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DD-15-11

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	)	

DIRECTOR'S DECISION UNDER 10 CFR 2.206

I. Introduction

By letter dated July 29, 2011 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML11213A030), 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), Section 2.206, "Request for action under this subpart." The Petitioner requested that the U.S. Nuclear Regulatory Commission (NRC or the Commission) 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, Part 50, 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.
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> [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:
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> (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.

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 the standard design criteria that are 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 almost never 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 that 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 SARs for nuclear power plants provides guidance for NRC staff review of the complete SAR included in license applications, as well as changes to the SAR 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 NRC 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 NRC 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 and provided reasonable assurance of adequate protection of public health and safety. Therefore, 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 the 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 decay heat removal 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 decay heat removal as supplemental SFP cooling at some facilities requires installation of short piping segments.
- The flow path connecting the decay heat removal to the SFP at some facilities does not have the same level of qualification as the residual heat removal system itself.

Nevertheless, these conditions allow use of the supplemental cooling capability for its design function, which is to provide additional cooling during infrequent refueling operations that involve transfer of all fuel assemblies from the reactor vessel to the spent fuel pool. For some facilities, the supplementary cooling capability supports continued cooling of the SFP under conditions where the normal SFP cooling system may not be available. However, for all facilities, the substantial heat capacity of the SFP coolant inventory and the availability of multiple pumps to provide forced cooling of the SFP support restoration of cooling before significant environmental effects would develop.

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

As discussed below, the NRC staff 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 staff accepted the SFP cooling system designs at all subject facilities because, in part, these designs

provide a reliable decay heat removal capability that reflects the importance to safety of decay and residual heat removal, 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 the PDC that are 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, “Definitions,” “Design bases” refers to information that identifies the specific functions to be performed by an 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 (LOCAs) 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 (EQ). These lists and supporting documents 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 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 creates a harsh environment for the purposes of EQ 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. Further, 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 have 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 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 decay heat removal 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 the 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 and the fuel is submerged. 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 degrees 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 cooler surfaces in the building.

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 leaving 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 the 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 safety-related standby gas treatment systems in BWRs with Mark I or Mark II containments would generally 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 the 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. Systems within the reactor building are important to reactor safety when the systems perform essential safe shutdown or accident mitigation functions, such as reactor decay heat removal, when fuel is located within the reactor vessel. This secondary containment configuration exists at the following BWRs with Mark I or Mark II containments: Browns Ferry Nuclear Power Plant (Browns Ferry), Units 1, 2, and 3; Dresden Nuclear Power Station (DNPS), Units 2 and 3; Edwin I. Hatch Nuclear Plant (Hatch), Unit 1 (the Hatch, Unit 2 reactor is in a secondary containment zone, separate from the zone containing the SFPs); LaSalle County Station (LaSalle), Units 1 and 2; Quad Cities Nuclear Power Station (QCNPS), Units 1 and 2; and Susquehanna Steam Electric Station (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 SFPs 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. To gain a better understanding of the plant-specific licensing basis at the remaining three facilities, the staff requested additional information.

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 (SAR) 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 DNPS and at the 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 at Browns Ferry, 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 bases for DNPS and QCNPS include 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 DNPS shutdown cooling system, which is powered from Class 1 electrical supplies, 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 a limited scope of the seismic qualifications 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. Based on its evaluation, the NRC staff agrees with the licensee's position.

TVA provided a similar description of the capability to provide SFP forced cooling at Browns Ferry. The three reactors at Browns Ferry each have dedicated SFPs, within a shared secondary containment structure. Each SFP is normally cooled by the design Class II fuel pool cooling system. SSCs 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 Browns Ferry SFPs. However, TVA described technical and administrative limitations on the availability of the SFP cooling assist mode of the RHR system. When the reactor is in the cold shutdown operating mode or in the refueling operating mode with the reactor vessel hydraulically disconnected from the SFP, the facility technical specifications require the RHR system to be in the shutdown cooling mode of operation. In this mode of RHR system operation, shared suction piping precludes simultaneous cooling of the associated SFP. Therefore, the RHR system would be available to provide forced SFP cooling under refueling conditions with the highest decay heat load (i.e., during transfer of irradiated fuel between the reactor and the SFP and when all irradiated fuel has been transferred from the reactor vessel to the SFP), but it may not be available when decay heat loads in the SFP are lower. Similar to the single unit BWRs without a shared secondary containment, the large coolant inventory and low decay heat load would allow substantial time for recovery of the normal SFP forced cooling system when the RHR SFP cooling assist mode was unavailable.

TVA also described the expected conditions that would develop within the secondary containment envelope if forced cooling of the SFP was not restored. 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 higher than other times with the reactor operating). 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. In order to evaluate protection against flooding, TVA evaluated the expected elevation of accumulated condensate assuming all condensate was directed to a single equipment zone and no water was removed. With these conservative assumptions, TVA determined that the accumulated condensate would not threaten operation of safety-related equipment in that equipment zone for at least 36 hours following the loss of forced SFP cooling. Using a more realistic distribution of condensate, 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 sustained loss of forced SFP cooling. Based on its evaluation, the NRC staff agrees with the licensee's position.

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.

After 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. Because 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. NTTF Recommendation 7.1 concerning reliable SFP instrumentation was determined to be a high-priority action. The NRC staff issued Order Enforcement Action (EA)-12-051, "Order Modifying Licenses with Regard to Reliable Spent Fuel Pool Instrumentation (Effective Immediately)," dated March 12, 2012, to address SFP instrumentation (ADAMS Accession No. ML12054A679). Order EA-12-049, "Issuance of Order to Modify Licenses with Regard to Requirements for Mitigation Strategies for Beyond-Design Basis External Events," issued on March 12, 2012 (ADAMS Accession No. ML12056A045), imposed additional requirements to ensure that strategies to maintain or restore core cooling, containment, and SFP cooling are available for a range of external initiating events. While compliance with existing regulations and guidance presumptively provides reasonable assurance of safe storage of spent fuel, the NRC's assessment of new insights from the events of Fukushima Dai-ichi leads the staff to conclude that additional requirements must be imposed on licensees to increase the capability of nuclear power plants to mitigate beyond-design basis external natural events. These new requirements enhance the capability of plant operators to maintain plant safety, consistent with the overall defense-in-depth philosophy, and therefore provide greater assurance that

challenges posed by beyond the design-basis external events do not pose an undue risk to public health and safety.

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 Nuclear Energy Institute 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. Plant specific information supporting this conclusion is readily available in the licensees’ SARs. 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 after 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 SFP cooling systems provide reliable decay heat removal (DHR) capability, 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 would result in a more robust system, 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 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 after 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 changes proposed by the petitioner would not warrant enhancements to either the design of the SFP forced cooling systems or the qualification of equipment in the reactor buildings of BWR Mark I and Mark II containments. 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 reactor and associated SFPs, when all fuel assemblies have been transferred from one reactor vessel to the SFP. At these sites, the NRC staff 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 2nd day of November 2015.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA/

William M. Dean, Director,
Office of Nuclear Reactor Regulation.

COMMENTS RECEIVED FROM THE PETITIONER

ON THE PROPOSED DIRECTOR'S DECISION

DATED APRIL 17, 2015

By letter dated April 17, 2015 (Agencywide Documents and Access and Management System (ADAMS) Package Accession No. ML12215A276), the U.S. Nuclear Regulatory Commission (NRC) staff issued the proposed director's decision (DD) for comments to David Lochbaum of the Union of Concerned Scientists (UCS). By letter dated May 8, 2015 (ADAMS Accession No. ML15128A388), Mr. Lochbaum of the UCS provided comments on the proposed DD. It is noted that each of the petitioner's comments are quoted in its entirety below. The NRC's response to the comments received is provided below.

UCS Comment No. 1

The proposed DD states that the NRC considers General Design Criterion (GDC) 61 to be applicable to the spent fuel pool (SFP) heat removal function and that the SFP cooling system need not comply with General Design Criterion 44.

UCS Comment: It is clear and unequivocal that nuclear power reactors licensed by the NRC must comply with all applicable regulatory requirements, including GDC 44 and GDC 61.

UCS agrees with the NRC staff that SFPs and their cooling and makeup systems must comply with GDC 61.

UCS also agrees with the NRC staff that containments and their cooling systems must comply with GDC 44.

Because the SFPs at boiling water reactors with Mark I and II containment designs, the subjects of our petition, are located inside containment, compliance with GDC 44 inherently includes handling the heat released from the SFPs and the cooling/makeup equipment operating to support them. In other words, one cannot pretend that these heat loads do not

exist and still be in compliance with GDC 44. There is an overlap between GDC 61 and GDC 44 forced by the location of the SFPs and support equipment within containment.

Similar overlaps exist between GDC 33, “Reactor Coolant Makeup,” and GDC 44, as well as between GDC 35 and GDC 44; the NRC requires compliance with GDC 33 and GDC 35, “Emergency Core Cooling,” as well as with GDC 44. The NRC must treat GDC 44 and GDC 61 the same way.

GDC 33 requires a system to provide makeup to the reactor vessel in the event of small breaks in the reactor coolant system. The High Pressure Coolant Injection (HPCI) system at most boiling-water reactors (BWRs) fulfills this requirement. Because the HPCI steam turbine, pump, and piping are located within secondary containment, GDC 44 calculations and evaluations must account for the heat loads from HPCI operation during an accident. If the heat loads from HPCI operation during an accident had been excluded from GDC calculations, it would be immaterial with respect to GDC 44 compliance as to whether the HPCI design complied with GDC 33. Both GDC must be met.

GDC 35 requires a system to provide “abundant emergency core cooling” in the event of larger breaks in the reactor coolant system. The Core Spray and Residual Heat Removal Systems at most BWRs fulfill this requirement. Because the pumps, electric motors, and piping for these systems are located within secondary containment, GDC 44 calculations and evaluations must account for heat loads from their operation during an accident. If the heat loads from core spray or residual heat removal (RHR) operation during an accident had been excluded from GDC calculations, it would be immaterial with respect to GDC 44 compliance whether the design complied with GDC 35. Both GDC must be met.

The underlying foundation for our petition is our contention that the heat loads from SFPs and their support equipment have not been accounted for within GDC 44 calculations and

evaluations. Thus, while GDC 61 is being met for the SFPs inside containment, GDC 44 might not be met. Both GDC must be met.

Our petition sought demands for information (DFIs) on how the applicable reactors complied with GDC 44. We assume and accept that the reactors comply with GDC 61. By whatever means are used to comply with GDC 61 (e.g., cooling by safety-related system, cooling by non-safety-related system, non-cooled with makeup to boiling pool, etc.), heat is released into secondary containment. GDC 44 cannot be met unless that heat load, along with the heat loads from all other sources, can be removed to maintain the design-basis conditions within secondary containment.

The heat loads from the SFP that may be unaccounted for are not insignificant. The matter first came to my attention while working on the power uprate project for Susquehanna in the early 1990s. The GDC 44 calculation for post-accident conditions within secondary containment assumed a total heat load of around 4.5 million British Thermal Units per hour (MBTU/hr). This calculation accounted for heat loads from HPCI, core spray, and RHR pump operation, heat losses from piping filled with warm water, and even from the heat emitted by lighting within the reactor building. But, it assumed that the SFP essentially disappeared during an accident. It considered neither the latent heat from the spent fuel nor any heat released from equipment operating inside secondary containment to cool the pool's water or provide makeup to a boiling pool. The design-basis heat load in each SFP at Susquehanna was 12.6 MBTU/hr, while the design-basis emergency heat load from a full core offload was 32.6 MBTU/hr. Thus, the GDC 44 calculation neglected heat loads 2.8 to 7.2 times greater than the total heat load considered. The results from the non-conservative GDC 44 calculation did not show abundant temperature profile margins in most reactor building areas.

The DFIs are still needed, and still requested, to answer the vital questions of how reactors with SFPs inside secondary containment comply with GDC 44.

NRC Response to UCS Comment No. 1

As discussed in the draft DD, GDC 44 applies specifically to cooling water systems. Cooling water systems designed to GDC 44 provide normal and post-accident cooling of the primary containment through various modes, heat removal from emergency systems through component cooling and room cooling, removal of decay and residual heat from the reactor vessel, and, sometimes, the SFP (consistent with the GDC 61 criterion of the "residual heat removal capability having reliability and testability that reflects the importance to safety of decay heat and other residual heat removal"). When the SFP cooling system heat exchangers are cooled by a cooling water system designed to GDC 44, the heat rejection is considered in the design of the cooling water system and the ultimate heat sink. However, some plant configurations have other cooling water systems and heat sinks providing cooling to the SFP.

The SFP is located within the secondary containment envelope, and is outside the primary containment. The primary heat removal path from the secondary containment is through the normal ventilation system and, following accidents and abnormal events, through the standby gas treatment system. This heat removal includes a portion of the decay heat from stored fuel that supports evaporation of water from the pool surface. If the SFP experienced a sustained loss of forced cooling, the decay heat of the fuel would raise the pool temperature until the heat transferred to the secondary containment by evaporation or boiling matched the decay heat. This heat would increase the temperature of the upper secondary containment and increase the heat removed by ventilation system operation and by transfer to the environment through conduction/convection mechanisms. These heat transfer paths are independent of the systems designed and licensed to the criteria of GDC 44.

The facility SARs consider the reliability of forced SFP cooling. For many facilities, these analyses recognize both the limits of the forced cooling systems and the large passive heat sink provided by the pool coolant inventory. Therefore, the staff accepted the existing cooling system designs, although they may lack the capability to provide nearly continuous forced cooling, provided by systems designed to the criteria of GDC 44.

UCS Comment No. 2

The proposed DD states, "The NRC staff has not identified another design-basis event [other than design bases earthquake] that would directly result in a sustained loss of forced [spent fuel pool] cooling."

UCS Comment: The NRC staff failed to identify other design-basis events that result in a sustained loss of forced SFP cooling.

Among the many examples we could cite is an example contained within the licensee event report (LER) dated March 21, 2012 (ADAMS Accession No. ML12083A194) for an oil leak from an emergency diesel generator (EDG) at Browns Ferry Unit 1. The small oil leak was reported because it "did not meet the 7-day mission time of the Unit 1/2 C EDG." Hence, the design basis for this NRC-licensed facility includes offsite power being unavailable for 7 days, a sustained period.

The RHR system at Browns Ferry is designed to provide forced cooling of the SFP (see letter dated December 31, 2014, ADAMS Accession No. ML14365A183) in fuel pool cooling assist mode. The RHR pumps can be powered from the electrical buses connected to the EDGs and thus would be available even when offsite power was not.

But, not all boiling water reactors have the RHR fuel pool cooling assist mode option. Even those reactors equipped with this option may be unable to use it when necessary. The Updated Final Safety Analysis Reports (FSARs) typically do not describe a loss of normal SFP

cooling as a design-basis accident or transient. Consequently, the valves that must be operated to establish the RHR fuel pool cooling assist mode are not guaranteed to be included within the periodic testing and inspection programs that provide reasonable assurance that the safety function can be performed.

If RHR fuel pool cooling assist is being relied upon to mitigate design-basis events, then all structures, systems, and components necessary to establish that alignment and sustain its operation must be included within appropriate testing and inspection programs. The responses to the DFIs sought by the petitioners would have identified what equipment was being relied upon, enabling determinations whether this equipment was adequately covered by testing and inspection programs.

The NRC should issue the DFIs to ensure that whatever means are being used to remove heat from secondary containments are likely to perform that safety function when needed.

NRC Response to UCS Comment No. 2

The NRC staff agrees that a loss of offsite power is a design-basis accident and that the design-basis for many fuel oil storage systems specifies a 7-day capacity. However, because the large passive heat sink provided by the pool coolant prevents substantial environmental effects for a period of several hours or more, SFP cooling systems, including those facilities where the system is non-safety-related, would be placed in operation using the emergency diesel generators as a power source. This electrical load is included among the loads used in determining required diesel fuel inventory that have a 7-day capacity as the licensing basis.

The NRC staff also agrees that testing of the SFP cooling mode of the RHR system is not required because this function is not safety-related. Similar to the electric power supply

issue, reasonable time is available to correct equipment failures that may affect this mode of RHR system operation.

Operating experience related to loss of offsite power events, similar to those considered as design-basis events, have resulted in a loss of one or more offsite power sources for periods as long as several days. However, these events have not resulted in increased pool temperatures that substantially affected the environment of the building housing the pool. This operating experience supports the continued acceptability of non-safety-related SFP cooling systems and operator action to restore cooling systems for the SFP.

UCS Comment No. 3

The proposed DD states, “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.”

UCS Comment: What does the NRC staff mean by “substantial time” – a day, a week, or a month? If it involves less than 30 days, it deviates from longstanding industry and NRC practice.

As stated in the LER dated December 31, 2012 (ADAMS Accession No. ML13002A391), for Pilgrim, “The mission time for the secondary containment system is 30 days.”

As stated in the LER dated September 29, 2004 (ADAMS Accession No. ML042810116), for Pilgrim, “The SGTS [standby gas treatment system] air accumulators (accumulator bank) function [is] to store sufficient pneumatic energy for operation of the SGTS for the 30-day mission time.”

As stated in the LER dated June 17, 2011 (ADAMS Accession No. ML11174A039), for Quad Cities Unit 1, the Residual Heat Removal Service Water System (RHRSW) has a “mission time of 30 days.”

As stated in the LER dated August 15, 2007 (ADAMS Accession No. ML072400342), for Susquehanna, “The design basis mission time for ESW [emergency service water] and RHRSW is defined in the FSAR as 30 days.”

As stated in the LER dated April 18, 2008 (ADAMS Accession No. ML081120106), for Hatch Unit 2, the reported problem “could have prevented the RHRSW system from meeting its 30-day mission time.”

The 30-day mission times for secondary containment, SGTS, RHRSW, and ESW all directly relate to GDC 44. In other words, one cannot comply with GDC for a “substantial time” that is less than 30 days.

The NRC has taken enforcement action in recent years because the 30-day mission time might not have been met:

By inspection report dated February 9, 2011 (ADAMS Accession No. ML110400431), the NRC issued a Green finding for a problem at Browns Ferry for an RHR pump motor problem that “would have prevented the pump from performing its intended safety functions during the system’s required mission time.” According to this NRC letter, “The mission time of the 1C RHR pump to perform its intended safety functions was 30 days.”

The NRC has not allowed BWR owners to only show adequate Net Positive Suction Head for emergency pumps during a “substantial time” of the accident. In Enclosure 1 to SECY-11-0014, dated January 31, 2011 (ADAMS Accession No. ML102110167), the NRC informed its Commissioners that, “The necessary time for a pump using containment accident pressure should include not only the duration of the accident when NPSH margin may be limited, but any additional time needed for operation of the pump … This additional time is usually taken as 30 days.”

The NRC also applied the 30-day mission time when evaluating the potential for post-accident debris to impair emergency pump operation: NUREG/CR-7011, "Evaluation of Treatment of Effects of Debris in Coolant on ECCS and CSS Performance in Pressurized Water Reactors and Boiling Water Reactors," dated May 2010 (ADAMS Accession No. ML100960388) explicitly stated the NRC expectation that licensees would evaluate the postulated design basis event assuming, "All ECCE and CSS pumps are in operation for an extended period (up to the maximum mission time)…"

The NRC rejected an industry comment seeking to neglect potential damage to emergency pumps from cavitation because it would likely occur after a substantial time: The nuclear industry commented on Draft Regulatory Guide DG-1234 that cavitation was a long-term degradation effect that should be excluded from post-accident assessments of emergency core cooling system (ECCS) pump performance. Its evaluation of public comments (ADAMS Accession No. ML111330292) stated, "The NRC staff disagrees with the comment. Cavitation over the post-LOCA mission time could affect pump performance."

Section 6.2.3, Secondary Containment Functional Design, of NUREG-0800, "Standard Review Plan" (ADAMS Accession No. ML063600406) Acceptance Criterion 1.G states, "Heat loads generated within the secondary containment (e.g., equipment heat loads) should be considered." There is no qualifier allowing heat loads to be ignored as long as GDC 61 is met. Hence, heat generated by the irradiated fuel in the SFPs and by equipment operated in support of the SFPs must be considered.

Section 9.2.5, Ultimate Heat Sink, of NUREG-0800, Standard Review Plan (ADAMS Accession No. ML070550048) Acceptance Criterion 3.A indicates that GDC 44 is met if the design provides, "The capability to transfer heat loads from safety-related SSCs to the heat sink under both normal and accident conditions." Secondary containment is a safety-related

structure. The last paragraph on page 9.2.5-5 directs the NRC reviewers to evaluate "the UHS design, including assumptions for heat loads…"

Compliance with GDC 44 requires a showing that heat loads within containment can be adequately removed over the entire 30-day mission time, not for a shorter period whether deemed substantial or not. As Abraham Lincoln might have observed, it is not sufficient to remove all the heat loads for some of the time, or to remove some of the heat loads all the time, but only to remove all the heat loads for the entire mission time.

The DFIs are still needed, and requested, to indicate how licensees are complying with this regulatory requirement.

NRC Response to UCS Comment No. 3

For certain safety analyses, the NRC staff has indicated that a 30-day evaluation period would be acceptable to the staff. In particular, Regulatory Guide 1.27, "Ultimate Heat Sink for Nuclear Power Plants," specifies evaluation of ultimate heat sink capacity for a 30-day evaluation period, unless the capability to replenish water to maintain continuous functionality within a shorter period is justified, considering postulated accident conditions. This evaluation considers the heat rejection from necessary equipment operation, residual heat associated with post-accident cool-down, and reactor decay heat. When the SFP rejects heat to this heat sink, the spent fuel decay heat would also be included in the evaluation. However, the 30-day evaluation period is not specified in any NRC regulation. A 30-day mission time is neither a requirement nor a common assumption for other accident analyses. Therefore, the NRC staff disagrees that a 30-day evaluation of SFP cooling is required.

UCS Comment No. 4

The proposed DD states, “A sustained loss of SFP forced cooling (i.e., the 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.”

UCS Comment: In the second full paragraph on page 9 of the Proposed Director’s Decision, the NRC staff states, “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,” for the EQ requirements of 10 CFR 50.49.

UCS agrees with these NRC statements, in fact, they form the basis for the request in our petition for DFI on how the applicable reactors comply with 10 CFR 50.49 for design-basis events.

As noted above, the mission time for many design basis accidents such as LOCA is 30-days. It is an undeniable fact that heat will be emitted from the SFP inside secondary containment throughout the mission time. Heat will also be released from equipment operating inside secondary containment to provide forced cooling of the SFP water or to provide makeup to the pool.

It is UCS’s contention that the calculations and evaluations that establish the temperature profiles for rooms and areas within secondary containment may not account for the heat released from the SFP and any associated equipment operation. As a result of these omissions, the EQ of electrical equipment within secondary containment may not ensure safety functions performed by this equipment are performed throughout their entire mission times.

Similar omissions of actual heat leads inside secondary containment have compromised EQ compliance in the past. For example, the LER dated June 29, 1998 (ADAMS Legacy Library Accession No. 9807070371) informed the NRC that the post-LOCA temperatures

calculated in the secondary containments for DNPS, Unit 2 and 3, were incorrect. The calculations had established 104 °F as the post-LOCA temperature. But, those calculations had assumed the post-LOCA heat loads were the same as the heat loads during normal operation. The error was in "ignoring the slow build-up of temperatures in the reactor building due to the combined effect of loss of ventilation due to the post-LOCA isolation of the secondary containment and heat load generated in the reactor building due to operating equipment and lighting." When the calculations were redone to correct these omissions, the resulting temperatures in secondary containment ranged from 121 °F to 152 °F. Electrical equipment was replaced at DNPS to restore compliance with 10 CFR 50.49.

The DFIs are still needed, and requested, to show how licensees comply with 10 CFR 50.49. When EQ calculations and evaluations properly include the post-accident heat loads from the SFP and any supporting equipment operation, their responses should be quite simple and straightforward. But, if the SFP and/or supporting equipment heat loads have been ignored in these calculations, the secondary containment temperatures will likely be non-conservative as they had been at DNPS.

NRC Response to UCS Comment No. 4

The NRC staff has found that the heat released from the SFP has been adequately considered in the evaluation of reactor building temperatures. The SFP forced cooling systems are reliable and provide reasonable assurance that forced cooling would be restored before substantial environmental effects could develop. Furthermore, the design-basis events that would initially result in a loss of SFP forced cooling, such as loss of offsite power events, would not challenge the environmental qualification of the equipment relied on to perform essential safety functions. Therefore, electrical equipment within the reactor building would be capable of performing its safety functions throughout the event.

COMMENTS RECEIVED FROM EXELON

ON THE PROPOSED DIRECTOR'S DECISION

DATED APRIL 17, 2015

Exelon provided comments by email dated May 15, 2015 (ADAMS Accession No. ML15138A323).

Exelon Comment No. 1

Section B, "Conformance with the Environmental Qualification Regulation," describes the considered environmental conditions required by 10 CFR 50.49 in two places, but does not appear to include radiation.

NRC Staff Response to Exelon Comment No. 1

Radiation is not a relevant environmental factor for events involving a loss of forced SFP cooling, because the provision of makeup water would maintain acceptable shielding throughout the course of the event.

Exelon Comment No. 2

Under the subsection, "Application of 10 CFR 50.49," the NRC indicates licensees have identified lists of equipment important to safety subject to environmental qualification and states the following: "…These lists identify the equipment, as well as the performance specifications, electrical characteristics, and environmental conditions for which the equipment was qualified…." Exelon believes that these lists alone may not contain all of the stated information; therefore, Exelon recommends that the NRC consider adding the phrase "and supporting documents" after "These lists" in the quoted excerpt above.

NRC Staff Response to Exelon Comment No. 2

The NRC staff agrees with the comment and will make the requested change.

Exelon Comment No. 3

Under the subsection, “Application of 10 CFR 50.49,” the NRC makes the statement, “Under the requirements of 10 CFR 50.49(e), the electric EQ program must include….” Exelon believes that word “electric” in the statement is unnecessary and would recommend that the word be deleted.

NRC Staff Response to Exelon Comment No. 3

The NRC staff agrees with the comment and will make the requested change.

Exelon Comment No. 4

In Section C, Safety Significance of a Sustained Loss of Forced Cooling, in the seventh paragraph under subsection "Effects of a Sustained Loss of Forced Spent Fuel Pool Cooling" (page 13), the NRC states: “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.” Exelon recommends further clarification as to the connection between systems important to safety being in the reactor building and higher water vapor generation.

NRC Staff Response to Exelon Comment No. 4

The NRC staff agrees with the need for further clarification. The staff will add the following sentence: “Systems within the reactor building are important to reactor safety when the systems perform essential safe-shutdown or accident mitigation functions, such as reactor decay heat removal, when fuel is located within the reactor vessel.”

Exelon Comment No. 5

Under the subsection, “Responses to Requests for Additional Information,” the discussion does contain a factual description of the DNPS fuel pool storage structure and cooling system. However, Exelon believes that it may be helpful to indicate that the shutdown cooling pumps are also powered from Class 1 electrical supplies.

NRC Staff Response to Exelon Comment No. 5

The NRC staff agrees with the comment and will add modify the subsection to state, “In addition, the DNPS shutdown cooling system, which is powered from Class I electrical supplies, 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.”