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Tennessee Valley Authority, 1101 Market Street, Chattanooga, Tennessee 37402

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10 CFR 50.4

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

Browns Ferry Nuclear Plant, Units 1, 2, and 3
Renewed Facility Operating License Nos. DPR-33, DPR-52, and DPR-68
NRC Docket Nos. 50-259, 50-260, and 50-296

Subject: **Response to Request for Additional Information Related to Potential Loss of Spent Fuel Pool Cooling**

References:
1. Letter from NRC to TVA, "Browns Ferry Nuclear Plant, Units 1, 2 and 3 - Request for Additional Information Related to Potential Loss of Spent Fuel Pool Cooling (TAC No. ME6761)," dated March 24, 2014
2. Letter from TVA to NRC, "Response to NRC Request for Additional Information Related to Potential Loss of Spent Fuel Cooling (TAC No. ME6761)," dated June 30, 2014

In Reference 1, the Nuclear Regulatory Commission (NRC) staff stated that an initial evaluation of a petition request concerning the reliability of the Spent Fuel Pool (SFP) cooling systems was complete. Using information contained in each subject facility's safety analysis report, the NRC staff concluded that additional information was necessary from certain facilities in order to evaluate the response of the facilities following design-basis events.

The staff selected facilities that shared a common secondary containment surrounding two SFPs for its initial information request. The NRC stated that these facilities were more likely to have a high decay heat load due to refueling in one of the SFPs, during a time when other equipment within the secondary containment may be essential for accident mitigation or safe shutdown of an adjacent operating unit.

The NRC requested that in order to better understand the reliability of the SFP cooling systems, the expected response of the affected facilities to their loss, and the safety significance of the SFP cooling function at the Browns Ferry Nuclear Plant (BFN), that the Tennessee Valley Authority (TVA) respond to the request for additional information provided.

Per Reference 1, TVA agreed to respond to the request for additional information within 90 days of the date of the letter, June 23, 2014. By Reference 2 and per telecom with Mr. John Lamb (NRC), the due date for the response was extended to August 14, 2014. Due to the complexity of the issue, additional time was required to provide a complete response.

In response to this request, TVA is providing the enclosed information.

There is one new regulatory commitment contained in this letter. Should you have any questions concerning this submittal, please contact Edward D. Schrull at (423) 751-3850.

Respectfully,

J. W. Shea
Vice President, Nuclear Licensing

Enclosure:
1. Response to Request for Additional Information Related to Potential Loss of Spent Fuel Pool Cooling
2. List of Commitments

cc (w/ Enclosure):

NRC Regional Administrator - Region II
NRC Senior Resident Inspector - Browns Ferry Nuclear Plant
NRC Project Manager - Browns Ferry Nuclear Plant
State Health Officer - Alabama Department of Public Health

**Response to Browns Ferry Nuclear Plant, Units 1, 2 and 3 -
Request for Additional Information Related to Potential Loss of Spent Fuel Pool Cooling
(TAC No. ME6761)**

Please respond to the following RAIs for the requested facility. The performance of structures, systems, and components should consider standard accident analysis methods and assumptions used in the safety analysis report, including loss of function under conditions beyond those considered in the design of the structure, system, or component (SSC) and consideration of additional single failures. Operator actions may be included when the action is specified in existing operating, alarm response, or emergency procedure and the personnel expected to execute the action have been properly trained.

NRC RAI 1

Describe the ability to maintain forced cooling of the spent fuel pool (SFP) using installed equipment following a design-basis earthquake with consequential loss of offsite power. Please consider SFP configurations encountered during routine refueling and normal operating conditions. The normal SFP cooling system and the SFP cooling assist mode of the residual heat removal system should be considered at a minimum, and, if a sustained loss of forced SFP cooling is expected, identify the expected range of times for the pool to reach saturation conditions.

Tennessee Valley Authority (TVA) Response

Introduction

At the Browns Ferry Nuclear Plant (BFN), the structure referred to as the Spent Fuel Pool (SFP) is called the Fuel Storage Pool (FSP). Also, the system referred to as the Spent Fuel Pool Cooling (SFPC) system at BFN is called the Fuel Pool Cooling (FPC) system. These terms were used interchangeably in past TVA correspondence with the NRC designating the same structure or system.

Under standard accident analysis methods and assumptions used in the Final Safety Analysis Report (FSAR), structures, systems, and components (SSCs) designed to Seismic Class I would perform their safety function following a Design Basis Earthquake (DBE). Seismic Class I includes those SSCs whose failure or malfunction might cause, or increase the severity of an accident which could endanger the public health and safety. This category includes those SSCs required for safe shutdown and isolation of the reactor.

SSCs designed to Seismic Class II are important to reactor operation, but are not essential for preventing an accident that could endanger the public health and safety, and are not essential for the mitigation of the consequences of such accidents. BFN Seismic Class II SSCs qualified for pressure boundary integrity are qualified to maintain pressure boundary integrity before, during, and after a seismic event at the site.

Ability to Provide Forced Cooling to the FSP

During normal operation, forced cooling to the FSP is provided by the FPC system that circulates FSP water through heat exchangers cooled by the Reactor Building Closed Cooling Water (RBCCW) system. The Auxiliary Decay Heat Removal (ADHR) system is also capable of providing forced cooling to the FSP. The ADHR system primary loop contains heat exchangers that are cooled by the ADHR secondary loop cooling towers. In addition, the Residual Heat Removal (RHR) system can be lined up to provide forced cooling to the FSP in the assist mode of operation by circulating FSP water through the RHR heat exchangers cooled by the Residual Heat Removal Service Water (RHRSW) system.

The entire FPC and ADHR systems have not been qualified to remain functional following a DBE, i.e., they are not Seismic Class I. Therefore, under standard accident analysis methods and assumptions used in the FSAR, these systems are not credited for forced cooling of the FSP.

The RHR/RHRSW systems and the portion of the FPC system that provides make-up water to the FSP are qualified to Seismic Class I requirements. Under standard accident analysis methods and assumptions used in the FSAR, the RHR/RHRSW systems can be credited for providing raw make-up water to the FSP by utilizing piping from the RHR system and FPC system. Also, the RHR/RHRSW systems are powered from the emergency diesel generators (EDGs) so they would remain functional when offsite power is not available, i.e., following a loss of offsite power (LOOP).

The suction flow path from the FSP to the RHR pumps includes a portion of the FPC system that is Seismic Class II, which is qualified for pressure boundary integrity. As such, this portion of the FPC system can not be credited for providing FSP water to the RHR pumps. Therefore, the RHR system cannot be credited for forced cooling of the FSP under standard accident analysis methods and assumptions used in the FSAR.

The Unit 2 FPC system that forms part of the flow path to the suctions of the RHR pumps has been evaluated to Seismic Class I design requirements but it is not considered Seismic Class I in the FSAR. As such, there is reasonable assurance that this piping would remain functional following a DBE. The portions of the Unit 1 FPC system and the Unit 3 FPC system that forms part of the flow path to the suctions of the RHR pumps is similar in configuration to the Unit 2 piping configurations. This similarity provides reasonable assurance that the Unit 1 and Unit 3 piping system would also remain functional following a DBE.

Provided the portion of the FPC system that forms part of the flow path to the suctions of the RHR pumps remains usable following a DBE, site procedures provide for the alignment of the RHR system to the FSP cooling assist mode of operation. In addition, the physical condition of the reactor building and contained SSCs following a DBE would allow operators to access the manually operated equipment necessary to put RHR into the FSP cooling mode of operation.

As described in the FSAR, there are two RHR Loops per Unit. Each loop has two Pumps and two Heat Exchangers. Each Unit has the ability to share one RHR Loop with the adjacent unit(s). Each RHR heat exchanger is cooled by the RHRSW system. The RHRSW system has eight shared pumps. There are eight EDGs that can power all the components in these systems needed for the required RHR pumps and heat exchangers to be functional. Under standard accident analysis methods and assumptions used in the FSAR, one RHR Loop is required for safe shutdown when each reactor has been in power operation. This allows for alignment of a separate RHR Loop to each Unit's associated FSP after the DBE. Due to the high interconnectivity of the RHR/RHRSW and onsite power systems, it is likely that for any station configuration and single failure, at least one RHR Loop would be available to maintain each reactor in safe shutdown and at least one RHR Loop would be available for forced cooling of each Unit's FSP.

Minimum Time to FSP Boiling

The above discussion provides two scenarios for the plant response to a DBE with consequential LOOP: 1. Under standard accident analysis methods and assumptions used in the FSAR, forced cooling of the FSP cannot be assured, however, make-up water would be provided to the FSP using Seismic Category I equipment; and 2. Because the FPC system that forms part of the flow path to the suctions of the RHR pumps has been designed and built to Seismic Class I requirements, forced cooling of the FSP can be realistically assumed. To the extent that forced cooling of the FSP cannot be assured under the strict FSAR assumptions, TVA has evaluated the time for the water in the FSP to boil following a DBE with consequential LOOP.

The minimum time to FSP boiling would occur just after the reactor has been refueled and the gates to the FSP have been closed. In this plant configuration and at this point in a refueling outage, there would be the least volume of water available in the FSP to absorb the heat released from the stored fuel and this would be when the FSP has the highest heat load due to the recently irradiated and discharged fuel. In this plant configuration and at this point in a refueling outage, the minimum time to FSP boiling would be more than twelve hours. However, twelve hours would be sufficient time to identify a loss of FSP cooling and align the RHR/RHRSW systems to cool the FSP. All other plant configurations before, during, or after a refueling outage would have more than this minimum time to FSP boiling. In the FSPs that have not just been loaded with recently irradiated fuel, the time to FSP boiling would be about 40 hours.

In summary, if the reactor were to experience a DBE with consequential LOOP and under standard accident analysis methods and assumptions used in the FSAR:

- Forced cooling of the FSP would not be available;

- For all plant configurations and times prior to, during, or after a refueling outage, the minimum time for the FSP to start to boil after a loss of forced FSP cooling would be twelve hours; and

- Make-up water to the FSP would be available to keep the fuel covered.

However, because of the design and construction of the return FPC piping, the number of RHR pump/heat exchanger pairs available to cool all the FSPs and the time available before FSP boiling would occur, there is reasonable assurance that forced cooling of the FSP would be maintained following a DBE with consequential LOOP.

NRC RAI 2

If the response to the above request determines that the SFP would experience a sustained loss of SFP forced cooling, describe the expected changes in environmental conditions within each affected secondary containment ventilation zone. Address the expected response of operators to manage environmental conditions, consistent with existing procedures, and describe the survivability of ventilation systems, such as the standby gas treatment system. Identify any secondary containment areas that could experience a harsh environment (i.e., an environment significantly more severe than the environment that would occur during normal plant operation with respect to radiation, temperature, humidity, or submergence of equipment as a result of accumulated condensate) as a result of the sustained loss of SFP forced cooling.

TVA Response

As discussed in the response to RAI 1, TVA has determined that there is reasonable assurance that forced cooling of the FSP would be maintained. However, under standard accident analysis methods and assumptions used in the FSAR, forced cooling of the FSP cannot be assumed following a DBE with consequential LOOP. With that assumption, the water in the FSP would start to boil no sooner than twelve hours after the event. The RHR/RHRSW systems would provide make-up water to the FSP so that the fuel would remain covered.

TVA performed a qualitative evaluation of the effect of boiling of the FSP water with the following results. Steam from the boiling FSP water would rise to the ceiling of the refueling zone where it would cool and condense. The surface area of the refueling zone walls and roof are so large that the steam generated by the boiling FSPs would be condensed in the refueling zone with the heat of condensation being transferred to the outside air via heat transfer through the walls and roof. The refueling zone would become a harsh environment (as defined in NRC RAI 2) with temperatures in some locations being equal to that of the boiling FSP water. In addition, the air in the refueling zone would be saturated with steam; condensate from the condensing steam would rain down or flow down the walls and accumulate on the refueling zone floor. The FSP water contains a low concentration of radionuclides, so dose rates in the refuel zone would not be significantly above those experienced during normal operation.

There are open equipment hatches and stairways from the refuel zone to the lower elevations of the reactor building. However, steam and hot air would not be expected to enter the lower elevations of the reactor building through these openings on the refueling floor. Rather, the steam would be condensed in the refueling zone and the air/steam in the refueling zone would be hot and rise rather than flow downward through these openings.

Hot condensate that would rain or spill through the equipment hatches and stairways would pool in the lowest elevation of the reactor building. The hot condensate would result in the air in the reactor building being hotter than during normal operation and saturated with moisture from the evaporating condensate.

Before the FSP starts to boil, i.e., a minimum of 12 hours after the DBE, operators would be able to access the portions of the refueling zone not in the vicinity of the hot FSP. Temperatures in the remainder of the reactor building would remain low enough before the FSP starts to boil to allow operator access. After hot condensate starts to flow into the reactor building, locations in the vicinity of the condensate may be too hot for operators to access. Under standard accident analysis methods and assumptions used in the FSAR, there are no actions that operators could take to manage the environment in the reactor building (e.g., reduce air temperature and humidity using safety-related SSCs) or remove the condensate that would accumulate in the lowest elevation of the reactor building.

Following a DBE and consequential LOOP, normal containment ventilation would be lost. The Primary Containment Isolation System provides a start signal to the Standby Gas Treatment (SBGT) system upon loss of power, high radiation, or Low Reactor Water Level. The SBGT system is Seismic Class I and would start on auxiliary power to maintain secondary containment vacuum and gaseous release control. This system takes suction from the southwest corner of the refueling zone and would take in steam and hot air if fuel pool boil-off did occur. The SBGT system is designed for LOCA conditions, which would bound the temperature and humidity experienced in the refueling zone for this event.

NRC RAI 3

If the response to the above request identifies harsh environmental conditions in any area of the facility secondary containment, describe the effect of these environmental conditions on important-to-safety electrical equipment within those areas necessary to maintain the integrity of the reactor coolant pressure boundary, the capability to shut down the reactor and maintain it in a safe shutdown condition or the capability to prevent or mitigate the consequences of accidents that could result in potential offsite exposure.

TVA Response

As discussed in the response to RAI 1, TVA has determined that there is reasonable assurance that forced cooling of the FSP would be maintained. However, under standard accident analysis methods and assumptions used in the FSAR, following a DBE with consequential LOOP, the water in the FSP would start to boil no sooner than twelve hours after the event. Consequently, the response to RAI 2 postulated that harsh environmental conditions could exist in the refuel zone due to air temperature, humidity and condensate, and in the remainder of the reactor building due to air temperature and humidity. The lowest elevation of the reactor building would also experience a harsh environment due to flooding.

Important-to-safety electrical equipment located in the refueling zone is not qualified to survive under standard accident analysis methods and assumptions used in the FSAR. However, no important-to-safety electrical equipment needed to maintain safe shutdown following a DBE and consequential LOOP is located in this area. Secondary containment isolation dampers located on the refuel floor close immediately upon reactor scram and are not required to re-open after environmental conditions degrade. As discussed in the response to RAI 2, no steam or hot air is expected to enter the reactor building below the refuel zone so it is likely that the air temperature in these areas would remain below 135°F. Below this temperature, environmentally induced failures would not occur. While the humidity in the reactor building is significantly above that during normal operation, postulated high energy line breaks (HELBs) in the reactor building produce 90% to 100% relative humidity for which important-to-safety electrical equipment has been environmentally qualified.

The lowest elevation of the reactor building contains the pumps needed for maintaining safe shutdown following a DBE and consequential LOOP. Under standard accident analysis methods and assumptions used in the FSAR, the pumps needed to maintain safe shutdown must be assumed to fail once the water level exceeded the 1.4 foot flood level for which they have been shown to be functional. The estimated time to flood to greater than this level and fail these pumps is greater than 36 hours from the DBE even should all the condensate preferentially pool in the lowest elevation of one reactor zone. Realistically, the condensate flow from the refueling zone would be more evenly distributed among the three reactor zones and the flood level needed to fail the pumps is likely three feet or more. Therefore, several days would be available to mitigate the flooding in this area and prevent loss of the pumps needed for safe shutdown.

NRC RAI 4

If the response to the above request identifies that electrical equipment necessary to shut down the reactor and maintain safe shutdown conditions could be adversely affected by a sustained loss of SFP forced cooling potentially resulting from a design-basis event, describe any corrective actions that will be implemented at the affected facility and the basis for concluding that those actions would acceptably resolve the described condition.

TVA Response

As discussed in the response to RAI 1, TVA has determined that there is reasonable assurance that forced cooling of the FSP would be maintained. However, under standard accident analysis methods and assumptions used in the FSAR, following a DBE with consequential LOOP, the water in the FSP would start to boil no sooner than twelve hours after the event. Consequently, the response to RAI 3 indicates that the pumps needed to maintain safe shutdown conditions could be adversely affected by a sustained loss of FSP forced cooling lasting more than 36 hours. However, the BFN Licensing Basis for a DBE with consequential LOOP is that safe shutdown must be maintained with no time limit. Therefore, the condition defined above represents a non-conformance with the BFN Licensing Basis.

This condition has been entered into TVA's corrective action program. TVA's current plans are to resolve the above non-conformance with its Licensing Basis by ensuring that for a DBE with consequential LOOP, forced cooling of the FSP would be maintained. In particular, the section of the FPC system comprising the suction flow path from the FSP to the RHR pumps would be qualified to remain functional following a DBE, i.e., that section would be qualified to Seismic Class I criteria, and a sufficient number of RHR/RHRSW and EDGs would be maintained functional during all station configurations. TVA will provide a revised response if the actions to resolve this non-conformance with the BFN Licensing Basis change. In the interim, TVA will provide a semiannual status letter to the NRC of the efforts to resolve the non-conformance.

Notwithstanding the requirement to use standard accident analysis methods and assumptions used in the FSAR, there currently exists a reasonable expectation that all equipment relied upon for safe shutdown would remain operable following a DBE and consequential LOOP. Because there is a reasonable expectation that forced FSP cooling would be available, a high level of plant safety is maintained and no report is required by 10 CFR 50.73 (i.e., this is an unanalyzed condition, but a serious degradation in plant safety does not exist).

ENCLOSURE 2

List of Commitments

TVA will provide a semiannual letter to the NRC of the efforts to resolve the non-conformance.