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

Braidwood Station, Units 1 and 2
Facility Operating License Nos. NPF-72 and NPF-77
NRC Docket Nos. STN 50-456 and STN 50-457

Byron Station, Units 1 and 2
Facility Operating License Nos. NPF-37 and NPF-66
NRC Docket Nos. STN 50-454 and STN 50-455

Subject: Response to NRC Request for Additional Information With Respect to Request for License Amendment Related to Application of Alternative Radiological Source Term

References:

(1) Letter from K. R. Jury (Exelon Generation Company, LLC) to NRC, "Request for License Amendment Related to Application of Alternative Radiological Source Term," dated February 15, 2005

(2) Letter from J. A. Bauer (Exelon Generation Company, LLC) to NRC, "Additional Information Related to Application of Alternative Radiological Source Term – Atmospheric Dispersion Coefficients," dated December 9, 2005

(3) Letter from J. A. Bauer (Exelon Generation Company, LLC) to NRC, "Additional Information Related to Application of Alternative Radiological Source Term – Dose Calculations," dated December 9, 2005

In Reference 1, Exelon Generation Company, LLC (EGC) requested an amendment to Appendix A Technical Specifications (TS), of Facility Operating License Nos. NPF-72, NPF-77, NPF-37, and NPF-66 for Braidwood Station, Units 1 and 2, and Byron Station, Units 1 and 2, respectively. The proposed amendment was requested to support application of an alternative source term methodology in accordance with 10 CFR 50.67, "Accident Source Term." During a conference call on December 1, 2005, NRC technical reviewers and EGC personnel discussed aspects of the Reference 1 submittal that required additional information to be provided for NRC review. Calculations supporting implementation of an alternative radiological source term were previously submitted to the NRC in References 2 and 3. Attachment 1 to this letter provides the EGC responses to the remainder of the NRC requests for information discussed during the

December 1, 2005, conference call. Attachment 2 contains a list of commitments made in this submittal.

This additional information does not affect the supporting analysis for the original license amendment request as described in Reference 1. No other information submitted with Reference 1 is affected by this additional information. The No Significant Hazards Consideration and the Environmental Consideration provided in Attachment 1 of Reference 1 are not affected by this additional information.

In accordance with 10 CFR 50.91(b), "State consultation," EGC is providing the State of Illinois with a copy of this letter and its attachments to the designated State Official.

If you have any questions about this letter, please contact David Chrzanowski at (630) 657-2816.

I declare under penalty of perjury that the foregoing is true and correct. Executed on the 13th day of February 2006.

Respectfully,

Joseph A. Bauer

Joseph A. Bauer
Manager – Licensing

Attachments:

Attachment 1: Response to NRC Request for Additional Information

Attachment 2: List of Commitments

Attachment 1

BRAIDWOOD STATION
UNITS 1 AND 2

Docket Nos. STN 50-456 and STN 50-457
License Nos. NPF-72 and NPF-77

and

BYRON STATION
UNITS 1 AND 2

Docket Nos. STN 50-454 and STN 50-455
License Nos. NPF-37 and NPF-66

Response to NRC Request for Additional Information

Attachment 1

Response to NRC Request for Additional Information

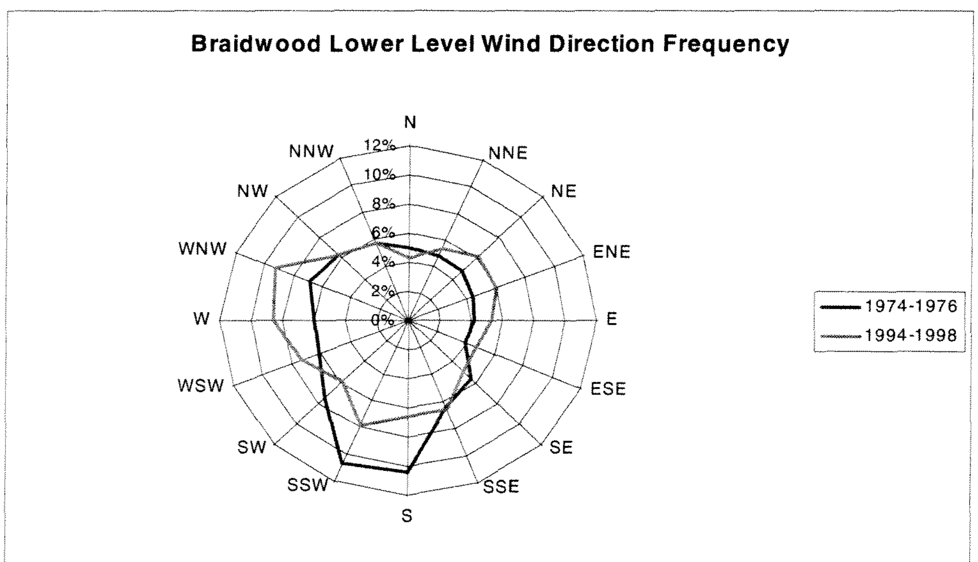
Meteorological Requests for Additional Information (RAIs)

1. The two methods used for determining atmospheric stability were delta-temperature (vertical temperature difference) and sigma theta (standard deviation of the horizontal wind direction). Delta-temperature was the principal method whereas sigma theta was used when delta-temperature data were not available. These two stability classification schemes do not always correlate well and the ARCON96 model is an empirical model based on field data conducted with stability data based on delta-temperature measurements as defined in Regulatory Guide (RG) 1.23, "Onsite Meteorological Programs." Please identify (a) the amount of stability class data based on sigma theta measurements as opposed to delta-temperature measurements, (b) the criteria chosen to implement the use of delta-temperature data in lieu of sigma theta data, and (c) the potential impact on the resulting dispersion analyses.

   Response:

   Calculation BYR04-050 & BRW-04-0044-M, "Calculation of Alternative Source Term Onsite and Offsite Atmospheric Dispersion Coefficients," Revision 1, dated November 30, 2004, was previously provided to the NRC in Reference 2. Substitution of delta-temperature with sigma theta information was not required or performed because the Byron Station and Braidwood Station joint data recoverability was consistently greater than the minimum required by RG 1.23 (i.e., 90%). Consequently, there is no impact on the resulting dispersion analyses.

2. A wind rose comparing the Braidwood lower level wind direction frequency distributions between the 1974 - 1976 data presented in Braidwood Updated Final Safety Analysis Report (UFSAR) Table 2.3-4 and the 1994 - 1998 data presented in Attachment 8 to your application letter dated February 15, 2005, is provided below:



This wind rose shows considerable discrepancies for the following wind direction sectors:

Period of Record	Wind Direction Sector			
	S	**SSW**	**W**	**WNW**
1974 - 1976	10.3%	10.7%	5.9%	6.8%
1994 - 1998	6.6%	7.7%	8.5%	9.1%

Please explain what might have caused these discrepancies in reported wind direction frequency distributions between the 1974 - 1976 and 1994 - 1998 data sets (e.g., changes in tower structure and/or boom orientation, vegetation growth, etc.) and the potential impact on the resulting atmospheric dispersion analyses.

Response:

Murray and Trettel, Inc. (the Exelon Generation Company, LLC (EGC) meteorological contractor) first began maintaining the meteorological equipment in 1975 and began providing reports based on data collected from May 1975 to present.

The period of record from 1974 through 1976 utilizes only three full years of data. The period of 1994 through 1998 utilizes five full years of data. The smaller data set associated with the 1974 through 1976 data may have had some impact on the wind direction sector frequencies. The wind direction sector frequencies for 1994 through 1998 are more consistent with what would be anticipated based upon known geographic location and topography.

The 1974 through 1976 data set was obtained prior to plant startup in 1987 and 1988. At that time, the man-made cooling lake had not been completely constructed and did not have any water inventory that could have affected local meteorology. Although this may only have affected the wind speed due to less ground friction, it is uncertain if it could also have affected the wind direction.

There were no changes in tower structure, boom orientation, or obvious instrumentation issues that could account for the lower level wind direction shift over the previous 20 years. There is no significant vegetation in the immediate vicinity of the meteorological monitoring tower. Routine calibrations and maintenance were performed on the required meteorological monitoring tower instruments. Murray and Trettel, Inc. noted that the lower level wind sensors were replaced in November 1993.

An additional review of the Braidwood Station meteorological tower (lower level) year-by-year wind direction percentage occurrence frequencies for 1994-1998 shows that there is a relatively higher degree of variability with respect to wind directions SSW, W and WNW than the other directions as shown in the Table below. These happen to be three of the four directions associated with the percentage occurrence "discrepancies" identified by the NRC. The Table shows that the maximum wind direction occurrence is either SSW or WNW in each of the five years. The Table also indicates that annual occurrence frequency varies more for these two directions than the others. West, the second most frequent wind occurrence direction, is third highest in annual variation.

While there may be other factors contributing to the differences identified by the NRC in the 1974-1976 and 1994-1998 Braidwood Station meteorological tower databases (e.g., topographical changes to the site environs), the natural climatological variability cannot be excluded as a significant contributor.

Braidwood Station
Meteorological Tower Lower Level Wind Direction Variability

Year	Wind Direction Category															
	N	NNE	NE	ENE	E	ESE	SE	SSE	S	SSW	SW	WSW	W	WNW	NW	NNW
1994	4.35	5.04	6.46	5.88	5.17	4.42	5.09	5.68	7.21	9.63	5.56	7.45	7.26	9.19	6.51	5.12
1995	3.48	4.31	5.57	5.95	5.23	4.80	4.51	6.41	7.15	6.18	5.53	7.90	9.92	10.22	6.84	5.89
1996	5.35	5.07	5.74	5.71	4.69	4.34	5.95	7.46	7.01	7.33	5.53	6.34	8.14	8.60	6.23	6.40
1997	3.78	5.79	7.16	6.24	5.29	4.29	4.32	6.14	5.64	6.98	6.49	6.87	9.04	10.31	5.91	5.67
1998	4.14	6.12	5.55	6.07	5.86	5.41	5.02	6.90	5.67	8.57	6.28	7.63	8.19	7.23	5.95	5.21
1994-1998 Avg.	**4.22**	**5.26**	**6.09**	**5.97**	**5.25**	**4.65**	**4.98**	**6.52**	**6.54**	**7.74**	**5.88**	**7.24**	**8.51**	**9.11**	**6.29**	**5.66**
Difference in Max - Min Years	**1.87**	**1.81**	**1.61**	**0.52**	**1.17**	**1.12**	**1.64**	**1.78**	**1.56**	**3.45**	**0.96**	**1.56**	**2.66**	**3.08**	**0.93**	**1.28**
Maximum Annual Deviation from 5-yr Avg.	**1.13**	**0.96**	**1.06**	**0.27**	**0.61**	**0.76**	**0.98**	**0.94**	**0.89**	**1.89**	**0.61**	**0.90**	**1.41**	**1.88**	**0.55**	**0.74**

The 1994-1998 Braidwood Station lower level wind direction data set is considered more representative of current conditions than the 1974-1976 data set.

3. Please identify the location of the air intake used during the control room maximum 100 percent outdoor air purge mode, and state whether this intake was considered as a potential unfiltered inleakage pathway.

<u>Response:</u>

The 100 percent outdoor air purge is performed using the same control room (CR) air intake as the normal CR fresh air intake depicted as Receptor Nos. R1 and R2 on Figure 4.1.1-1, "Byron/Braidwood Control Room Receptors and Release Points," in Attachment 1 of Reference 1. The purge is provided with a separate duct connection to the above outside air intake, and a separate isolation damper. The purge air intake isolation damper remains closed during normal and emergency operation. This purge air intake isolation damper and ductwork is considered as a potential unfiltered air inleakage pathway during normal operation. Any inleakage from it would have been identified during performance of the tracer gas test. However, this air inleakage occurs upstream of the recirculation filter, and therefore it is filtered before entering the CR envelope during the emergency operating mode.

4. Unfiltered leakage into the control room was apparently modeled using the outside air intake χ/Q values during the initial 30 minutes of each design basis accident and turbine building emergency air intake χ/Q values after this period. Please confirm that there are no potential unfiltered inleakage pathways during both normal and emergency operating modes that could result in χ/Q values that are higher than the control room air intake χ/Q values used.

   Response:

   For the first 30 minutes of the accident, the 1000 cfm of assumed unfiltered air inleakage was conservatively considered to be from the normal CR intake, with the CR Turbine Building Emergency Air Intake (i.e., Reference 1, Attachment 1, Figure 4.1.1-1 Receptor Nos. R3 and R4) considered for the remaining duration of the accident. The normal outside air intake is isolated after the first 30 minutes following an accident, and the outside air is then drawn in from Turbine Building Emergency Air Intake, which is filtered by the make-up filter unit before it is supplied to the CR Heating, Ventilating, and Air Conditioning (HVAC) system.

   Therefore the assumed CR unfiltered inleakage of 1000 cfm was modeled using the outside air normal fresh air intake χ/Q values during the initial 30 minutes of each design basis accident, and the turbine building emergency air intake χ/Q values after this period. Site walkdowns, relevant drawing reviews, and acceptable CR tracer gas test results have confirmed that there are no potential unfiltered inleakage pathways during either normal or emergency operating modes that could result in higher χ/Q values than the control room air intake χ/Q values used, with the exception of isolation dampers associated with the normal outside air intake. In the emergency operating mode, inleakage through these isolation dampers would be filtered by the recirculation filters. As a result, it is more conservative to treat this inleakage as unfiltered with the emergency intake χ/Q 's than as filtered with the normal outside air χ/Q's.

5. Please list and provide the basis for the χ/Q values used to model doses resulting from loss-of-coolant accident (LOCA) emergency core cooling system (ECCS) leakage into the auxiliary building.

   Response:

   Calculation BYR04-050 & BRW-04-0044-M and Calculation BYR04-051 & BRW-04-0038-M, "Re-analysis of Loss of Coolant Accident (LOCA) Using Alternative Source Terms," Revision 1, dated November 30, 2004, were previously provided to the NRC in References 2 and 3, respectively. The requested listing of χ/Q values used to model doses is provided on page 14 of the main body of Calculation BYR04-051 & BRW 04-0038-M. The bases for and derivation of these χ/Q values are provided in Calculation BYR04-050 & BRW 04-0044-M.

   For the post-LOCA ECCS leakage into the Auxiliary Building, the plant vent is the eventual release location (i.e., Release Point Nos. RP3 and RP4 as depicted on Figure 4.1.1-1, "Byron/Braidwood Control Room Receptors and Release Points," provided in Attachment 1 to Reference 1). All Auxiliary Building releases, both normal and accident, are captured by the Auxiliary Building ventilation system and exhausted through the plant vent. The CR Fresh Air Intake for the applicable CR is assumed as the

receptor location for the first 30 minutes and the CR Turbine Building Emergency Air Intake for the applicable CR is assumed as the receptor location for the remaining duration of the accident for the respective Units. The χ/Q values used represent the most conservative values for this Auxiliary Building release pathway. These pathways are the same as those used in the current license design basis described in the Byron Station and Braidwood Station UFSAR.

6. Please list, and provide the basis for, the χ/Q values used to model doses resulting from steam generator tube rupture (SGTR) releases through the condenser.

   Response:

   Calculation BYR04-050 & BRW-04-0044-M and Calculation BYR04-048 & BRW-04-0042-M, "Re-analysis of Steam Generator Tube Rupture (SGTR) Accident Using Alternative Source Terms," Revision 1, dated November 30, 2004, were previously provided to the NRC in References 2 and 3, respectively. The requested listing of χ/Q values used to model doses are provided on pages 12 and 13 of the main body of Calculation BYR04-048 & BRW-04-0042-M. The bases for and derivation of these χ/Q values are provided in Calculation BYR04-050 & BRW 04-0044-M.

   All of the post-SGTR releases, whether through the condenser or not, are considered to utilize the SG PORVs as the release location. The CR Fresh Air Intake is the applicable CR receptor location for the first 30 minutes and the CR Turbine Building Emergency Air Intake is the applicable CR receptor location for the remaining duration of the accident, for the respective Units. The SG PORV location closest to the CR Intakes for each Unit was utilized for conservatism since leakage through the condenser, in addition to being subject to more removal mechanisms, would exit through the plant vent which is a greater distance from the postulated CR receptor locations than the PORVs. Although the CR HVAC system will automatically align to the emergency mode, for conservatism, the analysis assumes that for the first 30 minutes, the SG PORV χ/Q is used with the normal CR outside air intake.

7. Braidwood and Byron each have four main steam lines for each unit. Each steam line has a set of five safety valves and one hydraulically operated atmospheric relief valve (steam generator power-operated relief valve (SG PORV)). The following questions concern the atmospheric dispersion modeling for the SG PORVs/safety valve release pathways:

   a. Please explain the criteria used to define the release locations for determining the horizontal distances and directions between the SG PORVs/safety valve release locations and the control room air intakes.

      Response:

      The release location distances utilized were "the shortest horizontal distance between the release point and the intake" as per RG 1.194, "Atmospheric Relative Concentrations for Control Room Radiological Habitability Assessments at Nuclear Power Plants," Section 3.4. Locations of the SG PORV/safety valves are based on design information and are indicated in calculation BYR04-050 & BRW-04-0044-M, Attachments E and F. The distances and directions used for the SG PORVs/safety

valve release locations and the control room air intakes are summarized in the tables in calculation BYR04-050 & BRW-04-0044-M, Attachments K and L.

b. A factor of five reduction in the ARCON96 χ/Q values was taken due to the energetic releases from the SG PORVs/safety valve release pathways. Guidance in RG 1.194 states that this credit is justified if (1) the release point is uncapped and vertically oriented and (2) the time-dependent vertical velocity exceeds the 95th percentile wind speed at the release point height by a factor of five. Both the locked rotor accident (LRA) and the SGTR analyses assume a SG PORV failure in the open position at the onset of the accident releases for 20 minutes. Please confirm that saturated steam choked sonic flow conditions continue to persist during these 20 minutes such that the SG PORV vertical exit velocity is always exceeding the 95th percentile wind speed by at least a factor of five for these accident scenarios.

Response:

As a precedent for the use of a factor of 5 credit applied to a plant similar to Byron and Braidwood Stations, the Virginia Electric and Power Company letter to the NRC dated November 20, 2003 (i.e., Letter from Leslie N. Hartz to U.S. NRC, Serial No. 03-464A, "Virginia Electric and Power Company North Anna Power Station Units 1 and 2 - Proposed Technical Specification Changes Implementation of Alternative Source Term RAI," ADAMS Accession number ML0333505160) was reviewed. The evaluation provided in this letter shows SG PORV steam exhaust releases are uncapped, vertically oriented, and have vertical velocities about an order of magnitude above the threshold needed to qualify for the factor of 5 reduction for the bounding choked sonic flow conditions of the saturated steam release. Like North Anna, the Byron Station and Braidwood Station SG PORVs are uncapped and vertically oriented with a 10-inch diameter exhaust stack. The 95th-percentile wind speeds for Byron Station are 7.61 meters per second and 6.89 meters per second at Braidwood Station for the 30 foot tower levels, and 11.26 meters per second at Byron Station and 10.92 meters per second at Braidwood Station for the 250 foot tower levels.

RETRAN computer program modeling of the Byron Station and Braidwood Station responses to a SGTR event indicate that the flow through the ruptured tube SG PORV remains choked for the duration of steaming (i.e., 20 minutes). An engineering evaluation using a RELAP5 model of the Byron Station and Braidwood Station SG PORVs confirmed choked flow conditions at the exit of the discharge piping with steam velocity exceeding 1200 feet per second (~375 meters per second) throughout the pressure range associated with SGTR and LRA. It is therefore concluded that the factor of 5 reduction is acceptable at Byron and Braidwood Stations for SGTR and LRA accidents involving SG PORV release from the steam generators.

c. Please provide the basis for not assuming a failed SG PORV for the control rod ejection accident and main steamline break (MSLB) analyses.

Response:

Calculation BYR04-045 & BRW-04-0039-M, "Re-analysis of Control Rod Ejection Accident (CREA) Using Alternative Source Terms," Revision 1, dated November 30, 2004, and Calculation BYR04-046 & BRW-04-0040-M, "Re-analysis of Main Steam Line Break (MSLB) Accident Using Alternative Source Terms," Revision 1, dated November 29, 2004, were previously provided to the NRC in Reference 3.

Consistent with the current licensing basis, a failed SG PORV is not assumed for the MSLB and CREA analyses. The current main steam line break and control rod ejection analyses are performed in accordance with the analytical methods and assumptions outlined in NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants," Section 15.1.5, "Steam System Piping Failures Inside and Outside of Containment (PWR)," and RG 1.77, "Assumptions Used for Evaluating a Control Rod Ejection Accident for Pressurized Water Reactors," May 1974, respectively.

8. The general description for the LRA states that an instantaneous seizure of a reactor coolant pump rotor results in a pressure increase throughout the reactor coolant system, opening the pressurizer power-operated relief valves (PORVs) and safety valves. Please explain how doses resulting from the activity release from the pressurizer PORVs and safety valves were modeled.

Response:

Calculation BYR04-049 & BRW-04-0043-M, "Re-analysis of Locked Rotor Accident (LRA) Using Alternative Source Terms," Revision 1, dated November 30, 2004, was previously provided to the NRC in Reference 3. All activity releases outside of containment are conservatively considered via the SG PORVs. A concurrent SG PORV failure is also evaluated. The pressurizer PORVs and safety valves are not considered as release paths. This is consistent with the existing license design basis described in UFSAR Chapter 15.3.3, "Decrease in Reactor Coolant System Flow Rate."

9. Concerning the atmospheric dispersion modeling for the MSLB release pathways: please explain the criteria used to define the release locations for determining the horizontal distances and directions between the MSLB release locations and the control room air intakes.

Response:

Calculation BYR04-050 & BRW-04-0044-M Section 2.3.1.1 (page 8 of 28) discusses the MSLB release point. Attachments E and F of the calculation show the relationship between release and intake locations. The current licensed design basis was utilized as a basis. The distances were determined by "taut string length" calculations as shown in calculation BYR04-050 & BRW-04-0044-M, Attachments I and J. The distances and directions used for the assumed MSLB release locations and the control room air intakes

are summarized in the tables in calculation BYR04-050 & BRW-04-0044-M, Attachments K and L.

10. The PAVAN atmospheric dispersion computer code produces the best results for five percentile X/Q values if the wind speed data are classified into a large number of categories at the lower wind speeds, thus preventing the data from being clustered into a few categories. The joint frequency distributions used for the PAVAN runs show that approximately 12 percent of the Braidwood wind speed data and 16 percent of the Byron wind speed data fall within the first wind speed class which ranges from 0.8 miles per hour (mph) (0.36 meters per second (m/sec)) to 3.5 mph (1.56 m/sec). The staff ran PAVAN using a finer wind speed category breakdown for the lower wind speeds and generated more conservative X/Q values. Consequently, please justify why a finer breakdown of wind speed categories (e.g., 0.5, 0.75, 1.0, 1.25, 1.5, 2.0, 3.0, 4.0, 5.0, 6.0, 8.0, and 10 m/sec as discussed in Section 4.6 of NUREG/CR-2858) should not be used in running the PAVAN computer code.

Response:

Based on existing NRC guidance, RG 1. 23, "Onsite Meteorological Programs," was used as the basis for Calculation BYR04-050 & BRW-04-0044-M, which utilizes seven wind speed categories in the PAVAN modeling.

The EAB/LPZ calculated doses, limits, and allowable margins to the respective limits are shown in the following Table. As can be seen in the table, considerable margin exists for each accident.

	EAB Doses and Limits		LPZ Doses and Limits		Margin to EAB Limit (REM TEDE)	Percentage Increase Required to Reach EAB Limit	Margin to LPZ Limit (REM TEDE)	Percentage Increase Required to Reach LPZ Limit
	Dose (REM TEDE)	Limit (REM TEDE)	Dose (REM TEDE)	Limit (REM TEDE)				
Loss of Coolant Accident	12.2	25	2.99	25	12.8	105%	22.01	736%
Main Steam Line Break	0.127	25	0.073	25	24.873	19585%	24.927	34147%
	0.175	2.5	0.406	2.5	2.325	1329%	2.094	516%
Control Rod Ejection Accident	4.647	6.3	1.983	6.3	1.653	36%	4.317	218%
Locked Rotor Accident	1.456	2.5	0.525	2.5	1.044	72%	1.975	376%
Steam Generator Tube Rupture	0.721	25	0.165	25	24.279	3367%	24.835	15052%
	0.327	2.5	0.077	2.5	2.173	665%	2.423	3147%
Fuel Handling Accident	4.24	6.3	0.356	6.3	2.06	49%	5.944	1670%

During a conference call on December 5, 2005, NRC representatives and EGC personnel agreed that the basis for determining X/Q values using the seven wind speed categories provided by the current revision of RG 1.23 is appropriate for the current

license amendment submittal. EGC, however, is committing to update calculation BYR04-050 & BRW-04-0044-M to reevaluate χ/Q values based on finer wind speed categories provided in the latest appropriate regulatory guidance the next time χ/Q calculations associated with the dose consequences of the above listed accidents are revised.

Design Basis Accidents (DBA) Dose Analysis RAIs

LOCA

1. Attachment 7, Table B of the February 15, 2005, submittal states that the Byron and Braidwood containments can be considered to be routinely purged during power operation, and that the purge dose contribution is summed with the postulated doses from other release paths. A discussion of the purge dose analysis does not appear in Attachments 1 or 6 of the submittal, nor does Table 4.5.3-1 in Attachment 1 list the purge dose results. Please provide the missing information on the purge dose calculation, including a discussion of the analysis and the analysis assumptions, inputs and results.

   Response:

   The current licensing basis analysis does not consider purge doses. A technical evaluation of the purge dose contribution was performed assuming an unfiltered release path via the containment miniflow purge system for a period of seven seconds until isolation occurs. This evaluation used modified Alternative Source Term (AST) LOCA analysis RADTRAD files to determine doses due to containment purge. These modifications include an updated nuclide inventory file to reflect the maximum coolant isotopic inventory permitted by Technical Specifications (TS), a release fraction and timing file that reflects iodine and noble gas isotopes, as well as a single containment volume (assuming no spray), which is released at a rate of 3000 cfm for a period of seven seconds (five seconds for valve closure and two seconds for instrument response).

   The results of this evaluation indicated a dose of approximately 5.24E-04 rem TEDE for the CR, 1.69E-04 for the EAB, and 1.42E-05 for the LPZ using this conservative seven second valve closure time from initiation of the event. None of these doses results in a significant change to the doses reported in the original submittal. These doses are consistent with those reported for Seabrook Station, a pressurized water reactor site comparable in size to Byron and Braidwood Stations, in Reference 4 (i.e., 3.84E-04 rem TEDE for the CR, 4.24E-04 for the EAB and 2.06E-04 for the LPZ).

2. RG 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents AT Nuclear Power Reactors," Appendix A, Section 5, "Assumptions on ESF System Leakage," states that the release source may also include leakage through valves isolating interfacing systems. See also Information Notice 91-56, "Potential Radioactive Leakage to Tank Vented to Atmosphere," dated September 19, 1991. The dose contribution from leakage from the ECCS does not include back leakage into the refueling water storage tank (RWST), which may be vented to atmosphere. Please provide a justification for not including this leakage pathway, or provide the missing information on the LOCA RWST back leakage pathway analysis, including a discussion of the analysis and the analysis assumptions, inputs and results.

   Response:

   Potential back leakage to the RWST was previously addressed in Reference 5.

Fuel-Handling Accident (FHA)

3. In Section 4.6.3 of Attachment 1 of the submittal, it is stated that an additional FHA analysis was performed for RECENTLY IRRADIATED FUEL with containment closure established or with the fuel handling building ventilation system operable, assuming a decay time of six hours. Please provide the missing information on the FHA analysis for RECENTLY IRRADIATED FUEL, including a discussion of the analysis and the analysis assumptions, inputs and results.

   Response:

   Calculation BYR04-047 & BRW-04-0041-M, "Re-analysis of Fuel Handling Accident (FHA) Using Alternative Source Terms," Revision 1, dated November 29, 2004, was previously provided to the NRC in Reference 3. Section 3 of the calculation (page 11) lists the assumptions/inputs used in the analysis of the FHA. The RADTRAD output in Attachment B of the calculation (beginning on page B-31) assumes a six hour decay period from the time of shutdown. This demonstrates that a decay period of only six hours prior to fuel movement would be acceptable, from a radiological standpoint, provided that containment closure is established and/or the fuel handling building ventilation system is operable as required by TS 3.9.4, "Containment Penetrations." Moving fuel within six hours of reactor shutdown is not acceptable from a radiological standpoint; however, fuel movement within six hours is not possible based on system constraints.

4. Would movement of RECENTLY IRRADIATED FUEL be prevented (i.e., what prevents movement of fuel before 48 hours of decay have elapsed)? If so, by what means is movement prevented?

   Response:

   Movement of RECENTLY IRRADIATED FUEL would be allowed by the proposed TS. As discussed above, a FHA assuming six hours of decay has elapsed and shows acceptable dose consequences provided that containment closure is established or the fuel handling building ventilation system is operable as required by the proposed TS. It should be noted that moving RECENTLY IRRADIATED FUEL in less than six hours from

the time of shutdown is not considered to be possible because of the activities that must be performed in order to allow movement of fuel in the reactor vessel to occur. It typically requires in excess of 54 hours to complete the activities that are necessary to allow movement of fuel in the reactor vessel. Cooldown from Mode 3 to Mode 5 alone typically requires approximately six hours. The Technical Requirements Manual (TRM) provides decay time restrictions. The decay time assures that the radiological assumptions for a fuel handling accident in containment are met. Another potentially more limiting consideration in determining an acceptable decay time is the impact of decay time on the spent fuel pool cooling requirements. An acceptable decay time is evaluated on a cycle-specific basis and considers the expected heat load in the spent fuel pool. Any changes to the TRM are evaluated in accordance with 10 CFR 50.59, "Changes, Tests, and Experiments."

5. Please provide discussion on the commitments to mitigate the consequences of a potential fuel handling accident in accordance with TSTF-51 and consistent with the guidance in NUMARC 93-01, "Safety Assessment for Removal of Equipment from Service During Shutdown Conditions," Revision 3. In particular, discuss the contingency plan to promptly close the containment equipment hatch, personnel airlock or other containment penetrations. How will the closure of the containment be achieved and what degree of closure will be attained? In what length of time are you committing to close the containment? Is the use of dedicated personnel proposed? Will the contingency plan be put into procedures? What training requirements are proposed?

   Response:

   The Braidwood Station and Byron Station containments are equipped with equipment hatches located at elevation 426'. The equipment hatches are connected to the Fuel Handling Building (FHB). The equipment hatch permits transfer of large equipment into and out of containment and is also furnished with a personnel airlock to facilitate containment access during power operation. An emergency hatch with a personnel airlock is also provided at ground level (i.e., elevation 401') opposite the equipment hatch side of containment and opens to the outside atmosphere. The emergency hatch is used as an emergency escape route, an alternate containment access at power, and as routine access for personnel and equipment into and out of the containment during cold shutdown, refueling mode and when the reactor is defueled.

   Reference 1, Attachment 5, provided a general description of how Braidwood Station and Byron Station would comply with the guidance provided in NUMARC 93-01, Section 11.3.6.5. The purpose of the NUMARC 93-01, Section 11.3.6.5 contingencies is to enable ventilation systems to draw the release from a postulated FHA in the proper direction such that it can be treated and monitored. It is not necessary, nor desirable, to maintain a contingency to close the containment equipment hatch or its associated personnel air lock doors during a postulated fuel handling accident to ensure that the purpose of NUMARC 93-01, Section 11.3.6.5 is met for Braidwood Station and Byron Station.

   The equipment hatches and associated personnel airlocks at Braidwood Station and Byron Station open into the FHB and do not communicate with the outside environment. The containment becomes a part of the FHB Ventilation System envelope when open (i.e., the equipment hatch and/or its associated personnel airlock is/are open). The FHB

Ventilation System is designed to mitigate the consequences of the FHA, by providing sufficient negative pressure to direct a release resulting from a postulated FHA to a filtered release pathway. Closure of the equipment hatch requires approximately four hours. In order to close the equipment hatch, a crane operator is required to man the containment polar crane for at least three hours. The release from a postulated FHA scenario would subject the crane operator to an un-inhabitable environment from an iodine dose perspective long before the task could be completed.

The current licensing basis permits the equipment hatch to remain open during handling of irradiated fuel in the containment. This basis assumes that the containment / FHB envelope is at a negative pressure with regards to the outside environment and the FHB Ventilation System is operable as required by TS 3.7.13, "Fuel Handling Building Exhaust Filter Plenum (FHB) Ventilation System." Following implementation of the proposed AST TS changes, TS 3.7.13 will only be applicable during movement of RECENTLY IRRADIATED FUEL, i.e., fuel that has occupied part of a critical reactor core within the previous 48 hours.

Reference 1, Attachment 5 provides a description of the contingencies related to the FHB Ventilation System that will be implemented during movement of irradiated fuel (i.e., irradiated fuel or RECENTLY IRRADIATED FUEL) in the containment or FHB and the actions that will be taken within one hour with respect to the FHB Ventilation System and penetrations, other than closing the equipment hatch and its associated personnel airlock, in the event of a FHA. The contingencies described in Reference 1, Attachment 5 ensure that releases resulting from a postulated FHA are directed to a filtered release pathway consistent with the guidance of NUMARC 93-01.

Byron and Braidwood Stations currently have procedures that track containment penetration status and identify the actions to achieve containment closure; BOP PC-1, "Containment Closure Tracking Capability," at Byron Station and 1/2BwOS XPC W-1, "Containment Penetration Status," at Braidwood Station. For implementation of the AST amendment these procedures will be revised to specify the one hour requirement for a fuel handling event and identify that for a fuel handling event, installation of the equipment hatch is not required provided the integrity of the containment / FHB ventilation envelope is established and a FHB charcoal plenum and booster fan are placed in operation.

The rationale for not closing the equipment hatch is as follows:

1. The equipment hatches at Byron and Braidwood Station, Units 1 and 2 open into the FHB and do not go to the outside environment. When open, the associated containment becomes a part of the FHB ventilation envelope, a system designed to mitigate the consequences of the FHA.

2. The current licensing basis permits the hatch to remain open for the containment FHA. This basis assumes that the containment-FHB envelope is at an initial pressure differential of –0.25 inches H_2O with reference to the atmospheric pressure and the FHB Ventilation System is operable or under an limiting condition for operation (LCO) provision with one fan in continuous operation. With implementation of the AST amendment, and in compliance with the Reference 1, Attachment 5 commitments, this condition will be established within one hour.

3. The installation of the equipment hatch requires a crane operator to man the containment polar crane for at least three hours. Unlike the loss of decay heat removal scenario where six or more hours would be provided until the polar crane operator is subjected to an un-inhabitable environment from heat load, the FHA scenario would subject the crane operator to an un-inhabitable environment from an iodine dose perspective long before the task could be completed.

Containment closure will be achieved through procedure execution. Byron Station and Braidwood Station currently have abnormal operating procedures 1/2BOA Refuel-1 (Byron Station) and 1/2BwOA Refuel-1 (Braidwood Station), "Fuel Handling Emergency." These procedures are executed from the control room at the direction of the Senior Reactor Operator (SRO). With implementation of the AST amendment the FHB charcoal booster fan will be required to be in operation as specified in the Reference 1, Attachment 5 commitments. Additionally, procedures 1/2 BOA Refuel-1 and 1/2 BwOA Refuel-1 will be modified to address auxiliary building charcoal booster fan operation to achieve the conditions specified in the Reference 1, Attachment 5 commitments.

The degree of containment closure is addressed by procedures BOP PC-1 and 1/2BwOSR XPC W-1 described above. Initial actions will originate from the control room. Local operator actions and verifications will be performed through the use of the associated checklists. Closure is satisfied when each required penetration is isolated. Methods of closure for the penetrations required for the Reference 1, Attachment 5 commitments are one or more of the following:

1. Component is intact, such as the secondary side of a steam generator (i.e., all secondary side manways and hand holes closed, vents and drains in containment closed).

2. The path is isolated by one or more valves or a blank flange in the potential release path.

3. One or more doors in the emergency hatch airlock are closed.

4. The penetration is filled with an approved sealant capable of withstanding 10 psig.

Containment penetrations that are in ventilation envelopes serviced by auxiliary building Nonaccessible Ventilation System would not be required to be closed to satisfy the Reference 1, Attachment 5 commitments. In the event a maintenance or testing activity has any of these penetrations configured such that a path is provided that would bypass their associated ventilation envelope, the subject penetrations will be identified prior to the start of the activity and tracked to ensure the penetrations will be closed per the closure methods specified above to meet the Reference 1, Attachment 5 commitments.

Current practice for containment closure is to have designated personnel for the local actions required for closure. This practice will be maintained. Control room actions for containment closure will continue to be performed by licensed reactor operators at SRO direction.

The procedures listed above are covered periodically as part of the operator training programs and will, in their revised form, continue to be addressed in these training programs with the implementation of the AST amendment.

Steam Generator Tube Rupture

6. In Table 8.1 in Attachment 6 of the submittal, iodine partitioning factors are decreased by a factor of 100 for the faulted steam generator flashed break flow for 0 - 445 seconds and for the secondary side stream release for 0 – 465 seconds. What is the basis for this reduction in the iodine release for those time periods? Please provide justification for the partitioning value.

   Response:

   What is shown in the "(Iodine) RCS to Environment (Flashed Break Flow)" portion of Table 8.1, Attachment 6, of Reference 1, is the 0.01 steam/water partitioning factor associated with flashed break flow iodine activity released from the faulted SG through the condenser. For the first 445 seconds of the accident the faulted SG PORV is assumed to be unopened and flashed break flow is transported to the environment through the condenser.

   After 445 seconds, with reactor trip and an assumed loss of offsite power, the condenser is isolated and its partition factor is no longer applicable; therefore, the portion of the faulted SG release attributed to flashed break flow after this time has no partition factor applied (i.e., a partition factor of 1.0 is applied). Since the faulted SG PORV is not modeled to open until 465 seconds into the event, the SGTR analysis inclusion of steam release, with no iodine partitioning, for the additional 20 second (445 - 465) time period (i.e., the time from reactor trip until the PORV for the ruptured SG opens) provides added conservatism.

   The balance of the primary coolant iodine which does not become airborne immediately (i.e., the "Ruptured SG to Environment (Steam Release)" portion of Table 8.1, Attachment 6, of the Reference 1 submittal), mixes with the SG secondary side water with an iodine steam/water partition factor of 0.01. For the first 445 seconds of the SGTR event, the flow of this steam release from the ruptured SG is transported to the environment through the condenser with the additional condenser iodine steam/water partition factor of 0.01. The combination of the SG secondary water and condenser steam/water partitioning factors results in a total iodine partition factor of 0.0001. As stated above, after 445 seconds the condenser is unavailable and steam flow to the environment would stop; however, the analysis conservatively continues to model steam flow to the environment through the condenser for an additional 20 seconds (i.e., between 445 seconds and 465 seconds).

   At 465 seconds, when the PORV for the ruptured SG opens, the condenser partition factor is no longer applicable and the portion of the faulted SG release attributed to steam release has only the SG secondary water 0.01 partition factor applied.

The SGTR event was evaluated with new AST assumptions using the same timing parameters, scenario assumptions, χ/Q release locations, CR receptor locations, partitioning factors, and other assumptions that were used in the existing analyses. The 0.01 iodine steam/water condenser partition factor is based on the Westinghouse Electric Company, LLC WCAP 8159, "Recommended Total Iodine and Entrainment Separation Factors for Normal Operation and Transients."

To determine the impact that the condenser partition factor has on CR dose, a qualitative assessment of the SGTR DBA was performed using a condenser partition factor of 0.1 instead of 0.01. The results of this assessment indicated that there was minimal impact on CR dose (i.e., an approximate total CR dose increase of only 0.03 rem TEDE)

EGC considers a 0.01 partition factor a suitable value to account for the plate-out of elemental iodine across the condenser volume. While alternative partition factor values could be considered, the qualitative assessment indicates values ranging from 0.01 to 0.1 have minimal impact on CR dose results for this DBA.

The iodine steam/water partition factor of 0.01 applied to the steam release flow through the SG secondary side water is provided in NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants," Section 15.6.3, "Radiological Consequences of Steam Generator Tube Failure," Item III.10. This 0.01 partition factor is also used as an assumption of RG 1.183, Appendix E, "Assumptions for Evaluation the Radiological Consequences of a PWR Main Steam Line Break Accident," paragraph 5.5.4.

7. In Table 8.2 in Attachment 6 of the submittal, iodine partitioning factors are decreased by a factor of 100 for the intact steam generators secondary side stream release for 0 - 2800 seconds. What is the basis for this reduction in the iodine release for that time period? Please provide justification for the partitioning value.

   Response:

   For the SGTR design basis accident, an iodine partitioning factor of 0.01 is assumed to be associated with the intact SG steam flow release path. What is shown in Attachment 6, Table 8.2, of the Reference 1 submittal, is the combination of the partitioning factors associated with steam release from the intact SGs and transport of this activity through the condenser for the first 2800 seconds of the accident. For this first 2800 seconds of the accident the intact SG PORVs are assumed to be unopened and steam flow is transported to the environment through the condenser. After 2800 seconds the 0.01 partitioning factor associated with the condenser is no longer applied, however the 0.01 partitioning factor associated with steam release through the open SG PORVs is still accounted for. For analytical and modeling purposes, these partitioning factors are applied directly to the flow rate of coolant through the stated pathways. As stated in the response to question SGTR 6 above, the timing parameters and partitioning factors are the same as those used in the existing analyses.

Control Room Dose Modeling

8. In the modeling of the control room for each of the design basis accidents (DBAs), the control room is assumed to be manually isolated at 30 minutes after the accident initiation, regardless of the automatic isolation/actuation that is expected to occur.

   a. Does this set of assumptions bound the conditions expected for a loss-of-offsite power coincident with the accident with automatic isolation/actuation of the emergency mode of operation for the control room ventilation system?

      Response:

      Upon high outside air radiation detection or initiation of a safety injection signal, the following automatic actions occur: the normal outside air intake is isolated, air is drawn in from the Turbine Building intake through the emergency make up filter unit (EMU), the recirculation charcoal filter bypass damper closes, and the recirculation filter inlet/outlet dampers open. The Byron Station and Braidwood Station CR HVAC systems are safety related and active components are designed with redundant equipment to meet single failure criteria. The redundant equipment is supplied with separate essential power sources and therefore a loss-of-offsite power will not affect the automatic actuation capability.

      During a DBA coincident with a loss-of-offsite power (LOOP) event, the CR HVAC system will automatically line up to the emergency mode. For conservatism, the analysis assumes a delayed alignment 30 minutes after accident initiation. Note that the 30 minute delay for emergency mode alignment is a conservative analysis assumption and does not credit manual operator action. The conditions expected for a LOOP, coincident with a DBA with automatic isolation and initiation of the emergency mode of the CR ventilation system, are bounded by the analysis.

   b. Does this set of assumptions bound the conditions expected for a loss-of-offsite power coincident with the accident without isolation or any ventilation system operation until the manual initiation of the emergency mode of the control room ventilation system?

      Response:

      See response to question 8a above for an overview of system operation. The conditions expected for a LOOP coincident with a DBA without automatic isolation or any ventilation system operation until the assumed delayed alignment of the emergency mode of the CR ventilation system are bounded by the analysis.

   c. What is the unfiltered inleakage for normal operation of the control room ventilation system?

      Response:

      For normal operation, an unfiltered inleakage value of 1000 cfm is assumed in addition to 6000 cfm (nominal) of unfiltered air intake during the initial 30 minute period following an accident for conservatism. Using the 30 minute time delay to

switch from normal to emergency isolation mode provides additional margin, since switchover is actually designed to occur almost instantaneously by the automatic actuation of the emergency mode. For the emergency mode, after 30 minutes following an accident and CR isolation, the unfiltered inleakage value used is 1000 cfm, which has been verified to be conservative by tracer gas test.

The inleakage identified during tracer gas testing in the emergency mode of operation was mainly through the normal outside air intake low leakage dampers; therefore, the inleakage identified through the outside air intake dampers is, in fact, part of the nominal 6000 cfm outside air intake flow used for normal operation. Since the normal mode and emergency mode make up air flow are approximately the same amount, the positive differential pressures throughout the CR Envelope would be the same during either mode. The unfiltered inleakage values obtained as part of the tracer gas test in the emergency mode will therefore conservatively bound any unfiltered inleakage during normal operation.

9. Please provide more information on the analysis of the control room filter loading and direct shine dose, including the methodology, inputs and assumptions used, and the dose results. Include the geometry (drawings, piping, etc.), source term, materials and assumptions used to determine the doses given.

Response:

Attachment K (i.e., page K-1 through K-27) of calculation BYR04-051 & BRW-04-0038-M contains an evaluation of the impact of AST on CR filter loading and direct shine dose.

Relevant drawings for information are UFSAR Figure 6.4-4, "Diagrammatic Representation of Total Control Room LOCA Dose," showing the CR Makeup Air Filter with respect to the occupied areas of the CR. Figure 6.4-4 is based on the pre-AST TID-14844, "Calculation of Distance Factors for Power and Test Reactor Sites," source terms, with 50% of the core halogens along with 100% of the core nobles distributed in the post-LOCA Primary Containment, as shown. UFSAR Figures 6.4-2, "Isometric View of Control Room," and 6.4-3, "Diagrammatic Representation of Radioactive Sources for Protected Area of Control Room During a LOCA," also present an isometric view of the CR and a diagrammatic representation of radioactive sources during a LOCA, respectively. UFSAR Table 6.4-1, "Expected Dose to Control Room Personnel Following a Loss-of-Coolant Accident (LOCA)," provides the resulting pre-AST CR doses, with 0.023 rem direct shine dose from containment, 0.013 rem from the radioactivity accumulated on the CR Makeup air filters, and 0.003 rem from the external plume shine, (i.e., all doses are whole body only). For AST containment release assumptions (i.e., 95% of the noble gas release delayed by one-half hour and only 40% of the core halogens released), the containment shine dose is bounded by the pre-AST result of 0.023 rem. The pre-AST 0.013 rem from the radioactivity accumulated on the CR Makeup air filters is also bounding as described below. The source for the small 0.003 rem external plume shine dose would also decrease with AST assumptions; however, the factor of two AST increase in assumed containment leak rate increases this dose contribution.

To account for this and other AST increased contributors such as ECCS iodine, the 0.003 rem whole-body pre-AST external plume shine dose to the CR is conservatively

multiplied by a factor of five to yield 0.015 rem TEDE. The total Direct Gamma Shine Dose considered in the total LOCA CR dose was therefore 0.05 rem TEDE, as derived from:

0.023 rem TEDE direct shine dose from Containment
+ 0.013 rem TEDE from the activity accumulated on the CR Makeup air filters
+ 0.015 rem TEDE from the external plume shine.

A further comparison is made between filter loading source terms based on AST source terms and those expected with TID-14844 / Regulatory Guide 1.4, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss of Coolant Accident for Pressurized Water Reactors," based source terms.

As a first step, the following RADTRAD Cases were run:

1. A Primary Containment (PC) Leakage case was modeled identical to that in the AST LOCA Calculation, except that a CR filter compartment was used in lieu of a CR. Flows were such that all aerosols, elemental iodine, and organic iodine that would have been brought into the CR are deposited in the "Control Room Filter." This includes unfiltered inleakage whose activity could be deposited during recirculation.

2. A TID-14844 / RG 1.4 evaluation of the above PC Leakage case was conducted. TID based release fraction timing was used. The same core source terms were used since any differences with respect to previously used values would be small.

3. An AST ECCS leakage case was run and modeled same as the PC Leakage case above.

A TID-14844 / RG 1.4 case for ECCS Leakage is not run because it would be much less significant using the historically assumed leak rate. Neglecting it is conservative in this before and after AST comparison.

For each of the above cases, the "Detailed output" feature of RADTRAD for the CR filter compartment was turned on. A simple program (RDTRDYYY) was developed to extract data in the form of isotope curie content of the filter at each time step from the RADTRAD runs, and to integrate this data to obtain curie-hours for each time step. The integrated data at times 24, 96 and 720 hours were extracted into Excel spreadsheets, and occupancy factors of 0.6 and 0.4 was applied to the 24 to 96 and 96 to 720 hour periods, respectively. The resulting integrated doses at the three time steps where summed to give the AST and pre-AST (i.e., TID-14844) Total Integrated Sources (TIS) for input into the MicroShield program.

Two MicroShield runs were made, one for the AST filter loading TIS and one for the Pre-AST TID filter loading TIS. Both runs used the UFSAR Table 6.4-1 identified eight inch shield thickness for the filter; however, a point source model was used for simplicity in order to develop a ratio between the two for comparison. The TIS in curie-hours was entered as "curies" in MicroShield, therefore the output in "mR/hr" is actually in mR.

The resulting relative ratio of operator doses from the filter loading for (TID Dose) / (AST Dose) = 1251 / 895.9; therefore, the pre-AST TID basis of filter loading is shown to be bounding.

Attachment 1

Response to NRC Request for Additional Information

References

(1) Letter from K. R. Jury (Exelon Generation Company, LLC) to NRC, "Request for License Amendment Related to Application of Alternative Radiological Source Term," dated February 15, 2005

(2) Letter from J. A. Bauer (Exelon Generation Company, LLC) to NRC, "Additional Information Related to Application of Alternative Radiological Source Term – Atmospheric Dispersion Coefficients," dated December 9, 2005

(3) Letter from J. A. Bauer (Exelon Generation Company, LLC) to NRC, "Additional Information Related to Application of Alternative Radiological Source Term – Dose Calculations," dated December 9, 2005

(4) Letter from M. E. Warner (FPL Energy) to NRC, "Seabrook Station License Amendment Request 03-02, 'Implementation of Alternate Source Term,' " dated October 6, 2003

(5) Letter from J. A. Bauer (Exelon Generation Company, LLC) to NRC, "Response to NRC Request for Additional Information With Respect to Request for License Amendment Related to Application of Alternative Radiological Source Term," dated November 28, 2005

Attachment 2

BRAIDWOOD STATION
UNITS 1 AND 2

Docket Nos. STN 50-456 and STN 50-457
License Nos. NPF-72 and NPF-77

and

BYRON STATION
UNITS 1 AND 2

Docket Nos. STN 50-454 and STN 50-455
License Nos. NPF-37 and NPF-66

List of Commitments

List of Commitments

The following table identifies those actions committed to by EGC in this document. Any other statements in this submittal are provided for information purposes and are not considered to be regulatory commitments.

Commitment	Scheduled Completion Date
EGC will update calculation BYR04-050 & BRW-04-0044-M to reevaluate χ/Q values based on finer wind speed categories provided in the latest appropriate regulatory guidance the next time calculations associated with the dose consequences of the accidents listed in the response to meteorological question 10 are revised (i.e., LOCA, MSLB, CREA, LRA, SGTR and FHA).	The next time calculations associated with the dose consequences of the listed accidents are revised.