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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

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

In the referenced letter, 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.

On July 24, 2003, the NRC requested additional information to support review of the referenced letter. 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 the referenced letter. 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.

A001

If you have any questions or require additional information, please contact Mr. Kenneth M. Nicely at (630) 657-2803.

I declare under penalty of perjury that the foregoing is true and correct.

Respectfully,

9/15/03
Executed on

Patrick R. Simpson
Manager – Licensing

Attachments:

1. Response to Request for Additional Information
2. Dresden Nuclear Power Station Marked-Up Technical Specifications for Proposed Changes
3. Quad Cities Nuclear Power Station Marked-Up Technical Specifications for Proposed Changes
4. Dresden Nuclear Power Station Retyped Technical Specifications for Proposed Changes
5. Quad Cities Nuclear Power Station Retyped Technical Specifications for Proposed Changes
6. Calculation DRE01-0040, "Site Boundary and Control Room Doses Following a Loss of Coolant Accident Using Alternative Source Terms," Revision 0, dated August 22, 2002
7. Calculation QDC-0000-N-1117, "Site Boundary and Control Room Doses Following a Loss of Coolant Accident Using Alternative Source Terms," Revision 0, dated August 22, 2002

cc: Regional Administrator - NRC Region III
NRC Senior Resident Inspector - Dresden Nuclear Power Station
NRC Senior Resident Inspector - Quad Cities Nuclear Power Station
Office of Nuclear Facility Safety - Illinois Department of Nuclear Safety

ATTACHMENT 1
Response to Request for Additional Information

I. Description of Proposed Changes

NRC Request

Proposed changes to acceptance criteria for methyl iodine penetration for laboratory test of charcoal for the SGTS and the CREVS are unacceptable. First, they are inconsistent with the guidance of Generic Letter 99-02. Second, the assumed adsorber efficiency is inconsistent with the guidance in Regulatory Guide 1.52. Revise the acceptance criteria and modify the assumed adsorber efficiencies in the analyses.

Response

The proposed changes to the acceptance criteria for methyl iodide penetration for laboratory test of a sample of the charcoal adsorber are being withdrawn. In addition, the following proposed changes for Quad Cities Nuclear Power Station (QCNPS) submitted in Reference 1 are being withdrawn: (1) revise Technical Specification (TS) Surveillance Requirement (SR) 3.6.4.3.1 to remove the requirement for operating the Standby Gas Treatment (SGT) System heaters during performance of the surveillance to operate each SGT subsystem for $\geq$ 10 continuous hours, (2) revise the test conditions in Section 5.5.7.c to increase the relative humidity from 70% to 95% for methyl iodide penetration testing, (3) revise Section 5.5.7.2 to delete the requirement for periodic SGT System heater testing.

II. Definition of Dose Equivalent ^{131}I

NRC Request

The proposed change in the a definition of Dose Equivalent ^{131}I adds committed effective dose equivalents dose factors for inhalation from Federal Guidance Report 11 to an existing definition that contains as acceptable dose conversion factors from Table E-7 of Regulatory Guide 1.109, Rev 1, Table III of TID 14844, and pages 192-212 of Supplement 1 to Part 1 of ICRP 30. This range of definitions has significantly different consequences when determining the curie content in reactor coolant. This would affect the consequences of the MSLB accident. Since the alternate source term is being adopted, the definition of Dose Equivalent ^{131}I should only include the inhalation committed effective dose equivalent for inhalation from Federal Guidance Report 11.

Response

The proposed change to revise the definition of dose equivalent I-131 in TS Section 1.1 has been modified to only include the inhalation committed effective dose equivalent for inhalation from Federal Guidance Report 11. Revised TS markups for Dresden Nuclear Power Station (DNPS) and QCNPS are provided in Attachments 2 and 3, respectively. Retyped TS pages for DNPS and QCNPS are provided in Attachments 4 and 5, respectively.

III. Safety Analysis

NRC Request 1

Section 2.2.2 discusses the post-accident dose rate in the control room and adjacent areas due to the shine from the refueling floor airborne source and an additional low-level post-accident external gamma ray dose rate component in the control room and adjacent areas due to shine from the refueling floor noble gas airborne source. These dose rates were calculated in the period 1980-1982. What is the difference between these two sources? Do the calculated dose rates include the contributions from the isotopes associated with the AST or are they only iodines and noble gases? If the calculations do not, provide an explanation why, for purposes of this amendment, the calculations should not include the AST isotopes. Provide the list of accidents for which these dose rates apply and the basis for including these dose rates only for those accidents. (Pg. 20, 28 & 29)

Response

Reference 1 provided control room post loss-of-coolant accident (LOCA) doses due to radioactivity within the control room in Tables 10a and 10b. A footnote to those tables states "The dose from external sources (e.g., refuel floor, passing cloud, and radioactivity accumulated on CREV/SGT System filters is expected to be much less than 0.5 rem TEDE."

Attachment A, Section 2.2.2, of Reference 1 states that these dose rates were calculated in the immediate post-TMI-2 period (i.e., 1980-1982).

A detailed description of the modeling used to determine the post-LOCA dose rates within the stations is provided in Updated Final Safety Analysis Report (UFSAR) Appendix 12A. Specific modeling details are provided in Section 12A.3. Section 12A.7 provides specific results for dose and dose rate to the control room. For QCNPS, the post-LOCA dose from external sources is 0.057 rad. For DNPS, the post-LOCA dose from external sources is 0.101 rad. These values are much less that than the 0.5 rem total effective dose equivalent (TEDE) conservatively assumed in the alternative source term (AST) analyses.

Concrete shielding provides the control room with additional protection against external sources, as shown in QCNPS UFSAR Section 6.4.2.5 and Figures 6.4-3 thru 6.4-9, and DNPS UFSAR Section 6.4.2.5, Section 12.3.2.2.4 and Figures 12.3-1 thru 12.3-5.

As noted above, the post-LOCA doses from external sources are much less than the bounding value of 0.5 rem TEDE assumed for the AST analyses. These external source doses were determined in the 1980-1982 timeframe using the TID-14844 source term. The doses from external sources were not recalculated using the AST source terms. The newer AST source term contains more nuclides than the TID-14844 list of noble gases and iodines. However removal mechanisms assumed in the AST analyses (i.e., plate-out and gravitational settling) will actually reduce the contributions due to radioiodines. Any additional contribution to the external dose from the remaining nuclides, in particulate or aerosol form, it is expected to be minimal. The net effect of

AST source terms and AST modeling is expected to result in a minimal change in dose from that calculated with the TID source term. Adequate margin between the calculated values (i.e., 0.101 and 0.057 rad for DNPS and QCNPS, respectively) and the assumed value of 0.5 rem TEDE is available to accommodate any increase in dose. If recalculated using the total concrete thicknesses identified above, the external dose would be reduced even further.

The additional post-LOCA doses from external sources were considered only for the LOCA. The consequences of the other accidents are lower than for the LOCA; therefore, the impact to the other accidents is expected to have a negligible effect.

In conclusion, the post-LOCA doses from external sources, if recalculated using the 60-isotope AST source terms, would be well within the 0.5 rem TEDE value assumed in the AST submittal.

NRC Request 2

Provide the analysis which supports the conclusion that suppression pool pH will remain above 7. (Pg. 20)

Response

The calculation supporting the conclusion that suppression pool pH will remain above 7 was submitted to the NRC in Reference 2.

NRC Request 3

It is stated that the impact of toxic gases on control room operators is limited because the control room is maintained at a positive pressure of 1/8 inch w.g. with respect to adjacent areas. Explain how the Dresden and Quad Cities designs accomplish this and how this is reflected in the hazardous chemical analyses for the two plants. (Pg. 22)

Response

Adverse interactions between the control room emergency zone and adjacent zones that may allow the transfer of toxic or radiological gases into the control room are minimized by maintaining the control room at a positive pressure, relative to adjacent areas, during normal operations. During the pressurization mode of operation, the CREV System maintains a positive pressure of at least 1/8 inch water gauge.

During a toxic gas event, the control room would be isolated by placing the CREV System in isolation mode. This would close outside air isolation dampers and place the system into full recirculation mode. Adverse interactions between the control room and adjacent areas that may allow the transfer of radiological gases into the control room are minimized by isolating outside air when the toxic chemical is detected. After the isolation, the control room is not maintained at a positive pressure.

NRC Request 4

What is the maximum containment pressure expected to be for the LOCA, rod drop and MSLB accidents? What pressure is the containment designed for? (Pg. 23)

Response

The peak calculated primary containment internal pressure for the design basis LOCA is 43.9 psig. The main steam line break (MSLB) accident analysis assumes that one main steam line outside containment is severed, thus containment pressure is not impacted by the MSLB accident. The control rod drop accident (CRDA) analysis does not assume a concurrent LOCA and the CRDA itself does not result in a pressurization of containment. Therefore, there is no impact to containment internal pressure for a CRDA. The containment design pressure is 62 psig.

NRC Request 5

Why are the MSIVs tested at 25 psig and 48 psig? (Pg. 23)

Response

Main steam isolation valves (MSIVs) are tested at 25 psig, not 48 psig. The LOCA analysis calculated MSIV leakage based on a pressure of 48 psig. Since the MSIVs are tested at 25 psig, not 48 psig, the proposed TS MSIV leakage acceptance criteria was scaled to 25 psig.

The NRC approved exemptions for DNPS and QCNPS to allow MSIV testing at 25 psig in References 3 and 4, respectively.

NRC Request 6

Provide the calculation and associated assumptions which conclude that the MSIV leakage would not exit the main steam line prior to 40 minutes following the LOCA? (Pg 23)

Response

Attachments 6 and 7 provide the LOCA calculations, with associated assumptions, for DNPS and QCNPS, respectively. In the calculation, it is estimated that environmental releases due to MSIV leakage will not occur until well over one hour. However, the analysis conservatively assumes that holdup of activity releases due to MSIV leakage in main steam lines (MSLs) is limited to 40 minutes (i.e., the time for CREV to be initiated).

NRC Request 7

It is stated that during the first 40 minutes following a LOCA the flow rate to the control room is 2000 cfm ± 10% through the CREV system. After 40 minutes the flow arrangement remains unchanged but now there is an additional 600 cfm of unfiltered inleakage into the CRE. These statements appear to be in conflict with other portions of

the amendment request. Other statements in the amendment request indicate that it is only at Quad Cities that the CREV system is initiated automatically upon receipt of a High Radiation signal on the refueling floor area or in the reactor building ventilation exhaust. For Dresden, operator actions are required to place the CREV system into operation. Clarify the actual operating configuration for the CRE following a LOCA. (Pg. 24)

Response

The CREV System at QCNPS automatically isolates the CREV System boundary upon receipt of a valid LOCA signal. Isolation of the CREV System boundary at DNPS requires manual action. At both DNPS and QCNPS, manual actions are required to place the CREV System into operation.

The 600 cfm unfiltered inleakage from unidentified sources is assumed for both the first 40 minutes following a LOCA and after 40 minutes. The CREV System provides 2000 cfm +/- 10% to the control room envelope. The LOCA calculation was performed assuming that in the first 40 minutes, the control room envelope receives 2200 cfm of unfiltered air through the CREV System and 600 cfm of unfiltered air through other unidentified sources. After 40 minutes, the analysis assumes that the CREV System provides 1800 cfm of filtered air to the control room envelope, and an additional 600 cfm of unfiltered air is assumed to enter the control room envelope from other unidentified sources.

Because CREV System filtration is only assumed to occur for the LOCA event (i.e., a 30 day event), any small change to the control room intake flow rate in the first 40 minutes does not significantly impact the results of the analysis.

The control room heating, ventilation, and air conditioning (HVAC) system consists of two independent HVAC subsystems sharing some common ductwork. The subsystems include one multizone subsystem (i.e., "A" Train) and one single zone subsystem (i.e., "B" Train). "A" Train is the primary temperature control and air distribution subsystem for the control room emergency zone. "B" Train is a backup subsystem that serves the control room emergency zone when "A" Train is not available.

The assumed 600 cfm of unfiltered inleakage is conservative for both DNPS and QCNPS. This is based on tracer gas testing and the CREV System design. At both DNPS and QCNPS, tracer gas testing of the control room boundary has been completed in both the "A" train filtration mode and "B" train filtration mode. The tracer gas testing results are summarized in the following table.

Train	DNPS Inleakage	QCNPS Inleakage
"A" Train (filtration mode)	162 +/- 91 cfm	222 +/- 75 cfm
"B" Train (filtration mode)	156 +/- 86 cfm	88 +/- 75 cfm

Tracer gas testing was not performed in the normal mode of CREV System operation. However, the inleakage during the normal mode would be lower than during the filtration mode and is already included in the existing analysis as described below.

In the normal mode, the ductwork and outside air damper is directly adjacent to the air handling unit (i.e., within 36 inches). In the filtration mode, the damper is closed when the control room is isolated, and airflow is directed to the air handling unit fan by the return air vane axial fan and the suction of the air handling unit centrifugal fan. The negative pressure exerted on the damper and ductwork tying it to the air handling unit would be lower in the normal mode because the damper would be open allowing free flow of makeup air to the air handling unit. Also, the negative pressure of the associated ductwork tying it to the air handling unit would be lower since makeup would be freely accepted since airflow follows the path of least resistance (i.e., open damper versus closed damper).

As noted above, tracer gas testing was performed in the "A" Train filtration mode of CREV System operation, not the normal mode. The inlet and exhaust ducts are the only additional regions to be tested in the normal mode. In the normal mode, inleakage at the inlet would be no different than the unfiltered outside air and would enter the boundary as a part of the "A" Train normal flow. Any inleakage that occurs at the exhaust section of the duct would be swept to outside air. Therefore, the impact is minimal because the test results are unaffected by inleakage in these areas, although the "A" Train (i.e., normal mode) inlet and exhaust have not been tracer gas tested.

NRC Request 8

There needs to be an inleakage number for the CRE during all accidents and a basis for that number. It is indicated that during the first 40 minutes following the LOCA the normal control room ventilation system is operating. During the operation of the normal system, unfiltered inleakage may also be occurring. It appears that no number has been provided. Provide the value for inleakage during the normal control room system's operation and the basis for that number. (Pg. 24)

Response

As noted above, the CREV System provides 2000 cfm +/- 10% to the control room envelope. The CRDA calculation was performed assuming that, for the duration of the accident, the control room envelope receives 2200 cfm through the ventilation system (i.e., unfiltered) and 600 cfm through other unidentified sources (i.e., unfiltered). For the FHA, it is conservatively assumed that the control room envelope receives 4000 cfm of unfiltered air. For the MSLB, the dose is calculated for an individual outside of the control room, thereby assuming no control room protection.

NRC Request 9

What is the removal mechanism for MSIV leakage in the vertical control volume? Provide the RADTRAD results for the MSIV leakage source term. (Pg. 24)

Response

The natural deposition constants for the MSLs are given in design inputs 14 and 15 of the DNPS and QCNPS LOCA calculations (i.e., Attachments 6 and 7). The constants

for aerosols account for gravitation settling and are applicable to horizontal pipe runs only. The constants for elemental iodine apply to both horizontal and vertical pipe runs and include plate-out. Tables 3 and 6 of the calculations provide RADTRAD results for the MSIV leakage source term.

NRC Request 10

How soon does the SGTS establish a negative 0.5 inch w.g. in the reactor building following a LOCA and rod drop accident? (Pg. 28)

Response

As described in the response to NRC Request 10 under Section IV, "Safety Analysis," the reactor building is maintained at a negative pressure during normal operation. At the beginning of the event, SGT automatically starts and maintains negative pressure. Therefore, the reactor building pressure remains negative and the analysis does not assume an explicit drawdown time.

AST assumes that there is insignificant release to the environment for the first two minutes during the coolant activity release phase. The onset of the gap activity release phase is two minutes after the initiation of the accident. The automatic start of the SGT System will ensure that the reactor building negative pressure will be maintained beyond the two-minute coolant activity release period.

NRC Request 11

It is indicated that in the event of a fuel handling accident the release will occur from the reactor building vent stack. Is the ventilation system associated with the reactor building vent stack safety related and will it operate in the event of a loss of offsite power? If no ventilation system is operating during the course of the accident, should it be assumed that the material will diffuse from the building? How is it assured that the release will be monitored? (Pg 30)

Response

DNPS and QCNPS each include a common refueling floor for both units at the upper elevation of the reactor building. This area is common to the stations' two units and has a shared ventilation system that exhausts both units to a short stack located on top of the reactor building. Under accident conditions, the reactor buildings are exhausted through the SGT System to the station chimney, which is an elevated release point. The release will be monitored since both the reactor building stack and the station chimney have effluent monitors that quantify radiological releases through these pathways.

Normal refueling is performed with the reactor building ventilation system running. This is a non-safety-related system. This system maintains the reactor building at a negative pressure (i.e., 0.25 inch water gauge) to preclude any unmonitored releases of radioactivity from the building.

Under the TS changes proposed for AST in Reference 1, both secondary containment and the SGT System are required to be operable during movement of recently irradiated fuel assemblies (i.e., fuel that has occupied part of a critical reactor core within the previous 24 hours) in the secondary containment. In the event of a FHA involving recently irradiated fuel, radioactivity that is released is captured by the ventilation system. If radioactivity levels exceed the setpoint of the Reactor Building Exhaust Radiation or Refueling Floor Radiation monitors, secondary containment isolation and actuation of the safety-related SGT System are initiated to support actions to limit the release of fission products. Any radioactivity is then filtered by the SGT System and released through the station chimney.

AST analyses have demonstrated that secondary containment and SGT System are not needed to mitigate a FHA involving fuel that has decayed greater than 24 hours. In the event of a FHA involving fuel that has decayed greater than 24 hours, the resulting airborne radioactivity is captured by the reactor building ventilation system and exhausted through the reactor building stack. The non-safety-related reactor building ventilation system and reactor building stack are not used as accident mitigation systems. Rather, this system is used for conservatism because this methodology assumes the radioactivity is quickly exhausted to the environment. In addition, the release is an unfiltered ground level release rather than an elevated filtered release through the SGT System.

If neither the reactor building ventilation system nor the SGT System is running, airborne radioactivity above the refuel floor from a FHA will remain there since there is no driving force to cause a release to the environment and subsequent radiological dose. In addition, releases of radioactive material by diffusion through the reactor building walls would be at a much slower rate (i.e., resulting in lower consequences) than assumed in the AST radiological dose assessment, since the assessment assumed an unfiltered ground level release exhausted by the reactor building ventilation system.

IV. Attachment A

NRC Request 1

The assumed adsorber efficiency for ESF filters should be consistent with the values in Table 1 of Revision 3 of Regulatory Guide 1.52. Licensees may select an efficiency of 90% for a two inch bed with justification. An assumption of 50% is unacceptable unless there are unique design features which would limit the efficiency to 50%. (Pg 25)

Response

See response to NRC request in Section I, "Description of Proposed Changes."

NRC Request 2

In the analyses, was the adsorber and particulate filter efficiencies reduced to account for the 1% bypass flow associated with the SGTS? (Pg 5.5-7 DNPS, Pg 5.5-x QCNPS, Pg 25)

Response

The efficiencies credited are within the maximum values permitted in accordance with Table 1 of Revision 3 to Regulatory Guide 1.52. The bypass flow associated with the SGT System remains unchanged. The LOCA analysis assumed a 50% efficiency.

NRC Request 3

If the High Pressure Coolant Injection (HPCI) System Gland Seal Condenser is vented through the SGT, what is the basis for ever allowing the SGT to be inoperable? (Pg 16)

Response

Allowed outage times are established to define a limiting time duration for which a system or component may be out of service. TS 3.5.1 specifies an allowed outage time of 14 days if the HPCI system is inoperable. TS 3.6.4.3 specifies an allowed outage time of 7 days if one SGT subsystem is inoperable and 1 hour if both subsystems are inoperable. For both TS 3.5.1 and TS 3.6.4.3, the affected unit must be in Mode 3 within 12 hours if the inoperable system or subsystem(s) are not restored to operable status within the specified allowed outage time (i.e., Completion Time). At DNPS, the HPCI Gland Seal Condenser vents to a common plenum, which leads to both SGT subsystems. Since the required actions for inoperable SGT subsystems are more restrictive than the actions for an inoperable HPCI system, it is acceptable to allow the SGT system to be inoperable without taking prior action for a concurrent HPCI system inoperability.

NRC Request 4

It is stated in Section 2.2.2 of that the post-accident dose rate in the control room and adjacent areas is due to the shine from the refueling floor airborne source. Is this for both the FHA and the LOCA? (Pg 20)

Response

The external source shine dose is calculated for the LOCA only. Because of the smaller source term for the FHA as compared to the LOCA, the concrete shielding between the refuel floor and the control room, and the decontamination of iodine by the water above the fuel, it is expected that the FHA external dose contribution will be negligible.

NRC Request 5

It is also stated in Section 2.2.2 of Attachment 2 that an additional low-level post-accident external source gamma ray dose rate component in the control room and adjacent areas is due to shine from the refueling floor noble gas airborne. These dose rates were calculated during the period 1980-1983 and are included in UFSAR Section 12A-3..2.3.4. What is the TEDE dose contribution for these sources and the other isotopes included with AST? (Pg 20)

Response

See response to NRC request 1 under Section III, "Safety Analysis."

NRC Request 6

Exclusion Area Boundary X/Q values should be used with a sliding two hour window to determine the start of the maximum time interval for the EAB dose. Was such a window used for DNPS and QCNPS?

Response

The exclusion area boundary dose is determined for the worst 2-hour X/Q. RADTRAD determines the maximum 2-hour window for exclusion area boundary dose. For DNPS and QCNPS, this period begins at T = 4 hours and ends at T = 6 hours.

NRC Request 7

Is the assumed control room envelope inleakage in the analyses 600 cfm through the CRE boundary, 600 cfm through control room ventilation system components and 10 cfm due to ingress/egress or 600 cfm in total from the CRE boundary, all control room ventilation system components and ingress and egress ?

Response

The unfiltered inleakage value to the control room envelope is through unidentified sources and is assumed to be 600 cfm total, which includes 10 cfm for ingress and egress.

NRC Request 8

Provide the calculations which determined the source term release via MSIV piping and the resultant doses. (Pg 24)

Response

Attachments 6 and 7 provide the requested calculations for DNPS and QCNPS, respectively.

NRC Request 9

It is indicated that a CRD accident at high power levels would result in a release via the augmented offgas (AOG) system. Would the AOG continue to operate in the event of an control rod drop accident?

Response

Three CRDA scenarios were assessed. The CRDA at high power level assumes that the steam jet air ejector (SJAE) is operational and the release is through the AOG

charcoal beds. This pathway was not analyzed since it was qualitatively determined to be bounded by the low power CRDA pathway with the release through the mechanical vacuum pump and station chimney. For this scenario, the AOG system is assumed to function normally throughout the event.

NRC Request 10

What is the basis for assuming that there will be no exfiltration from the reactor building in the event of a LOCA? (Pg 28)

Response

The reactor building ventilation system performs two secondary containment functions. First, it automatically controls the reactor building atmosphere at a slight negative pressure of approximately 1/10 to 1/4 inch water gauge with the exhaust fan dampers to assure intake of air so that exfiltration of airborne radioactive contamination is minimized. Second, it isolates on a secondary containment isolation signal.

The SGT System also maintains a negative reactor building pressure after an accident to minimize the release of unprocessed secondary containment atmosphere. The SGT System can reduce secondary containment pressure to –1/4 inch water gauge.

When SGT is in operation and the reactor building is completely isolated, a small average negative pressure is created in the reactor building which minimizes ground level release of airborne radioactivity. Two parallel trains are provided, each of which is capable of producing greater than 1/4 inch water gauge negative pressure. The system is designed to automatically start a single SGT System train or start both trains simultaneously. Redundant trains are provided to ensure the SGT System meets single failure criteria.

In the event of a design basis LOCA, secondary containment instrumentation automatically initiates closure of appropriate secondary containment isolation valves and starts the SGT System to limit fission product release. The reactor building is at a negative pressure at the beginning of the event, SGT automatically starts and maintains negative pressure, hence the reactor building pressure is always negative and no exfiltration will occur in the LOCA accident sequence.

In addition, during the conversion to Improved Technical Specifications (ISTS), ISTS SR 3.6.4.1.4, which requires verification that each SGT subsystem will draw down the secondary containment, was deleted. The justification for this deviation from NUREG-1433, Revision 1, was that deletion of the drawdown test is consistent with the current licensing basis, since the analysis does not assume an explicit drawdown time.

Additionally, AST assumes that there is insignificant release for the first two minutes during the coolant activity release phase. The onset of the gap activity release phase is two minutes after the initiation of the accident. The automatic start of the SGT System will ensure that the reactor building negative pressure will be maintained beyond the two-minute coolant activity release period.

NRC Request 11

Do the external sources, expected to contribute less than 0.5 rem TEDE during a LOCA, include all of the Regulatory Guide 1.183 isotopes. If not, why have these isotopes been excluded? (Pg 28)

Response

See response to NRC request 1 under Section III, "Safety Analysis."

NRC Request 12

What activity level is assumed in the steam released during the MSLB accident? (Pg 38)

Response

The activity levels assumed during the MSLB accident were provided in Reference 1, Attachment A, Table 6. These values correspond to the pre-accident spike iodine concentration of 4 μCi/gm I-131 equivalent and the maximum equilibrium iodine concentration of 0.2 μCi/gm I-131 equivalent.

NRC Request 13

Provide the basis for assuming credit for the offgas system for the removal of xenon, iodine and krypton isotopes resulting from a rod drop accident. (Pg 39, Pg 41)

Response

See response to NRC request 9 under Section IV, "Attachment A." The scenario that credited AOG operation was not analyzed because it was qualitatively determined to be bounded by the low power CRDA pathway with the release through the mechanical vacuum pump and station chimney.

V. <u>Attachment B</u>

NRC Request 1

It is indicated that your assessment complies with Note 11 of Table 3 of Regulatory Guide 1.183. Does this include the assumption of 10% iodine and noble gases in the gap of the fuel for a rod drop accident? (Pg 3)

Response

Yes, the calculation assumes that 10% of the core inventory of noble gases and iodines are released from the fuel gap.

NRC Request 2

What value (percentile) of the Power's model was used for natural deposition?

Response

The 10 percentile value was used.

NRC Request 3

What is the basis for assuming 50% mixing in the reactor building rather than assuming the leakage to the reactor building is exhausted directly to the atmosphere?

Response

The assumption of 50% mixing in the reactor building is consistent with the current NRC-approved licensing basis for DNPS and QCNPS for the calculation of control room operator thyroid doses. As described in Reference 5, the calculation of mixing for releases from the secondary containment to the environment credits 100% mixing in the secondary containment for DNPS for the calculation of control room operator thyroid dose. As described in Reference 6, the calculation of mixing for releases from the secondary containment to the environment credits 50% mixing in the secondary containment for QCNPS for the calculation of control room operator thyroid dose.

The assumption of 50% mixing in the reactor building is based on dilution due to mixing caused by SGT System operation. Based on the SGT System configuration and the large volume of the reactor building, the leakage cannot short circuit to the release point and hence the 50% mixing assumption is conservative and justified.

NRC Request 4

The ESF leakage rate should be a value based upon plant specific criteria and not typical industry leakage rates. Sometimes, the plant specific criteria is twice the value in the TMI Action Item III.D.1.1 Leakage Reduction Program where action is required to reduce leakage. Provide the basis for assuming that the industry value is appropriate for Dresden and Quad Cities. (Pg 17)

Response

There is no specific acceptance limit in the leakage reduction program at either site. However, both DNPS and QCNPS will continue to maintain the commitment to keeping this leakage as low as reasonably achievable. One of the goals of the AST submittal was to minimize changes to the way the plant is operated. With this in mind, there is no compelling basis for selecting a value less that specified in the AST regulatory guidance.

VI. Technical Specifications

NRC Request 1

The allowable penetration criteria in the VFTP was incorrectly calculated based upon the adsorber efficiencies presented in the accident analyses in Attachment A. Refer to Generic Letter 99-02. (Note that the assumed efficiencies should be consistent with Regulatory Guide 1.52.)

Response

See response to NRC request in Section I, "Description of Proposed Changes."

NRC Request 2

Why doesn't Technical Specification 3.1.7 contain an Action Statement which requires 3769.4 lbs of sodium pentaborate?

Response

As described in Reference 7, 3769.4 pounds of sodium pentaborate is equivalent to approximately 3000 gallons of 14% (by weight) of sodium pentaborate. SR 3.1.7.1 currently requires verification that the available volume of sodium pentaborate is within the limits of TS Figure 3.1.7-1 every 24 hours. The volume requirements of TS Figure 3.1.7-1 are more restrictive than the volume assumed in the AST analyses (i.e., 3769.4 pounds). Therefore, no changes to TS 3.1.7 are needed to ensure that the assumed volume of sodium pentaborate is available.

VII. References

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

2. Letter from P. R. Simpson (Exelon Generation Company, LLC) to U. S. Nuclear Regulatory Commission, "Additional Information Supporting the Request for License Amendment Related to Application of Alternative Source Term," dated March 28, 2003

3. Letter from D. G. Eisenhut (U. S. Nuclear Regulatory Commission) to L. DelGeorge (Commonwealth Edison Company), "Dresden Nuclear Power Station, Units 2 and 3," dated June 25, 1982

4. Letter from D. B. Vassallo (U. S. Nuclear Regulatory Commission) to D. L. Farrar (Commonwealth Edison Company), "Quad Cities Nuclear Power Station, Units 1 and 2," dated June 12, 1984

5. Letter from J. F. Stang (U. S. Nuclear Regulatory Commission) to I. Johnson (Commonwealth Edison Company), "Issuance of Amendments (TAC Nos. M98389 and M98390)," dated April 25, 1997

6. Letter from R. M. Pulsifer (U. S. Nuclear Regulatory Commission) to I. Johnson (Commonwealth Edison Company), "Issuance of Amendments (TAC Nos. M98227 and M98228)," dated March 27, 1997

7. Letter from P. R. Simpson (Exelon Generation Company, LLC) to U. S. Nuclear Regulatory Commission, "Additional Information Supporting the Request for License Amendment Related to Application of Alternative Source Term," dated March 21, 2003

ATTACHMENT 2

Dresden Nuclear Power Station
Marked-Up Technical Specifications for Proposed Changes

1.1 Definitions (continued)

CHANNEL FUNCTIONAL TEST	A CHANNEL FUNCTIONAL TEST shall be the injection of a simulated or actual signal into the channel as close to the sensor as practicable to verify OPERABILITY of all devices in the channel required for channel OPERABILITY. The CHANNEL FUNCTIONAL TEST may be performed by means of any series of sequential, overlapping, or total channel steps.
CORE ALTERATION	CORE ALTERATION shall be the movement of any fuel, sources, or reactivity control components, within the reactor vessel with the vessel head removed and fuel in the vessel. The following exceptions are not considered to be CORE ALTERATIONS: a. Movement of source range monitors, local power range monitors, intermediate range monitors, traversing incore probes, or special movable detectors (including undervessel replacement); and b. Control rod movement, provided there are no fuel assemblies in the associated core cell. Suspension of CORE ALTERATIONS shall not preclude completion of movement of a component to a safe position.
CORE OPERATING LIMITS REPORT (COLR)	The COLR is the unit specific document that provides cycle specific parameter limits for the current reload cycle. These cycle specific limits shall be determined for each reload cycle in accordance with Specification 5.6.5. Plant operation within these limits is addressed in individual Specifications.
DOSE EQUIVALENT I-131	DOSE EQUIVALENT I-131 shall be that concentration of I-131 (microcuries/gram) that alone would produce the same thyroid dose as the quantity and isotopic mixture of I-131, I-132, I-133, I-134, and I-135 actually present. The thyroid dose

(continued)

1.1 Definitions

DOSE EQUIVALENT I-131 (continued)	conversion factors used for this calculation shall be those listed in Table III of TID-14844, AEC, 1962, "Calculation of Distance Factors for Power and Test Reactor Sites;" Table E-7 of Regulatory Guide 1.109, Rev. 1, NRC, 1977; or ICRP 30, Supplement to Part 1, pages 192-212, Table titled, "Committed Dose Equivalent in Target Organs or Tissues per Intake of Unit Activity."
INSERT	
LEAKAGE	LEAKAGE shall be:

a. Identified LEAKAGE

1. LEAKAGE into the drywell, such as that from pump seals or valve packing, that is captured and conducted to a sump or collecting tank; or

2. LEAKAGE into the drywell atmosphere from sources that are both specifically located and known either not to interfere with the operation of leakage detection systems or not to be pressure boundary LEAKAGE;

b. Unidentified LEAKAGE

All LEAKAGE into the drywell that is not identified LEAKAGE;

c. Total LEAKAGE

Sum of the identified and unidentified LEAKAGE; and

d. Pressure Boundary LEAKAGE

LEAKAGE through a nonisolable fault in a Reactor Coolant System (RCS) component body, pipe wall, or vessel wall.

(continued)

INSERT

Federal Guidance Report 11, "Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion," 1989.

ATTACHMENT 3

Quad Cities Nuclear Power Station
Marked-Up Technical Specifications for Proposed Changes

1.1 Definitions

CHANNEL CHECK (continued)	status derived from independent instrument channels measuring the same parameter.
CHANNEL FUNCTIONAL TEST	A CHANNEL FUNCTIONAL TEST shall be the injection of a simulated or actual signal into the channel as close to the sensor as practicable to verify OPERABILITY of all devices in the channel required for channel OPERABILITY. The CHANNEL FUNCTIONAL TEST may be performed by means of any series of sequential, overlapping, or total channel steps.
CORE ALTERATION	CORE ALTERATION shall be the movement of any fuel, sources, or reactivity control components, within the reactor vessel with the vessel head removed and fuel in the vessel. The following exceptions are not considered to be CORE ALTERATIONS: a. Movement of source range monitors, local power range monitors, intermediate range monitors, traversing incore probes, or special movable detectors (including undervessel replacement); and b. Control rod movement, provided there are no fuel assemblies in the associated core cell. Suspension of CORE ALTERATIONS shall not preclude completion of movement of a component to a safe position.
CORE OPERATING LIMITS REPORT (COLR)	The COLR is the unit specific document that provides cycle specific parameter limits for the current reload cycle. These cycle specific limits shall be determined for each reload cycle in accordance with Specification 5.6.5. Plant operation within these limits is addressed in individual Specifications.
DOSE EQUIVALENT I-131	DOSE EQUIVALENT I-131 shall be that concentration of I-131 (microcuries/gram) that alone would produce the same thyroid dose as the quantity and isotopic mixture of I-131, I-132, I-133, I-134, and I-135 actually present. The thyroid dose

(continued)

1.1 Definitions

DOSE EQUIVALENT I-131 (continued)	conversion factors used for this calculation shall be those listed in Table III of TID-14844, AEC, 1962, "Calculation of Distance Factors for Power and Test Reactor Sites;" Table E-7 of Regulatory Guide 1.109, Rev. 1, NRC, 1977; or ICRP 30, Supplement to Part 1, pages 192-212, Table titled, "Committed Dose Equivalent in Target Organs or Tissues per Intake of Unit Activity."
INSERT	
LEAKAGE	LEAKAGE shall be:

a. Identified LEAKAGE

1. LEAKAGE into the drywell, such as that from pump seals or valve packing, that is captured and conducted to a sump or collecting tank; or

2. LEAKAGE into the drywell atmosphere from sources that are both specifically located and known either not to interfere with the operation of leakage detection systems or not to be pressure boundary LEAKAGE;

b. Unidentified LEAKAGE

All LEAKAGE into the drywell that is not identified LEAKAGE;

c. Total LEAKAGE

Sum of the identified and unidentified LEAKAGE; and

d. Pressure Boundary LEAKAGE

LEAKAGE through a nonisolable fault in a Reactor Coolant System (RCS) component body, pipe wall, or vessel wall.

(continued)

INSERT

Federal Guidance Report 11, "Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion," 1989.

ATTACHMENT 4

Dresden Nuclear Power Station
Retyped Technical Specifications for Proposed Changes

1.1 Definitions (continued)

CHANNEL FUNCTIONAL TEST	A CHANNEL FUNCTIONAL TEST shall be the injection of a simulated or actual signal into the channel as close to the sensor as practicable to verify OPERABILITY of all devices in the channel required for channel OPERABILITY. The CHANNEL FUNCTIONAL TEST may be performed by means of any series of sequential, overlapping, or total channel steps.
CORE ALTERATION	CORE ALTERATION shall be the movement of any fuel, sources, or reactivity control components, within the reactor vessel with the vessel head removed and fuel in the vessel. The following exceptions are not considered to be CORE ALTERATIONS: a. Movement of source range monitors, local power range monitors, intermediate range monitors, traversing incore probes, or special movable detectors (including undervessel replacement); and b. Control rod movement, provided there are no fuel assemblies in the associated core cell. Suspension of CORE ALTERATIONS shall not preclude completion of movement of a component to a safe position.
CORE OPERATING LIMITS REPORT (COLR)	The COLR is the unit specific document that provides cycle specific parameter limits for the current reload cycle. These cycle specific limits shall be determined for each reload cycle in accordance with Specification 5.6.5. Plant operation within these limits is addressed in individual Specifications.
DOSE EQUIVALENT I-131	DOSE EQUIVALENT I-131 shall be that concentration of I-131 (microcuries/gram) that alone would produce the same dose as the quantity and isotopic mixture of I-131, I-132, I-133, I-134, and I-135 actually present. The dose conversion factors used for this calculation shall be those listed in

(continued)

1.1 Definitions

DOSE EQUIVALENT I-131 (continued)	Federal Guidance Report 11, "Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion," 1989.
LEAKAGE	LEAKAGE shall be: a. <u>Identified LEAKAGE</u> 1. LEAKAGE into the drywell, such as that from pump seals or valve packing, that is captured and conducted to a sump or collecting tank; or 2. LEAKAGE into the drywell atmosphere from sources that are both specifically located and known either not to interfere with the operation of leakage detection systems or not to be pressure boundary LEAKAGE; b. <u>Unidentified LEAKAGE</u> All LEAKAGE into the drywell that is not identified LEAKAGE; c. <u>Total LEAKAGE</u> Sum of the identified and unidentified LEAKAGE; and d. <u>Pressure Boundary LEAKAGE</u> LEAKAGE through a nonisolable fault in a Reactor Coolant System (RCS) component body, pipe wall, or vessel wall.

(continued)

ATTACHMENT 5

Quad Cities Nuclear Power Station
Retyped Technical Specifications for Proposed Changes

1.1 Definitions

CHANNEL CHECK (continued)	status derived from independent instrument channels measuring the same parameter.
CHANNEL FUNCTIONAL TEST	A CHANNEL FUNCTIONAL TEST shall be the injection of a simulated or actual signal into the channel as close to the sensor as practicable to verify OPERABILITY of all devices in the channel required for channel OPERABILITY. The CHANNEL FUNCTIONAL TEST may be performed by means of any series of sequential, overlapping, or total channel steps.
CORE ALTERATION	CORE ALTERATION shall be the movement of any fuel, sources, or reactivity control components, within the reactor vessel with the vessel head removed and fuel in the vessel. The following exceptions are not considered to be CORE ALTERATIONS: a. Movement of source range monitors, local power range monitors, intermediate range monitors, traversing incore probes, or special movable detectors (including undervessel replacement); and b. Control rod movement, provided there are no fuel assemblies in the associated core cell. Suspension of CORE ALTERATIONS shall not preclude completion of movement of a component to a safe position.
CORE OPERATING LIMITS REPORT (COLR)	The COLR is the unit specific document that provides cycle specific parameter limits for the current reload cycle. These cycle specific limits shall be determined for each reload cycle in accordance with Specification 5.6.5. Plant operation within these limits is addressed in individual Specifications.
DOSE EQUIVALENT I-131	DOSE EQUIVALENT I-131 shall be that concentration of I-131 (microcuries/gram) that alone would produce the same dose as the quantity and isotopic mixture of I-131, I-132, I-133, I-134, and I-135 actually present. The dose conversion factors used for this calculation shall be those listed in

(continued)

1.1 Definitions

DOSE EQUIVALENT I-131 (continued)	Federal Guidance Report 11, "Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion," 1989.
LEAKAGE	LEAKAGE shall be: a. <u>Identified LEAKAGE</u> 1. LEAKAGE into the drywell, such as that from pump seals or valve packing, that is captured and conducted to a sump or collecting tank; or 2. LEAKAGE into the drywell atmosphere from sources that are both specifically located and known either not to interfere with the operation of leakage detection systems or not to be pressure boundary LEAKAGE; b. <u>Unidentified LEAKAGE</u> All LEAKAGE into the drywell that is not identified LEAKAGE; c. <u>Total LEAKAGE</u> Sum of the identified and unidentified LEAKAGE; and d. <u>Pressure Boundary LEAKAGE</u> LEAKAGE through a nonisolable fault in a Reactor Coolant System (RCS) component body, pipe wall, or vessel wall.

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