ENCLOSURE 2

M190077

Response to Requests for Additional Information

Non-Proprietary Information

IMPORTANT NOTICE

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

NRC RAI #1

The primary mitigation mechanism for the CRDA event is the Doppler reactivity effect as the fuel temperature increases. This effect is determined by [[

]]. More information is needed to fully understand [[

]]:

a. Provide a breakdown of the data from Figure 4-3 [[

b.

]].

GNF Response

As currently implemented for most other processes, [[

]] These lattice definitions in order of increasing height in the bundle are listed [[

]] in Table 1. [[

]]

The range of lattice enrichments, range of pin enrichments, range of gadolinium rod numbers, and range of gadolinium enrichments are presented for all lattices in Table 2. This significant range of parameters shows that [[ ]], while changes to gadolinium and enrichment are expected to have minimal impact.

[[

1.

2.

]]

The text shown below, [[ ]], is added to the end of Section 4.1.1.5 of the submitted LTR (Reference 1).

[[

]]

References

1. NEDE-33885P, “GNF CRDA Application Methodology,” Revision 0, February 2018.

Table 1: Lattice Types of Consideration [[ ]]

[[	
	]]

Table 2: Variation in Lattice Parameters [[ ]]

[[	
	]]

[[

]]

Figure 1: [[ ]]

[[

]]

Figure 2: [[ ]]

[[

]]

Figure 3: [[ ]]

[[

]]

Figure 4: Doppler Coefficient Comparison [[
]]

NRC RAI #2

The TR indicates [[

]]. Figure 4-2 in the TR shows a peak pin enthalpy distribution that suggests that some of the fuel assemblies [[

]] may have peak enthalpy rises that approach that experienced by the fuel assemblies located next to the control rod. This may be especially true for core configurations where control rods have been withdrawn such that there are additional neutronic coupling considerations. Provide justification [[

]].

GNF Response

[[

]]

The methodology just described is a deliberate transformation [[

]]

It is true that neutronic coupling needs to be taken into consideration. This is achieved by means of [[

]]

A secondary consideration [[

]]

NRC RAI #3

[[
]] Provide an extended discussion regarding the extent [[
]] are validated as acceptable [[
]]. Discuss how the area of applicability [[

]].

GNF Response

There are [[ ]] parameters that define the area of applicability [[
]]. Table 1 lists these parameters and describes how they are defined.

Table 1: GNF CRDA [[ ]] **Applicability Checklist**

[[	
	]]

The applicability [[

]] This checklist appears in Section 4.2.2 of the TR along with instructions for its use.

Discussion on Core Design and Operating Strategy

The CRDA evaluation that is performed [[ ]] accounts for variation in both core design and operating strategy by:

(1) [[

]]

(2) [[

]]

Furthermore, a plant's core design and operating strategy is constrained by the fuel product line definition [[ ]], as well as by the plant's Technical Specifications pertaining to reactivity control and power distribution limits.

[[

]]

Example Scenario [[

]]

The TR is modified as follows:

4.2.2 [[ ]] **Applicability Checklist**

[[

]]

Table 4-5: GNF CRDA [[ ]] **Applicability Checklist**

[[		

[[		
		]]

References:

1. NEDE-33885P, "GNF CRDA Application Methodology," Revision 0, February 2018.

NRC RAI #4

Section 4.3.4 of the TR describes several options [[

]] Describe why this requirement would be sufficient [[

]].

GNF Response

GNF has opted to [[

]] without introducing any new safety concerns.

[[

]]

The TR is modified as follows.

[[

]]

NRC RAI 5

The guidance provided in the TR does not discuss any specific constraints or recommendations that may be necessary for the time step size or explain why the recommended inputs from prior NRC approved applications of TRACG are acceptable for the CRDA event. The CRDA event is a very rapid prompt power excursion that occurs on a very short time scale, so the predicted neutronic response due to heatup of the fuel and moderator may be sensitive to the time step size. Provide a discussion regarding what time step inputs are to be used for the CRDA analysis, and why these inputs are acceptable for this intended application.

GNF Response

TRACG automatically determines the time step size in an attempt to maximize the accuracy of the calculation and minimize the computer time. As described in Section 8.2.4 of Reference [1], two basic criteria are used for this purpose: convergence and Rate-Of-Change (ROC).

Solutions to the thermal-hydraulic equations are required to converge within a prescribed set of convergence criteria. The calculations are aborted if convergence cannot be achieved by decreasing the time step size. If convergence is obtained within a low number of iterations, the time step size is allowed to increase provided the ROC criteria regulating temporal discretization are also satisfied. Conversely if many iterations are required, the time step size is reduced. TRACG will reduce the time step size before failure of the outer iterations. This allows the time step controller to maximize the time step size and computational efficiency while preventing excessive backtracking because of failures to converge in the hydraulics solution.

TRACG examines the ROC for the primary dependent variables in all cells and all nodes of all components. If the maximum ROC is low a quasi-steady-state condition exists, and the time step size increases. Conversely if the ROC is high, the time step size is reduced. The ROC criteria serve to prevent excessive changes in each of the dependent variables during a time step and thereby prevent excessive temporal discretization errors. The ROC dependent variables are: (1) total pressure, (2) void fraction, (3) gas temperature, (4) liquid temperature, (5) total non-condensable gas pressure ("air"), (6) vessel slab temperature, and (7) fuel rod temperature. The eighth ROC ratio is a complex quantity from the 3D kinetics model [[

]]. The time step size is automatically reduced to accommodate any rapid ROCs [[ ]] as they occur in the control rod drop simulation(s). Because of this automatic feature, the calculated results are not sensitive to the minimum and maximum limits that are input to the code.

Two examples are provided using information taken from Section 5.1.3 of Reference [2] [[
]]. Time step information is extracted from the TRACG cases that make up the Table 5-3 results. The minimum allowable time step size used for the TRACG cases is 1.0E-08 seconds and the maximum allowable time step size is 5.0E-03 seconds. The purpose of the maximum allowable value is to prevent the code from running at a pace that will not allow it to *see* the changing responses as they begin. This maximum limit comes into play primarily prior to the reactivity insertion and after the reactivity induced pulse has begun to settle back into a new quasi-steady condition.

Figure 1 presents the minimum and average time step size for each case compared to the minimum and maximum allowable time step size. The minimum time step size actually used was 8.07E-06 seconds which is roughly two orders of magnitude from the 1.0E-08 second minimum allowable time step size. The average time step size is approximately 3.76E-03 seconds. This example demonstrates that the specified lower limit on allowable time step size is sufficiently low that its value will not affect the results.

Figure 2 presents the individual time step sizes for the case with the highest overall ratio to either the pellet-cladding mechanical interaction or the high temperature cladding failure acceptance criteria in Table 5-3. [[

]] The minimum time step size actually used was 3.46E-05 seconds and the average time step size is approximately 2.40E-03 seconds. This example also demonstrates that the specified lower limit on allowable time step size is sufficiently low that its value will not affect the results.

References

[1] GE Hitachi Nuclear Energy, "TRACG Model Description," NEDE-32176P, Revision 4, January 2008.

[2] Global Nuclear Fuel, "GNF CRDA Application Methodology," NEDE-33885P, Revision 0, February 2018.

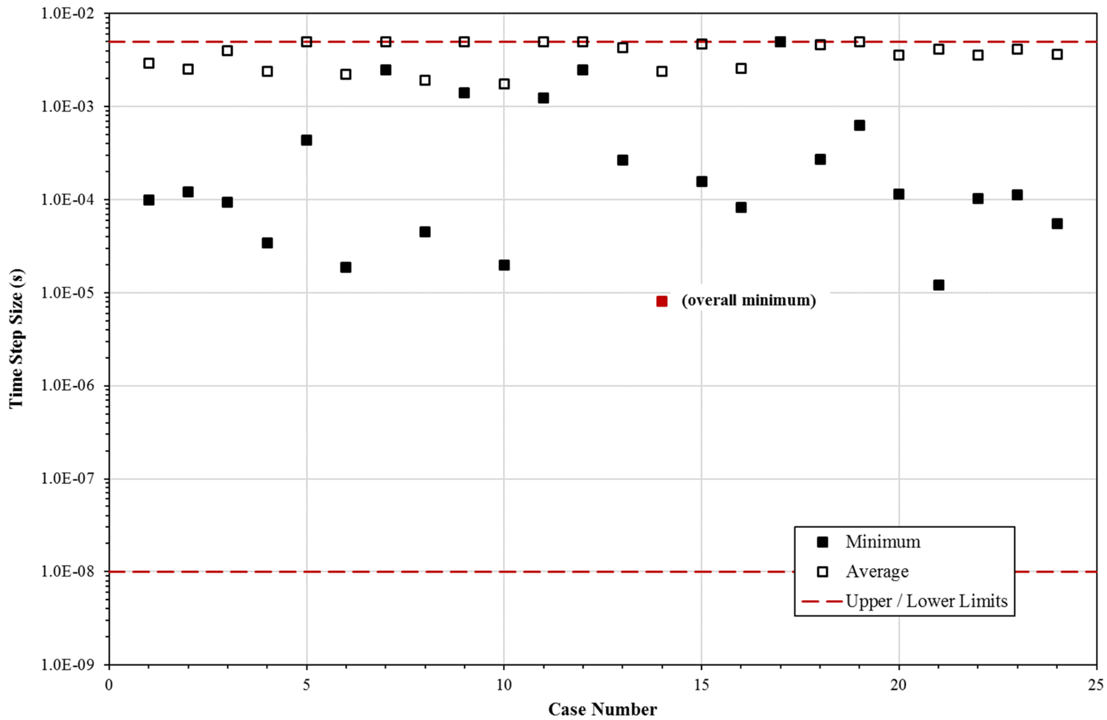

Figure 1: Time Step Sizes for Table 5-3 Cases

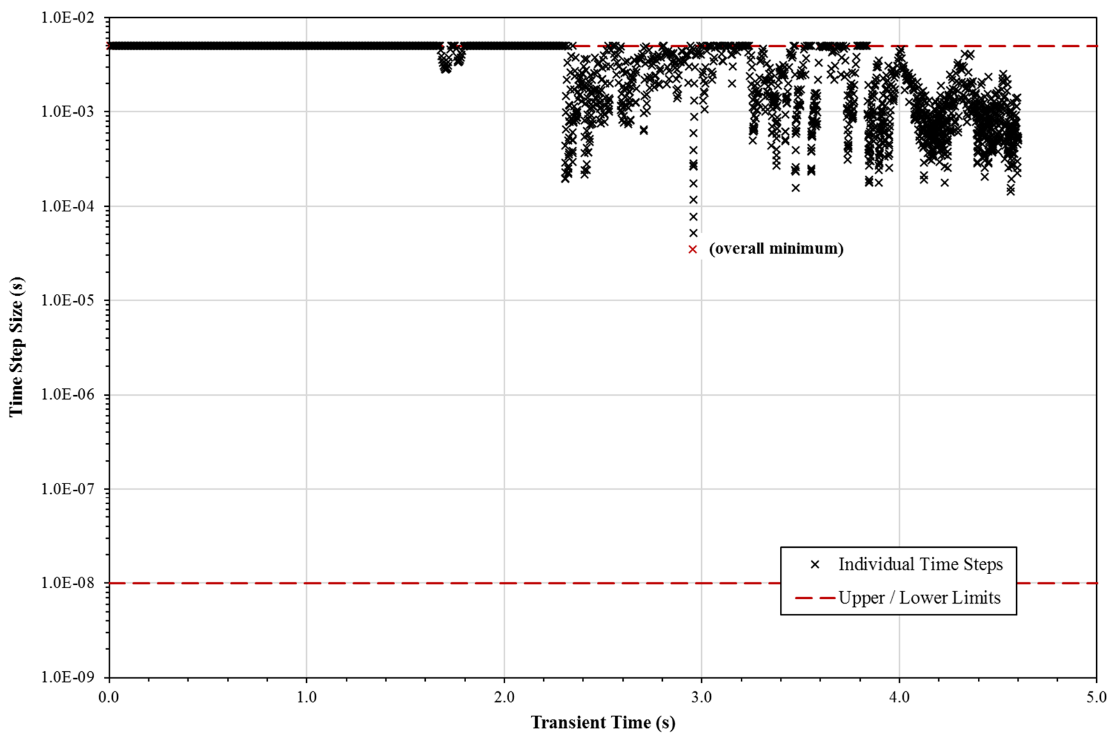

Figure 2: Time Step Sizes for Table 5-3 Limiting Case

NRC RAI #6

The guidance provided in the TR indicates that the control rods are to be modeled [[ ]] provide justification [[ ]].

GNF Response

When performing a calculation that models all the control rods [[ ]]. This means that for any given sequence [[ ]].

The following text is added to Section 4.3.3 of the TR.

When performing a calculation that models all the control rods [[ ]]. This means that for any given sequence [[ ]].

NRC RAI-7

A significant characteristic of the expected limiting CRDA event is that it involves a prompt critical excursion. In addition to the worth of the dropped control rod, another critical quantity to define the magnitude of the prompt critical excursion is the delayed neutron fraction. Provide a discussion of the uncertainty in the delayed neutron fraction as calculated by the neutronics model in PANACEA/TRACG, and explain how it is accommodated by the proposed CRDA analysis methodology.

GNF Response

The total delayed neutron fraction (β) is calculated by PANACEA [[

]]

Values of β depend primarily on exposure because of how the proportions of fissionable isotopes change with exposure. Initial lattice U235 enrichment also plays a less-significant roll. In addition to these two effects, the lattice calculations account for the changing neutron spectrum with exposure via the dependencies listed above. The capturing of the key elements influencing β together with the SPERT qualification and all the AOO transient qualification for TRACG using the PANAC 3D neutron kinetics supports the conclusion that there is no bias [[

]].

Uncertainty in β values are documented in Tables 1-13, 1-14 and 1-16 of Reference [7-1]. Text at the end of Section 1.4.1 of Reference [7-1] states the "more useful β values are given in Table 1-16" so that is the information used in this response. The first three rows of Table 7-3 of this response show the inputs that were used. Fission fractions for the key U235, U238, and Pu239 isotopes shown in columns 2, 3, and 4 were estimated using the correlations documented in Equations (9.3-14) through (9.3-16) of the TRACG Model Description Licensing Topical Report (LTR) (Reference [7-2]). Corresponding calculated weighted β values are shown in column 5. The 1-σ uncertainty values in column 6 were calculated as $\sigma = \sqrt{\sum_i w_i \sigma_i^2}$ where w_i is the fission fraction weighting fraction as given in columns 2, 3, and 4 of Table 7-3. The rightmost column (7) shows how the fractional uncertainty in the total value of β increases with exposure. This increase is caused by the increasing contribution of Pu239 for which the uncertainty fraction σ(β)/β is larger than for U235 and U238. Note that the weighted composite values for σ(β)/β can be larger than the values for individual isotopes because of how the weighted variances were used to calculate the composite value of σ(β).

Table 7-3 Determination of Uncertainty in Total [[ ]] β

[[

]]

The largest value of $\sigma(\beta)/\beta$=0.120 in the lower right corner of Table 7-3 will conservatively bound the uncertainty in total [[ ]] β regardless of [[ ]] exposures. Control Rod Drop Accident (CRDA) evaluations have demonstrated that the responses for fuel enthalpy rise, fuel absolute enthalpy, and peak cladding temperature (PCT) are [[ ]].

The TRACG code was used to evaluate the impacts of a 1-σ uncertainty of 12% in [[ ]] total β. These impacts were assessed for a representative CRDA calculation and include the impacts on calculated values for the reactor power, fuel rod enthalpies for the peak node of the peak rod, and PCT. [[ ]]

Reactor total powers [[ ]] are shown in Figure 7-5. [[ ]] When β is reduced the power pulse occurs slightly earlier, rises faster, and reaches a higher peak value. The key observation is that a larger power pulse associated with a reduced β also has a narrower pulse width so that the total energy of the pulse is essentially the same. This observation is consistent with what was observed for the SPERT qualification cases.

[[

]]

Figure 7-5 Total Reactor Power

The calculated fuel enthalpies for the peak rod in the peak node are shown in Figure 7-6. [[

]] The increase [[
]] is judged to be negligible [[
]].

[[

]]

Figure 7-6 Peak Hot Rod Enthalpy for Limiting Channel, Rod Group, Heated Node

The calculated PCT values are plotted in Figure 7-7. [[

]] This difference is judged to be negligible [[

]].

[[

]]

Figure 7-7 Peak Clad Temperature (from all rods in the simulated core)

The potential impacts of uncertainty in total β have been quantified. These impacts were assessed for a representative CRDA calculation and include the impacts on calculated values for the total reactor power, fuel rod enthalpy for the peak axial node of the peak rod, and PCT for all locations of all rods in the core. The insignificant impacts on these quantities supports the conclusion that the CRDA methodology described in the LTR to calculate the margin to the CRDA acceptance criteria that depend on these quantities is reasonably accurate [[ ]].

References

[7-1] Reactor Physics Constants, ANL-5800, Argonne National Laboratory, July 1963.

[7-2] TRACG Model Description LTR, NEDE-32176P, Revision 4, January 2008.

NRC RAI 8

The enthalpy for the limiting rod in each fuel assembly of interest is computed [[

]] enthalpies used to compare against the acceptance criteria are sensitive to uncertainties in the calculated rod and assembly powers [[ ]]. Provide a discussion of the uncertainty in the rod and assembly power distribution as calculated [[

]].

GNF Response

Reference [1] established the nuclear modeling uncertainties in the Safety Limit Minimum Critical Power Ratio (SLMCPR) and Linear Heat Generation Rate (LHGR) calculations. The pin power uncertainties total [[ ]] and the overall bundle power uncertainty is [[ ]]. Consequently, the SLMCPR and LHGR calculations incorporate a [[ ]] uncertainty. [[

]] This uncertainty is [[

]].

As described in Sections 3.1.3.5 and 3.1.4.3 of Reference [2], [[

]]. These thermal-mechanical limits are developed with [[

]] uncertainty [[ ]], see Section 8.6 of Reference [3]. [[

]] Thus, the uncertainty [[

]]

References

[1] GE Nuclear Energy, “Power Distribution Uncertainties for Safety Limit MCPR Evaluations,” NEDC–32694P-A, Revision 0, August 1999.

[2] Global Nuclear Fuel, “GNF CRDA Application Methodology,” NEDE-33885P, Revision 0, February 2018.

[3] Global Nuclear Fuel, “The PRIME Model for Analysis of Fuel Rod Thermal – Mechanical Performance Part 1 – Technical Bases,” NEDC-33256P-A Revision 1, September 2010.