UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

February 5, 2019

Mr. John P. Zimmerman
Deputy Manager, Idaho Cleanup Project
1955 Fremont Avenue, MS 1222
Idaho Falls, ID 83415

SUBJECT: REQUEST FOR CLARIFICATION OF RESPONSE TO ADDITIONAL INFORMATION FOR THE TECHNICAL REVIEW 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)

Dear Mr. Zimmerman:

By letter dated March 6, 2017, the U.S. Department of Energy, Idaho Operations Office (DOE-ID) submitted an application for renewal of License No. SNM-2508 for the Three Mile Island Unit 2 (TMI-2) Independent Spent Fuel Storage Installation (Agencywide Documents Access and Management System (ADAMS) Accession No. ML17075A198). The U.S. Nuclear Regulatory Commission (NRC) staff sent a request for additional information (RAI) related to the technical review of the renewal application on January 29, 2018 (ADAMS Accession No. ML18030A172), to which you responded on September 26, 2018 (ADAMS Accession No. ML18283A222). On October 2, November 15, and November 21, 2018 (ADAMS Accession Nos. ML18303A125, ML18331A337, and ML18331A262 respectively), you provided supplemental information that was referenced in the September 26, 2018, response.

The NRC staff reviewed DOE-ID's RAI responses and held a meeting with DOE-ID on December 6, 2018, to clarify the responses (see December 6, 2018, meeting summary at ADAMS Accession No. ML18360A186). The NRC staff considered the information provided by DOE-ID during the meeting and determined that the enclosed clarifications to the RAI responses are needed for the NRC staff to complete its technical review.

Discussion of this request for clarification occurred on January 22, 2019. We request that you provide this information by April 1, 2019. Inform us at your earliest convenience, but no later than March 18, 2019, if you are not able to provide the information by that date. To assist us in rescheduling your review, you should include a new proposed submittal date and the reasons for the delay.

Please reference Docket No. 72-20 and CAC/EPID Nos. 001028/L-2017-RNW-0019 and 000993/L-2017-LNE-0007 in future correspondence related to this request.

If you have any questions regarding this matter, please contact me at (301) 415-7116 or via Kristina.Banovac@nrc.gov.

Sincerely,

/RA/

Kristina L. Banovac, Project Manager
Renewals and Materials Branch
Division of Spent Fuel Management
Office of Nuclear Material Safety
and Safeguards

Docket No.: 72-20
License No.: SNM-2508
CAC/EPID Nos.: 001028/L-2017-RNW-0019
000993/L-2017-LNE-0007

Enclosure:
Request for Clarification

cc: TMI-2 ISFSI Service List

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TMI-2 ISFSI Service List

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SUBJECT: REQUEST FOR CLARIFICATION OF RESPONSE TO ADDITIONAL INFORMATION FOR THE TECHNICAL REVIEW 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), DOCUMENT DATE: February 5, 2019

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ADAMS Accession No.: ML19037A066

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Request for Clarification
U.S. Department of Energy, Idaho Operations Office
Docket No. 72-20
License No. SNM-2508
License Renewal

By letter dated March 6, 2017, the U.S. Department of Energy, Idaho Operations Office (DOE-ID) submitted an application for renewal of License No. SNM-2508 for the Three Mile Island Unit 2 (TMI-2) Independent Spent Fuel Storage Installation (ISFSI) (Agencywide Documents Access and Management System (ADAMS) Accession No. ML17075A198). The U.S. Nuclear Regulatory Commission (NRC) staff sent a request for additional information (RAI) related to the technical review of the renewal application on January 29, 2018 (ADAMS Accession No. ML18030A172), to which DOE-ID responded on September 26, 2018 (ADAMS Accession No. ML18283A222). On October 2, November 15, and November 21, 2018 (ADAMS Accession Nos. ML18303A125, ML18331A337, and ML18331A262 respectively), DOE-ID provided supplemental information that was referenced in the September 26, 2018, response. The NRC staff reviewed DOE-ID's RAI responses and held a meeting with DOE-ID on December 6, 2018, to clarify the responses (see December 6, 2018, meeting summary at ADAMS Accession No. ML18360A186).

This request for clarification of RAI responses identifies information needed by the NRC staff to complete its technical review of the license renewal application (LRA) and to determine whether the applicant has demonstrated compliance with the regulatory requirements. The requested information is listed by RAI response number.

In addition, the NRC staff requests the applicant to provide the revised updated final safety analysis report (UFSAR) Supplement (provided in Appendix C of the LRA), including any referenced tables, as revised through the responses to the January 29, 2018, RAI and this request for clarification. The NRC staff needs this information to reference in the proposed license condition # 17.

<u>RAI 2-2 and RAI 3-6 Follow-up</u>

Revise the supplemental shielding analyses in support of excluding the dry shielded canister (DSC) basket from the scope of renewal review (Orano Federal services Calculation CALC-3021323) to exclude the presence of the purge port shield block.

The supplemental shielding analyses in support of the exclusion of the DSC basket from the scope of renewal review rely on the presence of the purge port shield block. These analyses only look at dose rates for the vent port and filter housing because, as stated in the reports documenting the analyses, the purge port dose rates are bounded by the vent port dose rates (e.g., see Section 3.2, item 2 of LRA reference 3.11.216). That can only be so because of the presence of the purge port shield block, which like the shield block for the vent port, ensures against radiation streaming through the purge port and that a TMI-2 canister cannot be located directly beneath the purge port. However, the purge port shield block is scoped out of the renewal. Thus, the supplemental shielding analyses should include analyses for the purge port without credit for its shield block (in terms of both the shielding contribution and the relative placement of the TMI-2 canisters to the purge port). Alternatively, include the shield block in the renewal scope, provide an aging management review and any necessary corresponding time-limited aging analyses or aging management programs, and revise the UFSAR Supplement in Appendix C of the LRA, as appropriate.

This information is needed to determine compliance with Title 10 of the *Code of Federal Regulations* (10 CFR) 72.24(e), 72.122(h)(5), and 72.42(a).

<u>RAI 2-2, 3-4, and 3-6 Follow-up</u>

Provide the following additional information to clarify or support the responses to RAIs 2-2, 3-4, and 3-6.

a) An evaluation of dose rates at the horizontal storage module (HSM) rear access panel surface as a result of neglecting the TMI-2 fuel canister licon and the DSC basket's positioning of the TMI-2 canisters within the DSC (relative to the DSC vent and purge ports).

The combined effects of neglecting the licon in the TMI-2 fuel canisters and neglecting the positioning of the canisters in the DSC by the DSC's basket resulted in calculated dose rates at the DSC vent filter housing of about 175 mrem/hr. Given this result and the trend in dose rates in the vicinity of the housing to be larger than the dose rates on the housing surface (indicated in LRA references 3.11.102 and 3.11.103 and by DOE-ID in the December 6, 2018, meeting), it is not clear that the dose rate limit for the HSM rear access door that is specified in the technical specifications would not be exceeded. Thus, the evaluation of the dose rate impacts should include dose rates at the HSM rear access panel surface.

b) Provide an evaluation of occupational doses for operations for DSCs without a basket and TMI-2 fuel canisters without licon that demonstrates occupational doses will remain within, or consistent with, the design basis described in UFSAR Section 7.4.1 and other relevant UFSAR sections (e.g., Section 7.1.2) and occupational dose limits in 10 CFR Part 20.

The design bases in the UFSAR include an evaluation of occupational doses (see Section 7.4.1 of the UFSAR), which is based upon calculated dose rates from the UFSAR (see Section 7.3.2.2) and, for top end dose rates, appears to be based on LRA reference 2.4.6. Those calculated dose rates are in turn based on the DSC basket maintaining the positions of the TMI-2 canisters relative to each other and the DSC vent and purge ports. The dose rates are also based on the presence of the licon in the TMI-2 fuel canisters. To support the applicant's proposal to scope the DSC basket and TMI-2 fuel canister's licon out of renewal, the applicant conducted a supplemental analysis to evaluate the effects of neglecting the DSC's basket and TMI-2 fuel canister's licon on shielding and radiation protection. The applicant's supplemental analysis resulted in higher calculated dose rates for the top end areas of the DSC (i.e., 175 mrem/hr at the DSC vent filter housing calculated in the supplemental analysis, compared to ~ 3 mrem/hr calculated in LRA reference 2.4.6). Thus, it is not clear that the design basis occupational dose analysis in the UFSAR would be met or maintained for DSCs in this configuration (i.e., no DSC basket and no TMI-2 fuel canister licon).

The applicant's supplemental analysis did include some evaluation with regard to compliance with 10 CFR Part 20 occupational dose limits for DSCs in this configuration; however, the evaluation is limited in the activities it considered and is limited to a single DSC. The evaluation should consider relevant transfer operations, periodic maintenance activities involving or around the vent and purge port filter housings and the HSM rear access door, and activities required by technical specifications (e.g., surveillances for limiting conditions for operation 3.1.1 and 3.2.3). The evaluation should also account for operations within a given year involving not just one, but

multiple DSCs, up to all 29 loaded DSCs. Applying the calculated doses in the current supplemental analysis to all 29 DSCs indicates regulatory limits would be exceeded were a single individual to perform these activities for all 29 DSCs. Thus, the evaluation should describe the actions that would be taken, controls that would be imposed, or conditions that would assure that occupational dose limits will not be exceeded. Guidance provided in Section 11.4.3.1 of NUREG-1567, particularly the bulleted list at the end of the section, should be considered.

c) An explanation of the differences between the estimated dose rates for the vent port housing and the measured surface dose rates reported in LRA references 3.11.102 and 3.11.103.

It is not clear why measured dose rates on the vent filter housing surface, as reported in LRA references 3.11.102 and 3.11.103 (i.e., up to 15 mrem/hr), are higher than the dose rates estimated for that location in LRA reference 2.4.6 (i.e., ~ 3 mrem/hr).

d) An explanation of the locations of the measured 1 to 5 mrem/hr neutron dose rates reported in the text of the "Results" sections in LRA references 3.11.102 and 3.11.103.

The text of the "Results" section of LRA references 3.11.102 and 3.11.103 also states neutron dose rate measurements of 1 to 5 mrem/hr for HSMs 4 and 22; however, the location of these measured neutron dose rates (e.g., on the HSM rear access doors or the DSC purge and vent port filter housings) is unclear.

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

RAI 2-5 Follow-up

Propose a license condition or technical specification to use transfer cask spacers that are aged less than 20 years, and revise the UFSAR Supplement in Appendix C of the LRA, as appropriate.

The current TMI-2 design bases include the use of transfer cask spacers, as discussed in the UFSAR. Also, while the UFSAR shielding analyses do not credit the material of the transfer cask spacers, the analyses do credit the spacers' function of axially positioning the DSC within the transfer cask. The positioning of the DSC within the transfer cask has an important effect on transfer cask dose rates, both axially and radially for areas where the transfer casks' radial shielding changes. To address this effect and to support scoping out the transfer cask spacers, the applicant provided a supplemental analysis to evaluate the dose rate effects of neglecting the spacers to demonstrate that regulatory requirements would still be met when accounting for those effects. In addition, the applicant also noted in the RAI response that the spacers used during the initial ISFSI loading campaign no longer exist, and new spacers would need to be fabricated if the transfer cask were used in the future at the ISFSI (e.g., for retrieval operations from storage).

At the December 6, 2018, meeting, the NRC staff indicated it had questions regarding the supplemental analysis, and DOE-ID and NRC staff discussed the approach of a license condition limiting the age of spacers, given new spacers would need to be fabricated for future use. Under this approach, the spacers would scope into the renewal; however, the license condition would preclude the need to conduct an aging management review of the spacers.

The applicant should also include any corresponding changes to UFSAR Supplement in Appendix C of the LRA (e.g., scoping tables). The NRC staff is amenable to discussing alternative approaches upon DOE-ID request.

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

RAI 3-5 Follow-up

Revise the proposed UFSAR Supplement in Appendix C of the LRA to ensure that the HSM AMP adequately addresses the potential for aggregate reactions (alkali silica reaction, ASR).

In RAI 3-5, the staff requested that the applicant justify its conclusion that aging effects due to ASR in the HSM concrete are not credible. In its response, the applicant provided additional details of the petrographic characterization of cores obtained from HSMs affected by early design-related degradation. These results showed evidence of ASR (isolated small patches of white ASR gel) observed in one of six core samples taken during this evaluation in 2009. However, the applicant still concluded ASR to not be credible due to the limited identification of reactive aggregates per these results, the concrete mix specification used for HSM fabrication, and the limited presence of water during operations.

During the December 6, 2018, meeting, the NRC staff noted that it does not agree with the justification for excluding aggregate reactions as a credible aging mechanism, per the technical basis provided in Section 3.5.1.3 of NUREG-2214 (Managing Aging Processes In Storage (MAPS) Report), currently in final publication review. In NUREG-2214, the staff states that ASR, the most common aggregate reaction, is generally a slow degradation mechanism. ASR may take from 3 to more than 25 years to develop in concrete structures, depending on the nature (reactivity level) of the aggregates, the moisture and temperature conditions to which the structures are exposed, and the concrete alkali content. The delay in exhibiting deterioration indicates that there may be less reactive forms of silica that can eventually cause deterioration.

Operating experience has revealed degradation of the concrete in the Seabrook reactor containment as a result of ASR (see NRC Information Notice (IN) 2011-20, ADAMS Accession No. ML112241029). The concrete used at the Seabrook nuclear power plant passed all industry standard ASR screening tests at the time of construction. However, ASR-induced degradation was identified in August 2010. Per IN 2011-20, licensees that tested using American Society for Testing and Materials (ASTM) C227 and ASTM C289 could have concrete that is susceptible to ASR-induced degradation since these standard methods may not accurately predict aggregate reactivity when dealing with late- or slow-expanding aggregates containing strained quartz or microcrystalline quartz. In addition, ASR screening tests are generally not conducted on each aggregate source but rather in select batches, which increases the risk for use of aggregates of different reactivities when procured from different sources.

Due to the uncertainties in screening tests that can effectively be used to eliminate the potential for ASR and previous ASR operating experience at a nuclear facility, the staff considers the aging mechanism to be credible in concrete exposed to any environment with available moisture, and therefore, aging management is required during the 40-year timeframe. In its response to RAI 3-5, the applicant did not provide a technical basis to support that the conclusions in IN 2011-20 are not applicable to the TMI-2 HSM concrete (i.e., that the petrographic test methods used to assess aggregate reactivity of the TMI-2 HSM concrete aggregates is not subject to the uncertainties and risks of underestimated reactivities as discussed in IN 2011-20).

In the December 6, 2018, meeting, DOE-ID and the NRC staff discussed options to address the staff's concern and how the final approach should be reflected in the design bases (i.e., the UFSAR Supplement to be incorporated upon license renewal). In one approach, the applicant may choose to expand the scope of the HSM AMP to include aging effects due to aggregate reactions (i.e., define these aging effects to be credible). In an alternative approach, the applicant may credit the use of the American Concrete Institute 349.3R second tier acceptance criteria in the HSM AMP but predefine a corrective action that ensures that conditions not meeting these acceptance criteria will be evaluated to ensure ASR is not the apparent or root cause. Per NUREG-2214, the NRC staff considers the use of these acceptance criteria to be acceptable for managing ASR aging effects. However, the NRC staff wants to ensure that the HSM AMP for the TMI-2 ISFSI properly captures the potential for ASR. The NRC staff is amenable to discussing further alternative approaches upon DOE-ID request.

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

RAI 3-7 and RAI 3-8 Follow-up

Modify the benchmark analysis for the criticality calculations that support the responses to RAIs 3-7 and 3-8 to include sufficient trending with respect to key parameters and to include experiments with materials relevant to the criticality calculations. Also, confirm that the benchmark models used the same modeling techniques as the TMI-2 fuel canister analysis models.

Benchmark analyses for typical criticality evaluations consider trends on several parameters. However, the benchmark analysis for the calculations supporting the RAI responses only looks at trends for two parameters: the energy of average lethargy of fission and the uranium-235 number density in the fuel. While the calculations are not for a typical case, trends with respect to other parameters should have been considered, such as the ratio of hydrogen to fissile atoms. Also, the selected benchmark experiments include experiments with materials that are not relevant to the TMI-2 fuel canisters, such as cadmium. The benchmark analysis should include, to the extent practical, only experiments with materials that are in the analyzed system (i.e., the TMI-2 fuel canisters in the DSC in an HSM). Thus, the evaluation of trends in the benchmark analysis should determine the trends when experiments with the non-relevant materials are excluded to ensure an appropriate or bounding bias and bias uncertainty is used in the criticality evaluation.

In addition, the models for the benchmark analysis should use the same techniques as were used in the fuel canister analysis. This includes use of the cellmix option to create the material used in the geometry to represent the fuel both with and without water from the cell data card information as opposed to explicit modeling of the fuel pellets in the geometry. If the model for the benchmark analysis did not use the same techniques used in the fuel canister analysis, the applicant should revise the benchmark analysis accordingly.

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

RAI 3-9 Follow-up

Provide a clear and concise justification that the maximum water content assumed in the criticality analyses in the FSAR and in the supplementary criticality analysis (in response to RAIs 3-7 and 3-8) is not, or will not, be exceeded.

The maximum amount of water assumed in the analyses includes bound and unbound water that remained in the TMI-2 canisters after drying and water that could be reacquired during storage at the ISFSI, including for the requested period of extended operation. It is still not clear from the response to RAI 3-9 and the supporting references that the drying of the TMI-2 canisters ensured this maximum amount was not exceeded. In particular, LRA reference 3.11.5 discusses different conditions of drying (e.g., specific temperatures at specific pressures), which if met would ensure a certain level of dryness, or maximum residual bound and unbound water. It also appears to discuss different processes or indications (e.g., looking at plateaus in falling drying rates). However, the reference also appears to indicate that, in different instances, different criteria and indications were used for different canisters in ways that do not seem to be consistent. Also, while some canisters may have been checked for dryness, it is not clear how that assured all canisters were at the same level of dryness or did not exceed the residual water maximum amount. The licensee should provide a clear description of how the canisters were actually verified to not exceed the appropriate residual (bound and unbound) water maximum amount (including for the licon for the fuel canister since water here can get into the fuel cavity), the criteria that were used, and clear confirmation that the canisters were verified to meet those criteria. Any applicable references or appropriate supporting documentation should also be provided.

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