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PROPRIETARY INFORMATION – WITHHOLD UNDER 10 CFR 2.390

RS-21-065 10 CFR 50.90

October 25, 2021

U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, DC 20555-0001

Quad Cities Nuclear Power Station, Units 1 and 2
Renewed Facility Operating License Nos. DPR-29 and DPR-30
NRC Docket Nos. 50-254 and 50-265

Subject: License Amendment Request Regarding New Fuel Storage Vault and Spent Fuel Storage Pool Criticality Methodologies with Proposed Change to Technical Specifications Section 4.3.1

Reference: Public Pre-submittal Meeting Between Exelon Generation Company, LLC and the U.S. Nuclear Regulatory Commission, "Proposed License Amendment Request Associated with the Transition to a New Fuel Type and Vendor at LaSalle Station, Units 1 & 2 and Quad Cities Nuclear Power Station, Units 1 & 2," May 27, 2021 (ADAMS Accession Nos. ML21133A167 & ML21141A010)

In accordance with 10 CFR 50.90, "Application for amendment of license, construction permit, or early site permit," Exelon Generation Company, LLC (EGC) requests an amendment to Renewed Facility Operating License Nos. DPR-29 and DPR-30 for Quad Cities Nuclear Power Station (QCNPS), Units 1 and 2, respectively. Specifically, EGC is utilizing a new criticality safety analysis (CSA) methodology for performing the criticality safety evaluation for legacy fuel types in addition to the GNF3 reload fuel in the spent fuel pool (SFP). Use of the new SFP CSA methodology necessitates a change to the QCNPS Technical Specifications (TS) 4.3.1, "Criticality." EGC is also proposing a change to the new fuel vault (NFV) CSA to utilize the GESTAR II methodology for validating the NFV criticality safety for GNF3 fuel in the General Electric (GE) designed NFV racks.

EGC participated in a pre-submittal meeting with the NRC (see Reference) regarding the planned transition from the Framatome ATRIUM 10XM fuel design to the Global Nuclear Fuels - America, LLC (GNF) GNF3 fuel design at QCNPS. During this meeting, EGC's plans to submit this amendment request supporting both the use of a new CSA methodology for performing the criticality safety evaluation in the spent fuel pools and returning to the GNF GESTAR II coverage as the CSA basis in the NFV were discussed. A separate amendment request will be

Attachment 7 contains Proprietary Information. Withhold from public disclosure under 10 CFR 2.390. When separated from Attachment 7, this document is decontrolled.

submitted to support the transition to GNF3 fuel at QCNPS. While the revised SFP CSA and the altered NFV CSA basis support the planned transition to GNF3 fuel, neither the new analysis or the altered analysis basis is required to support the NRC review and approval of the separate fuel transition amendment request planned for submittal in late summer 2021.

The following attachments are included in support of this proposed license amendment:

Attachment 1: Evaluation of Proposed Changes

Attachment 2: Mark-up of QCNPS, Units 1 and 2 Technical Specifications Pages

Attachment 3: NEDO-33932, "Quad Cities Units 1 and 2 Fuel Storage Criticality Safety Analysis," Revision 1, dated October 2021 (**Non-Proprietary Version**)

Attachment 4: NEI 12-16 Criticality Analysis Checklist

Attachment 5: GNF 10 CFR 2.390 Affidavit

Attachment 6: Curtiss-Wright Corporation (Curtiss-Wright) 10 CFR 2.390 Affidavit

Attachment 7: NEDC-33932P, "Quad Cities Units 1 and 2 Fuel Storage Criticality Safety Analysis," Revision 1, dated October 2021 (**Proprietary Version**)

Attachment 7 contains information proprietary to GNF and Curtiss-Wright. As a result, this document is supported by signed affidavits from the owners of the information, which are included as Attachments 5 and 6, respectively. Each affidavit sets forth the basis on which the corporation's information may be withheld from public disclosure by the NRC and addresses with specificity the considerations listed in paragraph (b)(4) of 10 CFR 2.390, "Public inspections, exemptions, requests for withholding." Accordingly, it is respectfully requested that the information, which is proprietary to GNF and Curtiss-Wright be withheld from public disclosure. A redacted non-proprietary version of the report is provided in Attachment 3.

The proposed changes have been evaluated in accordance with 10 CFR 50.91(a)(1) using the criteria in 10 CFR 50.92(c), and it has been determined that the proposed changes involve no significant hazards consideration.

The proposed changes have been reviewed by the QCNPS Plant Operations Review Committee in accordance with the requirements of the EGC Quality Assurance Program.

EGC requests approval of the proposed changes by October 25, 2022. Once approved, the amendment will be implemented within 60 days. This implementation period will provide adequate time for the affected station documents to be revised using the appropriate change control mechanisms.

In accordance with 10 CFR 50.91, "Notice for public comment; State consultation," paragraph (b), a copy of this letter and its attachments, is being provided to the designated State Officials.

There are no regulatory commitments contained in this letter. Should you have any questions concerning this letter, please contact Ms. Rebecca L. Steinman at (630) 657-2831.

I declare under penalty of perjury that the foregoing is true and correct. Executed on the 25th day of October 2021.

Respectfully,

Patrick R. Simpson
Sr. Manager Licensing
Exelon Generation Company, LLC

Attachments:
1. Evaluation of Proposed Changes
2. Mark-up of QCNPS, Units 1 and 2 Technical Specifications Pages
3. NEDO-33932, "Quad Cities Units 1 and 2 Fuel Storage Criticality Safety Analysis," Revision 1, dated October 2021 (**Non-Proprietary Version**)
4. NEI 12-16 Criticality Analysis Checklist
5. Global Nuclear Fuels - Americas, LLC 10 CFR 2.390 Affidavit
6. Curtis-Wright Flow Control and Services Corporation 10 CFR 2.390 Affidavit
7. NEDC-33932P, "Quad Cities Units 1 and 2 Fuel Storage Criticality Safety Analysis," Revision 1, dated October 2021 (**Proprietary Version**)

cc: U.S. NRC Region III, Regional Administrator
U.S. NRC Senior Resident Inspector, Quad Cities Nuclear Power Station
Illinois Emergency Management Agency – Division of Nuclear Safety

Subject: License Amendment Request Regarding New Fuel Storage Vault and Spent Fuel Storage Pool Criticality Methodologies with Proposed Change to Technical Specifications Section 4.3.1

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1.0 SUMMARY DESCRIPTION

In accordance with 10 CFR 50.90, "Application for amendment of license, construction permit, or early site permit," Exelon Generation Company, LLC (EGC) requests an amendment to Renewed Facility Operating License Nos. DPR-29 and DPR-30 for Quad Cities Nuclear Power Station (QCNPS), Units 1 and 2, respectively. Specifically, EGC is utilizing a new criticality safety analysis (CSA) methodology (Reference 6.1) for performing the criticality safety evaluation for legacy fuel types in addition to the new GNF3 reload fuel design in the spent fuel pool (SFP). Use of the new SFP CSA methodology necessitates a change to the QCNPS Technical Specifications (TS) 4.3.1, "Criticality." EGC is also proposing a change to the new fuel vault (NFV) CSA to utilize the GESTAR II methodology (Reference 6.4, Section 3.5) for validating the NFV criticality safety for GNF3 fuel in the General Electric (GE) designed NFV racks.

2.0 DETAILED DESCRIPTION

2.1 Spent Fuel Pool Criticality Safety Analysis

EGC intends to transition from the ATRIUM 10XM fuel design to the GNF3 fuel design at QCNPS beginning in the spring of 2023. The previous SFP CSAs (see References 6.7) were prepared by Holtec International Inc. (Holtec). The CSA for the QCNPS SFPs is now being rebaselined by GNF to:

- Simplify the validation of GNF3 fuel designs against the CSA criteria. The new analysis will move QCNPS away from the need to validate the in-rack k_{inf} value for each new lattice design to now validating the in-core standard cold core geometry (SCCG) k_{inf} value against the defined limit. The SCCG k_{inf} value is generated for every lattice in each assembly design as part of the standard calculation set.

- Improve consistency among the Boiling Water Reactor (BWR) criticality safety analyses of record (AOR) methods utilized across the fleet. This also includes the methods utilized to verify new GNF3 fuel designs against the criticality safety AOR limitations as listed in the Technical Specifications.

The reason for this license amendment is the rebaselined SFP CSA's change from Holtec methodology to GNF methodology. The proposed methodology change requires NRC approval prior to using the CSA in support of storage of fuel in the QCNPS Unit 1 and Unit 2 SFPs. The QCNPS Unit 1 and Unit 2 SFP racks are designed to accommodate BWR fuel. Both pools' SFP racks credit Curtiss-Wright's NETCO-SNAP-IN rack inserts made of Boralcan. The SFP analysis does not credit any residual Boraflex material that may remain in the rack walls in the same manner as the previous NRC approved CSA for the introduction of rack inserts to the QCNPS SFPs (References 6.7). The revised analysis shows that the effective neutron multiplication factor (k_{eff}) in the SFP racks fully loaded with fuel of the highest anticipated reactivity, at a temperature corresponding to the highest reactivity, does not exceed the regulatory limit of 0.95 at a 95 percent probability, 95 percent confidence level as required by 10 CFR 50.68(b) (e.g., QCNPS complies with the requirements specified in 10 CFR 50.68(b) instead of maintaining monitoring systems as described in 10 CFR 70.24). Reactivity effects of abnormal and accident conditions are also evaluated to assure that under credible abnormal and accident conditions, the reactivity will not exceed the regulatory limit.

The SFP analysis is performed consistent with 10 CFR 50.68 requirements and industry guidance, including Nuclear Energy Institute (NEI) Report 12-16, Revision 4, "Guidance for Performing Criticality Analyses of Fuel Storage at Light-Water Reactor Power Plants" (Reference 6.2). Guidance pertaining to soluble boron in the SFP is not applicable because QCNPS is a BWR plant and has no soluble boron in the SFP. The calculations are performed using GNF's method of analyzing SCCG k_{inf} values and in-rack k_{inf} values and validating the linear correlation between these parameters across a wide range of k_{inf} values. The method then demonstrates that maintaining all fuel below the chosen SCCG k_{inf} upper limit results in an in-rack k_{eff} value no greater than 0.95 after accounting for biases and uncertainties (i.e., k_{max}(95/95) ≤ 0.95). A copy of the NEI 12-16 Criticality Analysis Checklist is included as Attachment 4 to identify the areas of the analysis that conform or do not conform to the guidance in NEI 12-16. Additional information is provided for any deviation from NEI 12-16 in the Attachment 4 checklist.

The change in the SFP CSA AOR necessitates a change to Technical Specifications Section 4.3.1, "Criticality." The specifics of the TS change are provided in Section 2.3.

2.2 New Fuel Vault Criticality Safety Analysis

The QCNPS NFV racks are GE designed low density racks with an interrack spacing of 11 inches (Reference 6.3, Section 9.1.1.2). The NFV rack CSA coverage for the new GNF3 fuel will be the GESTAR II (Reference 6.4) analysis for GE designed low density NFV racks upon approval of this proposed license amendment. The applicability of GESTAR II to the GNF3 fuel type is documented in the GNF3 GESTAR II validation report (Reference 6.6). The QCNPS NFV interrack pitch is ≥ 10.5 inches (the criteria listed in GESTAR II) and thus the racks may be utilized to store new GNF fuel with in-core SCCG k_{inf} ≤ 1.31 (Reference 6.4, Section 3.5). Past NFV CSA will no longer be applicable to QCNPS upon implementation of this license amendment because the only fuel to be delivered to the site for core reloads will be GNF3.

No TS change is needed for implementation of the GESTAR II NFV CSA methodology. The in-core SCCG limit of k_{inf} ≤ 1.31 is the GESTAR II basis NFV CSA limit for QCNPS storage of fresh GNF3 fuel.

2.3 Proposed Changes to Technical Specifications Section 4.3.1

The QCNPS, Units 1 and 2 TS requirements related to spent fuel storage are contained in TS Section 4.3, "Fuel Storage." TS 4.3.1 identifies requirements pertaining to the design of the SFP storage racks. Specifically, TS 4.3.1.1.a requires k_{eff} to be ≤ 0.95 if fully flooded with unborated water, which includes an allowance for uncertainties as described in Section 9.1.2 of the Updated Final Safety Analysis Report (UFSAR). TS 4.3.1.1.b requires a nominal 6.22-inch center-to-center distance between fuel assemblies placed in the SFP storage racks in both pools. TS 4.3.1.1.d provides the limit on the areal density for the neutron absorber in the rack inserts as ≥ 0.0116 $g^{10}B/cm^2$. None of these items require update due to the proposed change in CSA methodology.

The governing k_{inf} limit structure for acceptable SFP fuel storage in TS 4.3.1.1.c is replaced with a new condition in accordance with the new CSA basis as shown below.

Current TS 4.3.1.1.c	Proposed TS 4.3.1.1.c
The combination of U-235 enrichment and gadolinia loading shall be limited to ensure fuel assemblies have a maximum k-infinity of 0.8991 as determined at 4°C (39.2°F) in the normal spent fuel pool in-rack configuration; and	Fuel assemblies having a maximum k_{inf} of 1.29 in the normal reactor core configuration at cold conditions; and

A mark-up of the proposed TS change is provided in Attachment 2. There are no TS Bases associated with Chapter 4, "Design Features." The QCNPS Updated Final Safety Analysis Report (UFSAR) will be updated in accordance with 10 CFR 50.71(e) as part of implementation of the approved amendment. A summary of the proposed changes is provided below.

- Section 9.1.1.2, "Facilities Description," will be modified to reflect storage requirements of GNF3 fuel in the NFV, including receiving pre-channeled GNF3 fuel.
- Section 9.1.1.3, "Safety Evaluation," will be revised to reflect the NFV requirements of 10 CFR 50.68(b)(2) in this section and update cross-reference to ATRIUM 10XM with GNF3.
- Sections 9.1.2.3, "Safety Evaluation," and 9.1.2.3.1, "Safety Evaluation for Fuel," will be updated to reflect the characteristics of the new SFP CSA covering all fuel types.
- Section 9.1.5, "References," will be updated for consistency with other changes in Section 9.1.

3.0 TECHNICAL EVALUATION

3.1 Overview of System Design and Operation

The QCNPS UFSAR Section 9.1.2, "Spent Fuel Storage," documents the combined units' SFP safety design bases as follows:

A. A maximum k_{eff} of 0.95 is maintained with the racks fully loaded with fuel of the highest anticipated reactivity and flooded with unborated water at a temperature corresponding to the highest reactivity. The criticality analyses include allowance for uncertainty and are described in a criticality analysis report applicable to the spent fuel pools.

B. The spent fuel storage racks, containing their full complement of fuel assemblies (i.e., 7554 fuel assemblies) are designed to withstand earthquake loadings of a Class 1 structure.

C. There will be no release of contamination or exposure of personnel to radiation in excess of 10 CFR 20 limits.

D. It is possible, at any time, to perform limited work on irradiated components.

E. Pool storage space is provided for used control rods, flow channels, and other reactor components.

F. The fuel storage pools of Units 1 and 2 are connected by a double-gated transfer canal.

To achieve the safety design bases QCNPS has two joined SFPs which provide for storage of new unirradiated and irradiated fuel in a safe manner. The SFP facilities are designed to accept new unirradiated and irradiated fuel from both the Unit 1 and Unit 2 reactor cores (i.e., one unit's fuel may reside in either or both SFPs).

The QCNPS SFPs are identical in the types of SFP racks and neutron absorbing materials used. The SFPs contain high density spent fuel storage racks made up of 39 individual modules that have a combined capacity of 7554 fuel assemblies. The modules are a honeycomb arrangement of cells constructed of a series of cruciform shaped stainless steel elements. The high density spent fuel storage racks originally contained a nominal 0.070-inch-thick sheet of Boraflex neutron absorber material physically captured between the side walls of all adjacent boxes (cells). To provide space for the original neutron absorber sheet between each box wall, stainless steel spacer strips were placed between box wall plates.

The organic PDMS (polydimethylsiloxane) based Boraflex sheet material experienced premature degradation at QCNPS and across the industry. This was driven by high temperatures, high gamma radiation flux, and convection driven water flow that was able to enter and leave the areas between cells where the Boraflex resided. In response to the Boraflex degradation at QCNPS, all possible fuel storage cells in both SFP's racks had NETCO-SNAP-IN rack inserts installed (see Reference 6.7). A rack insert was installed in every rack cell location where a fuel bundle could be placed.

The rack inserts are made of a thin sheet of Rio Tinto Alcan's Boralcan metal matrix composite material (formed from molten aluminum with a very fine particle B_4C powder added) formed into a chevron shape that fully covers two of the interior sides of each rack cell in the axial range of the active fuel. All rack inserts were installed in the rack cells with the chevron corner in each cell's south-west corner. In this way all fuel in the rack cells will have one Boralcan neutron absorber insert wing between them. The one exception is in the fuel rack cells along the SFP's north and east most rack's edges. For these cells, the higher neutron radial leakage into the bulk pool water and surrounding structural materials helps offset the impact of having less neutron absorber. With the addition of the rack inserts, no negative reactivity credit is taken for residual Boraflex in the racks. The entire area that was originally occupied by Boraflex is now assumed in the CSA to contain water.

The specific NETCO-SNAP-IN rack inserts used at QCNPS have a minimum certified ^{10}B areal density of 0.0116 g/cm^2.

The spent fuel storage racks are designed to maintain the stored spent fuel in a spatial geometry that precludes the possibility of criticality. The spent fuel storage racks maintain this subcritical geometry when subjected to maximum earthquake conditions, dropped fuel assembly accident conditions, and any uplift forces generated by the fuel handling equipment.

3.2 Criticality Evaluation

In accordance with 10 CFR 50.68, a CSA for the QCNPS, Units 1 and 2 SFPs has been revised to support the purposes discussed in Section 2.1. The analysis, provided as Attachment 7, demonstrates that the maximum k_{eff} (i.e., k_{max}(95/95)) is less than the 10 CFR 50.68 limit of 0.95 for normal and credible abnormal operation with tolerances and computational uncertainties considered. All necessary requirements as outlined in NUREG-0800, Section 9.1.1 Revision 3 dated March 2007, have been met. NEI 12-16, Rev. 4 (Reference 6.2) was used as guidance for this analysis.

The revised CSA covers all legacy fuel in storage in either the QCNPS Unit 1 or Unit 2 SFP and the new GNF3 product line. A description of the GNF3 product line is provided in Section 4.1 of Attachment 7, while the disposition of legacy fuel is provided in Appendix B of Attachment 7.

The calculations are performed using GNF's peak in-core k_{inf} methodology. The peak in-core k_{inf} criterion method relies on a well-characterized relationship between the infinite lattice k_{inf} (in-core) for a given fuel design and a specific fuel storage rack k_{inf} (in-rack) containing that fuel. This methodology was shown to be appropriate for use at Quad Cities by validating that there exists a well-characterized, linear relationship between the infinite lattice k_{inf} (in-core) and fuel storage rack k_{inf} (in-rack). Appropriate application was also ensured by using a design basis lattice with conservative values of rack efficiency and in-core k_{inf} for all criticality analyses.

Appendix B of Attachment 7 shows that this method produces an in-core k_{inf} which correlates to an in-rack k_{inf} for GNF3 fuel that bounds the legacy fuel. This is in line with the requirements in 10 CFR 50.68(b) and NEI 12-16, Revision 4 (Reference 6.2). The CSA uses the minimum certified ^{10}B areal density of 0.0116 g/cm^2 in the Boralcan rack inserts at QCNPS.

The peak reactivity of the fuel in the QCNPS SFP storage racks was calculated using the computer codes TGBLA06 and MCNP-05P. In this evaluation, in-core k_{inf} values and exposure dependent, pin-by-pin isotopic specifications were generated using TGBLA06, the NRC-approved GE-Hitachi Nuclear Energy Americas LLC (GEH)/GNF BWR lattice physics code. The fuel storage criticality calculations were then performed using MCNP-05P, the GEH/GNF proprietary version of the Los Alamos National Laboratory Monte Carlo neutron transport code, MCNP5, using the TGBLA06 nuclide inventory as input. TGBLA06 uses ENDF/B-V cross-section data to perform coarse-mesh, broad-group, diffusion theory calculations. MCNP-05P uses ENDF/B-VII.0 pointwise (i.e., continuous) cross-section data, and all reactions in the cross-section evaluation are considered. MCNP-05P has been validated and verified for spent fuel pool storage rack evaluations in accordance with the NUREG/CR-6698 guidance (included as part of Attachment 7). The method of analysis is discussed in greater detail in Section 3.0 of Attachment 7. Validation of the codes and libraries is described in Section 3.4 and Appendix A of Attachment 7.

The use of TGBLA06 for BWR core depletion calculations has been reviewed and accepted by the NRC as part of the approval of Reference 6.4. The NRC has also approved the MCNP-05P/TGBLA06 code package for use in similar fuel pool criticality analyses. Reference 6.5 documents an example of a previously NRC approved use of this code package.

3.3 Accident Conditions

The spent fuel rack configuration was analyzed for credible accident scenarios. The scenarios considered are presented in the bulleted list that follows and are discussed in Section 5.5.3 of Attachment 7. Note that the missing rack insert is conservatively treated as a normal condition bias in Section 5.5.2.

- SFP temperature exceeding the normal range (moderator temperature/density changes)
- Dropped and dropped + damaged fuel assemblies
- Rack movement (seismic)
- Mislocated fuel assembly (an assembly in the wrong location outside a storage rack)

The criticality analysis for the storage of BWR assemblies in the QCNPS SFP racks with Boralcan rack inserts has been performed. The results for the normal condition show that k_{eff} is ≤ 0.95 with the storage racks fully loaded with fuel of the highest anticipated reactivity, at a temperature corresponding to the highest reactivity. The results for this bounding accident condition, i.e., the “Dropped/Damaged Fuel” (Case T14.B10), also show that k_{eff} is ≤ 0.95 with the storage racks fully loaded with fuel of the highest anticipated reactivity, at a temperature corresponding to the highest reactivity.

Reactivity effects of abnormal and accident conditions have been evaluated and assure that under all credible abnormal and accident conditions, the reactivity will not exceed the regulatory limit of 0.95 with a 95 percent probability at a 95 percent confidence level.

4.0 REGULATORY EVALUATION

4.1 Applicable Regulatory Requirements/Criteria

The regulations in 10 CFR 50.36, "Technical specifications," contain the requirements for the content of TSs. As required by 10 CFR 50.36(c)(4), "Design features," the Technical Specifications (TS) will include design features of the facility such as materials of construction and geometric arrangements, which, if altered or modified, would have a significant effect on safety and are not covered in categories described in paragraphs (c)(1), (2), and (3) of 10 CFR 50.36. QCNPS TS 4.0, "Design features", provides the QCNPS requirements for site location, the reactor core, and fuel storage meeting the intent of 10 CFR 50.36(c)(4). The governing k_{inf} limit structure for acceptable SFP fuel storage in TS 4.3.1.1.c is replaced with a new condition that is consistent with new CSA basis.

10 CFR 50.68, "Criticality accident requirements," paragraph (b)(4) states that the k_{eff} of the spent fuel storage racks loaded with fuel of the maximum fuel assembly reactivity and flooded with unborated water must not exceed 0.95, at a 95 percent probability, 95 percent confidence level. Further, paragraphs (b)(2) and (b)(3) state the equivalent neutron multiplication factor limit for the NFV, including the impact that an “optimum moderation” scenario might have. The requirements stated include that the k_{eff} of the fresh fuel in the fresh fuel storage racks loaded with fuel of the maximum fuel assembly reactivity and flooded with unborated water must not exceed 0.95, at a 95 percent probability, 95 percent confidence level. The regulation also states that for the optimum moderation case the k_{eff} must not exceed 0.98 at a 95 percent probability, 95 percent confidence level. The optimum moderation case is not applicable to the QCNPS NFV as it is a moderation controlled area (see Reference 6.3, Section 9.1.1.3). The QCNPS SFP criticality analysis, provided as Attachment 7 to this submittal along with the GESTAR II NFV criticality analysis in Reference 6.4, demonstrate that these requirements are met.

Paragraph (b)(7) of 10 CFR 50.68 states that the maximum nominal U-235 enrichment of the fresh fuel assemblies is limited to 5.0 percent by weight. QCNPS GNF3 fuel is below 5.0 percent by weight ^{235}U enrichment.

QCNPS, Units 1 and 2 were not licensed to the 10 CFR 50, Appendix A, General Design Criteria (GDC). The QCNPS, Units 1 and 2 UFSAR, Section 3.1, "Conformance with NRC General Design Criteria," provides an assessment against the 70 draft GDC published in 1967 and concluded that the plant specific requirements are sufficiently similar to the Appendix A GDC. Criterion 66, "Prevention of fuel storage criticality," states that criticality in the new and spent fuel storage shall be prevented by physical systems or processes, preferably by use of geometrically safe configurations. The evaluation of QCNPS's conformance with Criterion 66 is discussed in both Section 3.1.8.1, "Criterion 66 – Prevention of Fuel Storage Criticality" and Section 9.1, "Fuel and Storage Handling" of the QCNPS UFSAR. The racks in which new and spent fuel assemblies are placed are designed and arranged to ensure subcriticality in the vault and storage pool. The QCNPS criticality analysis demonstrates that, given the current spent fuel storage system design, k_{eff} will remain less than or equal to 0.95 for the legacy fuel types and the GNF3 fuel.

4.2 <u>Precedent</u>

The NRC has recently approved the use of the GNF CSA methodology to determine the acceptability of storing fresh and spent GNF3 fuel in other BWR spent fuel pools that contain NETCO-SNAP-IN rack inserts. These plants also utilized the NETCO-SNAP-IN rack inserts to provide fuel storage reactivity control in place of the degraded Boraflex material originally placed in the rack structure. One example of this is the acceptance of the CSA at Entergy’s River Bend Station as documented in the Reference 6.8 safety evaluation which was issued on December 31, 2019.

4.3 <u>No Significant Hazards Consideration</u>

In accordance with 10 CFR 50.90, "Application for amendment of license, construction permit, or early site permit," Exelon Generation Company, LLC (EGC) requests an amendment to Renewed Facility Operating License Nos. DPR-29 and DPR-30 for Quad Cities Nuclear Power

Station (QCNPS), Units 1 and 2, respectively. Specifically, EGC is utilizing a new criticality safety analysis (CSA) methodology for evaluating the legacy fuel types and the new GNF3 reload fuel design in the spent fuel pool (SFP). Use of the new SFP CSA methodology necessitates a change to the QCNPS Technical Specifications (TS) 4.3.1, "Criticality." EGC is also proposing a change to the new fuel vault (NFV) CSA to utilize the GESTAR II methodology for validating the NFV criticality safety for GNF3 fuel in the General Electric (GE) designed NFV racks. This methodology change does not require a change to the QCNPS TS.

According to 10 CFR 50.92, "Issuance of amendment," paragraph (c), a proposed amendment to an operating license involves no significant hazards consideration if operation of the facility in accordance with the proposed amendment would not:

(1) Involve a significant increase in the probability or consequences of an accident previously evaluated; or

(2) Create the possibility of a new or different kind of accident from any accident previously evaluated; or

(3) Involve a significant reduction in a margin of safety.

EGC has evaluated the proposed change for QCNPS using the criteria in 10 CFR 50.92, and has determined that the proposed change does not involve a significant hazards consideration. The following information is provided to support a finding of no significant hazards consideration.

1. Does the proposed amendment involve a significant increase in the probability or consequences of an accident previously evaluated?

Response: No

The proposed amendment involves a revised new fuel vault (NFV) criticality safety analysis (CSA) and a revised spent fuel pool (SFP) CSA for the QCNPS Units 1 and 2 spent fuel pools (SFPs) using new methodologies. Technical Specification 4.3.1.c requires revision to maintain consistency with the new methodology results. The proposed new CSA demonstrates adequate margin to criticality and therefore does not affect the consequences of any accident previously evaluated.

The impact of the CSA methodology change on the following four previously evaluated events and accidents was assessed:

- A fuel handling accident (FHA),
- A fuel mispositioning event,
- A seismic event, and
- A loss of SFP cooling event

This proposed amendment, covering only the change in CSA methodologies, does not change or modify the fuel handling processes, new and spent fuel storage racks, number of fuel assemblies that may be stored in the NFV or SFP, the assumed decay heat generation rate, or the SFP cooling and cleanup system. There is therefore no impact on the probability of an accident previously evaluated.

Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the proposed amendment create the possibility of a new or different kind of accident from any accident previously evaluated?

Response: No

Onsite storage of fresh and spent fuel assemblies in the QCNPS, Units 1 and 2 shared NFV and SFPs is a normal activity for which QCNPS has been designed and licensed. The proposed use of new methodologies for performing the QCNPS NFV CSA and SFP CSA does not change or modify the fuel handling processes, new or spent fuel racks, number of fuel assemblies that may be stored in the new fuel vault or spent fuel pool, decay heat generation rate, or the SFP cooling and cleanup system.

The limiting dropped/damaged fuel event does not represent a new or different type of accident. Having a dropped/damaged fuel assembly within the fuel storage racks has always been possible, it was just not previously identified as a bounding type of event. The associated analysis results show that the storage racks remain sub-critical following a worst-case dropped/damaged fuel event. Note that the missing rack insert event was conservatively modeled as part of the evaluation of normal conditions instead of as a separate accident.

Therefore, the proposed change does not create the possibility of a new or different kind of accident from any previously evaluated.

3. Does the proposed amendment involve a significant reduction in a margin of safety?

Response: No

QCNPS TS 4.3, "Fuel Storage," Specification 4.3.1.1.a requires the spent fuel storage racks to maintain the effective neutron multiplication factor, k_{eff}, less than or equal to 0.95 when fully flooded with unborated water, which includes an allowance for uncertainties. Therefore, for spent fuel pool criticality considerations, the required safety margin is 5 percent. The 10 CFR 50.68(b)(2) regulation also requires a k_{eff} of less than or equal to 0.95 in the NFV (the optimum moderation, 10 CFR 50.68(b)(3), case does not apply to Quad Cities due to countermeasures taken to prevent water fog entry into the NFV). Thus, the NFV also has a required safety margin of 5 percent.

The proposed change ensures, as verified by the associated criticality analysis, that k_{eff} continues to be less than or equal to 0.95, thus preserving the required safety margin of 5 percent. The updated GNF methodology analysis, which also adds the GNF3 fuel type, results in less margin to the k_{max}(95/95) ≤ 0.95 regulatory limit (i.e., the new k_{max}(95/95) value is less than 0.009 closer to the limit than the current k_{max}(95/95) value). In addition, using the in-core k_{inf} limit ensures that the SFP criticality analysis remains bounding for the fuel assemblies that are allowed to be stored in the SFP storage racks.

Therefore, the proposed change does not involve a significant reduction in a margin of safety.

Based on the above, EGC concludes that the proposed amendment does not involve a significant hazards consideration under the standards set forth in 10 CFR 50.92(c), and, accordingly, a finding of no significant hazards consideration is justified.

4.4 Conclusions

In conclusion, based on the considerations discussed above, (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

5.0 ENVIRONMENTAL CONSIDERATION

The proposed amendment would change a requirement with respect to installation or use of a facility component located within the restricted area, as defined in 10 CFR 20, or would change an inspection or surveillance requirement. However, the proposed amendment does not involve (i) a significant hazards consideration, (ii) a significant change in the types or significant increase in the amounts of any effluents that may be released offsite, or (iii) a significant increase in individual or cumulative occupational radiation exposure. Accordingly, the proposed amendment meets the eligibility criterion for categorical exclusion set forth in 10 CFR 51.22(c)(9). Therefore, pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the proposed amendment.

6.0 REFERENCES

6.1 NEDC-33932, "Quad Cities Units 1 and 2 Fuel Storage Criticality Safety Analysis," Revision 0, dated May 2021 (Attachments 3 and 7 to RS-21-065 for the non-proprietary and proprietary versions, respectively)

6.2 NEI 12-16, Revision 4, "Guidance for Performing Criticality Analyses of Fuel Storage at Light-Water Reactor Power Plants," dated September 2019. (ADAMS Accession Number ML19269E069)

6.3 QCNPS Updated Final Safety Analysis Report (UFSAR), Revision 15, dated October 2019

6.4 GE Licensing Topical Report NEDE-24011-P-A, "General Electric Standard Application for Reactor Fuel (GESTAR II, Main)," Revision 31, dated November 2020 (ADAMS Accession Number ML20330A199 for the non-proprietary version)

6.5 Final Safety Evaluation for GE Hitachi Nuclear Energy Licensing Topical Report NEDC-33374P, Revision 3, "Safety Analysis Report for Fuel Storage Racks Criticality Analysis for ESBWR Plants," dated September 21, 2010 (ADAMS Accession Number ML102430580 for the non-proprietary version)

6.6 NEDC-33879P, Revision 4, “GNF3 Generic Compliance with NEDE-24011-P-A (GESTAR II)," dated August 2020 (Enclosure to ADAMS Accession Number ML20244A104)

6.7 "Quad Cities Nuclear Power Station, Units 1 and 2 - Issuance of Amendments Regarding NETCO Inserts (TAC. NOS. MF2489 and MF2490) (RS-13-148)," dated December 31, 2014 (ADAMS Accession Number ML14346A306)

6.8 Final Safety Evaluation for River Bend Station, “River Bend Station, Unit 1 - Issuance of Amendment No. 201 RE: Change to the Neutron Absorbing Material Credited in Spent Fuel Pool for Criticality Control (EPID L-2018-LLA-0298)," dated December 31, 2019 (ADAMS Accession Number ML19357A009)

ATTACHMENT 2

QUAD CITIES NUCLEAR POWER STATION
UNITS 1 AND 2

Docket Nos. 50-254 and 50-265

Facility Operating License Nos. DPR-29 and DPR-30

MARK-UP OF QCNPS, UNITS 1 AND 2 TECHNICAL SPECIFICATIONS PAGES

4.0 DESIGN FEATURES (continued)

4.3 Fuel Storage

4.3.1 Criticality

4.3.1.1 The spent fuel storage racks are designed and shall be maintained with:

a. $k_{eff} \leq 0.95$ if fully flooded with unborated water, which includes an allowance for uncertainties as described in Section 9.1.2 of the UFSAR;

b. A nominal 6.22 inch center to center distance between fuel assemblies placed in the storage racks;

c. Fuel assemblies having a maximum k_{inf} of 1.29 in the normal reactor core configuration at cold conditions ~~The combination of U-235 enrichment and gadolinia loading shall be limited to ensure fuel assemblies have a maximum k-infinity of 0.8991 as determined at 4°C (39.2°F) in the normal spent fuel pool in-rack configuration~~; and

d. The installed neutron absorbing rack inserts having a Boron-10 areal density ≥ 0.0116 g/cm^2.

4.3.2 Drainage

The spent fuel storage pool is designed and shall be maintained to prevent inadvertent draining of the pool below elevation 666 ft 8.5 inches.

4.3.3 Capacity

The spent fuel storage pool is designed and shall be maintained with a storage capacity limited to no more than 3657 fuel assemblies for Unit 1 and 3897 fuel assemblies for Unit 2.

ATTACHMENT 3

NEDC-33932, "Quad Cities Units 1 and 2 Fuel Storage Criticality Safety Analysis," Revision 1, dated October 2021

(Non-Proprietary Version)

Global Nuclear Fuel

NEDO-33932
Revision 1
October 2021

Non-Proprietary Information

Quad Cities Units 1 and 2
Fuel Storage Criticality Safety Analysis

INFORMATION NOTICE

This is a non-proprietary version of the document NEDC-33932P, Revision 1, which has the proprietary information removed. Portions of the document that have been removed are indicated by an open and closed bracket as shown here [[ ]].

IMPORTANT NOTICE REGARDING CONTENTS OF THIS REPORT

Please Read Carefully

The design, engineering, and other information contained in this document is furnished for the purpose of providing the results of the spent fuel pool criticality analysis for Quad Cities Nuclear Power Station, Units 1 and 2 (Quad Cities). The only undertakings of GNF with respect to information in this document are contained in the contracts between Exelon Generation Company (Exelon) and GNF, and nothing contained in this document shall be construed as changing the contract. The use of this information by anyone other than Exelon, or for any purpose other than that for which it is intended is not authorized; and with respect to any unauthorized use, GNF makes no representation or warranty, express or implied, and assumes no liability as to the completeness, accuracy, or usefulness of the information contained in this document, or that its use may not infringe privately owned rights.

Revision Status

Revision Number	Date	Description of Change
0	May 2021	Initial release
1	October 2021	Revised marked proprietary content

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ACRONYMS

Term	Definition
2D	Two-Dimensional
AOA	Area of Applicability
BAF	Bottom of Active Fuel
BASE	Base Lattice
BOL	Beginning-of-Life
BWR	Boiling Water Reactor
CFR	Code of Federal Regulations
CW	Curtiss-Wright Flow Control Service, LLC
EALF	Energy of the Average Lethargy Causing Fission
[[	]]
GEH	GE-Hitachi Nuclear Energy Americas LLC
GNF	Global Nuclear Fuel - Americas, LLC
HTC	Haut Taux de Combustion
H/X	Hydrogen-to-Fissile Ratio
MID	Mid Lattice
MOX	Mixed Uranium-Plutonium Oxide
NCA	Nuclear Critical Assembly
NEI	Nuclear Energy Institute
NRC	Nuclear Regulatory Commission
SCCG	Standard Cold Core Geometry
SS	Stainless Steel
VAN	Vanished Lattice
WREC	Westinghouse Reactor Evaluation Center
UO_2	Uranium Dioxide

1.0 INTRODUCTION

This report describes the criticality analysis and results for the Quad Cities Boraflex spent fuel racks with credit for NETCO-SNAP-IN® neutron absorbing inserts. No credit for the Boraflex neutron absorber is taken in this analysis. The methodology and analytical models utilized in this criticality analysis confirm that the storage rack systems have been accurately and conservatively represented. This analysis covers the future GNF3 fuel product designs and all legacy fuel stored in Quad Cities spent fuel pools.

The racks are analyzed using the MCNP-05P Monte Carlo neutron transport program and ENDF/B-VII.0 cross-section library. The methodology used in this analysis is the peak Standard Cold Core Geometry (SCCG) in-core eigenvalue (k_∞) criterion methodology. A maximum SCCG, uncontrolled peak in-core k_∞ of 1.29 as defined by the lattice physics code TGBLA06 (Reference 1) is set as the limit for this analysis. As demonstrated in Table 1, the analysis resulted in a storage rack maximum k-effective (k_{max}(95/95)) less than 0.95 for normal and credible abnormal operation with tolerances and uncertainties taken into account.

Table 1 – Summary k_{max}(95/95) Result

Region	k_{max}(95/95)
Spent Fuel Pool Racks	0.94200

2.0 REQUIREMENTS

Title 10 of the Code of Federal Regulations (CFR) Part 50 defines the requirements for the prevention of criticality in fuel storage and handling at nuclear power plants. 10 CFR 50.68 details specifically that the storage rack k_{max}(95/95) for spent fuel storage racks must be demonstrated to be ≤ 0.95 for normal and credible abnormal operation with tolerances and computational uncertainties taken into account. The Standard Review Plan (Reference 2) outlines the standards that must be met for these analyses. All necessary requirements are met in this analysis. Nuclear Energy Institute (NEI) 12-16 (Reference 3) is used as the guidance document for this analysis.

3.0 METHOD OF ANALYSIS

In this evaluation, in-core k_∞ values and exposure dependent, pin-by-pin isotopic specifications are generated using the GE-Hitachi Nuclear Energy Americas LLC (GEH)/GNF lattice physics production code TGBLA06. TGBLA06 solves Two-Dimensional (2D) diffusion equations with diffusion parameters corrected by transport theory to provide system multiplication factors and perform burnup calculations.

The fuel storage criticality calculations are then performed using MCNP-05P, the GEH/GNF proprietary version of MCNP5 (Reference 4). 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, electron, or coupled transport involving all these particles, and computes the eigenvalue for neutron-multiplying systems. For the present application, only neutron transport is considered.

3.1 Cross-Sections

TGBLA06 uses ENDF/B-V cross-section data to perform coarse-mesh, broad-group, diffusion theory calculations. It includes thermal neutron scattering with hydrogen using an $S(\alpha,\beta)$ light water thermal scattering kernel.

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. For the present work, thermal neutron scattering with hydrogen was described using an $S(\alpha,\beta)$ light water thermal scattering kernel. The 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 libraries include very little data thinning and utilize resonance integral reconstruction error tolerances of 0.001%.

3.2 Geometry Treatment

TGBLA06 is a 2D lattice design computer program for Boiling Water Reactor (BWR) fuel bundle analysis. It assumes that a lattice is uniform and infinite along the axial direction and that the lattice geometry and material are reflecting with respect to the lattice boundary along the transverse directions.

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 pre-defined 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.

3.3 Convergence Checks

The use of TGBLA06 as a depletion code in this criticality analysis is consistent with its use for BWR fuel design and its associated user's manual. Convergence checks are encoded in the standard error routines and the absence of error messages was confirmed in all code output.

In this analysis, the following criticality code parameters were specified. At a minimum, all MCNP-05P cases were run with 20,000 neutrons per generation, 200 cycles skipped, and 500 total cycles run. Some cases were run for more cycles skipped and more total cycles in order to meet all the converge checks. For this analysis, the following MCNP-05P convergence checks were reviewed and confirmed passed for each case:

- Sampling of all cells that contain fissionable material
- Matching of first and second half eigenvalue
- Fission source entropy check

3.4 Validation and Computational Basis

MCNP-05P has been compared to [[ ]] critical experiments for validation purposes using ENDF/B-VII.0 nuclear cross-section data. The experiments cover a number of moderator-to-fuel ratios and poison materials that represent material and geometric properties similar to that of BWR fuel lattices both in and out of fuel racks. The critical experiments to which MCNP-05P has been compared are provided in Table 2. All are either low-enriched Uranium Dioxide (UO_2) or Mixed Uranium-Plutonium Oxide (MOX) pin lattice in water experiments. The Area of Applicability (AOA) considered covered by this validation is listed in Table 3, along with the parameters which characterize the spent fuel rack system for comparison. The critical experiment modeling results, along with the calculation of the associated bias and bias uncertainty terms at the 95/95 confidence level using NUREG/CR-6698 (Reference 5) guidance are provided in Appendix A. The study concluded that the appropriate bias to apply to systems covered by this AOA is [[ ]], and the appropriate uncertainty of that bias is [[ ]].

Table 2 – Summary of the Critical Benchmark Experiments

Experiment		Experiments	Year	Where
[[				
				]]

Table 3 – Area of Applicability Covered by Code Validation

Parameters	**Validation Area of Applicability**	**Spent Fuel Rack Characteristics**
Fissionable Material	Uranium, Plutonium	Uranium, Actinides
Chemical Form	UO_2, MOX	UO_2, MOX
Enrichment (wt.% ^{235}U)	wt.% $^{235}U \leq 4.9$	wt.% $^{235}U \leq 4.9$
Enrichment (wt.% ^{239}Pu)	wt.% $^{239}Pu \leq 5.3$	wt.% $^{239}Pu \leq 4.9$
Physical Form	Solid Compound	Solid Compound
Temperature	~20°C up to ~100°C	4-126°C
Moderator (in fuel region)	H_2O	H_2O
Physical Form	Solution	Solution
Temperature	~20°C up to ~100°C	4-126°C
Reflector (in fuel region)	H_2O	H_2O
Physical Form	Solution	Solution
Temperature	20°C	4-126°C
Absorbers	None/Boron/Gadolinium Stainless Steel (SS)/Copper	Boron/Gadolinium/ Fission Products
Neutron Energy Spectrum	Thermal	Thermal
Energy of Average Lethargy Causing Fission (MeV)	6.8E-8 – 8.6E-7	3.666E-07 (Limiting In-rack k_∞ Case)

Table 3 demonstrates that the AOA of this validation encompasses the majority of storage characteristics of new fuel in the spent fuel storage racks. [[

]]

For the storage of spent fuel, however, it is appropriate to add additional uncertainty terms to the k_{max}(95/95) result. Specifically, these items are:

- Uncertainty in fuel depletion calculations

  Consistent with NEI 12-16, a conservative approximation of the fuel depletion uncertainty was quantified by assessing the reactivity difference between a Beginning-of-Life (BOL) system and the exposure dependent, peak reactivity system of interest. Specifically, the cold, in-core, BOL reactivity of the spent fuel rack design basis bundle with no gadolinium present was compared to the reactivity of the exposed design basis bundle at its cold, in-core, peak reactivity statepoint. Both reactivities are calculated for comparison in the rack system. Five percent of the difference in reactivities between these two cases is included as an uncertainty to the spent fuel rack studies in Table 15 to cover the depletion isotopic benchmarking gap including gap for minor actinides and fission products.

- TGBLA06 eigenvalue uncertainty

  An additional uncertainty is also added to the fuel rack studies related to eigenvalue calculations performed using TGBLA06. A bias of [[ ]] and a 95/95 bias uncertainty of [[

  ]] This uncertainty is applied to the spent fuel rack's k_{max}(95/95) value to cover uncertainty in the assignment of in-core k_∞ values to fuel lattices.

3.5 In-Core k_∞ Methodology

The design of the fuel storage racks provides for a subcritical multiplication factor for both normal and credible abnormal storage conditions. In all cases, the storage rack eigenvalue must be ≤ 0.95. To demonstrate compliance with this limit, the peak in-core k_∞ method is utilized.

The peak in-core k_∞ criterion method relies on a well-characterized relationship between infinite lattice k_∞ (in-core) for a given fuel design and a specific fuel storage rack k_∞ (in-rack) containing that fuel. The use of an infinite lattice k_∞ criterion for demonstrating compliance to fuel storage criticality criteria has been used for all General Electric-supplied storage racks and is currently used for re-rack designs at a number of plants. This report demonstrates that the methodology is also appropriate for use at Quad Cities by presenting the following:

- A well-characterized, linear relationship between infinite lattice k_∞ (in-core) and fuel storage rack k_∞ (in-rack)
- The use of a design basis lattice with a conservative rack efficiency and in-core k_∞ for all criticality analyses

The analysis performed to calculate the lattice k_∞ to confirm compliance with the above criterion uses the Nuclear Regulatory Commission (NRC)-approved lattice physics methods encoded into the TGBLA06 engineering computer program. One of the outputs of the TGBLA06 solution is the lattice k_∞ of a specific nuclear design for a given set of input state parameters (e.g., void fraction, control state, fuel temperature).

Compliance of fuel with specified k_∞ limits will be confirmed for each new lattice as part of the bundle design process. Documentation that this has been met will be contained in the fuel design information report, which defines the maximum lattice k_∞ for each assembly nuclear design. The process for validating that specific assembly designs are acceptable for storage in the Quad Cities fuel storage racks is provided below.

1. Identify the unique lattices in each assembly design.
2. Deplete the lattices in TGBLA06 using the following conditions:
   a. Centered Assembly according to Quad Cities specific lattice spacing and zero leakage

   [[

]]

3. Ensure that the k_∞ values obtained from Step 3 for each lattice are less than or equal to the k_∞ limit of 1.29.

Documentation that all legacy fuel types at Quad Cities currently comply with this in-core limit is documented in Appendix B.

3.6 Definitions

Fuel Assembly – is a complete fuel unit consisting of a basic fuel rod structure that may include large central water rods. Several shorter rods may be included in the assembly. These are called "part-length rods". A fuel assembly includes the fuel channel.

Gadolinia – The compound Gd_2O_3. The gadolinium content in integral burnable absorber fuel rods is usually expressed in weight percentage gadolinia.

Lattice – An axial zone of a fuel assembly within which the nuclear characteristics of the individual rods are unchanged.

Base Lattice (BASE) – An axial zone of a fuel assembly typically located in the bottom half of the bundle within which all possible fuel rod locations for a given fuel design are occupied.

Mid Lattice (MID) – [[

]]

Vanished Lattice (VAN) – An axial zone of a fuel assembly typically in the upper half of the bundle within which a number of possible fuel rod locations are unoccupied.

Rack Efficiency – The ratio of a particular lattice statepoint in-rack eigenvalue (k_∞) to its associated lattice nominal in-core eigenvalue (k_∞). This value allows for a straightforward comparison of a rack's criticality response to varying lattice designs within a particular fuel product line. A lower rack efficiency implies increased reactivity suppression capability relative to an alternate design with a higher rack efficiency.

Design Basis Lattice – The lattice geometry, exposure history, and corresponding fuel isotopics for a fuel product line that result in the highest rack efficiency in a sensitivity study of reasonable fuel parameters at the desired in-core reactivity. This lattice is used for all normal, abnormal, and tolerance evaluations in the fuel rack analysis.

3.7 Assumptions and Conservatisms

The fuel storage rack criticality calculations are performed with the following assumptions to ensure the true system reactivity is always less than the calculated reactivity:

1. [[

2.

]]

3. Design basis lattices with in-core k_∞ values greater than the proposed 1.29 in-core k_∞ limit are used for all criticality analyses.

4. [[

]]

Sensitivity studies of the storage system reactivity to these depletion parameters are presented in Section 5.5. [[

]]

5. For conservatism, only positive reactivity differences from nominal conditions determined from depletion sensitivity and abnormal configuration, analyses are added as biases to the final storage rack k_{max}(95/95).

6. Neutron absorption in spacer grids, concrete, activated corrosion and wear products (CRUD) and axial blankets is ignored to limit parasitic losses in non-fuel materials.

7. TGBLA06 defined "lumped fission products" and Xe-135 are both conservatively ignored for MCNP-05P in-rack k_∞ calculations.

8. [[

]]

9. Only ^{10}B is modeled in the rack inserts. Each insert is assumed to contain the minimum areal density of 0.0116 g $^{10}B/cm^2$. All other insert material is ignored. Ignoring the other materials conservatively limits neutron absorption in the insert.

10. No credit is taken for the Boraflex in the storage racks in the analysis, and all material between the inner cell walls is modeled as water. Modeling this material as water is reasonable, as there is not a water tight seal between the Boraflex and pool environment, and therefore any significant gap formations within the poison material will be filled with water.

11. Each Boraflex rack cell will contain one chevron-shaped insert, and the inserts will be oriented uniformly in all the Boraflex racks. This analysis assumes the two neutron absorber panels in each chevron-shaped insert in the rack cells will be oriented such that one panel will be on the south side and the other panel will be on the west side. In this orientation there will be no insert absorber material between the outer edges of the Boraflex racks and the north and east pool walls.

4.0 FUEL DESIGN BASIS

This rack criticality analysis covers all legacy and current fuel in Quad Cities, and the planned GNF3 future fuel product line. The disposition for all legacy and current fuel is in Appendix B. The description of the GNF3 fuel product lines is found in Section 4.1. This product line is used to determine the design basis bundle in Section 5.3.

All fuel is UO_2 with some fuel rods containing gadolinia, Gd_2O_3.

This criticality analysis covers reconstituted fuel where a rod containing fuel is replaced with another fueled or non-fueled rod. Fuel where there are missing fuel rod locations that are not part of the normal fuel product line designs are explicitly assessed in Appendix B.

This criticality analysis also covers the storage of non-fuel items such as channels in spent fuel rack locations because this analysis covers peak reactivity fuel in every rack cell location.

4.1 GNF3 Fuel Description

The GNF3 fuel lattice configuration is a 10x10 fuel rod array [[

]] as shown in Figure 1, with corresponding dimensions in Table 4 and Table 5. Figure 1 also demonstrates the part-length rod locations. Fuel channel dimensions are provided in Figure 2 and Table 6. Pellet stack density is in Table 7. [[

]]

[[

]]

Figure 1 - GNF3 Lattice Configuration

Table 4 - Nominal Dimensions for GNF3 Fuel Lattice

Item			Dimension	
			mm	in
Channel	[[			
Fuel Rod				
Pellet				]]
[[				]]
Bundle Lattice	[[			]]

Table 5 – Cell Dimensions

Lattice Type	Channel Name	½ Wide Gap, Q		½ Narrow Gap, R		Control Blade Pitch, S	
		mm	in	mm	in	mm	in
[[							]]

[[

]]

Figure 2 – GNF3 Channel Dimensions

Table 6 - Nominal Channel Dimensions for GNF3 Lattice

	Channel Name	**83AV**				**93AV**			
	Channel Section	**Zone 1**		**Zone 2**		**Zone 1**		**Zone 2**	
	Dimension	**mm**	**in**	**mm**	**in**	**mm**	**in**	**mm**	**in**
[[									
									]]

Table 7 – GNF3 Fuel Stack Density as a Function of Gadolinia Concentration

Gadolinia Concentration (wt. fraction)	[[							
Pellet Density (g/cc)								]]

4.2 Fuel Model Description

The fuel models considered include 2D geometric modeling of all fuel material, cladding, water rods, and channels. In the depletion model, appropriate depletion time steps are used consistent with depletion timesteps used in BWR core design analyses. [[

]] Pin specific isotopic modeling as a function of exposure is performed based on the lattice physics code TGBLA06. To obtain the isotopic composition of the fuel pins, each lattice design considered is "burned" at reactor operating conditions [[ ]] and depleted through to a final exposure of [[

]] The isotopics utilized exclude Xe-135 and TGBLA06 defined "lumped fission products" [[ ]] An example of a GNF3 MID lattice model in MCNP-05P is depicted in Figure 3.

[[

]]

Figure 3 – GNF3 MID Lattice in MCNP-05P

The fuel loadings considered for each lattice span a range of exposures, average enrichments, number of gadolinia rods, gadolinia concentration, and void histories considered to be reasonably representative of any Quad Cities fuel designs. The lattice type and exposure history that results in the worst-case rack efficiency for an in-core k_∞ greater than the proposed limit is then used to define the design basis lattice. This lattice is assumed to be stored in every location in the rack being analyzed. Details on the determination of the design basis lattice using the process outlined above are presented in Section 5.3.

5.0 CRITICALITY ANALYSIS OF SPENT FUEL STORAGE RACKS

5.1 Description of Spent Fuel Storage Racks

The Quad Cities Boraflex storage racks manufactured by Joseph Oat Corporation consist of multiple modular cruciform, T-shaped, and L-shaped 304 SS structures that form rack cells with a center-to-center cell pitch of 6.22 inches. These structures contain 0.070-inch thick Boraflex panels sandwiched between the 0.075-inch SS outer walls of the modular shapes. A schematic of a Boraflex storage rack array without inserts installed is shown in Figure 4.

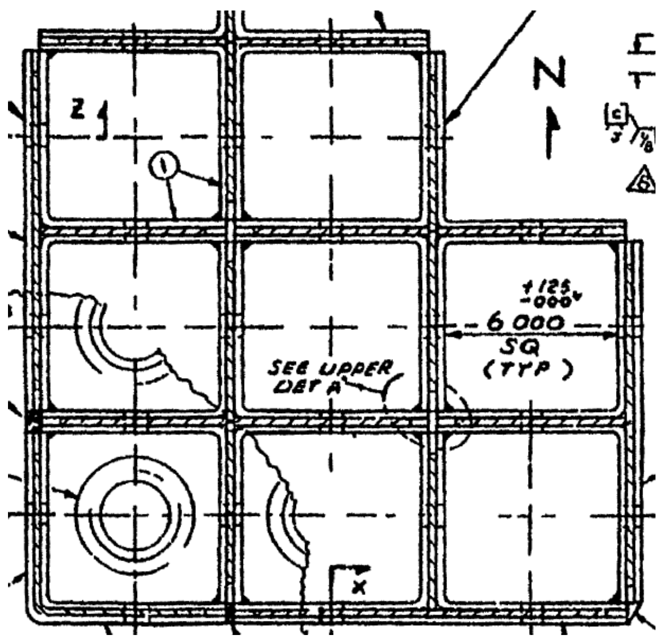

Figure 4 –Spent Rack Array Without Inserts

Originally, the racks employed thermal neutron absorption in the ^{10}B of the Boraflex as the primary mechanism of reactivity control; however, the Boraflex has been demonstrated to be degrading over time. Therefore, no credit is taken for the Boraflex in this analysis, and all material between the inner cell wall and outer wrapper is modeled as water. Modeling this material as water is reasonable, as the outer wrapper does not provide a water tight seal between the Boraflex and pool environment. Therefore, any significant gap formations within the poison material will be filled with water.

To supplement the reactivity suppression capability of the rack, chevron shaped neutron absorbing inserts (NETCO-SNAP-IN®) are installed in each of the storage cells in a storage rack module. These inserts extend over the full-length of the active fuel region of the storage assemblies. The inserts are manufactured from a Rio Tinto Alcan aluminum boron carbide metal matrix composite with a minimum certified areal density of 0.0116 g $^{10}B/cm^2$. The nominal designed wing length of the inserts is [[ ]] inches, and the nominal thickness is 0.085 inches. Each insert is installed with the same orientation. In this way, one leg of an insert exists between each bundle in the storage rack assembly. Figures 5 and 6 demonstrate where the inserts are located in each cell.

Based on the insert configuration, peripheral storage cells on the north and east sides of the storage pools will not be surrounded by four wings of the absorbing insert. The reactivity effect of this storage limitation is assessed in Section 5.5.

5.2 Spent Fuel Storage Rack Models

This analysis covers a single bounding storage configuration of maximum reactivity fuel in every storage location with a NETCO-SNAP-IN® insert in every storage location.

A 2D infinite storage array with periodic boundary conditions is modeled to conservatively represent the nominal spent fuel pool configuration. An image of a single element of the model is provided in Figure 5 and a zoomed in view of Figure 6, with dimensions and tolerances presented in Table 8. This single element is used to define a 10x10 rack array with periodic boundary conditions. This array is used in the design basis bundle selection process in Section 5.3.

[[

]]

[[

]]

Figure 5 – Storage Rack Model Schematic

[[

]]

Figure 6 – Zoomed Storage Rack Model Schematic

Table 8 – Storage Rack Dimensions

		Tolerances	
Rack Model Parameter	**Nominal**	**Plus**	**Minus**
	(inches)	***(inches)***	***(inches)***
Rack Cell Inner Dimension	6.00	0.125	0.000
Rack Pitch	6.22	0.125	0.000
Inner Cell Wall Thickness	0.075	0.004	0.004
Boraflex Thickness	0.070	0.007	0.007
Boraflex Width	5.80	-	-
Rack Insert Width	[[		]]
Rack Insert Thickness	0.085	0.005	0.005

5.3 Design Basis Lattice Selection

Table 9 defines the lattice designs and exposure histories that were explicitly studied in the spent fuel storage rack to determine the geometric configuration and isotopic composition that results in the worst rack efficiency. Note that void state is not a relevant parameter for zero exposure peak reactivity cases, and, therefore, only a single result is presented for these fuel loadings. Figure 7 presents a graph that demonstrates the linear nature of the in-core to in-rack results over all rack efficiency cases studied in the rack system. This figure also provides infinite in-core and in-rack eigenvalue pairs [[

]] to allow for the linear relationship to be demonstrated over a large range of exposures. The highest rack efficiency with an in-core k_∞ greater than the proposed limit of 1.29 is found to result from the parameters defined in Table 9 Case 12. The geometry and isotopics defined for this case are used to define all bundles in the remaining spent fuel rack analyses.

Table 9 – Fuel Parameter Ranges Studied in Spent Fuel Rack

Case	Lattice Type	Void	Average Lattice Enrichment (^{235}U wt.%)	Number of Gadolinia Rods	Gadolinia Concentration (Gd wt. %)	Peak-Reactivity Exposure (GWD/ST)	TGBLA06 Defined In-Core k∞	MCNP-05P Defined In-Rack k∞	Rack Efficiency
1	[[							0.87926	[[
2								0.86289	
3								0.90397	
4								0.89529	
5								0.88354	
6								0.90023	
7								0.89747	
8								0.89184	
9								0.88654	
10								0.88511	
11								0.88231	
12								0.91396	
13								0.90812	
14								0.89816	
15								0.90639	
16								0.90248	
17								0.89492	
18								0.87694	
19							]]	0.86723	]]

[[

]]

Figure 7 – Spent Fuel In-Core versus In-Rack Eigenvalues

5.4 Normal Configuration Analysis

5.4.1 Analytical Models

The most reactive normal configuration was determined by studying the reactivity effect of the following credible normal scenarios:

- Storage of non-channeled assemblies
- Eccentric loadings
  - When neutron absorber panels with an areal density above 0.01 g $^{10}B/cm^2$ are present on all four sides of the fuel assembly, a centrally located positioning of the fuel assembly in the storage cell is the most reactive configuration. Therefore, no eccentric loading cases were performed for this analysis consistent with NEI 12-16 (Reference 3).
- [[

  ]]
- Pool moderator temperature variation

As the non-channeled assembly evaluation demonstrated a decrease in reactivity when compared to nominal, channeled storage conditions, the studies are performed with channeled bundles.

5.4.2 Results

The results of the study are provided in Table 10. This information demonstrates that none of the normal configurations analyzed increase the system reactivity by a statistically significant amount over the nominal loading pattern. The in-rack k_∞ associated with this nominal combination of conditions is 0.91396, and is hereafter referred to as k_{Normal}. This configuration will be used for all abnormal and tolerance studies that are performed on an infinite basis. Any small, positive reactivity differences from this nominal condition are included in the calculation of the system bias in Section 5.5.4.

Table 10 – Spent Fuel Storage Rack In-Rack k_∞ Results – Normal Configurations

Term	Configuration	In-Rack k_∞	MCNP-05P Uncertainty (1σ)
Base	Nominal - Centered, channeled, [[ ]]	0.91396	[[
Δk_{N1}	Non-channeled assemblies	0.91075	
Δk_{N2}	[[	0.91471*	
Δk_{N2}	]]	0.91448	
Δk_{N3}	Moderator Temperature decrease to 4ºC (ρ=1 g/cc)	0.91511*	
Δk_{N3}	Moderator Temperature increase to 126ºC with 20% void (ρ=0.7508 g/cc)	0.87609	]]

* Largest positive reactivity increase from nominal case for each term is included in roll-up of Δk_{Bias}

5.5 Bias Cases

5.5.1 Depletion Bias Cases

The following configurations related to the depletion conditions of the stored bundles were explicitly considered, where each description defines a condition all bundles in storage experience over their entire exposure histories. These bound the conditions the bundles actually experience.

- [[
-
-
-
-
-
-
  ]]
- Depleted with clad creep

The following potential reactivity effect of changes that occur during depletion are considered:

a. Fuel rod changes (clad creep, fuel densification/swelling)

   Clad Creep - [[
   ]]

   Fuel Pellet Densification – [[
   ]]

b. Material dependent grid growth

   [[
   ]]

5.5.2 Normal Bias Cases

The following bias cases are included for normal conditions. As seen in Table 10, cases with a moderator temperature decrease [[ ]] resulted in the largest positive reactivity increases from the nominal case for their respective terms and are therefore included in Table 14.

- No inserts on rack periphery

  As discussed in Section 5.1 and illustrated in Figure 5, there will be assemblies loaded in storage cells on two sides that will not be surrounded by neutron absorbing inserts. [[

  ]] Results are provided in Table 11. The reactivity increase from this study is included in the final Δk_{Bias} term in Table 14.

Table 11 – Rack Periphery Study Results

Description	k_{eff}	MCNP-05P Uncertainty (1σ)	Δk
[[			]]
No Inserts on Rack Periphery	[[		]]

- Missing rack insert

  A missing insert from the 10x10 infinite array was analyzed to cover the periodic removal of an insert for inspection or an insert being accidently removed during fuel movement. The relative reactivity increase from this condition is included in the bias table in Table 14.

5.5.3 Abnormal/Accident Bias Cases

Additionally, perturbations of the normal spent fuel rack configuration were considered for credible accident scenarios. The scenarios considered are presented in the bulleted lists that follow, with explanations of the abnormal condition provided below each listing of similar configurations. The most limiting of these abnormal conditions is included in the final Δk_{Bias} term in Table 14.

- Dropped/damaged fuel

  Justification - The dropped/damaged fuel scenario [[

  ]] The relative reactivity change from this abnormal condition is included in Table 14.

- Abnormal positioning of a fuel assembly outside the fuel storage rack

  Justification – There is enough space for an abnormally positioned bundle between:

  - the south, east, and west sides of the Boraflex racks and the pool wall,
  - the north side one of the Boraflex racks and the dry cask storage pad, and
  - between the north side of one of the Boraflex racks and an adjacent Boraflex rack in the spent fuel pool.

  [[

  ]]

  A misplaced bundle outside the rack is analyzed on an edge of the rack [[

  ]] The calculation was then reperformed several times with a misplaced bundle oriented flush with a rack [[

  ]] the most limiting result generated is used to determine the Δk from the base case eigenvalue as shown in Table 12. The most limiting orientation identified in the study [[

  ]] depicted in Figure 8.

[[

]]

Figure 8 – Finite Misplaced Bundle Model Example

Table 12 – Results for a Misplaced Bundle

Description	**k_{eff}**	**MCNP-05P Uncertainty (1σ)**	**Δk**
[[			]]
Misplaced Bundle, in the most limiting location (Figure 8)	[[		]]

The following abnormal configurations are also considered bounded, with the justification provided:

- Dropped bundle on rack

  Justification – For a drop on the rack, the fuel assembly will come to rest horizontally on top of the rack with a minimum separation distance from the fuel in the rack of more than 12 inches. At this separation distance, the fissile material will be separated by enough neutron mean free paths to preclude neutron interactions that increase k_{eff}, and the overall effect on reactivity will be insignificant.

- Rack sliding due to seismic event which causes water gap between racks to close

  Justification – The racks modeled in this analysis are infinite in extent with no inter-module water gaps. This essentially assumes all racks are close-fitting and bounds possible reactivity effects of rack sliding.

- Loss of spent fuel pool cooling

  Justification – Normal sensitivity analysis results demonstrate that system reactivity decreases as moderator density decreases and pool temperature increases; therefore, reactivity effects of loss of spent fuel pool cooling are bounded by the nominal reactivity results.

Table 13 – Spent Fuel Storage Rack Abnormal Bias Summary

Description	k_{eff}	MCNP-05P Uncertainty (1σ)	Δk	Δk Uncertainty (2σ)
Dropped/Damaged Fuel	0.91498	[[ ]]	0.00102	[[ ]]
Misplaced Bundle*	[[			]]

* Per the double contingency principle (Reference 3), only the most limiting misplaced bundle case is included in the bias roll-up in Table 14.

5.5.4 Results

The results of the abnormal studies are provided in Table 14. The Δk term in this table represents the difference between the system reactivity with the specified bias case and k_{Normal} for terms Δk_{B1} through Δk_{B5}. [[
]] Δk_{B6} and Δk_{B7} are the normal condition cases that resulted in positive reactivity contributions. Δk_{B8} is extracted from Table 11. Δk_{B9} is the missing insert case and Δk_{B10} and Δk_{B11} are extracted from Table 13. The total contribution from these independent conditions to the k_{max}(95/95) of the spent fuel rack is calculated using Equation 1. In this equation, a Δk_{Bi} value must be both positive and the largest for its respective term to be considered.

$$\Delta k_{Bias} = \sum_{i=1}^{n} \Delta k_{Bi} \qquad (1)$$

Table 14 – Spent Fuel Storage Rack Bias Summary

Term	**Description**	**k_{eff}**	**MCNP-05P Uncertainty (1σ)**	**Δk***	**Δk Uncertainty (2σ)**
Δk_{B1}	[[	0.91264	[[	-0.00132	[[
Δk_{B2}		0.91549		0.00153	
Δk_{B2}		0.91517		0.00121	
Δk_{B3}		0.91485		0.00089	
Δk_{B3}		0.91480		0.00084	
Δk_{B4}		0.91494		0.00098	
Δk_{B4}	]]	0.91466		0.00070	
Δk_{B5}	Depleted with clad creep	0.91526		0.00130	
Δk_{B6}	[[ ]]	0.91471		0.00075	
Δk_{B7}	Moderator Temperature decrease to 4°C (ρ=1 g/cc)	0.91511	]]	0.00115	]]
Δk_{B8}	No inserts on rack periphery	[[			]]
Δk_{B9}	Missing insert	0.91853	[[	0.00457	[[
Δk_{B10}	Dropped/Damaged Fuel	0.91498	]]	0.00102	]]
Δk_{B11}	Misplaced Bundle	[[			]]
Δk_{Bias}				[[	]]

* For conservatism, only positive values that are the largest for their respective term are considered.

[[ ]]

5.6 Uncertainty Analysis

5.6.1 Analytic Models

The following tolerance study configurations were explicitly considered for the spent fuel rack:

- Fuel enrichment increases by [[ ]] ^{235}U
- Fuel pellet density increased by [[ ]] of nominal value
- Gadolinia concentration decreased by [[ ]]
- Rod cladding thickness increased by [[ ]] and rod cladding outer diameter increased by [[ ]]
- Rod cladding thickness decreased by [[ ]] and rod cladding outer diameter decreased by [[ ]]
- Channel thickness increase by [[ ]]
- Channel thickness decrease by [[ ]]
- Fuel pellet outer diameter increase by [[ ]]
- Fuel pellet outer diameter decrease by [[ ]]
- Fuel rod pin pitch increase by [[ ]]
- Fuel rod pin pitch decrease by [[ ]]
- Rack wall thickness decrease by 0.004 inches
- Rack wall thickness increase by 0.004 inches
- Rack pitch increase by 0.125 inches
- Rack insert thickness decrease by 0.005 inches
- Rack insert thickness increase by 0.005 inches
- Rack insert width decrease by [[ ]]
- Rack insert width increase by [[ ]]

All the tolerances used in these analyses are at least 2σ design limits. The models developed for these studies were all based on the normal configuration presented in Section 5.4.

There is no manufacturing tolerance for a decrease in rack pitch; therefore, no tolerance case was performed on the pitch decrease.

Because the Boraflex is modeled as water in this analysis, no tolerance cases are performed on the Boraflex thickness or width.

This analysis uses the certified minimum ^{10}B areal density; therefore, no tolerance case was performed on the insert ^{10}B density.

5.6.2 Results

The results of the tolerance studies and uncertainties are provided in Table 15. The values are summed using Equation 2, which is adopted from NEI 12-16 (Reference 3). The Δk_{Ti} terms in this table represent the difference between the system reactivity with the specified tolerance perturbation and k_{Normal}. In Equation 2, a Δk_{Ti} value must be both positive and the largest for its respective term to be considered. The Δk_{Ui} terms in the table represent the uncertainty contributions to $k_{max}(95/95)$ of the spent fuel rack and from the problem and code specific uncertainties, which are combined with the tolerance contributions (Δk_{Ti}) using Equation 2.

$$\Delta k_{Uncertainty} = \sqrt{\sum_{i=1}^{n} \Delta k_{Ti}^2 + \sum_{i=1}^{n} \Delta k_{Ui}^2} \qquad (2)$$

Table 15 – Spent Fuel Storage Rack Tolerance and Uncertainty Δk Values

Term	Description	k_{eff}	MCNP-05P Uncertainty (1σ)	Δk	Δk Uncertainty (2σ)+
Δk_{T1}	Fuel enrichment increase	0.91814	[[	0.00418	[[
Δk_{T2}	Fuel pellet density increase	0.91529		0.00133	
Δk_{T3}	Gadolinia wt.% decrease	0.91998		0.00602	
Δk_{T4}	Rod clad thickness/outer diameter increase	0.90908		-0.00488	
Δk_{T4}	Rod clad thickness/outer diameter decrease	0.91990		0.00594	
Δk_{T5}	Channel thickness increase	0.91489		0.00093*	
Δk_{T5}	Channel thickness decrease	0.91475		0.00079	
Δk_{T6}	Pellet outer diameter increase	0.91504		0.00108*	
Δk_{T6}	Pellet outer diameter decrease	0.91378		-0.00018	
Δk_{T7}	Fuel rod pin pitch increase	0.91618		0.00222*	
Δk_{T7}	Fuel rod pin pitch decrease	0.91334		-0.00062	
Δk_{T8}	Rack wall thickness increase	0.91476		0.00080*	
Δk_{T8}	Rack wall thickness decrease	0.91463		0.00067	
Δk_{T9}	Rack pitch increase	0.89850		-0.01546	
Δk_{T10}	Rack insert thickness decrease	0.91446		0.00050	
Δk_{T10}	Rack insert thickness increase	0.91554		0.00158*	
Δk_{T11}	Rack insert width decrease	0.91553		0.00157*	
Δk_{T11}	Rack insert width increase	0.91452	]]	0.00056	]]
Δk_{U1}	Critical benchmark bias uncertainty (95/95) (MCNP-05P versus critical experiments)	[[			
Δk_{U2}	TGBLA06 eigenvalue uncertainty (95/95)				]]
Δk_{U3}	Uncertainty on k_{Normal} (2 x 1σ value for base term in Table 10)	-	[[		
Δk_{U4}	Uncertainty of Δk bias contributors (2σ)	-			
Δk_{U5}	Uncertainty of Δk tolerance contributors (2σ)	-			
Δk_{U6}	Uncertainty in fuel depletion	-			]]
$\Delta k_{Uncertainty}$				[[	]]

* For conservatism, only positive values that are the largest for their respective term are considered.

[[ ]]

5.7 Maximum Reactivity

The maximum reactivity of the spent fuel rack without crediting Boraflex and with rack inserts installed, considering all biases, tolerances, and uncertainties, is calculated using Equation 3. The final values are presented in Table 16.

$$k_{max(95/95)} = k_{Normal} + \Delta k_{Bias} + \Delta k_{Uncertainty} \quad (3)$$

Table 16 – Spent Fuel Storage Rack Results Summary

Term	Value
k_{Normal}	0.91396
Δk_{Bias}	[[
$\Delta k_{Uncertainty}$	]]
$k_{max}(95/95)$	0.94200

[[

]]

6.0 CONCLUSIONS

The Quad Cities spent fuel racks have been analyzed for the storage of GNF3 fuel using the MCNP-05P Monte Carlo neutron transport program and the k_∞ criterion methodology. A maximum SCCG, uncontrolled peak in-core eigenvalue (k_∞) of 1.29 as defined by TGBLA06 is specified as the rack design limit for GNF3 fuel in the spent fuel racks with NETCO-SNAP-IN® rack inserts installed. The analyses resulted in a storage rack maximum k-effective (k_{max}(95/95)) less than the 10 CFR 50.68 limit of 0.95 for normal and credible abnormal operation with tolerances and computational uncertainties taken into account. Documentation that all legacy Quad Cities fuel types meet the k_{max}(95/95) limit is found in Appendix B.

7.0 REFERENCES

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2. NUREG-0800, “Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition,” Section 9.1.1, “Criticality Safety of Fresh and Spent Fuel Storage and Handling,” US NRC, Revision 3, March 2007. (NRC ADAMS Accession Number ML070570006).
3. NEI 12-16 Revision 4, “Guidance for Performing Criticality Analyses of Fuel Storage at Light-Water Reactor Power Plants,” September 2019. (NRC ADAMS Accession Number ML18088B400).
4. LA-UR-03-1987, “MCNP – A General Monte Carlo N-Particle Transport Code, Version 5,” April 2003.
5. NUREG/CR-6698, “Guide for Validation of Nuclear Criticality Safety Calculational Methodology,” US NRC, January 2001. (NRC ADAMS Accession Number ML050250061).
6. J.R. Taylor, “An Introduction to Error Analysis,” page 268-271, 2nd Edition, University Science Books, 1997.

APPENDIX A - MCNP-05P CODE VALIDATION

Table 17 presents the results of the benchmark calculations described in Section 3.4. Note that it is necessary to make an adjustment to the calculated k_{eff} value if the critical experiment being modeled was not at a critical state. This adjustment is done by normalizing the k_{calc} values to the experimental values, which is valid for small differences in k_{eff}. This normalization is reported as k_{norm} and is determined using Equation A-1. The combined uncertainty (σ_t) from the measurement and the calculation is also determined using Equation A-2.

$$k_{norm} = k_{calc}/k_{exp} \qquad \text{(A-1)}$$

$$\sigma_t = \sqrt{\sigma_{calc}^2 + \sigma_{exp}^2} \qquad \text{(A-2)}$$

Table 17 – MCNP-05P Results for the Benchmark Calculations

#	Experiment	Expt. #	Benchmark Eigenvalue (k_{exp})	Experimental Uncertainty (σ_{exp})	MCNP-05P Result (k_{calc})	MCNP-05P Uncertainty (σ_{calc})	Norm. Result (k_{norm})	Combined Uncertainty (σ_t)
[[								

#	Experiment	Expt. #	Benchmark Eigenvalue (k_{exp})	Experimental Uncertainty (σ_{exp})	MCNP-05P Result (k_{calc})	MCNP-05P Uncertainty (σ_{calc})	Norm. Result (k_{norm})	Combined Uncertainty (σ_t)

#	**Experiment**	**Expt. #**	**Benchmark Eigenvalue (k_{exp})**	**Experimental Uncertainty (σ_{exp})**	**MCNP-05P Result (k_{calc})**	**MCNP-05P Uncertainty (σ_{calc})**	**Norm. Result (k_{norm})**	**Combined Uncertainty (σ_t)**

#	Experiment	Expt. #	Benchmark Eigenvalue (k_{exp})	Experimental Uncertainty (σ_{exp})	MCNP-05P Result (k_{calc})	MCNP-05P Uncertainty (σ_{calc})	Norm. Result (k_{norm})	Combined Uncertainty (σ_t)

#	Experiment	Expt. #	Benchmark Eigenvalue (k_{exp})	Experimental Uncertainty (σ_{exp})	MCNP-05P Result (k_{calc})	MCNP-05P Uncertainty (σ_{calc})	Norm. Result (k_{norm})	Combined Uncertainty (σ_t)
								]]

A.1 - Trend Analysis

To determine if any trend is evident in this pool of experiments, the parameters listed in Table 18 were considered as independent variables.

Table 18 – Trending Parameters

Energy of the Average Lethargy causing Fission (EALF)
Uranium Enrichment (wt.% ^{235}U)
Plutonium Content (wt.% ^{239}Pu)
Atom ratio of hydrogen to fissile material (H/X)

Each parameter was plotted against the k_{norm} results independently for each case that was analyzed. These plots are provided in Figure 9 through Figure 12. This scatter plot of data was first analyzed by visual inspection to determine if any trends were readily apparent in the data. During this inspection, the axes of the graphs were modified to different scales to allow for a more thorough review. No clear evidence of a trend, linear or otherwise, was observed from this inspection.

[[

]]

Figure 9 – Scatterplot of k_{norm} versus EALF

[[

]]

Figure 10 – Scatterplot of k_{norm} versus ^{235}U wt.%{3}

[[

]]

Figure 11 – Scatterplot of k_{norm} versus ^{239}Pu wt.%

[[

]]

Figure 12 – Scatterplot of k_{norm} versus H/X

To further check for trends in the data, a linear regression was performed. The linear regression fitted equation is in the form y(x)= a +bx, where y is the dependent variable (k_{norm}) and x is any of the predictor variables from Table 18. Unweighted k_{norm} values were used in this evaluation, although it is noted that, due to the very similar σ values reported in Table 17, using weighted values would produce very similar results. This regression was performed using the built-in regression analysis tool in Excel. The fitted lines are included in Figure 9 through Figure 12. Again, it is noted through visual inspection that the trends do not appear to exhibit a strong correlation to the data. A useful tool to validate this claim is the linear correlation coefficient. This is a quantitative measure of the degree to which a linear relation exists between two variables. It is often expressed as the square term, r^2, and can be calculated directly using built in functions in Excel. The closer r^2 gets to the value of 1, the better the fit of data is expected to be to the linear equation. Results from this linear regression evaluation are summarized in Table 19.

A final method to test for goodness of fit is the chi squared test (χ^2). This method is explained in detail in Reference 6. In general, it can be stated that χ^2 is an indicator of the agreement between the observed (calculated) and expected (fitted) values for some variable. For linear goodness of fit testing using this method, Equation A-3 is utilized, where the expected value of $f(x_i)$ corresponds to the linear fitted equation for the trending parameter, x_i.

$$\chi^2 = \sum_1^N \left(\frac{k_{calc^i} - f(x_i)}{\sigma_{calc^i}}\right)^2 \qquad \text{(A-3)}$$

A more convenient way to report this result is the reduced chi squared value, which is denoted as $\tilde{\chi}^2$ and is defined by Equation A-4, where d is the degrees of freedom for the evaluation.

$$\tilde{\chi}^2 = \chi^2/d \qquad \text{(A-4)}$$

If a value of one or less is obtained for this equation, then there is no reason to doubt the expected (fitted) distribution is reasonable; however, if the value is much larger than one, the expected distribution is unlikely to be a good fit. Results for each trending parameter are summarized in Table 19.

Table 19 – Trending Results Summary

Trend Parameter	Intercept	Slope	r^2	$\tilde{\chi}^2$	Valid Trend
EALF	[[				No
^{235}U wt.%					No
^{239}Pu wt.%					No
H/X				]]	No

The results in Table 19 clearly demonstrate that there are no statistically significant or valid trends of k_{norm} with any of the trending parameters.

A.2 - Bias and Bias Uncertainty Calculation – Single Sided Tolerance Limit

As no trends are apparent in the critical experiment results, a weighted single-sided tolerance limit methodology is utilized to establish the bias and bias uncertainty for this AOA and code package combination. Use of this method requires the critical experiment results to have a normal statistical distribution. This was verified using the Anderson-Darling normality test. A graphical image of the results for this normality test, including the p-value for the distribution, is provided in Figure 13. Because the reported p-value is greater than 0.05, it is confirmed that the data fits a normal distribution, and the single sided tolerance limit methodology is confirmed to be applicable.

[[

]]

Figure 13 – Normality Test of k_{norm} Results

When using this method, the weighted bias and bias uncertainty are calculated using the following equations:

$$Bias = \bar{k}_{norm} - 1 \tag{A-5}$$

$$Bias\ Uncertainty = U \cdot S_p \tag{A-6}$$

$$\bar{k}_{norm} = \frac{\sum_{i=1}^{n} \frac{k_{norm\,i}}{\sigma_t^2}}{\sum_{i=1}^{n} \frac{1}{\sigma_t^2}} \tag{A-7}$$

$$S_P = \sqrt{s^2 + \bar{\sigma}^2} \tag{A-8}$$

$$\bar{\sigma}^2 = \frac{n}{\sum_{i=1}^{n} \frac{1}{\sigma_t^2}} \tag{A-9}$$

$$s^2 = \frac{\left(\frac{1}{n-1}\right)\sum_{i=1}^{n}\frac{1}{\sigma_t^2}\left(k_{norm\,i} - \bar{k}_{norm}\right)^2}{\frac{1}{n}\sum_{i=1}^{n}\frac{1}{\sigma_t^2}} \qquad \text{(A-10)}$$

where:

$\bar{k}_{norm}$ = Average weighted k_{norm}

S_P = Pooled standard deviation

s^2 = Variance about the mean

$\bar{\sigma}^2$ = Average total variance

U = one-sided tolerance factor for n data points at (95/95 confidence/probability level)

n = number of data points (=[[ ]])

Table 20 summarizes the results of these calculations.

Table 20 - Bias and Bias Uncertainty for MCNP-05P with ENDF/B-VII

Bias (weighted)	[[
Bias Uncertainty (95/95 level)	
Variance About the Mean	
Average Total Variance	
Pooled Standard Deviation (1σ)	
One-Sided Tolerance Factor	]]

Using the average weighted bias and pooled standard deviation; the upper one-sided 95/95-tolerance limit (bias uncertainty) was calculated for use in criticality calculations, in accordance with NUREG/CR-6698 (Reference 5) guidance. As seen in Figure 13, [[

]] As shown in Table 20, the MCNP-05P bias uncertainty (95/95) [[

]] Table 21 summarizes the recommended bias and bias uncertainty to be used in criticality calculations.

Table 21 – Recommended Bias and Bias Uncertainty in Criticality Analyses for MCNP-05P with ENDF/B-VII

Bias	[[
Bias Uncertainty (95/95)	]]

APPENDIX B - LEGACY FUEL STORAGE JUSTIFICATION

Exposure dependent, maximum, uncontrolled in-core k_∞ results for each fuel assembly in the Quad Cities spent fuel pools are confirmed to be less than 1.29. The in-core k_∞ values have been calculated using the process for validating that specific assembly designs are acceptable for storage in the Quad Cities fuel storage racks, as outlined in Section 3.5, and the in-core reactivity values are presented in Table 22. This information demonstrates that all fuel assemblies currently in the Quad Cities spent fuel pool have considerable margin to the reactivity of the GNF3 design basis bundle used in this analysis. Any GNF3 bundles in the Quad Cities core or spent fuel pool are covered by the design basis bundle study in Section 5.3.

The GNF3 design basis bundle with an in-core k_∞ value of 1.29 was shown to be below the 10 CFR 50.68 0.95 in-rack k-effective limit when analyzed in the storage racks. As represented in Table 22, the limiting legacy GNF fuel type and limiting legacy non-GNF fuel provided by Exelon have a significantly lower in-core k_∞ value than the GNF3 design basis bundle (i.e., less reactive than the design basis bundle). Therefore, it is confirmed that all legacy fuel bundles are safe for storage in the Quad Cities spent fuel storage racks with rack inserts installed.

Table 22 – Limiting Cold As-Designed Eigenvalue of Bundles Inserted Into Quad Cities

Bundle	Bundle Name	In-Core k_∞
[[		
		]]

Table 23 shows a list of legacy fuel bundles in Quad Cities spent fuel pools that have empty rod locations. The in-core k_∞ values have been calculated with all rods in their original locations using the methodology outlined in Section 3.5. The margin to safety was confirmed to exist in the storage racks by analyzing the bundles with the corresponding rods removed to reflect their current state. These bundles were analyzed under nominal storage conditions, as outlined in Section 5.4, and the in-rack reactivity values are presented in Table 23. The GNF3 design basis bundle, with a nominal in-rack k_∞ value reported in Table 10, was shown to be below the 10 CFR 50.68 k_{max}(95/95) limit of 0.95 when analyzed in the storage racks. As represented in Table 23, the legacy bundles with missing rod locations have a significantly lower in-rack k_∞ value than the GNF3 design basis bundle (i.e., less reactive than the design basis bundle). Therefore, it is confirmed that these legacy fuel bundles with missing rod locations are safe for storage in the Quad Cities spent fuel storage racks with rack inserts installed.

Table 23 – Legacy Bundles with Missing Rod Locations at Quad Cities

Assembly	Bundle Name & Description	# Rods Missing	In-Rack Nominal Reactivity
[[			
			]]

ATTACHMENT 4

NEI 12-16 Criticality Analysis Checklist

APPENDIX C: CRITICALITY ANALYSIS CHECKLIST

The criticality analysis checklist is completed by the applicant prior to submittal to the NRC. It provides a useful guide to the applicant to ensure that all the applicable subject areas are addressed in the application, or to provide justification/identification of alternative approaches.

The checklist also assists the NRC reviewer in identifying areas of the analysis that conform or do not conform to the guidance in NEI 12-16. Subsequently, the NRC review can then be more efficiently focused on those areas that deviate from NEI 12-16 and the justification for those deviations.

Subject	Included	Notes / Explanation
1.0 Introduction and Overview		
Purpose of submittal	YES	Section 1.0 of NEDC-33932P and Section 1.0 of Attachment 1
Changes requested	YES	Section 1.0 of NEDC-33932P and Sections 2.1, 2.2, and 2.3 of Attachment 1
Summary of physical changes	YES	Section 1.0 of NEDC-33932P and Sections 2.1, 2.2, and 2.3 of Attachment 1
Summary of Tech Spec changes	YES	Section 2.3 of Attachment 1 and Attachment 2
Summary of analytical scope	YES	Sections 1.0 and 3.0 of NEDC-33932P and Sections 2.1 and 2.2 of Attachment 1
2.0 Acceptance Criteria and Regulatory Guidance		
Summary of requirements and guidance	YES	Section 2.0 of NEDC-33932P and Sections 2.1 and 2.2 of Attachment 1
Requirements documents referenced	YES	Section 2.0 of NEDC-33932P and Sections 2.1 and 2.2 of Attachment 1
Guidance documents referenced	YES	Section 2.0 of NEDC-33932P and Sections 2.1 and 2.2 of Attachment 1
Acceptance criteria described	YES	Section 2.0 of NEDC-33932P and Sections 2.1 and 2.2 of Attachment 1
3.0 Reactor and Fuel Design Description		
Describe reactor operating parameters	NO	Not applicable for this analysis. See Sections 3.7 and 5.5 of NEDC-33932P for depletion parameters and assumptions.
Describe all fuel in pool	YES	Section 4.0 of NEDC-33932P

Subject	Included	Notes / Explanation
Geometric dimensions (Nominal and Tolerances)	YES	Section 4.1 of NEDC-33932P
Schematic of guide tube patterns	NO	Not applicable for BWR fuel
Material compositions	YES	Section 4.0 of NEDC-33932P
Describe future fuel to be covered	YES	Section 4.0 of NEDC-33932P
Geometric dimensions (Nominal and Tolerances)	YES	Section 4.1 of NEDC-33932P
Schematic of guide tube patterns	NO	Not applicable for BWR fuel
Material compositions	YES	Section 4.0 of NEDC-33932P
Describe all fuel inserts	NO	There are no fuel inserts in analysis NEDC-33932P.
Geometric Dimensions (Nominal and Tolerances)		
Schematic (axial/cross-section)		
Material compositions		
Describe non-standard fuel	YES	Section 4.0, Appendix B of NEDC-33932P
Geometric dimensions		
Describe non-fuel items in fuel cells	YES	Section 4.0 of NEDC-33932P
Nominal and tolerance dimensions	NO	Not applicable; analysis NEDC-33932P covers peak reactivity in every rack cell location
4.0 Spent Fuel Pool/Storage Rack Description		
New fuel vault & Storage rack description	YES	The new fuel vault analysis will be covered by the GESTAR II methodology and is not addressed in NEDC-33932P. See Section 2.2 of Attachment 1 for details.
Nominal and tolerance dimensions		
Schematic (axial/cross-section)		
Material compositions		
Spent fuel pool, Storage rack description	YES	Sections 5.1-5.2 of NEDC-33932P and Section 3.1 of Attachment 1
Nominal and tolerance dimensions		
Schematic (axial/cross-section)		
Material compositions		
Other Reactivity Control Devices (Inserts)	YES	Sections 5.1-5.2 of NEDC-33932P and Section 3.1 of Attachment 1
Nominal and tolerance dimensions		
Schematic (axial/cross-section)		
Material compositions		
5.0 Overview of the Method of Analysis		
New fuel rack analysis description	YES	The new fuel vault analysis will be covered by the GESTAR II methodology and is not addressed in NEDC-33932P. See Section 2.2 of Attachment 1 for details.
Storage geometries		
Bounding assembly design(s)		
Integral absorber credit		
Accident analysis		
Spent fuel storage rack analysis description	YES	Sections 3.5-3.7 and 5.0 of NEDC-33932P
Storage geometries	YES	Sections 5.1-5.2 of NEDC-33932P

Subject	Included	Notes / Explanation
Bounding assembly design(s)	YES	Section 5.3 of NEDC-33932P
Soluble boron credit	NO	Not applicable - No soluble boron credit in this BWR criticality analysis (NEDC-33932P)
Boron dilution analysis		
Burnup credit	NO	No burnup credit in BWR peak reactivity analysis NEDC-33932P – fuel is evaluated at peak reactivity
Decay/Cooling time credit	NO	No decay/cooling time credit in analysis of NEDC-33932P
Integral absorber credit	YES	Sections 5.1-5.2 of NEDC-33932P
Other credit	NO	No other credit in analysis NEDC-33932P
Fixed neutron absorbers	YES	Rack inserts - Sections 5.1-5.2 of NEDC-33932P
Aging management program	NO	Aging is not included in analysis NEDC-33932P; no credit is taken for Boraflex
Accident analysis	YES	Section 5.5.3 of NEDC-33932P
Temperature increase	YES	Sections 5.4-5.5 of NEDC-33932P
Assembly drop	YES	Section 5.5.3 of NEDC-33932P
Single assembly misload	YES	Section 5.5.3 of NEDC-33932P
Multiple misload	NO	Uniform pool, no opportunity for multiple misload
Boron dilution	NO	Not applicable - No soluble boron credit in this BWR criticality analysis (NEDC-33932P)
Other	YES	Sections 5.5.3 of NEDC-33932P
Fuel out of rack analysis	YES	Section 5.5 of NEDC-33932P considers worst case abnormal positioning of a fuel assembly outside the storage rack.
Handling		
Movement		
Inspection		
6.0 Computer Codes, Cross Sections and Validation Overview		
Code/Modules Used for Calculation of k_{eff}	YES	Section 3.0 of NEDC-33932P
Cross section library	YES	Section 3.1 of NEDC-33932P
Description of nuclides used	YES	Section 4.2 of NEDC-33932P
Convergence checks	YES	Section 3.3 of NEDC-33932P
Code/Module Used for Depletion Calculation	YES	Section 3.0 of NEDC-33932P
Cross section library	YES	Section 3.1 of NEDC-33932P
Description of nuclides used	YES	Section 4.2 of NEDC-33932P
Convergence checks	YES	Section 3.3 of NEDC-33932P
Validation of Code and Library	YES	Section 3.4, Appendix A of NEDC-33932P

Subject	Included	Notes / Explanation
Major Actinides and Structural Materials	YES	Section 3.4 of NEDC-33932P
Minor Actinides and Fission Products	YES	Section 3.4 of NEDC-33932P
Absorbers Credited	YES	Section 3.4 of NEDC-33932P
7.0 Criticality Safety Analysis of the New Fuel Rack		
Rack model	YES	The NFV rack CSA coverage for the new GNF3 fuel will be the GESTAR II analysis for the GE designed low density NFV racks upon approval of this license amendment. The QCNPS NFV racks are GE designed low density racks with an interrack spacing of 11 inches, which is ≥ 10.5 inches (the criteria listed in GESTAR II) and thus, the racks may be utilized to store new GNF fuel with in-core SCCG $k_{inf} \leq 1.31$. See Section 2.2 of Attachment 1 for details.
Boundary conditions		
Source distribution		
Geometry restrictions		
Limiting fuel design		
Fuel density		
Burnable Poisons		
Fuel dimensions		
Axial blankets		
Limiting rack model		
Storage vault dimensions and materials		
Temperature		
Multiple regions/configurations		
Flooded		
Low density moderator		
Eccentric fuel placement		
Tolerances		
Fuel geometry		
Fuel pin pitch		
Fuel pellet OD		
Fuel clad OD		
Fuel content		
Enrichment		
Density		
Integral absorber		
Rack geometry		
Rack pitch		
Cell wall thickness		
Storage vault dimensions/materials		
Code uncertainty		
Biases		
Temperature		
Code bias		
Moderator Conditions		
Fully flooded and optimum density moderator		

Subject	Included	Notes / Explanation
8.0 Depletion Analysis for Spent Fuel		
Depletion Model Considerations	YES	Sections 3.0, 3.3, 3.4, 3.7, and 4.2 of NEDC-33932P
Time step verification		
Convergence verification		
Simplifications		
Non-uniform enrichments		
Post Depletion Nuclide Adjustment		
Cooling Time		
Depletion Parameters		
Burnable Absorbers		
Integral Absorbers		
Soluble Boron		
Fuel and Moderator Temperature		
Power		
Control rod insertion		
Atypical Cycle Operating History		
9.0 Criticality Safety Analysis of Spent Fuel Pool Storage Racks		
Rack model	YES	Section 5.2 of NEDC-33932P
Boundary conditions		
Source distribution		
Geometry restrictions		
Design Basis Fuel Description	YES	Section 5.3 of NEDC-33932P
Fuel density	YES	Section 4.1 of NEDC-33932P
Burnable Poisons	YES	Section 5.2 of NEDC-33932P
Fuel assembly inserts	NO	No fuel assembly inserts in analysis of NEDC-33932P
Fuel dimensions	YES	Section 4.1 of NEDC-33932P
Axial blankets	NO	Section 3.7 of NEDC-33932P
Configurations considered	YES	Section 6.0 of NEDC-33932P
Borated	NO	Not applicable for this BWR analysis (NEDC-33932P)
Unborated	YES	BWR analysis NEDC-33932P considers unborated SFP
Multiple rack designs	NO	Only one spent fuel rack design is present at Quad Cities and the NFV racks are covered by the GESTAR II analysis (see Section 2.2 of Attachment 1).
Alternate storage geometry	NO	Not applicable for analysis of NEDC-33932P
Reactivity Control Devices	YES	Sections 5.1- 5.2 of NEDC-33932P
Fuel Assembly Inserts	NO	No fuel assembly inserts in analysis NEDC-33932P

Subject	Included	Notes / Explanation
Storage Cell Inserts	YES	Sections 5.1- 5.2 of NEDC-33932P
Storage Cell Blocking Devices	NO	No blocking devices in analysis NEDC-33932P
Axial burnup shapes	NO	Section 3.7 of NEDC-33932P
Uniform/Distributed	YES	Section 3.7 of NEDC-33932P
Nodalization	NO	Section 3.7 of NEDC-33932P
Blankets modeled	NO	Section 3.7 of NEDC-33932P
Tolerances/Uncertainties	YES	Section 5.6 of NEDC-33932P
Fuel geometry		
Fuel rod pin pitch		
Fuel pellet OD		
Cladding OD		
Axial fuel position	NO	Section 3.7 of NEDC-33932P
Fuel content	YES	Section 5.6 of NEDC-33932P
Enrichment		
Density		
Assembly insert dimensions and materials	NO	No fuel assembly inserts in analysis NEDC-33932P
Rack geometry	YES	Section 5.6 of NEDC-33932P
Flux-trap size (width)	NO	Not applicable to non-flux-trap racks
Rack cell pitch	YES	Section 5.6 of NEDC-33932P
Rack wall thickness	YES	Section 5.6 of NEDC-33932P
Neutron Absorber Dimensions	YES	Section 5.6 of NEDC-33932P
Rack insert dimensions and materials	YES	Section 5.6 of NEDC-33932P
Code validation uncertainty	YES	Sections 3.4, 5.6, and Appendix A of NEDC-33932P
Criticality case uncertainty	YES	Section 5.6 of NEDC-33932P
Depletion Uncertainty	YES	Sections 3.4, 5.6 of NEDC-33932P
Burnup Uncertainty	NO	Not applicable for BWR peak reactivity analysis NEDC-33932P
Biases	YES	Section 5.0 of NEDC-33932P
Design Basis Fuel design	YES	Section 5.3 of NEDC-33932P
Code bias	YES	Sections 3.4, 5.5 of NEDC-33932P
Temperature	YES	Section 5.4 of NEDC-33932P
Eccentric fuel placement	YES	Sections 5.4-5.5 of NEDC-33932P
Incore thimble depletion effect	NO	Not applicable for analysis NEDC-33932P
NRC administrative margin	NO	Not applicable for analysis NEDC-33932P
Modeling simplifications	YES	Sections 3.7, 4.2 of NEDC-33932P
Identified and described		

Subject	Included	Notes / Explanation
10.0 Interface Analysis		
Interface configurations analyzed	NO	N/A, the spent fuel pool is uniform with rack inserts in every cell. Only one rack design.
Between dissimilar racks	NO	
Between storage configurations within a rack	NO	
Interface restrictions	NO	
11.0 Normal Conditions		
Fuel handling equipment	NO	Not in the scope and does not impact results of criticality analysis NEDC-33932P.
Administrative controls	NO	No new administrative controls included in NEDC-33932P or Attachment 1.
Fuel inspection equipment or processes	NO	Not in the scope and does not impact results of criticality analysis NEDC-33932P.
Fuel reconstitution	YES	Section 4.0 of NEDC-33932P
12.0 Accident Analysis		
Boron dilution	NO	Not applicable - No soluble boron credit in this BWR criticality analysis (NEDC-33932P)
Normal conditions		
Accident conditions		
Single assembly misload	YES	Section 5.5 of NEDC-33932P
Fuel assembly misplacement	YES	Section 5.5 of NEDC-33932P
Neutron Absorber Insert Misload	YES	Section 5.5 of NEDC-33932P
Multiple fuel misloads	NO	Uniform pool, single storage configuration, no opportunity for multiple misloads
Dropped assembly	YES	Section 5.5 of NEDC-33932P
Temperature	YES	Section 5.4 of NEDC-33932P
Seismic event/other natural phenomena	YES	Section 5.5 of NEDC-33932P
13.0 Analysis Results and Conclusions		
Summary of results	YES	Section 6.0 of NEDC-33932P
Burnup curve(s)	NO	Not applicable for BWR peak reactivity analyses, including NEDC-33932P
Intermediate Decay time treatment	NO	Not applicable for BWR peak reactivity analyses, including NEDC-33932P
New administrative controls	NO	No new administrative controls included in NEDC-33932P or Attachment 1.
Technical Specification markups	YES	Section 2.3 of Attachment 1 and Attachment 2

Subject	Included	Notes / Explanation
14.0 References	YES	**Section 7.0** of NEDC-33932P
Appendix A: Computer Code Validation:		**Appendix A** of NEDC-33932P
Code validation methodology and bases	YES	Appendix A of NEDC-33932P
New Fuel		
Depleted Fuel		
MOX		
HTC		
Convergence		
Trends		
Bias and uncertainty		
Range of applicability	YES	Described in Section 3.4 of NEDC-33932P
Analysis of Area of Applicability coverage	YES	Described in Section 3.4 of NEDC-33932P

ATTACHMENT 5

Global Nuclear Fuels - Americas, LLC 10 CFR 2.390 Affidavit

Global Nuclear Fuel – Americas

AFFIDAVIT

I, Brian R. Moore, state as follows:

(1) I am General Manager, Core & Fuel Engineering, Global Nuclear Fuel – Americas, LLC ("GNF-A"), and have been delegated the function of reviewing the information described in paragraph (2) which is sought to be withheld, and have been authorized to apply for its withholding.

(2) The information sought to be withheld is contained in GNF-A proprietary report, NEDC-33932P, "Quad Cities Units 1 and 2 Fuel Storage Criticality Safety Analysis," Revision 1, October 2021. GNF-A proprietary information within the text and tables is identified by a dotted underline placed within double square brackets. [[This sentence is an example.{3}]] Figures and large objects containing GNF-A proprietary information are identified with double square brackets before and after the object. In all cases, the superscript notation {3} refers to Paragraph (3) of this affidavit, which provides the basis for the proprietary determination.

(3) In making this application for withholding of proprietary information of which it is the owner or licensee, GNF-A relies upon the exemption from disclosure set forth in the Freedom of Information Act ("FOIA"), 5 USC Sec. 552(b)(4), and the Trade Secrets Act, 18 USC Sec. 1905, and NRC regulations 10 CFR 9.17(a)(4), and 2.390(a)(4) for "trade secrets" (Exemption 4). The material for which exemption from disclosure is here sought also qualify under the narrower definition of "trade secret", within the meanings assigned to those terms for purposes of FOIA Exemption 4 in, respectively, Critical Mass Energy Project v. Nuclear Regulatory Commission, 975 F2d 871 (DC Cir. 1992), and Public Citizen Health Research Group v. FDA, 704 F2d 1280 (DC Cir. 1983).

(4) Some examples of categories of information which fit into the definition of proprietary information are:

a. Information that discloses a process, method, or apparatus, including supporting data and analyses, where prevention of its use by GNF-A's competitors without license from GNF-A constitutes a competitive economic advantage over other companies;

b. Information which, if used by a competitor, would reduce his expenditure of resources or improve his competitive position in the design, manufacture, shipment, installation, assurance of quality, or licensing of a similar product;

c. Information which reveals aspects of past, present, or future GNF-A customer-funded development plans and programs, resulting in potential products to GNF-A;

d. Information which discloses patentable subject matter for which it may be desirable to obtain patent protection.

The information sought to be withheld is considered to be proprietary for the reasons set forth in paragraphs (4)a. and (4)b. above.

(5) To address 10 CFR 2.390 (b) (4), the information sought to be withheld is being submitted to NRC in confidence. The information is of a sort customarily held in confidence by GNF-A, and is in fact so held. The information sought to be withheld has, to the best of my knowledge and belief, consistently been held in confidence by GNF-A, no public disclosure has been made, and it is not available in public sources. All disclosures to third parties including any required transmittals to NRC, have been made, or must be made, pursuant to regulatory provisions or proprietary agreements which provide for maintenance of the information in confidence. Its initial designation as proprietary information, and the subsequent steps taken to prevent its unauthorized disclosure, are as set forth in paragraphs (6) and (7) following.

(6) Initial approval of proprietary treatment of a document is made by the manager of the originating component, the person most likely to be acquainted with the value and sensitivity of the information in relation to industry knowledge, or subject to the terms under which it was licensed to GNF-A.

(7) The procedure for approval of external release of such a document typically requires review by the staff manager, project manager, principal scientist or other equivalent authority, by the manager of the cognizant marketing function (or his delegate), and by the Legal Operation, for technical content, competitive effect, and determination of the accuracy of the proprietary designation. Disclosures outside GNF-A are limited to regulatory bodies, customers, and potential customers, and their agents, suppliers, and licensees, and others with a legitimate need for the information, and then only in accordance with appropriate regulatory provisions or proprietary agreements.

(8) The information identified in paragraph (2), above, is classified as proprietary because it contains details of GNF-A's fuel design and licensing methodology. The development of this methodology, along with the testing, development and approval was achieved at a significant cost to GNF-A or its licensor.

The development of the fuel design and licensing methodology along with the interpretation and application of the analytical results is derived from the extensive experience database that constitutes a major GNF-A asset.

(9) Public disclosure of the information sought to be withheld is likely to cause substantial harm to GNF-A's competitive position and foreclose or reduce the availability of profit-making opportunities. The information is part of GNF-A's comprehensive BWR safety and technology base, and its commercial value extends beyond the original development cost. The value of the technology base goes beyond the extensive physical database and analytical methodology and includes development of the expertise to determine and apply the appropriate evaluation process. In addition, the technology base includes the value derived from providing analyses done with NRC-approved methods.

The research, development, engineering, analytical, and NRC review costs comprise a substantial investment of time and money by GNF-A.

The precise value of the expertise to devise an evaluation process and apply the correct analytical methodology is difficult to quantify, but it clearly is substantial.

GNF-A's competitive advantage will be lost if its competitors are able to use the results of the GNF-A experience to normalize or verify their own process or if they are able to claim an equivalent understanding by demonstrating that they can arrive at the same or similar conclusions.

The value of this information to GNF-A would be lost if the information were disclosed to the public. Making such information available to competitors without their having been required to undertake a similar expenditure of resources would unfairly provide competitors with a windfall, and deprive GNF-A of the opportunity to exercise its competitive advantage to seek an adequate return on its large investment in developing and obtaining these very valuable analytical tools.

I declare under penalty of perjury that the foregoing is true and correct.

Executed on this 15th day of October 2021.

Brian R. Moore
General Manager, Core & Fuel Engineering
Global Nuclear Fuel – Americas, LLC
3901 Castle Hayne Road
Wilmington, NC 28401
Brian.Moore@ge.com

ATTACHMENT 6

Curtiss-Wright Corporation 10 CFR 2.390 Affidavit

CURTISS-WRIGHT

Nuclear Division
44 Shelter Rock Road
Danbury, CT 06810
T: 203.448.3310 | F: 203.437.6279
www.cwnuclear.com

AFFIDAVIT

I, Matthew C. Harris, Segment Manager of NETCO, business segment of Scientech, Curtiss-Wright Corporation, do hereby affirm and state:

1. I am the Segment Manager of the NETCO, business segment of Scientech, Curtiss-Wright Corporation, and am authorized to execute this affidavit on its behalf. I am further authorized to review information submitted to the Nuclear Regulatory Commission (NRC) and apply to the NRC for the withholding of information from disclosure.

2. The information sought to be withheld is contained in the GNF Report "Quad Cities Units 1 and 2: Fuel Storage Criticality Safety Analysis" – NEDC-33932P, Revision 1, October 2021. Curtiss-Wright Flow Control Service, Co. LLC confidential proprietary information is identified by a solid underline inside double square brackets. [[This sentence is an example. {C}]] Curtiss-Wright proprietary information in Figures and large objects is identifiable by double square brackets before and after the object.

3. In making this application for withholding of proprietary information of which it is the owner, NETCO relies on provisions of NRC regulation 10 CFR 2.390(a)(4). The information for which exemption from disclosure is sought is confidential commercial information.

4. The proprietary information provided by NETCO should be held in confidence by the NRC pursuant to the policy reflected in 10 CFR 2.390(a)(4) because:

   a) The information sought to be withheld in the Report (see paragraph 2 above) is and has been held in confidence by NETCO.

   b) This information is of a type that is customarily held in confidence by NETCO, and there is a rational basis for doing so because the information contains methodology, data and supporting information developed by NETCO that could be used by a competitor as a competitive advantage.

   c) This information is being transmitted to the NRC in confidence.

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d) This information sought to be withheld, to the best of my knowledge and belief, is not available in public sources and no public disclosure has been made.

e) The information sought to be withheld contains developed, patented, product fabrication data and supporting information that could be used by a competitor as a competitive advantage, and would result in substantial harm to the competitive position of NETCO. This information would reduce the expenditure of resources and improve his competitive position in the implementation of a similar product. Third party agreements have been established to ensure maintenance of the information in confidence. The development of the methodology, data and supporting information was achieved at a significant cost to NETCO. Public disclosure of this information sought to be withheld is likely to cause substantial harm to NETCO's competitive position and reduce the availability of profit-making opportunities.

5. Initial approval of proprietary treatment of a document is made by the Segment Manager of NETCO, business segment of Scientech, the person most likely to be familiar with the value and sensitivity of the information and its relation to industry knowledge. Access to such information within NETCO is on a "need to know" basis.

6. Accordingly, NETCO requests that the designated document be withheld from public disclosure pursuant to 10 CFR 2.390(a)(4).

I declare under penalty of perjury that the foregoing affidavit and statements therein are true and correct to the best of my knowledge, information and belief.

Matthew C. Harris
Segment Manager, NETCO , business segment of Scientech
Curtiss-Wright Corporation

Date: __10/15/21__________