TMI-2 *SOLUTIONS*

September 29, 2022

TMI2-RA-COR-2022-0019

10 CFR 50.90
10 CFR 50.91
10 CFR 70.17
10 CFR 70.24

U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, DC 20555-0001

Three Mile Island Nuclear Station, Unit 2 (TMI-2)
NRC Possession Only License No. DPR 73
NRC Docket No. 50-320

Subject: License Amendment Request – Three Mile Island, Unit 2, Decommissioning Technical Specifications, Response to Request for Additional Information

References:

1) Letter to Sauger, J.T. (TMI-2 Solutions, LLC) from Snyder, A.M. (U.S. NRC), "Three Mile Island, Unit 2 – Request for Additional Information for Requested Licensing Action Regarding Decommissioning Technical Specifications," dated 29 July 2022 (ML22210A080, pkg; RAI Enclosure ML22210A088)
2) Letter TMI2-RA-COR-2021-0002 from van Noordennen, G.P. (TMI-2 Solutions, LLC) to Document Control Desk (U.S. NRC), "License Amendment Request – Three Mile Island, Unit 2, Decommissioning Technical Specifications," dated 19 February 2021 (ML21057A046)
3) Letter TMI2-RA-COR-2022-0002 from van Noordennen, G.P. (TMI-2 Solutions, LLC) to Document Control Desk (U.S. NRC), "License Amendment Request – Three Mile Island, Unit 2, Decommissioning Technical Specifications, Supplemental Information," dated 7 January 2022 (ML22013A177)
4) "TMI-2 Accident Analysis Questions," dated 7 February 2022
5) Letter TMI2-RA-COR-2022-0008 from van Noordennen, G.P. (TMI-2 Solutions, LLC) to Document Control Desk (U.S. NRC), "Supplemental Information to License Amendment Request- Three Mile Island, Unit 2, Decommissioning Technical Specifications," dated 7 April 2022 (ML22101A077)
6) Letter TMI2-RA-COR-2022-0007 from van Noordennen, G.P. (TMI-2 Solutions, LLC) to Document Control Desk (U.S. NRC), "License Amendment Request – Three Mile Island, Unit 2, Decommissioning Technical Specifications, Response to Questions," dated 8 April 2022.
7) Letter TMI2-RA-COR-2022-0013 from Lackey, M.B. (Energy*Solutions*), "License Amendment Request – Three Mile Island, Unit 2, Decommissioning Technical Specifications, Response to Questions," dated 16 May 2022 (ML22138A285)

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Reference 1 requested additional information to aid in the review of potential events at TMI-2 described by Reference 2, which was supplemented by Reference 3. The NRC asked preliminary questions in Reference 4, and TMI-2 Solutions provided responses and supplemental information in References 5 through 7. The Request for Additional Information in Reference 1 was in response to the latest information provided.

Attachment 1 contains TMI-2 Solutions' responses to the questions in Reference 1. Attachment 2 contains a Request for Exemption from the requirements of 10 CFR 70.24, "Criticality Accident Requirements." Attachment 3 contains the list of Regulatory Commitments included in this submittal.

In accordance with 10 CFR 50.91(7)(b)(1), a copy of this submittal has been sent to the Commonwealth of Pennsylvania.

In the event that the NRC has any questions with respect to the content of this document, please contact me at 509-420-3078 or Mr. Tim Devik, TMI-2 Licensing Manager, at 603-384-0239.

I declare under penalty of perjury that the foregoing is true and correct. Executed on 29 September 2022.

Sincerely,

Mike Lackey
Senior Vice President
D&D Operations
Energy*Solutions*

Attachments:

1. Responses to Request for Additional Information, Questions 1-16
2. Request for Exemption from 10 CFR 70.24, Criticality Accident Requirements
3. List of Regulatory Commitments

cc: w/Attachments

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NRC Lead Inspector – Three Mile Island Nuclear Station – Unit 2
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THREE MILE ISLAND, UNIT No. 2 – REQUEST FOR ADDITIONAL INFORMATION FOR REQUESTED LICENSING ACTION REGARDING DECOMMISSINING TECHNICAL SPECIFICATIONS EPID: L-2021-LLA-0038

ACCIDENT ANALYSIS:

By letter dated February 19, 2021 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML21057A046), TMI-2 Solutions, LLC (TMI-2 Solutions or licensee) submitted a License Amendment Request (LAR) to remove certain requirements from the TMI-2 Technical Specifications (TS) that restrict activities in the TMI-2 Reactor Building during Post-Defueling Monitored Storage (PDMS). The licensee would like to progress to actively decommissioning the remaining structures, systems, and components that were contaminated in the 1979 accident. Previously, the licensee had evaluated the impacts of a fire in a High Integrity Container (HIC) containing spent ion exchange resins. Subsequently, the licensee determined that the HIC fire scenario was not representative of the activities that would be occurring during decommissioning and submitted supplemental information on January 7, 2022 (ML22013A177). The U.S. Nuclear Regulatory Commission (NRC) staff provided preliminary questions on the information on February 7, 2022 (ML22038A936). The licensee provided a response on April 7, 2022 (ML22101A077), including references and additional analyses on May 8, 2022 (ML22138A302). This request for additional information (RAI) is in response to the latest information provided.

Fire is arguably one of the largest risks at a nuclear facility (U.S. Department of Energy (DOE), 1994). Fire risk is a product of the likelihood of a fire occurring and the consequences if a fire were to occur. Though minor in impact, fires have occurred at nuclear reactors undergoing decommissioning (e.g., Crystal River, Ft. Calhoun, Indian Point). By the introduction of fuel and energy sources combined with the diverse activities that are necessary to complete decommissioning, the frequency of occurrence of fires has been higher during decommissioning than during operations or, in the case of TMI-2, PDMS.

When responding to RAIs, the licensee may identify alternative approaches such as management controls, procedures, calculations, or conditions that will ensure the impacts from potential fires during decommissioning will meet established criteria for protection of human health.

RAI 1 Fractional Airborne Release Factor (ARF)

Comment: Insufficient basis was provided for using the revised ARF of 1.5×10^{-4} based on the 1973 reference.

Basis: The license revised the fractional ARF from a previously used value of 1×10^{-3} to a new value of 1.5×10^{-4}. The revised value was indicated to be more appropriate and is based on information found in NUREG/CR-0130 (1978), *Technology, Safety and Costs of Decommissioning a Reference Pressurized Water Reactor Power Station*, which in turn references Battelle-Pacific Northwest Laboratories (BNWL)-1730 (1973). The data in BNWL-1730 were developed from measurements of burning different types of uranium (dioxide powder, nitrate solution) containing materials (e.g., cardboard, paper, plastic, rubber) in a small enclosure. The fire produced conditions inside the enclosure were very smoky and some material did not burn well suggesting perhaps the oxygen flow was not sufficient. ARF's were measured from 3×10^{-5} to 5×10^{-4}. Wall deposition was cited as being as high as 2.3×10^{-3}.

Though not extensively studied, the importance of the ARF to accident risk analysis has been recognized. The ARF is likely to depend on material type, condition and form of the material, and projected fire magnitude. New information is available for a variety of different materials and conditions, and the new material reflects a broader consideration of materials and conditions. In NUREG-1140 (1988), *A Regulatory Analysis on Emergency Preparedness for Fuel Cycle and Other Radioactive Material Licenses*, a value is provided of 1 x 10^{-3} for uranium (U), plutonium (Pu), americium (Am), and curium (Cm). In NUREG-1940 (2012), *RASCAL 4: Description of Models and Methods*, values for dry process waste, a packed waste fire (solids), a loose waste fire (solids), and high efficiency particulate air (HEPA) filters are provided of 1 x 10^{-3}, 5 x 10^{-4}, 5 x 10^{-2}, and 1 x 10^{-4}, respectively. In the 1994 DOE Handbook compiling various ARFs (DOE-HDBK-3010-94, ML13078A031), a wide variety of measurements are summarized and discussed. Values cited that may be relevant to TMI-2 range from 5 x 10^{-2} for plastics to 5 x 10^{-4} for packaged waste or a burning container – Figure 5-11 is especially informative. Recently, Hubbard et. al (SAND2019-12565J, 2019), measured ARF's for uranium containing materials and various surrogates. For uranium, the ARF was 9.6 x 10^{-4} with a standard deviation of 7.1 x 10^{-4}. For surrogates, the ARF's were about a factor of 100 lower, though the authors expected the surrogate data may have been impacted by precipitation from solution.

The licensee indicated that in the case of TMI-2, the loose contamination is primarily on non-combustible metal and concrete surfaces. The licensee stated that the contamination will not be involved in the fire but could be swept up into it. Fire of non-combustible materials is not of concern aside from the potential for a large fire to volatize cesium (Cs) associated with pore water of the contaminated concrete. The concrete walls of the reactor building basement are one of the largest sources of radioactivity remaining outside of debris contained within the reactor.

From examination of historical photos, some materials located within the reactor building are combustible. In addition, fuel and other combustible materials will be introduced to facilitate decommissioning. A critical assessment of materials present, and appropriate ARF's for those materials, may help support selection of ARF's or help determine if additional controls are necessary for certain materials.

Path Forward: Describe the form and material types that may be subject to a fire with emphasis on combustible materials. Provide additional basis for the ARF selected addressing the more recent observations noted in the basis section. As necessary, revise the ARF to be consistent with the ranges and uncertainties.

RESPONSE

Summary

As noted in the response below, the 1E-3 ARF used in the original calculation was from SAND2019-12565J, 2019, and Battelle-Pacific Northwest Laboratories (BNWL)-1730 (1973) which were based on testing for scenarios that are evaluations whose intended purpose, accident scenarios, and contaminant chemical forms are not representative of the Reactor Building contaminants or Dry Active Waste (DAW) fire scenario applicable to the TMI-2 decommissioning. As noted in the Table below, the 1.5E-4 ARF is within the range applicable to DAW fires with non-combustible powers dispersed in the area and within the combustible DAW materials.

Comment Response

The radioactive material in the combustible waste present is Dry Active Waste (waste bags, and materials, disposable protective clothing, etc.) contaminated with removable contamination present in the Reactor Building. The removable contamination is generally in the form of non-combustible dust particles or powders that get consumed or entrained in the fire. The ARF used is from NUREG/CR-0130 Vol 2 page J-47, "release fraction is assumed to be equal to the release fraction from a contaminated waste fire, or 1.5 x 10-4." This value was chosen based upon a review of other recent references (as described in the table below) for combustion ARFs of particulate, removable contamination shown in **Table 1**. Note that the loose polystyrene ARF of 1E-02 is not considered applicable because it is for "loose polystyrene" and polystyrene resins are not being used to process liquid radioactive waste during the decommissioning. Zeolite, which is a non-combustible mineral similar to rock dust, was used in the clean-up phase to prepare for PDMS and will be used for high activity water processing in the DECON phase. Therefore, the ARF of 1.5E-04 used by NUREG/CR-0130 in Appendix J for a DAW fire is considered to be the appropriate ARF. As seen in **Table 1** below it is also in the range of other ARFs such as those used for similar events.

Table 1 - Summary of Recent NRC Guidance for Airborne Release Fractions (ARF) and Respirable Fractions (RFs) for Powders and Surface Contamination During Combustion (e.g., Thermal Stress)

NUREG-1887 RASCAL 3.0.5 Table 3.11	ARF	RF	Respirable Particle ARF
Table 3.11 Fire Release Fractions by Compound Form of RASCAL 3.0.5 published by the NRC gives a fire release fraction for a non-volatile solid as 0.0001	1.00E-04	(RF NVA_d)	1.00E-04
HEPA Filters High Temperature	1.00E-04	1	1.00E-04
NUREG/CR 6410 Nuclear Fuel Cycle Facility Accident Analysis Handbook Table 3-1	**ARF**	**RF**	
3.3.2.11 Solid, contaminated HEPA filters-Passage of heated air up to 400 °C	1.00E-04	(RF NVA_d)	1.00E-04
3.3.2.12 Solid, contaminated combustible- Powders USDOE 1994, Subsection 5.2.1.1 Packaged waste, burns to self-extinguishment	5.00E-04	(RF NVA_d)	5.00E-04
3.3.2.13 Solid, contaminated combustible USDOE 1994, Subsection 5.2.1.2	1.00E-02	1	1.00E-02
b. Loose polystyrene			
For contaminated combustible materials – Median heated/burned in packages with largely non-contaminated exterior surfaces (e.g., packaged in bags, compact piles, pails, drums), the following values are assessed to be bounding.	8.00E-05	1	8.00E-05

For contaminated combustible materials Bounding heated/burned in packages with largely non-contaminated exterior surfaces (e.g., packaged in bags, compact piles, pails, drums), the following values are assessed to be bounding.	5.00E-04	1	5.00E-04
Uncontained Plastics. The following values apply to burning of unpackaged contaminated combustible plastics.	5.00E-02	1	5.00E-02
Polystyrene: Based upon maximum experimentally determined ARF and RF for a limited set of experiments involving polystyrene contaminated with uranium nitrate hexahydrate (UNH) solution. The value selected is based on rounding upward the maximum value from the data set:	1.00E-02	1	1.00E-02
Contaminated, Noncombustible Solids Bounding values were selected based on reasoned judgment that the suspension of surface contamination (most probably in the form of a sparse population of particles attached to the surface) under thermal stress is bounded by the suspension of non-reactive powders under thermal stress in a flowing airstream (see subsection 4.4.1.1).	**6.00E-03**	**0.01**	6.00E-05
HEPA Filters Thermal Stress	1.00E-04	1	1.00E-04
DOE-HDBK-3010-94 Airborne Release Fractions	**ARF**	**RF**	**Respirable Particle ARF**
For contaminated combustible materials - Median heated/burned in packages with largely non-contaminated exterior surfaces (e.g., packaged in bags, compact piles, pails, drums), the following values are assessed to be bounding.	8.00E-05	1	8.00E-05
For contaminated combustible materials Bounding heated/burned in packages with largely non-contaminated exterior surfaces (e.g., packaged in bags,	5.00E-04	1	5.00E-04

compact piles, pails, drums), the following values are assessed to be bounding.			
Uncontained Plastics. The following values apply to burning of unpackaged contaminated combustible plastics.	5.00E-02	1	5.00E-02
Contaminated, Noncombustible Solids Bounding values were selected based on reasoned judgment that the suspension of surface contamination (most probably in the form of a sparse population of particles attached to the surface) under thermal stress is bounded by the suspension of non-reactive powders under thermal stress in a flowing airstream (see subsection 4.4.1.1).	6.00E-03	0.01	6.00E-05

Given the range of 1E-04 to 6E-05 ARFs for similar fire scenarios, the NUREG/CR-130 Appendix J value of 1.5E-04 is appropriate for use in the overall ARFs for the scenario being evaluated.

With regard to other references mentioned by NRC in the RAI, those references were from evaluations whose intended purpose (accident scenarios and contaminant chemical forms), are not representative of the Reactor Building contaminants or Dry Active Waste (DAW) fire scenarios applicable to the TMI-2 decommissioning.

NUREG-1140 (1988) is not applicable, as it is a study conducted by NRC to determine the possession limit source terms for byproduct material (Part 30 licensees), source material (Part 40 licensees), special nuclear material, (Part 70 licensees), and spent fuel storage (Part 72 licensees). To this end, the NUREG states in Section 2.1.5, “A Discussion of the Conservatism in the Calculations,” the Commission's policy is that, "Emergency planning should be based on realistic assumptions regarding severe accidents... The absorbed dose calculated in this Regulatory Analysis have been conservatively calculated. Exposure to a population near a plant experiencing a severe accident is likely to be far below the absorbed dose in this analysis, probably by an order of magnitude or more.” With regard to the Airborne Release Fractions chosen in the analysis methodology of the NUREG, the NUREG states to use “typical weather” for the specific scenarios evaluated.

Worst-case release fractions are not applicable. The release fractions due to fires (accidents with highest potential release) were determined from experiments designed to maximize releases. In such experiments a finely powdered material is typically placed on top of a large amount of combustible material. Having the entire licensed inventory unenclosed on top of a large quantity of combustible material would be most unusual. Radioactive materials are usually within shielded "pigs" and kept in metal safes or well shielded hot cells or glove boxes. Amounts of combustible materials present are generally kept low. The powdered contaminants in the TMI-2 DAW fire would not be piled on top of the combustible waste but would be mixed within the waste volume or be located on the wall and floor surfaces of the fire as removable contamination.

SAND2019-12565J (2019), "Airborne Release Fractions from Surrogate Nuclear Waste Fires Containing Lanthanide Nitrates and Depleted Uranium Nitrate in 30% Tributyl Phosphate in Kerosene," is also not applicable as it evaluates contaminant forms and combustion scenarios that are not analogous to the radioactive contaminant forms or combustion scenarios applicable to the events under consideration for the TMI-2 decommissioning, since they evaluated the release fractions of plutonium and uranium nitrates associated with the PUREX® chemical separation process in solution in a kerosene fire. The TMI-2 uranium and plutonium oxides forms are insoluble and re not present in combustible fluids.

Battelle-Pacific Northwest Laboratories (BNWL)-1730 (1973) is not applicable because the reference fire is for Uranium and Plutonium being combusted in a gasoline fire from a traffic accident. These scenarios are for outdoor settings with gasoline burning in the wind and are not applicable to the radioactive contaminant form or combustible forms in a Reactor Building Fire scenario. BNWL-1730 documents experiments with uranium dioxide particles or uranium nitrate in solution were deposited on various materials ranging from a smooth metal surface to soil. Gasoline was added to these materials then ignited in ducting at various flow rates to simulate wind. In the burning experiments in which a uranyl nitrate solution was deposited on a stainless steel plate, 11% was made airborne with an air flow of 23 mph. This release was the largest for the burning experiments; however, as much as 24% of uranium dioxide powder was aerodynamically entrained from dry, sandy soil by air at a velocity of 20 mph. Thus, these ARFs are not analogous to the Reactor Building fire scenario being evaluated.

Based on the above, the NUREG/CR-130 Appendix J Value of 1.5E-04 is appropriate for use in the overall ARFs for the scenario being evaluated.

As described in the Response to RAI 2, as decommissioning progresses, combustible material and radioactive material will be relocated to different buildings, elevations, and areas. TMI-2 Solutions has implemented procedures and processes to ensure plant modifications and decommissioning work do not introduce a new limiting scenario and do not invalidate any assumptions or requirements of the Fire Protection Program.

Combustibles and fire hazards were removed or de-energized prior to entry into PDMS to the extent practicable. The TMI-2 Fire Protection Program Evaluation includes a list of combustibles used to calculate the potential fire severity in each Fire Zone as well as a description of work process controls implemented to minimize fire risk. The remaining combustibles primarily consist of cable insulation, miscellaneous hoses/plastics, and small quantities of oil/grease. Control of the introduction of combustible material is described in the response to RAI 2.

References

Smith et. al, "Technology, Safety and Costs of Decommissioning a Reference Pressurized Water Reactor Power Station," NUREG/CR-0130, Vol. 2, Battelle Pacific Northwest Laboratory, Richland WA, 1978.

Mishima, J. and L. C. Schwendiman, "Fractional Airborne Release of Uranium (Representing Plutonium) During the Burning of Contaminated Wastes," BNWL-1730, Battelle Pacific Northwest Laboratories, Richland WA, 1973.

U.S. NRC, "RASCAL 3.0.5: Description of Models and Methods, U.S. Nuclear Regulatory Commission," NUREG/CR-1887, U.S. Nuclear Regulatory Commission, Washington, DC, 2007.

U.S. NRC, "Nuclear Fuel Cycle Facility Accident Analysis Handbook," NUREG/CR-6410, U.S. Nuclear Regulatory Commission, Washington, DC, 1998.

U.S. DOE, “Airborne Release Fractions/Rates and Respirable Fractions for non-Reactor Nuclear Facilities,” DOE-HDBK-3010-94, U.S. Department of Energy, Washington DC, 1994. (ML13078A031)

U.S. NRC, “A Regulatory Analysis on Emergency Preparedness for Fuel Cycle and Other Radioactive Material Licensees,” NUREG-1140, U.S. Nuclear Regulatory Commission, Washington, DC 1991. (ML101460227)

Hubbard et. al, “Airborne Release Fractions from Surrogate Nuclear Waste Fires Containing Lanthanide Nitrates and Depleted Uranium Nitrate in 30% Tributyl Phosphate in Kerosene,” SAND2019-12565J, Sandia National Laboratories, Albuquerque, NM, 2019.

RAI 2 Fire Scenarios

Comment: Fire scenarios evaluated do not encompass the range of operating configurations that may occur during decommissioning.

Basis: The licensee evaluated different fire scenarios. The scenarios were designed to account for fires in different locations and with different structures, systems, and components in place to mitigate the impacts of a hypothetical fire. The locations of the fires were assumed to be the operating platform, A & B D-rings, fuel transfer channel, and the reactor building basement (BAS). The licensee analysis considered three scenarios: a fire while the reactor building purge is running (Case 1), a fire while the reactor building is held at a slightly negative pressure (Case 2), and a fire while the reactor building is under passive ventilation (Case 3). The licensee stated that Case 1 would be the limiting case.

Potential offsite fire impacts are a function of how much material is released and over what duration as well as how long a person is exposed to that release. The release point is also important as the atmospheric dispersion and dilution can vary with release point. The licensee estimated that ground level releases would lead to larger doses (by about a factor of 2) compared to a release at height under similar atmospheric conditions.

In case 2, the licensee assumed that the radioactive material released by the fire would be contained and released slowly over a 14-hour duration. In case 3, it was assumed that release would occur while the reactor building was under passive ventilation. A small fire could result in material passing through the filters (being filtered) while a large fire would trigger a pressure differential resulting in the breather isolation valve closing and sealing off the reactor. The magnitude of the fire was not otherwise discussed.

A large fire should result in the release of more radioactive material because more contaminated material would be consumed. HEPA filters will not perform indefinitely in response to a fire. Filters may clog with soot and debris and fail. The analysis should consider the actions of fire personnel responding to a fire. It is unlikely that the fire impacts will be insensitive to fire magnitude. A primary objective of personnel responding to a fire is to extinguish the fire and to do that fire personnel must be able to see what is happening. A large fire is more likely to have ingress and egress as well as actions taken to increase visibility. The three cases analyzed do not seem to encompass the set of reasonable permutations (e.g., the building purge may be inoperable or deactivated and ingress/egress may lead to ground release) associated with active decommissioning as opposed to PDMS, and Case 1 is not clearly bounding.

Path Forward: Please provide a discussion and analysis of alternative cases that may occur as systems are dismantled and deactivated or discuss management controls that will be used to ensure the limiting case examined (Case 1) is bounding.

RESPONSE

TMI-2 Solutions has implemented management controls discussed below to ensure the limiting case examined is bounding.

The PDMS SAR establishes Case 1 (a fire in the Reactor Building while the purge is running) as the limiting scenario while in PDMS. The Technical Support Document TSD 21-077 (previously provided via TMI2-RA-COR-2022-0007) reviews all currently known scenarios and validates that, based on the existing source term, Case 1 remains the limiting fire scenario as TMI-2 enters DECON and begins active decommissioning.

As stated in TMI2-RA-COR-2022-0007, the most limiting scenario, a Reactor Building fire, is not based on any specific event (e.g., purge inoperable or ingress/egress situations). Its main purpose is to demonstrate that even if the RBEVS was bypassed, the event would not exceed 100 mrem to the maximally-exposed individual. In the event of a fire in areas that contain significant radioactive material outside of closed non-combustibles containers (e.g., the RB), the building will be evacuated until the fire burns out if the fire cannot be suppressed in its incipient stage and the ventilation will be shutdown. Fire fighters will not enter these areas for fire suppression. It is assumed that all equipment in a fire area fails due to a fire. In the event of a fire, regardless of whether the filters fail, the ventilation system will be shut down, which remains bound by Case 1 where the purge remains in operation.

As decommissioning progresses, combustible material and radioactive material will be relocated to different buildings, elevations, and areas. TMI-2 Solutions has implemented procedures and processes to ensure plant modifications and decommissioning work do not introduce a new limiting scenario and do not invalidate any assumptions or requirements of the Fire Protection Program.

For significant radioactive materials available to a fire, programmatic measures include engineering approvals prior to relocating the material to a new fire zone or performing the decommissioning work associated with it. This process utilizes limits from TSD 21-077 to ensure the quantity of radioactive material available to a fire in any fire zone remains within the limits established to ensure that a fire could not result in a radioactive material release which exceeds the limits in the Fire Hazards Analysis and 10 CFR 20.1301 limits. The program's process includes: (1) screening criteria to determine which activities require further evaluation, (2) standards for characterizing radioactive material in terms of the limits, and (3) a formal authorization process for changes to the amount of radioactive material in a fire zone which could be released by a fire.

For combustible materials, programmatic processes are in place to control the introduction of new combustibles or flammables. The processes were developed in accordance with the requirements of NRC Regulatory Guide 1.191 and are similar to the measures implemented by standard operating nuclear power plants. The processes include typical permit approval, combustible free zones, and engineering design reviews to ensure the possibility of a fire is minimized and any changes do not reduce the effectiveness of fire protection for facilities, systems, and equipment that could result in a radiological hazard, taking into account the decommissioning plant conditions and activities.

References

Three Mile Island Nuclear Station, Unit 2 (TMI-2) "Revised Update 14 of Post-Defueling Monitored Storage Safety Analysis Report," TMI2-RA-COR-2021-0014, dated September 29, 2021.

Three Mile Island Nuclear Station, Unit 2 (TMI-2), "License Amendment Request – Three Mile Island, Unit 2, Decommissioning Technical Specifications, Response to Questions," TMI2-RA-COR-2022-0007, Attachment 4, RSCS TSD No. 21-077 Rev 00 "TMI-2 Source Term Limitations and Administrative Controls for the TMI-2 Decommissioning Fire Protection Program from Letter," dated April 8, 2022.

U.S. NRC, "Fire Protection Program for Nuclear Power Plants During Decommissioning and Permanent Shutdown, Rev 1," Regulatory Guide 1.191, U.S. Nuclear Regulatory Commission. Washington, DC, January 2021.

RAI 3 Offsite Dose Calculations

Comment: The offsite dose calculations lack transparency and traceability.

Basis: Offsite doses resulting from a potential fire were described in TMI2-RA-COR-2022-0007, LAR TMI- 2 "GPU Nuclear Calculation 4440-7380-90-017, Revision 4, PDMS Safety Analysis Report (SAR) Section 8.2.5 Fire Analysis Source Terms") of ML22138A302 (May 13, 2022, Attachment 2). The licensee described modifications to previous calculations (revision 3) to account for additional decay and ingrowth, the presence of additional loose contamination, and use of updated dose conversion factors (revision 4). These changes were sufficiently described and appropriate.

Staff were able to verify the dose conversion factors that were used and most other parameters, as well as the calculated decay and ingrowth. However, the approach taken for the amount of source material (inventory) that is released as a result of the fire was not clear. In revision 3 of the analysis, the amount released was calculated as a product of two factors: the amount of material available and the fraction of available material that was released to the air. The amount of fuel elements available (e.g., Pu, Am) was assumed to be 100 percent or a fraction of 1.0. The amount of Cs and Sr available was assumed to be 1 percent or a fraction of 0.01 for a fire in the reactor basement. These were then multiplied by factors of 8×10^{-4} for actinides and apparently 1.5×10^{-3} for Cs and Sr. Staff could only replicate the basement fire dose of 0.889 mrem by using these factors. The impact is the dose for new ARF of 1.5×10^{-4} does not decrease by a factor of 6.67 but instead would be 0.80 mrem for the basement fire (note RAI #1 on the basis for the ARF).

Path Forward: Please verify the combined factors of material available and airborne fraction released in revision 3 of the fire analysis source terms and update the revision 4 analyses as appropriate.

RESPONSE

Conversations between TMI-2 Solutions and the NRC were held regarding RAI 3. The NRC agreed that TMI-2 Solutions can provide the response to RAI 3 at a later date that will be communicated to the NRC independently of this submittal.

RAI 4 Basis for Inventory

Comment: Additional information is necessary to adequately support the inventory of radionuclides assumed in the fire analyses scenarios.

Basis: Fire analyses were previously performed as part of the PDMS safety analyses. The previous analyses assumed most of the transuranic radionuclides remained with or was present in different areas of the plant proportional to how much fuel was estimated to be present. This assumption was appropriate when characterization data was limited. Characterization data has been developed which suggests the inventory assumed for the BAS fire may have been underestimated. Because of high radiation fields in some areas of the plant, characterization data was difficult to obtain and in many cases was limited to total radiation (e.g., R/hr). The reactor building basement was flooded to a depth of approximately 2.6 m (8 ft) during the accident and subsequent recovery activities. Table 1 is the inventory of select radionuclides estimated to be present at the time of the accident (1979) and in 1990. Only about 1 percent (1.3 kg out of 100 kg) of fuel is estimated to remain in the reactor building basement (about 1 percent of the total transuranics shown).

Table 1 Inventory of Select Radionuclides Pre- and Post-Defueling

Radionuclide	Inventory at Accident (1979) Ci	Inventory in 1990 Ci
^{90}Sr	760,000	2,400
^{137}Cs	820,000	43,000
^{241}Pu	160,000	950
^{239}Pu	9,000	90
^{241}Am	19	22

In the current submittal for the analysis of potential fires during decommissioning, the license revised the inventory (using the same starting point/inventory) to account for additional radioactive decay and ingrowth as well as the presence of additional surface contamination that was identified after the previous analyses. Whereas radioactive decay decreased the amount of ^{137}Cs and ^{90}Sr as well as many transuranics, the amount of some radionuclides increased. For example, ^{241}Pu has a 14.4-year half-life and decays into ^{241}Am. The amount of ^{241}Am present was estimated to have increased significantly thereby offsetting the decay of other radionuclides.

In GEND-INF-011-Vol3 (1983), samples of the liquid and sediment/debris in the reactor building basement were obtained and characterized. The measured plutonium in the solids averaged 3.4 mg/g. In other documents, various estimates of solids in the basement were provided and about half of the solids was reported as being removed. The remaining solids are on the order of 3000 kg (6600 lbs). Approximately 90 percent of the plutonium would be expected to be ^{241}Pu based on the inventory assigned by the licensee in the accident analysis. The specific activity of ^{241}Pu is approximately 103.35 Ci/g. The measurement data corresponds toapproximately 1,000 Ci of ^{241}Pu, which is significantly larger than the approximate 10 Ci (1% of the 950 Ci shown in Table 1) included in the analyses.

The source term (inventory) used in the BAS fire scenario (Attachment 2, TMI2-RA-COR-2022-0007, LAR TMI- 2 "GPU Nuclear Calculation 4440-7380-90-017, Revision 4, PDMS SAR Section 8.2.5 Fire Analysis Source Terms", ML22138A302) used the inventory from previous analyses as the starting point.

The enrichment of the fuel at the time of the accident is not precisely known (Cragnolino, 1997). A value of 2.57 percent in U-235 was used (the core average). A higher enrichment may be conservative with respect to criticality analysis, but a lower enrichment is conservative with respect to other accident analyses because of the increase in ^{241}Pu which decays into ^{241}Am. The inventory used in the fire accident analysis is apparently based on the 2.57 percent value.

Path Forward: Please address the apparent discrepancy between the characterization data (concentrations) provided in GEND-INF-011-Vol3 and the assumed basement inventory applied in the fire analyses. Please address the assumed fuel enrichment and how it yields a conservative starting inventory for fire accident analyses.

RESPONSE

Summary
GEND-INF-011-Vol3 (1983) (Reference 1) samples provided an initial characterization of radionuclide inventory in the RB basement with a high degree of uncertainty. Since that time, additional samples of basement materials, shipments of removed basement sediment, visual inspections, and specific fuel characterization have been performed and have reduced uncertainty of the initial characterization.

GEND-INF-011 results were included with additional samples of basement sediment and water and summarized in section 2.10 of Reference 2, "Three Mile Island Nuclear Station Unit 2 (TMI-2), Reactor Building Characterization." With the additional samples included in the TPO/TMI-125-R2 report, the average plutonium concentrations from all samples were reduced from 3.4 to 2.1 microcurie/gram. These results showed the conclusions in GEND-INF-011 (Reference 1) were conservative.

Comment Response
In late June of 2021, focused drone inspections of the TMI-2 Reactor Building basement were conducted and noted that the overall condition of the basement floor areas were generally clear with no appreciable sedimentation observed. Areas that were not accessed by the desludging program had light coatings of dried sediment. GEND-INF-011 assumed solids distribution was homogenous on the basement floor and the drone inspections showed this assumption to be conservative.

The TMI-2 Post-Defueling Survey Report for the Reactor Building Basement (Estimate of Record) (ML20248B795) (Reference 3) noted "although many samples have been taken in the RB basement from August 28, 1979, forward, most are suspect due to insufficient volumes to provide a representative sample." Because of uncertainty of previous samples, the estimate of record was based on gamma spectroscopy measurements of multiple basement locations. The gamma spectroscopy correlated gamma energy peaks of Ce144 to mass of UO_2; a common method used to characterize remaining fuel material after defueling. The resulting Estimate of Record of 1.3 kg of UO_2 represents the most accurate and up to date estimate of uranium and plutonium present in the basement and shows the results in GEND-INF-011 (Reference 1) were conservative.

The "Validation of Estimated Quantities of Plutonium Remaining in TMI Unit 2" report (Reference 4) provided a summary of plutonium species based on the results of an ORIGEN burnup calculation reproduced in the report which concluded there is 0.0264 grams of Pu241 present with the 1.3kg of UO_2 in the RB basement. The 0.0264 grams of Pu241 (at 102.3 Ci/gram per Rad Pro Calculator) equates to 2.7 Ci of Pu241 activity in the RB basement, showing the results in GEND-INF-011 (Reference 1) were conservative and the approximately 10 Ci of Pu241 used in the fire analyses is also conservative.

The initial homogenized enrichment of U235 in the initial core loading was approximately 2.54% (60 assemblies loaded at 2.96%, 61 assemblies loaded at 2.64%, and 56 assemblies loaded at 1.98% = 2.54% total). The fuel was burned from initial criticality until time of the accident and an ORIGEN burnup calculation was performed to calculate the total radionuclide inventory of TMI2 following the accident. The analytical results strongly correlated to sample results (GEND-INF-075) (Reference 5). The ORIGEN calculated radionuclide inventory has been decay-corrected and used for multiple purposes, including as source data for the Final Programmatic Environmental Impact Statement related to decontamination and disposal of radioactive wastes resulting from March 28,1979, accident Three Mile Island Nuclear Station, Unit 2 (Reference 6), the Post Defueling Monitored Storage Safety Analysis Report from which the fire dose calculation draws data, and the safe fuel mass limit calculation (Reference 7). Regardless of the delta between the starting core enrichment (2.54%) and the enrichment at the time of the accident (approximately 2.24%), the radionuclide activity source data referenced in various analyses is the same, correlates to sample results, and is the best available data as input for both the safe fuel mass limit and fire accident analyses.

References

1. Cox et. al, "Reactor Building Basement Radionuclide and Source Distribution Studies," GEND-INF-011-Vol 3, US DOE, 1983.

2. Three Mile Island Nuclear Station Unit 2 (TMI-2), Reactor Building Characterization, TPO/TMI-125, Revision 2, 1989.

3. "SNM Accountability," Three Mile Island Nuclear Station Unit 2 (TMI-2) letter 4410-89-L-0097, dated September 22, 1989 – Contents include "TMI-2 Post-Defueling Survey Report for the Reactor Building Basement (Estimate of Record)" (ML20248B795).

4. McKamey, "Validation of Estimated Quantities of Plutonium Remaining in TMI Unit 2," June 2019.

5. Akers et al "TMI-2 Core Debris Grab Samples –Examination and Analysis" GEND-INF-075 – Part-2, US DOE, 1986.

6. "Programmatic Environmental Impact Statement related to decontamination and disposal of radioactive wastes resulting from March 28, 1979 accident Three Mile Island Nuclear Station Unit 2, Final Supplement Dealing with Post-Defueling Monitored Storage and Subsequent Cleanup" NUREG-0683 Supplement No. 3, US NRC, August 1989.

7. TMI2-RA-COR-2022-0008, "Supplemental Information to License Amendment Request- Three Mile Island, Unit 2, Decommissioning Technical Specifications," April 7, 2022 contents include "Determination of the Safe Fuel Mass Limit for Decommissioning TMI-2," TMI2-EN-RPT-0001, Rev 1, 2022.

RAI 5 Buildup of Radiolytic Gas

Comment: The licensee did not provide sufficient information of radiolytic gases (primarily Hydrogen (H_2)) that could pose an explosion hazard.

Basis: Interaction of radiation with water or other materials can result in the production of radiolytic gases, primarily hydrogen. In sufficient concentrations and with oxygen present, hydrogen is flammable. Through operation of the Submerged Demineralizer System and packaging of the generated waste for disposal, it was observed that TMI-2 debris could generate H_2 in short-term storage that could reach Lower Flammability Limits (LFL). Licensing of the dry cask storage system in Idaho for TMI-2 debris applied multiple controls and systems in order to prevent buildup of H_2 gas to the LFL (ML18296A527). Canisters were vacuumed dried prior to storage and the systems included a HEPA filter to vent hydrogen. Monitoring of hydrogen levels is performed (ML19259A017) and observed hydrogen levels have been around 0.04 percent where the LFL with oxygen present is 5 percent - the venting has been very effective but hydrogen generation is continual.

Though a large fraction of the radioactivity has been removed from the TMI-2 systems, high radiation fields remain. The deactivated reactor systems have dead end and closed portions (e.g., high points in unused piping) where H_2 gas could collect. Significant moisture is present in many systems and components. Decades have passed since the accident where H_2 could be generated.

Path Forward: Please demonstrate the impacts of a hydrogen explosion initiated by decommissioning activities is bound by the fire scenarios evaluated, or please describe management controls and procedures such as circulation of air and monitoring for flammable gases that will be used prior to cutting or introduction of flame to systems being decommissioned.

RESPONSE

In preparation for entry into PDMS, the plant systems were vented, drained, and the remaining water volumes were processed for disposal. As a result, there are no significant water volumes remaining in TMI-2. However, TMI-2 Solutions recognizes there may be small, localized hydrogen gas pockets remaining within the highly contaminated portions of plant systems and components that could lead to hydrogen production. TMI-2 Solutions will establish a work planning instruction which will evaluate specific hydrogen concerns relevant to a given scope of work and include appropriate hydrogen mitigation measures appropriate for that work.

RAI 6 Dust Explosion and Exothermic Reaction Hazard

Comment: The licensee did not address the potential for dust explosions or discuss management controls that would be used to ensure a dust explosion will not occur.

Basis: In typical decommissioning of a reactor, contamination is primarily present in a fixed or embedded form that is not easily dispersed. During the accident at TMI-2, aggressive conditions occurred. Fuel, cladding, and other materials were melted and distributed, primarily within the reactor pressure vessel, but with some material (estimates range from 0.5 to 1 percent) distributed outside the pressure vessel. Contaminated coolant leaked and flowed to different areas of the plant including the reactor basement. During the accident approximately 40 percent of the fuel melted which would have produced approximately 13 tons of Zirconium (Zr). If approximately 1 percent of the melted material exited the pressure vessel, that means approximately 130 kg (290 lbs) was deposited throughout the systems and about 1 kg (2 lbs) would be < 5 um powder based on measured particle size distributions.

Metallic Zr with sufficient surface area is pyrophoric and numerous accidents have occurred (Atomic Energy Commission (AEC), 1956). Many other powders, especially powdered metals such as aluminum, magnesium, sodium, lithium, potassium, and titanium, can be highly reactive. The minimum explosible concentration (MEC) is dependent on the form of the material and the particle size. For iron dust, which is generally viewed as being somewhat inert, the MEC is on the order of 100 to 200 g/m^3 for 4 mm particles (Cashdollar, 2000). Dust explosions can occur when the "fire triangle" is achieved: a fuel, an oxidizer (usually air), and a heat or ignition source is present. It is expected that most of the material deposited outside of the pressure vessel would have been oxidized during the event. However, characterization data is limited in some areas due to high radiation fields.

Pyrophoricity studies were completed prior to defueling and pyrophoricity of debris was not observed (Clark et al., 1984). However, those studies evaluated debris from inside the pressure vessel and focused on larger particle sizes of the core debris (lower specific surface area). Because the debris deposited outside the reactor vessel has smaller particle sizes, the surface area to volume ratio will be higher.

Path Forward: Please describe controls that will be used to minimize the risk of dust explosions or other exothermic reactions during decommissioning. Please summarize characterization data and other studies that demonstrate that reactive dusts are not present in sufficient quantities to present an explosion or fire hazard.

RESPONSE

Summary

Based on historical data the following conclusions can be reached:

- If fines were released from the reactor vessel to the Reactor Building Basement via pressurizer relief valves, they were exposed to oxygen in the water for several years after the accident and then to the atmosphere after water was removed and are thoroughly oxidized.
- If any unoxidized fines exist, they would be mixed with river water sediment, concrete dust, and dirt which would act as a diluent and would minimize any potential for ignition and propagation.
- Pyrophoricity of TMI-2 sediment was not a safety concern during cleanup operations and a further 30 years of oxidation has occurred.

Thus, reactive dusts are not present in sufficient quantities to present an explosion or fire hazard. Therefore, specific controls to prevent dust explosions are not required.

Comment Response
The study referenced above, GEND-INF-044, "TMI-2 LEADSCREW DEBRIS PYROPHORICITY STUDY" (i.e. (Clark et al., 1984)) represents only a portion of the material studied. GEND-043 "TMI-2 PYROPHORICITY STUDIES" tested several samples of debris material. Samples from the leadscrew and plenum cover represent material that was displaced from the TMI-2 core and would be similar to material displaced elsewhere. When subjected to testing neither of these samples indicated a pyrophoric tendency.

Theoretical pyrophoricity was also evaluated. As discussed in the GPU Nuclear "Safety Evaluation Report for Early Defueling of the TMI-2 Reactor Vessel" dated May 20, 1985. (ML20127L978)

> The concern over pyrophoric materials is presently focused on the potential for metallic zircalloy and zirconium hydride fines existing in the dewatered canisters. The manner in which the fuel deteriorated during the accident makes the presence of these species, in a pyrophoric form, highly unlikely in the present configuration of the core rubble bed. Zircalloy, being a ductile metal even after irradiation, would not break up into small particles under the high temperature steam environment of the TMI-2 accident. Rather, the material oxidizes, and it is the oxide which breaks up as a consequence of thermal shock or abrasion. However, during the early defueling process, it is possible, as a result of cutting operations, that fresh (i.e., unoxidized) metal surfaces, including small chips and fines, could be created.
> ...
>
> Considerable analyses have been conducted since the pyrophoric concern was initially raised and are summarized in (TPO/TMI-127 "Technical Plan for Pyrophoricity," December 1984). The analyses indicate that three conditions must exist to initiate and maintain a pyrophoric reaction:
>
> (1) The pyrophoric material must have a high surface to volume ratio of the nature of powder. Experience indicates that moist zirconium fines of less than 10 microns will burn. However, existing analysis of core debris indicates only about 1.5 % of the particulate matter is less than 45 microns. The early defueling activities are not likely to generate significant additional quantities of fines in the size range of concern.
>
> (2) The pyrophoric material must exist in an oxygen depleted environment and then be suddenly exposed to oxygen. The surface of the core pyrophoric material has been exposed to oxygen in the water since the accident. Thus, oxidation that has already occurred would limit a pyrophoric reaction to material that is freshly exposed. The early defueling process is not likely to expose significant quantities of debris in the size range specified In item (1) above. Any additional exposure of pyrophoric material due to the early defueling activities would initially be underwater, where oxidation would again occur at some rate.
>
> (3) The oxidation rate must exceed the heat transfer rate to the surrounding environment. The oxidized debris that will be mixed with any pyrophoric material acts as a diluent and minimizes the potential for ignition and propagation.

The NRC in approving the GPU Nuclear Safety Evaluation via THREE MILE ISLAND PROGRAM OFFICE SAFETY EVALUATION OF EARLY DEFUELING OF THE TMI-2 REACTOR VESSEL, dated November 12, 1985 (ML20136B809) stated in part:

> ... Despite the fact that some debris may be exposed to oxygen, the potential for a pyrophoric reaction is still very small for the following reasons: significant quantities of potentially pyrophoric material (zirconium hydride) are not postulated to exist in sizes small enough to spontaneously ignite (10 microns); unoxidized surfaces must be newly exposed to an oxygen environment to undergo a pyrophoric reaction and any new surfaces exposed in the course of defueling will be in contact with water, thus oxidizing before canister dewatering occurs; and the rate of oxidation must exceed the heat transfer rate of the material for ignition to occur. We conclude that the potential for a pyrophoric event during early defueling activities is extremely unlikely...

This basic conclusion in various forms was continued throughout the cleanup process. With respect to defueling the Reactor Vessel. Also, of note are the GPU Nuclear "Sediment Transfer and Processing Operations Safety Evaluation Report" dated March 18, 1986 (ML20140B692) and the NRC "Sediment Transfer and Processing Operations Safety Evaluation Report" dated September 25, 1986 (ML20210P148). This Safety Evaluation approved the removal of sediment from the Reactor, Auxiliary and Fuel Handling Buildings. As stated in the GPU Nuclear Safety Evaluation Report:

> As a result of the 1979 accident, radioactive water and core debris particles were released to the Reactor Building and AFHB In various tanks, sumps, and on the reactor building basement floor. Consequently, radioactive sediment Is located in these areas which consists primarily of river water sediment, concrete dust, and dirt.

Pyrophoricity was not an accident of concern with respect to this process.

Based on the above the following conclusions can be reached:
- If fines were released from the reactor vessel to the Reactor Building Basement via pressurizer relief valves, they were exposed to oxygen in the water for several years after the accident and then to the atmosphere after water was removed and are thoroughly oxidized.
- If any unoxidized fines exist, they would be mixed with river water sediment, concrete dust, and dirt which would act as a diluent and would minimize any potential for ignition and propagation.
- Pyrophoricity of TMI-2 sediment was not a safety concern during cleanup operations and a further 30 years of oxidation has occurred.

References

Clark et al, "TMI-2 Leadscrew Debris Pyrophoricity Study," GEND-INF-044, Pacific Northwest Laboratory, Richland WA, 1984.

Baston et al, "TMI-2 Pyrophoricity Studies." GEND-043, EG&G Idaho, Inc., Idaho Falls, ID, 1984

Three Mile Island Nuclear Station Unit 2 (TMI-2), "Safety Evaluation Report for Early Defueling of the TMI-2 Reactor Vessel" dated May 20, 1985. (ML20127L978)

Three Mile Island Nuclear Station Unit 2 (TMI-2), TPO/TMI-127 "Technical Plan for Pyrophoricity", December 1984.

US NRC, Three Mile Island Program Office, "Safety Evaluation of Early Defueling of the TMI-2 Reactor Vessel", dated November 12, 1985. (ML20136B809)

Three Mile Island Nuclear Station Unit 2 (TMI-2), "Sediment Transfer and Processing Operations Safety Evaluation Report" dated March 18, 1986. (ML20140B692)

US NRC, Three Mile Island Program Office, "Sediment Transfer and Processing Operations Safety Evaluation Report" dated September 25, 1986. (ML20210P148)

RAI 7 Cork Seams

Comment: A potential fire in the contaminated cork seams and associated materials was not addressed.

Basis: The TMI-2 cork seam is a construction joint between the various major facility structures. During the 1979 accident, the cork seam was immersed in contaminated water. The cork and associated polyurethane sheet material are potential fuel sources, that when exposed during decommissioning, could become a source of contaminated fuel consumed in a fire. The release pathways may be more limiting than the evaluated fire scenarios.

Path Forward: Please describe controls that will be used to prevent a fire in the cork seams and associated materials during decommissioning or complete analyses to demonstrate that the evaluated fire scenarios, subject to the technical comments provided in the RAI, are more limiting.

RESPONSE

The response provided above for RAI 2 also applies to RAI 7, as RAI 2 is regarding all other possible fire scenarios and RAI 7 is regarding a fire scenario specific to the contaminated cork seam. As a result, the same processes for control of combustible materials and radioactive material available to a fire apply to the cork seam.

The cork seam construction joint consists of combustible materials (polysulfide sealant, polyurethane foam, and the cork itself, but the following measures minimize the possibility of an adverse fire involving the cork seam: (1) the work process controls identified in the response to RAI 2, (2) the cork seam itself is partially saturated with water, (3) the cork seam is only present in the basement elevation, which has concrete walls and ceilings as barriers, (4) the majority of the cork seam location is isolated from the major decommissioning impacted areas (e.g., inside Locked High Radiation Area cubicles in the Auxiliary Building, etc.), (5) a portion of the cork seam is in an area with ventilation that exhausts through HEPA filtration, (6) of the total volume of cork seam, only a narrow ~1 inch width is exposed and available to a potential fire, and (7) there are presently no major combustibles within the areas containing the cork seam.

TECHNICAL SPECIFICATIONS

RAI 8 Annual Effluent Monitoring Report

In TMI-2 Solutions' application, as supplemented, TMI-2 Solutions proposed deletion of this Technical Specification below and relocation to the Decommissioning Quality Assurance Plan:

> ANNUAL RADIOACTIVE EFFLUENT
> RELEASE REPORT
>
> "6.8.1.2 The Annual Radiological Effluent Release Report covering the operation of the ~~unit~~ facility during the previous calendar year shall be submitted before May 1 each year. The report shall include a summary of the quantities of radioactive liquid and gaseous effluents and solid waste released from the facility. The material provided shall be (1) consistent with the objectives outlined in the ODCM and (2) in conformance with 10 CFR 50.36a and Section IV. 8.1 of Appendix I to 10 CFR Part 50."

Comment: The licensee did not address the requirement that the annual effluent monitoring reporting is required by regulation to be in the technical specifications.

Basis: As TMI-2 Solutions holds a part 50 license, then Title 10 of the *Code of Federal Regulations* (10 CFR) 50.36a(a)(2), "Technical specifications on effluents from nuclear power reactors," continues to require TMI-2's TS to contain this TS. This is because 50.36a states:

a) [E]ach licensee of a nuclear power reactor ... will include technical specifications that ... require that:

   (1) ...

   (2) Each holder of an operating license ... shall submit a report to the Commission annually that specifies the quantity of each of the principal radionuclides released to unrestricted areas in liquid and in gaseous effluents during the previous 12 months, including any other information as may be required by the Commission to estimate maximum potential annual radiation doses to the public resulting from effluent releases. The report must be submitted as specified in § 50.4, and the time between submission of the reports must be no longer than 12 months. If quantities of radioactive materials released during the reporting period are significantly above design objectives, the report must cover this specifically. On the basis of these reports and any additional information the Commission may obtain from the licensee or others, the Commission may require the licensee to take action as the Commission deems appropriate.

Path Forward: Therefore, granting TMI-2 Solutions LAR (for removal of TS Section 6.8.1.2) would cause the TS to cease meeting 50.36a(a)(2); the license may only be amended in the requested fashion only if the licensee is first exempted from 50.36(a)(2).

Alternatively, TMI-2 Solutions may supplement its application to request including TS Section 6.8.1.2 in its TS for the staff's consideration in its review of the February 21, 2021 amendment application, as amended. TMI-2 Solutions should include a markup of the proposed TS change, if it decides to pursue this option.

RESPONSE

TMI-2 Solutions will supplement its License Amendment Request (LAR) to retain Technical Specification Section 6.8.1.2 in its Technical Specifications. An amended LAR will be submitted under separate cover which will include a markup of the proposed Technical Specification change.

Reference

Three Mile Island Nuclear Station Unit 2 (TMI-2), "License Amendment Request – Three Mile Island, Unit 2, Decommissioning Technical Specifications," TMI2-RA-COR-2021-0002, dated February 19, 2021 (ML21057A046)

SAFE FUEL MASS LIMITS

RAI 9 Criticality

Comment: It is not clear in the application how the requirements of 10 CFR 50.68(a), "Criticality accident requirements," are satisfied.

Basis: Paragraph a to 10 CFR 50.68 states that the applicant shall comply with the requirements of 10 CFR 70.24, "Criticality accident requirements," or meet certain alternative requirements, as described in 10 CFR 50.68(b), in lieu of maintaining a criticality accident alarm system (CAAS) as described in 10 CFR 70.24.

10 CFR 70.24(a) requires, in part, that each licensee authorized to possess special nuclear material (SNM) in a quantity exceeding 700 grams of contained uranium-235 (U-235), 520 grams of U-233, 450 grams of plutonium, 1.5 kilograms of contained U-235 if no uranium enriched to more than 4 wt. percent U-235 is present, or 450 grams of any combination thereof, maintain in each area in which such licensed SNM is handled, used, or stored, a CAAS.

Attachment 1 to TMI-RA-COR-2022-0001, "Supplemental Information to License Amendment Request, Three Mile Island, Unit 2, NRC Possession Only License No. DPR-73, Response to Questions on [Safe Fuel Mass Limit] Analysis," states that the use of a traditional CAAS is not planned due to the low likelihood of inadvertent criticality. However, it is not clear how the requirements of 10 CFR 50.68(a) are satisfied.

Path Forward: Provide information that demonstrates compliance with 10 CFR 50.68 or request an exemption from the requirements of 10 CFR 50.68 in accordance with 10 CFR 50.12, "Specific exemptions."

RESPONSE

TMI-2 has a current exemption under the PDMS License (Reference 1) associated with the requirements of 10 CFR 70.24 as approved on June 15, 1992. As noted in the exemption:

> "...it is appropriate to request an exemption from 10 CFR 70.24 if an evaluation determines that a potential for criticality does not exist, as for example where the quantities or form of special nuclear material make criticality practically impossible or where geometric spacing is used to preclude criticality"

This exemption was granted on the basis of both an evaluation on the potential for criticality and geometric spacing. Specifically, within the PDMS condition, the NRC determined, as described in Inspection Report 50-320/90-03 dated June 14, 1990 (Reference 2) that the safe fuel mass limit (SFML) was 93 kg UO_2. An administrative SFML applied a 25% conservatism at 70 kg and further administrative controls were applied on geometric separation distances. At the time of the 10 CFR 70.24 exemption, the defueling activities had concluded and ~99% of the fuel material had been removed from the core leaving a balance of 1097 kg UO_2 of fuel bearing material (described in the PDMS SAR). Because the balance of fuel bearing material remaining on site was greater than the SFML at the time, additional administrative controls (i.e., geometric controls) were necessary to preclude criticality.

TMI-2 Solutions considers that an exemption to 10 CFR 70.24 for a criticality monitoring system to be appropriate under the DECON licensing basis because TMI2-RA-COR-2022-0008, "Supplemental Information to License Amendment Request- Three Mile Island, Unit 2, Decommissioning Technical Specifications" provides a calculation which shows the SFML associated with remaining fuel bearing material at TMI-2 is 1361 kg UO_2. That SFML is 24% higher than the 1097 kg UO_2 estimate of record for remaining fuel bearing material at TMI-2 which analytically precludes a criticality accident at TMI-2. The 1361 kg UO_2 SFML result represents a significant improvement over the 1990 SFML calculation result. This improvement was achieved by taking credit for impurities and actual enrichment based on the results of physical samples taken during the defueling effort.

Administrative controls for geometric spacing are not necessary to further preclude a criticality accident because there is not enough kg UO_2 at TMI-2 to assemble an optimal critical configuration. However, as part of its Fuel Bearing Material Management Program TMI-2 Solutions will be implementing local administrative controls for the purpose of defense in depth on the activities which will handle the highest quantities of fuel bearing material (e.g., segmenting the reactor vessel internals which represent 925 kg UO_2 or 68% of the SFML). These defense in depth controls will include control on the physical location of segmentation equipment and limiting the number of waste receptacles (i.e., physical manifestations of controls on geometric spacing). Based on a current evaluation which shows the DECON SFML to be higher than the existing UO_2 inventory at TMI-2, this RAI submittal includes TMI-2's request for exemption from the criticality monitoring requirement of 10 CFR 70.24 under the DECON condition in Attachment 2.

References

US NRC "Exemption From 10 CFR 70.42 Regarding Criticality Accident Requirements at Three Mile Island Unit 2 (TAC No. M65512)" Dated June 15, 1992. (ML20210D723)

NRC Inspection Report 50-320/90-03, E. C. Wenzinger to R. L. long, dated June 14, 1990. (ML20043G083)

Three Mile Island Nuclear Station, Unit 2 (TMI-2) "Revised Update 14 of Post-Defueling Monitored Storage Safety Analysis Report," TMI2-RA-COR-2021-0014, dated September 29, 2021.

Three Mile Island Nuclear Station, Unit 2 (TMI-2), "Supplemental Information to License Amendment Request- Three Mile Island, Unit 2, Decommissioning Technical Specifications," TMI2-RA-COR-2022-0008, dated April 7, 2022. (ML22101A077)

MATERIAL CONTROL AND ACCOUNTING (MC&A)

RAI 10 Accounting for Debris Material

Comment: It is not clear in the LAR how TMI-2 Solutions plans to control and account for Debris Material throughout decommissioning.

Basis: Debris Material must be controlled and accounted for at all times during decommissioning because the Debris Material contains large quantities of SNM, including uranium-235 and plutonium. Once TMI-2 has entered DECON, the applicant has stated that SNM will be retrieved, aggregated, and placed into dry cask storage using various shapes and sizes of containers to place into a basket and canister. To minimize aggregating the remaining SNM, the core debris will be generally packaged and loaded as it is retrieved. These canisters will then be transferred to the expanded Independent Spent Fuel Storage Installation (ISFSI) inside the Three Mile Island Station, Unit No. 1 ("TMI-1"), ISFSI fence to store the canisters after TMI-1 completes their spent fuel transfer campaign to the ISFSI. In addition, estimates of the quantities and form of SNM at TMI-2 provided by the applicant indicate that the site may need more detailed plans for material control and accounting during decommissioning, compared to sites where SNM is generally restricted to undamaged spent fuel assemblies. 10 CFR Part 74, "Material Control and Accounting of Special Nuclear Material," establishes requirements for the control and accounting of SNM at fixed sites and for documenting the transfer of SNM. General reporting requirements as well as specific requirements for certain licensees possessing SNM of low strategic significance, special nuclear material of moderate strategic significance, and formula quantities of strategic special nuclear material are included.

Path Forward: Describe how TMI-2 Solutions will control and account for Debris Material being removed from the Reactor Building to the Three Mile Island ISFSI throughout the decommissioning process in order to meet the applicable requirements of 10 CFR Part 74. Describe TMI-2 plans to refine current rough estimates of radionuclide content in Debris Material in existing reports and provide more accurate information on quantities of SNM as materials are packaged and removed.

RESPONSE

Material Control and Accountability Program Description:
TMI-2 Solutions has developed a program to ensure proper accounting of SNM is conducted throughout the decommissioning process, meeting the applicable requirements of 10 CFR 74, with applicable exemptions. Current rough estimates of radionuclide content will be refined using LLRW characterizations and the site final status survey as explained below.

Overview of TMI-2 NMCA Processes
SNM is a constituent of Fuel Bearing Material and is dispersed as shown in the discrete quantities of UO_2 identified in Tables 4.3-1 and 4.3-2 of the PDMS SAR (Reference 1). Each of these quantities of fuel was estimated during post defueling survey reports and, together with nine discrete and packaged items turned over from TMI-1, represent the material balance and starting inventory for the TMI-2 Decommissioning Project reported annually (Reference 2).

The accident at TMI-2 caused a significant portion of the fuel to derange, melt, oxidize, and combine with core materials and the total amount of post-accident material was estimated to be 133,000 Kgs (~293,000 pounds) (Reference 3).

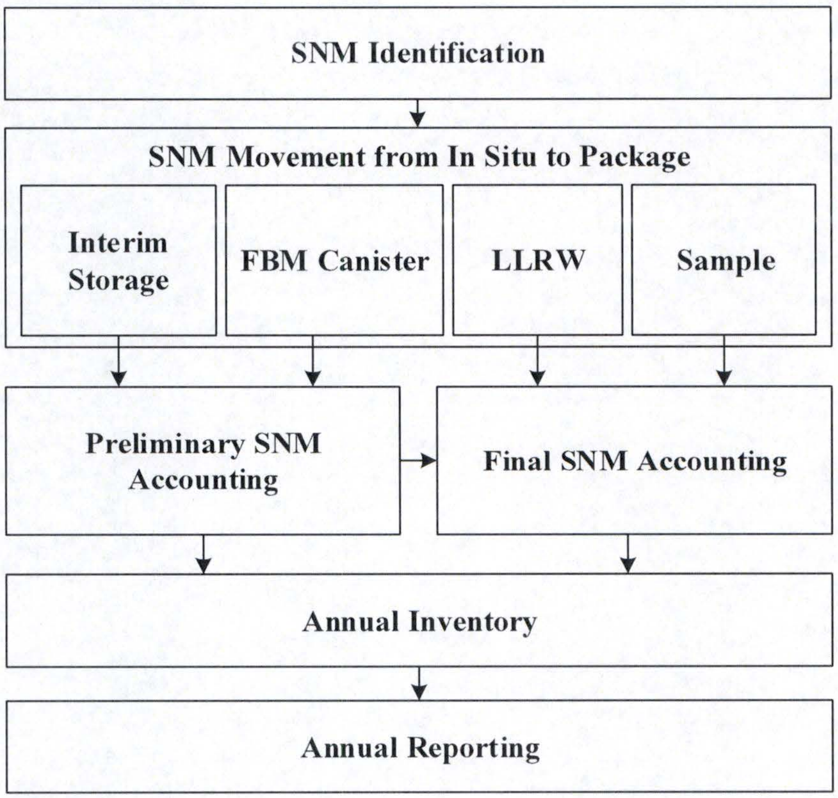
TMI-2 was exempted from multiple SNM Control and Accountability requirements associated with transfer and disposal, including physical inventory (Reference 4). In lieu of the requirements, TMI-2 provided information describing the physical contents of the shipments made to DOE.

Following completion of damaged fuel removal, post-defueling characterization surveys identified 1097 kg of UO_2 remaining in various locations in TMI-2 with a total uncertainty of +/- 40% (References 5-12). The capability to significantly reduce the 40% uncertainty would require characterizing the collected fuel debris in each container using sophisticated hot cell and laboratory facilities with the means to homogenize, sample, weigh, and analyze the contents of each canister. Such facilities did not (and do not) exist at TMI-2. The results of the post defueling survey reports were reviewed and approved by the NRC in Reference 13.

TMI-2 NMCA Process

Figure 1 provides a high-level flow chart for MC&A requirements from the existing configuration through the decommissioning project.

Figure 1



SNM Identification:

- Each discrete location containing fuel bearing material to determine optimal decommissioning sequences using engineering assessments created per written procedures. The engineering assessments integrate inputs of contemporaneous regulatory waste classification and shipping requirements with known historical reports and data. The engineering assessments generate operational assumptions and constraints, cut and packaging plans, expected waste packages, waste classification, estimated SNM in waste packages and Dry Storage Canisters, evaluate 10 CFR 37 radioisotopes of concern in resulting packages, evaluate whether confirmatory radiological measurements are required, and describe the approach for characterization of low-level radioactive waste (LLRW) packages.

SNM Movement from In Situ to a Package:

- The starting in situ physical condition of SNM as a constituent of Fuel Bearing Material (FBM) is that SNM has been best-estimated using state of the art techniques, many of which were developed specifically for conditions at TMI-2 in post defueling survey reports (References 5-12). Uncertainties associated with individual discrete estimates vary between 17-104% and weighted average to a total of +/- 40% uncertainty. Physical inventories have not occurred since the post defueling surveys were completed because they were exempted (Reference 4).

- In-situ physical configuration of FBM varies and will generally be one of: films or fines plated onto previous water to metal interfaces, loose debris or gravel-like particulate of varying size, artifacts which are tightly adhered to larger plant components, or captured in varying types of filter media.

- Physical inventory of SNM as a constituent of FBM in situ is not required under the current exemption (Reference 4).

- SNM movement from its in-situ condition to a confined package shall be to one of:

    - Interim Storage: FBM which is planned to be disposed in a Dry Storage Canister and is removed from its in-situ configuration prior to availability of a Dry Storage Canister shall be placed in an approved interim storage location per written procedure.

    - FBM Canister: FBM which will be removed from its in-situ configuration and directly placed into Dry Storage Canisters per written procedures.

    - LLRW: It is anticipated much of the waste produced at TMI-2 will contain trace amounts of SNM. LLRW packages which have been characterized for waste class, shipping criteria, and disposal site waste acceptance criteria and do not contain more than 1 gram of SNM do not require control and accountability actions. LLRW waste packaging procedures which contain more than 1 gram of SNM will require material accountability and control actions per written procedures.

    - Sample: It is anticipated most samples produced at TMI-2 and sent off site for analysis will contain trace amounts of SNM. While samples generally expected to contain very small total quantities of materials, any sample which contains more than 1 gram of SNM will require material control and accountability actions. Samples which do not contain more than 1 gram of SNM will not require control and accountability actions.

Preliminary SNM Accounting:

- SNM as a constituent of FBM which has been either placed in interim storage containers or into Dry Storage Canisters will have preliminary SNM accounting performed per a written procedure to analytically determine SNM content based on the engineering assessment process. This will remain preliminary until after the final status survey for the decommissioning project is complete.

Final SNM Accounting: Accountability of SNM as a constituent of FBM disposed in Dry Storage Canisters will be based on the net of starting inventory reported in the PDMS SAR (Reference 1) less SNM disposed in LLRW or sent as a Sample and reconciled using the TMI-2 site Final Status Survey data. This is similar to the reporting method following the defueling effort in 1990.

- Final Material Transaction Reports will be performed per a written procedure for LLRW and Sample shipments containing more than 1 gram of SNM.

- The Final Status Survey will validate via an engineered sampling scheme that any material remaining on site is below the derived gross contamination limits (DGCLs). The final SNM accounting will reconcile the total SNM shipped and the SNM loaded into dry storage canisters to the final status survey. Final material balance reports will be generated at that time by written procedure.

Annual Inventory: An inventory will be conducted per a written procedure annually as follows:

- SNM as a constituent of FBM which has not been removed from its in-situ configuration during the decommissioning process will not be physically inventoried. Remaining in-situ material will be tracked per written procedure through decommissioning processes.

- SNM in any packages physically on site (to include LLRW or samples awaiting shipment, interim storage of FBM, and FBM in Dry Storage Canisters) at the time of the annual inventory will be physically inventoried.

Annual Reporting: TMI-2 will report the SNM inventory on-site annually using the results of the annual physical inventory (as determined above), analytically evaluated remaining in-situ material and shipments (LLRW and samples) containing SNM per written procedure.

RAI 10 Specific Response
The Program Description above describes how TMI-2 Solutions will control and account for Debris Material being removed during decommissioning processes in a manner compliant with 10 CFR 74 with existing exemptions.

For material being removed from TMI-2 systems which will be moved to dry cask storage, each canister will be analytically evaluated per written procedures based on the estimates of record (References 5-12) and an estimated quantity of SNM will be assigned which is associated with estimates from the cargo being packaged. These estimates for each dry canister will not improve on existing uncertainties.

The remaining TMI-2 fuel bearing material represents approximately two standard spent fuel modules of fuel and fission product materials and will be stored within approximately 14 dry storage canisters; each dry storage canister is designed to contain up to 37 spent fuel modules. TMI-2 fuel bearing material will be aggregated to quantities significantly lower than the dry storage systems at most reactor plants.

The SNM estimate for each dry canister will be considered as a preliminary estimate and will be refined by, and finalized, upon completion of the TMI-2 decommissioning against the small quantities shipped as samples, LLRW and the Final Status Survey.

For low quantities of fuel bearing material being removed from TMI-2 systems which will be disposed as sample materials or within low level radioactive waste, the waste characterization process required by 10 CFR 61 will be per written procedure and will improve upon the current estimates with additional sampling and characterization processes. This will not have an appreciable effect on the refinement of the estimates for the material in the DCSs.

References

1. Three Mile Island Nuclear Station, Unit 2 (TMI-2), "Revised Update 14 of Post-Defueling Monitored Storage Safety Analysis Report," TMI2-RA-COR-2021-0014, dated September 29, 2021

2. Exelon Generation, "Transmittal of DOE/NRC Form 742, Material Balance Report for Three Mile Island," NF210273, dated August 16, 2021

3. Three Mile Island Nuclear Station, Unit 2 (TMI-2), Defueling Completion Report, Final Submittal," 4410-90-L-0012, February 22, 1990 (ML20011F539)

4. US NRC "Approval of Exemption from 10 CFR 30.51, 40.61, 70.51(d), and 70.53" dated October 17, 1985 (ML20138D392)

5. Three Mile Island Nuclear Station, Unit 2 (TMI-2), "SNM Accountability," 4410-88-L-0162, dated September 30, 1988 (ML20207M226) – (Reactor Vessel Plenum)

6. Three Mile Island Nuclear Station, Unit 2 (TMI-2), "SNM Accountability," 4410-89-L-0097, dated September 22, 1989 (ML20248B795) – (Letdown Coolers, Pressurizer, Reactor Building Basement)

7. Three Mile Island Nuclear Station, Unit 2 (TMI-2), "SNM Accountability," 4410-90-L-0019, dated March 14, 1990 (20012D159) – (Reactor Vessel Head)

8. Three Mile Island Nuclear Station, Unit 2 (TMI-2), "SNM Accountability," C312-91-2045, dated June 6, 1991 (ML20077F496) – (Auxiliary and Fuel Handling Buildings)

9. Three Mile Island Nuclear Station, Unit 2 (TMI-2), "SNM Accountability," C312-91-2052, dated June 18, 1991 - (Reactor Building Miscellaneous Components)

10. Three Mile Island Nuclear Station, Unit 2 (TMI-2), "SNM Accountability," C312-91-2055, dated July 3, 1991. (ML20082A241) – (Reactor Coolant System)

11. Three Mile Island Nuclear Station, Unit 2 (TMI-2), "SNM Accountability," C312-91-2064, dated August 20, 1991 – ('A' and 'B' Once-Through Steam Generators (OTSGs))

12. Three Mile Island Nuclear Station, Unit 2 (TMI-2), "SNM Accountability," C312 -93-2004, dated February 1, 1993 - (Reactor Vessel)

13. US NRC, "Post-Defueling Survey Report Reviews" dated November 4, 1994 (ML20078H309)

RAI 11 Reports of Loss or Theft of SNM

Comment: In the LAR TMI-2 Solutions does not address reporting of loss, theft, or attempted theft of SNM.

Basis: 10 CFR 74.11(a), "Reports of loss or theft or attempted theft or unauthorized production of special nuclear material," requires each licensee who possesses one gram or more of contained uranium-235, uranium-233 or plutonium to notify the NRC Operations Center within 1 hour of discovery of any loss or theft or other unlawful diversion of SNM which the licensee is licensed to possess, or any incident in which an attempt has been made to commit a theft or unlawful diversion of SNM.

Path Forward: Provide a description of the MC&A activities that are performed or the measures in place to show how the reporting requirement of 10 CFR 74.11(a) is met.

RESPONSE

The TMI-2 Materials Security Plan establishes a security zone for which:

- Personnel access is controlled
- Random searches of personnel and equipment upon exiting the security zone are conducted. These random searches include radiation detection which will detect radioisotopes present with SNM.

Security personnel are notified upon detection of loss or theft or attempted theft of SNM.

The TMI-2 Materials Security Plan is implemented using security personnel under contract from TMI-1. Security notifications of less than 4 hours are made by the TMI-1 security organization using the TMI-1 processes and procedures. TMI-2 has the requirements of 10 CFR 74.11(a) included in TMI2-RA-PR-005, Reporting of Events and Conditions. This procedure details the one-hour reporting requirement and meets 10 CFR 74.11(a).

RAI 12 Material Status Reports

Comment: In the LAR TMI-2 Solutions does not address completion or submission of Material Balance Reports or Physical Inventory Listing Reports.

Basis: 10 CFR 74.13(a), "Material status reports," requires each licensee possessing SNM in a quantity totaling 1 gram or more of contained uranium-235, uranium-233, or plutonium to complete and submit, in computer-readable format Material Balance Reports concerning SNM that the licensee has received, produced, possessed, transferred, consumed, disposed, or lost. The Physical Inventory Listing Report must be submitted with each Material Balance Report.

Path Forward: Provide a description of the MC&A activities that are performed or the measures in place to show how the reporting requirements of 10 CFR 74.13(a) are met.

RESPONSE

The Reference 1 exemption required the preparation of Material Balance Reports and Physical Inventory Listings of remaining SNM at TMI-2. Reference 2 provided a report by the NRC reviewing and accepting the post-defueling survey report results, including the overall uncertainty of +/- 40%. Accounting for that uncertainty, TMI-2 Solutions will execute reporting to written procedures in two stages: Preliminarily for material which is packaged for dry storage at an onsite ISFSI during decommissioning processes, and finally, for both fuel bearing material packaged for samples or disposal as low-level radioactive waste and for packaged dry storage containers upon completion of decommissioning activities reconciled using final status survey data.

Preliminary SNM Accounting:

> SNM as a constituent of FBM which has been either placed in interim storage containers or into Dry Storage Canisters will have preliminary SNM accounting performed per a written procedure to analytically determine SNM content based on the engineering assessment process. This will remain preliminary until after the final status survey for the decommissioning project is complete.

Final SNM Accounting: Accountability of SNM as a constituent of FBM disposed in Dry Storage Canisters will be based on the net of starting inventory reported in the PDMS SAR (Reference 3) less SNM disposed in LLRW or sent as a Sample and reconciled using the TMI-2 site Final Status Survey data. This is similar to the reporting method following the defueling effort.

> Final Material Transaction Reports will be performed per a written procedure for LLRW and Sample shipments containing more than 1 gram of SNM.
>
> The Final Status Survey will validate via an engineered sampling scheme that any material remaining on site is below the derived gross contamination limits (DGCLs). The final SNM accounting will reconcile the total SNM shipped and the SNM loaded into dry storage canisters to the final status survey. Final material balance reports will be generated at that time by written procedure.

References

1. US NRC "Approval of Exemption from 10 CFR 30.51, 40.61, 70.51(d), and 70.53" dated October 17, 1985 (ML20138D392)

2. US NRC, "Post-Defueling Survey Report Reviews" dated November 4, 1994 (ML20078H309)

3. Three Mile Island Nuclear Station, Unit 2 (TMI-2) "Revised Update 14 of Post-Defueling Monitored Storage Safety Analysis Report," TMI2-RA-COR-2021-0014, dated 29 September 2021

RAI 13 Nuclear Material Transaction Reports

Comment: In the LAR TMI-2 Solutions does not address completion of Nuclear Material Transaction Reports.

Basis: 10 CFR 74.15(a), "Nuclear material transaction reports," requires each licensee who transfers or receives SNM in a quantity of 1 gram or more of contained uranium-235, uranium-233, or plutonium to complete, in computer-readable format, a Nuclear Material Transaction Report. In addition, each licensee who adjusts the inventory in any manner, other than for transfers and receipts, shall submit a Nuclear Material Transaction Report, in computer-readable format, to coincide with the submission of the Material Balance Report. Each licensee who transfers SNM shall submit a Nuclear Material Transaction Report no later than the close of business the next working day. Each licensee who receives SNM shall submit a Nuclear Material Transaction Report within 10 days after the material is received.

Path Forward: Provide a description of the MC&A activities that are performed or the measures in place to show how the reporting requirements of 10 CFR 74.15(a) are met.

RESPONSE

As discussed in RAI 10, Material Transaction Reports for all packages containing more than 1 gram of SNM which are shipped from TMI-2 will be created in accordance with written procedures.

For LLRW waste packages, characterization prior to the shipment will meet 10 CFR 74.15(a) requirements.

For SNM in dry cask storage, characterization will meet 10 CFR 74.15(a) requirements after completion of final status survey for TMI-2 per written procedure.

RAI 14 MC&A Records

Comment: In the LAR TMI-2 Solutions does not address MC&A recordkeeping.

Basis: 10 CFR 74.19(a), "Recordkeeping," requires licensees to keep records showing the receipt, inventory (including location and unique identity), acquisition, transfer, and disposal of all SNM in its possession regardless of its origin or method of acquisition. Each record relating to material control or material accounting must be maintained and retained for the period specified by the appropriate regulation or license condition. Each record of receipt, acquisition, or physical inventory of SNM must be retained as long as the licensee retains possession of the material and for 3 years following transfer or disposal of the material. Each record of transfer of SNM to other persons must be retained by the licensee who transferred the material until the Commission terminates the license authorizing the licensee's possession of the material.

Path Forward: Provide a description of the MC&A activities that are performed or the measures in place to show how the MC&A records requirements of 10 CFR 74.19(a) are met.

RESPONSE

The Program Description, included in the response to RAI 10, describes the overall material control and accountability program, requirements for records meeting 10 CFR 74.19(a) are included in the associated written procedures.

10 CFR 74.19(a)(1): TMI-2 does not intend to receive or acquire SNM thus there is no program or procedures addressing receipt of SNM.

Physical inventory is exempted for in situ material per Reference 1. Upon removal from its in situ condition and packaged for dry storage or packaged in small quantities within LLRW or samples, physical inventory will occur, including the associated records per written procedure. Records for transfer or disposal will be generated per written procedure.

10 CFR 74.19(a)(2) through (4): record retention requirements associated with SNM records are included in the TMI-2 Records Procedure which establishes record retention requirements meeting 10 CFR74.19(a)(2) through (4).

Reference

1. US NRC "Approval of Exemption from 10 CFR 30.51, 40.61, 70.51(d), and 70.53" dated October 17, 1985 (ML20138D392)

RAI 15 Written MC&A Procedures

Comment: In the LAR TMI-2 Solutions does not address the use of written MC&A procedures to enable the licensee to account for the SNM in its possession.

Basis: 10 CFR 74.19(b) requires each licensee authorized to possess SNM in a quantity exceeding one effective kilogram to establish, maintain, and follow written MC&A procedures that are sufficient to enable the licensee to account for the SNM in its possession under license.

Path Forward: Provide a description of the MC&A activities that are performed or the measures in place to show how the procedure requirements of 10 CFR 74.19(b) are met, if applicable.

RESPONSE

The Program Description, included in the response to RAI 10, describes the overall material control and accountability program and associated written procedures which will meet requirements of 10 CFR 74.19(b).

RAI 16 Annual Physical Inventory of SNM

Comment: In the LAR TMI-2 Solutions does not address conduct of an annual physical inventory of SNM.

Basis: 10 CFR 74.19(c) requires certain licensees who are authorized to possess SNM in a quantity greater than 350 grams of contained uranium-235, uranium-233, or plutonium, to conduct a physical inventory of all SNM in its possession under license at intervals not to exceed 12 months. The results of these physical inventories shall be retained in records by the licensee until the Commission terminates the license authorizing the possession of the material.

Path Forward: Provide a description of the MC&A activities that are performed or the measures in place to show how the inventory requirements of 10 CFR 74.19(c) are met.

RESPONSE

The NRC has provided a response to the applicability of the existing exemptions for MC&A (Reference 1). This explanation states that the existing TMI-2 exemption for 70.53 received in 1985 (Reference 2) is equivalent to an exemption from the current 10 CFR 74.19(c).

Notwithstanding the current exemption applicability, upon packaging material into a container TMI-2 Solutions will perform physical inventory of the resulting containers per written procedures. This will include SNM in any packages physically on site (to include LLRW or samples awaiting shipment, interim storage of FBM, and FBM in Dry Storage Canisters) at the time of the annual inventory.

References

1. Email from Amy Snyder (NRC) to Timothy Devik (TMI2S) "Exemptions- TMI-2" dated 22 September 2022

2. US NRC "Approval of Exemption from 10 CFR 30.51, 40.61, 70.51(d), and 70.53" dated October 17, 1985 (ML20138D392)

NRC REFERENCES

AEC, "Zirconium Fire and Explosion Hazard Evaluation," Accident and Fire Prevention Information, Issue No. 45, Atomic Energy Commission, 1956.

Cashdollar, K. L., "Overview of Dust Explosibility Characteristics," Pittsburgh Research Laboratory, Pittsburgh PA, 2000.

Clark et al, "TMI-2 Leadscrew Debris Pyrophoricity Study," GEND-INF-044, Pacific Northwest Laboratory, Richland WA, 1984.

Cragnolino, G. "Characteristics of the Three Mile Island Unit 2 Fuel Debris – Final Report," CNWRA 98-002, Center for Nuclear Waste Regulatory Analyses, San Antonio, TX, 1997.

Cox et. al, "Reactor Building Basement Radionuclide and Source Distribution Studies," GEND-INF-011-Vol3, GEND, US DOE, 1983.

DOE, "Airborne Release Fractions/Rates and Respirable Fractions for non-Reactor Nuclear Facilities, DOE-HDBK-3010-94, US Department of Energy, Washington DC, 1994 (ML13078A031).

Hubbard et. al, "Airborne Release Fractions from Surrogate Nuclear Waste Fires Containing Lanthanide Nitrates and Depleted Uranium Nitrate in 30% Tributyl Phosphate in Kerosene," SAND2019-12565J, Sandia National Laboratories, Albuquerque, NM 2019.

Mishima, J. and L. C. Schwendiman, "Fractional Airborne Release of Uranium (Representing Plutonium) During the Burning of Contaminated Wastes," BNWL-1730, Battelle Pacific Northwest Laboratories, Richland WA, 1973.

AREVA Federal Services and Spectra Tech, Inc., prepared for DOE-Idaho Office, REDACTED TMI-2 Independent Spent Fuel Storage Installation Application for 10 CFR 72 Specific License Renewal, Special Nuclear Materials License Number SNM-2508, Revision I, dated August 26, 2018 (ML18296A527).

TMI-2 ISFSI Renewed Technical Specifications [Letter to J. Zimmerman re: Issuance of Renewed Materials License No. SNM-2508 for the Three Mile Island Unit 2 Independent Spent Fuel Storage Installation], dated August 16, 2019 (ML19259A017).

Three Mile Island Nuclear Station, Unit 2 (TMI-2) Response to Request for Additional Information, Attachment 2, TMI-2-RA-COR-2022-0007, License Amendment Request Three Mile Island Nuclear Station, Unit 2 "GPU Nuclear Calculation 4440-7380-90-017, Revision 4, PDMS SAR Section 8.2.5 Fire Analysis Source Terms", dated (ML22138A302, Pkg).

Three Mile Island Nuclear Station, Unit 2 (TMI-2) - License Amendment Request - Three Mile Island, Unit 2, Decommissioning Technical Specifications, Response to Questions, dated May 16, 2022 (ML21057A046 Pkg - Non-public).

Three Mile Island Nuclear Station, Unit 2 (TMI-2) License Amendment Request - Three Mile Island, Unit 2, Decommissioning Technical Specifications, Supplemental Information, dated January 7, 2022 (ML22013A177).

Request for Additional Information - TMI-2 Accident Analyses Questions, dated February 7, 2022 (ML22038A936).

Three Mile Island, Unit 2, Supplemental Information to License Amendment Request, Decommissioning Technical Specifications, dated April 19, 2022 (ML22101A077).

Smith et. al, "Technology, Safety and Costs of Decommissioning a Reference Pressurized Water Reactor Power Station, NUREG/CR-0139, Vol 2, Battelle Pacific Northwest Laboratory, Richland WA, 1978.

U.S. NRC, "RASCAL 4: Description of Models and Methods," NUREG-1940, U.S. Nuclear Regulatory Commission, Washington, DC 2012 (ML13031A448).

U.S. NRC, "A Regulatory Analysis on Emergency Preparedness for Fuel Cycle and Other Radioactive Material Licensees," NUREG-1140, U.S. Nuclear Regulatory Commission, Washington, DC 1991 (ML101460227).

ATTACHMENT 2

REQUEST FOR EXEMPTION FROM 10 CFR 70.24, CRITICALITY ACCIDENT REQUIREMENTS

1.0 SPECIFIC EXEMPTION REQUEST

10 CFR 70.24 requires, in part, that each licensee authorized to possess special nuclear material (SNM) in a quantity exceeding 700 grams of contained uranium-235 (U-235) 520 grams of uranium-233, 450 grams of plutonium, 1,500 grams of contained uranium-235 if no uranium enriched to more than 4 wt. percent U-235 is present, or 450 grams of any combination thereof, shall maintain in each area in which such licensed SNM is handled, used, or stored a criticality accident alarm system (CAAS).

Pursuant to 10 CFR 70.24(d), TMI-2 Solutions, LLC requests an exemption from the requirements of 10 CFR 70.24 for Three Mile Island Unit-2 as authorized by 10 CFR 70.17(a), "Specific Exemptions." The proposed action would exempt TMI-2 Solutions from the requirements of 10 CFR 70.24 to maintain a radiation monitoring system in each area where licensed special nuclear material is handled, used, or stored that will energize clearly audible alarm signals if accidental criticality occurs.

2.0 BACKGROUND

The NRC granted TMI-2 an exemption from the requirements of 10 CFR 70.24, criticality accident requirements for SNM storage areas, on June 15, 1992 (ML20210D729). As noted in the exemption:

> "...it is appropriate to request an exemption from 10 CFR 70.24 if an evaluation determines that a potential for criticality does not exist, as for example where the quantities or form of special nuclear material make criticality practically impossible or where geometric spacing is used to preclude criticality."

This exemption was granted on the basis of both an evaluation on the potential for criticality and geometric spacing. Specifically, within the PDMS condition, the NRC determined, as described in Inspection Report 50-320/90-03 dated June 14, 1990 (ML20043G083) that the safe fuel mass limit (SFML) was 93 kg UO_2. An administrative SFML applied a 25% conservatism at 70 kg and further administrative controls were applied on geometric separation distances. At the time of the 10 CFR 70.24 exemption, the defueling activities had concluded that ~99% of the fuel material had been removed from the core leaving a balance of 1097 kg UO_2 of fuel bearing material (described in the PDMS SAR) (ML21236A288). Because the balance of fuel bearing material remaining on site was greater than the SFML at the time, additional administrative controls (i.e., geometric controls) were necessary to preclude criticality.

3.0 DISCUSSION

TMI-2 Solutions considers a 10 CFR 70.24 exemption for criticality monitoring system appropriate under the DECON license basis because the TMI2-RA-COR-2022-0008, "Supplemental Information to License Amendment Request – Three Mile Island, Unit 2, Decommissioning Technical Specifications," (ML22101A077) provides a calculation which shows the SFML associated with remaining fuel bearing material at TMI-2 is 1361 kg UO_2. That SFML is 24% higher than the 1097 kg UO_2 estimate of record for remaining fuel bearing material at TMI-2 which analytically precludes a criticality accident at TMI-2. The 1361 kg UO_2 SFML result represents a significant improvement over the 1990 SFML calculation result. This improvement was achieved by taking credit for impurities and actual enrichment based on the results of physical samples taken during the defueling effort.

Administrative controls for geometric spacing are not necessary to further preclude a criticality accident because there is not enough kg UO_2 at TMI-2 to assemble an optimal critical configuration. However, as part of its Fuel Bearing Material Management Program, TMI-2 Solutions will be implementing local administrative controls for the purpose of defense in depth on the activities which will handle the highest quantities of fuel bearing material (e.g., segmenting the reactor vessel internals which represent 925 kg UO_2 or 68% of the SFML). These defense in depth controls will include control on the physical location of segmentation equipment and limiting the number of waste receptacles (i.e., physical manifestations of controls on geometric spacing).

4.0 JUSTIFICATION FOR EXEMPTION

10 CFR 70.17(a) states that the Commission may, upon application of any interested person or upon its own initiative, grant such exemptions from the requirements of the regulations in this part as it determines are authorized by law and will not endanger life or property or the common defense and security and are otherwise in the public interest.

A. The exemption is authorized by law

The requested exemption to the requirements of 10 CFR 70.24 is authorized by law. As noted in the 1992 exemption – consistent with Regulatory Guide 8.12, Rev. 2 Section C – if an evaluation does not determine that a potential for criticality exists, it is appropriate to request an exemption from 10 CFR 50.24.

B. The exemption will not endanger life or property

Granting of the requested exemption to the requirements of 10 CFR 70.24 would not endanger life or property. As stated above, the SFML associated with the remaining fuel bearing material at TMI-2 is 24% higher than the estimate of record for remaining fuel bearing material at TMI-2 which analytically precludes a criticality accident at TMI-2. Administrative controls, though not necessary, have been implemented through the TMI-2 Fuel Bearing Materials Program and will provide defense in depth assurances to further preclude a criticality accident. Additionally, the amount of SNM would not change as a consequence of the proposed exemption and, therefore, would not result in any significant radiological impacts.

C. The exemption will not endanger the common defense and security

The requested exemption to the requirements of 10 CFR 70.24 does not involve information or activities that could potentially impact the common defense and security. The SFML calculation determined that there is not enough material that remains at TMI-2 to cause a criticality accident in either amount or geometrical configuration, and the existing administrative restrictions described in the TMI-2 Fuel Bearing Material Program prevent proliferation and limit aggregation. Therefore, the requested exemption to require a criticality monitor will not endanger the common defense and security.

D. The exemption is otherwise in the public interest

Granting of the exemption will reduce the burden of installation and maintenance of a criticality monitoring system that would not provide any additional benefit or protection. As such, the reduced burden created by granting the exemption is otherwise in the public interest.

5.0 CONCLUSION

Based on the above and in accordance with 10 CFR 70.17(a), the requested exemption to the requirements of 10 CFR 70.24 is authorized by law and will not endanger life or property or the common defense and security and is otherwise in the public interest.

ATTACHMENT 3

LIST OF REGULATORY COMMITMENTS

The table included in this attachment identifies the regulatory commitments in this document. The type of commitment and associated schedule for implementation are provided. Any other statements in this submittal represent intended or planned actions. They are provided for information purposes and are not considered to be regulatory commitments.

Regulatory Commitment	Type		Scheduled Completion Date
	One-Time Action	Continuing Compliance	
TMI-2 Solutions will supplement its License Amendment Request (LAR) to retain Technical Specification Section 6.8.1.2 in its Technical Specifications. An amended LAR will be submitted under separate cover which will include a markup of the proposed Technical Specification change.	X		October 31, 2022
TMI-2 Solutions will establish a work planning instruction which will evaluate specific hydrogen concerns relevant to a given scope of work and include appropriate hydrogen mitigation measures appropriate for that work.	X		December 31, 2022