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Byron Station, Units 1 and 2
Facility Operating License Nos. NPF-37 and NPF-66
NRC Docket Nos. STN 50-454 and STN 50-455

Subject: Response to a Request for Information Regarding the Basis of a Revision to the Updated Final Safety Analysis Report Addressing Isolation of the Thermal Barrier Cooler Return Line

Reference: Letter from G. F. Dick (NRC) to O. D. Kingsley (Exelon Generation Company, LLC), "Isolation of a Thermal Barrier Cooler Return Line – Byron Station, Units 1 and 2," dated September 26, 2001

In the referenced letter the NRC requested that we provide the basis for our conclusion that a revision to the Updated Final Safety Analysis Report regarding the isolation capability of the reactor coolant pump thermal barrier component cooling return line could be implemented without submitting a license amendment request to the NRC. Attachment A to this letter contains our response.

The original response was due to the NRC on November 12, 2001. During a November 1, 2001, telephone conversation between G. F. Dick (NRC) and J. A. Bauer (Exelon), the due date was extended to November 16, 2001.

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Rec'd 01/14/02

Should you have any questions regarding this submittal, please contact Mr. William Grundmann, Regulatory Assurance Manager, at (815) 234-5441, extension 2800.

Respectfully,

Richard P. Lopriore
Site Vice President
Byron Nuclear Generating Station

RPL/JL/dpk

Attachment A: Byron Station, Justification of Conclusions, Updated Final Safety Analysis Report Revision, Isolation of a Thermal Barrier Cooler Return Line

cc: Regional Administrator – NRC Region III
NRC Senior Resident Inspector – Byron Station
NRC Project Manager – NRR – Byron Station
Office of Nuclear Facility Safety – Illinois Dept. of Nuclear Safety

Attachment A

Byron Station
Justification of Conclusions

Updated Final Safety Analysis Report Revision
Isolation of a Thermal Barrier Cooler Return Line

ATTACHMENT A

NRC REQUEST FOR INFORMATION

Please provide the basis upon which it was concluded that the changes to the isolation capability of the thermal barrier cooler return line and the associated UFSAR change could be done without submitting a license amendment to the NRC in accordance with 10 CFR 50.59 and 50.90.

RESPONSE

Background

The Component Cooling (CC) Water, motor operated valves (MOVs), 1/2CC9438 and 1/2CC685 are the inboard and outboard containment isolation valves installed on the Reactor Coolant Pump (RCP) thermal barrier cooling return line. The valves are associated with containment penetration P-24. As indicated in Byron/Braidwood Updated Final Safety Analysis Report (UFSAR), Table 6.2-58, valves 1/2CC9438 and 1/2CC685 were designed to the requirements of General Design Criteria (GDC) 56. The 1/2CC9438 and 1/2CC685 MOVs receive a Phase B Containment Isolation signal. A Phase B containment isolation signal is actuated either manually by plant operators or automatically upon receiving a high containment pressure signal (i.e., High-3 signal). The Phase B containment isolation signal results in an autoclose signal to both the 1/2CC9438 and 1/2CC685 valves.

In addition to the Phase B isolation signal, the 1/2CC685 valves receive a close signal upon indication of high CC flow (i.e., greater than 240 gpm) in the thermal barrier cooling return line. The high flow return line automatic isolation of the 1/2CC685 valve is provided to mitigate a postulated break of the RCP thermal barrier. The 1/2CC9438 valves do not receive an autoclose signal in the event of high flow in the thermal barrier cooling return line.

A review of a record copy of the Byron/Braidwood Final Safety Analysis Report (FSAR), Table 6.2-58, "Containment Isolation Provisions" shows that the plants were originally licensed with the 1/2CC9438 valves designed to the requirements of GDC 56, "Primary Containment Isolation." The GDC 56 designation for the RCP thermal barrier return containment penetration is in accordance with Westinghouse System Standard Design Criteria 1.14, "NSSS Containment Isolation". The use of GDC 56 for the CC return lines was accepted by the NRC in the Byron Safety Evaluation Report, NUREG 0876, Section 6.2.4, "Containment Isolation System." FSAR Figure 9.2-3, "Component Cooling System," and Section 9.2.2.4.4, "Shared Function" indicate that the high CC flow autoclose signal is only provided to the 1/2CC685 MOVs. Single automatic isolation in response to a postulated RCP thermal barrier break was reviewed and accepted as the design for the plant.

The 1/2CC685 MOVs were modified in the early 1990s to meet the recommendations of Generic Letter 89-10, "Safety-Related Motor-Operated Valve Testing and Surveillance." The original 1/2CC685 valve actuator did not have adequate thrust capability to close the valve under postulated worst case conditions. The modification replaced the existing valve and motor operator with a smaller diameter valve and larger motor operator. The 1/2CC9438 MOVs were not modified because; 1) the valves were fully capable of performing their containment isolation function without modification, 2) the valves did not receive the same auto close signal as the 1/2CC685 valves, and 3) no specific instructions existed at the time in the Emergency Operating Procedures (EOPs) to close 1/2CC9438 valves should the 1/2CC685 valves fail to auto isolate.

ATTACHMENT A

In 1997, an internal audit identified an apparent UFSAR discrepancy in that the wording in UFSAR, Section 9.2.2.4.4, took credit for manually closing the 1/2CC9438 MOVs if the 1/2CC685 MOVs did not close in response to a thermal barrier break; however, the 1/2CC9438 valves did not have the same design capacity as the 1/2CC685 valves. The audit questioned whether the 1/2CC9438 MOVs should also be capable of closing against the maximum differential pressure associated with a break of the RCP thermal barrier. In response to this internal audit, an operability evaluation was prepared in response to the apparent discrepancy and the EOPs were revised to provide more guidance on how to respond to a failure of the 1/2CC685 valves to autoclose. This revision to the EOPs included direction to attempt to manually close the 1/2CC9438 valves.

As a corrective action for the apparent UFSAR discrepancy, Section 9.2.2.4.4 of the UFSAR was revised in 1998. The change was implemented under UFSAR Draft Revision Package (DRP) Number 7-257. A 10 CFR 50.59 Safety Evaluation, 6G-98-0200, was prepared to support the UFSAR change. The Safety Evaluation concluded that the proposed activity could be implemented under the provisions of 10 CFR 50.59. Prior to the change, the UFSAR stated the following:

> *"Safety-related indication of component cooling water flow from the reactor coolant pump motor oil coolers is provided and alarmed in the main control board. The reactor coolant pump (RCP) thermal barrier outlet header has a flow indicating switch which causes a motor-operated valve to close in this line in the event of high flow (an indication of a broken RCP thermal barrier). Should the valve or switch not operate properly, an increasing level is noted in the CCWS surge tank, resulting in high level alarm, if not isolated. A second motor-operated valve in series with the previously mentioned valve is available for manual isolation of the line if required. Additionally, two level instruments are provided on each surge tank, both of which will give a high alarm in the control room."*

The UFSAR wording was revised to the following:

> *"Safety-related indication of component cooling water flow from the reactor coolant pump motor oil coolers is provided and alarmed in the main control board. The reactor coolant pump (RCP) thermal barrier outlet header has a flow indicating switch, which causes a motor-operated valve (_CC685) to close in this line in the event of high flow (an indication of a broken RCP thermal barrier). Should the valve or switch not operate properly, an increasing level is noted in the CCWS surge tank, resulting in high level alarm, if not isolated. Emergency operating procedures would direct plant operators to initiate actions to isolate the leak via manual closure of the _CC685 valve from the control room, attempted closure from the control room of a second motor operated valve (1/2CC9438) which is located upstream of the _CC685 valve inside containment, or dispatch of an operator locally to facilitate closure of the _CC685 valve. Note that the _CC9438 valve meets the provisions of 10CFR50 Appendix B General Design Criteria for containment isolation following an accident which releases radioactive material inside containment. However, the _CC9438 valve does not receive an automatic signal on high CCWS flow and is not provided with the same design capability as the _CC685 valve for isolation of the CCWS RCP thermal barrier cooling line following a thermal barrier*

break. Additionally, two level instruments are provided on each surge tank, both of which will give a high alarm in the control room."

Basis for Processing the UFSAR Change without Prior NRC Approval

The UFSAR change was effectively a change in the manual actions to be taken in response to a failure of the 1/2CC685 valve to auto-isolate on high RCP thermal barrier return flow. The UFSAR change does not substitute manual actions in place of automatic functions. The original design only provided one automatic isolation valve to autoclose in the event of a thermal barrier break. Upon a failure of the 1/2CC685 valve to autoclose, manual operator action would be required to mitigate the failure. The original FSAR description stated: *"Should the valve or switch not operate properly, an increasing level is noted in the CCWS [Component Cooling Water System] surge tank, resulting in high level alarm, if not isolated. A second motor-operated valve in series with the previously mentioned valve is available for manual isolation of the line if required."* The second motor-operated valve in series with the autoclose 1/2CC685 valves are the 1/2CC9438 valves.

The UFSAR change made was to revise the manual action description to 1) attempt to manually close the 1/2CC685 valves from the control room; 2) attempt to isolate the leak by manually closing the 1/2CC9438 valves; and 3) dispatch an operator locally to manually close the 1/2CC685 valve. Since the 1/2CC685 valves are postulated to fail and the 1/2CC9438 valves may not be fully capable of closing under the worst case postulated differential pressure, the third action, i.e., dispatching an operator locally to manually close the 1/2CC685 valves, becomes the new design basis method of manually mitigating the postulated event and single failure.

As discussed in NRC Information Notice 97-78, "Crediting of Operator Actions in Place of Automatic Actions and Modification of Operator Actions, Including Response Times", changes in manual actions described in the FSAR must be evaluated under 10 CFR 50.59 to determine whether an Unreviewed Safety Question (USQ) is involved and whether NRC review and approval is required before implementation. IN 97-78 provides the following guidelines for review of changes in manual actions: 1) the specific operator actions required; 2) the potentially harsh or inhospitable environmental conditions expected; 3) a general discussion of the ingress/egress paths taken by the operators to accomplish the functions, 4) the procedural guidance for required actions; 5) the specific operator training necessary to carry out actions, including any operator qualifications required to carry out actions; 6) any additional support personnel and/or equipment required by the operator to carry out actions; 7) a description of information required by the control room staff to determine whether such operator action is required, including qualified instrumentation used to diagnose the situation and to verify that the required action has successfully been taken; 8) the ability to recover from credible errors in performance of manual actions, and the expected time required to make such a recovery; and 9) consideration of the risk significance of the proposed operator actions. The following discusses each of these guidelines:

ATTACHMENT A

1. The specific operator actions required.

The specific operator actions required are clearly described in the revised wording in Section 9.2.2.4.4 of the UFSAR and in the operating procedures. Specifically, in Attachment B of Byron Operating Abnormal (BOA) procedure, 1/2BOA PRI-6, "Component Cooling Malfunction", the operator action in response to increasing CC surge tank level is to check 1/2CC685 with the "expected response" of "closed;" the "response not obtained" actions are: 1) Close 1/2CC685, 2) If 1/2CC685 can not be closed then close 1/2CC9438, and 3) If neither the 1/2CC685 or 1/2CC9438 can be closed, then perform the following: a) Deenergize 1/2CC685 at motor control cent (MCC) 1/232X4 D4 and b) Locally close 1/2CC685.

These actions are consistent with other operator actions taken in response to similar equipment malfunctions. Adequate time is available for the operators to take these actions because the leak rate from a postulated thermal barrier break is expected to be relatively small. The postulated maximum inleakage from a failure of the RCP thermal barrier is 285 gpm. Westinghouse plants did not have a specific safety analysis addressing this event but the thermal barrier rupture with the postulated maximum inleakage is, in effect, a small break loss of coolant accident (SBLOCA). UFSAR Section 5.4.1.3.10, "Shaft Seal Leakage," discusses the impact of a similar SBLOCA associated with a postulated failure of a RCP seal with a 300 gpm leakrate. For a break of this size, the RCS will slowly depressurize and a reactor trip and a safety injection (SI) signal will be generated. If only one train of SI pumps is available (i.e., minimum safeguards), the RCS will continue to depressurize until the high-head SI flow is capable of matching the liquid phase break flow. The RCS pressure will stabilize at this equilibrium pressure, which will be greater than the secondary side pressure. If two trains of SI pumps are available (i.e., best estimate safeguards), the RCS will begin to refill and repressurize subsequent to SI flow delivery. The RCS pressure will rise to a level such that SI termination and throttling will be necessary. For either "minimum" or "best estimate" safeguards capability, the steam generators would not drain for a break of this size and continuous circulation of reactor coolant will be provided by the RCP or by natural circulation, if the RCP had tripped. The reactor core would remain covered and adequately cooled for a break of this size. Given the available inventory in the Refueling Water Storage Tank (RWST), adequate inventory is available for SI injection until the manual actions can be taken to isolate the thermal barrier break.

If the 1/2CC685 valve fails to auto-close, the inleakage of RCS into the CC system may result in overfill of the CC surge tank to the Auxiliary Building equipment floor drains. Assuming the maximum inleakage from a thermal barrier break, the CC surge tank would overflow in approximately four minutes. The existing flood analysis for the Auxiliary Building area where the CC surge tank is located is based on a limiting break size of 448 gpm, as the drain system is capable of removing this volume of water. Thus, for the worst case thermal barrier break, the analyzed flood height remains bounded.

For the specific case of an overflow of the CC surge tanks, the water would collect in the berm area surrounding the CC surge tank. There are two 4" drains in the berm area, rated at 100 gpm. Assuming a 285 gpm leak, the water would back up into the berm area at a rate of 185 gpm. A conservative estimate of the berm size indicates that it would take approximately 22 minutes to overfill the berm. After the berm area is full, the water would flow to other nearby floor drains.

The drains from the berm area and the nearby local floor drains transfer water to the Unit 2 Auxiliary Building Floor Drain Sump Room, which has a volume of approximately 2000 cubic feet. Considering the maximum inleakage from a thermal barrier break and approximately 4,070 gallons of water in the full CC surge tank berm, approximately 67 minutes is available to isolate the leak before the sump room is filled and the water overflows into other rooms within the Auxiliary Building. This amount of time is judged to be adequate for taking the required operator actions to locally manually isolate the 1/2CC685 valve.

2. **The potentially harsh or inhospitable environmental conditions expected.**

Dispatching a plant operator to manually isolate the 1/2CC685 valve would not result in significant radiation exposure to the operator. The RCS coolant that would potentially be flowing through the RCP thermal barrier and the CC return line would not be expected to have high coolant activity in excess of the Technical Specification (TS) allowable limits. The postulated maximum inleakage from a failure of the RCP thermal barrier is 285 gpm. The thermal barrier rupture with the postulated maximum inleakage is, in effect, a SBLOCA. UFSAR Section 5.4.1.3.10 discusses the impact of a similar SBLOCA associated with a postulated failure of a RCP seal with a 300 gpm leakrate. For a SBLOCAs of this size, the reactor core would remain covered and adequately cooled; accordingly, RCS activity levels are not expected to exceed normal operating levels.

3. **A general discussion of the ingress/egress paths taken by the operators to accomplish the functions.**

The MCCs are located in the Auxiliary Building Electrical Penetration Area on elevation 426'. Operator access can be achieved from either the Fuel Handling Building or from the Auxiliary Building. The 1/2CC685 valves are located in the containment penetration area of the Auxiliary Building. Operator access to the penetration area is from 401' elevation down the stairway to the 364' elevation. The CC surge tank is located on the 426' elevation of the Auxiliary Building. As discussed above, the overflow of potentially contaminated CC water is contained in the CC surge tank berm area and the sump room for at least 22 minutes. Thus, operator ingress/egress paths to accomplish the necessary functions are not adversely affected.

4. **The procedural guidance for required actions.**

Procedural guidance for the required actions is provided in procedures 1/2 BOA PRI-6, as discussed above.

5. **The specific operator training necessary to carry out actions, including any operator qualifications required to carry out actions.**

All field operators are trained in the steps to manually trip a breaker and manually close a motor operated valve locally using the valve handwheel.

6. **Any additional support personnel and/or equipment required by the operator to carry out actions.**

No additional support personnel or equipment is required to manually trip the breaker and locally close the 1/2CC685 valve.

7. **A description of information required by the control room staff to determine whether such operator action is required, including qualified instrumentation used to diagnose the situation and to verify that the required action has been successfully been taken.**

   Indications of a thermal barrier leak includes radiation monitors, high CC return flow from the RCP thermal barrier, and CC surge tank level. Verification that the required action has been successfully been taken would be available from the CC flow instrumentation, CC surge tank level instrumentation or termination of overflow from the CC surge tank.

8. **The ability to recover from credible errors in performance of manual actions and the expected time required to make such a recovery.**

   The operator could make an error when manually operating the hand switches in the control room, de-energizing the CC685 valve, or in manually closing the valve locally. Evidence of an error would be apparent from the available CC flow and level instruments. Since the credible errors are reversible, recovery from an error is not a concern and the time available for taking the manual action is sufficient to accommodate recovery from any credible error.

9. **Consideration of the risk significance of the proposed operator actions.**

   The Core Damage Frequency (CDF) for this event is so low that it screens out of the Byron probabilistic risk assessment model. The CDF is low because the probability of the thermal barrier break is small and there is a high probability that the event will be mitigated without core damage.

   In summary, based on the low risk significance of the postulated event, the guidance provided in the plant procedures and operator training programs, the high probability that the proposed operator actions can be completed in the time required, and the ability to recover from credible errors, the change in description of assumed manual operator actions did not result in a USQ. Thus, we believe the change was made properly under the provisions of 10 CFR 50.59 without submitting a license amendment to the NRC in accordance with 10 CFR 50.90.

<u>References</u>

1. UFSAR Draft Revision Package Number 7-257
2. Problem Identification Form (PIF) Number B1997-03693
3. Byron/Braidwood UFSAR, Revision 8
4. Byron/Braidwood FSAR, Amendment 46, January 1985
5. Westinghouse Standard System Design Criteria 1.14, NSSS Containment Isolation
6. MOV Design Basis Document, 20897-DB-BYR-CC Revision 1, Component Cooling Water System
7. Westinghouse Fluid System Engineering Memorandum, FSE-CW&BOP-91-1537, dated April 24, 1991

ATTACHMENT A

8. Byron Operability Assessment 97-71, completed November 26, 1997
9. Byron Abnormal Operating Procedures 1/2BOA PRI-6, "Component Cooling Malfunction"
10. NRC Information Notice 97-78, "Crediting of Operator Actions in Place of Automatic Actions and Modification of Operator Actions, Including Response Times", dated October 23, 1997
11. Calculation 3C8-1281-001, Revision 12, "Auxiliary Building Flood Level Calculations"