August 19, 2004

Mr. Christopher M. Crane, President
and Chief Nuclear Officer
Exelon Generation Company, LLC
4300 Winfield Road
Warrenville, IL 60555

SUBJECT: 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)

Dear Mr. Crane:

By letter dated October 10, 2002, Exelon Generation Company (Exelon) submitted an amendment request to support application of an alternative source term (AST) at Dresden Nuclear Power Station, Units 2 and 3, and Quad Cities Nuclear Power Station, Units 1 and 2. By letter dated September 15, 2003, Exelon responded to a Nuclear Regulatory Commission (NRC) staff request for additional information (RAI) on the dose assessment supporting the AST amendment request. Our continuing review of the AST amendment request and dose assessment RAI response has identified the need for a second round of RAIs in the dose assessment area. These questions are attached and follow the same numbering as Exelon's September 15, 2003, response. This RAI was mailed electronically to your staff on June 18, 2004, and it was discussed with your staff on July 1, 2004. Several response dates were discussed with your staff but have passed. Please respond to this RAI within 30 days.

Please contact me at 301-415-2863 if your staff has any questions about this RAI.

Sincerely,

/RA/

Lawrence W. Rossbach, Project Manager, Section 2
Project Directorate III
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket Nos.: 50-237, 50-249, 50-254, and 50-265

Enclosure: Request for Additional Information

cc w/encl: See next page

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DRESDEN NUCLEAR POWER STATION, UNITS 2 AND 3

AND QUAD CITIES NUCLEAR POWER STATION, UNITS 1 AND 2

REQUEST FOR ADDITIONAL INFORMATION

ALTERNATIVE SOURCE TERM AMENDMENT REQUEST

The following request for additional information (RAI) follows the same numbering as in Exelon's September 15, 2003, RAI response.

II. Definition of Dose Equivalent Iodine 131

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.

III. Safety Analysis

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 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 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 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 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?

IV. Attachment A

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 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 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 off-gas 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?"

V. Attachment B

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.