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BROWNS FERRY NUCLEAR PLANT (BFN) - 10 CFR 50, APPENDIX R

As requested in the NRC/TVA March 23, 1987 meeting held in Bethesda, Maryland, we are providing the enclosed material to resolve certain Appendix R draft safety evaluation report open items.

It is expected that following the review of this material a final safety evaluation for BFN will be issued. Should additional information on this subject be required, please refer any questions to James D. Wolcott, BFN Site Licensing, at (205) 729-2689.

Very truly yours,

TENNESSEE VALLEY AUTHORITY

Original Signed By
R. L. Gridley

R. Gridley, Director
Nuclear Safety and Licensing

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ENCLOSURE 1
BROWNS FERRY NUCLEAR PLANT
RESPONSE TO NRC REQUEST FOR ADDITIONAL INFORMATION
FROM MARCH 23, 1987 APPENDIX R MEETING IN BETHESDA, MARYLAND

Item 3.4.3-1 Containment Atmosphere Dilution (CAD) Valve Alignment

REQUEST

1. Provide justification for changing manual action to align CAD system valves from one hour to two hours.

2. Show that alignment of CAD system is needed only for non-automatic depressurization system (ADS) valves.

3. Show that the operator can achieve manual actions to align CAD system valves after a fire.

4. Show that each fire area has at least one ADS valve available during a fire event.

5. Show that safe shutdown can be achieved with one relief valve after initial reactor depressurization.

RESPONSE

The manual action of aligning the CAD system is to provide pneumatic (nitrogen) supply to operate the main steam relief valves (MSRVs) for safe shutdown after a fire event. The MSRVs have two safe shutdown functions. The MSRVs are required to depressurize the reactor vessel which allows the RHR system to operate in the low pressure coolant injection (LPCI) mode and maintain coolant inventory. For this function, three MSRVs are required within the first 20 minutes of the fire event. The initial pneumatic supply for these MSRVs is from the two receiver tanks of the drywell air control system, each with a 57 ft^3 capacity. MSRVs with the ADS function have their own accumulators as backup pneumatic supply. These ADS accumulators are sized for five valve operations. Only one valve operation (to open valve) is required of the MSRVs (both ADS and non-ADS). It is estimated that the receiver tanks of the drywell air control system are capable of holding the MSRVs open for at least one hour. The reactor will be sufficiently depressurized by then. Therefore, no manual action is required to provide pneumatic supply to the MSRVs for the first safe shutdown function of the MSRVs.

After the initial depressurization, the MSRVs are used to provide a flow path for removing decay heat from the reactor vessel to the suppression pool. Only one MSRV is required for this function (See table 4.1 of NEDC-31119). For each fire area or fire zone, at least one MSRV used for the initial depressurization will be an ADS valve. This valve will be sufficient to satisfy the second safe shutdown function since it has its own accumulator sized for five valve operations. Except for minor leakage through the solenoid valve to the MSRV pneumatic actuator, the operation of holding the MSRV open will not expend any pneumatic supply. It is estimated that the ADS

Item 3.4.3-1 Containment Atmosphere Dilution (CAD) Valve Alignment (Continued)

RESPONSE (Continued)

accumulator can keep the valve open for at least 2.5 hours (refer to SER dated 7/24/85, Subject NUREG-0737, Item II.K.3.28, "Qualification of ADS Accumulator"). Therefore, no manual action is required to ensure the flow path for residual heat removal circulation from the reactor vessel to the suppression pool within the first 2.5 hours of the blowdown.

The manual action which aligns the CAD system to provide nitrogen to operate the MSRVs within one hour is a conservative requirement which allows the flexibility to use the non-ADS valves for the second safe shutdown function. The required manual actions involve opening two one-inch manual valves and closing another two-inch manual valve which can be performed within two to three minutes. The equivalent fire loads for the fire zones in question are approximately 30 to 40 minutes. The area containing the valves (Elevation 565 of each Reactor Building, Fire Zones 1-1, 1-2, 2-1, 2-2, 3-1, and 3-2) has area detection and suppression, emergency lighting, limited combustibles in the area, a high ceiling (20 to 30 feet) and large open area, and the presence of an open hatch about 50 feet from the CAD location. Portable smoke ejectors, self-contained breathing apparatus, and portable lanterns will be available for use by the operators, if necessary, to minimize any potential problems with smoke obscuration. To alleviate any possible concern on this matter, the one-hour requirement in the manual action tables will be changed to two hours. Therefore, the operator will have a significant amount of time after the fire to perform the manual action requiring only two to three minutes.

Item 3.4.3-2 - Manual Actions

REQUEST

1. Show that each fire area (with the exception of fire area 16 of the Control Building) has at least one emergency equipment cooling water (EECW) pump auto-start capability.

2. Provide analysis to support extending the time limit for manual start of EECW pumps.

3. Commit to update correction to manual action calculation table for starting EECW pumps.

4. Show cost impact of performing modification to provide EECW pump auto-start feature for a Control Building fire.

RESPONSE

The current post-fire shutdown procedure for BFN has a manual action requirement to manually verify the EECW pump availability in five minutes after the start of a diesel generator, normally assumed to be at the initiation of the fire event.

RESPONSE (Continued)

Recent review of the plant configuration and the EECW pump start logic indicates that the previous approach and assumptions went beyond the design basis for Appendix R fire events. The EECW pump logic includes an automatic start circuitry based on diesel generator start recognition signals and a manual start and stop circuitry for each pump. The circuitry for the diesel generator start recognition signals is inside the Reactor Buildings, Diesel Generator Buildings, and the respective shutdown board rooms. The circuitry for the manual start and stop signals is inside the Control building and the respective shutdown board rooms and locally in the intake pump station.

Analysis demonstrates that for a fire in any of the fire areas (4 through 15 and 17 through 24), at least two EECW pumps would start automatically upon a start signal with its associated diesel generators, thus eliminating the need to manually verify the EECW pump start from outside the control room for these areas.

Analysis also indicates that the auto-start circuitries associated with the diesel generator run recognition for the two divisions of the EECW pumps (A3 and C3 versus B3 and D3) are separated from each other by the wall between the unit 2 and unit 3 Reactor Buildings (figure 1). Thus, EECW pumps A3 and C3 would be available for a fire in unit 1 or 2. For most locations in units 1 and 2, the A3 and C3 pumps would start automatically. However, for fire zones 1-3 and 2-3, because the control power feeder to 4 kV Shutdown Board 3EA is located in these two zones, the automatic start of the A3 pump cannot be ensured. The A3 pump can be started by manual action outside of the control room. The automatic start of the C3 pump is ensured for these two fire zones. For fire zone 1-3, only unit 3 diesel generators are used for safe shutdown. The C3 pump provides direct cooling water to the unit 3 diesel generators and no spurious operation of the sectionalizing valves on the EECW header would defeat this direct cooling capability. For fire zone 2-3, one of the unit 1 and 2 diesel generators (DG A) is used for supplying power to unit 1 RHR pump and other required ac equipment. Since unit 1 is not a fire affected unit for a fire in fire zone 2-3, the diesel generator DG A which provides power to unit 1 will not be required for the first two hours of the event. A spurious operation of the sectionalizing valve in unit 2 (FCV-67-21) would not affect the unit 3 diesel generators which supply power to unit 2; however, it would cut off the EECW flow from EECW Pump C3 to the units 1 and 2 diesel generators. If this happens, the operator can trip the required diesel generator DG A from the control room upon receiving the high temperature alarm. The operator would then manually start the EECW pump before restarting the diesel generator at 30 minutes. Should the operator fail to trip the diesel generator and should the diesel generator subsequently fail, the operator can align another unit 3 diesel generator to supply the necessary ac power to support the safe shutdown function in unit 1. For both fire zones 1-3 and 2-3, if the EECW flow from EECW pump C3 does not provide adequate cooling water to the required diesel generators, the operator could close the unit 3 sectionalizing valves from the main control room to isolate unessential loads. The operator would then manually start the EECW pump A3 locally when

RESPONSE (Continued)

required to operate the unit 1 and unit 2 diesel generators. Therefore, the Reactor Buildings (fire areas 1 through 3) will only require a manual action inside the control room to ensure an adequate cooling water supply to the diesel generators.

The intake pumping station (part of fire area 25) will have a two-pump auto-start capability; however, a spurious trip of one pump would leave a single pump to supply the required diesel generators. To ensure adequate flow, the operator may be required to either close the EECW header sectionalizing valves or manually start a second EECW pump at the shutdown board. The circuits to sectionalizing valves would be unaffected by a fire in the intake pumping station and could be operated from the control room.

The Control Building (fire area 16) contains only the manual start and stop circuitry for all of the EECW pumps. Consequently, a fire in the Control Building will not affect the auto-start circuitry of the EECW pumps and the EECW pumps would receive the automatic start signal if the diesel generators should start. An EECW pump would not start during a Control Building fire only if the fire damages the manual stop circuitry causing the EECW pumps to spuriously stop. It would require four simultaneous and identical spurious operations to cause a total loss of the EECW pumps. It would require even more spurious operations to cause a total loss of the EECW pumps along with fully loading up the diesel generators. These conditions are beyond the required spurious operation assumptions given in Generic Letter 86-10.

One of the proposed fixes to ensure the diesel generator availability is to provide an automatic trip of the diesel generators based on high jacket water temperature. A simple trip system would have two safety grade temperature sensors with dual contacts at each diesel generator and connecting the logic to a local terminal block for the trip relays. The cost estimate for this simple system is approximately $589,770 for all three units. This estimate includes all engineering, materials, and construction costs. The estimated schedule is approximately 30 weeks for engineering and 60 days for construction. However, this modification is not desirable because the added protection for the diesel generators will also contribute to the unavailability of the diesel generators during a loss of power event. This high cost cannot be justified in view of the fact that the trip may reduce the reliability of the diesel generators.

Based on the above, it is evident that the five minute manual action requirement for the EECW pumps is unnecessary. Analysis demonstrates that, for a fire in fire area locations 4 through 24, at least two EECW pumps would start automatically upon a start signal from its associated diesel generator. All fire zones in fire areas 1, 2, and 3, except for zones 1-3 and 2-3, would have the auto start capability of at least two EECW pumps. For fire area 25 and fire zones 1-3 and 2-3 spurious pump trip or valve closure would leave a single EECW pump which is adequate until appropriate control room (or local) manual actions are taken. The manual actions table will be changed to remove

RESPONSE (Continued)

the 5-minute action accordingly making the soonest operator action 10 minutes. The 10-minute actions are either to verify the scram and isolation functions or to disable the HPCI system if it is operating unreliably and overfilling the reactor vessel.

Item 3.4.6-2 - Credit for Drywell Coolers

REQUEST

1. State that drywell coolers are not needed or taken credit for during an Appendix R event.

2. State that drywell coolers will not be deliberately tripped unless the entry conditions are met.

RESPONSE

The drywell coolers are not required for an Appendix R safe shutdown event. However, the drywell cooling function will not be deliberately removed unless the following three entry conditions for the safe shutdown procedure are met:

1. A confirmed fire occurrence of significant severity in any plant location in which an immediate fire extinguishment cannot be achieved.

2. Loss of adequate high pressure makeup capability such that no reasonable alternate exists but to proceed to a low pressure source of makeup.

3. Inability to provide normal or emergency power to vital equipment.

Since the drywell blowers are not an essential safety function, they were not separated per Appendix R requirements. To preclude the possibility of the drywell blower cables from contributing to a high impedance fault condition, the breaker to the blower is tripped when its respective reactor motor operated valve board is aligned. For those cases in which the RMOV board is not used for the Appendix R event and in which it is essential that the drywell cooling function be defeated, either power to the blowers or source power to the board will be removed.

Item 3.4.3-4 - Scram Verification

REQUEST

1. State that shutdown procedures will allow scram verification to be verified by tripping reactor pressure system (RPS) motor generator (MG) set breaker from the battery board room.

RESPONSE

The Appendix R Procedure will require either opening the RPS MG set generator output breaker on the battery board or opening the feeder breaker to the RPS MG set motor on the 480V reactor motor operated valve board. This action will ensure power is removed from the reactor protection system which causes an automatic scram on loss of power and verifies the scram will take place.

Item 4.0.A.3 - Torus Level and Temperature Instrumentation

REQUEST

1. Summarize the previously made case (NEDC 31119, November 21, 1986 submittal) that the instrument circuits will survive a fire event.

2. Summarize the previously made case (NEDC 31119, November 21, 1986 submittal) for not requiring the use of the instruments.

3. Provide cost impact of performing modification to ensure instrument indication availability during a fire event on elevation 593' of reactor building.

RESPONSE

Documentation has been provided in NEDC-31119 and in the November 21, 1986 submittal which demonstrates that the suppression pool level and temperature instrumentation are not required to mitigate the design basis Appendix R event. In particular, the analysis has shown that the instrumentation will not be required to determine any operator actions to preserve pool level or temperature. All operator actions are based on reactor conditions (reactor pressure and reactor water level) and preanalyzed worse-case conditions in which actions are performed regardless of the torus conditions. Operator action in the Appendix R Shutdown Procedure will not depend upon knowledge of torus level or temperature. All sources which are capable of overfilling or draining the torus have been determined and these sources will be isolated. The procedures will direct the operators to depressurize the reactor and establish a heat removal path (alternate shutdown cooling) based upon low reactor water level or high torus temperature or the inability to determine torus temperature. Analysis demonstrates that the peak torus temperature will remain below torus temperature limits given the worst timeframe assumed (two to three hours) for performance of these operator actions. Therefore, torus temperature and level process variables are not necessary to perform or control the functions given in III.L.2 of Appendix R.

A description of the cable routing for the torus instrumentation was provided in the November 21, 1986 submittal. The cables for the torus instrumentation inside the Reactor Building stay within its own unit. Separation between the

<u>RESPONSE</u> (Continued)

redundant trains exists on the 519 and 565 elevations of each Reactor Building. However, the north area of the 593 elevation of the Reactor Buildings is the area of the closest proximity for most of the torus instrumentation cables. Consequently, the situation is that for a fire in any area other than the north area (593 elevation) of each Reactor Building, the torus instrumentation is available to the operator if he desires to confirm the torus condition. Since a separate loop of the torus instrumentation is available at the backup control panel, the torus instrumentation is available for a fire event in the Control Building. For the north area of the 593 elevation, it is expected that sufficient instrumentation would survive a fire on this elevation at this particular location where the redundant trains of torus instrumentation are at the closest proximity. This is because the north area of 593 elevation is presently protected by an existing open head spray system and will be protected by a sprinkler system designed to NFPA-13 requirements. The major combustible in this area is cable insulation which is either in conduit or coated liberally with Flamastic.

If the instrumentation did not survive, alternate means exist to determine the torus conditions if the operator desires so. For example, the RHR heat exchanger inlet temperature can be used to determine the pool temperature. The cables for the RHR heat exchanger inlet temperature with indication in the control room is approximately 10 feet from the closest proximity for the pool temperature indications. If the cables are also lost during the fire, thermowells are available to provide a local determination of the torus temperature. Direct torus water level indication can be determined by taking differential pressure readings from the instrument taps at the local torus level transmitter. Therefore, various alternate means can be used to determine the torus conditions should the operator desire such information.

The cost to separate the torus instrumentation is estimated to be approximately $2,238,876 for all three units. This estimate includes all engineering, materials, and construction costs. The estimated schedule duration is approximately 30 weeks per unit for engineering which could be escalated by 50 percent due to Class 1E and safety-related procurement and 30 to 60 days for construction per unit.

In summary, the high cost of providing further protection for torus instrumentation at the 593 elevation cannot be justified based on the following considerations.

a. The benefit of the torus instrumentation has been demonstrated to be minimal.

b. It is highly probable that the existing instrumentation would survive a fire in the critical area.

c. Alternate means of determining torus conditions are available to the operators.

Item 4.0.A-4 - Three Phase Shorts on HI-LO Pressure Interface Circuit for Residual Heat Removal (RHR) Shut Down Cooling (SDC) Flow Control Valve (FCV) 74-48

REQUEST

1. Commit to tag out the breaker for FCV 74-48 to ensure removal of motive power.

2. State what steps are necessary to ensure the valve is in its proper position prior to tagging out and what controls exist to ensure the breaker is periodically verified for proper tagged out position.

RESPONSE

For HI-LO pressure interface valve FCV 74-48 motive power will be removed by opening the breaker disconnect switch and placing a valve hold order tag on the breaker and the control room switch. Before tagging out the equipment, position indicating lights are available to ensure the valve's proper tagged out position. The tagout is verified by a Semiannual Hold Order Audit which includes verification of the position of the tagged equipment.

Item 4.0.B.4 - Manual Actions in Fire Areas 4, 8, 9, 12, and 13

REQUEST

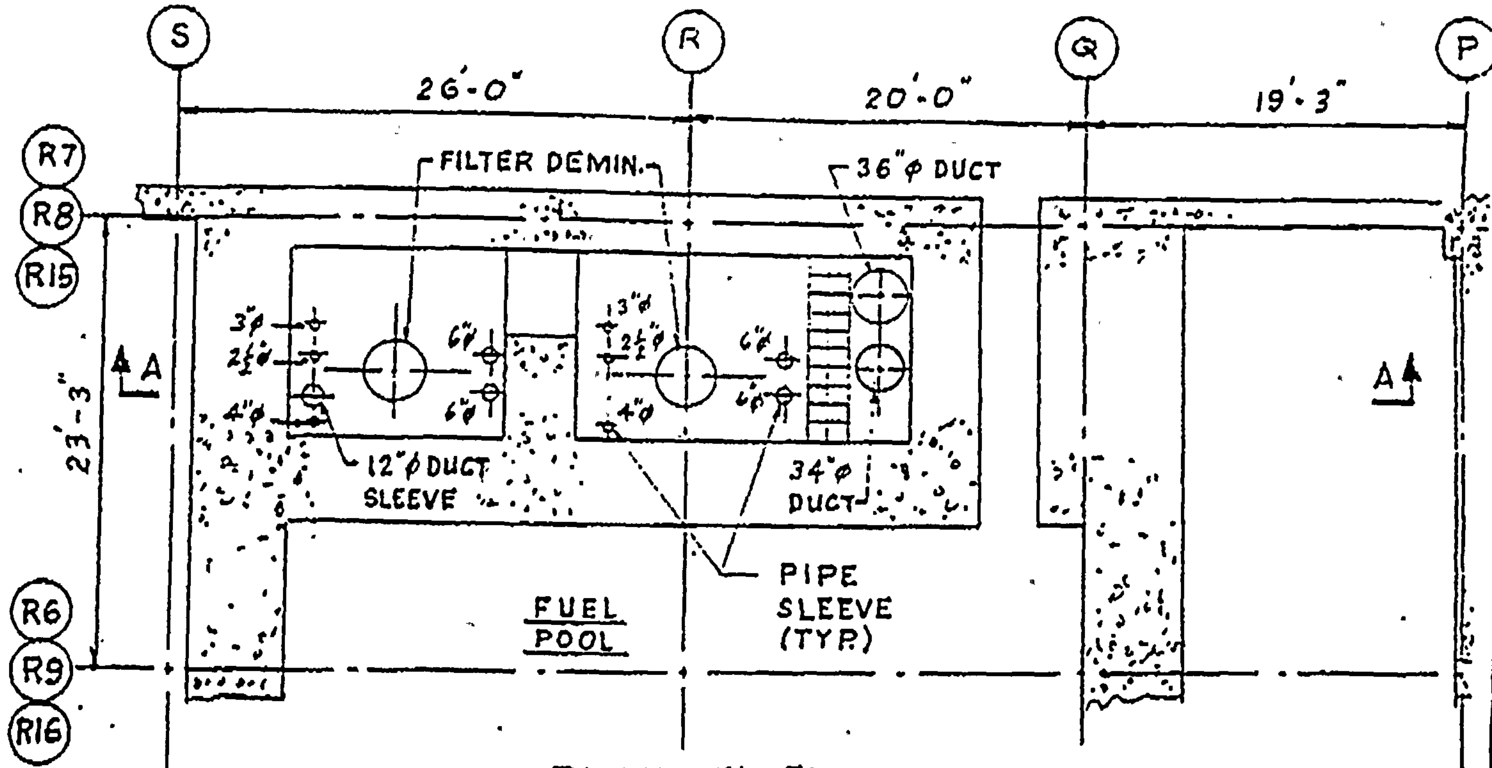
1. Confirm that the manual actions performed in these fire areas do not require cold shutdown repairs to accomplish.

2. Clarify what manual actions are performed in these fire areas.

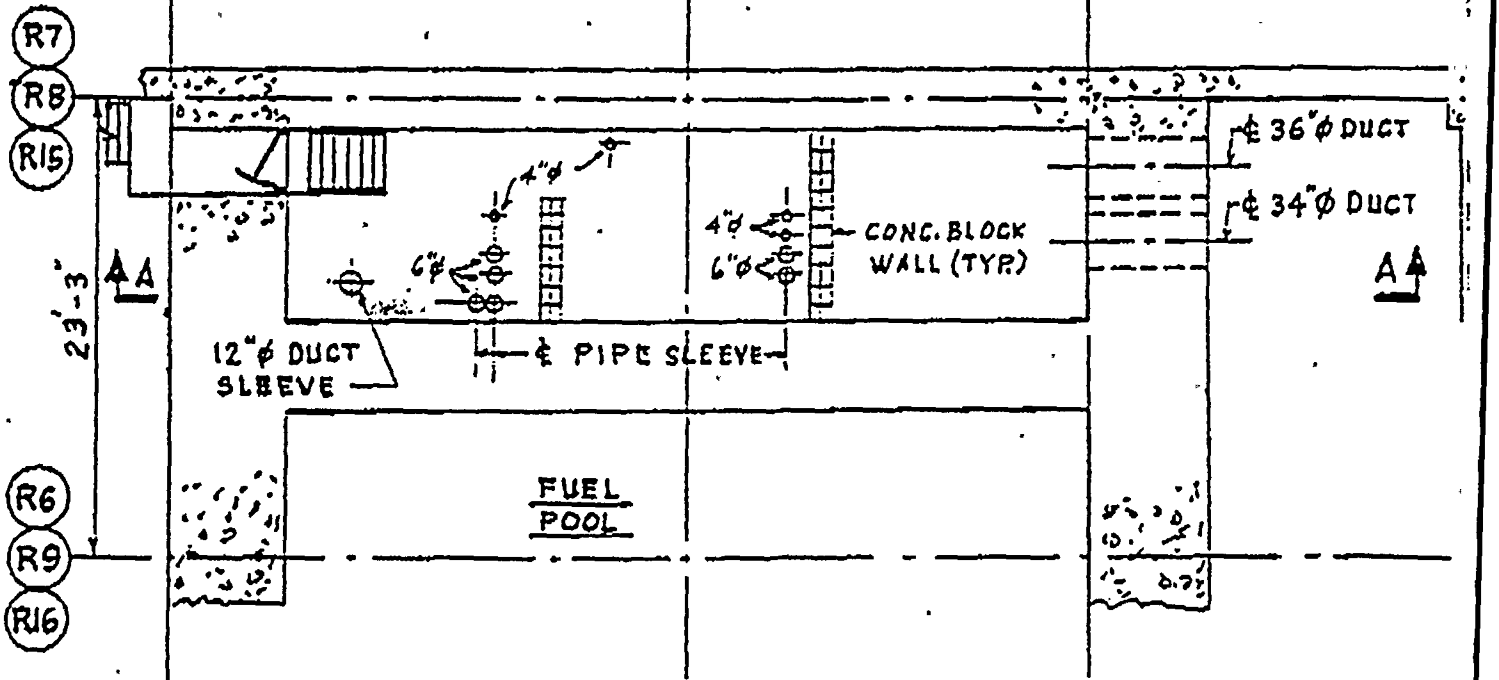
3. Commit to correcting mistakes identified in manual action calculations (Enclosure 4 of November 21, 1986 submittal to NEDC 31119) for these fire areas.

RESPONSE

Enclosure 4 of the November 21, 1986 submittal provided a table of manual actions for each fire area. The table for fire areas 4, 8, 9, 12, and 13 mistakenly identified an action to be performed on a potentially fire damaged circuit breaker. This error was identified during development of the operating procedure and will be corrected in manual action calculations. The valves will be disabled by verifying that power to the board is removed from its power source located in a different fire area. The valves are disabled to prevent spurious operation and allow the operator to manually position the valve using the local handwheel. No repair or repair procedure is required to obtain hot or cold shutdown conditions for these areas.



PLAN - FL. EL. 639.0
RWCU FILTER DEMIN. ROOMS
UNIT-1 - OPPOSITE HAND
UNITS 2 & 3 - AS SHOWN



PLAN - FL. EL. 621.25
RWCU VALVE ROOM
UNIT-1 - OPPOSITE HAND
UNITS 2 & 3 - AS SHOWN

Item 4.0.B.5 - III.G.1 and III.G.3 Fire Areas

REQUEST

1. Confirm that III.G.1 fire areas identified in November 21, 1986 submittal to NEDC 31119 do not contain any redundant cables, equipment, or components.

2. Clarify the withdrawal of the III.G.3 exemption for the turbine building.

RESPONSE

The following information is provided to clarify fire area III.G designations.

A fire area designated solely as III.G.1 will have available redundant equipment, cabling, and systems for Appendix R that is not located in that fire area.

The November 21, 1986 submittal clarified the definition for alternate shutdown such that the turbine building (part of fire area 25, subject of exemption i) would now be classified as III.G.2 fire area. Exemption i includes a III.G.3 exemption for the turbine building which is no longer needed. Therefore, that portion of Exemption i will be withdrawn.

The intake pumping station fire area classification of III.G.1 as shown in the November 21, 1986 enclosure 3 table should be III.G.2.

Item 4.0.B.6 - Testing of Safe Shutdown Systems

REQUEST

1. Describe the initial, postmodification, and periodic testing of remote shutdown panel 25-32.

RESPONSE

The backup control panel was initially tested in the BFN preoperational test program using General Electric test procedures. When modifications are performed, postmodification testing is required to ensure the affected equipment will operate as designed. The remote shutdown panel and electrical distribution system are currently tested each operating cycle. This testing includes checks to determine remote operation capability and circuit isolation from the Control Building.

New Item from Section 3.4.3 of Draft Safety Evaluation Report

REQUEST

State what manual actions are being taken to ensure the following objectives can be accomplished within 10 minutes.

1. Transfer to local control (if needed) and ensure closure of main steam isolation valves (MSIV).

2. Close the high pressure coolant injection (HPCI) steam supply shutoff valve (in the fire affected unit) to prevent water intrusion into the main steam lines.

RESPONSE

Manual actions have been provided to verify that the main steam isolation valves are closed following the automatic or manual control room isolation. This action involves accessing the remote control panel (panel 25-32), transferring the control switches, and placing the control switch for each MSIV to the closed position. There are eight valves (four inboard and four outboard) and the action on the panel requires only seconds to accomplish.

The HPCI system can potentially operate spuriously requiring an action to prevent filling the reactor vessel up to the main steam lines. To prevent this occurrence, the HPCI steam line will be isolated by closing the steam supply valve to the HPCI turbine. This action requires the operator to transfer the breaker to emergency on the 250V reactor motor operated valve board A and then operate the control switch to the closed position. The access and action time should only require one or two minutes.