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

April 17, 2015

To Those on the Enclosed List

Dear Sir or Madam:

By letter dated July 29, 2011, the Union of Concerned Scientists submitted a petition under Section 2.206, "Requests for Action under this subpart," of Title 10 of the *Code of Federal Regulations* to the U.S. Nuclear Regulatory Commission (NRC). The petition requested that the NRC issue a demand for information to a number of boiling-water reactor licensees with Mark I and Mark II containment designs. The petition has been reviewed by the NRC staff and the staff's proposed director's decision on the petition is enclosed. I request that you provide comments to me on any portions of the decision that you believe involve errors or any issues in the petition that you believe have not been fully addressed. The staff is making a similar request of the petitioner. The staff will then review any comments provided by you and the petitioner and consider them in the final version of the director's decision with no further opportunity to comment.

Please provide your comments within 30 days from the date of this letter. Please feel free to contact Mr. John G. Lamb, Petition Manager, at 301-415-3100, to discuss any questions related to this petition.

Sincerely,

Louise Lund, Acting Director
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket Nos. 50-259, 50-260, 50-296, 50-325, 50-324, 50-298, 50-237, 50-249, 50-331, 50-321, 50-366, 50-341, 50-354, 50-333, 50-263, 50-220, 50-410, 50-219, 50-277, 50-278, 50-293, 50-254, 50-265, 50-397, 50-373, 50-374, 50-352, 50-353, 50-387, 50-388, and 50-271

Enclosures:
1. List of Licensee's with Mark I & II Containment Design
2. Proposed Director's Decision

cc: Listserv

ENCLOSURE 1

LIST OF LICENSEES WITH MARK I AND II CONTAINMENT DESIGN

LIST OF LICENSEES WITH MARK I AND MARK II CONTAINMENT DESIGN

Browns Ferry Nuclear Plant, Units 1, 2 & 3 Docket Nos. 50-259, 50-260 & 50-296 Mr. Joseph W. Shea Vice President, Nuclear Licensing Tennessee Valley Authority 1101 Market Street, LP 3D-C Chattanooga, TN 37402-2801	Brunswick Steam Electric Plant Units 1 & 2 Docket Nos. 50-325 & 50-324 Mr. William R. Gideon Brunswick Steam Electric Plant Vice President Brunswick Steam Electric Plant P.O. Box 10429 Southport, NC 28461
Columbia Generating Station Docket No. 50-397 Mr. Mark E. Reddemann, Chief Executive Officer Energy Northwest MD 1023 76 North Power Plant Loop P.O. Box 968 Richland, WA 99352	Duane Arnold Energy Center Docket No. 50-331 Mr. Thomas A. Vehec Next Era Energy Duane Arnold Energy Center 3277 DAEC Road Palo, IA 52324-9785
Cooper Nuclear Station Docket No. 50-298 Mr. Oscar A. Limpias Vice President-Nuclear and CNO Nebraska Public Power District Cooper Nuclear Station 72676 648A Avenue, P.O. Box 98 Brownville, NE 68321	Fermi, Unit 2 Docket No. 50-341 Mr. Paul Fessler Senior VP and Chief Nuclear Officer DTE Electric Company Fermi 2 - 210 NOC 6400 North Dixie Highway Newport, MI 48166
James A. FitzPatrick Nuclear Power Plant Docket No. 50-333 Mr. Jeffrey Forbes Entergy Nuclear Operations, Inc. 1340 Echelon Parkway Jackson, MS 39213	Hope Creek Generating Station Docket No. 50-354 Mr. Robert Braun PSEG Nuclear LLC - N09 P. O. Box 236 Hancocks Bridge, NJ 08038
Edwin I. Hatch Nuclear Plant Units 1 & 2 Docket Nos. 50-321 & 50-366 Mr. C. R. Pierce Regulatory Affairs Director Southern Nuclear Operating Co., Inc. P.O. Box 1295 / BIN B038 Birmingham, AL 35201-1295	Nine Mile Point Nuclear Station, Units 1 and 2 Docket Nos. 50-220 & 50-410 Mr. Peter Orphanos, Site Vice President Nine Mile Point Nuclear Station 348 Lake Road Oswego, NY 13126

Monticello Nuclear Generating Plant Docket No. 50-263 Mr. Peter A. Gardner Site Vice President Northern States Power Company - Minnesota Monticello Nuclear Generating Plant 2807 West County Road 75 Monticello, MN 55362-9637	Pilgrim Nuclear Power Station Docket No. 50-293 Mr. John Dent, Jr. Pilgrim Nuclear Power Station 600 Rocky Hill Road Plymouth, MA 02360-5508
Susquehanna Steam Electric Station, Units 1 & 2 Docket Nos. 50-387 & 50-388 Mr. Timothy S. Rausch Senior Vice President and Chief Nuclear Officer PPL Susquehanna, LLC 769 Salem Boulevard, NUCSB3 Berwick, PA 18603-0467	Vermont Yankee Nuclear Power Station Docket No. 50-271 Mr. Michael A. Perito Entergy Nuclear Operations, Inc. 440 Hamilton Avenue White Plains, NY 10601
Dresden Nuclear Power Station, Units 2 & 3 Docket Nos. 50-237 & 50-249 LaSalle County Station, Units 1 & 2 Docket Nos. 50-373 & 50-374 Limerick Generating Station, Units 1 & 2 Docket Nos. 50-352 & 50-353 Oyster Creek Nuclear Generating Station Docket No. 50-219 Peach Bottom Atomic Power Station, Units 2 & 3 Docket Nos. 50-277 & 50-278 Quad Cities Nuclear Power Station, Units 1 & 2 Docket Nos. 50-254 & 50-265 Mr. Bryan Hanson President and Chief Nuclear Office Exelon Nuclear 4300 Winfield Road Warrenville, IL 60555	

ENCLOSURE 2

PROPOSED DIRECTOR'S DECISION

ML12215A314

DD-15-xxxx

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION
OFFICE OF NUCLEAR REACTOR REGULATION

William M. Dean, Director

In the Matter of	)	
	)	
Boiling-Water Reactor	)	(10 CFR 2.206)
Operating-Power Reactors	)	
With Mark I and Mark II	)	
Containment Designs	)	

PROPOSED DIRECTOR'S DECISION UNDER 10 CFR 2.206

I. Introduction

By letter dated July 29, 2011, Mr. David Lochbaum, on behalf of the Union of Concerned Scientists (the Petitioner), filed a petition in accordance with Title 10 of the *Code of Federal Regulations* (10 CFR) 2.206, "Request for Action under This Subpart." The Petitioner requested that the U.S. Nuclear Regulatory Commission (NRC) issue a demand for information (DFI) to a number of boiling-water reactor (BWR) licensees with Mark I and Mark II containment designs. The Petitioner requested that the DFI compel the licensees to describe how their individual facilities comply with 10 CFR, Appendix A, General Design Criterion (GDC) 44, "Cooling Water," and with 10 CFR 50.49, "Environmental Qualification of Electric Equipment Important to Safety for Nuclear Power Plants."

As the basis for the request, the Petitioner stated, in part, the following:

> The spent fuel pool in BWRs with Mark I and II containments is located within the reactor building, also called the secondary containment. The reactor building is a structure important to safety—it houses the emergency core cooling system pumps, as well as the control rod drive system pumps and the reactor core isolation cooling system pump, which are also capable of supplying makeup

> water to the reactor vessel. Following a design and licensing bases event, decay heat from irradiated fuel stored in the spent fuel pool is among the "combined heat load" within the reactor building that must be transferred to the ultimate heat sink to satisfy GDC 44. When system(s) prevent the spent fuel pool from boiling, the heat from piping losses, motor operation, etc., falls among the "combined heat loads." When system(s) cannot prevent the spent fuel pool from boiling following a design and licensing bases event, the heat emitted from the boiling pool falls among the "combined heat loads." One way or another, GDC 44 requires that the heat load from irradiated fuel stored in the spent fuel pools inside the reactor building at BWR Mark I and II plants be transferred to the ultimate heat sink. If GDC 44 is not satisfied, the plant's response to design and licensing bases events may be impaired or degraded. The licensees' responses to the DFI we seek would describe how they satisfy this GDC requirement, or not.
>
> [W]hen a spent fuel pool is prevented from boiling following a design and licensing bases event, the heat losses from piping and equipment used to achieve that outcome must be included or accounted for within the environmental qualification (EQ) programs mandated by 10 CFR 50.49. When a spent fuel pool cannot be prevented from boiling following a design and licensing bases event, the temperature, humidity and submergence conditions created by the boiling pool must be included or accounted for within the EQ programs. If 10 CFR 50.49 is not satisfied, the plant's response to design and licensing bases events may be impaired or degraded. The licensees' responses to the DFI we seek would describe how they satisfy this 10 CFR 50.49 requirement, or not.

The Petitioner requested that the NRC issue a DFI requiring the subject licensees to provide specific information about compliance with GDC 44 and 10 CFR 50.49. The administrative action of issuing a DFI is described in 10 CFR 2.204, "Demand for Information," as described, in part, as follows:

> (a) The Commission may issue to a licensee or other person subject to the jurisdiction of the Commission a demand for information for the purpose of determining whether an order under § 2.202 should be issued, or whether other action should be taken, which demand will:
>
> (1) Allege the violations with which the licensee or other person is charged, or the potentially hazardous conditions or other facts deemed to be sufficient ground for issuing the demand; and
>
> (2) Provide that the licensee must, or the other person may, file a written answer to the demand for information under oath or affirmation within twenty (20) days of its date, or such other time as may be specified in the demand for information.

This proposed director's decision (DD) is being issued to solicit comments on the NRC's disposition of the petition.

II. Discussion

The petition requests that the NRC issue a DFI to BWR licensees who use Mark I and Mark II containment designs to seek information on compliance with GDC 44 and 10 CFR 50.49. Criterion 44 of the GDC requires, in part, that a system be provided to transfer heat from structures, systems, and components (SSCs) important to safety to an ultimate heat sink under normal operating and accident conditions, and that this system be able to perform this safety function with either onsite or offsite power, assuming a single component failure. The regulations in 10 CFR 50.49 require licensees to establish an environmental qualification program for electrical equipment important to safety that would be exposed to harsh environmental conditions expected to develop as a result of design-basis accidents. In the following subsections, the staff discusses the applicability of these regulations to BWR spent fuel pools (SFPs).

A. Conformance with General Design Criterion 44

Regulatory Framework for the General Design Criteria

The licensees for BWRs within the scope of this petition (or their predecessors) received construction permits for these reactors between 1964 and 1974. Reviews of construction permits issued during that time period evolved from case-by-case evaluations without standard design criteria to reviews against standard design criteria applicable to light water reactors. The licensing basis documents for all reactor licensees contain a description of the degree of conformance of the plant design with certain design criteria.

The Atomic Energy Commission (AEC), the forerunner of the NRC, initiated rulemaking to enhance the rigor of the construction permit reviews. The AEC issued a draft set of design criteria for comment on November 22, 1965. This set of design criteria was significantly expanded based on the ongoing reviews and comments, and the AEC published a revised set of design criteria in the *Federal Register* (32 FR 10213) as part of a proposed rule on July 11, 1967. Under the provisions of Paragraph (a)(3)(i) of 10 CFR 50.34, "Contents of applications; technical information," which became effective on January 16, 1969, an applicant for a construction permit must include the principal design criteria (PDC) for a proposed facility and the relationship of the design bases to the PDC as part of the preliminary safety analysis report (SAR). For construction permits issued on or after May 21, 1971, the GDC in 10 CFR Part 50, Domestic Licensing of Production and Utilization Facilities," Appendix A, "General Design Criteria for Nuclear Power Plants," established the minimum requirements for the PDC for water-cooled nuclear power plants.

When the licensees applied for operating licenses, they transferred these PDC and the facility design bases from the preliminary SAR into the final SAR supporting plant operation. The licensees that were not required to include the PDC in the construction permit application have since completed evaluations demonstrating that the plant design was consistent with the intent of the GDC and included information in the final SAR describing the extent of this evaluation.

The regulations in 10 CFR 50.34(b) required each applicant for an operating license to present the design bases of the facility and safety analyses of the facility as a whole in the final SAR. For each plant, the NRC staff reviewed and accepted the design bases of the facility and the associated safety analyses in the course of the operating license review.

The reactors within the scope of this petition began commercial operation between 1969 and 1990. Although some reactors began commercial operation under provisional operating

licenses, all reactors within the scope of this petition were issued full operating licenses between 1971 and 1991. Therefore, the NRC staff has considered and accepted the design capabilities of the facility with respect to the applicable design criteria. Furthermore, the NRC has established regulations in 10 CFR 50.71(e) requiring licensees to update the final SAR periodically throughout the licensed period of a plant's operation.

The NRC staff considers GDC 61, "Fuel Storage and Handling and Radioactivity Control," applicable to the SFP heat removal function. Criterion 61 requires, in part, that the fuel storage system be designed to prevent significant reduction in coolant inventory under accident conditions and with a residual heat removal (RHR) capability having reliability and testability that reflects the importance to safety of decay heat and other residual heat removal. This criterion is appropriate for the SFP heat removal function because the SFP contains a very large inventory of coolant that mitigates the effect of any temporary loss of the SFP heat removal function, and the heat that the cooling system is intended to remove from the SFP only rarely approaches the design heat removal rate.

The NRC staff considers GDC 44, "Cooling Water," applicable to the heat removal function for normal operation of reactor support systems, for reactor decay heat removal under normal operating conditions, and for containment heat removal under reactor accident conditions. Criterion 44 of the GDC requires, in part, the provision of a system that transfers heat from SSCs which are important to safety, to an ultimate heat sink under normal operating and accident conditions. That system must also be able to perform this safety function with either onsite or offsite power, assuming a single component failure. These requirements provide greater assurance of reliability, which is important because the reactor and essential support systems have a high likelihood, if called upon for service, to be required to remove heat at or near the design rate to support essential safety functions.

The NRC Standard Review Plan (SRP) regarding the Review of the Safety Analysis Reports for Nuclear Power Plants provides guidance for NRC staff review of the complete safety analysis report included in license applications, as well as changes to the safety analysis report associated with license amendment requests. The NRC SRP guidance includes acceptance criteria derived from applicable GDC and other NRC regulations and a method acceptable to the staff to demonstrate compliance with those acceptance criteria. In Revision 2 to SRP Section 9.1.3, "Spent Fuel Pool Cooling and Cleanup System," the staff based the acceptance criteria for coolant inventory control and SFP decay heat removal capability on GDC 61. Previous versions of SRP, Section 9.1.3, included GDC 44 among the acceptance criteria for SFP decay heat removal, The staff removed the reference to GDC 44 from the review acceptance criteria in Revision 2 of SRP, Section 9.1.3, because the cooling system criterion of GDC 61 was consistent with the accepted SFP cooling system designs. The licensees for the subject facilities have not linked the SFP decay heat removal function to GDC 44 in the final SARs.

Typical Spent Fuel Pool Forced Cooling Systems at BWRs

Accepted SFP cooling system designs at BWRs typically consist of a normal forced cooling system and the capability to align the RHR system or a comparable system to provide backup or supplementary SFP cooling. The normal SFP cooling systems all have redundant pumps. The availability of redundant pumps supports prompt restoration of cooling after identification of a failure of the operating pump. The safety-related residual heat removal system or another comparable system generally can be aligned to provide backup or supplemental SFP cooling.

The qualifications of the cooling system designs vary. The normal cooling systems often consist of standard industrial systems without enhanced quality measures. However, several facilities have safety-related normal SFP cooling systems. Supplementary cooling systems

typically have enhanced quality measures, because the safety-related RHR system often performs this function. However, the staff identified the following configuration issues at some facilities that may limit the availability of the supplementary cooling systems:

- The supplementary cooling capability provided by the RHR system would be available at some facilities only during refueling when the reactor vessel is open and connected to the SFP through a flooded refueling cavity and open gates.
- The use of the RHR system as supplemental SFP cooling at some facilities requires installation of short piping segments.
- The flow path connecting the RHR system to the SFP at some facilities does not have the same level of qualification as the residual heat removal system itself.

Facility Design Basis and Applicability of Principal Design Criteria to Spent Fuel Pool Cooling

As discussed below, the NRC has determined that the heat removal design of all SFPs provides adequate assurance of public health and safety. During the operating license review and subsequent operating license amendment reviews involving SFP cooling, the NRC staff accepted the SFP cooling system designs at all subject facilities, because, in part, these designs provided a reliable RHR capability that reflects the importance to safety of decay and RHR, which is consistent with the criteria in GDC 61. As described previously, these designs generally do not ensure a continuous capability to transfer heat from the SFP to an ultimate heat sink because that capability is not necessary. The heat capacity of the SFP, combined with a low decay heat generation rate of the stored fuel, ensures a slow response of SFP temperature to loss of forced cooling events. Furthermore, the necessary heat removal decreases rapidly throughout the course of the operating cycle. Accordingly, the NRC staff has accepted operator action to realign systems and components necessary to restore forced cooling of the SFP.

The SARs for BWR facilities within the scope of the petition describe conformance to the facility PDC in a consistent manner. The SARs that include a comparison of the facility design

with the GDC of 10 CFR Part 50, Appendix A, provide a comparison of the design of the forced SFP cooling capability with GDC 61. Similarly, SARs that include a comparison of the facility against plant-specific PDC provide a comparison of the design of the forced SFP cooling capability with a principal design criterion specific to spent fuel and radioactive waste storage. Comparisons with GDC 44 or similar PDC for cooling water address the design of the cooling water systems essential for accident mitigation system operation and removal of reactor decay heat, but not the SFP cooling system.

Therefore, the design of the forced SFP cooling system is consistent with the PDC used for each facility. The PDC for forced SFP cooling (i.e., GDC 61 or a plant-specific criterion) reflect consideration of the low-decay heat rate and high-thermal capacity of the SFP. The greater capability specified by GDC 44 appropriately reflects the design basis for cooling of the reactor accident mitigation and the immediacy of the cooling water flow requirement for those events. Accordingly, GDC 44 or similar plant-specific PDC have not been applied to the design of the forced SFP cooling systems, and, therefore, compliance of the SFP cooling system design with GDC 44 is not required.

B. Conformance with the Environmental Qualification Regulation

Regulatory Framework for Environmental Qualification

The regulations in 10 CFR 50.49 require licensees to establish an environmental qualification program for electrical equipment important to safety that would be exposed to harsh environmental conditions expected to develop as a result of design-basis accidents. As stated in 10 CFR 50.49(c), these requirements do not apply to equipment in a mild environment, which is an environment that would, at no time, be significantly more severe than the environment that would occur during normal plant operation, including anticipated operational occurrences. The considered environmental conditions include temperature and pressure;

humidity; submergence (if the equipment could be subject to submergence), that could result from a design-basis accident; chemical effects; aging; synergistic effects; and margins.

As defined in 10 CFR 50.2, "Design bases" mean that information that identifies the specific functions to be performed by a SSC of a facility, and the specific values or ranges of values chosen for controlling parameters as reference bounds for design. Design-basis events are defined in 10 CFR 50.49 as conditions of normal operation, including anticipated operational occurrences, design basis-accidents, external events, and natural phenomena for which the plant must be designed to ensure the integrity of the reactor coolant pressure boundary, the capability to shut down the reactor and maintain it in a safe shutdown condition, and the capability to prevent or mitigate the consequences of accidents that could potentially result in significant offsite exposures.

Application of 10 CFR 50.49

In the licensing of the operating reactor fleet, the NRC staff has considered design basis high-energy line breaks, such as reactor loss-of- coolant accidents (LOCA) and steam-line break accidents, as causes of harsh environments. These events cause an immediate change in the environmental conditions and may result in harsh environments in areas housing equipment essential to the mitigation of these design basis events. Furthermore, conditions assumed to develop during the mitigation of these accidents, such as high radiological dose rates and pump seal leakage could lead to harsh environments, affecting mitigation equipment in areas of the plant not directly affected by the high-energy line break. Under the requirements of 10 CFR 50.49(d), licensees have identified and the NRC has accepted lists of equipment important to safety subject to environmental qualification. These lists identify the equipment, as well as the performance specifications, electrical characteristics, and environmental conditions for which the equipment was qualified. Under the requirements of 10 CFR 50.49(e), the electric

EQ program must include and be based on temperature and pressure; humidity; submergence (if the equipment could be subject to submergence), that could result from a design-basis accident; chemical effects; aging; synergistic effects; and margins.

Conversely, a sustained loss of SFP forced cooling (i.e., heating of the SFP to the extent that the reactor building environment would be substantially changed) has not been considered among the design basis events that create a harsh environment, for the purposes of environmental qualification of equipment. Because the SFP forced cooling systems at several facilities have not been qualified to withstand the effects of the site design-basis earthquake, a design basis earthquake with its attendant loss of offsite power could be postulated to result in a sustained loss of forced cooling. The NRC staff has not identified another design basis event that would directly result in a sustained loss of forced cooling. A sustained loss of SFP forced cooling would be unlikely to affect equipment essential to safely shutting down the reactor following a design basis seismic event. The low decay heat rate within the SFP would require substantial time to heat the SFP coolant inventory to near saturation conditions, and restoration of the forced cooling function has a high probability of success during that time. Although the SFP forced cooling system equipment at many facilities has not been designed to withstand the effects of the design-basis seismic event, the NRC staff has found that mechanical and electrical equipment designed and installed to satisfy general industrial standards would remain capable of performing its function after the design-basis earthquake (NUREG-0933, "Resolution of Generic Safety Issues," specifically, Task Action Plan Items A-40, "Seismic Design Criteria," and A-46, "Seismic Qualification of Equipment in Operating Plants"). This finding was based on performance of equipment used at non-nuclear facilities during and after seismic events, and is supported by the experience at nuclear facilities during recent earthquakes at nuclear facilities in Japan and the United States (see Reports at Agencywide Documents Access Management System (ADAMS) Accession Nos. ML11347A454, ML12103A092, and ML11308B406).

Therefore, the NRC staff does not expect a design-basis seismic event to result in a sustained loss of SFP forced cooling.

Because the design criterion applicable to SFP cooling specifies reliability of the RHR function, consistent with its importance to safety, the staff determined that the probability of damage to the SFP forced cooling system, the redundancy of components, and the time available for recovery of the cooling function are appropriate considerations in assessing the consequences of design basis events. With those considerations, the staff found that the existing designs of SFP forced cooling systems adequately protect against a sustained loss of the cooling function. Consequently, electrical equipment important to safety is not required to be qualified for the environmental effects of sustained SFP boiling, under 10 CFR 50.49, because that state has not been included within the design bases of the subject facilities.

C. Safety Significance of a Sustained Loss of Forced Cooling

Effects of a Sustained Loss of Forced Spent Fuel Pool Cooling

A sustained loss of SFP forced cooling would allow the pool temperature to increase, but the stored fuel would remain adequately cooled as long as an adequate SFP coolant inventory is maintained. Without forced cooling, coolant temperature at the pool surface is limited by evaporative cooling from the free surface of the pool to a value no higher than the boiling temperature (100° Celsius (C) (212° Fahrenheit (F))). The design of the pool storage racks ensures that natural circulation of the coolant will maintain the fuel cool. With the coolant at its normal level, the rack design provides for adequate natural circulation of coolant to prevent nucleate boiling on the fuel cladding surface because the cladding temperature remains below the saturation temperature at the depth of the fuel (about 116° C (241° F)). Therefore, forced cooling is not required to protect the fuel cladding integrity when adequate water level is maintained and makeup is supplied to compensate for coolant inventory loss.

As the pool surface temperature approaches 100°C (212° F), the water vapor leaving the pool surface can add a significant amount of latent heat and water vapor to the atmosphere of the building surrounding the SFP. Depending on the ventilation system design and capability, the added heat and water vapor may not be adequately removed from the building atmosphere. If not removed, the added heat and vapor could increase the building temperature and result in condensation on the surfaces.

The configuration of the reactor building in BWRs with Mark I and II containments limits the effect of the heat and condensation on systems important to reactor safety. These reactor buildings have very large free volumes over the SFP and limited openings for flow to lower-level volumes that contain the systems important to reactor safety. The water vapor evolving from the SFP surface will heat the air as it mixes. Without operating ventilation, this mixture (i.e., water vapor and heated air) would remain in the upper level of the reactor building, because it is less dense than the air elsewhere in the reactor building secondary containment volume.

For the BWR reactor building secondary containment structures enclosing a single reactor and its associated SFP, the rate of production of water vapor would be relatively low when the reactor was fully fueled and, with the reactor fully fueled and in operation, systems located within the reactor building would be important to reactor safety. These conditions exist only after reactor refueling is complete, the spent fuel has been discharged to the SFP, and the SFP has been isolated from the refueling cavity. Under these conditions, the fuel pool would not reach temperatures necessary for substantial vapor production for days, because the decay heat rate of the fuel would be low, as a result of the small amount of fuel discharged during each refueling and the time necessary to return to operation.

Once substantial water vapor generation begins, the vapor would either be vented through an operating ventilation system, or would collect in the upper reactor building, above the

refueling floor. The staff expects the safety-related standby gas treatment system in BWRs with Mark I or Mark II containments to be capable of venting much of the vapor for hours or days, depending on the decay heat rate.

Remaining vapor collecting in the upper portion of the reactor building would condense at a rate likely to prevent pressurization of the upper elevations of the reactor building, thereby preventing the heat and water vapor from being forced into the lower elevations of the reactor building. The NRC staff expects a small part of this condensation to return to the SFP or be held in other cavities on the refueling floor, and the remainder to flow through floor drains to the reactor building sumps. The sumps contain level alarms to notify operators of increasing water level and pumps to transfer water to the radioactive waste system for treatment. The flow of condensed water to the sump would not exceed the capacity of the sump pumps. Therefore, the NRC staff concludes that sustained loss of forced cooling in BWRs with single unit Mark I and II containment designs would be unlikely to produce a harsh environment that adversely affects equipment important to reactor safety.

For BWR secondary containment structures enclosing more than one reactor and its associated SFP, the decay heat rate and, consequently, the water vapor generation rate could be much higher when systems within the reactor building would be important to reactor safety. This condition exists because the entire core could be transferred to one SFP while an adjacent reactor operates at power. This secondary containment configuration exists at the following BWRs with Mark I or Mark II containments: Browns Ferry, Units 1, 2, and 3; Dresden, Units 2 and 3; Hatch, Unit 1 (The Hatch Unit 2 reactor is in a secondary containment zone separate from the zone containing the SFPs.); LaSalle, Units 1 and 2; Quad Cities, Units 1 and 2; and Susquehanna, Units 1 and 2. However, based on current operating practices, placement of the full core fuel inventory in the SFP is an infrequent condition. (Based on sampling of recent

operating reports, the NRC staff estimates the full core is present in BWR spent fuel pools less than 3 days per reactor year on average.) At three of these sites (i.e., Hatch, LaSalle, and Susquehanna), the RHR system has been qualified to provide at least one train of RHR dedicated to SFP cooling following a design-basis seismic event, and, therefore, adequate forced cooling of the SFP would be reasonably assured when the full core is in the SFP.

In 1996, the NRC staff developed and implemented a generic action plan for ensuring the safety of spent fuel storage pools (ADAMS Accession No. ML003706364). The impetus for the action plan included a concern that the design of the Susquehanna Steam Electric Station (Susquehanna) failed to meet GDC 44 with respect to sustained loss of the forced cooling function. This condition most likely resulted from a LOCA and a coincident loss of offsite power. Design-basis radiological conditions could have prevented recovery of the normal SFP cooling system after this accident. The heat and water vapor added to the reactor building atmosphere by subsequent SFP boiling could have caused failure of accident mitigation or other safety equipment and an associated increase in the consequences of the initiating event. Using probabilistic and deterministic methods, the NRC staff evaluated these issues as they related to Susquehanna and concluded that public health and safety were protected adequately on the basis of the existing design features and operating practices at Susquehanna. The NRC staff also determined that the postulated sequence of events was not within the scope of events that the facility was required to be designed to withstand. However, the NRC staff conducted a broader evaluation of the potential for this type of event to occur at other facilities. Details of the NRC staff's evaluations for all issues can be found in a memorandum dated September 30, 1997 (ADAMS Accession No. ML003706412).

Responses to Requests for Additional Information

Based on the above discussion, the NRC staff requested additional information about the reliability of the SFP forced cooling function and the safety significance of its loss at BWR

facilities with shared secondary containments, where safety analysis report information showed limited backup capability for forced cooling of the SFP. By letter dated November 25, 2013 (ADAMS Accession No. ML13269A287), the NRC requested Exelon Generation Company (Exelon) to address the reliability of SFP cooling at Dresden Nuclear Power Station (DNPS) and at the Quad Cities Nuclear Power Station (QCNPS). Exelon responded by letter dated March 5, 2014 (ADAMS Accession No. ML14064A526). Subsequently, the NRC staff also requested information from the Tennessee Valley Authority (TVA) regarding the reliability of SFP forced cooling for the Browns Ferry Nuclear Plant, Units 1, 2, and 3, by letter dated March 24, 2014 (ADAMS Accession No. ML14055A295). By letter dated June 30, 2014 (ADAMS Accession No. ML14182A695), TVA requested additional time to evaluate the NRC staff's information request. TVA provided its complete response by letter dated September 3, 2014 (ADAMS Accession No. ML14248A681), and TVA revised a portion of that response by letter dated December 31, 2014 (ADAMS Accession No. ML14365A183).

Exelon described that the design basis for DNPS and QCNPS includes only structures and equipment whose failure could directly result in a significant release of radioactivity within design Class I, which applies to structures and equipment designed to function during and following the design basis seismic event. Each DNPS and QCNPS unit has a dedicated SFP cooled by a normal SFP cooling system, which consists of two circulating pumps, two heat exchangers, and additional equipment. These heat exchangers reject heat to the reactor building closed cooling water (RBCCW) system. Exelon described that the SFP structures at both DNPS and QCNPS are Class I structures, but the normal SFP cooling and RBCCW systems are not Class I systems. However, both the SFP cooling and RBCCW system pumps are powered from Class I electrical supplies. In addition, the shutdown cooling system could be aligned to provide supplemental SFP cooling at DNPS, and the Class I RHR system could be aligned to provide SFP cooling at QCNPS. As noted previously, the NRC staff has evaluated

the limited scope of seismic qualification at older licensed facilities, such as DNPS and QCNPS, and determined that the existing classification provided an acceptable assurance of safety. Although the normal SFP cooling systems are not designated as Class I systems at DNPS and QCNPS, Exelon stated that the facilities have sufficient diversity of equipment powered from Class I electrical buses, with emergency diesel generator backup, that a sustained loss of SFP forced cooling would not be expected. The NRC staff agreed.

TVA provided a similar description of the capability to provide SFP forced cooling at the Browns Ferry Nuclear Plant (BFNP). The three reactors at BFNP each have dedicated SFPs within a shared secondary containment structure. Each SFP is normally cooled by the design Class II fuel pool cooling system. Structures, systems, and components designated as design Class II have not been qualified to remain functional during and following a design basis earthquake. However, the portion of the fuel pool cooling system used for SFP makeup from the Class I RHR system has been qualified to design Class I. Other portions of the fuel pool cooling system, used in combination with the RHR system for the SFP cooling assist mode, have been qualified as seismic design Class II to retain pressure boundary integrity following a design basis earthquake. TVA stated that this other section of piping, which supplies water from the SFP to the suction of the RHR system pumps, has been evaluated to seismic Class I design requirements for the Unit 2 SFP, and the configuration of this piping section for the Unit 1 and Unit 3 SFPs is similar. Therefore, there is reasonable assurance that this piping would remain functional following a design basis earthquake, and, therefore, the fuel pool cooling assist mode of the RHR system would function to cool the BFNP SFPs. However, TVA described technical and administrative limitations on the availability of the SFP cooling assist mode of the RHR system, when the associated reactor is in the cold shutdown or refueling operating modes, and when the reactor vessel is hydraulically disconnected from the SFP. Overall, TVA concluded

that there is reasonable assurance that forced cooling of the SFPs would be maintained. The NRC staff agreed.

TVA also described the expected conditions that would develop within the secondary containment envelope if forced cooling of the SFP was not maintained. TVA determined that the minimum time for the SFP to reach saturation conditions would exceed 12 hours, and that the minimum time to boil would occur with the SFP isolated from the refueling cavity at the conclusion of refueling (i.e., the water volume was small and the decay heat was high). Similar to the NRC staff expectations described above, TVA concluded that the environment on the refueling floor would be most affected by the sustained loss of forced cooling. TVA determined that the essential equipment required to respond to a design basis earthquake with an associated loss of offsite power would not be challenged by the temperature or humidity conditions created by a sustained loss of SFP cooling, in part, because the equipment is qualified for higher temperatures and relative humidity associated with high-energy line breaks. Assuming all condensate was directed to a single equipment zone and the minimum elevation for which pump operation is assured, approximately 36 hours after the loss of SFP forced cooling would be available before the condensate accumulation would affect the essential equipment. For more realistic distribution of condensate and condensate accumulation limits, TVA estimated that several days would be available to mitigate the condensate accumulation. In addition, TVA determined that the standby gas treatment system could be aligned to draw on the refueling zone and that the system function would not be challenged because it is designed for high-energy line break conditions, that bound the conditions, resulting from a sustained loss of SFP forced cooling. Accordingly, TVA concluded that there is a reasonable expectation that all equipment relied upon for safe shutdown would remain operable following a design basis earthquake with a consequential loss of offsite power. The NRC staff agreed.

Additional Measures to Improve Spent Fuel Pool Safety

The NRC staff has imposed additional requirements to maintain the safety of the reactor and stored spent fuel since the completion of the Spent Fuel Storage Action Plan. In accordance with 10 CFR 50.54 (hh)(2), licensees have been required to develop and implement guidance and strategies intended to maintain or restore core cooling, containment, and SFP cooling capabilities under the circumstances associated with the loss of large areas of the plant caused by explosions or fire. This strategy for SFP cooling typically involves the capability to replace water lost from the SFP or the capability to spray water into the SFP, not the ability to provide forced cooling of the SFP water.

Following the earthquake and tsunami at the Fukushima Dai-ichi nuclear power plant in March 2011, the NRC established a senior-level task force—referred to as the Near-Term Task Force (NTTF). The NTTF conducted a systematic and methodical review of the NRC regulations and processes to determine whether the agency should make safety improvements, in light of the events in Japan. As a result of this review, the NTTF issued SECY-11-0093, "Near-Term Report and Recommendations for Agency Actions following the Events in Japan" (ADAMS Accession No. ML11186A950). Subsequently, SECY 11-0124, "Recommended Actions To Be Taken Without Delay from the Near-Term Task Force Report," (ADAMS Accession No. ML112911571), and SECY-11-0137, "Prioritization of Recommended Actions To Be Taken in Response to Fukushima Lessons Learned," (ADAMS Accession No. ML11272A111) were issued to establish the NRC staff's prioritization of the recommendations. Recommendation 7.1, concerning reliable SFP instrumentation, was determined to be a high-priority action. The NRC staff issued Order EA-12-051, dated March 12, 2012, to address SFP instrumentation (ADAMS Accession No. ML12054A679). Order EA-12-049, issued on March 12, 2012 (ADAMS Accession No. ML12056A045), imposed additional requirements to ensure strategies to maintain or restore core cooling, containment, and SFP cooling are

available for a range of external initiating events. These requirements enhance the capability of plant operators to maintain plant safety under conditions beyond the design basis of plant operation.

The NRC staff issued guidance for developing, implementing, and maintaining the requirements of Order EA 12-049 in JLD-ISG-2012-01, "Compliance with Order EA-12-049, Order Modifying Licenses with Regard to Requirements for Mitigation Strategies for Beyond-Design-Basis External Events," Revision 0 (ADAMS Accession No. ML12229A174). In the attachment to JLD-ISG-2012-01, the NRC staff endorsed NEI-12-06, "Diverse and Flexible Coping Strategies (FLEX) Implementation Guide," Revision 0 (ADAMS Accession No. ML12242A378), as an acceptable method to develop strategies and guidance for SFP cooling. This document addresses the potential for environmental effects resulting from loss of SFP forced cooling. Specifically, Table C-3, "Summary of Performance Attributes for BWR SFP Cooling Function," and Table D-3, "Summary of Performance Attributes for PWR [Pressurized Water Reactor] SFP Cooling Functions," in Appendices C and D, respectively, identify the ability to vent steam and condensate from the SFP as a baseline capability of the SFP cooling strategy. The identified purpose of this capability is to avoid access and equipment problems that could result from accumulation and subsequent condensation of steam from a boiling SFP.

III. Conclusion

The NRC denies the petition because the NRC staff has reasonable assurance that the design and operation of SFP cooling systems for BWRs with Mark I and Mark II containment designs satisfy the current design and licensing basis. The NRC staff has found that existing SSCs related to the storage of irradiated fuel provide adequate protection for public health and safety. Information supporting this conclusion is readily available in the licensees' SAR, supporting safe operation of each facility. Additionally, the requirements imposed by the NRC

on licensees, per NRC Orders EA-12-049 and EA-12-051, provide enhanced mitigation capabilities to ensure that core cooling, containment, and SFP cooling can be maintained or restored following beyond-design-basis external events. Guidance for implementation of these capabilities includes measures to manage the environmental effects created by a SFP at saturation conditions. While these NRC Orders required licensees to have enhanced mitigation strategies for beyond-design-basis external events, it is reasonable to assume that these mitigation capabilities would also be available for plant operators to use following design basis events.

In compliance with the GDC, the spent fuel pool cooling systems provide decay heat removal (DHR) capability with reliability, consistent with the importance to safety of SFP DHR. This capability is consistent with the criteria included in GDC 61, which specifically apply to fuel storage. Although the capability for continuous DHR from the SFP provided by a system satisfying the criteria of GDC 44 is desirable, this capability is unnecessary because of the very large heat capacity of the pool and the low-decay heat rate of the stored fuel. Systems to which the criteria of GDC 44 specifically apply (e.g., systems removing heat from emergency diesel generators and post-accident containment heat removal systems) require the essentially continuous heat removal capability specified by GDC 44, because of a much lower heat capacity, relative to the design-basis heat generation rate. Thus, the subject facilities comply with an appropriate design criterion for SFP forced cooling systems.

The NRC staff determined that the environmental qualification (EQ) requirements of 10 CFR 50.49 would not apply to a sustained pool boiling event because that event is outside the spectrum of events considered within the SAR. The final SARs of some of the facilities within the scope of this petition discuss the potential for the pool to reach saturation, but the effects of continued boiling have not been evaluated. Other facilities have the capability to maintain SFP forced cooling with no more than short interruptions following design-basis

events. The NRC staff found that redundant pumps in the SFP forced cooling systems at all facilities would support early restoration of cooling and, regardless of cooling restoration, pool boiling was unlikely to create a harsh environment in the vicinity of electric equipment within the scope of the regulation because of the low-decay heat rate and the configuration of the reactor building.

The NRC staff concludes that the safety significance of the design differences would not warrant enhancements to either the design of the SFP forced cooling systems or the EQ of equipment in the reactor building. The conditions in which a sustained loss of SFP cooling and subsequent pool boiling could substantially affect the systems necessary to shut down the reactor safely and maintain safe shutdown conditions are limited to those sites with a shared structure enclosing more than one operating reactors and associated SFPs, when all fuel assemblies have been transferred from one RV to the SFP. At these sites, the NRC staff had previously determined that the probability of a sustained loss of cooling was sufficiently small and that no additional forced cooling capability was necessary. Under other conditions, the NRC staff expects that the rate of vapor generation resulting from decay heat from the fuel stored in the pool would be too low to substantially affect the systems necessary to maintain the safe shutdown conditions.

As provided in 10 CFR 2.206(c), a copy of this DD will be filed with the Secretary of the Commission for the Commission to review. As provided for by this regulation, the decision will constitute the final action of the Commission 25 days after the date of the decision unless the Commission, on its own motion, institutes a review of the decision within that time.

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

FOR THE NUCLEAR REGULATORY COMMISSION

William M. Dean, Director
Office of Nuclear Reactor Regulation

Letter to Those on the Enclosed List dated April 17, 2015

SUBJECT: DRAFT DIRECTOR'S DECISION - PETITION OF JULY 29, 2011

DISTRIBUTION: G20110563
PUBLIC
LPL1-2 R/F
RidsNrrDorl Resource
RidsNrrDorlLpl1-2 Resource
RidsNrrDpr Resource
RidsNrrPMBrownsFerry Resource
RidsNrrPMBrunswick Resource
RidsNrrPMColumbia Resource
RidsNrrPMCooper Resource
RidsNrrPMDresden Resource
RidsNrrPMDuane Arnold Resource
RidsNrrPMFermi Resource
RidsNrrPMFitzpatrick Resource
RidsNrrPMHatch Resource
RidsNrrPMHope Creek Resource
RidsNrrPMLimerick Resource
RidsNrrPMLaSalle Resource
RidsNrrPMMonticello Resource
RidsNrrPMNineMilePoint Resource
RidsNrrPMOysterCreek Resource
RidsNrrPMPeach Bottom Resource
RidsNrrPMPilgrim Resource
RidsNrrPMQuadCities Resource
RidsNrrPMVermontYankee Resource
RidsNrrPMSusquehanna Resource
RidsNrrLAABaxter Resource
RidsNrrMailCenter Resource
RidsOgcMailCenter Resource
RidsEdoMailCenter Resource
RidsOpaMail Resource
RidsRgn1MailCenter Resource
RidsRgn2MailCenter Resource
RidsRgn3MailCenter Resource
RidsRgn4MailCenter Resource
RidsNrrOd Resource
RidsOcaMailCenter Resource

To Those on the Enclosed List

Dear Sir or Madam:

By letter dated July 29, 2011, the Union of Concerned Scientists submitted a petition under Section 2.206, "Requests for Action under this subpart," of Title 10 of the *Code of Federal Regulations* to the U.S. Nuclear Regulatory Commission (NRC). The petition requested that the NRC issue a demand for information to a number of boiling-water reactor licensees with Mark I and Mark II containment designs. The petition has been reviewed by the NRC staff and the staff's proposed director's decision on the petition is enclosed. I request that you provide comments to me on any portions of the decision that you believe involve errors or any issues in the petition that you believe have not been fully addressed. The staff is making a similar request of the petitioner. The staff will then review any comments provided by you and the petitioner and consider them in the final version of the director's decision with no further opportunity to comment.

Please provide your comments within 30 days from the date of this letter. Please feel free to contact Mr. John G. Lamb, Petition Manager, at 301-415-3100, to discuss any questions related to this petition.

Sincerely,
/RA/
Louise Lund, Acting Director
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket Nos. 50-259, 50-260, 50-296,
50-325, 50-324, 50-298, 50-237, 50-249,
50-331, 50-321, 50-366, 50-341, 50-354,
50-333, 50-263, 50-220, 50-410, 50-219,
50-277, 50-278, 50-293, 50-254, 50-265,
50-397, 50-373, 50-374, 50-352, 50-353,
50-387, 50-388, and 50-271

Enclosures:
1. List of Licensee's with Mark I & II Containment Design
2. Proposed Director's Decision

cc: Listserv

Distribution: See next page
ADAMS ACCESSION NOS: Package: ML12215A276; Incoming: ML11213A030; Letter: ML12215A283; Proposed DD: ML12215A314 Licensee Letter: ML15069A112 * via email

OFFICE	DORL/LPLI-2/PM	DORL/LPLI-2/LA	DPR*	DSS/SPBP/BC	DORL/LPL4-2/BC*	JLD/TSD/JERB/BC*
NAME	JLamb	ABaxter	MBanic	GCasto	MKhanna	BPham
DATE	02/27/15	03/11/15	02/27/15	03/04/15	03/01/15	03/10/15
OFFICE	Tech Editor*	DE/D*	OGC	DORL/D	NRR/D	DORL/D
NAME	CHsu	JLubinski	CHair	MEvans	WDean	LLund
DATE	03/09/15	03/09/15	03/11/15	3 /16 /15	4/16/15	4/17/15

OFFICIAL RECORD COPY