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May 31, 1991

U.S. Nuclear Regulatory Commission
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ULNRC-2416
NRC TAC No. 68524

Gentlemen:

CALLAWAY PLANT
DOCKET NUMBER 50-483
STATION BLACKOUT

References: 1) ULNRC-1973, dated April 12, 1989
2) ULNRC-2182, dated March 29, 1990

A telecon was held on May 9, 1991 between Union Electric and NRC/SAIC to discuss the Callaway Station Blackout (SBO) submittal. The results of this telecon, in the form of NRC question and Union Electric response, are contained herein.

Please contact us if there are any questions concerning this information.

Very truly yours,

AlPassalt
for Donald F. Schnell

WEK/dls

Attachment

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STATE OF MISSOURI)
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CITY OF ST. LOUIS)

Alan C. Passwater, of lawful age, being first duly sworn upon oath says that he is Manager, Licensing and Fuels (Nuclear) for Union Electric Company; that he has read the foregoing document and knows the content thereof; that he has executed the same for and on behalf of said company with full power and authority to do so; and that the facts therein stated are true and correct to the best of his knowledge, information and belief.

By Alan C Passwater
Alan C. Passwater
Manager, Licensing and Fuels
Nuclear

SUBSCRIBED and sworn to before me this 31st day
of May, 1991.

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NRC Q.1 Justify the claim that the two preferred sources of offsite power to the ESF buses are independent.

UE A.1 As depicted in Figure 8.2-5 of the Callaway FSAR Site Addendum and in Figure 8.3-1 of the Standard Plant FSAR, the two Engineered Safety Features (ESF) transformers with their supply circuits from the 345-kV switchyard provide two independent sources of offsite power for the Class 1E buses.

ESF Transformer XNB01 is supplied by one of the two 345/13.8 kV Safeguard transformers in the Switchyard. A Safeguard Transformer is connected directly to each 345-kV bus through a disconnect switch. Each Safeguard Transformer has two low side breakers connected so that either transformer may supply XNB01 via underground duct. The 13.8-kV breakers are electrically interlocked so that the low side windings of the Safeguard Transformers cannot be connected together. XNB01 is normally supplied by Safeguard Transformer B with the capability for manual transfer to Safeguard Transformer A.

ESF Transformer XNB02 is supplied from one of the secondary windings of Start-up Transformer XMR01. XMR01 is supplied power from a 345-kV circuit from the switchyard. The 345-kV breakers connecting this circuit to switchyard buses A and B are all normally closed.

Normally, Class 1E Load Group 1 (Bus NB01) is supplied by ESF Transformer XNB01 and Load Group 2 (Bus NB02) is supplied by ESF Transformer XNB02. In the event of the loss of a preferred source, the affected load group would be automatically supplied by its associated emergency diesel generator. However, if required, the incoming preferred power supply associated with one load group can supply the 4.16-kV bus of the other load group. This manual transfer is accomplished by operator action in the control room. Each preferred source is sized to supply both load groups simultaneously.

NRC Q.2 Explain what will be stripped, and when it will be stripped, to ensure that the batteries will last for the four-hour SBO without charging. (The UFSAR states that the batteries will last for 200 minutes or 3.3 hours.)

UE A.2 To support the SBO coping assessment for Callaway, calculations employing the methodology of IEEE STD 485 were performed to demonstrate that station batteries have adequate capacity for the four hour coping duration. These calculations assumed a 60°F electrolyte temperature and used a 25 percent margin for aging.

For the Class 1E batteries, no loads are required to be shed to achieve a four-hour capacity. To be prudent, procedural guidance is provided to allow the operators to de-energize the ESF Status Panels in order to conserve battery capacity. These panels may be re-energized if necessary to evaluate equipment status.

The nonsafety-related batteries do not supply any loads necessary for decay heat removal during an SBO but do provide breaker control power to restore offsite power to ESF Transformer XNB02. A non-vital inverter will be shed within one hour after the onset of the SBO to assure the capability to operate the supply breaker to XNB02.

The 200 minute Class 1E battery loading cycle provided in FSAR Table 8.3-2 is the design load cycle for the batteries. A footnote will be added to this table to clarify that the batteries have been analyzed for a 240 minute loading cycle to support the SBO coping analysis.

NRC Q.3 Loss of HVAC (more detailed explanation is needed):

NRC Q.3.a What are the assumed initial temperatures?

UE A.3.a The assumed initial temperatures are:

Turbine Driven Auxiliary Feed Water (TD AFW) Pump Room	113°F	(1)
Battery & Inverter Rooms	90°F	
Control Room/I&C Cabinet Room	N/A	(2)
Main Steam/Main Feedwater (MS/MFW) Tunnel	120°F	

(1) NUMARC 87-00 guidance was used, but plant design is based on loss of offsite power/ventilation for this room which does not have safety related coolers. The original A/E steady state equilibrium calculation shows 142°F final temperature with loss of power (LOP).

(2) The initial temperature is not applicable since a steady state equilibrium calculation was used instead of NUMARC equation. The NUMARC equation is not appropriate for room construction.

NRC Q.3.b Explain the control room calculations.

UE A.3.b Since NUMARC methodology is not appropriate due to control room construction, a steady state equilibrium calculation was used. A combined calculation was done for the control room proper and the I&C cabinet room since they are in the same structural enclosure but separated by the control board. The control room/I&C cabinet room is enclosed by heavy concrete construction with two exterior walls (SE and NE exposure), one wall with air conditioned adjacent space and one wall adjacent to the control room HVAC equipment room in the aux building. Air conditioned cable spreading rooms are above and below. Using the appropriate coefficient of heat transfer "U" values and Q in = Q out, final equilibrium temperatures were calculated.

NRC Q.3.c What heat loads were used in the AFW and control room analyses?

UE A.3.c Heat loads in the TD AFW pump room were not given since it was acknowledged that the original design bases for the room was LOP and that the nonsafety-related room coolers would be inoperable. Heat load in the control room is 8.6 KW and in the I&C cabinet room 32.6 KW, for a total of 41.2 KW.

NRC Q.3.d When, specifically, does procedure OTO-GK-00001 require opening instrument cabinet doors?

UE A.3.d OTO-GK-00001 is being revised to comply with NUMARC 87-00 2.7.1.2a criteria of "within approximately 30 minutes of the event (loss of all AC power) onset".

NRC Q.3.e What assures that fully grouted concrete block walls (used as heat sinks [ ] equivalent to poured concrete walls) have enough mass to approximate concrete?

UE A.3.e The only concrete block walls involved are in the battery and inverter rooms. The cores in these blocks were required to be completely filled with grout to achieve the required fire rating. In addition, the walls are seismic category II/I. The construction procedure that governed the erection of these walls required inspections by field engineering personnel. These documented inspections included verification that the walls were fully grouted per the design documents.

NRC Q.3.f What are the final room temperatures?

UE A.3.f The final room temperatures are as follows:

TD AFW Pump Room	142°F (Original A/E calc)(1)
Inverter Rooms	103.9°F
Battery Rooms	93.7°F
Control Room	111.5°F
I&C cabinet Room	98.1°F
MS/MFW Tunnel	202.2°F

(1) NUMARC 87-00 Equation E-18 results in 136.4°F.

NRC Q.4 What is the expected temperature of the drywell? Does it pose equipment operability problems?

UE A.4 A plant specific containment analyses was performed for the Callaway large dry containment. Two cases were run; one with 111 gpm Reactor Coolant System (RCS) leakage (i.e., 25 gpm/Reactor Coolant Pump (RCP), 10 gpm identified leakage, 1 gpm unidentified) and one case with no RCS leakage. The results were as follows:

T_{MAX} with leakage - 166°F
T_{MAX} no leakage - 173°F

The difference is due to the improved heat transfer due to humidity. Both temperatures are well below the Environmental Qualification envelope temperature of 384.9°F for Main Steam Line Break. Therefore, containment temperature is not a concern for SBO.

NRC Q.5 Explain the containment isolation valve (CIV) analysis and how CIVs are treated in SBO procedures. Additionally, when are the excluded CIVs operated or tested and do they have electrical indication?

UE A.5 The containment isolation valve analysis was performed by reviewing the containment isolation valves identified in FSAR Figure 6.2.4-1 against the exclusion criteria specified in Reg. Guide 1.155 Position C.3.2.7 and the exclusions in NUMARC 87-00. Once the valves that clearly fell under these exclusions were eliminated, the remaining valves were evaluated to determine whether they should be excluded for other reasons. The following provides some of the specific considerations that went into the reviews as discussed in the NRC telecon.

- When considering exclusion b) for valves that fail closed on a loss of power, valves were not excluded unless they had some mechanical mechanism, such as springs, that force the valve to close regardless of what position it was in at the time of power failure. Motor operated valves that fail as-is were not excluded. The valves that were excluded are air operated valves and solenoid valves that fail closed using spring force. Air operated valves that use DC powered air supply solenoids were not excluded since on loss of AC power these valves will not fail closed.

- For the exclusion on non-radioactive closed-loop systems not expected to be breached in a station blackout, we excluded valves in penetrations for the Essential Service Water, Component Cooling Water, and Secondary Side of the Steam Generator systems.

- Fourteen valves were excluded because they are in penetrations which would be isolated by some other valve, generally a check valve. This is based on the allowance that we do not have to assume a single failure. The check valves taken credit for were either containment isolation check valves or Reactor Coolant System Pressure Isolation Valves (PIV's) which are leak tested per our Technical Specifications. We did not take credit for other valves that were not containment isolation valves.

- A specific analysis was performed in order to exclude the Residual Heat Removal (RHR) suction isolation valves from the RCS hot legs. Although they do not meet the specific exclusions for normally locked closed or for fail closed valves, due to the design of the controls for these valves they could not be open at the onset of a station blackout. These valves have interlocks which prevent them from being opened when RCS pressure is above 425 PSIG. It would take a failure of these interlocks in order for these valves to be open and the SBO analysis does not assume single failures. NUMARC 87-00 assumption 2.2.1 states that the SBO analysis be performed assuming the SBO occurs at 100% power which would mean that our RCS pressure would be approximately 2235 psig.

- A specific analysis was performed for the RHR suction isolation valves from the containment sumps. These valves are verified to be closed once every month per plant Technical Specifications. These valves are maintained closed during all power operations and opening the valves would result in entry into Technical Specification action statements. The valves are only opened for surveillance testing during refueling outages in Mode 5 or 6. These valves have interlocks which prevent them from being opened when the RHR suction isolation valves from the Refueling Water Storage Tank are open. Therefore it would again take a failure of the interlocks in order for these valves to be open at the onset of a SBO. Based on the same discussion as above, these valves were excluded.

- A specific analysis was performed for the Containment Spray suction isolation valves from the containment sumps. These valves are maintained closed during all power operations and opening the valves would result in entry into Technical Specification action statements. The valves are only opened for surveillance testing during refueling outages in Mode 5 or 6. The valves are verified to be closed once every month per plant Technical Specifications. The valves are encapsulated inside tanks that are designed as an extension of the containment boundary. Although DC powered indication is available in the control room, these encapsulations will prevent taking manual control to operate the valves. However, the containment spray system was designed to contain radioactive fluid following a LOCA. As discussed in FSAR Figure 6.2.4-1, page 13, a single active or passive failure can be accommodated since the system is closed outside the containment and is designed and constructed commensurate with the design and construction of the containment. The system is tested periodically for leaks as part of our Technical Specification 6.8.4.a Reactor Coolant sources Outside of Containment leakage reduction program. In addition, the system is maintained full of water and isolated from all other systems, which would prevent releases from containment. Based on the low probability of the valves being open and the system is closed outside of containment, it is acceptable that these valves are not capable of being manually closed following a SBO. The valves are included in the emergency response procedure to verify the valves are closed.

In the SBO procedures, the CIVs that need to be verified closed are identified. Procedure ECA-0.0, Loss of AC Power, directs operators to ensure all of these valves are closed using the control room Engineered Safety Features (ESF) status panels which are DC powered. If any of the valves are not closed, it directs operators to manually align the components.

The question on when the excluded CIVs are operated and tested was only discussed in the telecon with regard to the valves that were excluded due to being normally locked closed during operation. Our locked closed valves are not operated or surveilled during power operation. In addition, most of these valves do not have electrical indication. Every valve that receives an automatic containment isolation signal has electrical DC powered indication in the control room. The emergency procedure on loss of AC power verifies all of the valves that have this indication are closed.

NRC Q.6 Identify the assumptions and describe the approach to the plant-specific reactor coolant inventory analysis.

UE A.6 The assumptions used to verify the core would remain covered during an SBO event were:

RCS leakage of 11 gpm
10 gpm identified (allowable per Tech Specs)
1 gpm unidentified (allowable per Tech Specs)

RCP Seal leakage of 100 gpm total
25 gpm per RCP

Letdown Losses 167 ft^3
125 gpm for 10 min. until letdown isolation

RCS shrinkage due to cooldown of 2390 ft^3.

Therefore, total system losses for the 4 hour period are 6118 ft^3. Total volume available to cover top of fuel is 9290 ft^3. Therefore 3172 ft^3 or 23,726 gallons of margin exists.

NRC Q.7 Is there enough compressed air to operate valves needed to cooldown the plant?

UE A.7 The capacity of the nitrogen accumulators for the Steam Generator Atmospheric Steam Dump Valves and AFW control valves was examined to ensure sufficient pressure is required to assure valve operations during the 4 hour coping period. The design nominal pressure will provide sufficient nitrogen for an 8 hour period with each ARV being stroked every 10 minutes, and each AFW control valve stroked 3 times per hour. The minimum allowed pressure will provide air for 5 hours with the same frequency of operation. Therefore, adequate backup air capacity exists.

NRC Q.8 Provide the assumptions used in the CST inventory calculation.

UE A.8 The question was raised as to how Union Electric performed the calculation which determined the required Condensate Storage Tank (CST) Volume. The loads considered in our calculation are:

Decay heat removal (7.43×10^8 BTU for 4 hours)
Sensible heat removal from RCS for cooldown (1.13×10^8 BTU)
Sensible heat removal from the steam generator (S/G) fluid
Restoration of S/G levels to hot zero power conditions

The removal of decay heat and sensible heat from the RCS and S/Gs required approximately 91,000 gallons. Restoration of S/G levels required approximately 40,000 gallons. The calculation then adds a 20% margin which brings the total required water volume to 158,000 gallons. This assumed an initial CST temperature of 120°F. No further actions were required since the current technical specification limit on CST inventory is 281,000 gallons.