RS-04-096

June 30, 2004

U. S. Nuclear Regulatory Commission
ATTN: Document Control Desk
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Dresden Nuclear Power Station, Units 2 and 3
Facility Operating License Nos. DPR-19 and DPR-25
NRC Docket Nos. 50-237 and 50-249

Quad Cities Nuclear Power Station, Units 1 and 2
Facility Operating License Nos. DPR-29 and DPR-30
NRC Docket Nos. 50-254 and 50-265

Subject: Additional Information Supporting the Request for License Amendment Related to Application of Alternative Source Term

References:

1. Letter from K. R. Jury (Exelon Generation Company, LLC) to U. S. NRC, "Request for License Amendments Related to Application of Alternative Source Term," dated October 10, 2002

2. Letter from L. W. Rossbach (U. S. NRC) to C. M. Crane (Exelon Generation Company, LLC), "Dresden Nuclear Power Station, Units 2 and 3, and Quad Cities Nuclear Power Station, Units 1 and 2 – Request for Additional Information (RAI) Regarding Alternative Source Term Amendment Request (TAC Nos. MB6530, MB6531, MB6532, and MB6533)," dated June 3, 2004

In Reference 1, Exelon Generation Company, LLC (EGC) requested an amendment to the facility operating licenses for Dresden Nuclear Power Station, Units 2 and 3, and Quad Cities Nuclear Power Station, Units 1 and 2. The proposed changes support application of an alternative source term methodology. To support the proposed changes, EGC evaluated the four design basis accidents (i.e., loss-of-coolant, main steam line break, fuel handling, and control rod drop accidents) that could potentially result in main control room or offsite doses.

In Reference 2, the NRC requested additional information related to crediting the standby liquid control system for pH control of the suppression pool. The attachment provides the requested information.

EGC has reviewed the information supporting a finding of no significant hazards consideration that was previously provided to the NRC in Attachment C of Reference 1. The

supplemental information provided in this submittal does not affect the bases for concluding that the proposed license amendment does not involve a significant hazards consideration.

If you have any questions concerning this letter, please contact Mr. Kenneth M. Nicely at (630) 657-2803.

I declare under penalty of perjury that the foregoing is true and correct. Executed on the 30th day of June 2004.

Respectfully,

Patrick R. Simpson
Manager – Licensing

Attachment:
Response to Request for Additional Information

cc: Regional Administrator - NRC Region III
NRC Senior Resident Inspector - Dresden Nuclear Power Station
NRC Senior Resident Inspector - Quad Cities Nuclear Power Station
Illinois Emergency Management Agency - Division of Nuclear Safety

ATTACHMENT
Response to Request for Additional Information

Request 1

1. Please identify whether the SLC system is classified as a safety-related system as defined in 10 CFR 50.2, and whether the system satisfies the regulatory requirements for such systems. If the SLC system is not classified as safety-related, please provide the information requested in Items 1.1 to 1.5 below to show that the SLC system is comparable to a system classified as safety-related. If any item is answered in the negative, please explain why the SLC system should be found acceptable for pH control agent injection.

    1.1 Is the SLC system provided with standby AC power supplemented by the emergency diesel generators?

    1.2 Is the SLC system seismically qualified in accordance with Regulatory Guide (RG) 1.29 and Appendix A to 10 CFR Part 100 (or equivalent used for original licensing)?

    1.3 Is the SLC system incorporated into the plant's ASME Code inservice inspection and inservice testing programs based upon the plant's code of record (10 CFR 50.55a)?

    1.4 Is the SLC system incorporated into the plant's Maintenance Rule program consistent with 10 CFR 50.65?

    1.5 Does the SLC system meet 10 CFR 50.49 and Appendix A to 10 CFR 50 (General Design Criteria 4, or equivalent used for original licensing)?

Response to Request 1

The standby liquid control (SLC) systems at Dresden Nuclear Power Station (DNPS) and Quad Cities Nuclear Power Station (QCNPS) are classified as safety-related as defined in 10 CFR 50.2, and satisfy the regulatory requirements for such systems. The SLC systems at DNPS and QCNPS meet the following items.

1.1 As stated in Attachment B of Reference 1, the SLC systems at DNPS and QCNPS are powered by emergency power. This includes the capability to supply power from the emergency diesel generators.

1.2 The SLC components required for the alternative source term (AST) function are seismically qualified.

1.3 The SLC system is incorporated into the DNPS and QCNPS American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code inservice inspection (ISI) and inservice testing (IST) programs as required by 10 CFR 50.55a, "Codes and standards."

1.4 The SLC system is incorporated into the Maintenance Rule programs at DNPS and QCNPS, consistent with 10 CFR 50.65, "Requirements for monitoring the effectiveness of maintenance at nuclear power plants."

1.5 The SLC system is not currently subject to the requirements of 10 CFR 50.49, "Environmental qualification of electric equipment important to safety for nuclear power plants," since it does not provide any function to accidents that would cause a harsh environment. A review of the zone maps contained in Figures 3.11-5 and 3.11-1 of the DNPS and QCNPS Updated Final Safety Analysis Reports (UFSARs), respectively, shows that the floor where the SLC pumps and associated electrical equipment is located is a mild environment. The motor control centers (MCCs) that provide power for the SLC pumps are included in the Environmental Qualification program. Therefore, the SLC system can be considered to comply with the requirements of 10 CFR 50.49 for the new AST function.

DNPS and QCNPS were designed and built prior to the implementation of 10 CFR 50, Appendix A, "General Design Criteria for Nuclear Power Plants." The SLC systems were designed to General Electric Specification APED 4600, "Nuclear Safety Criteria for Boiling Water Reactors." A comparison of APED 4600 to the draft general design criteria (GDC) was performed and concluded the SLC design implemented the intent of Appendix A. This review was documented in Appendix B of the Final Safety Analysis Report (FSAR) and is currently in Section 3.1 of the UFSAR. The design basis of the SLC is described in Section 9.3.5 of the UFSAR.

<u>Request 2</u>

2. Please describe proposed changes to plant procedures that implement SLC sodium pentaborate injection as a pH control additive. In addition, please address Items 2.1 to 2.5 below in your response. If any item is answered in the negative, please explain why the SLC system should be found acceptable for pH control additive injection.

2.1 Are the SLC injection steps part of a safety-related plant procedure?

2.2 Are the entry conditions for the SLC injection procedure steps symptoms of imminent or actual core damage?

2.3 Does the instrumentation cited in the procedure entry conditions meet the quality requirements for a Type E variable as defined in RG 1.97 Tables 1 and 2?

2.4 Have plant personnel received initial and periodic refresher training in the SLC injection procedure?

2.5 Have other plant procedures (e.g., Emergency Response Guildelines/Severe Accident Guidelines) that call for termination of SLC as a reactivity control measure been appropriately revised to prevent blocking of SLC injection as pH control measure? (For example, the override before Step RC/Q-1, *"If while executing the following steps:....It has been determined that the reactor will*

remain shutdown under all conditions without boron, terminate boron injection and...")

Response to Request 2

Details regarding the plant procedures that implement SLC sodium pentaborate injection as a pH control additive were provided to the NRC in Reference 2. Exelon Generation Company, LLC (EGC) has evaluated the need for procedure changes related to SLC injection. As described in Reference 2, when the DNPS emergency operating procedures (EOPs) direct use of alternate injection systems, the operator is referenced to DEOP 500-3, "Alternate Water Injection Systems," for the specific steps to perform. This procedure currently contains steps that may prevent the operator from injecting boron or meeting the assumed concentration when manually initiating SLC. Therefore, as stated in Reference 2, DEOP 500-3 will be revised prior to implementation of the AST amendment to specifically direct boron injection without dilution until the required amount of boron is injected for pH control following a loss-of-coolant accident (LOCA).

No changes to QCNPS procedures are needed since the cues and required actions for SLC initiation are not changing for implementation of AST.

2.1 The procedures described in Attachment 2 to Reference 2 that implement SLC sodium pentaborate injection are controlled procedures that are prepared, reviewed, approved, and used in accordance with the EGC Quality Assurance Topical Report.

2.2 The specific cues that alert the operator to take actions to inject SLC were provided to the NRC in Attachment 2 to Reference 2. Specifically, DNPS EOP DEOP 100, "RPV Control," and QCNPS EOP QGA 100, "RPV Control," are entered with reactor pressure vessel (RPV) water level below the scram setpoint, RPV pressure above the high pressure scram setpoint, drywell pressure above the scram setpoint, or reactor power above the low power alarm with a scram signal present. The RPV low level and the drywell high pressure entry conditions ensure that these procedures are entered for a LOCA.

Attachment 2 to Reference 2 also described the entry conditions for the severe accident management guidelines (SAMGs). The entry conditions to the SAMGs are listed within the EOPs. When conditions defined in the EOPs indicate that adequate core cooling cannot be restored and maintained, for any reason, then SAMG entry is directed. Adequate core cooling is defined as any of the following: core submersion, spray cooling, steam cooling with injection, or steam cooling without injection. In the reactor power leg of SAMG-2, "RPV, Containment, and Radioactivity Release Control," SLC initiation is directed upon entry to the SAMGs with no qualifying or conditional restrictions.

The specific cues described above that alert the operator to take actions to inject SLC are symptoms of imminent core damage.

2.3 The drywell high pressure and reactor water level instruments meet the quality requirements for Type A or B variables as defined in Regulatory Guide 1.97, Tables 1 and 2.

2.4 Licensed operators have received initial training on the EOPs and SAMGs, and will continue to receive periodic refresher training. Additionally, as discussed above, DNPS procedure DEOP 500-3 will be revised prior to implementation of the AST amendment to specifically direct boron injection without dilution until the required amount of boron is injected for pH control following a LOCA. As stated in Reference 2, DNPS will train operators on the revised procedure.

2.5 At QCNPS, when the EOPs or SAMGs direct SLC initiation, the operator is referenced to operating procedure QCOP 1100-02, "Injection of Standby Liquid Control," for the specific steps to perform. QCOP 1100-02 contains steps that direct the operator to inject the entire SLC tank if a LOCA is in progress, which ensures that SLC injection will not be terminated during a LOCA.

As described above and in Reference 2, DNPS procedure DEOP 500-3 will be revised prior to implementation of the AST amendment to specifically direct boron injection without dilution until the required amount of boron is injected for pH control following a LOCA.

Request 3

3. Please provide a description of the analysis assumptions, inputs, methods, and results that show that a sufficient quantity of sodium pentaborate can be injected to raise and maintain the suppression pool greater than pH 7 within 24 hours of the start of the event. (See also Position 2 of Appendix A to RG 1.183.) In your response, please discuss the adequacy of recirculation of suppression pool liquid via emergency core cooling systems through the reactor vessel and the break location and back to the suppression pool in meeting the transport and mixing assumptions in the chemical analyses. Assume a large break LOCA.

Response to Request 3

In Reference 3, EGC responded to an NRC request for additional information regarding the control of suppression pool pH for the 30-day period after a large-break LOCA. Attachment 3 of Reference 3 provided the calculation that documents the analysis assumptions, inputs, and methods.

On June 23, 2004, the NRC amended Request 3 to request additional details regarding the adequacy of recirculation of suppression pool liquid via emergency core cooling systems through the reactor vessel and the break location and back to the suppression pool. In a telephone call with the NRC on June 29, 2004, EGC agreed to address the adequacy of recirculation in a separate submittal by July 9, 2004.

Request 4

4. Please show that the SLC system has suitable redundancy in components and features to assure that for onsite or offsite electric power operation its safety function of injecting sodium pentaborate for the purpose of suppression pool pH control can be accomplished assuming a single failure. For this purpose, the check value is considered an active device since the check valve must open to inject sodium pentaborate. If the SLC system cannot be considered redundant with respect to its active components, the licensee should implement one of the three options described below, providing the information specified for that option for staff review.

4.1 Option 1 Show acceptable quality and reliability of the non-redundant active components and/or compensatory actions in the event of failure of the non-redundant active components. If you choose this option, please provide the following information to justify the lack of redundancy of active components in the SLC system:

4.1.1 Identify the non-redundant active components in the SLC system and provide their make, manufacturer, and model number.

4.1.2 Provide the design-basis conditions for the component and the environmental and seismic conditions under which the component may be required to operate during a design-basis accident. Environmental conditions include design-basis pressure, temperature, relative humidity and radiation fields.

4.1.3 Indicate whether the component was purchased in accordance with Appendix B to 10 CFR Part 50. If the component was not purchased in accordance with Appendix B, provide information on the quality standards under which it was purchased.

4.1.4 Provide the performance history of the component both at the licensee's facility and in industry databases such as equipment performance and information exchange system (EPIX) and nuclear plant reliability data system (NPRDS).

4.1.5 Provide a description of the component's inspection and testing program, including standards, frequency, and acceptance criteria.

4.1.6 Indicate potential compensating actions that could be taken within an acceptable time period to address the failure of the component. An example of a compensating action might be the ability to jumper a switch in the control room to overcome its failure. In your response please consider the availability of compensating actions and the likelihood of successful injection of the sodium pentaborate when non-redundant active components fail to perform their intended functions.

4.2 Option 2 Provide for an alternative success path for injecting chemicals into the suppression pool. If you chose this option, please provide the following information.

4.2.1 Provide a description of the alternative injection path, its capabilities for performing the pH control function, and its quality characteristics.

4.2.2 Do the components which make up the alternative path meet the same quality characteristics required of the SLC system as described in Items 1.1 to 1.5, 2 and 3 above?

4.2.3 Does the alternate injection path require actions to be taken in areas outside the control room? How accessible will these areas be? What additional personnel would be required?

4.3 Option 3 Show that 10 CFR 50.67 dose criteria are met even if pH is not controlled. If you chose this option, demonstrate through analyses that the projected accident doses will continue to meet the criteria of 10 CFR 50.67 assuming that the suppression pool pH is not controlled. The dissolution of Cesium Iodide and its re-evolution from the suppression pool as elemental iodine must be evaluated by a suitably conservative methodology. The analysis of iodine speciation should be provided for staff review. The analysis documentation should include a detailed description and justification of the analysis assumptions, inputs, methods, and results. The resulting iodine speciation should be incorporated into the dose analyses The calculation may take credit for the mitigating capabilities of other equipment, for example the standby gas treatment system (SGTS), if such equipment would be available. A description of the dose analysis assumptions, inputs, methods, and results should be provided. Licensees proposing this approach should recognize that this option will incur longer staff review times and will likely involve fee-billable support from national laboratories.

Response to Request 4

The DNPS and QCNPS SLC systems can be considered redundant with respect to its active components, except as outlined below. This limited lack of redundancy is offset as described in the justifications provided. The following information is provided in accordance with Option 1 above. Therefore, Options 2 and 3 are not applicable to DNPS and QCNPS.

4.1.1 The non-redundant active components of the SLC system are (1) the check valves (i.e., two in series) located on the SLC injection line, and (2) the SLC initiation control switch in the main control room. Details regarding these components are listed below.

DNPS				
Unit	**Component**	**Description**	**Manufacturer**	**Model #**
2	2-1101-15	Piston Check Valve	Rockwell Edwards	D36174(F316)T4
2	2-1101-16	Lift Check Valve	Crane Nuclear, Inc.	3888U
3	3-1101-15	Lift Check Valve	Crane Nuclear, Inc.	3888U
3	3-1101-16	Lift Check Valve	Crane Nuclear, Inc.	3888U
2	2-1130-301	Control Switch with Key Lock	General Electric	SB-1
3	3-1130-301	Control Switch with Key Lock	General Electric	SB-1

QCNPS				
Unit	**Component**	**Description**	**Manufacturer**	**Model**
1	1-1101-15	Spring Loaded Check Valve	Rockwell Edwards	D36174/44A1C
1	1-1101-16	Lift Check Valve	Crane Nuclear, Inc.	3888U-1.5-1250-SR-A-N
2	2-1101-15	Spring Loaded Check Valve	Rockwell Edwards	33AFO
2	2-1101-16	Spring Loaded Check Valve	Rockwell Edwards	34AFO
1	1-1130-0301	Control Switch with Key Lock	General Electric	Q16SB1EB95SSM4C43
2	2-1130-0301	Control Switch with Key Lock	General Electric	Q16SB1EB95SSM4C43

4.1.2 The environmental data for the control switch and check valves are listed in the following tables. These components are seismically qualified.

DNPS				
Component	**Pressure (psia)**	**Temperature (°F)**	**Relative Humidity (%)**	**Radiation (1 hour dose in Rads)**
2-1101-15	63	295	100	2.8E07
2-1101-16	14.7	118	100	<1.0E04
3-1101-15	63	295	100	2.8E07
3-1101-16	14.7	118	100	<1.0E04
2-1130-301 (CS)	14.7	95	90	<1.0E04
3-1130-301 (CS)	14.7	95	90	<1.0E04

QCNPS				
Component	**Pressure (psia)**	**Temperature (°F)**	**Relative Humidity (%)**	**Radiation (1 hour dose in Rads)**
1-1101-15	62	294	100	2.8E07
1-1101-16	15.7	214	100	<1.0E04
2-1101-15	62	294	100	2.8E07
2-1101-16	14.8	201	100	4.3E05
1-1130-0301 (CS)	14.7	80	90	<1.0E04
2-1130-0301 (CS)	14.7	80	90	<1.0E04

4.1.3 The SLC system was designed and installed prior to 10 CFR 50, Appendix B. The system was designed to General Electric Specification GE APED 4600. The piping and valves were installed per American Standards Association (ASA) B31.1, "Code for Pressure Piping." The SLC system is safety related, and all repair, replacement, and modification work is performed in accordance with the EGC Quality Assurance Topical Report, which implements 10 CFR 50, Appendix B.

4.1.4 The containment isolation check valves are stainless steel 1½ inch valves, mounted horizontally in the injection line. For an anticipated transient without scram event, the containment isolation check valves are designed to open against full reactor pressure. For the AST function, the system operating requirements are reduced since the reactor pressure is much lower following a design basis LOCA.

In accordance with the IST program, these check valves are inspected and tested during scheduled refueling outages. Technical Specifications require a system flow test to the vessel, which demonstrates the operability of the integrated system, at least once every 24 months on a staggered test basis. A review of the IST databases at DNPS and QCNPS was performed, and no instances of the SLC check valves failing to open were identified.

Reference 4 documents a review and evaluation of check valve failures. The review found that the overall failure rate for all check valves was 0.00996 per year. The failure rate of the check valves < 2 inches was 0.00706 per year. In the distribution of failures, the restricted flow and failed closed modes were each responsible for only 7 percent of the valve failures. In the < 2 inches size group, the fraction of stuck closed failures was approximately 0.15 resulting in a failed closed failure rate of 0.001059 per year. This data indicates that check valves < 2 inches are very reliable.

A review of the EPIX database was performed, and no failures of the check valves failing to open were identified.

Additionally, no failures of the SLC control switch were identified in the EPIX database. Previous SLC system functional testing at DNPS and QCNPS has not identified any problems with the switch that would preclude SLC system injection. Based on the above, any failure that would prevent the start of at least one pump is highly improbable and EGC expects the SLC control switch to operate when required.

Based on this information, the stuck closed failure of these valves in the common SLC system discharge line, and the failure of the SLC control switch are highly unlikely.

4.1.5 The check valves listed in the response to request 4.1.1 above are tested in accordance with Technical Specifications and IST program requirements. These valves have an open function to support injection of sodium pentaborate and a close function for primary containment isolation.

SR 3.1.7.8 requires verification of flow through one SLC subsystem from a pump into the RPV every 24 months on a staggered test basis. This allows the system to be tested for complete continuity during a shutdown when demineralized water can be pumped into the RPV. During the test, one of the subsystems, including an explosive valve, is initiated, and it is verified that a flow path from the pump to the RPV is available. This testing necessitates replacement of the explosive charge in the shear plug valves. Both complete flow paths are tested every 48 months. This test verifies the flowpath to the RPV and particularly the proper operation of the check valves in the drywell and the control switch in the control room.

Finally, SR 3.1.7.9 requires verification that all heat traced piping between the storage tank and pump suction is unblocked every 24 months and once within 24 hours after piping temperature is restored within limits.

A leakage test is performed on the containment isolation valves (i.e., two check valves located in series near the drywell penetration) to verify containment valve isolation capability. At DNPS, the test is performed every refueling outage, and at QCNPS, the test frequency is performance-based and may vary. At DNPS, the current required action range is leakage greater than 0.75 gpm. At QCNPS, the current required action range is leakage greater than 15 scfh.

Additionally, at QCNPS the open function of the check valves are also tested under an operating surveillance. This surveillance is performed during every cold shutdown of sufficient duration as part of the IST program. The acceptance criterion for this surveillance is flow into the RPV as indicated by either the flow light being lit or a noticeable increase in reactor water level.

In summary, the required Technical Specifications and ASME Code Section XI testing provide assurance of a high degree of system reliability and confidence that the system injection function would perform satisfactorily if called upon following a design basis LOCA.

4.1.6 As discussed above, both the check valves and control switches have indicated a high degree of reliability. Due to the inherent reliability of these components, no compensatory actions are needed. However, given the location of the control switch, a jumper can easily be installed to bypass a failed switch and initiate SLC injection. Given the mission time of 24 hours, there is a high likelihood of successful injection of the sodium pentaborate if the control switch were to fail.

Additionally, alternative means of injecting boron are available using the condensate system at DNPS and the reactor water cleanup system at QCNPS. Although these systems could potentially be available for use, the AST analysis for DNPS and QCNPS do not credit these alternative methods for pH control. EGC believes compensating actions are not warranted due to the reliability of the non-redundant components of the SLC system.

References

1. Letter from K. R. Jury to U. S. NRC, "Request for License Amendments Related to Application of Alternative Source Term," dated October 10, 2002

2. Letter from P. R. Simpson to U. S. NRC, "Additional Information Supporting the Request for License Amendment Related to Application of Alternative Source Term," dated March 21, 2003

3. Letter from P. R. Simpson to U. S. NRC, "Additional Information Supporting the Request for License Amendment Related to Application of Alternative Source Term," dated March 28, 2003

4. NUREG/CR-5944, "A Characterization of Check Valve Degradation and Failure Experience in the Nuclear Power Industry," dated September 1993