. It is anticipated that a similar approach will be pursued for GNF2. Since cladding corrosion is expected to be low [ ]. Extension [ ] requires further justification, established corrosion limits, and NRC review and approval.

Audit Finding #5:

The GNF2 design maintains an allowance for fuel centerline melting during local AOOs. While this approach is consistent with GESTAR II, little data is available to validate fuel swelling models at melting conditions, especially for higher burnup fuel. In addition, little data is available to validate fuel performance models for future operation with fuel rods which have previously undergone melting. If GNF desires to maintain this approach, then validation of these models against measured data should be included in the ongoing PRIME review.

In its response, GEH states that the GSTR-M application methodology is such that melting during local AOOs is precluded for any fuel design and that current reloads do not utilize the GESTAR II allowance for limited fuel melting. The NRC staff considers this issue resolved for GNF2 fuel.

Audit Finding #10: Open Items.

Amendment 32 provides a response to the open items identified in the NRC staff's GNF2 audit (Reference 3). The first open item requested information related to GNF2 channel design's susceptibility to shadow corrosion induced channel bow. In its response, GEH stated that the minor differences between GNF2 channels and GE14 channels will not exacerbate channel bow. Further, GNF continues to manage channel distortion via the cell friction methodology, which minimizes the likelihood of control blade interference. Based upon ongoing efforts to control channel bow and no significant differences in channel design (which would exacerbate the issue), the NRC staff finds the GNF2 channel design acceptable.

In a second open item, the NRC staff requested information related to the effect of GNF2 design features on flow induced vibration. In its response, GEH stated that there were no known occurrences of grid to rod fretting failures in GNF BWR fuel designs over several decades of deployment. To date, inspections on GNF2 LUAs have shown no abnormal indications near grid locations. The NRC staff finds the application of GESTAR II and use of GNF2 fuel for the

[ ] acceptable based on the limited LUA operating experience (especially fuel rod wear inspections under grid straps). Extension beyond the [ ] requires further justification that assembly design features (e.g., introduction of mixing vanes) do not introduce fuel rod vibration and the potential for grid-to-rod fretting and NRC review and approval.

In a third open item, the NRC staff requested information related to the inclusion of water holes in the water rod structural analysis. In its response, GEH concluded that while the water rod holes were not explicitly modeled in the finite element analysis (FEA), the amount of conservatism in the structural calculations assuming all loads are applied at the minimum water rod diameter offsets the reduced cross-sectional effect of both sets of water rod holes. During a recent ESBWR audit, the NRC staff questioned similar engineering judgments for the GE14E fuel design. GEH, following its corrective action program, is performing detailed FEA calculations (modeling the water rod holes) to investigate its conclusion. The GNF2 fuel design does not introduce any new design features which exacerbate this potential problem. As such, the NRC staff considers this issue to be generic in scope and not specific to its approval of Amendment 32 or the GNF2 fuel design.

In a fourth open item, the NRC staff requested information related to the applicability of power ramp test results to GNF2 fuel. In its response, GEH stated that current GNF2 fuel designs have the standard barrier cladding design. Historically, the inclusion of the zirconium barrier has been an effective method on minimizing vulnerability to pellet cladding interaction (PCI)/stress corrosion cracking (SCC). A comparison of power ramp test results with barrier cladding (Figure 1 of Reference 1) shows that PCI/SCC failure would not be expected at or below the GNF2 rod power envelope. In its response, GEH provides a discussion of the applicability of the power ramp test results to the GNF2 design. GEH states that the local cladding stresses are driven by the change in local power (and resulting pellet strain) and independent of rod diameter and cladding thickness. One item not discussed is the initial pellet-to-cladding gap size between the older test rods and GNF2. For a given power change, initial gap size will impact cladding stresses. This item requires further investigation prior to removing the [ ] limit.

The GNF2 design includes a non-barrier option. Due to the limited scope of this review and schedule restrictions, the NRC staff was unable to reach a safety finding with respect to the acceptability of a non-barrier GNF2 fuel rod design. Hence, the staff's approval of Amendment 32 for GNF2 is limited to the zirconium barrier fuel rod design.

In a fifth open item, the NRC staff requested information related to local cladding hydrogen concentration near the Alloy X-750 grid spacers. In its response, GEH concludes that the performance of GNF2 will not be adversely affected by shadow corrosion and hydriding at spacer locations, especially given the rod exposure limit. Based upon anticipated corrosion (and hydrogen pickup) during the limited rod exposure, the NRC staff finds this response acceptable. However, further data needs to be provided to justify extended [ ].

As indicated in Table 1 of Reference 1, Audit Findings #11 through #22 do not require any actions or response.

Table 2 of Reference 2 provides “Commitments to Changes in GESTAR II and the GNF2 Compliance Report.” The commitments include changes to GESTAR II to incorporate the PRIME thermal-mechanical methodology (currently under NRC staff review) and to address audit findings. Since these commitments involve future changes to an NRC-approved TR, they are outside the NRC staff’s review of Amendment 32 and must be submitted separately for NRC staff review and approval.

Based upon the disposition of the GNF2 audit findings above, the NRC staff finds the application of GESTAR II and use of GNF2 fuel for the [ ] acceptable. As noted above, extension [ ] requires further justification and NRC review and approval.

4.0 LIMITATIONS AND CONDITIONS

Licensees referencing TR NEDE-24011-P-A/NEDO-24011-A, for batch loading of GNF2 fuel assemblies must ensure compliance with the following conditions and limitations:

1. The GNF2 fuel assembly design is approved for [ ].

2. The NRC staff review and approval is limited to the zirconium barrier GNF2 fuel rod design.

3. The application of GESTAR II to the GNF2 fuel assembly design is approved for [ ].

5.0 CONCLUSION

Based upon its review described above, the NRC staff finds Amendment 32 to NEDE-24011-P-A/NEDO-24011-A, entitled “General Electric Standard Application for Reload Fuel (GESTAR II),” acceptable. Licensees referencing TR NEDE-24011-P-A/NEDO-24011-A need to comply with the conditions listed in Section 4.0 of this SE.

6.0 REFERENCES

1. Letter from A. Lingenfelter (GNF) to U.S. Nuclear Regulatory Commission, “Amendment 32 to NEDE-24011-P-A, General Electric Standard Application for Reactor Fuel (GESTAR II),” FLN-2008-011, October 15, 2008 (ADAMS Package Accession No. ML082910505).

2. Letter from A. Lingenfelter (GNF) to U.S. Nuclear Regulatory Commission, “GNF2 Advantage Generic Compliance with NEDE-24011-P-A (GESTAR II), NEDC-33270P, March 2007, and GEXL17 Correlation for GNF2 Fuel, NEDC-33292P, March 2007,” FLN-2007-011, March 14, 2007 (ADAMS Accession No. ML070780335).

3. NRC Memorandum, “Audit Report for GNF2 Advanced Fuel Assembly Design GESTAR II Compliance Report,” September 2008 (ADAMS Package Accession No. ML082690382).

4. Letter from J. Post (GEH) to U. S. Nuclear Regulatory Commission, “Part 21 Notification: Adequacy of GE Thermal-Mechanical Methodology, GESTR-M,” MFN 07-040, January 21, 2007 (ADAMS Package Accession No. ML072290203).

5. Letter from D. Porter (GEH) to U. S. Nuclear Regulatory Commission, “Part 21 Notification: Adequacy of GE Thermal-Mechanical Methodology, GESTR-M – Supplement 1,” MFN 07-040 Supplement 1, January 4, 2008 (ADAMS Accession No. ML080100670).

6. Letter from D. Porter (GEH) to U. S. Nuclear Regulatory Commission, “Part 21 Notification: Adequacy of GE Thermal-Mechanical Methodology, GESTR-M – Supplement 2,” MFN 07-040 Supplement 2, August 28, 2008 (ADAMS Package Accession No. ML082420309).

7. Letter from A. Lingenfelter (GNF) to U. S. Nuclear Regulatory Commission, “GNF2 Advantage Generic Compliance with NEDE-24011-P-A (GESTAR II), NEDC-33270P, Revision 1, August 2008,” FLN-2008-008, August 29, 2008 (ADAMS Accession No. ML082460763).

8. Letter from H. Nieh (USNRC) to A. Lingenfelter (GNF), “Final Safety Evaluation for Global Nuclear Fuel (GNF) Amendment 30 to Topical Report (TR) NEDE-24011P-A/NEDO-24011-A, ‘General Electric Standard Application for Reload Fuel (GESTAR II)’,” February 11, 2008 (ADAMS Accession No. ML080310007).

Principle Contributor: P. Clifford

Attachment: Comment Resolution Table

Date: July 30, 2009

Comment Resolution Table

Location	Comment	Resolution
Through out document	Correct GE Hitachi brand name.	Incorporated.
Page 3, Line 15	Add space between "to" and "NEDE-".	Incorporated.
Page 4, Line 16 and 32	The term NCLO used in these 2 locations should be Clad Lift-Off (CLO) or else change the sentence to reflect the negative context.	Changed NCLO to CLO.
Page 5, Line 11	The second sentence implies it should follow from the first, but it does not.	Reworded first sentence of Audit Finding #4.

NRC
Safety Evaluation Report
Approving Amendment 33
to
NEDE-24011-P

August 30, 2010

Mr. Andrew A. Lingenfelter, Manager
GNF Engineering
Global Nuclear Fuel - Americas, LLC
P.O. Box 780, M/C F12
Wilmington, NC 28402

SUBJECT: FINAL SAFETY EVALUATION FOR AMENDMENT 33 TO GLOBAL NUCLEAR FUEL TOPICAL REPORT NEDE-24011-P, "GENERAL ELECTRIC STANDARD APPLICATION FOR REACTOR FUEL (GESTAR II)" (TAC NO. ME3525)

Dear Mr. Lingenfelter:

By letter dated March 5, 2010, (Agencywide Documents and Access Management System (ADAMS) Accession No. ML100700432), as supplemented by letter dated May 27, 2010 (ADAMS Accession No. ML101481056), Global Nuclear Fuel - Americas, LLC (GNF) submitted Amendment 33 to Topical Report (TR) NEDE-24011-P, "General Electric Standard Application for Reactor Fuel (GESTAR II)" to the U. S. Nuclear Regulatory Commission (NRC) staff. By letter dated August 6, 2010 (ADAMS No. ML102140572), an NRC draft safety evaluation (SE) regarding our approval of Amendment 33 to NEDE-24011-P was provided for your review and comment. By letter dated August 9, 2010 (ADAMS No. ML102220340), GNF replied stating that it did not find any factual errors or clarity concerns in the draft SE and provided a non-proprietary version of the draft SE to indicate the information that GNF considers proprietary to GNF.

The NRC staff has found that Amendment 33 to NEDE-24011-P is acceptable for referencing in licensing applications for GNF-designed fuel for boiling water reactors to the extent specified and under the limitations delineated in the TR and in the enclosed final SE. The final SE defines the basis for acceptance of the TR.

Our acceptance applies only to material provided in the subject TR. We do not intend to repeat our review of the acceptable material described in the TR. When the TR appears as a reference in license applications, our review will ensure that the material presented applies to the specific plant involved. License amendment requests that deviate from this TR will be subject to a plant-specific review in accordance with applicable review standards.

In accordance with the guidance provided on the NRC website, we request that GNF publish accepted proprietary and non-proprietary versions of this TR within three months of receipt of this letter. The accepted versions shall incorporate this letter and the enclosed final SE after the title page. Also, they must contain historical review information, including NRC requests for additional information and your responses. The accepted versions shall include an "-A" (designating accepted) following the TR identification symbol.

Document transmitted herewith contains proprietary information. When separated from enclosures, this document is decontrolled.

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As an alternative to including the RAIs and RAI responses behind the title page, if changes to the TR were provided to the NRC staff to support the resolution of RAI responses, and the NRC staff reviewed and approved those changes as described in the RAI responses, there are two ways that the accepted version can capture the RAIs:

1. The RAIs and RAI responses can be included as an Appendix to the accepted version.
2. The RAIs and RAI responses can be captured in the form of a table (inserted after the final SE) which summarizes the changes as shown in the approved version of the TR. The table should reference the specific RAIs and RAI responses which resulted in any changes, as shown in the accepted version of the TR.

If future changes to the NRC's regulatory requirements affect the acceptability of this TR, GNF and/or licensees referencing it will be expected to revise the TR appropriately, or justify its continued applicability for subsequent referencing.

Sincerely,

/RA/

Thomas B. Blount, Deputy Director
Division of Policy and Rulemaking
Office of Nuclear Reactor Regulation

Project No. 712

Enclosures:
1. Proprietary version of the Final SE
2. Non-proprietary version of the Final SE

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/RA/

Thomas B. Blount, Deputy Director
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ADAMS ACCESSION NOs.:
Package:
Non-Prop Final SE: ML102280591
Cover letter: ML102280144
Prop Final SE: ML102280588
***No major changes to SE input. NRR-043**

OFFICE	PLPB/PM	PLPB/PM	PLPB/LA	SNPB/BC*	PLPB/BC	DPR/DD
NAME	SPhilpott	JRowley	EHylton	AMendiola	JJolicoeur	TBlount
DATE	8/17/10	8/20/10	8/23/10	8/25/10	8/24/10	8/30/10

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Global Nuclear Fuel Project No. 712

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FINAL SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

AMENDMENT 33 TO TOPICAL REPORT NEDE-24011-P-A / NEDO-24011-A

"GENERAL ELECTRIC STANDARD APPLICATION FOR REACTOR FUEL (GESTAR II)"

GLOBAL NUCLEAR FUEL

PROJECT NO. 712

1.0 INTRODUCTION AND BACKGROUND

By letter dated March 5, 2010 (Reference 1), as supplemented by letter dated May 27, 2010 (Reference 2), Global Nuclear Fuel (GNF) submitted Amendment 33 to topical report (TR) NEDE-24011-P, "General Electric Standard Application for Reactor Fuel (GESTAR II)." GESTAR II provides a fuel design and core reload process used extensively by licensees with GNF or GE Hitachi Nuclear Energy Americas (GEH) fuel designs. This U.S. Nuclear Regulatory Commission (NRC)-approved process allows GNF to modify fuel assembly designs without undergoing a formal NRC submittal and review. As part of this process, GNF provides written notification outlining the new design and acknowledging compliance with the requirements of GESTAR II. Upon notification, the NRC staff may conduct an audit of the engineering calculations supporting the new fuel design. Amendment 32 to GESTAR II (Reference 3) was necessitated by an NRC staff audit of the GNF2 fuel design GESTAR II Compliance Report (Reference 4). Amendment 33 resolves remaining NRC staff concerns from its review of Amendment 32, incorporates the recently approved PRIME fuel rod thermal-mechanical (T-M) methods (Reference 5), and removes a more restrictive exposure limit imposed by Amendment 32 and the corresponding NRC staff safety evaluation (SE) (Reference 6) for the GNF2 fuel design.

2.0 REGULATORY EVALUATION

TR NEDE-24011-P-A/NEDO-24011-A provides an approved fuel design and core reload process. The approved methodology and acceptance criteria detailed within TR NEDE-24011 are cited within many boiling water reactor (BWR) technical specifications as references in the core operating limits reports.

Regulatory guidance for the review of fuel rod cladding materials and fuel system designs and adherence to Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50, Appendix A, General Design Criteria (GDC)-10 "Reactor Design," GDC-27 "Combined Reactivity Control Systems Capability," and GDC-35 "Emergency Core Cooling" is provided in NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants" (SRP), Section 4.2, "Fuel System Design" (Reference 7). In accordance with

ENCLOSURE 2

SRP Section 4.2, the objectives of the fuel system safety review are to provide assurance that:

- The fuel system is not damaged as a result of normal operation and anticipated operational occurrences (AOOs),
- Fuel system damage is never so severe as to prevent control rod insertion when it is required,
- The number of fuel rod failures is not underestimated for postulated accidents, and
- Coolability is always maintained.

GESTAR II provides a licensed reload methodology, including an NRC-approved fuel T-M design methodology utilized to demonstrate compliance with the fuel design criteria in SRP Section 4.2. The NRC staff reviewed Amendment 33 to GESTAR II to confirm that the fuel design criteria in SRP Section 4.2 will continue to be satisfied with the changes introduced by this amendment.

3.0 TECHNICAL EVALUATION

Building on the NRC staff's review of the Economic Simplified BWR (ESBWR) GE14E fuel design, evaluations of GEH notifications pursuant to 10 CFR Part 21 related to General Electric Stress and Thermal Analysis of Reactor Rods - Mechanical (GESTR-M) (References 8, 9, and 10), and the GNF2 GESTAR II Compliance Report (References 4 and 11), GESTAR II Amendment 32 (Reference 3) and the corresponding NRC staff SE (Reference 6) imposed an exposure limitation for the GNF2 fuel design. This SE will repeat, and then build upon, several findings from the NRC staff's audit report (Reference 12) for the GNF2 GESTAR II Compliance Report and the corresponding sections of the NRC staff's SE for Amendment 32. Enclosure 1 of Reference 1 documents GNF's response to NRC staff concerns and limitations resulting from its audit of the GNF2 fuel design and its review of Amendment 32.

Audit Finding #1 (Reference 12):

> Based on limited lead use assembly (LUA) operating history and the lack of a post-irradiation examination (PIE) to validate in-reactor performance up to end-of-life (EOL) exposure, GEH has neither met the intent of the GESTAR-II LUA requirement, nor satisfied established regulatory practice.

Amendment 32 SE (Reference 6):

> In its response, GEH states that the LUA program for GNF2 "…is completely consistent with GESTAR II and with the long history of LUAs applied under GESTAR II." Further, GEH states that the "…evolutionary changes from an experience base of over 26,000 GE14 and GE12 bundles, not warranting more extensive LUA exposure or examinations."

> The NRC staff does not accept this position and expects further in-reactor experience and inspection prior to batch application.
>
> GEH acknowledges that continued inspections at interim exposures are planned and will reveal any unanticipated behavior well before GNF2 reload bundles reach similar exposures. In addition, the exposure of GNF2 LUAs will always lead the reloads by a substantial margin.
>
> Approval of the application of GESTAR II and use of GNF2 fuel for the [ ] is supported by the limited LUA operating experience documented in the compliance report and in Amendment 32. Extension [ ] requires further LUA operating experience and inspection along with NRC review and approval.

Amendment 33 SE:

In Enclosure 1 of Reference 1, GNF provided the details of additional LUA operating experience and inspections beyond that reported in Amendment 32. No indications of fretting wear or unusual corrosion were observed during these inspections while compiling this more extensive LUA inspection database. Based upon the additional LUA operating experience and inspection results, the NRC staff finds that GNF has met the requirements of GESTAR II and that the more restrictive GNF2-specific exposure limit may be removed.

Audit Findings #2, #6, #7, #8, and #9 (Reference 12):

All findings are related to adequacy of GESTR-M methods above [ ] and GNF2 No Clad Liftoff (NCLO) rod internal pressure design calculations.

Amendment 32 SE (Reference 6):

> In addition to the detailed information provided by GEH, the NRC staff has performed independent calculations using the FRAPCON-3 fuel thermal-mechanical model. The NRC staff's calculations are documented in Table 3 of the audit report (Reference [12]). The FRAPCON-3 calculations confirm that the GNF2 fuel rod design satisfies all thermal-mechanical design criteria except NCLO rod internal pressure criteria. Independent calculations reveal that the NCLO criteria (cladding creep outward, fuel pellet/cladding gap opening) are violated prior to the approved EOL [ ].
>
> The FRAPCON-3 calculations confirm earlier concerns regarding the adequacy of GSTR-M at higher burnup. Specifically, GSTR-M calculations under predict UO_2 fuel temperature, which results in an under prediction of fission gas release and rod internal pressure. Hence, GSTR-M calculations do not predict clad liftoff (CLO) for the GNF2 fuel rod design.
>
> In Item 2 of Table 1 (Reference 1), GNF states that GE11 fuel rods were licensed at [ ]. Based upon the concerns discussed above, this would bring into question the adequacy of the GSTR-M calculations for these higher power fuel rods. During a past audit, the NRC staff discussed the impact of the GSTR-M 10 CFR Part 21 concerns on the

GE11/13 rod designs. Crediting the larger fuel rod plenum region of the GE11/13 (relative to the GE14 design), GEH provided sample rod internal pressure calculations, which demonstrate significant pressure margin to the NCLO criteria. GEH stated that millions of GE11/13 rods have operated to design exposures with no indications of problems due to high internal rod pressure. This design is now being phased out in BWR/3-6 reactors, but is still being supplied to BWR/2 reactors. However, its application to BWR/2 reactors is limited to a peak linear heat generation rate [ ] due to the loss-of-coolant accident response characteristics for these reactors. The NRC staff accepts the disposition of this issue for GE11/13 fuel rods designs.

The NRC staff's independent calculations predict CLO of the GNF2 fuel rod design, but at an exposure [ ]. Based upon the GEH thermal-mechanical analyses and the NRC staff's independent calculations, the NRC staff finds the application of GSTR-M to GNF2 fuel acceptable up to a peak pellet exposure of [ ]. Extension [ ] requires further justification. This may involve using an approved PRIME methodology and/or a modified GNF2 fuel rod design. NRC review and approval is required to [ ].

Amendment 33 SE:

The incorporation of the approved PRIME methodology into GESTAR II and its application to GNF2 fuel designs addresses previous concerns with GESTR-M. In addition to other enhancements, the PRIME fuel thermal conductivity model has been benchmarked against an extensive database and accurately predicts thermal conductivity degradation with increasing fuel burnup.

As indicated above, from the NRC staff's SE for Amendment 32, independent NRC staff calculations with FRAPCON-3 confirmed that the GNF2 fuel rod designs satisfy all T-M design criteria except NCLO rod internal pressure criteria. The NRC staff repeated the limiting rod internal pressure cases (full length, UO_2 and [ ] Gadolinium rod designs) using the revised linear heat generation rate (LHGR) envelopes provided in Appendix B to the revised GNF2 GESTAR II Compliance Report (Enclosure 7 of Reference 1). Consistent with earlier audit calculations, the NRC staff deterministically applied worst case manufacturing tolerances (to maximize fuel temperature and fission gas release and minimize void volume) and applied a 10 percent rod power penalty in lieu of modeling uncertainties.

Examination of the revised GNF2 LHGR power envelopes revealed that the UO_2 rod power limit is [ ] at beginning of life. These changes appear to be prompted by differences between GESTR-M and PRIME fuel thermal conductivity models and the resulting calculations of fuel temperature and fission gas release. Independent NRC staff calculations confirmed that rod internal pressure remained below the critical value which would produce an outward creep of the cladding and re-open the clad-to-fuel pellet gap (i.e., cladding liftoff) at these new rod power envelopes. Unlike the earlier audit calculations, the calculated rod internal pressure is

[ ]

The modifications to GESTAR II proposed in Amendment 33 include the addition of the recently approved PRIME fuel rod T-M methodology, along with the following implementation plan.

GEH Commitment to Transition from GESTR-M to PRIME (Reference 2):

> GNF will transition from the GESTR-M to the PRIME Thermal-Mechanical (T-M) methodology basis as quickly as practical. Beginning with the GNF2 fuel product line, the fuel T-M design will use the PRIME methodology. Fuel products preceding GE14 (e.g., GE11 and GE12), which are currently operating, may continue to use the GESTR-M basis. GNF is no longer loading these older fuel products, but some may remain in operating plants for several more cycles. GNF will implement the PRIME T-M basis for the GE14 fuel product line, including GE14 currently in operation, in the reload workscope for new fuel cycle designs initiated following the completion of the downstream codes implementation, which is anticipated in early 2011. The GE14 GESTAR II Compliance Report will be amended to include PRIME based T-M limits that include consideration of the revised design criteria in Amendment 33.

The NRC staff finds the inclusion of PRIME in GESTAR II and the transition plan described above acceptable. Based upon this migration to PRIME and independent calculations, the NRC staff finds that the more restrictive GNF2-specific exposure limit may be removed.

<u>Audit Finding #3 (Reference 12):</u>

> The GNF design continues to use the [ ] design criteria. While this approach is consistent with GESTAR II, it does not address issues identified by the NRC staff during the Economic Simplified BWR (ESBWR) review. Note that GEH plans to revise the fuel rod cladding strain design criteria for the ESBWR fuel design (GE14E). GEH needs to demonstrate, via empirical data, that GNF2 fuel rod cladding is capable of achieving the [ ] at EOL conditions or revisit the criterion.

Amendment 32 SE (Reference 6):

> This item is discussed in Supplement 2 of the Part 21 Notification (Reference [10]). GEH states that the exposure-dependent strain limits proposed for ESBWR are consistent with the analyzed strain criteria [ ]. The NRC staff agrees with this statement. However, the GNF2 Alloy X-750 grid [spacers] provide an additional source of hydrogen pickup for the fuel rod cladding which must be considered when setting the exposure-dependent breakpoint. Extension [ ] requires further justification for the exposure-dependent strain limits for GNF2 and NRC review and approval.

Amendment 33 SE:

Section 2.2.2.7 of GESTAR II Amendment 33 (Reference 1) originally indicated that the revised, hydrogen (burnup)-dependent cladding strain failure criteria and associated corrosion limits would apply to future fuel designs beginning with GNF2. Based on NRC staff concerns, Amendment 33 was revised to indicate that these revised criteria and corrosion limits apply to all fuel designs (Reference 2). The basis for the revised hydrogen (burnup)-dependent cladding strain criteria and corrosion limits (detailed in Reference 13) was previously approved by the NRC staff as part of its review of the ESBWR GE14E fuel design.

Attachment 2 of Enclosure 1 ("Amendment 32 Safety Evaluation Follow-on Items and GNF Response") of the GESTAR II Amendment 33 submittal (Reference 1) provides results of visual inspections, pool-side corrosion measurements, and hot-cell examinations to support the application of the new corrosion limits to the GNF2 fuel design with its Alloy X-750 grid spacers. Section 3.2.10 of the revised GNF2 GESTAR II Compliance Report (NEDC-33270P, Revision 3 – Enclosure 7 of Reference 1) details the conformance of the GNF2 fuel rod designs to the revised cladding strain limits. The NRC staff completed independent FRAPCON-3 calculations as part of the original GNF2 compliance audit which confirmed the calculated cladding strain and mechanical overpower limits.

Based upon the corrosion data provided in Reference 1, the NRC staff finds the introduction of the previously approved revised cladding strain criteria and corrosion limits in GESTAR II and their application to the GNF2 fuel design acceptable. As such, the NRC staff finds that the more restrictive GNF2-specific exposure limit may be removed.

<u>Audit Finding #4 (Reference 12):</u>

> The GNF2 fuel rod design needs to include limits for cladding corrosion. While this approach is consistent with GESTAR-II, corrosion limits are required to ensure that key assumptions related to fuel performance analyses remain applicable. Specifically, an upper limit on local cladding oxidation (corresponding to oxide spallation) and an upper limit on local cladding hydrogen content (corresponding to the strain limit) need to be provided.

Amendment 32 SE (Reference 6):

> This item is being addressed for the GE14E fuel assembly design in the ESBWR design review. It is anticipated that a similar approach will be pursued for GNF2. Since cladding corrosion is expected to be low [
>
>]. Extension
> [ ] requires further justification, established corrosion limits, and NRC review and approval.

Amendment 33 SE:

This item was addressed above as part of audit finding #3.

Audit Finding #5 (Reference 12):

> The GNF2 design maintains an allowance for fuel centerline melting during local AOOs. While this approach is consistent with GESTAR II, little data is available to validate fuel swelling models at melting conditions, especially for higher burnup fuel. In addition, little data is available to validate fuel performance models for future operation with fuel rods which have previously undergone melting. If GEH desires to maintain this approach, then validation of these models against measured data should be included in the ongoing PRIME review.

Amendment 32 SE (Reference 6):

> In its response, GEH states that the GSTR-M application methodology is such that melting during local AOOs is precluded for any fuel design and that current reloads do not utilize the GESTAR II allowance for limited fuel melting. The NRC staff considers this issue resolved for GNF2 fuel.

Amendment 33 SE:

Section 2.2.2.5 of GESTAR II Amendment 33 (Reference 2) stipulates that fuel centerline temperature during normal operation and AOOs does not exceed melting temperature. Section 3.2.9 of the revised GNF2 GESTAR II Compliance Report (NEDC-33270P, Revision 3 – Enclosure 7 of Reference 1) details the conformance of the GNF2 fuel rod designs to this fuel centerline melting criteria. The NRC staff completed independent FRAPCON-3 calculations as part of the original GNF2 compliance audit which confirmed the calculated fuel temperature and thermal overpower limits.

The NRC staff finds the introduction of the revised fuel melting strain criteria in GESTAR II and their application to the GNF2 fuel design acceptable.

Audit Finding #10 (Reference 12): Open Items

Audit finding #10 referred to five open items concerning additional detail needed by the NRC staff to complete its review of the GNF2 fuel assembly design evaluations. Amendment 32 provided responses to the open items identified in the NRC staff's GNF2 audit. In Enclosure 1 of Reference 1, GNF provided additional information related to these open items.

The first open item requested information related to GNF2 channel design's susceptibility to shadow corrosion induced channel bow. This item was addressed in Amendment 32 and the NRC staff's corresponding SE. The NRC staff considers this item closed.

Amendment 32 SE (Reference 6):

> In a second open item, the NRC staff requested information related to the effect of GNF2 design features on flow induced vibration. In its response, GEH stated that there were no known occurrences of grid to rod fretting failures in GNF BWR fuel designs over several

decades of deployment. To date, inspections on GNF2 LUAs have shown no abnormal indications near grid locations. The NRC staff finds the application of GESTAR II and use of GNF2 fuel for the [ ] acceptable based on the limited LUA operating experience (especially fuel rod wear inspections under grid straps). Extension beyond the [ ] requires further justification that assembly design features (e.g., introduction of mixing vanes) do not introduce fuel rod vibration and the potential for grid-to-rod fretting and NRC review and approval.

Amendment 33 SE:

In Enclosure 1 of Reference 1, GNF details additional LUA operating experience and inspections beyond that reported in Amendment 32. Individual fuel rods were removed and inspected, including visual inspection adjacent to grid strap locations. No indications of fretting wear or unusual corrosion were observed while compiling this more-extensive LUA inspection database. Based upon the additional LUA operating experience and inspections, the NRC staff finds that GNF has met the requirements of GESTAR II and that the more restrictive GNF2-specific exposure limit may be removed.

Amendment 32 SE (Reference 6):

In a third open item, the NRC staff requested information related to the inclusion of water holes in the water rod structural analysis. In its response, GEH concluded that while the water rod holes were not explicitly modeled in the finite element analysis (FEA), the amount of conservatism in the structural calculations assuming all loads are applied at the minimum water rod diameter offsets the reduced cross-sectional effect of both sets of water rod holes. During a recent ESBWR audit, the NRC staff questioned similar engineering judgments for the GE14E fuel design. GEH, following its corrective action program, is performing detailed FEA calculations (modeling the water rod holes) to investigate its conclusion. The GNF2 fuel design does not introduce any new design features which exacerbate this potential problem. As such, the NRC staff considers this issue to be generic in scope and not specific to its approval of Amendment 32 or the GNF2 fuel design.

Amendment 33 SE:

Section 3.2.1 of the GNF2 compliance report (NEDC-33270P, Revision 3 – Enclosure 7 of Reference 1) and Enclosure 1 of Reference 1 include a description of the detailed ANSYS FEA calculations (with explicit modeling of the water holes) which demonstrate that the GNF2 water rod will not buckle under the maximum handling and shipping loads. The NRC staff agrees with this conclusion and considers this item closed.

Amendment 32 SE (Reference 6):

In a fourth open item, the NRC staff requested information related to the applicability of power ramp test results to GNF2 fuel. In its response, GEH stated that current GNF2 fuel designs have the standard barrier cladding design. Historically, the inclusion of the zirconium barrier has been an effective method on minimizing vulnerability to pellet

cladding interaction (PCI)/stress corrosion cracking (SCC). A comparison of power ramp test results with barrier cladding (Figure 1 of Reference 1) shows that PCI/SCC failure would not be expected at or below the GNF2 rod power envelope. In its response, GEH provides a discussion of the applicability of the power ramp test results to the GNF2 design. GEH states that the local cladding stresses are driven by the change in local power (and resulting pellet strain) and independent of rod diameter and cladding thickness. One item not discussed is the initial pellet-to-cladding gap size between the older test rods and GNF2. For a given power change, initial gap size will impact cladding stresses. This item requires further investigation prior to removing the [ ] limit.

The GNF2 design includes a non-barrier option. Due to the limited scope of this review and schedule restrictions, the NRC staff was unable to reach a safety finding with respect to the acceptability of a non-barrier GNF2 fuel rod design. Hence, the [NRC] staff's approval of Amendment 32 for GNF2 is limited to the zirconium barrier fuel rod design.

Amendment 33 SE:

Attachment 1 to Enclosure 1 of Reference 1 provides justification for the continued applicability of earlier power ramp test data to the GNF2 fuel design, including consideration of the potential effects of differences in initial cladding-to-fuel pellet gap size. GNF concluded that, despite differences in fuel rod diameter, cladding thickness, pellet diameter, and initial gap size, the ramp test data is applicable to GNF2 fuel rod designs.

To assess the validity of the GNF conclusion, the NRC staff performed independent calculations with FRAPCON-3 simulating power ramps on both 8x8 and 10x10 fuel rod designs. The results indicate that the smaller gap size closes faster and calculated cladding strain is slightly larger for the 10x10 design for an identical power ramp. However, these differences were not significant. Further evaluations may be necessary to assess the applicability of the power ramp test data to future fuel designs that exhibit more substantial differences in design specifications and whether it would be necessary to scale test data and consider the impact of these differences on power maneuvering restrictions and predicted fuel rod performance during AOOs.

The zirconium barrier design has proven to reduce PCI/SCC susceptibility both during in-reactor operations and during power ramp testing. As documented in Enclosure 1 of Reference 1, GNF has not provided additional information to support the non-barrier option and accepts the continued application of the Amendment 32 limitation (i.e., the NRC staff review and approval of GESTAR II is limited to the zirconium barrier GNF2 fuel rod design.). With the continued inclusion of the barrier design, the NRC staff's concerns related to slight differences in predicted cladding stress and strain and the applicability of the power ramp test data is diminished. Non-barrier fuel rod designs or future designs with more substantial differences (e.g., fuel rod specifications, pellet composition, barrier alloy) may require a more detailed assessment.

Based upon the review detailed above, the NRC staff finds that the more restrictive GNF2-specific exposure limit may be removed.

Amendment 32 SE (Reference 6):

> In a fifth open item, the NRC staff requested information related to local cladding hydrogen concentration near the Alloy X-750 grid spacers. In its response, GEH concludes that the performance of GNF2 will not be adversely affected by shadow corrosion and hydriding at spacer locations, especially given the rod exposure limit. Based upon anticipated corrosion (and hydrogen pickup) during the limited rod exposure, the NRC staff finds this response acceptable. However, further data needs to be provided to justify extended [ ].

Amendment 33 SE:

This item was addressed in a previous section of this SE (see audit finding #3 above).

Amendment 32 imposed a limitation of [ ] on the approval of the GNF2 fuel assembly design and on the application of GESTAR II to the GNF2 fuel assembly design. This limitation was prompted by several issues, including the lack of LUA data, NRC staff concerns with the GESTR-M thermal conductivity model and UO_2 rod internal pressure, and necessary updates to cladding strain criteria and corrosion limits. As described above, all of these issues have been dispositioned to the satisfaction of the NRC staff. As such, the cycle and exposure limitation is no longer necessary. The approved burnup limit for the GNF2 fuel assembly design is [ ].

4.0 <u>LIMITATIONS AND CONDITIONS</u>

There are no additional limitations and conditions associated with Amendment 33 to GESTAR II. Licensees referencing NEDE-24011-P (GESTAR II) must continue to comply with all previous NRC limitations and conditions except for the Amendment 32 limitations (#1 and #3) associated with [ ] for batch loading of GNF2 fuel assemblies.

5.0 <u>CONCLUSION</u>

Based upon its review described above, the NRC staff finds Amendment 33 to NEDE-24011-P-A / NEDO-24011-A, "General Electric Standard Application for Reactor Fuel (GESTAR II)," acceptable. Licensees referencing GESTAR II need to comply with the conditions listed in Section 4.0 of this SE.

11

6.0 REFERENCES

1. Letter from GNF to NRC, MFN 10-045, "Amendment 33 to NEDE-24011-P-A, General Electric Standard Application for Reactor Fuel (GESTAR II) and GNF2 Advantage Generic Compliance with NEDE-24011-P-A (GESTAR II), NEDC-33270P, Revision 3, March 2010," dated March 5, 2010. (ADAMS Package Accession Number ML100700464)

2. Letter GNF to NRC, MFN 10-045 Supplement 1, "Amendment 33 to NEDE-24011-P, General Electric Standard Application for Reactor Fuel (GESTAR II)," dated May 27, 2010. (ADAMS Package Accession Number ML101481067)

3. Letter from GNF to NRC, FLN-2008-011, "Amendment 32 to NEDE-24011-P-A, General Electric Standard Application for Reactor Fuel (GESTAR II)," dated October 15, 2008. (ADAMS Package Accession No. ML082910505)

4. Letter from GNF to NRC, FLN-2007-011, "GNF2 Advantage Generic Compliance with NEDE-24011-P-A (GESTAR II), NEDC-33270P, March 2007, and GEXL17 Correlation for GNF2 Fuel, NEDC-33292P, March 2007," dated March 14, 2007. (ADAMS Accession No. ML070780335)

5. Final SE for GNF TR NEDC-33256P, NEDC-33257P, and NEDC-33258P, "The PRIME Model for Analysis of Fuel Rod Thermal-Mechanical Performance," dated January 22, 2010. (ADAMS Package Accession No. ML100210284)

6. Final SE for GNF Amendment 32 to TR NEDE-24011-P-A / NEDO-24011-A, "General Electric Standard Application for Reactor Fuel (GESTAR II)," dated July 30, 2009. (ADAMS Package Accession No. ML091680754)

7. NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants," Section 4.2, Revision 3, "Fuel System Design," dated March 2007. (ADAMS Accession No. ML070740002)

8. Letter from GEH to NRC, MFN 07-040, "Part 21 Notification: Adequacy of GE Thermal-Mechanical Methodology, GESTR-M," dated January 21, 2007. (ADAMS Package Accession No. ML072290203)

9. Letter from GEH to NRC, MFN 07-040 Supplement 1, "Part 21 Notification: Adequacy of GE Thermal-Mechanical Methodology, GESTR-M – Supplement 1," dated January 4, 2008. (ADAMS Accession No. ML080100670)

10. Letter from GEH to NRC, MFN 07-040 Supplement 2, "Part 21 Notification: Adequacy of GE Thermal-Mechanical Methodology, GESTR-M – Supplement 2," dated August 28, 2008. (ADAMS Package Accession No. ML082420309)

11. Letter from GNF to NRC, FLN-2008-008, “GNF2 Advantage Generic Compliance with NEDE-24011-P-A (GESTAR II), NEDC-33270P, Revision 1, August 2008,” dated August 29, 2008. (ADAMS Accession No. ML082460763)

12. NRC Memorandum, “Audit Report for GNF2 Advanced Fuel Assembly Design GESTAR II Compliance Report,” dated September 25, 2008. (ADAMS Package Accession No. ML082690382)

13. Letter from GEH to NRC, MFN 08-347, “Response to Portion of NRC Request for Additional Information Letter No. 110 - Related to ESBWR Design Certification Application - RAI Numbers 4.2-2 Supplement 3, 4.2-4 Supplement 2 and 4.8-6 Supplement 1,” dated May 9, 2008. (ADAMS Package Accession No. ML081350404)

Principal Contributor: Paul M. Clifford

Date: August 30, 2010

NRC
Safety Evaluation Report
Approving Amendment 34
to
NEDE-24011-P

April 20, 2011

Mr. Andrew A. Lingenfelter
Vice President, Fuel Engineering
Global Nuclear Fuel–Americas, LLC
P.O. Box 780, M/C A-55
Wilmington, NC 28401

SUBJECT: FINAL SAFETY EVALUATION FOR AMENDMENT 34 TO GLOBAL NUCLEAR FUEL - AMERICAS TOPICAL REPORT NEDE-24011P-A/NEDO-24011-A, "GENERAL ELECTRIC STANDARD APPLICATION FOR REACTOR FUEL (GESTAR II)" (TAC NO. ME3594)

Dear Mr. Lingenfelter:

By letter dated February 9, 2010 (Agencywide Documents and Access Management System (ADAMS) Package Accession No. ML100491466), and subsequently revised by letter dated March 8, 2011 (ADAMS Accession No. ML110680170), Global Nuclear Fuel – Americas, LLC (GNF) submitted Amendment 34 to Topical Report (TR) NEDE-24011P-A/NEDO-24011-A, entitled "General Electric Standard Application for Reactor Fuel (GESTAR II)" to the U.S. Nuclear Regulatory Commission (NRC) staff for review.

The NRC staff has found that Amendment 34 to GESTAR II is acceptable for referencing in licensing applications for General Electric-designed boiling water reactors to the extent specified in the enclosed final safety evaluation (SE). The final SE defines the basis for acceptance of the TR.

Our acceptance applies only to material provided in the subject TR. We do not intend to repeat our review of the acceptable material described in the TR. When the TR appears as a reference in license applications, our review will ensure that the material presented applies to the specific plant involved. License amendment requests that deviate from this TR will be subject to a plant-specific review in accordance with applicable review standards.

In accordance with the guidance provided on the NRC website, we request that GNF publish accepted proprietary and non-proprietary versions of this TR within three months of receipt of this letter. The accepted versions shall incorporate this letter and the enclosed final SE after the title page. Also, they must contain historical review information, including NRC requests for additional information and your responses. The accepted versions shall include a "-A" (designating accepted) following the TR identification symbol.

If future changes to the NRC's regulatory requirements affect the acceptability of this TR, GNF and/or licensees referencing it will be expected to revise the TR appropriately, or justify its continued applicability for subsequent referencing.

Sincerely,

Thomas B. Blount, Deputy Director
Division of Policy and Rulemaking
Office of Nuclear Reactor Regulation

Project No. 712

Enclosure:
Final SE

cc w/encl: See next page

A. Lingenfelter - 2 -

If future changes to the NRC's regulatory requirements affect the acceptability of this TR, GNF and/or licensees referencing it will be expected to revise the TR appropriately, or justify its continued applicability for subsequent referencing.

Sincerely,

Thomas B. Blount, Deputy Director
Division of Policy and Rulemaking
Office of Nuclear Reactor Regulation

Project No. 712

Enclosure:
Final SE

cc w/encl: See next page

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ADAMS ACCESSION NO.: ML111010231 **NRR-043**

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SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

AMENDMENT 34 TO TOPICAL REPORT NEDE-24011P-A/NEDO-24011-A

"GENERAL ELECTRIC STANDARD APPLICATION FOR REACTOR FUEL (GESTAR II)"

GLOBAL NUCLEAR FUEL – AMERICAS, LLC

PROJECT NO. 712

1.0 INTRODUCTION AND BACKGROUND

By letter dated February 9, 2010 (Reference 1), and subsequently revised by letter dated March 8, 2011 (Reference 2), Global Nuclear Fuel – Americas, LLC (GNF) submitted Amendment 34 to Topical Report (TR) NEDE-24011-P-A/NEDO-24011-A, entitled "General Electric Standard Application for Reactor Fuel (GESTAR II)" to the U.S. Nuclear Regulatory Commission (NRC) staff for review. The changes in Amendment 34 are administrative in nature and include an updated revision reference, the addition of an approved operating flexibility option for boiling water reactors (BWRs), and a figure to illustrate BWR operating domains.

2.0 EVALUATION

The NRC staff has evaluated each of the proposed changes below.

Updated Reference 4-40 of GESTAR II Main

Reference 4-40 of GESTAR II Main has been updated to include the accepted (-A) versions of TRs NEDE–32906P, Supplement 1-A, "TRACG Application for Anticipated Transients without Scram [(ATWS)]Overpressure Transient Analyses," November 2003, and NEDE-32906P, Supplement 3-A, "Migration to TRACG04/PANAC11 from TRACG02/PANAC10 for TRACG AOO [(anticipated operational occurrence)] and ATWS Overpressure Transients," Revision 1, April 2010. These TRs have been previously reviewed and approved (References 3 and 4). Reference 4-40 was also updated to reflect the latest approved revision of NEDE–32906P–A, "TRACG Application for Anticipated Operational Occurrences Transient Analyses" (Revision 3). NEDE–32906P–A, Revision 3, was previously reviewed and approved by the NRC staff (Reference 5). The NRC staff agrees that updating this reference is an administrative change and, therefore, finds it acceptable.

Add Figure S-5 - Power-Flow Operating Domain Illustration in GESTAR II U.S. Supplement

This amendment to GESTAR II also incorporates an illustrative power-flow domain figure showing the changes that have taken place over many years relative to the original BWR

power-flow map. The NRC staff requested that GNF include this figure as an update to the TR to provide clarity to future NRC staff reviewers with regard to the various BWR operating domains when referencing this TR. The added figure is not specific to any particular BWR type and is included only for general information. The NRC staff has reviewed this figure and agrees that including it for general information is acceptable.

Add Reference to TR NEDC-33006P-A, Revision 3 to Sections S.5.2 and S.5.2.13

This change revises Section S.5.2 and adds Section S.5.2.13 to include reference to the approved TR NEDC-33006P-A, Revision 3, "General Electric Boiling Water Reactor Maximum Extended Load Line Limit Analysis Plus" (MELLLA+). Section S.5.2.13 provides a brief description of MELLLA+ as an available operating flexibility option for BWRs. The NRC staff has reviewed and agrees with this general description of MELLLA+. This amendment also adds the TR as Reference S-107 in Section S.6 of GESTAR II. This TR has been previously reviewed and approved (Reference 6). The NRC staff agrees that adding this reference and general description of MELLLA+ is an administrative change and, therefore, finds it acceptable.

3.0 CONCLUSION

Based on the above, the NRC staff finds that the changes proposed in Amendment 34 to TR NEDE 24011-P-A/NEDO-24011-A are administrative in nature and, therefore, are acceptable and appropriate for inclusion in NEDE 24011-P-A/NEDO-24011-A.

4.0 REFERENCES

1. Letter from GNF to NRC, MFN 10-051, "Administrative Amendment 34 to GESTAR II to Implement the Referencing of NEDC-33006P, Maximum Extended Load Line Limit Analysis Plus and NEDE-32906P, Supplement 3-A, Migration to TRACG04 / PANAC11 from TRACG02 / PANAC10 for TRACG AOO and ATWS Overpressure Transients," dated February 9, 2010. (ADAMS Package Accession No. ML100491466)

2. Letter from GNF to NRC, MFN 10-051 Revision 1, "Revision to Administrative Amendment 34 to GESTAR II to Implement the Referencing of NEDC-33006P, Maximum Extended Load Line Limit Analysis Plus and NEDE-32906P, Supplement 3-A, Migration to TRACG04 / PANAC11 from TRACG02/PANAC10 for TRACG AOO and ATWS Overpressure Transients," dated March 8, 2011. (ADAMS Accession No. ML110680170)

3. GNF TR NEDE–32906P, Supplement 1-A, "TRACG Application for Anticipated Transients without Scram Overpressure Transient Analyses," dated November 2003. (ADAMS Package Accession No. ML033381073)

4. GNF TR NEDE-32906P, Supplement 3-A, "Migration to TRACG04 / PANAC11 from TRACG02 / PANAC10 for TRACG AOO and ATWS Overpressure Transients," Revision 1, April 2010. (ADAMS Package Accession No. ML110970401)

5. GNF TR NEDE-32906P-A Revision 3," TRACG Application for Anticipated Operational Occurrences Transient Analyses," dated March 2006. (ADAMS Package Accession

No. ML062720163)

6. GNF TR NEDC-33006P-A, “General Electric Boiling Water Reactor Maximum Extended Load Line Limit Analysis Plus,” dated June 2009. (ADAMS Package Accession No. ML091800530)

Principal Contributor: Stephen Philpott

Date:

Global Nuclear Fuel

Project No. 712

cc:

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NRC
Safety Evaluation Report
Approving Amendment 35
to
NEDE-24011-P

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

May 14, 2012

Mr. Andrew A. Lingenfelter
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Global Nuclear Fuel–Americas, LLC
P.O. Box 780, M/C A-55
Wilmington, NC 28401

SUBJECT: FINAL SAFETY EVALUATION FOR AMENDMENT 35 TO GLOBAL NUCLEAR FUEL - AMERICAS TOPICAL REPORT NEDE-24011-P-A/NEDO-24011-A, "GENERAL ELECTRIC STANDARD APPLICATION FOR REACTOR FUEL (GESTAR II)" (TAC NO. ME7253)

Dear Mr. Lingenfelter:

By letter dated September 23, 2011 (Agencywide Documents and Access Management System (ADAMS) Accession No. ML112660430), and revised by letters dated March 15 and 22, 2012 (ADAMS Accession Nos. ML12076A035 and ML12083A002), Global Nuclear Fuel – Americas, LLC (GNF) submitted Amendment 35 to Topical Report (TR) NEDE-24011-P-A/ NEDO-24011-A, entitled "General Electric Standard Application for Reactor Fuel (GESTAR II)" to the U.S. Nuclear Regulatory Commission (NRC) staff for review.

The NRC staff has found that Amendment 35 to GESTAR II is acceptable for referencing in licensing applications for General Electric-designed boiling water reactors to the extent specified in the enclosed final safety evaluation (SE). The final SE defines the basis for acceptance of the TR.

Our acceptance applies only to material provided in the subject TR. We do not intend to repeat our review of the acceptable material described in the TR. When the TR appears as a reference in license applications, our review will ensure that the material presented applies to the specific plant involved. License amendment requests that deviate from this TR will be subject to a plant-specific review in accordance with applicable review standards.

In accordance with the guidance provided on the NRC website, we request that GNF publish accepted proprietary and non-proprietary versions of this TR within three months of receipt of this letter. The accepted versions shall incorporate this letter and the enclosed final SE after the title page. Also, they must contain historical review information, including NRC requests for additional information and your responses. The accepted versions shall include a "-A" (designating accepted) following the TR identification symbol.

If future changes to the NRC's regulatory requirements affect the acceptability of this TR, GNF and/or licensees referencing it will be expected to revise the TR appropriately, or justify its continued applicability for subsequent referencing.

Sincerely,

Sher Bahadur, Deputy Director
Division of Policy and Rulemaking
Office of Nuclear Reactor Regulation

Project No. 712

Enclosure:
Final SE

cc w/encl: See next page

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

AMENDMENT 35 TO TOPICAL REPORT NEDE-24011-P-A/NEDO-24011-A

"GENERAL ELECTRIC STANDARD APPLICATION FOR REACTOR FUEL (GESTAR II)"

GLOBAL NUCLEAR FUEL – AMERICAS, LLC

PROJECT NO. 712

1.0 INTRODUCTION AND BACKGROUND

By letter dated September 23, 2011 (Reference 1), and revised by letters dated March 15 and 22, 2012 (References 2 and 3), Global Nuclear Fuel – Americas, LLC (GNF) submitted Amendment 35 to Topical Report (TR) NEDE-24011-P-A/NEDO-24011-A, entitled "General Electric Standard Application for Reactor Fuel (GESTAR II)" to the U.S. Nuclear Regulatory Commission (NRC) staff for review. The changes in Amendment 35 are administrative in nature and include updated references and the addition of references to recently approved TR supplements.

2.0 EVALUATION

The NRC staff has evaluated each of the proposed changes below.

Add Reference to Three Supplements to TR NEDC-33173P

Reference 2-21 of GESTAR II Main has been updated to include the approved (-A) version of TR NEDO–33173, Supplement 4-A, "Implementation of PRIME Models and Data in Downstream Methods," Revision 0, September 2011. This TR has been previously reviewed and approved by the NRC staff (Reference 3). Therefore, inclusion of this reference is acceptable.

Section S.5.2 of the US Supplement to GESTAR II was revised to implement referencing of three supplements to GE-Hitachi Nuclear Energy (GEH) TR NEDC-33173P-A, which expand and modify the applicability of NEDC-33173P-A. These three supplements have been previously reviewed and approved by the NRC staff (References 4, 5, and 6). These approved supplements were added to Section S.6 of the US Supplement to GESTAR II as references S-108, S-109, and S-110, respectively. With the submittal of Reference 2, GNF further revised the paragraph in Section S.5.2 that now references these approved supplements to NEDC-33173P-A. The NRC staff finds that the revisions to this paragraph accurately reflect the approvals of these supplements to TR NEDC-33173P-A. Reference S-101 of Section S.6 was also updated to reflect the latest approved revision (Revision 2) of NEDC-33173P-A.

ENCLOSURE

NEDC-33173P-A, Revision 2, was previously reviewed and approved by the NRC staff with NEDC-33173P, Supplement 2, Parts 1-3 (Reference 5). With the submittal of the approved (-A) version of NEDC-33173P, Supplement 2, GEH also submitted Revision 3 to NEDC-33173P-A (Reference 7). Revision 3 is the combination of NRC-approved changes in Revisions 1 and 2 (Revision 2 merged with Revision 1) with some additional minor corrections identified by GEH. If the NRC staff approves NEDC-33173P-A, Revision 3, reference S-101 will need to be updated to reflect Revision 3 of NEDC-33173P-A and its date of publication in the next revision of GESTAR II. The NRC staff agrees that these changes to GESTAR II are administrative and, therefore, are acceptable.

Revised Reference to NEDO-32465-A

With the submittal of Reference 3, GNF proposed to update Section S.6 of the US Supplement to GESTAR II, Reference S-85 to add Supplement 1 to TR NEDO-32465-A, "Migration to TRACG04/PANAC11 from TRACG02/PANAC10 for Reactor Stability Detect and Suppress Solutions Licensing Basis Methodology for Reload Applications." This supplement is currently under review by the NRC staff. Since NEDO-32465, Supplement 1, is not yet approved, the NRC staff does not approve this proposed change to Reference S-85. If the NRC staff approves Supplement 1 to TR NEDO-32465-A, then GNF will need to submit a subsequent amendment to request approval to update Reference S-85.

3.0 CONCLUSION

Based on the evaluation stated above, the NRC staff finds that the changes proposed in Amendment 35 to TR NEDE 24011-P-A/NEDO-24011-A, with the exception of the proposed change submitted in Reference 3, are administrative in nature and, therefore, are acceptable and appropriate for inclusion in NEDE-24011-P-A/NEDO-24011-A.

4.0 REFERENCES

1. Letter from GNF to NRC, MFN 11-213, "Administrative Amendment 35 to GESTAR II to Implement the Referencing of NEDC-33173 Supplements 2P-A and 3P-A and NEDO-33173 Supplement 4-A," dated September 23, 2011. (ADAMS Accession No. ML112660430)

2. Letter from GNF to NRC, MFN 11-213 Sup 1, "Revised Word Usage on Pg US-44 of US Supplement to GESTAR II in Administrative Amendment 35 Submittal," dated March 15, 2012. (ADAMS Accession No. ML12076A035)

3. Letter from GNF to NRC, MFN 11-213 Sup 2, "Additional Reference Change in US Supplement to GESTAR II to be included in Administrative Amendment 35 (TAC No. ME7253)," dated March 22, 2012. (ADAMS Accession No. ML12083A002)

4. TR NEDC-33173, Supplement 3P-A, "Applicability of GE Methods to Expanded Operating Domains – Supplement for GNF2 Fuel," Revision 1, dated July 2011. (ADAMS Package Accession No. ML111960462)

5. TR NEDC-33173, Supplement 2, Part 1P-A, Revision 1, "Applicability of GE Methods to Expanded Operating Domains – Power Distribution Validation for Cofrentes Cycle 13," Part 2P-A, Revision 1, "Applicability of GE Methods to Expanded Operating Domains – Pin-by-Pin Gamma Scan at FitzPatrick October 2006," and Part 3P-A, Revision 1, "Applicability of GE Methods to Expanded Operating Domains – Power Distribution Validation for Cofrentes Cycle 15," dated April 2012. (ADAMS Package Accession No. ML121150469)

6. TR NEDO–33173, Supplement 4-A, "Implementation of PRIME Models and Data in Downstream Methods," dated September 2011. (ADAMS Accession No. ML112660155)

7. TR NEDC-33173P-A, Revision 3, "Applicability of GE Methods to Expanded Operating Domains," dated April 2012. (ADAMS Package Accession No. ML121150469)

Principal Contributor: Stephen Philpott

Date: May 14, 2012