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

Dresden Nuclear Power Station, Units 2 and 3
Renewed 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
Renewed 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 P. R. Simpson (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 M. Banerjee (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 July 22, 2005

In Reference 1, Exelon Generation Company, LLC (EGC) requested an amendment to the facility operating licenses for Dresden Nuclear Power Station (DNPS), Units 2 and 3, and Quad Cities Nuclear Power Station (QCNPS), Units 1 and 2. The proposed changes support application of an alternative source term (AST) methodology.

In Reference 2, the NRC requested additional information related to dose assessment and meteorological areas. The attachments to this letter provide the requested information.

EGC has developed a set of tables that compares the changes in parameters and methods used in the current licensing basis, the Reference 1 submittal, and the final supplemented analyses. These tables are provided as Attachments 18 and 19 for

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DNPS and QCNPS, respectively. A summary of the revised analysis results is also included in Attachments 18 and 19. The tables provided in these attachments supersede Tables 1 through 16 provided in Attachment A of Reference 1.

Additionally, EGC has determined that some of the responses previously provided to support NRC requests for additional information require revision as a result of the revised AST analyses submitted herein. Attachment 22 provides a matrix of the previous responses that require revision, with references to the submittal date for the previous response, the original response, and the revised response.

EGC has reviewed the information supporting a finding of no significant hazards consideration that was previously provided to the NRC in Attachment C of Reference 1. The supplemental information provided in this submittal does not affect the bases for concluding that the proposed license amendment does not involve a significant hazards consideration.

Attachment 23 provides a list of regulatory commitments being made in this submittal. 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 August 2005.

Respectfully,

Patrick R. Simpson

Patrick R. Simpson
Manager – Licensing

Attachments:

1. Response to Request for Additional Information
2. Calculation DRE02-0037, "Re-analysis of Control Rod Drop Accident (CRDA) Using Alternative Source Terms," Revision 1
3. Calculation QDC-0000-N-1268, "Re-analysis of Control Rod Drop Accident (CRDA) Using Alternative Source Terms," Revision 1
4. Calculation DRE02-0036, "Re-analysis of Fuel Handling Accident (FHA) Using Alternative Source Terms," Revision 1
5. Calculation QDC-0000-N-1267, "Re-analysis of Fuel Handling Accident (FHA) Using Alternative Source Terms," Revision 1
6. Calculation DRE02-0035, "Re-analysis of Main Steam Line Break (MSLB) Accident Using Alternative Source Terms," Revision 3
7. Calculation QDC-0000-N-1266, "Re-analysis of Main Steam Line Break (MSLB) Accident Using Alternative Source Terms," Revision 3
8. Calculation DRE05-0048, "Dresden Units 2 & 3 Post-LOCA EAB, LPZ, and CR Dose – AST Analysis," Revision 0
9. Calculation QDC-0000-N-1481, "Quad Cities Units 1 & 2 Post-LOCA EAB, LPZ, and CR Dose – AST Analysis," Revision 0

10. Technical Evaluation to Support Control Room Unfiltered Inleakage up to 60,000 cfm at the Dresden Nuclear Power Station (DNPS)
11. Technical Evaluation to Support Control Room Unfiltered Inleakage up to 60,000 cfm at the Quad Cities Nuclear Power Station (QCNPS)
12. Marked-up Technical Specifications Page for DNPS
13. Marked-up Technical Specifications Page for QCNPS
14. Retyped Technical Specifications Page for DNPS
15. Retyped Technical Specifications Page for QCNPS
16. Calculation DRE04-0030, "Atmospheric Dispersion Factors (X/Qs) for Accident Release," Revision 1
17. Calculation QDC-0000-M-1408, "Atmospheric Dispersion Factors (X/Qs) for Accident Release," Revision 1
18. DNPS Plant Parameters Tables
19. QCNPS Plant Parameters Tables
20. Technical Evaluation to Evaluate the Impact of Accounting for Terrain Effects in Determining the Post-LOCA Dispersion Coefficients at the Dresden Nuclear Power Station (DNPS)
21. Technical Evaluation to Evaluate the Impact of Accounting for Terrain Effects in Determining the Post-LOCA Dispersion Coefficients at the Quad Cities Nuclear Power Station (QCNPS)
22. Updated Responses to Previous NRC Requests for Additional Information
23. Summary of Regulatory Commitments

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

NRC Request 1

For the control rod drop accident analysis, the results of the staff's confirmatory analysis do not match Exelon's conclusions. Please provide your calculations. In addition, please discuss the assumptions used for radionuclide deposition and plateout during the period of mechanical vacuum pump operation.

Response

The calculations for the Dresden Nuclear Power Station (DNPS) and Quad Cities Nuclear Power Station (QCNPS) control rod drop accident analysis using alternative source term (AST) are provided as Attachments 2 and 3, respectively. These calculations were revised to remove the discussion of Main Steam Line Radiation Monitor setpoint determination. These setpoints continue to assure tripping of the mechanical vacuum pumps at essentially the initiation of a CRDA event, and therefore no releases or deposition/plateout of radioactivity via the mechanical vacuum pump pathway are included. The calculations for the control rod drop accident, included as Attachments 2 and 3, have been revised to reflect this change.

NRC Request 2

For the fuel handling accident analysis, the staff's confirmatory analysis does not match Exelon's conclusions. This disparity may be caused by the assumed release rate. Please confirm that Exelon's analysis assumes that the entire activity released from the fuel pool is released to the environment at the end of two hours. Please provide the calculation for the pool decontamination factor.

Response

The calculations for the DNPS and QCNPS fuel handling accident analysis using AST are provided as Attachments 4 and 5, for comparison to the NRC's confirmatory analysis. The assumptions related to the release rate are described in Section 2.4. The analyses for DNPS and QCNPS assume that effectively the entire activity released from the fuel pool (i.e., 99.97%, calculated using an exponential equation) is released to the environment by the end of two hours. The calculated pool decontamination factor (DF) is 135 for a depth of 19 feet and 200 for 23 feet or greater. The DF calculation uses Staff Technical Paper, "Evaluation of Fission Product Release and Transport," G. Burley, 1971. Additional information is provided in Section 2.3 of the DNPS and QCNPS calculations.

The total doses calculated for the fuel handling accident at each station are shown below.

	DNPS		Quad Cities	
Water Coverage:	23 feet	19 feet	23 feet	19 feet
EAB (rem TEDE)	0.709	0.967	3.84	5.24
LPZ (rem TEDE)	0.074	0.101	0.294	0.401
CR (rem TEDE)	1.35	1.99	1.22	1.80

NRC Request 3

For the main steamline break accident analysis, the contributions from noble gases and cesium do not appear to be included in Exelon's analysis. Please provide the results of an analysis considering cesium and noble gas contributions. Please provide the assumptions utilized in determining the effect of cesium and noble gases upon dose.

Response

The DNPS and QCNPS calculations for the main steam line break accident have been revised to consider cesium and noble gas contributions. The revised calculations are provided as Attachments 6 and 7 for DNPS and QCNPS, respectively. The 140,000 pounds of reactor coolant water is conservatively utilized from NUREG–0800, Standard Review Plan 15.6.4, as a bounding value for all conditions, and is only used for the control room and offsite dose analysis. The discharge values stated in the UFSAR remain the basis for the determination that there is no fuel damage as a result of the main steam line break accident. Section 6.3 of the calculations contains information regarding the determination of the quantity of cesium used in the analyses. Cesium releases are based on the Regulatory Guide 1.183 assumption (i.e., Appendix C, paragraph 3.6) that a single cesium atom will accompany 95% of the released iodine atoms. For Cs-133, Cs-134, Cs-135, and Cs-137, isotopic data for end of cycle conditions from a suppression pool pH assessment were used. For shorter lived isotopes such as Cs-136 and Cs-138, the ratio of their concentration values in reactor water to that of Cs-137 in American Nuclear Society Standard (ANS) 18.1-1999 "Radioactive Source Terms for Normal Operation of Light Water Reactors," Table 5, is used to predict their relative concentrations. Releases reflect this distribution, with the molar fractions converted to curie quantities based on the isotope's decay constant. Cs-133, representing about 38% of the cesium, is stable. The control room dose determination uses a hemispherical cloud model using a spreadsheet calculation. Iodine, noble gases, and cesium are treated in the same manner. Attachment A of each calculation contains the spreadsheets used to calculate the effects due to iodine, noble gases, and cesium. The effects due to cesium are negligible.

The results of the revised main steam line break dose calculations are tabulated below.

Location	DNPS Case 1 (Normal equilibrium limit of 0.2 μCi/gm) Dose (rem TEDE)	DNPS Case 2 (Iodine spike limit of 4.0 μCi/gm) Dose (rem TEDE)
EAB	3.25E-02	6.46E-01
LPZ	4.06E-03	8.06E-02
CR	9.20E-02	1.84E+00

Location	QCNPS Case 1 (Normal equilibrium limit of 0.2 µCi/gm) Dose (rem TEDE)	QCNPS Case 2 (Iodine spike limit of 4.0 µCi/gm) Dose (rem TEDE)
EAB	6.38E-02	1.27E+00
LPZ	6.42E-03	1.27E-01
CR	9.20E-02	1.84E+00

NRC Request 4a – Loss-of-Coolant Accident

Exelon's analysis assumes credit for radionuclide deposition in vertical main steamline components. It is inappropriate to consider deposition and plateout in such sections of piping. Please provide a revised analysis which does not take credit for vertical removal, or provide a justification for why credit for a vertical run of pipe is appropriate. Please provide Stone & Webster Calculation 086457.022-UR(B)-001, Rev. 0, "Modeling Gravitational Settling/Plateout in Main Steam Lines at Dresden 2 & 3 and Quad Cities 1 & 2," if it is still being used and relied upon in your analysis [sensitive information – redacted].

Response

The DNPS and QCNPS calculations for the loss-of-coolant accident have been revised, and the Stone & Webster calculation referenced above is no longer used as the analyses of record. The revised calculations for DNPS and QCNPS, provided as Attachments 8 and 9, respectively, do not credit aerosol settling in vertical main steam line segments. However, deposition of elemental iodine, which does not act as a particulate and therefore is not gravity dependent, is credited in all available seismically qualified piping (i.e., horizontal and vertical). Neither aerosol settling nor elemental iodine deposition is credited between the reactor vessel nozzle and the inboard main steam isolation valve (MSIV) in the steam line that is assumed to fail.

The aerosol deposition removal efficiencies for the main steam lines are determined based on the methodology in Appendix A of AEB 98-03 using the projected area (i.e., diameter x length) of horizontal pipe only.

The revised LOCA calculations assumed a leakage rate for MSIVs equal to 150 standard cubic feet per hour (scfh) for all four main steam lines, and 60 scfh maximum through any one steam line, including the line with the failed MSIV. These leakage rates are based on a pressure of 48 psig. A conversion factor of 1.73 is used to determine the corresponding leakage rates at the test pressure of $\geq$ 25 psig. These assumptions for leakage rates are different than those in the calculations that formed the basis for the DNPS and QCNPS AST submittal dated October 10, 2002. Therefore, the proposed Technical Specification changes related to MSIV leakage require revision. Specifically, the proposed changes to Technical Specification Surveillance Requirement (SR) 3.6.1.3.10 are being revised to read:

> "Verify the leakage rate through each MSIV leakage path is $\leq$ 34 scfh when tested at $\geq$ 25 psig, and the combined leakage rate for all MSIV leakage paths is $\leq$ 86 scfh when tested at $\geq$ 25 psig."

Revised Technical Specification markups indicating this change are provided in Attachments 12 and 13 for DNPS and QCNPS, respectively. The retyped Technical Specification pages for DNPS and QCNPS are provided in Attachments 14 and 15, respectively.

For elemental iodine deposition, the natural removal efficiency for elemental iodine in each steam line volume is assumed to be 50% as recommended in AEB 98-03, Appendix B. This treatment of elemental iodine includes the re-suspension and fixation of elemental iodine from the pipe surface. The mechanism for elemental iodine removal is due to chemical adsorption and is independent of surface orientation. Therefore, consideration of vertical piping for elemental iodine deposition is appropriate.

NRC Request 4b – Loss-of-Coolant Accident

Please provide the bases for the reduction in the drywell and main steam isolation valve leakage by 50 percent after 24 hours, including the applicable reactor pressure vessel and drywell pressures used in this analysis.

Response

The revised LOCA calculations for DNPS and QCNPS (i.e., Attachments 8 and 9) do not credit any reduction in drywell pressure or the MSIV leakage rate of 150 scfh. Leakage rates are held constant for the duration of the accident for conservatism.

The revised analyses consider mixing of the drywell and suppression pool air spaces after the initial two hours following a LOCA. At two hours, Emergency Core Cooling System (ECCS) injection is assumed. This injection causes mixing due to the extremely turbulent conditions generated by the injection of cold ECCS liquid onto the hot surfaces in the core. This turbulence causes a blowdown of fission products from the drywell to the suppression pool thereby causing mixing. Return to the drywell is through the drywell to torus vacuum breakers.

The containment leak rate in the Technical Specifications is based on weight percent. However, the calculations are based on volume percent. If containment release rate is modeled in percent per day, the use of either weight percent per day or volume percent per day produces the same activity release rate because the terms mass or volume cancel to establish the activity release rate.

NRC Request 4c – Loss-of-Coolant Accident

For the drawdown time for the standby gas treatment system (SGTS), it is the staff's understanding that the actual performance at Dresden has shown the reactor building at positive pressure for a period of time following the SGTS receiving a start signal. This appears to be contrary to information provided in responses to a previous RAI and to the alternate source term submittal. Please provide information on those occurrences where the reactor building was not at a negative pressure. Provide an analysis which accounts for the reactor building being positive, or provide justification for the staff to conclude that the reactor building pressure is being maintained negative.

Response

Recent equipment problems have allowed reactor building differential pressure (dP) to drop, following SGTS receiving a start signal, with secondary containment not restored to ≥ 0.25 inch of vacuum water gauge in a timely manner. This situation occurred in early 2005 where, on one occasion, the drawdown time was approximately 56 minutes. During this event, the reactor building remained under negative pressure. Investigation determined the cause of the problem was due to a failed damper. The damper was repaired and modifications to the reactor building dP control circuit and several reactor building dampers were made. Several subsequent surveillance tests confirmed that the drawdown time has shortened to between one and 15 minutes and that the reactor building remains at a negative pressure at all times.

There were two recent instances where the reactor building dP indicated positive pressure upon a SGTS auto start signal. However, EGC believes that the reactor building pressure did not go positive. Investigation found faulty instrumentation that has since been repaired. Since that repair, DNPS has auto started SGTS and the building remained under negative pressure.

Based on the above, the assumption that the reactor building remains at a negative pressure is still valid and no analysis accounting for the reactor building being positive is required.

NRC Request 4d – Loss-of-Coolant Accident

The licensee's analysis assumes a leakage rate of one gallon per minute (gpm) for emergency core cooling system leakage contributions, but it is unclear how this number is supported by plant procedures. Please provide the bases for the assumed leakage rate, including a discussion of applicable plant procedures and surveillance practices which assure that actions are taken to limit actual leakage to one gpm.

Response

Procedure QCTP 0820-08, "Leakage Reduction," for QCNPS describes the methods utilized to monitor, survey, and document the preventive maintenance program for the reduction of leakage for those systems outside containment that could contain highly radioactive fluids during and after a serious transient or accident. The scope of the procedure includes systems which could be used for long-term post-LOCA service. The overall objective of this leak reduction program is to prevent leakage from systems to the building atmosphere, and thus minimize possible releases of excessive radioactivity to the environs.

Procedure QCTP 0820-08 states that the criteria for allowable leakage is "as low as practical." Section F.2 requires leakage that exceeds 10 gallons per hour (gph) to be captured in the Corrective Action Program by initiation of an Issue Report (IR). Section F.2 further requires the IR to specifically identify that the AST limit leakage has been exceeded if leakage exceeds 60 gph (i.e., one gpm). As stated in the procedure, an operability evaluation is required in this case due to exceeding the AST limit.

For DNPS, the corresponding procedure is DPT 09, "Leak Detection and Reduction Program." Procedure DPT 09 requires initiation of an IR for any leak discovered. Further, the procedure requires a followup inspection on all components exhibiting leakage to be conducted after maintenance to the component has been concluded. If the followup inspection continues to show leakage, then a new IR is initiated for repair.

Although procedure DPT 09 does not contain specific steps to address actions in the event of exceeding the AST limit of one gpm, the procedure clearly requires prompt corrective actions in the event of any leakage identified. Additionally, the Exelon Generation Company, LLC (EGC) Corrective Action Program requires all IRs to be reviewed for impact on operability. EGC will evaluate the need to revise DPT 09 to include wording similar to QCTP 0820-08 upon implementation of the AST license amendment.

NRC Request 4e – Loss-of-Coolant Accident

The staff's calculations do not result in a decontamination factor of 200 for elemental iodine in the drywell at 3.1 hours. Please provide additional details on the analytical assumptions and initial conditions used in determining the value of 3.1 hours.

Response

Section 7.7 of the revised LOCA calculations for DNPS and QCNPS (i.e., Attachments 8 and 9) provides the calculation for the time period that results in a decontamination factor of 200 for elemental iodine. The termination time for elemental iodine removal by wall surface deposition is now calculated to be 3.615 hours as shown below.

Natural deposition on containment surfaces (i.e., plateout) of the elemental iodine released to containment is calculated using the methodology outlined in NUREG-0800, Standard Review Plan 6.5.2 (i.e., page 6.5.2-10) as follows.

The equation for the wall deposition is:

$\lambda_w = K_w \times A/V$

Where:

λ_w = first order removal coefficient by wall deposition
K_w = mass transfer coefficient = 4.9 m/hr
A = wetted surface area = 32,250 ft^2
V = drywell net free air volume = 1.58E+05 ft^3
$\lambda_w = K_w \times A/V$ = 4.9 m/hr x (3.2808 ft/m) (32,250 ft^2) / (1.58E+05 ft^3) = 3.28 hr^{-1}

Maximum DF of elemental iodine = 200

$1/DF = e^{-(\lambda wt)}$
$1/200 = e^{-(3.28t)}$
$0.005 = e^{-(3.28t)}$

$\ln(0.005) = -3.28t$
$-5.298 = -3.28t$
$t = 1.615$ hr

The maximum iodine activity concentration takes place in the containment at the end of the early-in-vessel release phase (RG 1.183, Appendix A, Section 3.3), which is at 2.0 hr after the onset of a LOCA (RG 1.183, Tables 1 and 4).

Therefore, the termination time for elemental iodine removal by wall surface deposition is:

2.0 hr + 1.615 hr = 3.615 hr

NRC Request 5

With respect to control room dose calculations, Exelon indicated in a response to a previous request for information that operation of the control room ventilation system was assumed during all or parts of all accidents. Given that tracer gas inleakage testing was only done for emergency ventilation system operation, please provide information as to how Exelon has confirmed the inleakage characteristics of the control room envelope in the normal operating mode. Please discuss how the system will operate in the event of a loss of offsite power.

Response

In a public meeting with the NRC on August 17, 2005, EGC discussed the results of sensitivity studies that were performed to evaluate the impact of additional unfiltered inleakage into the DNPS and QCNPS control rooms prior to the operator action to manually align the Control Room Emergency Ventilation (CREV) system to the emergency mode of operation. The LOCA calculations for DNPS and QCNPS (i.e., Attachments 8 and 9) credit the normal mode of the control room HVAC system with an assumed unfiltered inleakage of 2000 cfm. During previous discussions with the NRC, the NRC questioned the adequacy of this assumption since tracer gas testing has not been performed specifically for the normal mode of operation. The sensitivity studies demonstrate that control room doses remain acceptable even with essentially equilibrium airborne activity concentrations between the control room air and the outside air for the assumed 40-minute period prior to initiation of the emergency mode of CREV system operation. Attachments 10 and 11 provide details regarding the assumptions, methodology, and conclusions of the sensitivity studies that demonstrate control room doses remain within allowable dose limits under these conditions.

NRC Request 6

Please discuss whether the control room X/Q values listed in Attachments 2 and 3 to the November 5, 2004, response to an RAI are limiting. Please address whether there are penetrations or other possible release locations that are closer to the control room intake which would result in higher X/Q values and whether factors such as loss of offsite power or single-failure (e.g., loss of forced ventilation) would result in more limiting X/Q values. Page 7 of Attachment 4 states that only one release scenario was considered for each of the design-basis accidents. Please discuss the bounding nature of these scenarios for each design-basis accident.

Response

Release points have been chosen for each pathway and are indicated in the X/Q calculations provided in Attachments 16 and 17 (i.e., Attachments C and E of the calculations for QCNPS and DNPS, respectively). Only the worst-case release points were selected. The release points for the LOCA are identified as the station chimney (i.e., elevated release) and the steam tunnel (i.e., MSIV ground level release). For the CRDA, the release points are the elevated stack (i.e., for potential steam jet air ejector releases considered), and the MSIV release point (i.e., steam tunnel at grade level). An unobstructed release point is considered to bound pathways that would involve the assumed 1% per day condenser leakage releases to an unventilated turbine building with eventual releases chiefly though grade level openings. For the FHA, the release point is the reactor building exhaust stack (i.e., unfiltered ground level release). This unobstructed release point is considered to bound pathways that would involve diffuse releases to an unventilated building with eventual releases chiefly though grade level openings. There are no roof penetrations closer to the intake than the reactor building exhaust stack. The MSLB uses a hemispherical cloud model and is not dependent on X/Q.

NRC Request 7

Please discuss whether the elevated release X/Q values in Attachments 2 and 3 to the November 5, 2004, letter adequately reflect reduced effective plume heights due to terrain. The PAVAN input files provided as an enclosure to the November 5, 2004, letter, do not include arrays of terrain heights as a function of distance and direction. Page 2 of the attachment to an October 31, 2003, response to an RAI states that there is a general lack of complexity to the terrain surrounding both sites, but estimates a maximum terrain height of about 25 feet at an unspecified location. Therefore, please provide a table of terrain heights as a function of distance and direction suitable for input into PAVAN. In addition, please discuss the results for downwind sectors with increases in terrain height and whether the resultant X/Q values in those sectors are still lower than the limiting X/Q values presented in Attachments 2 and 3 of the November 5, 2004, letter.

Response

A 100 ft bluff located 1300 meters in the north through east sectors at DNPS is the only significant terrain feature historically considered for the station in the Updated Final Safety Analysis Report (UFSAR). The QCNPS UFSAR states "there are no topographical features which significantly affect the local meteorology." Therefore, terrain effects were not previously considered for Exclusion Area Boundary (EAB) and Low Population Zone (LPZ) X/Q value determinations. The X/Q calculations for DNPS and QCNPS have now been revised to consider terrain effects for the EAB X/Q values for the Station Chimney release pathway. This has resulted in changes to the X/Q values. The revised calculations for DNPS and QCNPS are provided as Attachments 16 and 17, respectively. Section 3.2.1.2 of the revised calculations provides a table of terrain heights in the vicinity of the EAB as a function of distance and direction for PAVAN, and details regarding the impact of these terrain effects. Additionally, the revised EAB X/Q values have been applied in the revised radiological dose calculations for DNPS and QCNPS.

For terrain effects for the Station Chimney release pathway on LPZ X/Q values, a bounding sensitivity analysis was performed using the conservative assumption of a terrain height equal to the Station Chimney height of 94.6 meters at the QCNPS and DNPS LPZ distances of 4828 meters and 8000 meters, respectively. The impact of the LPZ X/Q values for this terrain assumption was assessed for the previously analyzed LOCA doses, and as expected, increased the calculated doses. However, this increase only impacts elevated releases, and even with these values the LPZ doses remain less than the EAB doses for the LOCA. All calculated offsite doses remain below the 25 rem TEDE dose limit specified in 10 CFR 50.67. Details regarding the sensitivity analysis described above are provided in Attachments 20 and 21 for DNPS and QCNPS, respectively.

Therefore, the historical practice of not applying detailed terrain consideration at and beyond the LPZ boundary remains acceptable. Because of the controlling nature of the EAB dose, such detailed LPZ investigations are not necessary.

No credit is taken for an elevated release for any other design basis accidents.

NRC Request 8

For purposes of considering the effects of possible downwash in the ARCON96 X/Q calculations, please provide the station chimney diameter(s) and estimated effluent vertical velocities and stack flows during the applicable design-basis accidents.

Response

The effects of stack downwash were considered in the revised X/Q calculations for DNPS and QCNPS (i.e., Attachments 16 and 17, respectively). Additional ARCON96 model runs were completed with downwash incorporated by setting the vertical velocity (i.e., derived from the stack flow divided by the stack top internal area), stack radius (i.e., to the outside edge at the stack top), and stack flow (i.e., the design SGTS flow) to 0.21 m/s, 1.86 m, and 1.89 m^3/s, respectively, based on values from the DNPS and QCNPS UFSARs. This has resulted in changes to the X/Q values. Section 2.2.1 and Table 2-3 of the revised calculations provide details regarding the effects of stack downwash. Additionally, the revised X/Q values have been applied in the revised radiological dose calculations for DNPS and QCNPS.

NRC Request 9

Please confirm that the direction from the Dresden control room intake looking toward the station chimney is approximately 48 degrees from true north (i.e., in a northeasterly direction).

Response

The DNPS plant north and true north are the same. Therefore, the direction from the DNPS control room intake looking toward the station chimney is approximately 48 degrees from true north (i.e., in a northeasterly direction).

<u>NRC Request 10</u>

Please provide a reference citing when Regulatory Guide 1.5 was previously used and reviewed by the staff to estimate the exclusion area boundary and low population zone X/Q values for the main steamline break accident dose assessments.

<u>Response</u>

The DNPS and QCNPS UFSARs refer to Regulatory Guide 1.5 (i.e., Safety Guide 5) for main steam line break accident evaluations (i.e., page 15.6-23 of the DNPS UFSAR and page 15.6-11 of the QCNPS UFSAR). The DNPS and QCNPS AST main steam line break analyses (i.e., Attachments 6 and 7, respectively) are being provided to the NRC for confirmatory review of where the Regulatory Guide 1.5 approach, as referenced in the UFSAR, is utilized, consistent with the historic manner of main steam line break accident dose analyses per the current licensing design basis.