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

March 30, 2015

Mr. Terry D. Hobbs
Decommissioning General Manager
Crystal River Nuclear Plant (NA2C)
15760 W. Power Line Street
Crystal River, FL 34428-6708

SUBJECT: CRYSTAL RIVER UNIT 3 – EXEMPTIONS FROM CERTAIN EMERGENCY PLANNING REQUIREMENTS AND RELATED SAFETY EVALUATION (TAC NO. MF2981)

Dear Mr. Hobbs:

The U.S. Nuclear Regulatory Commission (NRC) has approved the enclosed exemptions from specific requirements of Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50.47, "Emergency plans," and Appendix E, "Emergency Planning and Preparedness for Production and Utilization Facilities," to 10 CFR Part 50. This action is in response to your application for exemptions dated September 26, 2013, "Crystal River Unit 3 – License Amendment Request #315, Revision 0, Permanently Defueled Emergency Plan and Emergency Action Level Scheme and Request for Exemption to Certain Radiological Emergency Response Plan Requirements Defined by 10 CFR 50," supplemented by letters dated March 28, May 7, May 23 and August 28, 2014.

A copy of the exemptions and the NRC staff's safety evaluation are also enclosed. The exemptions will be forwarded to the Office of the Federal Register for publication.

Sincerely,

Thomas J. Wengert for

Michael D. Orenak, Project Manager
Plant Licensing IV-2 and Decommissioning
Transition Branch
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-302

Enclosures:
1. Exemptions
2. Safety Evaluation

cc w/encls: Distribution via Listserv

ENCLOSURE 1

EXEMPTIONS

DUKE ENERGY FLORIDA, INC.

CRYSTAL RIVER UNIT 3 NUCLEAR GENERATING PLANT

DOCKET NO. 50-302

[7590-01-P]

NUCLEAR REGULATORY COMMISSION

[Docket No. 50-302; NRC-20XX-XXXX]

Duke Energy Florida, Inc.; Crystal River Unit 3 Nuclear Generating Station

AGENCY: Nuclear Regulatory Commission.

ACTION: Exemption; issuance.

SUMMARY: The U.S. Nuclear Regulatory Commission (NRC) is granting exemptions in response to a request from Duke Energy Florida, Inc. (DEF or the licensee) regarding certain emergency planning (EP) requirements. The exemptions will eliminate the requirements to maintain an offsite radiological emergency plan and reduce the scope of onsite emergency planning activities at the Crystal River Unit 3 Nuclear Generating Station (CR-3) based on the reduced risks of accidents that could result in an offsite radiological release at a decommissioning nuclear power reactor.

ADDRESSES: Please refer to Docket ID <**INSERT:** NRC-20YY-XXXX> when contacting the NRC about the availability of information regarding this document. You may obtain publicly-available information related to this document using any of the following methods:

- **Federal Rulemaking Web Site:** Go to http://www.regulations.gov and search for Docket ID <**INSERT:** NRC-20YY-XXXX>. Address questions about NRC dockets to Carol Gallagher; telephone: 301-415-3463; e-mail: Carol.Gallagher@nrc.gov. For technical

questions, contact the individual(s) listed in the FOR FURTHER INFORMATION CONTACT section of this document.

- **NRC's Agencywide Documents Access and Management System (ADAMS):** You may obtain publicly available documents online in the ADAMS Public Documents collection at http://www.nrc.gov/reading-rm/adams.html. To begin the search, select "ADAMS Public Documents" and then select "Begin Web-based ADAMS Search." For problems with ADAMS, please contact the NRC's Public Document Room (PDR) reference staff at 1-800-397-4209, 301-415-4737, or by e-mail to pdr.resource@nrc.gov. The ADAMS accession number for each document referenced in this document (if that document is available in ADAMS) is provided the first time that a document is referenced.
- **NRC's PDR:** You may examine and purchase copies of public documents at the NRC's PDR, Room O1-F21, One White Flint North, 11555 Rockville Pike, Rockville, Maryland 20852.

FOR FURTHER INFORMATION CONTACT: Michael Orenak, Office of Nuclear Reactor Regulation, U.S. Nuclear Regulatory Commission, Washington DC 20555-0001; telephone: 301-415-3229; e-mail: Michael.Orenak@nrc.gov.

I. Background.

The CR-3 facility is a decommissioning power reactor located in Citrus County, Florida. The licensee, DEF, is the holder of CR-3 Facility Operating License No. DPR-72. The license provides, among other things, that the facility is subject to all rules, regulations, and orders of the NRC now or hereafter in effect.

By letter dated February 20, 2013 (ADAMS Accession No. ML13056A005), DEF submitted to the NRC a certification in accordance with section 50.82(a)(1)(i) of Title 10 of the

Code of Federal Regulations (10 CFR) indicating it would permanently cease power operations, and 10 CFR 50.82(a)(1)(ii) that it had permanently defueled the reactor vessel at CR-3. On May 28, 2011, DEF completed the final removal of fuel from the reactor vessel at CR-3. As a permanently shutdown and defueled facility, and in accordance with section 50.82(a)(2), DEF is no longer authorized to operate the reactor or emplace nuclear fuel into the reactor vessel. CR-3 is still authorized to possess and store irradiated (i.e., spent) nuclear fuel. The spent fuel is currently being stored onsite in a spent fuel pool (SFP).

During normal power reactor operations, the forced flow of water through the reactor coolant system (RCS) removes heat generated by the reactor. The RCS, operating at high temperatures and pressures, transfers this heat through the steam generator tubes converting non-radioactive feedwater to steam, which then flows to the main turbine generator to produce electricity. Many of the accident scenarios postulated in the updated safety analysis reports (USARs) for operating power reactors involve failures or malfunctions of systems, which could affect the fuel in the reactor core, which in the most severe postulated accidents, would involve the release of large quantities of fission products. With the permanent cessation of reactor operations at CR-3 and the permanent removal of the fuel from the reactor vessel, such accidents are no longer possible. The reactor, RCS, and supporting systems are no longer in operation and have no function related to the storage of the spent fuel. Therefore, EP provisions for postulated accidents involving failure or malfunction of the reactor, RCS, or supporting systems are no longer applicable.

Based on the time that CR-3 has been permanently shutdown (approximately 64 months), there is no longer any possibility of an offsite radiological release from a design-basis accident that could exceed the U.S. Environmental Protection Agency's (EPA) Protective Action Guidelines (PAGs) at the exclusion area boundary.

The EP requirements of 10 CFR 50.47, "Emergency plans," and appendix E to 10 CFR part 50, "Emergency Planning and Preparedness for Production and Utilization Facilities,"

continue to apply to nuclear power reactors that have permanently ceased operation and have removed all fuel from the reactor vessel. There are no explicit regulatory provisions distinguishing EP requirements for a power reactor that is permanently shutdown and defueled from a reactor that is authorized to operate. In order for DEF to modify the CR-3 emergency plan to reflect the reduced risk associated with the permanently shutdown and defueled condition of CR-3, certain exemptions from the EP regulations must be obtained before the CR-3 emergency plan can be amended.

II. Request/Action.

By letter dated September 26, 2013 (ADAMS Accession No. ML13274A584), "Crystal River Unit 3 - License Amendment Request #315, Revision 0, Permanently Defueled Emergency Plan and Emergency Action Level Scheme, and Request for Exemption to Certain Radiological Emergency Response Plan Requirements Defined by 10 CFR 50," DEF requested exemptions from certain EP requirements of 10 CFR part 50 for CR-3. More specifically, DEF requested exemptions from certain planning standards in 10 CFR 50.47(b) regarding onsite and offsite radiological emergency plans for nuclear power reactors; from certain requirements in 10 CFR 50.47(c)(2) that require establishment of plume exposure and ingestion pathway emergency planning zones for nuclear power reactors; and from certain requirements in 10 CFR 50, appendix E, section IV, which establishes the elements that make up the content of emergency plans. In a letter dated March 28, 2014 (ADAMS Accession No. ML14098A072), DEF provided responses to the NRC staff's request for additional information (RAI) concerning the proposed exemptions. In a letter dated May 7, 2014 (ADAMS Accession No. ML14139A006), DEF provided an additional supplemental response to a separate set of RAIs, which contained information applicable to the SFP inventory makeup strategies for mitigating the potential loss of water inventory due to a beyond-design-basis accident. In a letter dated

August 28, 2014 (ADAMS Accession No. ML14251A237), CR-3 provided a supplement, which amended its request to align with the exemptions recommended by the NRC staff and approved by the Commission in staff requirements memorandum (SRM) to SECY-14-0066, "Request by Dominion Energy Kewaunee, Inc. for Exemptions from Certain Emergency Planning Requirements," dated August 7, 2014 (ADAMS Accession No. ML14219A366). The information provided by DEF included justifications for each exemption requested. The exemptions requested by DEF will eliminate the requirements to maintain formal offsite radiological emergency plans, reviewed by the Federal Emergency Management Agency (FEMA) under the requirements of 44 CFR part 350, and reduce the scope of onsite emergency planning activities. DEF stated that application of all of the standards and requirements in 10 CFR 50.47(b), 10 CFR 50.47(c) and 10 CFR part 50, appendix E is not needed for adequate emergency response capability based on the reduced risks at the permanently shutdown and defueled facility. If offsite protective actions were needed for a very unlikely accident that could challenge the safe storage of spent fuel at CR-3, provisions exist for offsite agencies to take protective actions using a comprehensive emergency management plan (CEMP) under the National Preparedness System to protect the health and safety of the public. A CEMP in this context, also referred to as an emergency operations plan (EOP), is addressed in FEMA's Comprehensive Preparedness Guide 101, "Developing and Maintaining Emergency Operations Plans." Comprehensive Preparedness Guide 101 is the foundation for State, territorial, Tribal, and local emergency planning in the United States. It promotes a common understanding of the fundamentals of risk-informed planning and decision making and helps planners at all levels of government in their efforts to develop and maintain viable, all-hazards, all-threats emergency plans. An EOP is flexible enough for use in all emergencies. It describes how people and property will be protected; details who is responsible for carrying out specific actions; identifies the personnel, equipment, facilities, supplies and other resources available; and outlines how all

actions will be coordinated. A CEMP is often referred to as a synonym for "all hazards planning."

III. Discussion.

In accordance with 10 CFR 50.12, "Specific exemptions," the Commission may, upon application by any interested person or upon its own initiative, grant exemptions from the requirements of 10 CFR part 50 when: (1) the exemptions are authorized by law, will not present an undue risk to public health or safety, and are consistent with the common defense and security; and (2) any of the special circumstances listed in 10 CFR 50.12(a)(2) are present. These special circumstances include, among other things, that the 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.

As noted previously, the current EP regulations contained in 10 CFR 50.47(b) and appendix E to 10 CFR part 50 apply to both operating and shutdown power reactors. The NRC has consistently acknowledged that the risk of an offsite radiological release at a power reactor that has permanently ceased operations and removed fuel from the reactor vessel is significantly lower, and the types of possible accidents are significantly fewer, than at an operating power reactor. However, current EP regulations do not recognize that once a power reactor permanently ceases operation, the risk of a large radiological release from credible emergency accident scenarios is significantly reduced. The reduced risk for any significant offsite radiological release is based on two factors. One factor is the elimination of accidents applicable only to an operating power reactor, resulting in fewer credible accident scenarios. The second factor is the reduced short-lived radionuclide inventory and decay heat production due to radioactive decay. Due to the permanently defueled status of the reactor, no new spent fuel will be added to the SFP and the radionuclides in the current spent fuel will continue to

decay as the spent fuel ages. The irradiated fuel will produce less heat due to radioactive decay, increasing the available time to mitigate the SFP inventory loss. The NRC's NUREG-1738, "Technical Study of Spent Fuel Pool Accident Risk at Decommissioning Nuclear Power Plants," dated February 2001 (ADAMS Accession No. ML010430066), confirmed that for permanently shutdown and defueled power reactors bounded by the assumptions and conditions in the report, the risk of offsite radiological release is significantly less than for an operating power reactor.

EP exemptions similar to those requested by DEF were granted to permanently shutdown and defueled power reactor licensees, such as for Zion Nuclear Power Station in 1999 (ADAMS Legacy Accession No. 9909070079) and Kewaunee Power Station in 2014 (ADAMS Accession No. ML14261A223). However, the exemptions did not relieve the licensees of all EP requirements. Rather, the exemptions allowed the licensees to modify their emergency plans commensurate with the credible site-specific risks that were consistent with a permanently shutdown and defueled status. Specifically, approval of the prior exemptions was based on demonstrating that: (1) the radiological consequences of design-basis accidents would not exceed the limits of the EPA PAGs at the exclusion area boundary, and; (2) in the unlikely event of a beyond-design-basis accident resulting in a loss of all modes of heat transfer from the fuel stored in the SFP, there is sufficient time to initiate appropriate mitigating actions, and if needed, for offsite authorities to implement offsite protective actions using a CEMP approach to protect the health and safety of the public.

With respect to design-basis accidents at CR-3, the licensee provided analyses demonstrating that none would warrant an offsite radiological emergency plan meeting the requirements of 10 CFR part 50.

With respect to beyond-design-basis accidents at CR-3, the licensee analyzed two bounding beyond-design-basis accidents that have a potential for a significant offsite release. One of these beyond-design-basis accidents involves a complete loss of SFP water inventory,

where cooling of the spent fuel would be primarily accomplished by natural circulation of air through the uncovered spent fuel assemblies. The licensee's analysis of this accident shows that as of September 26, 2013, air cooling of the spent fuel assemblies was sufficient to keep the fuel within a safe temperature range indefinitely without fuel damage or offsite radiological release. The second beyond-design-basis accident analysis performed by the licensee could not completely rule out the possibility of a radiological release from a SFP. This more limiting analysis assumes an incomplete drain down of the SFP water, or some other catastrophic event (such as a complete drainage of the SFP with rearrangement of spent fuel rack geometry and/or the addition of rubble to the SFP) that would effectively impede any decay heat removal through all possible modes of cooling. This analysis is commonly referred to as an adiabatic heat-up. The licensee's analysis demonstrates that as of September 26, 2013, there would be at least 19.7 hours under adiabatic heat-up conditions before the spent fuel cladding would reach a temperature where the potential for a significant offsite radiological release could occur. This analysis conservatively does not consider the period of time from the initiating event causing a loss of SFP water inventory until all cooling means are lost.

The NRC staff has verified DEF's analyses and its calculations. The analyses provide reasonable assurance that in granting the requested exemptions to DEF, there is no design-basis accident that will result in an offsite radiological release exceeding the EPA PAGs at the exclusion area boundary. In the unlikely event of a beyond-design-basis accident affecting the SFP that results in adiabatic heat-up conditions (i.e., a complete loss of heat removal via all modes of heat transfer), the NRC staff has reviewed and verified that there will be at least 19.7 hours available before an offsite release might occur and, therefore, at least 19.7 hours to initiate appropriate mitigating actions to restore a means of heat removal to the spent fuel. If a radiological release were projected to occur under this unlikely scenario, a minimum of 10 hours is considered sufficient time for offsite authorities to implement protective actions using a CEMP approach to protect the health and safety of the public.

The NRC staff reviewed the licensee's justification for the requested exemptions against the criteria in 10 CFR 50.12(a) and the bases for prior EP exemption request approvals, as discussed above. The staff determined, as described below, that the criteria in 10 CFR 50.12(a) are met, and that the exemptions should be granted. Assessment of the DEF EP exemptions is described in SECY-14-0118, "Request by Duke Energy Florida, Inc., for Exemptions from Certain Emergency Planning Requirements," dated October 29, 2014 (ADAMS Accession No. ML14219A444). The Commission approved the NRC staff's intention to grant the exemptions in the SRM to SECY-14-0118, dated December 30, 2014 (ADAMS Accession No. ML14364A111). Descriptions of the specific exemptions requested by DEF and the NRC staff's basis for granting each exemption are provided in SECY-14-0118 and summarized in a table at the end of this document. The staff's detailed review and technical basis for the approval of the specific EP exemptions are provided in the NRC staff's safety evaluation enclosed in an NRC letter dated March 30, 2015 (ADAMS Accession No. ML15058A906).

A. Authorized by Law

The licensee has proposed exemptions from certain EP requirements in 10 CFR 50.47(b), 10 CFR 50.47(c)(2), and 10 CFR 50, appendix E, section IV, that would allow DEF to revise the CR-3 Emergency Plan to reflect the permanently shutdown and defueled condition of the station. As stated above, in accordance with 10 CFR 50.12, the Commission may, upon application by any interested person or upon its own initiative, grant exemptions from the requirements of 10 CFR part 50. The NRC staff has determined that granting of the licensee's proposed exemptions will not result in a violation of the Atomic Energy Act of 1954, as amended, or the NRC's regulations. Therefore, the exemptions are authorized by law.

B. No Undue Risk to Public Health and Safety

As stated previously, DEF provided analyses that show the radiological consequences of design-basis accidents will not exceed the limits of the EPA PAGs at the exclusion area boundary. Therefore, formal offsite radiological emergency plans required under 10 CFR part 50 are no longer needed for protection of the public beyond the exclusion area boundary.

Although very unlikely, there is one postulated beyond-design-basis accident that might result in significant offsite radiological releases. However, NUREG-1738 confirms that the risk of beyond-design-basis accidents is greatly reduced at permanently shutdown and defueled reactors. The NRC staff's analyses concludes that the event sequences important to risk at permanently shutdown and defueled power reactors are limited to large earthquakes and cask drop events. For EP assessments, this is an important difference relative to operating power reactors where typically a large number of different sequences make significant contributions to risk. Per NUREG-1738, relaxation of offsite EP requirements under 10 CFR part 50 a few months after shutdown resulted in only a small change in risk.

NUREG-1738 further concludes that the change in risk due to relaxation of offsite EP requirements is small because the overall risk is low, and because even under current EP requirements for operating power reactors, EP was judged to have marginal impact on evacuation effectiveness in the severe earthquakes that dominate SFP risk. Specifically, for ground motion levels that correspond to SFP failure in the central and eastern United States, it is expected that electrical power would be lost and more than half of the bridges and buildings (including those housing communication systems and emergency response equipment) would be unsafe even for temporary use within at least 10 miles of the plant. This approach is also consistent with previous Commission rulings on San Onofre and Diablo Canyon in which the Commission found that for those risk-dominant earthquakes that cause very severe damage to both the plant and the offsite area, emergency response would have marginal benefit because of offsite damage. All other sequences including cask drops (for which offsite radiological

emergency plans are expected to be more effective) are too low in likelihood to have a significant impact on risk.

Therefore, granting exemptions that eliminate the requirements of 10 CFR part 50 to maintain offsite radiological emergency plans and reducing the scope of onsite emergency planning activities will not present an undue risk to the public health and safety.

C. Consistent with the Common Defense and Security

The requested exemptions by DEF only involve EP requirements under 10 CFR part 50 and will allow DEF to revise the CR-3 Emergency Plan to reflect the permanently shutdown and defueled condition of the facility. Physical security measures at CR-3 are not affected by the requested EP exemptions. The discontinuation of formal offsite radiological emergency plans and the reduction in scope of the onsite emergency planning activities at CR-3 will not adversely affect DEF's ability to physically secure the site or protect special nuclear material. Therefore, the proposed exemptions are consistent with common defense and security.

D. Special Circumstances

Special circumstances, in accordance with 10 CFR 50.12(a)(2)(ii), are present whenever application of the regulation in the particular circumstances is not necessary to achieve the underlying purpose of the rule. The underlying purpose of 10 CFR 50.47(b), 10 CFR 50.47(c)(2), and 10 CFR part 50, appendix E, section IV, is to provide 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 radiological emergency plans. The standards and requirements in these regulations were developed by considering the risks associated with operation of a power reactor at its licensed

full-power level. These risks include the potential for a reactor accident with offsite radiological dose consequences.

As discussed previously, because CR-3 is permanently shutdown and defueled, there is no longer a risk of offsite radiological release from a design-basis accident and the risk of a significant offsite radiological release from a beyond-design-basis accident is greatly reduced when compared to an operating power reactor. The NRC staff has confirmed the reduced risks at CR-3 by comparing the generic risk assumptions in the analyses in NUREG-1738 to site specific conditions at CR-3 and determined that the risk values in NUREG-1738 bound the risks presented by CR-3. Furthermore, the staff has recently concluded in NUREG-2161, "Consequence Study of a Beyond-Design-Basis Earthquake Affecting the Spent Fuel Pool for a U.S. Mark I Boiling Water Reactor," dated September 2014 (ADAMS Accession No. ML14255A365), that, consistent with earlier research studies, SFPs are robust structures that are likely to withstand severe earthquakes without leaking cooling water and potentially uncovering the spent fuel. The NUREG-2161 study shows the likelihood of a radiological release from spent fuel after the analyzed severe earthquake at the reference plant to be about one time in 10 million years or lower.

The licensee has analyzed site-specific spent fuel air-cooling and adiabatic heat-up beyond-design-basis accident scenarios to determine the risk of cladding damage, and the time to rapid cladding oxidation. The air-cooling analysis shows that as of September 26, 2013, in the event of a complete SFP drain down due to a loss of water inventory, assuming that natural circulation of air through the spent fuel racks was available, the peak fuel clad temperature would remain below 1049°F (565°C), the temperature at which incipient cladding failure may occur. Therefore, in this postulated accident, fuel cladding remains intact.

The beyond-design-basis adiabatic heat-up accident analysis of the spent fuel evaluates a postulated condition involving a very unlikely scenario where the SFP is drained in such a way that all modes of cooling or heat transfer are assumed to be unavailable. DEF analysis of this

beyond-design-basis accident shows that as of September 26, 2013, 19.7 hours would be available between the time the fuel is uncovered (at which time adiabatic heat-up begins), until the fuel cladding reaches a temperature of 1652°F (900°C), the temperature associated with rapid cladding oxidation and the potential for a significant radiological release.

Exemptions from the offsite EP requirements in 10 CFR part 50 have previously been approved by the NRC when the site-specific analyses show that at least 10 hours is available following a loss of SFP coolant inventory accident with no air cooling (or other methods of removing decay heat) until cladding of the hottest fuel assembly reaches the zirconium rapid oxidation temperature. The NRC staff concluded in its previously granted exemptions, as it does with the DEF requested EP exemptions, that if a minimum of 10 hours is available to initiate mitigative actions consistent with plant conditions, or if needed, for offsite authorities to implement protective actions using a CEMP approach, then formal offsite radiological emergency plans, required under 10 CFR part 50, are not necessary at permanently shutdown and defueled facilities.

Additionally, DEF committed to maintaining SFP makeup strategies in its letter to the NRC dated May 7, 2014 (ADAMS Accession No. ML14139A006). The multiple strategies for providing makeup to the SFP include: using existing plant systems for inventory makeup; supplying water through hoses to connections to the existing SFP piping using the diesel-driven fire service pump; and using a diesel-driven portable pump to take suction from CR-3 intake and discharge canals. These strategies will continue to be required as license condition 2.C.(14), "Mitigation Strategy License Condition." Considering the very low probability of beyond-design-basis accidents affecting the SFP, these diverse strategies provide multiple methods to obtain additional makeup or spray to the SFP before the onset of any postulated offsite radiological release.

For all the reasons stated above, the NRC staff finds that the licensee's requested exemptions to meet the underlying purpose of all of the standards in 10 CFR 50.47(b), and

requirements in 10 CFR 50.47(c)(2) and 10 CFR part 50, appendix E, acceptably satisfy the special circumstances in 10 CFR 50.12(a)(2)(ii) in view of the greatly reduced risk of offsite radiological consequences associated with the permanently shutdown and defueled state of the CR-3 facility.

The NRC staff has concluded that the exemptions being granted by this action will maintain an acceptable level of emergency preparedness at CR-3 and, if needed, that there is reasonable assurance that adequate offsite protective measures can and will be taken by State and local government agencies using a CEMP approach in the unlikely event of a radiological emergency at the CR-3 facility. Since the underlying purposes of the rules, as exempted, would continue to be achieved, even with the elimination of the requirements under 10 CFR part 50 to maintain formal offsite radiological emergency plans and reduction in the scope of the onsite emergency planning activities at CR-3, the special circumstances required by 10 CFR 50.12(a)(2)(ii) exist.

E. Environmental Considerations

In accordance with 10 CFR 51.31(a), the Commission has determined that the granting of this exemption will not have a significant effect on the quality of the human environment as discussed in the NRC staff's Finding of No Significant Impact and associated Environmental Assessment published March 2, 2015 (80 FR 11233).

IV. Conclusions.

Accordingly, the Commission has determined, pursuant to 10 CFR 50.12(a), that DEF's request for exemptions from certain EP requirements in 10 CFR 50.47(b), 10 CFR 50.47(c)(2), and 10 CFR part 50, appendix E, section IV, and as summarized in the table at the end of this document, are authorized by law, will not present an undue risk to the public health and safety,

and are consistent with the common defense and security. Also, special circumstances are present. Therefore, the Commission hereby grants DEF exemptions from certain EP requirements of 10 CFR 50.47(b), 10 CFR 50.47(c)(2), and 10 CFR part 50, appendix E, section IV, as discussed and evaluated in detail in the staff's safety evaluation dated March 30, 2015. The exemptions are effective as of March 30, 2015.

Dated at Rockville, Maryland, this 30th day of March, 2015.

For the Nuclear Regulatory Commission.

Michele G. Evans

Michele G. Evans, Director,
Division of Operating Reactor Licensing,
Office of Nuclear Reactor Regulation.

Table of Exemptions Granted to DEF.

10 CFR 50.47	NRC Staff Basis for Exemption
10 CFR 50.47(b) The NRC is granting exemptions from portions of the rule language that would otherwise require offsite emergency response plans.	In the Statement of Considerations (SOC) for the final rule for emergency planning (EP) requirements for independent spent fuel storage installations (ISFSIs) and for monitor retrievable storage installations (MRS) (60 *Federal Register* (FR) 32430; June 22, 1995), the Commission responded to comments concerning offsite EP for ISFSIs or a MRS and concluded that, "the offsite consequences of potential accidents at an ISFSI or a MRS would not warrant establishing Emergency Planning Zones [EPZ]." In a nuclear power reactor's permanently defueled state, the accident risks are more similar to an ISFSI or a MRS than an operating nuclear power plant. The EP program would be similar to that required for an ISFSI under section 72.32(a) of 10 CFR when fuel stored in the spent fuel pool (SFP) has more than 5 years of decay time and would not change substantially when all the fuel is transferred from the SFP to an onsite ISFSI. Exemptions from offsite EP requirements have previously been approved when the site-specific analyses show that at least 10 hours is available until the hottest fuel assembly reaches 900°C from a partial drain-down event without any spent fuel cooling. The technical basis that underlied the approval of the exemption request is based partly on the analysis of a time period that spent fuel stored in the SFP is unlikely to reach the zirconium ignition temperature in less than 10 hours. This time period is based on a heat-up calculation, which uses several simplifying assumptions. Some of these assumptions are conservative (adiabatic conditions), while others are non-conservative (no oxidation below 900°C). Weighing the conservatisms and non-conservatisms, the NRC staff judges that this calculation reasonably represents conditions, which may occur in the event of an SFP accident. The staff concluded that if 10 hours were available to initiate mitigative actions, or if needed, offsite protective actions using a comprehensive emergency management plan (CEMP), formal offsite radiological emergency plans are not necessary

	for these permanently defueled nuclear power reactor licensees. As supported by the licensee's SFP analysis, the NRC staff believes an exemption to the requirements for formal offsite radiological emergency plans is justified for a zirconium fire scenario considering the low likelihood of this event together with time available to take mitigative or protective actions between the initiating event and before the onset of a postulated fire. The Duke Energy Florida, Inc. (DEF) analysis has demonstrated that due to the considerable time since shutdown, approximately 4 years as of the date of the analysis, the radiological consequences of design-basis accidents will not exceed the limits of the U.S. Environmental Protection Agency's (EPA) Protective Action Guidelines (PAGs) at the exclusion area boundary. These analyses also show that for beyond-design-basis events where the SFP is drained, air cooling will prevent the fuel from reaching the lowest temperature where incipient cladding failure may occur (565°C). In the event that air cooling is not possible, 19.7 hours is available to take mitigative or, if needed, offsite protective actions using a CEMP from the time the fuel is uncovered until it reaches the auto-ignition temperature of 900°C. DEF has also furnished information on its SFP inventory makeup strategies for mitigating the loss of water inventory. Several sources of makeup to the pools are available, such as the fire service system, using the diesel-driven fire service pump for loss of electrical power. If available fresh water sources are depleted, salt water sources with inexhaustible inventory from the Crystal River Unit 3 (CR-3) intake and discharge canal, using portable diesel powered pumps are available. Pool inventory addition can be implemented without accessing the elevation of the pool deck. In a letter dated May 7, 2014, "Crystal River Unit 3 - Response to Requests for Additional Information and Supplement 1 to License Amendment Request #316, Revision 0" (ADAMS Accession No. ML14139A006), DEF withdrew its

	request to remove License Condition 2.C.(14), "Mitigation Strategy License Condition," from its Facility Operating License. This license condition requires CR-3 to maintain its SFP inventory makeup strategies as discussed above.
10 CFR 50.47(b)(1) The NRC is granting exemptions from portions of the rule language that would otherwise require the need for Emergency Planning Zones (EPZs).	Refer to basis for 10 CFR 50.47(b).
10 CFR 50.47(b)(3) The NRC is granting exemptions from portions of the rule language that would otherwise require the need for an Emergency Operations Facility (EOF).	Considering the time available to take mitigative or, if needed, offsite protective actions using a CEMP between the initiating event and before the onset of a postulated fire, decommissioning power reactors present a low likelihood of any credible accident resulting in a radiological release. As such, an emergency operations facility would not be required. The "nuclear island," control room, or other onsite location can provide for the communication and coordination with offsite organizations for the level of support required. Also refer to basis for 10 CFR 50.47(b).
10 CFR 50.47(b)(4) The NRC is granting exemptions from portions of the rule language that would otherwise require reference to formal offsite radiological emergency response plans.	Considering the time available to take mitigative or if needed, offsite protective actions using a CEMP between the initiating event and before the onset of a postulated fire, decommissioning power reactors present a low likelihood of any credible accident resulting in a radiological release. As such, formal offsite radiological emergency response plans are not required. The Nuclear Energy Institute (NEI) document NEI 99-01, "Development of Emergency Action Levels for Non-Passive Reactors" (Revision 6), was found to be an acceptable method for development of emergency action levels (EALs) and was endorsed by the U.S. Nuclear Regulatory Commission (NRC) in a letter dated March 28, 2013 (ADAMS Accession No. ML12346A463). NEI 99-01 provides EALs for non-passive operating nuclear power reactors, permanently defueled reactors, and ISFSIs. Also refer to basis for 10 CFR 50.47(b).

10 CFR 50.47(b)(5) The NRC is granting exemptions from portions of the rule language that would otherwise require early notification of the public and a means to provide instructions to the public within the plume exposure pathway EPZ.	Refer to basis for 10 CFR 50.47(b).
10 CFR 50.47(b)(6) The NRC is granting exemptions from portions of the rule language that would otherwise require prompt communications with the public.	Refer to basis for 10 CFR 50.47(b).
10 CFR 50.47(b)(7) The NRC is granting exemptions from portions of the rule language that would otherwise require information to be made available to the public on a periodic basis about how they will be notified and what their initial protective actions should be.	Refer to basis for 10 CFR 50.47(b).
10 CFR 50.47(b)(9) The NRC is granting exemptions from portions of the rule language that would otherwise require the capability for monitoring offsite consequences.	Refer to basis for 10 CFR 50.47(b).
10 CFR 50.47(b)(10) The NRC is granting exemptions from portions of the rule language that would reduce the range of protective actions developed for radiological emergencies. Consideration of evacuation, sheltering, or the use of potassium iodide will no longer be necessary. Evacuation time estimates (ETEs) will no longer need to developed or updated. Protective actions for the ingestion exposure pathway EPZ will not need to be developed.	In the unlikely event of an SFP accident, the iodine isotopes, which contribute to an offsite dose from an operating reactor accident, are not present, so potassium iodide distribution would no longer serve as an effective or necessary supplemental protective action. The CR-3 SFP is considered an ISFSI and is licensed under 10 CFR part 72, subpart K, "General License for Storage of Spent Fuel at Power Reactor Sites." The Commission responded to comments in its SOC for the final rule for EP requirements for ISFSIs and MRS facilities (60 FR 32435), and concluded that, "the offsite consequences of potential accidents at an ISFSI or an MRS would not warrant establishing EPZs." Additionally, in the SOC for the final rule for EP requirements for ISFSIs and for MRS

	facilities (60 FR 32430), the Commission responded to comments concerning site-specific EP 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 refer to basis for 10 CFR 50.47(b) and 10 CFR 50.47(b)(2).
10 CFR 50.47(c)(2) The NRC is granting exemptions from portions of the rule language that would otherwise require the establishment of a 10 mile radius plume exposure pathway EPZ and a 50 mile radius ingestion pathway EPZ.	Refer to basis for 10 CFR 50.47(b)(10).

10 CFR part 50, appendix E, section IV	**NRC Staff Basis for Exemption**
10 CFR part 50, appendix E, section IV.1. The NRC is granting exemptions from portions of the rule language that would otherwise require onsite protective actions during hostile action.	The EP Rule published in the *Federal Register* (76 FR 72560; November 23, 2011), amended certain requirements in 10 CFR part 50. Among the changes, the definition of "hostile action" was added as an act directed toward a nuclear power plant or its personnel. This definition is based on the definition of "hostile action" provided in NRC Bulletin 2005-02, "Emergency Preparedness and Response Actions for Security-Based Events." NRC Bulletin 2005-02 was not applicable to nuclear power reactors that have permanently ceased operations and have certified that fuel has been removed from the reactor vessel. The NRC excluded non-power reactors from the scope of "hostile action" at the time of the rulemaking because, as defined in 10 CFR 50.2, a non-power reactor is not considered a nuclear power reactor and a regulatory basis had not been developed to support the inclusion of non-power reactors within the scope of "hostile action." Similarly, a decommissioning power reactor or an ISFSI is not a "nuclear reactor" as defined in 10 CFR part 50. A decommissioning power reactor also has a low likelihood of a credible accident resulting in radiological releases requiring offsite protective measures. For all of these reasons, the NRC staff concludes

	that a decommissioning power reactor is not a facility that falls within the scope of "hostile action." Similarly, for security, risk insights can be used to determine which targets are important to protect against sabotage. A level of security commensurate with the consequences of a sabotage event is required and is evaluated on a site-specific basis. The severity of the consequences declines as fuel ages and, thereby, removes over time the underlying concern that a sabotage attack could cause offsite radiological consequences. Although, this analysis provides a justification for exempting CR-3 from "hostile action" related requirements, some EP requirements for security-based events are maintained. The classification of security-based events, notification of offsite authorities and coordination with offsite agencies under a CEMP concept are still required.
10 CFR part 50, appendix E, section IV.2. The NRC is granting exemptions from portions of the rule language concerning the evacuation time analyses within the plume exposure pathway EPZ for the licensee's initial application.	Refer to basis for 10 CFR 50.47(b)(10).
10 CFR part 50, appendix E, section IV.3. The NRC is granting exemptions from portions of the rule language that would otherwise require use of NRC-approved ETEs and updates to State and local governments when developing protective action strategies.	Refer to basis for 10 CFR part 50, appendix E, section IV.2 and 10 CFR 50.47(b).
10 CFR part 50, appendix E, section IV.4. The NRC is granting exemptions from portions of the rule language that would otherwise require licensees to update evacuation time estimates based on the most recent census data and submit the ETE analysis to the NRC prior to providing it to State and local government for developing protective action	Refer to basis for 10 CFR part 50, appendix E, section IV.2 and 10 CFR 50.47(b).

strategies.	
10 CFR part 50, appendix E, section IV.5. The NRC is granting an exemption from portions of the rule language that would otherwise require licensees to estimate the EPZ permanent resident population changes once a year between decennial censuses.	Refer to basis for 10 CFR part 50, appendix E, section IV.2 and 10 CFR 50.47(b).
10 CFR part 50, appendix E, section IV.6. The NRC is granting an exemption from portions of the rule language that would otherwise require the licensee to submit an updated ETE analysis to the NRC based on changes in the resident population that result in exceeding specific evacuation time increase criteria.	Refer to basis for 10 CFR part 50, appendix E, section IV.2 and 10 CFR 50.47(b).
10 CFR part 50, appendix E, section IV.A.1. The NRC is granting an exemption from the word "operating" in the requirement to describe the normal plant organization.	Based on the permanently shutdown and defueled status of the reactor, a decommissioning reactor is not authorized to operate under 10 CFR 50.82(a). Because the licensee cannot operate the reactors, the licensee does not have a "plant operating organization."
10 CFR part 50, appendix E, section IV.A.3. The NRC is granting an exemption from the requirement to describe the licensee's headquarters personnel sent to the site to augment the onsite emergency response organization.	The number of staff at decommissioning sites is generally small but is 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 the licensee's headquarters personnel.
10 CFR part 50, appendix E, section IV.A.4. The NRC is granting exemptions from portions of the rule language that would otherwise require the licensee to identify a position and function within its organization, which will carry the responsibility for making offsite dose projections.	Although, the likelihood of events that would result in doses in excess of the EPA PAGs to the public beyond the exclusion area boundary based on the permanently shutdown and defueled status of the reactor is extremely low, the licensee still must be able to determine if a radiological release is occurring. If a release is occurring, then the licensee staff should 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 based

	on comprehensive EP. Also refer to basis for 10 CFR 50.57(b)
10 CFR part 50, appendix E, section IV.A.5. The NRC is granting an exemption from the requirement for the licensee to identify individuals with special qualifications, both licensee employees and non-employees, for coping with emergencies.	The minimal systems and equipment needed to maintain the spent nuclear fuel in the SFP in a safe condition requires minimal personnel and is governed by the technical specifications. As such, additional employees or other persons with special qualifications are not anticipated. Refer to basis for 10 CFR part 50, appendix E, section IV.A.3
10 CFR part 50, appendix E, section IV.A.7. The NRC is granting exemptions from portions of the rule language that would otherwise require a description of the assistance expected from State, local, and Federal agencies for coping with a hostile action.	Offsite emergency measures are limited to support provided by local police, fire departments, and ambulance and hospital services, as appropriate. Due to the low probability of design-basis accidents or other credible events to exceed the EPA PAGs, protective actions such as evacuation should not be required, but could be implemented at the discretion of offsite authorities using a CEMP. Refer to basis for 10 CFR part 50, appendix E, section IV.1 and 10 CFR 50.47(b).
10 CFR part 50, appendix E, section IV.A.8. The NRC is granting an exemption from the requirement to identify the State and local officials for ordering protective actions and evacuations.	Offsite emergency measures are limited to support provided by local police, fire departments, and ambulance and hospital services, as appropriate. Due to the low probability of design-basis accidents or other credible events to exceed the EPA PAGs, protective actions such as evacuation should not be required, but could be implemented at the discretion of offsite authorities using a CEMP. Also refer to basis for 10 CFR 50.47(b).
10 CFR part 50, appendix E, section IV.A.9. The NRC is granting an exemption from the requirement for the licensee to provide an analysis demonstrating that on-shift personnel are not assigned responsibilities that would prevent performance of their assigned emergency plan functions.	Responsibilities should be well defined in the emergency plan and procedures, regularly tested through drills and exercises audited and inspected by the licensee and the NRC. The duties of the on-shift personnel at a decommissioning reactor facility are not as complicated and diverse as those for an operating power reactor. The NRC staff considered the similarity between the staffing levels at a permanently shutdown

	and defueled reactor and staffing levels at an operating power reactor site. The minimal systems and equipment needed to maintain the spent nuclear fuel in the SFP or in an ISFSI in a safe condition requires minimal personnel and is governed by Technical Specifications. In the EP final rule published in the *Federal Register* (76 FR 72560; November 23, 2011), the NRC concluded that the staffing analysis requirement was not necessary for non-power reactor licensees due to the small staffing levels required to operate the facility. The NRC staff also examined the actions required to mitigate the very low probability design-basis events for the SFP. Several sources of makeup to the pools are available, such as the fire service system, using the diesel-driven fire service pump for loss of electrical power. If available fresh water sources are depleted, salt water sources with inexhaustible inventory from the CR-3 intake and discharge canal, using portable diesel powered pumps are available. Pool inventory addition can be implemented without accessing the elevation of the pool deck. DEF believes these diverse strategies provide defense-in-depth and ample time to provide makeup or spray to the SFP prior to the onset of zirconium cladding ignition when considering very low probability beyond-design-basis events affecting the SFP. In a letter dated May 7, 2014, DEF withdrew its request to remove License Condition 2.C.(14), "Mitigation Strategy License Condition," from its Facility Operating License. This license condition requires CR-3 to maintain its SFP inventory makeup strategies as discussed above.
10 CFR part 50, appendix E, section IV.B.1. The NRC is granting exemptions from portions of the rule language that would otherwise require offsite emergency actions levels and offsite protective measures and associate offsite monitoring for the emergency conditions. In addition, the NRC is granting exemption from portions of the rule language that would otherwise require	NEI 99-01, Revision 6, was found to be an acceptable method for development of EALs. No offsite protective actions are anticipated to be necessary, so classification above the alert level is no longer required, which is consistent with ISFSI facilities. Also refer to basis for 10 CFR part 50, appendix E, section IV.1 and 10 CFR 50.47(b).

emergency action levels based on hostile action.	
10 CFR part 50, appendix E, section IV.C.1. The NRC is granting exemptions from portions of the rule language that would otherwise require emergency actions levels based on operating reactor concerns, such as offsite radiation monitoring, pressure in containment, and the response of the emergency core cooling system. In addition, the NRC is striking language that would otherwise require offsite emergency action levels of a site area emergency and a general emergency.	Containment parameters do not provide an indication of the conditions at a defueled facility and emergency core cooling systems are no longer required. SFP level, SFP temperature, and area radiation monitors indicate the conditions at CR-3. In the SOC for the final rule for EP requirements for ISFSIs and MRS facilities (60 FR 32430), the Commission responded to comments concerning a general emergency at an ISFSI and a MRS, and concluded that, "...an essential element of a General Emergency is that a release can be reasonably expected to exceed EPA PAGs 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 available time for event mitigation and, if necessary, implementation of offsite protective actions using a CEMP. NEI 99-01, Revision 6, was found to be an acceptable method for development of EALs. No offsite protective actions are anticipated to be necessary, so classification above the alert level is no longer required. Also, refer to the basis for 10 CFR 50.47(b).

10 CFR part 50, appendix E, section IV.C.2. The NRC is granting exemptions from portions of the rule language that would otherwise require the licensee to assess, classify, and declare an emergency condition within 15 minutes.	In the EP rule published in the *Federal Register* (76 FR 72560), non-power reactor licensees were not required to assess, classify and declare an emergency condition within 15 minutes. An SFP and an ISFSI are also not nuclear power reactors as defined in the NRC's regulations. A decommissioning power reactor has a low likelihood of a credible accident resulting in radiological releases requiring offsite protective measures. For these reasons, the NRC staff concludes that a decommissioning power reactor should not be required to assess, classify and declare an emergency condition within 15 minutes.
10 CFR part 50, appendix E, section IV.D.1. The NRC is granting exemptions from portions of the rule language that would otherwise require the licensee to reach agreement with local, State, and Federal officials and agencies for prompt notification of protective measures or evacuations. In addition, the NRC is granting an exemption from identifying the associated titles of officials to be notified for each agency within the EPZs.	Refer to basis for 10 CFR 50.47(b), 10 CFR 50.47(b)(2) and 10 CFR 50.47(b)(6).
10 CFR part 50, appendix E, section IV.D.2. The NRC is granting an exemption from the requirement for the licensee to annually disseminate general information on emergency planning and evacuations within the plume exposure pathway EPZ. In addition, the NRC is granting an exemption for the need for signage or other measures to address transient populations in the event of an accident.	Refer to basis for 10 CFR 50.47(b) ,10 CFR 50.47(b)(2) and 10 CFR 50.47(b)(5).
10 CFR part 50, appendix E, section IV.D.3. The NRC is granting exemptions from portions of the rule language that would otherwise require the licensee to have the	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

capability to make notifications to State and local government agencies within 15 minutes of declaring an emergency.	provided. DEF's exemption request provides that CR-3 will make notifications to the State of Florida and the NRC within 60 minutes of declaration of an event. The State Watch Office will perform the notification to the County (Citrus), as well as the Florida Department of Emergency Management. In the permanently defueled condition of the reactor, the rapidly developing scenarios associated with events initiated during reactor power operation are no longer credible. Also refer to basis for 10 CFR 50.47(b) and 10 CFR 50.47(b)(2).
10 CFR part 50, appendix E, section IV.D.4. The NRC is granting an exemption from the requirement for the licensee to obtain FEMA approval of its backup alert and notification capability.	Refer to basis for 10 CFR part 50, appendix E, section IV.D.3 regarding the alert and notification system requirements.
10 CFR part 50, appendix E, section IV.E.8.a.(i) The NRC is granting exemptions from portions of the rule language that would otherwise require the licensee to have an onsite technical support center and emergency operations facility.	Due to the low probability of design-basis accidents or other credible events to exceed the EPA PAGs at the exclusion area boundary, the available time for event mitigation at a decommissioning reactor and, if needed, to implement offsite protective actions using a CEMP, an EOF and a technical support center (TSC) would not be required to support offsite agency response. Onsite actions may be directed from the control room or other location, without the requirements imposed on a TSC.

10 CFR part 50, appendix E, section IV.E.8.a.(ii) The NRC is granting exemptions from portions of the rule language that would otherwise require the licensee to have an onsite operational support center.	NUREG-0696, "Functional Criteria for Emergency Response Facilities" (ADAMS Accession No. ML051390358) 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 decommissioning power reactor, an OSC is no longer required to meet its original purpose of an assembly area for plant logistical support during an emergency. The OSC function can be incorporated into another facility. Also refer to the basis for 10 CFR part 50, appendix E, section IV.E.8.a.(i).
10 CFR part 50, appendix E, section IV.E.8.b. and subpart sections IV.E.8.b.(1) - E.8.b.(5) The NRC is granting exemptions from the requirements related to an offsite emergency operations facility's location, space and size, communications capability, access to plant data and radiological information, and access to copying and office supplies.	Refer to basis for 10 CFR 50.47(b)(3) and 10 CFR part 50, appendix E, section IV.E 8.a.(i) .
10 CFR part 50, App. E, section IV E.8.c. and sections IV E.8.c.(1) - E.8.c.(3) The NRC is granting exemptions from the requirements to have an emergency operations facility with the capabilities to obtain and display plant data and radiological information; the capability to analyze technical information and provide briefings; and the capability to support events occurring at more than one site (if the emergency operations center supports more than one site).	Refer to basis for 10 CFR 50.47(b)(3) and 10 CFR part 50, appendix E, section IV.E 8.a.(i).
10 CFR part 50, App. E, section IV E.8.d The NRC is granting exemptions from the requirements to have an alternate facility 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.	Refer to basis for 10 CFR part 50, appendix E, section IV.1; 10 CFR part 50, appendix E, section IV.E 8.a.(i); and 10 CFR 50, appendix E, section IV.E.8.a.(ii).

10 CFR part 50, appendix E, section IV.E.8.e. The NRC is granting an exemption from the need for the licensee to comply with paragraph 8.b of this section that details EOFs requirements.	Because of the low probability of design-basis accidents or other credible events that would be expected to exceed the EPA PAGs and the available time for event mitigation and, if needed, implementation of offsite protective actions using a CEMP, there is no need for the EOF. Refer to basis for 10 CFR 50.47(b)(3) and 10 CFR part 50, appendix E, section IV.E 8.a.(i).
10 CFR part 50, appendix E, section IV.E.9.a. The NRC is granting exemptions from portions of the rule language that would otherwise require the licensee to have communications with contiguous State and local governments that are within the plume exposure pathway EPZ.	The Plume exposure pathway EPZ is no longer required by the exemption granted to 10 CFR 50.47(b)(10). The State and the local governments in which the nuclear facility is located will still need to be informed of events and emergencies, so lines of communication must be maintained. Refer to basis for 10 CFR 50.47(b)(2) and 10 CFR 50.47(b)(10).
10 CFR part 50, appendix E, section IV.E.9.c. The NRC is granting exemption from the requirements for communication and testing provisions between the control room, the onsite TSC, State/local emergency operations centers, and field assessment teams.	Because of the low probability of design-basis accidents or other credible events that would be expected to exceed the EPA PAGs and the available time for event mitigation and, if needed, implementation of offsite protective actions using a CEMP, there is no need for the TSC, EOF, offsite field assessment teams, and the communication and testing provisions that refer to them. Refer to justification for 10 CFR 50.47(b)(3) and 10 CFR part 50, appendix E, section IV.E 8.a.(i). Communication with State and local emergency operation centers is maintained to coordinate assistance on site if required.
10 CFR part 50, appendix E, section IV.E.9.d. The NRC is granting exemptions from portions of the rule language that would otherwise require provisions for communications from the control room, onsite TSC, and EOF with NRC Headquarters and the appropriate Regional Operations Center.	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. The licensee is still required to maintain monthly communication tests with NRC Headquarters and the appropriate Regional Operations Center. Also refer to basis for 10 CFR 50.47(b); 10 CFR 50, appendix E, section IV.E.8.a.(i); and 10 CFR 50, appendix E, section IV.E.8.a.(ii).

10 CFR part 50, appendix E, section IV.F.1. and section IV F.1.viii. The NRC is granting exemptions from portions of the rule language that would otherwise require the licensee to provide training and drills for the licensee's headquarters personnel, Civil Defense personnel, or local news media.	Decommissioning power reactor sites typically have a level of emergency response that does not require additional response by the licensee's headquarters personnel. Therefore, the NRC staff considers exempting licensee's headquarters personnel from training requirements to be reasonable. Due to the low probability of design-basis accidents or other credible events to exceed the EPA PAGs, offsite emergency measures are limited to support provided by local police, fire departments, and ambulance and hospital services, as appropriate. Local news media personnel no longer need radiological orientation training since they will not be called upon to support the formal Joint Information Center. The term "Civil Defense" is no longer commonly used; references to this term in the examples provided in the regulation are, therefore, not needed. Also refer to basis for 10 CFR 50.47(b).
10 CFR part 50, appendix E, section IV.F.2. The NRC is granting exemptions from portions of the rule language that would otherwise require testing of a public alert and notification system.	Because of the low probability of design-basis accidents or other credible events that would be expected to exceed the limits of EPA PAGs and the available time for event mitigation and offsite protective actions from a CEMP, the public alert and notification system are not needed and, therefore, require no testing. Also refer to basis for 10 CFR 50.47(b).
10 CFR part 50, appendix E, section IV.F.2.a. and sections IV.F.2.a.(i) through IV.F.2.a.(iii) The NRC is granting exemptions from the requirements for full participation exercises and the submittal of the associated exercise scenarios to the NRC.	Due to the low probability of design-basis accidents or other credible events that would be expected to exceed the limits of EPA PAGs, the available time for event mitigation and, if necessary, implementation of offsite protective actions using a CEMP, no formal offsite radiological emergency plans are required and full participation emergency plan exercises that test the State and local emergency plans are not necessary. The intent of submitting exercise scenarios at an operating power reactor site is to ensure that licensees utilize different scenarios in order to prevent the preconditioning of responders at power reactors. For decommissioning power reactor sites, there are limited events that could occur, and as such, the submittal of exercise

	scenarios is not necessary. The licensee would be exempt from 10 CFR part 50, appendix E, section IV.F.2.a.(i)-(iii) because the licensee would be exempt from the umbrella provision of 10 CFR part 50, appendix E, section IV.F.2.a. Also, refer to the basis for 10 CFR 50.47(b) and 10 CFR part 50, appendix E, section IV.C.1.
10 CFR part 50, appendix E, section IV.F.2.b. The NRC is granting exemptions from portions of the rule language that would otherwise require the licensee to submit scenarios for its biennial exercises of its onsite emergency plan. In addition, the NRC is granting exemption from portions of the rule language that requires assessment of offsite releases, protective action decision making, and references to the TSC, OSC, and EOF.	The intent of submitting onsite exercise scenarios at an operating power reactor site is to ensure that licensees utilize different scenarios in order to prevent the preconditioning of responders at power reactors. For decommissioning power reactor sites, there are limited events that could occur, and as such, the submittal of exercise scenarios is not necessary. Biennial exercises are not required per the exemption from 10 CFR part 50, appendix E, section IV.F.2.c. The low probability of design basis accidents or other credible events that would exceed the EPA PAGs, the available time for event mitigation and, if necessary, implementation of offsite protective actions using a CEMP, render a TSC, OSC and EOF unnecessary. The principal functions required by regulation can be performed at an onsite location that does not meet the requirements of the TSC, OSC, or EOF. Refer to basis for 10 CFR part 50, appendix E, section IV.F.2.a; 10 CFR part 50, appendix E, section IV.E 8.a.(i); 10 CFR part 50, appendix E, section IV.E 8.a.(ii); and 10 CFR 50.47(b).
10 CFR part 50, appendix E, section IV.F.2.c. and sections IV F.2.c.(1) through F.2.c.(5) The NRC is granting exemptions from the requirements regarding the need for the licensee to exercise offsite plans biennially with full participation by each offsite authority having a role under the radiological response plan. The NRC is also granting exemptions from the conditions for conducting these exercises (including hostile action exercises) if two different licensees have facilities on the	Refer to basis for 10 CFR part 50, appendix E, section IV.F.2.a and 10 CFR 50.47(b).

same site or on adjacent, contiguous sites, or share most of the elements defining co-located licensees.	
10 CFR part 50, appendix E, section IV.F.2.d. The NRC is granting exemptions from the requirements to obtain State participation in an ingestion pathway exercise and a hostile action exercise, with each State that has responsibilities, at least once per exercise cycle.	Refer to basis for 10 CFR 50, appendix E, section IV.F.2.a.
10 CFR part 50, appendix E, section IV.F.2.e. The NRC is granting exemptions from portions of the rule language that would otherwise require the licensee to allow participation exercise in licensee drills by any State and local Government in the plume exposure pathway EPZ when requested.	Refer to basis for 10 CFR 50.47(b)(2) and 10 CFR 50.47(b)(10).
10 CFR part 50, appendix E, section IV.F.2.f. The NRC is granting exemptions from portions of the rule language that would otherwise require FEMA to consult with the NRC on remedial exercises. The NRC is granting exemption from portions of the rule language that discuss the extent of State and local participation in remedial exercises.	FEMA is responsible for evaluating the adequacy of offsite response during an exercise. No action is expected from State or local government organizations in response to an event at a decommissioning power reactor site other than onsite firefighting, law enforcement and ambulance/medical services support. A memorandum of understanding is in place for those services. Offsite response organizations will continue to take actions on a comprehensive emergency planning basis to protect the health and safety of the public as they would at any other industrial site. Also, refer to the basis for 10 CFR 50, appendix E, section IV.F.2.a.

10 CFR part 50, appendix E, section IV.F.2.i. The NRC is granting exemptions from portions of the rule language that would otherwise require the licensee to engage in drills and exercises for scenarios that include a wide spectrum of radiological release events and hostile action.	Due to the low probability of design-basis accidents or other credible events to exceed the EPA PAGs, the available time for event mitigation and, if needed, implementation of offsite protective actions using a CEMP, the previously routine progression to general emergency in power reactor site scenarios is not applicable to a decommissioning site. Therefore, the licensee is not expected to demonstrate response to a wide spectrum of events. Also refer to basis for 10 CFR part 50, appendix E, section IV.1 regarding hostile action.
10 CFR part 50, appendix E, section IV.F.2.j. The NRC is granting exemptions from the requirements regarding the need for the licensee's emergency response organization to demonstrate proficiency in key skills in the principal functional areas of emergency response. In addition, the NRC is granting an exemption during an eight calendar year exercise cycle, from demonstrating proficiency in the key skills necessary to respond to such scenarios as hostile actions, unplanned minimal radiological release, § 50.54(hh)(2) implementation strategies, and scenarios involving rapid escalation to a site area emergency or general emergency.	With the permanently shutdown defueled and conditions of the site, where only the SFP and its related support systems, structures, and components remain, there are no other facilities in which emergency response organization personnel could demonstrate proficiency. Also refer to basis for 10 CFR part 50, appendix E, section IV.F.2.i.
10 CFR part 50, appendix E, section IV.I The NRC is granting exemptions from the requirements regarding the need for the licensee to develop a range of protective action for onsite personnel during hostile actions.	Refer to basis for 10 CFR part 50, appendix E, section IV.1.

ENCLOSURE 2

SAFETY EVALUATION RELATED TO

DUKE ENERGY FLORIDA, INC.

CRYSTAL RIVER UNIT 3 NUCLEAR GENERATING PLANT

REQUEST FOR EXEMPTIONS FROM PORTIONS OF

10 CFR 50.47 AND 10 CFR PART 50, APPENDIX E

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO REQUEST FOR EXEMPTIONS FROM PORTIONS OF

10 CFR 50.47 AND 10 CFR PART 50 APPENDIX E

DUKE ENERGY FLORIDA, INC., ET AL.

CRYSTAL RIVER UNIT 3 NUCLEAR GENERATING PLANT

DOCKET NO. 50-302

1.0 INTRODUCTION

The Crystal River Unit 3 Nuclear Generating Plant (CR-3) is a decommissioning power reactor located in Red Level, Florida, in Citrus County, about 5 miles south of Levy County. The site is 7.5 miles northwest of Crystal River, Florida, and 90 miles north of St. Petersburg, Florida. CR-3 is situated on the Gulf of Mexico, within the Crystal River Energy Complex. Duke Energy Florida, Inc. (DEF, the licensee), is the holder of the CR-3 Facility Operating License No. DPR-72, issued pursuant to the Atomic Energy Act of 1954, as amended, and Part 50, "Domestic Licensing of Production and Utilization Facilities," of Title 10 of the *Code of Federal Regulations* (10 CFR).

CR-3 has been shut down since September 26, 2009, and the final removal of fuel from the reactor vessel was completed on May 28, 2011. By letter dated February 20, 2013 (Reference 1), the licensee submitted a certification to the U.S. Nuclear Regulatory Commission (NRC) of permanent cessation of power operations and the removal of fuel from the reactor vessel, pursuant to 10 CFR 50.82(a)(1)(i) and 10 CFR 50.82(a)(1)(ii). Upon docketing of the certification, the 10 CFR Part 50 license for CR-3 no longer authorizes operation of the reactor or emplacement or retention of fuel into the reactor vessel, as specified in 10 CFR 50.82(a)(2). CR-3 is authorized to possess and store irradiated (i.e., spent) nuclear fuel. Spent fuel is currently stored onsite in a spent fuel pool (SFP).

By letter dated September 26, 2013 (Reference 2), the licensee requested exemptions from specific emergency planning (EP) standards of 10 CFR 50.47, "Emergency planning," and specific requirements of Appendix E to 10 CFR Part 50, "Emergency Planning and Preparedness for Production and Utilization Facilities," for CR-3. By letters dated March 28, May 7, May 23, and August 28, 2014 (References 3, 4, 5, and 6, respectively), the licensee supplemented the original September 26, 2013, request. In the March 28, 2014, supplement (Reference 3), the licensee responded to a request for additional information (RAI) from the NRC staff regarding the requested exemptions. In the May 7, 2014, supplement (Reference 4), the licensee reaffirmed the continuation of SFP inventory makeup strategies for mitigating the

loss-of-water inventory. In the May 23, 2014, supplement (Reference 5), the licensee responded to an RAI regarding the EP amendment request. In the August 28, 2014, supplement (Reference 6), the licensee amended its request to align with the exemptions approved in staff requirements memorandum (SRM) to SECY-14-0066 (Reference 7). The NRC staff found the application complete and the licensee's associated technical justification provided a basis for the Commission's consideration of the requested exemption.

In accordance with 10 CFR 50.12, the licensee stated that this exemption request and its impact on the corresponding emergency plan: (1) is authorized by law; (2) will not present an undue risk to the public health and safety; and (3) is consistent with the common defense and security.

1.1 Discussion

The regulations that require each nuclear power reactor licensee to establish and maintain emergency plans and preparedness are set forth in 10 CFR 50.47 and Appendix E to 10 CFR Part 50. The regulations include standards for both onsite and offsite radiological emergency plans. However, when compared to an operating nuclear power plant, the regulations do not take into account the reduced risk of an offsite radiological release at a permanently shutdown and defueled reactor.

With the termination of reactor operations at CR-3 and the permanent removal of the fuel from the reactor core, most of the accident scenarios postulated for operating reactors are no longer possible. The irradiated fuel is currently stored in the SFP and CR-3 plans to complete the move of all irradiated fuel to an onsite dry cask independent spent fuel storage installation (ISFSI) by August 2019, according to the CR-3 Post-Shutdown Decommissioning Activities Report (Reference 8). The irradiated fuel will remain onsite until it can be moved offsite for long-term storage or disposal. The CR-3 reactor, reactor coolant system (RCS), and supporting systems are no longer in operation and have no function related to the storage of the irradiated fuel. Therefore, postulated accidents involving failure or malfunction of the reactor, RCS, and the systems supporting reactor operation are no longer applicable.

During reactor decommissioning, the principal public safety concerns involve the perceived radiological risks associated with the storage of spent fuel onsite. For a period of time after fuel has been irradiated in a power reactor and is being stored in an SFP, a highly unlikely accident scenario has been postulated where a loss-of-water inventory from the SFP could result in a significant heat-up of the spent fuel, culminating in substantial zirconium cladding oxidation and fuel damage, also known as a zirconium fire.

In August 1997, the NRC published NUREG/CR-6451, "A Safety and Regulatory Assessment of Generic BWR [boiling water reactor] and PWR [pressurized water reactor] Permanently Shutdown Nuclear Power Plants" (Reference 9), which provided recommendations on operationally-based regulations that could be partially or totally removed for decommissioning power reactor licensees without impacting public health and safety. It recommended that licensees apply for exemptions from certain EP requirements after the spent fuel is no longer susceptible to substantial zirconium oxidation and the fuel cladding remains intact when the SFP is drained.

In the late 1990s, the NRC staff developed a thermal-hydraulic criterion for determining when reductions in EP requirements at decommissioning power reactors could be permitted. The criterion was used on a case-by-case basis to grant exemptions from certain EP requirements. The underlying technical basis was a demonstration that: (1) the radiological consequences of applicable design-basis accidents (DBAs) would not exceed radiological release limits at the site exclusion area boundary (EAB); and (2) for a highly unlikely beyond design-basis accident[1] (beyond-DBA) where the SFP is drained and no cooling (air or water) of the fuel is taking place, the spent fuel stored in the SFP would not reach the zirconium ignition temperature in fewer than 10 hours starting from the time at which the accident was initiated. The NRC staff concluded that if 10 hours were available to initiate mitigation actions, or if needed, offsite protective actions using a comprehensive emergency management plan[2] (CEMP) approach, then formal offsite radiological emergency plans would not be necessary for permanently defueled power reactor licensees.

The analysis and 10-hour criterion for mitigating the potential consequences of beyond-DBAs at a SFP does not credit the natural air cooling and water cooling in the SFP after the event, as a modeling simplification. It assumes that the fuel immediately begins to heat up without removing any of its energy (often referred to as an adiabatic heatup). These assumptions include the simplified treatment of the thermal-hydraulic response and the use of often bounding configurations that do not allow for thermal radiation between high powered bundles and low power bundles and from the spent fuel assemblies to the SFP wall liner. In a more realistic calculation, as provided in the recent NUREG-2161 "Consequence Study of a Beyond-Design-Basis Earthquake Affecting the Spent Fuel Pool for a U.S. Mark I Boiling-Water Reactor," dated September 2014 (Reference 10), thermal radiation heat transfer (in addition to air cooling) can play a significant role. For example, it could take more than 10 hours for the fuel to heat up to the zirconium cladding ignition temperature (900 degrees Celsius (C)) after only one month of being moved from the reactor to the SFP, for the reference plant, if the assemblies most recently removed from the reactor are distributed among older, cooler, fuel assemblies. It should be noted that this assessment applies to BWR fuel only. Due to the much higher mass and slightly higher burnup of a typical PWR assembly, the time to reach an air-coolable configuration is significantly longer for PWR fuel using similar analytical methods and assumptions.

The 10-hour time frame is not intended to be the time in which it would take to repair all key safety systems or to repair a large SFP breach. Rather considering the very low probability of beyond-DBAs affecting the SFP, in the NRC staff's judgment, 10 hours provides a reasonable time period to implement pre-planned mitigation measures to provide makeup or spray to the

---

[1] Beyond Design-Basis Accidents - This term refers to accident sequences that are possible but unlikely and are considered beyond the scope of design-basis accidents that a nuclear facility must be designed and built to withstand.

[2] A comprehensive emergency management plan in this context, also referred to as an emergency operations plan (EOP), is addressed in the Federal Emergency Management Agency's (FEMA) Comprehensive Preparedness Guide (CPG) 101, "Developing and Maintaining Emergency Operations Plans". CPG 101 is the foundation for State, territorial, Tribal, and local emergency planning in the United States. It promotes a common understanding of the fundamentals of risk-informed planning and decision making and helps planners at all levels of government in their efforts to develop and maintain viable, all-hazards, all-threats emergency plans. An EOP is flexible enough for use in all emergencies. It describes how people and property will be protected; details who is responsible for carrying out specific actions; identifies the personnel, equipment, facilities, supplies and other resources available; and outlines how all actions will be coordinated. A comprehensive emergency management plan is often referred to as a synonym for "all hazards planning."

SFP before the onset of zirconium cladding ignition and, if necessary, for offsite authorities to implement protective actions using a CEMP (all-hazards) approach.

In February 2001, the NRC prepared NUREG-1738, "Technical Study of Spent Fuel Pool Accident Risk at Decommissioning Nuclear Power Plants" (Reference 11), to provide a technical basis for potential rulemaking, including EP, for nuclear power reactors that are permanently shut down and defueled. Although the rulemaking was later deferred in light of higher priority work after the terrorist attacks of September 11, 2001, NUREG-1738 provides insights that the NRC staff continues to find helpful for the evaluation of exemption requests regarding EP requirements.

NUREG-1738 identified a zirconium fire resulting from a substantial loss-of-water inventory from the SFP as the only postulated scenario at a decommissioning plant that could result in a significant offsite radiological release. The scenarios that lead to this condition have very low probabilities of occurrence and are considered beyond-DBAs, but the consequences of such accidents could lead to an offsite radiological dose in excess of the U.S. Environmental Protection Agency's (EPA's) Protective Action Guidelines (PAGs) (Reference 12) at the EAB. However, the risk associated with zirconium cladding fire events decreases as the spent fuel ages, decay time increases, decay heat decreases, and short-lived radionuclides decay away. After a certain amount of time, the overall risk of a zirconium cladding fire becomes extremely low due to two factors: (1) the amount of time available for preventative and mitigating actions; and (2) the increased likelihood that the fuel is air coolable. The NRC staff also notes that the results of research conducted for NUREG-1738 and NUREG-2161 suggest that, while other radiological consequences can be extensive, a postulated accident scenario leading to a SFP zirconium fire, where the fuel has significant decay time, will have little potential to cause offsite early fatalities regardless of the type of offsite EP response.

Although the risk of sabotage is not considered in any standard reactor risk analyses, the NRC staff cannot rule out radiological sabotage (which is not quantifiable) as an insignificant risk contributor relative to other zirconium cladding fire initiators. Therefore, permanently shutdown and defueled reactors must continue to provide a high assurance of adequate protection from the design-basis threat of radiological sabotage under the plant's Physical Security Plan. Physical security for special nuclear material at fixed sites, including decommissioning power reactors, is required by 10 CFR Part 73, "Physical Protection of Plants and Materials." Decommissioning power reactor licensees are required by 10 CFR 73.55(f) to develop target sets for use in the development and implementation of security strategies that protect against spent fuel sabotage. When compared to operating power reactors, the number of target sets at a decommissioning reactor is significantly reduced. Implementation of the protective strategy at a decommissioning reactor takes into account this reduction in target sets.

In Enclosure 6 of the September 26, 2013, letter (Reference 2), the licensee provided a permanently defueled accident analysis, which included: (1) a fuel handling accident; (2) a radioactive waste handling accident; (3) a loss of SFP normal cooling (boil off); (4) a loss of SFP inventory with air cooling; (5) loss of SFP inventory with an adiabatic heatup of the hottest fuel assembly; and (6) a loss of SFP inventory radiation dose. The loss of SFP inventory events are considered beyond-DBAs. The analyses demonstrate that in all cases, with the exception of (5) adiabatic heatup of the hottest fuel assembly, radiation exposure levels at the site's EAB would be less than the EPA PAGs.

The loss of SFP normal cooling analysis includes a description of two locations where the SFP inventory can be replenished on an extended loss-of-cooling without additional radiation exposure to plant personnel. These locations provide connection points where the fire service system can be used with a diesel-driven fire pump. In the event that fresh water supplies are exhausted, the intake and discharge canals can provide water using portable diesel-driven pumps. The licensee has committed to maintain the availability of trained personnel, procedures and the equipment to perform these makeup strategies. In Enclosure 1 of the letter dated May 23, 2014 (Reference 5), the licensee stated that timed validation studies, based on in-field performance, demonstrated that the required personnel can retrieve the portable pump, setup suction and discharge hoses, and begin adding inventory to the SFP in less than 2 hours.

The analysis for the event in which the SFP is drained and air cooling is possible demonstrated that, as of September 26, 2013, fuel cladding would not reach the lowest temperature at which incipient failure may occur (565 degrees C). In the case of an adiabatic heatup, at least 19.7 hours would be available, from the time the fuel is uncovered and starts to heat up until the hottest fuel assembly reaches 900 degrees C, to take mitigative actions consistent with plant conditions, and if necessary, for offsite authorities to implement appropriate protective measures using a CEMP (all-hazards) approach. The loss of SFP inventory radiation dose analysis indicated that dose rates in the control room would be approximately 7.7E-06 roentgen equivalent man (rem)/hour, and less than 1.9E-07 rem/hour at the EAB. This assessment demonstrated that the dose rate at the EAB would be sufficiently low to allow ample time to implement offsite actions without planning to prevent exceeding the EPA PAGs.

The licensee provided evaluations indicating the potential of a significant loss-of-coolant inventory event was very small because:

- makeup water could be easily added over the extended period necessary for a loss of normal heat removal to cause a loss-of-coolant inventory;
- the CR-3 site is in a seismically inactive zone and the SFP is a seismic class I structure; and
- the cask handling system at CR-3 will be designed and licensed as single-failure-proof prior to moving a spent fuel cask (see Section 3.2.1 IDC No. 1 below).

To ensure adequate protection of the public health and safety, the licensee will maintain a defense-in-depth philosophy at CR-3 that applies successive compensatory measures to prevent accidents or mitigate damage if a malfunction, accident, or naturally caused event occurs at the facility. As provided in the supplemental letter dated May 7, 2014 (Reference 4), the licensee committed to maintaining its makeup strategies in the event of a loss-of-coolant inventory in the SFP as part of their license condition to develop and maintain mitigation strategies for addressing large fires and explosions, as required under 10 CFR 50.54(hh)(2). The multiple strategies for providing makeup to the SFP include: using existing plant systems for inventory makeup, supplying water through hoses to connections to the existing SFP piping using the diesel-driven fire service pump, and using a diesel-driven portable pump to take suction from CR-3 intake and discharge canals. The licensee stated that, considering the very low-probability of beyond-DBAs affecting the SFP, these diverse strategies provide defense-in-

depth and time to provide makeup or spray to the SFP before the onset of zirconium cladding ignition.

In addition, in the unlikely situation that a radiological release occurs, elements of the EP requirements, for which the exemptions are requested, would still facilitate notification of and coordination with offsite authorities because the licensee will use the State Hot Ringdown System to notify State Watch Office Tallahassee (SWOT), as designated in the CR-3 Permanently Defueled Emergency Plan (PDEP), of a declared emergency. Notification of an emergency is provided verbally or electronically to the SWOT based on the content of the Florida Nuclear Plant Emergency Notification Form. The content of the initial notification and follow-up message form has been established in conjunction with the State of Florida. The Florida Nuclear Plant Emergency Notification Form contains the date and time of the incident, the class of the emergency, and the applicable emergency action levels (EALs). The NRC staff also notes that the proposed exempted EP requirements will necessitate that the licensee be able to determine if a radiological release is occurring. If a release is postulated to occur, the licensee's staff would be in a position to promptly communicate that information to offsite authorities for their consideration in determining an appropriate response.

The NRC staff provided an evaluation of DEF's exemption requests to the Commission in SECY-14-0118, "Request by Duke Energy Florida, Inc., for Exemptions from Certain Emergency Planning Requirements," dated October 29, 2014 (Reference 13), which was approved by the Commission in the SRM to SECY-14-0118, dated December 30, 2014 (Reference 14).

2.0 REGULATORY EVALUATION

The regulations at 10 CFR 50.12(a)(2)(ii) provide that the NRC may, on application by a licensee or on its own initiative, grant exemptions from the requirements of the regulations in circumstances in which application of the regulation would not serve the underlying purpose of the rule or is not necessary to achieve the underlying purpose of the rule[3].

The underlying purpose of the planning standards in 10 CFR 50.47(b), the requirements in 10 CFR 50.47(c)(2), and certain requirements of 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 (EPZs) for nuclear power plants; and to ensure that licensees maintain effective offsite and onsite radiological emergency response plans.

The NRC staff relied on past precedent to assess whether the CR-3 request for EP exemptions satisfied the underlying purpose of the EP rules. The last exemptions that eliminated the requirements for formal offsite radiological EP were approved in October 2014 for Kewaunee Power Station. Prior to the Kewaunee Power Station, the last approved exemptions that eliminated the requirements for formal offsite radiological EP were for the Zion Nuclear Power Station in 1999 (Reference 15). The staff recognizes that the planning standards in 10 CFR

[3] Notwithstanding the special circumstances of the exemption request, 10 CFR 50.12(a)(1) requires that the exemption must be authorized by law, not present an undue risk to the public health and safety, and be consistent with the common defense and security.

50.47(b), the requirements in 10 CFR 50.47(c)(2), and certain requirements in 10 CFR 50, Appendix E, Section IV, were developed taking into consideration the risks associated with accidents that have the potential for significant offsite radiological dose consequences during operation of a nuclear power reactor at its licensed full-power level. As provided in Section 1.1 of this safety evaluation (SE), the staff has concluded that after a reactor is permanently shut down and defueled, the risks associated with accidents that have a potential for offsite radiological release, are significantly reduced for those licensees that are reasonably aligned with the analyses presented in NUREG-1738 (Reference 11). This position has been further informed by recent SFP studies provided in NUREG-2161 (Reference 10).

Based on the low risk of postulated beyond-DBAs that will result in significant offsite radiological consequences, the NRC staff considers that the special circumstances condition of 10 CFR 50.12(a)(2)(ii) can be met by demonstrating that CR-3 satisfies the two criteria provided below. Specifically, the requested exemptions to the planning standards in 10 CFR 50.47(b), the requirements in 10 CFR 50.47(c)(2), and certain requirements in 10 CFR 50, Appendix E, Section IV, eliminating prescribed offsite EP and reducing the scope of the onsite EP activities, are considered by the staff to satisfy an underlying purpose of the EP regulations if CR-3 site-specific analyses demonstrate:

1. An offsite radiological release will not exceed the EPA PAGs at the EAB for a DBA; and

2. In the unlikely event of a beyond-DBA resulting in a loss of all modes of cooling for the spent fuel stored in the SFP, there is a minimum of 10 hours for the hottest fuel assembly to reach the 900 degrees C, the critical temperature threshold for self-sustained oxidation of cladding in air. This will ensure that sufficient time exists to initiate appropriate mitigating actions and, if needed, sufficient time is available for offsite agencies to take protective actions using a CEMP (all-hazards) approach to protect the health and safety of the public.

2.1 Design-Basis Accidents

Pursuant to the change process permitted by 10 CFR 50.59, DEF has revised the CR-3 Final Safety Analysis Report (FSAR) to reflect the permanently shutdown and defueled condition of the facility. Chapter 14 of the FSAR describes the DBAs and transient scenarios that could apply to CR-3. The CR-3 FSAR no longer contains any transients that continue to apply to CR-3. The only accident scenarios still evaluated in the FSAR, based on the permanently shutdown and defueled status of the facility, are an accidental release of waste liquid and a fuel handling accident (FHA). The licensee's analysis demonstrated that the dose consequences from those accidents do not exceed the thresholds of the EPA PAGs or a site area emergency (SAE) at the EAB. The licensee previously evaluated a waste gas release DBA, but since the waste gas decay tanks have been released and the tanks are permanently vented to the atmosphere, a rupture of these components would no longer be an applicable initiator or source of such an accident.

The NRC staff evaluated the radiological consequences of the postulated FHA DBA against the dose criteria specified in 10 CFR 50.67, "Accident source term," and described using the following guidance:

- Regulatory Guide (RG) 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors," issued July 2000 (Reference 16);

- RG 1.194, "Atmospheric Relative Concentrations for Control Room Habitability Assessments at Nuclear Power Plants," issued June 2003 (Reference 17);

- RG 1.145, "Atmospheric Dispersion Models for Potential Accident Consequences at Nuclear Power Plant," issued February 1983 (Reference 18);

- EPA Federal Guidance Report (FGR) No. 11, "Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion," issued September 1988 (Reference 19);

- EPA FGR 12, "External Exposure to Radionuclides in Air, Water, and Soil," issued September 1993 (Reference 20);

- WASH-1238, "Environmental Survey of Transportation of Radioactive Materials to and from Nuclear Power Plants" (Reference 21);

- RG 1.23, "Meteorological Monitoring Programs for Nuclear Power Plants;" Revision 1, issued March 2007 (Reference 22);

- NUREG/CR-2858, "PAVAN: An Atmospheric Dispersion Program for Evaluating Design Basis Accidental Releases of Radiological Materials from Nuclear Power Stations" issued November 1982 (Reference 23); and

- NUREG/CR-6331, Revision 1, "Atmospheric Relative Concentrations in Building Wakes," May 1997 (Reference 24).

The FHA dose acceptance criteria are specified in NUREG-0800, "Standard Review Plan [SRP] for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR [Light-Water Reactor] Edition" SRP, Section 15.0.1, "Radiological Consequence Analyses Using Alternative Source Terms," issued July 2000 (Reference 25). The dose acceptance criteria for the FHA are a total effective dose equivalent (TEDE) of 6.3 rem at the EAB for the worst 2 hours, 6.3 rem at the outer boundary of the low population zone (LPZ), and 5 rem in the control room for the duration of the accident. RG 1.183 provides guidance to licensees on an acceptable application of alternative source term (AST) submittals, including acceptable radiological analysis assumptions for use in conjunction with the accepted AST. The NRC staff also considered relevant information in the CR-3 FSAR. The NRC approved implementation of the AST methodology for the FHA dose consequence analysis at CR-3 by License Amendment No. 199 dated September 17, 2001, to Facility Operating License DPR-72 (Reference 26).

The EPA's "Protective Action Guide and Planning Guidance for Radiological Incidents," Draft for Interim Use and Public Comment, issued March 2013 (Reference 12), provides radiological protection criteria for application to all incidents that would require consideration of protective actions, with the exception of nuclear war. This manual provides recommended numerical PAGs for the principal protective actions available to public officials during a radiological

incident. The EPA developed this manual to assist public officials in planning for emergency response to radiological incidents.

The Nuclear Energy Institute (NEI) document NEI 99-01, "Development of Emergency Action Levels for Non-Passive Reactors," Revision 6, dated November 2012 (Reference 27), provides guidance for development of EALs for reactors in a permanently defueled condition. NEI 99-01 was endorsed by the NRC in a letter dated March 28, 2013 (Reference 28). NEI 99-01 states that the accident analysis necessary to adopt the permanently defueled EAL scheme must confirm that the source terms and release motive forces are not sufficient to warrant classification of an SAE or General Emergency (GE), resulting in the maximum classification level of an Alert during an accident. An SAE would be declared for any events where exposure levels beyond the EAB are expected to exceed 10 percent of the EPA PAGs, which are a projected dose of 1 to 5 rem TEDE in four days for sheltering or evacuation of the public, and a projected dose of 5 rem child thyroid dose from radioactive iodine for administration of prophylactic drugs (potassium iodide). Correspondingly, NEI 99-01 established the SAE classification threshold as 100 millirem (mrem) TEDE or 500 mrem thyroid committed dose equivalent.

2.2 Beyond-Design-Basis Accidents

The NRC staff has long recognized that the frequency of a large radiological release at a decommissioning power reactor storing irradiated fuel in an SFP is lower than the frequency of a large offsite radiological release at an operating reactor. As stated previously in this SE, the NRC staff completed a detailed study of decommissioning SFP risk, which is documented in NUREG-1738 (Reference 11). For spent fuel that has aged one year, the NRC staff estimated the frequency of fuel uncovery to range from 5.8 per 10 million years to 2.4 per million years for the plants studied. The frequency of fuel uncovery was used as a simplifying and conservative surrogate for the overall frequency of severe fuel damage resulting from inadequate cooling following a loss-of-coolant inventory. Beyond-design-basis seismic initiating events dominate the fuel uncovery frequency estimates. Fuel cask handling accidents were also significant contributors to the frequency estimate. Because the configuration of the fuel, the storage racks, and the pool structure could be affected in unpredictable ways by a major seismic event or cask drop, the associated consequence evaluation could not rule out conditions where air cooling would be inadequate, even after many years of decay. Assuming that a worst-case spent fuel uncovery accident was to occur at CR-3, the licensee assessed the time that would be available for response measures before the onset of a potential radiological release. The licensee's analyses considered both situations where the heat of oxidation affected the heatup rate in air-cooled configurations and the adiabatic heatup in configurations where air cooling would be precluded. In this SE, the staff verifies the licensee's assumptions, calculations, and overall analyses of these two beyond-DBAs in support of the licensee's justification for the EP exemption requests in accordance with the criteria discussed in Section 2.0 of this SE.

3.0 TECHNICAL EVALUATION

3.1 Design-Basis Accidents

Since CR-3 is in a permanently defueled condition, the licensee reevaluated the most limiting (bounding) types of possible DBAs and determined that the limiting accidents are a radioactive waste handling accident and a FHA in the auxiliary building.

3.1.1 Radioactive Waste Handling Accident

The licensee evaluated postulated radioactive waste handling accidents under permanently defueled conditions to determine the most limiting radioactive waste handling accident. The bounding radioactive waste accident analysis was the evaluation of a drop (spill) of a high integrity container (HIC) on the south berm, adjacent to the auxiliary building, such that its entire contents of radioactive dewatered demineralizer resin escapes. The licensee determined that the consequences of a dropped spent resin HIC bounds all postulated radioactive waste handing accidents.

The licensee assigned the source term of radioactivity in a spent resin HIC based on review of recent radioactive waste shipments over the past 5½ years, selecting the highest. The NRC staff reviewed this methodology and finds that this source term is the largest potential source term available for release from a radioactive waste accident since the plant is in a permanently defueled plant status.

Using this source term, the licensee performed an evaluation of the offsite dose impact assuming a 100 percent loss of the contents from the HIC. The licensee concluded that an airborne release is not expected to occur with a loss of contents due to a container drop because of the low flammability and reactivity of the resin. However, for conservative evaluation purposes, the licensee assumed that 10 percent of the radioactive material is released as an airborne effluent. The NRC staff finds that a 10 percent release as an airborne effluent is a conservative assumption for an airborne release from a resin spill based on a review of WASH-1238 (Reference 21). WASH-1238 concludes that most resin waste is in solid form and the radioactive material is tightly bound to the resin. WASH-1238 also concludes that an expected airborne release rate would be less than 0.1 percent of the contents, and that even in a fire involving solid material, most of the radioactivity will remain in the ash.

The licensee evaluated the dispersion of the airborne release to the downwind EAB to determine the radioactivity concentrations using the NRC computer code PAVAN. To determine the downwind EAB concentrations, the licensee used the plant's site specific meteorological data and guidance from RG 1.194 and RG 1.145. The NRC staff finds this approach acceptable since the evaluations were performed using NRC-approved guidance on dispersion of gaseous radioactive effluent releases.

The licensee performed dose assessments to an individual at the EAB using the guidance in RG 1.183 (Reference 16), which states that the dose factors should be derived from data in EPAs FGR 11 and FGR 12. The NRC finds these calculational methods acceptable because

the calculations use guidance approved by the EPA and NRC for use in making dose assessments.

The licensee determined that the dose to the most limiting individual at the EAB from the radioactive waste handling accident is 40 mrem TEDE. The licensee also evaluated the thyroid dose from both radioactive iodine nuclides and for other radionuclides present in radioactive waste shipments. For thyroid dose from radioactive iodine, the licensee concluded that since there has been over four years of radioactive decay of the radioactive iodine isotopes, there is no substantial iodine source term and no substantial thyroid dose from iodine. For other radionuclides, the licensee evaluated the thyroid dose by comparing the dose factors in EPA FGR 11 for the TEDE dose to the dose factors for the thyroid. The licensee concluded that the thyroid dose from the non-iodine isotopic mix present in the primary resin in radioactive waste is less than (i.e., bound by) the TEDE value, since the committed dose equivalent dose factors for the thyroid organ are smaller than the TEDE dose factors. The NRC staff verified that the thyroid dose is less than the TEDE dose by comparing the dose factors in EPA FGR 11 for the TEDE dose for cobalt-60 and cesium-137 to the dose factors for the thyroid.

Based on the NRC staff's verifications of the licensee's calculations and that 40 mrem TEDE bounds the thyroid dose, the staff concludes that the maximum consequences at the EAB of the radioactive waste handling accident are below both the EPA PAGs threshold and the more limiting SAE threshold.

3.1.2 Fuel Handling Accident

In the current AST evaluation during power operation, the radiological consequence analysis evaluated the radiological consequences of a postulated FHA in the containment (without containment isolation) and in the auxiliary building. Since the assumptions and parameters used for an FHA inside containment are identical to those for an FHA in the auxiliary building, the resulting radiological consequences are the same regardless of the location of the accident. Since the fuel has been permanently removed from the containment building, an FHA in the reactor containment building is no longer possible. However, an FHA in the auxiliary building (including the SFP) is still possible.

The licensee defined the FHA in the auxiliary building as the dropping of a spent fuel assembly onto the SFP racks that hold the spent fuel such that the cladding of all the fuel rods in one assembly ruptures. The licensee assumed that the gap activity in the damaged rods is instantaneously released into the SFP. The licensee used an overall decontamination factor of 100 for iodine in elemental and particulate forms in the SFP water because a damaged assembly that lies across the top of the fuel racks would be submerged by slightly less than 23 feet of water, which is designed to mitigate the release by absorbing the radioactivity released from the damaged assembly. No decontamination factor was used for noble gases. The licensee postulated that the activity released from the SFP mixes with the auxiliary building atmosphere before being released directly to the environment through the auxiliary building vent. The licensee also assumed that the auxiliary building exhaust rate was sufficient to release the activity in a 2-hour time period. No credit was taken for control room isolation or filtered recirculation of control room air in the FHA analysis. The NRC staff finds that these assumptions are consistent with the current licensing basis FHA analysis, which does not credit any filtration by the auxiliary and the fuel handling building charcoal exhaust system or the

control room emergency ventilation system for accident mitigation, but does credit a decontamination factor of 100 for iodine released into the SFP.

A fission product decay period of 4 years was assumed. The NRC staff finds this assumption to be conservative because CR-3 ceased operation on September 26, 2009, which is greater than 4 years ago.

The licensee evaluated the radiological consequences resulting from the postulated FHA for the permanently defueled condition at CR-3 using the NRC computer code RADTRAD. The licensee evaluated the dispersion of the airborne release to the downwind EAB using the PAVAN atmospheric dispersion computer code (Reference 23). The licensee concluded that the radiological consequences at the EAB, LPZ, and in the control room are within the dose criteria for design-basis accidents specified in 10 CFR 50.67 and SRP Section 15.0.1 (Reference 25). The licensee also concluded that the radiological consequences are less than the dose criteria for declaration of a SAE as specified in NEI 99-01, Revision 6.

The NRC staff reviewed the methods, parameters, and assumptions that the licensee used in its radiological dose consequence analyses and found that they are consistent with the guidance provided in RG 1.183 (Reference 16). The staff compared the doses estimated by the licensee to the applicable criteria identified in the SRP and NEI 99-01, Revision 6, and concludes that the maximum consequences at the EAB of the FHA are below both the EPA PAGs threshold and the more limiting SAE threshold.

3.1.3 Atmospheric Dispersion Factors

The licensee evaluated two DBAs, a FHA and a radioactive waste handling accident, in their September 26, 2013, submittal. For the FHA, the licensee calculated χ/Q values for the control room, the EAB, and the LPZ. For the radioactive waste handling accident, the licensee calculated χ/Q values for the EAB. The licensee supplemented their original submittal with a complete list of calculated χ/Q values in the May 23, 2014, letter (Reference 5).

3.1.3.1 Meteorological Data

As part of the May 23, 2014, supplement, the licensee provided the hourly meteorological data set for 2003 through 2007 formatted for input into the ARCON96 atmospheric dispersion computer code (Reference 24). The licensee also provided the meteorological data in the form of a joint wind speed, wind direction, and atmospheric stability frequency distribution for the 2003 through 2007 time period for input to the PAVAN code. The NRC staff reviewed the overall quality of the meteorological data and found it to be consistent with the guidance outlined in RG 1.23 (Reference 22). The staff concludes that the data provides an acceptable basis for making estimates of atmospheric dispersion in support of the September 26, 2013, request.

3.1.3.2 Control Room Atmospheric Dispersion Factors

DEF generated χ/Q values (calculations that model the concentration and deposition of radioactive releases dispersed through the atmosphere) using the ARCON96 computer code and guidance provided in RG 1.194 to assess the control room post-accident atmospheric dispersion conditions. The NRC staff evaluated the applicability of the ARCON96 model for the

FHA and the radioactive waste handling accident and found no unusual siting, building arrangements, release characterization, release-receptor configuration, meteorological regimes, or terrain conditions that would preclude the use of the model for CR-3.

The NRC staff reviewed the licensee's assessments of the control room post-accident dispersion conditions and the ARCON96 atmospheric dispersion modeling and found that the licensee's inputs and assumptions were generally consistent with site configuration drawings and NRC staff practice. In addition, the staff generated sample comparative χ/Q value estimates and found the resultant χ/Q values to be similar to those calculated by the licensee. Based on the above, the staff concludes that the χ/Q values used by the licensee are acceptable for use in making the control room radiological consequences assessments for the FHA and radioactive waste handling accident.

3.1.3.3 Offsite Atmospheric Dispersion Factors

The licensee calculated the EAB and LPZ χ/Q values using guidance provided in RG 1.145 (Reference 18) and the PAVAN atmospheric dispersion computer code (Reference 23). The NRC staff performed a qualitative review of the inputs and assumptions used in the PAVAN computer calculations and of the resulting χ/Q values. The staff calculated comparative χ/Q values and found the results to be similar to the EAB and LPZ χ/Q values calculated by the licensee. Based on the similarities of the NRC staff's and licensee's calculated offsite EAB and LPZ χ/Q values, the staff concludes that the licensee's χ/Q values are acceptable for use in the offsite FHA and radioactive waste handling accident radiological consequences assessments.

3.1.4 Design-Basis Accidents Conclusion

As described above, the NRC staff reviewed the assumptions, inputs, and methods used by the licensee to assess the radiological consequences of DBAs for the permanently defueled condition at CR-3. The staff finds that the licensee used analysis methods and assumptions consistent with the conservative regulatory requirements and guidance identified in Section 2.1 of this SE. The staff compared the doses estimated by the licensee to the applicable criteria identified in the SRP and NEI 99-01, Revision 6. The staff finds that given the permanently shutdown and defueled condition of CR-3, with spent fuel stored in the SFP, the radiological consequences of DBAs are well below the limits of offsite radiological release and exposure limits. The staff further finds that sufficient safety margins and adequate defense-in-depth exist at CR-3 to address unanticipated events and to compensate for uncertainties in accident progression and analysis assumptions and parameters. The NRC staff finds with respect to the consequences of the remaining DBAs at CR-3, any offsite radiological release will not exceed the EPA PAGs at the EAB. Therefore, the underlying purposes of the regulations applicable to EP would still be achieved if the requested EP exemptions were granted as discussed in Section 2.0 of this SE.

3.2 Beyond Design-Basis Accidents

In Section 3.1 of Enclosure 1 to the September 26, 2013, letter, DEF discussed beyond-DBA scenarios involving the loss of pool coolant inventory with air-cooled heatup, the adiabatic heatup of the hottest fuel assembly, and the dose rates associated with a complete loss of SFP coolant inventory. In Enclosure 6 to the September 26, 2013, letter, the licensee provided

evaluations of these events in Sections 6.0, 7.0, and 8.0, respectively. The NRC staff has reviewed the licensee's beyond-DBAs and found that the scope of these postulated events reasonably encompasses the events with the greatest potential for significant radiological release from the CR-3 SFP. The NRC staff focused its review on the evaluation of beyond-DBAs involving substantial loss of SFP coolant inventory because these events, although very unlikely, have the greatest potential to result in a significant offsite release and challenge emergency response capabilities. The staff performed a review of the calculation summaries concerning: 1) an evaluation of time to the potential onset of fuel damage with air cooling available, and; 2) an evaluation of the time for the hottest fuel assembly to heat adiabatically to a temperature at which runaway oxidation of the cladding is possible. The results of the licensee's analyses show that only the adiabatic heatup of the hottest assembly would be expected to reach temperatures associated with a significant release. The assessment of the adiabatic heatup is important because it is also a criteria used by the staff in its finding of special circumstances related to the EP regulation exemptions.

3.2.1 Implementation of Supporting Actions and Commitments

In accordance with the safety analysis in NUREG-1738 (Reference 11), the beyond-design-basis event sequences that dominate risk at a decommissioning power reactor are large earthquake and cask-drop events. This is an important difference relative to an operating power reactor, where typically a large number of different initiating events make significant contributions to risk.

Assurance that the results of the NUREG-1738 analysis bound the plant-specific conditions at CR-3 can be established by assessing the facility against certain design and operational characteristics that were assumed in the risk analysis. These characteristics were identified in the NUREG-1738 study as recovery, mitigation, and emergency response activities assumptions that were relied on to evaluate the likelihood of success in event sequences. These characteristics are referred to as Industry Decommissioning Commitments (IDCs) and Staff Decommissioning Assumptions (SDAs) in the NUREG-1738 analyses.

The March 28, 2014 (Reference 3), supplement described the conformance of the CR-3 facility and operations with the IDCs and the SDAs. Included in the licensee's discussion of the IDCs and SDAs, the licensee addressed measures in place to minimize the potential risk from event sequences that dominate risk at a decommissioning reactor with fuel stored in a SFP (for example, those IDCs and SDA related to fuel cask handling activities and seismic events).

The NRC staff evaluation focuses on the conformance with IDCs and SDAs that are related to the design and operation of structures, systems, and components associated with the SFPs. The following provides the IDC and SDA item text, the licensee's response, and the staff's assessment:

IDC #1 states: 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).

To provide for safe handling of heavy loads in the vicinity of the SFP, DEF has developed procedures for handling heavy loads that comply with NUREG-0612 guidelines (Reference 29).

The licensee stated that, in accordance with License Amendment No. 239, issued December 27, 2011 (Reference 30), and License Amendment No. 241, issued June 26, 2012 (Reference 31), CR-3 will complete replacement of the auxiliary building cask handling crane with a single failure proof crane prior to moving a spent fuel shipping cask. The NRC staff finds that the qualification and operation of the CR-3 cask handing crane as single-failure-proof handling system satisfies the conditions assumed in the staff's analysis presented in NUREG-1738 with respect to protection from potential cask drop events.

IDC #2 states: Procedures and training of personnel will be in place to ensure that onsite and offsite resources can be brought to bear during an event.

IDC #3 states: Procedures will be in place to establish communication between onsite and offsite organizations during severe weather and seismic events.

IDC #4 states: 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.

The licensee described how the proposed CR-3 PDEP; the associated implementing procedures EM-202, "Duties of the Emergency Coordinator," and EM-503, "Conduct of the Emergency Mitigation Coordinator"; and the Off-site Support Directory would provide for access to offsite resources, including provisions for training, communications, and coordination to obtain offsite resources. The NRC staff concludes that the licensee has adequate procedures to satisfy the conditions assumed in the NUREG-1738 analysis regarding effective use of onsite and offsite resources to respond to events affecting the SFP.

IDC #5 states: Spent fuel pool instrumentation will include readouts and alarms in the control room (or where personnel are stationed) for spent fuel pool temperature, water level, and area radiation levels.

The licensee stated that independent narrow range SFP level indication is available in the main control room for both the "A" SFP and the "B" SFP, which are normally connected through an open gate. In addition, the facility is equipped with control room alarms for high and low SFP level, SFP high temperature via the plant computer, and high SFP area radiation levels. Therefore, the NRC staff finds that the licensee will maintain adequate SFP monitoring instrumentation to satisfy the conditions assumed in the NUREG-1738 analysis regarding monitoring events affecting the SFP.

IDC #6 states: Spent fuel pool 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 could not occur.

The spent fuel storage area contains two gates. One gate isolates the "A" SFP from the "B" SFP and one gate isolates the "B" SFP from the Cask Area. The licensee stated that the gates are not normally installed and the bottom of each SFP gate opening is located about 1 foot above fuel stored in rack modules. Therefore, the configuration of the gate openings limits the leakage from the storage pools. The NRC staff finds that the described design

features that limit the potential for drainage through the gate openings are consistent with the assumptions used in the analysis presented in NUREG-1738.

IDC #7 states: Procedures or administrative controls to reduce the likelihood of rapid drain-down events will include (1) prohibitions on the use of pumps that lack adequate siphon protection (2) controls for pump suction and discharge points. The functionality of anti-siphon devices will be periodically verified.

The licensee stated that both design features and administrative controls are in place to reduce the likelihood of rapid drain-down events. The licensee addressed permanently installed systems to ensure no drain paths are present. The installed SFP cooling loops draw through piping more than 20 feet above the top of the racks and, therefore, cannot rapidly drain the SFPs. The cask loading area contains a drain line that is normally isolated by a valve under administratively controls, but if unmitigated flow were to occur, the lowest pool elevation that could be reached would be at an elevation about 5 feet above the top of the fuel racks. The NRC staff finds that the described design features that minimize the potential for rapid drainage through permanent systems are consistent with the assumptions used in the analysis presented in NUREG-1738.

IDC #8 states: An onsite restoration plan will be in place to provide repair of the spent fuel pool cooling systems or to provide access for makeup water to the spent fuel pool. The plan will provide for remote alignment of the makeup source to the spent fuel pool without requiring entry to the refuel floor.

The licensee stated that the demineralized water system provides normal makeup to the SFP and an alternate means to provide makeup water to the SFPs without requiring entry to the refueling floor has been developed using portable equipment. Administrative controls ensure the equipment is periodically inventoried and procedures govern its use during loss of SFP level events. The NRC staff finds that the planned SFP cooling and makeup water availability conformed to the capabilities assumed for the staff's analysis presented in NUREG-1738.

IDC #9 states: Procedures will be in place to control SFP operations that have the potential to rapidly decrease SFP inventory. These administrative controls may require additional operations or management review, management physical presence for designated operations or administrative limitations such as restrictions on heavy load movements.

The licensee stated that procedures govern SFP operations, such as water transfer or heavy load movements, which have the potential to rapidly decrease SFP inventory. Fuel handling activities will not occur without a senior reactor operator or certified fuel handler providing oversight. Additionally, the licensee's measures described in IDCs #1, #6, and #7 provide supplemental assurance against a rapid decrease of SFP inventory event. The NRC staff finds that the procedures described in the application conform to the administrative controls considered in the staff's analysis presented in NUREG-1738.

IDC #10 states: 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.

The licensee stated that alternate makeup may be provided by fire service pumps that can supply makeup water to the SFP via the fire water system and fire hoses. The licensee stated that administrative controls for these components, including surveillance and operability requirements, were contained in the CR-3 Fire Protection Program. The NRC staff finds that the administrative controls described in the application conform to those considered in the staff's analysis presented in NUREG-1738.

SDA #1 states: 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 licensee described the SFP cooling and makeup water capabilities that will be maintained, consisting of two redundant motor-driven pumps, two redundant heat exchangers, an ultimate heat sink, a demineralized water system tank for makeup water, a filtration system, and isolation valves. Both motor and diesel-driven fire pumps will additionally be available to provide makeup water to the SFP. The NRC staff finds the cooling and makeup capabilities described in the application exceeded the capabilities considered in the staff's analysis presented in NUREG-1738.

SDA #2 states: Walkdowns 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.

The licensee stated that procedures require operations personnel to record SFP level and temperature in the control room and visually observe the pools at least once per shift. Abnormal procedures provide the necessary guidance to address loss of SFP cooling and loss of level conditions. The NRC staff finds that the proposed monitoring of the SFP systems is consistent with that assumed for the staff's analysis presented in NUREG-1738.

SDA #3 states: 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 an emergency.

The licensee described that level and temperature instrumentation that is based on direct measurement of the relevant parameters was provided in the main control room. The SFP high temperature and high radiation alarms have been included as potential entry conditions for EALs in the defueled state. The NRC staff finds that the SFP monitoring capability is consistent with the commitments and assumptions in the analysis presented in NUREG-1738.

SDA #4 states: Licensee determines that there are no drain paths in the SFP that could lower the pool level (by draining, suction, or pumping) more than 15 feet below the normal pool operating level.

The licensee reported that a drain path penetration in the cask loading area could inadvertently syphon the SFP level as much as 20 feet below the normal pool operating level. However, this syphon path has an administratively controlled syphon break to prevent inadvertent syphoning. In addition, the licensee also indicated that the syphon path line is small and that there would be ample time for identification and recovery, should syphoning take place through this line. Furthermore, Technical Specification 4.3.2, "Drainage," for CR-3 states that the SFP is designed and shall be maintained to prevent inadvertent draining of the pool below the level of the cask loading area drain path. Therefore, the SFP design reasonably protects against drainage consistent with the assumptions used in the NRC staff's analysis presented in NUREG-1738.

SDA #5 states: Load drop consequence analysis will be performed for facilities with non-single failure-proof systems. The analyses and any mitigative actions necessary to preclude catastrophic damage to the SFP that would lead to a rapid pool draining would be sufficient to demonstrate that there is high enough confidence in the facilities ability to withstand a heavy load drop.

As discussed under IDC #1, the licensee will use single-failure proof cranes for such loads. Therefore, the NRC staff finds that the protection against heavy load drops is consistent with the assumptions considered in the NRC staff's analysis presented in NUREG-1738.

SDA #6 states: Each decommissioning plant will successfully complete the seismic checklist provided in Appendix 2B to 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 NUREG-1738 ($<1 \times 10^{-5}$ per year including non-seismic events).

As noted in NUREG-1738, severe seismic events with relatively low frequencies of occurrence have been found to be the dominant challenge to spent fuel pool structural integrity. Attachment 1 to Appendix 2B of NUREG-1738 presents a seismic checklist to establish a high-confidence of a low probability of SFP structural failure as a result of seismic events below 1.2g peak ground acceleration. Item 10 of the NUREG-1738 seismic checklist provides an alternative to the detailed seismic analysis that specifies a delay in any reduction in EP capability until plant-specific analyses suggest a zirconium cladding fire is no longer a credible concern. The licensee has provided site-specific analyses indicating that, as of September 26, 2013, a zirconium fire would no longer be a credible outcome of events that lead to a complete draining of the CR3 SFP and allow development of natural circulation air cooling. The NRC staff's evaluation of that site-specific analysis is provided in Section 3.2.2 of this safety evaluation.

The NRC staff found that, as of September 26, 2013, the potential scenario of seismically-initiated radiological releases from the SFP is no longer credible. The staff's analysis presented in NUREG-1738 demonstrated that the fuel would either be adequately cooled by air or heat so

slowly that ample time for implementation of release prevention and mitigation measures would be available. Therefore, the conditions at CR-3 satisfy the alternative to verification of substantial seismic structural margin.

SDA #7 states: 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 CR-3 SFPs contain no Boraflex panels in the SFP racks for nuclear criticality control. The NRC finds that the criticality prevention methods at CR-3 satisfy the assumption regarding the integrity of solid neutron absorbing panels assumed in the analysis presented in NUREG-1738.

Based on the above evaluations, the NRC staff concludes that the design and operation of structures, systems, and components associated with SFP storage provide for safe storage of spent fuel and are consistent with the capabilities assumed in the staff's analysis presented in NUREG-1738.

3.2.2 Site Specific Analyses

The licensee performed site-specific quantitative analyses of beyond-DBAs affecting fuel stored in the SFPs at CR-3. In Enclosure 1 to the licensee's letter dated September 26, 2013 (Reference 2), the licensee provided a summary of the analyses used to determine (1) the date where decay heat would be low enough to preclude the fuel cladding from reaching a temperature of 565 degrees C, which is associated with the onset of fuel damage, and (2) the time for the hottest fuel assembly to heat adiabatically from its normal storage temperature to a temperature of 900 degrees C, where runaway zirconium oxidation may begin. Between 565 degrees C and 900 degrees C, zirconium oxidation may generate significant additional heat if oxygen can freely react with the zirconium cladding. Enclosure 6 to the licensee's letter dated September 26, 2013, included the permanently defueled accident analyses.

A. Analysis of Onset of Fuel Damage

In Section 6.0 of Enclosure 6 to the letter dated September 26, 2013, the licensee provided a summary of a calculation evaluating air-cooling of spent fuel in the CR-3 SFP following a complete loss of coolant inventory. This analysis used a GOTHIC (Generation of Thermal-Hydraulic Information for Containments) model of the CR-3 auxiliary building fuel handling area to determine quasi steady-state air temperatures in specific locations within the building, including the down-comer region around the spent fuel. Using these air temperatures, a separate model using the COBRA code determined the maximum fuel cladding temperature assuming a completely drained SFP and the resulting natural circulation air flow through the fuel assemblies. The COBRA code was developed by Pacific Northwest National Laboratories in the 1990s to model the heat transfer within and between fuel assemblies in storage and transportation systems.

The GOTHIC model was used to evaluate fuel handling area temperatures for yearly high summer outdoor temperature conditions with the fuel decay heat calculated for the date of September 26, 2013. The GOTHIC air temperature analysis used plant-specific parameters for the configuration of the CR-3 auxiliary building and included a number of assumptions regarding

heat sources and sinks. The decay heat rate for the fuel stored in the CR-3 SFPs was calculated to be 381 kilowatts (kWs) (1.3 million British Thermal Units (BTUs) per hour) in pool "A" and 440 kWs (1.5 million BTUs per hour) in pool "B". The ventilation system was modeled as shutdown, which minimized air exchange with the outdoor environment. The analysis also included consideration of heat absorption by structures, heat transfer through the structures to the environment, heat generation by electrical equipment, and the heat gain from solar radiation. With these assumptions, the maximum temperature in the space between the fuel storage racks and the pool wall was found to be 164 degrees C (328 degrees Fahrenheit (F)). The licensee assumed the air would absorb additional heat as it traveled down between the fuel storage racks and the pool wall such that the inlet temperature to the fuel storage cells would be 173 degrees C (344 degrees F). These temperatures were then used to establish the fuel channel entry air temperatures using the COBRA code.

The determination of the highest cladding temperature using the COBRA code assumed no heat transfer from the limiting fuel assembly to adjacent assemblies. The limiting assembly was one assembly from the final discharge that defueled the reactor following the last operating cycle (Cycle 16) with a decay heat rate of 1.7 kW (5700 BTU per hour). The licensee determined this modeled configuration was conservative based on evaluation of the fuel storage distribution in the CR-3 SFPs, where the Cycle 16 assemblies are arranged such that no two are adjacent to each other and only four assemblies were determined to have the limiting decay heat rate.

The results of the COBRA evaluation showed that the peak cladding temperature would not exceed the temperature selected to represent the onset of potential cladding damage, 565 degrees C (1049 degrees F). The COBRA results show that the peak cladding temperature would be 547 degrees C (1016 degrees F) for the limiting Cycle 16 assembly. This temperature is below the temperature associated with the onset of potential cladding damage, and far below temperatures of 900 degrees C (1652 degrees F) associated with rapid cladding oxidation and the potential for a significant radiological release.

Based on the provided analysis and the results of previous studies, the NRC staff concludes that there is reasonable assurance that, as of September 26, 2013, the spent fuel stored in the CR-3 SFPs would remain at temperatures below those associated with the onset of cladding damage following a complete loss of coolant inventory. Furthermore, the analysis demonstrated ample margin to temperatures where runway zirconium oxidation would be a concern, assuming the fuel can be air-cooled. Therefore, the NRC staff finds air cooling to be a credible method to maintain the fuel cladding temperature below that associated with the onset of cladding damage.

B. Heat-Up Analysis Assuming No Air Cooling

The licensee presented its evaluation of the response of the hottest fuel assemblies under conditions where the heat generated within the assembly would be retained within the assembly (i.e., adiabatic heat up, in Section 7.0 of Enclosure 6 to the license amendment request dated September 26, 2013). The calculation used an assumed initial temperature, the calculated thermal capacity of the fuel assembly within the heated length of the assembly, and an estimated decay heat rate for the hottest fuel assembly. From this information, the licensee

calculated the time to reach 900 degrees C (1652 degrees F), which corresponds to runaway cladding oxidation and the potential for a large radiological release.

An initial fuel assembly temperature of 30 degrees C (86 degrees F) was assumed based on precedent evaluations when the heat load has been similarly low. The time for the fuel assembly to reach specified temperatures was calculated assuming the fuel assembly was dry at the initial temperature, which is conservative relative to the actual conditions following a rare and challenging event that could lead to a loss of SFP water. For these events, water would be expected to be present for a significant time, considering the large volume of water initially in the pool, and absorb nearly all the decay heat generated during that time.

The thermal capacity of the fuel assembly was calculated based on the dimensions and materials used for the most recent fuel assembly design, which are also the assemblies producing the highest decay heat. With 4 years decay for the limiting assembly and the unlikely condition that results in an adiabatic heat up, as of September 26, 2013, the time to reach 900 degrees C (1652 degrees F), which represents the onset of runaway zirconium oxidation, would be 19.7 hours.

The NRC staff finds the adiabatic heatup calculation adequate to demonstrate that over 10 hours would be available before a significant radiological release might occur following an accident leading to loss of SFP water with no air cooling. The adiabatic heatup calculation is a simplified method for determining the minimum time available for deployment of mitigation equipment and, if necessary, implementation of offsite measures using a CEMP (all hazards) approach.

C. Assessment of Specific Events at CR-3

The NRC staff also reviewed additional low probability event assessments provided by the licensee as described below:

1. Extended Loss of Normal (Design) Heat Removal Capability

As of July 1, 2013, approximately 107.7 hours would be available before the SFP commences boiling and 19.9 days would be available to restore water cooling to the SFP before the SFP water level reaches 10 feet above the top of the fuel (additional time would be available before fuel is uncovered). Because of the relative ease with which alternative means of supplying cooling water to the SFP can be established, it is not reasonable to postulate that fuel damage can occur due to a loss of normal cooling capability to the SFP.

2. Shine from an Empty Spent Fuel Pool

Although a significant release of radioactive material from the spent fuel is highly unlikely in the absence of water cooling, the potential exists for radiation exposure to an offsite individual in the event that shielding of the fuel is lost (a beyond-design-basis event). The gamma radiation dose rate at the EAB would be sufficiently low, such that it would take more than 500 years for the event to exceed the EPA early-phase PAG of 1 rem. The EPA early-phase PAG is defined as the period beginning at the projected or actual initiation of a release and extending a few days later. The PAGs were developed to respond to a mobile airborne plume that could

transport and deposit radioactive material over a large area. In contrast, the radiation field formed by scatter from a drained SFP would be stationary rather than moving and would not cause transport or deposition of radioactive materials. This would allow sufficient time to develop and implement onsite mitigative actions and provide confidence that additional offsite measures could be taken using a CEMP (all-hazards) approach if efforts to re-establish shielding over the spent fuel are delayed.

3.2.3 Conclusion Concerning Beyond-Design-Basis Loss of SFP Cooling Water Inventory Accidents (with and without Air Cooling)

The NRC staff has confirmed the licensee's analysis showing that as of September 26, 2013, there will be well over 10 hours from the initiation of the accident until the cladding reaches a temperature where a zirconium fire is possible. Under conditions where cooling air flow can develop and air exchange between the fuel building and outside environment occurs, suitably conservative calculations indicate that, as of September 26, 2013, the fuel would remain at temperatures where the cladding would be undamaged for an unlimited period. Furthermore, as discussed in Section 3.1 of this safety evaluation, the design and operation of structures, systems, and components associated with SFP storage provide for safe storage of spent fuel and are consistent with the capabilities assumed in the analysis presented in NUREG-1738. This confirms that there is sufficient time available to support deployment of mitigation equipment consistent with plant conditions and if needed, for offsite agencies to take protective actions using a comprehensive emergency plan to protect the health and safety of the public.

4.0 EXEMPTIONS

Pursuant to 10 CFR 50.12, the Commission may, upon application by any interested person or upon its own initiative, grant exemptions from the requirements of 10 CFR Part 50 when: (1) the exemptions are authorized by law, will not present an undue risk to public health or safety, and are consistent with the common defense and security; and (2) when special circumstances are present. The NRC staff reviewed this request to determine whether the specific exemptions should be granted.

This section reflects the NRC staff's technical evaluation of the DEF's exemptions as provided to the Commission in SECY-14-0118, "Request by Duke Energy Florida, Inc. for Exemptions from Certain Emergency Planning Requirements," dated October 29, 2014 (Reference 13), which was approved by the Commission in the SRM to SECY-14-0118, dated December 30, 2014 (Reference 14).

4.1 Specific Exemptions for 10 CFR 50.47

The licensee's letter dated September 26, 2013 (Reference 2), as supplemented by letters dated March 28, 2014 (Reference 3), and August 28, 2014 (Reference 6), requested an exemption from certain requirements (as indicated by strikeout and bolded text) of the planning standards in 10 CFR 50.47 for CR-3.

4.1.1 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:

The licensee has provided revised radiological analyses that show that as of September 26, 2013, the radiological consequences of applicable DBAs at CR-3 will not exceed the limits of the EPA PAGs at the EAB. Additionally, analyses developed for beyond-DBA loss of water inventory events for the SFP show that, as of September 26, 2013, the events' radiological consequences will not exceed the limits of the EPA PAGs at the EAB, or sufficient time is available to initiate mitigative actions consistent with plant conditions, and if necessary, for offsite authorities to implement protective measures using a CEMP (all-hazards) approach to protect the health and safety of the public.

The licensee's analyses determined that if all cooling were lost to the SFP, as of July 1, 2013, it would take 107.7 hours (4.5 days) to boil and a total of 19.9 days for the SFP water inventory to reach a level of 10 feet from the top of the fuel due to boil off. Additionally, it concluded that as of September 26, 2013, in the event of a complete loss of SFP cooling water, but with natural circulation of air through the spent fuel racks, peak fuel clad temperature for the spent fuel would be less than 565 degrees C. At this temperature, fuel cladding remains intact and an offsite release would not occur.

Exemptions from offsite EP requirements have previously been approved when the site-specific analyses show that at least 10 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 degrees C. The technical basis that underlies the approval of the exemption requests is based, in part, on the analysis that spent fuel stored in the SFP is unlikely to reach the zirconium ignition temperature in less than 10 hours. This analysis uses several simplifying assumptions. Some of these assumptions are conservative (adiabatic conditions), while others are non-conservative (no oxidation below 900 degrees C). Weighing the conservatisms and non-conservatisms, while unlikely, the NRC staff judges that this calculation reasonably represents conditions that may occur in the event involving a loss of SFP water inventory due to a beyond-DBA. The staff concluded that if 10 hours were available to initiate mitigative actions consistent with plant conditions, and if necessary, for offsite authorities to implement appropriate protective measures using CEMP (all-hazards) approach, then formal offsite radiological EP is not necessary for a permanently shutdown and defueled nuclear power reactor. The licensee provided an analysis of uncovered spent fuel with no air cooling possible (adiabatic heatup) at CR-3. The results of the calculations are, as of September 26, 2013, that it would take 19.7 hours to reach a self-sustaining oxidation temperature for zirconium cladding of 900 degrees C.

The licensee provided descriptions of multiple strategies for providing makeup to the SFP, including: (1) using existing plant systems for inventory makeup; (2) supplying water through hoses to connections to the existing SFP piping using the diesel-driven fire service pump; and (3) using a diesel-driven portable pump to take suction from CR-3 intake and discharge canals. The licensee has committed to maintaining the mitigation strategies for the loss of large areas of the plant due to an explosion or a fire previously required under 10 CFR 50.54(hh)(2), which will continue to be required as a license condition. Considering the very low-probability of beyond-

DBAs affecting the SFP, these diverse strategies provide defense-in-depth against the events, which could lead to the onset of zirconium cladding ignition.

For all the reasons stated, and consistent with the NRC staff's regulatory evaluation basis in Section 2.0 of this safety evaluation, the NRC staff finds the licensee's requested exemptions meet the underlying purpose of the planning standards in 10 CFR 50.47(b) and requirements in Appendix E to Part 50, and acceptably satisfies the special circumstances in 10 CFR 50.12(a)(2)(ii) in view of the greatly reduced risk of offsite radiological consequences associated with the permanently shutdown and defueled state of the plant.

4.1.2 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.

NUREG-0396, "Planning Basis for the Development of State and Local Government Radiological Emergency Response Plans in Support of Light Water Nuclear Power Plants," dated December 1978 (Reference 32), provided that emergency response plans should be useful for responding to any accident that would produce offsite radiological doses in excess of the EPA PAGs. Additionally, it introduced the concept of generic plume exposure pathway zones as a basis for the planning of response actions, which would result in dose savings in the vicinities of nuclear facilities in the event of a serious power reactor accident. As previously discussed in Section 4.1.1 of this SE, the licensee has provided revised radiological analyses that show that, as of September 26, 2013, the radiological consequences for applicable DBAs at CR-3 will not exceed the limits of the EPA PAGs at the EAB. Based on the above analysis and the analysis in Section 4.1.1, the NRC staff concludes that CR-3 has met the criteria under 10 CFR 50.12 for an exemption from the requirements of 10 CFR 50.47(b)(1) specifying "within the Emergency Planning Zones."

4.1.3 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.

The NRC requires a level of licensee EP commensurate with the potential consequences to public health and safety and common defense and security at the licensee's site. With the termination of reactor operations at CR-3 and the permanent removal of the fuel from the reactor core, most of the accident scenarios postulated for operating reactors are no longer possible. The irradiated fuel is now stored in the SFP and will remain onsite until it can be moved offsite for long-term storage or disposal. The reactor, RCS, and secondary systems are no longer in operation and have no function related to the storage or the irradiated fuel. Therefore postulated accidents involving failure or malfunction or the reactor, RCS, or

supporting systems are no longer applicable. During reactor decommissioning, the principal public safety concerns involve the possible radiological risks associated with the storage of spent fuel onsite. As such, a separate emergency operations facility (EOF) would not be required. Onsite operations staff will continue to maintain and provide for communication and coordination capabilities with offsite organizations for the level of support required for the remaining DBAs and the prompt implementation of mitigative actions in response to a SFP accident.

Based on the above analysis and the analysis provided in Section 4.1.1 of this SE, the NRC staff concludes that CR-3 has met the criteria under 10 CFR 50.12 for an exemption from the requirements of 10 CFR 50.47(b)(3) specifying "arrangements to accommodate State and local staff at the licensee's Emergency Operations Facility have been made."

4.1.4 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~~.

The NRC requires a level of licensee EP commensurate with the potential consequences to public health and safety, and common defense and security at the licensee's site. The licensee's exemption request included radiological analyses to show that, as of September 26, 2013, the radiological consequences of DBAs will not exceed the limits of the EPA PAGs at the EAB. As discussed previously, the licensee furnished information to supplement its exemption request concerning its SFP inventory makeup strategies. Considering the very low-probability of beyond-DBAs affecting the SFP, and with the time available to initiate mitigative actions consistent with plant conditions or, if needed, implementation of protective actions by offsite authorities using a CEMP (all-hazards) approach between the initiating event and before the onset of a postulated zirconium cladding fire, formal offsite radiological emergency plans are not needed. Therefore, reliance on information provided by CR-3 for initial offsite response measures, based on emergency classification, would not be required.

NEI 99-01, Revision 6 (Reference 27), provides EALs for non-passive operating nuclear power reactors, permanently defueled power reactors, and ISFSIs (when licensees elect to use their 10 CFR Part 50 emergency plan to fulfill the requirements of 10 CFR 72.32 for a stand-alone ISFSI).

Based on the above analysis and the analysis provided in Section 4.1.1 of this SE, the NRC staff concludes that CR-3 has met the criteria under 10 CFR 50.12 for an exemption from the requirements in 10 CFR 50.47(b)(4) that "and State and local response plans call for reliance on information provided by facility licensees for determinations of minimum initial offsite response measures."

4.1.5 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~~.

The NRC requires a level of licensee EP commensurate with the potential consequences to public health and safety, and common defense and security at the licensee's site. The licensee's exemption request included radiological analyses to show that, as of September 26, 2013, the radiological consequences of DBAs will not exceed the limits of the EPA PAGs at the EAB. Unlike operating reactor accident sequences potentially leading to a large early release, accident scenarios at decommissioning plant SFPs evolve slowly and provide a longer time period to deal with both SFP mitigative actions and protective actions, including public evacuation, if necessary. As discussed previously, DEF furnished information to supplement its exemption request concerning its SFP inventory makeup strategies. Considering the very low probability of beyond-design-basis events affecting the SFP, and with the time available between the initiating event and before the onset of a postulated zirconium cladding fire to initiate mitigative actions consistent with plant conditions or, if needed, implementation of protective actions by offsite authorities using a CEMP (all-hazards) approach, formal offsite radiological emergency plans are not needed. Therefore, a means to provide early notification and clear instruction to the populace within a designated EPZ is not required.

Based on the above analysis and the analysis provided in Section 4.1.1 of this SE, the NRC staff concludes that CR-3 has met the criteria under 10 CFR 50.12 for an exemption from the requirements in 10 CFR 50.47(b)(5) specifying "and the public," and "and means to provide early notification and clear instruction to the populace within the plume exposure pathway EPZ have been established."

4.1.6 10 CFR 50.47(b)(6)

> Provisions exist for prompt communications among principal response organizations to emergency personnel ~~and to the public~~.

The NRC requires a level of licensee EP commensurate with the potential consequences to public health and safety and common defense and security at the licensee's site. The licensee's exemption request included radiological analyses to show that, as of September 26, 2013, the radiological consequences of DBAs will not exceed the limits of the EPA PAGs at the EAB. Unlike operating reactor accident sequences potentially leading to a large early release, accident scenarios at decommissioning plant SFPs evolve slowly and provide a longer time period to deal with both SFP mitigative actions and protective actions, including public evacuation if necessary. As discussed previously, DEF furnished information to supplement its exemption request concerning its SFP inventory makeup strategies. Considering the very low-probability of beyond-design-basis events affecting the SFP, and with the time available between the initiating event and before the onset of a postulated zirconium cladding fire to initiate mitigative actions consistent with plant conditions or, if needed, implementation of

protective actions by offsite authorities using a CEMP (all-hazards) approach, formal offsite radiological emergency plans are not needed. Therefore, the requirement to provide prompt communication to the public in regards to initial or predetermined protective actions is not required.

Based on the above analysis and the analysis provided in Section 4.1.1 of this SE, the NRC staff concludes that CR-3 has met the criteria under 10 CFR 50.12 for an exemption from the requirements in 10 CFR 50.47(b)(6) to provide prompt communications "and to the public."

4.1.7 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.

The NRC requires a level of licensee EP commensurate with the potential consequences to public health and safety and common defense and security at the licensee's site. The licensee's exemption request included radiological analyses to show that, as of September 26, 2013, the radiological consequences of DBAs will not exceed the limits of the EPA PAGs at the EAB. Unlike operating reactor accident sequences potentially leading to a large early release, accident scenarios at decommissioning plant SFPs evolve slowly, and provide a longer time period to deal with both SFP mitigative actions and protective actions, including public evacuation if necessary. As discussed previously, DEF furnished information to supplement its exemption request concerning its SFP inventory makeup strategies. Considering the very low-probability of beyond-design-basis events affecting the SFP, and with the time available to initiate mitigative actions consistent with plant conditions or if needed, implementation of protective actions by offsite authorities using a CEMP (all-hazards) approach between the initiating event and before the onset of a postulated zirconium fire, formal offsite radiological emergency plans are not needed. Therefore, the requirement to provide periodic information to the public on how they will be notified and what their initial or predetermined protective actions should be in an emergency is not required.

Based on the above analysis and the analysis provided in Section 4.1.1 of this SE, the NRC staff concludes that CR-3 has met the criteria under 10 CFR 50.12 for an exemption from the requirements of 10 CFR 50.47(b)(7) that "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)," and "(including the physical location or locations)" of points of contact with the news media.

4.1.8 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.

The NRC requires a level of licensee EP commensurate with the potential consequences to public health and safety and common defense and security at the licensee's site. The licensee's exemption request included radiological analyses to show that, as of September 26, 2013, the radiological consequences of DBAs will not exceed the limits of the EPA PAGs at the EAB. As discussed previously, DEF furnished information to supplement its exemption request concerning its SFP inventory makeup strategies. Considering the very low-probability of beyond-design-basis events affecting the SFP, and with the time available to initiate mitigative actions consistent with plant conditions or if needed, implementation of protective actions by offsite authorities using a CEMP approach between the initiating event and before the onset of a postulated zirconium fire, formal offsite radiological emergency plans are not needed. Therefore, the requirement for assessing or monitoring offsite consequences beyond the EAB is not required.

Based on the above analysis and the analysis provided in Section 4.1.1 of this SE, the NRC staff concludes that CR-3 has met the criteria under 10 CFR 50.12 for an exemption from the requirements in 10 CFR 50.47(b)(9) related to assessing and monitoring actual or potential "offsite" consequences of a radiological emergency condition.

4.1.9 10 CFR 50.47(b)(10)

> A range of protective actions has been developed for the ~~exposure pathway EPZ for~~ emergency workers and the public. ~~In developing this range of actions, consideration has been given to evacuation, sheltering, and, as a supplement to these, the prophylactic use of potassium iodide (KI), as appropriate. Evacuation time estimates have been developed by applicants and licensees. Licensees shall update the evacuation time estimates on a periodic basis. Guidelines for the choice of protective actions during an emergency, consistent with Federal guidance, are developed and in place, and protective actions for the ingestion exposure pathway EPZ appropriate to the locale have been developed.~~

CR-3 is no longer considered a "nuclear reactor" as defined in the NRC's regulations due to its permanently shutdown and defueled state. The CR-3 SFP, licensed as a general license under 10 CFR Part 72, Subpart K, "General License for Storage of Spent Fuel at Power Reactor Sites," is not considered a co-located ISFSI due to CR-3's current status. In 1995, the Commission provided its view on evacuation planning for an ISFSI, not co-located at a reactor site, in its Statement of Considerations (SOC) for the Final Rule for EP requirements for ISFSIs and monitored retrievable storage installations (MRS) (60 *Federal Register* (FR) 32439), "The Commission does not agree that as a general matter emergency plans for an ISFSI must include evacuation planning."

The NRC staff finds the licensee's proposal to discontinue formal offsite radiological EP activities, thus relying on a CEMP (all-hazard) planning approach, and reduce the scope of onsite EP is acceptable in view of the greatly reduced offsite radiological consequences associated with the permanently shutdown and defueled state of the power reactor. The NRC staff has determined that no credible events would result in doses to the public that would exceed the EPA PAGs at the EAB. Therefore, designated plume exposure pathway and ingestion pathway EPZs and the associated protective actions developed from evacuation time estimates (ETE) are no longer required. Additionally, the staff concludes that the licensee provides for an acceptable level of emergency preparedness at CR-3 in its permanently shutdown and defueled condition, and also provides reasonable assurance that adequate protective measures can and will be taken in the event of a radiological emergency at CR-3. In the unlikely event of an SFP accident, the iodine isotopes, which contribute to an offsite dose from an operating reactor power accident, are not present, so KI distribution would no longer serve as an effective or necessary supplemental protective action.

The 2011 Final Rule "Enhancements to Emergency Preparedness Regulations" (EP Final Rule) published in the *Federal Register* (76 FR 72560) on November 23, 2011 (Reference 33), changed the regulation by adding to 10 CFR 50.47(b)(10) the requirements, "Evacuation time estimates have been developed by applicants and licensees. Licensees shall update the evacuation time estimates on a periodic basis." These requirements to develop and update an ETE are primarily used to inform offsite protective action decision making.

Although formal offsite radiological emergency planning has typically been exempted for decommissioning sites, offsite organizations will continue to be relied upon for firefighting, law enforcement, ambulance and medical services in support of the licensee's (onsite) emergency plan. Additionally, the licensee is responsible for the control of activities within the EAB, including public access.

Based on the above analysis and the analysis provided in Sections 4.1.1 and 4.1.2 of this SE, the NRC staff concludes that CR-3 has met the criteria under 10 CFR 50.12 for an exemption from the requirements in 10 CFR 50.47(b)(10) for "plume exposure pathway EPZ for" and "In developing this range of actions, consideration has been given to evacuation, sheltering, and, as a supplement to these, the prophylactic use of potassium iodide (KI), as appropriate. Evacuation time estimates have been developed by applicants and licensees. Licensees shall update the evacuation time estimates on a periodic basis. Guidelines for the choice of protective actions during an emergency, consistent with Federal guidance, are developed and in place, and protective actions for the ingestion exposure pathway EPZ appropriate to the locale have been developed."

4.1.10 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.**~~

Based on the analysis in Sections 4.1.2 and 4.1.9 of this SE the NRC staff concludes that CR-3 has met the criteria under 10 CFR 50.12 for an exemption from having an EPZ. Specifically, the NRC staff concludes that CR-3 has met the criteria under 10 CFR 50.12 for an exemption from the requirements in the first sentence of 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." In addition, the NRC staff concludes that CR-3 has met the criteria under 10 CFR 50.12 for an exemption from the last sentence of 10 CFR 50.47(c)(2): "The plans for the ingestion pathway shall focus on such actions as are appropriate to protect the food ingestion pathway." "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" is not applicable to CR-3, and therefore, requires no exemption.

4.2 Specific Exemptions for 10 CFR Part 50, Appendix E, Section IV

The licensee's letter dated September 26, 2013, requested an exemption from certain requirements (as indicated by strikeout and bolded text) of Appendix E to 10 CFR 50 for CR-3.

4.2.1 10 CFR Part 50, Appendix E, Section IV.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, 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.

In the EP Final Rule (Reference 33), the Commission defined "hostile action" as, in part, an act directed toward a nuclear reactor or its personnel. The NRC excluded non-power reactors from the scope of "hostile action" at the time of the rulemaking because, as defined in 10 CFR 50.2, a non-power reactor is not considered a nuclear power reactor and a regulatory basis had not been developed to support the inclusion of non-power reactors in the scope of "hostile action." Similarly, a decommissioning power reactor or ISFSI is not a "nuclear reactor" as defined in the NRC's regulations. A decommissioning power reactor also has a low likelihood of a credible

accident resulting in radiological releases requiring offsite protective measures. For these reasons, the NRC staff concludes that a decommissioning power reactor is not a facility that falls within the scope of "hostile action" in the EP Final Rule.

Similarly, for security, risk insights can be used to determine which targets are important to protect against sabotage. A level of security commensurate with the consequences of a sabotage event is required and is evaluated on a site-specific basis. The severity of the consequences declines as fuel ages and, thereby, removes over time the underlying concern that a sabotage attack could cause offsite radiological consequences.

Although, this analysis provides a justification for exempting CR-3 from "hostile action" related requirements, some EP requirements for security-based events are maintained. The classification of security-based events, notification of offsite authorities, and coordination with offsite agencies under a CEMP (all-hazards) approach are still required.

Based on the above analysis, the NRC staff concludes that CR-3 has met the criteria under 10 CFR 50.12 for an exemption from the requirement in 10 CFR Part 50, Appendix E, Section IV.1 for "onsite protective actions during hostile action."

4.2.2 10 CFR Part 50, Appendix E, Section IV.2

> ~~**This nuclear power reactor license applicant shall also provide an analysis of the time required to evacuate various sectors and distances within the plume exposure pathway EPZ for transient and permanent populations using the most recent U.S. Census Bureau data as of the date the applicant submits its application to the NRC.**~~

Based on the analysis in Section 4.1.9 of this SE, the NRC staff concludes that CR-3 has met the criteria under 10 CFR 50.12 for an exemption from the requirements in 10 CFR Part 50, Appendix E, Section IV.2 to have an plume exposure pathway EPZ.

4.2.3 10 CFR Part 50, Appendix E, Section 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.**~~

Based on the analysis in Sections 4.1.1 and 4.1.9 of this SE, the NRC staff concludes that CR-3 has met the criteria under 10 CFR 50.12 for an exemption from the requirements in 10 CFR Part 50, Appendix E, Section IV.3 to develop and update ETEs and from having an offsite emergency response plan.

4.2.4 10 CFR Part 50, Appendix E, Section 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.~~

Based on the analysis in Sections 4.1.1 and 4.1.9 of this SE, the NRC staff concludes that CR-3 has met the criteria under 10 CFR 50.12 for an exemption from the requirements in 10 CFR Part 50, Appendix E, Section IV.4 to develop and update ETEs and from having an offsite emergency plan.

4.2.5 10 CFR Part 50, Appendix E, Section IV.5

> ~~During the years between decennial censuses, nuclear power reactor licensees shall estimate EPZ permanent resident population changes once a year, but no later than 365 days from the date of the previous estimate, using the most recent U.S. Census Bureau annual resident population estimate and State/local government population data, if available. These licensees shall maintain these estimates so that they are available for NRC inspection during the period between decennial censuses and shall submit these estimates to the NRC with any updated ETE analysis.~~

Based on the analysis in Sections 4.1.1 and 4.1.9 of this SE, the NRC staff concludes that CR-3 has met the criteria under 10 CFR 50.12 for an exemption from the requirements in 10 CFR Part 50, Appendix E, Section IV.5 to have an EPZ and update the ETEs and from having an offsite emergency plan..

4.2.6 10 CFR Part 50, Appendix E, Section 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.~~

Based on the analysis in Sections 4.1.1 and 4.1.9 of this SE, the NRC staff concludes that CR-3 has met the criteria under 10 CFR 50.12 for an exemption from the requirements in 10 CFR Part 50, Appendix E, Section IV.6 to have an EPZ and update the ETEs and from having an offsite emergency plan..

4.2.7 10 CFR Part 50, Appendix E, Section IV.A.1

> A description of the normal plant ~~operating~~ organization.

Based on the licensee's certifications that it has permanently shutdown and defueled the CR-3 reactor, 10 CFR 50.82(a)(2) no longer authorizes DEF to operate the reactor. Because the licensee cannot operate the reactors, the licensee does not have a "plant operating organization." Based on this rational, the NRC staff concludes that CR-3 has met the criteria under 10 CFR 50.12 for an exemption from the requirement in 10 CFR Part 50, Appendix E, Section IV.A.1 to describe an "operating" organization.

4.2.8 10 CFR Part 50, Appendix E, Section IV.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 decommissioning sites is generally small, but is commensurate with the need to safely store spent fuel at the facility in a manner that is protective of public health and safety. Decommissioning power reactor sites typically have a level of emergency response that does not require a response by the licensee's headquarters organization. However, this would not preclude the use of the licensee's staff normally located offsite to augment the on-shift organization, if needed. Based on this rationale, the NRC staff concludes that CR-3 has met the criteria under 10 CFR 50.12 for an exemption from the requirements of 10 CFR Part 50, Appendix E, Section IV.A.3, which specifically identifies licensee's headquarters personnel.

4.2.9 10 CFR Part 50, Appendix E, Section IV.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 entities.

The license's analysis demonstrated that no DBAs result in doses in excess of the EPA PAGs to the public beyond the EAB. The likelihood of beyond-DBAs that would result in doses in excess of the EPA PAGs to the public beyond the EAB is extremely low; however, the licensee still must be able to determine if a radiological release is occurring. If a release is occurring, then the licensee's staff should 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 based on a CEMP (all-hazards) approach, rather than that based on a detailed formal offsite radiological emergency plan.

Based on the above analysis and the analysis in Section 4.1.1 of this SE, the NRC staff concludes that CR-3 has met the criteria under 10 CFR 50.12 for an exemption from the

requirement in 10 CFR Part 50, Appendix E, Section IV.A.4 specific to the "offsite" aspect of dose projections.

4.2.10 10 CFR Part 50, Appendix E, Section IV.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 number of the licensee's staff at decommissioning sites is generally smaller than that for an operating power reactor, but is still commensurate with the need to operate the facility in a manner that is protective of public health and safety. The NRC staff considered the similarity between the staffing levels at a permanently shutdown and defueled reactor and staffing levels at an operating power reactor site, since the spectrum of accidents at a decommissioning facility is greatly reduced requiring less specialized qualifications. The minimal systems and equipment needed to maintain the spent nuclear fuel in the SFP in a safe condition requires minimal personnel and is governed by the technical specifications. As such, additional employees or other persons with special qualifications are not anticipated. Sufficient on-shift staffing is maintained to support the timely implementation of SFP mitigation actions, if needed.

Based on the above analysis and the analysis provided in Section 4.2.8 of this SE, the NRC staff concludes that CR-3 has met the criteria under 10 CFR 50.12 for an exemption from the requirements in 10 CFR Part 50, Appendix E, Section IV.A.5.

4.2.11 10 CFR 50, Appendix E, Section IV.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.

Although the NRC has previously exempted decommissioning power reactors from "hostile action" enhancements, based on the applicability as stated in the SOC for the EP Final Rule (Reference 33), the licensee's physical security plan must continue to provide high assurance against a potential security event impacting a designated target set. Therefore, some EP requirements for security-based events are maintained, such as the classification of security-based events, notification of offsite authorities, and coordination for the response of offsite organizations (i.e., law enforcement, firefighting, medical assistance) onsite.

Based on the analysis provided in Sections 4.1.1 and 4.2.1 of this SE, the NRC staff concludes that CR-3 has met the criteria under 10 CFR 50.12 for an exemption from the requirements in 10 CFR Part 50, Appendix E, Section IV.A.7, "By June 23, 2014"; "and a description of the"; and "hostile action at the site. For purposes of this appendix, 'hostile action' is defined as".

4.2.12 10 CFR 50, Appendix E, Section IV.A.8

> ~~**Identification of the State and/or local officials responsible for planning for, ordering, and controlling appropriate protective actions, including evacuations when necessary.**~~

For a decommissioning facility, offsite emergency measures are limited to support provided by local police, fire departments, and ambulance and hospital services, responding onsite. Due to the inability of a DBA to exceed the EPA PAGs at the EAB, the low probability of beyond-DBAs to exceed the EPA PAGs at the EAB, and slow progression and long time period available to deal with both the accident, pre-planned protective actions, such as evacuation, formal radiological emergency plans are not required, but could be implemented at the discretion of offsite authorities using a CEMP (all-hazards) approach.

Based on the above analysis and the analysis provided in Section 4.1.1 of this SE, the NRC staff concludes that CR-3 has met the criteria under 10 CFR 50.12 for an exemption from the requirements in 10 CFR Part 50, Appendix E, Section IV.A.8.

4.2.13 10 CFR Part 50, Appendix E, Section IV.A.9

> ~~**By December 24, 2012, for nuclear power reactor licensees, a detailed analysis demonstrating that on-shift personnel assigned emergency plan implementation functions are not assigned responsibilities that would prevent the timely performance of their assigned functions as specified in the emergency plan.**~~

Responsibilities should be well defined in the emergency plan and procedures, regularly tested through drills and exercises that are audited by the licensee, and inspected by the NRC. The duties of the on-shift personnel at a decommissioning reactor facility are not as complicated and diverse as those for an operating power reactor.

The NRC staff considered the similarity between the staffing levels at a permanently shutdown and defueled reactor and staffing levels at a non-power reactor site. The minimal systems and equipment needed to maintain the spent nuclear fuel in the SFP or in a dry cask storage system in a safe condition requires minimal personnel and is governed by the technical specifications. In the 2011 EP Final Rule, the NRC concluded that the staffing analysis requirement was not necessary for non-power reactor licensees due to the small staffing levels required to operate the facility. However, adequate on-shift staffing will be maintained to ensure the prompt implementation of SFP mitigation actions, if needed.

The NRC staff also examined the actions required to mitigate the very low probability beyond-DBAs for the SFP. The licensee provided information stating that the assigned on-shift operators were trained in the use of the procedures and adequate in number to carry out the

actions required for restoring SFP cooling/level in accordance with their procedures. The licensee also provided descriptions of multiple strategies for providing makeup to the SFP, including: (1) using existing plant systems for inventory makeup; (2) supplying water through hoses to connections to the existing SFP piping using the diesel-driven fire service pump; and (3) using a diesel-driven portable pump to take suction from CR-3 intake and discharge canals. The licensee has committed to maintaining the important mitigation strategies for the loss of large areas of the plant due to explosion or fire previously required under 10 CFR 50.54(hh)(2). These strategies will continue to be required as a license condition. DEF believes that, considering the very low-probability of beyond-DBAs affecting the SFP, these diverse strategies provide defense-in-depth against the events which could lead to the onset of zirconium cladding ignition.

Based on the above analysis, the NRC staff concludes that CR-3 has met the criteria under 10 CFR 50.12 for an exemption from the requirement of 10 CFR Part 50, Appendix E, Section IV.A.9.

4.2.14 10 CFR Part 50, Appendix E, Section IV.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 [S]tate 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.

NEI 99-01, Revision 6 (Reference 27), is an acceptable method for the development of an EAL scheme for non-passive operating nuclear power reactors, permanently defueled power reactors, and ISFSIs (when licensees elect to use their 10 CFR Part 50 emergency plan to fulfill the requirements of 10 CFR 72.32 for a stand-alone ISFSI). No offsite protective actions are anticipated to be necessary, so classification above the Alert level is no longer required, which is consistent with exemptions for previous decommissioning power reactors and the requirements of 10 CFR 72.32(a)(3) for ISFSIs.

Based on the above analysis and the analysis provided in Sections 4.1.1 and 4.2.1 of this SE, the NRC staff has concluded that CR-3 has met the criteria under 10 CFR 50.12 for an exemption from having an offsite emergency plan and "hostile action" enhancements. Therefore, the NRC staff concludes that CR-3 has met the criteria under 10 CFR 50.12 for an exemption from the requirements in 10 CFR Part 50, Appendix E, Section IV.B.1 related to:

"and outside"; "and offsite"; and "By June 20, 2012, for nuclear power reactor licensees, these action levels must include hostile action that may adversely affect the nuclear power plant."

4.2.15 10 CFR Part 50, Appendix E, Section IV.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.

Containment parameters do not provide an indication of the conditions for a permanently shutdown and defueled power reactor, and emergency core cooling systems are no longer required. Other indications, such as SFP level, SFP temperature, and area radiation monitors indicate the conditions at CR-3.

In the SOC for the Final Rule for EP requirements for ISFSIs and for MRS facilities (60 FR 32430), 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."

As discussed in Section 3.1 of this SE, the licensee's analysis demonstrates that no DBA would reach the dose criteria for the declaration of an SAE or a GE. An Alert will be the highest emergency classification possible. As discussed in Section 3.2, the probability of a beyond-DBA condition that could reach emergency classifications of an SAE or a GE is very low. In the event of a beyond-DBA consisting of the catastrophic loss of the SFP water inventory and a complete loss of air cooling, as of September 26, 2013, it would take 19.7 hours from the time the fuel attains an adiabatic condition until it reaches a temperature of 900 degrees C to initiate mitigative actions consistent with plant conditions, and if necessary, for offsite authorities to implement appropriate protective measures using a CEMP (all-hazards) approach.

Based on the above analysis and the analysis provided in Section 4.1.1 of this SE, the NRC staff concludes that CR-3 has met the criteria under 10 CFR 50.12 for an exemption from the requirements in 10 CFR Part 50, Appendix E, Section IV.C.1 related to: "and offsite," "such as the pressure in containment and the response of the Emergency Core Cooling System," and "(3) site area emergency and (4) general emergency."

4.2.16 10 CFR 50, Appendix E, Section IV.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 2011 EP Final Rule (Reference 33), non-power reactor licensees were not required to assess, classify, and declare an emergency condition within 15 minutes. A SFP and an ISFSI are also not nuclear power reactors as defined in the NRC's regulations. Non-power reactors do not have the same potential impact on public health and safety as do power reactors and non-power reactor licensees do not require complex offsite emergency response activities. Similarly, a decommissioning power reactor has a low likelihood of a credible accident resulting in radiological releases requiring offsite protective measures. Unlike operating reactor accident sequences potentially leading to a large early release, accident scenarios at decommissioning plant SFPs evolve slowly and provide a longer time period to deal with both SFP mitigative actions and protective actions, including public evacuation if necessary. For these reasons, the NRC staff concludes that a decommissioning power reactor should not be required to assess, classify, and declare an emergency condition within 15 minutes.

The licensee must still ensure that timely communication exists with appropriate offsite response organizations for the possibility of an event leading to an offsite release. In the unlikely event that the SFP water inventory is catastrophically lost, and air cooling is not possible, as of September 26, 2013, it would take 19.7 hours from the time the fuel is uncovered until it reaches a temperature of 900 degrees C to initiate mitigative actions consistent with plant conditions, and if necessary, for offsite authorities to implement appropriate protective measures using a CEMP (all-hazards) approach.

Based on the above analysis, the NRC staff concludes that CR-3 has met the criteria under 10 CFR 50.12 for an exemption from the requirements in 10 CFR Part 50, Appendix E, Section IV.C.2 for: "By June 20, 2012, nuclear power reactor"; and "within 15 minutes."

4.2.17 10 CFR 50, Appendix E, Section IV.D.1

> Administrative and physical means for notifying local, State, and Federal officials and agencies ~~**and agreements reached with these officials and agencies for the prompt notification of the public and for public evacuation or other protective measures, should they become necessary,**~~ shall be described.

> This description shall include identification ~~of the appropriate officials, by title and agency,~~ of the State and local government agencies ~~within the EPZs.~~

Based on the analyses in Sections 4.1.1, 4.1.2, and 4.1.6 of this SE the NRC staff has concluded that CR-3 has met the criteria under 10 CFR 50.12 for an exemption from the requirements to have an offsite emergency plan, EPZ, and prompt notification of the public. Therefore, the NRC staff concludes that CR-3 has also met the criteria under 10 CFR 50.12 for an exemption from the requirements in 10 CFR 50, Appendix E, Section IV.D.1 for: "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 of the appropriate officials, by title and agency"; and "within the EPZs."

4.2.18 10 CFR 50, Appendix E, Section IV.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.**~~

In the permanently shutdown and defueled condition of the power plant, the plume exposure pathway EPZ is no longer applicable. Based on the analyses in Sections 4.1.1, 4.1.2, and 4.1.5 of this SE, the NRC staff has concluded that CR-3 has met the criteria under 10 CFR 50.12 for an exemption from the requirements to have an EPZ and the means to provide early notification and clear instruction to the populace within a designated EPZ. Therefore, the NRC staff concludes that CR-3 has met the criteria under 10 CFR 50.12 for an exemption from the requirements in 10 CFR 50, Appendix E, Section IV.D.2.

4.2.19 10 CFR 50, Appendix E, Section IV.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 the site, each nuclear power reactor licensee shall demonstrate that administrative and physical means have been established for alerting and providing prompt instructions to the public with the plume exposure pathway EPZ. The design objective of the prompt public alert and notification system shall be to have the capability to essentially complete the initial alerting and notification of the public within the plume exposure pathway EPZ within about 15 minutes. The use of this alerting**~~

> ~~and notification capability will range from immediate alerting and notification of the public (within 15 minutes of the time that State and local officials are notified that a situation exists requiring urgent action) to the more likely events where there is substantial time available for the appropriate governmental authorities to make a judgment whether or not to activate the public alert and notification system. The alerting and notification capability shall additionally include administrative and physical means for a backup method of public alerting and notification capable of being used in the event the primary method of alerting and notification is unavailable during an emergency to alert or notify all or portions of the plume exposure pathway EPZ population. The backup method shall have the capability to alert and notify the public within the plume exposure pathway EPZ, but does not need to meet the 15 minute design objective for the 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.~~

While the capability needs to exist for the notification of designated offsite government authorities within a specified time period following an emergency declaration, 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. As discussed previously in Section 1.1 of this SE, CR-3 proposes to complete emergency notification through the SWOT and the NRC within 60 minutes of declaration of an emergency or a change in classification. SWOT will assume responsibility for notification of Citrus County.

In the permanently shutdown and defueled condition of the reactor, the rapidly developing scenarios associated with events initiated during reactor power operation are no longer credible. The slow progression of SFP events allows greater time for the licensee to successfully mitigate the accidents and, if necessary, for offsite authorities to protect the health and safety of the public using a CEMP (all-hazards) approach. In the permanently shutdown and defueled condition of the power plant, the alert and notification system is no longer required, and the alert and notification duties will be performed by SWOT.

Based on the above analysis and the analyses in Sections 4.1.1 and 4.1.2 of this SE, the NRC staff has concluded that CR-3 has met the criteria under 10 CFR 50.12 for an exemption from the requirements to maintain formal offsite radiological EP and an EPZ. Therefore, the NRC staff concludes that CR-3 has also met the criteria under 10 CFR 50.12 for an exemption from the requirements of "15" [minutes] and the remainder of the requirements in 10 CFR 50, Appendix E, Section IV.D.3 as indicated.

4.2.20 10 CFR Part 50, Appendix E, Section IV.D.4

> ~~If FEMA [Federal Emergency Management Agency] has approved a nuclear power reactor site's alert and notification design report, including the~~

~~**backup alert and notification capability, as of December 23, 2011, then the backup alert and notification capability requirements in Section IV.D.3 must be implemented by December 24, 2012. If the alert and notification design report does not include a backup alert and notification capability or needs revision to ensure adequate backup alert and notification capability, then a revision of the alert and notification design report must be submitted to FEMA for review by June 24, 2013, and the FEMA-approved backup alert and notification means must be implemented within 365 days after FEMA approval. However, the total time period to implement a FEMA-approved backup alert and notification means must not exceed June 22, 2015.**~~

Based on the analysis in Section 4.2.19 of this SE. the NRC staff has concluded that a primary alert and notification system is not required and CR3 has met the criteria under 10 CFR 50.12 for an exemption from the primary alert and notification system. , Therefore, the NRC staff concludes that CR-3 is not required to have a backup alert and notification capability and has met the criteria under 10 CFR 50.12 for an exemption from 10 CFR Part 50, Appendix E, Section IV.D.4.

4.2.21 10 CFR Part 50, Appendix E, Section IV.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 the inability of a DBA to exceed the EPA PAGs at the EAB, the low probability of beyond-DBAs to exceed the EPA PAGs at the EAB, and the available time to take mitigation actions consistent with plant conditions, and, if necessary, for offsite authorities to implement appropriate protective measures using a CEMP (all-hazards) approach, an EOF would not be required to support an offsite agency response. Coordination with offsite authorities and response organizations can be coordinated from the control room or another onsite location. In addition, onsite actions may be directed from the control room or another onsite location, without the requirements imposed on a technical support center (TSC). Due to the reduced size of on-shift and Emergency Response Organization (ERO) staffing for a permanently shutdown and defueled power reactor, separate facilities to accommodate emergency response staffing are no longer required. As such, greater efficiency and coordination is gained by locating staff in a central onsite facility.

Based on the above analysis, the NRC staff concludes that CR-3 has met the criteria under 10 CFR 50.12 for an exemption from the requirements in 10 CFR Part 50, Appendix E, Section IV.E 8.a.(i) for a "onsite technical support center and an emergency operations facility."

4.2.22 10 CFR 50, Appendix E, Section IV.E.8.a.(ii)

> ~~**For nuclear power reactor licensees, a licensee onsite operational support center;**~~

NUREG-0696, "Functional Criteria for Emergency Response Facilities," dated February 1981 (Reference 34), 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. The OSC should provide a location where plant logistic support can be coordinated during an emergency and restrict control room access to those support personnel specifically requested by the shift supervisor.

With the permanently shutdown and defueled status of the CR-3 reactor and the storage of the spent nuclear fuel in the SFP, an OSC is no longer required to meet its original purpose during an emergency, nor to support initial SFP mitigation actions if needed. A central facility for licensee staff to perform repair and corrective action support can improve efficiency and coordination.

Based on the above analysis and the analysis in Section 4.2.21, the NRC staff concludes that CR-3 has met the criteria under 10 CFR 50.12 for an exemption from the requirement in 10 CFR Part 50, Appendix E, Section IV.E.8.a.(ii).

4.2.23 10 CFR 50, Appendix E, Section IV.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:**~~
>
> ~~**(1) Space for members of an NRC site team and Federal, State, and local responders;**~~
>
> ~~**(2) Additional space for conducting briefings with emergency response personnel;**~~
>
> ~~**(3) Communication with other licensee and offsite emergency response facilities;**~~
>
> ~~**(4) Access to plant data and radiological information; and**~~
>
> ~~**(5) Access to copying equipment and office supplies;**~~

Based on the analysis in Sections 4.1.3 and 4.2.21 of this SE, the NRC staff has concluded that CR-3 has met the criteria under 10 CFR 50.12 for an exemption from the requirement to have an EOF. Therefore, the NRC staff concludes that CR-3 has also met the criteria under 10 CFR 50.12 for an exemption from the requirement in 10 CFR Part 50, Appendix E, Section IV.E.8.b.

4.2.24 10 CFR 50, Appendix E, Section IV.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;**~~
>
> ~~**(2) The capability to analyze plant technical information and provide technical briefings on event conditions and prognosis to licensee and offsite response organizations for each reactor at a nuclear power reactor site and for each nuclear power reactor site that the facility serves; and**~~
>
> ~~**(3) The capability to support response to events occurring simultaneously at more than one nuclear power reactor site if the emergency operations facility serves more than one site; and**~~

Based on the analysis in Sections 4.1.3 and 4.2.21 of this SE, the NRC staff has concluded that CR-3 is exempt from the requirement to have an EOF. Therefore, the NRC staff concludes that CR-3 has also met the criteria under 10 CFR 50.12 for an exemption from the requirement in 10 CFR Part 50, Appendix E, Section IV.E.8.c.

4.2.25 10 CFR 50, Appendix E, Section IV.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.**~~

Based on the above analysis and the analyses in Sections 4.2.1, 4.2.21, and 4.2.22 of this SE, the NRC staff has concluded that CR-3 has met the criteria under 10 CFR 50.12 for an exemption from the requirements to have "hostile action" enhancements and to have an EOF, TSC, and OSC. Therefore, the NRC staff concludes that CR-3 has also met the criteria under 10 CFR 50.12 for an exemption from the requirements in 10 CFR Part 50, Appendix E, Section IV.E.8.d.

4.2.26 10 CFR 50, Appendix E, Section IV.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;**~~

Due to the inability of a DBA to exceed the EPA PAGs at the EAB, the low probability of beyond-DBAs to exceed the EPA PAGs at the EAB, and the available time to take mitigation actions consistent with plant conditions, and, if necessary, for offsite authorities to implement appropriate protective measures using a CEMP (all-hazards) approach, an EOF would not be required to support an offsite agency response. Coordination with offsite authorities and response organizations can be coordinated from the control room or another onsite location.

Based on the above analysis and the analysis in Sections 4.1.3 and 4.2.21 of this SE, the NRC staff concludes that CR-3 has met the criteria under 10 CFR 50.12 for an exemption from the requirements in 10 CFR Part 50, Appendix E, Section IV.E.8.e.

4.2.27 10 CFR Part 50, Appendix E, Section IV.E.9.a

> Provision for communications with contiguous State/local governments ~~**within the plume exposure pathway EPZ**~~. Such communications shall be tested monthly.

Based on the analysis in Sections 4.1.2 and 4.1.9 of this SE, the NRC staff has concluded that CR-3 has met the criteria under 10 CFR 50.12 for an exemption from the requirement to have an EPZ. Therefore, the NRC staff concludes that CR-3 is exempt from the requirements in 10 CFR Part 50, Appendix E, Section IV.E.9.a, related to: "within the plume exposure pathway EPZ."

4.2.28 10 CFR Part 50, Appendix E, Section IV.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.**~~

Communication with State and local governments is maintained and tested in 10 CFR Part 50, Appendix E, Section IV.E.9.a above. Because of the low probability of DBAs or other credible events that would be expected to exceed the EPA PAGs at the EAB and the available time to initiate mitigative actions consistent with plant conditions and, if necessary, for offsite authorities

to employ their CEMP (all-hazards) to implement protective actions, there is no need for the TSC, EOF, or offsite field assessment teams. With the elimination of the requirements for a TSC, EOF and the field assessment teams, the requirements to perform annual testing is no longer required. Communications with State/local governments will continue to be tested monthly under 10 CFR Part 50, Appendix E, Section IV.E.9.a.

Based on the above analysis and the analyses in Sections 4.1.3 and 4.2.21 of this SE, the NRC staff concludes that CR-3 has met the criteria under 10 CFR 50.12 for an exemption from the requirements in 10 CFR Part 50, Appendix E, Section IV.E.9.c.

4.2.29 10 CFR Part 50, Appendix E, Section IV.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 near-site emergency operations**~~ facility. Such communications shall be tested monthly.

The NRC staff concludes that the functions of the control room, EOF, TSC and the 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.

Based on the above analysis and the analysis in Sections 4.1.1, 4.2.21, and 4.2.22 of this SE, the NRC staff concludes that CR-3 has met the criteria under 10 CFR 50.12 for an exemption from the requirements in 10 CFR Part 50, Appendix E, Section IV.E.9.d for "nuclear power reactor control room, the onsite technical support center, and the near-site emergency operations."

4.2.30 10 CFR Part 50, Appendix E, Section IV.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:
>
> i. Directors and/or coordinators of the plant emergency organization;
>
> ii. Personnel responsible for accident assessment, including control room shift personnel;
>
> iii. Radiological monitoring teams;
>
> iv. Fire control teams (fire brigades);
>
> v. Repair and damage control teams;

vi. First aid and rescue teams;

vii. Medical support personnel;

viii. ~~Licensee's headquarters support personnel~~;

ix. Security personnel.

In addition, a radiological orientation training program shall be made available to local services personnel; e.g., local emergency services**~~/Civil Defense~~**, local law enforcement personnel**~~, local news media persons~~**.

The number of staff required at decommissioning sites is generally small, but is commensurate with the need to safely store spent fuel at the facility in a manner that ensures public health and safety. Decommissioning sites typically have a level of emergency response that does not require additional response by headquarters personnel. Training for licensee personnel responding from company locations offsite will still be required to be trained based on ERO positions specified above. Therefore, the NRC staff considers exempting licensee's headquarters personnel from training requirements reasonable.

"Civil Defense" is an outdated term and no longer used. The category of offsite responders, which could be expected to respond onsite, is captured under "local emergency services" and "local law enforcement."

Radiological orientation training for local news media will be performed, as needed, in accordance with DEF Corporate Public Affairs protocols.

Based on the above analysis and the analysis in Section 4.1.1 of this SE, the NRC staff concludes that CR-3 has met the criteria under 10 CFR 50.12 for an exemption from the requirement in 10 CFR Part 50, Appendix E, Section IV.F.1 to provide a training program for "licensee's headquarters personnel," "Civil Defense" and "local news media persons."

4.2.31 10 CFR Part 50, Appendix E, Section IV.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.

Based on the analysis in Section 4.2.19 of this SE that concluded CR-3 is exempt from the requirements to have a public alert and notification system, the NRC staff concludes that CR-3 has met the criteria under 10 CFR 50.12 for an exemption from the requirements in 10 CFR Part 50, Appendix E, Section IV.F.2 for: "test the public alert and notification system."

4.2.32 10 CFR 50, Appendix E, Section IV.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.**~~

Due to the inability of a DBA to exceed the EPA PAGs at the EAB, the low probability of beyond-DBAs to exceed the EPA PAGs at the EAB, and the available time to take mitigation actions consistent with plant conditions, and if necessary, for offsite authorities to implement appropriate protective measures using a CEMP (all-hazards) approach, no formal offsite radiological emergency planning is required and full participation emergency plan exercises are not necessary. The intent of submitting exercise scenarios at an operating power reactor site is to check that licensees utilize different scenarios in order to prevent the preconditioning of responders at power reactors. For decommissioning power reactor sites, there are limited events that could occur, and as such, the submittal of exercise scenarios is not necessary.

Based on above analysis and the analysis in Sections 4.1.1 and 4.2.15 of this SE, the NRC staff concludes that CR-3 has met the criteria under 10 CFR 50.12 for an exemption from the requirements in 10 CFR Part 50, Appendix E, Section IV.F.2.a.

4.2.33 10 CFR 50, Appendix E, Section IV.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 have controllers intervene, and the drills may focus on the onsite exercise training objectives.

The intent of submitting exercise scenarios at an operating power reactor site is to check that licensees utilize different scenarios in order to prevent the preconditioning of responders at power reactors. For decommissioning power reactor sites, there are limited events that could occur, and as such, the submission of exercise scenarios would not fulfil the intent of the rule at decommissioning sites.

Based on the analysis in Sections 4.1.1, 4.2.21, 4.2.22, and 4.2.32 of this SE that concluded CR-3 is exempt from the requirements to have an offsite emergency plan, EOF, TSC, OSC, and to have full participation emergency plan exercises, the NRC staff concludes that CR-3 has met the criteria under 10 CFR 50.12 for an exemption from the requirements in 10 CFR 50, Appendix E, Section IV.F.2.b for "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"; "and offsite"; "protective action recommendation development, protective action decision making"; and "(Technical Support Center (TSC), Operations Support Center (OSC), and the Emergency Operations Facility (EOF))." The remaining requirements of 10 CFR Part 50, Appendix E. Section IV.F.2.b apply to CR-3.

4.2.34 10 CFR 50, Appendix E, Section IV.F.2.c.

> **~~Offsite plans for each site shall be exercised biennially with full participation by each offsite authority having a role under the radiological response plan. Where the offsite authority has a role under a radiological response plan for more than one site, it shall fully participate in one exercise every two years and shall, at least, partially participate in other offsite plan exercises in this period. If two different licensees each have licensed facilities located either on the same site or on adjacent, contiguous sites, and share most of the elements defining co-located licensees, then each licensee shall:~~**
>
> **(1) ~~Conduct an exercise biennially of its onsite emergency plan;~~**
>
> **(2) ~~Participate quadrennially in an offsite biennial full or partial participation exercise;~~**
>
> **(3) ~~Conduct emergency preparedness activities and interactions in the years between its participation in the offsite full or partial participation exercise with offsite authorities, to test and maintain interface among the affected State and local authorities and the licensee. Co-located licensees shall also participate in emergency preparedness activities and interaction with offsite authorities for the period between exercises;~~**
>
> **(4) ~~Conduct a hostile action exercise of its onsite emergency plan in each exercise cycle; and~~**
>
> **(5) ~~Participate in an offsite biennial full or partial participation hostile action exercise in alternating exercise cycles.~~**

Based on the analysis in Sections 4.1.1 and 4.2.32 of this SE, the NRC staff concludes that CR-3 is exempt from the requirements to have an offsite emergency plan and to participate in full participation emergency plan exercises. Therefore, the NRC staff concludes that CR-3 has met the criteria under 10 CFR 50.12 for an exemption from the requirements of 10 CFR 50, Appendix E, Section IV.F.2.c.

4.2.35 10 CFR 50, Appendix E, Section IV.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.**~~

Based on the analysis in Section 4.2.32 of this SE, the NRC staff has concluded that CR-3 has met the criteria under 10 CFR 50.12 for an exemption from the requirement to have full participation emergency plan exercises. Therefore, the NRC staff concluded that CR-3 has also met the criteria under 10 CFR 50.12 for an exemption from the requirements of 10 CFR 50, Appendix E, Section IV.F.2.d

4.2.36 10 CFR 50, Appendix E, Section IV.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.

Based on the analysis in Sections 4.1.2 and 4.1.9 of this SE, the NRC staff has concluded that concluded CR-3 has met the criteria under 10 CFR 50.12 for an exemption from the requirement to have a plume exposure pathway EPZ. Therefore, the NRC staff concludes that CR-3 has also met the criteria under 10 CFR 50.12 for an exemption from requirements in 10 CFR Part 50, Appendix E, Section IV.F.2.e of the "located within the plume exposure pathway EPZ."

4.2.37 10 CFR 50, Appendix E, Section IV.F.2.f.

> Remedial exercises will be required if the emergency plan is not satisfactorily tested during the biennial exercise, such that NRC~~**, in consultation with FEMA,**~~ cannot (1) find reasonable assurance that adequate protective measures can and will be taken in the event of a radiological emergency or

> (2) determine that the Emergency Response Organization (ERO) has maintained key skills specific to emergency response. ~~**The extent of State and local participation in remedial exercises must be sufficient to show that appropriate corrective measures have been taken regarding the elements of the plan not properly tested in the previous exercises.**~~

Based on the analysis in Section 4.2.32 of this SE, the NRC staff concluded that full participation emergency plan exercises are not required. FEMA does not have responsibilities related to onsite emergency preparedness, so NRC consultation with FEMA is not necessary. Based on these reasons, the staff concludes that CR-3 has met the criteria under 10 CFR 50.12 for an exemption from the requirements of 10 CFR 50, Appendix E, Section IV.F.2.f "in consultation with FEMA" and "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."

4.2.38 10 CFR 50, Appendix E, Section IV.F.2.i.

> Licensees shall use drill and exercise scenarios that provide reasonable assurance that anticipatory responses will not result from preconditioning of participants. ~~**Such scenarios for nuclear power reactor licensees must include a wide spectrum of radiological releases and events, including hostile action.**~~ Exercise and drill scenarios as appropriate must emphasize coordination among onsite and offsite response organizations.

In the SOC for the EP Final Rule (Reference 33), the NRC discussed the addition of a new Section IV.F.2.i to Appendix E to require all nuclear power reactor licensees to include hostile action in biennial evaluated exercises. The EP Final Rule also ensures that scenarios will be sufficiently varied by requiring the use of a wide spectrum of radiological releases and events to properly train responders to respond to events more realistic than those currently used in training, and to avoid preconditioning the responders to success with inappropriate anticipatory responses.

In the EP Final Rule, the NRC identified this requirement as specific for power reactor licensees. The NRC staff considered the similarity between the CR-3 facility and a non-power reactor for the low likelihood of any credible accident resulting in radiological releases requiring offsite protective measures. The results of analyses of applicable DBAs and hypothetical accident conditions evaluated for CR-3 show that there is a substantial design margin for safety to the public and onsite personnel. Unlike nuclear power reactors, permanently shutdown and defueled plants have a low risk of a radiological release and a smaller spectrum of possible events.

Based on the above analysis and the analyses provided in Section 4.2.1 of this SE, the NRC staff has concluded that CR-3 has met the criteria under 10 CFR 50.12 for an exemption from the requirements to have "hostile action" enhancements. Therefore, the NRC staff concludes that CR-3 has also met the criteria under 10 CFR 50.12 for an exemption from requirement of 10 CFR Part 50, Appendix E, Section IV.F.2.i for: "Such scenarios for nuclear power reactor licensees must include a wide spectrum of radiological releases and events, including hostile action."

4.2.39 10 CFR 50, Appendix E, Section IV.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 required by Section IV.F.2.a.~~

In the SOC for the EP Final Rule, the NRC discussed the addition of a new Section IV.F.2.j to Appendix E to require all nuclear power reactor licensees to provide an opportunity for the ERO to demonstrate proficiency in response to a wide spectrum of scenarios, including a "hostile action" and a loss of large areas of the plant due to fire or explosion. It further provides that the ERO must demonstrate key skills specific to emergency response duties in the control room, TSC, OSC, EOF, and joint information center.

In the EP Final Rule, the NRC identified this requirement as specific for nuclear power reactor licensees. Additionally, with the current conditions of the site, where only the SFP and its related support systems, structures, and components remain, there are no other facilities in which ERO personnel could demonstrate proficiency. Based on the above analysis and the analysis in Section 4.2.38 of this SE, the NRC staff concludes that CR-3 has met the criteria under 10 CFR 50.12 for an exemption from the requirements in 10 CFR Part 50, Appendix E, Section IV.F.2.j.

4.2.40 10 CFR 50, Appendix E, Section IV.I.

> ~~**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.**~~

Based on the analysis provided in Section 4.2.1 of this SE, the NRC staff has concluded that CR-3 has met the criteria under 10 CFR 50.12 for an exemption from the requirement to implement the security enhancements for a "hostile action." Therefore, the NRC staff concludes that CR-3 has also met the criteria under 10 CFR 50.12 for an exemption from the requirement in 10 CFR Part 50, Appendix E, Section IV.I.

5.0 STATE CONSULTATION

In accordance with the Commission's regulations, the Florida State official was notified of the proposed exemption on February 10, 2015. The State official had no comments.

6.0 ENVIRONMENTAL CONSIDERATION

Finding Of No Significant Impact

In accordance with 10 CFR 51.31(a), the Commission has determined that the granting of this exemption will not have a significant effect on the quality of the human environment as discussed in the NRC staff's Finding of No Significant Impact and associated Environmental Assessment published March 2, 2015 (80 FR 11233).

7.0 CONCLUSION

The NRC staff concluded that the licensee's request for an exemption from certain requirements of 10 CFR 50.47(b), 10 CFR 50.47(c), and Appendix E to 10 CFR Part 50, as specified in this safety evaluation, is acceptable in view of the greatly reduced offsite radiological consequences associated with the permanently shutdown CR-3. The conclusion is consistent with the staff's evaluation as provided to the Commission in SECY-14-0118 (Reference 13), which was approved by the Commission in the SRM to SECY-14-0118 (Reference 14).

The review considered the permanently shutdown and defueled status of CR-3 and the low likelihood of any credible accident resulting in radiological releases requiring offsite protective measures. This safety evaluation was supported by the licensee's analyses and NRC staff's assessment of both DBAs and beyond-DBAs. The NRC staff concludes that the emergency planning requirements for CR-3, as modified by the exemptions described in this safety evaluation, would provide: (1) an adequate basis for an acceptable state of emergency preparedness; and (2) in conjunction with arrangements made with offsite response agencies, reasonable assurance that adequate protective measures can and will be taken in the event of a radiological emergency at CR-3.

Accordingly, the NRC staff has determined that, pursuant to 10 CFR 50.12(a), the exemptions evaluated above are authorized by law, will not present an undue risk to the public health and safety, and is consistent with the common defense and security. Also, special circumstances are present. Specifically, the NRC staff finds the licensee's requested exemptions meet the underlying purpose of the planning standards in 10 CFR 50.47 and requirements in Appendix E to 10 Part 50, and acceptably satisfies the special circumstances in 10 CFR 50.12(a)(2)(ii) in view of the reduced risk of offsite radiological consequences associated with the permanently shut down and defueled state of the plant.

The CR-3 PDEP incorporating these exemptions will be reviewed separately under the 10 CFR 50.90 license amendment process.

8.0 REFERENCES

1. Letter from Jon A. Franke, Duke Energy, to the U.S. Nuclear Regulatory Commission, "Crystal River Unit 3 - Certification of Permanent Cessation of Power Operations and that Fuel Has Been Permanently Removed from the Reactor," dated February 20, 2013 (ADAMS Accession No. ML13056A005).

2. Letter from John Elnitsky, Duke Energy, to U.S. Nuclear Regulatory Commission, "Crystal River Unit 3 – License Amendment Request #315, Revision 0, Permanently Defueled Emergency Plan and Emergency Action Level Scheme, and Request for Exemption to Certain Radiological Emergency Response Plan Requirements Defined by 10 CFR 50," dated September 26, 2013 (ADAMS Accession No. ML13274A584).

3. Letter from John Elnitsky, Duke Energy, to U.S. Nuclear Regulatory Commission, "Crystal River Unit 3 – Exemptions to Radiological Emergency Response Plan Requirements Defined by 10 CFR 50.47 and Appendix E to Part 50, Revision 1," dated March 28, 2014 (ADAMS Accession No. ML14098A072).

4. Letter from John Elnitsky, Duke Energy, to U.S. Nuclear Regulatory Commission, "Crystal River Unit 3 - Response to Requests for Additional Information and Supplement 1 to License Amendment Request #316, Revision 0," dated May 7, 2014 (ADAMS Accession No. ML14139A006).

5. Letter from John Elnitsky, Duke Energy, to U.S. Nuclear Regulatory Commission, "Crystal River Unit 3 - Permanently Defueled Emergency Plan and Emergency Action Level Scheme, Revision 1 and Response to Request for Additional Information," dated May 23, 2014. (*Not publicly available*)

6. Letter from R. R. Reising, Duke Energy, to U.S. Nuclear Regulatory Commission, "Crystal River Unit 3 – Exemptions to Radiological Emergency Response Plan Requirements Defined by 10 CFR 50.47 and Appendix E to Part 50, Supplement," dated August 28, 2014 (ADAMS Accession No. ML14251A237).

7. U.S. Nuclear Regulatory Commission, "Staff Requirements Memorandum – SECY-14-0066 – Request by Dominion Energy Kewaunee, Inc. for Exemptions from Certain Emergency Planning Requirements," dated August 7, 2014 (ADAMS Accession No. ML14219A366).

8. Letter from John Elnitsky, Duke Energy, to U.S. Nuclear Regulatory Commission, "Crystal River Unit 3 - Post-Shutdown Decommissioning Activities Report," dated December 2, 2013 (ADAMS Accession No. ML13340A009).

9. NUREG/CR-6451, "A Safety and Regulatory Assessment of Generic BWR and PWR Permanently Shutdown Nuclear Power Plants," August 1997 (ADAMS Accession No. ML082260098).

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

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

12. U.S. Environmental Protection Agency's "Protective Action Guide and Planning Guidance for Radiological Incidents," Draft for Interim Use and Public Comment dated March 2013 (PAG Manual).

13. U.S. Nuclear Regulatory Commission, "Request by Duke Energy Florida, Inc., for Exemptions from Certain Emergency Planning Requirements," SECY-14-0118, dated October 29, 2014 (ADAMS Accession No. ML14219A444).

14. U.S. Nuclear Regulatory Commission, Staff Requirements - SECY-14-0118 - "Request by Duke Energy Florida, Inc., for Exemptions from Certain Emergency Planning Requirements," dated December 30, 2014 (ADAMS Accession No. ML14364A111).

15. U.S. Nuclear Regulatory Commission, Approval of Defueled Station Emergency Plan and Exemption from Certain Requirements of 10 CFR 50.47 for Zion Nuclear Power Station, dated August 31, 1999 (ADAMS Legacy Accession No. 9909070079).

16. Regulatory Guide (RG) 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors," July 2000 (ADAMS Accession No. ML003716792).

17. Regulatory Guide (RG) 1.194, "Atmospheric Relative Concentrations for Control Room Radiological Habitability Assessments at Nuclear Power Plants," June 2003 (ADAMS Accession No. ML031530505).

18. Regulatory Guide (RG) 1.145, "Atmospheric Dispersion Models for Potential Accident Consequence Assessments at Nuclear Power Plants, Revision 1, dated February 1983 (ADAMS Accession No. ML003740205).

19. U.S. Environmental Protection Agency Federal Guidance Report No. 11, "Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion" [EPA 520/1-88-020], September 1988.

20. Environmental Protection Agency Federal Guidance Report No. 12, "External Exposure to Radionuclides in Air, Water, and Soil" [EPA 402-R-93-81], September 1993.

21. WASH-1238, "Environmental Survey of Transportation of Radioactive Materials to and from Nuclear Power Plants," December 1972

22. Regulatory Guide (RG) 1.23, "Meteorological Monitoring Programs for Nuclear Power Plants;" Revision 1, March 2007 (ADAMS Accession No. ML070350028).

23. NUREG/CR-2858, "PAVAN: An Atmospheric Dispersion Program for Evaluating Design Basis Accidental Releases of Radiological Materials from Nuclear Power Stations," issued November 1982 (ADAMS Accession No. ML12045A149).

24. NUREG/CR-6331, Revision 1, "Atmospheric Relative Concentrations in Building Wakes dated May 1997."

25. NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR [Light-Water Reactor] Edition," SRP, Section 15.0.1, "Radiological Consequence Analyses Using Alternative Source Terms," July 2000 (ADAMS Accession No. ML003734190).

26. Letter from John M. Goshen, U.S. Nuclear Regulatory Commission to Mr. Dale E. Young, Crystal River Nuclear Plant, "Crystal River Unit 3 – Issuance of Amendment Regarding Alternative Source Term and Control Room Ventilation System (TAC No. MB0241)," Approval of License Amendment No. 199, dated September 17, 2001 (ADAMS Accession No. ML012430210).

27. Nuclear Energy Institute (NEI) 99-01, Revision 6, "Development of Emergency Action Levels for Non-Passive Reactors," dated November 21, 2012 (ADAMS Accession No. ML12326A805).

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

29. NUREG-0612, "Control of Heavy Loads at Nuclear Power Plants: Resolution of Generic Technical Activity A-36," July 1980 (ADAMS Accession No. ML070250180).

30. Letter from Farideh E. Saba, U.S. Nuclear Regulatory Commission, to Mr. Jon A. Franke, Crystal River Nuclear Plant, "Crystal River Unit 3 Nuclear Generating – Issuance of Amendment Regarding Departure from a Method for Evaluation the Auxiliary Building Overhead Crane and Revisions to Associated Commitments (TAC No. ME5208)," License Amendment No. 239 issued December 27, 2011 (ADAMS Accession No. ML11321A165).

31. Letter from Farideh E. Saba, U.S. Nuclear Regulatory Commission, to Mr. Jon A. Franke, Crystal River Nuclear Plant, "Crystal River Unit 3 Nuclear Generating – Issuance of Amendment Regarding Departure From a Method for Evaluating the Auxiliary Building Overhead Crane and Revisions to Associated Commitments (TAC No. ME8234)," License Amendment No. 241 issued was issued June 26, 2012 (ADAMS Accession No. ML12136A392).

32. NUREG-0396, "Planning Basis for the Development of State and Local Government Radiological Emergency Response Plans in Support of Light Water Nuclear Power Plants, " dated December 1978 (ADAMS Accession No. ML051390356).

33. 2011 EP Final Rule, "Enhancements to Emergency Preparedness Regulations," published in the *Federal Register* (76 FR 72560) and associated Statements of Consideration (76 FR 72589) dated November 23, 2011.

34. NUREG-0696, "Functional Criteria for Emergency Response Facilities," dated February 1981, (ADAMS Accession No. ML051390358).

Principal Contributors: M. Norris, NSIR
M. Wasem, NSIR
S. Jones, NRR
M. Kichline, NRR
S. Garry, NRR
J. White, NRR

Date: March 30, 2015

March 30, 2015

Mr. Terry D. Hobbs
Decommissioning General Manager
Crystal River Nuclear Plant (NA2C)
15760 W. Power Line Street
Crystal River, FL 34428-6708

SUBJECT: CRYSTAL RIVER UNIT 3 NUCLEAR GENERATING PLANT – EXEMPTIONS FROM CERTAIN EMERGENCY PLANNING REQUIREMENTS AND RELATED SAFETY EVALUATION (TAC NO. MF2981)

Dear Mr. Hobbs:

The U.S. Nuclear Regulatory Commission (NRC) has approved the enclosed exemptions from specific requirements of Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50.47, "Emergency plans," and Appendix E, "Emergency Planning and Preparedness for Production and Utilization Facilities," to 10 CFR Part 50. This action is in response to your application for exemptions dated September 26, 2013, "Crystal River Unit 3 – License Amendment Request #315, Revision 0, Permanently Defueled Emergency Plan and Emergency Action Level Scheme and Request for Exemption to Certain Radiological Emergency Response Plan Requirements Defined by 10 CFR 50," supplemented by letters dated March 28, May 7, May 23 and August 28, 2014.

A copy of the exemptions and the NRC staff's safety evaluation are also enclosed. The exemptions will be forwarded to the Office of the Federal Register for publication.

Sincerely,
/RA Thomas Wengert for/
Michael D. Orenak, Project Manager
Plant Licensing IV-2 and Decommissioning
Transition Branch
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-302

Enclosures:
1. Exemptions
2. Safety Evaluation

cc w/encls: Distribution via Listserv

DISTRIBUTION:
PUBLIC
LPL4-2 R/F
RidsNrrPMCrystalRiver Resource
RidsRgn2MailCenter Resource
RidsNrrDorlLpl4-2 Resource
RidsNrrLAPBlechman Resource
RidsAcrsAcnw_MailCTR Resource
JWhite, NRR
MWaters, EDO Region II
RidsNrrDraArcb Resource
RidsNrrDssSpsb Resource
RidsNsirDprDdep Resource
RidsNrrDraAphb Resource
RidsNrrDssSbpb Resource
RidsNsirDprOrlob Resource
MNorris, NSIR
MWasem, NSIR
MKichline, NRR
SGarry, NRR
SJones, NRR
RidsOgcMailCenter Resource
RidsNrrDorl Resource

ADAMS Accession Nos.: Letter ML15058A906; Exemption: ML15040A090 ****via memorandum**

OFFICE	NRR/DORL/LPL4-2/PM*	NRR/DORL/LPL4-2/LA	NSIR/DPR/ORLOB/BC**	NRR/DSS/SBPB/BC**	NRR/DRA/ARCB/BC**
NAME	MOrenak	PBlechman*	JAnderson	GCasto	UShoop
DATE	3/09/15	3/11/15	1/13/15	2/5/15	12/8/14
OFFICE	OGC	NRR/DORL/LPL4-2/BC	NRR/DORL/D	NRR/DORL/LPL4-2/PM	
NAME	SUttal - NLO	MKhanna	MEvans	MOrenak (TWengert for)	
DATE	3/30/15	3/30/15	3/30/15	3/30/15	

OFFICIAL RECORD COPY ***see previous**