ENCLOSURE 2

MFN 16-053

GNF Response to RAIs

Non-Proprietary Information – Class I (Public)

IMPORTANT NOTICE

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

Additional Change in GESTAR II Section 1.2.3.G

Section 1.2.3.G of GESTAR II includes text repeated from Section 3.5 that is not necessary. Section 1.2.3.G will be modified as follows to simply reference Section 3.5:

G. *The effective multiplication factor for new fuel stored under normal and abnormal conditions shall be shown to meet fuel storage limits by demonstrating that the peak uncontrolled lattice k–infinity calculated in a normal reactor core configuration meets the limits provided in Section 3 for GE designed regular or high density storage racks.*

~~The basic criterion associated with the storage of both irradiated and new fuel is that the effective multiplication factor of fuel stored under normal conditions will be less than or equal to 0.90 for regular density racks and less than or equal to 0.95 for high density racks. Abnormal storage conditions are limited to a k_{eff} of less than or equal to 0.95 for both high and regular density designs.~~ For GE designed fuel storage racks, the~~se~~ storage criteria are satisfied if the uncontrolled lattice k–infinity calculated in the normal reactor core configuration meets the conditions documented in Subsection 3.5.

Revised in-core k_∞ limit for low-density spent fuel storage racks in GESTAR II Section 3.5

The previous low-density spent fuel rack criticality analysis assumed the important effects of moderator temperature are captured by modifying only the density of the water in the calculation. Therefore, the 20°C cross section library and S(α,β) water scattering kernel were used for all moderator temperature cases.

It has recently been discovered that for low-density spent fuel racks, the temperature of the S(α,β) water scattering kernel has an effect on in-rack reactivity. Therefore, the criticality safety analysis of GE low-density spent fuel storage racks was updated using a 100°C cross section library and S(α,β) water scattering kernel for the 100°C moderator temperature case.

Due to the increased reactivity of using the 100°C cross section library and S(α,β) water scattering kernel, it is necessary to reduce the uncontrolled lattice k_∞ limit for GE low-density spent fuel racks to 1.28.

This results in the following change to the GESTAR II Section 3.5 changes proposed in Amendment 37 (Reference 1):

(a) $k_\infty \leq$ ~~1.31~~**1.28** for low-density spent fuel storage racks with an interrack spacing ≥ 11.70 inches.

It has been confirmed that GE/GNF fuel at plants with GE low-density spent fuel racks meet this limit.

Reference

1 Letter, Andrew A. Lingenfelter (GNF) to NRC Document Control Desk, Subject: Amendment 37 to NEDE-24011-P-A-19 and NEDE-24011-P-A-19-US, General Electric Standard Application for Reactor Fuel (GESTAR II) and the US Supplement, MFN 13-006, February 13, 2013.

Amendment 37 Requests for Additional Information

RAI-1

Section 1.2.7 indicates that "New coefficients for the critical power will be provided in the critical power correlation *report* for each fuel product line." Further, Section 1.2.7 C indicates that the listed criteria for "establishing the new correlation are those which were used in establishing the current GEXL and GEXL-PLUS correlations approved by NRC."

The above mentioned section of the GESTAR II document indicates that General Electric Hitachi (GEH) does not need NRC approval for a new critical power correlation even for a new product line but only to send a report to the NRC on the new correlation except for a situation where the form of the correlation is different. The staff would like GEH to verify whether it has explicit assurance from NRC that a new correlation does not need NRC staff approval.

Response

The approval of the subject section was part of the Amendment 22 process proposed in 1989 and approved by the NRC in 1990. The purpose of this process was to provide a framework of criteria and guidelines that when followed by GEH (or subsequently GNF) would allow a for a new fuel design to be effectively licensed without NRC review and approval. From Section 1.0 of the Safety Evaluation (SE):

> "If a fuel design complies with the fuel acceptance criteria, it is acceptable for licensing applications without the explicit staff review."

The characteristics of the GEXL model, requirements, and constraints were defined as part of the proposed process and became Section 1.1.7 and 1.2.7 of GESTAR. Therefore, the 1990 SE approving the Amendment 22 process also approved the GEXL process included therein.

RAI-2

Page 2-2 of NEDE-24011-P-Draft indicates that [[

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Please respond to the following request:

Describe how the lattice-dependent Maximum Average Peak Linear Heat Generation Rate (MAPLHGR)/or Linear Heat Generation Rate (LHGR) is different from the current methodology by which MAPLHGR is calculated.

Response

For the vast majority of plants (and cores), the MAPLHGR limit is set at the limiting exposure so to assure compliance by Emergency Core Cooling System – Loss-of-Coolant Accident (ECCS-LOCA) analysis to the acceptance criteria. The ECCS-LOCA evaluation model used with the SAFER code casts the axial power distribution conservatively by considering a limiting bundle with full length fuel rods and with the limits set to extremes for that single bundle. With the standard nodal modeling in the core, the power distribution is applied so that it is consistent with the integral of the power traversing up through the bundle. This is sufficient rigor to obtain bounding results for compliance purposes, as has been approved for application. This is the general condition where the LHGR limit is monitored directly.

As stated in a prior paragraph (page 2-1 of NEDE-24011-P-Draft), the model has been approved so that "Lattice local power and exposure peaking factors may be applied…" This instance occurs for plants and cores which more prominently challenge acceptance criteria and is applied by utilization of the approved CORCL code as part of the ECCS-LOCA evaluation model. The CORCL code offers a more detailed assessment of the core/bundles, taking into account fuel rod groupings, with attendant peaking variation, as affected by features such as part-length rods and radiation components to heat transfer. For these applications, the greater detail of the model identifies the spans between lattices and fuel rod condition as a function of exposure, and assigns a MAPLHGR value for each node. This additional detail is then coupled with the output from the SAFER code in calculating the results of the event for the specific bundle and for calculation of PCT and maximum local oxidation; the explicit MAPLHGR values which are shown to demonstrate compliance are identified for the bundle and spans therein. As a practical matter, the only plants which rely upon this additional modeling detail are the BWR/2 designs: Nine Mile Point Unit 1 and Oyster Creek. These are calculated with the MAPLGHR across each lattice and axial span, over a range of exposures, which supports the overall limiting – and acceptable – Peak Cladding Temperature (PCT) and maximum local oxidation result, so that the core is in compliance overall.

RAI-3

Please provide details for the k_∞ calculations for the lattices listed in Section 3.5, Reactivity of Fuel in Storage, for low density and high-density spent fuel storage racks as well as for low-density new fuel vault storage with various interrack spacing.

Response

The peak, cold in-core lattice infinite multiplication factor (k_∞) criterion for demonstrating compliance to the 10 CFR 50.68 fuel storage criticality criterion has been used for all GE-supplied fuel storage racks, and is currently used for re-rack designs at a number of plants. This methodology was presented in the recent Peach Bottom spent fuel pool criticality submittal and accepted by the Nuclear Regulatory Commission (NRC) in the Peach Bottom Safety Evaluation Report in Reference 3-1.

In-core eigenvalues and exposure dependent, pin-by-pin isotopics are generated using the GEH/GNF lattice physics code TGBLA. TGBLA solves two-dimensional (2D) diffusion equations with diffusion parameters corrected by transport theory to provide system multiplication factors and perform burnup calculations. The in-rack k_{eff} calculations in fuel storage criticality analyses are performed using MCNP. MCNP is a Los Alamos Monte Carlo program for solving the neutron transport equation for a fixed source or eigenvalue problem. MCNP implements a robust geometry representation that can correctly model complex components in two or three dimensions. GEH/GNF has compared MCNP to a large number of critical experiments for validation purposes using ENDF/B-VII.0 nuclear cross-section data. [[

]] This code validation has been accepted by the NRC in Reference 3-1.

The methods and analytical approach employed are consistent with the most current NRC guidance for performing spent fuel pool criticality analyses listed in DSS-ISG-2010-01 (Reference 3-2). The design basis lattice is used to study all credible abnormal configurations, bundle and rack tolerances, depletion parameter sensitivities, and computational and validation related uncertainties.

The well-characterized linear relationship between in-core k_∞ and in-rack k_{eff} is evaluated for each rack. A conservative lattice with a peak, cold in-core k_∞ value at or above the intended storage limit is used in the criticality analyses. The design basis lattice is selected for each rack that represents the highest rack efficiency for the given in-core k_∞ limit based on a study of:

1. [[

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Rack efficiency is the ratio of a particular lattice statepoint in-rack eigenvalue (k_{eff}) to its associated lattice nominal in-core eigenvalue (k_∞). A lower rack efficiency implies increased reactivity suppression capability relative to an alternate lattice design with a higher rack efficiency. This value allows for a straightforward comparison of a rack's criticality response to varying lattice designs.

Compliance of every lattice in every bundle for a given plant with a GE rack is confirmed to be at or below the specified GE rack k_∞ limits as part of the bundle design process for each reload. A separate criticality analysis is performed for each of the rack types described in GESTAR II Section 3.5 and confirms that the k_∞ limits would result in a k_{max} value compliant with 10 CFR 50.68. A criticality analysis is performed for each new GNF fuel product line per GESTAR II Section 1.1.3.G.

The design basis lattices used in the criticality analyses are conservative for several reasons:

1. [[

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3. Third, there is even additional margin in the in-core k_∞ limits themselves. [[

]] while the k_∞ limits for the GE racks are all above 1.28.

References

3-1 Letter, Richard B. Ennis (NRC) to Michael J. Pacilio (Exelon Nuclear), Subject: Peach Bottom Atomic Power Station, Units 2 and 3 – Issuance of Amendments RE: Use of Neutron Absorbing Inserts in Spent Fuel Pool Storage Racks (TAC Nos. ME7538 and ME7539), May 21 2013 (ADAMS Accession Number ML13122A423).

3-2 DSS-ISG-2010-01, Revision 0, Final Division of Safety Systems Interim Staff Guidance, Staff Guidance Regarding the Nuclear Criticality Safety Analysis for Spent Fuel Pools.

RAI-4

Please provide additional details of the sensitivity study performed to determine the effect of bundle power, bundle flow, subcooling, R-factor, and subcooling on critical power ratio for fuel designs as mentioned in Section S.2.2.1.

Response

The results of the sensitivity study referenced in Section S.2.2.1 are illustrated in Table S-1. The sensitivity of the critical power ratio (CPR) to a variety of parameters associated with the GEXL methodology was developed by exercising the GEXL01 correlation, applicable to GE 7×7 and early 8×8 fuels in 1977. Subsequently, only values applicable to the early 8x8 fuels were retained in Table S-1 of GESTAR II. The process to calculate the effect of thermal-hydraulic parameters, such as bundle power, bundle flow, subcooling, R-factor, and pressure CPR is described in the following paragraphs.

An approximate nominal value of each thermal hydraulic parameter was estimated and the reference values for the range of each parameter were used for the sensitivities summarized in RAI Table 4-1. The sensitivity was calculated from the following equation:

$$\text{Sensitivity} = \frac{\frac{(CPR_{PER} - CPR_{REF})}{CPR_{REF}}}{\frac{(X_{PER} - X_{REF})}{X_{REF}}}$$

where:

CPR = (Critical power calculated by GEXL) / (Bundle Power)

X = input parameter such as bundle power, bundle flow, subcooling, R-factor, and pressure

REF = reference condition

PER = perturbed condition

In RAI Table 4-1, $\Delta CPR = CPR_{PER} - CPR_{REF}$, and $\Delta Parameter = X_{PER} - X_{REF}$.

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For illustration, these sensitivities have been determined for other GEXL correlations (GEXL02, 07, 14, and 17 correlations) developed for newer fuel product lines. The same nominal conditions and power shape, which were used in the original Table S-1 of GESTAR II, have been maintained. The sensitivity results are presented in RAI Table 4-1. These results illustrate that [[

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RAI Table 4-1
Sensitivity Analysis of CPR to Various Thermal Hydraulic Parameters by Various GEXL Correlations

Parameters	Unit	Nominal Parameter		(ΔCPR ÷ Nominal CPR) ÷ (ΔParameter ÷ Nominal Parameter)					
		Shown in Table S-1	Medium Value	Shown in Table S-1	GEXL01 (8×8)	GEXL02 (8×8)	GEXL07 (GE11)	GEXL14 (GE14)	GEXL17 (GNF2)
Bundle Power	MWth	[[							
Coolant Flow	Mlbm / hr-ft^2								
Inlet Subcooling	BTU / lbm								
R-factor	-								
Core Pressure	psia								]]

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