Exelon Nuclear

Exelon Generation
4300 Winfield Road
Warrenville, IL 60555

www.exeloncorp.com

RS-09-130

September 28, 2009

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

LaSalle County Station, Units 1 and 2
Facility Operating License Nos. NPF-11 and NPF-18
NRC Docket Nos. 50-373 and 50-374

Subject: Additional Information Supporting Request for License Amendment Regarding Application of Alternative Source Term

References:

1. Letter from P. R. Simpson (Exelon Generation Company, LLC) to U.S. NRC, "Additional Information Regarding Request for License Amendment Regarding Application of Alternative Source Term," dated October 23, 2008

2. Letter from C. S. Goodwin (U.S. NRC) to C. G. Pardee (Exelon Nuclear), "LaSalle County Station, Units 1 and 2 – Request for Additional Information Related to Alternative Source Term License Amendment Request (TAC Nos. ME0068 – ME0069)," dated August 27, 2009

In Reference 1, Exelon Generation Company, LLC (EGC) requested an amendment to Facility Operating License Nos. NPF-11 and NPF-18 for LaSalle County Station (LSCS), Units 1 and 2, respectively. Specifically, the proposed change revises Technical Specifications (TS) to support the application of alternative source term (AST) methodology with respect to the loss-of-coolant accident and the fuel handling accident.

The NRC requested additional information to complete review of the proposed license amendment in Reference 2, and in an e-mail that was sent to EGC on August 31, 2009 and a fax that was sent to EGC on September 18, 2009. In response to this request, EGC is providing the attached information.

EGC has reviewed the information supporting a finding of no significant hazards consideration, and the environmental consideration, that were previously provided to the NRC in Attachment 1 of Reference 1. The additional information provided in this submittal does not affect the bases for concluding that the proposed license amendment does not involve a significant hazards consideration. In addition, the additional information provided in this submittal does not affect the bases for concluding that neither an environmental impact statement nor an environmental assessment needs to be prepared in connection with the proposed amendment.

Should 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 28th day of September 2009.

Respectfully,

Patrick R. Simpson
Manager – Licensing

Attachments:

1. Response to Request for Additional Information
2. Drawing M-25, Sheet 5

cc: NRC Regional Administrator, Region III
NRC Senior Resident Inspector – LaSalle County Station
Illinois Emergency Management Agency – Division of Nuclear Safety

CONTAINMENT AND VENTILATION BRANCH

RAI-1

Describe the emergency core cooling system (ECCS) flow paths that will be in use in LGA-001 when the SLC system is activated.

Response

As described in Section 6.3 of the Updated Final Safety Analysis Report (UFSAR), during a loss-of-coolant accident (LOCA), all ECCS with electrical power from the running diesel generators (DGs) would be injecting with suction from the suppression pool. This includes the High Pressure Core Spray (HPCS) system, Low Pressure Core Spray (LPCS) system, and all three Residual Heat Removal (RHR) subsystems in the Low Pressure Core Injection (LPCI) mode. Simplified flow diagrams are contained in UFSAR Figures 6.3-1, 6.3-4, and 6.3-8 (i.e., sheet 3). The worst case design basis accident assumes the loss of the Division 1 DG. In this case, injection would be provided by HPCS, B LPCI and C LPCI. Water is drawn from the suppression pool through the operating pumps and injected into the core.

RAI-2

Describe the SLC flow path from injection into the RPV, through the RPV, and from the RPV to the suppression pool in conjunction with the ECCS flow path(s) in use per LGA-001.

Response

In the event of a LOCA requiring activation of the Standby Liquid Control (SLC) system, the sodium pentaborate solution from the SLC storage tank would be injected into the reactor pressure vessel (RPV) below the core, where the solution would mix with ECCS water. Reactor water, including the sodium pentaborate solution, would flow out of the break and onto the drywell floor. This mixture of water would then drain through the 98 downcomers, each with an inner diameter of 23.5 inches, located on the drywell floor into the suppression pool. Each downcomer's outlet is 12 feet 4 inches below the suppression pool water level. This mixture would distribute and mix with the suppression pool water and will be injected back into the RPV via the ECCS, and the cycle would repeat itself.

RAI-3

Please describe how the EOPs will be changed to assure that adequate sodium pentaborate reaches the suppression pool to maintain a pH of at least 7.

Response

In a large break LOCA, the current Emergency Operating Procedure (EOP) guidance requires injection of the sodium pentaborate solution via the SLC system when RPV level reaches -150 inches (i.e., 11 inches above the top of active fuel). This level should be reached within the first few minutes of the accident.

The current EOP guidance for shutting down the SLC system is either when the tank is empty or when it is no longer needed (i.e., level has recovered above -150 inches). As committed to in Reference 1, this guidance will be revised to ensure no steps would terminate the injection during a design basis accident LOCA prior to emptying the SLC system boron solution storage tank (i.e., injection of the full content into the RPV) for pH control in the suppression pool.

RAI-4

Describe how LSCS will verify the changes to the EOPs assure completion of the injection of sodium pentaborate to the suppression pool within the required time limit.

Response

Section 3.6.11 of Attachment 1 to Reference 1 identifies that completion of the SLC system injection of its sodium pentaborate solution is required for pH control within 3.5 hours of the start of the LOCA. Technical Specification (TS) Section 3.1.7, "Standby Liquid Control (SLC) System," Surveillance Requirements require verification of available volume and concentration of the SLC storage tank and that the minimum pump flow rate is 41.2 gallons per minute (gpm). TS Figure 3.1.7-1, "Sodium Pentaborate Solution Volume/Concentration Requirements," provides the acceptable volumetric ranges for a given concentration. As described in UFSAR Section 7.4.2, the injection time is from 50 to 125 minutes from initiation of the system, depending on the tank volume and concentration.

A maximum time of 125 minutes is required for full injection. The AST analysis assumes complete injection of the sodium pentaborate solution within 210 minutes (i.e., 3.5 hours) following the start of the LOCA. The time difference of 85 minutes is the available margin (i.e., the margin of time after the start of the LOCA within which SLC system activation is required). Based on the above response to RAI-3, activation of the SLC system would occur well before 85 minutes, and the changes to the EOPs discussed will assure completion of the injection.

RAI-5

The proposed license revision indicates that the LSCS SLC system cannot be considered redundant with respect to its active components. The submittal indentified the inboard and outboard check valves as non-redundant components. Exelon Generation Company has determined that the 1(2) C41-F006 and 1(2) C41-F007 check valves have an acceptable performance history at LSCS.

Please describe how the inboard and outboard check valves are protected from foreign material when maintenance is performed on the SLC flow path.

Response

The SLC system check valves would be designated foreign material exclusion area (FMEA) 1. Exelon Generation Company, LLC (EGC) procedures define FMEA 1 as follows.

> FMEA 1: The highest level of FME control imposed on a system or component. An Area 1 is established in situations where a final visual inspection of internal cleanliness prior to system closure is not possible due to configuration or other circumstances, such as ALARA concerns, etc. Additionally, FMEA 1 controls may be applied based upon risk to systems, components, or other structures where introduction of any materials could be irreversible.

The first line supervisor (FLS) and crew during the job walk down and preparation would determine how they could mitigate the introduction of foreign material into the system, by installing an FME device, such as a plug or bladder into the piping on both the upstream side and the downstream side of the valve. The FMEA would then be designated as FMEA 2, which is defined as follows.

> FMEA 2: An Area 2 is established in situations where a final visual inspection of internal cleanliness prior to system closure is possible.

At job execution, when the bonnet of the check valve is removed, the FLS and crew would perform an initial inspection for any as found foreign material and document this data in the work order. They would then immediately install the FME device (e.g., plug or bladder) into the upstream and downstream side of the valve. The check valve inspection would continue with measurements being taken or any internal repairs to the valve that are required.

When all repairs are completed, the mechanics would clean out the valve body, by vacuuming or wiping. The FLS or lead worker will perform an inspection of the valve body and if it were deemed to be clean, the FME device (e.g., plug or bladder) would be removed. A final inspection would then be performed by the FLS prior to installing the disc and bonnet. If the inspection shows the cleanliness to be adequate, the disc and bonnet will be reinstalled.

MECHANICAL AND CIVIL ENGINEERING BRANCH

RAI-1

Page 1.183-8 of Regulatory Guide (RG) 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors," (Section 1.3.2, Re-Analysis Guidance) denotes the responsibility of licensees to evaluate the radiological and nonradiological impacts of an alternative source term (AST) implementation and the impacts of any associated plant modifications. Section 3.6 of your October 23, 2008, submittal, indicates that credit will be taken for portions of the Control Room (CR) and Auxiliary Electric Equipment Room Heating Ventilation and Air Conditioning (HVAC) systems in support of the proposed AST implementation at LSCS. Please discuss the methodologies used to demonstrate the seismic ruggedness and/or the seismic qualification of the aforementioned HVAC systems and their components. Additionally, please provide the references which provide the regulatory acceptance bases of these methodologies.

Response

No changes to the existing CR or Auxiliary Electric Equipment Room (AEER) HVAC systems are required to support implementation of AST. The existing systems are safety related and are designed and installed to meet Seismic Category I requirements, with the exception of heating equipment which is not Seismic Category I, but is seismically supported. These HVAC systems are described in UFSAR Section 9.4.1. Seismic methodologies are described in UFSAR Sections 3.7, 3.8, 3.9 and 3.10.

Regulatory acceptance of the seismic methodology and CR and AEER HVAC systems is documented in NUREG-0519 (i.e., Reference 2), including the associated supplements. Specific sections documenting the regulatory acceptance are listed below.

Seismic Methodology
- NUREG-0519, Sections 3.7, 3.8, 3.9, and 3.10
- NUREG-0519, Supplement No. 1, Sections 3.9, and 3.10
- NUREG-0519, Supplement No. 2, Sections 3.9, and 3.10
- NUREG-0519, Supplement No. 5, Sections 3.8, 3.9, and 3.10
- NUREG-0519, Supplement No. 6, Section 3.10

CR and AEER HVAC Systems
- NUREG-0519, Sections 6.4, 6.5, and 9.4.1

RAI-2

Please verify whether the secondary containment bypass piping is credited for the proposed AST. If affirmative, please confirm whether these piping runs have been seismically evaluated and discuss the methodologies used in these evaluations.

Response

As discussed in Section 3.6 of Attachment 1 to Reference 1, there are three release pathways (i.e., the primary containment, ECCS, and main steam isolation valve (MSIV)). The ECCS and MSIV pathways consist of piping subsystems. Each pathway piping is described below, along with the associated seismic qualification.

ECCS Piping

ECCS includes HPCS, LPCS, and LPCI. These systems are classified as safety related, Seismic Category I. As discussed in UFSAR 6.3.2.3, all piping systems and components for the ECCS comply with the applicable codes, addenda, code cases, and errata in effect at the time the equipment is procured. UFSAR Tables 3.2-1, 3.2-2, 3.2-3, and 3.2-4 contain information regarding the code requirements pertaining to components and systems. UFSAR Tables 3.2-1, 3.2-2, and 3.2-3 list code editions in effect at the time of original equipment procurement.

The piping and components of the ECCS subsystems within the containment, and out to and including the pressure retaining injection valve, are Class I. All other piping and components are Class 2, 3, or non-Code. Subsection NA, NB, NC, and ND of the American Society of Mechanical Engineers Code apply to the ECCS.

The methodologies to seismically qualify the ECCS piping are described in UFSAR Sections 3.7, 3.8, 3.9, and 3.10. Regulatory acceptance of these methodologies is described in NUREG-0519 (i.e., Reference 2), including the associated supplements. Specific sections documenting the regulatory acceptance of the seismic methodologies are listed above in response to Mechanical and Civil Engineering Branch RAI-1.

<u>MSIV Leakage and Main Steam Piping Outside of the Primary Containment</u>

As discussed in Section 3.6.7.2 of Attachment 1 to Reference 1, the primary components relied upon for the treatment of MSIV leakage are the main steam lines and main condenser. The NRC previously concluded, in a license amendment that was issued to support elimination of the MSIV Leakage Control System (i.e., Reference 3), that the alternate leakage treatment pathway that includes the main steam lines and main condenser is seismically qualified, and the condenser shell and internal components are seismically rugged and capable of transferring safe shutdown earthquake (SSE) loads to the supporting structure. As such, the NRC concluded in Reference 3 that the turbine building has an acceptable design margin of safety for SSE loading. No changes to the components credited in the analyses for this alternate leakage treatment pathway are being made to support implementation of AST.

<u>RAI-3</u>

Please provide a list of any additional mechanical equipment or systems (i.e. air handling units, motor control centers), fans and I&C cabinets, etc.) not mentioned above which will be credited as part of the implementation of the AST at LSCS. For these items:

a) Indicate whether the equipment is new or existing.

b) Describe the location of the equipment and the seismic qualification method employed to demonstrate the seismic ruggedness of this equipment, such as the plant licensing basis or a Nuclear Regulatory Commission (NRC)-endorsed industry standard.

c) Summarize the results of the seismic qualification of the equipment, indicating whether any modifications or re-design will be necessary in support of the AST.

<u>Response</u>

There are no additional mechanical equipment or systems that will be credited as part of the implementation of the AST, other than those previously discussed in the responses to Mechanical and Civil Engineering Branch RAI-1 and RAI-2 above.

ACCIDENT DOSE BRANCH

RAI-1

Please explain how the LaSalle 1998 through 2003, hourly meteorological data provided in support of the October 23, 2008, AST license amendment request (LAR), in general, was processed from the raw measurements and discuss the LaSalle site meteorological characteristics. The following are some NRC staff observations and concerns with respect to the reported atmospheric stability and wind speed measurements.

- There appears to be a relatively low frequency of reported unstable conditions. This is particularly noticeable when compared with historic data in the LaSalle Updated Final Safety Analysis Report (UFSAR). In addition, extremely stable conditions (G) were reported almost 12 percent of the time between 1998 and 2003 for measurements between the 61.0 and 10.1 meter levels.

Information Source Data Period Delta-T Interval	Atmospheric Stability Frequency Occurrence (%)							
	A	B	C	D	E	F	G	Total
LaSalle UFSAR 10/1976 – 09/1978 114.3 -10.1 m	3.5	3.6	4.7	45.8	24.3	14.1	4.2	100
NRC estimate 1998 – 2003 114.3 - 10.1 m	0.1	0.4	1.6	48.4	29.2	14.8	5.5	100
NRC estimate 1998 – 2003 61.0 - 10.1 m	1.9	3.4	6.4	37.2	26.2	13.3	11.7	100

- LaSalle appears to be a relatively high wind speed site for all stability categories. In particular, staff estimated average wind speed during G stability at about 7.5 miles per hour (mph) and maximum yearly values ranging from about 16 mph to almost 30 mph at the 10.1 meter level. Such wind speeds seem too high to support G stability conditions. Reported wind speeds also seemed generally high for other stable categories.

- Wind speed and atmospheric stability data for 1995 through 1998, from the Dresden and Quad Cities sites in eastern and western northern Illinois, respectively, seem more similar to each other than the 1998 through 2003, meteorological data from LaSalle which is located between these two sites in central northern Illinois.

Response

The X/Q calculation supporting AST was submitted to the NRC as Attachment 6 to Reference 1. Section 3.3.1 of the calculation discusses that the LaSalle County Station (LSCS) meteorological data from the six-year period, 1998-2003, as supplied by Murray and Trettel were used in the PAVAN analysis. The format of PAVAN meteorological input consists of a joint wind direction based on sixteen 22.5 degree sectors, wind speed (i.e., 14 intervals), and stability class (i.e., seven classes) occurrence frequency distribution.

Each such meteorological joint frequency distribution for input to PAVAN was transformed as data for a joint wind-stability occurrence frequency distribution. The fourteen wind speed categories were derived based on Section 4 of the NRC Regulatory Issue Summary 2006-04 (i.e., Reference 4) with the first category identified as "calm" as shown in Table 3-2 of the X/Q calculation (i.e., Attachment 6 to Reference 1). The minimum (i.e., non-calm) wind speed was assumed to be 0.7 miles per hour (mph), based on the higher of the wind speed and direction instrument starting thresholds. Calm wind speed occurrences were assigned a value of 0.3 mph.

Section 2.3.2 of the LSCS UFSAR provides a description of the LSCS site meteorological characteristics.

The frequency of unstable conditions is lower for the period of 1998 through 2003 when compared to the two year period of October 1976 through September 1978. The major contributing factor to this difference is that the meteorological tower was in a different location in the 1970s and had a greater potential for winds influenced by obstructions than it does now. The values being compared are not like for like.

A higher percentage of extremely stable conditions is anticipated when using the lower level delta T (i.e., 61.0 meters) versus the upper level (i.e., 114.3 meters). Higher wind speeds at a height nearly twice as high would create less stable conditions.

The wind speeds recorded at LSCS are much higher than any of the other 18 nuclear-related facilities Murray and Trettel has provided data-related services to over the past several years. The upper level wind and temperature measurement level at LSCS (i.e., 114.3 meters or 375 feet) supports the extremely stable delta T conditions at night (i.e., cold and calm conditions at the surface, and warmer and windier conditions at higher elevations – an inversion indicating stable conditions).

Wind speed and atmospheric stability data are different at LSCS than it is at the Dresden Nuclear Power Station (DNPS) and Quad Cities Nuclear Power Station (QCNPS). Murray and Trettel reviews data from all three sites on a daily basis and differences are routine. The LSCS meteorological tower is located at a site with fewer obstructions to wind flow, as compared to DNPS or QCNPS, which also leads to higher average wind speeds at LSCS. In addition, Murray and Trettel recently performed a study for Clinton Power Station (CPS) which indicated that conditions (i.e., wind and stability) at CPS were more consistent with DNPS and Braidwood Station than they were at LSCS.

The LSCS meteorological tower is calibrated every four months by Murray and Trettel. Valid data recovery levels at LSCS routinely exceed 98 percent on an annual basis.

RAI-2

With regard to page H1 of Calculation No. L-003063, "Alternative Source Term Onsite and Offsite X/Q Values," Rev. No.1, Attachment H, which direction is true north? What is the scale of the figure? Where are the reactor building hatch access to auxiliary building roof and reactor building truck bay door release locations? How was the release point shown for the integrated leak rate test location determined to be appropriate for use in calculating the atmospheric dispersion factors (X/Q values)? Where is the assumed location for releases from the reactor building wall? Where is the control room located and why is it appropriate to assume that use of the control room air intake X/Q values as the input to the dose assessments is appropriate for unfiltered inleakage into the control room?

Response

As shown on pages J2 and J5 of the X/Q calculation (i.e., Attachment 6 to Reference 1), true north is the same as plant north.

The drawing previously submitted to the NRC in Reference 1 (i.e., page H1 of calculation L-003063, Revision 1) includes distances along the bottom, top, and right side of the drawing. However, EGC recognizes that the actual numbers are difficult to read, and as such, a larger version of this drawing is provided in Attachment 2 for clarification.

The location of the Reactor Building hatch access to Auxiliary Building roof, relative to the CR intake, is shown on page G2 of the X/Q calculation. Note that the Reactor Building hatch access to Auxiliary Building roof is annotated on page G2 as "Auxiliary Building Roof Access." Details regarding the release location distance and direction, relative to the CR, is provided in Table 2-3 (i.e., sheet 3 of 6) of the X/Q calculation.

The location of the Reactor Building truck bay door, relative to the CR intake, is shown on page H2 of the X/Q calculation. Details regarding the release location distance and direction, relative to the CR, is provided in Table 2-3 (i.e., sheet 4 of 6) of the X/Q calculation.

Since the integrated leak rate test (ILRT) penetrations could be open during fuel movement, the X/Q value for this location was calculated to confirm a potential release from this location was not bounding. The ILRT release location consists of two fixed penetrations that are adjacent to each other.

As discussed in Section 2.3.2 of the X/Q Calculation, the release locations identified for investigation were the stack (i.e., inclusive of the Standby Gas Treatment (SGT) system stack), the Unit 1 and Unit 2 MSIV pathway through the turbine/condenser, the Reactor Building hatch access to the Auxiliary Building roof (i.e., the Auxiliary Building roof access door at elevation 893'-6" extending north of column 8.9), the Reactor Building truck bay door, the Reactor Building wall, and the location for the ILRT penetrations. The stack, Unit 1 and Unit 2 MSIV, the Reactor Building hatch access to the Auxiliary Building roof, the Reactor Building truck bay door, and the ILRT release scenarios are modeled as point sources. The Reactor Building wall is modeled as a diffuse area source.

Calculations associated with the distance and direction from the CR/AEER intakes to the Reactor Building wall diffuse area source are contained in Attachment J of the X/Q calculation.

In accordance with NRC Regulatory Guide 1.194 (i.e., Reference 5), Section 3.2.4.5, the diffuse area source representation in ARCON96 requires the building cross-sectional area to be calculated from the maximum building dimensions projected onto a vertical plane perpendicular to the line of sight from the building center to the intake.

Figure 2 of NRC Regulatory Guide 1.194 specifies that, for a diffuse area source, only that part of the structure above grade or an enclosing building should be included in the building height. For the Reactor Building wall scenarios, the portion of the Reactor Building above the Auxiliary Building roof height was utilized for determining the release height, building area, and vertical diffusion coefficient (σ_z).

NRC Regulatory Guide 1.194 also requires the diffuse area source release height to be assumed at the vertical center of the projected area, and initial horizontal (σ_y) and vertical (σ_z) diffusion coefficients to be specified.

The diffuse source release area representing a portion of the Reactor Building wall, as calculated in accordance with NRC Regulatory Guide 1.194, is shown in Attachment J of the X/Q calculation.

The ARCON96 input parameter values were set in accordance with Table A-2 of NRC Regulatory Guide 1.194 (i.e., surface roughness length equals 0.2 meters; wind direction window equals 90 degrees, 45 degrees on either side of the line of sight from source to receptor; minimum wind speed equals 0.5 meters per second; and sector averaging constant equals 4.3).

The CR is located in the Auxiliary Building at elevation 768'-0". UFSAR Figure 6.4-3 shows the relationship of the CR intakes to the CR. As shown in the figure, the CR HVAC system's ductwork is also located in the Auxiliary Building below the intakes. The unfiltered inleakage would be into the negative pressure portion of the ductwork between the intake filter and the air handling unit. The use of the X/Q values associated with the CR intakes (i.e., outside of the building) is conservative in that the intakes are closer to the significant release sources than the CR itself or the negative pressure portions of the ductwork.

RAI-3

Section 3.3.2.2 of Attachment 1, "Evaluation of Proposed Change," to the LAR states that the stack was modeled as a ground-level release to obtain X/Q values to be utilized for the fuel-handling accident (FHA) only. Why was this calculation made? Why was a diffuse release from secondary containment assumed to be appropriate for the FHA? Section 3.2.4.1 of RG 1.194, "Atmospheric Relative Concentrations for Control Room Radiological Habitability Assessments at Nuclear Power Plants," states that diffuse source modeling should be used only for those situations in which the activity being released is homogeneously distributed throughout the building, and when the assumed release rate from the building surface would be reasonably constant over the surface of the building. Please confirm that no other source/receptor pair was more limiting.

Response

Since the base of the stack is located close to the CR air intake, the CR doses resulting from use of the stack release X/Q values at stack discharge elevation and ground level to the closest CR intake were assumed consistent with secondary containment integrity without filtration or any assumed diffuse area release. The X/Q value for a release from the stack at ground level was determined to be an artificial worst-case for these two release points. However, the diffuse area source was determined to be bounding for all permitted release points.

The wall area surface above the Reactor Building refueling floor is a potential release path best characterized as a diffuse area source. Due to the potential for leakage, the worst-case X/Q value to the CR calculated, with relaxed secondary containment requirements during fuel movement, for this possibility is this diffuse area source. This worst-case diffuse area source is from the wall of the Reactor Building above the refueling floor facing the closest (i.e., south) CR air intake in the dose calculation. Note that secondary containment integrity is still required if either LSCS Unit 1 or Unit 2 is in Modes 1, 2, or 3.

As discussed in Attachment 9 to Reference 1, the Reactor Building volume is artificially set to one cubic foot. Additionally, the exhaust rate is set artificially high at 0.1 air change per minute to assure an essentially complete release within two hours. This evacuates 99.9994 percent of the activity within two hours. Since there is no direct penetration for this exhaust, or realistic driving force to remove the activity at this rate, the diffuse area source was used and determined to be the bounding condition for all permitted release points.

RAI-4

For each DBA, which source/receptor pairs were compared to determine the limiting control room X/Q values? Which X/Q values were input into each of the limiting dose assessments?

Response

Table 6 of the FHA calculation (i.e., page 21), which was submitted to the NRC as Attachment 9 to Reference 1, provides a comparison of CR dose results for assumed release pathways.

For the LOCA, there are only a limited number of release locations to consider. These were evaluated by each specific pathway. The MSIV release location was assumed to be from the turbine/condenser, and the X/Q values were calculated accordingly. For the containment and ECCS leakage pathways, the release is from the drywell/piping into the Reactor Building and then through the main stack via SGT (i.e., unfiltered during the drawdown). There are no other primary or secondary containment bypass pathways. Since these are the only release pathways (i.e., pathways that are considered to occur simultaneously with dose consequences considered separately and then added together), the highest value for either unit is assumed in the calculation. Table 4-1 of the X/Q calculation contains a summary of all X/Q values calculated for LSCS.

RAI-5

Regarding the DBAs analyzed in support of this LAR, please confirm that the generated X/Q values model the limiting doses and all potential release scenarios were considered, including those due to loss of offsite power or other single failures.

Response

A plant walk-down was performed to identify potential release points as described below for the two accidents analyzed (i.e., FHA and LOCA).

For the FHA

CR atmospheric dispersion factors were derived for potential pathways of assumed intentional opening of two doors in series during the FHA for the Reactor Building access to the Auxiliary Building roof, the Reactor Building ILRT penetrations, or truck bay doors. Table 6 of the FHA calculation (i.e., page 21), provides a comparison of CR dose results from use of these dispersion factors to that used for the results, as determined using the ratio of the pathway dispersion factor to the diffuse area release dispersion factor times the diffuse area release CR dose. Table 6 indicates the opening of the two doors in series of the Reactor Building access to the Auxiliary Building roof cannot be allowed during fuel handling, but the temporary opening of the truck bay and/or an ILRT penetration can be allowed (e.g., during a dual unit shutdown). Also shown are the CR doses resulting from use of the stack release atmospheric dispersion factors at stack discharge elevation and ground level to the closest CR intake (i.e., an artificial worst-case), consistent with secondary containment integrity without filtration or any assumed diffuse area release.

For the LOCA:

There are no new release locations identified for the LOCA. The generated X/Q values model the limiting doses and all potential release scenarios were considered, including those due to loss of offsite power or other single failures. The release pathways associated with the LOCA are discussed on pages 47-48 of the LOCA calculation, which was submitted to the NRC as Attachment 7 to Reference 1.

RAI-6

Why does the joint frequency distribution for ground level releases to the exclusion area boundary and outer boundary of the low population zone use stability classifications based upon measurements between the 114.3 to 10.1-meter intervals rather than between the 61.0 and 10.1 meter levels? RG 1.145, "Atmospheric Dispersion Models for Potential Accident Consequence Assessments at Nuclear Power Plants," Section C.1.1, states that atmospheric stability should be determined by temperature difference between the release height and the 10-meter level.

Response

The LSCS meteorological tower upper delta temperature measurement (i.e., 375-33 feet, or 114.3-10.1 meters) was considered to uniformly best represent the onsite local stability conditions for all postulated releases to the downwind exclusion area boundary and low population zone receptor distances. As the Reactor Building complex is clearly the dominant site structure, and the much smaller elevation Turbine Building is adjacent to it, it is expected that any Turbine Building releases are highly subject to the aerodynamic dispersive effects of the Reactor Building in the surrounding immediate vicinity. NRC Regulatory Guide 1.145 (i.e., Reference 9) guidance and EPA modeling guidance have long been in general agreement that a building complex exerts aerodynamic influences on the atmospheric dispersion from the ground level potentially up to 2.5 times a controlling structure's elevation. Such elevation at LSCS would be approximately equal to that of the meteorological tower's upper level. Overall, the aerodynamic effect of structures on ambient atmospheric dispersion is the tendency to produce more vertical mixing and uniformity in the temperature stratification, and hence, less stable conditions. Therefore, the delta temperature stability including the entire measurement layer (i.e., 375-33 feet) was assumed to be more representative than the lower (i.e., 61.0-10.1 meter) delta temperature.

These factors are essentially independent of stability classes as measured over varying height intervals at the isolated meteorological tower.

RAI-7

Section 4.3 of Appendix A, "Assumptions for Evaluating the Radiological Consequences of a LWR Loss-of-Coolant Accident," to RG 1.183 states that ambient temperatures used in assessments on the ability of the secondary containment to maintain a negative pressure should be the 1-hour average value that is exceeded only 5 percent or 95 percent of the total numbers of hours in the data set, whichever is conservative for the intended use (e.g., if high temperatures are limiting, use those exceeded only 5 percent). What is the limiting temperature for the LaSalle site?

Response

Information regarding the ability of the secondary containment to maintain a negative pressure during periods of high wind is provided in Section 2.3.1 of the X/Q calculation. Additional information is included in Attachment G of the LOCA calculation. This evaluation was performed with no reference to temperature, consistent with the approach used for other stations in the EGC fleet.

The highest temperature recorded at Ottawa, Illinois as mentioned in Section 2.3.1.1 of the Final Safety Analysis Report is 112°F. This is based on the "Decennial Census of United States Climate - Climatic Summary of the United States - Supplement for 1951 through 1960 - Illinois." The climatography of the United States No. 86-9 U.S. Department of Commerce, Weather Bureau, Washington, D.C., 1964 indicates that the highest temperature recorded at Ottawa was 102°F, based on a 67-year period of record. An additional analysis was performed on the daily maximum temperatures at Ottawa, Illinois for 29 years (i.e., July 1948 through December 1976) based on "Climatological Data - Illinois," published by U.S. Department of Commerce, National Oceanic and Atmospheric Administration Environmental Data Service. The analysis indicated that during this period:

a. no temperature reading was recorded at or above 105°F, and
b. 22 temperature readings were recorded at or above 100°F.

Due to the low frequency of these temperatures which may persist for only a limited period of time, they are not used as a design-basis maximum temperature for the HVAC system. The outside design maximum temperature of 95°F dry bulb and 78°F wet bulb was used at LSCS as recommended in "1977 Fundamentals Handbook," published by the American Society of Heating, Refrigerating and Air-Conditioning Engineers, Inc. These design conditions are also supported by the Air Force Environmental Technical Application Center.

RAI FORWARD IN AN E-MAIL ON AUGUST 31, 2009

RAI

On page 15 of the submittal, Section 3.4, "NUREG-0737," the first paragraph states that the licensee "has determined that continued compliance will be maintained with NUREG-0737, Item II.B.2..." However, no basis is given for this statement. It is unclear whether the licensee has recalculated the doses with the AST source term, or if they are claiming that the current dose assessment based on the TID source term (as required by II.B.2) is bounding. Several applicants for the TID source term have determined that the time frame that personnel will be required to access the plant vital areas is early enough in the course of the accident when the dose rates from the TID nuclide mix bounds the dose rates from the AST nuclide assumptions.

Response

Attachment E of the LOCA calculation (i.e., Attachment 7 to Reference 1) contains information regarding vital area accessibility. Note that in the Table on page E-1, the AST disposition for pipe containing source dose effects refers to Reference E-1, which was not clearly identified. Reference E-1 is an NRC memorandum from J. E. Rosenthal to A. C. Thadani, "Initial Screening of Candidate Generic Issue 187, 'The Potential Impact of Postulated Cesium Concentration on Equipment Qualification in the Containment Sump,'" dated April 30, 2001.

LSCS vital areas, and their shielding protection for post accident operation, are addressed in UFSAR Section 12.3.2.5 for pre-AST conditions. This UFSAR discussion and supporting analyses are based on NRC Regulatory Guide 1.3 (i.e., Reference 6) source terms, and NUREG-0578 and NUREG-0737 (i.e., References 7 and 8, respectively) shielding review guidance. UFSAR Section 12.3.2.5 indicates that three locations constituted post-accident vital areas that require continuous occupancy. These were the CR, Technical Support Center (TSC), and the AEER where the remote shutdown panels are located. Additionally, UFSAR Table 12.3-7 shows dose rate contributions from pipes containing sources that are ECCS fluids, and Table 12.3-8 shows dose rates for essential post-accident paths.

The following table provides supporting information to address each of these areas.

Vital Area Dose Issue	AST Disposition
CR Dose	CR dose determinations using AST assumptions were performed in Attachment A of the LOCA calculation.
TSC Dose	TSC dose determinations were performed in Attachment E of the LOCA calculation using the Containment Leakage, ECCS Leakage, and MSIV Leakage RADTRAD models, as developed for the CR, where TSC X/Q and HVAC parameters replace the CR parameters.
AEER Dose	AEER doses were calculated based on the assumed required occupancy. For this design basis LOCA AST evaluation, safe shutdown can and will be accomplished from the CR. Therefore, the AEER remote shutdown panels do not require access. An AEER mission dose was calculated to support the action to start fans that provide containment air mixing for post-LOCA combustible gas control purposes.
Pipe Containing Source Dose Effects	The existing TID-14844 based evaluation of doses from ECCS piping shine is slightly conservative for AST because of the AST delay in release of certain isotopes from the reactor, and the reduction in total iodine core inventory release fractions from 50 percent to 30 percent. This is offset to a degree by the AST increase in assumed core releases to greater than 1 percent for the cesium, tellurium, strontium, and barium isotopic groups. However, based on Reference E-1 (i.e., discussed above), TID-based source terms for ECCS piping are conservative over the 30-day duration of these analyses. Therefore, the existing analyzed doses remain conservative and were not reanalyzed.
Pathway Dose Considerations	The evaluation of pathway doses included piping shine and shine from the activity in the secondary containment (i.e., refuel floor). At worst, the consequences would increase proportional to the increase in primary containment leakage, or by the ratio 1.0/0.635 or 1.575.

RAI FAXED TO EGC ON SEPTEMBER 18, 2009

RAI

The release from the upper secondary containment to the environment was modeled as a diffuse source assuming a vertical cross-sectional area about 92 meters wide and 16 meters tall. Section 3.2.4.1 of Regulatory Guide 1.194, "Atmospheric Relative Concentrations for Control Room Radiological Habitability Assessments at Nuclear Power Plants," states that diffuse source modeling should be used only for those situations in which the activity being released is homogeneously distributed throughout the building and when the assumed release rate from the building surface would be reasonably constant over the surface of the building. What ensures such a homogeneous distribution within and constant release of activity over the surface of the upper secondary containment wall at LaSalle? If a ventilation system would be used to facilitate the distribution, please provide appropriate figures and a discussion of how this would be accomplished. Also, please confirm that modeling the release as a diffuse source from the upper containment wall is more limiting than as a release through any penetration.

Response

The LSCS AST analyses do not assume the activity being released for an FHA is homogeneously distributed throughout the Reactor Building, and there is no ventilation system credited to facilitate distribution. Rather, the following methodology was used for conservatism to ensure that dose results from the diffuse source modeling were maximized.

The AST analyses assume that all of the activity released by the dropped fuel bundle is instantaneously distributed into the entire refueling floor volume. For LSCS, this volume is assumed to be one cubic foot to conservatively model the release without mixing in this artificially small volume. In order to postulate that virtually all of the activity is released out to the environment in a two hour period, the analyses assume that there is an artificially high exhaust rate using the standard decay equation and solving for lambda. Leakage through the building wall is not expected. However, any potential leakage through the side of the building is expected to be small with little driving force. In addition, there are no penetrations through the wall to the outside. Therefore, the analyses assume that there is a diffuse source for conservatism.

If the analyses assumed a homogeneous distribution of the activity throughout the building, rather than assuming all the activity is contained within a volume equal to one cubic foot, dose results would be lower. Therefore, the methodology described above was used for conservatism.

References

1. Letter from P. R. Simpson (Exelon Generation Company, LLC) to U.S. NRC, "Additional Information Regarding Request for License Amendment Regarding Application of Alternative Source Term," dated October 23, 2008

2. NUREG-0519, "Safety Evaluation Report Related to the Operation of the LaSalle County Station, Units 1 and 2," dated March 1981

3. Letter from M. D. Lynch (U.S. NRC) to D. L. Farrar (Commonwealth Edison Company), "Issuance of Amendments (TAC Nos. M93597 and M93598)," dated April 5, 1996

4. NRC Regulatory Issue Summary 2006-04, "Experience With Implementation of Alternative Source Terms," dated March 7, 2006

5. NRC Regulatory Guide 1.194, "Atmospheric Relative Concentrations for Control Room Radiological Habitability Assessments at Nuclear Power Plants"

6. NRC Regulatory Guide 1.3, "Assumptions for Evaluating the Potential Radiological Consequences of a Loss of Coolant Accident for Boiling Water Reactors"

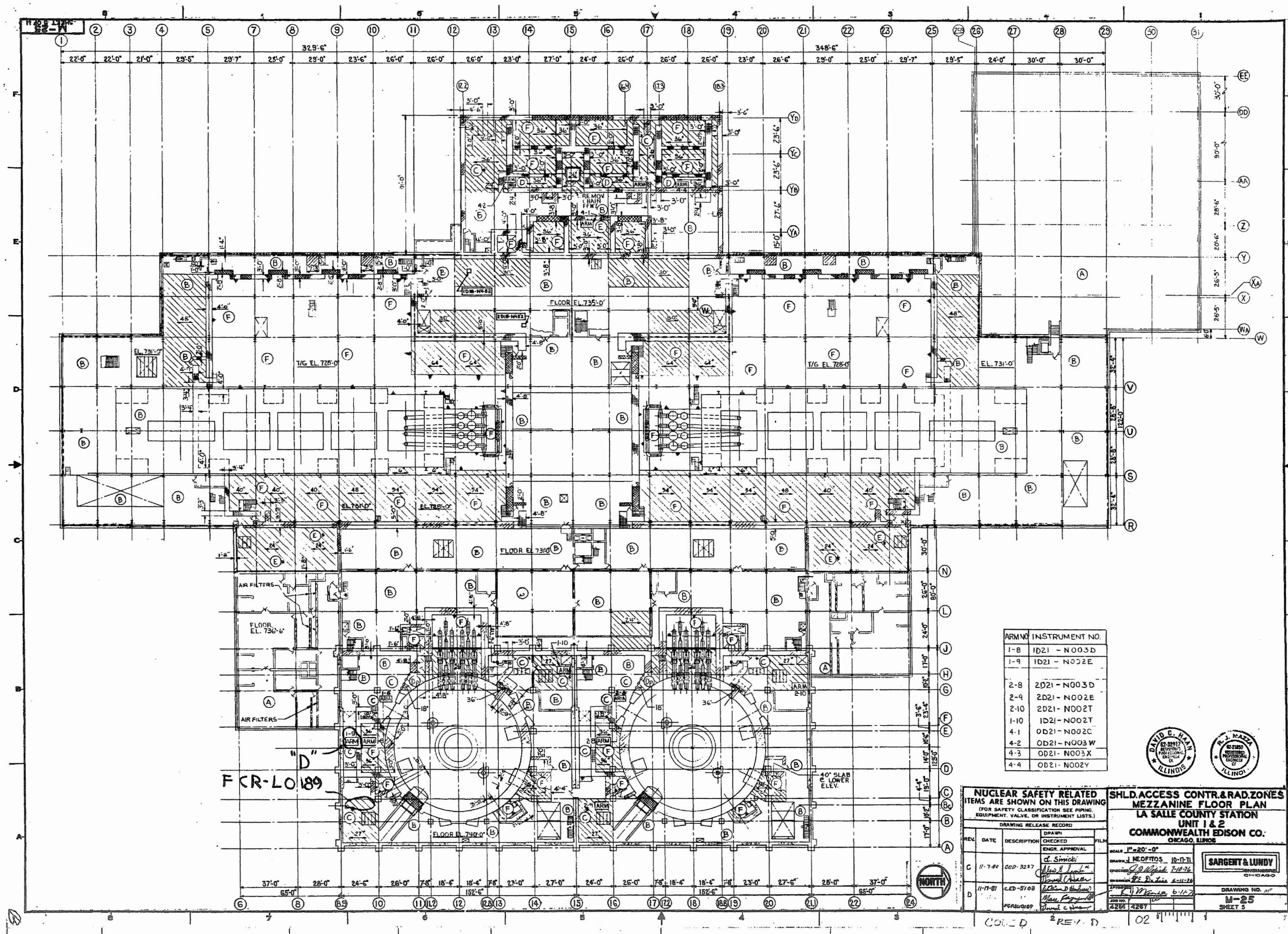
7. NUREG-0578, "TMI-2 Lessons Learned Task Force Status Report and Short-Term Recommendations"

8. NUREG-0737, "Clarification of TMI Action Plan Requirements"

9. NRC Regulatory Guide 1.145, "Atmospheric Dispersion Models for Potential Accident Consequence Assessments at Nuclear Power Plants"

ATTACHMENT 2

Drawing M-25, Sheet 5


NUCLEAR SAFETY RELATED
ITEMS ARE SHOWN ON THIS DRAWING
(FOR SAFETY CLASSIFICATION SEE PIPING, EQUIPMENT, VALVE, OR INSTRUMENT LISTS.)
SHLD. ACCESS CONTR. & RAD. ZONES
MEZZANINE FLOOR PLAN
LA SALLE COUNTY STATION
UNIT 1 & 2
COMMONWEALTH EDISON CO.
CHICAGO, ILLINOIS
SARGENT & LUNDY
CHICAGO
M-25
SHEET 5
ARM NO. INSTRUMENT NO.
1-8 1D21 - N003D
1-9 1D21 - N002E
2-8 2D21 - N003D
2-9 2D21 - N002E
2-10 2D21 - N002T
1-10 1D21 - N002T
4-1 0D21 - N002C
4-2 0D21 - N003W
4-3 0D21 - N003X
4-4 0D21 - N002Y
T/G EL. 726'-0"
FLOOR EL. 735'-0"
FLOOR EL. 731'-0"
FLOOR EL. 740'-0"
FLOOR EL. 736'-6"
EL. 731'-0"
AIR FILTERS
40" SLAB
LOWER ELEV.
NORTH
"D"
FCR-LO189