Exelon Generation®

Michael P. Gallagher
Exelon Nuclear
Vice President
License Renewal and Decommissioning

200 Exelon Way
Kennett Square, PA 19348

610 765 5958 Office
610 765 5658 Fax
www.exeloncorp.com

michaelp.gallagher@exeloncorp.com

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

RA-17-048

August 22, 2017

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

Oyster Creek Nuclear Generating Station
Renewed Facility Operating License No. DPR-16
NRC Docket Nos. 50-219 and 72-15

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

Reference: Letter from Keith R. Jury, Exelon Generation Company, LLC to U.S. Nuclear Regulatory Commission - "Permanent Cessation of Operations at Oyster Creek Nuclear Generating Station," dated January 7, 2011, RA-11-007 (ML110070507)

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 Oyster Creek Nuclear Generating Station (OCNGS). The requested exemptions would allow OCNGS to reduce emergency planning requirements consistent with the permanently defueled condition of the station.

By letter dated January 7, 2011 (Reference), Exelon provided formal notification to the U.S. Nuclear Regulatory Commission (NRC) of Exelon's contingent determination to permanently cease power operations at OCNGS no later than December 31, 2019. Once the certifications for permanent cessation of power operations and of permanent removal of fuel from the reactor vessel are submitted to the NRC pursuant to 10 CFR 50.82(a)(1)(i) and (ii), NRC regulations stipulate in 10 CFR 50.82(a)(2) that the 10CFR 50 license no longer authorizes operation of the reactor or emplacement or retention of fuel in the reactor vessel.

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 OCNGS 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 OCNGS 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 12 months 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 OCNGS without any additional compensatory actions. Following the OCNGS shutdown, which is expected by the end of 2019 (Reference 1), 12 months after shutdown will occur in early January 2021. The bounding analysis is contained in Attachment 2.

OCNGS 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 February 22, 2019. Exelon requests that the approved exemptions become effective 12 months following the docketing of the certification required by 10 CFR 50.82(a)(1)(ii) that OCNGS has been permanently defueled. Approval of these exemptions by February 22, 2019 will allow OCNGS 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 support of this exemption request and the associated amendment for the PDEP, numerous discussions, both electronic and in person, have been held with the cognizant state (New Jersey) and local response organizations. On June 30, 2017, the New Jersey Department of Environmental Protection Bureau of Nuclear Engineering (BNE) met with the OCNGS Decommissioning Transition Organization to review emergency planning during the permanently defueled decommissioning phase. A draft of the exemption request (Attachment 1 of this letter) was provided to the BNE. The BNE performed a review of the draft exemption request and had no substantive comments. The BNE has been provided draft copies of the proposed License Amendment Request (LAR) for the PDEP and associated EALs. The results of their review are expected to be received shortly. It is Exelon's intent to address, as appropriate, BNE's comments prior to submittal of the PDEP LAR. Any correspondence regarding BNE's comments and Exelon's resolution of those comments will be included as an attachment to the LAR.

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

If you have any questions concerning this submittal, please contact Paul Bonnett at (610) 765-5264.

Respectfully,

Michael P. Gallagher

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

Attachments:
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. Oyster Creek Nuclear Generating Station Zirconium Fire Analysis for Drained Spent Fuel Pool (Calculation C-1302-226-E310-457)

cc: w/Attachments

Regional Administrator - NRC Region I
NRC Senior Resident Inspector - Oyster Creek Nuclear Generating Station
NRC Project Manager, NRR - Oyster Creek Nuclear Generating Station
Director, Bureau of Nuclear Engineering - New Jersey Department of Environmental Protection
Mayor of Lacey Township, Forked River, NJ

Attachment 1

OYSTER CREEK NUCLEAR GENERATING STATION

DOCKET NUMBERS 50-219 & 72-15 / LICENSE NUMBER DPR-16

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 Oyster Creek Nuclear Generating Station (OCNGS):

- 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 requested exemptions would allow Exelon to reduce emergency planning requirements and subsequently revise the OCNGS 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 OCNGS. 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

OCNGS site is located near the Atlantic Ocean within the State of New Jersey. The facility site, approximately 152 acres, is in Lacey and Ocean Townships, Ocean County. OCNGS is about two miles inland from the shore of Barnegat Bay and seven miles west-northwest of Barnegat Light on the Atlantic shorefront. The site is approximately nine miles south of Toms River, New Jersey, about fifty miles east of Philadelphia, Pennsylvania, and sixty miles south of Newark, New Jersey. Exelon owns approximately 708 acres of property to the east of Route 9 extending to the Barnegat Bay. Water access to the site is provided by the Intercostal Waterway, which runs through Barnegat Bay.

Section 15 of the OCNGS Updated Final Safety Analysis Report (UFSAR) describes the design basis accident (DBA) scenarios that are applicable to OCNGS. Many of the accident scenarios postulated in the UFSAR for operating power reactors involve failures or malfunctions of systems, which could affect the fuel in the reactor vessel, which in the most severe postulated accidents, would involve the release of large quantities of fission products. With the termination of reactor operations and the permanent removal of fuel from the reactor vessel, such accidents are no longer possible. Therefore, the postulated accidents involving failure or malfunction of the reactor, reactor cooling system, steam system, or turbine generator are no longer applicable. The remaining accident is the Fuel Handling Accident (FHA) that takes place in the spent fuel pool (SFP) located in the Reactor Building.

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 6) beyond the site boundary. Exelon will maintain the version of the EPA PAGs as specified in the current and proposed OCNGS Emergency Plan.

By letter dated January 7, 2011 (Reference 2), 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 OCNGS no later than December 31, 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 OCNGS 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).

In accordance with the New Jersey Department of Environmental Protection's Administrative Order No. 2011-06 (Reference 35), OCNGS will submit a Post-Shutdown Decommissioning Activities Report (PSDAR) by December 31, 2018, which will identify OCNGS's selected method of decommissioning. By December 31, 2019, the OCNGS reactor will be permanently shut down. After the reactor is shut down, all fuel assemblies will be removed from the reactor vessel and placed in the SFP. The irradiated fuel will be stored in the SFP and the Independent Spent Fuel Storage Installation (ISFSI) until it is all placed into ISFSI or shipped offsite in accordance with the schedules described in the PSDAR and updated Spent Fuel Management Plan.

With the reactor defueled, the reactor vessel assembly and supporting structures and systems are no longer in operation and have no function related to the safe storage and management of irradiated fuel in the SFP. A fuel pool cooling and clean-up system is provided to remove decay heat from spent fuel stored in the SFP and to maintain a specified water temperature, purity, clarity, and level.

3.0 BASIS FOR EXEMPTION REQUEST

In order to allow a reduction in emergency planning requirements commensurate with the hazards associated with OCNGS'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 12 months after permanent cessation of power operations, the spent fuel in the SFP will have decayed to the extent that the requested exemptions can be implemented at OCNGS without any compensatory actions (Reference 16). This analysis is contained in Attachment 2. Considering that the shutdown date is December 31, 2019, 12 months following permanent cessation of power operations would occur in January 2021. Exelon plans to submit a permanently defueled emergency plan (PDEP) by August 29, 2017, 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 OCNGS 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 OCNGS.

The proposed emergency plan will be based on the exemptions requested herein. Exelon requests approval of these exemption requests within 18 months of the date of this submittal with an effective date of January 2, 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 to OCNGS 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. 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 OCNGS. 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 FROM 10 CFR 50.47

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 3), 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 Management Plan (CEMP), formal offsite radiological emergency plans would not be necessary for a permanently defueled nuclear power reactor licensee.

TABLE 1
EXEMPTIONS FROM 10 CFR 50.47

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Item	10 CFR 50.47 Emergency Plans	Basis for Exemption
		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. OCNGS has an analysis (Reference 15) that demonstrates that 33 days after 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 16) also shows that 12 months after shutdown for an unlikely event of a beyond-DBA where the hottest fuel assembly adiabatic heat-up occurs, 10 hours are available to take 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. Several systems will be available to provide makeup water to the SFP, such as Torus water, Firewater, and portable FLEX and B.5.b pumps. These systems provide diversity with electrical driven pumps, installed diesel and portable diesel pumps. Water sources are from various tanks, fire pond, and intake or discharge canal water. OCNGS 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 are maintained in accordance with License Condition 2.C.(8) of the OCNGS Renewed Facility Operating 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.

TABLE 1
EXEMPTIONS FROM 10 CFR 50.47

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Item	10 CFR 50.47 Emergency Plans	Basis for Exemption
		Training of the on-shift staff will be maintained and 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 made,~~** and other organizations capable of augmenting the planned response have been identified.	Discontinuing offsite emergency planning activities and reducing the scope of onsite emergency planning is acceptable given the significantly reduced offsite consequences when OCNGS is in the permanently defueled condition. The OCNGS 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).

TABLE 1
EXEMPTIONS FROM 10 CFR 50.47

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Item	10 CFR 50.47 Emergency Plans	Basis for Exemption
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. OCNGS 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 4), endorsed by the NRC in a letter dated March 28, 2013 (Reference 5). OCNGS analysis (Reference 16) shows that after the spent fuel has decayed for 12 months, 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. 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 31), 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 32) 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

TABLE 1
EXEMPTIONS FROM 10 CFR 50.47

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Item	10 CFR 50.47 Emergency Plans	Basis for Exemption
		only occur during the first several days after the fuel was removed from the reactor. As previously indicated, an OCNGS analysis shows that after the spent fuel has decayed for 12 months, 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).
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	Refer to basis for 10 CFR 50.47(b)

TABLE 1
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Item	10 CFR 50.47 Emergency Plans	Basis for Exemption
	~~offsite~~ consequences of a radiological emergency condition are in use.	
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 the ingestion exposure pathway EPZ appropriate to the locale have been developed.~~**	OCNGS has developed an analysis indicating that 12 months after permanent cessation of power operations, no credible accident at OCNGS will result in radiological releases requiring offsite protective actions. The analysis of the potential radiological impact of the postulated accident for OCNGS 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 OCNGS 12 months after permanent cessation of power operations, evacuation planning, including evacuation time estimates, is not needed since OCNGS will meet the criteria for an exemption from offsite EP 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 40.47(b)(11) through (b)(16)	No exemption requested.
13	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~~**	OCNGS has developed an analysis indicating that 12 months after permanent cessation of power operations, no credible accident at OCNGS will result in radiological releases requiring offsite protective actions. The analysis of the potential radiological impact of the postulated accident for OCNGS in a permanently defueled condition indicates that any releases beyond the site boundary are limited to small fractions of the EPA PAG exposure levels.

TABLE 1
EXEMPTIONS FROM 10 CFR 50.47

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Item	10 CFR 50.47 Emergency Plans	Basis for Exemption
	~~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.~~	OCNGS is not a gas cooled reactor and is not authorized for power operation. Refer to basis for 10 CFR 50.47(b)(10).

TABLE 2
EXEMPTIONS 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	**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), OCNGS 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 7), 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 8). 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, notification of offsite authorities and coordination with offsite agencies

TABLE 2
EXEMPTIONS 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
		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 OCNGS and NPRs show that the OCNGS facility should be treated in a similar fashion as an NPR. Similar to NPRs, OCNGS will pose lower radiological risks to the public from accidents than do power reactors because: (1) OCNGS 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 OCNGS 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 accident at OCNGS will result in radiological releases requiring offsite protective actions.
2	IV.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	IV.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)

TABLE 2
EXEMPTIONS FROM 10 CFR PART 50, APPENDIX E

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Item	10 CFR PART 50, APPENDIX E, SECTION IV	Basis for Exemption
4	IV.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	IV.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	IV.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~~**	Refer to basis for 10 CFR 50.47(b)(10)

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EXEMPTIONS FROM 10 CFR PART 50, APPENDIX E

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Item	10 CFR PART 50, APPENDIX E, SECTION IV	Basis for Exemption
	~~shall update the ETE analysis to reflect the impact of that population 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	IV.7	No exemption requested.
8	*A Organization* The organization for coping <…> A.1. A description of the normal plant **~~operating~~** organization.	Once OCNGS is permanently shut down and defueled, a decommissioning reactor will not be authorized to operate under 10 CFR 50.82(a). Because the OCNGS 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.
9	A.2.	No exemption requested.
10	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 OCNGS 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. OCNGS 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).
11	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	OCNGS has developed an analysis indicating that 12 months after permanent cessation of power operations, no credible accident at OCNGS will result in radiological releases requiring offsite protective actions. OCNGS will maintain the capability to determine if a radiological release is occurring. If a release is occurring, OCNGS will promptly communicate that information to offsite

TABLE 2
EXEMPTIONS FROM 10 CFR PART 50, APPENDIX E

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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.	authorities for their consideration. The offsite organizations are responsible for deciding what, if any, protective actions should be taken based on a CEMP. Also refer to basis for 10 CFR 50.47(b).
12	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 OCNGS adiabatic heatup 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 OCNGS 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).
13	A.6.	No exemption requested.
14	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 OCNGS from "hostile action" related requirements, some EP requirements for security-based events will be maintained. The classification of security-based events,

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EXEMPTIONS FROM 10 CFR PART 50, APPENDIX E

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		notification of offsite authorities, and coordination with offsite agencies under a CEMP concept will still be required. OCNGS will maintain appropriate actions for the protection of onsite personnel in a security-based event. The scope of protective actions will be appropriate for the defueled plant status, but will not be the same as actions necessary for an operating power plant. Refer to basis for 10 CFR Part 50, Appendix E, Section IV.1.
15	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 analysis has been developed indicating that 12 months 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).
16	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 OCNGS 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 7), 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

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Item	10 CFR PART 50, APPENDIX E, SECTION IV	Basis for Exemption
		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 mitigate an emergency without impeding timely performance of emergency plan functions. For all of these reasons, it can be concluded that a decommissioning NPP is exempt from the requirement of 10 CFR Part 50, Appendix E, Section IV.A.9.
17	*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.	OCNGS will develop EALs consistent with the Permanently Defueled EALs detailed in Appendix C of NEI 99-01, Revision 6 (Reference 4). OCNGS proposes to continue to review EALs with the State of New Jersey 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."
18	B.2.	No exemption requested.

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Item	10 CFR PART 50, APPENDIX E, SECTION IV	Basis for Exemption
19	*C. Activation of Emergency Organization* 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.	The Permanently Defueled EALs, developed consistent with Appendix C of NEI 99-01, Revision 6 (Reference 4), will be adopted, as previously described. This 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 5). No offsite protective actions are anticipated to be necessary, so 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 3), 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 OCNGS 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

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Item	10 CFR PART 50, APPENDIX E, SECTION IV	Basis for Exemption
		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 9) for 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.
20	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 7), 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. OCNGS will maintain the capability to assess, classify, and declare an emergency condition within 30 minutes after the availability of indications to operators that an EAL threshold has been reached. Emergency declaration is required to be made as soon as conditions warranting classification are present and recognizable, but within 30 minutes. 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 classify and declare an emergency is unnecessarily restrictive.

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Item	10 CFR PART 50, APPENDIX E, SECTION IV	Basis for Exemption
		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.
21	*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).
22	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).
23	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~~**	OCNGS proposes to complete emergency notifications within 60 minutes after the availability of indications to operators that an EAL threshold has been reached. This timeframe is consistent with the 10 CFR 50.72(a)(3) notification to

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	~~shall demonstrate that the appropriate governmental authorities have the capability to make a public alerting and notification decision promptly on being informed 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 with 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~~	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. Because of the geographic location of OCNGS, emergency planning and responsibilities have historically involved coordination with the State of New Jersey. Decommissioning-related emergency plan submittals for OCNGS have been discussed with offsite response organizations since Exelon provided notification that it would permanently cease power operations. These discussions have addressed changes to onsite and offsite emergency preparedness throughout the decommissioning process, including the proposed time of 60 minutes to notify the state after the availability of indications to operators that an EAL threshold has been reached. New Jersey Emergency Management officials have been able to review and concur with this proposal. The State will provide a letter with the Emergency Plan submittal acknowledging the notification period. Also refer to basis for 10 CFR 50.47(b) and 50.47(b)(10).

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	~~primary prompt public alert and notification system. When there is a decision to activate the alert and 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.~~	
24	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.
25	E1 thru E7	No exemption requested.
26	*E. Emergency Facilities and Equipment*	The OCNGS analysis indicates that within 12 months after shutdown, no design-basis accidents or other credible event at OCNGS will exceed the EPA PAGs. Due to the low probability of design-basis accidents or other credible events to

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	E.8.a.(i) A licensee **~~onsite technical support center and an emergency operations~~** facility from which effective direction can be given and effective control can be exercised during an emergency;	exceed the EPA PAGs at the site boundary, the available time for event mitigation at a 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 OCNGS 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).
27	E.8.a.(ii) **~~For nuclear power reactor licensees, a licensee onsite operational support center;~~**	NUREG-0696, "Functional Criteria for Emergency Response Facilities," (Reference 33) 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. A single onsite facility will continue to be maintained at OCNGS, from which Control Room support, emergency mitigation, radiation monitoring, and effective control may be exercised during an emergency.
28	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~~**	In accordance with paragraph 8.e., the requirements of paragraph 8.b.(1) – (5) do not apply to the OCNGS 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 OCNGS in a permanently shutdown and defueled condition. Refer to basis for 10 CFR 50.47(b)(3).

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	~~**emergency operations facility more than 25 miles 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;**~~	
29	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).

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	(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~~**	
30	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 exception of the capability for staging emergency response organization personnel at the alternative~~**	Refer to basis for 10 CFR Part 50, Appendix E, Section IV.1. regarding "hostile action."

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	~~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.~~	
31	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 50.47(b)(3).
32	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). OCNGS will maintain communications with the State of New Jersey and the NRC. Note, the State and local officials, and agencies for which provisions will be maintained are those which OCNGS is currently committed to.
33	E.9.b	No exemption requested
34	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.~~**	OCNGS has developed an analysis indicating that 12 months after permanent cessation of power operations, no credible accident at OCNGS will result in radiological releases requiring offsite protective actions, there is no need for the TSC, EOF, or field assessment teams. An onsite facility will continue to be maintained, from which effective direction can be given and effective control can be exercised during an emergency. OCNGS will also continue to test communication systems used to contact the State Emergency Operations Center (EOC) on an annual basis. Also refer to justification for 10 CFR 50.47(b)(3). Communication with State and local EOCs is maintained to coordinate assistance on site if required.
35	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 functions of the control room, EOF, TSC and OSC are intended to be combined into one or more locations 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

TABLE 2
EXEMPTIONS 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 emergency operations~~ facility. Such communications shall be tested monthly.	direction can be given and effective control can be exercised during an emergency. OCNGS will maintain communication with the NRC. Also refer to basis for 10 CFR 50.47(b).
36	*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; 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;~~**	viii. The number of staff at OCNGS 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. OCNGS 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 OCNGS 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. 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.

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	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~~**.	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.
37	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.	OCNGS analyses demonstrate that 12 months after permanent cessation of power operations, no remaining postulated accidents at OCNGS 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 comprehensive emergency management plan. 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)
38	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 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.	Refer to basis for 10 CFR 50.47(b). OCNGS will continue to invite the State of New Jersey and local support to participate in the periodic drills and exercises conducted to assess their ability to perform responsibilities related to an emergency at OCNGS, to the extent defined by the OCNGS 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 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

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EXEMPTIONS FROM 10 CFR PART 50, APPENDIX E

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		Emergency in power reactor site scenarios is not applicable to a decommissioning site. Exelon considers OCNGS to be exempt from 10 CFR Part 50, Appendix E, Section F.2.a.(i)-(iii) because OCNGS will be exempt from the umbrella provision of 10 CFR Part 50, Appendix E, Section IV.F.2.a.
39	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~~** impact of radiological ~~r~~eleases**~~, 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	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 OCNGS 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. OCNGS will continue to conduct biennial exercises and will invite the State of New Jersey 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 OCNGS to the extent defined by the OCNGS emergency plan.

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	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.	
40	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 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~~**	See basis for 10 CFR Part 50, Appendix E, Section IV.1 and 10 CFR Part 50, Appendix E, Section IV.F.2.a.

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Item	10 CFR PART 50, APPENDIX E, SECTION IV	Basis for Exemption
	~~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.~~**	
41	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 reactor plume exposure pathway EPZ should rotate this participation from site to site.~~**	Refer to basis for 10 CFR 50.47(b)(10).
42	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).
43	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	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.

TABLE 2
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	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.~~**	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.
44	F.2.g and F.2.h	No Exemption requested.
45	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 OCNGS 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, OCNGS 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.
46	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~~**	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). Periodic drills and exercises will be completed to demonstrate ERO proficiency in key skills necessary to implement the principal functional areas of emergency response as applicable for the permanently defueled plant status. Critiques will follow each drill or exercise activity. OCNGS will continue to invite the State of New Jersey and local support organizations to participate in the periodic drills and exercises to assess their ability to perform responsibilities related to an emergency at OCNGS to the extent defined by the OCNGS emergency plan.

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Item	10 CFR PART 50, APPENDIX E, SECTION IV	Basis for Exemption
	~~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.~~ **~~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.~~**	
47	G. Maintaining Emergency Preparedness H. Recovery	No exemptions requested.
48	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~~**	Refer to basis for 10 CFR Part 50, Appendix E, Section IV. 1.

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Item	10 CFR PART 50, APPENDIX E, SECTION IV	Basis for Exemption
	~~ensure the continued ability of the licensee to safely shut down the reactor and perform the functions of the licensee's emergency plan.~~	

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 OCNGS 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 OCNGS 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 OCNGS 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);	The postulated accident that will remain applicable to OCNGS and could contribute to dose upon implementation of the requested exemptions is the fuel handling accident (FHA) in the Reactor Building, where the SFP is located. The results of the analysis indicate that the dose at the EAB would not exceed the EPA PAGs within 33 days after permanent cessation of power operations (Reference 15). Exelon will maintain the version of the EPA PAGs as specified in the current and proposed OCNGS Emergency Plan. This analysis is described in Section 5.2.
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 16) 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 12 months 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);	OCNGS performed an analysis (Reference 17) to determine the offsite radiological impact of a complete loss of SFP water. It was determined that the gamma radiation dose rate at the EAB would be limited to small fractions of the EPA PAG exposure levels at 12 months after shutdown. This analysis is described in Section 5.4 and is included in Attachment 2.
4	Considering the site-specific seismic hazard, either an evaluation demonstrating a high confidence of a low-probability (less than 1 x 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).	OCNGS 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 10). The seismic evaluation included all structures including the SFP, and was prepared and submitted for NRC review. The Exelon submittal (Reference 11) 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 12. The NRC staff concluded

TABLE 3
INTERIM STAFF GUIDANCE-02 COMPARISON

Analysis	ISG-02 Description	Response
		that the assessment was performed consistent with the NRC-endorsed (Reference 13) SFP Evaluation Guidance Report (Reference 14) 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 OCNGS 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.	OCNGS 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.(8) of the OCNGS Renewed Facility Operating 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

Fuel Handling Accident

The postulated design basis accident that will remain applicable to OCNGS in its permanently shut down and defueled condition is the FHA in the Reactor Building where the SFP is located. An analysis based on the FHA was performed to determine the dose to operators in the control room and the public at the Exclusion Area Boundary (EAB or "Site Boundary") as a function of time after shutdown. The analysis shows that the dose at the EAB 33 days after shutdown (with no credit for containment) is less than 1 rem TEDE and 5 rem Thyroid, which are the EPA PAG thresholds for recommended evacuation (Reference 15). Due to the amount of decay assumed (33 days), the results of this analysis may be applied after February 2, 2020, assuming a OCNGS shutdown by December 31, 2019.

5.3 Consequences of Beyond Design Basis Events

Hottest Fuel Assembly Adiabatic Heat-up (Zirconium Fire)

The analysis (Reference 16) is provided in Attachment 2 to compare the conditions for the hottest fuel assembly stored in the OCNGS fuel pool to a criterion proposed in SECY-99-168 "Improving Decommissioning Regulations for Nuclear Power Plants" (Reference 18), 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 degrees Celsius (°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 heatup analysis presented in Attachment 2, at 12 months (365 days) after permanent cessation of power operations (12 months decay time), 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 9), 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 heatup 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 Other Analyzed Events

Loss of Spent Fuel Pool Normal Cooling

OCNGS analyzed a drain down event of the SFP to determine a dose rate curve at the EAB and Control Room. NUREG-0586, "Final Generic Environmental Impact Statement on Decommissioning of Nuclear Facilities," (Reference 34) Supplement 1, Section 4.3.9, identifies that a SFP drain down event is a beyond design basis event. Although the analysis described in Section 5.3 above, demonstrated a significant release of radioactive material from the spent fuel is not possible in the absence of water cooling after 365 days (1 year) following permanent cessation of power operations, the potential exists for radiation exposure to an offsite individual in the event that shielding of the fuel is lost. The SFP water and the concrete pool structure serve as radiation shielding. 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. The offsite and Control Room radiological impacts of a postulated complete loss of SFP water were assessed in Calculation C-1302-226-E310-458, "Dose at Exclusion Area Boundary and Control Room Due to Shine from Drained Spent Fuel Pool During SAFSTOR" (Reference 17). It was determined that the gamma radiation dose rate at the EAB would be limited to small fractions of the EPA PAGs. 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.

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 OCNGS 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 9).

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 19). 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 OCNGS 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 9) 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 6) 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 20), 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 Comprehensive Emergency Management Plan).

The purpose of NUREG-2161 (Reference 20) 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 MWt and the SFP contained 2844 fuel assemblies. OCNGS was licensed to generate 1930 MWt, and the SFP has the capacity to hold 3035 fuel assemblies. The SFP is expected to contain 2529 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 OCNGS are lower than those in the NUREG- 2161 study.

OCNGS 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 11) 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 12. The NRC staff concluded that the assessment was performed consistent with the NRC-endorsed (Reference 13) SFP Evaluation Guidance Report (Reference 14) 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 OCNGS 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 OCNGS.

Based on the above, OCNGS has demonstrated that no credible 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).	The OCNGS design is in alignment with this description. Procedure 131 "Oyster Creek Load Lift Management Procedure," controls the handling of heavy loads to meet the guidance provided in NUREG-0612 (Reference 21). The cask handling crane (i.e. reactor building bridge crane) trolley was upgraded to address Phase I requirements of NUREG-0612 to provide redundancy in the load carrying path from the cask to the crane trolley itself, so that no single failure would allow the cask to drop. In addition to the trolley replacement, a comprehensive maintenance program and strict administrative control of all cask handling was implemented. The NRC Safety Evaluation Report for the acceptance of NUREG-0612 Phase I actions is documented in Reference 22. As documented in Reference 23 the NRC considered Phase II to be an enhancement and completed without requiring completion of implementation actions identified during the Phase II review.
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.	OCNGS procedures are in place to ensure onsite and offsite resources can be brought to bear during an event, including: • Abnormal Operating Procedure, ABN-16, "Loss of Spent Fuel Pool Cooling" • Abnormal Operating Procedure, ABN-41, "Security Event" • FSG-00, "Extended Loss of AC Power FLEX Strategy Implementation • EDMG-01-FC1, "Extensive Damage Mitigation Guidelines Flow Chart 1" • ERO activation in accordance with the OC Permanently Defueled Emergency Plan. 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 OCNGS 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. Once OCNGS is shut down and defueled, the on-shift plant operators, including Certified Fuel Handlers (CFH), and fire brigade members 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 OCNGS 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
		OCNGS CFH training program was approved by the NRC by letter dated September 6, 2016 (Reference 24). Emergency Plan drills are conducted with biennial participation of the Offsite Response Organizations to maintain proficiency in response to a plant event.
3	Procedures will be in place to establish communication between onsite and offsite organizations during severe weather and seismic events.	OCNGS maintains the following procedures to provide guidance for establishing and maintaining communications between offsite agencies and the onsite ERO during severe weather and seismic events: ● Abnormal Operating Procedure, ABN-31, "High Winds" ● Abnormal Operating Procedure, ABN-38, "Station Seismic Event" ● OC Permanently Defueled Emergency Plan, EP-OC-1001
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 resources could be obtained in a timely manner.	OCNGS has multiple portable pumps and portable emergency generators that meet Extensive Damage Mitigation Guidelines (EDMG) and FLEX requirements. These can be used as required by abnormal procedures. In addition, offsite resources are committed and implemented by: Procedure EDMG-01, "Extensive Damage Mitigation Guidelines" which provides coordination for utilizing offsite fire truck equipment. Procedure FSG-23, "Transfer of FLEX Strategy to SAFER" which provides details of resources that can be brought to the site to supplement or replace beyond design base events. CC-OC-118-1002, "SAFER Response Plan" procedure provides the contacts and logistics for the offsite organization's response.
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.	OCNGS design meets the intent of this IDC. There is a continuous temperature monitor that reads out locally without any power. It provides a high temperature alarm function in the Control Room (CR) (powered by vital DC). The temperature gage is read via dedicated video to the CR. Fuel pool level is monitored by the CR with continuous dedicated video display of surge tank level and surge tank lo and lo-lo level annunciators. Fuel pool low level annunciator in the CR is actuated by a continuous level instrument that senses level below the surge tank weir height. Additionally, there are two channels of continuous remote indication of the SFP water level indicators in the cable spreading room above the control room that have been added for reliable SFP level indication (post-Fukushima).

TABLE 4
INDUSTRY DECOMMISSIONING COMMITMENTS (IDCS)

IDC	Industry Commitments	Response
		There are two channels of continuous remote indication of Refueling Floor area radiation in the control room. Each of these channels provide high area radiation annunciation in the control room. A local alarm to notify personnel of high area radiation levels is also in place. In addition, each of these channels provides input to the plant computer.
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 OCNGS SFP gate has static seals between the inner and outer gate. There is no credible catastrophic failure mechanism for these seals. If SFP inventory were to leak due to seal rupture or degradation, level would not go below the top of the spent fuel racks. The fixed top elevation of the refueling slot between the SFP and reactor vessel where a removable refueling slot plug is placed over is at elevation 94'-9". The top elevation of a spent fuel rack in the SFP is 94'-6".
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.	OCNGS Procedure 311, "Fuel Pool Cooling System," allows specified volumes to be pumped or letdown from the SFP. The procedure meets the requirements of this IDC by controlling the suction and discharge points. Additionally, the ISFSI equipment design is such that there are no ISFSI related SFP operations that have the potential to cause a rapid drain down. Procedure EN-HU-106, "Procedure and Work Instruction Use and Adherence," establishes the expectations and requirements for procedure adherence and usage for all personnel performing activities. Additionally, all work activities are subject to the work process controls and integrated risk management where the activities are analyzed and managed for risk. (e.g. address SFP activities.) The OCNGS SFP has active and passive anti-siphon devices.
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.	The onsite restoration plan is in place to repair SFP cooling systems to provide maintenance to normal systems. If necessary the following procedures provide direction to add makeup or additional cooling: ABN-16, "Loss of Fuel Pool Cooling" FSG-09, "Makeup to the Fuel Pool" (FLEX) EDMG-SP-X2, "External Makeup to the Fuel Pool" (B.5.b) EDMG-SP-X11 "Spraying the Plume or Fuel Pool Using the Portable Pump and Ladder Truck" FSG-23, "Transfer of FLEX Strategy to SAFER" The multiple makeup sources from onsite and offsite that includes: • Torus water

TABLE 4
INDUSTRY DECOMMISSIONING COMMITMENTS (IDCS)

IDC	Industry Commitments	Response
		• Fire Water system • Intake water There are multiple ways to add makeup water to the SFP with or without entry to the refuel floor.
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 limitations such as restrictions on heavy load movements.	OCNGS Procedure 311, "Fuel Pool Cooling System," allows specified volumes to be pumped or letdown from the SFP. The procedure meets the requirements of this IDC by controlling the suction and discharge points. Additionally, the Independent Spent Fuel Storage Installation (ISFSI) transfer equipment design is such that there are no ISFSI related SFP operations that have the potential to cause a rapid drain down. Procedure EN-HU-106, "Procedure and Work Instruction Use and Adherence," establishes the expectations and requirements for procedure adherence and usage for all personnel performing activities. Additionally, all work activities are subject to the work process controls and integrated risk management where the activities are analyzed and managed for risk. (e.g. address SFP activities.) Heavy loads requirements are controlled under the Procedure 131 "Oyster Creek Load Lift Management Procedure." Fuel moves and heavy load moves that could affect the safe handling and storage of nuclear fuel require approval by the Shift Manager.
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.	OCNGS has multiple systems and sources to provide alternate makeup to the fuel pool. There are electric-driven fire pumps and a diesel-driven fire pumps that can supply makeup water to the SFP via the SW system or the Fire Water system. The OCNGS fire protection program provides controls for operation with equipment out of service and periodic functionality testing. OCNGS also has two diesel driven engine emergency makeup pumps capable of taking suction from the intake water to satisfy FLEX requirements. Additionally, a B.5.b diesel driven pump that also takes suction from the intake can be aligned for SPF makeup or cooling. The B.5.b and FLEX systems provide 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 OCNGS design aligns with the intent of this description. The OCNGS SFP cooling system design is based, in part, on Regulatory Guide 1.13, ""Spent Fuel Storage Facility Design Basis," which included, in part, a Seismic Category I makeup system to add coolant to the SFP. The design basis requirement for SFP cooling is provided which is a Nuclear Safety Design Class I system (i.e. it is designed to withstand design basis earthquake seismically induced load) protected by a Nuclear Safety Design Class I structure. The SFP cooling system heat exchangers are cooled by Reactor Building Closed Cooling Water (RBCCW), which has redundant pumping capacity and is provided by redundant power sources adequate to provide makeup at the required capacity. The RBCCW pumps are normally powered from offsite power, but can be supplied from an alternate reliable power source. RBCCW is cooled by either SW or ESW ultimate heat sink systems. The stations design also includes an electric-driven fire pump and a diesel-driven fire pump, both of which will be maintained until all fuel is removed from the SFP. Each fire pump has the capability to deliver 500 gallons per minute (gpm) of makeup water to the SFP.
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 OCNGS performs a walk-down of SFP systems once per day due to dose considerations associated with an operating reactor. The frequency of these walk-downs may be increased following final plant shutdown and permanent defueling of the reactor. There are other methods available in the control room to alert operators to potential SFP events, such as annunciators and level indication. OCNGS procedures meet the requirements of this SDA by providing the guidance on the capability and availability of onsite and offsite makeup sources. ABN-38, "Stations Seismic Event," directs the inspection of the SFP and cooling systems following a seismic event. Procedure 311 "Fuel Pool Cooling System" and ABN-16 "Loss of Fuel Pool Cooling", establishes multiple makeup sources from onsite and offsite that includes: Fire Water system FSG-09 – "Makeup to the Fuel Pool with FLEX" EDMG-SP-X2 – "External Makeup to the Fuel Pool using the B.5.b. Portable Pump" Prior to final shutdown, OCNGS will establish the timelines required to initiate the various onsite and offsite SFP makeup sources based on expected system configurations and availability.

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.	OCNGS design meets the intent of this SDA. There is a continuous temperature monitor that reads out locally without any power. It provides a high temperature alarm function in the Control Room (CR) (powered by vital DC). The temperature gage is read via dedicated video to the CR. Fuel pool level is monitored by the CR with continuous dedicated video display of surge tank level and surge tank 'Lo' and 'Lo-Lo' level annunciators. Fuel pool 'Lo Level' annunciator in the CR is actuated by a continuous level instrument that senses level below the surge tank weir height. Additionally, there are two channels of continuous remote indication of the SFP water level indicators in the cable spreading room above the control room that have been added for reliable SFP level indication (post-Fukushima). OCNGS has procedures in place to respond to an abnormally 'Lo Level' in the SFP to direct the plant staff to take appropriate actions to provide the necessary SFP makeup; first through normal means, then by utilizing all available onsite resources, including both design basis and defense- in-depth capabilities. Refer to the OCNGS responses for IDC 2 and IDC 4 for details associated with calling in offsite resources. Regarding the declaration of a general emergency, OCNGS will be employing Shutdown 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, 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 OCNGS SFP design is consistent with this SDA. • The OCNGS normal SFP cooling system suction lines are from weir feed surge tanks. Weir lip is at 118'-2" which is approximately 24' above Top of Active Fuel (TAF). • The OCNGS return line is protected from siphoning by check valves and passive vacuum break holes. This is seismically rated piping. • The OCNGS Fuel Pool lowest drain path is via the 3" drain line located between in the inboard and outboard SFP gates, located at elevation 94'-6" which is located approximately 24' below normal SFP water level. Drain-down to this elevation, which is 3" above the top of the spent fuel racks, prevents uncover of fuel. However, this path requires a gross failure of the inboard gate sealing gasket as well as failure of the 3" drain line.

TABLE 5
STAFF DECOMMISSIONING ASSUMPTIONS (SDAS)

SDA	Staff Assumptions	Response
		Therefore, this drain path is not considered to be a credible failure mode for inventory loss given the assumption that inventory loss is not the result of catastrophic failures.
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 confidence in the facilities ability to withstand a heavy load drop.	The OCNGS design is in alignment with this description. The OCNGS heavy loads program, Procedure 131 "Oyster Creek Load Lift Management Procedure," controls the handling of heavy loads to meet the guidance provided in NUREG-0612. The cask handling crane (i.e. reactor building bridge crane) trolley was upgraded to address Phase I requirements of the NUREG to provide redundancy in the load carrying path from the cask to the crane trolley itself, so that no single failure would allow the cask to drop. In addition to the trolley replacement, a comprehensive maintenance program and strict administrative control of all cask handling was implemented via implementation of. The NRC Safety Evaluation Report for the acceptance of NUREG-0612 Phase I actions is documented in Reference 22.
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).	OCNGS 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 11) 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 12. The NRC staff concluded that the assessment was performed consistent with the NRC-endorsed (Reference 13) SFP Evaluation Guidance Report (Reference 14) 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 OCNGS has high density spent fuel racks that utilize two types of neutron poison. Ten BORAFLEX racks and four BORAL racks of similar design. As described in Section 9.1.2.3.9.1 of the OCNGS UFSAR, an aging management program is in place to manage loss of material and reduction of neutron absorption capacity of BORAFLEX neutron absorption panels in the spent fuel racks. The loss of material and the reduction of the neutron-absorbing capacity will be determined through in-situ testing and NRC accepted RACKLIFE modeling (Reference 25). BORAL performance is assessed by the surveillance program that utilizes test coupons.

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.

As discussed in this request, revised radiological analyses have been developed that show that, 33 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 EAB. In addition, analyses have been developed for beyond design basis events related to the SFP which show that, 12 months 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 exclusion area boundary (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 35 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 OCNGS's ability to physically secure the site or protect special nuclear material. Physical security measures at OCNGS 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 OCNGS 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 OCNGS 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, 33 days after shutdown, the radiological consequences of design basis accidents will not exceed the limits of the EPA PAGs at the EAB. In addition, analyses have been developed for beyond design basis events related to the SFP which show that, 12 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. 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 (Reference 16).

Since the underlying purposes of the rules would continue to be achieved even with OCNGS 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 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), and Zion 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 OCNGS 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 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 ENO for VY (Reference 26); to Southern California Edison Company for SONGS, Units 1, 2, and 3 (Reference 27); to Duke Energy Florida, Inc. for CR3 (Reference 28); and to Dominion Energy Kewaunee, Inc. for KPS (Reference 29). 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.

Exelon proposes that OCNGS should not be required to plan for an offsite impact resulting from hostile action because (1) the facility poses a lower radiological risk to the public than does a power reactor, and (2) the facility has a low likelihood of a postulated accident resulting in radiological releases requiring offsite protective measures.

Additionally, the specific exemption request for the regulation that involves a shift staffing analysis is consistent with the exemption approved by the NRC for a shutdown facility with an Independent Spent Fuel Storage Installation by letter dated March 18, 2013 (Reference 30).

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 amendment.

   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 OCNGS emergency response organization commensurate with the reduction in consequences of radiological events that will be possible at OCNGS 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. Letter from Keith R. Jury, Exelon Generation Company, LLC to U.S. Nuclear Regulatory Commission, "Permanent Cessation of Operations at Oyster Creek Nuclear Generating Station," dated January 7, 2011 (ADAMS Accession No. ML110070507)
3. 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
4. NEI 99-01, Revision 6 "Methodology for the Development of Emergency Action Levels for Non-Passive Reactors," dated November 2012 (ADAMS Accession No. ML12326A805).
5. 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)
6. U.S. Environmental Protection Agency, EPA 400-R-92-001, "Manual of Protective Action Guides and Protective Actions for Nuclear Incidents," dated October 1991 (reprinted May 1992)
7. Federal Register Notice, Vol. 76, No. 226 (76 FR 72560), Enhancements to Emergency Preparedness Regulations, dated November 23, 2011
8. U.S. Nuclear Regulatory Commission, Bulletin 2005-002, "Emergency Preparedness and Response Actions for Security-Based Events," dated July 18, 2005 (ADAMS Accession No. ML051740058)
9. NUREG-1738, "Technical Study of Spent Fuel Accident Risk at Decommissioning Nuclear Power Plants," dated February 2001 (ADAMS Accession No. ML010430066)
10. 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. 12073A348)
11. 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. ML14090A241)

12. Letter from U.S. Nuclear Regulatory Commission to Mr. Bryan C. Hanson (Exelon Generation Company, LLC), "Oyster Creek Nuclear Generating Station - 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. MF5257)," dated February 17, 2016 (ADAMS Accession No. ML15350A353)
13. 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)
14. EPRI, "Seismic Evaluation Guidance: Spent Fuel Pool Integrity Evaluation," Electric Power Research Institute Technical Update 3002007148, dated February 2016 (ADAMS Accession No. ML16055A021)
15. C-1302-226-E310-460, EAB, LPZ, and CR Dose Due to Fuel Handling Accident (FHA) - Post Cessation of Power Operations, dated August 9, 2017
16. C-1302-226-E310-457, "Oyster Creek Nuclear Generating Station Zirconium Fire Analysis for Drained Spent Fuel Pool," dated June 19, 2017
17. C-1302-226-E310-458, Dose at Exclusion Area Boundary and Control Room Due to Shine from Drained Spent Fuel Pool During SAFSTOR," dated June 16, 2017
18. 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)
19. 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)
20. 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)
21. U.S. Nuclear Regulatory Commission, NUREG-0612, "Control of Heavy Loads at Nuclear Power Plants." dated January 1980 (ADAMS Accession No. ML070205180)
22. U.S. Nuclear Regulatory Commission Letter to OCNGS, "Control of Heavy Loads (Phase I) – NUREG-0612 – Oyster Creek Nuclear Generating Station," dated June 21, 1983, Docket No. 50-219/LS05-83-06-045
23. U.S. Nuclear Regulatory Commission Letter to OCNGS, "Completion of Phase II of Control of Heavy Loads at Nuclear Power Plants' NUREG-0612. (Generic Letter 85-11)," dated June 28, 1985 (ADAMS Accession No. ML031150689)
24. Letter from U.S. Nuclear Regulatory Commission to Mr. Bryan C. Hanson (Exelon

Generation Company, LLC), "Oyster Creek Nuclear Generating Station; Clinton Power Station, Unit No. 1; and Quad Cities Nuclear Power Station, Units 1 and 2 - Approval of Certified Fuel Handler Training and Retraining Program (CAC NOS. MF8109, MF8138, MF8139, and MF8140)," dated July 18, 2016 (ADAMS Accession No. ML16200A236)

25. U.S. Nuclear Regulatory Commission, NUREG-1875, "Safety Evaluation Report Related to the License Renewal of Oyster Creek Generating Station," dated March 2007 (ADAMS Accession No. ML071310246)
26. 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
27. 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
28. 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
29. Federal Register Notice, Vol. 79, No. 214 (79 FR 65715), Dominion Energy Kewaunee, Inc.; Kewaunee Power Station, Exemption; issuance, dated November 5, 2014
30. Letter from USNRC to Connecticut Yankee Atomic Power Company, "Response to Exemption Request for Portions of Title 10 of the Code of Federal Regulations Part 50 Appendix E, and Title 10 of the Code of Federal Regulations Part 50.47 for the Haddam Neck Plant (TAC No. L24663)," dated March 18, 2013 (ADAMS Accession No. ML13064A374)
31. 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)
32. 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)
33. U.S. Nuclear Regulatory Commission, NUREG-0696, "Functional Criteria for Emergency Response Facilities," dated February 1981 (ADAMS Accession No. ML051390358)
34. NUREG-0586, "Final Generic Environmental Impact Statement on Decommissioning of Nuclear Facilities," dated October 2002
35. New Jersey Department of Environmental Protection Administrative Order No. 2011-06, dated May 6, 2011

<u>Attachment 2</u>

OYSTER CREEK NUCLEAR GENERATING STATION

DOCKET NUMBERS 50-219 & 72-15 / LICENSE NUMBER DPR-16

OYSTER CREEK NUCLEAR GENERATING STATION ZIRCONIUM FIRE ANALYSIS FOR DRAINED SPENT FUEL POOL (CALCULATION C-1302-226-E310-457)

Design Analysis		Last Page No. [6] 15	
Analysis No.: [1] C-1302-226-E310-457		Revision: [2] 0 Major ☒	Minor ☐
Title: [3] Oyster Creek Nuclear Generating Station Zirconium Fire Analysis for Drained Spent Fuel Pool			
EC/ECR No.: [4] 619639		Revision: [5] 0	

Station(s): [7]	OCNGS	Component(s): [14]	
Unit No.: [8]	1	N/A	
Discipline: [9]	A5352NESDM		
Descrip. Code/Keyword: [10]	N/A		
Safety/QA Class: [11]	Non-Safety Related		
System Code: [12]	226		
Structure: [13]	SFP		

CONTROLLED DOCUMENT REFERENCES [15]

Document No.:	From/To	Document No.:	From/To
C-1302-226-E310-458, R0	From		
GEH-0000-0118-3544, R1	From		

Is this Design Analysis Safeguards Information? [16] Yes ☐ No ☒ If yes, see SY-AA-101-106

Does this Design Analysis contain Unverified Assumptions? [17] Yes ☐ No ☒ If yes, ATI/AR#: ____

This Design Analysis SUPERCEDES: [18] N/A in its entirety.

Description of Revision (list changed pages when all pages of original analysis were not changed): [19]
Original Revision

Preparer: [20] Blake Holton (Print Name) — Blake Holton (Sign Name) — 6/15/17 (Date)

Method of Review: [21] Detailed Review ☒ Alternate Calculations (attached) ☐ Testing ☐

Reviewer: [22] Dwayne Blaylock (Print Name) — [signature] (Sign Name) — 6/15/17 (Date)

Review Notes: [23] Independent review ☒ Peer review ☐

(For External Analyses Only)
External Approver: [24] Joanne G. Morris (Print Name) — Joanne G. Morris (Sign Name) — 6/15/17 (Date)

Exelon Reviewer: [25] Robert Csillag (Print Name) — Robert Csillag (Sign Name) — 6/19/17 (Date)

See Pg 3 of Owner's Acceptance checklist for comments.

Independent 3rd Party Review Reqd? [26] Yes ☐ No ☒

Exelon Approver: [27] * Herbert Tritt (Print Name) — [signature] (Sign Name) — 6/19/17 (Date)

* SMOE, F.H. Ray, has authorized Herbert Tritt to approve engineering products as D.E.M.

Attachment 2

Owner's Acceptance Review checklist for External Design Analysis

Page 1 of 3

Design Analysis No.: C-1302-226-E310-457 Rev: 0
Contract #: 00511303 Release #: 00368

No	Question	Instructions and Guidance	Yes / No / N/A
1	Do assumptions have sufficient documented rationale?	All Assumptions should be stated in clear terms with enough justification to confirm that the assumption is conservative. For example, 1) the exact value of a particular parameter may not be known or that parameter may be known to vary over the range of conditions covered by the Calculation. It is appropriate to represent or bound the parameter with an assumed value. 2) The predicted performance of a specific piece of equipment in lieu of actual test data. It is appropriate to use the documented opinion/position of a recognized expert on that equipment to represent predicted equipment performance. Consideration should also be given as to any qualification testing that may be needed to validate the Assumptions. Ask yourself, would you provide more justification if you were performing this analysis? **If** yes, the rationale is likely incomplete.	☒ ☐ ☐
2	Are assumptions compatible with the way the plant is operated and with the licensing basis?	Ensure the documentation for source and rationale for the assumption supports the way the plant is currently or will be operated post change and they are not in conflict with any design parameters. If the Analysis purpose is to establish a new licensing basis, this question can be answered yes, if the assumption supports that new basis.	☒ ☐ ☐
3	Do all unverified assumptions have a tracking and closure mechanism in place?	**If** there are unverified assumptions without a tracking mechanism indicated, **then** create the tracking item either through an ATI or a work order attached to the implementing WO. Due dates for these actions need to support verification prior to the analysis becoming operational or the resultant plant change being op authorized.	☐ ☐ ☒
4	Do the design inputs have sufficient rationale?	The origin of the input, or the source should be identified and be readily retrievable within Exelon's documentation system. If not, then the source should be attached to the analysis. Ask yourself, would you provide more justification if you were performing this analysis? **If** yes, the rationale is likely incomplete.	☒ ☐ ☐
5	Are design inputs correct and reasonable with critical parameters identified, if appropriate?	The expectation is that an Exelon Engineer should be able to clearly understand which input parameters are critical to the outcome of the analysis. That is, what is the impact of a change in the parameter to the results of the analysis? If the impact is large, then that parameter is critical.	☒ ☐ ☐
6	Are design inputs compatible with the way the plant is operated and with the licensing basis?	Ensure the documentation for source and rationale for the inputs supports the way the plant is currently or will be operated post change and they are not in conflict with any design parameters.	☒ ☐ ☐

Attachment 2

Owner's Acceptance Review checklist for External Design Analysis

Page 2 of 3

Design Analysis No.: C-1302-226-E310-457 Rev: 0

No	Question	Instructions and Guidance	Yes / No / N/A
7	Are Engineering Judgments clearly documented and justified?	See Section 2.13 in CC-AA-309 for the attributes that are sufficient to justify Engineering Judgment. Ask yourself, would you provide more justification if you were performing this analysis? **If** yes, the rationale is likely incomplete.	☐ ☐ ☒
8	Are Engineering Judgments compatible with the way the plant is operated and with the licensing basis?	Ensure the justification for the engineering judgment supports the way the plant is currently or will be operated post change and is not in conflict with any design parameters. **If** the Analysis purpose is to establish a new licensing basis, **then** this question can be answered yes, if the judgment supports that new basis.	☐ ☐ ☒
9	Do the results and conclusions satisfy the purpose and objective of the Design Analysis?	Why was the analysis being performed? Does the stated purpose match the expectation from Exelon on the proposed application of the results? **If** yes, **then** the analysis meets the needs of the contract.	☒ ☐ ☐
10	Are the results and conclusions compatible with the way the plant is operated and with the licensing basis?	Make sure that the results support the UFSAR defined system design and operating conditions, or they support a proposed change to those conditions. If the analysis supports a change, are all of the other changing documents included on the cover sheet as impacted documents?	☒ ☐ ☐
11	Have any limitations on the use of the results been identified and transmitted to the appropriate organizations?	Does the analysis support a temporary condition or procedure change? Make sure that any other documents needing to be updated are included and clearly delineated in the design analysis. Make sure that the cover sheet includes the other documents where the results of this analysis provide the input.	☐ ☐ ☒
12	Have margin impacts been identified and documented appropriately for any negative impacts (Reference ER-AA-2007)?	Make sure that the impacts to margin are clearly shown within the body of the analysis. If the analysis results in reduced margins ensure that this has been appropriately dispositioned in the EC being used to issue the analysis.	☐ ☐ ☒
13	Does the Design Analysis include the applicable design basis documentation?	Are there sufficient documents included to support the sources of input, and other reference material that is not readily retrievable in Exelon controlled Documents?	☒ ☐ ☐
14	Have all affected design analyses been documented on the Affected Documents List (ADL) for the associated Configuration Change?	Determine if sufficient searches have been performed to identify any related analyses that need to be revised along with the base analysis. It may be necessary to perform some basic searches to validate this.	☒ ☐ ☐
15	Do the sources of inputs **and** analysis methodology used meet committed technical and regulatory requirements?	Compare any referenced codes and standards to the current design basis and ensure that any differences are reconciled. If the input sources **or** analysis methodology are based on an out-of-date methodology or code, additional reconciliation may be required if the site has since committed to a more recent code	☒ ☐ ☐

Attachment 2
Owner's Acceptance Review checklist for External Design Analysis
Page 3 of 3

Design Analysis No.: C-1302-226-E310-457 Rev: 0

No	Question	Instructions and Guidance	Yes / No / N/A
16	Have vendor supporting technical documents and references (including GE DRFs) been reviewed when necessary?	Based on the risk assessment performed during the pre-job brief for the analysis (per HU-AA-1212), ensure that sufficient reviews of any supporting documents not provided with the final analysis are performed.	☐ ☐ ☒
17	Do operational limits support assumptions and inputs?	Ensure the Tech Specs, Operating Procedures, etc. contain operational limits that support the analysis assumptions and inputs.	☐ ☐ ☒

Create an SFMS entry as required by CC-AA-4008. SFMS Number: 59199

Exelon Reviewer Comments:

An HU-AA-1212 pre-job brief for owner's acceptance was held on 3/15/17 with the DEM and Exelon acceptance reviewer. The overall risk ranking was a '1,' therefore, existing in-process reviews are sufficient. An ITPR is not required. The brief did identify that additional technical expertise was required as allowed by CC-AA-103-1003 in the form of a review committee. Technical experts in Radiological Engineering (Jack McCarthy), Nuclear Fuels (Jill Fisher), Reactor Engineering (Jim Frank), and Oyster Creek Emergency Plan (Jim Frank) were utilized to support the Exelon owner's acceptance review. The Exelon acceptance reviewer (Robert Csillag) acted as chair of this committee and coordinated all Exelon reviews performed per CC-AA-103-1003, Attachment 2 and resolution of all comments.

Enercon was verified to be on the approved vendor list as an EOC per CC-AA-12, Attachment 1, therefore, a design review by Exelon is not required.

Design qualifications were verified to be current for the Exelon owner's acceptance reviewer as was verification of being part of the ESP population.

Critical inputs and assumptions were scrutinized as was the veracity of the conclusions. Comments were supplied by the Exelon owner review committee. All comments were resolved to the satisfaction of the Exelon owner's review committee by the EOC.

Exelon Reviewer: Robert Csillag (See signature and date on design analysis cover sheet).

Contents

1.0 Purpose and Scope

The purpose of this calculation is to conservatively evaluate the length of time it takes for an uncovered spent fuel assembly in the spent fuel pool to reach the temperature where the zirconium cladding would fail. This analysis supports decommissioning of Oyster Creek Nuclear Generating Station (OCNGS). Specifically, this analysis will be used to support LAR submittal once the hottest fuel assembly decay time is sufficient and is demonstrated to reach 900°C in 10 hours which supports the requirements of ISG-02, Section 5, Item 2 (Ref. 9).

The number of hours it takes for the fuel to heat up (the heat-up time) is determined as a function of the decay time after shutdown. The heat load from a GNF2 bundle is used in this analysis as determined in Attachment 8 of Reference 1.

NUREG-0586 Supplement 1, Section 4.3.9, identifies that a spent fuel pool drain down event is beyond design basis (Reference 4.1). This calculation is non-safety related as it is beyond design basis in support of SAFSTOR.

> “Radiological accidents considered in licensing nuclear power plants are classified as design basis accidents (DBAs) and severe (beyond design basis) accidents. DBAs are those accidents that both the licensee and the NRC staff evaluate to ensure that the plant can withstand normal and abnormal transients and a broad spectrum of postulated accidents without undue hazard to the health and safety of the public. Severe accidents are those that are beyond the design basis of the plant. They are more severe than DBAs because they may result in substantial damage to the fuel, whether or not there are serious offsite consequences. For the most part, DBAs focus on reactor operation and are not applicable to plants undergoing decommissioning. The only DBAs or severe accidents (beyond design basis) applicable to a decommissioning plant are those involving the spent fuel pool. These postulated accidents are not expected to occur during the life of the plant, but are evaluated to establish the design basis for the preventive and mitigative safety systems of the spent fuel storage facility.”

2.0 Acceptance Criteria

There are no specific acceptance criteria for this analysis; however, SECY-99-168 (Ref. 8) reports that a plant specific EP exemption determined "10 hours will be sufficient time to take mitigative action" and that for BWRs, 2 years is expected to be the decay time needed to reach a 10 hour heat-up time from 30 °C to 900 °C. NUREG-1738 shows that a 10 hour heat up time for a BWR requires less than 2 years of cooling time (Ref. 7, Fig. 2-1).

NUREG/CR-6451 (Ref. 6) presents several studies discussing the maximum allowable temperature of zirconium cladding that will ensure that failure of the zirconium cladding will not occur. NUREG/CR-6451 states 565 °C (1049 °F) as the lowest temperature where incipient cladding failure might occur. Per NUREG-1738 (Ref. 7, pg. 3-7), 900°C (1652 °F) is the temperature where "runaway oxidation" (zirconium fire) is expected to occur. These two temperatures are the failure temperatures of interest for this calculation.

3.0 Assumptions

3.1 The heat-up time is conservatively assumed to start when the spent fuel pool has been completely drained. This is conservative as the drain down time (boil-off) would increase the time to cladding failure.

3.2 The starting temperature for the heat-up analysis is assumed to be uniform and 125 °F. The temperature 125 °F is the maximum initial pool temperature (Ref. 3). SECY-99-168 (Ref. 8) and NUREG-1738 (Ref. 7) both set the starting water temperature at 30 °C (86 °F) so setting the initial temperature to the maximum pool temperature is conservative because the heat-up time would be increased with a lower starting temperature.

3.3 The structural components in the fuel assembly are not credited to absorb any of the heat. The thermal mass of the channel walls is also neglected in this analysis. This is conservative because the structural components and channel walls thermal inertia would reduce the required assembly decay time.

3.4 OCNGS final cycle 27 contains 548 bundles of GNF2 fuel and 12 bundles of GE11 fuel (Ref. 13). The GNF2 bundles are limiting in terms of heat load (Ref. 2), therefore, the analysis herein will only evaluate the heat up of GNF2 fuel assemblies and not any other assembly type in the spent fuel pool because the offloaded fuel directly after a cycle contains the assemblies with the highest decay heat (referred to as the hottest fuel assembly here-in).

3.5 The specific heat for uranium dioxide and the zircaloy-2 cladding are determined at a temperature of 500°F. A temperature of 500 °F is in the temperature range for this analysis. From Reference 4 (Attachment 1), the specific heat capacity slightly increases with an increase in temperatures; at higher temperatures, the uranium dioxide and zircaloy-2 would heat up more slowly. Thus, using a temperature around or less than the midpoint for material properties is conservative with respect to the assembly heat-up. This temperature is used as representative of the full temperature range in this analysis.

3.6 This analysis conservatively assumes that there is no air cooling of the assemblies (i.e., adiabatic conditions): the flow paths that would provide natural circulation cooling are assumed to be blocked.

3.7 The mass of uranium dioxide used in the analysis does not include gadolinium. The mass of zircaloy-2 used in the analysis does not include the tie plates. This is conservative as it neglects thermal mass in the bundle which would increase the heat-up time.

3.8 The diameter of the water rods is based on the diameter in Zone 2, as this is the zone where the heated length of rods are located (Reference 12, Table 6).

3.9 Only the heated length of the rod is analyzed. This is conservative as it neglects thermal mass in the fuel bundle which would increase the heat-up time.

4.0 References

1. C-1302-226-E310-458, Rev. 0, "Dose at Exclusion Area Boundary and Control Room Due to Shine from Drained Spent Fuel Pool During SAFSTOR".
2. TODI NF172795, Rev.0, "Transmit GNF2 Dimensional Data for Oyster Creek".
3. OCNGS Technical Specification 5.3.1, Amendment No. 223.
4. Glasstone and Sesonske, "Nuclear Reactor Engineering", Van Nostrand Reinhold Company, 1981 (Attachment 1).
5. GEH-0000-0118-3544, Rev. 1, "GNF2 Fuel Design Cycle-Independent Analyses for Exelon Oyster Creek Generating Station".
6. NUREG/CR-6451, "A Safety and Regulatory Assessment of Generic BWR and PWR Permanently Shutdown Nuclear Power Plants", August 1997 (ML082260098).
7. NUREG-1738, "Technical Study of Spent Fuel Pool Accident Risk at Decommissioning Nuclear Power Plants", February 2001 (ML010430066).
8. SECY-99-168, "Improving Decommissioning Regulations for Nuclear Power Plants", June 30, 1999 (ML992800087).
9. NSIR/DPR-ISG-02, "Emergency Planning Exemption Requests for Decommissioning Nuclear Power Plants", May 11, 2015.
10. ORNL/TM-2005/39, SCALE 6.1, User's Manual, June 2011.
11. Incropera & DeWitt, "Fundamentals of Heat and Mass Transfer", Third Edition, John Wiley & Sons, Inc., 1990.
12. Oyster Creek Design Analysis DB-0011.03, Revision 1, "GNF2 Design Basis".
13. TODI NF172711, Rev. 0, "Oyster Creek Unit 1 cycle 27 Preliminary EOC E-Plan Calculation Inputs-December 31st Shutdown".

5.0 Input Data

5.1 Zirconium Properties

The cladding for the GNF2 fuel is zircaloy-2 (Ref. 5). The specific heat capacity of zircaloy-2 at 500°F (533 K) (Assumption 3.5) is 0.0761 Btu/ lbm-°F (Ref. 5, Table 10-5) and the density of zircaloy-2 is 6.56 g/cm^3 (409.53 lb/ft^3) (Ref. 10).

5.2 Uranium Properties

The specific heat capacity of uranium dioxide at 500 °F (533 K) (Assumption 3.5) is 0.0683 Btu/ lbm-°F (Ref. 5, Table 10-5). The mass of uranium dioxide in the fuel bundle is 457.25 lbs (Ref. 5, Table 10-5).

5.3 Geometry for Limiting Assemblies

The table below shows the geometry inputs for the GNF2 fuel bundles evaluated in this analysis (Assumption 3.4). Table 1 contains fuel assembly input data for a GNF2 fuel bundle.

Table 1: Fuel Bundle Inputs for GNF2 Fuel

Number of Heated Rods	92 rods	Reference 5
Number of Water Rods	2 rods	Reference 5
Number of 2/3 Length Part Length Rod	8 rods	Reference 5
Number of 3/8 Length Part length Rod	6 rods	Reference 5
Heated Length of 2/3 Part Length Rod	102 inches	Reference 2 and 5
Heated Length of 3/8 Part Length Rod	54 inches	Reference 2 and 5
Outer Diameter of Water Rods	0.980 inches	Reference 2
Inner Diameter of Water Rods	0.920 inches	Reference 2
Outer Diameter of Cladding	0.404 inches	Reference 2
Inner Diameter of Cladding	0.3567 inches	Reference 2
Heated Length of Full Length Rods	145.24 inches	Reference 2

5.4 Heat Load

Attachment 8 of Reference 1 determines the maximum heat load from a single fuel assembly. The assembly with the highest heat load will have the shortest heat-up time. The table showing the maximum fuel assembly heat generation rates for several years is located in Table A8-2 of Reference 1 and is reproduced in Table 2 below.

Table 2: Decay Heat Source Terms from ORIGEN-ARP

Decay Time	1 Year Decay (W/MTU)	1.25 Year Decay (W/MTU)	1.5 Year Decay (W/MTU)	2 Year Decay (W/MTU)	3 Year Decay (W/MTU)	5 Year Decay (W/MTU)
Cycle 27, max. burnup, min. enrichment, max. MTU	9.27E+03	7.93E+03	6.93E+03	5.50E+03	3.86E+03	2.56E+03

The worst-case (hottest) bundle is one that was discharged at the end of Cycle 27 and has been cooling for one year. From Table 2, it has a heat load of 9.27E+03 W/MTU. The maximum 0.1820 MTU/assembly value was derived from Cycle 27 data (Reference 1). The worst-case heat per assembly is calculated as follows:

$$Worst-case\ bundle\ heat\ load = \frac{9.27E+03\ W}{MTU} \times \frac{0.1820\ MTU}{assembly}$$
$$= 1687.14\ \frac{W}{assembly} = 5{,}756.66\ \frac{BTU/hr}{assembly}$$

The worst case bundle heat load is determined at the remaining decay times (1.25 years, 1.5 years, etc), using the same methodology.

6.0 Identification of Computer Codes

N/A

7.0 Method of Analysis

This analysis determines the heat-up time of the fuel assembly using the thermal capacity of materials.

Equation 7-1 (Ref. 11, Ch. 8):

$$\dot{q} = m \times c_p \times \frac{\Delta T}{t}$$

Where:

$\dot{q}$ is the heat generation rate in BTU/hr
m is the mass of material in lbm. (=ρV)
c_p is the specific heat in BTU/lb- °F
ΔT is the temperature increase in °F
t is the heat-up time in hr
ρ is density in lbm/ft^3
V is volume in ft^3

For this analysis, there are two materials that are considered: the uranium dioxide fuel pellets and the zircaloy-2 cladding. The zircaloy-2 is in the cladding and the water tubes, which are also being heated. Under adiabatic conditions, zircaloy-2 and the uranium dioxide are modeled as heating up at the same rate, so the $\frac{\Delta T}{t}$ will be the same for both materials.

Equation 7-2:

$$\dot{q} = \frac{\Delta T}{t} \times \left(m_u \times c_{p,u} + \rho_z \times V_z \times c_{p,z}\right)$$

Where:

X_u signifies the property is for uranium dioxide
X_z signifies the property is for zirconium

This calculation seeks the heat-up time, so Equation 7-2 is solved for t.

Equation 7-3:

$$t = \frac{\Delta T}{\dot{q}} \times \left(m_u \times c_{p,u} + \rho_z \times V_z \times c_{p,z}\right)$$

The mass of uranium is given in Reference 5 so the volume of uranium dioxide does not need to be determined.

The volume of zircaloy-2 in the heated rods, and water rods, are given below. The length of the cladding, and water rods, that are heated is conservatively modeled as being the same as the heated length of uranium dioxide. In reality, they are longer than the length of the uranium dioxide pellets.

Equation 7-4:

$$V_{z,cl} = \left(\left(\pi \times \frac{{D_{c,o}}^2 - {D_{c,i}}^2}{4}\right) N_{FL} \times L_{FL}\right) + \left(\left(\pi \times \frac{{D_{c,o}}^2 - {D_{c,i}}^2}{4}\right) N_{\left(\frac{2}{3}\right)} \times L_{\left(\frac{2}{3}\right)}\right) + \left(\left(\pi \times \frac{{D_{c,o}}^2 - {D_{c,i}}^2}{4}\right) N_{\left(\frac{3}{8}\right)} \times L_{\left(\frac{3}{8}\right)}\right)$$

Where:

$V_{z,cl}$ is the volume of zircaloy-2 in the cladding of heated tubes
$D_{c,o}$ is the outer diameter of the cladding
$D_{c,i}$ is the inner diameter of the cladding

Equation 7-5:

$$V_{z,wr} = \left(\pi \times \frac{{D_{w,o}}^2 - {D_{w,i}}^2}{4}\right) N_{wr} \times L$$

Where:

$V_{z,wr}$ is the volume of zircaloy-2 in the water rods
$D_{w,o}$ is the outer diameter of the water rods
$D_{w,i}$ is the inner diameter of the water rods
N_{wr} is the number of water rods

Equation 7-6:

$$V_z = V_{z,cl} + V_{z,wr}$$

The temperature increase (ΔT) for this analysis is taken to be from the initial temperature of the pool, 125°F (Assumption 3.2), to the zirconium cladding failure temperatures of interest, 1049°F (565°C) and 1652°F (900°C) (Acceptance Criteria, Section 2).

The heat-up time is calculated as a function of the decay time for each of the times in Section 5.4.

The hottest assembly source term methodology is described in Attachment 8 of Reference 1.

8.0 Numeric Analysis

The volume of zircaloy-2 in the cladding is determined below using Equation 7-4:

$$V_{z,cl} = \left(\left(\pi \times \frac{(\frac{0.404\ in}{12\ in/ft})^2 - (\frac{0.3567\ in}{12\ in/ft})^2}{4} \right) 78\ rods \times \frac{145.24\ in}{12\ in/ft} \right) + \left(\left(\pi \times \frac{(\frac{0.404\ in}{12\ in/ft})^2 - (\frac{0.3567\ in}{12\ in/ft})^2}{4} \right) 8\ rods \times \frac{102\ in}{12\ in/ft} \right) + \left(\left(\pi \times \frac{(\frac{0.404\ in}{12\ in/ft})^2 - (\frac{0.3567\ in}{12\ in/ft})^2}{4} \right) 6\ rods \times \frac{54\ in}{12\ in/ft} \right) = 0.204 ft^3$$

The volume of zircaloy-2 in the water rods is determined below using Equation 7-5.

$$V_{z,wr} = \left(\pi \times \frac{(\frac{0.980\ in}{12\ in/ft})^2 - (\frac{0.920\ in}{12\ in/ft})^2}{4} \right) 2\ rods \times \frac{145.24\ in}{12\ in/ft} = 0.015\ ft^3$$

The total zircaloy-2 volume is then determined below from Equation 7-6:

$$V_z = 0.204\ ft^3 + 0.015\ ft^3 = 0.219\ ft^3$$

The heat-up time is then determined for end temperatures of 565°C (1049°F) and 900°C (1652°F) using the maximum bundle heat load at different decay times with Equation 7-3. The heat-up time to 1049°F for the one year decay maximum bundle (Section 5.4) is shown below; the heat-up for the remaining decay times is solved in the same exact manner (i.e., changing the heat load and keeping the remaining inputs constant) and the results are reported in Section 9.

$$t = \frac{(1049°F - 125°F)}{5756.66\,BTU/hr}\left(457.25\ lbm \times 0.0683\frac{BTU}{lbm - °F} + 409.53\frac{lb}{ft^3} \times 0.219\ ft^3 \times 0.0761\frac{BTU}{lbm - °F}\right) = 6.11\ hours$$

9.0 Results and Conclusions

The results are shown in Table 3.

Table 3: Results

End Temperature (°C, °F)	Decay Time (years)	Heat-Up Time (hours)
565, 1049	1	6.11
565, 1049	1.25	7.14
565, 1049	1.5	8.17
565, 1049	2	10.29
565, 1049	3	14.67
900, 1652	1	10.09
900, 1652	1.25	11.80
900, 1652	1.5	13.50
900, 1652	2	17.01
900, 1652	3	24.24

The 10 hour heat-up time to a temperature of 565°C (1049 °F) occurs at a decay time of about 2 years, which is the expected decay time to a temperature of 900°C (1652 °F) stated in SECY-99-168 (Ref. 8). The 10 hour heat-up time to a temperature of 900°C (1652°F) occurs at a decay time just before one year after shutdown, which is less than the decay time calculated in NUREG-1738 (Ref. 7, pg. 2-3). Based on the results of this analysis, EP staffing could potentially be reduced one year after shutdown.

Figure 1 below shows the heat-up time vs decay time for both of the temperatures of interest.

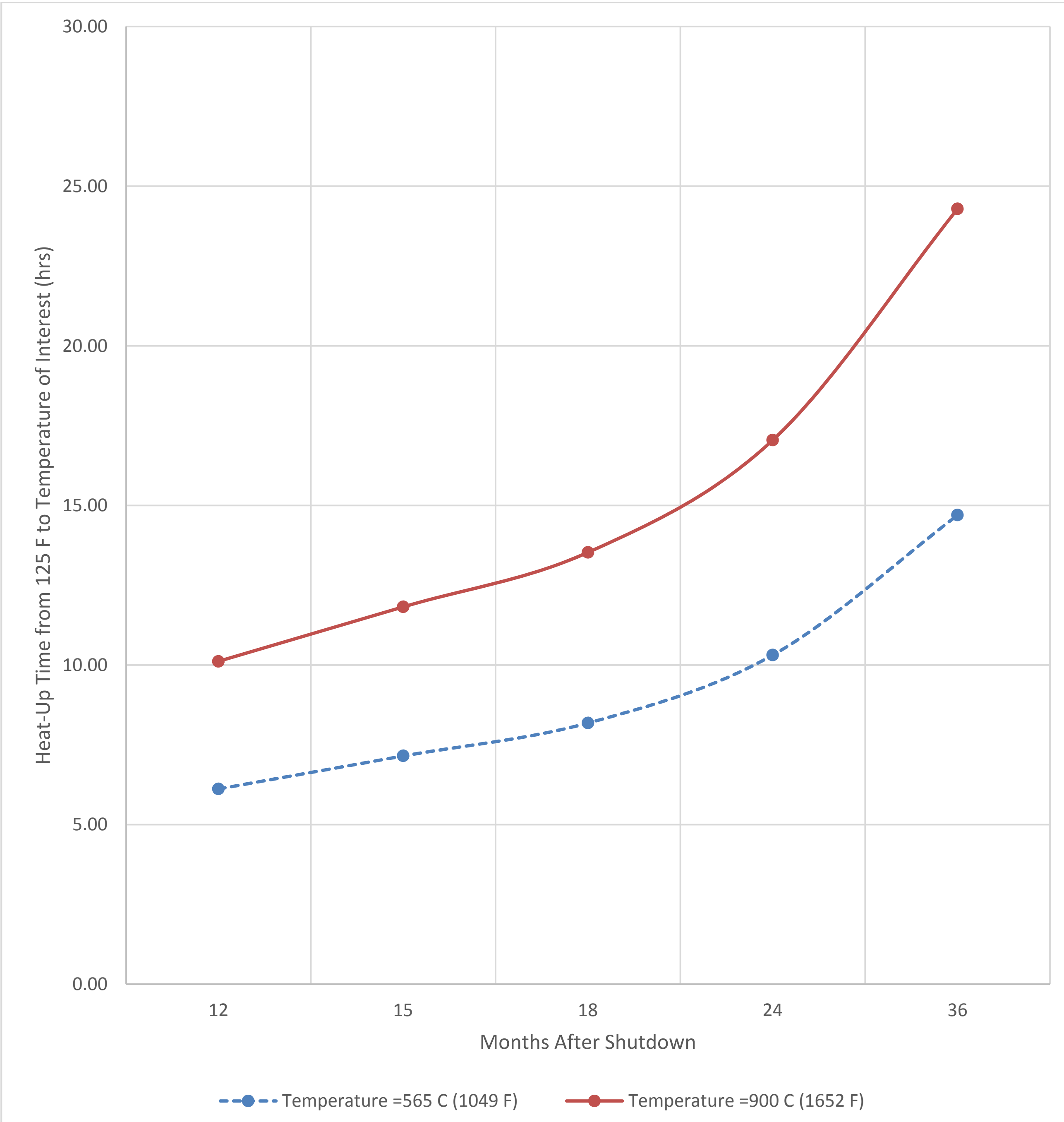

Figure 1: Heat-Up Time vs. Decay Time

TABLE A.6. PHYSICAL PROPERTIES OF SOME REACTOR MATERIALS
(Average values for preliminary calculations only; not to be used for design purposes.)

Material	Temperature (K)	Density (kg/m^3)	Coefficient of Linear Thermal Expansion* (per K × 10^6)	Specific Heat (J/kg·K × 10^{-3})	Thermal Conductivity (W/m·K)	Ultimate Tensile Strength (MPa)	Yield Strength or Compressive Strength (C) (MPa)	Young's Modulus (10^9 Pa)	Poisson's Ratio
Graphite (average nuclear grade)	300	1700	–	0.71	156	13.8	58 (C)	–	–
	500	–	3.6†	1.25	118	15.9	–	9.0	–
	800	–	5.0†	1.67	73	17.2	–	–	–
	1400	–	6.3†	1.88	36	19.3	–	–	–
	2500	–	8.5†	–	31	27.6	–	–	–
Steel, carbon (A 533-B)	300	7860	–	0.50	52	550	340	207	0.28
	600	–	10.2	0.59	43	530	280	182	–
	750	–	10.4	0.63	38	450	240	172	–
	800	–	10.4	0.67	35	390	200	169	–
Steel, stainless (type 347)	300	7950	–	0.50	14	520	210	–	–
	500	7860	16.9	0.52	17	420	–	173	0.30
	700	7710	17.4	0.55	20	400	150	166	0.31
	800	–	18.5	0.57	22	390	–	157	0.32
Uranium carbide (UC)	300	13,630	–	0.15	–	–	372 (C)	214	–
	800	–	–	–	23	–	–	–	–
	1250	–	10.3	–	23	–	–	–	–
Uranium dioxide	300	10,980	–	0.23	8.0	–	960 (C)	183	–
	500	–	9.0	0.28	6.1	–	–	–	–
	800	–	10.1	0.30	4.1	–	–	165	–
	1100	–	–	0.31	2.6	–	–	–	–
	1400	–	12.8	0.32	2.2	–	–	–	–
	2300	–	–	0.42	2.3	–	–	–	–
Zircaloy-2	300	6560	–	0.28	12.7	490	300	95	0.43
	500	–	–	0.31	15.2	280	170	90	0.38
	600	–	6.5	0.33	16.5	210	117	78	–
	800	–	–	0.35	18.9	–	–	–	–
	1000	–	–	0.37	21.6	–	–	–	–

*Average values from 20°C to indicated temperatures.
†Average of longitudinal and transverse properties.