September 7, 2010

Mr. Jerald G. Head
Senior Vice President, Regulatory Affairs
GE Hitachi Nuclear Energy
3901 Castle Hayne Road MC A-18
Wilmington, NC 28401

SUBJECT: FINAL SAFETY EVALUATION REVISION 1 FOR GE HITACHI NUCLEAR ENERGY LICENSING TOPICAL REPORT NEDC-33326P REVISION 1, "GE14E FOR THE ECONOMIC SIMPLIFIED BOILING WATER REACTOR INITIAL CORE NUCLEAR DESIGN REPORT"

Dear Mr. Head:

On August 24, 2005, GE Hitachi (GEH) Nuclear Energy submitted the Economic Simplified Boiling Water Reactor (ESBWR) design certification application to the staff of the U.S. Nuclear Regulatory Commission. Subsequently, in support of the design certification, GEH submitted the license topical report (LTR) NEDC-33326P Revision 1, "GE14E for ESBWR Initial Core Nuclear Design Report." The staff has now completed its review of NEDC-33326P Revision 1.

The staff finds NEDC-33326P Revision 1, acceptable for referencing for the ESBWR design certification to the extent specified and under the limitations delineated in the LTRs and in the associated safety evaluation (SE). The SE, which is enclosed, defines the basis for acceptance of the LTR.

The staff requests that GEH publish the revised version of the LTRs listed above within 1 month of receipt of this letter. The accepted version of NEDC-33326P shall incorporate this letter and the enclosed SE and add an "-A" (designated accepted) following the report identification number.

If NRC's criteria or regulations change, so that its conclusion that the LTR is acceptable is invalidated, GEH and/or the applicant referencing the LTR will be expected to revise and resubmit its respective documentation, or submit justification for continued applicability of the LTR without revision of the respective documentation.

Document transmitted herewith contains sensitive unclassified information. When separated from the enclosures, this document is "DECONTROLLED."

Pursuant to 10 CFR 2.390, we have determined that the enclosed SE contains proprietary information. We will delay placing the non-proprietary version of this document in the public document room for a period of 10 working days from the date of this letter to provide you with the opportunity to comment on the proprietary aspects only. If you believe that any additional information in Enclosure 1 is proprietary, please identify such information line by line and define the basis pursuant to the criteria of 10 CFR 2.390.

The Advisory Committee on Reactor Safeguards (ACRS) subcommittee, having reviewed the subject LTR and supporting documentation, agreed with the staff's recommendation for approval following the May 18, 2010 ACRS subcommittee meeting.

Sincerely,

/RA Frank Akstulewicz for:/

David B. Matthews, Director
Division of New Reactor Licensing
Office of New Reactors

Docket No. 52-010

Enclosure:
1. Safety Evaluation (Non-Proprietary)
2. Safety Evaluation (Proprietary)

cc: See next page (w/o enclosure)

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DATE	06/28/10	08/12/10	09/07/10	09/07/10	08/19/10	09/07/10

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FINAL SAFETY EVALUATION REPORT (SER) REVISION 1 FOR GE HITACHI NUCLEAR ENERGY LICENSING TOPICAL REPORT (LTR) NEDC-33326P REVISION 1, "GE14E FOR THE ECONOMIC SIMPLIFIED BOILING WATER REACTOR INITIAL CORE NUCLEAR DESIGN REPORT"

1 INTRODUCTION

The staff based its review of the nuclear design on information contained in Licensing Topical Report (LTR) NEDC-33326P (Reference 1), the Economic Simplified Boiling Water Reactor (ESBWR) design control document (DCD) (Reference 2), responses to staff requests for additional information (RAIs), and other supporting topical reports referenced by the applicant. The staff conducted its evaluation in accordance with the guidelines provided by Standard Review Plan (SRP) Section 4.3, "Nuclear Design."

2 REGULATORY CRITERIA

DCD Tier 2, Section 4.3, "Nuclear Design," presents the ESBWR nuclear design bases. The nuclear design must not exceed the specified acceptable fuel design limits during normal operation, including anticipated operational occurrences (AOOs), and the effects of postulated reactivity accidents will not cause significant damage to the reactor coolant pressure boundary (RCPB) or impair the capability to cool the core, or sustain unstable core conditions. To meet these objectives, the nuclear design must conform to the following general design criteria (GDC):

- GDC 10, "Reactor Design," requiring the reactor design (reactor core, reactor coolant system, control and protection systems) to assure that specified acceptable fuel design limits are not exceeded during any condition of normal operation, including AOOs.

- GDC 11, "Reactor Inherent Protection," requiring a net negative prompt feedback coefficient in the power operating range.

- GDC 12, "Suppression of Reactor Power Oscillations," requiring that power oscillations that can result in conditions exceeding specified acceptable fuel design limits are not possible, or can be reliably and readily detected and suppressed.

- GDC 13, "Instrumentation and Control," requiring a control and monitoring system to monitor variables and systems over their anticipated ranges for normal operation, AOOs, and accident conditions.

- GDC 20, "Protection System Functions," requiring, in part, a protection system that automatically initiates a rapid control rod insertion to assure that fuel design limits are not exceeded as a result of AOOs.

- GDC 25, "Protection System Requirements for Reactivity Control Malfunctions," requiring protection systems designed to assure that specified acceptable fuel design limits are not exceeded for any single malfunction of the reactivity control systems.

- GDC 26, “Reactivity Control System Redundancy and Capability,” requiring, in part, a reactivity control system capable of holding the reactor subcritical under cold conditions.
- GDC 27, “Combined Reactivity Control Systems Capability,” requiring, in part, a control system designed to control reactivity changes during accident conditions in conjunction with poison addition by the emergency core cooling system (ECCS).

- GDC 28, “Reactivity Limits,” requiring, in part, that the reactivity control systems be designed to limit reactivity accidents so that the reactor coolant system boundary is not damaged beyond limited local yielding.

3 NUCLEAR DESIGN DESCRIPTION

3.1 Summary of Technical Information

3.1.1 Core Description

The topical report NEDC-33326P, (Reference 1) describes the ESBWR initial core design. The 4500 MWth ESBWR core consists of 1132 fuel bundles and 269 control blades. The core design given in Figure 3-1 of Reference 1 characterizes the initial full core design. Five types of fuel bundles, which are similar except for differences in enrichment and burnable poison content, are loaded in the reference pattern. Bundle differences allow for a flatter radial power distribution across the core and provide low reactivity bundles similar in the neutronic behavior to partially burnt bundles.

The applicant provided a description of the ESBWR fuel bundle designs including lattice information in Reference 1. The bundle designs included several zones that vary axially throughout the bundle. A two-dimensional lattice describes each zone. Multiple lattices describe the variation of the fuel bundle axially as the design includes part-length rods, vanished fuel rods[1], variations in burnable poison loadings, and enrichment.

Typically, bundle nuclear properties in a core vary both axially and radially. As a result, the core is modeled with several nodes that account for these differences as well as the influence of individual nodes on the neighboring nodes. The staff safety evaluation report (SER) evaluates the modeling techniques and qualifications in References 3 and 4 for application to the ESBWR.

3.1.2 Power Distribution

The acceptance criteria in the area of nuclear design, specifically power distributions, are based on meeting the relevant requirements of the GDC related to the reactor core and the reactivity control systems.

The nuclear design basis for control requirements is that maximum linear heat generation rate (MLHGR) and the minimum critical power ratio (MCPR) constraints shall be met during operation. The operating limit MCPR and MLHGR limit are determined such that the fuel rods do not exceed required licensing limits during AOOs.

[1] Vanished fuel rods refer to those rod locations within the bundle lattice above the part-length fuel rod plena. Within the bundle, at these specific rod locations and within this upper axial span there are no physical rods obstructing coolant flow.

3.1.3 Safety and Operating Limits

The MLHGR is the maximum local linear heat generation rate (LHGR), more specifically the fuel rod with the highest surface heat flux at any nodal plane in a fuel bundle in the core. The MLHGR operating limit is bundle-type dependent and the staff SER for LTR NEDC-33242P (Reference 5) evaluates the limit. The staff SER evaluates the LHGR, ensuring that it meets all mechanical design assumptions. The reactor cannot be operated with the fuel at LHGR values greater than acceptable values within the body of the safety analysis under normal operating conditions. Under abnormal conditions, including the maximum overpower condition, the MLHGR will not exceed the strain limit or cause fuel melting.

The MCPR is the minimum critical power ratio of all of the fuel bundles. The critical power ratio (CPR) for any bundle is the ratio of the bundle power that would result in transition boiling to the current bundle power. Therefore, the bundle with the smallest CPR has the smallest margin to transition boiling. The CPR is a function of several parameters; the most important are bundle power, bundle flow, the local power distribution and the details of the bundle mechanical design.

The plant operating limit MCPR (OLMCPR) is established by considering the limiting AOOs for each operating cycle. The OLMCPR determines that 99.9 percent of the rods avoid boiling transition during the limiting analyzed AOO, as discussed in the staff SER for LTR NEDC-33237P (Reference 6).

The design bases affecting power distribution of the ESBWR include the following parameters:

- Under abnormal conditions (including maximum overpower), the MLHGR will not exceed mechanical design limits for the fuel.

- The MCPR during normal operation will remain greater than the OLMCPR to avoid boiling transition during normal operation and AOOs.

GDC 13 provides the required criteria to evaluate core monitoring. In-core nuclear instrumentation performs core monitoring, in part, to ensure that the core operates within these limits. According to DCD Tier 2, Section 7.9, information from the core monitoring instrumentation is used by the 3D MONICORE system to determine the margin to operational limits. The 3D MONICORE system has two components, the Monitor and the Predictor. In each case, the calculational engine is the PANAC11 three-dimensional, quasi-steady-state core simulator. The staff's SER on References 3 and 4, discuss the analytical capabilities of the PANAC11 core simulator.

The 3D MONICORE system has several adaption methods. Adaption methods improve monitoring accuracy by incorporating live plant data from the neutron monitoring system in the core simulator.

The calculation of the bundle CPR and the nodal LHGR during operation are performed by the PANAC11 core simulator. 3D MONICORE reports the minimum bundle CPR and maximum nodal LHGR relative to their respective limits (OLMCPR and MLHGR limit, respectively). The thermal margin information is also passed to the automatic thermal limits monitor (ALTM), the rod worth minimizer (RWM) and the Multi-channel Rod Block Monitor (MRBM) subsystems of the rod control and information system (RC&IS). Instrumentation signals also inform the reactor

protection system (RPS). The RPS utilizes signals to initiate a reactor SCRAM when these signals exceed a specified setpoint.

3.1.4 Neutron Monitoring System

Nuclear instrumentation monitors variables affecting the nuclear fission process. Appropriate controls ensure that the reactor operates within acceptable ranges. Specifically, the nuclear instrumentation monitors the reactor power and ensures that it does not exceed acceptable design limits. To meet these objectives, the nuclear design must conform to GDC 10 and 13.

The staff reviewed the nuclear instrumentation design in accordance with SRP Section 4.3. In DCD Tier 2 (Reference 2) and NEDE-33197P (Reference 4), the applicant describes how the instrumentation for power and power shape monitoring and calibration meets the requirements set forth in GDC 10 and GDC 13.

3.1.4.1 Description of the Instrumentation

The ESBWR core monitoring is accomplished with several in-core nuclear instruments that cover the expected ranges for normal operation, AOOs, and accident conditions. The neutron monitoring system is comprised of three separate measurement systems: the source range monitor, the local power range monitor, and the automatic fixed in-core probe. The power range neutron monitoring system (PRNM) receives signals from several local detectors. These in-core nuclear instruments include the local power range monitors (LPRMs) as well as automatic fixed in-core gamma thermometers (GTs). For low powers characteristic of the source range through a normal startup (greater than 10 percent of rated thermal power) the source range neutron monitoring system (SRNM) monitors the core neutron flux.

The LPRMs are arranged in 64 strings, each with four detectors, and distributed throughout the core. The locations of LPRM strings are shown in Figure 7.2-7 of Reference 2. For every four by four array of bundles, there are four LPRM strings (one at each corner). The LPRM strings are comprised of four LPRM detectors that are spaced evenly axially throughout the core. The LPRM detectors are polarized fission chambers.

Inside the LPRM instrument guide tube, there are seven automatic fixed in-core probes (AFIP). The AFIP is a gamma thermometer instrument that is used to periodically calibrate the LPRM signal. Figure 7.2-8 of DCD Tier 2 (Reference 2) shows the axial elevation of the AFIPs. There are seven AFIPs in each LPRM instrument string. There is one AFIP at the same elevation as the midplane of each of the LPRM detectors. In between each LPRM detector there is another AFIP. The AFIPs are evenly distributed between the uppermost and bottommost LPRMs at 381 mm (15 inch) intervals for a total of seven AFIPs.

The neutron monitoring system instruments measure the neutron flux and monitor the fission process. The number and types of instruments included in the design are sufficient to monitor the flux over the entire range of operation between startup (low power), normal operation, and transient conditions (high power). When the reactor power is low, monitoring the startup process calls for increased instrument sensitivity. According to Chapter 7 of the DCD Tier 2 (Reference 2), the SRNM is comprised of 12 detectors. These detectors are fixed in-core regenerative fission chamber sensors. The 12 detectors are spaced evenly throughout the core and located at the core midplane axially; Figure 7.2-6, Reference 2, shows the radial locations.

The detectors are inside the pressure barrier tubes. The SRNM detectors measure the reactor flux over ten decades, from a flux level of approximately 10^3 n/cm^2/sec to 10^{13} n/cm^2/sec. This range extends to approximately 10 percent of rated power. The LPRM monitoring capability overlaps this range as the LPRMs can monitor core power from the startup range through the power range: from one percent of power to greater than rated thermal power (Reference 2).

3.1.4.2 Rod Control and Information System

The Rod Control and Information System (RC&IS) is a non-safety-related system. The RC&IS is a logic system that provides controls on reactor maneuvering through control rod motion during normal operation and maintains status information regarding the current control rod configuration for the reactor.

Using local power indications from the LPRM detectors, the RC&IS subsystems issue rod blocks to ensure that control rod motion does not exceed safety and operating limits. The ATLM and MRBM work together above the low power set point to inhibit rod withdrawals when local detectors indicate power changes that challenge the MLHGR limit or the OLMCPR. The MRBM, unlike conventional rod block monitors, uses several channels of LPRM indications throughout the core to simultaneously monitor each region of the core where control rods are being withdrawn during ganged withdrawal sequences. Below the low power set point, the RWM compares the sequence to withdraw the control rod at low power to a preprogrammed control rod withdrawal pattern. In cases where the control rod's withdrawal is different, the RWM enforces control rod insertions and withdrawals at low power to reduce the available reactivity worth of a control rod to mitigate the consequences of a control rod drop accident during low power operation (Reference 2).

Upon receipt of a SCRAM signal by the RPS, the RC&IS initiates a fast fine motion control rod drive (FMCRD) run-in as a backup to the hydraulic SCRAM through the diverse protection system (DPS). The RC&IS also sends selected control rod run-in (SCRRI) signals to the DPS following specific AOOs, namely load rejection, turbine trip and loss of feedwater heating (Reference 2).

Another important function of the RC&IS is to interface with the plant computer to perform LPRM calibration and plant simulator adaption. This function is performed by using AFIP signals in conjunction with three dimensional nuclear models to determine gain adjustments and nodal parameter corrections. The AFIP signals are input into the 3D MONICORE system to perform adaption. The staff's SER on References 3 and 4 evaluate the calibration and adaption features of 3D MONICORE with the ESBWR specific AFIP design.

3.1.5 Reactivity Coefficients

The reactivity coefficients express the effects of changes in the core conditions, such as power, fuel and moderator temperature, and moderator density, on core reactivity. These coefficients vary with fuel exposure and power level. The applicant has provided calculated values of the coefficients in Reference 1.

Reactivity coefficients, the differential changes in reactivity produced by differential changes in core conditions, use external disturbances to predict the response of the core. The base initial condition of the system and the postulated initiating event determine which of the several defined coefficients are significant in evaluating the response of the reactor. The coefficients of interest are the Doppler coefficient, the void reactivity coefficient, and the moderator temperature coefficient. The combination of these reactivity coefficients dictates the power

reactivity coefficient. A combination of negative coefficients ensures that the reactor will have an inherent negative reactivity feedback with increasing power.

Reference 3 evaluates the computational tools employed by the applicant to calculate the reactivity coefficients. The coefficients calculated by the applicant are not used in steady state or transient analyses, but are meant to demonstrate compliance with GDC 11.

3.1.5.1 Doppler Reactivity Coefficient

In order to demonstrate that the Doppler reactivity coefficient remains negative in the power operating range, the applicant calculated temperature dependent eigenvalues for each of the five fuel bundle types for the dominant zone lattice. At each point in exposure, the temperature was increased and the change in eigenvalue was shown to be negative at all points in exposure. The Doppler reactivity coefficient is predominantly driven by the uranium-238 and plutonium-240 content in the fuel and, while an inherent feature of the fuel, this coefficient does not vary significantly among BWR fuel designs. The ESBWR initial core calculated Doppler coefficient is approximately [[ ]], which is slightly greater (in magnitude) than typical values of operating reactor Doppler coefficients. The applicant attributes the difference to a lower initial enrichment (Reference 1).

3.1.5.2 Void Reactivity Coefficient

The applicant estimated the void reactivity coefficient for both the power range of operation and for cold shutdown conditions. The applicant's analyses indicate a negative trend of core eigenvalue with increasing core average void content in the power range of operation, indicating inherent negative reactivity feedback under these conditions. The magnitude of the void reactivity coefficient, however, decreases with decreasing void content. Therefore, the applicant identified the cold shutdown condition as a limiting case, particularly at the end of the cycle following depletion of burnable poisons. The end of cycle conditions are typically over-moderated and, given that the core is entirely fresh, there are no significant plutonium driven spectral effects. The analysis for the limiting condition verifies that the void reactivity coefficient is negative. The value of the coefficient is calculated to be about [[ ]] at the most limiting condition (Reference 1).

3.1.5.3 Moderator Temperature Coefficient

Lastly, the applicant calculated the moderator temperature coefficient. During normal operation, the coolant is only subcooled near the core inlet and remains at a near constant temperature once reaching saturated conditions. The end of the reference cycle was identified as the condition with the least negative moderator temperature coefficient. The results indicate that at temperatures above [[ ]] the core eigenvalue decreases with increasing water temperature (Reference 1).

The moderator temperature coefficient decreases in magnitude over cycle exposure with the withdrawal of control rods and the depletion of gadolinia burnable poisons. Late in the cycle, the reduction in the poison content leads to potential conditions where the reactor is over-moderated, thereby yielding a positive moderator temperature coefficient for cold conditions. While the end of cycle (EOC) moderator temperature coefficient is positive, it is small compared to the effects of the void reactivity feedback. The applicant's calculations show that the

moderator temperature coefficient at the EOC may be positive and on the order of [[ ]] (Reference 1).

3.1.6 Control Requirements

The control rod system is designed to provide shutdown margin and reactivity control of maximum excess reactivity anticipated during cycle operation. The control rods provide reactivity changes that compensate for the reactivity effects of the fuel and water density changes accompanying power level changes over the range from full load to no load and allow for control of the power distribution within the core.

The reference rod patterns for the ESBWR core are similar to current BWR rod patterns. For the first half of the cycle, burnable poisons in conjunction with a conventional rod pattern control the hot excess reactivity. During the middle of the cycle, very little rod movement is needed because of a nearly constant hot excess reactivity during this part of the cycle. In the latter part of the cycle the other half of the rods are employed in a similar checkerboard pattern. During the progression towards the EOC the control rods in the second set are withdrawn, and finally the first set is used for control towards the EOC where there is just a small amount of hot excess reactivity.

This reference rod pattern is used in the analyses of the core power distribution and is an input into the Chapter 6 and 15 analyses that describe accidents and transients starting from different times during cycle operation. The reference rod pattern for the initial cycle also factors into the determination of the radial and axial power distributions in the core. As is common for currently operating BWRs, the axial power shape is bottom peaked at the beginning of cycle (BOC) and evolves into a slightly top peaked distribution at the EOC.

Margins to thermal operating limits (Maximum Fraction of Limiting Power Density and the CPR Ratio) were analyzed over the initial operating cycle.

In addition to providing the means for controlling core reactivity for power maneuvering, the control rods provide the minimum shutdown margin following any AOO and are capable of making the core subcritical rapidly enough to prevent exceeding specified acceptable fuel design limits. The control rods automatically insert hydraulically upon receipt of a SCRAM signal from the reactor protection system.

The applicant has provided an analysis in Figure 3-44 of Reference 1 showing that the control rod worth is sufficient to ensure a subcritical configuration for xenon-free, cold shutdown conditions at the beginning of the cycle. The BOC condition is often limiting in terms of available shutdown margin. The analysis presented indicates a minimum shutdown margin of [[ ]] at the BOC.

The applicant provided analyses of the shutdown margin for the reactor during several points in exposure. At the beginning, middle, and end of cycle (BOC, MOC, and EOC) exposure points, the applicant calculated the shutdown margin assuming one control rod withdrawn, and repeated the calculation for each control rod. Therefore, this calculation identifies the highest worth control rod at each exposure point, the shutdown margin with the highest worth rod withdrawn, and the relative worth of the remaining control rods. As is expected, the shutdown margins are greatest when the low worth peripheral rods are stuck out. During cycle exposure, the core radial power shape tends to shift outward, and this is consistent with decreasing shutdown margins assuming stuck-out control rods near the core edge towards EOC.

The control rods are backed up by the standby liquid control system (SLCS). The SLCS is a second reactivity control system meant to provide a diverse and redundant capability to the control rods. The SLCS is an accumulator-driven boron injection system. The SLCS is designed to provide the capability of bringing the reactor, at any time in a cycle, from full power with a minimum control rod inventory (which is defined to be at the peak of the xenon transient) to a subcritical condition with the reactor in the most reactive xenon-free state if the control rods fail to insert.

The applicant analyzed the capability of the SLCS system to inject sufficient boron into the reactor coolant system so that the resultant equivalent uniform boron concentration ensures that the reactor is subcritical (with [[ ]] margin) under cold shutdown, xenon-free conditions from its most critical state with the control rods fully withdrawn.

3.1.7 Stability

GDC 12 requires that power oscillations that could result in exceeding the specified acceptable fuel design limits be prevented or readily detected and suppressed.

DCD Tier 2, Section 4.3.3.6, “Stability Evaluation,” discusses the stability of the reactor with respect to xenon-induced power distribution oscillations. The strong negative reactivity feedback from the void reactivity coefficient damps xenon-induced power distribution oscillations. The applicant presented considerations of thermal-hydraulic stability in DCD Tier 2, Appendix 4D for the equilibrium core. The staff evaluation addresses the thermal-hydraulic stability of the ESBWR equilibrium core in SER Section 4A and for the initial core in the staff safety evaluation report for NEDO-33337 (Reference 7).

In NEDO-33337, the applicant references an approved NRC methodology for performing stability analyses. The approval of the methodology, as described in Reference 8, however, is contingent upon demonstrating accuracy in the PANACEA provided cross sections to the TRACG 3D kinetics model. The staff’s SER evaluates the efficacy of the nuclear design methodology in Reference 3. Approval of NEDC-33239P constitutes approval of PANAC11 to generate nuclear data for use by TRACG04.

3.1.8 Reactivity Accidents

GDC 28 requires that the reactivity control system be designed in such a way as to preclude reactivity accidents of sufficient magnitude to impair core coolability or reactor coolant pressure boundary (RCPB) integrity.

The consequences of a postulated control rod drop accident are sensitive to the core management loading and specific core design. Factors such as control blade worth and radial power peaking are key parameters in assessing the consequences of such an accident. Therefore, the staff requested in RAI 4.6-38 that GEH evaluate the consequences of a control rod drop accident for the initial core design.

The response to RAI 4.6-38 refers to the analysis performed in response to RAI 4.6-23, Supplement 2. The response briefly describes a reload licensing screening approach, analysis procedures, and analytical results. The analyses were performed using the PANAC11 (PANACEA version 11) three dimensional simulator in a transient mode with six delayed neutron groups. PANAC11 calculates the fuel enthalpy rise according to an adiabatic model (by

integrating transient power) and explicitly accounts for blade worth, nominal blade pull during startup, and radial power shapes.

Calculated fuel enthalpy rise for the ESBWR initial core design indicates significant margin to the interim criteria in SRP Section 4.2, Revision 3. The maximum calculated fuel enthalpy rise for the initial core is [[ ]]. The limiting enthalpy rise is for the BOC for a static blade worth of [[ ]] (Reference 9). For low exposure the cladding failure fuel enthalpy rise limit is [[ ]]. The initial core CRDA analysis indicates significant margin to cladding failure. Therefore barrier integrity is ensured as the analysis indicates that no fuel rods fail. The radiological consequences of such an accident are bounded by the analyses in DCD Section 15.3.1.5, which is based on 1000 failed fuel rods (Reference 2).

4 STAFF EVALUATION

The applicant provided several analyses to demonstrate ESBWR initial core compliance with the prescribed GDC in SRP Section 4.3. The staff's SER on References 3 and 4 reviews the results of these analyses in the following sections.

The staff evaluated the information contained in the subject LTR and supporting topical reports as it relates to ESBWR design compliance with GDC 10, 11, 12, 13, 20, 25, 26, 27 and 28. The staff described the review of the applicant's analyses in the following sections as they relate to concerns regarding power distribution and operating limits, reactivity feedback, and reactivity control. The staff reviewed compliance with GDC 12 as it relates to thermal hydraulic stability in its review of the initial core transients, LTR NEDO-33337 (Reference 7).

4.1 Power and Operating Limits

As set forth above, GDC 10 and 13 specify the requirements for the core operating power and instrumentation.

The MLHGR limit and OLMCPR are determined such that operation within these limits prevents fuel damage from melting, excessive strain, or boiling transition. These limits are determined such that there is adequate margin to account for the effects of AOOs. The staff documented the review of the methods for determining the MLHGR limit and the OLMCPR in the staff's safety evaluation of References 6 and 7 respectively. Operation by the 3D MONICORE system based on fuel specific analyses, the PANAC11 computational engine, and plant instrumentation determine the limits. The uncertainties in the methodology and plant instrumentation are addressed in the NRC approved methodologies (References 3, 4, 7, 10, 11, and 12). In its review the staff considered the modifications made to the methods for expanded operating domains (Reference 18) and their applicability to the ESBWR operating conditions. The staff also considered aspects of the methods unique to the ESBWR plant instrumentation (Reference 4). In response to staff RAI 4.4-68, the applicant verified that the bundle R-factor is determined using limiting axial power and void fraction profiles (Reference 13). The staff reviewed these profiles and finds them acceptable for determining a conservative R-factor for the initial core. The staff's SER on References 3 and 4 describes the review of the incorporation of uncertainties into the limits.

The staff reviewed the nuclear design performance over the initial cycle, as analyzed by the applicant. The methods for determining the margin to limits are described in NEDC-33237P, NEDC-33242, and NEDC-33239P (which references NEDE-33197P). The staff acceptance of

the uncertainties and methodology will be documented in the staff's safety evaluation of NEDE-33197P, NEDC-33239P, NEDC-33237P, and NEDC-33242P. The cycle analysis performed by the applicant shows that the reference loading pattern and control rod withdrawal sequence for the ESBWR indicate that there is margin to both of these limits during normal operation.

To account for the effects of AOOs, the applicant provided analyses demonstrating the capability of the reactor protection system (RPS) and associated control rod system to perform its SCRAM function in NEDO-33337. The staff documented its evaluation of NEDO-33337 in the staff's DCD SER Section 15. The automatic function of the RPS is to prevent exceeding acceptable fuel design limits in the event of AOOs.

The applicant bases its analyses on the actuation of the RPS in response to input signals in excess of a setpoint value. The applicant provided a description of the methodology for the determination of setpoints. The determination process includes margin associated with uncertainties in the instrumentation. The staff's evaluation of the setpoint methodology is provided in Chapter 7 of the staff's safety evaluation of the ESBWR DCD.

The applicant's cycle calculations show adequate margin to both the MLHGR and OLMCPR limits during normal operation, and evaluated the function of RPS to SCRAM the reactor prior to exceeding any specified acceptable fuel design limits (SAFDLs). Therefore, the staff approval of NEDO-33337 in conjunction with the information provided in Reference 1 sufficiently demonstrates compliance with GDC 10.

4.2 Neutron Monitoring

The neutron monitoring system is designed to meet the requirements of GDC 13. Specifically the PRNM and SRNM are designed to monitor the fission process over the range of anticipated operation and accident conditions. The PRNM is comprised of several LPRM detectors with the capability of monitoring the neutron flux in the reactor between one percent of rated core power and well over 100 percent of the rated core power (125 percent). The SRNM is designed to monitor the neutron flux at very low levels (~10^3 n/sq-cm/sec) or approximately 10 decades below the normal operating level. The combination of these two neutron monitoring subsystems allows for an overlapping monitoring capability over the full range of neutron flux levels under normal operation including startup and anticipated operational occurrences. The LPRM capability extends to higher neutron flux levels allowing for monitoring of the reactor core power during accident conditions and anticipated transients without SCRAM. Therefore, the staff finds that the ESBWR neutron monitoring system is acceptable in that it provides sufficient capability and adequately monitors the neutron flux levels in the reactor over the necessary ranges.

The in-core ESBWR neutron monitoring system is based on a series of distributed local power range monitors. Substantially, the polarized fission gas chambers are the same as those instruments widely applied within the operating fleet of BWRs. The design differences between the ESBWR and conventional BWRs will not impact the fundamental operation of the LPRMs so long as the steady state bypass void fraction remains below 5 percent. These instruments interface with the 3D MONICORE system to determine the operating characteristics of the core.

The neutron monitoring system includes in-core gamma thermometers (GTs) replacing the function of the traversing in-core probe (TIP) system for conventional reactors. The gamma thermometers determine, much as gamma TIP instruments do, the axial power shape based on local gamma flux indications. There are two primary differences between the gamma TIP and

GT instruments. First, the GT instrument operates by inferring the local gamma flux based on heat deposition in the instrument. Second, the GTs do not move through the core.

The staff reviewed the information provided by the applicant in regard to the GT design and found that with regular calibration the GT can be used to determine the local gamma flux. When combined with coupled transport calculations to determine the detector response kernels (or signal to power ratios) the GT indication may adequately determine the local nodal power in surrounding nodes. The GT instruments are spaced within the core alongside the LPRMs, giving a complete radial mapping capability if the core power distribution is quadrant symmetric.

The 3D MONICORE system determines the margin to limits based on input from the neutron monitoring system, adaption, and input from the core thermal hydraulic instrumentation (i.e. core flow). The 3D MONICORE system is based on the PANAC11 calculational engine. The staff's SER on References 3 and 4 documents staff review of the PANAC11 code.

However, GDC 13 also requires that appropriate controls are in place to ensure that the reactor core is operated within prescribed safety and operating limits. The GDC 13 requirements for the NMS are fulfilled by prescribing limits that account for instrument and measurement uncertainties. Of key importance to the prescription of these limits is the accuracy of the neutron flux measurements. The pedigree of LPRM measurements in particular is related to the efficacy of the AFIPs and process computer to effectively and accurately calibrate the local indications of the neutron flux level. Additionally, the core monitoring system is used to adapt predictive calculations performed by the 3D MONICORE system to determine the local power distribution. The staff's SER for References 3 and 4 describes the reviewed and evaluated methods used to account for any uncertainties in the measurement, calibration, and adaption of the core neutronic modeling in the MLHGR limit and OLMCPR.

Therefore, the in-core instrumentation meets the requirements of GDC 13 by providing monitoring capability over the range of expected operation and providing sufficient information, given the capabilities of the 3D MONICORE system, to monitor core operating parameters relative to associated operating limits.

The staff finds that the ESBWR initial core design adequately meets the requirements of GDC 10 and 13, and is therefore acceptable.

4.3 Reactivity Feedback

As set forth above, GDC 11 requires that the core be designed with inherent negative reactivity feedback.

The applicant provided several analyses to indicate the nature and magnitude of the reactivity feedback coefficients for the reference ESBWR core. The staff's SER on References 3 and 4 reviewed the applicant's nuclear methods. In each case the applicant performed the analysis by perturbing the steady state calculation to determine the change in eigenvalue as a result of a change in the fuel temperature, coolant temperature, or coolant void.

In general, the Doppler coefficient is a strong function of fertile heavy metal content and spectrum hardness. For the ESBWR, the enrichment and planar fuel geometry are similar to operating BWRs. However, the bundle enrichment is slightly lower and the bundle pitch is slightly greater. The greater pitch and lower enrichment soften the neutron spectrum. A softened spectrum reduces the fertile resonance integral and consequently would serve to

reduce the Doppler coefficient in magnitude. However, the reduced enrichment results in a smaller positive reactivity effect from enhanced fissile resonance absorption. The net effect of these differences is an increase in the Doppler coefficient magnitude. The applicant's calculations are consistent with this expectation.

The increased assembly spacing also affects the moderator temperature coefficient. The increased hydrogen to heavy metal ratio decreases the magnitude of the moderator temperature coefficient and leads to slightly positive values for cold (zero power) conditions at the EOC where the neutron spectrum is very soft (thus, over-moderated). The positive nature of the moderator temperature coefficient is of minor concern, due to the relatively slow nature of the moderator temperature change (relative to fuel temperature change), and, at normal operating conditions, the core dynamic behavior is driven predominantly by the strong, negative void reactivity feedback. This condition is only for low temperatures and is not of sufficient magnitude to cause operational concerns during startup and shutdown operations, or a reactivity insertion problem.

The applicant provided a series of core calculations to determine the estimated void coefficient. As the void reactivity coefficient is stronger for higher void fractions, the applicant performed calculations for cold shutdown conditions. This calculation is conservative because the spectrum at cold shutdown conditions is over-moderated. The applicant simulated the effects of voids in the subcooled coolant using the PANAC11 core simulator, and found that in the most limiting case the void reactivity coefficient was negative.

The power reactivity coefficient is a combination of the Doppler, void, and moderator temperature reactivity coefficients. While the design differences of the ESBWR make the moderator temperature and Doppler coefficients less negative than for an operating BWR, the increased void, higher enrichment, and higher burnable poison loading result in an overall negative power coefficient. In the case of the ESBWR, the void coefficient is not significantly different from operating reactors and a dominant contributor to the power coefficient. The staff finds these values for the reactivity coefficients to be acceptable because they are negative in the power operating range, ensure a negative power reactivity coefficient, and therefore meet the requirements of GDC 11.

4.4 Reactivity Control

As set forth above, GDC 20, 25, 26, 27, and 28 specify the requirements for the reactivity control systems.

The reactivity control worth calculations were performed using the TGBLA06 and PANAC11 codes. These calculations show that the rod values are similar to operating BWR control rod worth. For the middle of the ESBWR initial cycle the hot excess reactivity is nearly constant, therefore requiring a high in-core rod density during normal operation near the beginning of cycle. The beginning of cycle hot excess is comparable to the peak hot excess and the neutron spectrum is the hardest (due to the presence of large quantities of burnable absorber), limiting individual control rod worth. Therefore, this condition is limiting as the additional available rod density for control is small and each rod worth is relatively low compared to other points during cycle exposure. By demonstrating shutdown margin with the strongest control rod withdrawn in the limiting condition, the applicant has demonstrated the system can fully control the core reactivity given the failure of a single control rod to insert.

Additionally, the applicant calculated the shutdown margin at several exposure points during the cycle to demonstrate that the BOC condition is the limiting point. Towards the EOC the shutdown margin decreases, due to the buildup of plutonium and depletion of burnable poisons leading to an increase in the hot excess over the course of the cycle. However, the increase in the hot excess is sufficiently small such that the shutdown margin at the BOC is the most limiting.

On this basis, the staff determined that the control system has adequate negative reactivity worth to ensure shutdown capability, assuming that the most reactive control rod is stuck in the fully withdrawn position.

The control rod system automatically inserts control blades to shut down the reactor on receipt of a SCRAM signal. The negative reactivity worth of the control rods is sufficient to bring the reactor to a cold shutdown condition at any point during exposure. The staff's SER for Chapters 7 and 15 of the ESBWR DCD for LTR NEDO-33337 document the staff review of the RPS design to adequately prompt automatic control rod insertion during AOOs. Therefore, the design meets the requirements of GDC 20.

Additionally, control rod assignments to particular hydraulic control units shall maintain sufficient distance between rods such that there is essentially no neutronic coupling between the control cells, such that there is no significant impact on the shutdown margin given a failure of a single hydraulic control unit. When the reactor is shut down, the core is filled with liquid water and the mean free paths for neutrons are much smaller than at power, where the presence of voids allows for increased neutron transport during slowing down. Therefore, control cell neutronic coupling is effectively limited to nearby neighboring control cells. The assignment of control rods to hydraulic control units, such that no hydraulic control unit drives two nearby control rods would preclude neutronic coupling. Without any coupling, there is no synergistic effect of a dual control rod insertion failure which could result in local criticality. The HCU mapping is provided in Reference 1. The HCU mapping indicates that the control rods assigned to a specific HCU are distanced apart from each other within the core. The mapping indicates that the distance between control rods sharing an HCU is between five and seven control rod locations. As the mean free path for even higher energy neutrons at normal operating conditions ranges of about 15 - 30 cm, and the mean free path is reduced when the core is under cold conditions with control rods inserted, the staff finds that the HCU assignments adequately preclude the possibility of synergistic reactivity effects. Therefore, local criticality based on the failure of any particular HCU is not a concern if the remainder of the control rods inserted provide sufficient negative reactivity to ensure that the reactor is shut down and subcritical under cold conditions at its most reactive point.

The staff considered the design basis cold critical eigenvalue that is used to determine the analytical shutdown margin. The initial core nuclear design predicted shutdown margin is very large ([[ ]] at the limiting exposure point). This margin well exceeds the design requirement of 1 percent and includes sufficient additional margin to bound (with 95 percent confidence) the variability in cold critical eigenvalue (based on operating fleet experience) without consideration of the bias. In response to RAI 4.3-11, the applicant provided a qualification of the startup cold eigenvalue design basis against Plant A (a large BWR/4 restart core with modern (GE14) fuel) (Reference 13). The results indicate that the BOC and mid-cycle eigenvalues are conservative relative to the design basis and trend similarly with standard reload eigenvalue bases. The staff therefore finds that the shutdown margin is sufficiently large to provide reasonable assurance that the requirements of GDC 25 are met considering the failure of a single rod to insert.

The DCD Tier 2 in Section 4.3.1.2 and Appendix 4B state that compliance with GDC 26 is demonstrated by showing margin to criticality in the most reactive cold condition with the strongest rod pair withdrawn. The applicant provided the assignment of control rods to individual HCUs in Section 4.3.3 of the DCD Tier 2 (Reference 2). The staff has evaluated the calculation of the shutdown margin and reactivity margin to criticality at cold conditions assuming the strongest rod pair withdrawn. The staff finds that the shutdown margin

calculations in NEDC-33326P provide reasonable assurance that the requirements of GDC 26 are met.

The SLCS meets the requirements for diverse and redundant control systems per GDC 26 and the combined reactivity control system requirements per GDC 27. The staff has determined that the SLCS is adequate for bringing the reactor to a cold shutdown condition at any point in exposure, and therefore acts as a fully redundant control system. The system is diverse in that it is a dissolved-poison, passive, liquid injection system, thereby adequately satisfying GDC 26. As the SLCS is fully capable of controlling the reactivity and is an emergency core cooling system, it provides sufficient negative worth to compensate for a partial failure of the control rod system, thereby adequately satisfying GDC 27. The analysis indicates a very large margin (~9 percent). The margin is sufficiently large that it ensures subcriticality considering any additional uncertainty in the determination of the cold critical eigenvalue (maximum uncertainty of ~0.5 percent).

Compliance with GDC 28 is demonstrated by analysis of the consequences of a postulated control rod drop accident. The staff notes some conservatism in the initial core analysis, in particular the adiabatic assumption precludes any void formation (which would insert negative reactivity during the accident). Also, the calculations are performed assuming that the worth of the dropped rod, regardless of its position during the startup withdrawal sequence, is added to a critical reactor.

The analysis appropriately assumes that the control rod is dropped from its full inserted position to the position of the drive and accounts for the effects of exposure explicitly.

The staff notes that neither operator error nor calculational biases and uncertainties were included in the calculation. The staff, however, has reviewed the applicability of PANAC11 to evaluating nuclear characteristics for the ESBWR in its review of Reference 8. The staff found that PANAC11 is suitable for calculations of blade worth for the ESBWR. The staff approved previous versions of PANACEA to provide control blade worth and control rod drop shape information to downstream transient evaluations (References 14, 15, 16, and 17). Therefore, the staff is reasonably assured that the calculations are indicative of the expected ESBWR behavior; however, the staff does not find that the brief description of the reload licensing methodology for CRDA is adequate for staff review for generic application to all ESBWR reload licensing evaluations. Therefore, the staff's acceptance of the analytical CRDA results for the initial core design does not constitute staff approval of the reload licensing methodology for CRDA outlined in the RAI response generically.

The staff found that the low enthalpy rises are a result of low blade worth (less than 80 cents in all cases). Therefore, the staff finds that the calculational results indicating large margin are expected. There is reasonable assurance that consideration of modeling biases, uncertainty, and operator error would not result in changes to the analytic result on the order of magnitude of the available margin. The large margins to cladding failure for the ESBWR initial core provide

the staff reasonable assurance that, for the core design described in the subject LTR, the radiological consequences are bounded by the DCD analyses and that barrier integrity has been demonstrated.

On the basis of its review of the information provided in NEDC-33326P, as described above, the staff concludes that the functional design of the ESBWR reactivity control systems meets the requirements of GDC 20, 25, 26, 27 and 28 and, therefore, is acceptable

4.5 Stability and Transient Calculations

The applicant provided extensive analyses demonstrating that the ESBWR is not susceptible to xenon-induced power oscillations in Reference 3. The staff has found that the ESBWR, generally, is not susceptible to xenon-induced power oscillations due to the strong void power coupling through the void reactivity coefficient. Therefore, the staff finds that the ESBWR compliance with GDC 12 need only consider thermal hydraulic instability.

The review of References 3 and 4 is documented in the associated staff SER, which addresses the acceptability of the use of the nuclear design methodology to provide information to the transient reactor analyses.

The staff reviewed the results of the stability analyses in DCD, SER Section 4A for LTR NEDO-33337 (Reference 7).

CONCLUSION

To allow for changes in reactivity from reactor heat up, changes in operating conditions, fuel burnup, and fission product buildup, the applicant has designed a significant amount of excess reactivity into the core. The applicant provided substantial information about core reactivity balances for the initial cycle, and has shown that the design incorporates methods to control excess reactivity at all times. The applicant has shown that sufficient control rod worth would be available at any time during the cycle to shut down the reactor, assuming that the most reactive control rod is stuck in the fully withdrawn position.

The applicant's assessment of reactivity control requirements over the initial cycle is suitably conservative, and the control system has adequate negative worth to ensure shutdown capability.

With respect to the requirements applicable to the nuclear design of the ESBWR, the staff finds the following:

- The applicant has satisfied the requirements of GDC 10, 20, and 25 with respect to fuel design limits by demonstrating that the ESBWR design meets the following objectives:
    - No fuel damage occurs during normal operation, including the effects of AOOs (GDC 10).
    - Automatic initiation of the reactivity control system ensures that fuel design criteria are not exceeded as a result of AOOs and those systems and components important to safety will automatically operate under accident conditions (GDC 20).

- No single malfunction of the reactivity control system will violate the fuel design limits (GDC 25).

- The staff reviewed the results of the applicant's calculations with respect to the Doppler, void, and moderator coefficients of reactivity are negative in the power operating range. The calculations indicated the relative magnitude of the coefficients and nature. They are generally similar to operating BWRs. Accordingly, the applicant has satisfied the requirements of GDC 11 with respect to nuclear feedback characteristics.

- The staff reviewed the applicant's analysis of xenon-induced power oscillations and has determined that the analysis is suitably conservative and performed with appropriate ESBWR inputs. Furthermore, the staff has evaluated how nuclear parameters are translated into TRACG through PANACEA, and finds this method acceptable. Therefore, the staff concluded that xenon-induced power oscillations are not a concern in the requirements of GDC 12, and the PANACEA Wrap-up file is adequate for the purposes of calculating reactor kinetic behavior for stability analyses. The acceptability of the design in terms of GDC 12 is addressed in the staff's SER for LTR NEDO-33337 in regard to thermal-hydraulic instabilities.

- The staff reviewed the applicant's core monitoring system, and finds that the applicant has satisfied the requirements of GDC 13 by providing instrumentation and controls to monitor the fission process. The applicant has also demonstrated the ability of the PANAC11 calculational engine to acceptably determine the margin to safety limits based on plant live data.

- The ESBWR design includes a standby liquid control system, which provide the following capabilities:

  - Reliable shutdown of the reactor during normal operating conditions and during AOOs in the event of multiple failures in the control rod drive system

  - Adequate boron injection capability to maintain safe-shutdown at all times during the reference cycle.

  Accordingly, the staff concludes that the applicant has satisfied the requirements of GDC 26 by providing two independent reactivity control systems of different design and GDC 27 by including a system that injects dissolved absorber through the accumulator driven SLCS.

- The ESBWR control rod system design includes many design features to limit the possibility of a control rod drop accident. Calculations were performed that demonstrate that there are no fuel cladding failures as a result of a postulated control rod drop accident for the initial core design. Therefore, the requirements of GDC 28 are met.

For the reasons set forth above, the staff concludes that the ESBWR nuclear design satisfies the requirements of GDC 10, 11, 13, 20, 25, 26, 27 and 28, and therefore is acceptable.

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