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September 22, 2004

U. S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, DC 20555-0001

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 P. R. Simpson (Exelon Generation Company, LLC) to U. S. NRC, "Additional Information Supporting the Request for License Amendment Related to Application of Alternative Source Term," dated September 15, 2003

3. 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 Regarding Alternative Source Term Amendment Request (TAC Nos. MB6530, MB6531, MB6532, and MB6533)," dated August 19, 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. In Reference 2, EGC submitted a response to a request for additional information related to the dose assessment supporting the amendment request. In Reference 3, the NRC requested additional information related to the dose assessment. Attachment 1 to this letter 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

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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 22nd day of September 2004.

Respectfully,

Patrick R. Simpson

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

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 1
Response to Request for Additional Information

The following requests for additional information (RAI) and responses follow the same numbering as in the Reference 1 RAI response.

II. Definition of Dose Equivalent Iodine 131

NRC Request

The September 15, 2003, response on the definition of Dose Equivalent Iodine 131 (^{131}I) stated that the proposed TS change had been modified to indicate the inhalation committed dose equivalent from Federal Guidance Report 11. However, a review of the revised TS markup pages shows that only Federal Guidance Report 11 was referenced and the inhalation dose conversion factors were not specified. Please revise the definition to indicate that it is the inhalation committed dose conversion factors of Federal Guidance Report 11.

Response

The proposed change to the definition of dose equivalent iodine I-131 has been revised to indicate that the inhalation committed dose conversion factors from Federal Guidance Report 11 will be used. The revised Technical Specifications (TS) markups for Dresden Nuclear Power Station (DNPS) and Quad Cities Nuclear Power Station (QCNPS) are provided in Attachments 2 and 3, respectively. Retyped TS pages for DNPS and QCNPS are provided in Attachments 4 and 5, respectively. These proposed changes supersede those previously provided in References 1 and 2.

III. Safety Analysis

NRC Request – Response to Request 1

The staff has reviewed Updated Final Safety Analysis Report (UFSAR) Section 12A.3, Section 6.4.2.5 and Dresden Section 12.3.2.2.4 and has concluded that the shine dose to the control room operators needs to incorporate Regulatory Guide (RG) 1.183 isotopes. Please revise the shine dose to control room operators to include the RG 1.183 isotopes.

Response

Exelon Generation Company, LLC (EGC) has performed a calculation to determine the shine dose to DNPS control room operators following a loss-of-coolant accident (LOCA). This calculation uses the 60-isotope library (i.e., Regulatory Guide 1.183 isotopes) from the RADTRAD computer code for the evaluation of dose consequences. The results of the calculation conclude that the total TEDE dose due to direct shine from the reactor building is 105.1 mrem.

In Reference 2, EGC stated that the dose from external sources is expected to be much less than 0.5 rem TEDE. As stated above, the results of the calculation for DNPS conclude that the total TEDE dose due to direct shine from the reactor building is 105.1 mrem. Due to the similarities between DNPS and QCNPS, the assumption that

the dose from external sources is expected to be much less than 0.5 rem TEDE for QCNPS is justified.

Additionally, Reference 2 stated that the post-LOCA control room doses were 4.53 rem and 3.9 rem for DNPS and QCNPS, respectively. With the additional 105.1 mrem due to direct shine from the reactor building, the post-LOCA control room doses remain below the regulatory limit of 5 rem TEDE.

NRC Request – Response to Request 7

The September 15, 2003, response to this request indicates that inleakage during the normal mode of operation will be lower than during the filtration mode and explains why inleakage through dampers and in ducts would be less than during normal operation. However, the response does not address inleakage through the four walls, ceiling and floor and why it would be less during normal operation than it would during the emergency mode of operation. Has it been confirmed through measurements that the inleakage characteristics of the Dresden and Quad Cities control rooms will remain the same when the normal ventilation systems are operating. Did such measurements account for adjacent area ventilation systems being configured in their accident mode of operation while the control room ventilation systems remain in their normal mode of operation.

Based upon the information provided in the December 9, 2003, letter responding to Generic Letter 2003-01, does the operation of the Quad Cities control room ventilation system Train B isolate on the same signals which isolate Train A? If it does not, what signals does it isolate on and is one train more limiting than the other with respect to the time of exposure to the control room operators?

Response

The four walls, ceiling, and floor provide a substantial barrier. Inleakage through these structures is not anticipated to change appreciably between the normal and emergency modes of operation. In the normal mode of operation, all air intakes are unfiltered. For this reason, a tracer gas test was not performed in the normal mode. Adjacent area ventilation systems were aligned in configurations to maximize inleakage potential with the control room ventilation system in the emergency mode.

Additionally, the 600 cfm unfiltered inleakage value is also assumed in the normal mode until the emergency mode is in operation. This value is assumed during the entire event, regardless of the mode.

The QCNPS Train B control room ventilation isolates on the same signals that isolate Train A.

NRC Request – Response to Request 8

The September 15, 2003, response to NRC Request 8 has not provided an adequate basis for the assumed value for inleakage during the time period in which normal

ventilation system is operating. The inleakage characteristics of the control room envelope (CRE) while the normal control room ventilation system is operating will be a function of the pressures established in the areas adjacent to the CRE and in those areas where the control room ventilation systems are located. The CRE inleakage will also be affected by the control room ventilation system ductwork pressures and the pressures in the rooms in which the ductwork passes and by the pressures in the ductwork of the non-control room ventilation systems which traverse the control room envelope. There appears to have been no confirming measurement that the value of 600 cfm represents the performance characteristics of the control room's normal ventilation system under accident conditions nor is it certain that the limiting condition for that particular mode of operation has been identified. Guidance on the determination of limiting conditions may be found in RG 1.197. Provide confirmation that 600 cfm is the limiting inleakage value with the control room's normal ventilation system is operating.

Response

The control room's normal ventilation system is not in operation during accident situations except during the switchover period from the normal mode of operation to the emergency mode of operation. During this period, full-flow, unfiltered conditions are assumed.

As stated in Reference 1, tracer gas testing was not performed in the normal mode of Control Room Emergency Ventilation (CREV) system operation. However, the inleakage during the normal mode would be lower than during the filtration mode for the reasons described in EGC's response to NRC Request 7 in Reference 1. Inleakage values determined through tracer gas testing in the filtration mode were provided in Reference 1. Based on these test results, the assumed inleakage value of 600 cfm is justified.

NRC Request – Response to Request 10

The September 15, 2003, response to this request addressed the ability of the Standby Gas Treatment System (SGTS) to establish and maintain the reactor building at a negative 0.5 inch w.g. following a LOCA. During a May 5, 2004, Loss of Offsite Power event at Dresden the required vacuum for secondary containment (shared by both Unit 2 and Unit 3) could not be maintained. It appears that this was a result of the Unit 2's Drywell Vent and Purge System operating and not receiving a Division II isolation signal while Unit 3 received a Division II isolation signal which initiated the SGTS automatically and secured Unit 3's Drywell Vent and Purge system. It is our understanding that the final cause of this event remains unknown. However, possible causes may have been inadequate procedures (not securing the opposite unit's drywell vent and purge); inadequate design (not auto securing the opposite unit's drywell vent and purge); or inadequate material condition of the Unit 2 non-safety related drywell vent and purge system which may have affected the ability of the SGTS maintain the required vacuum in the secondary containment. Nevertheless, based upon the May 5th event, what actions have been taken to assure that the negative 0.5 inch w.g. pressure may be maintained in the reactor building in the event of an accident?

Response

The NRC request above does not accurately describe the SGT System. Specifically, the SGT System does not establish and maintain the reactor building at a negative 0.5 inch w.g. following a LOCA. EGC's response to NRC Request 10 was previously provided in Reference 1. Reference 1 states "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."

Details regarding the DNPS event described above were described in Reference 4. Specifically, the root cause of the low secondary containment vacuum was determined to be a degraded secondary containment boundary that was not detected due to an inadequate leak rate test procedure. The degraded secondary containment boundary resulted from air in-leakage into the Unit 2 Drywell and Torus Purge Exhaust (DTPE) filter housings. At the time of the event, Unit 2 was in a maintenance outage and the DTPE fans were in operation due to activities in the Unit 2 drywell. The DTPE fans are not normally in operation and the secondary containment leak rate test procedure does not test with the DTPE fans operating as a part of the secondary containment barrier.

Two corrective actions to prevent reoccurrence were described in Reference 4. The first is to modify the current design to trip the DTPE fans on both units following an automatic SGT system initiation from either unit, rather than operate the DTPE fans during the secondary containment leak rate test. The second action is to develop a source document that clearly identifies the secondary containment boundaries.

NRC Request – Response to Request 11

It is stated that the SGTS will be OPERABLE whenever fuel handling operations occur which involve "recently irradiated fuel". "Recently irradiated fuel" has been defined as any fuel which has not decayed for at least 24 hours. BWRs are presumed to be unable to begin fuel handling operations until 24 hours following the reactor becoming sub-critical. Therefore, it appears that the SGTS will never have to be OPERABLE during fuel handling operations. Based upon the above, does Exelon agree that the SGTS will never be OPERABLE during fuel handling operations? If you agree, then what assurances will there be that all releases to the reactor building will be processed and discharged through a radiation monitor?

It appears that only one fuel handling accident (FHA) analysis was performed when two FHA analyses should have been performed. One analysis should have assumed the dropping of a fuel assembly with no decay time and release through SGTS to the station chimney. The second analysis should have assumed the release of the contents of a fuel assembly with 24 hours decay and release occurring as a ground-level release via the reactor building vent stack. Please provide analyses which cover both situations.

At Dresden, the reactor building stack seems to be further from the control room intake than the Unit 2 reactor building. If the reactor building ventilation system is not operating and the release from a fuel handling accident is via diffusion, it would appear that such a

diffuse release source would result in a greater concentration at the control room intake and in the control room compared to a release occurring with the reactor building ventilation system operating. A similar situation may exist at Quad Cities Unit 1 reactor building. Would a diffuse release from the reactor building due to a FHA result in higher doses to the control room operators?

Response

These proposed changes related to the applicability requirements during movement of irradiated fuel assemblies are consistent with Technical Specification Task Force Traveler (TSTF)-51, "Revise Containment Requirements During Handling Irradiated Fuel and Core Alterations," Revision 2. TSTF-51, Revision 2, was approved by the NRC on October 15, 1999. TSTF-51 changes the TS operability requirements for engineered safety features such that they are not applicable after sufficient radioactive decay has occurred to ensure that offsite doses remain within limits.

Although the SGT system may not be needed for handling irradiated fuel that has decayed greater than 24 hours, this provision would only be used if both units at either DNPS or QCNPS were shutdown, since the SGT system is common to both units. However, in the case where the SGT system is not operable, releases would continue to be monitored since the Offsite Dose Calculation Manual requires the station chimney and reactor building vent stack effluent monitors at all times.

Since fuel movement is not permitted until 24 hours after shutdown, calculation of radiological consequences for decay periods less than 24 hours is therefore not warranted.

It is not reasonable to assume that all fuel floor activity can be released via a diffuse release in a 2-hour period. For this to occur, the wind speeds required would have to be extremely high. With such high winds, the site would most likely be in an Alert emergency classification, which would result in suspension of fuel movement. Additionally, with such high winds, X/Q values would be very low, making it such that very little activity would be drawn into the control room intake. The winds would drive the activity past the intake at a higher rate than the intake could draw it in (i.e., wind speed higher than intake flow).

IV. Attachment A

NRC Request – Response to Request 2

The accident analyses which involve HEPA filters and charcoal adsorbers with an approved 1% allowable bypass for the in-place test should account for the reduction in filter and adsorber efficiency by reducing the effective filtration and adsorption rates. Your dose consequence methodology should account for the 1% bypass. Please provide revised dose assessments for those accidents which assume filtration and adsorption to reduce the consequences of an accident and for which the filter or adsorber providing such a mitigating affect has an allowable 1% penetration for the in-place filter or charcoal adsorber test.

Response

The 1% in-place test (i.e., particulate) bypass flow is accounted for by using an efficiency of 99% per Regulatory Guide 1.52. The charcoal adsorbers are tested in a laboratory using approved methods with a safety factor of 2 which bounds the 1% bypass flow. Therefore, the existing dose assessments are acceptable.

NRC Request – Response to Request 5

Under the Safety Analysis Response to Request 1, Exelon has been requested to provide the TEDE dose to the control room operators due to shine based upon RG 1.183 isotopes.

Response

As stated above, EGC has performed a calculation to determine the shine dose to the control room operators due to the isotopes contained in RG 1.183. The results are discussed above in EGC's response to NRC Request – Response to Request 1.

NRC Request – Response to Request 9

It does not appear that the September 15, 2003, response answered the staff's RAI. Information was requested which asked, "Would the augmented offgas (AOG) system continue to operate in the event of a CRDA?" The answer appears to be "No".

A review of the UFSAR has led the staff to conclude that since the main steam line radiation monitor (MSLRM) trip function and the main steam line (MSL) isolation functions have been removed for all modes of operation except during the operation of the mechanical vacuum pump, the AOG will continue to operate unless the radioactivity exceeds the limit established in accordance with the offsite dose calculation manual (ODCM). If that limit is exceeded, the holdup line of the off-gas system is automatically isolated after a 15-minute delay. From UFSAR Section 10.4.2.5, it appears that the AOG is isolated as noted above but that the steam jet air ejector is not isolated. Section 10.4.3 of the UFSAR indicates that the holdup of the offgas provides sufficient time between detection and isolation to prevent release. From this description, it appears that the AOG will be isolated. If this is true, then question becomes, "What happens to the radioactivity following a CRDA if the AOG is isolated?"

Response

There were three scenarios considered for the control rod drop accident (CRDA) analyses. All three assumed the reactor was in the startup/power ascension mode.

The first scenario assumes the mechanical vacuum pump (MVP) is in operation for condenser vacuum. Gland sealing (GS) steam flow is in operation. Main steam line radiation monitor (MSLRM) high alarms in the control room within seconds. MSLRM high-high radiation initiates MVP shutdown within seconds. In this scenario, 0.15% of the activity is released through the GS condenser to the station chimney with no delay.

No AOG operation is assumed. The balance of activity is released from the condenser at a rate of 1% volume per day for 24 hours as a ground level release. The dose for this scenario was not calculated since this is bounded by the second scenario, described below, which has a significantly larger and quicker release.

The second scenario also assumes the MVP and GS are in operation. There is possibly a MSLRM high radiation alarm in the control room. The MSLRM high-high radiation setpoint is not reached, thus there is no automatic MVP trip. The reactor trip and other neutron instrument responses alert operators of the CRDA. In response to MSLRM alarms and other indications, the operator trips the MVP within 10 minutes. Again, 0.15% of the activity is released through the GS condenser to the station chimney with no delay. No AOG operation is assumed. The activity is exhausted at high flow from the condenser through the station chimney using the MVP for 10 minutes before manual isolation. The balance of activity is released from the condenser at a rate of 1% volume per day for 24 hours as a ground level release.

The third scenario assumes the GS and steam jet air ejectors are in operation. There is no dependence on MSLRM response. In this scenario, 0.15% of the activity is released through the GS condenser to the station chimney with no delay. Activity from the condenser is exhausted through the AOG system, minimizing iodine releases and delaying noble gas releases for decay. Dose implications were not calculated for this scenario since it is bounded by the second scenario, since the second scenario assumes a significantly quicker release and includes iodines.

If an offgas high-high radiation signal is received, the chimney isolation and holdup volume drain valves automatically close after a 15-minute delay. Isolation of the offgas system will result in a loss of main condenser vacuum due to the inability to remove non-condensables from the main condenser. The activity will then be released from the condenser at a rate of 1% volume per day for 24 hours as a ground level release. In this case, the dose consequences would be bounded by the second scenario.

V. Attachment B

NRC Request – Response to Request 4

The leakage reduction program should have an acceptance criterion of 1 gpm. While you can have an acceptance criteria of as low as reasonably achievable (ALARA), a maximum value of 1 gpm needs to be specified. If there is not a limitation of 1 gpm, the facility could find themselves outside their licensing basis.

Response

Procedures DOS 0040-14, "Leak Detection and Reduction," and QCTP 0820-08, "Leakage Reduction," specify the acceptance criterion for leakage from primary coolant sources outside containment for DNPS and QCNPS, respectively. For DNPS, the acceptance criterion is no leakage, and for QCNPS, the acceptance criterion is 10 gallons per hour. If leakage is discovered that exceeds these criteria, a work request and/or condition report is initiated to correct the issue. Additionally, QCTP 0820-08

requires that if the leakage exceeds 60 gallons per hour, the condition report identify that the alternative source term limit has been exceeded and that an operability evaluation is required.

References

1. Letter from P. R. Simpson (Exelon Generation Company, LLC) to U. S. NRC, "Additional Information Supporting the Request for License Amendment Related to Application of Alternative Source Term," dated September 15, 2003

2. 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

3. NRC Regulatory Guide 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors," dated July 2000

4. Letter from D. G. Bost (Exelon Generation Company, LLC) to U. S. NRC, "Licensee Event Report 2004-003-00, 'Unit 3 Scram Due to Loss of Offsite Power and Subsequent Inoperability of the Standby Gas Treatment System for Units 2 and 3,'" dated July 6, 2004

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."
LEAKAGE	LEAKAGE shall be:

INSERT

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

the inhalation committed dose conversion factors in 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.
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(continued)

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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."
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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

the inhalation committed dose conversion factors in 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.
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(continued)

1.1 Definitions

DOSE EQUIVALENT I-131 (continued)	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.
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(continued)

1.1 Definitions

DOSE EQUIVALENT I-131 (continued)	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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