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10 CFR 50.12
10 CFR 50.47
10 CFR 50, Appendix E

TMI-19-040

July 1, 2019

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

Three Mile Island Nuclear Station, Unit 1
Renewed Facility Operating License No. DPR-50
NRC Docket No. 50-289

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

Subject: Request for Exemptions from Portions of 10 CFR 50.47 and 10 CFR Part 50, Appendix E

Reference:

1. Letter from J. Bradley Fewell (Exelon Generation Company, LLC) to U.S. Nuclear Regulatory Commission, "Certification of Permanent Cessation of Power Operations for Three Mile Island Nuclear Station, Unit 1," dated June 20, 2017 (ML17171A151)
2. Letter from U.S. Nuclear Regulatory Commission to Bryan C. Hanson, (Exelon Generation Company, LLC), "Three Mile Island Nuclear Station, Units 1 and 2 – Issuance of Amendment No. 296 for Unit 1 RE: Changes to Emergency Plan for Post-Shutdown and Permanently Defueled Condition (EPID L-2018-LLA-0073), dated April 18, 2019 (ML19065A114)

Pursuant to 10 CFR 50.12, "Specific exemptions," Exelon Generation Company, LLC (Exelon) requests exemptions from portions of 10 CFR 50.47(b), 10 CFR 50.47(c)(2), and 10 CFR Part 50, Appendix E for Three Mile Island Nuclear Station (TMI). The requested exemptions would allow TMI to reduce emergency planning requirements consistent with the permanently defueled condition of the station.

By letter dated June 20, 2017 (Reference 1), Exelon provided formal notification to the U.S. Nuclear Regulatory Commission (NRC) pursuant to 10 CFR 50.4(b)(8) and 10 CFR 50.82(a)(1)(i) of Exelon's determination to permanently cease operations at TMI, Unit 1 (TMI-1) on or about September 30, 2019.

Once the certifications for permanent cessation of operations and permanent removal of fuel from the reactor vessel are submitted to the NRC pursuant to 10 CFR 50.82(a)(1)(i) and (ii), and pursuant to 10 CFR 50.82(a)(2), the 10 CFR 50 license will no longer authorize operation of the reactor or placement or retention of fuel in the reactor vessel.

By letter April 18, 2019 (Reference 2), the NRC issued the approved changes to the TMI site emergency plan (SEP) to support the planned permanent cessation of operation and permanent defueling at the TMI-1 reactor. The approved changes revise the TMI SEP emergency response organization (ERO) on-shift and augmented staffing, commensurate with the reduced spectrum of credible accidents for a permanently shutdown and defueled nuclear power reactor facility.

Three Mile Island, Unit 2 (TMI-2), has a possession only license and is currently maintained in accordance with the NRC approved SAFSTOR condition (method in which a nuclear facility is placed and maintained in a condition that allows it to be safely stored and subsequently de-contaminated) known as Post-Defueling Monitored Storage (PDMS). Exelon maintains the emergency planning responsibilities for TMI-2, which is owned by First Energy Corporation, through a service agreement. This request for exemptions does not impact Exelon's ability to maintain the service agreement.

The requested exemptions are permissible under 10 CFR 50.12 because they are authorized by law, will not present an undue risk to the public health and safety, are consistent with the common defense and security, and present special circumstances.

More specifically, application of the portions of the regulations from which exemptions are sought is not necessary to ensure adequate emergency response capability for TMI and to achieve the underlying purpose of the rules. Furthermore, continued application of these portions of the regulations from which exemptions are sought would result in an undue hardship or other costs to the TMI-1 Decommissioning Trust Fund by requiring continued implementation of unnecessary emergency response capabilities. Finally, granting the requested exemptions would result in benefit to the public health and safety and would not result in a decrease in safety, because they would enhance the ability of the emergency response organization to respond to credible scenarios.

The exemption requests are contained in Attachment 1 to this letter. Exelon has performed analyses which show that 488 days after permanent cessation of power operations, the spent fuel stored in the spent fuel pool will have decayed to the extent that the requested exemptions may be implemented at TMI-1. Following the TMI-1 shutdown, which is expected by the end of September 2019 (Reference 1), 488 days after shutdown is expected to be about January 30, 2021. The bounding analysis is contained in Attachment 2.

TMI-1 plans to submit a Permanently Defueled Emergency Plan (PDEP), containing a Permanently Defueled Emergency Action Level (EAL) scheme, for NRC review and approval pursuant to 10 CFR 50.54(q)(4) and 10 CFR 50, Appendix E, Section IV.B.2. The proposed emergency plan will be based on the exemptions requested herein.

Exelon requests review and approval of this exemption request by August 30, 2020. Exelon requests that the approved exemptions become effective 488 days following the permanent shutdown of TMI-1. TMI will provide the permanent shutdown date in the certification required by 10 CFR 50.82(a)(1)(ii) that TMI-1 has been permanently shutdown and defueled. Approval

of these exemptions by August 30, 2020, will allow TMI-1 adequate time to implement changes to the emergency plan and emergency response organization by the requested effective date.

This letter contains no new regulatory commitments.

In accordance with 10 CFR 50.91 "Notice for public comment; State consultation" paragraph (b), Exelon is notifying the State of Pennsylvania of this request for exemption by transmitting a copy of this letter and its attachments to the designated State Official.

On May 30, 2019, the Commonwealth of Pennsylvania – Department of Environmental Protection Bureau of Radiation Protection (PA-BRP) received the draft proposed changes of the TMI Permanently Defueled Emergency Plan and EAL scheme. On June 24, 2019, the PA-BRP and Pennsylvania Emergency Management Agency (PEMA) met with representatives of TMI-1 and provided comments on the PDEP and associated EALs. An acknowledgement from the Commonwealth of Pennsylvania confirming that they completed their review of the proposed TMI Emergency Plan/EALs and comments were resolved to their satisfaction will be included in attachment to the License Amendment Request for the proposed changes to the TMI Permanently Defueled Emergency Plan (PDEP) and Emergency Action Level scheme.

If you have any questions concerning this submittal, please contact Leslie Holden at (630) 657-2524.

Respectfully,

Michael P. Gallagher

Michael P. Gallagher
Vice President, License Renewal & Decommissioning
Exelon Generation Company, LLC

Attachment:
1. Request for Exemptions from Portions of 10 CFR 50.47(b), 10 CFR 50.47(c)(2) and 10 CFR Part 50, Appendix E
2. Three Mile Island Nuclear Station Zirconium Fire Analysis for Drained Spent Fuel Pool (Calculation C-1101-202-E410-476, Revision 1)

cc: w/Attachment

NRC Regional Administrator, Region I
NRC Senior Resident Inspector – Three Mile Island Nuclear Station – Unit 1
NRC Project Manager, NRR – Three Mile Island Nuclear Station – Unit 1
NRC Project Manager, NMSS/DUWP/RDB – Three Mile Island – Unit 2
Director, Bureau of Radiation Protection - PA Department of Environmental Resources

bcc: w/o Attachment

Sr. Vice President – Mid-Atlantic Operations
Site Vice President – TMI-1
Plant Manager – TMI-1
Director, Operations – TMI-1
Director, Training – TMI-1

w/ Attachment

Vice President – License Renewal and Decommissioning – KSA
Vice President – Licensing and Regulatory Affairs
Director, Licensing and Regulatory Affairs (East)
Senior Manager, Emergency Preparedness – Cantera
Senior Manager, Decommissioning – Cantera
Decommissioning – Licensing - Cantera
Site Decommissioning Director – TMI-1
Regulatory Assurance Manager – TMI-1
Manager, Licensing and Regulatory Affairs – KSA
Exelon Document Control Desk Licensing (Hard Copy)
Commitment Tracking Coordinator – East
Records Management – KSA
R. R. Brady – TMI-1
M. D. Fitzwater – TMI-1
E. M. Carreras – TMI-1
C. W. Smith – TMI-1
D. H. Walker – Emergency Preparedness – KSA
T. J. Barton – Emergency Preparedness – KSA
M. J. Casey, FENOC Fleet Project Management, GPU TMI-2 Project Manager

ATTACHMENT 1

THREE MILE ISLAND NUCLEAR STATION

REQUEST FOR EXEMPTIONS FROM

PORTIONS OF 10 CFR 50.47(b), 10 CFR 50.47(c)(2)

AND 10 CFR PART 50, APPENDIX E

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1.0 SPECIFIC EXEMPTION REQUEST

Pursuant to 10 CFR 50.12 "Specific exemptions," Exelon Generating Company, LLC (Exelon) requests exemptions from the following for Three Mile Island Nuclear Station:

- Certain standards in 10 CFR 50.47(b) regarding onsite and offsite emergency response plans for nuclear power reactors;
- Certain requirements of 10 CFR 50.47(c)(2) to establish plume exposure and ingestion pathway emergency planning zones for nuclear power plants; and
- Certain requirements of 10 CFR 50, Appendix E, which establish the elements that make up the content of emergency plans.

The Emergency Plan encompasses both Three Mile Island (TMI), Unit 1 (TMI-1), and TMI, Unit 2 (TMI-2). Exelon maintains the emergency planning responsibilities for TMI-2, which is owned by First Energy Corporation, through a service agreement. This exemption request does not impact Exelon's ability to maintain the service agreement.

The requested exemptions would allow Exelon to reduce emergency planning requirements and subsequently revise the TMI Emergency Plan to reflect the permanently defueled condition of the station. The current 10 CFR Part 50 regulatory requirements for emergency planning (developed for operating reactors) ensure safety at TMI. However, once the station is permanently shut down and defueled, and a sufficient decay of the spent fuel has occurred in a state of decommissioning, some of these requirements exceed what is necessary to protect the health and safety of the public.

The requested exemptions and justification for each are based on and consistent with Interim Staff Guidance NSIR/DPR-ISG-02, Emergency Planning Exemption Requests for Decommissioning Nuclear Power Plants, which was issued May 11, 2015 (Reference 1).

2.0 BACKGROUND

Three Mile Island Nuclear Station is located in an area of low population density about 12 miles southeast of Harrisburg, Pennsylvania. The area is in Londonderry Township, Dauphin County, about 2.5 miles from the southern tip of Dauphin County, where the county is coterminous with York and Lancaster Counties. The TMI site is part of an 814-acre tract consisting of Three Mile Island and several adjacent islands, which were purchased by a predecessor. The island, which is situated about 900 feet from the east bank and approximately one mile from the west bank of the Susquehanna River, is elongated parallel to the flow of the river with its longest axis oriented approximately due north and south. The north and south ends of the island have access bridges, which connect the island to State Highway Route 441. The north access bridge is used daily. Route 441 is a two-lane highway, which runs parallel to TMI on the east bank of the Susquehanna River and is more than 2,000 feet from the TMI reactors at the closest point. The exclusion area for TMI is a 2,000-foot radius, and for the purposes of Emergency Planning, the exclusion area and the site boundary are considered the same.

Section 6, "Safety Analysis," of the TMI-1 Defueled Safety Analysis Report (DSAR) describes the design basis accident (DBA) scenarios that are applicable to TMI-1. After the reactor is defueled, the spent fuel will be stored in the Spent Fuel Pool (SFP) located in the Fuel Handling Building. While spent fuel is stored in the SFP, the remaining accident is the Fuel Handling Accident (FHA) that takes place in the SFP.

The analyses of the potential radiological impact of accidents while the plant is in a permanently defueled condition indicate that no design basis accident or reasonably conceivable beyond design basis accident will be expected to result in radioactive releases that exceed U.S. Environmental Protection Agency (EPA) Protective Action Guidelines (PAGs) (Reference 2) beyond the site boundary. Exelon will maintain the version of the EPA PAGs as specified in the current and proposed TMI Emergency Plan.

TMI-2 has a possession only license and is currently maintained in accordance with the NRC approved SAFSTOR condition (method in which a nuclear facility is placed and maintained in a condition that allows it to be safely stored and subsequently decontaminated) known as Post-Defueling Monitored Storage (PDMS). All fuel assemblies have been removed from the TMI-2 reactor and spent fuel pool.

By letter dated June 20, 2017 (Reference 3), pursuant to 10 CFR 50.82(a)(1)(i), Exelon submitted a certification to the NRC indicating its intention to permanently cease power operations at TMI-1 on or about September 30, 2019. Once fuel has been permanently removed from the reactor vessel, Exelon will submit a written certification to the NRC, in accordance with 10 CFR 50.82(a)(1)(ii) that meets the requirements of 10 CFR 50.4(b)(9). Upon docketing of these certifications, the 10 CFR Part 50 license for TMI-1 will no longer authorize operation of the reactor or emplacement or retention of fuel into the reactor vessel, as specified in 10 CFR 50.82(a)(2).

By letter dated April 18, 2019 (Reference 4), the NRC issued the Post-Shutdown Emergency Plan (PSEP) which approved changes to the TMI site emergency plan (SEP) to support the planned permanent cessation of operations and permanent defueling of the TMI-1 reactor. The PSEP revised the TMI SEP emergency response organization (ERO) on-shift and augmented staffing, to be commensurate with the reduced spectrum of credible accidents for a permanently shutdown and defueled nuclear power reactor facility. The PSEP maintains effectiveness of the TMI SEP in accordance with 10 CFR 50.47 and 10 CFR 50, Appendix E.

Pursuant to 10 CFR 50.82(a)(4)(i), TMI-1 submitted a Post-Shutdown Decommissioning Activities Report (PSDAR) (Reference 5), which identified SAFSTOR as TMI-1's selected method of decommissioning. With the reactor permanently defueled, the reactor vessel assembly and supporting structures and systems will no longer be in operation and will have no function related to the safe storage and management of irradiated fuel in the SFP. The irradiated fuel will be stored in the SFP and later in the Independent Spent Fuel Storage Installation (ISFSI) (when built) until it is shipped offsite in accordance with the schedules described in the PSDAR and Spent Fuel Management Plan (Reference 6).

3.0 BASIS FOR EXEMPTION REQUEST

In order to allow a reduction in emergency planning requirements commensurate with the hazards associated with TMI's permanently defueled condition, exemptions from portions of 10 CFR 50.47(b), 50.47(c)(2), and 10 CFR 50, Appendix E, are needed. Exelon has performed an analysis indicating that 488 days after permanent cessation of power operations at TMI-1, a minimum of 10 hours is available before fuel cladding temperature reaches 900°C with a complete loss of SFP water inventory with no heat loss (adiabatic heat up). After the 488-day period, there is sufficient time within the 10 hours described in the supporting analysis to mitigate events that could lead to a zirconium cladding fire (herein referred to as the Zirc-Fire Window) (Reference 7). This analysis is contained in Attachment 2. Considering a shutdown date of September 30, 2019, 488 days following

permanent cessation of power operations would occur January 30, 2021. Exelon plans to submit a permanently defueled emergency plan (PDEP) by July 1, 2019, including a Permanently Defueled Emergency Action Level scheme for NRC review and approval pursuant to 10 CFR 50.54(q)(4) and 10 CFR 50, Appendix E, Section IV.B.2.

Based on the analyses detailed in Section 5.0, below, Exelon has concluded that the portions of 10 CFR 50.47(b), 10 CFR 50.47(c)(2), and 10 CFR Part 50, Appendix E identified in Tables 1 and 2 will not be necessary to protect the health and safety of the public when TMI-1 is in the permanently defueled condition and would be unduly burdensome. Approval of the exemptions requested in Tables 1 and 2 would not present an undue risk to the public or prevent an appropriate response in the event of an emergency at TMI.

The proposed emergency plan will be based on the exemptions requested herein. Exelon requests approval of these exemption requests by August 30, 2020 with an effective date of meeting the Zirc-Fire Window at 488 days after shutdown, which is expected to be about January 30, 2021. Approval of these exemptions by the requested date will enable Exelon adequate time to implement changes to the emergency preparedness program and emergency response organization.

4.0 EXEMPTIONS TO EMERGENCY PLAN REQUIREMENTS DEFINED BY 10 CFR 50.47 AND 10 CFR PART 50, APPENDIX E

Exelon requests exemptions from portions of 10 CFR 50.47(b) and (c)(2) and Appendix E to 10 CFR Part 50 to the extent that these regulations apply to specific provisions of onsite and offsite emergency planning that will no longer be applicable once the certifications required by 10 CFR 50.82(a)(1)(i) and (ii) have been submitted and sufficient decay of the spent fuel has occurred for TMI-1. The specific portions of 10 CFR 50.47 and 10 CFR Part 50, Appendix E from which exemptions are being requested are identified using **bold ~~strikethrough~~** text in Table 1 (Exemptions Requested from 10 CFR 50.47(b) and (c)(2)) and Table 2 (Exemptions Requested from 10 CFR Part 50, Appendix E), below. The portions of regulation that are not identified using **bold ~~strikethrough~~** text (i.e., those portions for which exemption is not being requested), will remain applicable to TMI. Details related to specific exemption requests are provided in the Basis for Exemption column.

The requested exemptions and justification for each are based on, and consistent with NSIR/DPR-ISG-02 (Reference 1).

TABLE 1
EXEMPTIONS REQUESTED FROM 10 CFR 50.47(b) AND (c)(2)

Bold ~~strikethrough~~ text identifies the proposed exemption with respect to the regulation. The basis for the exemption explains the scope of the exception.

Item	10 CFR 50.47 Emergency Plans	Basis for Exemption
1	10 CFR 50.47(b) The onsite **~~and, except as provided in paragraph (d) of this section, offsite~~** emergency response plans for nuclear power reactors must meet the following standards:	In the Statement of Considerations (SOCs) for the final rule for EP requirements for independent spent fuel storage installations (ISFSIs) and for monitored retrievable storage (MRS) facilities (60 FR 32430; June 22, 1995) (Reference 8), the Commission responded to comments concerning offsite emergency planning for ISFSIs or MRS and concluded that, "the offsite consequences of potential accidents at an ISFSI or an MRS would not warrant establishing Emergency Planning Zones (EPZs)." As discussed in ISG-02 (Reference 1), in a nuclear power reactor's permanently defueled state, the accident risks are more similar to an ISFSI or MRS than an operating nuclear power plant. The EP program would be similar to that required for an ISFSI under 10 CFR 72.32(a) when fuel stored in the SFP has more than five years of decay time and would not change substantially when all the fuel is transferred from the SFP to an onsite ISFSI. Exemptions from offsite EP requirements have previously been approved when the site-specific analyses show that in a partial drain-down event, at least 10 hours is available from the time when cooling of the spent fuel is not effective until the hottest fuel assembly reaches the zirconium ignition temperature of 900 degrees Celsius (°C). The technical basis that underlies the approval of the exemption request is based partly on the analysis of a time period that spent fuel stored in the SFP is unlikely to reach the zirconium ignition temperature in less than 10 hours. This time period is based on a heat up calculation which uses several simplifying assumptions. Some of these assumptions are conservative (adiabatic conditions), while others are non-conservative (no oxidation below 900°C). Weighing the conservatisms and non-conservatisms, the staff judges that this calculation reasonably represents conditions which may occur in the event of an SFP accident. The NRC staff concluded that if 10 hours were available to initiate mitigative actions, or if needed, offsite protective actions using Comprehensive Emergency

TABLE 1
EXEMPTIONS REQUESTED FROM 10 CFR 50.47(b) AND (c)(2)

Bold ~~strikethrough~~ text identifies the proposed exemption with respect to the regulation. The basis for the exemption explains the scope of the exception.

Item	10 CFR 50.47 Emergency Plans	Basis for Exemption
		Management Plan (CEMP), formal offsite radiological emergency plans would not be necessary for a permanently defueled nuclear power reactor licensee. As supported by the licensee's SFP analysis, the NRC staff considers an exemption from the requirements for formal offsite radiological emergency plans is justified for a zirconium fire scenario considering the low likelihood of this event together with time available to take mitigative or protective actions between the initiating event and before the onset of a postulated fire. TMI-1 has an analysis (Reference 9) that demonstrates that 365 days after permanent shutdown, the radiological consequences of the analyzed design basis accident (DBA) will not exceed the limits of the U.S. Environmental Protection Agency's (EPA's) Protective Action Guides (PAGs) at the exclusion area boundary (EAB). An additional analysis (Reference 7) also shows that 488 days after shutdown for an unlikely event of a beyond-DBA where the hottest fuel assembly adiabatic heat up occurs, 10 hours are available to initiate mitigative or if needed, offsite protective actions, using a CEMP from the time the fuel is uncovered until it reaches the auto-ignition temperature of 900°C. TMI-1 maintains several strategies implemented by procedures for mitigating the loss of SFP water inventory. These mitigative strategies are maintained in accordance with License Condition 2.c.(17) of the TMI-1 Renewed Facility License. These diverse strategies provide defense-in-depth and can be implemented in ample time to provide makeup water or spray to the SFP prior to the onset of zirconium cladding ignition should a very low probability beyond design basis event affect the SFP. Several means will be available to provide makeup water to the SFP, such as the Fire Service (FS) System and the portable equipment maintained in accordance with Extensive Damage Mitigating Guidelines (EDMGs) (in support of License Condition 2.c.(17)). There are diverse means to provide makeup water to the SFP

TABLE 1
EXEMPTIONS REQUESTED FROM 10 CFR 50.47(b) AND (c)(2)

Bold ~~strikethrough~~ text identifies the proposed exemption with respect to the regulation. The basis for the exemption explains the scope of the exception.

Item	10 CFR 50.47 Emergency Plans	Basis for Exemption
		with installed FS electrical and diesel driven pumps, as well as EDMG portable diesel pumps. Water sources are from the river and alternate fire service sources. Three (3) trained on-shift individuals can implement the established procedures to remove debris, route hoses, and establish an operating portable diesel pump to supply makeup water to the SFP within 4 hours, well within a 10-hour period. The three (3) on-shift individuals are assigned to perform this task; they do not have other assigned required emergency preparedness (EP) activities during the performance of this task. Direction and selection of these tasks will continue to be directed by the Certified Fuel Handler and Non-Certified Operator. Training of the on-shift staff will be maintained, and they will implement such strategies and plans to mitigate the consequences of an event involving a catastrophic loss-of-water inventory concurrently from the SFP.
2	10 CFR 50.47(b)(1) Primary responsibilities for emergency response by the nuclear facility licensee and by State and local organizations **~~within the Emergency Planning Zones~~** have been assigned, the emergency responsibilities of the various supporting organizations have been specifically established, and each principal response organization has staff to respond and to augment its initial response on a continuous basis.	Refer to basis for 10 CFR 50.47(b).
3	10 CFR 50.47(b)(2)	No exemption requested.
4	10 CFR 50.47(b)(3) Arrangements for requesting and effectively using assistance resources have been made, **~~arrangements to accommodate State and local staff at the licensee's Emergency Operations Facility have been~~**	Discontinuing offsite emergency planning activities and reducing the scope of onsite emergency planning is acceptable given the significantly reduced offsite consequences when TMI-1 is in the permanently defueled condition. The TMI

TABLE 1
EXEMPTIONS REQUESTED FROM 10 CFR 50.47(b) AND (c)(2)

Bold ~~strikethrough~~ text identifies the proposed exemption with respect to the regulation. The basis for the exemption explains the scope of the exception.

Item	10 CFR 50.47 Emergency Plans	Basis for Exemption
	~~made,~~ and other organizations capable of augmenting the planned response have been identified.	emergency plan will continue to maintain arrangements for requesting and using assistance resources from offsite support organizations. Decommissioning power reactors present a low likelihood of any credible accident resulting in a radiological release together with the time available to take mitigative or, if needed, offsite protective actions using a CEMP between the initiating event and before the onset of a postulated fire. As such, an Emergency Operations Facility would not be required. The Control Room or other onsite location can provide for the communication and coordination with offsite organizations for the level of support required. Also refer to basis for 10 CFR 50.47(b).
5	10 CFR 50.47(b)(4) A standard emergency classification and action level scheme, the basis of which includes facility system and effluent parameters, is in use by the nuclear facility licensee**~~, and State and local response plans call for reliance on information provided by facility licensees for determinations of minimum initial offsite response measures~~**.	Decommissioning power reactors present a low likelihood of any credible accident resulting in a radiological release together with the time available to take mitigative or, if needed, offsite protective actions using a CEMP between the initiating event and before the onset of a postulated fire. As such, formal offsite radiological emergency response plans are not required. TMI will adopt the Permanently Defueled Emergency Action Levels (EALs) consistent with those detailed in Appendix C of Nuclear Energy Institute (NEI) 99-01, "Development of Emergency Action Levels for Non-Passive Reactors," Revision 6 (Reference 10), endorsed by the NRC in a letter dated March 28, 2013 (Reference 11). A site-specific TMI-1 analysis (Reference 7) shows that after the spent fuel has decayed for 488 days, for beyond design basis events where the SFP is drained, and air cooling is not possible, 10 hours is available to take mitigative or, if needed, offsite protective actions using a comprehensive approach to emergency planning from the time spent fuel cooling is lost until the hottest fuel assembly reaches a temperature of 900°C. No offsite protective actions are anticipated to be necessary. Therefore, classification above the Alert level (e.g., Site Area Emergency or General Emergency) will no longer be required.

TABLE 1
EXEMPTIONS REQUESTED FROM 10 CFR 50.47(b) AND (c)(2)

Bold ~~strikethrough~~ text identifies the proposed exemption with respect to the regulation. The basis for the exemption explains the scope of the exception.

Item	10 CFR 50.47 Emergency Plans	Basis for Exemption
		Also refer to basis for 10 CFR 50.47(b).
6	10 CFR 50.47(b)(5) Procedures have been established for notification, by the licensee, of State and local response organizations and for notification of emergency personnel by all organizations; the content of initial and follow up messages to response organizations **~~and the public~~** has been established**~~; and means to provide early notification and clear instruction to the populace within the plume exposure pathway Emergency Planning Zone have been established~~**.	Per SECY-00-0145 (Reference 12), after approximately 1 year of spent fuel decay time (and as supported by the SFP analysis), the NRC staff considers an exception to the offsite EPA PAG standard is justified for a zirconium fire scenario considering the low likelihood of this event together with time available to take mitigative or protective actions between the initiating event and before the onset of a postulated fire. SECY-13-0112, "Consequence Study of a Beyond-Design-Basis Earthquake Affecting the Spent Fuel Pool for a U.S. Mark I Boiling Water Reactor," (Reference 13) provides that depending on the size of the pool liner leak, releases could start anywhere from eight hours to several days after the leak starts, assuming that mitigation measures are unsuccessful. If 10 CFR 50.54(hh)(2)-type mitigation measures are successful, releases could only occur during the first several days after the fuel was removed from the reactor. As previously indicated, a TMI-1 analysis shows that after the spent fuel has decayed for 488 days, for beyond design basis events where the SFP is drained, and air cooling is not possible, 10 hours is available to take mitigative or, if needed, offsite protective actions using a comprehensive approach to emergency planning from the time spent fuel cooling is lost until the hottest fuel assembly reaches a temperature of 900°C. Therefore, offsite emergency plans for the populace within the plume exposure pathway Emergency Planning Zone are not necessary for permanently defueled nuclear power plants. Refer to basis for 10 CFR 50.47(b).
7	10 CFR 50.47(b)(6) Provisions exist for prompt communications among principal response organizations to emergency personnel **~~and to the public~~**.	Refer to basis for 10 CFR 50.47(b).

TABLE 1
EXEMPTIONS REQUESTED FROM 10 CFR 50.47(b) AND (c)(2)

Bold ~~strikethrough~~ text identifies the proposed exemption with respect to the regulation. The basis for the exemption explains the scope of the exception.

Item	10 CFR 50.47 Emergency Plans	Basis for Exemption
8	10 CFR 50.47(b)(7) **~~Information is made available to the public on a periodic basis on how they will be notified and what their initial actions should be in an emergency (e.g., listening to a local broadcast station and remaining indoors),~~** [T]he principal points of contact with the news media for dissemination of information during an emergency **~~(including the physical location or locations)~~** are established in advance, and procedures for coordinated dissemination of information to the public are established.	Refer to basis for 10 CFR 50.47(b).
9	10 CFR 50.47(b)(8)	No exemption requested.
10	10 CFR 50.47(b)(9) Adequate methods, systems, and equipment for assessing and monitoring actual or potential **~~offsite~~** consequences of a radiological emergency condition are in use.	Refer to basis for 10 CFR 50.47(b)
11	10 CFR 50.47(b)(10) A range of protective actions has been developed **~~for the plume exposure pathway EPZ~~** for emergency workers and the public. **~~In developing this range of actions, consideration has been given to evacuation, sheltering, and, as a supplement to these, the prophylactic use of potassium iodide (KI), as appropriate. Evacuation time estimates have been developed by applicants and licensees. Licensees shall update the evacuation time estimates on a periodic basis. Guidelines for the choice of protective actions during an emergency, consistent with Federal guidance, are developed and in place, and protective actions for~~**	TMI-1 has developed an analysis indicating that 488 days after permanent cessation of power operations, no credible or beyond design basis accident at TMI-1 will result in radiological releases requiring offsite protective actions. The analysis of the potential radiological impact of the postulated accident for TMI-1 in a permanently defueled condition indicates that any releases beyond the site boundary are limited to small fractions of the EPA PAG exposure levels. In the unlikely event of a SFP accident, the iodine isotopes which contribute to an offsite dose from an operating reactor accident are not present, so potassium iodide (KI) distribution offsite would no longer serve as an effective or necessary supplemental protective action. Because it is not possible for PAGs to be exceeded at TMI-1 488 days after permanent cessation of power operations, evacuation planning, including

TABLE 1
EXEMPTIONS REQUESTED FROM 10 CFR 50.47(b) AND (c)(2)

Bold ~~strikethrough~~ text identifies the proposed exemption with respect to the regulation. The basis for the exemption explains the scope of the exception.

Item	10 CFR 50.47 Emergency Plans	Basis for Exemption
	~~the ingestion exposure pathway EPZ appropriate to the locale have been developed.~~	evacuation time estimates, is not needed since TMI-1 will meet the criteria for an exemption from offsite emergency preparedness requirements as discussed in the exemption from 10 CFR 50.47(b). Also refer to basis for 10 CFR 50.47(b).
12	10 CFR 50.47(b)(11) through (b)(16)	No exemption requested.
13	10 CFR 50.47(c)(1)	No exemption requested.
14	10 CFR 50.47(c)(2) **~~Generally, the plume exposure pathway EPZ for nuclear power plants shall consist of an area about 10 miles (16 km) in radius and the ingestion pathway EPZ shall consist of an area about 50 miles (80 km) in radius. The exact size and configuration of the EPZs surrounding a particular nuclear power reactor shall be determined in relation to local emergency response needs and capabilities as they are affected by such conditions as demography, topography, land characteristics, access routes, and jurisdictional boundaries.~~** The size of the EPZs **~~also~~** may be determined on a case-by-case basis for gas-cooled nuclear reactors and for reactors with an authorized power level less than 250 MW thermal. **~~The plans for the ingestion pathway shall focus on such actions as are appropriate to protect the food ingestion pathway.~~**	TMI-1 has developed an analysis indicating that 488 days after permanent cessation of power operations, no credible or beyond design basis accident at TMI-1 will result in radiological releases requiring offsite protective actions. The analysis of the potential radiological impact of the postulated accident for TMI-1 in a permanently defueled condition indicates that any releases beyond the site boundary are limited to small fractions of the EPA PAG exposure levels. Refer to basis for 10 CFR 50.47(b)(10).

TABLE 2
EXEMPTIONS REQUESTED FROM 10 CFR PART 50, APPENDIX E

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Item	10 CFR PART 50, APPENDIX E, SECTION IV	Basis for Exemption
1	**IV Content of Emergency Plans** 1. The applicant's emergency plans shall contain, but not necessarily be limited to, information needed to demonstrate compliance with the elements set forth below, i.e., organization for coping with radiological emergencies, assessment actions, activation of emergency organization, notification procedures, emergency facilities and equipment, training, maintaining emergency preparedness, [and] recovery**~~, and onsite protective actions during hostile action~~**. In addition, the emergency response plans submitted by an applicant for a nuclear power reactor operating license under this Part, or for an early site permit (as applicable) or combined license under 10 CFR Part 52, shall contain information needed to demonstrate compliance with the standards described in § 50.47(b), and they will be evaluated against those standards.	Following docketing of the "Certification of Permanent Removal of Fuel from the Reactor Vessel," in accordance with 10 CFR 50.82(a)(1)(i) and (ii), TMI-1 will become a permanently shutdown facility with spent fuel stored in the SFP. In the EP Final Rule (76 FR 72560, Nov. 23, 2011) (Reference 14), the NRC defined "hostile action" as, in part, an act directed toward a nuclear power plant or its personnel. This definition is based on the definition of "hostile action" provided in NRC Bulletin 2005-02, "Emergency Preparedness and Response Actions for Security-Based Events," dated July 18, 2005 (Reference 15). NRC Bulletin 2005-02 was not applicable to nuclear power reactors that have permanently ceased operations and have certified that fuel has been removed from the reactor vessel. The NRC excluded non-power reactors from the definition of "hostile action" at the time of the rulemaking because, as defined in 10 CFR 50.2, a non-power reactor is not considered a nuclear power reactor and a regulatory basis had not been developed to support the inclusion of non-power reactors (NPR) in the definition of "hostile action." Similarly, a decommissioning power reactor or ISFSI is not a "nuclear reactor" as defined in the NRC's regulations. A decommissioning power reactor also has a low likelihood of a credible accident resulting in radiological releases requiring offsite protective measures. For all of these reasons, the NRC staff has concluded that a decommissioning power reactor is not a facility that falls within the definition of "hostile action." Similarly, for security, risk insights can be used to determine which targets are important to protect against sabotage. A level of security commensurate with the consequences of a sabotage event is required and is evaluated on a site-specific basis. The severity of the consequences declines as fuel ages and, thereby, removes over time the underlying concern that a sabotage attack, under the current definition, could cause offsite radiological consequences. Although, this analysis provides a justification for an exemption to include the definition for a "hostile action" and its related requirements, elements for security-based events would be maintained. The classification of security-based events,

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EXEMPTIONS REQUESTED FROM 10 CFR PART 50, APPENDIX E

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Item	10 CFR PART 50, APPENDIX E, SECTION IV	Basis for Exemption
		notification of offsite authorities and coordination with offsite agencies under a CEMP would still be required. Other security-related requirements in the EP Final Rule would be exempted such as, on-shift staffing analysis, emergency response organization (ERO) augmentation and alternative facilities, protection of onsite personnel, and challenging drills and exercises due to the reduced radiological risk for a decommissioning power reactor. The following similarities between TMI and NPRs show that the TMI facility should be treated in a similar fashion as an NPR. Similar to NPRs, TMI will pose lower radiological risks to the public from accidents than do power reactors because: 1) TMI-1 will be a permanently shutdown facility (with fuel stored in the SFP and ISFSI) and will no longer generate fission products; 2) fuel stored in the TMI-1 SFP will have lower decay heat resulting in lower risk of fission product release in the event of a beyond design basis boil off or drain down event; and 3) no credible or beyond design basis accident at TMI-1 will result in radiological releases requiring offsite protective actions.
2	2. **~~This nuclear power reactor license applicant shall also provide an analysis of the time required to evacuate various sectors and distances within the plume exposure pathway EPZ for transient and permanent populations, using the most recent U.S. Census Bureau data as of the date the applicant submits its application to the NRC~~**.	Refer to basis for 10 CFR 50.47(b)(10)
3	3. **~~Nuclear power reactor licensees shall use NRC approved evacuation time estimates (ETEs) and updates to the ETEs in the formulation of protective action recommendations and shall provide the ETEs and ETE updates to State and local governmental authorities for use in developing offsite protective action strategies.~~**	Refer to basis for 10 CFR 50.47(b)(10)

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4	4. **~~Within 365 days of the later of the date of the availability of the most recent decennial census data from the U.S. Census Bureau or December 23, 2011, nuclear power reactor licensees shall develop an ETE analysis using this decennial data and submit it under § 50.4 to the NRC. These licensees shall submit this ETE analysis to the NRC at least 180 days before using it to form protective action recommendations and providing it to State and local governmental authorities for use in developing offsite protective action strategies.~~**	Refer to basis for 10 CFR 50.47(b)(10)
5	5. **~~During the years between decennial censuses, nuclear power reactor licensees shall estimate EPZ permanent resident population changes once a year, but no later than 365 days from the date of the previous estimate, using the most recent U.S. Census Bureau annual resident population estimate and State/local government population data, if available. These licensees shall maintain these estimates so that they are available for NRC inspection during the period between decennial censuses and shall submit these estimates to the NRC with any updated ETE analysis.~~**	Refer to basis for 10 CFR 50.47(b)(10).
6	6. **~~If at any time during the decennial period, the EPZ permanent resident population increases such that it causes the longest ETE value for the 2-mile zone or 5-mile zone, including all affected Emergency Response Planning Areas, or for the entire 10-mile EPZ to increase by 25 percent or 30 minutes, whichever is less, from the nuclear power reactor licensee's currently NRC approved or updated ETE, the licensee shall update the ETE analysis to reflect the impact of that population~~**	Refer to basis for 10 CFR 50.47(b)(10)

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Item	10 CFR PART 50, APPENDIX E, SECTION IV	Basis for Exemption
	~~increase. The licensee shall submit the updated ETE analysis to the NRC under § 50.4 no later than 365 days after the licensee's determination that the criteria for updating the ETE have been met and at least 180 days before using it to form protective action recommendations and providing it to State and local governmental authorities for use in developing offsite protective action strategies.~~	
7	After an applicant for a combined license <...>	No exemption requested.
8	*A. Organization* The organization for coping <...>	No exemption requested.
9	A.1. A description of the normal plant **~~operating~~** organization.	Once TMI-1 is permanently shut down and defueled, a decommissioning reactor will not be authorized to operate under 10 CFR 50.82(a). Because the TMI-1 cannot operate the reactor, a "plant operating organization" will no longer be required. Rather, the facility will be maintained by a defueled on-shift staff.
10	A.2.	No exemption requested.
11	A.3. **~~A description, by position and function to be performed, of the licensee's headquarters personnel who will be sent to the plant site to augment the onsite emergency organization.~~**	The number of staff at TMI-1 during decommissioning will be small but commensurate with the need to safely store spent fuel at the facility in a manner that is protective of public health and safety. TMI-1 will have a level of emergency response that does not require response by headquarters personnel. The on-shift and emergency response positions will be defined in the Permanently Defueled Emergency Plan (PDEP).
12	A.4. Identification, by position and function to be performed, of persons within the licensee organization who will be responsible for making **~~offsite~~** dose projections, and a description of how these projections will be made and the	TMI-1 has developed an analysis indicating that 488 days after permanent cessation of power operations, no credible or beyond design basis accident at TMI-1 will result in radiological releases requiring offsite protective actions. TMI-1 will maintain the capability to determine if a radiological release is occurring. If a release is occurring, TMI-1 will promptly communicate that information to offsite authorities

TABLE 2
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Item	10 CFR PART 50, APPENDIX E, SECTION IV	Basis for Exemption
	results transmitted to State and local authorities, NRC, and other appropriate governmental entities.	for their consideration. The offsite organizations are responsible for deciding what, if any, protective actions should be taken based on a CEMP.
13	A.5. **~~Identification, by position and function to be performed, of other employees of the licensee with special qualifications for coping with emergency conditions that may arise. Other persons with special qualifications, such as consultants, who are not employees of the licensee and who may be called upon for assistance for emergencies shall also be identified. The special qualifications of these persons shall be described.~~**	As indicated by the TMI-1 adiabatic heat up analysis, the time available to initiate compensatory actions in the event of a loss of SFP cooling or inventory precludes the need to identify and describe the special qualifications of these individuals in the emergency plan. The number of staff at TMI-1 during decommissioning will be small but commensurate with the need to maintain the facility in a manner that is protective of public health and safety. Also refer to basis for 10 CFR 50.47(b).
14	A.6.	No exemption requested.
15	A.7. **~~By June 23, 2014,~~** [I]dentification of**~~, and a description of~~** the assistance expected from, appropriate State, local, and Federal agencies with responsibilities for coping with emergencies, including **~~hostile action at the site. For purposes of this appendix, "hostile action" is defined as~~** an act directed toward a nuclear power plant or its personnel that includes the use of violent force to destroy equipment, take hostages, and/or intimidate the licensee to achieve an end. This includes attack by air, land, or water using guns, explosives, projectiles, vehicles, or other devices used to deliver destructive force.	A decommissioning power reactor has a low likelihood of a credible accident resulting in radiological releases requiring offsite protective measures. For this reason and those described in the basis for 10 CFR Part 50, Appendix E, Section IV.1, a decommissioning power reactor is not a facility that falls within the definitions of "hostile action." Similarly, for security, risk insights can be used to determine which targets are important to protect against sabotage. A level of security commensurate with the consequences of a sabotage event is required and is evaluated on a site-specific basis. The severity of the consequences declines as fuel ages, and over time, the underlying concern that a sabotage attack could cause offsite radiological consequences is removed. Although the analysis described above and in the basis for 10 CFR Part 50, Appendix E, Section IV.1 provides a justification for exempting TMI-1 from "hostile action" related requirements, some EP requirements for security-based events will be maintained. Protective actions are maintained for onsite personnel through the classification of security-based events, notification of offsite authorities, and

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Item	10 CFR PART 50, APPENDIX E, SECTION IV	Basis for Exemption
		coordination of offsite response organizations (i.e., local law enforcement, firefighting, medical assistance) onsite under a CEMP concept. Refer to basis for 10 CFR Part 50, Appendix E, Section IV.1.
16	A.8. **~~Identification of the State and/or local officials responsible for planning for, ordering and controlling appropriate protective actions, including evacuations when necessary.~~**	Offsite emergency measures are limited to support provided by local police, fire departments, and ambulance and hospital services, as appropriate. Because an analysis has been developed indicating that 488 days after permanent cessation of power operations and due to the low probability of design basis accidents or other credible events to exceed the EPA PAGs, protective actions such as evacuation should not be required, but could be implemented at the discretion of offsite authorities using a CEMP. Also refer to basis for 50.47(b)(10).
17	A.9. **~~By December 24, 2012, for nuclear power reactor licensees, a detailed analysis demonstrating that on-shift personnel assigned emergency plan implementation functions are not assigned responsibilities that would prevent the timely performance of their assigned functions as specified in the emergency plan.~~**	Responsibilities of the on-shift and emergency response personnel will be detailed in the Permanently Defueled Emergency Plan and implementing procedures and will be regularly tested through drills and exercises, and audited and inspected by Exelon and the NRC. The duties of the on-shift personnel at a decommissioning reactor facility are not as complicated and diverse as those for an operating power reactor. In the EP Final Rule (Reference 14), the NRC acknowledged that the staffing analysis requirement was not necessary for non-power reactor licensees because staffing at non-power reactors is generally small, which is commensurate with operating the facility in a manner that is protective of the public health and safety. The minimal systems and equipment needed to maintain the spent nuclear fuel in the SFP or in a dry cask storage system in a safe condition requires minimal personnel and is governed by Technical Specifications. Because of the slow rate of the event scenarios postulated in the design basis accident and postulated beyond design basis accident analyses and because the duties of the on-shift personnel at a decommissioning reactor facility are not as complicated and diverse as those for an operating reactor, significant time is available to complete actions necessary to

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		mitigate an emergency without impeding timely performance of emergency plan functions. For all these reasons, it can be concluded that a decommissioning nuclear power plant (NPP) is exempt from the requirement of 10 CFR Part 50, Appendix E, Section IV.A.9.
18	*B. Assessment Actions* B.1. The means to be used for determining the magnitude of, and for continually assessing the impact of, the release of radioactive materials shall be described, including emergency action levels that are to be used as criteria for determining the need for notification and participation of local and State agencies, the Commission, and other Federal agencies, and the emergency action levels that are to be used for determining when and what type of protective measures should be considered within **~~and outside~~** the site boundary to protect health and safety. The emergency action levels shall be based on in-plant conditions and instrumentation in addition to onsite **~~and offsite~~** monitoring. **~~By June 20, 2012, for nuclear power reactor licensees, these action levels must include hostile action that may adversely affect the nuclear power plant.~~** The initial emergency action levels shall be discussed and agreed on by the applicant or licensee and State and local governmental authorities, and approved by the NRC. Thereafter, emergency action levels shall be reviewed with the State and local governmental authorities on an annual basis.	TMI EALs will be developed consistent with the Permanently Defueled EALs detailed in Appendix C of NEI 99-01, Revision 6 (Reference 10), which the NRC found to be an acceptable method for development of EALs. TMI-1 will continue to review EALs with the Commonwealth of Pennsylvania on an annual basis. However, based upon the reduced scope of EALs for the permanently defueled facility, the scope of the annual review of EALs is expected to be limited (i.e., informal mailings, etc.). Also, refer to basis for 10 CFR Part 50, Appendix E, Section IV.1 for the justification from the requirements in Appendix E related to "hostile action."
19	B.2.	No exemption requested.
20	*C. Activation of Emergency Organization*	The Permanently Defueled EALs, developed consistent with Appendix C of NEI 99-01, Revision 6 (Reference 10), will be adopted, as previously described. This

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	C.1. The entire spectrum of emergency conditions that involve the alerting or activating of progressively larger segments of the total emergency organization shall be described. The communication steps to be taken to alert or activate emergency personnel under each class of emergency shall be described. Emergency action levels (based not only on onsite **~~and offsite~~** radiation monitoring information but also on readings from a number of sensors that indicate a potential emergency**~~, such as the pressure in containment and the response of the Emergency Core Cooling System~~**) for notification of offsite agencies shall be described. The existence, but not the details, of a message authentication scheme shall be noted for such agencies. The emergency classes defined shall include: (1) notification of unusual events, (2) alert**~~, (3) site area emergency, and (4) general emergency~~**. These classes are further discussed in NUREG-0654/FEMA-REP-1.	scheme eliminates the Site Area Emergency and General Emergency event classifications. Additionally, the need to base EALs on containment parameters is no longer appropriate since these parameters do not provide indication of the conditions at a defueled facility and emergency core cooling systems are no longer required. Other indications, such as SFP level or temperature, can be used at sites where there is spent fuel in the SFPs. The EAL scheme presented in NEI 99-01, Revision 6 was endorsed by the NRC in a letter dated March 28, 2013 (Reference 11). No offsite protective actions are anticipated to be necessary, since classification above the Alert (e.g., Site Area Emergency or General Emergency) level is no longer required. In the event of an accident at a defueled facility that meets the conditions for relaxation of emergency planning requirements, there will be available time for event mitigation, and if necessary, implementation of offsite protective actions using a comprehensive approach to emergency planning. See the basis for 10 CFR 50.47(b) detailing the low likelihood of any credible accident resulting in radiological releases requiring offsite protective measures. In the Statement of Considerations for the Final Rule for EP requirements for ISFSIs and for MRS facilities (60 FR 32430; June 22, 1995) (Reference 8), the Commission responded to comments concerning a general emergency at an ISFSI and MRS, and concluded that, "...an essential element of a General Emergency is that a release can be reasonably expected to exceed EPA Protective Action Guidelines exposure levels off site for more than the immediate site area." The probability of a condition reaching the level above an emergency classification of Alert is very low. In the event of an accident at TMI that meets the criteria for an exemption from the NRC's offsite EP requirements, there will be time available to initiate mitigative actions consistent with plant conditions, and if necessary, for offsite authorities to employ their CEMP to take protective actions. As stated in NUREG-1738, "Technical Study of Spent Fuel Pool Accident Risk at Decommissioning Nuclear Power Plants" (February 2001) (Reference 16) for

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		instances of small SFP leaks or loss of cooling scenarios, these events evolve very slowly and generally leave many days for recovery efforts. Offsite radiation monitoring will be performed as the need arises. Due to the decreased risks associated with defueled plants, offsite radiation monitoring systems are not required. Refer to basis for 10 CFR Part 50, Appendix E, Section IV.B.1.
21	C.2. **~~By June 20, 2012, nuclear power reactor~~** Licensees shall establish and maintain the capability to assess, classify, and declare an emergency condition **~~within 15 minutes~~** after the availability of indications to plant operators that an emergency action level has been exceeded and shall promptly declare the emergency condition as soon as possible following identification of the appropriate emergency classification level. Licensees shall not construe these criteria as a grace period to attempt to restore plant conditions to avoid declaring an emergency action due to an emergency action level that has been exceeded. Licensees shall not construe these criteria as preventing implementation of response actions deemed by the licensee to be necessary to protect public health and safety provided that any delay in declaration does not deny the State and local authorities the opportunity to implement measures necessary to protect the public health and safety.	In the Statement of Consideration for the EP Final Rule published in the Federal Register (76 FR 72560) (Reference 14), non-power reactor licensees were not required to assess, classify and declare an emergency condition within 15 minutes. A SFP and an ISFSI are also not nuclear power reactors as defined in the NRC's regulations. A decommissioning power reactor has a low likelihood of a credible accident resulting in radiological releases requiring offsite protective measures. For these reasons, the staff concludes that a decommissioning power reactor should not be required to assess, classify, and declare an emergency condition within 15 minutes. TMI will maintain the capability to assess, classify, and declare an emergency condition. Emergency declaration is required to be made as soon as conditions warranting classification are present and recognizable, but within 30 minutes after the availability of indications to operators that an EAL threshold has been reached. In the permanently defueled condition, the rapidly developing scenarios associated with events initiated during reactor power operation are no longer credible. The consequences resulting from the only remaining events (e.g., fuel handling accident) develop over a significantly longer period. As such, the 15-minute requirement to assess, classify and declare an emergency is unnecessarily restrictive. See the basis for 10 CFR 50.47(b) detailing the low likelihood of any credible accident resulting in radiological releases requiring offsite protective measures and 10 CFR Part 50, Appendix E, Section IV.1.

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22	*D. Notification Procedures* D.1. Administrative and physical means for notifying local, State, and Federal officials and agencies **~~and agreements reached with these officials and agencies for the prompt notification of the public and for public evacuation or other protective measures, should they become necessary,~~** shall be described. This description shall include identification **~~of the appropriate officials, by title and agency,~~** of the State and local government agencies **~~within the EPZs~~**.	Refer to basis for 10 CFR 50.47(b) and 50.47(b)(10).
23	D.2. **~~Provisions shall be described for yearly dissemination to the public within the plume exposure pathway EPZ of basic emergency planning information, such as the methods and times required for public notification and the protective actions planned if an accident occurs, general information as to the nature and effects of radiation, and a listing of local broadcast stations that will be used for dissemination of information during an emergency. Signs or other measures shall also be used to disseminate to any transient population within the plume exposure pathway EPZ appropriate information that would be helpful if an accident occurs.~~**	Refer to basis for 10 CFR 50.47(b) and 50.47(b)(10).
24	D.3. A licensee shall have the capability to notify responsible State and local governmental agencies **~~within 15 minutes~~** after declaring an emergency. **~~The licensee shall demonstrate that the appropriate governmental authorities have the capability to make a public alerting and notification decision promptly on being informed~~**	TMI-1 proposes to complete emergency notifications within 30 minutes after the event classification has been made. This timeframe is consistent with the 10 CFR 50.72(a)(3) notification to the NRC and is appropriate because in the permanently defueled condition, the rapidly developing scenarios associated with events initiated during reactor power operation are no longer credible and there is no need for State or local response organizations to implement any protective actions.

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	~~by the licensee of an emergency condition. Prior to initial operation greater than 5 percent of rated thermal power of the first reactor at the site, each nuclear power reactor licensee shall demonstrate that administrative and physical means have been established for alerting and providing prompt instructions to the public within the plume exposure pathway EPZ. The design objective of the prompt public alert and notification system shall be to have the capability to essentially complete the initial alerting and notification of the public within the plume exposure pathway EPZ within about 15 minutes. The use of this alerting and notification capability will range from immediate alerting and notification of the public (within 15 minutes of the time that State and local officials are notified that a situation exists requiring urgent action) to the more likely events where there is substantial time available for the appropriate governmental authorities to make a judgment whether or not to activate the public alert and notification system. The alerting and notification capability shall additionally include administrative and physical means for a backup method of public alerting and notification capable of being used in the event the primary method of alerting and notification is unavailable during an emergency to alert or notify all or portions of the plume exposure pathway EPZ population. The backup method shall have the capability to alert and notify the public within the plume exposure pathway EPZ, but does not need to meet the 15-minute design objective for the primary prompt public alert and notification system. When there is a decision to activate the alert and~~	Because of the geographic location of TMI, emergency planning and responsibilities have historically involved coordination with the Commonwealth of Pennsylvania. Decommissioning-related emergency plan submittals for TMI have been discussed with offsite response organizations since Exelon provided notification that it would permanently cease power operations at TMI-1. These discussions have addressed changes to onsite and offsite emergency preparedness throughout the decommissioning process, including the proposed time of 30 minutes to notify the state after the event classification has been made. Pennsylvania Emergency Management officials have been able to review and concur with this proposal. The State's acknowledgement of their review will be provided with the PDEP submittal. Also refer to basis for 10 CFR 50.47(b) and 50.47(b)(10).

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	~~notification system, the appropriate governmental authorities will determine whether to activate the entire alert and notification system simultaneously or in a graduated or staged manner. The responsibility for activating such a public alert and notification system shall remain with the appropriate governmental authorities.~~	
25	D.4. **~~If FEMA has approved a nuclear power reactor site's alert and notification design report, including the backup alert and notification capability, as of December 23, 2011, then the backup alert and notification capability requirements in Section IV.D.3 must be implemented by December 24, 2012. If the alert and notification design report does not include a backup alert and notification capability or needs revision to ensure adequate backup alert and notification capability, then a revision of the alert and notification design report must be submitted to FEMA for review by June 24, 2013, and the FEMA-approved backup alert and notification means must be implemented within 365 days after FEMA approval. However, the total time period to implement a FEMA-approved backup alert and notification means must not exceed June 22, 2015.~~**	Refer to basis for 10 CFR Part 50, Appendix E, Section IV.D.3. regarding the alert and notification system requirements.
26	*E. Emergency Facilities and Equipment* E.1 thru E.7	 No exemption requested.
27	E.8.a.(i) A licensee **~~onsite technical support center and an emergency operations~~** facility from which effective	The TMI-1 analysis indicates that within 488 days after shutdown, no design basis accidents or other credible event at TMI-1 will exceed the EPA PAGs. Due to the low probability of design basis accidents or other credible events to exceed the EPA PAGs at the site boundary, the available time for event mitigation at a

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	direction can be given and effective control can be exercised during an emergency;	decommissioning power reactor and, if needed, to implement offsite protective actions using a CEMP, an emergency operations facility (EOF) would not be required to support offsite agency response. Onsite actions may be directed from the Control Room or other location, without the requirements imposed on a technical support center (TSC). An onsite facility will continue to be maintained, from which effective direction can be given and effective control may be exercised during an emergency. The TMI emergency plan will continue to maintain arrangements for requesting assistance and using resources from appropriate offsite support organizations. Refer to basis for 10 CFR 50.47(b)(3).
28	E.8.a.(ii) **~~For nuclear power reactor licensees, a licensee onsite operational support center;~~**	NUREG-0696, "Functional Criteria for Emergency Response Facilities," (Reference 17) provides that the operational support center (OSC) is an onsite area separate from the Control Room and the TSC where licensee operations support personnel will assemble in an emergency. For a permanently shutdown and defueled power plant, an OSC is no longer required to meet its original purpose of an assembly area for plant logistical support during an emergency. The Control Room is the single onsite facility that provides support, emergency mitigation, radiation monitoring, and effective control that will be exercised during an emergency.
29	E.8.b. **~~For a nuclear power reactor licensee's emergency operations facility required by paragraph 8.a of this section, either a facility located between 10 miles and 25 miles of the nuclear power reactor site(s), or a primary facility located less than 10 miles from the nuclear power reactor site(s) and a backup facility located between 10 miles and 25 miles of the nuclear power reactor site(s). An emergency operations facility may serve more than one nuclear power reactor site. A licensee desiring to locate an emergency operations facility more than 25 miles~~**	In accordance with paragraph E.8.e., the requirements of paragraph 8.b do not apply to the TMI-1 EOF because it was an approved facility prior to December 23, 2011. However, the exemption is requested to clearly reflect that the requirement no longer applies to TMI-1 in a permanently shutdown and defueled condition. Refer to basis for 10 CFR 50.47(b)(3).

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Bold ~~strikethrough~~ text identifies the proposed exemption with respect to the regulation. The basis for the exemption explains the scope of the exception.

Item	10 CFR PART 50, APPENDIX E, SECTION IV	Basis for Exemption
	~~from a nuclear power reactor site shall request prior Commission approval by submitting an application for an amendment to its license.~~ **~~For an emergency operations facility located more than 25 miles from a nuclear power reactor site, provisions must be made for locating NRC and offsite responders closer to the nuclear power reactor site so that NRC and offsite responders can interact face-to-face with emergency response personnel entering and leaving the nuclear power reactor site. Provisions for locating NRC and offsite responders closer to a nuclear power reactor site that is more than 25 miles from the emergency operations facility must include the following:~~** (1) **~~Space for members of an NRC site team and Federal, State, and local responders;~~** (2) **~~Additional space for conducting briefings with emergency response personnel;~~** (3) **~~Communication with other licensee and offsite emergency response facilities;~~** (4) **~~Access to plant data and radiological information; and~~** (5) **~~Access to copying equipment and office supplies;~~**	
30	E.8.c. **~~By June 20, 2012, for a nuclear power reactor licensee's emergency operations facility required by paragraph 8.a of this section, a facility having the following capabilities:~~**	Refer to basis for 10 CFR Part 50, Appendix E, Section IV.E.8.a.(i) and 10 CFR 50.47(b)(3).

TABLE 2
EXEMPTIONS REQUESTED FROM 10 CFR PART 50, APPENDIX E

Bold ~~strikethrough~~ text identifies the proposed exemption with respect to the regulation. The basis for the exemption explains the scope of the exception.

Item	10 CFR PART 50, APPENDIX E, SECTION IV	Basis for Exemption
	(1) **~~The capability for obtaining and displaying plant data and radiological information for each reactor at a nuclear power reactor site and for each nuclear power reactor site that the facility serves;~~** (2) **~~The capability to analyze plant technical information and provide technical briefings on event conditions and prognosis to licensee and offsite response organizations for each reactor at a nuclear power reactor site and for each nuclear power reactor site that the facility serves; and~~** (3) **~~The capability to support response to events occurring simultaneously at more than one nuclear power reactor site if the emergency operations facility serves more than one site; and~~**	
31	E.8.d. **~~For nuclear power reactor licensees, an alternative facility (or facilities) that would be accessible even if the site is under threat of or experiencing hostile action, to function as a staging area for augmentation of emergency response staff and collectively having the following characteristics: the capability for communication with the emergency operations facility, control room, and plant security; the capability to perform offsite notifications; and the capability for engineering assessment activities, including damage control team planning and preparation, for use when onsite emergency facilities cannot be safely accessed during hostile action. The requirements in this paragraph 8.d must be implemented no later than December 23, 2014, with the~~**	Refer to basis for 10 CFR Part 50, Appendix E, Section IV.1. regarding "hostile action."

TABLE 2
EXEMPTIONS REQUESTED FROM 10 CFR PART 50, APPENDIX E

Bold ~~strikethrough~~ text identifies the proposed exemption with respect to the regulation. The basis for the exemption explains the scope of the exception.

Item	10 CFR PART 50, APPENDIX E, SECTION IV	Basis for Exemption
	~~exception of the capability for staging emergency response organization personnel at the alternative facility (or facilities) and the capability for communications with the emergency operations facility, control room, and plant security, which must be implemented no later than June 20, 2012.~~	
32	E.8.e. **~~A licensee shall not be subject to the requirements of paragraph 8.b of this section for an existing emergency operations facility approved as of December 23, 2011;~~**	Refer to basis for 10 CFR Part 50, Appendix E, Section IV.E.8.b and 10 CFR 50.47(b)(3).
33	E.9.a. Provisions for communications with contiguous State/local governments **~~within the plume exposure pathway EPZ~~**. Such communication shall be tested monthly.	Refer to basis for 10 CFR 50.47(b) and (b)(10). TMI-1 will maintain communications with the Commonwealth of Pennsylvania and the NRC. Existing commercial phone lines will to be used to communicate EP notifications to the Commonwealth of Pennsylvania and will continue to be functionally tested monthly.
34	E.9.b	No exemption requested
35	E.9.c. **~~Provision for communications among the nuclear power reactor control room, the onsite technical support center, and the emergency operations facility; and among the nuclear facility, the principal State and local emergency operations centers, and the field assessment teams. Such communications systems shall be tested annually.~~**	TMI-1 has developed an analysis indicating that 488 days after permanent cessation of power operations, no credible accident at TMI-1 will result in radiological releases requiring offsite protective actions; or in the event of beyond design basis accidents, 10 hours is available to take mitigative actions, and if needed, implement offsite protective actions using CEMP concept. Therefore, there is no need for the Technical Support Center (TSC), Emergency Operations Facility (EOF), or field assessment teams. Additionally, there is no need to maintain and test committed provisions for communications with State and local emergency operations centers (EOCs) with these facilities. An onsite facility will continue to be maintained, from which effective command and control can be maintained during an emergency. Communication with State and

TABLE 2
EXEMPTIONS REQUESTED FROM 10 CFR PART 50, APPENDIX E

Bold ~~strikethrough~~ text identifies the proposed exemption with respect to the regulation. The basis for the exemption explains the scope of the exception.

Item	10 CFR PART 50, APPENDIX E, SECTION IV	Basis for Exemption
		local EOCs is maintained to coordinate assistance on site if required. Testing will be as described in justification for 10 CFR 50, Appendix E, Section IV.E.9.a Refer to justification for 10 CFR 50.47(b)(3) and 10 CFR Part 50, Appendix E, Section IV.E.8.a.(i).
36	E.9.d. Provisions for communications by the licensee with NRC Headquarters and the appropriate NRC Regional Office Operations Center from the **~~nuclear power reactor control room, the onsite technical support center, and the emergency operations~~** facility. Such communications shall be tested monthly.	The functions of the Control Room, EOF, TSC and OSC are intended to be combined into an onsite facility due to the smaller facility staff and the greatly reduced required interaction with State and local emergency response facilities. An onsite facility will continue to be maintained, from which effective direction can be given and effective control can be exercised during an emergency. TMI-1 will maintain communication with the NRC. Also refer to basis for 10 CFR 50.47(b).
37	*F. Training* F.1. The program to provide for: (a) The training of employees and exercising, by periodic drills, of radiation emergency plans to ensure that employees of the licensee are familiar with their specific emergency response duties, and (b) The participation in the training and drills by other persons whose assistance may be needed in the event of a radiation emergency shall be described. This shall include a description of specialized initial training and periodic retraining programs to be provided to each of the following categories of emergency personnel: i. Directors and/or coordinators of the plant emergency organization; ii. Personnel responsible for accident assessment, including control room shift personnel;	viii. The number of staff at TMI-1 during the decommissioning process will be small but commensurate with the need to safely store spent fuel at the facility in a manner that is protective of public health and safety. TMI-1 will maintain a level of emergency response that does not require additional response by headquarters personnel. The on-shift and emergency response positions are defined in the Permanently Defueled Emergency Plan and will be regularly tested through drills and exercises, audited, and inspected by Exelon and the NRC. Also see the basis for 10 CFR 50.47(b). Therefore, exempting licensee's headquarters personnel from training requirements is considered to be reasonable.

TABLE 2
EXEMPTIONS REQUESTED FROM 10 CFR PART 50, APPENDIX E

Bold ~~strikethrough~~ text identifies the proposed exemption with respect to the regulation. The basis for the exemption explains the scope of the exception.

Item	10 CFR PART 50, APPENDIX E, SECTION IV	Basis for Exemption
	iii. Radiological monitoring teams; iv. Fire control teams (fire brigades); v. Repair and damage control teams; vi. First aid and rescue teams; vii. Medical support personnel; viii. **~~Licensee's headquarters support personnel;~~** ix. Security personnel. In addition, a radiological orientation training program shall be made available to local services personnel; e.g., local emergency services**~~/Civil Defense~~**, local law enforcement personnel**~~, local news media persons~~**.	Due to the low probability of design basis accidents or other credible events to exceed the EPA PAGs, offsite emergency measures are limited to support provided by local police, fire departments and medical services, as appropriate. Therefore, the term "Civil Defense" is no longer a commonly used term and is no longer applicable as an example in the regulation. Local news media personnel no longer need radiological orientation training since they will not be called upon to support the formal Joint Information Center.
38	F.2. The plan shall describe provisions for the conduct of emergency preparedness exercises as follows: Exercises shall test the adequacy of timing and content of implementing procedures and methods, test emergency equipment and communications networks, **~~test the public alert and notification system,~~** and ensure that emergency organization personnel are familiar with their duties.	TMI-1 analyses demonstrate that 488 days after permanent cessation of power operations, no remaining postulated accidents at TMI-1 will result in radiological releases requiring offsite protective actions, or in the event of beyond design basis accidents, 10 hours is available to take mitigative actions, and if needed, implement offsite protective actions using a CEMP. Therefore, the public alert and notification system will not be used, and no testing would be required. Also refer to basis for 10 CFR 50.47(b).
39	F.2.a. **~~A full participation exercise which tests as much of the licensee, State, and local emergency plans as is reasonably achievable without mandatory public participation shall be conducted for each site at which a power reactor is located. Nuclear power reactor licensees shall submit exercise scenarios under § 50.4~~**	Refer to basis for 10 CFR 50.47(b). TMI-1 will continue to invite the Commonwealth of Pennsylvania and local support to participate in the periodic drills and exercises conducted to assess their ability to perform responsibilities related to an emergency at TMI, to the extent defined by the TMI emergency plan. Because the need for offsite emergency planning is relaxed due to the low probability of the postulated accident or other credible events that would be expected to result in an offsite radioactive release that would exceed

TABLE 2
EXEMPTIONS REQUESTED FROM 10 CFR PART 50, APPENDIX E

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Item	10 CFR PART 50, APPENDIX E, SECTION IV	Basis for Exemption
	~~at least 60 days before use in a full participation exercise required by this paragraph 2.a.~~ F.2.a.(i), (ii), and (iii) are not applicable.	the EPA PAGs and the available time for event mitigation, no formal offsite radiological emergency plans will be in place to test. The intent of submitting exercise scenarios for use by power reactor licensees is to check that licensees utilize different scenarios in order to prevent the preconditioning of responders at power reactors. For defueled sites, there are limited events that could occur and the previously routine progression to General Emergency in power reactor site scenarios is not applicable to a decommissioning site. Exelon considers TMI to be exempt from 10 CFR Part 50, Appendix E, Section F.2.a.(i)-(iii) because TMI will be exempt from the umbrella provision of 10 CFR Part 50, Appendix E, Section IV.F.2.a.
40	F.2.b. Each licensee at each site shall conduct a subsequent exercise of its onsite emergency plan every 2 years. **~~Nuclear power reactor licensees shall submit exercise scenarios under § 50.4 at least 60 days before use in an exercise required by this paragraph 2.b. The exercise may be included in the full participation biennial exercise required by paragraph 2.c. of this section.~~** In addition, the licensee shall take actions necessary to ensure that adequate emergency response capabilities are maintained during the interval between biennial exercises by conducting drills, including at least one drill involving a combination of some of the principal functional areas of the licensee's onsite emergency response capabilities. The principal functional areas of emergency response include activities such as management and coordination of emergency response, accident assessment, event classification, notification of offsite authorities, and assessment of the onsite **~~and offsite~~**	Refer to basis for 10 CFR Part 50, Appendix E, Section IV.F.2.a. The low probability of design basis accidents or other credible events that would result in an offsite radioactive release that would exceed the EPA PAGs and the available time for event mitigative actions at TMI-1 during decommissioning render the TSC, OSC and EOF unnecessary. The principal functions required by regulation can be performed at an onsite location that does not meet the requirements of the TSC, OSC or EOF. TMI-1 will continue to conduct biennial exercises and will invite the Commonwealth of Pennsylvania and local support organizations (firefighting, law enforcement, and ambulance/medical services) to participate in periodic drills and exercises to assess its ability to perform responsibilities related to an emergency at TMI to the extent defined by the TMI emergency plan.

TABLE 2
EXEMPTIONS REQUESTED FROM 10 CFR PART 50, APPENDIX E

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Item	10 CFR PART 50, APPENDIX E, SECTION IV	Basis for Exemption
	impact of radiological releases~~, protective action recommendation development, protective action decision making, plant~~ system repair and mitigative action implementation. During these drills, activation of all of the licensee's emergency response facilities ~~(Technical Support Center (TSC), Operations Support Center (OSC), and the Emergency Operations Facility (EOF))~~ would not be necessary, licensees would have the opportunity to consider accident management strategies, supervised instruction would be permitted, operating staff in all participating facilities would have the opportunity to resolve problems (success paths) rather than have controllers intervene, and the drills may focus on the onsite exercise training objectives.	
41	F.2.c. ~~Offsite plans for each site shall be exercised biennially with full participation by each offsite authority having a role under the radiological response plan. Where the offsite authority has a role under a radiological response plan for more than one site, it shall fully participate in one exercise every two years and shall, at least, partially participate in other offsite plan exercises in this period. If two different licensees each have licensed facilities located either on the same site or on adjacent, contiguous sites, and share most of the elements defining co-located licensees, then each licensee shall:~~ (1) ~~Conduct an exercise biennially of its onsite emergency plan;~~ (2) ~~Participate quadrennially in an offsite biennial~~	See basis for 10 CFR Part 50, Appendix E, Section IV.1 and 10 CFR Part 50, Appendix E, Section IV.F.2.a.

TABLE 2
EXEMPTIONS REQUESTED FROM 10 CFR PART 50, APPENDIX E

Bold ~~strikethrough~~ text identifies the proposed exemption with respect to the regulation. The basis for the exemption explains the scope of the exception.

Item	10 CFR PART 50, APPENDIX E, SECTION IV	Basis for Exemption
	~~full or partial participation exercise;~~ (3) **~~Conduct emergency preparedness activities and interactions in the years between its participation in the offsite full or partial participation exercise with offsite authorities, to test and maintain interface among the affected State and local authorities and the licensee. Co-located licensees shall also participate in emergency preparedness activities and interaction with offsite authorities for the period between exercises;~~** (4) **~~Conduct a hostile action exercise of its onsite emergency plan in each exercise cycle; and~~** (5) **~~Participate in an offsite biennial full or partial participation hostile action exercise in alternating exercise cycles.~~**	
42	F.2.d. **~~Each State with responsibility for nuclear power reactor emergency preparedness should fully participate in the ingestion pathway portion of exercises at least once every exercise cycle. In States with more than one nuclear power reactor plume exposure pathway EPZ, the State should rotate this participation from site to site. Each State with responsibility for nuclear power reactor emergency preparedness should fully participate in a hostile action exercise at least once every cycle and should fully participate in one hostile action exercise by December 31, 2015. States with more than one nuclear power~~**	Refer to basis for 10 CFR 50.47(b)(10).

TABLE 2
EXEMPTIONS REQUESTED FROM 10 CFR PART 50, APPENDIX E

Bold ~~strikethrough~~ text identifies the proposed exemption with respect to the regulation. The basis for the exemption explains the scope of the exception.

Item	10 CFR PART 50, APPENDIX E, SECTION IV	Basis for Exemption
	~~reactor plume exposure pathway EPZ should rotate this participation from site to site.~~	
43	F.2.e. Licensees shall enable any State or local Government **~~located within the plume exposure pathway EPZ~~** to participate in the licensee's drills when requested by such State or local Government.	Refer to basis for 10 CFR 50.47(b)(10).
44	F.2.f. Remedial exercises will be required if the emergency plan is not satisfactorily tested during the biennial exercise, such that NRC**~~, in consultation with FEMA,~~** cannot (1) find reasonable assurance that adequate protective measures can and will be taken in the event of a radiological emergency or (2) determine that the Emergency Response Organization (ERO) has maintained key skills specific to emergency response. **~~The extent of State and local participation in remedial exercises must be sufficient to show that appropriate corrective measures have been taken regarding the elements of the plan not properly tested in the previous exercises.~~**	The Federal Emergency Management Agency (FEMA) is responsible for the evaluation of an offsite response exercise. No action is expected from State or local government organizations in response to an event at a decommissioning site other than firefighting, law enforcement, and ambulance/medical services. Memoranda of understanding will continue to be in place for those services. Offsite response organizations will continue to take actions to protect the health and safety of the public as they would at any other industrial site.
45	F.2.g and F.2.h	No exemption requested.
46	F.2.i. Licensees shall use drill and exercise scenarios that provide reasonable assurance that anticipatory responses will not result from preconditioning of participants. **~~Such scenarios for nuclear power reactor licensees must include a wide spectrum of radiological releases and events, including hostile action.~~** Exercise and drill scenarios as appropriate must emphasize coordination among onsite and offsite response organizations.	At TMI-1 there will be limited events that could result in radioactive releases that exceed the EPA PAGs and the previously routine progression to General Emergency in power reactor site scenarios will not be applicable. Therefore, TMI-1 is not expected to demonstrate response to a wide spectrum of events. Also refer to basis for 10 CFR Part 50, Appendix E, Section IV.1 regarding "hostile action."

TABLE 2
EXEMPTIONS REQUESTED FROM 10 CFR PART 50, APPENDIX E

Bold ~~strikethrough~~ text identifies the proposed exemption with respect to the regulation. The basis for the exemption explains the scope of the exception.

Item	10 CFR PART 50, APPENDIX E, SECTION IV	Basis for Exemption
47	F.2.j. **~~The exercises conducted under paragraph 2 of this section by nuclear power reactor licensees must provide the opportunity for the ERO to demonstrate proficiency in the key skills necessary to implement the principal functional areas of emergency response identified in paragraph 2.b of this section.~~** **~~Each exercise must provide the opportunity for the ERO to demonstrate key skills specific to emergency response duties in the control room, TSC, OSC, EOF, and joint information center. Additionally, in each eight calendar year exercise cycle, nuclear power reactor licensees shall vary the content of scenarios during exercises conducted under paragraph 2 of this section to provide the opportunity for the ERO to demonstrate proficiency in the key skills necessary to respond to the following scenario elements: hostile action directed at the plant site, no radiological release or an unplanned minimal radiological release that does not require public protective actions, an initial classification of or rapid escalation to a Site Area Emergency or General Emergency, implementation of strategies, procedures, and guidance developed under § 50.54(hh)(2), and integration of offsite resources with onsite justification. The licensee shall maintain a record of exercises conducted during each eight year exercise cycle that documents the content of scenarios used to comply with the requirements of this paragraph. Each licensee shall conduct a hostile action exercise for each of its sites no later than December 31, 2015.~~**	Refer to basis for 10 CFR Part 50, Appendix E, Section IV.F.2. Also refer to basis for 10 CFR Part 50, Appendix E, Section IV.1 regarding "hostile action" and 10 CFR 50.47(b)(5) regarding § 50.54(hh)(2).

TABLE 2
EXEMPTIONS REQUESTED FROM 10 CFR PART 50, APPENDIX E

Bold ~~strikethrough~~ text identifies the proposed exemption with respect to the regulation. The basis for the exemption explains the scope of the exception.

Item	10 CFR PART 50, APPENDIX E, SECTION IV	Basis for Exemption
	~~The first eight-year exercise cycle for a site will begin in the calendar year in which the first hostile action exercise is conducted. For a site licensed under Part 52, the first eight-year exercise cycle begins in the calendar year of the initial exercise required by Section IV.F.2.a.~~	
48	G. Maintaining Emergency Preparedness and H. Recovery	No exemptions requested.
49	I. Onsite Protective Actions During Hostile Action **~~By June 20, 2012, for nuclear power reactor licensees, a range of protective actions to protect onsite personnel during hostile action must be developed to ensure the continued ability of the licensee to safely shut down the reactor and perform the functions of the licensee's emergency plan.~~**	Refer to basis for 10 CFR Part 50, Appendix E, Section IV.1.

NOTE: Appendix E to 10 CFR Part 50, Section VI.2 exempts permanently or indefinitely shutdown plants from the requirement to provide hardware to support the Emergency Response Data System (ERDS). Therefore, specific exemptions from Appendix E to 10 CFR Part 50, sections VI.1, 3, 4 and 10 CFR 50.72(a)(4) are not required.

5.0 TECHNICAL EVALUATION

5.1 Accident Analysis Overview

10 CFR 50.82(a)(2) specifies that the 10 CFR Part 50 license no longer authorizes operation of the reactor or emplacement or retention of fuel in the reactor vessel after docketing the certifications for permanent cessation of operations and permanent removal of fuel from the reactor vessel in accordance with 10 CFR 50.82(a)(1). Following the termination of reactor operations at TMI-1 and the permanent removal of the fuel from the reactor vessel, the postulated accidents involving failure or malfunction of the reactor and supporting structures, systems and components are no longer applicable.

A summary of the postulated radiological accidents analyzed for the permanently shutdown and defueled condition of TMI-1 is presented below and are in accordance with NRC ISG-02 (Reference 1).

Section 5.0 of ISG-02 indicates that site-specific analyses should demonstrate that: (1) the radiological consequences of the remaining applicable postulated accidents would not exceed the limits of the EPA PAGs at the EAB; (2) in the event of a beyond design basis event resulting in the drain down of the SFP to the point that cooling is not effective, there is at least 10 hours (assuming an adiabatic heat up) from the time that the fuel is no longer being cooled until the hottest fuel assembly reaches 900°C; (3) adequate physical security is in place to assure implementation of security strategies that protect against spent fuel sabotage; and (4) in the unlikely event of a beyond design basis events resulting in a loss of all SFP cooling, there is sufficient time to implement pre-planned mitigation measures to provide makeup or spray to the SFP before the onset of a zirconium cladding ignition.

Table 3 contains a listing of seven analyses that are expected to be evaluated by a decommissioning power reactor licensee requesting exemption of emergency planning requirements. The table also contains a description of how TMI-1 addresses each of these analyses.

TABLE 3
INTERIM STAFF GUIDANCE-02 COMPARISON

Analysis	ISG-02 Description	Response
1	Applicable design DBAs (i.e., fuel handling accident in the spent fuel storage facility, waste gas system release, and cask handling accident if the cask handling system is not licensed as single-failure-proof) (Indicates that any radiological release would not exceed the limits of EPA PAGs at EAB);	As discussed in Section 5.2, the postulated design basis accident that will remain applicable to TMI-1 and could contribute to dose upon implementation of the requested exemptions is the fuel handling accident (FHA) in the Fuel Handling Building, where the SFP is located. The results of the analysis indicate that the dose at the EAB would not exceed the EPA PAGs 365 days after permanent cessation of power operations (Reference 9). Exelon will maintain the version of the EPA PAGs as specified in the current and proposed TMI Emergency Plan.
2	Complete loss of SFP water inventory with no heat loss (adiabatic heatup) demonstrating a minimum of 10 hours is available before any fuel cladding temperature reaches 900 degrees Celsius from the time all cooling is lost (Demonstrates sufficient time to mitigate events that could lead to a zirconium cladding fire);	Exelon performed an analysis (Reference 7) that conservatively evaluated the length of time (in hours) it takes for uncovered spent fuel assemblies in the SFP to reach the temperature at which the zirconium cladding would fail. The analysis concluded that a decay time of 488 days after permanent cessation of power operations is the period that the hottest fuel assembly would reach 900°C in 10 hours after the assemblies have been uncovered. This analysis is described in Section 5.3 and is included in Attachment 2.
3	Loss of SFP water inventory resulting in radiation exposure at the EAB and control room; (Indicates that any release is less than EPA PAGs at EAB);	TMI-1 performed an analysis (Reference 18) to determine the radiological impact of a complete loss of SFP water. It was determined that the gamma radiation dose rate at the EAB and the Control Room would be less that regulatory defined limits at 488 days after shutdown. This analysis is described in Section 5.4.
4	Considering the site-specific seismic hazard, either an evaluation demonstrating a high confidence of a low-probability (less than 1×10^{-5} per year) of seismic failure of the spent fuel storage pool structure or an analysis demonstrating the fuel has decayed sufficiently that natural air flow in a completely drained pool would maintain peak cladding temperature below 565 degrees Celsius (the point of incipient cladding damage) (Indicates that any release is less than EPA PAGs at EAB).	TMI-1 conducted a seismic evaluation in response to a NRC request for information pursuant to 10 CFR 50.54(f) regarding Recommendation 2.1 of the Near-Term Task Force (NTTF) Review of Insights from the Fukushima Dai-ichi Accident (Reference 19). The seismic evaluation included all structures including the SFP, and was prepared and submitted for NRC review. The Exelon submittal (Reference 20) documents the seismic evaluation in conformance with NTTF Recommendation 2.1 including the high-confidence-of-low-probability-of-failure (HCLPF) values and the 1×10^{-5} per year hazard level. The NRC Staff review of the NTTF submittal, specifically for the SFP Evaluation associated with the reevaluated seismic hazard implementing NTTF

TABLE 3
INTERIM STAFF GUIDANCE-02 COMPARISON

Analysis	ISG-02 Description	Response
		Recommendation 2.1 is documented in Reference 21. The NRC staff concluded that the assessment was performed consistent with the NRC-endorsed (Reference 22) SFP Evaluation Guidance Report (Reference 23) and provided sufficient information, including the SFP integrity evaluation, to meet the SFP Evaluation Guidance (Item 9 in Enclosure 1 of the NRC's 50.54(f) letter), thus supporting SDA No. 6 of NUREG-1738.
5	The analyses and conclusions described in NUREG-1738 are predicated on the risk reduction measures identified in the study as Industry Decommissioning Commitments (IDC) and Staff Decommissioning Assumptions (SDA), listed in Tables 4.1-1 and 4.1-2 of that document. The staff should ensure that the licensee has addressed these IDCs and SDAs for the decommissioning site if they are storing fuel in an SFP.	IDCs and SDAs are addressed in Section 5.5 and Tables 4 and 5.
6	Verify that the licensee presents a determination that there is sufficient resources and adequately trained personnel available on-shift to initiate mitigative actions within the 10-hour minimum time period that will prevent an offsite radiological release that exceeds the EPA PAGs at the EAB.	The onsite restoration plans for repair of the SFP cooling system and to provide makeup water to the SFP are incorporated into TMI-1 procedures. There are multiple ways to initiate mitigative actions and add makeup water to the SFP within the 10-hour minimum time period with or without entry to the SFP floor. Refer to SDA 2 in Table 5.
7	Verify that mitigation strategies are consistent with that required by the Permanently Defueled Technical Specifications or by retained license conditions.	TMI-1 maintains procedures and strategies for the movement of any necessary portable equipment that will be relied upon for mitigating the loss of SFP water. These mitigative strategies were developed in response to 10 CFR 50.54(hh)(2) and are maintained in accordance with License Condition 2.c.(17) of the TMI-1 Renewed Facility License. These diverse strategies provide defense-in-depth and ample time to provide makeup water or spray to the SFP prior to the onset of zirconium cladding ignition when considering very low probability beyond design basis events affecting the SFP. Refer to SDA 4 in Table 5.

5.2 Consequences of Design Basis Events

5.2.1. TMI-1

As described in the license amendment request for proposed changes to the TMI Technical Specifications reflecting the Permanently Defueled condition (Reference 24), the applicable remaining design basis accidents were (1) a Fuel Handling Accident in the Spent Fuel Pool, (2) a Waste Gas Tank Rupture, and (3) a Cask Drop Accident.

As of the end of Zirc-Fire Window all waste gas generated will have been released and the Waste Gas Tank Rupture will no longer be applicable. As stated in the PSDAR (Reference 5), TMI is constructing an ISFSI to support dry fuel storage until the DOE takes possession of the irradiated fuel.

As part of the ISFSI project the Spent Fuel Handling Building Crane is being replaced/upgraded to a 'single failure proof' design, and therefore will no longer require a Cask Drop Analysis. Therefore, the only design basis accident that remain applicable will be the fuel handling accident.

The FHA is defined as the dropping of a single spent fuel assembly in the SFP during fuel handling activities, such that the entire outer row of fuel rods in the assembly, 56 of 208, suffers mechanical damage to the cladding. This accident is postulated to occur despite the administrative controls and physical limitations imposed on fuel-handling operations. The gap activity in the damaged rods is instantaneously released into the SFP. The release occurs under 23 feet of water, which acts as a filter.

The Post Permanent Shutdown FHA (Reference 9) was evaluated using the methodology described in Regulatory Guide 1.183 (Reference 25). This new analysis did not credit the function of any structure, system, or component (SSC) or active mitigation measures. The analysis credits the decontamination of the 23 feet of water over the fuel assemblies in the SFP (i.e., 99.5% (or a Decontamination Factor (DF) of 200) of the iodine released from the fuel assembly is assumed to remain in the water).

The FHA analysis shows that the dose at the EAB 365 days after shutdown (with no credit for safety systems) is 1.78×10^{-4} rem TEDE and 5.95×10^{-13} rem Thyroid (Reference 9). This is less than the EPA PAG of 1 rem TEDE and 5 rem Thyroid, and the accepted 10% EPA PAG for declaration of Site Area Emergency per NEI 99-01, Rev.6 (Reference 10).

5.2.2. TMI-2

The bounding event for TMI-2 is a fire in the Reactor Building (RB) with the RB Purge System in operation. Per the TMI-2 Fire Protection Program Evaluation Report (Reference 26) the dose at the exclusion area boundary is 13.5 mrem expressed as a bone dose. Due to the isotopic mix (e.g., negligible amounts of iodine) and the nature of potential releases (i.e., particulate matter), a more restrictive basis (i.e., the critical organ) for comparison was selected for reporting dose for TMI-2 fires.

This is also less than the EPA PAGs and the accepted 10% EPA PAG for declaration of Site Area Emergency per NEI 99-01, Rev.6 (Reference 10).

5.3 Hottest Fuel Assembly Adiabatic Heat Up (Zirconium Fire)

The analysis (Reference 7) is provided in Attachment 2 to compare the conditions for the hottest fuel assembly stored in the TMI-1 fuel pool to a criterion proposed in SECY-99-168 "Improving Decommissioning Regulations for Nuclear Power Plants" (Reference 27), applicable to offsite emergency response for the unit in the decommissioning process. This criterion considers the time for the hottest assembly to heat up from 30 °C to 900°C adiabatically. If the heat up time is greater than 10 hours, then offsite emergency preplanning involving the plant is not necessary.

Based on the limiting fuel assembly for decay heat and adiabatic heat-up analysis presented in Attachment 2, at 488 days (approximately 16 months) after permanent cessation of power operations, the time for the hottest fuel assembly to reach 900°C is 10 hours after the assemblies have been uncovered. As stated in NUREG-1738, "Technical Study of Spent Fuel Pool Accident Risk at Decommissioning Nuclear Power Plants" (February 2001) (Reference 16), 900°C is an acceptable temperature to use for assessing onset of fission product release under transient conditions (to establish the critical decay time for determining availability of 10 hours to evacuate) if fuel and cladding oxidation occurs in air.

Because of the length of time it would take for the adiabatic heat up to occur, there is ample time to respond to any drain down event that might cause such an occurrence by restoring cooling or makeup or providing spray. As a result, the likelihood that such a scenario would progress to a zirconium fire is not deemed credible.

5.4 Consequences of Beyond Design Basis Events

NUREG-0586, "Final Generic Environmental Impact Statement on Decommissioning of Nuclear Facilities," (Reference 28) Supplement 1, Section 4.3.9, identifies that a SFP drain down event is a beyond design basis event. The premise of the required adiabatic heat-up analysis was the rapid drain down that would expose the fuel to air cooling. The requirements of the analysis were to determine the decay time required to limit the heat up to 900°C at 10 hours, which would define the mitigation window and event duration.

The offsite and Control Room radiological impacts of a postulated complete loss of SFP water were assessed in Technical Evaluation 623073, "TMI Spent Fuel Pool Draindown Shine Dose Rate Evaluation, Revision 0," (Reference 18). A loss of water shielding above the fuel could increase the offsite radiation levels because of the gamma rays streaming up out of the SFP being scattered back to a receptor at the site boundary. With a decay of 365 days from shutdown the dose rate at the EAB would be 4.04×10^{-1} mrem/hr not crediting the shielding from the Fuel Handling Building (FHB) roof. Crediting the FHB roof structure, the dose rate at the EAB would be 4.6×10^{-10} mrem/hr. The resultant dose rates if taken over the 10-hour accident duration would be less than the EPA PAGs and the Site Area Emergency Fraction provided by NEI 99-01, Rev. 6 (Reference 10).

It should be noted that the EPA PAGs were developed to respond to a mobile airborne plume that could transport and deposit radioactive material over a large area. In contrast, the radiation field formed by gamma scatter from a drained SFP would be stationary rather than moving and would not cause transport or deposition of radioactive materials. The extended period required to exceed the EPA PAG limit of 1 rem TEDE would allow sufficient time to develop and implement onsite mitigative actions and provide confidence that additional offsite measures could be taken without planning if efforts to reestablish shielding over the fuel are delayed.

Additionally, the Control Room radiological impacts at 365-days of a postulated complete loss of SFP water determined that the gamma radiation dose rate in the Control Room will be below 0.1 mrem/hr.

5.5 Comparison to NUREG-1738 Industry Decommissioning Commitments and Staff Decommissioning Assumptions

Although the limited scope of design and beyond design basis accidents that remain applicable to TMI-1 justify a reduction in the necessary scope of emergency response capabilities, Exelon also evaluated the industry decommissioning commitments (IDCs) and staff decommissioning assumptions (SDAs) contained in NUREG-1738 (Reference 16).

NUREG-1738 contains the results of the NRC staff's evaluation of the potential accident risk in spent fuel pools at decommissioning plants in the United States. As stated therein, the study was undertaken to support development of a risk-informed technical basis for reviewing exemption requests and a regulatory framework for integrated rulemaking. The NRC staff performed analyses and sensitivity studies on evacuation timing to assess the risk significance of relaxed offsite emergency preparedness requirements during decommissioning. The staff based its sensitivity assessment on the guidance in Regulatory Guide (RG) 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis" (Reference 29). The staff's analyses and conclusions apply to decommissioning facilities with SFPs that meet the design and operational characteristics assumed in the risk analysis.

The NUREG-1738 study found that the risk at decommissioning plants is low and well within the Commission's Safety Goals. The risk is low because of the very low likelihood of a zirconium fire (resulting from a postulated irrecoverable loss of SFP cooling water inventory) even though the consequences from a zirconium fire could be serious.

The study provided the following assessment:

> *"The staff found that the event sequences important to risk at decommissioning plants are limited to large earthquakes and cask drop events. For emergency planning (EP) assessments, this is an important difference relative to operating plants where typically a large number of different sequences make significant contributions to risk. Relaxation of offsite EP a few months after shutdown resulted in only a "small change" in risk, consistent with the guidance of RG 1.174. Figures ES-1 and ES-2 [in NUREG-1738] illustrate this finding. The change in risk due to relaxation of offsite EP is small because the overall risk is low, and because even under current EP requirements, EP was judged to have marginal impact on evacuation effectiveness in the severe earthquakes that dominate SFP risk. All other sequences including cask drops (for which emergency planning is expected to be more effective) are too low in likelihood to have a significant impact on risk. For comparison, at operating reactors, additional risk-significant accidents for which EP is expected to provide dose savings are on the order of $1x10^{-5}$ per year, while for decommissioning facilities, the largest contributor for which EP would provide dose savings is about two orders of magnitude lower (cask drop sequence at $2x10^{-7}$ per year)."*

The Executive Summary in NUREG-1738 states, in part,

> *"the staff's analyses and conclusions apply to decommissioning facilities with SFPs that meet the design and operational characteristics assumed in the risk analysis. These*

characteristics are identified in the study as IDCs and SDAs. Provisions for confirmation of these characteristics would need to be an integral part of rulemaking."

The IDCs and SDAs are listed in Tables 4.1-1 and 4.1-2, respectively, of NUREG-1738. The tables below show how the TMI-1 SFP meets or compares with each of these IDCs (Table 4) and SDAs (Table 5).

5.6 Consequences of a Beyond-Design Basis Earthquake

NUREG-1738 (Reference 16) identifies beyond design basis seismic events as the dominant contributor to events that could result in a loss of SFP coolant that uncovers fuel for plants in the Central and Eastern United States. Additionally, NUREG-1738 identifies a zirconium fire resulting from a substantial loss-of-water inventory from the SFP, as the only postulated scenario at a decommissioning plant that could result in a significant offsite radiological release. The scenarios that lead to this condition have very low frequencies of occurrence (i.e., on the order of one to tens of times in a million years) and are considered beyond design basis events because the SFP and attached systems are designed to prevent a substantial loss of coolant inventory under accident conditions. However, the consequences of such accidents could potentially lead to an offsite radiological dose in excess of the EPA PAGs (Reference 2) at the EAB.

However, the risk associated with zirconium cladding fire events decreases as the spent fuel ages, decay time increases, decay heat decreases, and short-lived radionuclides decay away. As decay time increases, the overall risk of a zirconium cladding fire continues to decrease due to two factors: (1) the amount of time available for preventative actions increases, which reduces the probability that the actions would not be successful; and (2) the increased likelihood that the fuel is able to be cooled by air, which decreases the reliance on actions to prevent a zirconium fire. The results of research conducted for NUREG-1738 and NUREG-2161, "Consequence Study of a Beyond-Design-Basis Earthquake Affecting the Spent Fuel Pool for a U.S. Mark I Boiling Water Reactor," dated September 2014 (Reference 30), suggest that, while other radiological consequences can be extensive, a postulated accident scenario leading to a SFP zirconium fire, where the fuel has had significant decay time, will have little potential to cause offsite early fatalities, regardless of the type of offsite response (i.e., formal offsite radiological emergency preparedness plan or CEMP).

The purpose of NUREG-2161 (Reference 30) was to determine if accelerated transfer of older, colder spent fuel from the SFP at a reference plant to dry cask storage significantly reduces the risks to public health and safety. The study states that:

"this study's results are consistent with earlier research studies' conclusions that SFPs are robust structures that are likely to withstand severe earthquakes without leaking cooling water and potentially uncovering the spent fuel. The study shows the likelihood of a radiological release from the spent fuel after the analyzed severe earthquake at the reference plant to be about one time in 10 million years or lower. If a leak and radiological release were to occur, this study shows that the individual cancer fatality risk for a member of the public is several orders of magnitude lower than the Commission's Quantitative Health Objective of two in one million (2 x 10^{-6}/year). For such a radiological release, this study shows public and environmental effects are generally the same or smaller than earlier studies."

The reference plant for the study (a General Electric Type 4 BWR with a Mark I containment) generated approximately 3500 Megawatt-thermal (MWt) and the SFP contained 2844 fuel assemblies. TMI-1 was licensed to generate 2568 MWt, and the SFP has the capacity to hold 1987 fuel assemblies. The SFP is expected to contain 1666 fuel assemblies following permanent cessation of power operations and transfer of all fuel from the reactor vessel to the SFP. Based on these differences, the risk and the consequences of an event involving the SFP at TMI-1 are lower than those in the NUREG-2161 study.

The final off-load into the spent fuel pool will be constrained to ensure that the requirements of Exelon procedure NF-AP-309, "PWR Special Nuclear Material and Core Component Move Sheet Development." Attachment 1, Section 3 – Thermal Management Guidelines in Support of Permanent Shutdown, are met. The off-loaded fuel assemblies (hot cells) will be arranged so that all four face-adjacent cells will have assemblies that have been discharged for at least 5 years (cold cells). Additionally, two or more hot cells may not take credit for the same cold cell. Storing spent fuel in a such a dispersed pattern in SFP promotes air coolability of the spent fuel in the unlikely event of a loss of water. This ensures that fuel distribution in the SFP will be bounded by that assumed in NUREG-2161.

TMI-1 conducted a seismic evaluation in response to a NRC request for information pursuant to 10 CFR 50.54(f) regarding Recommendation 2.1 of the NTTF Review of Insights from the Fukushima Dai-ichi Accident. The seismic evaluation included all structures including the SFP, and was prepared and submitted for NRC review. The Exelon submittal (Reference 20) documents the seismic evaluation in conformance with NTTF Recommendation 2.1 including the high-confidence-of-low-probability-of-failure (HCLPF) values and the 1×10^{-5} per year hazard level. The NRC staff review of the NTTF submittal, specifically for the SFP Evaluation associated with the reevaluated seismic hazard implementing NTTF Recommendation 2.1 is documented in Reference 21. The NRC staff concluded that the assessment was performed consistent with the NRC-endorsed (Reference 22) SFP Evaluation Guidance Report (Reference 23) and provided sufficient information, including the SFP integrity evaluation, to meet the SFP Evaluation Guidance (Item 9 in Enclosure 1 of the NRC's 50.54(f) letter), thus supporting SDA No. 6 of NUREG-1738.

6.0 CONCLUSION

Exelon has concluded, based on the analysis and actions described above, that the health and safety of the public are protected once TMI-1 is in the permanently defueled condition. Approval of the exemptions requested above would not present an undue risk to the public or prevent appropriate response in the event of an emergency at TMI.

Based on the above, TMI-1 has demonstrated that no credible or beyond design basis accident will result in radiological releases requiring offsite protective actions. Additionally, there is sufficient time, resources and personnel available to initiate mitigative actions that will prevent an offsite release that exceeds EPA PAGs.

TABLE 4
INDUSTRY DECOMMISSIONING COMMITMENTS (IDCS)

IDC	Industry Commitments	Response
1	Cask drop analyses will be performed or single failure-proof cranes will be in use for handling of heavy loads (i.e., phase II of NUREG-0612 will be implemented).	Currently TMI-1 has analyzed the Fuel Cask Drop Accident in UFSAR Section 14.2.2.8 (Reference 31). A fuel cask drop accident is defined as the dropping of a fuel cask through the maximum drop height during transfer operations of a fuel cask onto a rail car. A fuel cask drop into the spent fuel pool is prevented by the Technical Specification requirement that the key operated travel interlock system for automatically limiting the travel area of the Fuel Handling Building crane shall be imposed whenever loads in excess of 15 tons are lifted and transported. As discussed in the PDSAR (Reference 5), as part of the ISFSI project the current Spent Fuel Handling Building Crane will be upgraded (or replaced) to a single-failure proof design to handle the spent fuel casks. Since the Spent Fuel Handling Crane will be single-failure proof, the cask drop event will not be considered credible and a cask drop analysis will no longer be required.
2	Procedures and training of personnel will be in place to ensure that onsite and offsite resources can be brought to bear during an event.	TMI-1 procedures are in place to ensure onsite and offsite resources can be brought to bear during an event, including: • Abnormal Operating Procedure, OP-TM-AOP-035, "Loss of Spent Fuel Pool Cooling" • Abnormal Operating Procedure, OP-TM-AOP-008, "Security Threat/Intrusion" • Abnormal Operating Procedure OP-TM-AOP-020, "Loss of Station Power" • Abnormal Operating Procedure, OP-TM-AOP-002, "Flood" • Abnormal Operating Procedure, OP-TM-AOP-003, "Earthquake" • Abnormal Operating Procedure, OP-TM-AOP-004, "Tornado/High Winds" • ERO activation in accordance with the TMI Permanently Defueled Emergency Plan, EP-TM-1001. These procedures are required by NRC Regulations and will be implemented as necessary depending on the type of event. Once TMI-1 is shut down and defueled, the on-shift plant operators, including Certified Fuel Handlers (CFH), and Non-Certified Operators (NCOs) will continue to be appropriately trained on the various actions needed to provide makeup to the SFP based on a systematic approach to training. Once TMI-1 is no longer operating, maintaining SFP cooling and inventory would be the highest priority activity; therefore, the personnel needed to perform these actions will be available at all times. The

TABLE 4
INDUSTRY DECOMMISSIONING COMMITMENTS (IDCS)

IDC	Industry Commitments	Response
		TMI-1 CFH training program was approved by the NRC by letter dated December 29, 2017 (Reference 32). Emergency Plan drills will be conducted to maintain proficiency in response to a plant event as described in the PDEP.
3	Procedures will be in place to establish communication between onsite and offsite organizations during severe weather and seismic events.	TMI-1 maintains procedures to provide guidance for establishing and maintaining communications between offsite agencies and the onsite ERO during severe weather and seismic events. The following Abnormal Operating Procedures (AOPs) address severe weather and seismic events actions: • Abnormal Operating Procedure, OP-TM-AOP-002, "Flood" • Abnormal Operating Procedure, OP-TM-AOP-003, "Earthquake" • Abnormal Operating Procedure, OP-TM-AOP-004, "Tornado/High Winds" The AOPs direct entry into the following procedures: • OP-AA-108-111-1001, "Severe Weather and Natural Disaster Guidelines" • OP-TM-108-111-1001, "TMI Severe Weather and Site Inaccessibility Guidelines" These procedures provide direction for additional actions and communications with onsite and offsite stakeholders if the event does not reach the threshold for entry into the PDEP. If the severity of the event requires entry into the PDEP, communications with onsite and offsite organizations will be directed by the TMI PDEP and associated procedures. These procedures are required by NRC Regulations and will be implemented as necessary depending on the type of event. Communications are described in the procedures for onsite and offsite communications, they are not specifically referenced in the existing TMI-1 Emergency Plan and will not be included in the planned Permanently Defueled Emergency Plan (to be submitted for NRC approval). Therefore, it is not necessary for them to be specifically referenced in the Emergency Plan. Equipment requirements are specified in the pertinent procedures.
4	An offsite resource plan will be developed which will include access to portable pumps and emergency power to supplement onsite resources. The plan would principally identify organizations or suppliers where offsite	TMI-1 has multiple portable pumps and emergency generators that meet Extensive Damage Mitigation Guidelines (EDMG) requirements. These can be used as required by abnormal procedures. In addition, offsite resources are available from other Exelon Facilities in the nearby vicinity.

TABLE 4
INDUSTRY DECOMMISSIONING COMMITMENTS (IDCS)

IDC	Industry Commitments	Response
	resources could be obtained in a timely manner.	
5	SFP instrumentation will include readouts and alarms in the control room (or where personnel are stationed) for SFP temperature, water level, and area radiation levels.	Spent Fuel Pool Temperature is monitored on the Plant Process Computer and has a high temperature alarm function in the Control Room. There are Low Level alarm functions available in the TMI-1 Control Room. Additionally, there are two channels of continuous remote indication of the SFP water level indicators in the 322' elevation Control Tower that have been added for reliable SFP level indication (post-Fukushima). Radiation channel RM-G-9 located in the Fuel Handling Building provides radiation levels in the spent fuel storage area and is monitored and alarmed in the Control Room.
6	SFP seals that could cause leakage leading to fuel uncovery in the event of seal failure shall be self-limiting to leakage or otherwise engineered so that drainage cannot occur.	The TMI-1 SFP is contained in the Fuel Handling Building and is connected to the Fuel Transfer Canal via two fuel transfer tubes. There are no seals in the SFP that would be subject to leakage. When not actively performing refueling the SFP is isolated from the Fuel Transfer Canal with two blank flanges on the Fuel Transfer Tubes on the Reactor Building side, and two locked closed gate valves on the SFP side. Failure of the Spent Fuel Pool Cooling pump seals will not cause a total drain-down of SFP and is discussed in more detail in IDCS #7.
7	Procedures or administrative controls to reduce the likelihood of rapid drain down events will include (1) prohibitions on the use of pumps that lack adequate siphon protection or (2) controls for pump suction and discharge points. The functionality of anti-siphon devices will be periodically verified.	The design of the SFP and its cooling system and connections to the pool are such that the SFP cannot be drained below the level of the top of the stored fuel when in its storage rack. The most serious failure of the Spent Fuel Cooling System would be complete loss of water from both spent fuel storage pools. To protect against this possibility, the cooling water inlet and outlet connections to spent fuel pool B all enter slightly below, or at, the normal water level in the pool. Fuel pool A has a drain connection from the spent fuel cooling system extending downward from 10 feet above the top of fuel stored in this pool (330 foot elevation) to 2 inches above the bottom of the pool. This line has a syphon breaker with a normally locked open valve to prevent water from syphoning from the pool below 330-foot elevation in the highly unlikely event that the line should break outside the pool. A combination drain/fill line enters the spent fuel cask pit at elevation 332 ft (approximately 12 ft

TABLE 4
INDUSTRY DECOMMISSIONING COMMITMENTS (IDCS)

IDC	Industry Commitments	Response
		above the top of the spent fuel stored in pool B). This line extends down inside the pit to elevation 323 ft 6 inches. There is a syphon breaker on this line with a normally locked open valve to prevent draining the spent fuel cask pit below elevation 332 ft in the unlikely event that the line should break outside the pit. The locked open valve is administratively controlled and periodically verified to locked open by operator rounds. Therefore, it is concluded that the Spent Fuel Cooling System provides adequate protection against serious depression of the water level in either of the spent fuel pools in the highly unlikely event of the rupture of any of its lines.
8	An onsite restoration plan will be in place to provide repair of the SFP cooling systems or to provide access for makeup water to the SFP. The plan will provide for remote alignment of the makeup source to the SFP without requiring entry to the refuel floor.	There are multiple ways to add makeup water to the SFP with or without entry to the refuel floor. OP-TM-AOP-035, "Loss of Fuel Pool Cooling", provides the initial response to the abnormal conditions in the Spent Fuel Pool. The following procedures describes the spent fuel makeup strategies: • Makeup from Fire Service (OP-TM-251-901, "High Capacity Fire Service Makeup to Spent Fuel Pool") • Makeup from raw water sources (OP-TM-919-922, "FSG-6 – Makeup from Raw Water Sources") • Fuel Pool Makeup from FX-P-2A/B (OP-TM-919-914) "Spent Fuel Pool Makeup Using FX-P-2A or FX-P-2B." This method does not require access to the spent fuel pool refueling floor. • Spent Fuel Pool Spray (OP-TM-251-902, "Spent Fuel Pool Spray") • Spent Fuel Pool Spray from outside the SFP Building (OP-TM-251-904, "Spent Fuel Pool Building (External) Spray"), including using an off-site fire truck.
9	Procedures will be in place to control SFP operations that have the potential to rapidly decrease SFP inventory. These administrative controls may require additional operations or management review, management physical presence for designated operations or administrative	WC-DC-100, "Decommissioning Work Control Process" dictates the review and approval of work conducted while in decommissioning. This procedure directs performance of integrated risk assessment per OP-DC-104, "Decommissioning Integrated Risk Management," that provides for evaluation of potential operational risk. Heavy loads are controlled through MA-AA-716-022, "Control of Heavy Loads Program." Fuel moves and heavy load moves that could affect the safe handling and storage of nuclear fuel require approval by the Shift Manager.

TABLE 4
INDUSTRY DECOMMISSIONING COMMITMENTS (IDCS)

IDC	Industry Commitments	Response
	limitations such as restrictions on heavy load movements.	Additionally, the ISFSI transfer equipment will be designed such that there will be no ISFSI related SFP operations that will have the potential to cause a rapid drain down of the SFP.
10	Routine testing of the alternative fuel pool makeup system components will be performed and administrative controls for equipment out of service will be implemented to provide added assurance that the components would be available, if needed.	TMI-1 has multiple systems and sources to provide alternate makeup to the fuel pool. There is an electric-driven fire pump (FS-P-2) and a diesel-driven fire pump (FS-P-3) that can supply makeup water to the SFP via the Fire Service System. The TMI-1 fire protection program provides controls for operation with equipment out of service and periodic functional testing. TMI-1 also has two diesel driven engine emergency makeup pumps capable of taking suction from the river to satisfy the EDMG requirements. The EDMG equipment provides defense-in-depth and have testing and out of service requirements controlled by their program procedures.

TABLE 5
STAFF DECOMMISSIONING ASSUMPTIONS (SDAS)

SDA	Staff Assumptions	Response
1	Licensee's SFP cooling design will be at least as capable as that assumed in the risk assessment, including instrumentation. Licensees will have at least one motor-driven and one diesel-driven fire pump capable of delivering inventory to the SFP.	The TMI-1 design aligns with the intent of this description of the standard system in NUREG-1738. The TMI-1 Spent Fuel Pool Cooling System (SF) design has two independent trains of spent fuel pool cooling. Each train of spent fuel cooling rejects its heat to the Nuclear Service Closed Cooling Water System (NSCCW), which in turn rejects its heat to the Susquehanna River (Ultimate Heat Sink) via the Nuclear River Water System (NR). Normal makeup to the SFP to provide for evaporation losses is provided by Reclaimed Water. To provide makeup to address abnormal loss in the spent fuel pool, there are multiple means available. The primary method would be to use Fire Service (FS) water to provide makeup via hoses to the spent fuel pool. The Fire Service System includes a motor driven fire service pump (FS-P-2) and a diesel driven fire pump (FS-P-3), both take suction from the Susquehanna River. Each FS pump has the capability to deliver 500 gallons per minute (gpm) of makeup water to the SFP. In addition to the river, the fire service system has a water storage tank (Altitude Tank), which provides an additional 100,000-gallon water source to the FS system.
2	Walk-downs of SFP systems will be performed at least once per shift by the operators. Procedures will be developed for and employed by the operators to provide guidance on the capability and availability of onsite and offsite inventory makeup sources and time available to initiate these sources for various loss of cooling or inventory events.	Currently TMI-1 performs a walk-down of SFP systems once per day. Once the reactor is permanently shutdown, shiftly operator rounds will include spent fuel cooling system operating parameters, availability (status) of EDMG and availability of onsite makeup sources. Additionally, there are other methods available in the Control Room to alert operators to potential SFP events, such as annunciators and level indication. TMI-1 procedure 1104-6, "Spent Fuel Pool Cooling System," describes the normal operation of the Spent Fuel Pool Cooling system. OP-TM-AOP-035, "Loss of Fuel Pool Cooling," provides the initial response to the abnormal conditions in the Spent Fuel Pool. This AOP will direct mitigation actions related to restoring SFP cooling and/or makeup water. See response for IDC#8 for more details on procedures for SFP mitigation strategies. The ability to use EDMG strategies to provide makeup from the river using portable pumps have been demonstrated to be capable of being implemented within 4 hours. The operation and control of the Spent Fuel Pooling Cooling Systems and mitigation of a loss of spent fuel pool cooling will be addressed in the Certified Fuel Handling and Non-Certified Operator training programs.

TABLE 5
STAFF DECOMMISSIONING ASSUMPTIONS (SDAS)

SDA	Staff Assumptions	Response
3	Control room instrumentation that monitors SFP temperature and water level will directly measure the parameters involved. Level instrumentation will provide alarms at levels associated with calling in offsite resources and with declaring a general emergency.	TMI-1 design meets the intent of this SDA. Spent Fuel Pool temperature is monitored on the plant process computer and has a high temperature alarm function in the Control Room. There are low level alarm functions available in the TMI-1 Control Room. Additionally, there are two channels of continuous remote indication of the SFP water level indicators in the 322' Control Tower that have been added for reliable SFP level indication (post-Fukushima). Radiation channel RM-G-9 located in the Fuel Handling Building provides radiation levels in the spent fuel storage area and is monitored and alarmed in the Control Room. Refer to the TMI-1 responses for IDC 2 and IDC 4 for details associated with calling in offsite resources. Regarding the declaration of a general emergency, the result of the dose calculations for both the Fuel Handling Accident and the beyond design basis event of a total loss of water inventory in the SFP, do not approach the Protective Action Guideline to support a classification of greater than an Alert. TMI-1 will be employing Permanently Defueled EALs using an approved NRC EAL Scheme. based on Appendix C of NEI 99-01, "Development of Emergency Action Levels for Non-Passive Reactors," Revision 6 (Reference 10). Consistent with the NEI 99-01 Permanently Defueled EALs scheme, it is expected that station conditions will not have the capacity to reach any threshold requiring the declaration of a site area emergency nor general emergency.
4	Licensee determines that there are no drain paths in the SFP that could lower the pool level (by draining, suction, or pumping) more than 15 feet below the normal pool operating level and that licensee must initiate recovery using offsite sources.	The TMI-1 SFP design is consistent with this SDA. See discussion for IDC #s 6 and 7.
5	Load Drop consequence analyses will be performed for facilities with non-single failure-proof systems. The analyses and any mitigative actions necessary to preclude catastrophic damage to the SFP that would lead to a rapid pool draining would be sufficient to demonstrate that there is high	The TMI-1 design is in alignment with this description. See discussion for IDC #1.

TABLE 5
STAFF DECOMMISSIONING ASSUMPTIONS (SDAS)

SDA	Staff Assumptions	Response
	confidence in the facilities ability to withstand a heavy load drop.	
6	Each decommissioning plant will successfully complete the seismic checklist provided in Appendix 2B to this study [NUREG-1738]. If the checklist cannot be successfully completed, the decommissioning plant will perform a plant specific seismic risk assessment of the SFP and demonstrate that SFP seismically induced structural failure and rapid loss of inventory is less than the generic bounding estimates provided in this study (<1 x10^{-5} per year including non-seismic events).	TMI-1 conducted a seismic evaluation in response to a NRC request for information pursuant to 10 CFR 50.54(f) regarding Recommendation 2.1 of the NTTF Review of Insights from the Fukushima Dai-ichi Accident. The seismic evaluation included all structures including the SFP, and was prepared and submitted for NRC review. The Exelon submittal (Reference 20) documents the seismic evaluation in conformance with NTTF Recommendation 2.1 including the high-confidence-of-low-probability-of-failure (HCLPF) values and the 1 x 10^{-5} per year hazard level. The NRC staff review of the NTTF submittal, specifically for the SFP evaluation associated with the reevaluated seismic hazard implementing NTTF Recommendation 2.1 is documented in Reference 21. The NRC staff concluded that the assessment was performed consistent with the NRC-endorsed (Reference 22) SFP Evaluation Guidance Report (Reference 23) and provided sufficient information, including the SFP integrity evaluation, to meet the SFP Evaluation Guidance (Item 9 in Enclosure 1 of the NRC's 50.54(f) letter), thus supporting SDA No. 6 of NUREG-1738
7	Licensees will maintain a program to provide surveillance and monitoring of Boraflex in high-density spent fuel racks until such time as spent fuel is no longer stored in these high-density racks.	The TMI-1 Spent Fuel Pool "A" contains high density storage racks that employ neutron absorber material (Boral and Metamic). There are three coupon trees located in the high-density racks. Two are located in the Region II racks containing Boral; one is located in the Region II racks containing Metamic. Procedure NF-TM-600-1000, "TMI Spent Fuel Rack Boral/Metamic Coupon Program," defines and tracks a surveillance program to verify the long-term integrity of the neutron absorber material used in high-density Spent Fuel Pool storage racks. This program is a license renewal aging management program commitment that has been maintained after permanent cessation of power operations and is required by License Condition 2.(c).21.

7.0 JUSTIFICATION FOR EXEMPTIONS AND SPECIAL CIRCUMSTANCES

10 CFR 50.12 states that the Commission may, upon application by any interested person or upon its own initiative, grant exemptions from the requirements of the regulations of Part 50 which are authorized by law, will not present an undue risk to the public health and safety, and are consistent with the defense and security. 10 CFR 50.12 also states that the Commission will not consider granting an exemption unless special circumstances are present. As discussed below, this exemption request satisfies the provisions of Section 50.12.

7.1 Exemptions

A. The exemptions are authorized by law

10 CFR 50.12 allows the NRC to grant exemptions from the requirements of 10 CFR Part 50. The proposed exemption would not result in a violation of the Atomic Energy Act of 1954, as amended, or the Commission's regulations. Therefore, the exemption is authorized by law.

B. The exemptions will not present an undue risk to public health and safety

The underlying purpose of 10 CFR 50.47(b), 10 CFR 50.47(c)(2), 10 CFR 50, Appendix E, Section IV is to ensure that there is reasonable assurance that adequate protective measures can and will be taken in the event of a radiological emergency, to establish plume exposure and ingestion pathway emergency planning zones for nuclear power plants, and to ensure that licensees maintain effective offsite and onsite emergency plans.

The requested exemptions and justification for each are based on and consistent with Interim Staff Guidance NSIR/DPR-ISG-02, Emergency Planning Exemption Requests for Decommissioning Nuclear Power Plants, which was issued May 11, 2015.

As discussed in this request, revised radiological analyses have been developed that show that, 365 days after shutdown, the radiological consequences of design basis accidents will not exceed the limits of the Environmental Protection Agency (EPA) Protective Action Guides (PAGs) at the exclusion area boundary (EAB). In addition, analyses have been developed for beyond design basis events related to the spent fuel pool (SFP) which show that, 488 days after permanent cessation of power operation, the analyzed event is either not credible, is capable of being mitigated, or the radiological consequences of the event will not exceed the limits of the EPA PAGs at the EAB.

Additionally, the offsite and Control Room radiological impacts of a postulated complete loss of SFP water were assessed. It was determined that the gamma radiation dose rate at the EAB would be limited to small fractions of the EPA PAG exposure levels and the dose rate in the Control Room will be below 0.1 mRem/hr.

Therefore, offsite emergency response plans will no longer be needed for protection of the public beyond the EAB. Based on the reduced consequences of radiological events possible at the site when it is in the permanently defueled condition, the scope of the onsite emergency preparedness organization and corresponding requirements in the emergency plan may be accordingly reduced without an undue risk to the public health and safety.

Therefore, the underlying purpose of the regulations will continue to be met. Since the underlying purpose of the rules will continue to be met, the exemptions will not present an undue risk to the public health and safety.

C. The exemptions are consistent with the common defense and security

The reduced consequences of radiological events that will remain possible at the site once it is in the permanently defueled condition allows for a corresponding reduction in the scope of the onsite emergency preparedness organization and associated reduction of requirements in the emergency plan. These reductions will not adversely affect TMI-1's ability to physically secure the site or protect special nuclear material. Physical security measures at TMI are not affected by the requested exemption. Therefore, the proposed exemptions are consistent with the common defense and security.

7.2 Special Circumstances

Pursuant to 10 CFR 50.12(a)(2), the NRC will not consider granting an exemption to its regulations unless special circumstances are present. Exelon has determined that special circumstances are present as discussed below.

Special circumstances will exist at TMI because the plant will be permanently shut down and defueled and the radiological source term at the site will be reduced from that associated with reactor power operation. With the reactor power plant permanently shut down and defueled, the design basis accidents and transients postulated to occur during reactor operation will no longer be possible. In particular, the potential for a release of a large radiological source term to the environment from the high pressures and temperatures associated with reactor operation will no longer exist.

A. Application of the regulation in the particular circumstances would not serve the underlying purpose of the rule or is not necessary to achieve the underlying purpose of the rule. (10 CFR 50.12(a)(2)(ii))

The underlying purpose of 10 CFR 50.47(b), 10 CFR 50.47(c)(2), and 10 CFR 50, Appendix E, Section IV is to ensure that there is reasonable assurance that adequate protective measures can and will be taken in the event of a radiological emergency, to establish plume exposure and ingestion pathway emergency planning zones for nuclear power plants, and to ensure that licensees maintain effective offsite and onsite emergency plans.

The standards and requirements in 10 CFR 50.47(b), 10 CFR 50.47(c)(2), and 10 CFR 50, Appendix E, Section IV were developed taking into consideration the risks associated with operation of a nuclear power reactor at its licensed full power level. These risks include the potential for a reactor accident with offsite radiological dose consequences.

The radiological consequences of accidents that will remain possible at TMI-1 are substantially lower than those at an operating plant. The upper bound of offsite dose consequences limits the highest attainable emergency class to the Alert level. In addition, because of the reduced consequences of radiological events that will still be possible at the site, the scope of the onsite emergency preparedness organization may be reduced accordingly. Thus, the underlying purpose of the regulations will not be adversely affected by eliminating offsite emergency planning activities or reducing the scope of onsite emergency planning as described in this request.

Revised radiological analyses have been developed that show that, 365 days after shutdown, the radiological consequences of design basis accidents will not exceed the limits of the EPA PAGs at the EAB (Reference 9). In addition, analyses have been developed for beyond design basis events related to the SFP which show that, 488 days (approximately 16 months) after shutdown, the analyzed event is either not credible, is capable of being mitigated, or the radiological consequences of the event will not exceed the limits of the EPA PAGs at the EAB (Reference 7).

Therefore, application of all of the standards and requirements in 10 CFR 50.47(b), 10 CFR 50.47(c)(2), and 10 CFR 50, Appendix E, Section IV are not necessary to achieve the underlying purpose of those rules.

Since the underlying purposes of the rules would continue to be achieved even with TMI being permitted to reduce the scope of emergency preparedness requirements consistent with placing the facility in the permanently defueled condition, the special circumstances are present as defined in 10 CFR 50.12(a)(2)(ii).

B. Compliance would result in undue hardship or other costs that are significantly in excess of those contemplated when the regulation was adopted, or that are significantly in excess of those incurred by others similarly situated. (10 CFR 50.12(a)(2)(iii))

Application of all of the standards and requirements in 10 CFR 50.47(b), 10 CFR 50.47(c)(2), and 10 CFR 50, Appendix E, Section IV is not needed for adequate emergency response capability and is excessive for a permanently defueled facility. Application of all of these standards and requirements would result in undue costs being incurred for the maintenance of an emergency response organization in excess of that actually needed to respond to the diminished scope of credible events. Other licensees similarly situated, such as Omaha Public Power District's (OPPD) Fort Calhoun Station, Unit 1 (FCS); Entergy Nuclear Operation, Inc.'s (ENO) Vermont Yankee Nuclear Power Station (VY); Southern California Edison Company's San Onofre Nuclear Generating Station (SONGS); Duke Energy Florida, Inc.'s Crystal River Unit 3 Nuclear Generating Station (CR3); and Dominion Energy Kewaunee, Inc.'s Kewaunee Power Station (KPS) have been granted similar exemptions.

Therefore, compliance with the rule would result in an undue hardship or other costs that are significantly in excess of those contemplated when the regulation was adopted, or that are significantly in excess of those incurred by others similarly situated and the special circumstances required by 10 CFR 50.12(a)(2)(iii) exist.

C. The exemptions would result in benefit to the public health and safety that compensates for any decrease in safety that may result from the grant of the exemptions. (10 CFR 50.12(a)(2)(iv))

The plant will be permanently shut down and defueled and the radiological source term at the site will be reduced from that associated with reactor power operation. With the reactor power plant permanently shutdown and defueled, the design basis accidents and transients postulated to occur during reactor operation will no longer be possible. In particular, the potential for a release of a large radiological source term to the environment from the high pressures and temperatures associated with reactor operation will no longer exist.

The proposed exemptions would allow TMI to revise the station emergency plan to correspond to the reduced scope of remaining accidents and events. As such, the plan would no longer need to address response actions for events that would no longer be possible. The revised plan would thereby enhance the ability of the emergency response organization to respond to those scenarios that remain credible since emergency preparedness training and drills would focus only on applicable activities. Elimination of requirements for classification of emergency action levels for events that were no longer possible would enhance the ability of the Emergency Response Organization (ERO) to correctly classify those events that remain credible. As the proposed exemption will enhance the ability of the organization to respond to credible events, a resultant benefit to the public health and safety is realized.

Therefore, since the granting the exemptions would result in benefit to the public health and safety and would not result in a decrease in safety, the special circumstances required by 10 CFR 50.12(a)(2)(iv) exist.

8.0 PRECEDENT

The exemption requests for 10 CFR 50.47(b), 10 CFR 50.47(c)(2) and 10 CFR Part 50, Appendix E, requirements are consistent with exemptions on the same emergency planning requirements that recently have been issued by the NRC for other nuclear power reactor facilities beginning decommissioning. Specifically, the NRC granted similar exemptions to OPPD for FCS (Reference 33), ENO for VY (Reference 34); to Southern California Edison Company for SONGS, Units 1, 2, and 3 (Reference 35); to Duke Energy Florida, Inc. for CR3 (Reference 36); and to Dominion Energy Kewaunee, Inc. for Kewaunee Power Station (Reference 37). Similar to the current request, these precedents each resulted in exemptions from certain emergency planning requirements in 10 CFR 50.47(b); 10 CFR 50.47(c)(2); and 10 CFR Part 50, Appendix E, related to the elimination of offsite radiological emergency plans and reduction in the scope of the onsite emergency planning activities. For the same reasons that the NRC recently issued these exemptions, Exelon seeks approval of the enclosed proposed exemption requests.

9.0 ENVIRONMENTAL ASSESSMENT

The proposed exemption meets the eligibility criterion for categorical exclusion set forth in 10 CFR 51.22(c)(25), because the proposed exemption involves: (i) no significant hazards consideration; (ii) no significant change in the types or significant increase in the amounts of any effluents that may be released offsite; (iii) no significant increase in individual or cumulative public or occupational radiation exposure; (iv) no significant construction impact; (v) no significant increase in the potential for or consequences from radiological accidents; and (vi) the requirements from which the exemption is sought involve requirements of an administrative, managerial, or organizational nature. Therefore, pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the proposed exemption.

(i) No Significant Hazards Consideration Determination

Exelon has evaluated the proposed exemption to determine whether or not a significant hazards consideration is involved by focusing on the three standards set forth in 10 CFR 50.92 as discussed below:

1. <u>Does the proposed exemption involve a significant increase in the probability or consequences of an accident previously evaluated?</u>

   The proposed exemptions have no effect on structures, systems, and components (SSCs) and no effect on the capability of any plant SSC to perform its design function. The proposed exemptions would not increase the likelihood of the malfunction of any plant SSC.

   When the exemptions become effective, there will be no credible events that would result in doses to the public beyond the exclusion area boundary that would exceed the Environmental Protection Agency (EPA) Protective Action Guides (PAGs). The probability of occurrence of previously evaluated accidents is not increased, since most previously analyzed accidents will no longer be able to occur and the probability and consequences of the remaining Fuel Handling Accident (FHA) are unaffected by the proposed exemption.

   Therefore, the proposed exemption does not involve a significant increase in the probability

or consequences of an accident previously evaluated.

2. <u>Do the proposed exemptions create the possibility of a new or different kind of accident from any accident previously evaluated?</u>

   The proposed exemption does not involve a physical alteration of the plant. No new or different type of equipment will be installed and there are no physical modifications to existing equipment associated with the proposed exemption. Similarly, the proposed exemption will not physically change any SSCs involved in the mitigation of any accidents. Thus, no new initiators or precursors of a new or different kind of accident are created. Furthermore, the proposed exemption does not create the possibility of a new accident as a result of new failure modes associated with any equipment or personnel failures. No changes are being made to parameters within which the plant is normally operated, or in the setpoints which initiate protective or mitigative actions, and no new failure modes are being introduced.

   Therefore, the proposed exemption does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. <u>Do the proposed exemptions involve a significant reduction in a margin of safety?</u>

   The proposed exemption does not alter the design basis or any safety limits for the plant. The proposed exemption does not impact station operation or any plant SSC that is relied upon for accident mitigation.

   Therefore, the proposed exemption does not involve a significant reduction in a margin of safety.

Based on the above, Exelon concludes that the proposed exemption presents no significant hazards consideration, and, accordingly, a finding of "no significant hazards consideration" is justified.

(ii) There is no significant change in the types or significant increase in the amounts of any effluents that may be released offsite.

There are no expected changes in the types, characteristics, or quantities of effluents discharged to the environment associated with the proposed exemption. There are no materials or chemicals introduced into the plant that could affect the characteristics or types of effluents released offsite. In addition, the method of operation of waste processing systems will not be affected by the exemption. The proposed exemption will not result in changes to the design basis requirements of SSCs that function to limit or monitor the release of effluents. All the SSCs associated with limiting the release of effluents will continue to be able to perform their functions. Therefore, the proposed exemption will result in no significant change to the types or significant increase in the amounts of any effluents that may be released offsite.

(iii) There is no significant increase in individual or cumulative public or occupational radiation exposure.

The exemption will result in no expected increases in individual or cumulative occupational radiation exposure on either the workforce or the public. There are no expected changes in normal occupational doses. Likewise, design basis accident dose is not impacted by the proposed exemption.

(iv) There is no significant construction impact.

No construction activities are associated with the proposed exemption.

(v) There is no significant increase in the potential for or consequences from radiological accidents.

See the no significant hazards considerations discussion in Item (i)(1) above.

(vi) Requirements of an administrative, managerial, or organizational nature.

The proposed exemptions will form the basis for a reduction in size of the TMI-1 emergency response organization commensurate with the reduction in consequences of radiological events that will be possible at TMI-1 once the facility is in the permanently defueled condition. They also will modify the requirements for emergency planning. Therefore, the exemptions address requirements of an administrative, managerial, or organizational nature.

10.0 REFERENCES

1. NSIR/DPR-ISG-02, Interim Staff Guidance, "Emergency Planning Exemption Requests for Decommissioning Nuclear Power Plants," dated May 11, 2015 (ADAMS Accession No. ML14106A057)
2. U.S. Environmental Protection Agency, EPA 400-R-92-001, "Manual of Protective Action Guides and Protective Actions Guidelines for Nuclear Incidents," dated October 1991 (reprinted May 1992)
3. Letter from J. Bradley Fewell (Exelon Generation Company, LLC) to U.S. Nuclear Regulatory Commission, "Certification of Permanent Cessation of Power Operations for Three Mile Island Nuclear Station, Unit 1," dated June 20, 2017 (NRC Accession No. ML17171A151)
4. Letter from U.S. Nuclear Regulatory Commission to Bryan C. Hanson, (Exelon Generation Company, LLC), "Three Mile Island Nuclear Station, Units 1 and 2 – Issuance of Amendment No. 296 for Unit 1 RE: Changes to Emergency Plan for Post-Shutdown and Permanently Defueled Condition (EPID L-2018-LLA-0073), dated April 18, 2019 (ADAMS Accession No. ML19065A114)
5. Letter from Michael P. Gallagher, (Exelon Generation Company, LLC) to U.S. Nuclear Regulatory Commission – "Three Mile Island Nuclear Station, Unit 1 – Post-Shutdown Decommissioning Activities Report," dated April 5, 2019 (ADAMS Accession No. ML19095A041)
6. Letter from Michael P. Gallagher, (Exelon Generation Company, LLC) to U.S. Nuclear Regulatory Commission – "Spent Fuel Management Plan for Three Mile Island Nuclear Station – Unit 1," dated April 5, 2019 (ADAMS Accession No. ML19095A009)
7. C-1101-202-E410-476, "DECOM Spent Fuel Pool Thermohydraulic Analysis," Revision 1, dated June 10, 2019
8. Federal Register Notice, Vol. 60, No. 120, (60 FR 32430-32442) "Emergency Planning Licensing Requirements for Independent Spent Fuel Storage Facilities (ISFSI) and Monitored Retrievable Storage Facilities (MRS)," dated June 22, 1995
9. C-1101-900-E000-088, "Fuel Handling Accident Dose Consequence - Post Permanent Shutdown," Revision 0, dated May 11, 2018

10. Nuclear Energy Institute (NEI) 99-01, Revision 6 "Methodology for the Development of Emergency Action Levels for Non-Passive Reactors," dated November 2012 (ADAMS Accession No. ML12326A805).

11. Letter from Mark Thaggard (USNRC) to Susan Perkins-Grew (NEI), "U.S. Nuclear Regulatory Commission Review and Endorsement of NEI 99-01, Revision 6, Dated November 2012 (TAC No. D92368)," dated March 28, 2013 (ADAMS Accession No. ML12346A463)

12. U.S. Nuclear Regulatory Commission, Commission Paper SECY-00-0145, "Integrated Rulemaking Plan for Nuclear Power Plant Decommissioning," dated June 28, 2000 (ADAMS Accession No. ML003721626)

13. U.S. Nuclear Regulatory Commission, Commission Paper SECY-13-0112, "Consequence Study of a Beyond-Design-Basis Earthquake Affecting the Spent Fuel Pool for a U.S. Mark I Boiling Water Reactor," dated October 9, 2013 (ADAMS Accession No. ML13256A339)

14. Federal Register Notice, Vol. 76, No. 226 (76 FR 72560), Enhancements to Emergency Preparedness Regulations, dated November 23, 2011

15. U.S. Nuclear Regulatory Commission, Bulletin 2005-02, "Emergency Preparedness and Response Actions for Security-Based Events," dated July 18, 2005 (ADAMS Accession No. ML051740058)

16. NUREG-1738, "Technical Study of Spent Fuel Accident Risk at Decommissioning Nuclear Power Plants," dated February 2001 (ADAMS Accession No. ML010430066)

17. U.S. Nuclear Regulatory Commission, NUREG-0696, "Functional Criteria for Emergency Response Facilities," dated February 1981 (ADAMS Accession No. ML051390358)

18. Technical Evaluation 623073, "TMI Spent Fuel Pool Draindown Shine Dose Rate Evaluation, Revision 0," dated May 28, 2018

19. Letter from U.S. Nuclear Regulatory Commission to All Power Reactor Licensees, "Request for Information Pursuant to Title 10 of the Code of Federal Regulations 50.54(f) Regarding Recommendations 2.1, 2.3, And 9.3, of The Near-Term Task Force Review of Insights from The Fukushima Dai-Ichi Accident," dated March 12, 2012 (ADAMS Accession No. ML12073A348)

20. Letter from Mr. James Barstow (Exelon Generation Company, LLC) to U.S. Nuclear Regulatory Commission, "Exelon Generation Company, LLC, Seismic Hazard and Screening Report (Central and Eastern United States (CEUS) Sites), Response to NRC Request for Information Pursuant to 10 CFR 50.54(f) Regarding Recommendation 2.1 of the Near-Term Task Force Review of Insights from the Fukushima Dai-ichi Accident," dated March 31, 2014 (ADAMS Accession No. ML14090A271)

21. Letter from U.S. Nuclear Regulatory Commission to Mr. Bryan C. Hanson (Exelon Generation Company, LLC), "Three Mile Island Nuclear Station, Unit 1 - Staff Assessment of Information Provided Pursuant To Title 10 of the Code of Federal Regulations Part 50, Section 50.54(f), Seismic Hazard Reevaluations for Recommendation 2.1 Of The Near-Term Task Force Review Of Insights From The Fukushima Dal-ichi Accident (CAC NO. MF3905)," dated August 14, 2015 (ADAMS Accession No. ML15223A215)

22. Letter, Jack R. Davis (USNRC) to Joseph E. Pollock (NEI), "Endorsement of Electric Power Research Institute Report 3002007148, Seismic Evaluation Guidance: Spent Fuel Pool Integrity Evaluation," dated March 17, 2016 (ADAMS Accession No. ML15350A158)

23. EPRI, "Seismic Evaluation Guidance: Spent Fuel Pool Integrity Evaluation," Electric Power Research Institute Technical Update 3002007148, dated February 2016 (ADAMS Accession No. ML16055A021)

24. Letter from Michael P. Gallagher, (Exelon Generation Company, LLC) to U.S. Nuclear Regulatory Commission – "License Amendment Request – Proposed Defueled Technical Specifications and Revised License Condition for Permanently Defueled Conditions," dated July 25, 2018 (ADAMS Accession No. ML18206A545)

25. U.S. Nuclear Regulatory Commission, Regulatory Guide 1.183, Revision 0, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors," dated July 2000 (ADAMS Accession No. ML003716792)

26. 990-3017, "Three Mile Island Unit No. 2 Fire Protection Program Evaluation, Revision 12, dated May 18, 2018

27. U.S. Nuclear Regulatory Commission, Commission Paper SECY-99-168, "Improving Decommissioning Regulations for Nuclear Power Plants," dated June 30, 1999 (ADAMS Accession No. ML992800087)

28. U.S. Nuclear Regulatory Commission, NUREG-0586, "Final Generic Environmental Impact Statement on Decommissioning of Nuclear Facilities," dated October 2002

29. U.S. Nuclear Regulatory Commission, Regulatory Guide 1.174, Revision 2, "An Approach for Using Probabilistic Risk Assessment in Risk Informed Decisions on Plant-Specific Changes to the Licensing Basis," dated May 2011 (ADAMS Accession No. ML100910006)

30. U.S. Nuclear Regulatory Commission, NUREG-2161, "Consequence Study of a Beyond-Design-Basis Earthquake Affecting the Spent Fuel Pool for a U.S. Mark I Boiling Water Reactor," dated September 2014 (ADAMS Accession No. ML14255A365)

31. Three Mile Island Nuclear Station, Unit 1, Updated Final Safety Analysis Report, Chapter 14, "Safety Analysis," Section 14.2.2.8, "Fuel Cask Drop Accident," Revision 24, April 2018

32. Letter from U.S. Nuclear Regulatory Commission to Mr. Bryan C. Hanson (Exelon Generation Company, LLC), "Three Mile Island Nuclear Station; Unit 1 - Approval of Certified Fuel Handler Training and Retraining Program (CAC NOS. MF9960, EPID L-2017-LLL-0013)" dated December 29, 2017 (ADAMS Accession No. ML17228A729)

33. U.S. Nuclear Regulatory Commission, Omaha Public Power District, Fort Calhoun Station, "Fort Calhoun Station, Unit No. 1 - Exemptions from Certain Emergency Planning Requirements and Related Safety Evaluation (CAC NO. MF9067; EPID L-2016-LLE-0003)," Dated December 11, 2017, (ADAMS Accession No. ML17263B191)

34. Federal Register Notice, Vol. 80, No. 242 (80 FR 78776), Entergy Nuclear Operations, Inc.; Vermont Yankee Nuclear Power Station, Exemption; issuance, dated December 17, 2015

35. Federal Register Notice, Vol. 80, No. 113 (80 FR 33558), Southern California Edison Company; San Onofre Nuclear Generating Station, Units 1, 2, and 3, and Independent Spent Fuel Storage Installation, Exemption; issuance, dated June 12, 2015

36. Federal Register Notice, Vol. 80, No. 69 (80 FR 19358), Duke Energy Florida, Inc.; Crystal River Unit 3 Nuclear Generating Station, Exemption; issuance, dated April 10, 2015

37. Federal Register Notice, Vol. 79, No. 214 (79 FR 65715), Dominion Energy Kewaunee, Inc.; Kewaunee Power Station, Exemption; issuance, dated November 5, 2014

ATTACHMENT 2

THREE MILE ISLAND NUCLEAR STATION

THREE MILE ISLAND NUCLEAR STATION ZIRCONIUM FIRE ANALYSIS FOR DRAINED SPENT FUEL POOL

(CALCULATION C-1101-202-E410-476, Revision 1)

Design Analysis Cover Sheet

Design Analysis		**Last Page No.**	66
Analysis No.:	C-1101-202-E410-476	**Revision:**	1 Major ☒ Minor ☐
Title:	DECOM Spent Fuel Pool TH Analysis		
EC No.:	623197	**Revision:**	0

		Component(s):	
Station(s):	Three Mile Island		
Unit No.:	01	N/A	
Discipline:	TEDM		
Descrip. Code/Keyword:	Spent Fuel		
Safety/QA Class:	Augmented Quality		
System Code:	202		
Structure:	N/A		

CONTROLLED DOCUMENT REFERENCES

Document No.:	From/To	Document No.:	From/To
ER-TM-TSC-0016 Rev 7	From		
HI-89407 Rev 6	From		

Is this Design Analysis Safeguards Information? Yes ☐ No ☒ If yes, see SY-AA-101-106

Does this Design Analysis contain Unverified Assumptions? Yes ☐ No ☒ If yes, ATI/AR#:

This Design Analysis SUPERCEDES: None in its entirety.

Description of Revision (list changed pages when all pages of original analysis were not changed): See Revision Summary page.

	Print Name	Sign Name	Date
Preparer:	Dan Sells / Bill McSorley	Dan A Sells / [signature]	5/30/19 / 6/3/19

Method of Review: Detailed Review ☒ Alternate Calculations (attached) ☐ Testing ☐

	Print Name	Sign Name	Date
Reviewer:	Mara Levy / Greg Heasley	[signature] / [signature]	5/30/19 / 5-30-19

Review Notes: Independent review ☒ Peer review ☐

An independent review of the product was performed IAW CC-AA-309, Rev. 11 and CC-AA-309-1001, Rev. 9. The minimum decay time after shut down of TMI-1 Cycle 22 in Attachment 1 were determined to be acceptable given the inputs, assumptions, and methodology used.

(For External Analyses Only)

	Print Name	Sign Name	Date
External Approver:	N/A	N/A	N/A
Exelon Reviewer:	N/A	N/A	N/A

Independent 3rd Party Review Reqd? Yes ☐ No ☒

	Print Name	Sign Name	Date
Exelon Approver:	Patrick Brady / Patrick Bennett	Patrick Brady / Patrick Bennett	6/4/19 / 6/10/19

Table of Contents

Revision Summary

Revision 1

- Added the following to the available mass to heat up
  - (7) M5 Spacer Grids
  - (1) HMP Spacer Grid
  - Upper End Fitting
  - Lower End Fitting
- Adjusted the lengths of the fuel rods, guide tubes, and instrument tube to be their actual lengths instead of just the fuel stack length.
- Added references for fuel rod and instrument tube lengths.
- Repeated the Purpose #2 calculation at 1 year after shutdown as a revised input for Purpose #5.
- Changed the initial temperature for Purpose #5 to just above the maximum design temperature calculated using the decay heat at 1 year.
- Added references for heat capacity/specific heat for CF3 stainless steel (end fittings) and Inconel Alloy 718 (HMP Spacer Grid).
- Changed the temperature used to calculate the heat capacities to the midpoint temperature rather than the 500F value and recalculated the heat capacities.
- Updated the specific heat study (Attachment 2) with the addition of CF3 stainless steel and Inconel Alloy 718.
- Revised the Decay Heat Generation Rate calculation to include CF3 stainless steel and Inconel Alloy 718 components.
- Adjusted section and equations numbers as necessary.

1. Purpose

1. Establish the total heat generation rate in the spent fuel versus time after permanent shutdown in 2019 and all fuel is loaded into the SFP.
2. Determine spent fuel pool water temperature with one or both cooling trains in operation, assuming the reactor has been shut down for 14 days, and NSCCW temperature to the spent fuel coolers is 95°F. This supports DSAR.
3. Determine the "time to boil" (TTB) and "time to Top of Active Fuel" (TTAF) versus time after permanent shutdown in 2019 and all fuel is loaded into the SFP.
4. Determine the spent fuel assembly with the highest heat generation rate, and determine the heat generation rate of that assembly versus time after reactor shutdown.
5. Determine the minimum decay time for the limiting spent fuel assembly (purpose #4) to heat up to 900°C (1652°F/1173 K) in 10 hours after a loss of all cooling. This is required for EPlan exemption in accordance with Reference 3.3.

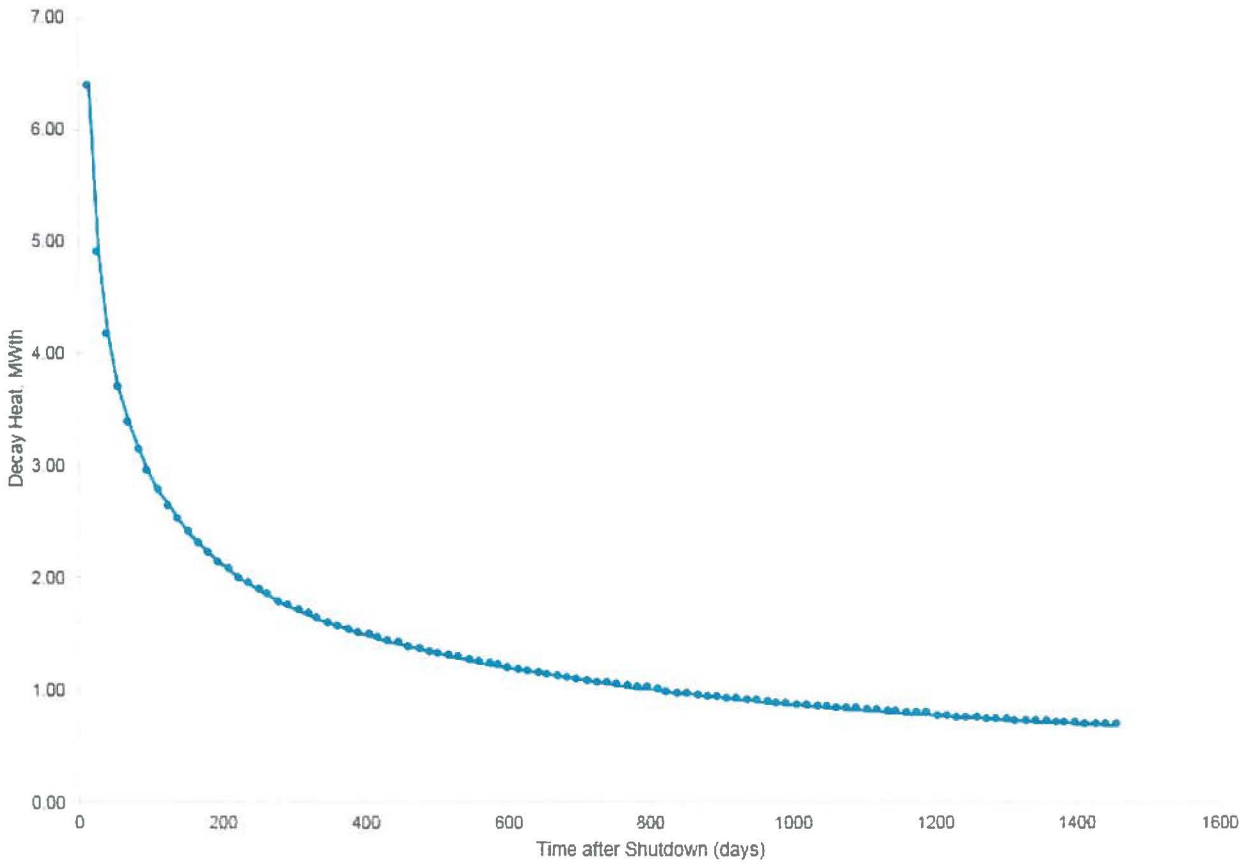
Background:

1. Purpose #1 is required to perform calculations for Purpose #2 and #3. This result is not required for a purpose outside this analysis.
2. Purpose #2 is required for the Defueled Safety Analysis Report (DSAR) section 3.2. This DSAR section is a revision of UFSAR section 9.4. This revision replaces the maximum SF pool temperatures based on the "bounding" design basis SFP heat load with the temperatures based to the maximum heat with the defueled spent fuel pool heat load. This result will be used to change a design basis parameter in the UFSAR/DSAR. The spent fuel pool temperature will also be calculated at 1 year after shutdown to support Purpose #5.
3. Purpose #3 is required for evaluation of loss of spent fuel pool cooling mitigating strategies, and the required response time. The cause for the loss of spent fuel pool cooling includes fire, aircraft impact, multiple equipment failures, loss of offsite power, etc. where the spent fuel pool integrity is not affected. This result does not change a design basis parameter.
4. Purpose #4 is required to perform calculations for Purpose #5. This result is not required for a purpose outside this analysis.
5. Purpose #5 is required to support a request for exemption to Eplan requirements. The regulatory basis for this exemption is described in reference 3.3 section 5 item 2. This result does not change a design basis parameter.

2. Results and Conclusions

2.1. The total heat generation rate in the spent fuel versus time after permanent shutdown in 2019 and all fuel is loaded into the SFP is shown in Figure 2.1 below. Tabular values of this result are in Appendix 7.1.

Figure 2.1 - Post-Defueled SFP Decay Heat



The spent fuel pool decay heat at 14 days after shutdown is 6.38 MW_{th}, and the spent fuel pool decay heat at 1 year after shutdown is 1.57 MW_{th} (input for purpose #2).

2.2. Maximum SFP Temperature Results

2.2.1. With one SF cooling train in service at 14 days after shutdown, the maximum SF pool temperature is 179.7°F.

2.2.2. With both SF cooling trains in service at 14 days after shutdown, the maximum SF pool temperature is 137.4°F.

2.2.3. The maximum "expected" SFP temperature at 14 days after shutdown is 122.4°F (input for purpose #3).

2.2.4. With one SF cooling train in service at 1 year after shutdown, the maximum SF pool temperature is 115.9°F.

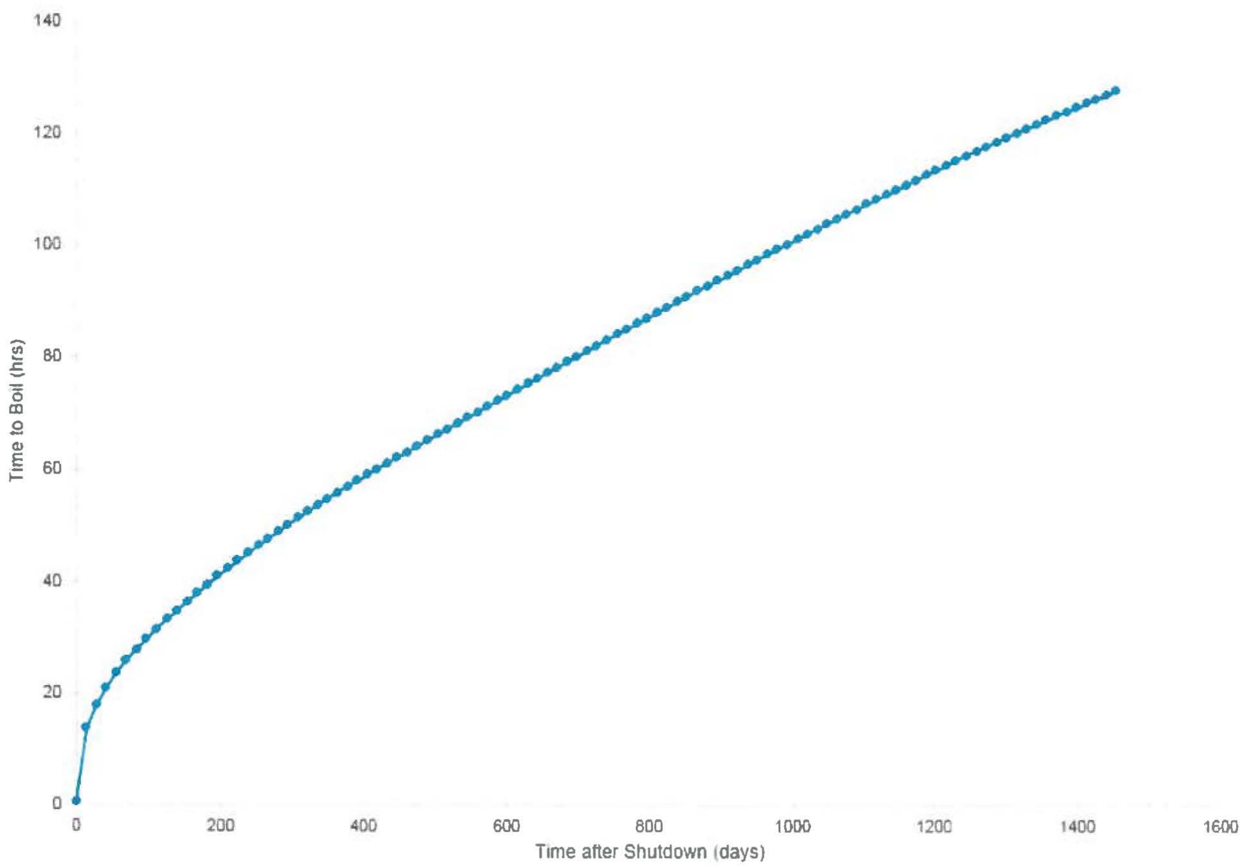
2.2.5. With both SF cooling trains in service at 1 year after shutdown, the maximum SF pool temperature is 105.4°F (input for purpose #5).

2.2.6. The maximum "expected" SFP temperature at 1 year after shutdown is 90.4°F.

2.3. SFP TTB/TTAF

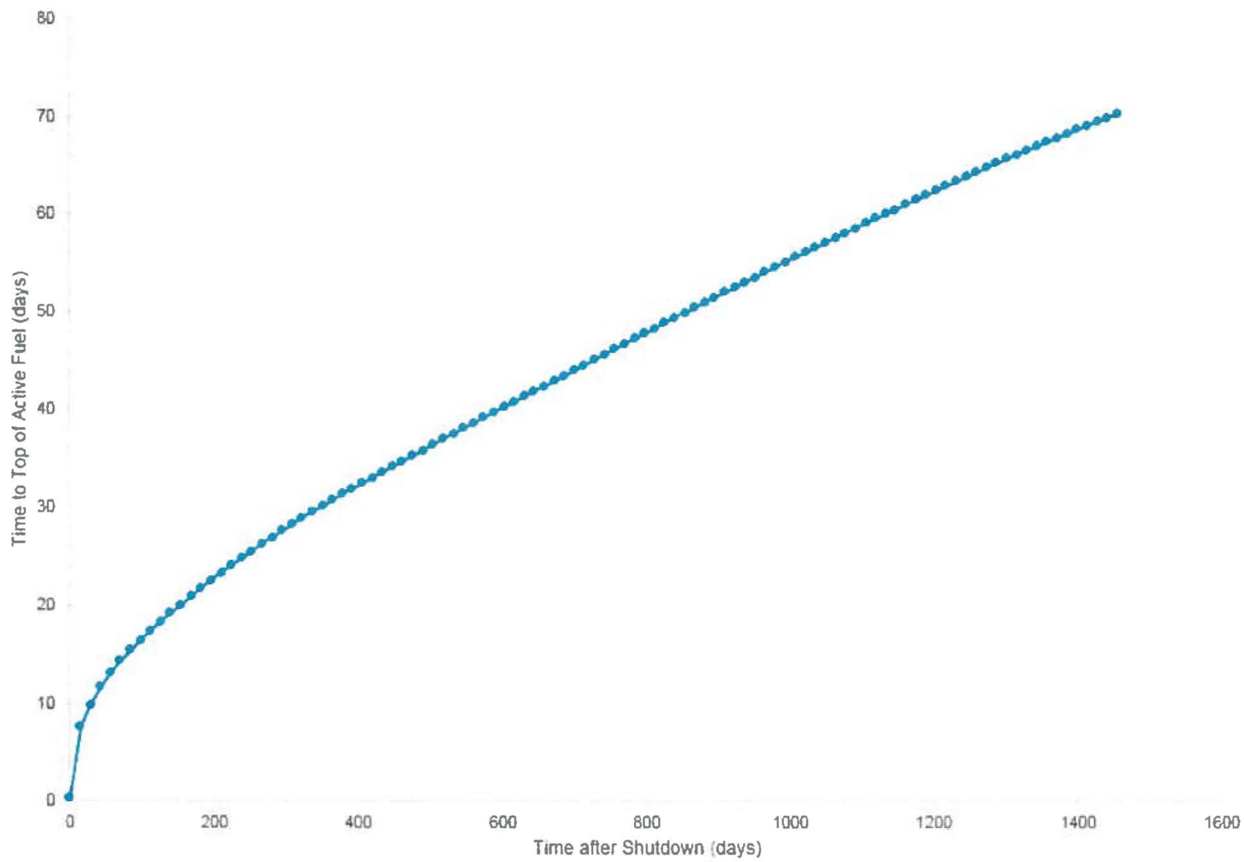
2.3.1. The time to boil after permanent shutdown is shown in Figure 2.3.1 below. Tabular values of this result are in Appendix 7.3.

Figure 2.3.1 - Post-Defueled Time to Boil



2.3.2. The time to top of active fuel after permanent shutdown is shown in Figure 2.3.2 below. Tabular values of this result are in Appendix 7.3.

Figure 2.3.2 - Post-Defueled Time to Top of Active Fuel



2.4. A best estimate decay heat load for the fuel assembly with the maximum heat load in the TMI SFP following TMI-1 Cycle 22 shutdown has been calculated using the ORIGEN2 computer code. Per Attachment 1 Section 7.3 of this calculation, the decay heat load from this Batch 22A fuel assembly as a function of decay time (in days) after Cycle 22 shutdown is shown in the table below. This decay heat load is appropriate to be used as input to the Beyond Design Basis zirconium fire event for a drained SFP.

Maximum Fuel Assembly Decay Heat (Watts) for Various Decay Times After TMI-1 Cycle 22 Shutdown

	Decay Time (Days)				
	365.0D	548.0D	730.0D	913.0D	1095.0D
Decay Heat (W)	6.44E+03	4.70E+03	3.65E+03	2.94E+03	2.44E+03

The following polynomial fit represents this data for decay heat (x) between 2.44 and 6.44 kW. The polynomial fit is in good agreement with the individual ORIGEN2 data points. A 1% multiplier is applied to the Excel fit for conservatism.

$$\mathbf{Y\ (days) = 1.01 * [-9.22479E\text{-}09*x^3 + 1.59163E\text{-}04*x^2 - 1.01332E{+}00*x + 2.75333E{+}03]}$$

2.5. The minimum decay time for the limiting spent fuel assembly (Batch 22A) to heat up to 900°C (1652°F/1173K) in 10 hours after a loss of all cooling is 488 days.

3. References and Computer Programs

3.1. CC-AA-309 "Control of Design Analyses"

3.2. CC-AA-309-1001, "Guidelines for Preparation and Processing of Design Analyses"

3.3. NISR/DPR-ISG-02 "Emergency Planning Exemption Requests for Decommissioning Nuclear Power Plants"

3.4. FS1-0030101, Rev. 1, TMI-1 Cycle 22 Final Fuel Cycle Design

3.5. EX0009886 RE Decay Heat Version 0

3.6. Exelon Nuclear Fuels TMI Fuel Database, "C22database.accdb," 1/03/2018.

3.7. Exelon TODI NF173266, Rev. 0, "TMI End-of-Cycle 21 Data and FIDMS Files," 9/18/2017.

3.8. Exelon TODI NF173357, Rev. 0, "TMI-1 Cycle 22 Control Rod Drop Times and Start-up Data," 10/23/2017.

3.9. UFSAR Section 9.4

3.10. HI-89402, Rev. 2, Thermal Hydraulic Analysis of TMI-1 Spent Fuel Pool, dated September 30, 1990.

3.11. HI-89407 Holtec International, Licensing Report for Pool A Reracking, Rev. 6, dated October 30, 1990 (UFSAR Chapter 9 Reference 10)

3.12. C-1101-202-E270-438, Rev. 1, Core and SFP Time to Boil/Uncover

3.13. 1505-1, Rev. 62, Fuel and Control Component Shuffles

3.14. ER-TM-TSC-0016, Rev. 7, RCS and SFP Heatup and Inventory Boiloff Following Loss of Active Decay Heat Removal

3.15. FS1-0029697, Rev. 1, Task 76 Inputs, Mechanical Design Information for Safety Analysis, and Regulatory Compliance for TMI1-22

3.16. "Chart of the Nuclides", Thirteenth Edition. General Electric, 1984.

3.17. FS1-0036046-1.0 Confirmation of Material Properties for TMI

3.18. Kim, Choong S. "Thermophysical Properties of Stainless Steels." Argonne National Laboratory, 1975.

3.19. Farwick, D. G., and R. N. Johnson. "Thermophysical Properties of Selected Wear-Resistant Alloys." U.S. Dept. of Energy, 1980.

3.20. AREVA Dwg. 9017973-000, MK-B-HTP Fuel Bundle Assembly

3.21. Framatome Dwg. 1238726, Rev 1, Instrumentation Tube (MONOBLOC)

3.22. Esmaili, H. "Spent Fuel Assembly Heat Up Calculations in Support of Task 2 of User Need NSIR-2015-01," April 2016 (ADAMS Accession No. ML16110A431).

4. Assumptions

4.1. Purpose #1 Assumptions

4.1.1. Fuel batch burnups are taken from NEMO calculations documented in Tables 38-39 and Figure 34 of Reference 3.4. NEMO is the design code for TMI-1 core simulations and calculated burnups are representative of actual burnups measured by the core monitoring system. Burnups prior to Cycle 21 are based on core follow calculations that are depleted to actual cycle lengths. Cycle 21 burnups are based on a design depletion to 647 EFPD; differences from the actual Cycle 21 length of 648.8 EFPD (Reference 3.7) are negligible with respect to this decay heat calculation. Cycle 22 burnups are based on 720 EFPD, which is the maximum licensed cycle length for Cycle 22. Higher fuel assembly burnups are conservative for decay heat calculations.

4.1.2. Reference 3.4 lists fuel batch enrichments based on the nominal base enrichment of the batch. Since lower enrichments yield higher decay heat results, actual enrichments based on NRC 741 forms are used in this calculation. This includes segments of the fuel assemblies that have lower than nominal enrichments (i.e., Gad rods and axial blankets). The actual enrichments were obtained from Reference 3.6.

4.1.3. End-of-Cycle 22 date: 30 September 2019

Basis: This date is conservative and corresponds to the maximum cycle length based on a licensed cycle length of 720 EFPD.

4.2. Purpose #2 Assumptions

4.2.1. No assumptions were made in this section.

4.3. Purpose #3 Assumptions

4.3.1. The initial SFP temperature is 125°F.

Basis: This value is conservative based on the maximum "expected" SFP temperature of 122.4°F determined in section 6.2.1 of this calculation. Setting the initial temperature at the maximum "expected" pool temperature is conservative because the heat-up time would be increased with a lower starting temperature.

4.3.2. The initial SFP level is assumed to be at the low-level limit of 343'-6".

Basis: This value is conservative based on the normal operating band of the SFP of 344'-5" to 344'-9" (Reference 3.13 Figure 1) because less water would result in a lower heat-up time.

4.4. Purpose #4 Assumptions

See Attachment 2 Section 4.

4.5. Purpose #5 Assumptions

4.5.1. No assumptions were made in this section.

5. Design Input

5.1. Purpose #1 Design Inputs

5.1.1. Batch-average burnups for fuel loaded in Cycle 22 are taken from Tables 38-39 of Reference 3.4.

5.1.2. Actual batch enrichments for fuel loaded in Cycle 22 are taken from Reference 3.6. For each batch, the lowest as-built enrichment of all fuel assemblies in the batch is used to represent the batch.

5.1.3. Inputs for every fuel assembly currently stored in the spent fuel pool (burnup, enrichment, cycles in core, batch number) are taken from the TMI Fuel Database (Reference 3.6).

5.1.4. End-of-Cycle 21 date/time (reactor shutdown): 18 September 2017 at 05:16 (Reference 3.7).

5.1.5. Beginning-of-Cycle 22 date/time (initial criticality): 07 October 2017 at 17:43 (Reference 3.8).

5.2. Purpose #2 Design Inputs

5.2.1. SFP Heat generation rate at 14 days after shutdown is 6.38 MW_{th}.

Basis: This input assumes that the core offload is complete and the plant is ready to initiate Decom Phase 2 at 14 days after reactor shutdown. The total spent fuel pool heat load at 14 days was determined in section 6.1.1 of this calculation. This assumption is conservative for all times after the reactor has been shut down for at least 14 days.

5.2.2. SFP Heat generation rate at 1 year after shutdown is 1.57 MW_{th}.

Basis: This input assumes that all of the spent fuel remains in the spent fuel pool (no ISFSI campaigns). The total spent fuel pool heat load at 1 year was determined in section 6.1.1 of this calculation. This assumption is conservative for all times after the reactor has been shut down for at least 1 year.

5.2.3. Cooler flow rate (GPM) is 1000 GPM.

Basis: This is the design flow rate as described in UFSAR section 9.4.

5.2.4. "Design" NSCCW Inlet temperature (°F) is 95. The maximum "expected" NSCCW inlet temperature is 80°F.

Basis: The design NSCCW inlet temperature is described in UFSAR section 9.4 and 9.6.

The maximum expected NSCCW temperature is based on the planned shutdown no later than 30 September 2019. The river water temperature in October (or later) would be colder than the design temperature. The PPC data for RW inlet temperature (A0089) was reviewed for 2017 and 2016. River water temperatures varied between 56°F and 72°F in 2017, and between 54°F and 72°F in 2016. The maximum expected NSCCW temperature of 80°F is based on (1) a conservative river water temperature for October of 75°F and (2) lower NSCCW to NR differential temperature (of 5°F) because the spent fuel pool will be the only significant heat load on the NSCCW/NR systems.

5.2.5. Spent Fuel Cooler HX Effectiveness is 0.516 at 140°F or above, and 0.514 between 130 & 140°F.

Basis: The design maximum spent fuel pool temperatures described in UFSAR section 9.4 were determined in Reference 3.11. The same analytical model used in Reference 3.11 (section 5) is used in Reference 3.10. Reference 3.10 Table 5.1 provides the "cooler effectiveness" coefficients used for the design analysis. Using a value of 0.516 is conservative for all temperatures at or above 140°F.

5.3. Purpose #3 Design Inputs

5.3.1. SFP Heat Generation rate calculated from section 6.1.1. All the fuel assemblies currently in the SFP and the core are assumed to stay in the pool indefinitely following permanent shutdown.

Basis: This is conservative because ISFSI campaigns will be scheduled and completed in the years following permanent shutdown to move the fuel assemblies from the spent fuel pool to dry cask storage. Each ISFSI campaign would remove additional heat loads from the pool resulting in higher TTB and TTAF times.

5.3.2. Mass of SFP water available for TTB/TTAF is $3.43E^6$ lbm (Reference 3.12).

Basis: The volume of SFP water available for TTB and TTAF is the water above the elevation of the fuel (318.75 ft) up to the low-level limit (343.5 ft). This is conservative because a large quantity of water is available for heat-up in the fuel region as well as the north end of the "A" pool being empty due to inaccessibility. The available mass for TTAF is different than Reference 3.12 available mass for boiloff because this analysis is calculating the time to top of active fuel (318.75 ft) while Reference 3.12 was calculating time to 7 feet above the fuel (325.75 ft).

5.4. Purpose #4 Design Inputs

See Attachment 1 Section 5.

5.5. Purpose #5 Design Inputs

5.5.1. The minimum uranium loading of the limiting fuel assembly in the Cycle 22 Core is 485.75 kgU (Reference 3.4).

5.5.2. The atomic weight of uranium is 238.029 g/mole (Reference 3.16).

5.5.3. The atomic weight of oxygen is 15.999 g/mole (Reference 3.16).

5.5.4. The density of M5 is 6.50 g/cm^3 at 70°F (Reference 3.17).

5.5.5. Fuel Mechanical Data (Reference 3.15):

Guide Tubes
- Number per bundle = 16
- Material = M5
- Guide Tube Length = 156.249 inches
- Guide Tube ID = 0.498 inches
- Guide Tube OD = 0.530 inches

Instrument Tube
- Number per bundle = 1
- Material = M5
- Instrument Tube Length (Reference 3.21) = 155.285 inches
- Instrument Tube ID* = 0.400 inches
- Instrument Tube OD = 0.493 inches

Fuel Rods
- Number per bundle = 208 rods
- Material = M5
- Fuel Rod Length (Reference 3.20) = 155.00 inches
- Cladding OD = 0.430 inches
- Cladding ID = 0.380 inches

HTP Spacer Grids
- Number per bundle = 7
- Material = M5
- Mass = 1088.7 grams

HMP Spacer Grid
- Number per bundle = 1
- Material = Inconel Alloy 718
- Mass = 1197.7 grams

Upper End Fitting
- Material = CF3 stainless steel
- Mass** = 21.9 lbs

Lower End Fitting
- Material CF3 stainless steel
- Mass** = 14.9 lbs

* The MONOBLOC™ Instrument Tube has a smaller inner diameter (0.352 inches) at the upper end of the tube (Reference 3.15). Assuming the larger inner diameter is uniform over the active length of the fuel is conservative because it results in a smaller mass of cladding available for heat-up.

** The masses for the upper and lower end fittings include all small components such as stop pins, number plates, shims, spacers, cruciform springs, bolts, and nuts.

5.5.6. The initial fuel temperature is 110°F (316 K).

Basis: This value is conservative based on the maximum SFP temperature of 105.4°F determined in section 6.2.1 of this calculation. Setting the initial temperature to the maximum pool temperature at 1 year after shutdown is conservative because the spent fuel pool temperature at the minimum decay time of 488 days would be lower than at 1 year after shutdown, thus increasing the heat-up time.

5.5.7. The heat capacity for all fuel component materials are determined at the midpoint between the initial and final fuel temperatures, 881°F (745 K).

Basis: As shown in Attachment 2, the heat capacities for M5 and UO_2 increase with respect to temperature. This means that, as the fuel assembly temperature increases, more time would be required to heat the fuel assembly mass to 900°C. Therefore, using the midpoint temperature for material properties is conservative with respect to the assembly heat-up.

6. Method of Analysis & Numerical Analysis

6.1. Purpose #1 Methodology

6.1.1. Post-Defueled SFP Heat Load

The methodology described in ANSI/ANS-5.1 Decay Heat Power in Light Water Reactors was used to determine the total spent fuel pool heat load. The calculations were performed in a DTQSA controlled spreadsheet. A separate spreadsheet (Appendix 7.1) was created to yield the SFP heat load at various times after shutdown. This was then converted into a plot of decay heat vs. time after shutdown (Figure 2.1).

6.2. Purpose #2 Methodology

6.2.1. Maximum SFP Temperature

The methodology described in Reference 3.11 section 5.5 for determination of "Bulk Pool Temperature" is applied for this calculation. For the case of the maximum pool temperature the heat generated in the spent fuel equals the quantity of heat being transferred to the NSCCW system. In this case

Equation 6.2.1

$$Q_{SF} = Q_{SFC} = \dot{m} * H_x * (T_{POOL} - T_{NS})$$

Where:

Q_{SF} = heat generated in spent fuel, BTU/HR
Q_{SFC} = heat transferred through spent fuel cooler, BTU/HR
$\dot{m}$ = flow rate through spent fuel cooler, lbm/hr
H_x = heat exchanger effectiveness
T_{POOL} = bulk temperature in SF pool, °F
T_{NS} = NSCCW temperature into Spent Fuel Cooler

A spreadsheet (Appendix 7.2) was used to determine the maximum SF pool temperatures for one SF cooling train in service, both SF cooling trains in service, and "expected" NSCCW conditions at both 14 days and 1 year after shutdown.

6.3. Purpose #3 Methodology

6.3.1. Time to Boil

The methodology described in Reference 3.14 Enclosure 6.13 for Time to Boil was used in this calculation. A spreadsheet (Appendix 7.3) was used to calculate the TTB at various times after shutdown and converted to a plot of TTB vs. time after shutdown (Figure 2.3.1).

Equation 6.3.1

$$\Delta t\,(TTB) = 2.93E^{-7} * \left(\frac{mass}{DH}\right) * (212 - \text{Initial T})$$

Where:

Δt (TTB) = time to boil, hr
Mass = Mass available for TTB, lbm
DH = Decay Heat, MW_{th}
Initial T = Initial Temperature, °F

6.3.2. Time to Top of Active Fuel

The methodology described in Reference 3.14 Enclosure 6.13 for Time to Boiloff was used in this calculation. The calculation assumes that the spent fuel pool has already reached saturation. This time is conservative because it would take additional time for the pool to reach saturation before beginning to boiloff. A spreadsheet (Appendix 7.3) was used to calculate the TTAF at various times after shutdown and converted to a plot of TTAF vs. time after shutdown (Figure 2.3.2).

Equation 6.3.2

$$\Delta t\,(\mathrm{TTAF}) = \left(3.37\mathrm{E}^{-4} * \left(\frac{\mathrm{mass}}{\mathrm{DH}}\right)\right) \div \left(24\frac{\mathrm{hr}}{\mathrm{day}}\right)$$

Where:

Δt (TTAF) = time to top of active fuel, days
Mass = Mass available for TTAF, lbm
DH = Decay Heat, MW_{th}

6.4. Purpose #4 Methodology

See Attachment 1 Sections 6 and 7.

6.5. Purpose #5 Methodology

Section 6.5.10 determines the decay heat ($\dot{q}$) required to raise the fuel assembly mass from the initial temperature to 900°C in 10 hours. This calculation conservatively ignores any heat transfer from the fuel assembly to the environment around the assembly and assumes that the heat-up time begins when the SFP is completely drained ignoring the time required to drain-down/boil-off the SFP. Equations in sections 6.5.2 and 6.5.7 below are used to develop a conservative mass for the UO_2 fuel, fuel rods, guide tubes, and instrument tube. The FA heat up rate is determined using Equation 6.5.10. The specific heat of Uranium Dioxide, M5, CF3 stainless steel, and Inconel Alloy 718 are determined in sections 6.5.3, 6.5.4, 6.5.5, and 6.5.6, respectively.

This decay heat generation rate determined in section 6.5.10 is used as an input into the equation determined in Attachment 1 section 7.2 to find the minimum decay time required to achieve this decay heat.

The methodology is recognized to be very conservative. The analysis does not credit the mass of the racks, and the spent fuel is loaded in the pool in a 1-in-5 pattern (Reference: RIN 3150-AJ59 "Regulatory Improvements for Power Reactors Transitioning to Decommissioning" Appendix A).

6.5.1. Temperature Rise

The temperature rise for this analysis begins at the assumed initial temperature of 110°F (316 K) (Input 5.5.6) and ends at final temperature of 900°C (1173 K). Therefore, the temperature rise is:

Equation 6.5.1

$$\Delta T = T_{final} - T_{initial}$$

Using the values above yields the temperature rise:

$$\Delta T = 857 \text{ K}$$

6.5.2. Mass and Atomic Weight of Uranium Dioxide

Since the mass of the uranium in the uranium dioxide is known (Input 5.5.1), the total mass of uranium dioxide can be determined based on the chemical composition using Equation 6.5.2-1. The limiting fuel batch (22A) has 16 pins that contain gadolinium; however, the gadolinium is not included in this analysis. Neglecting the mass of the gadolinium is conservative because the additional thermal mass would increase the decay heat required to heat-up the assembly; thus, reducing the decay time.

Equation 6.5.2-1

$$UO_2 \rightarrow U + 2O$$

$$m_{UO_2} = \left(m_U + 2 * m_U\left(\frac{A_O}{A_U}\right)\right) * \left(1000\frac{g}{kg}\right)$$

Where:

m_{UO2} = mass of uranium dioxide, g
m_U = mass of uranium, kg
A_O = atomic weight of oxygen, g/mole
A_U = atomic weight of uranium, g/mole

Inputting the above values yields the mass of uranium dioxide:

$$m_{UO_2} = 5.51E^5 \text{ g}$$

The atomic weight of uranium dioxide is equal to the sum of the individual atomic weights of the uranium and oxygen.

Equation 6.5.2-2

$$A_{UO_2} = A_U + 2 * A_O$$ ~~Equation 6.5.2-2~~

Where:

A_{UO2} = atomic weight of uranium dioxide, g/mole
A_O = atomic weight of oxygen, g/mole
A_U = atomic weight of uranium, g/mole

Inputting the above values yields the uranium dioxide atomic weight:

$$A_{UO_2} = 270.027 \text{ g/mole}$$

6.5.3. Specific Heat of Uranium Dioxide

Uranium Dioxide specific heat is governed by the following equation (Reference 3.17):

Equation 6.5.3

$$c_{p,UO_2} = \left(\frac{K_1 * \theta^2 * \exp\left(\frac{\theta}{T_k}\right)}{T_k^2 * \left(\exp\left(\frac{\theta}{T_k}\right) - 1\right)^2} + 2K_2 * T_k + \frac{K_3 * E_D}{R * T_k^2} * \exp\left(\frac{-E_D}{R * T_k}\right) \right) \div A_{UO_2}$$

$$\text{for } 298\text{K} \leq T_k \leq 3120\text{K}$$

Where:

$c_{p,UO2}$ = specific heat of UO_2, J/g-K
Θ = 535.85 K
E_D = $157.7707*10^3$ J/mol
K_1 = 80.1314 J/mol-K
K_2 = $32.845*10^{-4}$ J/mol-K^2
K_3 = $23.62183*10^6$ J/mol
R = 8.134 J/mol-K (ideal gas constant)
T_k = Temperature, K
A_{UO2} = atomic weight of uranium dioxide, g/mole

Inputting the above values yields uranium dioxide specific heat:

$$c_{p,UO_2} = 0.302 \frac{\text{J}}{\text{g} - \text{K}}$$

6.5.4. Specific Heat of M5

Specific heat for M5 is governed by the following equation (Reference 3.17):

Equation 6.5.4

$$c_{p,M5} = 0.2375 + 0.0001591 * T \quad \text{for } 273\text{K} \leq T \leq 1100\text{K}$$

Where:

$c_{p,M5}$ = specific heat of M5, J/g-K

T = Temperature, K

Inputting the above values yields the M5 specific heat:

$$c_{p,M5} = 0.356 \frac{J}{g-K}$$

6.5.5. Specific Heat of CF3 Stainless Steel

CF3 stainless steel is thermodynamically the same as 304L stainless steel; therefore, the specific heat is governed by the same equation (Reference 3.18):

Equation 6.5.5

$$c_{p,SS} = (0.1122 + 3.222x10^{-5} * T) * \left(\frac{4.184\ J}{1\ cal}\right)$$

Where

$c_{p,SS}$ = specific heat of CF3 stainless steel, J/g-K
T = Temperature, K

Inputting the above values yields the CF3 stainless steel specific heat:

$$c_{p,SS} = 0.570 \frac{J}{g-K}$$

This value will be applied to the total mass of the upper and lower end fittings even though the masses include all the small components that may not be made up of CF3 stainless steel. The effects of these small components (shims, spacers, cruciform springs, bolts, and nuts) are considered inconsequential to the overall calculation due to their minimal masses.

6.5.6. Specific Heat of Inconel Alloy 718

The specific heat for Inconel Alloy 718 was found using the table in Appendix 7.5.

Interpolating and converting units (1 BTU/lb-F = 4.186798 J/g-K) yields the Inconel Alloy 718 specific heat, $c_{p,HMP}$:

$$c_{p,HMP} = 0.583 \frac{J}{g-K}$$

6.5.7. M5 Volume

The volume of M5 is determined by summing the volume of each of the credited M5 components (fuel rods, guide tubes, instrument tube) within the fuel assembly using the following equation. The M5 HTP grids are credited separately because their mass is known.

Equation 6.5.7

$$V_{M5,rods} = (V_{FR} + V_{GT} + V_{IT}) * \left(16.3871 \frac{cm^3}{in^3}\right)$$

Where

$V_{M5,rods}$ = total volume of M5 rods, guide tubes, and instrument tube, cm^3
V_{FR} = volume of M5 in the fuel rods, in^3
V_{GT} = volume of M5 in the guide tubes, in^3
V_{IT} = volume of M5 in the instrument tube, in^3

Inputting the values calculated below yields a total M5 volume:

$$V_{M5,rods} = 18029.6\ cm^3$$

6.5.7.1. Fuel Rods

Equation 6.5.7.1

$$V_{FR} = \left(\pi * \frac{D_{FR,o}^2 - D_{FR,i}^2}{4}\right) * N_{FR} * L_{FR}$$

Where

$D_{FR,o}$ = fuel rod outer diameter, inches
$D_{FR,i}$ = fuel rod inner diameter, inches
N_{FR} = number of fuel rods
L_{FR} = length of fuel rod, inches

Inputting the above values yields a total fuel rod M5 volume:

$$V_{FR} = 1025.5\ in^3$$

6.5.7.2. Guide Tubes

Equation 6.5.7.2

$$V_{GT} = \left(\pi * \frac{D_{GT,o}^2 - D_{GT,i}^2}{4}\right) * N_{GT} * L_{GT}$$

Where

$D_{GT,o}$ = guide rod outer diameter, inches
$D_{GT,i}$ = guide tube rod inner diameter, inches
N_{GT} = number of guide tubes
L_{GT} = length of guide tube, inches

Inputting the above values yields a total guide tube M5 volume:

$$V_{GT} = 64.6\ in^3$$

6.5.7.3. Instrument Tube

Equation 6.5.7.3

$$V_{IT} = \left(\pi * \frac{D_{IT,o}^2 - D_{IT,i}^2}{4}\right) * N_{IT} * L_{IT}$$

Where
$D_{IT,o}$ = instrument tube outer diameter, inches
$D_{IT,i}$ = instrument inner diameter, inches
N_{IT} = number of instrument tubes
L_{IT} = length of instrument tube, inches

Inputting the above values yields an instrument tube M5 volume:

$$V_{IT} = 10.1 \text{ in}^3$$

6.5.8. Mass of HTP Spacer Grids

The total mass of the M5 spacer grids can be found by multiplying the known mass of an individual grid by the number of grids per bundle.

Equation 6.5.8

$$m_{M5,HTP} = N_{HTP} * m_{HTP\ grid}$$

Where
$m_{M5,HTP}$ = total mass of HTP spacer grids, grams
N_{HTP} = number of HTP spacer grids per bundle
$M_{HTP\ grid}$ = mass of a single HTP spacer grid, grams

Inputting the above values yields a total HTP spacer grid mass:

$$m_{M5,HTP} = 7620.9 \text{ g}$$

6.5.9. Mass of End Fittings

The total mass of the upper and lower end fittings can be found by summing the individual masses.

Equation 6.5.9

$$m_{SS} = (m_{UEF} + m_{LEF}) * \left(\frac{453.592 \text{ gram}}{1 \text{ lb}}\right)$$

Where
m_{SS} = total mass of the end fittings, grams
m_{UEF} = mass of the upper end fitting, lbs
m_{LEF} = mass of the lower end fitting, lbs

Inputting the above values yields the total mass of the end fittings:

$$m_{SS} = 16692.2 \text{ g}$$

6.5.10. Decay Heat Generation Rate

The required decay heat is determined using the thermal capacity of materials.

Equation 6.5.10-1

$$\dot{q} = \left(m * c_p * \frac{\Delta T}{t}\right) * \left(0.000278 \frac{W}{J/hr}\right)$$

Where:

$\dot{q}$ = heat generation, W
m = ρ*V = mass of material, g
c_p = specific heat, J/g-K
ΔT = temperature rise, K
t = heat-up time, hr
ρ = density, g/cm^3
V = volume, cm^3

For this analysis, there are four materials that are considered: uranium dioxide, M5, CF3 stainless steel, and Inconel Alloy 718. The fuels pellets are uranium dioxide. The fuel rod cladding, guide tubes, instrument tube, and HTP spacer grids are M5. The upper and lower end fittings are CF3 stainless steel. The lower HMP grid is Inconel Alloy 718. Under adiabatic conditions, the materials are modeled as heating up at the same rate consistent with Assumption 3 in Reference 3.22; therefore, ΔT/t will be the same for all materials. This is justified because there is direct contact, thus conductive heat transfer, between the fuel rods and the lower end fitting and additional conductive heat transfer through the welded cage structure to both end fittings. Separating out the materials and substituting density and volume for M5 rods results in the following equation:

Equation 6.5.10-2

$$\dot{q} = \left(\frac{\Delta T}{t} * \left(m_{UO_2} * c_{p,UO_2} + \left(\rho_{M5} * V_{M5,rods} + m_{M5,HTP}\right) * c_{p,M5} + m_{SS} * c_{p,SS} + m_{HMP} * c_{p,HMP}\right)\right) * \left(0.000278 \frac{W}{J/hr}\right)$$

Where:

X_{UO2} signifies the property is for uranium dioxide
X_{M5} signifies the property is for M5
X_{SS} signifies the property is for CF3 stainless steel
X_{HMP} signifies the property is for Inconel Alloy 718

Inputting the values above yields a required decay heat:

$$\dot{q} = 5270\ \mathrm{W}$$

6.5.11. Minimum Decay Time

The decay heat determined using Equation 6.5.10-2 was input into the equation determined in Attachment 1 shown below.

$$Y\ (\text{days}) = 1.01 * [-9.22479E{-}09*x^3 + 1.59163E{-}04*x^2 - 1.01332E{+}00*x + 2.75333E{+}03]$$

Where:

Y = Decay Time, days
x = Decay Heat, W

Inputting the decay heat from section 6.5.10 yields the required decay time:

$$Y = 488\ \text{days}$$

7. Appendices

    7.1. Purpose #1 Excel Calculations and Results

    7.2. Purpose #2 Excel Calculations and Results

    7.3. Purpose #3 Excel Calculations and Results

    7.4. Purpose #5 Excel Calculations and Results

    7.5. Specific Heats of Inconel 706 and Inconel 718 Alloys

Appendix 7.1 Purpose #1 Excel Calculations and Results

Time after Shutdown (days)	SFP Decay Heat (MW_{th})	Time after Shutdown (days)	SFP Decay Heat (MW_{th})	Time after Shutdown (days)	SFP Decay Heat (MW_{th})
14	6.38	504	1.32	994	0.87
28	4.89	518	1.30	1008	0.87
42	4.16	532	1.28	1022	0.86
56	3.70	546	1.26	1036	0.85
70	3.38	560	1.25	1050	0.84
84	3.14	574	1.23	1064	0.84
98	2.94	588	1.21	1078	0.83
112	2.78	602	1.20	1092	0.82
126	2.64	616	1.18	1106	0.82
140	2.51	630	1.16	1120	0.81
154	2.41	644	1.15	1134	0.80
168	2.31	658	1.13	1148	0.80
182	2.22	672	1.12	1162	0.79
196	2.14	686	1.11	1176	0.78
210	2.07	700	1.09	1190	0.78
224	2.00	714	1.08	1204	0.77
238	1.94	728	1.07	1218	0.77
252	1.89	742	1.05	1232	0.76
266	1.84	756	1.04	1246	0.75
280	1.79	770	1.03	1260	0.75
294	1.75	784	1.02	1274	0.74
308	1.70	798	1.01	1288	0.74
322	1.67	812	1.00	1302	0.73
336	1.63	826	0.98	1316	0.73
350	1.60	840	0.97	1330	0.72
364	1.57	854	0.96	1344	0.72
378	1.54	868	0.95	1358	0.71
392	1.51	882	0.94	1372	0.71
406	1.48	896	0.94	1386	0.71
420	1.46	910	0.93	1400	0.70
434	1.43	924	0.92	1414	0.70
448	1.41	938	0.91	1428	0.69
462	1.39	952	0.90	1442	0.69
476	1.37	966	0.89	1456	0.69
490	1.34	980	0.88		

Appendix 7.2 Purpose #2 Excel Calculations and Results

	Design Maximum (both SFP trains)	Design Maximum (only one SFP train)	Maximum "expected" pool temperature
SFP Heat generation rate (MW)	6.38	6.38	6.38
SFP Heat generation rate (BTU/HR)	21768560	21768560	21768560
# of cooler trains in service	2	1	2
Cooler flow rate (GPM)	1000	1000	1000
Cooler flow rate (LBM/HR)	499800	499800	499800
NSCCW Inlet temperature (F)	95	95	80
HX Effectiveness	0.514	0.514	0.514
Pool water temperature (F)	137.4	179.7	122.4

	Design Maximum (both SFP trains)	Design Maximum (only one SFP train)	Maximum "expected" pool temperature
SFP Heat generation rate MW	1.57	1.57	1.57
SFP Heat generation rate BTU/HR	5356840	5356840	5356840
# of cooler trains in service	2	1	2
Cooler flow rate (GPM)	1000	1000	1000
Cooler flow rate (LBM/HR)	499800	499800	499800
NSCCW Inlet temperature (F)	95	95	80
HX Effectiveness	0.514	0.514	0.514
Pool water temperature (F)	105.4	115.9	90.4

Appendix 7.3 Purpose #3 Excel Calculations and Results

Time after Shutdown (days)	SFP Decay Heat (MWth)	TTB (hrs)	TTAF (days)
14	6.38	13.71	7.55
28	4.89	17.87	9.84
42	4.16	21.04	11.59
56	3.70	23.65	13.03
70	3.38	25.88	14.26
84	3.14	27.87	15.35
98	2.94	29.71	16.37
112	2.78	31.47	17.33
126	2.64	33.15	18.26
140	2.51	34.77	19.16
154	2.41	36.35	20.03
168	2.31	37.89	20.87
182	2.22	39.39	21.70
196	2.14	40.85	22.50
210	2.07	42.27	23.28
224	2.00	43.64	24.04
238	1.94	45.00	24.79
252	1.89	46.33	25.52
266	1.84	47.62	26.23
280	1.79	48.88	26.92
294	1.75	50.10	27.60
308	1.70	51.30	28.26
322	1.67	52.46	28.90
336	1.63	53.60	29.52
350	1.60	54.71	30.14
364	1.57	55.80	30.74
378	1.54	56.87	31.33
392	1.51	57.92	31.91
406	1.48	58.96	32.48
420	1.46	59.98	33.04
434	1.43	61.00	33.60
448	1.41	62.00	34.15
462	1.39	63.00	34.70
476	1.37	64.04	35.27
490	1.34	65.07	35.85

Time after Shutdown (days)	SFP Decay Heat (MWth)	TTB (hrs)	TTAF (days)
504	1.32	66.10	36.41
518	1.30	67.13	36.98
532	1.28	68.14	37.54
546	1.26	69.15	38.09
560	1.25	70.16	38.65
574	1.23	71.16	39.20
588	1.21	72.15	39.74
602	1.20	73.14	40.29
616	1.18	74.12	40.83
630	1.16	75.10	41.37
644	1.15	76.08	41.91
658	1.13	77.05	42.44
672	1.12	78.01	42.97
686	1.11	78.98	43.50
700	1.09	79.95	44.04
714	1.08	80.95	44.59
728	1.07	81.95	45.14
742	1.05	82.94	45.69
756	1.04	83.92	46.23
770	1.03	84.90	46.77
784	1.02	85.88	47.31
798	1.01	86.85	47.84
812	1.00	87.82	48.37
826	0.98	88.78	48.90
840	0.97	89.73	49.43
854	0.96	90.68	49.95
868	0.95	91.63	50.47
882	0.94	92.56	50.99
896	0.94	93.49	51.50
910	0.93	94.41	52.01
924	0.92	95.32	52.51
938	0.91	96.27	53.03
952	0.90	97.22	53.56
966	0.89	98.17	54.07
980	0.88	99.10	54.59

Time after Shutdown (days)	SFP Decay Heat (MWth)	TTB (hrs)	TTAF (days)
994	0.87	100.03	55.10
1008	0.87	100.96	55.61
1022	0.86	101.87	56.12
1036	0.85	102.78	56.62
1050	0.84	103.68	57.11
1064	0.84	104.58	57.61
1078	0.83	105.46	58.09
1092	0.82	106.34	58.58
1106	0.82	107.20	59.05
1120	0.81	108.06	59.53
1134	0.80	108.91	60.00
1148	0.80	109.76	60.46
1162	0.79	110.61	60.93
1176	0.78	111.51	61.42
1190	0.78	112.39	61.91
1204	0.77	113.27	62.40
1218	0.77	114.14	62.87
1232	0.76	115.00	63.35
1246	0.75	115.85	63.82
1260	0.75	116.69	64.28
1274	0.74	117.53	64.74
1288	0.74	118.35	65.19
1302	0.73	119.17	65.64
1316	0.73	119.97	66.09
1330	0.72	120.76	66.52
1344	0.72	121.54	66.95
1358	0.71	122.32	67.38
1372	0.71	123.08	67.80
1386	0.71	123.84	68.21
1400	0.70	124.58	68.63
1414	0.70	125.32	69.03
1428	0.69	126.05	69.43
1442	0.69	126.77	69.83
1456	0.69	127.48	70.22

Appendix 7.4 Purpose #5 Excel Calculations and Results

Initial Temperature	110	F
	43	C
	316	K
Final Temperature	1652	F
	900	C
	1173	K
Delta Temperature	857	K
Specific Heat Temperature	881	F
	472	C
	745	K
Mass of UO2		
Mass of U	485.75	kgU
Atomic Mass of U	238.029	g/gmole
Atomic Mass of O	15.999	g/gmole
Mass of O	65.30	kgO2
Mass of UO2	551.05	kgUO2
Mass of UO2	5.51E+05	gUO2
Molar Mass UO2	270.027	g/gmole
Heat Capacity (UO2)		
θ	535.85	K
E_D	157770.7	J/mol
K_1	80.1314	J/mol-K
K_2	3.2845E-03	J/mol-K
K_3	2.362183E+07	J/mol
R	8.314	J/mol-K
Temperature	745	K
Molar Mass	270.027	g/mol
Specific Heat	81.7	J/mol-K
Heat Capacity	0.302	J/g-K

Heat Capacity of M5

Heat Capacity	0.356	J/g-K

Heat Capacity of Alloy 718

Heat Capacity	0.5830	J/g-K

Heat Capacity of CF3 stainless steel

Heat Capacity	0.570	J/g-K

Volume of M5

Fuel Rods	Value	Units
Number of Rods	208	
Inner Diameter	0.380	in
Outer Diameter	0.430	in
Fuel Rod Length	155.00	in
Volume	1025.5	in^3

Guide Tubes	Value	Units
Number of Tubes	16	
Inner Diameter	0.498	in
Outer Diameter	0.530	in
GT Assembly Length	156.249	in
Volume	64.6	in^3

Instrument Tube	Value	Units
Number of Tubes	1	
Inner Diameter	0.400	in
Outer Diameter	0.493	in
Length	155.285	in
Volume	10.1	in^3

Volume of M5	1100.2	in^3
Volume of M5	18029.6	cm^3

M5 Spacer Grids		
Number of Grids	7	
Mass of each grid	1088.7	g
Total Mass	7620.9	g
HMP Spacer Grid		
Number of Grids	1	
Mass of Each Grid	1197.7	g
Total Mass	1197.7	g
Upper End Fitting		
Mass	21.9	lbs
Mass	9933.7	g
Lower End Fitting		
Mass	14.9	lbs
Mass	6758.52	g
Total Mass of End Fittings		
Mass of UEF	9933.7	g
Mass of LEF	6758.5	g
Total Mass	16692.2	g
Decay Heat		
Delta Temperature	856.7	K
Time	10	hrs
Mass of UO2	5.51E+05	gUO2
Heat Capacity of UO2	0.302	J/g-K
Density of M5	6.50	g/cm^3
Volume of M5	18029.571	cm^3
Mass of M5 Spacer Grids	7620.9	g
Heat Capacity of M5	0.356	J/g-K
Mass of End Fittings	16692.186	g
Heat Capacity of CF3	0.570	J/g-K
Mass of HMP Grid	1197.7	g
Heat Capacity of Alloy 718	0.583	J/g-K
Decay Heat	5270	W

Decay Time

Constants	-9.22479E-09	x^3
	1.59163E-04	x^2
	-1.01332E+00	x
	2.75333E+03	
Decay Time	**488**	**days**

Appendix 7.5 Specific Heats of Inconel 706 and Inconel 718 Alloys

TABLE 5

SPECIFIC HEATS OF INCONEL 706 AND INCONEL 718 ALLOYS

Temperature (°F)	Specific Heat (Btu lb^{-1} F^{-1})	
	Inconel 706	Inconel 718
70	0.111	0.107
100	0.112	0.109
150	0.114	0.112
200	0.116	0.114
250	0.117	0.116
300	0.119	0.119
350	0.120	0.121
400	0.121	0.123
450	0.122	0.125
500	0.123	0.126
550	0.123	0.128
600	0.124	0.130
650	0.125	0.131
700	0.125	0.133
750	0.126	0.135
800	0.128	0.137
850	0.132	0.138
900	0.136	0.140
950	0.141	0.142
1000	0.144	0.143
1050	0.148	0.145
1100	0.149	0.149
1150	0.150	0.160
1200	0.151	0.166
1250	0.151	0.169
1300	0.150	0.169
1350	0.149	0.169
1400	0.148	0.168
1450	0.147	0.167
1500	0.146	0.166

ATTACHMENT 1
ORIGEN2 Decay Heat Calculation for Fuel Assembly with Maximum Decay Heat Load

1.0 PURPOSE

To calculate the decay heat from the fuel assembly determined to produce the highest decay heat load of all fuel assemblies in the TMI Spent Fuel Pool (SFP) between 1 and 3 years after shut down of TMI-1 Cycle 22. The resultant decay heat load is to be used as input to the Beyond Design Basis zirconium fire event for a drained SFP. Best estimate decay heat loads will be calculated using the ORIGEN2 code based on reactor rated power of 2568 MWt and a Cycle 22 length of 720 EFPD, which is the maximum licensed length for Cycle 22.

2.0 SUMMARY OF RESULTS

A best estimate decay heat load for the fuel assembly with the maximum heat load in the TMI SFP following TMI-1 Cycle 22 shutdown has been calculated using the ORIGEN2 computer code. Per Section 7.3 of this calculation, the decay heat load from this Batch 22A fuel assembly as a function of decay time (in days) after Cycle 22 shutdown is shown in the table below. This decay heat load is appropriate to be used as input to the Beyond Design Basis zirconium fire event for a drained SFP.

Maximum Fuel Assembly Decay Heat (Watts) for Various Decay Times After TMI-1 Cycle 22 Shutdown

	Decay Time (Days)				
	365.0D	**548.0D**	**730.0D**	**913.0D**	**1095.0D**
Decay Heat (W)	**6.44E+03**	**4.70E+03**	**3.65E+03**	**2.94E+03**	**2.44E+03**

The following polynomial fit represents this data for decay heat (x) between 2.44 and 6.44 kW. The polynomial fit is in good agreement with the individual ORIGEN2 data points. A 1% multiplier is applied to the Excel fit for conservatism.

$$\mathbf{Y\ (days) = 1.01 * [-9.22479E\text{-}09*x^3 + 1.59163E\text{-}04*x^2 - 1.01332E{+}00*x + 2.75333E{+}03]}$$

3.0 REFERENCES

3.1 RSIC Code Package CCC-371, "ORIGEN 2.1, Isotope Generation and Depletion Code Matrix Exponential Method," May 1999.

3.2 ORNL/TM-11018, "Standard- and Extended-Burnup PWR and BWR Reactor Models for the ORIGEN2 Computer Code," S. Ludwig, J. Renier, December 1989.

3.3 AREVA Calculation FS1-0030101, Rev. 1, "TMI-1 Cycle 22 FFCD," 2/03/2017.

3.4 Exelon Nuclear Fuels TMI Fuel Database, "C22database.accdb," 1/03/2018.

3.5 Exelon TODI NF173266, Rev. 0, "TMI End-of-Cycle 21 Data and FIDMS Files," 9/18/2017.

3.6 Exelon DTSQA document, "ORIGEN Version 2.1, EX0004724 Release Notes," October 2014.

3.7 AREVA Doc. FS1-0029697, Rev. 1, "Task 76 Inputs, Mechanical Design Information for Safety Analysis, and Regulatory Compliance for TMI1-22," 12/19/2016.

3.8 AREVA Doc. BAW-10227P-A, Rev. 1, "Evaluation of Advanced Cladding and Structural Material (M5) in PWR Reactor Fuel," June 2003.

3.9 ASTM Specification B637 for Alloy 718 (see Appendix B)

4.0 ASSUMPTIONS

4.1 Fuel batch burnups are taken from NEMO calculations documented in Tables 38-39 and Figure 34 of Reference 3.3. NEMO is the design code for TMI-1 core simulations and calculated burnups are representative of actual burnups measured by the core monitoring system. Burnups prior to Cycle 21 are based on core follow calculations that are depleted to actual cycle lengths. Cycle 21 burnups are based on a design depletion to 647 EFPD; differences from the actual Cycle 21 length of 648.8 EFPD (Reference 3.5) are negligible with respect to this decay heat calculation. Cycle 22 burnups are based on 720 EFPD, which is the maximum licensed cycle length for Cycle 22. Higher fuel assembly burnups are conservative for decay heat calculations.

4.2 Reference 3.3 lists fuel batch enrichments based on the nominal base enrichment of the batch. Since lower enrichments yield higher decay heat results, actual enrichments based on NRC 741 forms are used in this calculation. This includes segments of the fuel assemblies that have lower than nominal enrichments (i.e., Gad rods and axial blankets). The actual enrichments were obtained from Reference 3.4.

4.3 For fuel burned in more than one cycle, ORIGEN2 runs ignored refueling outages. This has no impact on short-lived isotopes which reach equilibrium concentrations shortly after cycle startup and has a conservative, albeit minimal, impact on long-lived isotopes which continually increase in concentration as a function of exposure; ignoring intermediate decay periods will increase the final concentrations.

4.4 The ORIGEN2 cross-section library, PWRUE.LIB, is used in this calculation as this is most representative of TMI-1 two-year cycles. The library is based on an "extended cycle" reactor model where fuel achieves 50 GWd/mtU burnup in three cycles.

4.5 The decay heat loads calculated herein using ORIGEN2 are to be used as input to a Beyond Design Basis analysis. Therefore, ORIGEN2 results can be used directly as best estimate values, without applying any additional uncertainty factor.

4.6 Only fuel operating in Cycle 22 needs to be evaluated for maximum decay heat loads following Cycle 22 shut down. All other fuel in the SFP will have decayed for ≥ 2 years with a significant reduction in decay heat.

4.7 Only structural materials in the active fuel region receive sufficient flux to generate activation products with any appreciable decay heat load with respect to this calculation.

5.0 DESIGN INPUT

5.1 Batch-average burnups for fuel loaded in Cycle 22 are taken from Tables 38-39 and Figure 34 (for Batch 24A only) of Reference 3.3. The tables also contain the effective full power days (EFPD) of operation for all TMI cycles.

5.2 Actual batch enrichments for fuel loaded in Cycle 22 are taken from Reference 3.4. For each batch, the lowest as-built enrichment of all fuel assemblies in the batch is used to represent the batch.

5.3 Uranium loading for each fuel batch in Cycle 22 is taken from Table 14 of Reference 3.3.

5.4 Fuel assembly structural materials and dimensional data are taken from Reference 3.7.

5.5 Material specifications for M5 and Alloy 718 are taken from References 3.8 and 3.9, respectively. The Alloy 718 specification is copied to Appendix B; the maximum Cobalt content is assumed and elements less than 0.50% are ignored.

6.0 OVERALL APPROACH AND METHODOLOGY

A fuel assembly's decay heat is a function of the fuel's power level (which affects the equilibrium concentration of both long and short-lived isotopes) and the exposure of the fuel (which affects long-lived isotopes that continue to accumulate). Maximizing these parameters will result in a higher decay heat load. Fuel assembly U235 enrichment has a second order effect on decay heat, with lower enrichments resulting in higher decay heat loads.

The fuel batches in Cycle 22 will be examined to determine which batches can be eliminated based on comparison of the three parameters of interest. The remaining batches will be evaluated using ORIGEN2 to calculate decay heat loads for various decay times after Cycle 22 shutdown. The batch with the highest decay heat loads after 1 year of decay (expected minimum time for

the zirc fire analysis) will be selected to provide the maximum fuel assembly decay heat loads to the downstream zirc fire analysis.

Since the purpose of this calculation is to identify the fuel assembly with the highest heat load after 1 year of decay, batches that reside in only one eighth-core symmetric location (typically batches containing only 1, 4, or 8 fuel assemblies) can be treated as essentially the same. Larger batches will be examined to determine if one of the eighth-core symmetric locations for that batch has a higher burnup (and therefore specific power) than the others.

6.1 Computer Codes

The DOS-based ORIGEN2 code, Version 2.1 (DTSQA application EX0004724) is used to calculate decay heat. The code was run on the **nfw-ksq-01** Windows 2008 virtual server where it was approved for release in Reference 3.6. DOSBox was used to run ORIGEN2 on the Windows server with D:\dosprogs mounted as the D: drive.

The dto.bat batch file used to execute ORIGEN2 for this calculation provides the paths and filenames of the executable program and libraries that were called. A typical example of this batch file is listed in Appendix A; only the batch ID was changed in the input/output filenames for the various batches analyzed.

All cases were run in directory C:\Users\u001rpj\ORIGEN21\TMI\DTO.

7.0 CALCULATIONS

7.1 Comparison of Fuel Batches Operating in TMI-1 Cycle 22

The fuel batches operating in TMI-1 Cycle 22 are listed in Table 14 of Reference 3.3, which is copied below.

Table 14: TMI-1 Cycle 22 Fuel Inventory

Batch ID	Base Enrichment (%)	Gadolinia Pins	Pin Layout	FA Design	Number of FA's	Design Loading (kgU)
19E3	4.95	12 x 2.0	Figure 12	Mark-B-HTP	8	489.18
20A3	1.40	None	N/A	Mark-B-HTP-1	1	489.83
22A1	4.57	12 x 3.0 8 x 8.0	Figure 13	Mark-B-HTP-1	4	485.75
22B2	4.76	12 x 2.0 8 x 8.0	Figure 14	Mark-B-HTP-1	8	486.07
22D	4.90	12 x 2.0	Figure 15	Mark-B-HTP-1	8	487.77
22E2	4.90	16 x 2.0	Figure 16	Mark-B-HTP-1	8	487.56
23A2	4.10	None	Figure 17	Mark-B-HTP-1	4	488.43
23B	4.10	12 x 3.0 8 x 8.0	Figure 18	Mark-B-HTP-1	12	485.75
23C	4.30	None	Figure 17	Mark-B-HTP-1	8	488.43
23D	4.30	16 x 2.0 4 x 6.0	Figure 19	Mark-B-HTP-1	4	486.91
23E	4.50	None	Figure 17	Mark-B-HTP-1	8	488.43
23F	4.50	8 x 2.0	Figure 20	Mark-B-HTP-1	16	487.99
23G	4.50	16 x 2.0	Figure 21	Mark-B-HTP-1	8	487.56
23H	4.50	12 x 2.0 8 x 8.0	Figure 22	Mark-B-HTP-1	8	486.07
24A	4.36	12 x 3.0 8 x 8.0	Figure 23	Mark-B-HTP-1	16	485.75
24B	4.75	None	Figure 17	Mark-B-HTP-1	4	488.43
24C	4.75	8 x 2.0	Figure 24	Mark-B-HTP-1	12	487.99
24D	4.75	16 x 2.0	Figure 25	Mark-B-HTP-1	16	487.56
24E	4.75	8 x 3.0 8 x 8.0	Figure 26	Mark-B-HTP-1	8	486.07
24F	4.88	8 x 2.0	Figure 27	Mark-B-HTP-1	8	487.99
24G	4.88	8 x 3.0	Figure 28	Mark-B-HTP-1	8	487.78

The batch average burnups for these batches are listed in Tables 38 and 39 from Reference 3.3, and are copied below. Also, EOC-22 quarter-core assembly burnups from Figure 34 of Ref. 3.3 are copied below. Inspection of these tables and figure allows the following batches in Cycle 22 to be eliminated from consideration:

- **Batch 20A3 – total burnup and final cycle power level considerably lower than other batches**
- **Batch 22E2 – total burnup and enrichment bounded by Batch 22A1**
  - From Figure 34, Batch 22E2 consists of quarter-core locations, O-13 & H-15. Location O-13 has a slightly higher burnup (51,783 MWd/mtU) than Batch 22A1 (51,255 MWd/mtU). However, the enrichment difference (0.33 w/o) is considered more significant than the small burnup difference, especially after 1 year of decay.
- **Batches 23B thru 23H – total burnups less than or equivalent to Batch 23A2, and enrichment bounded by Batch 23A2**
  - From Figure 34, Batch 23B consists of two eighth-core locations, H-12 & N-14. Burnups from both locations are bounded by Batch 23A2.
  - Batch 23E consists of two quarter-core locations, H-10 & H-11. Location H-10 has a slightly higher burnup (49,681 MWd/mtU) than Batch 23A2 (48,504 MWd/mtU). However, the enrichment difference (0.40 w/o) is considered more significant than the small burnup difference, especially after 1 year of decay.
  - Batch 23F consists of two quarter-core and one eighth-core locations, M-11, N-12 & M-12. Burnups from all locations are bounded by Batch 23A2.
- **Batches 24B thru 24G – total burnups less than or equivalent to Batch 24A, and enrichment bounded by Batch 24A**
  - From Figure 34, Batch 24A consists of two eighth-core locations, K-12 & L-11. **The higher burnup from L-11 (28,667 MWd/mtU) will be used for Batch 24A.**
  - Batch 24C consists of one quarter-core and one eighth-core locations, H-09 & K-10. Burnups from both locations are bounded by Batch 24A.
  - Batch 24D consists of two eighth-core locations, K-14 & L-14. Burnups from both locations are bounded by Batch 24A.
  - Batch 24F has a burnup equivalent to Batch 24A, but has a uranium loading ~0.5% higher. Since fuel assembly specific powers are determined by multiplying burnup by uranium loading, Bach 24F would have a specific power ~0.5% higher than Batch 24A. However, the enrichment difference (0.52 w/o) is considered more significant than the small power difference, especially after 1 year of decay.

Table 38: TMI-1 Fuel Burnup (MWd/mtU) Distribution by Batch ID (17-20 Only) and by Cycle

Batch ID	Base Enrichment (%)	Number of FA by Batch	Rated Thermal Power = 2568 MW_{th}								EFPD
			679.16	694.90	700.30[7]	628.31	685.24	690.35	647[6]	720[6]	
			Cycle 15	Cycle 16	Cycle 17	Cycle 18	Cycle 19	Cycle 20	Cycle 21	Cycle 22	Batch-Average
17A[3]	4.45	28	26707	21323							48030
17B[3]	4.78	8	23197	24155	6303						53655
17C1[3]	4.78	8	26876	22787							49663
17C2[3]	4.78	8	26056	9407	12701						48164
17D1[3]	4.90	1	23179	23271							46450
17D2[3]	4.90	15	23405	19993	10925						54323
18A[3]	4.45	4		26777	17802	12985					57564
18B[3]	4.45	16		27260	22236						49496
18C[3]	4.60	12		27142	21999						49141
18D1[3]	4.60	8		27804	21564						49368
18D2[3]	4.60	8		25837	9196	18546					53579
18E1[3]	4.80	12		23275	22796						46071
18E2[3]	4.80	4		23819	12028	18198					54045
18F[3]	4.80	8		23045	24305						47350
19A[3]	4.60	16			27927	19538					47465
19B[3]	4.70	4			27524	19099					46623
19C1[3]	4.70	8			27336	20456					47792
19C2[3]	4.70	8			26105	8921	12206				47232
19D[3]	4.80	12			28056	14701					42757
19E1[3]	4.95	4			24459	11262					35721
19E2[3]	4.95	12			23860	21022	8025				52907
19E3[3]	4.95	8			23522	22570				6562	52654
20A1	1.40	18				6549	10075				16624
20A2	1.40	1				6323	6141		15749		28213
20A3	1.40	1				6323	6141			17578	30042
20B	2.50	1				17426	19258	15683			52367
20C1[3]	4.60	4				24462		23056			47518
20C2[3]	4.60	4				24404	22101	7349			53854
20D[3]	4.60	4				24178	22811				46989
20E1[3]	4.60	8				24943	22933				47876
20E2[3]	4.60	8				24178	8805	12600			45583
20F[3]	4.70	8				24385	23583				47968
20G[3]	4.70	8				24389	22684				47073
20H1[3]	4.95	4				22246	23538				45784
20H2[3]	4.95	4				22281	22019	8718			53018
20I[3]	4.95	8				22476	23262	7851			53589
20J[3]	4.95	8				21869	24576	6424			52869

Table 39: TMI-1 Fuel Burnup (MWd/mtU) Distribution by Batch ID (21-24 Only) and by Cycle

Batch ID	Base Enrichment (%)	Number of FA by Batch	Rated Thermal Power = 2568 MW_{th}						
			685.24	690.35	647[6]	720[6]	---	---	EFPD
			Cycle 19	Cycle 20	Cycle 21	Cycle 22	Cycle 23	Cycle 24	Batch-Average
21A[3]	4.10	4	26531	21650					48181
21B[3]	4.40	12	27083	21955					49038
21C1[3]	4.40	4	26579		21179				47758
21C2[3]	4.40	12	26943	21976					48919
21C3[3]	4.40	8	26528	21749	5434				53711
21D[3]	4.82	8	25790	8980	11941				46711
21E1[3]	4.90	4	23015	23614					46629
21E2[3]	4.90	4	23022	19643	7366				50031
21F1[3]	4.95	4	24089	22407					46496
21F2[3]	4.95	12	22822	22815	7593				53230
22A1[3]	4.57	4		27343		23912			51255
22A2[3]	4.57	12		27299	21254				48553
22B1[3]	4.76	12		27334	21305				48639
22B2[3]	4.76	8		25806	8421	13021			47248
22C[3]	4.76	12		27555	21510				49065
22D[3]	4.90	8		24541	21524	7929			53994
22E1[3]	4.90	8		23090	22355				45445
22E2[3]	4.90	8		21991	19839	8520			50350
23A1	4.10	4			24244				24244
23A2	4.10	4			24241	24263			48504
23B[3]	4.10	12			25063	13031			38094
23C	4.30	8			24528	24138			48666
23D[3]	4.30	4			25789	18251			44040
23E	4.50	8			25571	23385			48956
23F[3]	4.50	16			20958	23659			44617
23G[3]	4.50	8			22388	24549			46937
23H[3]	4.50	8			24083	23750			47833
24A[3]	4.36	16				28400			28400
24B	4.75	4				27711			27711
24C[3]	4.75	12				28429			28429
24D[3]	4.75	16				24279			24279
24E[3]	4.75	8				26986			26986
24F[3]	4.88	8				28473			28473
24G[3]	4.88	8				22845			22845

Figure 34: TMI-1 Cycle 22 Assembly-Average Burnup Distribution (GWd/mtU)

	8	9	10	11	12	13	14	15
H	20A3 30.042 12.464 17.578	24C 28.466 0.000 28.466	23E 49.681 25.578 24.103	23E 48.232 25.565 22.667	23B 46.860 24.735 22.125	24B 27.711 0.000 27.711	23D 44.040 25.789 18.251	22E2 48.918 40.910 8.008
K	24C 28.466 0.000 28.466	22A1 51.255 27.343 23.912	24C 28.408 0.000 28.408	23H 47.853 24.083 23.770	24A 28.250 0.000 28.250	23G 46.898 22.308 24.590	24D 24.874 0.000 24.874	22D 53.955 46.020 7.935
L	23E 49.681 25.578 24.103	24C 28.413 0.000 28.413	23A2 48.504 24.241 24.263	24A 28.667 0.000 28.667	23C 48.708 24.510 24.198	24F 28.494 0.000 28.494	24D 23.705 0.000 23.705	19E3 52.632 46.065 6.567
M	23E 48.232 25.565 22.667	23H 47.811 24.082 23.729	24A 28.516 0.000 28.516	23F 45.859 21.447 24.412	23F 44.537 20.474 24.063	24E 26.967 0.000 26.967	22B2 47.281 34.279 13.002	
N	23B 46.860 24.735 22.125	24A 28.166 0.000 28.166	23C 48.624 24.547 24.077	23F 44.480 20.502 23.978	23F 43.591 21.408 22.183	24G 22.782 0.000 22.782	23B 33.781 25.335 8.446	
O	24B 27.711 0.000 27.711	23G 46.977 22.468 24.509	24F 28.452 0.000 28.452	24E 27.005 0.000 27.005	24G 22.908 0.000 22.908	22E2 51.783 42.751 9.032		
P	23D 44.040 25.789 18.251	24D 24.846 0.000 24.846	24D 23.692 0.000 23.692	22B2 47.214 34.174 13.040	23B 33.640 25.121 8.519	**Batch ID** **720.0 EFPD** **0.0 EFPD** **Delta BU**		
R	22E2 48.918 40.910 8.008	22D 54.035 46.111 7.924	19E3 52.677 46.120 6.557					

7.2 Structural Materials for ORIGEN2 Input

Per assumption 4.7, only structural materials in the active fuel region will be modelled in ORIGEN2. This consists of fuel cladding, guide tubes, instrument tube, and spacer grids. Quantities, materials, and dimensions for these components are taken from Ref. 3.7, as are spacer grid weights. The length for fuel rods, guide tubes, and instrument tube are set equal to the active fuel length of 143 inches. The density for M5 material is taken from Ref 3.8, page A-3. These parameters are shown in the following table, along with calculated volumes and weights.

Description	Quantity	Material	Density (g/cc)	Dimensions	Volume (cc) per unit	Weight (g) per unit	Total weight (g)
HTP Grids	7	M5	---	---	---	1088.7	7620.9
HMP Grids	1	Alloy 718	---	---	---	1197.7	1197.7
Fuel Rods	208	M5	6.48	ID = 0.380" OD = 0.430" Length = 143.0"	74.5	483.0	100,466.3
Guide Tubes	16	M5	6.48	ID = 0.498" OD = 0.530" Length = 143.0"	60.5	392.3	6,277.2
Instrument Tube	1	M5	6.48	ID = 0.400" OD = 0.493" Length = 143.0"	152.8	990.5	990.5

In the table above, tube volumes are calculated using the following formula:

$$Volume\ (cc) = \left(\frac{\pi*(OD*2.54)^2}{4} - \frac{\pi*(ID*2.54)^2}{4}\right) * (Length * 2.54)$$

Tube weights are calculated using the formula:

Unit Weight (g) = Volume * Density

Total weights are calculated using the formula:

Total Weight (g) = Unit weight * Quantity

Based on the table above, the total weight of M5 and Alloy 718 in the active fuel region is:

115,354.85 g of M5

1197.7 g of Alloy 718

The following M5 alloy nominal chemistry is taken from Ref. 3.8, page A-3. Minor impurities are ignored.

M5	
98.875%	Zr
1.00%	Nb
0.125%	O

The following Alloy 718 nominal chemistry is taken from Ref. 3.9 (see Appendix B). Minor impurities are ignored.

Alloy 718	
52.50%	Ni
19.00%	Cr
5.125%	Nb
3.05%	Mo
0.90%	Ti
0.50%	Al
1.00%	Co
17.925%	Fe

Multiplying the total weights of M5 and Alloy 718 by their constituent elements, the following elemental weights are calculated.

M5 (g)		Alloy 718 (g)	
114057.1	Zr	628.8	Ni
1153.5	Nb	227.6	Cr
144.2	O	61.4	Nb
		36.5	Mo
		10.8	Ti
		6.0	Al
		12.0	Co
		214.7	Fe

Combining elemental weights yields the following totals for ORIGEN2 input:

Totals for ORIGEN2

Weight (g)		Atomic #	
144.19	O	8	Add to weight of Oxygen from UO2
5.99	Al	13	
10.78	Ti	22	
227.56	Cr	24	
214.69	Fe	26	
11.98	Co	27	
628.79	Ni	28	
114057.11	Zr	40	
1214.93	Nb	41	
36.53	Mo	42	

7.3 Calculation of Decay Heat Using ORIGEN2.1

The fuel batches operating in TMI-1 Cycle 22 that will be evaluated for decay heat loads are listed in the following table. The table also includes additional information required for simulating the fuels' irradiation using ORIGEN2:

- Since the purpose of this calculation is to determine the decay heat load for the highest heat load assembly, the basis of the calculation is 1 fuel assembly.
- As noted in Assumption 4.2, the actual enrichments from Reference 3.4 are used to minimize enrichments and maximize decay heat.
- Average burnups are taken from Tables 38 and 39 above, with the corresponding Cycle #'s and EFPD for the cycles listed. Per Section 7.1, Batch 24A burnup is taken from core location L-11 in Figure 34.
- The fuel assembly's specific power for each cycle is calculated by dividing the burnup by the EFPD, and multiplying by the MTU loading
- The MTU loading for a fuel assembly in each batch is taken from Table 14 above.
- The grams of U235, U238, and O from UO_2 for each batch are determined as follows:

  U235 wt (gms) = Batch loading * (Avg. Enr./100) * 10^6

  U238 wt (gms) = Batch loading * (1 - Avg. Enr./100) * 10^6

  Oxygen wt (gms) = (U235 wt + U238 wt)/238*2*15.9994
- The grams for structural materials are taken from Section 7.2

Batch	# of FA	Avg. Enr. (w/o U235)	Avg. Burnup per Cycle (MWd/mtU)	Cycle of Operation	Cycle EFPD	Power (MW)	Loading (MTU)	U235 wt. (gms)	U238 wt. (gms)	Oxygen wt. (gms)
19E3	1	4.689	23522 22570 6562 ***52654***	17 18 22	700 628 720	16.4 17.6 4.5	0.48918	22937.65	466242.35	65769.63
22A1	1	4.261	27343 23912 ***51255***	20 22	690 720	19.2 16.1	0.48575	20697.81	465052.19	65308.48
22B2	1	4.427	25806 8421 13021 ***47248***	20 21 22	690 647 720	18.2 6.3 8.8	0.48607	21518.32	464551.68	65351.50
22D	1	4.641	24541 21524 7929 ***53994***	20 21 22	690 647 720	17.3 16.2 5.4	0.48777	22637.41	465132.59	65580.06
23A2	1	3.945	24241 24263 ***48504***	21 22	647 720	18.3 16.5	0.48843	19268.56	469161.44	65668.80
24A	1	4.077	28667 ***28667***	22	720	19.3	0.48575	19804.03	465945.97	65308.48

The decay heat loads are calculated using ORIGEN2 and the batch-specific parameters from the table above. The input decks (identifiable by batch number) and a typical batch file, **dto.bat**, are listed in Appendix A.

The ORIGEN2 input deck is set up to deplete each fuel batch for the requisite number of cycles, and then to decay the isotopic mix after Cycle 22 shutdown. The ORIGEN2 decay heat results are shown in the following table, with decay times in days.

Decay Heat (Watts) for Various Decay Times after TMI-1 Cycle 22 Shutdown

Batch	T=0	30.0D	90.0D	180.0D	365.0D	548.0D	730.0D	913.0D	1095.0D
B19E	2.73E+05	1.04E+04	7.18E+03	5.43E+03	3.80E+03	2.99E+03	2.47E+03	2.10E+03	1.85E+03
B22A	9.63E+05	2.59E+04	1.55E+04	**1.05E+04**	**6.44E+03**	**4.70E+03**	**3.65E+03**	**2.94E+03**	**2.44E+03**
B22B	5.28E+05	1.53E+04	9.46E+03	6.54E+03	4.16E+03	3.11E+03	2.48E+03	2.06E+03	1.76E+03
B22D	3.26E+05	1.17E+04	7.95E+03	5.93E+03	4.09E+03	3.20E+03	2.63E+03	2.23E+03	1.95E+03
B23A	9.82E+05	2.61E+04	**1.56E+04**	**1.05E+04**	6.38E+03	4.63E+03	3.58E+03	2.86E+03	2.37E+03
B24A	**1.16E+06**	**2.62E+04**	1.42E+04	8.54E+03	4.53E+03	3.07E+03	2.26E+03	1.72E+03	1.37E+03
Max	1.16E+06	2.62E+04	1.56E+04	1.05E+04	6.44E+03	4.70E+03	3.65E+03	2.94E+03	2.44E+03

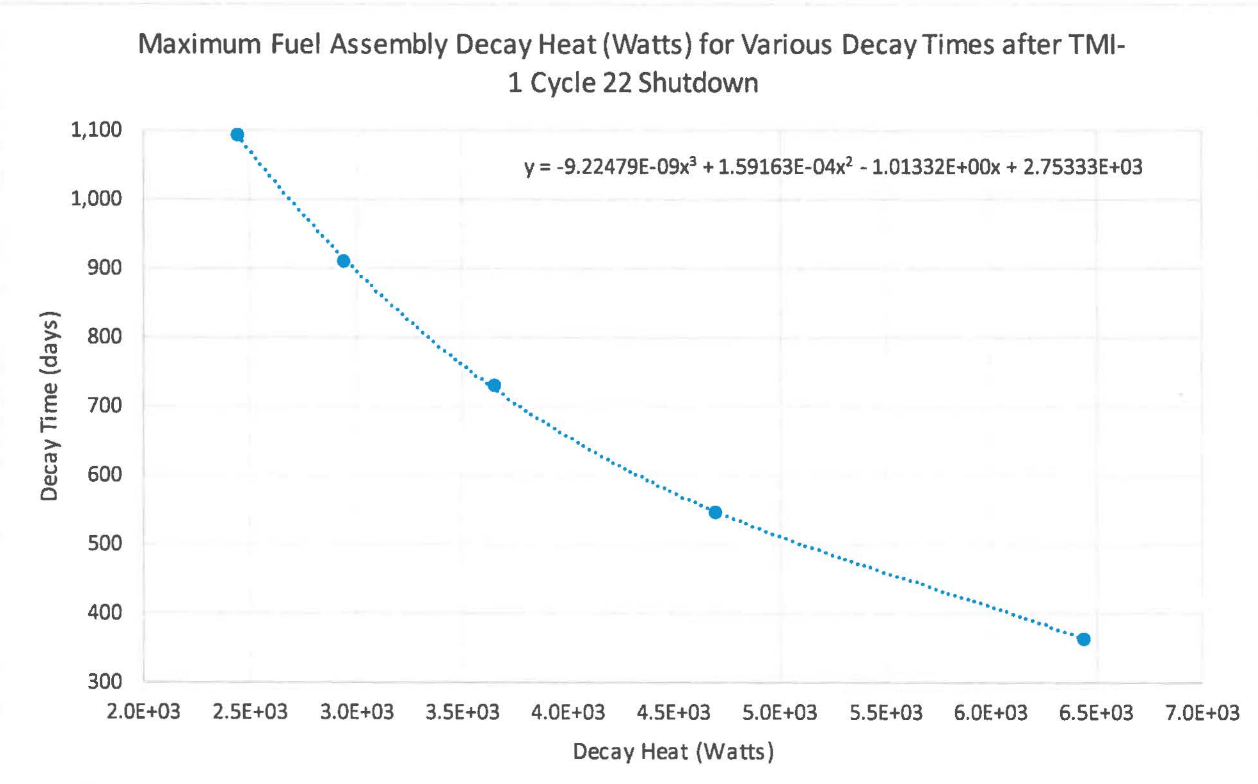
For decay times of 180 days and beyond, Batch 22A yields the highest decay heat loads, which will therefore be the basis for the downstream zirc fire analysis.

The Batch 22A results from 365 to 1095 days of decay were plotted in Excel and fit with a 3rd order polynomial trendline (see figure below).

ORIGEN2 Decay Heat (Watts)	Decay Time (days)	Polynomial Fit Decay Time (days)	Difference %
6.44E+03	365	364.9	0.0
4.70E+03	548	549.0	0.2
3.65E+03	730	727.0	-0.4
2.94E+03	913	916.4	0.4
2.44E+03	1095	1093.6	-0.1

The following polynomial fit represents this data for decay heat (x) between 2.44 and 6.44 kW. As seen in the table above, the polynomial fit is in good agreement with the individual ORIGEN2 data points. A 1% multiplier is applied to the Excel fit for conservatism.

Y (days) = 1.01 * [-9.22479E-09*x^3 + 1.59163E-04*x^2 - 1.01332E+00*x + 2.75333E+03]


Maximum Fuel Assembly Decay Heat (Watts) for Various Decay Times after TMI-1 Cycle 22 Shutdown
y = -9.22479E-09x³ + 1.59163E-04x² - 1.01332E+00x + 2.75333E+03
Decay Time (days)
1,100
1,000
900
800
700
600
500
400
300
2.0E+03
2.5E+03
3.0E+03
3.5E+03
4.0E+03
4.5E+03
5.0E+03
5.5E+03
6.0E+03
6.5E+03
7.0E+03
Decay Heat (Watts)

APPENDIX A

ORIGEN2 Input Decks and Typical Job Batch File

dto-b19e.inp

```
-1
-1
-1
BAS     Grams of Heavy Metal for 1 Mk-B-HTP Fuel Assembly
RDA     PLACE FUEL into vectors -1
LIP     0  0  0
LIB     0  1  2  3   604   605   606  9  3  0  1 39
PHO     0 0 0 10
RDA     READ FUEL COMPOSITION
INP     -1  1  -1 -1  1  1
RDA TIT     IRRADIATION OF TMI Batch 19E Fuel to EOC-22
MOV     -1  1  0  1.0
HED     1    CHARGE
RDA     BURN TO EOC-22
BUP
IRP      50.0     16.4    1   2   4 2
IRP     100.0     16.4    2   3   4 0
IRP     150.0     16.4    3   4   4 0
IRP     200.0     16.4    4   5   4 0
IRP     250.0     16.4    5   2   4 0
IRP     300.0     16.4    2   3   4 0
IRP     350.0     16.4    3   4   4 0
IRP     400.0     16.4    4   5   4 0
IRP     450.0     16.4    5   6   4 0
IRP     500.0     16.4    6   3   4 0
IRP     550.0     16.4    3   4   4 0
IRP     600.0     16.4    4   5   4 0
IRP     650.0     16.4    5   6   4 0
IRP     700.0     16.4    6   7   4 0
IRP     750.0     17.6    7   4   4 0
IRP     800.0     17.6    4   5   4 0
IRP     850.0     17.6    5   6   4 0
IRP     900.0     17.6    6   7   4 0
IRP     950.0     17.6    7   8   4 0
IRP    1000.0     17.6    8   5   4 0
IRP    1050.0     17.6    5   6   4 0
IRP    1100.0     17.6    6   7   4 0
IRP    1150.0     17.6    7   8   4 0
IRP    1200.0     17.6    8   9   4 0
IRP    1250.0     17.6    9   6   4 0
IRP    1300.0     17.6    6   7   4 0
IRP    1328.0     17.6    7   8   4 0
IRP    1378.0      4.5    8   9   4 0
IRP    1428.0      4.5    9  10   4 0
IRP    1478.0      4.5   10   7   4 0
IRP    1528.0      4.5    7   8   4 0
IRP    1578.0      4.5    8   9   4 0
IRP    1628.0      4.5    9  10   4 0
IRP    1678.0      4.5   10  11   4 0
```

```
IRP   1728.0      4.5   11   8    4 0
IRP   1778.0      4.5    8   9    4 0
IRP   1828.0      4.5    9  10    4 0
IRP   1878.0      4.5   10  11    4 0
IRP   1928.0      4.5   11  12    4 0
IRP   1978.0      4.5   12   9    4 0
IRP   2013.0      4.5    9  10    4 0
IRP   2048.0      4.5   10  11    4 0
BUP
OPTL    4*8 5 8 8 8 7 15*8
OPTA    4*8 5 8 5 8 7 15*8
OPTF    4*8 5 8 5 8 7 15*8
OUT    -11  1  -1  0
MOV     11  1  0  1.0       MOVE EOC22 To Vector 1
HED     1    CHARGE
RDA     Decay of Discharge Fuel
DEC     30.0    1   2   4 2
DEC     90.0    2   3   4 0
DEC    180.0    3   4   4 0
DEC    365.0    4   5   4 0
DEC    548.0    5   6   4 0
DEC    730.0    6   7   4 0
DEC    913.0    7   8   4 0
DEC   1095.0    8   9   4 0
HED      1     T=0
OUT    -9   1  -1  0
END
2  922350  22937.65  922380  466242.35       0     0.0                        UO2
4  080000  65913.82  130000       5.99  220000    10.78  240000     227.56  Fuel
4  260000    214.69  270000      11.98  280000   628.79  400000 114057.11  Fuel
4  410000   1214.93  420000      36.53       0     0.0                        Fuel
0
END
```

dto-b22a.inp

```
-1
-1
-1
BAS     Grams of Heavy Metal for 1 Mk-B-HTP Fuel Assembly
RDA     PLACE FUEL into vectors -1
LIP     0  0  0
LIB     0  1  2  3   604   605   606  9  3  0  1 39
PHO     0 0 0 10
RDA     READ FUEL COMPOSITION
INP     -1  1  -1 -1  1  1
RDA TIT     IRRADIATION OF TMI Batch 22A Fuel to EOC-22
MOV     -1  1  0  1.0
HED     1    CHARGE
RDA     BURN TO EOC-22
BUP
```

```
IRP     50.0      19.2    1   2    4 2
IRP    100.0      19.2    2   3    4 0
IRP    150.0      19.2    3   4    4 0
IRP    200.0      19.2    4   5    4 0
IRP    250.0      19.2    5   2    4 0
IRP    300.0      19.2    2   3    4 0
IRP    350.0      19.2    3   4    4 0
IRP    400.0      19.2    4   5    4 0
IRP    450.0      19.2    5   6    4 0
IRP    500.0      19.2    6   3    4 0
IRP    550.0      19.2    3   4    4 0
IRP    600.0      19.2    4   5    4 0
IRP    650.0      19.2    5   6    4 0
IRP    690.0      19.2    6   7    4 0
IRP    740.0      16.1    7   4    4 0
IRP    790.0      16.1    4   5    4 0
IRP    840.0      16.1    5   6    4 0
IRP    890.0      16.1    6   7    4 0
IRP    940.0      16.1    7   8    4 0
IRP    990.0      16.1    8   5    4 0
IRP   1040.0      16.1    5   6    4 0
IRP   1090.0      16.1    6   7    4 0
IRP   1140.0      16.1    7   8    4 0
IRP   1190.0      16.1    8   9    4 0
IRP   1240.0      16.1    9   6    4 0
IRP   1290.0      16.1    6   7    4 0
IRP   1340.0      16.1    7   8    4 0
IRP   1375.0      16.1    8   9    4 0
IRP   1410.0      16.1    9  10    4 0
BUP
OPTL     4*8  5  8  8  8  7  15*8
OPTA     4*8  5  8  5  8  7  15*8
OPTF     4*8  5  8  5  8  7  15*8
OUT     -10  1  -1  0
MOV      10  1  0  1.0       MOVE EOC22 To Vector 1
HED      1    CHARGE
RDA      Decay of Discharge Fuel
DEC      30.0    1   2   4 2
DEC      90.0    2   3   4 0
DEC     180.0    3   4   4 0
DEC     365.0    4   5   4 0
DEC     548.0    5   6   4 0
DEC     730.0    6   7   4 0
DEC     913.0    7   8   4 0
DEC    1095.0    8   9   4 0
HED       1      T=0
OUT     -9   1  -1  0
END
2  922350  20697.81  922380  465052.19       0      0.0                      UO2
4  080000  65452.67  130000       5.99  220000    10.78  240000     227.56  Fuel
4  260000    214.69  270000      11.98  280000   628.79  400000 114057.11  Fuel
```

```
4  410000   1214.93  420000      36.53       0     0.0                    Fuel
0
END
```

dto-b22b.inp

```
-1
-1
-1
BAS     Grams of Heavy Metal for 1 Mk-B-HTP Fuel Assembly
RDA     PLACE FUEL into vectors -1
LIP     0  0  0
LIB     0  1  2  3   604   605   606  9  3  0  1 39
PHO     0 0 0 10
RDA     READ FUEL COMPOSITION
INP     -1  1  -1 -1  1  1
RDA TIT     IRRADIATION OF TMI Batch 22B Fuel to EOC-22
MOV     -1  1  0  1.0
HED     1    CHARGE
RDA     BURN TO EOC-22
BUP
IRP      50.0      18.2     1   2   4 2
IRP     100.0      18.2     2   3   4 0
IRP     150.0      18.2     3   4   4 0
IRP     200.0      18.2     4   5   4 0
IRP     250.0      18.2     5   2   4 0
IRP     300.0      18.2     2   3   4 0
IRP     350.0      18.2     3   4   4 0
IRP     400.0      18.2     4   5   4 0
IRP     450.0      18.2     5   6   4 0
IRP     500.0      18.2     6   3   4 0
IRP     550.0      18.2     3   4   4 0
IRP     600.0      18.2     4   5   4 0
IRP     650.0      18.2     5   6   4 0
IRP     690.0      18.2     6   7   4 0
IRP     740.0       6.3     7   4   4 0
IRP     790.0       6.3     4   5   4 0
IRP     840.0       6.3     5   6   4 0
IRP     890.0       6.3     6   7   4 0
IRP     940.0       6.3     7   8   4 0
IRP     990.0       6.3     8   5   4 0
IRP    1040.0       6.3     5   6   4 0
IRP    1090.0       6.3     6   7   4 0
IRP    1140.0       6.3     7   8   4 0
IRP    1190.0       6.3     8   9   4 0
IRP    1240.0       6.3     9   6   4 0
IRP    1290.0       6.3     6   7   4 0
IRP    1337.0       6.3     7   8   4 0
IRP    1387.0       8.8     8   9   4 0
IRP    1437.0       8.8     9  10   4 0
IRP    1487.0       8.8    10   7   4 0
```

```
IRP   1537.0       8.8     7   8    4 0
IRP   1587.0       8.8     8   9    4 0
IRP   1637.0       8.8     9  10    4 0
IRP   1687.0       8.8    10  11    4 0
IRP   1737.0       8.8    11   8    4 0
IRP   1787.0       8.8     8   9    4 0
IRP   1837.0       8.8     9  10    4 0
IRP   1887.0       8.8    10  11    4 0
IRP   1937.0       8.8    11  12    4 0
IRP   1987.0       8.8    12   9    4 0
IRP   2022.0       8.8     9  10    4 0
IRP   2057.0       8.8    10  11    4 0
BUP
OPTL     4*8 5 8 8 8 7 15*8
OPTA     4*8 5 8 5 8 7 15*8
OPTF     4*8 5 8 5 8 7 15*8
OUT     -11  1  -1  0
MOV      11  1  0  1.0       MOVE EOC22 To Vector 1
HED      1    CHARGE
RDA      Decay of Discharge Fuel
DEC      30.0    1   2   4 2
DEC      90.0    2   3   4 0
DEC     180.0    3   4   4 0
DEC     365.0    4   5   4 0
DEC     548.0    5   6   4 0
DEC     730.0    6   7   4 0
DEC     913.0    7   8   4 0
DEC    1095.0    8   9   4 0
HED       1     T=0
OUT     -9   1  -1  0
END
2  922350  21518.32  922380  464551.68       0     0.0                     UO2
4  080000  65495.69  130000       5.99  220000    10.78  240000     227.56 Fuel
4  260000    214.69  270000      11.98  280000   628.79  400000 114057.11 Fuel
4  410000   1214.93  420000      36.53       0     0.0                     Fuel
0
END
```

dto-b22d.inp

```
-1
-1
-1
BAS      Grams of Heavy Metal for 1 Mk-B-HTP Fuel Assembly
RDA      PLACE FUEL into vectors -1
LIP      0  0  0
LIB      0  1  2  3   604   605   606  9  3  0  1 39
PHO      0 0 0 10
RDA      READ FUEL COMPOSITION
INP      -1  1  -1 -1  1  1
RDA TIT      IRRADIATION OF TMI Batch 22D Fuel to EOC-22
```

```
MOV     -1  1  0  1.0
HED     1    CHARGE
RDA     BURN TO EOC-22
BUP
IRP     50.0       17.3     1    2    4 2
IRP    100.0       17.3     2    3    4 0
IRP    150.0       17.3     3    4    4 0
IRP    200.0       17.3     4    5    4 0
IRP    250.0       17.3     5    2    4 0
IRP    300.0       17.3     2    3    4 0
IRP    350.0       17.3     3    4    4 0
IRP    400.0       17.3     4    5    4 0
IRP    450.0       17.3     5    6    4 0
IRP    500.0       17.3     6    3    4 0
IRP    550.0       17.3     3    4    4 0
IRP    600.0       17.3     4    5    4 0
IRP    650.0       17.3     5    6    4 0
IRP    690.0       17.3     6    7    4 0
IRP    740.0       16.2     7    4    4 0
IRP    790.0       16.2     4    5    4 0
IRP    840.0       16.2     5    6    4 0
IRP    890.0       16.2     6    7    4 0
IRP    940.0       16.2     7    8    4 0
IRP    990.0       16.2     8    5    4 0
IRP   1040.0       16.2     5    6    4 0
IRP   1090.0       16.2     6    7    4 0
IRP   1140.0       16.2     7    8    4 0
IRP   1190.0       16.2     8    9    4 0
IRP   1240.0       16.2     9    6    4 0
IRP   1290.0       16.2     6    7    4 0
IRP   1337.0       16.2     7    8    4 0
IRP   1387.0        5.4     8    9    4 0
IRP   1437.0        5.4     9   10    4 0
IRP   1487.0        5.4    10    7    4 0
IRP   1537.0        5.4     7    8    4 0
IRP   1587.0        5.4     8    9    4 0
IRP   1637.0        5.4     9   10    4 0
IRP   1687.0        5.4    10   11    4 0
IRP   1737.0        5.4    11    8    4 0
IRP   1787.0        5.4     8    9    4 0
IRP   1837.0        5.4     9   10    4 0
IRP   1887.0        5.4    10   11    4 0
IRP   1937.0        5.4    11   12    4 0
IRP   1987.0        5.4    12    9    4 0
IRP   2022.0        5.4     9   10    4 0
IRP   2057.0        5.4    10   11    4 0
BUP
OPTL     4*8  5  8  8  8  7  15*8
OPTA     4*8  5  8  5  8  7  15*8
OPTF     4*8  5  8  5  8  7  15*8
OUT     -11   1   -1   0
```

```
MOV      11   1   0   1.0        MOVE EOC22 To Vector 1
HED      1     CHARGE
RDA      Decay of Discharge Fuel
DEC      30.0      1    2    4 2
DEC      90.0      2    3    4 0
DEC     180.0      3    4    4 0
DEC     365.0      4    5    4 0
DEC     548.0      5    6    4 0
DEC     730.0      6    7    4 0
DEC     913.0      7    8    4 0
DEC    1095.0      8    9    4 0
HED       1      T=0
OUT     -9    1   -1   0
END
2  922350   22637.41  922380  465132.59        0      0.0                       UO2
4  080000   65724.25  130000       5.99   220000    10.78  240000      227.56   Fuel
4  260000     214.69  270000      11.98   280000   628.79  400000  114057.11   Fuel
4  410000    1214.93  420000      36.53        0      0.0                       Fuel
0
END
```

<u>dto-b23a.inp</u>

```
-1
-1
-1
BAS      Grams of Heavy Metal for 1 Mk-B-HTP Fuel Assembly
RDA      PLACE FUEL into vectors -1
LIP      0   0   0
LIB      0   1   2   3    604    605    606   9   3   0   1  39
PHO      0  0  0  10
RDA      READ FUEL COMPOSITION
INP      -1   1   -1  -1   1   1
RDA TIT      IRRADIATION OF TMI Batch 23A Fuel to EOC-22
MOV      -1   1   0   1.0
HED      1     CHARGE
RDA      BURN TO EOC-22
BUP
IRP      50.0      18.3     1    2    4 2
IRP     100.0      18.3     2    3    4 0
IRP     150.0      18.3     3    4    4 0
IRP     200.0      18.3     4    5    4 0
IRP     250.0      18.3     5    2    4 0
IRP     300.0      18.3     2    3    4 0
IRP     350.0      18.3     3    4    4 0
IRP     400.0      18.3     4    5    4 0
IRP     450.0      18.3     5    6    4 0
IRP     500.0      18.3     6    3    4 0
IRP     550.0      18.3     3    4    4 0
IRP     600.0      18.3     4    5    4 0
IRP     647.0      18.3     5    6    4 0
```

```
IRP    697.0      16.5     6    7    4 0
IRP    747.0      16.5     7    4    4 0
IRP    797.0      16.5     4    5    4 0
IRP    847.0      16.5     5    6    4 0
IRP    897.0      16.5     6    7    4 0
IRP    947.0      16.5     7    8    4 0
IRP    997.0      16.5     8    5    4 0
IRP   1047.0      16.5     5    6    4 0
IRP   1097.0      16.5     6    7    4 0
IRP   1147.0      16.5     7    8    4 0
IRP   1197.0      16.5     8    9    4 0
IRP   1247.0      16.5     9    6    4 0
IRP   1297.0      16.5     6    7    4 0
IRP   1332.0      16.5     7    8    4 0
IRP   1367.0      16.5     8    9    4 0
BUP
OPTL     4*8 5 8 8 8 7 15*8
OPTA     4*8 5 8 5 8 7 15*8
OPTF     4*8 5 8 5 8 7 15*8
OUT     -9  1  -1  0
MOV      9  1  0  1.0       MOVE EOC22 To Vector 1
HED      1     CHARGE
RDA      Decay of Discharge Fuel
DEC      30.0     1    2    4 2
DEC      90.0     2    3    4 0
DEC     180.0     3    4    4 0
DEC     365.0     4    5    4 0
DEC     548.0     5    6    4 0
DEC     730.0     6    7    4 0
DEC     913.0     7    8    4 0
DEC    1095.0     8    9    4 0
HED       1      T=0
OUT     -9   1  -1   0
END
2  922350  19268.56  922380  469161.44        0      0.0                   UO2
4  080000  65812.99  130000       5.99   220000    10.78  240000    227.56  Fuel
4  260000    214.69  270000      11.98   280000   628.79  400000 114057.11  Fuel
4  410000   1214.93  420000      36.53        0      0.0                   Fuel
0
END
```

<u>dto-b24a.inp</u>

```
-1
-1
-1
BAS      Grams of Heavy Metal for 1 Mk-B-HTP Fuel Assembly
RDA      PLACE FUEL into vectors -1
LIP      0  0  0
LIB      0  1  2  3   604   605   606  9  3  0  1 39
PHO      0 0 0 10
```

```
RDA     READ FUEL COMPOSITION
INP     -1  1  -1 -1  1  1
RDA TIT     IRRADIATION OF TMI Batch 24A Fuel to EOC-22
MOV     -1  1  0  1.0
HED     1    CHARGE
RDA     BURN TO EOC-22
BUP
IRP      50.0      19.3    1   9   4 2
IRP     100.0      19.3    9   2   4 0
IRP     150.0      19.3    2   9   4 0
IRP     200.0      19.3    9   3   4 0
IRP     250.0      19.3    3   9   4 0
IRP     300.0      19.3    9   4   4 0
IRP     350.0      19.3    4   9   4 0
IRP     400.0      19.3    9   5   4 0
IRP     450.0      19.3    5   9   4 0
IRP     500.0      19.3    9   6   4 0
IRP     550.0      19.3    6   9   4 0
IRP     600.0      19.3    9   7   4 0
IRP     650.0      19.3    7   9   4 0
IRP     685.0      19.3    9   8   4 0
IRP     720.0      19.3    8   9   4 0
BUP
OPTL    4*8 5 8 8 8 7 15*8
OPTA    4*8 5 8 5 8 7 15*8
OPTF    4*8 5 8 5 8 7 15*8
OUT    -9  1  -1  0
MOV     9  1  0  1.0       MOVE EOC22 To Vector 1
HED     1    CHARGE
RDA     Decay of Discharge Fuel
DEC     30.0     1   2   4 2
DEC     90.0     2   3   4 0
DEC    180.0     3   4   4 0
DEC    365.0     4   5   4 0
DEC    548.0     5   6   4 0
DEC    730.0     6   7   4 0
DEC    913.0     7   8   4 0
DEC   1095.0     8   9   4 0
HED      1     T=0
OUT    -9   1  -1   0
END
2  922350    19804.03   922380    465945.97       0   0.0             UO2
4  080000  65452.67  130000       5.99  220000     10.78  240000    227.56  Fuel
4  260000    214.69  270000      11.98  280000    628.79  400000 114057.11  Fuel
4  410000   1214.93  420000      36.53       0     0.0                      Fuel
0
END
```

dto.bat

Note: DOSBox was used to run ORIGEN2 on the nfw-ksq-01 server. D:\dosprogs was mounted as the D: drive when this batch file was run.

```
echo off
echo ***********************************************************************
echo ***********************************************************************
echo **                                                                  **
echo **                           O R I G E N 2                          **
echo **           Oak Ridge Isotope GENeration and Depletion Code        **
echo **                        Version 2.1 (8-1-91)                      **
echo **                                                                  **
echo ***********************************************************************
echo **                                                                  **
echo **   Developed by:  Oak Ridge National Laboratory                   **
echo **                  Chemical Technology Division                    **
echo **                                                                  **
echo **   Technical Contact:  Scott B. Ludwig                            **
echo **                       (615) 574-7916   FTS 624-7916              **
echo **                                                                  **
echo **   Distributed by:  Radiation Shielding Information Center (RSIC) **
echo **                    Oak Ridge National Laboratory                 **
echo **                    P.O. Box 2008                                 **
echo **                    Oak Ridge, TN  37831                          **
echo **                    (615) 574-6176   FTS 624-6176                 **
echo ***********************************************************************
echo ***********************************************************************
pause
echo ** Execution continuing ...                                         **
echo ***********************************************************************
echo ***********************************************************************
echo **                                                                  **
echo **     Version 2.1 (8-1-91) for mainframes and 80386 or 80486 PCs   **
echo **                                                                  **
copy dto-b24a.inp tape5.inp >nul
REM  (NOT USED IN THIS CASE) copy samp_2.u3 tape3.inp >nul
copy d:\origen2\libs\decay.lib+d:\origen2\libs\pwrue.lib tape9.inp >nul
copy d:\origen2\libs\gxuo2brm.lib tape10.inp >nul
d:\origen2\code\origen2
rem  combine and save files from run
copy tape12.out+tape6.out dto-b24a.u6 >nul
copy tape13.out+tape11.out dto-b24a.out >nul
ren tape7.out dto-b24a.pch
ren tape15.out dto-b24a.dbg
ren tape16.out dto-b24a.vxs
ren tape50.out dto-b24a.ech
rem  cleanup files
del tape*.inp
del tape*.out
echo ***********************************************************************
```

```
echo ******************* O R I G E N 2 - Version 2.1 ***********************
echo *********************** Execution Completed ***************************
echo ***********************************************************************
echo on
```

APPENDIX B

Alloy 718 Chemistry

B637 – 16

TABLE 1 Chemical Requirements

Element	Composition Limits, %							
	UNS N07022	UNS N07208	UNS N07252 (Formerly Grade 689)	UNS N07001 (Formerly Grade 685)	UNS N07500 (Formerly Grade 684)	UNS N07740	UNS N07750 (Formerly Grade 688)	UNS N07718 (Formerly Grade 718)
Carbon	0.010 max	0.04–0.08	0.10–0.20	0.03–0.10	0.15 max	0.005–0.08	0.08 max	0.08 max
Manganese	0.5 max	0.3 max	0.50 max	1.00 max	0.75 max	1.00 max	1.00 max	0.35 max
Silicon	0.08 max	0.15 max	0.50 max	0.75 max	0.75 max	1.00 max	0.50 max	0.35 max
Phosphorus	0.025 max	0.015 max	0.015 max	0.030 max	0.015 max	0.030 max	...	0.015 max
Sulfur	0.015 max	0.015 max	0.015 max	0.030 max	0.015 max	0.030 max	0.01 max	0.015 max
Chromium	20.0–21.4	18.5–20.5	18.00–20.00	18.00–21.00	15.00–20.00	23.50–25.50	14.00–17.00	17.0–21.0
Cobalt	1.0 max	9.0–11.0	9.00–11.00	12.00–15.00	13.00–20.00	15.00–22.00	1.00 max[A]	1.0 max[A]
Molybdenum	15.5–17.4	8.0–9.0	9.00–10.50	3.50–5.00	3.00–5.00	2.00 max	...	2.80–3.30
Columbium (Nb) + tantalum	...	...	...	...	...	...	0.70–1.20	4.75–5.50
Titanium	...	1.90–2.30	2.25–2.75	2.75–3.25	2.50–3.25	0.50–2.50	2.25–2.75	0.65–1.15
Aluminum	0.5 max	1.38–1.65	0.75–1.25	1.20–1.60	2.50–3.25	0.20–2.00	0.40–1.00	0.20–0.80
Zirconium	...	0.020 max	...	0.02–0.12	...	...	...	...
Boron	0.006 max	0.003–0.010	0.003–0.01	0.003–0.01	0.003–0.01	0.0008–0.006	...	0.006 max
Iron	1.8 max	1.5 max	5.00 max	2.00 max	4.00 max	3.00 max	5.00–9.00	remainder[B]
Copper	0.5 max	0.1 max	...	0.50 max	0.15 max	0.50 max	0.50 max	0.30 max
Nickel	remainder[B]	remainder[B]	remainder[B]	remainder[B]	remainder[B]	remainder[B]	70.00 min	50.0–55.0
Tantalum	0.2 max	0.1 max	...	...	...	...	...	...
Columbium (Niobium)	...	0.2 max	...	...	...	0.50–2.50	...	...
Tungsten	0.8 max	0.5 max	...	...	...	...	...	...

Element	UNS N07080 (Formerly Grade 80A)	UNS N07752	UNS N09925	UNS N07725
Carbon	0.10 max	0.020–0.060	0.03 max	0.03 max
Manganese	1.00 max	1.00 max	1.0 max	0.35 max
Silicon	1.00 max	0.50 max	0.5 max	0.20 max
Phosphorus	...	0.008 max	0.03 max	0.015 max
Sulfur	0.015 max	0.003 max	0.03 max	0.010 max
Chromium	18.00–21.00	14.50–17.00	19.5–22.5	19.00–22.50
Cobalt	...	0.050 max	...	...
Molybdenum	...	...	2.5–3.5	7.00–9.50
Columbium (Nb) + tantalum	...	0.70–1.20	0.5 max (Nb only)	2.75–4.00
Titanium	1.80–2.70	2.25–2.75	1.9–2.40	1.00–1.70
Aluminum	0.50–1.80	0.40–1.00	0.1–0.5	0.35 max
Boron	...	0.007 max	...	...
Iron	3.00 max	5.00–9.00	22.0 min	remainder[B]
Copper	...	0.50 max	1.5–3.0	...
Zirconium	...	0.050 max	...	...
Vanadium	...	0.10 max	...	...
Nickel	remainder[B]	70.0 min	42.0–46.0	55.0–59.0

[A] If determined.
[B] The element shall be determined arithmetically by difference.

ATTACHMENT 2
Specific Heat Study

A study was performed to determine a suitable temperature to be used to calculate the specific heats for the materials. Each material's specific heat is governed by an equation (or set of equations) provided in References 3.17, 3.18, and 3.19 (shown below):

$$c_{p,UO_2} = \left(\frac{K_1 * \theta^2 * \exp\left(\frac{\theta}{T_k}\right)}{T_k^2 * \left(\exp\left(\frac{\theta}{T_k}\right) - 1\right)^2} + 2K_2 * T_k + \frac{K_3 * E_D}{R * T_k^2} * \exp\left(\frac{-E_D}{R * T_k}\right) \right) \div A_{UO_2}$$

$$\text{for } 298\text{K} \leq T_k \leq 3120\text{K}$$

Where:

$c_{p,\,UO2}$ = specific heat of UO_2, J/g-K
Θ = 535.85 K
E_D = 157.7707*10^3 J/mol
K_1 = 80.1314 J/mol-K
K_2 = 32.845*10^{-4} J/mol-K^2
K_3 = 23.62183*10^6 J/mol
R = 8.134 J/mol-K (ideal gas constant)
T_k = Temperature, K
A_{UO2} = atomic weight of uranium dioxide, g/mole

$$c_{p,M5} = 0.2375 + 0.0001591 * T \quad \text{for } 273\text{K} \leq T \leq 1100\text{K}$$
$$c_{p,M5} = -13.4034 + 0.01256 * T \quad \text{for } 1100\text{K} \leq T \leq 1140\text{K}$$
$$c_{p,M5} = 6.9160 - 0.005264 * T \quad \text{for } 1140\text{K} \leq T \leq 1250\text{K}$$
$$c_{p,M5} = 0.2141 + 0.0000975 * T \quad \text{for } 1250\text{K} \leq T \leq 1600\text{K}$$

Where:

$c_{p,\,M5}$ = specific heat of M5, J/g-K
T = Temperature, K

$$c_{p,SS} = (0.1122 + 3.222\text{x}10^{-5} * T) * \left(\frac{4.184\ \text{J}}{1\ \text{cal}}\right)$$

Where

$c_{p,SS}$ = specific heat of CF3 stainless steel, J/g-K
T = Temperature, K

Inconel Alloy 718 specific heats are provided in Appendix 7.5.

In order to determine how these parameters affect the heat-up time for the fuel assembly, the heat transfer equation (Equation 6.5.10-2) was solved for time:

$$t = \frac{\Delta T}{\dot{q}} * \left(m_{UO_2} * c_{p,UO_2} + \left(\rho_{M5} * V_{M5,Rods} + m_{M5,Grids}\right) * c_{p,M5} + m_{SS} * c_{p,SS} + m_{HMP} * c_{p,HMP}\right)$$

For simplicity, the values calculated for ΔT, $\dot{q}$, m_{UO2}, ρ_{M5}, V_{M5}, $m_{M5,Grids}$, m_{SS}, and $m_{Inconel}$ in section 6.5 of this calculation are used and assumed to be constant over the temperature range.

ΔT = 857 K

$\dot{q}$ = 5270 W

m_{UO2} = 5.51E^5 g

ρ_{M5} = 6.50 g/cm^3

$V_{M5,Rods}$ = 18029.6 cm^3

$m_{M5,Grids}$ = 7620.9 g

m_{SS} = 16692.2 g

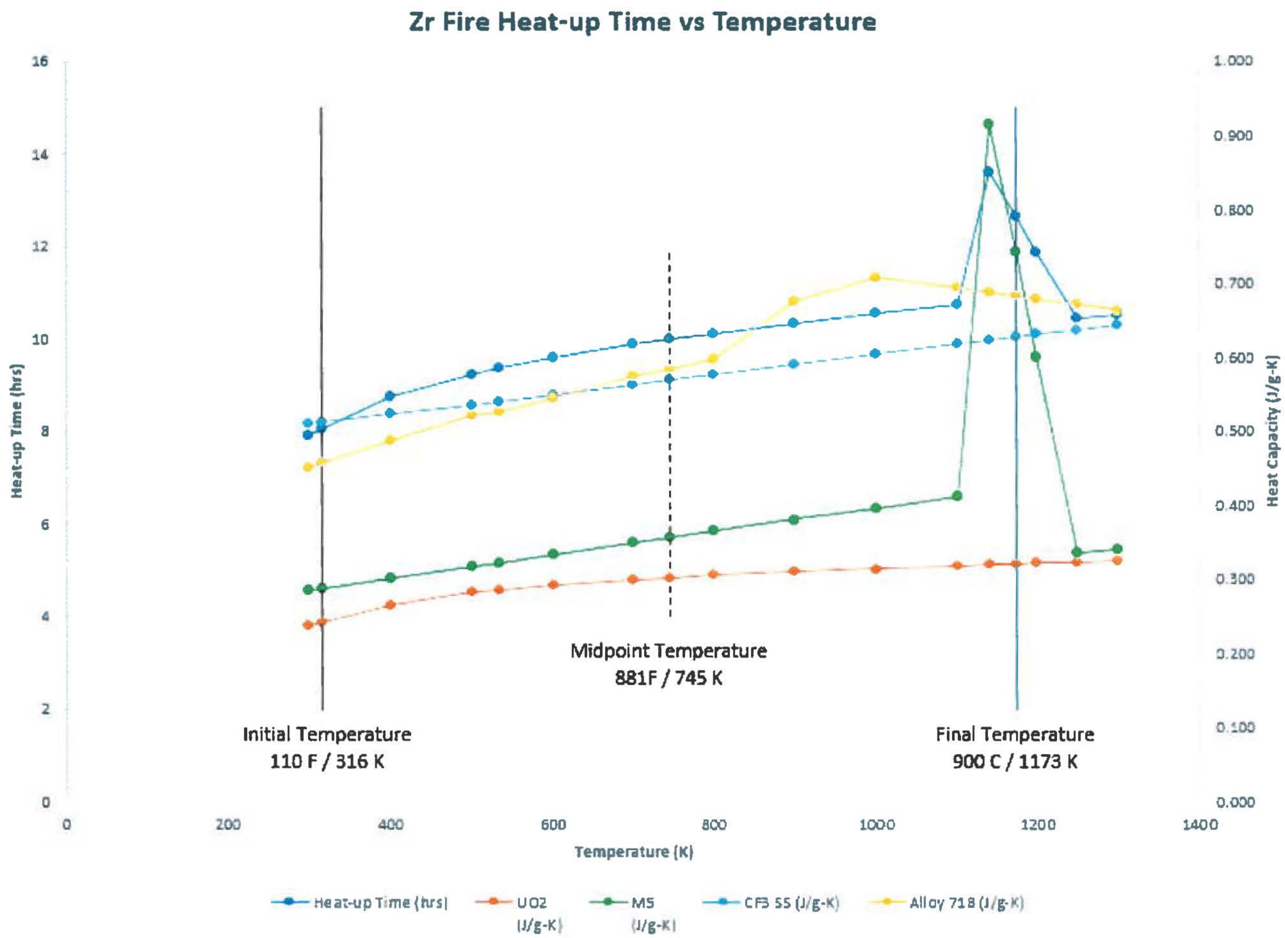
m_{HMP} = 1197.7 g

The following table calculates the heat-up time for the fuel assembly using the temperature dependent specific heat values.

Temperature (K)	UO2 (J/g-K)	M5 (J/g-K)	CF3 SS (J/g-K)	Alloy 718 (J/g-K)	Heat-up Time (hrs)
300	0.236	0.285	0.510	0.451	7.90
316	0.242	0.288	0.512	0.459	8.07
400	0.266	0.301	0.523	0.488	8.74
500	0.282	0.317	0.537	0.522	9.25
533	0.286	0.322	0.541	0.528	9.38
600	0.292	0.333	0.550	0.546	9.60
700	0.300	0.349	0.564	0.574	9.89
745	0.302	0.356	0.570	0.583	10.00
800	0.305	0.365	0.577	0.597	10.13
900	0.310	0.381	0.591	0.675	10.35
1000	0.314	0.397	0.604	0.708	10.55
1100	0.318	0.413	0.618	*0.693*	10.74
1140	0.319	0.915	0.623	*0.687*	13.62
1173	0.320	0.741	0.627	*0.682*	12.67
1200	0.321	0.599	0.631	*0.678*	11.89
1250	0.323	0.336	0.638	*0.671*	10.45
1300	0.325	0.341	0.645	*0.663*	10.53

Note: The specific heat values for Inconel Alloy 718 above 1088 K are extrapolated based on the last two values in the Appendix 7.5 table.

The values from the table are plotted in the figure below.



The figure clearly shows that the heat-up time increases with increasing material temperature for all materials except for Inconel Alloy 718. However, even though the specific heat for Inconel decreases at the higher end of the temperature band, it does not have a significant impact on the heat-up time due to the minimal amount of mass it is applying to in the calculation (0.8% of total mass). Overall, this means that as the materials in the fuel assembly reach higher temperatures, they would heat up more slowly. Therefore, using a temperature at or below the midpoint of the temperature range would be conservative with respect to the assembly heat-up. For the purposes of this analysis, the specific heats were calculated at the midpoint temperature of 881°F (745K).