Entergy

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BVY 14-009

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

March 14, 2014

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

SUBJECT: Request for Exemptions from Portions of 10 CFR 50.47 and 10 CFR 50, Appendix E
Vermont Yankee Nuclear Power Station
Docket No. 50-271
License No. DPR-28

REFERENCES: 1. Letter, Entergy Nuclear Operations, Inc. to USNRC, "Notification of Permanent Cessation of Power Operations," BVY 13-079, dated September 23, 2013

Dear Sir or Madam:

Pursuant to 10 CFR 50.12, Entergy Nuclear Operations, Inc. (ENO) requests exemptions from portions of 10 CFR 50.47(b), 10 CFR 50.47(c)(2) and 10 CFR 50, Appendix E for Vermont Yankee Nuclear Power Station (VY). The requested exemptions would allow VY to reduce emergency planning requirements and subsequently revise the VY Emergency Plan consistent with the anticipated permanently defueled condition of the station.

On September 23, 2013, ENO informed the NRC that VY will permanently cease operations in the fourth quarter of 2014 (Reference 1). Once VY permanently ceases operations and submits the certifications required by 10 CR 50.82(a)(1)(i) and (ii), pursuant to 10 CFR 50.82(a)(2), the 10 CFR Part 50 license for VY will no longer authorize 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 will not present an undue risk to the public health and safety, and application of the regulations in this particular circumstance is not necessary to achieve the underlying purpose of the rules. More specifically, application of the portions of the regulations from which exemptions are sought is not necessary to ensure adequate emergency response capability for VY. Furthermore, continued application of these portions of the regulations from which exemptions are sought would impose a burden on ENO and the VY Decommissioning Trust Fund by requiring continued implementation of unnecessary emergency response capability.

The exemption requests are contained in Attachment 1 to this letter. ENO has performed an analysis which shows that, within 15.4 months after shutdown, the spent fuel stored in the spent fuel pool will have decayed to the extent that the requested exemptions may be implemented at VY without any additional compensatory actions. Following the VY shutdown, which is expected by the end of 2014 (Reference 1), 15.4 months after shutdown will occur near the middle of April 2016. This analysis is contained in Attachment 2.

VY plans to submit a permanently defueled emergency plan, containing 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. The proposed emergency plan will be based on the exemptions requested herein.

ENO requests review and approval of this exemption request by December 1, 2015 with an effective date of April 15, 2016. Approval of these exemptions by December 1, 2015 will allow VY adequate time to implement changes to the emergency plan and emergency response organization by the requested effective date of April 15, 2016.

Attachment 3 of this letter contains new regulatory commitments.

Should you have any questions concerning this letter or require additional information, please contact Mr. Coley Chappell at 802-451-3374.

Sincerely,

CJW/plc

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. Vermont Yankee Maximum Cladding Temperature Analysis for an Uncovered Spent Fuel Pool with no Air Cooling
3. List of Regulatory Commitments

cc list next page

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Attachment 1

Vermont Yankee Nuclear Power Station

Request for Exemptions from Portions of 10 CFR 50.47(b), 10 CFR 50.47(c)(2) and 10 CFR Part 50, Appendix E

Vermont Yankee Nuclear Power Station
Requests for Exemptions from Portions of 10 CFR 50.47(b), 50.47(c)(2) and 10 CFR Part 50, Appendix E

I. Summary Description

Pursuant to 10 CFR 50.12 "Specific exemptions," Entergy Nuclear Operations, Inc. (ENO) requests exemptions from the following for Vermont Yankee Nuclear Power Station (VY):

- 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 establishes the elements that make up the content of emergency plans.

The requested exemptions would allow ENO to reduce emergency planning requirements and subsequently revise the VY 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 VY. However, once the station is permanently shutdown, defueled, and 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 draft Interim Staff Guidance NSIR/DPR-ISG-02, Emergency Planning Exemption Requests for Decommissioning Nuclear Power Plants, which was issued for public comment in December 2013 (Reference 18).

II. Detailed Description

By letter dated September 23, 2013 (Reference 1), pursuant to 10 CFR 50.82(a)(1)(i), ENO submitted a certification to the NRC indicating its intention to permanently cease power operations at VY at the end of the current operating cycle, which is expected to occur near the end of December 2014. ENO stated its intention to submit a supplement to Reference 1 certifying the date on which operations have ceased, or will cease, in accordance with 10 CFR 50.82(a)(1)(i) and 10 CFR 50.4(b)(8). Once fuel has been permanently removed from the reactor vessel, ENO 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 VY 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 order to allow a reduction in emergency planning requirements commensurate with the hazards associated with VY'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. ENO has performed an analysis indicating that, within 15.4 months after shutdown, the spent fuel in the spent fuel pool (SFP) will have decayed to the extent that the requested exemptions can be implemented at VY without any compensatory actions. This analysis is included in Attachment 2. Because VY expects shutdown to occur by the end of December 2014, 15.4 months after shutdown will occur near the middle of April 2016. ENO plans to submit a permanently defueled emergency plan by May 31, 2014, 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. The proposed emergency plan will be based on the exemptions requested herein. ENO requests approval of these exemption requests by December 1, 2015 with an effective date of April 15, 2016. Approval of these exemptions by December 1, 2015 will enable ENO adequate time to implement changes to the emergency preparedness program and emergency response organization by April 15, 2016.

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

ENO 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 VY 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 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 strikethrough text (i.e., those portions for which exemption is not being requested), will remain applicable to VY. Details related to specific exemption requests are provided in the Basis for Exemption column.

Table 1

Exemptions Requested from 10 CFR 50.47(b) and 50.47(c)(2)

Item #	Regulation in 10 CFR 50.47	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 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 19), the Commission responded to comments concerning offsite emergency planning for ISFSIs or an MRS and concluded that, "the offsite consequences of potential accidents at an ISFSI or a MRS [monitored retrievable storage installation] would not warrant establishing Emergency Planning Zones." 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 draft proposed rulemaking in SECY-00-0145 (Reference 20) suggested that after at least one year of spent fuel decay time, the decommissioning licensee would be able to reduce its EP program to one similar to that required for an MRS under 10 CFR 72.32(b) and additional EP reductions would occur when: (1) approximately five years of spent fuel decay time has elapsed; or (2) a licensee has demonstrated that the decay heat level of spent fuel in the pool is low enough that the fuel would not be susceptible to a zirconium fire for all spent fuel configurations. 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

Table 1

Exemptions Requested from 10 CFR 50.47(b) and 50.47(c)(2)

Item #	Regulation in 10 CFR 50.47	Basis for Exemption
		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 been approved when the specific site analyses show that at least ten hours is available from a partial drain down event where cooling of the spent fuel is not effective until the hottest fuel assembly reaches 900°C. Because ten hours allows sufficient time to initiate mitigative actions to prevent a zirconium fire in the SFP or to initiate ad hoc offsite protective actions, offsite EP plans are not necessary for these permanently defueled nuclear power plant licensees.
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.	See basis for 50.47(b).
3	10 CFR 50.47(b)(2): On-shift facility licensee responsibilities for emergency response are unambiguously defined, adequate staffing to provide initial facility accident response in key functional areas is maintained at all times, timely augmentation of response capabilities is available and the interfaces among various onsite response activities and offsite support and response activities are specified.	No exemption is 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 once VY is in the permanently defueled condition. The VY emergency plan will continue to maintain arrangements for requesting and using assistance resources from offsite support

Table 1 **Exemptions Requested from 10 CFR 50.47(b) and 50.47(c)(2)**		
Item #	**Regulation in 10 CFR 50.47**	**Basis for Exemption**
		organizations. Decommissioning power reactors present a low likelihood of any credible accident resulting in radiological releases requiring offsite protective measures because of the permanently shut down and defueled status of the reactor. An emergency operations facility is not required. The control room or other location can provide for the communication and coordination with offsite organizations for the level of support required. Also see basis for 50.47(b).
5	10 CFR 50.47(b)(4): A standard emergency classification and action level scheme, the bases of which include 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~~.	VY will adopt the Permanently Defueled Emergency Action Levels (EALs) detailed in Appendix C of Nuclear Energy Institute (NEI) 99-01, "Development of EALs for Non-Passive Reactors," Revision 6 (Reference 2), endorsed by the NRC in a letter dated March 28, 2013. No offsite protective actions are anticipated to be necessary, so classification above the Alert level will no longer be required. Also see basis for 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 followup 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 20), after approximately 1 year of spent fuel decay time [and as supported by the SFP analysis], the NRC staff believes 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. The spent fuel scoping study (Reference 3) 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 of mitigation measures are successful, releases could only occur during the first several days after the fuel came out of the reactor. Therefore, offsite EP plans are not

Table 1 Exemptions Requested from 10 CFR 50.47(b) and 50.47(c)(2)		
Item #	**Regulation in 10 CFR 50.47**	**Basis for Exemption**
		necessary for permanently defueled nuclear power plants.
7	10 CFR 50.47(b)(6): Provisions exist for prompt communications among principal response organizations to emergency personnel ~~and to the public.~~	See basis for 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.	See basis for 50.47(b).
9	10 CFR 50.47(b)(8): Adequate emergency facilities and equipment to support the emergency response are provided and maintained.	No exemption is requested.
10	10 CFR 50.47(b)(9): Adequate methods, systems, and equipment for assessing and monitoring actual or potential ~~offsite~~ consequences of a radiological emergency condition are in use.	See basis for 50.47(b).
11	10 CFR 50.47(b)(10): ~~A range of protective actions has been developed for the plume exposure pathway EPZ for emergency workers and the public. In developing this range of actions, consideration has been given to evacuation, sheltering, and, as a supplement to these, the prophylactic use of potassium iodide (KI), as appropriate. Evacuation time estimates have been developed by applicants and licensees. Licensees shall update the evacuation time estimates on a periodic basis. Guidelines for the choice of protective actions during an emergency, consistent with Federal guidance, are developed and in place, and protective actions for~~	In the unlikely event of a SFP accident, the iodine isotopes which contribute to an off-site dose from an operating reactor accident are not present, so potassium iodide (KI) distribution off-site would no longer serve as an effective or necessary supplemental protective action. The Commission responded to comments in its Statement of Considerations for the Final Rule for emergency planning requirements for ISFSIs and MRS facilities (60 FR 32435) (Reference 19), and concluded that, "the offsite consequences of potential accidents at an ISFSI or a MRS would not warrant establishing Emergency Planning

Table 1

Exemptions Requested from 10 CFR 50.47(b) and 50.47(c)(2)

Item #	Regulation in 10 CFR 50.47	Basis for Exemption
	~~the ingestion exposure pathway EPZ appropriate to the locale have been developed.~~	Zones." Additionally, in the Statement of Considerations for the Final Rule for EP requirements for ISFSIs and for MRS facilities (60 FR 32430) (Reference 19), the Commission responded to comments concerning site-specific emergency planning that includes evacuation of surrounding population for an ISFSI not at a reactor site, and concluded that, "The Commission does not agree that as a general matter emergency plans for an ISFSI must include evacuation planning." Also see basis for 50.47(b).
12	10 CFR 50.47(b)(11): Means for controlling radiological exposures, in an emergency, are established for emergency workers. The means for controlling radiological exposures shall include exposure guidelines consistent with EPA Emergency Worker and Lifesaving Activity Protective Action Guides.	No exemption is requested.
13	10 CFR 50.47(b)(12): Arrangements are made for medical services for contaminated injured individuals.	No exemption is requested.
14	10 CFR 50.47(b)(13): General plans for recovery and reentry are developed.	No exemption is requested.
15	10 CFR 50.47(b)(14): Periodic exercises are (will be) conducted to evaluate major portions of emergency response capabilities, periodic drills are (will be) conducted to develop and maintain key skills, and deficiencies identified as a result of exercises or drills are (will be) corrected.	No exemption is requested.
16	10 CFR 50.47(b)(15): Radiological emergency response training is provided to those who may be called on to assist in an emergency.	No exemption is requested.
17	10 CFR 50.47(b)(16): Responsibilities for plan development and review and for distribution of emergency plans are established, and	No exemption is requested.

Table 1

Exemptions Requested from 10 CFR 50.47(b) and 50.47(c)(2)

Item #	Regulation in 10 CFR 50.47	Basis for Exemption
	planners are properly trained.	
18	10 CFR 50.47(c)(2)~~: Generally, the plume exposure pathway EPZ for nuclear power plants shall consist of an area about 10 miles (16 km) in radius and the ingestion pathway EPZ shall consist of an area about 50 miles (80 km) in radius. The exact size and configuration of the EPZs surrounding a particular nuclear power reactor shall be determined in relation to local emergency response needs and capabilities as they are affected by such conditions as demography, topography, land characteristics, access routes, and jurisdictional boundaries.~~ The size of the EPZs ~~also~~ may be determined on a case-by-case basis for gas cooled nuclear reactors and for reactors with an authorized power level less than 250 MW thermal. ~~The plans for the ingestion pathway shall focus on such actions as are appropriate to protect the food ingestion pathway.~~	Analyses have been developed indicating that, within 15.4 months after shutdown, no credible accident at VY will result in radiological releases requiring offsite protective actions. The analysis of the potential radiological impact of an accident for VY in a permanently defueled condition indicates that any releases beyond the site boundary are limited to small fractions of the EPA PAG exposure levels. According to the EPA's "Protective Action Guides and Planning Guidance for Radiological Incidents, Draft for Interim Use and Public Comment," dated March 2013 (PAG Manual), "EPZs are not necessary at those facilities where it is not possible for PAGs to be exceeded off-site." (Reference 4). Also see basis for 50.47(b).

Table 2
Exemptions Requested from 10 CFR 50, Appendix E

Item #	Regulation in Appendix E to Part 50	Basis for Exemption
19	10 CFR 50 App E: **III. The Final Safety Analysis Report; Site Safety Analysis Report** The final safety analysis report or the site safety analysis report for an early site permit that includes complete and integrated emergency plans under § 52.17(b)(2)(ii) of this chapter shall contain the plans for coping with emergencies. The plans shall be an expression of the overall concept of operation; they shall describe the essential elements of advance planning that have been considered and the provisions that have been made to cope with emergency situations. The plans shall incorporate information about the emergency response roles of supporting organizations and offsite agencies. That information shall be sufficient to provide assurance of coordination among the supporting groups and with the licensee. The site safety analysis report for an early site permit which proposes major features must address the relevant provisions of 10 CFR 50.47 and 10 CFR part 50, appendix E, within the scope of emergency preparedness matters addressed in the major features. The plans submitted must include a description of the elements set out in Section IV for the emergency planning zones (EPZs) to an extent sufficient to demonstrate that the plans provide reasonable assurance that adequate protective measures can and will be taken in the event of an emergency.	No exemption is requested.

Table 2

Exemptions Requested from 10 CFR 50, Appendix E

Item #	Regulation in Appendix E to Part 50	Basis for Exemption
20	10 CFR 50 App E 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 its "Certification of Permanent Removal of Fuel from the Reactor Vessel," in accordance with 10 CFR 50.82(a)(1)(ii), VY will become a permanently shutdown facility with spent fuel stored in the SFP. In the EP Final Rule (76 FR 72596, Nov. 23, 2011) (Reference 5), 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. 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 (NPRs) from the definition of "hostile action" at that time because an NPR is not a nuclear power plant and a regulatory basis had not been developed to support the inclusion of NPR in that definition. Likewise, SFPs are not a nuclear power plant. The following similarities between VY and NPRs show that the VY facility should be treated in a similar fashion as an NPR. Similar to NPRs, VY will pose lower radiological risks to the public from accidents than do power reactors because: (1) VY will be a permanently shutdown facility (with fuel stored in the SFP) and will no longer generate fission products; 2) Fuel stored in the VY SFP will have lower decay heat resulting in lower risk of fission product release in the event of a non-credible boil off or drain down event; and 3) no credible accident at VY will result in radiological releases requiring offsite protective actions.
21	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.~~	See basis for 50.47(b)(10).

Table 2

Exemptions Requested from 10 CFR 50, Appendix E

Item #	Regulation in Appendix E to Part 50	Basis for Exemption
	~~Census Bureau data as of the date the applicant submits its application to the NRC.~~	
22	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.~~	See basis for IV.2.
23	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.~~	See basis for IV.2.
24	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~~	See basis for IV.2.

Table 2

Exemptions Requested from 10 CFR 50, Appendix E

Item #	Regulation in Appendix E to Part 50	Basis for Exemption
	~~ETE analysis.~~	
25	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 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.~~	See basis for IV.2.
26	IV 7 After an applicant for a combined license under part 52 of this chapter receives its license, the licensee shall conduct at least one review of any changes in the population of its EPZ at least 365 days prior to its scheduled fuel load. The licensee shall estimate EPZ permanent resident population changes using the most recent U.S. Census Bureau annual resident population estimate and State/local government population data, if available. If 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	No exemption is requested. VY is not an applicant for a combined license, and therefore, this regulation is not applicable to VY.

Table 2
Exemptions Requested from 10 CFR 50, Appendix E

Item #	Regulation in Appendix E to Part 50	Basis for Exemption
	less, from the licensee's currently approved ETE, the licensee shall update the ETE analysis to reflect the impact of that population increase. The licensee shall submit the updated ETE analysis to the NRC for review under § 50.4 of this chapter no later than 365 days before the licensee's scheduled fuel load.	
27	A Organization The organization for coping with radiological emergencies shall be described, including definition of authorities, responsibilities, and duties of individuals assigned to the licensee's emergency organization and the means for notification of such individuals in the event of an emergency. Specifically, the following shall be included:	No exemption is requested.
28	A.1. A description of the normal plant ~~operating~~ organization.	Appendix A to 10 CFR Part 50, "General Design Criteria for Nuclear Power Plants," states in part: "… there may be water-cooled nuclear power units for which fulfillment of some of the General Design Criteria may not be necessary or appropriate. For plants such as these, departures from the General Design Criteria must be identified and justified." In Appendix A, a nuclear power unit is defined as a nuclear power reactor and associated equipment necessary for electric power generation and includes those structures, systems, and components required to provide reasonable assurance that the facility can be operated without undue risk to the health and safety of the public. Following docketing of the certifications required by 10 CFR 50.82(a)(1), VY will not be a facility that can be operated to generate electrical power. Therefore, VY will not have a "plant operating organization." Rather, the station will

Table 2

Exemptions Requested from 10 CFR 50, Appendix E

Item #	Regulation in Appendix E to Part 50	Basis for Exemption
		be maintained by a defueled on-shift staff.
29	A 2. A description of the onsite emergency response organization (ERO) with a detailed discussion of: a. Authorities, responsibilities, and duties of the individual(s) who will take charge during an emergency; b. Plant staff emergency assignments; c. Authorities, responsibilities, and duties of an onsite emergency coordinator who shall be in charge of the exchange of information with offsite authorities responsible for coordinating and implementing offsite emergency measures.	No exemption is requested.
30	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 VY 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. Decommissioning sites typically have a level of emergency response that does not require response by headquarters personnel.
31	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 results transmitted to State and local authorities, NRC, and other appropriate governmental	Analyses have been developed indicting that, within 15.4 months after shutdown, no credible accident at VY will result in radiological releases requiring offsite protective actions. VY will still be able to determine if a radiological release is occurring. If a

Table 2

Exemptions Requested from 10 CFR 50, Appendix E

Item #	Regulation in Appendix E to Part 50	Basis for Exemption
	entities.	release is occurring, then VY will evaluate the need to promptly communicate that information to offsite authorities for their consideration. The offsite organizations are responsible for deciding what, if any, protective actions should be taken.
32	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.~~	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 qualification of these individuals in the emergency plan. The number of staff at VY once it is in the permanently defueled state will be small but will be commensurate with the need to operate the facility in a manner that is protective of public health and safety.
33	A 6. A description of the local offsite services to be provided in support of the licensee's emergency organization.	No exemption is requested.
34	A 7. ~~By June 23, 2014,~~ identification 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.~~	Requiring a licensee for a decommissioning site to provide a description of the assistance expected from appropriate State, local, and Federal agencies with responsibilities for coping with emergencies is an unnecessary burden on the licensee, in light of the low risk of emergencies necessitating offsite assistance. Requiring an identification and description of the assistance expected from appropriate State, local, and Federal agencies with responsibilities for coping with hostile action at the site is unnecessary because, as explained in section IV.1, a decommissioning power reactor is exempt from requirements in Appendix E related to a "hostile action."

Table 2

Exemptions Requested from 10 CFR 50, Appendix E

Item #	Regulation in Appendix E to Part 50	Basis for Exemption
35	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 analyses have been developed indicating that within 15.4 months after shutdown, no credible accident at VY will result in radiological releases requiring offsite protective actions, protective actions such as evacuation should not be required. Also see basis for 50.47(b)(10).
36	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.~~	In the EP Final Rule (Reference 5), 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 spent fuel pool or in a dry cask storage system in a safe condition requires minimal personnel and is governed by Technical Specifications. Because of the slow rate of the event scenarios postulated in the design basis accident and postulated beyond design basis accident analyses and because the duties of the on-shift personnel at a decommissioning reactor facility are not as complicated and diverse as those for an operating reactor, significant time is available to complete actions necessary to 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.

Table 2
Exemptions Requested from 10 CFR 50, Appendix E

Item #	Regulation in Appendix E to Part 50	Basis for Exemption
37	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.	VY will adopt the Permanently Defueled EALs detailed in Appendix C of NEI 99-01, Revision 6. VY proposes to continue to review EALs with the State of Vermont 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 reduced (informal mailings, etc.). Also see basis in section IV.1 for the justification from the requirements in Appendix E related to "hostile action."
38	B.2. A licensee desiring to change its entire emergency action level scheme shall submit an application for an amendment to its license and receive NRC approval before implementing the change. Licensees shall follow the change process in § 50.54(q) for all other emergency action level changes.	No exemption is requested.

Table 2

Exemptions Requested from 10 CFR 50, Appendix E

Item #	Regulation in Appendix E to Part 50	Basis for Exemption
39	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, detailed in Appendix C of NEI 99-01, Revision 6, will be adopted. This scheme eliminates the Site Area Emergency and General Emergency event classifications. Additionally, the need to base EALs on containment pressure and the response of the ECCS is no longer appropriate for notification of offsite agencies. Containment parameters do not provide an 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 will be used while there is spent fuel in the SFP. In the Statement of Considerations for the Final Rule for EP requirements for ISFSIs and for MRS facilities (60 FR 32430) (Reference 19), 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 a defueled facility that meets the conditions for relaxation of EP requirements, there will be time to take ad hoc measures to protect the public. As stated in NUREG-1738, 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

Table 2
Exemptions Requested from 10 CFR 50, Appendix E

Item #	Regulation in Appendix E to Part 50	Basis for Exemption
		defueled plants, offsite radiation monitoring systems are not required.
40	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 Proposed Rule (74 FR 23254) (Reference 21) to amend certain emergency planning requirements for 10 CFR Part 50, the NRC asked for public comment on whether the NRC should add requirements for non-power reactor licensees to assess, classify, and declare an emergency condition within 15 minutes and promptly declare an emergency condition. The NRC received several comments on these issues. The NRC believed there may be a need for the NRC to be aware of security related events early on so that an assessment can be made to consider the likelihood that the event is part of a larger coordinated attack. However, the NRC determined that further analysis and stakeholder interactions are needed prior to changing the requirements for non-power reactor licensees. Therefore, the NRC did not include requirements in the 2011 EP Final Rule (Reference 5) for non-power reactor licensees to assess, classify, and declare an emergency condition within 15 minutes and promptly declare an emergency condition. See basis in section IV.1 for discussion on the similarity between a permanently defueled reactor and a non-power reactor for the low likelihood of any credible accident resulting in radiological releases requiring offsite protective measures.
41	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	See basis for 50.47(b) and 50.47(b)(10).

Table 2
Exemptions Requested from 10 CFR 50, Appendix E

Item #	Regulation in Appendix E to Part 50	Basis for Exemption
	include identification ~~of the appropriate officials, by title and agency,~~ of the State and local government agencies ~~within the EPZs~~.	
42	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.~~	See basis for section IV.D.1.
43	D.3. A licensee shall have the capability to notify responsible State and local governmental agencies within ~~15~~ minutes after declaring an emergency. ~~The licensee shall demonstrate that the appropriate governmental authorities have the capability to make a public alerting and notification decision promptly on being informed by the licensee of an emergency condition. Prior to initial operation greater than 5 percent of rated thermal power of the first reactor at a site, each nuclear power reactor licensee shall demonstrate that administrative and physical means have been established for alerting and providing prompt instructions to the public within the plume exposure pathway EPZ. The design objective of the prompt public alert and notification system shall be to have the capability to essentially complete the initial alerting and initiate notification of the public within the plume exposure pathway EPZ within about 15~~	While the capability needs to exist for the notification of offsite government agencies within a specified time period, previous exemptions have allowed for extending the State and local government agencies' notification time up to 60 minutes based on the site-specific justification provided. VY proposes to complete emergency notification within 60 minutes after an emergency declaration or a change in classification to the State of Vermont. This timeframe is consistent with the 10 CFR 50.72(a)(3) notification to the NRC and is appropriate because in the permanently defueled condition, the rapidly developing scenarios associated with events initiated during reactor power operation are no longer credible and there is no need for State or local response organizations to implement

Table 2 **Exemptions Requested from 10 CFR 50, Appendix E**		
Item #	**Regulation in Appendix E to Part 50**	**Basis for Exemption**
	~~minutes. The use of this alerting and notification capability will range from immediate alerting and notification of the public (within 15 minutes of the time that State and local officials are notified that a situation exists requiring urgent action) to the more likely events where there is substantial time available for the appropriate governmental authorities to make a judgment whether or not to activate the public alert and notification system. The alerting and notification capability shall additionally include administrative and physical means for a backup method of public alerting and notification capable of being used in the event the primary method of alerting and notification is unavailable during an emergency to alert or notify all or portions of the plume exposure pathway EPZ population. The backup method shall have the capability to alert and notify the public within the plume exposure pathway EPZ, but does not need to meet the 15-minute design objective for the primary prompt public alert and notification system. When there is a decision to activate the alert and 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.~~	any protective actions. Also see basis for 50.47(b) and 50.47(b)(10).
44	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~~	See basis for section IV D.3 regarding the alert and notification system requirements.

Table 2 **Exemptions Requested from 10 CFR 50, Appendix E**		
Item #	**Regulation in Appendix E to Part 50**	**Basis for Exemption**
	~~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.~~	
45	E. Emergency Facilities and Equipment Adequate provisions shall be made and described for emergency facilities and equipment, including: E.1. Equipment at the site for personnel monitoring;	No exemption is requested.
46	E.2. Equipment for determining the magnitude of and for continuously assessing the impact of the release of radioactive materials to the environment;	No exemption is requested.
47	E.3. Facilities and supplies at the site for decontamination of onsite individuals;	No exemption is requested.
48	E.4. Facilities and medical supplies at the site for appropriate emergency first aid treatment;	No exemption is requested.

Table 2
Exemptions Requested from 10 CFR 50, Appendix E

Item #	Regulation in Appendix E to Part 50	Basis for Exemption
49	E.5. Arrangements for medical service providers qualified to handle radiological emergencies onsite;	No exemption is requested.
50	E.6. Arrangements for transportation of contaminated injured individuals from the site to specifically identified treatment facilities outside the site boundary;	No exemption is requested.
51	E.7. Arrangements for treatment of individuals injured in support of licensed activities on the site at treatment facilities outside the site boundary;	No exemption is requested.
52	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;	Due to analyses indicting that, within 15.4 months after shutdown, no credible accident at VY will result in radiological releases requiring offsite protective actions, offsite agency response will not be required at an emergency operations facility (EOF) and 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 VY emergency plan will continue to maintain arrangements for requesting assistance and using resources from appropriate offsite support organizations.
53	E.8.a (ii) ~~For nuclear power reactor licensees, a licensee onsite operational support center;~~	NUREG-0696, "Functional Criteria for Emergency Response Facilities," (Reference 22) 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

Table 2 Exemptions Requested from 10 CFR 50, Appendix E		
Item #	**Regulation in Appendix E to Part 50**	**Basis for Exemption**
		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 VY, from which control room support, emergency mitigation, radiation monitoring, and effective control may be exercised during an emergency.
54	E.8.b. For a nuclear power reactor licensee's emergency operations facility required by paragraph 8.a of this section, either a facility located between 10 miles and 25 miles of the nuclear power reactor site(s), or a primary facility located less than 10 miles from the nuclear power reactor site(s) and a backup facility located between 10 miles and 25 miles of the nuclear power reactor site(s). An emergency operations facility may serve more than one nuclear power reactor site. A licensee desiring to locate an emergency operations facility more than 25 miles 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:	No exemption is requested.
55	E.8.b. (1) Space for members of an NRC site team and Federal,	

Table 2

Exemptions Requested from 10 CFR 50, Appendix E

Item #	Regulation in Appendix E to Part 50	Basis for Exemption
	State, and local responders	
56	E.8.b. (2) Additional space for conducting briefings with emergency response personnel;	
57	E.8.b.(3) Communication with other licensee and offsite emergency response facilities;	
58	E.8.b.(4) Access to plant data and radiological information; and	
59	E.8.b.(5) Access to copying equipment and office supplies;	
60	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:~~ ~~(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;~~	See basis for 50.47(b)(3).
61	~~E.8.c (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~~	

Table 2
Exemptions Requested from 10 CFR 50, Appendix E

Item #	Regulation in Appendix E to Part 50	Basis for Exemption
62	E.8.c (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~~	
63	~~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 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.~~	See basis for section IV.1 regarding hostile action.
64	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;~~	See basis for 50.47(b)(3).
65	E.9. At least one onsite and one offsite communications system; each system shall have a backup power source. All communication	See basis for 50.47(b) and (b)(10).

Table 2

Exemptions Requested from 10 CFR 50, Appendix E

Item #	Regulation in Appendix E to Part 50	Basis for Exemption
	plans shall have arrangements for emergencies, including titles and alternates for those in charge at both ends of the communication links and the primary and backup means of communication. Where consistent with the function of the governmental agency, these arrangements will include: E.9.a. Provision for communications with contiguous State/local governments ~~within the plume exposure pathway EPZ.~~ Such communications shall be tested monthly.	VY will maintain communications with the State of Vermont and the NRC. The onsite response facilities will be combined into a single facility.
66	E.9.b. Provision for communications with Federal emergency response organizations. Such communications systems shall be tested annually.	No exemption is requested.
67	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.	Due to analyses indicting that, within 15.4 months after shutdown, no credible accident at VY 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. VY will also continue to test communication systems used to contact the State EOCs on an annual basis. Also see justification for 50.47(b)(3).

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Item #	Regulation in Appendix E to Part 50	Basis for Exemption
68	E.9.d. Provisions for communications by the licensee with NRC Headquarters and the appropriate NRC Regional Office Operations Center from the ~~nuclear power reactor control room, the onsite technical support center, and the emergency operations~~ facility. Such communications shall be tested monthly.	The functions of the control room, EOF, TSC and OSC may 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 direction can be given and effective control may be exercised during an emergency. VY will maintain communications with the NRC. Also see basis for 50.47(b).
69	F. Training F.1. The program to provide for: (a) The training of employees and exercising, by periodic drills, of 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 radiological 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	No exemption is requested.
70	F.1. i. Directors and/or coordinators of the plant emergency organization;	
71	F.1. ii. Personnel responsible for accident assessment, including control room shift personnel;	

Table 2 Exemptions Requested from 10 CFR 50, Appendix E		
Item #	**Regulation in Appendix E to Part 50**	**Basis for Exemption**
72	F.1. iii Radiological monitoring teams;	
73	F.1. iv. Fire control teams (fire brigades);	
74	F.1. v. Repair and damage control teams;	
75	F.1. vi. First aid and rescue teams;	
76	F.1. vii. Medical support personnel;	
77	F.1. viii. ~~Licensee's headquarters support personnel;~~	The number of staff at VY 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. Decommissioning sites typically have a level of emergency response that does not require additional response by headquarters personnel. Therefore, exempting licensee's headquarters personnel from training requirements is considered to be reasonable.
78	F.1. ix. Security personnel.	No exemption is requested.
79	F.1 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.~~	Because there will no longer be any expected actions that must be taken by the public during an emergency, it is no longer necessary to pre-plan the dissemination of this information to the public or to provide radiological orientation training to local news media persons.

Table 2
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Item #	Regulation in Appendix E to Part 50	Basis for Exemption
		The phrase "Civil Defense" is no longer a commonly used term and is no longer applicable as an example in the regulation.
80	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.	Due to analyses indicting that, within 15.4 months after shutdown, no credible accident at VY will result in radiological releases requiring offsite protective actions, the public alert and notification system will not be used and therefore requires no testing. Also see basis for 50.47(b).
81	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.~~	VY will continue to invite the State of Vermont and local support organizations to participate in the periodic drills and exercises conducted to assess its ability to perform responsibilities related to an emergency at VY to the extent defined by the VY emergency plan and State emergency plans. Because the need for off-site emergency planning is relaxed due to the low probability of design-basis accidents or other credible events that would be expected to result in an offsite radioactive release that would exceed the limits of EPA PAGs and the available time for event mitigation, no off-site emergency plans will be in place to test. The intent of submitting exercise scenarios at power reactors is to check that licensees utilize different scenarios in order to prevent the preconditioning of responders at power reactors. For defueled sites, there are limited events that could occur and the previously routine progression to General Emergency in power reactor site scenarios is not applicable to
82	F.2.a(i) For an operating license issued under this part, this exercise must be conducted within two years before the issuance of the first operating license for full power (one authorizing operation above 5 percent of rated power) of the first reactor and shall include participation by each State and local government within the plume exposure pathway EPZ and each state within the ingestion exposure pathway EPZ. If the full participation exercise is conducted more than 1 year prior to issuance of an operating	

Table 2 **Exemptions Requested from 10 CFR 50, Appendix E**		
Item #	**Regulation in Appendix E to Part 50**	**Basis for Exemption**
	licensee for full power, an exercise which tests the licensee's onsite emergency plans must be conducted within one year before issuance of an operating license for full power. This exercise need not have State or local government participation.	a decommissioning site. ENO considers VY to be exempt from F.2.a.(i)-(iii) because VY will be exempt from the umbrella provision of F.2.a.
83	F 2.a.(ii) For a combined license issued under part 52 of this chapter, this exercise must be conducted within two years of the scheduled date for initial loading of fuel. If the first full participation exercise is conducted more than one year before the scheduled date for initial loading of fuel, an exercise which tests the licensee's onsite emergency plans must be conducted within one year before the scheduled date for initial loading of fuel. This exercise need not have State or local government participation. If FEMA identifies one or more deficiencies in the state of offsite emergency preparedness as the result of the first full participation exercise, or if the Commission finds that the state of emergency preparedness does not provide reasonable assurance that adequate protective measures can and will be taken in the event of a radiological emergency, the provisions of § 50.54(gg) apply.	
84	F 2.a (iii) For a combined license issued under part 52 of this chapter, if the applicant currently has an operating reactor at the site, an exercise, either full or partial participation, shall be conducted for each subsequent reactor constructed on the site. This exercise may be incorporated in the exercise requirements of Sections IV.F.2.b. and c. in this appendix. If FEMA identifies one or more deficiencies in the state of offsite emergency preparedness as the result of this exercise for the new reactor, or if the Commission finds that the state of emergency preparedness does	

Table 2
Exemptions Requested from 10 CFR 50, Appendix E

Item #	Regulation in Appendix E to Part 50	Basis for Exemption
	not provide reasonable assurance that adequate protective measures can and will be taken in the event of a radiological emergency, the provisions of § 50.54(gg) apply.	
85	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, assessment of the onsite ~~and offsite~~ impact of radiological releases, ~~protective action recommendation development, protective action decision making, plant~~ system repair and mitigative action implementation. During these drills, activation of all of the licensee's emergency response facilities ~~(Technical Support Center (TSC), Operations Support Center (OSC), and the Emergency Operations Facility (EOF))~~ would not be necessary, licensees would have the opportunity to consider accident management strategies, supervised instruction would be permitted, operating staff in all participating facilities would have the opportunity to resolve problems (success paths) rather than	See basis for section IV.F.2.a. The low probability of a design-basis accident or other credible events that would result in an offsite radioactive release that would exceed the EPA PAGs and the available time for event mitigation at VY during decommissioning render TSCs, OSCs and EOFs 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. The onsite response facilities at VY will be combined into a single facility. VY will continue to conduct biennial exercises and will invite the State of Vermont 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 VY to the extent defined by the VY and State emergency plans.

Table 2

Exemptions Requested from 10 CFR 50, Appendix E

Item #	Regulation in Appendix E to Part 50	Basis for Exemption
	have controllers intervene, and the drills may focus on the onsite exercise training objectives.	
86	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:~~	See basis for section IV.F.2.a.
87	F 2.c.(1) ~~Conduct an exercise biennially of its onsite emergency plan;~~	
88	F 2.c.(2) ~~Participate quadrennially in an offsite biennial full or partial participation exercise;~~	
89	F 2.c.(3) ~~Conduct emergency preparedness activities and interactions in the years between its participation in the offsite full or partial participation exercise with offsite authorities, to test and maintain interface among the affected State and local authorities and the licensee. Co-located licensees shall also participate in emergency preparedness activities and interaction with offsite authorities for the period between exercises;~~	

Table 2

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Item #	Regulation in Appendix E to Part 50	Basis for Exemption
90	F 2.c.(4) ~~Conduct a hostile action exercise of its onsite emergency plan in each exercise cycle; and~~	
91	F 2.c.(5) ~~Participate in an offsite biennial full or partial participation hostile action exercise in alternating exercise cycles.~~	
92	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.~~	See basis for section IV.2.
93	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.	See basis for section IV.2.
94	F 2.f. Remedial exercises will be required if the emergency plan is not satisfactorily tested during the biennial exercise, such that NRC~~, in consultation with FEMA~~, cannot (1) find reasonable assurance that adequate protective measures can and will be taken in the event of a radiological emergency or (2) determine that	The U.S. Federal Emergency Management Agency (FEMA) is responsible for the evaluation of an offsite response exercise. No action is expected from State or local government organizations in response to an event at a decommissioning site other than firefighting, law enforcement and ambulance/medical services. Memoranda of

Table 2

Exemptions Requested from 10 CFR 50, Appendix E

Item #	Regulation in Appendix E to Part 50	Basis for Exemption
	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.~~	understanding will continue to be in place for those services. Offsite response organizations will continue to take ad hoc actions to protect the health and safety of the public as they would at any other industrial site.
95	F 2.g. All exercises, drills, and training that provide performance opportunities to develop, maintain, or demonstrate key skills must provide for formal critiques in order to identify weak or deficient areas that need correction. Any weaknesses or deficiencies that are identified in a critique of exercises, drills, or training must be corrected.	No exemption is requested.
96	F 2.h. The participation of State and local governments in an emergency exercise is not required to the extent that the applicant has identified those governments as refusing to participate further in emergency planning activities, pursuant to § 50.47(c)(1). In such cases, an exercise shall be held with the applicant or licensee and such governmental entities as elect to participate in the emergency planning process.	No exemption is requested.
97	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	For defueled sites, there are limited events that could occur and the previously routine progression to General Emergency in power reactor site scenarios is not applicable to a decommissioning site. Therefore, defueled sites are not expected to demonstrate response to a wide spectrum of events.

Table 2

Exemptions Requested from 10 CFR 50, Appendix E

Item #	Regulation in Appendix E to Part 50	Basis for Exemption
	among onsite and offsite response organizations.	Also see basis for section IV.1 regarding hostile action.
98	F 2.j. ~~The exercises conducted under paragraph 2 of this section by nuclear power reactor licensees must provide the opportunity for the ERO to demonstrate proficiency in the key skills necessary to implement the principal functional areas of emergency response identified in paragraph 2.b of this section. Each exercise must provide the opportunity for the ERO to demonstrate key skills specific to emergency response duties in the control room, TSC, OSC, EOF, and joint information center. Additionally, in each eight calendar year exercise cycle, nuclear power reactor licensees shall vary the content of scenarios during exercises conducted under paragraph 2 of this section to provide the opportunity for the ERO to demonstrate proficiency in the key skills necessary to respond to the following scenario elements: hostile action directed at the plant site, no radiological release or an unplanned minimal radiological release that does not require public protective actions, an initial classification of or rapid escalation to a Site Area Emergency or General Emergency, implementation of strategies, procedures, and guidance developed under § 50.54(hh)(2), and integration of offsite resources with onsite response. 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~~	See basis for section IV.F.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. The VY emergency plan will discuss exercise and drill types and frequencies of occurrence. Scenarios will be developed to test all major elements of the emergency plan within an eight (8) year period. These elements include management and coordination of emergency response, accident assessment, and system repair and corrective action. VY will continue to include the State of Vermont and local support organizations in the periodic drills and exercises to assess its ability to perform responsibilities related to an emergency at VY to the extent defined by the VY and State emergency plans.

Table 2

Exemptions Requested from 10 CFR 50, Appendix E

Item #	Regulation in Appendix E to Part 50	Basis for Exemption
	~~required by Section IV.F.2.a.~~	
99	G. Maintaining Emergency Preparedness Provisions to be employed to ensure that the emergency plan, its implementing procedures, and emergency equipment and supplies are maintained up to date shall be described.	No exemption is requested.
100	H. Recovery Criteria to be used to determine when, following an accident, reentry of the facility would be appropriate or when operation could be resumed shall be described.	No exemption is requested.
101	I. Onsite Protective Actions During Hostile Action ~~By June 20, 2012, for nuclear power reactor licensees, a range of protective actions to protect onsite personnel during hostile action must be developed to ensure the continued ability of the licensee to safely shut down the reactor and perform the functions of the licensee's emergency plan.~~	See basis for section IV.1.
102	10CFR 50 App E V. Implementing Procedures No less than 180 days before the scheduled issuance of an operating license for a nuclear power reactor or a license to possess nuclear material, or the scheduled date for initial loading of fuel for a combined license under part 52 of this chapter, the applicant's or licensee's detailed implementing procedures for its emergency plan shall be submitted to the	No exemption is requested.

Table 2
Exemptions Requested from 10 CFR 50, Appendix E

Item #	Regulation in Appendix E to Part 50	Basis for Exemption
	Commission as specified in § 50.4. Licensees who are authorized to operate a nuclear power facility shall submit any changes to the emergency plan or procedures to the Commission, as specified in § 50.4, within 30 days of such changes.	
103	10CFR 50 App E VI. Emergency Response Data System 1. The Emergency Response Data System (ERDS) is a direct near real-time electronic data link between the licensee's onsite computer system and the NRC Operations Center that provides for the automated transmission of a limited data set of selected parameters. The ERDS supplements the existing voice transmission over the Emergency Notification System (ENS) by providing the NRC Operations Center with timely and accurate updates of a limited set of parameters from the licensee's installed onsite computer system in the event of an emergency. When selected plant data are not available on the licensee's onsite computer system, retrofitting of data points is not required. The licensee shall test the ERDS periodically to verify system availability and operability. The frequency of ERDS testing will be quarterly unless otherwise set by NRC based on demonstrated system performance. 2. Except for Big Rock Point and all nuclear power facilities that are shut down permanently or indefinitely, onsite hardware shall be provided at each unit by the licensee to interface with the NRC receiving system. Software, which will be made available by the NRC, will assemble the data to be transmitted and transmit data	The regulation that identifies the requirement to maintain the Emergency Response Data System (ERDS) is not applicable to nuclear power facilities that are permanently shutdown. Once VY is permanently defueled, this system will no longer be necessary to transmit safety system parameter data. No exemption is requested since this change in the ERDS data requirement is identified in 10 CFR 50 Appendix E, VI. 2.

Table 2

Exemptions Requested from 10 CFR 50, Appendix E

Item #	Regulation in Appendix E to Part 50	Basis for Exemption
	from each unit via an output port on the appropriate data system.	

Based on the analysis detailed below, ENO has concluded that the portions of 10 CFR 50.47(b), 10 CFR 50.47(c)(2) and 10 CFR 50, Appendix E, identified in the above tables will not be necessary to protect the health and safety of the public once VY is in the permanently defueled condition, and would be unduly burdensome. 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 VY.

III. Background

VY is located in the town of Vernon, Vermont in Windham County on the west shore of the Connecticut River immediately upstream of the Vernon Hydrostation. VY is a boiling water reactor with a rated thermal power of 1912 MWt. The station is located on approximately 125 acres in Windham County, and is owned by Entergy Nuclear Vermont Yankee, LLC, with the exception of a narrow strip of land between the Connecticut River and the VY property for which Entergy Nuclear Vermont Yankee, LLC has perpetual rights and easements from the owner.

The site is bounded by the Connecticut River (Vernon Pond) on the east, by farm and pasture land mixed with wooded areas on the north and south, and by the town of Vernon on the west. Warwick and Northfield State Forests (approximately 8 miles southwest of the site), Green Mountain National Forest (approximately 18 miles southwest of the site) and the Pisgah Mountain Range (northeast of the site) limit the population density and land use within a 50-mile radius of the site. Most of the land around the site is undeveloped. The developed land is used for agricultural, dairying, and for residential areas within small villages. The primary agricultural crop is silage corn, which is stored for year-round feed for milk cows.

The area within 10 miles of the site has only one urban area, the city of Brattleboro, Vermont (2000 population 12,005), which is located about 5 miles upriver. The remainder of this area is rural and contains several small villages with populations between 1,000 and 3,000. The average population density within a 10-mile radius of VY for 2000 was estimated to be 126 people per square mile.

Section 14 of the VY Updated Final Safety Analysis Report (UFSAR) describes the design basis accident (DBA) scenarios that are applicable to VY during power operations and describe the accidents with the greatest potential for radiation exposure of any accident considered under the same assumptions. The most severe postulated accidents for nuclear power plants involve damage to the nuclear reactor core and the release of large quantities of fission products. The UFSAR accident scenarios include a Control Rod Drop Accident (CRDA), a Loss-of-Coolant Accident (LOCA), a Refueling/Fuel Handling Accident (RA/FHA) and a Main Steam Line Break Accident.

Following docketing of its "Certification of Permanent Removal of Fuel from the Reactor Vessel," in accordance with 10 CFR 50.82(a)(1)(ii), and within two years following cessation of operations, VY will submit a Post-Shutdown Decommissioning Activities Report (PSDAR), which will identify VYs selected method of decommissioning. At the end of the current operating cycle, the VY 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 ISFSI until it is shipped off-site in accordance with the schedules described in the PSDAR and updated Irradiated Fuel Management Plan. The PSDAR will identify most fluid systems drained and the plant in a stable condition until final decontamination and dismantlement.

When the reactor is permanently defueled, the SFP and its supporting systems will be modified and dedicated only to spent fuel storage. 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 SFPs. A fuel pool cooling and clean-up system is provided to remove decay heat from spent fuel stored in the fuel pool and to maintain a specified water temperature, purity, clarity, and level.

IV. Technical Evaluation

A. 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 VY 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 VY is presented below. According to the EPA, "Protective Action Guides and Planning Guidance for Radiological Incidents, Draft for Interim Use and Public Comment," dated March 2013 (Reference 4), Section 2.3.5, "PAGs and Nuclear Facilities Emergency Planning Zones (EPZ)," EPZs are not necessary at those facilities where it is not possible for PAGs to be exceeded off-site.

A. Consequences of Design Basis Events

The postulated design basis accident that will remain applicable to VY in its permanently shutdown and defueled condition is the FHA in the reactor building where the SFP is located. A new 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") and Low Population Zone (LPZ) as a function of time after shutdown. The analysis shows that the dose at the EAB and LPZ 17 days after shutdown (with open containment) is less than 1 rem TEDE, which is below the EPA PAG threshold of 1 rem for recommended evacuation.

Due to the amount of decay assumed (17 days), the results of this analysis may be applied after January 17, 2015, assuming a VY shutdown by the end of December 2014. The analysis was submitted for NRC review in Reference 6.

B. Consequences of Beyond Design Basis Events

a. Hottest Fuel Assembly Adiabatic Heatup - Beyond Design Basis Event

The analysis in Attachment 2 is provided to compare the conditions for the hottest fuel assembly stored in the VY fuel pools to a criterion proposed in SECY-99-168 (Reference 7) 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, at 15.4 months after shutdown (15.4 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 8), 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 partial 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.

C. Consequences of Other Analyzed Events

a. Loss of Spent Fuel Pool Normal Cooling

This analysis assesses the time available to initiate compensatory measures in the event of a loss of spent fuel pool inventory as well as the radiological impact. From Engineering Change 47710, the initiating event is postulated to be an external event that results in a prolonged loss of all Alternating Current (AC) power. In this scenario, there is no active cooling of the spent fuel pool, nor is there the ability to maintain pool water inventory with normal plant systems. This evaluation determined that 15.4 months following shutdown, the time to reach 212 degrees Fahrenheit will be 74 hours, and the total time from the loss of cooling to boil off inventory to 3 feet above the top of the fuel assemblies will be 16 days. Although no fuel damage is expected while the water level remains above the top of the fuel, a level of 3 feet above the top of the fuel was chosen for ease of comparison to the corresponding information contained in NUREG-1738. Three feet of water continues to provide sufficient shielding from radiation to any personnel involved in responding to the event. Due to the slow rate of spent fuel pool water boil-off, adequate time will be available to restore cooling or makeup, either through restoration of normal systems or through readily available mitigation measures, without significant radiological consequences for plant workers in the Reactor Building.

b. Radioactive Waste Handling Accident

This analysis evaluated the drop of a high integrity container (HIC). The accident evaluated the drop of the largest liner containing the highest concentration of radioactive materials (dewatered resin containing 19,415 curies of 25 various radionuclides representing the highest activity waste at the facility). The calculation postulates that the container is dropped 250 meters (820 feet) from the closest site boundary with subsequent container failure with 1% of the liner contents released and 0.5% of the release becoming aerosolized and carried in the direction of the closest Site Boundary. The resulting two hour integrated dose at the Site Boundary is projected to be 16.1 millirem TEDE, which is below the EAB limit of 1 rem TEDE.

B. 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 VY justify a reduction in the necessary scope of emergency response capabilities, ENO also evaluated the industry decommissioning commitments (IDCs) and staff decommissioning assumptions (SDAs) contained in NUREG-1738.

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 9). 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 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 following tables show how the VY SFP meets or compares with each of these IDCs (Table 3) and SDAs (Table 4). Attachment 3 includes a new regulatory commitment to update the VY UFSAR with this information.

C. Consequences of a Beyond-Design Basis Earthquake

In June 2013, a draft study, entitled "Consequence Study of a Beyond-Design-Basis Earthquake Affecting the Spent Fuel Pool for a U.S. Mark 1 Boiling Water Reactor," was published for public comment (Reference 17). The purpose of the consequence study was to determine if accelerated transfer of older, colder spent fuel from the SFP at a reference plant to dry cask storage significantly reduces risks to public health and safety. The specific reference plant used for the study was a General Electric (GE) Type 4 BWR with a Mark I containment. VY is a GE BWR/4 with a Mark I containment.

The study states: "Past risk studies have shown that storage of spent fuel in a high-density configuration is safe and risk of a large release due to an accident is very low. This study's results are consistent with earlier research conclusions that spent fuel pools are robust structures that are likely to withstand severe earthquakes without leaking. The NRC continues to believe, based on this study and previous studies that spent fuel pools protect public health and safety."

The study also estimated that the likelihood of a radiological release from the SFP resulting from the selected severe seismic event analyzed in the study was on the order of one time in 10 million years or lower. The study analyzed two cases for each scenario: one where mitigation measures of 10 CFR 50.54(hh)(2) were credited, and one where they were not used or were unsuccessful. It showed that successful mitigation reduces the likelihood of a release and that the likelihood of a release was equally low for both high- and low-density loading in the SFP. The study did not consider the post-Fukushima mitigation measures required by Orders EA-12-049 (Mitigating Strategies Order) and EA-12-051 (Reliable Hardened Containment Vents Order).

D. Conclusion

Based on the above, VY 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 3
Industry Decommissioning Commitments (IDCs) Comparison

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 VY design is in alignment with this description. Program Procedure PP-7023 "Control of Heavy Loads Program Document" 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 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 10. As documented in Reference 11 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.	VY procedures are in place to ensure onsite and offsite resources can be brought to bear during an event, including: ON-3177, *Operations Response to an Aircraft Threat*, OPOP-SECU-3132, *Operations Department Response to Security Events*, OPOP-PHEN-3127, *Natural Phenomena*, and ON-3157, *Loss of Fuel Pool Level/Cooling*. Additionally, EPOP-CR-3540, *Control Room Actions During an Emergency*, directs the Shift Manager to notify the Security Shift Supervisor to implement OP 3547, *Security Actions during an Emergency*, which activates the VY Pager System. This action notifies Emergency Personnel to report to the proper location. EPOP-CR-3540 also directs the ERO Notification System Activation. These procedures are not specifically referenced in the existing VY Emergency Plan and will not be included in the planned Permanently Defueled Emergency Plan (to be submitted for NRC approval). These procedures are required by Technical Specification 6.4. Therefore, it is not necessary for them to be specifically referenced in the Emergency Plan. Equipment requirements are specified in the pertinent procedures. Once VY is shutdown and defueled, the on-shift plant operators, including Certified Fuel Handlers (CFH), and fire brigade members will be appropriately trained on the various actions needed to provide makeup to the SFP based on a systematic approach to training. Once VY 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 VY CFH training program was submitted for NRC review and approval by letter dated October 31, 2013

		(Reference 12). Finally, quarterly Emergency Plan drills are conducted with frequent 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.	Procedures OP 3504, *Emergency Communications*, EPOP-CR-3540, *Control Room Actions During an Emergency*, OPOP-PHEN-3127, *Natural Phenomena*, AP 0153, *Operations Department Communications and Log Maintenance*, and AP 3150, *Extensive Damage Mitigation Strategy for Establishing Command and Control*, provide guidance for initiating and maintaining communications between offsite agencies and the onsite Emergency Response Organization during severe weather and seismic events.
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.	Appendix G, *Loss of Large Areas of the Plant Due to Fire or Explosion*, of Procedure PP 7019, *Severe Accident Management Program*, contains an offsite resource list which shows providers, their capabilities, and a contact telephone number as well as actions for spent fuel pool damage and supplying water to off-site fire support for external make-up.
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.	VY design meets the intent of this IDC. There are two narrow range channels of continuous remote indication of spent fuel pool water level in the control room. Each of these channels provides high and low annunciation on the Fuel Pool Filter Demineralizer Panel with a system trouble alarm and high and low level indicator lights in the control room. In addition, each of these channels provides input to the plant computer and there is local water level indication on the side of the SFP. There are two channels of continuous remote indication of the spent fuel pool water temperature in the control room. There are two channels of SFP water temperature to a common recorder that provides high temperature annunciation in the control room. In addition, there are two channels of SFP water temperature that provide input to the plant computer. 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 VY 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 321.5 feet. The top elevation of a spent fuel rack in the SFP is 321.29 feet.
7	Procedures or administrative controls to reduce the likelihood of rapid draindown 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.	VY procedure, OPOP-NFPC-2184, *Normal 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 draindown. 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 VY SFP does not have 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 incorporated into procedure ON-3157, *Loss of Fuel Pool Level/Cooling*, and PP-7019, Appendix G, *Loss of Large Areas of the Plant Due to Fire or Explosion*. ON-3157 establishes multiple makeup sources from onsite and offsite that includes: • Fire Water system • Service Water (SW) system via the Residual Heat Removal system • Cooling Tower #2 deep basin via an engine driven emergency makeup pump 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	VY procedure OPOP-NFPC-2184, Normal Fuel Pool Cooling System, allows specified volumes to be pumped or letdown from the spent fuel pool. The procedure meets the requirements of this IDC by controlling the suction and discharge points. Additionally, the Independent Spent Fuel Storage Installation (ISFSI) equipment design is such that there are no ISFSI related SFP operations that have the potential to cause a rapid draindown.

	physical presence for designated operations or administrative limitations such as restrictions on heavy load movements.	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 procedures EN-MA-119, *Material Handling Program*, and the VY heavy loads program PP-7023, *Control of Heavy Loads Program Document*. Heavy Loads in the vicinity of the SFP are addressed in PP-7023. 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.	VY practices align with this description. The SW system has redundant pumping capability and power supplies to ensure alternative fuel pool makeup function. The SW system runs continuously thus allowing for constant monitoring. Additionally, there is an electric-driven fire pump and a diesel-driven fire pump that can supply makeup water to the SFP via the SW system or the Fire Water system. The VY Technical Requirements Manual (TRM) contains administrative controls for the fire pumps that address operation with equipment out of service and periodic functionality testing. All sources discussed above take suction from the Connecticut River. VY also has an engine driven emergency makeup pump capable of taking suction from the Cooling Tower #2 deep basin to provide an alternate source of makeup water to the SFP. The systems provide defense-in-depth. The systems and components necessary to implement the emergency makeup strategies are routinely tested to ensure capability is maintained.

TABLE 4
Staff Decommissioning Assumptions (SDAs) Comparison

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 VY design aligns with the intent of this description. The VY SFP cooling system design is based, in part, on Regulatory Guide 1.13 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 by the SW (SW) system, 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 SW which has redundant pumping capacity and is provided by redundant power sources adequate to provide makeup at the required capacity. The SW pumps are normally powered from offsite power, but can be supplied from an alternate reliable power source. 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. All sources discussed above take suction from the Connecticut River. The SW system runs continuously thus allowing for constant monitoring. VY also has an engine driven emergency makeup pump capable of taking suction from the Cooling Tower #2 deep basin to provide an alternate source of makeup water to the SFP at a minimum of 300 gpm. When the discharge of this pump is connected to a pumper type fire truck, the fire truck is able to provide a minimum of 200 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 VY 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. VY procedures meet the requirements of this SDA by providing the guidance on the capability and availability of onsite and offsite makeup sources. OPOP-PHEN-3127, *Natural Phenomena*, directs the inspection of the SFP and cooling systems following a seismic event. ON-3157, *Loss of Fuel Pool Level/Cooling*, establishes multiple makeup sources from onsite and offsite that

		includes: • Fire Water system • SW system via the Residual Heat Removal system • Cooling Tower #2 deep basin via an engine driven emergency makeup pump Prior to final shutdown, VY will establish the timelines required to initiate the various onsite and offsite SFP makeup sources based on expected system configurations and availability.
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.	VY design meets the intent of this SDA. There are two narrow range channels of continuous remote indication of SFP water level in the control room. Level is determined by measuring the hydrostatic pressure from a sensor located in the SFP. Each of these channels provides high and low annunciation on the Fuel Pool Filter Demineralizer Panel with a System Trouble Alarm and high and low level indicator lights in the control room. In addition, each of these channels provides input to the plant computer and there is local water level indication on the side of the SFP. There are two channels of continuous remote indication of the SFP water temperature in the control room from temperature elements located in the SFP. There are two channels of SFP water temperature to a common recorder that provides high temperature annunciation in the control room. In addition, there are two channels of SFP water temperature that provide input to the plant computer. VY has procedures in place to respond to an abnormally low 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 VY responses for IDC 2 and IDC 4 for details associated with calling in offsite resources. Regarding the declaration of a general emergency, VY 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 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	The VY SFP design is consistent with this SDA. • The VY normal SFP cooling system suction lines are located at elevation 336'-6", which is approximately 7'-6" below normal SFP water level. • The VY maximum depth of siphon path is at elevation 334'-0", which is approximately 10'

	normal pool operating level and that licensee must initiate recovery using offsite sources.	below normal SFP water level. However, this line is isolated with a normally locked-closed valve to prevent possible siphoning. This is seismically rated piping. • The VY maximum drain path is via the 3" drain line located between in the inboard and outboard SFP gates, located at elevation 319.85' which is located approximately 24' below normal SFP water level. Drain-down to this elevation would result in uncovery of the top 1.5' of the spent fuel racks. However, there would need to be a gross failure of the inboard gate sealing gasket as well as failure of the 3" drain line. 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 nonsingle 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 VY design is in alignment with this description. The VY heavy loads program PP-7023, *Control of Heavy Loads Program Document*, 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 PP-7023. The NRC Safety Evaluation Report for the acceptance of NUREG-0612 Phase I actions is documented in Reference 10. The VY TRM contains functionality requirements for the reactor building crane, specifically, TRM 3.12, "Refueling and Spent Fuel Handling," which is subject to the provisions of 10 CFR 50.59.
6	Each decommissioning plant will successfully complete the seismic checklist provided in Appendix2B 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).	A VY-specific seismic risk estimate of the SFP was performed in support of development of NUREG-1738. The risk estimate and results are documented in Attachment 2 to Appendix 2B of the NUREG. Table 2 of the attachment shows that the risk of a seismic-induced SFP failure was estimated to be $8.9x10^{-7}$ per year. This value is less than the NRC Pool Performance Guideline (PPG) of $1x10^{-5}$ per year specified in Section 2 of Appendix 2B. Also, a separate study performed under NUREG/CR-5176 that is referenced in Appendix 2B of NUREG-1738 includes a VY-specific seismic risk assessment of the SFP. The results of this assessment indicate SFP failure was $6.7x10^{-6}$ per year, which is below the PPG guideline as well. Finally, item 10 of the seismic checklist provides an alternative in which the licensee delays requesting of a waiver (i.e. EP exemptions) until the plant specific danger of a zirconium fire is no longer a concern. As detailed in Section II of this submittal, VY is requesting that the exemptions become effective on April 15, 2016, which occurs approximately 15.4 months following the final VY reactor shutdown scheduled for December 29, 2014. VY's zirconium fire analysis (Attachment

		2) determined that 15.4 months after shutdown, it would take 10 hours for the spent fuel to reach 900°C following uncovering of the fuel.
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 VY spent fuel racks utilize Boral, rather than Boraflex, as the neutron absorbing material. As described in Section 15.2.40 of the VY UFSAR, an aging management program is in place to manage loss of material and reduction of neutron absorption capacity of Boral neutron absorption panels in the spent fuel racks. The loss of material and the reduction of the neutron-absorbing capacity will be determined through coupon testing, direct in situ testing or both. Such testing will include periodic verification of boron loss through areal density measurement of coupons or through direct in situ techniques, such as measurement of boron areal density, measurement of geometric changes in the material (blistering, pitting and bulging), and detection of gaps through blackness testing. As part of License Renewal Commitment 52, VY plans to perform neutron attenuation testing using an in-situ method prior to the end of 2014.

V. 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.

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, 17 days after shutdown, the radiological consequences of design basis accidents will not exceed the limits of the Environmental Protection Agency (EPA) Protective Action Guides at the EAB. In addition, analyses have been developed for beyond design basis events related to the SFP which show that, within 15.4 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 Protective Action Guides at the exclusion area boundary (EAB).

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 VY's ability to physically secure the site or protect special nuclear material. Physical security measures at VY are not affected by the requested exemption. Therefore, the proposed exemptions are consistent with the common defense and security.

D. 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. ENO has determined that special circumstances are present as discussed below.

Special circumstances will exist at VY because the plant will be permanently shutdown 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.

1. **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 VY 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.

   Revised radiological analyses have been developed that show that, 17 days after shutdown, the radiological consequences of design basis accidents will not exceed the limits of the EPA Protective Action Guides at the EAB. In addition, analyses have been developed for beyond design basis events related to the SFP which show that, within 15.4 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 Protective Action Guides 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.

Since the underlying purposes of the rules would continue to be achieved even with VY 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).

2. **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 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.

3. **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 shutdown 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 VY 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.

E. Precedents

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 changes to emergency plans approved by the NRC for transition to a permanently defueled condition, implemented in 1998, as identified in References 13 and 14 (with respect to exemptions requested for regulations that were in place in 1998). Specific exemption requests for regulations that involve hostile action and offsite planning are consistent with exemptions approved by the NRC for a shutdown facility with an Independent Spent Fuel Storage Installation by letter dated May 2, 2013 (Reference 15). ENO proposes that VY 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 16).

VI. 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

ENO 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. Does the proposed exemption involve a significant increase in the probability or consequences of an accident previously evaluated?

   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 EPA 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 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, ENO 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 VY emergency response organization commensurate with the reduction in consequences of radiological events that will be possible at VY once the facility is in the permanently defueled condition.

References

1. Letter, Entergy Nuclear Operations, Inc. to USNRC, "Notification of Permanent Cessation of Power Operations," BVY 13-079, dated September 23, 2013
2. NEI 99-01, "Development of Emergency Action Levels for Non-Passive Reactors," Revision 6
3. 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," October, 2013
4. Environmental Protection Agency Protective Action Guides and Planning Guidance for Radiological Incidents, Draft for Interim Use and Public Comment, dated March 2013
5. Federal Register Notice, Vol. 76, No. 226 (76 FR 72596), Enhancements to Emergency Preparedness Regulations, dated November 23, 2011
6. Letter, Entergy Nuclear Operations, Inc. to USNRC, "Technical Specifications Proposed Change No. 306, Eliminate Certain ESF Requirements during Movement of Irradiated Fuel," BVY 13-097, dated November 14, 2013

7. Commission Paper SECY-99-168, Improving Decommissioning Regulations for Nuclear Power Plants, dated June 30, 1999
8. NUREG-1738, "Technical Study of Spent Fuel Pool Accident Risk at Decommissioning Nuclear Power Plants," dated February 2001
9. Regulatory Guide 1.174, "An Approach for Using Probabilistic Risk Assessment In Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis"
10. Letter, USNRC to Vermont Yankee Nuclear Power Corporation (VYNPC), "Control of Heavy Loads (Phase I)," NVY 84-139, dated June 27, 1984
11. Letter, USNRC to VYNPC, "Completion of Phase II of 'Control of Heavy Loads at Nuclear Power Plants' NUREG 0612. (Generic Letter 85-11)," NVY 85-138, dated June 28, 1985
12. Letter, Entergy Nuclear Operations, Inc. to USNRC, "Request for Approval of Certified Fuel Handler Training Program," BVY 13-095, dated October 31, 2013
13. Letter from USNRC to Zion Nuclear Power Station, Unit Nos. 1 and 2, "Request For Approval of Defueled Station Emergency Plan and Exemption from Certain Requirements of 10 CFR 50.47, "Emergency Plans"- Zion Nuclear Power Station, Unit Nos. 1 and 2 (TAC NOS MA5253 and MA5254)," dated August 31, 1999. (ADAMS Accession No. 9909070079)
14. Letter from USNRC to Maine Yankee Atomic Power Company, "Exemption from Certain Requirements of 10 CFR 50.54(q), 10 CFR 50.47(b) and (c), And Appendix E to 10 CFR Part 50 at Maine Yankee Atomic Power Station (TAC No. MA0069)," dated September 3, 1998. (ADAMS Accession No. 9809140214)
15. Letter from USNRC to Maine Yankee Atomic Electric Company, "Response to Exemption Request For Portions of Title 10 of the Code of Federal Regulations Part 50, Appendix E, and Section 50.47 of Title 10 of the Code of Federal Regulations for the Maine Yankee Atomic Power Station (TAC NO. L24661)," dated May 2, 2013. (ADAMS Accession No. ML13112A842)
16. 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)
17. USNRC, "Consequence Study of a Beyond-Design-Basis Earthquake Affecting the Spent Fuel Pool for a U.S. Mark I Boiling Water Reactor" (Draft Report for Comment) June 2013, ADAMS Accession No. ML13133A132.
18. Draft NSIR/DPR-ISG-02, Interim Staff Guidance, Emergency Planning Exemption Requests for Decommissioning Nuclear Power Plants
19. Federal Register Notice, Vol. 60, No. 120, Emergency Planning Licensing Requirements for Independent Spent Fuel Storage Facilities (ISFSI) and Monitored Retrievable Storage Facilities (MRS), dated June 22, 1995
20. USNRC, "Integrated Rulemaking Plan for Nuclear Power Plant Decommissioning," Commission Paper SECY-00-0145, June 28, 2000 (ADAMS Accession No. ML003721626)
21. Federal Register Notice, Vol. 74, No. 94, Enhancements to Emergency Preparedness Regulations, dated May 18, 2009
22. NUREG-0696, "Functional Criteria for Emergency Response Facilities," February 1981

Attachment 2

Vermont Yankee Nuclear Power Station

Vermont Yankee Maximum Cladding Temperature Analysis for an Uncovered Spent Fuel Pool with no Air Cooling

DESIGN CONTROL SUMMARY			
CLIENT:	Entergy	UNIT: 1, 2	PAGE NO.: 1
PROJECT NAME:	Vermont Yankee Nuclear Power Plant	S&L NUCLEAR QA PROGRAM	
PROJECT NO.:	12680-007	APPLICABLE ☒ YES ☐ NO	
CALC. NO.:	2013-13016	SAFETY RELATED ☒ YES ☐ NO	
TITLE:	Vermont Yankee Maximum Cladding Temperature Analysis for an Uncovered Spent Fuel Pool with no Air Cooling		
EQUIPMENT NO.:			

IDENTIFICATION OF PAGES ADDED/REVISED/SUPERSEDED/VOIDED & REVIEW METHOD

Initial Issue. The main body is pages 1 through 10. The final page is Attachment B, page B2.

INPUTS/ ASSUMPTIONS
☒ VERIFIED
☐ UNVERIFIED

REVIEW METHOD:	Detailed Review	REV.:	0
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PREPARER:	Matthew M. Ross	DATE:	12/11/2013
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APPROVER:	Robert J. Peterson	DATE:	12-12-13

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6. **Methodology** 8
7. **Results** 11
8. **Conclusions and Recommendations** 12

Attachments: No. of Pages:
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Attachment B: Analysis 2

1. Purpose and Scope

1.1. Purpose

The purpose of this calculation is to conservatively evaluate the length of time (number of hours) it takes for uncovered spent fuel assemblies to reach the temperature at which the zirconium cladding would fail. The time to failure is calculated for various decay times after shutdown. This analysis conservatively assumes that there is no radiative or air cooling of the assemblies: the flow paths that would provide natural circulation cooling are assumed to be blocked.

1.2. Scope

The number of hours that it takes for the fuel to reach the failure temperature (the heat-up time) is determined as a function of the decay date after shutdown (the decay time).

The zirconium cladding must remain below the failure temperature. Per NUREG/CR-6451 (Ref. 2.1, see Design Input 4.1), 565 °C (1049 °F) is the lowest temperature where incipient cladding failure might occur with an expected failure at 671°C (1240°F). Per SECY-99-168 (Ref. 2.4), 800°C (1472°F) is the lowest temperature where self-sustained oxidation would occur and 565°C is the minimum temperature where clad swelling might occur. NUREG-1738 (Ref. 2.7, pgs. 3-7, A1B-5) states that runaway oxidation of zirconium occurs at 900 °C and 565°C is associated with the 10 hour creep rupture time. For this analysis, the NUREG/CR-6451 temperature (565 °C, 1049 °F) and the NUREG-1738 temperature (900 °C, 1652 °F) are the temperatures of interest for the zirconium cladding.

There are no specific acceptance criteria for this analysis, however, SECY-99-168 (Ref. 2.4) suggests that "10 hours (is) sufficient time to take mitigative action." SECY-99-168 also performed a generic analysis that found 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 to 900 °C for a PWR would occur at less than 2 years (Ref. 2.7, Fig. 2-2). A 10 hour heat up time for a BWR would occur sooner (Ref. 2.7, Fig. 2-1).

2. References

2.1. NUREG/CR-6451, "A Safety and Regulatory Assessment of Generic BWR and PWR Permanently Shutdown Nuclear Power Plants," August 1997.

2.2. Incropera, Frank P., and David P. DeWitt, Introduction to Heat Transfer, Fourth Edition, John Wiley & Sons.

2.3. Design Input Record EC47013, "Heat Generation Rate After Shutdown and Adiabatic Heatup Analysis," Rev. 0, dated 10/17/2013. (Included as Attachment A)

2.4. SECY-99-168, "Improving Decommissioning Regulations for Nuclear Power Plants," June 30, 1999.

2.5. Sargent & Lundy Calculation 2013-13824, "Decay Heat Rate Analysis for a Bounding Discharged Fuel Assembly," Rev. 0.

2.6. ARS 21002, CRP 9-4, "Alarm Response Sheets," Rev. 14.

2.7. NUREG-1738, "Technical Study of Spent Fuel Pool Accident Risk at Decommissioning Nuclear Power Plants," February 2001.

2.8. NUREG/CR-7024, "Material Property Correlations: Comparisons between FRAPCOM-3.4, FRAPTRAN 1.4, and MATPRO," March 2011.

3. Definitions

3.1. Decay Time

The decay time is the time since the reactor was shut down.

3.2. Heat-up Time

The heat-up time is the amount of time between when the fuel becomes uncovered and when the zirconium cladding reaches the failure temperatures of interest, 565 °C (1049 °F) and 900 °C (1652 °F).

4. **Input Data**

4.1. Maximum Zirconium Temperature

Several studies are presented in NUREG/CR-6451 (Ref. 2.1) discussing the maximum allowable temperature of zirconium cladding that will ensure that failure of the zirconium cladding will not occur. Per NUREG/CR-6451 (Ref. 2.1), 565 °C (1049 °F) is the lowest temperature where incipient cladding failure might occur with an expected failure at 671°C (1240°F). Per SECY-99-168 (Ref. 2.4), 800°C (1472°F) is the lowest temperature where self-sustained oxidation would occur and 565°C is the minimum temperature where clad swelling might occur. NUREG-1738 uses 900 °C (1652 °F) as the temperature where "runaway oxidation" is expected to occur and 565°C is associated with the 10 hour creep rupture time (Ref. 2.7, pgs. 3-7, A1B-5). Temperatures of 565°C and 900°C are the failure temperatures of interest for this calculation

4.2. Zirconium Properties

The specific heat of zirconium at 640 K (692 °F) is 331 J/kg-K (Ref. 2.3, pg. 11). A temperature of 692 °F is in the temperature range (roughly the midpoint for both ranges) of this analysis. From References 2.2 and 2.3, the specific heat slightly increases with an increase in temperature for most of the range of temperatures in this analysis. At higher temperatures, the zirconium would heat up more slowly. This temperature is representative of the full temperature range for this analysis.

4.3. Spent Fuel Pool Temperature

The spent fuel pool cooling system alarm response is set at 120 °F (Ref. 2.6). This is the maximum long term temperature in the SFP. As the decay time increases (i.e. as the plant has been shutdown longer) the fuel heat generation rate will be lower and the maximum spent fuel pool temperature would likely be lower.

4.4. Geometry for Limiting Assemblies

The table below shows the geometry inputs for the fuel assemblies used in this analysis. Reference 2.3 shows data for GE14 and GNF2 fuel. Both fuel types are evaluated in this analysis and the results for the worse of the two are presented.

Table 4-1: Fuel Assembly Inputs (from Ref. 2.3)

Fuel Type	GNF2	GE14
Fuel Pellet Diameter (mm)	8.88	8.76
Outer Diameter of Cladding (mm)	10.26	10.26
Inner Diameter of Cladding (mm)	9.1	8.94
Number of Full Length Rods	78	78
Number of Partial Length Rods	8 long, 6 short	14

Fuel Type	GNF2	GE14
Length of Full Length Rods (mm)	3810	3810
Length of Partial Length Rods (mm)	2591, 1372	2134
Number of Water Rods	2	2
Outer Diameter of Water Rods (cm)	2.489	2.489
Inner Diameter of Water Rods (cm)	2.337	2.337
Fuel Density (g/cm^3)	10.6	10.6
Cladding Density (lb/in^3)	0.237	0.237

4.5. Heat Load

Reference 2.5 determines the maximum heat load from a single assembly. Per Reference 2.5, a representative assembly was used for both Vermont Yankee fuel types. The assembly with the highest heat load will have the shortest heat-up time. The table showing the maximum fuel assembly heat generation rate for several years is below.

Table 4-2: Heat Generated by Highest Heat Load Assembly (Ref. 2.5)

Days Since Shutdown	Decay Heat (watts)
30	8,278
60	6,038
90	4,972
120	4,276
150	3,761
180	3,359
210	3,038
240	2,778
270	2,561
300	2,379
330	2,222
360	2,087
390	1,967
420	1,860
450	1,764
480	1,677
510	1,597
540	1,524
570	1,456
600	1,393

Days Since Shutdown	Decay Heat (watts)
630	1,335
660	1,282
690	1,230
720	1,183
750	1,139
780	1,097
810	1,058
840	1,021
870	987
900	954
930	923
960	894
990	867
1020	841
1050	817
1080	794
1110	772
1140	752
1170	732
1200	714

Per Reference 2.3, the reactor contains 368 fuel assemblies and has a 100% reactor power level of 1912 MWt. The average heat generation for an assembly in the reactor is 5.2 megawatts (= 1912 / 368). The heat loads in the table above are on the order of 0.1% to 0.01% of the peak reactor power. A heat load of 1000 watts is very low compared with heat loads commonly analyzed at nuclear power plants.

5. Assumptions

5.1. The properties of pure zirconium are used for the specific heat and density of the zirconium alloy cladding. Based on an examination of alloys of some metals (e.g. aluminum, nickel, or steel) in Table A.1 of Reference 2.2, the density and specific heat are not significantly impacted by alloying.

5.2. The heat-up time is assumed to start when the spent fuel pool has been completely drained. This is conservative. It is likely that site personnel will start to respond to an incident when draindown starts.

6. Methodology and Numerical Analysis

This analysis determines the heat-up time of the fuel assembly using the thermal capacity of materials (Based on Section 2.3 of Ref. 2.2).

$$\dot{q} = \rho \times V \times c_p \times \frac{\Delta T}{t} \qquad \text{Equation 6-1}$$

Where:

$\dot{q}$ is the heat generation rate in BTU/hr
ρ is the density of the material in lb/ft^3
V is the volume of the material in ft^3
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

For this analysis, there are two materials being heated: the uranium dioxide fuel pellets and the zirconium alloy cladding. The zirconium is in the cladding and the guide tubes, which are also being heated. The zirconium and the uranium dioxide are modeled as heating up at the same rate, so the $\Delta T/t$ will be the same for both materials.

$$\dot{q} = \frac{\Delta T}{t} \times \left(\rho_u \times V_u \times c_{p,u} + \rho_z \times V_z \times c_{p,z}\right) \qquad \text{Equation 6-2}$$

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 6-2 is solved for t.

$$t = \frac{\Delta T}{\dot{q}} \times \left(\rho_u \times V_u \times c_{p,u} + \rho_z \times V_z \times c_{p,z}\right) \qquad \text{Equation 6-3}$$

The volume of uranium dioxide is given below.

$$V_u = \left(\pi \times \frac{D_p^{\ 2}}{4}\right) N_{hr} \times L \qquad \text{Equation 6-4}$$

Where:

D_p is the diameter of the uranium dioxide pellet in ft
N_{hr} is the number of heated rods per assembly
L is the heated length of the rods in ft

The volumes of zirconium in the heated rods and in the guide tubes are given below. The length of the cladding and guide tubes that are heated is conservatively modeled as being the same as the heated length of uranium dioxide. In reality, the guide tubes and cladding are longer than the length of the uranium dioxide pellets (i.e. the heated length of the fuel is less than the total length of the assembly).

$$V_{z,c} = \left(\pi \times \frac{D_{c,o}^{2} - D_{c,i}^{2}}{4} \right) N_{hr} \times L \qquad \text{Equation 6-5}$$

$$V_{z,g} = \left(\pi \times \frac{D_{g,o}^{2} - D_{g,i}^{2}}{4} \right) N_{gt} \times L \qquad \text{Equation 6-6}$$

$$V_{z} = V_{z,g} + V_{z,c} \qquad \text{Equation 6-7}$$

Where:

$V_{z,c}$ is the volume of zirconium in the cladding of heated tubes in ft^3
$V_{z,g}$ is the volume of zirconium in the guide tubes in ft^3
$D_{c,o}$ is the outer diameter of the cladding in ft
$D_{c,i}$ is the inner diameter of the cladding in ft
$D_{g,o}$ is the outer diameter of the guide tubes in ft
$D_{g,i}$ is the inner diameter of the guide tubes in ft
N_{gt} is the number of guide tubes per assembly

The temperature increase (ΔT) for this analysis is from the initial temperature of the pool, 120 °F (Input 4.3), to the zirconium cladding failure temperatures of interest, 1049 °F and 1652 °F (Input 4.1). Since the heat generation rate is low (see Input 4.5), the diameter of the rod is small, and the spacing between rods is small, temperatures in an assembly can be modeled as uniform during the event. The heat-up time is calculated as a function of the decay time.

The use of the maximum initial pool temperature is appropriate to use as the starting cladding temperature for this analysis. In the generic analyses in both SECY-99-168 (Ref. 2.4) and NUREG-1738 (Ref. 2.7), the starting water temperature was set at 30 °C (86 °F). Both documents state that the analysis starts at the time of fuel uncovery. As stated above, the heat generation rate is low (see Input 4.5), the diameter of the rod is small, and the spacing between rods is small. Therefore, temperatures in an assembly can be modeled as uniform before the event. Therefore, using 120 °F as the starting temperature of the analysis is conservative compared to the methods used in the guidance documents.

The specific heat of uranium dioxide (UO_2) is calculated using Equation 2.2-1 in Reference 2.8.

$$Cp = \frac{K_1\theta^2\,exp(\theta/T)}{T^2\left[exp(\theta/T)-1\right]^2} + K_2T + \frac{YK_3E_D}{2RT^2}exp(-E_D/RT) \qquad \text{Equation 6-8}$$

Where:

Cp is the specific heat in J/kg-K
K_1, K_2, K_3, are constants for UO_2 of 296.7 J/kg-K, $2.43\text{x}10^{-2}$ J/kg-K^2, and $8.745\text{x}10^7$ J/kg
θ is the Einstein temperature and is 535.285 K for UO_2
T is the temperature of the UO_2, in K
Y is the oxygen to metal ratio (2 for UO_2)
E_D is the activation energy for Frenkel Defects and is $1.577\text{x}10^5$ J/mol
R is the universal gas constant 8.3143 J/mol-K

The third term in this summation is negligible at temperatures below 900 K and is ignored. Ignoring the term is conservative because it very slightly reduces the specific heat, which reduces the heat-up time. Per Figure 2.2-1 in Reference 2.8, the heat capacity of uranium is concave over the range of temperatures of this analysis (120 °F is 322 K and 1049 °F is 838 K). Therefore, a conservatively low value for specific heat for the range of temperatures considered is selected as the average of the specific heats at those two temperatures.

$$Cp = \frac{296.7\times 535.285^2\,exp(535.285/322)}{322^2\left[exp(535.285/322)-1\right]^2} + 2.43\times 10^{-2}\times 322 = 245\,\tfrac{J}{kg-K}$$

$$Cp = \frac{296.7\times 535.285^2\,exp(535.285/838)}{838^2\left[exp(535.285/838)-1\right]^2} + 2.43\times 10^{-2}\times 838 = 307\,\tfrac{J}{kg-K}$$

$$Cp = \left(245\,\tfrac{J}{kg-K} + 307\,\tfrac{J}{kg-K}\right)\times \tfrac{1}{2} = 276\,\tfrac{J}{kg-K}$$

7. Results

The detailed analysis is documented in Attachment B. Per Attachment B, GNF2 fuel heats up slightly faster than GE14 fuel. The results shown in Table 7-1 below are for GNF2 fuel which bound the results for GE14 fuel.

Table 7-1: Results

Decay Time (Days since Shutdown)	Heat-Up Time to 565 °C (hours)	Heat-Up Time to 900 °C (hours)
240	3.7	6.2
300	4.4	7.2
360	5.0	8.2
420	5.6	9.2
480	6.2	10.2
540	6.8	11.3
600	7.5	12.3
660	8.1	13.4
720	8.8	14.5
780	9.5	15.6
840	10.2	16.8
900	10.9	18.0
960	11.6	19.2

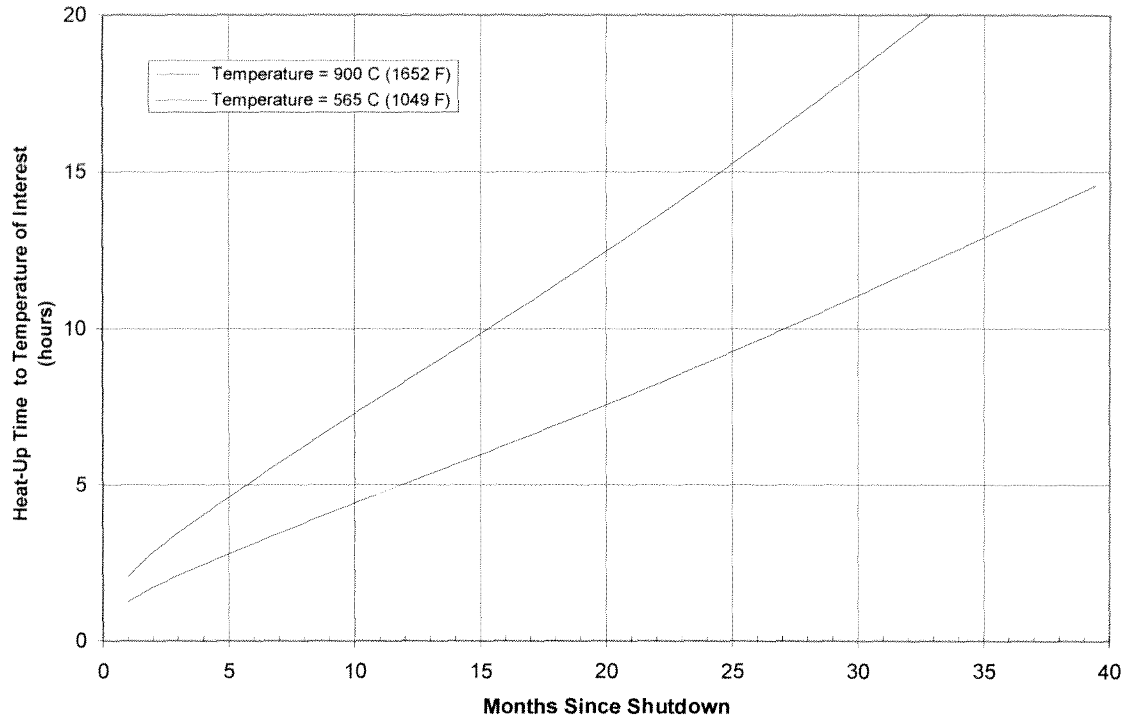
By interpolating, the heat-up time to 565 °F is 10 hours at a decay time of 823 days (27 months) after shutdown. By interpolating the heat-up time to 900 °C is 10 hours at a decay time of 468 days (15.4 months) after shutdown.

The 10 hour heat-up time to a temperature of 565 °C (1049 °F) occurs at a decay time of under 2.5 years, while the 10 hour heat-up time to a temperature of 900 °C (1652 °F) occurred at a decay time of under 1.5 years.

As stated above, SECY-99-168 performed a generic analysis that found 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 to 900 °C for a PWR would occur at less than 2 years (Ref. 2.7, Fig. 2-2) and that the 10 hour heat up time for a BWR would occur sooner (Ref. 2.7, Fig. 2-1). The results calculated here are more favorable than these generic analyses since the calculated decay time for a 10 hour heat-up time is less than what the generic analyses predict.

A plot showing the heat-up time to the temperatures of interest as a function of decay time is Figure 7-1.

Figure 7-1: Heat-Up Time vs. Decay Time



8. Conclusions and Recommendations

The Vermont Yankee results are more favorable than the generic analyses performed for SECY-99-168 (Ref. 2.4) and NUREG-1738 (Ref. 2.7). There are no acceptance criteria for this analysis. There are no specific recommendations for this analysis.

Attachment 3

Vermont Yankee Nuclear Power Station

List of Regulatory Commitments

List of Regulatory Commitments

This table identifies actions discussed in this letter for which Entergy commits to perform. Any other actions discussed in this submittal are described for the NRC's information and are **not** commitments.

COMMITMENT	TYPE (Check one)		SCHEDULED COMPLETION DATE (If Required)
	ONE-TIME ACTION	CONTINUING COMPLIANCE	
Revise UFSAR to include a description of how the VY spent fuel pool design and operational characteristics meets or compares with the NUREG-1738 Industry Decommissioning Commitments (IDC) and Staff Decommissioning Assumptions (SDA).	x		Complete in accordance with next scheduled UFSAR update following exemption approval
VY will establish the timelines required to initiate the various onsite and offsite SFP makeup sources based on expected system configurations and availability. (SDA 2)	x		Prior to permanent cessation of operations