

SAFETY EVALUATION OF GE HITACHI NUCLEAR ENERGY LICENSING TOPICAL REPORT NEDC-33374P, "SAFETY ANALYSIS REPORT FOR FUEL STORAGE RACKS CRITICALITY ANALYSIS FOR ESBWR PLANTS"

1.0 Introduction

By letter dated November 9, 2007 (Reference 1), and in support of its application for design certification of the economic simplified boiling-water reactor (ESBWR), GE Hitachi Nuclear Energy (GEH), in conjunction with its affiliate company, Global Nuclear Fuels (GNF), submitted Licensing Topical Report NEDC-33374P, Revision 0, "Safety Analysis Report for Fuel Storage Racks Criticality Analysis for ESBWR Plants." The submittal was revised by letters dated December 3, 2008, August 22, 2009, and December 5, 2009 (References 2, 3, and 4) for review and approval by the staff of the U.S. Nuclear Regulatory Commission (NRC). Unless otherwise noted, references hereafter to NEDC-33374P indicate Revision 3 (Reference 4). The staff issued requests for additional information (References 5-7), to which the applicant responded by letters dated November 24, 2008, July 22, 2009, July 28, 2009, August 13, 2009, August 22, 2009, and December 4, 2009 (References 8-13). The ESBWR Design Control Document (DCD), Tier 2, Revision 7, Sections 9.1.1.7, 9.1.2.2, and 9.1.2.8 (Reference 14), refers to NEDC-33374P to address criticality control for the new and spent fuel storage racks for the ESBWR.

2.0 Regulatory Criteria

The staff reviewed NEDC-33374P in accordance with NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants" (hereafter referred to as the SRP), Section 9.1.1, Revision 3, "Criticality Safety of Fresh and Spent Fuel Storage and Handling," issued March 2007 (Reference 15). The staff's acceptance of the criticality safety of fresh and spent fuel storage and handling is based on compliance with the following requirements:

- General Design Criterion (GDC) 62, "Prevention of Criticality in Fuel Storage and Handling," in Appendix A, "General Design Criteria for Nuclear Power Plants," to Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50, "Domestic Licensing of Production and Utilization Facilities" (Reference 16), as it relates to the prevention of criticality by physical systems or processes using geometrically safe configurations.
- 10 CFR 50.68, "Criticality Accident Requirements," as it relates to preventing a criticality accident and to mitigating the radiological consequences of a criticality accident.

Acceptance criteria adequate to meet the above requirements include the following:

- American National Standards Institute/American Nuclear Society (ANSI/ANS) Standards 57.1 ("Design Requirements for Light Water Reactor Fuel Handling Systems"), 57.2 ("Design Requirements for Light Water Reactor Spent Fuel Storage Facilities at Nuclear Power Plants"), and 57.3 ("Design Requirements for New Fuel Storage Facilities at Light Water Reactor Plants") (References 17-19) specify the criteria for GDC 62, as they relate to the prevention of criticality accidents in fuel storage and handling. Compliance with GDC 62 requires the prevention of criticality in the fuel storage and handling system through the use of physical systems or processes, with preference given to the application of geometrically safe configurations.

- ANSI/ANS 57.1, ANSI/ANS 57.2, and Regulatory Guide 1.13, “Spent Fuel Storage Facility Design Basis,” (Reference 20) provide guidance acceptable to the staff for meeting the requirements associated with spent fuel storage and handling.
- Compliance with 10 CFR 50.68 requires that the licensee either maintain monitoring systems capable of detecting a criticality accident as described in 10 CFR 70.24, “Criticality Accident Requirements,” (Reference 21) thereby reducing the consequences of a criticality accident, or meet the requirements specified in 10 CFR 50.68(b), thereby reducing the likelihood that a criticality accident will occur.

3.0 Summary of Technical Information

In NEDC-33374P, the applicant described the criticality analyses and results for the ESBWR spent fuel and buffer pools for the storage of 10x10 GE14E fuel bundles in new and spent fuel storage racks to be supplied by Equipos Nucleares SA (ENSA). NEDC-33374P provides details on the methodology and analytical models used in the criticality analysis to represent the storage rack systems.

In NEDC-33374P, the applicant generated in-core k-infinity values and exposure-dependent, pin-by-pin isotopic specifications using the GEH/GNF lattice physics production code TGBLA06A (Reference 22). TGBLA06A solves two-dimensional diffusion equations with diffusion parameters corrected by transport theory to provide system multiplication factors and to perform burnup calculations.

The fuel storage criticality calculations are then performed using MCNP-05P, the GEH/GNF proprietary version of MCNP5 (Reference 23). MCNP-05P is a Monte Carlo program for solving the linear neutron transport equation for a fixed source or an eigenvalue problem. The code implements the Monte Carlo process for neutron, photon, or electron transport or coupled transport involving all these particles, and can compute the eigenvalue for neutron-multiplying systems. For the present application, only neutron transport was considered.

The racks are analyzed using the MCNP-05P Monte Carlo neutron transport program (Reference 23) and the cold in-core k-infinity criterion methodology. A maximum cold, uncontrolled peak in-core eigenvalue k-infinity of 1.32, as defined using the lattice physics code TGBLA06A (Reference 22), is specified as the rack design limit for GE14E fuel in both the new and spent fuel racks. For both racks, the analyses resulted in a storage rack maximum k-effective ($k(95/95)$) less than 0.95 for normal and credible abnormal operation, with tolerances and uncertainties taken into account.

The analyses provided in the topical report demonstrate that, for both the new and spent fuel storage racks, the k-effective value is less than 0.95, with a 95-percent probability and a 95-percent confidence level, for all normal and credible abnormal conditions, with tolerances and computational uncertainties taken into account.

4.0 Staff Evaluation

The staff reviewed NEDC-33374P in accordance with the guidance of SRP Section 9.1.1, Revision 3 (Reference 15). The staff also compared the information in NEDC-33374P to DCD, Tier 2, Revision 7, Chapter 9 and Section 4.3.1 of the technical specifications, issued March 2010 (Reference 14).

The staff verified that the design of the new and spent fuel storage racks complied with the requirements of GDC 62. The applicant addressed the guidelines of Regulatory Guide 1.13, ANSI/ANS 57.1-1992 and ANSI/ANS 57.2-1983 for spent fuel storage, and ANSI/ANS 57.3-1983 for new fuel storage in the design certification, and therefore, these guidelines are outside the scope of NEDC-33374P. The staff evaluation of these guidelines is in Section 9.1 of the safety evaluation report (SER) for the design certification.

The criticality evaluation for the handling of fresh and spent fuel is outside the scope of NEDC-33374P as it is addressed by combined license (COL) item 9.1.4-A, "Fuel Handling Operations," in the design certification. The staff evaluation of the COL item 9.1-4-A is in the SER for the design certification.

The evaluation of the performance effectiveness of neutron-absorbing materials was limited to aspects relevant to the criticality analysis presented in the topical report. The review did not consider other issues, such as the effectiveness of the periodic surveillance program, material compatibility, and structural integrity.

4.1 Fuel Assembly Design Data

The staff verified that the criticality analysis used the appropriate fuel design data. NEDC-33374, Section 4, describes fuel design basis, which is the GE14E fuel design, and the fuel criticality model. The staff reviewed GE14E fuel design specification documents and drawings during two audits in 2009 (Reference 24) and determined that the fuel design basis is consistent with the detailed fuel design information and that appropriate fuel design information was incorporated into the criticality models. Accordingly, the staff finds that appropriate fuel assembly data, including design tolerances, were used in the criticality analysis.

4.2 Fuel Storage Rack Design Data

The staff verified that the criticality analysis used the appropriate fuel rack design data. Sections 5 and 6 of NEDC-33374 describe new and spent fuel racks and their criticality models. The staff reviewed the fuel rack design specification documents and drawings during two audits (Reference 24) and determined that the fuel rack design descriptions are consistent with the detailed fuel rack design information and that appropriate fuel rack design information was incorporated into the criticality models. Accordingly, the staff finds that appropriate fuel storage rack data, including design tolerances, were used in the criticality analysis.

4.3 Identification of Computational Methods

The staff verified that the criticality analysis uses appropriate computational methods and describes the computational methods used.

NEDC-33374P, Section 3, describes the two computational methods used in the criticality analysis. The applicant's lattice design code, TGBLA06, was used to calculate burned fuel

compositions and the “in-core K_{∞} ” values. The burned fuel compositions were then used in the MCNP-05P calculations to obtain fuel storage rack k-effective values.

TGBLA06A is a two-dimensional lattice design computer program for boiling-water reactor (BWR) fuel bundle analysis. It assumes that a lattice is uniform and infinitely long along the axial direction and that the lattice geometry and material are reflecting with respect to the lattice boundary along the transverse directions.

TGBLA06A uses ENDF/B-V (evaluated nuclear data file) cross-section data to perform coarse-mesh, broad-group, and diffusion theory calculations. It includes thermal neutron scattering with hydrogen using an $S(a,\beta)$ light-water thermal scattering kernel.

The staff previously reviewed and accepted the use of TGBLA06 for BWR core depletion calculations, similar to those described in Section 3.3 of NEDE-33374P, as part of the approval of Amendment 26 to NEDE-24011-P-A, “GESTAR II—Implementing Improved GE Steady-State Methods,” dated November 10, 1999 (Reference 25), for operating BWRs. The staff reviewed and accepted the use of TGBLA06 for ESBWR applications in the SER for NEDC-33239P, Revision 4, “GE14 for ESBWR Nuclear Design Report” (Reference 26). The staff confirmed that the use of TGBLA06 for the analyses in NEDE-33374P is within the scope of the prior reviews, considering that the analysis is applicable only to GE14E fuel in the ESBWR design.

MCNP-05P implements a robust geometry representation that can correctly model complex components in three dimensions. An arbitrary three-dimensional configuration is treated as geometric cells bounded by first- and second-degree surfaces and some special fourth-degree elliptical tori. The cells are described in a Cartesian coordinate system and are defined by the intersections, unions, and complements of the regions bounded by the surfaces. Surfaces are defined by supplying coefficients to the analytic surface equations or, for certain types of surfaces, known points on the surfaces. Rather than combining several predefined geometrical bodies in a combinatorial geometry scheme, MCNP-05P has the flexibility of defining geometrical shapes from all the first- and second-degree surfaces of analytical geometry and elliptical tori and then combining them with Boolean operators. The code performs extensive checking for geometry errors and provides a plotting feature for examining the geometry and material assignments.

MCNP-05P uses point-wise (i.e., continuous) cross-section data, and all reactions in a given cross-section evaluation (e.g., ENDF/B-VII.0) are considered. Cross-section tables include all details of the ENDF representations for neutron data. The code requires that all the cross sections be given on a single union energy grid suitable for linear interpolation; however, the cross-section energy grid varies from isotope to isotope.

The staff has routinely approved MCNP-05P for benchmarking nuclear design methods, including TGBLA06 comparisons for the ESBWR, as noted in the SER for NEDC-33239P (Reference 26). MCNP is also a generally accepted code used for criticality analyses, provided that it is properly validated (discussed in Section 4.4 below). The staff considers the general use of MCNP for the criticality analyses of spent fuel racks to be similar to these other applications and therefore acceptable.

Based on the above, the staff finds that acceptable computational methods are used in the criticality analysis and are adequately described.

4.4 Evaluation of Computational Method Validation

The staff verified that the validation study is thorough and uses benchmark critical experiments that are similar to the normal-conditions and credible abnormal-conditions models and to confirm that the k-effective bias and bias uncertainty values are properly determined. In addition, the criticality analysis should include a detailed description of the validation study supporting the criticality analysis.

In NEDC-33374P, Section 3.3 and Appendix B describe the computational method validation. Section 3.3 describes the critical experiments used in the validation study that form the validation basis for the computational method. This section includes a summary of the critical benchmark experiments and the area of applicability covered by the code validation. The validation set did not include critical experiments with plutonium, uranium, and fission product compositions adequately similar to the plutonium, uranium, and fission product compositions for burned fuel. The depletion uncertainty validation gap was addressed by adopting a decrement penalty of 5 percent of the reactivity, as suggested in the Kopp memorandum (Reference 27). The NRC has accepted this practice in past reviews. The fission product and actinide cross section uncertainties were addressed in a similar manner by adopting a decrement penalty of 5 percent of the reactivity worth of these isotopes.

The staff reviewed validation critical experiment information during two audits (Reference 24) and determined that the summary validation information in NEDC-33374, Section 3.3, is consistent with the detailed validation critical experiment information. The validation critical experiment set included critical experiments with energy of average lethargy of neutrons causing fission (EALF) that are similar to the normal and limiting abnormal-conditions models. A small set of mixed plutonium dioxide and uranium dioxide critical experiments were included in the validation database. No anomalous results were observed that would indicate a problem with the plutonium cross sections.

NEDC-33374P, Appendix B, describes the determination of the bias and bias uncertainty. The staff reviewed the method used and confirmed the results by comparison to limits calculated using an alternate method, the single-sided upper tolerance limit method, as described in NUREG/CR-6698, "Guide for Validation of Nuclear Criticality Safety Computational Methodology," issued January 2001 (Reference 28). The staff finds that the bias and bias uncertainty were correctly determined from the validation database.

Based on the above, the staff finds the computational method validation acceptable because the validation was performed using an appropriate set of critical experiments and because the applicant determined the bias and bias uncertainty by using valid statistical analysis methods.

4.5 Identification of Normal Conditions

The staff verified that the criticality analysis provides descriptions and definitions of normal conditions associated with the storage of fresh and spent fuel. In addition to being used to evaluate the subcriticality of normal conditions, the described normal conditions serve as the basis of initial conditions for evaluating abnormal conditions. The staff verified that the descriptions of normal conditions are comprehensive and that the logic used to envelop one or more classes of normal conditions is sound.

NEDC-33374P, Section 5.4, describes the applicant's normal condition analysis for new fuel storage. The applicant evaluated three classes of normal configurations: (1) fuel assembly

channeling, (2) eccentric loadings, and (3) pool moderator temperature variation. Through this evaluation, the applicant identified the bounding normal condition for new fuel storage, which is listed in NEDC-33374P, Table 4, and shown in Figure 7.

NEDC-33374P, Section 6.4, describes the applicant's normal condition analysis for spent fuel storage. The applicant evaluated four classes of normal configurations: (1) fuel assembly channeling, (2) eccentric loadings, (3) rotated bundles, and (4) pool moderator temperature variation. Through this evaluation, the applicant identified the bounding normal condition for spent fuel storage, which is listed in NEDC-33374P, Table 11, and shown in Figure 15.

The staff reviewed the detailed normal condition evaluations during the two audits (Reference 24) and determined that the summary information in NEDC-33374P, Section 5.4, is consistent with the detailed information on normal conditions. The staff finds that the normal conditions are appropriately described and comprehensive. The logic used to identify bounding normal conditions is sound and appropriate. The staff also finds that the normal conditions serve as the basis of initial conditions for evaluating abnormal conditions.

4.6 Analysis of Normal-Conditions Models

The staff evaluated the normal-conditions models to verify that normal conditions are modeled correctly and that all modeling approximations and assumptions are appropriate.

NEDC-33374P, Section 5, describes the new fuel storage rack normal conditions, including the dimensions and tolerances considered in the criticality analysis. The new fuel storage area in the buffer pool comprises 17 2x14 fuel racks. Each rack is fabricated from stainless steel plates forming 2x7 lattices, with side doors to introduce the fuel bundles. Two 2x7 fuel racks are joined and installed together forming a 2x14 rack. At the bottom of the rack cells is a circular drilled plate, placed above the base plate, to allocate the bottom fuel nozzle. The base plate rises 11 centimeters (cm) above the floor and is attached to it; therefore, the only possible displacements are rack bow at the upper edge and bundle displacement inside the rack cells.

The new fuel storage racks are distributed in three rows, one with nine racks and the other two with four racks each. There is a minimum distance of 30 cm between the largest row and the group of the other two. The two groups of four racks are assumed to be in contact. The minimum distance between side surfaces of the racks and pool walls is 30 cm, while the minimum distance between the racks' end and walls is 10 cm. NEDC-33374P, Figure 4, illustrates the general new fuel rack arrangement in the buffer pool. NEDC-33374P, Figure 5, illustrates a single storage element of the new fuel rack, with dimensions and tolerances presented in NEDC-33374P, Table 3. A two-dimensional, infinite model was used to conservatively describe the new fuel rack storage system in MCNP-05P. The only structural material included in the model is the stainless steel rack wall, as the cell doors do not cover the complete active fuel length.

NEDC-33374P, Section 6, describes the spent fuel storage rack normal conditions, including the dimensions and tolerances considered in the criticality analysis. Each spent fuel rack is fabricated using an array of borated stainless steel storage cells with an outer frame of stainless steel. All spent fuel bundles are placed in spent fuel racks from the top. At any given axial height, each borated stainless steel cell is bounded by four interlocking panels that run the horizontal length of the array. Five of these borated panels, stacked one on top of another, extend from the base plate of the storage rack to above the top of active fuel. These interlocking panels that form the fuel element interior matrix are type 304B7 borated stainless

steel, consistent with American Society of Testing and Materials (ASTM) A887, "Standard Specification for Borated Stainless Steel Plate, Sheet, and Strip for Nuclear Application," (Reference 29). An additional plate made of stainless steel makes up the sixth and top row of the fuel rack, which extends to 358.7 cm above the top of the base plate. The panels are locked in place by lining up slots of perpendicular panels with one another and locking the tab of one panel into its respective hole on the neighboring panel above or below it. The exterior panels on each rack module are made of ordinary (nonborated) stainless steel. NEDC-33374P, Figure 10, illustrates how the slotted steel panels are assembled to construct a spent fuel storage rack module.

The scope of NEDC-33374P is limited to fuel storage in the fresh and spent fuel storage racks. Fuel handling is covered only in regard to how it affects fuel in the fuel storage racks. According to NEDC-33374P, no more than one fuel assembly will be handled around or above the fuel storage racks at any one time. Normal conditions evaluated include nominal conditions, eccentric assembly placement, assembly grouping, assembly rotations, and an assembly placed next to the storage racks. Spent fuel storage rack spacing has been restricted to prevent placement of an assembly between storage racks.

NEDC-33374P, Section 3.6, identifies modeling assumptions and conservatisms used in the criticality analysis. This section describes assumptions regarding boundary conditions (neutron leakage and absorption), depletion calculations, omitted materials, and material and geometry simplifications. The staff finds that this section identifies assumptions consistent with the guidelines in SRP Section 9.1.1.

The staff reviewed a few representative examples of the MCNP input files for normal conditions during the two audits (Reference 24) and determined that the MCNP model input information is consistent with the information on normal conditions in NEDC-33374P, Sections 5 and 6. The staff also determined that the MCNP input models implemented the modeling assumptions and simplifications identified in NEDC-33374P, Section 3.6.

The staff finds that the modeling of design and manufacturing tolerances and uncertainties is adequately addressed either by modeling some tolerances and uncertainties directly using conservatively bounding parameters or by including them in the rack-specific uncertainty analyses. Accordingly, the staff finds that the normal conditions are modeled correctly and that the modeling simplifications, approximations, and assumptions are appropriate.

4.7 Identification of Abnormal Conditions

The staff evaluated abnormal conditions identified by the applicant to verify that the scope of considered abnormal conditions is comprehensive.

In NEDC-33374P, Sections 5 and 6 identify abnormal conditions related to fuel assemblies being dropped, misallocated fuel assemblies, inadvertent release of fuel from new fuel storage racks, storage rack sliding or tipping, and loss of boron from the borated stainless steel in the spent fuel storage racks. Loss of boron from the borated stainless steel was screened as not being a credible failure.

The staff examined whether the miscalculation or misassignment of the lattice-specific, cold, in-core k-infinity value for an assembly should be considered a credible failure. This parameter is compared with the limiting value of 1.32 to determine acceptability for storage in fresh or spent fuel racks. NEDC-33374P, Section 3.4, describes the process the applicant uses to determine

that specific GE14E assembly designs meet the lattice-specific, cold, in-core k-infinity value. During two audits (Reference 24), the staff reviewed supporting information on the determination of the lattice-specific, cold, in-core k-infinity value. The staff has previously approved the use of the lattice-specific, cold, in-core k-infinity value as a controlling parameter for spent fuel pool criticality as part of the approval of Amendment 26 to NEDE-24011-P-A (Reference 25). In addition, the lattice-specific, cold, in-core k-infinity value is included in the design certification technical specifications. Based on the controls identified by the applicant and previous precedents accepting this approach, and with no evidence to the contrary, the staff finds this approach acceptable.

Soluble boron is not present under normal conditions in BWRs and fuel storage facilities. Consequently, the staff finds that there is no need to evaluate inadvertent deboration of the fresh and spent fuel pools.

Abnormal conditions related to dropped and misallocated fuel assemblies and related to spent fuel storage rack sliding were evaluated. The spacing between spent fuel storage racks is limited such that a fuel assembly cannot be inserted between rack modules. The applicant modeled fuel assemblies at various locations next to the fresh and spent fuel storage racks. Because of the configuration of the fresh and spent fuel storage racks, the staff finds that this is an appropriate means of modeling dropped and misallocated fuel assemblies.

Based on the above, the staff finds the scope of the abnormal conditions considered in the topical report to be comprehensive and thus acceptable.

4.8 Analysis of Abnormal Conditions

The staff evaluated the abnormal-conditions models to verify that abnormal conditions were modeled correctly and that all modeling approximations and assumptions were appropriate.

NEDC-33374P, Section 5, describes the new fuel storage rack abnormal conditions, including the dimensions and tolerances considered in the criticality analysis. The abnormal conditions include a single fuel assembly in various positions external to the new fuel storage racks, as shown in NEDC-33374P, Figures 8 and 9. The applicant described how other potential abnormal configurations of new fuel storage racks are bounded by these configurations.

NEDC-33374P, Section 6, describes the spent fuel storage rack abnormal conditions, including the dimensions and tolerances considered in the criticality analysis. The abnormal conditions include alternative depletion conditions, modified rack spacing, a single fuel assembly in various positions external to the spent fuel storage racks, and filling various part-length rod locations with fuel material to represent dropped or damaged fuel configurations. The applicant described how these configurations bound other potential abnormal configurations of spent fuel storage racks.

The applicant accounted for the abnormal configurations by including them in the rack-specific bias analyses. For the new fuel storage racks, this was done by determining the most reactive external assembly location. NEDC-33374P, Table 12, identifies the abnormal configurations included in the bias analyses. The treatment of modeling assumptions and conservatism and the modeling of design and manufacturing tolerances for abnormal conditions are the same as for normal conditions and are considered acceptable, since they are consistent with the guidelines of SRP Section 9.1.1.

The staff reviewed representative examples of the MCNP input files for abnormal conditions during the two audits (Reference 24) and determined that the MCNP model input information is consistent with the information on abnormal conditions in NEDC-33374P, Sections 5 and 6. The staff also determined that the MCNP input models implemented the modeling assumptions and simplifications identified in NEDC-33374P, Section 3.6.

Based on the above, the staff finds that the abnormal-conditions models conservatively simulate the actual abnormal conditions, and the modeling approximations and assumptions are appropriate.

4.9 Analysis of Normal and Credible Abnormal-Conditions Logic

The staff evaluated the analysis of normal and credible abnormal conditions to verify that the analysis is complete and logically sound and that assumptions, limits, and controls are clearly stated.

The staff reviewed the logic behind the analysis and conclusions. In NEDC-33374P, Sections 5 and 6 describe the applicant's analysis of normal and credible abnormal conditions. As described in Sections 4.5 to 4.8 of this report, the staff finds that sound, technically based logic was followed to determine potential configurations and that the analysis of normal and abnormal conditions is sound, logical, and complete. NEDC-33374P, Section 3.6, clearly and appropriately describes simplifications, approximations, and assumptions.

The limits and controls are related primarily to the new and spent fuel storage rack designs and to storage of only the GE14E fuel design. NEDC-33374P, Appendix A, identifies the essential parameters for subcriticality regarding the new and spent fuel storage racks and GE14E fuel design. The limits and controls are based on geometrically safe configurations consistent with the guidelines of SRP Section 9.1.1. Accordingly, the staff finds that the analysis of normal and credible abnormal conditions is acceptable and that assumptions, limits, and controls are clearly identified.

4.10 Analysis Conclusions

The staff verified that the criticality analysis supports the applicant's conclusions and that the conclusions are consistent with the applicable requirements.

In NEDC-33374P, Tables 7 and 15 provide the analytical results of the criticality analyses for the new and spent fuel storage racks, respectively. Based on these results and the preceding discussions, the staff agrees with the applicant's conclusion that the normal and credible abnormal conditions were properly identified and were evaluated to be acceptable (i.e., k-effective less than 0.95, with a 95-percent probability with a 95-percent confidence level) for the fresh and spent fuel storage racks in the reactor building buffer pool and for the spent fuel racks in the fuel building spent fuel pool.

Compliance with General Design Criterion 62

Based on the above, the staff finds that the design of the new and spent fuel storage racks provides reasonable assurance that fuel will remain subcritical under all normal and credible abnormal conditions. Accordingly, the staff finds that new and spent fuel storage racks meet the applicable requirements of GDC 62.

Compliance with 10 CFR 50.68

The staff verified that the design complies with the requirements of 10 CFR 50.68. The applicant has addressed the requirements of 10 CFR 50.68 by compliance with the additional design and analysis requirements specified in 10 CFR 50.68(b)(2) and (b)(4).

The regulation in 10 CFR 50.68(b)(2) requires that the estimated ratio of neutron production to neutron absorption and leakage (k-effective) of the fresh fuel in the fresh fuel storage racks be calculated assuming that the racks are loaded with fuel of the maximum fuel assembly reactivity and flooded with unborated water, and must not exceed 0.95, at a 95-percent probability, 95-percent confidence level. The ESBWR new (fresh) fuel storage racks are designed for unborated water and meet the k-effective criteria as discussed above. Accordingly, the staff finds that the new fuel storage racks meet the requirements of 10 CFR 50.68(b)(2).

The regulation in 10 CFR 50.68(b)(4) requires that, if no credit is taken for soluble boron, the k-effective of the spent fuel storage racks loaded with fuel of the maximum fuel assembly reactivity must not exceed 0.95, at a 95-percent probability, 95-percent confidence level, if flooded with unborated water. The ESBWR spent fuel storage racks are designed for unborated water and meet the k-effective criteria as discussed above. Accordingly, the staff finds that the spent fuel storage racks meet the requirements of 10 CFR 50.68(b)(4).

Based on the above, the staff finds that new and spent fuel storage racks meet the applicable requirements of 10 CFR 50.68.

Determination of NEDC-33374P as a Tier 2* Report

The criticality analysis is performed specifically for the GE14E fuel design. The criticality analysis is dependent on the dimensional and material tolerances ranges specified in NEDC-33374P, Appendix A. Exceptions to the fuel design or the storage rack dimensional and material tolerances could impact the conclusions of the criticality analyses and therefore must be submitted to the NRC for review prior to use. Accordingly, the staff determined that NEDC-33374P should be designated as Tier 2*. The staff notes that NEDC-33374P is designated as Tier 2* in the design certification.

5.0 Conclusions

The criticality analyses in NEDC-33374P show that for both the new and spent fuel storage racks the k-effective value is less than 0.95, with a 95-percent probability and a 95-percent confidence level, for all normal and credible abnormal conditions, including unborated water, and with tolerances and computational uncertainties taken into account. Accordingly, the staff finds that the design of the new and spent fuel storage racks provides reasonable assurance that fuel will remain subcritical under all normal and credible abnormal conditions and therefore will meet the applicable requirements of GDC 62 and 10 CFR 50.68. The staff also finds that NEDC-33374P should be designated Tier 2*.

6.0 References

1. Letter from James C. Kinsey, GE Hitachi Nuclear Energy, to the U.S. Nuclear Regulatory Commission, "Transmittal of Licensing Topical Report NEDC-33374P, 'Safety Analysis Report for Fuel Storage Racks Criticality Analysis for ESBWR Plants,' November 2007," MFN 07-603, November 9, 2007 (ADAMS Accession No. ML073180182, transmittal; ML073180186, NEDO-33374, Revision 0; ML073180187, NEDC-33374, Revision 0).
2. Letter from Richard E. Kingston, GE Hitachi Nuclear Energy, to the U.S. Nuclear Regulatory Commission, "Submittal of Licensing Topical Report NEDC-33374P, Revision 1, 'Safety Analysis Report for Fuel Storage Racks Criticality Analysis for ESBWR Plants,' December 2008," MFN 08-933, December 3, 2008 (ADAMS Accession No. ML083460076, transmittal, ML083460077, NEDC-33374, Revision 1).
3. Letter from Richard E. Kingston, GE Hitachi Nuclear Energy, to the U.S. Nuclear Regulatory Commission, "Transmittal of GEH Licensing Topical Report NEDC-33374P, 'Safety Analysis Report for Fuel Storage Racks Criticality Analysis for ESBWR Plants,' Revision 2, August 2009," MFN 09-561, August 22, 2009 (ADAMS Accession No. ML092380441, transmittal, ML092380442, NEDC-33374, Revision 2).
4. Letter from Richard E. Kingston, GE Hitachi Nuclear Energy, to the U.S. Nuclear Regulatory Commission, "Transmittal of GEH Licensing Topical Report NEDC-33374P, 'Safety Analysis Report for Fuel Storage Racks Criticality Analysis for ESBWR Plants,' Revision 3, August 2009," MFN 09-764, December 5, 2009 (ADAMS Accession No. ML093421411, transmittal, ML093421412, NEDC-33374, Revision 3).
5. Letter from Bruce Baval, U.S. Nuclear Regulatory Commission, to Robert E. Brown, GE Hitachi Nuclear Energy, "Request for Additional Information Letter No. 217 Related to NEDC-33374P, 'Safety Analysis Report for Fuel Storage Racks Criticality Analysis for ESBWR Plants,'" MFN 08-551, June 25, 2008 (ADAMS Accession No. ML081770230).
6. Letter from Dennis Galvin, U.S. Nuclear Regulatory Commission, to Jerald G. Head, GE Hitachi Nuclear Energy, "Request for Additional Information Letter No. 304 Related to Design Control Document (DCD) Revision 5," MFN 09-227, March 31, 2009 (ADAMS Accession No. ML090330425).
7. Letter from Dennis Galvin, U.S. Nuclear Regulatory Commission, to Jerald G. Head, GE Hitachi Nuclear Energy, "Request for Additional Information Letter No. 380 Related to Design Control Document (DCD) Revision 6," October 28, 2009 (ADAMS Accession No. ML092960405).
8. Letter from Richard E. Kingston, GE Hitachi Nuclear Energy, to the U.S. Nuclear Regulatory Commission, "Response to NRC Request for Additional Information Letter No. 217 Related to Licensing Topical Report NEDC-33374P, Revision 0, 'Safety Analysis Report for Fuel Storage Racks Criticality Analysis for ESBWR Plants'—RAI Numbers 9.1-77 through 9.1-95," MFN 08-912, November 24, 2008 (ADAMS Accession No. ML083310721).
9. Letter from Richard E. Kingston, GE Hitachi Nuclear Energy, to the U.S. Nuclear Regulatory Commission, "Response to Portion of NRC Request for Additional Information Letter No. 304 Related to Design Control Document (DCD) Revision 5—

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