August 22, 2016

MEMORANDUM TO: Kevin Hsueh, Chief
Licensing Processes Branch
Division of Policy and Rulemaking
Office of Nuclear Reactor Regulation

FROM: Joseph A. Golla, Project Manager ***/RA/***
Licensing Processes Branch
Division of Policy and Rulemaking
Office of Nuclear Reactor Regulation

SUBJECT: GENERAL ELECTRIC-HITACHI NUCLEAR ENERGY – TRACG APPLICATION FOR EMERGENCY CORE COOLING SYSTEMS/LOSS-OF-COOLANT-ACCIDENT ANALYSES FOR BWR/2-6 – NEDE-33005P – NUCLEAR PERFORMANCE AND CODE REVIEW BRANCH AUDIT REPORT (TAC NO. ME5405)

By letter dated January 27, 2011, General Electric (GE)-Hitachi Nuclear Energy (GEH) submitted Topical Report NEDE-33005P, Revision 0, “TRACG Application for Emergency Core Cooling Systems/Loss-of-Coolant-Accident Analyses for BWR/2-6,” for U.S. Nuclear Regulatory Commission (NRC) staff review.

The NRC staff technical review has required several rounds of correspondence with GEH. In its review, the NRC staff transmitted a request for additional information (RAI) by letter dated October 19, 2012. By letter dated October 7, 2014, GEH provided its response. The NRC staff transmitted a second round RAI by letter dated September 15, 2015, to which GEH responded by letter dated February 19, 2016. The NRC staff transmitted a third round RAI to GEH by letter dated March 24, 2016, to which GEH has not yet responded.

The NRC staff conducted an audit, following Office of Nuclear Reactor Regulation Office Instruction LIC-111, “Regulatory Audits.” The audit was conducted Wednesday and Thursday, April 27 and 28, 2016, at GEH offices in Washington, DC. A non-proprietary version of the NRC staff Regulatory Audit Report is enclosed.

Project No. 710

Enclosure:
As stated

CONTACT: Benjamin T. Parks, NRR/DSS
(301) 415-0979

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DATE	08/15/2016	08/15/2016	08/18/2016	08/22/2016

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REGULATORY AUDIT REPORT

NUCLEAR REGULATORY COMMISSION STAFF REVIEW OF NEDE-33005P

"TRACG APPLICATION FOR EMERGENCY CORE COOLING SYSTEM/

LOSS-OF-COOLANT-ACCIDENT ANALYSES FOR BWR/2-6"

TAC NO. ME5405

1.0 BACKGROUND

By letter dated January 27, 2011, General Electric (GE)-Hitachi Nuclear Energy (GEH) submitted Topical Report NEDE-33005P, Revision 0, "TRACG Application for Emergency Core Cooling Systems/Loss-of-Coolant-Accident Analyses for BWR/2-6,"[1,2] for U.S. Nuclear Regulatory Commission (NRC) staff review.

The NRC staff technical review has required ongoing correspondence with GEH. In its review, the NRC staff transmitted a request for additional information (RAI) by letter dated October 19, 2012.[3] By letter dated October 7, 2014, GEH provided its response.[4] The NRC staff transmitted a second round RAI by letter dated September 15, 2015, to which GEH responded by letter dated February 19, 2016.[5,6] The NRC staff transmitted a third round RAI to GEH by letter dated March 24, 2016, to which GEH has not yet responded.[7]

2.0 REGULATORY AUDIT BASES

The TRACG loss-of-coolant accident (LOCA) evaluation model (TRACG-LOCA) was developed in accordance with the regulatory requirements established in Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50, "Domestic Licensing of Production and Utilization Facilities," Section 46, "Acceptance criteria for emergency core cooling systems for light-water nuclear power reactors" (10 CFR 50.46). In developing TRACG-LOCA, GEH considered guidance contained in two NRC Regulatory Guides (RGs). These include: (1) RG 1.157, "Best-Estimate Calculations of Emergency Core Cooling System Performance," and (2) RG 1.203, "Transient and Accident Analysis Methods."[8,9]

The NRC staff is reviewing NEDE-33005P to determine whether TRACG-LOCA is an acceptable evaluation model as set forth in 10 CFR 50.46. In its review, the NRC staff relies on the regulatory guidance described above, as well as applicable chapters contained in NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants, LWR [light-water reactor] Edition." These chapters include Chapter 6.3, "Emergency Core Cooling System," Chapter 15.0.2, "Review of Transient and Accident Analysis Methods," and Chapter 15.6.5, "Loss-of-Coolant-Accidents Resulting From Spectrum of Postulated Piping Breaks Within the Reactor Coolant Pressure Boundary."[10,11,12]

3.0 REGULATORY AUDIT SCOPE/OBJECTIVES

The NRC staff regulatory audit scope focused on specific open items associated with NEDE-33005. The goal of the audit was to obtain clarification on the open items, and to identify what additional information, if any, would be submitted to complete the review.

3.1. MISCELLANEOUS OPEN ITEMS

3.1.1. Draft Responses to RAIs 99 and 100

GEH continues to draft responses to RAIs 99 and 100, regarding the overall combination of uncertainty and the use of the Cathcart-Pawel correlation to obtain a result for maximum local cladding oxidation.

During the audit, GEH and the staff discussed these RAIs. GEH made draft responses available for audit.

The topic discussed in RAI 99 relates to the statistical meaning of the results. In the draft response, GEH clarified that [

] Nonetheless, using a biased break size introduces more conservatism than an approach whereby the break size is randomly sampled.

The topic discussed in RAI 100 relates to the acceptance criterion promulgated at 10 CFR 50.46(b)(2). Specifically, the RAI seeks clarification regarding the 17-percent acceptance criterion and its basis using the Baker-Just oxidation kinetics correlation. GEH discussed a draft response to the RAI, which outlined several approaches to adjusting the acceptance criterion based on the selected cladding oxidation kinetics equation. GEH also provided sample results indicating the effect that use of the Cathcart-Pawel correlation, with an analogous 13-percent acceptance limit, would have on an emergency core cooling system (ECCS) evaluation.

3.1.2. Revision to RAI 33 Response

The letter transmitting the round 2 RAI response dated February 19, 2016, indicated that a revision is planned for the response to RAI 33, which addresses the model and uncertainty for cladding strain and perforation. The NRC staff audited the data and related material supporting this planned RAI response.

3.1.3. Burnup Limitation

The response to RAI 97, which pertains to fuel rod swelling, rupture, and relocation, appears to be constrained by data that do not exceed current licensed burnup limitations on fuel. This may be the case with additional models or inputs, such as the fuel rod thermal-mechanical performance data that are obtained from the PRIME code. The NRC staff discussed potential limitations on the TRACG-LOCA methodology that may result from the use of burnup-dependent databases.

3.1.4. Licensing Approaches for Expanded Operating Domains

Generally, an NRC staff approval of NEDE-33005P would be accompanied by an amendment to GESTAR-II, which would permit the application of TRACG-LOCA on a plant-specific basis without a need to request a licensing action from the NRC. The NRC staff anticipates, however, that TRACG-LOCA analyses may be used to support licensing requests, such as those associated with expanded operating domains like Maximum Extended Load Line Limit Analysis Plus (MELLLA+) and extended power uprate (EPU). The NRC staff audited the expected content of portions of such applications that would be supported by TRACG-LOCA analyses, relative to applicable conditions and limitations, which apply to requests for operating domain expansions.

In its audit, the NRC staff identified and discussed several licensing topical reports whose limitations, conditions, and dispositions will require revision or update in order to accommodate the use of TRACG-LOCA. The limitations and conditions apply to the following GEH licensing topical reports (LTRs):

Extended Power Uprate (EPU)

GEH maintains a suite of licensing topical reports that are referenced by licensees in requests to implement EPUs. These include:

- NEDC-32424P-A, “Generic Guidelines for General Electric Boiling Water Reactor Extended Power Uprates,” commonly referred to as EPU Licensing Topical Report (ELTR) 1,[13]
- NEDC-32523P-A, “Generic Evaluations for General Electric Boiling Water Reactor Extended Power Uprates,” commonly referred to as ELTR 2, [14] and
- NEDE-33004P-A, “Licensing Topical Report: Constant Pressure Power Uprate,” commonly referred to as the Constant Pressure Power Uprate (CPPU) LTR, or CLTR.[15]

ELTR 1

The NRC staff safety evaluation (SE) approving ELTR 1 (Reference 13) noted the following with regard to ECCS performance:

1. Reanalysis of the LOCA response utilizing the SAFER/GESTR computer model will improve the apparent safety margins, especially the margin between the calculated peak cladding temperature (PCT) and the 2200-degree Fahrenheit limit contained in 10 CFR 50.46. Information contained in ELTR 1 implies that this margin can be used in the relaxation of inputs to the ECCS codes, which corresponds to changes in operational requirements of ECCS equipment. It is the staff’s position that this “additional” margin must be used judiciously. Therefore, the staff will not support changes that would relax equipment requirements, such as emergency diesel generator start times, pump flow requirements, and so on, as part of an amendment request for an EPU. The staff suggests that the relaxation of these parameters might be pursued as a boiling-water reactor (BWR) Owners Group initiative, but in any case must be treated separately from the generic BWR power uprate program.

2. If a licensee updates a plant from a previous ECCS computer code to the SAFER/GESTR code, a baseline run using SAFER/GESTR at the present power level must be included so that the true effect of the power uprate can be assessed. Inclusion of SAFER/GESTR results obtained using relaxed input parameters (as previously discussed) may be included in the power uprate amendment request, but must be accompanied by corresponding results obtained by using the present input parameters and uprated power.

Appendix D to ELTR 1 provides the basis for EPU ECCS evaluations performed in accordance with this LTR. While the content of Appendix D is generally based on the use of SAFER methods, it is more broadly constrained by a requirement to use an NRC-approved evaluation model, and a requirement to evaluate the entire break spectrum and limiting single failure for uprated operating conditions.

ELTR 2

The NRC staff SE of ELTR 2 (Reference 14) states the following, with regard to design basis accidents:

> Plant-specific analyses will continue to demonstrate the ability of each plant to cope with the full spectrum of hypothetical pipe break sizes from breaks as small as instrument lines to breaks in the largest recirculation, steam, feedwater and ECCS lines… Challenges to the fuel and containment, as well as potential radiological releases to the environment, will be assessed on a plant-specific basis using NRC-approved methods.

The evaluations presented in ELTR 2, including those discussed in ELTR 2, Supplement 1, Volume 1, conclude that the ECCS equipment performs acceptably relative to EPU operation, given acceptable results from an NRC-approved ECCS evaluation model like SAFER/GESTR-LOCA.

CLTR

The CLTR builds on the guidelines and evaluations provided in ELTR 1 and 2, and on the experience from EPU applications that had been reviewed and approved between the approval of ELTR 1 and 2, and the approval of the CLTR in 2003. Based on this experience, the NRC staff approved a limited set of analyses that would be used to support conclusions about ECCS performance at EPU plants.

Key considerations included that: (1) the implementation of an EPU tended not to affect the LOCA break spectrum and limiting single failure, and (2) the implementation of an EPU tended to increase the predicted PCT on the order of 20 °F or less. Based on these considerations, the NRC staff approved a disposition requiring the evaluation of the limiting break and a sub-set of smaller break sizes for the ECCS performance evaluation at uprated conditions.

The CLTR is also limited, to some extent, to GE14 fuel designs. The NRC understands that some recent NRC licensees using more advanced fuel designs have used a hybrid of CLTR and ELTR 1/ELTR 2 dispositions. The TRACG-LOCA evaluation model specifically delineates its applicability to the EPU operating domains, such that an EPU application could be supported by a TRACG-LOCA break spectrum ECCS evaluation.

Interim Methods Licensing Topical Report (IMLTR)

Some plants incorporate NEDC-33173P, "Applicability of GE Methods to Expanded Operating Domains,"[16] or the so-called Interim Methods Licensing Topical Report (IMLTR) in the licensing bases. The IMLTR imposes limitations based on the database against which GEH methods are qualified. Several of the limitations, identified below, warrant additional discussion within the framework of TRACG-LOCA.

Power-to-Flow Ratio

The IMLTR SE (Reference 16) imposes a limitation on the power-to-flow operating domain, as follows:

> Plant-specific EPU and expanded operating domain applications will confirm that the core thermal power to core flow ratio will not exceed 50 MWt/Mlbm/hr at any statepoint in the allowed operating domain. For plants that exceed the power-to-flow value of 50 MWt/Mlbm/hr, the application will provide power distribution assessment to establish that neutronic methods axial and nodal power distribution uncertainties have not increased.

Relative to the above, the NRC staff review of TRACG-LOCA is being performed with consideration to its applicability within the MELLLA+ operating domain, such that this core thermal power-to-core flow ratio constraint would apply to the TRACG-LOCA approval.

ECCS Evaluations

> For applications requesting implementation of EPU or expanded operating domains, including MELLLA+, the small and large break ECCS-LOCA analyses will included top-peaked and mid-peaked power shape in establishing the MAPLHGR [maximum average planar linear heat-generation rate] and determining the PCT. This limitation is applicable to both the licensing bases PCT and the upper bound PCT. The plant-specific applications will report the limiting small and large break licensing basis and upper bound PCTs.
>
> The ECCS-LOCA will be performed for all statepoints in the upper boundary of the expanded operating domain, including the minimum core flow statepoints, the transition statepoint as defined in [NEDE-33006P-A (Reference 17)] and the 55 percent core flow statepoint. The plant-specific application will report the limiting ECCS-LOCA results as well as the rated power and flow results. The SRLR [supplemental reload licensing report] will include both the limiting statepoint ECCS-LOCA results and the rated conditions ECCS-LOCA results.

The limitations identified above would remain applicable, once the NRC review of TRACG-LOCA is complete. As appropriate, and drawing on Section 9.9 of NEDE-33005, the TRACG-LOCA SE will offer the staff interpretation regarding compliance with these limitations, considering appropriate initial conditions and operating parameters for the TRACG-LOCA evaluation model and related results. For example, the upper bound PCT is a result specific to the SAFER/GESTR-LOCA evaluation model and, presumably, the TRACG-LOCA equivalent would be an upper tolerance limit.

<u>MELLLA+</u>

Several limitations related to concurrent changes, set forth in Section 12.4 of the staff SE approving MELLLA+, require ECCS performance evaluation. TRACG-LOCA is an acceptable method for performing such evaluations, provided that the conditions being evaluated remain within the TRACG-LOCA qualification range.

Several limitations are specific to ECCS-LOCA evaluations, specifically those discussed in Sections 12.10 through 12.14 of the staff SE approving MELLLA+. These include limitations related to the ECCS-LOCA off-rated multiplier (Section 12.10), the axial power distributions required for analysis in the MELLLA+ operating domain (Section 12.11), reporting of ECCS performance evaluation results (Section 12.12), requirements for small break LOCA analysis (Section 12.13), and requirements for break spectrum analysis (Section 12.14).

<u>ECCS-LOCA Off-Rated Multiplier Limitations</u>

a) The plant-specific application will provide the 10 CFR Part 50, Appendix K, and the nominal PCTs calculated at the rated EPU power/rated CF [core flow], rated EPU power/minimum CF, at the low-flow MELLLA+ boundary (Transition Statepoint). For the limiting statepoint, both the upper bound and the licensing PCT will be reported. The M+SAR[*] will justify why the transition statepoint ECCS-LOCA response bounds the 55 percent CF statepoint. The M+SAR will provide discussion on what power/flow combination scoping calculations were performed to identify the limiting statepoints in terms of DBA[design-basis accident]-LOCA PCT response for the operation within the MELLLA+ boundary. The M+SAR will justify that the upper bound and licensing basis PCT provided is in fact the limiting PCT considering uncertainty applications to the non-limiting statepoints.

b) LOCA analysis is not performed on cycle-specific basis; therefore, the thermal limits applied in the M+SAR LOCA analysis for the 55 percent CF MELLLA+ statepoint and/or the transition statepoint must be either bounding or consistent with cycle-specific off-rated limits. The COLR [core operating limit report] and the SRLR will contain confirmation that the off-rated limits assumed in the ECCS-LOCA analyses bound the cycle-specific off-rated limits calculated for the MELLLA+ operation. Every future cycle reload shall confirm that the cycle-specific off-rated thermal limits applied at the 55 percent CF and/or the transition

[*] The M+SAR, or MELLLA+ Safety Analysis Report, is typically enclosed in a license amendment request seeking approval to implement the MELLLA+ operating domain. It provides all the supporting safety analyses, including the ECCS-LOCA performance evaluation.

statepoints are consistent with those assumed in the plant-specific ECCS-LOCA analyses.

c) Off-rated limits will not be applied to the minimum CF statepoint.

d) If credit is taken for these off-rated limits, the plant will be required to apply these limits during core monitoring.

ECCS-LOCA Axial Power Distribution Evaluation

For MELLLA+ applications, the small and large break ECCS-LOCA analyses will include top peaked and mid-peaked power shape in establishing the MAPLHGR and determining the PCT. This limitation is applicable to both the licensing bases PCT and the upper bound PCT. The plant-specific applications will report the limiting small and large break licensing basis and upper bound PCTs.

ECCS-LOCA Reporting

a) Both the nominal and Appendix K PCTs should be reported for all of the calculated statepoints, and

b) The plant-variable and uncertainties currently applied will be used, unless the NRC staff specifically approves a different plant variable uncertainty method for application to the non-rated statepoints.

Small-Break LOCA [Analysis]

Small-break LOCA analysis will be performed at the MELLLA+ minimum CF and the transition statepoints for those plants that: (1) are small-break LOCA limited based on small-break LOCA analysis performed at the rated EPU conditions; or (2) have margins of less than or equal to **[ ]** relative to the Appendix K or the licensing basis PCT.

[LOCA] Break Spectrum

The scope of small-break LOCA analysis for MELLLA+ operation relies upon the EPU small-break LOCA analysis results. Therefore, the NRC staff concludes that for plants that will implement MELLLA+, sufficient small break sizes should be analyzed at the rated EPU power level to ensure that the peak PCT break size is identified.

Similar to those associated with IMLTR, these conditions and limitations will require an appropriate interpretation within the TRACG-LOCA analytic framework, relative to: (1) TRACG-LOCA initial conditions, operating parameters, and analytic output, and (2) the discussion contained in Section 9.9 of the LTR.

<u>Conclusion</u>

These conditions and limitations were discussed with GEH during the audit. No additional information is required or requested from the vendor in order to complete the staff SE.

3.2. <u>DECAY HEAT</u>

Regulatory Position 3.2, "Sources of Heat During a Loss-of-Coolant Accident," of RG 1.157 provides guidance regarding acceptable decay heat models.

The decay heat model included in TRACG04 is enhanced over what was previously provided in TRACG02. Whereas the TRACG02 code included the May-Witt decay heat model, TRACG04 adds both the 1979 and 1994 American Nuclear Society (ANS) decay heat models. The SE approving Supplement 3 to NEDE-32906P notes that the 1994 standard is used in anticipated operational occurrence (AOO), anticipated transient without scram (ATWS), and American Society of Mechanical Engineers overpressure analysis; however, NEDE-33005P indicates that the 1979 ANS decay heat model is used for ECCS evaluation. The NRC staff notes that this approach is in line with the guidance contained in RG 1.157.

The vendor notes, in Table 2.5-1, on Page 2-8 of NEDE-33005P, that, "Calculations are made in accordance with the 1979 ANS Standard. Sensitivities to variations in voids, enrichment and operating history are shown in Appendix B of [NEDE-30996P-A]." The TRACG04 decay heat model is also described in Section 9.3 of NEDE-32176P, Revision 4.

In addition, the LTR states, [

] Further explanation is provided on Page 5-31 of NEDE-33005P, which states, in part, that "the decay heat curve becomes a function of the fuel design, depletion environment and power history. In theory, each point in the reactor has a unique decay heat curve. Fortunately, the variations in decay heat due to the above effects are small and a generic curve can be defined to cover all locations with little loss in accuracy."

From the remarks in NEDE-33005P, it is not possible to determine whether different decay heat curves are calculated for each bundle group, as the text in Chapter 2 of the LTR suggests, or whether a generic curve is used for the entire core, as suggested in Chapter 5. In addition, the LTR is not clear as to whether the effects of fuel exposure, depletion power density, irradiation time, fuel enrichment, and void fraction, are explicitly modeled on a bundle-by-bundle basis, or whether the modeling is simplified to a point that permits the use of more generic modeling approaches.

In addition to the above, GEH references NEDE-30996P-A, Appendix B, for discussion regarding the sensitivities to significant operating conditions. The NRC copy of NEDE-30996P-A does not include an Appendix B; however, NEDE-23785-1-PA does.

In addition to these documentation issues, the uncertainty treatment for decay heat also warrants discussion. A succinct and specific discussion of the uncertainty treatment does not appear in the LTR; rather, Section 5.1.3.31 states that the decay heat curve, with $\pm 1\sigma$ curves, is shown for a bundle average exposure of 11 GWd/MTU (per response to RAI 54). The decay heat entry in Table 5.1-2 refers the reader back to the text in Section 5.1.3.31. The LTR does not specifically define the approach used to address decay heat uncertainty.

The uncertainty treatment for decay heat within a realistic model warrants discussion, because the decay heat uncertainties are somewhat different from other phenomenological uncertainties. This is because the process itself occurs, stochastically, in time. Thus, the appropriateness of applying a single uncertainty multiplier for the entire time-dependent curve is questionable, and the use of a multiplier that skews the curve below the nominal value for the entire transient requires additional justification as either a realistic or conservative approach.

Discussion During Audit:

The decay heat model is based on an auxiliary computer program called DECAY-01P. The program was implemented to address concerns identified in NRC Information Notice 1996-39.[18] The computer program is used by SAFER, and is the basis for concurrent TRACG applications, such as for operating plant AOO analysis.

The TRACG AOO application may make use of any of three decay heat models, which include the May-Witt model, which is the default model for TRACG, or either the 1979 or 1994 ANS standard decay heat models. This topic is discussed in Appendix A of the staff SE approving NEDE-32906P-A, Supplement 3-A, “Migration from TRACG04/PANAC11 from TRACG02/PANAC10 for TRACG AOO and ATWS Overpressure Transients,” in the staff evaluation of the response to RAI 23.[19] During the audit, GEH clarified that the 1979 model is most commonly used for TRACG-AOO licensing calculations. By contrast, the TRACG-LOCA LTR specifically states that the 1979 model will be used for consistency with RG 1.157.

During the discussion on decay heat, GEH agreed with the staff observation that the decay heat model discussion that appears in NEDE-33005P would benefit from additional clarification, and agreed to revise the relevant passages in the LTR to provide a more succinct description of the modeling and uncertainty approach. The vendor also stated that the reference to NEDE-30996P-A is incorrect, and that the reference should be to NEDE-23785P-A, Appendix B, instead.

During the audit, the NRC staff also discussed the approach that GEH uses to account for the uncertainty associated with the decay heat model. [

]

GEH intends to submit a more detailed summary discussion of the decay heat model and revise the LTR as described above.

3.3. LICENSING APPLICATION REVIEW

The NRC staff audited portions of an in-process TRACG-LOCA analysis to support a licensing application.

4.0 TEAM ASSIGNMENTS

The audit team consisted of Benjamin T. Parks in the Nuclear Performance and Code Review Branch, Division of Safety Systems, Office of Nuclear Reactor Regulation; and Joseph Golla in the Licensing Processes Branch, Division of Policy and Rulemaking, Office of Nuclear Reactor Regulation.

5.0 LOGISTICS AND SCHEDULE

The audit will took place at GEH facilities in Washington, DC, from Wednesday, April 27, 2016, to Thursday, April 28, 2016.

6.0 REFERENCES

1. Harrison, James F., GE-Hitachi Nuclear Energy Americas (GEH), letter to U.S. Nuclear Regulatory Commission, “NEDE-33005P and NEDO-33005, ‘Licensing Topical Report TRACG Application for Emergency Core Cooling Systems/Loss-of-Coolant-Accident Analyses for BWR/2-6,’” MFN 11-001, Project No. 710, January 27, 2011, Agencywide Documents Access and Management System (ADAMS) Accession No. ML110280323.

2. GEH, “TRACG Application for Emergency Core Cooling Systems/Loss-of-Coolant Accident Analyses for BWR/2-6,” NEDE-33005P, Enclosure 1 to MFN 11-001, Project No. 710, January 27, 2011, ADAMS Accession No. ML110280326 (proprietary). See also non-proprietary NEDO-33005, Enclosure 2 to MFN 11-001, ADAMS Accession No. ML110280325.

3. Philpott, Stephen S., U.S. Nuclear Regulatory Commission, letter to Jerald G. Head, GEH, "Request for Additional Information Re: GE-Hitachi Nuclear Energy Americas Topical Report NEDE-33005P, Revision 0, 'TRACG Application for Emergency Core Cooling Systems/Loss-of-Coolant-Accident Analyses for BWR/2-6' (TAC No. ME5405)," Project No. 710, October 19, 2012, ADAMS Accession No. ML12242A571. See also proprietary Enclosure 2, ML12242A572.

4. Harrison, James F., GEH, letter to U.S. Nuclear Regulatory Commission, "Response to Request for Additional Information Re: GE-Hitachi Nuclear Energy Americas Topical Report (TR) NEDE-33005P, Revision 0, 'TRACG Application for Emergency Core Cooling Systems/Loss-of-Coolant-Accident Analyses for BWR/2-6,' (TAC No. ME5405)," MFN 14-064, Project No. 710, October 7, 2014, ADAMS Accession No. ML14281A018. See also proprietary Enclosure 1, ML14281A016, and non-proprietary Enclosure 2, ML14281A015.

5. Golla, Joseph A., U.S. Nuclear Regulatory Commission, letter to Jerald G. Head, GEH, "Request for Additional Information Regarding Review of Licensing Topical Reports NEDE-33005P and NEDO-33005, "Licensing Topical Report TRACG Application for Emergency Core Cooling Systems/Loss-of-Coolant Accident Analyses for BWR/2-6' (TAC No. ME5405)," Project No. 710, September 15, 2015, ADAMS Accession No. ML15204A582. See also proprietary ML15204A571.

6. Harrison, James F., GEH, letter to U.S. Nuclear Regulatory Commission, "Response to Request for Additional Information Regarding Review of Licensing Topical Reports NEDE-33005P and NEDO-33005, 'Licensing Topical Report TRACG Application for Emergency Core Cooling Systems/Loss-of-Coolant Accident Analyses for BWR/2-6' (TAC No. ME5405)," MFN 16-008, Project No. 710, February 19, 2016, ADAMS Accession No. ML16050A139. See also proprietary Enclosure 1, ML16050A141 and non-proprietary Enclosure 2, ML16050A142.

7. Golla, Joseph A., U.S. Nuclear Regulatory Commission, letter to Jerald G. Head, GEH, "Request for Additional Information Regarding Review of Licensing Topical Report NEDE-33005P and NEDO-33005, 'TRACG Application for Emergency Core Cooling Systems/Loss-of-Coolant Accident Analyses for BWR/2-6' (TAC No. ME5405)," Project No. 710, March 24, 2016, ADAMS Accession No. ML16062A061.

8. U.S. Nuclear Regulatory Commission, Regulatory Guide 1.157, "Best-Estimate Calculations of Emergency Core Cooling System Performance," May 1989, ADAMS Accession No. ML003739584.

9. U.S. Nuclear Regulatory Commission, Regulatory Guide 1.203, "Transient and Accident Analysis Methods," December 2005, ADAMS Accession No. ML053500170.

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