THREE MILE ISLAND ALERT

Submittal of Questions by Mile Island Alert, Inc. in Response to Additional Nuclear Regulatory Commission Questions Pertaining to the Technical Review of the Three Mile Island Unit 2 Independent Spent Fuel Storage Installation License Renewal Application.

Re: Three Mile Island Unit-2 Independent Spent Fuel Storage Installation Application for 10 CFR 72 Specific License Renewal Special Nuclear Material License Number SNM-2508 (Docket-No. 72-20) Prepared for the United States Department of Energy-Idaho Office. Prepared by Orano Federal Services, LLC and Spectra Tech, Inc. Revision 3.

Submitted by, Eric Epstein, Chairman
Three Miles Island Alert, Inc.
4100 Hillsdale Road
Harrisburg, PA 17112

Dated: July 29, 2019.

NMSS26
NMSS

July 29, 2019

Department of Energy
Steven Whanschaffe, License Manager
Idaho Operations Office
1955 Fremont Avenue
Idaho Falls, Idaho 83415

Dear Mr. Whanschaffe:

As outlined in Section 1.21, Department of Energy Idaho Operations Office ("DOE"-"ID") has prepared this License Renewal Application ("LRA") in accordance with applicable requirements in title 1p of the Code of Federal Regulations and the guidance contained in the Nuclear Regulatory Commission ("NRC") Technical Report (NUREG-1927) [1.4.4]. and Nuclear Energy Institute ("NEI") Guidance document (NEI 14-03) [1.4.5]. This application supports license renewal for an additional 20-year period beyond the end of the current license term of the Special Nuclear Materials (SNM) License Number SNM-2508, (Docket No., 72-20) [1.4.1].

Three Mile Island Alert submits the following comments and questions relating to the proposed License Renewal Application. Specifically, TMIA is following up on the "Submittal of Reponses to Additional Nuclear Regulatory Commission Questions Pertaining to the Technical Review of the Three Mile Island Unit 2 Independent Spent Fuel Storage Installation License Renewal and Application." (Enclosure 1: Director, Division of Spent Fuel Management, Office of Nuclear Material Safety and Safeguards, Washington, D.C., 20555-0001, May 21, 2019).

Follow-up Questions 1 and 2:
Re: Additional material: 1-5-2.

1) Please describe the "additional material" that may be stored at the ISFSI. 1.2.2.2. (1)

2) Please provide analyses, data and reports confirming that the site's original design can be expanded by an additional 20 years. 1.2.2.2 (5)

Follow-up Question 3:
Re: INL Site: 1-6-8.

Please provide maps for the "central portion" of the site as well as all relevant effluent plume pathways, geographical, and geological mapping, sediment and soil analyses, etc. (1.2.2.2) (7)

Follow-up Question 4:
Re: Content estimates: 1-12.

Please provide an analysis and descriptions of "the 303,653 pounds of damaged fuel materials shipped from TMI-2 to the INL site; an estimated 7,936 pounds was from the "no-core material" (i.e., core handling debris [1.4.42]". According to a summary of shipments from TMI-2, this non-core material consisted of core baskets and casings (4,260 pounds), drill strings, debris buckets, and diatomaceous earth [1.4.42]. Therefore, the term "as used in the LRA" is considered synonymous with any of the contents of the TMI-2 canisters, including "core handling debris."

Follow-up Question 5: 1-14.
Re: NUHOMS.

Please describe the "adaptation" for the Standardized NUHOMS system. (1.3.1)

Follow-up Question 6: 1-4-17.
Re: Amendment 1.

Please clarify the disposition of the "...incorrect canisters" and "number of fuel canisters." (1-17)

Follow-up Question 7: 1-4-17.
Re: Amendment 1.

DOE is "...not authorized to add additional fuel" as a licensing condition. Where will the additional "core debris" and fuel from TMI-2 be stored? (1-17.)

Follow-up Question 8: 1-4-17.
Re: Amendment 1.

What changes are contained in the Safeguards Contingency Plan? (1-17 to 1-18.)

Follow-up Question 9: 1-19.
Re: Aging implications.

Please define, "security related materials..."

Follow-up question 9: 1-21.
Re: Amendment 4: DSC and seals.

"Modified LCO (Limiting Condition for Operation) condition removed the requirement to transport affected DSC (Dry Shielded Canister] to TAN (Test Area North) or another facility if the seals could not be adequately repaired or replaced to meet the LCO leakage limit." [Description 1-4-20.]

Where would this material be transported, and for what period of time?

Follow-up Question 10:
Exemption 12c.

1-24. Please qualify and quantify the term "when practicable" when applied to the "methods of criticality control." (p. 1-24)

Respectfully submitted,

Eric Epstein, Chairman
Three Miles Island Alert, Inc.
4100 Hillsdale Road
Harrisburg, PA 17112
lechambon@comcast.net
717-635-8615

Enclosure

cc:

U.S. Nuclear Regulatory Commission
Office of Nuclear Materials Safety and Safeguards
Director, Division of Spent Fuel Management
Washington, D.C. 20555-0001
Attn: Document Control Desk

U.S. Nuclear Regulatory Commission
Region I
Physics Branch Division of Nuclear Materials Safety
Raymond Powell, Chief Decommissioning, ISFSI,
Reactor Health
2100 Renaissance Blvd., Suite 100
King of Prussia, PA 19406

Gregory H. Halnon President & CNO,
GPU Nuclear, Inc.,
VP Nuclear Regulatory Affairs, FirstEnergy
Services FirstEnergy Corporation
Mail Stop: A-WAC-B1 3
41 White Pond Drive
Akron, Ohio 44320

Follow up questions from DOE

Re: NRC Information Notice 2013-07: Premature Degradation of Spent Fuel Degradation of Spent Fuel Storage Cask Structures and Components from Environmental Moisture, (April 16, 2013).

In 2017, the licensee determined that the HSMs appeared to be prematurely deteriorating and that continued crack growth could impact the ability of the HSMs to fulfill their originally planned 50-year design service life. Subsequent evaluations by the licensee initiated the development of an annual inspection plan for the HSMs and base mat as well as an examination of the inside of the HSMs.

The evaluation also recommended that the licensee retain the services of a company experienced and qualified in testing and evaluating concrete to determine the degradation mechanism and make recommendations both for repairs and to prevent further degradation. Although the cracking was discussed with the storage system vendor, the licensee chose an independent vendor to perform an evaluation of the HSMs and base mat concrete in 2009.

The evaluation included a field investigation and laboratory analysis to evaluate the concrete material quality, strength, and long-term durability potential. The conclusion reached was that water had entered the anchor bolt block out holes on the roof of the HSMs.

Subsequent freeze and thaw cycles initiated the crack formation. Repetition of the process resulted in both continued crack growth and the efflorescence growth identified in 2007. In addition to identifying the root cause of the cracking, the report also suggested repairs (injecting resin into the cracks), preventative actions (e.g., installing caps over the anchor bolt block out holes), and monitoring (use of crack gauges). The licensee incorporated the suggested corrective actions.

1) Is the root cause of the crack growth the only causal agent or have additional issues been identified?

2) What is the status of the implementation of the corrective actions preventive actions, repairs and monitoring?

Re:

Three Mile Island, Unit 2 ISFSI at the Idaho National Laboratory Site The Three Mile Island, Unit 2 ISFSI uses NUHOMS-12T horizontal storage modules (HSMs). The HSMs were delivered to the Idaho National Laboratory site in 1999 as precast concrete IN 2013-07 Page 3 of 5 components. The storage system consists of an external rectangular reinforced concrete vault (i.e., HSM) with a storage canister resting horizontally on internal rails inside the HSM.

The prefabricated modules consist of a body and a roof joined together by anchor bolts. All sections were a minimum of 0.6-meters (2-feet) thick. In 2000, the licensee noted cracks in the HSMs, and concluded they were cosmetic and insignificant.

However, in 2007, the licensee observed continued cracking, crazing and spalling as well as increased efflorescence on the HSM surfaces. The efflorescence was a solid, whitish crystalline material which was determined through sampling and analysis to be calcium carbonate.

The licensee performed an evaluation in 2007, during which it determined that the HSMs were capable of performing their design basis functions. In 2008, the licensee noted that 28 of the 30 HSMs had cracks, mostly emanating from the anchor bolt block out holes with widths up to 0.95 centimeters (0.38 inches).

At that time, the licensee determined that the HSMs appeared to be prematurely deteriorating and that continued crack growth could impact the ability of the HSMs to fulfill their originally planned 50-year design service life. Subsequent evaluations by the licensee initiated the development of an annual inspection plan for the HSMs and base mat as well as an examination of the inside of the HSMs.

The evaluation also recommended that the licensee retain the services of a company experienced and qualified in testing and evaluating concrete to determine the degradation mechanism and make recommendations both for repairs and to prevent further degradation. Although the cracking was discussed with the storage system vendor, the licensee chose an independent vendor to perform an evaluation of the HSMs and base mat concrete in 2009.

The evaluation included a field investigation and laboratory analysis to evaluate the concrete material quality, strength, and long-term durability potential. The conclusion reached was that water had entered the anchor bolt block out holes on the roof of the HSMs.

Subsequent freeze and thaw cycles initiated the crack formation. Repetition of the process resulted in both continued crack growth and the efflorescence growth identified in 2007. In addition to identifying the root cause of the cracking, the report also suggested repairs (injecting resin into the cracks), preventative actions (e.g., installing caps over the anchor bolt block out holes), and monitoring (use of crack gauges). The licensee incorporated the suggested corrective actions.

Additional information is available in "Three Mile Island, Unit 2, ISFSI—NRC Inspection of the Independent Spent Fuel Storage Installation—Inspection Report 07200020/2012-001," dated August 14, 2012 (ADAMS Accession No. ML12228A457).

Mark Lombard, Director Division of Spent Fuel Storage and Transportation Office of Nuclear Material Safety and Safeguards

Follow up questions from DOE's Response to U. S. Nuclear Regulatory Commission Questions Regarding the Condition of Three Mile Island Unit-2 Horizontal Storage Modules

Please provide an update on the e following corrective actions which are being performed to further mitigate the concrete, degradation.

1) Verify the effectiveness of the polyurethane foam fill in the HSM anchor block outs to prevent water intrusion (previous completed action),

2) Install nut, bolt, and plate covers over the HSM anchor block outs,

3) Repair the cracks on the HSMs; and,

4) Apply moisture coating to the top surfaces of the HSMs.

Re: Department of Energy Idaho Operations
Office 1955 Fremont Avenue
Idaho Falls, ID 83415
May 3, 2011

U.S. Nuclear Regulatory Commission Attn: Document Control Desk
Washington, D.C. 20555-0001
Subject: Response to U.S. Nuclear Regulatory Commission Questions Regarding Condition of Three Mile Island Unit-2 Horizontal Storage Modules (EM-FMDP- 11-049)

Dear Sir or Madame:

U.S. Nuclear Regulatory Commission (NRC) Region IV Inspection Report "Three Mile Island Unit-2 ISFSI - NRC Inspection Report 072-020/2011-001 and Notice of Deviation," dated April 7, 2011, identified a concern with respect to the condition of the Three Mile Island Unit-2 (TMI-2) Horizontal Storage Module (HSM) concrete. The report requested the Department of Energy (DOE) provide to the NRC information

on the actions which will, be taken to stabilize the. concrete-degradation,-when the actions will be complete, and provisions that will be established to monitor and confirm the actions taken were effective. The NRC report noted several actions have been taken to. stabilize and mitigate the concrete degradation.

In addition to those actions, noted by the NRC, DOE has identified the following corrective actions which are being performed to further mitigate the concrete, degradation.

1) Verify the effectiveness of the polyurethane foam fill in the HSM anchor block outs to prevent water intrusion (previous completed action),

2) Install nut, bolt, and plate covers over the HSM anchor block outs,

3) Repair the cracks on the HSMs, and,

4) Apply moisture coating to the top surfaces of the HSMs.

The corrective actions identified above have been approved and scheduled. It is anticipated these action will be completed during the FY 2011 construction season. In all cases, final corrective actions will be completed no later than the FY2012 construction season.

DOE is developing an "Aging Management Program" in accordance with NUREG- 1927 to support submittal of a license renewal application for the TMI-2 ISFSI in 2017. Provisions (including a concrete surface monitoring program) will be established to monitor and confirm the effectiveness of the corrective maintenance' actions in stabilizing the concrete degradation

U.S. Nuclear Regulatory Commission
EM-FMDP-1 1-049

Finally, The HSM concrete condition has been screened in accordance with 10 CFR 72.48 and determined to not adversely affect the design functions of the HSM as described in the Safety Analysis Report.

If you have questions or comments relating to stabilization of HSM concrete degradation, please contact me at 208-526-4151.

Enclosures:

cc:

Document Control Desk
Kristin Banovac, NRC
Bernard White, NRC
Meraj Rahimi. NRC
Nicholas DiNuzio, DOE HQ
Greg Sossom, DOE HQ

Department of Energy
Idaho Operations Office
1955 Fremont Avenue
Idaho Falls, ID 83415

May 21, 2019

Attn: Document Control Desk
Director, Division of Spent Fuel Management
Office of Nuclear Material Safety and Safeguards
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

SUBJECT: Submittal of Reponses to Additional Nuclear Regulatory Commission Questions Pertaining to the Technical Review of the Three Mile Island Unit 2 Independent Spent Fuel Storage Installation License Renewal Application, Docket 72-0020 (CLN191270)

REFERENCES:
1) Letter: DOE-ID to NRC (EM-NRC-17-007), Subject: Submittal of Application for Renewal of TMI-2 ISFSI License SNM-2508, dated March 6, 2017

2) Letter: DOE-ID to NRC (CLN190779), Subject: DOE-ID Submittal of Responses to NRC Request for Clarification of Response to Request for Additional Information for the Technical Review of the Three Mile Island Unit 2 Independent Spent Fuel Storage Installation License Renewal Application, Docket 72-0020

3) Letter: NRC to DOE-ID, Subject: Request For Additional Clarification of the Application for Renewal of the Three Mile Island Unit 2 Independent Spent Fuel Storage Installation License No. SNM-2508 (CAC/EPID Nos. 001028/L-2017-RNW-0019 and 000993/L-2017-LNE-0007), dated May 13, 2019

Dear Sir or Madam:

On March 6, 2017, the Department of Energy, Idaho Operations Office (DOE-ID) submitted a license renewal application (LRA) requesting a 20-year renewal of Three Mile Island Unit 2 (TMI-2) Independent Spent Fuel Storage Installation (ISFSI) specific license SNM-2508 (Reference 1) (ADAMS Accession Nos. ML17089A501 and ML17075A199 through ML17075A201). The Nuclear Regulatory Commission (NRC) acknowledged acceptance of the LRA on May 5, 2017 (ADAMS Accession No. ML17125A284).

On April 16, 2019 DOE-ID discussed with the NRC additional questions and clarifications pertaining to the Reference 2 response that the staff requires to complete its Safety Evaluation

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Report on the LRA. That conversation and the additional questions are documented in the NRC's Conversation Record dated April 22, 2019 (ADAMS Accession No. ML19112A158) and the Reference 3 letter (ADAMS Accession No. ML19134A042), respectively. Enclosure 1 to this letter contains the NRC questions and provides the DOE-ID responses. Also enclosed with this submittal is Revision 3 of the LRA (Enclosure 2), which reflects the changes required as a result of the responses and a small number of editorial corrections.

In addition, we have also taken the opportunity to revise proposed License Condition 18 in LRA Appendix D to comport with the discussion held on April 16, 2019 and documented in the associated Conversation Record.

Should you have questions or require additional information, please contact me at (208) 526-4993, or Steve Ahrendts at (208) 526-8888.

Sincerely,

Steven Wahnschaffe
License Manager

Enclosures: 1) Responses to Additional NRC Requests for Clarification on TMI-2 ISFSI LRA
2) TMI-2 ISFSI License Renewal Application, Revision 3

cc: Kristina Banovac, NRC (w/ electronic enclosures)
Bernard White, NRC (w/ electronic enclosures)
Meraj Rahimi, NRC (w/ electronic enclosures)
Nicholas DiNunzio, DOE HQ (w/ electronic enclosures)
Greg Sosson, DOE HQ (w/ electronic enclosures)

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~~wahnfctdeid.doe.gov~~
wahnfctdeid.doe.gov

Department of Energy
Idaho Operations Office
1955 Fremont Avenue
Idaho Falls, ID 83415

May 21, 2019

Attn: Document Control Desk
Director, Division of Spent Fuel Management
Office of Nuclear Material Safety and Safeguards
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

SUBJECT: Submittal of Reponses to Additional Nuclear Regulatory Commission Questions Pertaining to the Technical Review of the Three Mile Island Unit 2 Independent Spent Fuel Storage Installation License Renewal Application, Docket 72-0020 (CLN191270)

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Sincerely,

Steven Wahnschaffe
License Manager

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2) TMI-2 ISFSI License Renewal Application, Revision 3

cc: Kristina Banovac, NRC (w/ electronic enclosures)
Bernard White, NRC (w/ electronic enclosures)
Meraj Rahimi, NRC (w/ electronic enclosures)
Nicholas DiNunzio, DOE HQ (w/ electronic enclosures)
Greg Sosson, DOE HQ (w/ electronic enclosures)

NRC RAI 2-2, 3-4, and 3-6 Additional Follow-up Question

Provide additional information to justify scoping the dry shielded canister (DSC) basket and DSC purge port block out of the renewal review. Alternatively, revise the renewal application to scope the basket, and purge port block into the renewal review and perform an appropriate aging management review for those components.

The licensee proposed to scope out of the renewal review the DSC's purge port block and the DSC's basket, since it did not identify that these components met any of the criteria in Section 2.4.2 of NUREG-1927 for scoping into the renewal review. In response to the staff's questions, the licensee performed a supplemental shielding analysis to justify its scoping out of these components. However, the staff found that the supplemental analysis was not adequate to justify the licensee's scoping determination because it did not demonstrate that the design basis for the ISFSI would be maintained when these components and their functions were neglected, including with respect to the following items:

1. The supplemental shielding analysis focused only on the dose rates at the surfaces of the HEPA filter housings on the DSC ports and compliance with the technical specification limit for that location. The design basis in terms of shielding and radiation protection also includes dose rates at the horizontal storage module (HSM) rear access door and the effects of dose rate changes on the occupational and public dose assessments in the UFSAR.

2. In evaluating to support the assumption that the vent port dose rates bound the purge port dose rates, the analysis does not address reasonable worst case configurations and impacts due to neglect of the purge port block. For example, the analysis does not include a debris canister placed nearly centered directly beneath the purge port. Also, the licensee appears to disregard a significantly higher dose rate configuration included in the analysis, without justification.

3. The referenced analysis to show the HSM rear access door dose rates do not exceed the technical specification limit is not consistent with the configurations evaluated in the supplemental analysis for the vent and purge port surface dose rates with the basket and purge port block neglected. The referenced analysis smears the fuel debris contents throughout the DSC cavity, which is more consistent for crediting the positioning of the contents by the basket and does not account for the vent and purge port configurations. Given the significantly higher dose rates at the vents' filter housings calculated in the supplemental analysis (compared to the design-basis analysis), it is not clear that an appropriate analysis of HSM rear access door dose rates, which accounts for the configurations neglecting the positional function of the basket and the functions of the purge port block, would be below the technical specification limit.

4. The impacts of the dose rate increases in the supplemental analysis on the occupational doses were not adequately considered. For example, impacts on the evaluation in Section 7.4.1 of the UFSAR were not evaluated. Additionally, for analyses that were provided in the RAI responses, comparisons of doses for actions involving a single DSC were compared against regulatory limits. However, these actions are likely to be performed on multiple DSCs

within a given year. Thus, evaluations of doses for actions performed on a reasonable number of DSCs expected over a given year should be compared against the regulatory limits since those limits are annual limits.

5. It is not clear that other dose rates calculated in the UFSAR that would be affected by the neglect of these components have been considered and evaluated for the renewal.

In addition, given the significant increases in the dose rates calculated in the supplemental analysis versus those for the design basis analysis, the staff finds that, even if technical specification limits are not exceeded, the design basis in the UFSAR appears to be significantly exceeded or changed when the basket and purge port block are neglected. The supplemental analysis did not account for the decay of the source term over the initial license period of 20 years. However, based on staff calculations, the decrease in the design basis source term due to the 20 years decay is not sufficient to compensate for the neglect of the basket and the purge port and the Licon in the debris fuel canisters, the effects of which were included in the analysis for the basket and purge port. Per those calculations, the source term only decreases by about 40% across much of the spectrum, particularly that portion of the spectrum that contributes most to dose rates, though some parts of the spectrum decrease by about 60%. This is enough to compensate for the effects of the Licon alone, but not for the effects of neglecting the basket and the purge port block. The staff notes that ideally, a demonstration that the design basis is maintained would be done by showing the design basis dose rates in the UFSAR that would be affected by the neglect of components and their functions are not exceeded.

Thus, the licensee should revise its renewal application to provide an analysis that demonstrates that the design basis is maintained when accounting for the effects of neglecting the basket and purge port block in conjunction with neglecting the Licon. The analysis should address the issues described above. Alternatively, the licensee could scope in the basket and purge port block and perform an appropriate aging management review for those components. Since the materials and environment for them are the same as for the vent port shield block, which the licensee did scope in, the staff expects that the aging management review results and aging management activities for the vent port shield block will apply to the basket and purge port block.

This information is needed to determine compliance with 10 CFR 72.24(e), 72.122(h)(5), and 72.42(a).

DOE-ID Response

DOE-ID has revised the license renewal application (LRA) to change the scoping determination for the DSC basket and purge port block to "in-scope" for renewal and has performed an appropriate aging management review for the affected items. These two components are comprised of Items 1 through 6 on DSC Drawing 219-02-1000 and Item 4 on Drawing 219-02-1003. As discussed in the April 16, 2019 telephone conversation, these changes to the scoping evaluation affect the scoping and aging management review (AMR) sections of the LRA but do not affect the proposed aging management program (AMP) previously proposed for the DSC.

The appropriate changes have been made to the following LRA information in Revision 3 to the document to reflect the change in scoping determination for these components:

- Section 2.3.2.1
- Tables 2-3 and 2-4
- Tables 3-4 and 3-5
- Sections 3.4.1 through 3.4.4
- Section 3.8
- Section A1.1
- Section A1.3

All changes to the LRA are indicated with revision bars in the right margin. DOE-ID has also made three editorial corrections to the LRA, as follows: In LRA Section 3.7.2.1 the number '1' was added to the end of the cited work order. In the last line of LRA Section A1.4, the cross-reference language was clarified. Lastly, LRA Table 2-3 was updated for Item 5 on Drawing 219-02-1002. This item has always been in scope for renewal but the table inadvertently did not indicate which intended function was applicable. The table has been revised to indicate "Yes" for the confinement function for this item.

As discussed in the cover letter, LRA Section D.2.2 has also been revised to reflect our desired language for proposed License Condition 18 that was discussed in the April 16, 2019 conference call. Please also note that no changes to the proposed revisions to TMI-2 ISFSI FSAR information in LRA Appendix C were required because the proposed FSAR tables in that appendix simply refer to the associated LRA tables, some of which were revised with this submittal. The latest versions of the LRA tables in the body of LRA Revision 3 that are referred to in LRA Appendix C will be incorporated into the TMI-2 ISFSI FSAR.

NRC RAI 3-9 Additional Follow-up

Provide the following with respect to the debris canister drying process for ensuring that residual water in the debris canisters does not exceed the bound and unbound water limits in the UFSAR, which are based on the water amounts evaluated in the criticality analyses.

This information is needed to determine compliance with 10 CFR 72.42(a).

DOE-ID Response

Sub-questions and responses are provided individually below.

NRC Sub-question a

Confirmation of the staff's understanding of the criteria and process for ensuring the residual water in the debris canisters is below the UFSAR limits for bound and unbound water, specifically that:

The falling rate is used for ensuring unbound water is removed whereas a drying time is used, after reaching a first plateau in the falling rate, for ensuring bound water is adequately removed. Reaching the first, or a, plateau in the falling rate indicates that nearly all unbound water is gone. The drying time after getting to that plateau ensures that the fuel debris temperature exceeds the 170°F minimum for ensuring the bound water does not exceed the limit specified for bound water in the UFSAR analyses. It appears that the fuel temperature at the centerline reaches or exceeds that minimum temperature before the falling rate plateau is reached for the unbound water, hence only an hour of drying beyond the point at which that plateau is reached is needed.

DOE-ID Response to Sub-question a

The staff's understanding is correct for unbound water removal, using the falling rate drying acceptance criteria. However, see Response 'b' for clarification of the term "plateau," because the bulk of the unbound water has been removed by the time falling rate drying begins. For unbound water removal, complete removal is assessed by comparison to one of the three methods indicated in Section 7.4.2.2 of EDF-1466 (LRA Reference 3.11.5). The three methods require attaining either an acceptable change in slope of the falling rate drying curve or a total decline in the falling rate of unbound water removal.

The staff's understanding is also correct for the bound water removal acceptance criteria, except that the plateau marks the end of falling rate drying, not the start. From Figure 23 of EDF-1466, which represents the model in EDF-1469, the core debris fuel temperature reaches the 170°F minimum temperature requirement at less than nine hours of total drying time. However, this nine total hour marker is independent of the time it takes to reach the first plateau for complete unbound water removal. Rather, the core fuel debris temperature necessarily reaches the 170°F minimum temperature requirement, because the unbound water removal acceptance criterion ensures the core debris has been heated sufficiently to raise the core debris temperature, thus allowing for bound water removal. During falling rate drying to remove any residual unbound water, leaving only bound water, the heat added to the debris will continue to raise its temperature, thus removing increasing amounts of the bound water content. Reaching the 170°F minimum temperature shows an adequate amount of bound water has been removed, supporting a maximum of 2.13 liters of bound water remaining in the canister.

NRC Sub-question b

Confirmation that ensuring the plateau in the falling rate is reached also ensures that all free water in the debris canister, including in parts of the debris canister where fuel debris is not located (e.g., in areas around the hydrogen recombiners), is removed from the canister.

DOE-ID Response to Sub-question b

The staff's understanding is correct. The term "plateau" as used in EDF-1466 marks the end of falling rate drying, indicating the furnace isolation pressure remains constant as total furnace drying time increases. The removal of unbound water is dependent on meeting one of the three methods indicated in Section 7.4.2.2 of EDF-1466 (see Response 'a') for falling rate

drying acceptance criteria, which effectively represents reaching the first plateau. However, the plateau itself may not be evident, especially for intact or near-intact fuel assemblies; hence the second method in Section 7.4.2.2 may be used. This is evident in Figure 25 of EDF-1466 for HVDC-010.

NRC Sub-question c

Clarification as to why the falling rate would have two plateaus and whether the second plateau indicates that essentially all bound water is removed and occurs at debris temperatures that significantly exceed the 170°F minimum.

DOE-ID Response to Sub-question c

The staff's understanding is correct. The second falling rate curve is explained by the final removal of remaining bound water and the second plateau represents essentially no remaining bound water.

NRC Sub-question d

Confirmation that the operations steps in TPR-1190 Rev. 15 were followed for all debris canister drying operations for both furnaces, even though the TPR for the second furnace is a separate document.

DOE-ID Response to Sub-question d

The staff's understanding is correct. While the specific language and step numbers may have varied in other revisions of TPR-1190 and TPR-6596, the drying operations described in the responses above for unbound and bound water removal were followed for all TMI-2 Canister drying operations, regardless of specific drying procedure or revision used.

NRC Sub-question e

Clarification as to how operations for which some steps were marked as 'N/A' and therefore not performed still meet the conditions used in the analyses to show the maximum residual water (bound and unbound) will not exceed the UFSAR limits.

DOE-ID Response to Sub-question e

Particular steps (e.g., TMI-2 Canister Loading or Unloading operations) may not be applicable to a given drying campaign or are covered in ancillary procedures or other working copies of the drying procedures, and thus would have the Not Applicable (i.e., "N/A") sign-off for those steps. For example, step 3.1.5 in Revision 15 of TPR-1190 is specifically for TMI-2 Filter or Knockout Canisters. If only TMI-2 Fuel Canisters were being dried, this step would not apply. In addition, steps not required for each separate canister may be marked "N/A" if already performed during a previous canister loading for that drying cycle In addition, some steps may have been previously performed in other procedures or in other working copies of

the drying procedures, and are therefore not applicable to that particular sign/off copy.

If steps are marked as "N/A," these steps are either unrelated to compliance or do not factor into compliance with any of the dryness acceptance criteria. The criticality analyses prescribe bounding FSAR water volumetric limits that were used in driving the dryness acceptance criteria specified in EDF-1466. These dryness acceptance criteria were implemented via the drying procedural steps for each drying campaign. It follows that as long as the procedural dryness acceptance criteria are met, the FSAR water volumetric limits would not be exceeded.

NRC Sub-question f

Confirmation of the location in the fuel debris in the debris canisters of the coolest debris temperature (axial and radial) and that the minimum temperature at that location exceeded the 170°F minimum. The references discuss a centerline temperature; however, it is not clear this is the centerline of each debris canister. Also, with four canisters dried at the same time in a furnace, it is not clear the debris canister centerline would be the point of the coolest debris temperature.

DOE-ID Response to Sub-question f

The centerline of the TMI-2 Canister is the coolest spot radially during drying. In terms of axial heat distribution, as discussed in the EDF-1469 summary, the top head of the furnace was water cooled. Thus, the top end of the TMI-2 Canister was conservatively considered to be cooler than lower elevations within the furnace. This factor could not be modeled directly in the two-dimensional ABAQUS model. Therefore, this was accounted for in the model by reasonably matching heater output power with actual drying data. Because 50 kilowatts represented the actual full heater output, a match was attained by modeling only 70% of this heat output for drying campaign HVDC-005 and 50% for HVDC-009. In any case, the dryness acceptance criteria discussed in Response 'a' above ensure that the minimum core fuel debris temperatures at any TMI-2 Canister location would exceed the 170°F minimum threshold.

NRC Sub-question g

Confirmation that the debris temperatures reached at least the 170°F minimum in drying run -010.

DOE-ID Response to Sub-question g

In drying run -010, the canister was heated long enough to remove the bound water. See Figure 25 in EDF-1466, where the second plateau is reached after about 12 hours of heating. Again, the dryness acceptance criteria discussed in Response 'a' above ensure that the minimum core fuel debris temperatures at any TMI-2 Canister location would exceed the 170°F minimum threshold.