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10 CFR 50.90

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U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
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Clinton Power Station, Unit 1
Facility Operating License No. NPF-62
NRC Docket No. 50-461

Subject: License Amendment Request to Incorporate Revised Alternative Source Term Dose Calculation

Reference: Letter from K. N. Jabbour (NRC) to C. M. Crane (AmerGen), "Clinton Power Station, Unit 1 – Issuance of an Amendment – re: Application of an Alternative Source Term Methodology (TAC No. MB8365)," dated September 19, 2005 (NRC Accession No. ML052570461)

In accordance with 10 CFR 50.90, "Application for amendment of license, construction permit, or early site permit," Exelon Generation Company, LLC (EGC) requests an amendment to Facility Operating License No. NPF-62 for Clinton Power Station (CPS), Unit 1. The proposed change consists of modifications to the Loss-of-Coolant Accident (LOCA) dose calculation in the CPS Updated Safety Analysis Report (USAR) and a revision to the CPS Technical Specifications (TS). Specifically, NRC approval is requested for: 1) a TS change for the acceptance criteria for feedwater penetration leakage, 2) a revised element of analysis methodology pertaining to mixing of activity in the Secondary Containment, and 3) a more than minimal increase in Control Room post-LOCA dose consequences. The current LOCA dose calculation methodology was submitted to, and approved by, the NRC in Amendment 167 to NPF-62 (i.e., referenced letter), which implemented an alternative source term (AST) methodology in accordance with 10 CFR 50.67, "Accident source term."

This proposed change is in response to an NRC violation, and is necessary to restore compliance with existing regulatory requirements. The change removes a reduction factor credit for dual remote Control Room outside air intakes that had been previously misapplied. This change results in an increase in the Control Room atmospheric dispersion values (X/Qs) based on the elimination of a factor of four reduction factor from the dose analyses. This results in an increase in the post-accident dose for the Control Room. Although the resultant Control Room dose is below the current guideline value defined in NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants:

[Light Water Reactor] LWR Edition," Section 15.0.1, "Radiological Consequences Analyses Using Alternative Source Terms," Revision 0, the increase in consequences is more than minimal, and as such requires prior NRC approval in accordance with 10 CFR 50.59(c)(1)(iii). An operability evaluation that assessed the current configuration and determined that Control Room Ventilation system remains operable when considering the non-conforming condition described above will remain in-place until this issue is resolved. The proposed change also incorporates a new radionuclide core inventory that encompasses fuel cycle lengths from 12 up to 24 months.

The attachment and associated enclosures provide an evaluation of the proposed change, and includes a marked-up TS page, and marked-up USAR and TSs Bases pages, for information only.

The proposed amendment has been approved by the CPS Plant Operations Review Committee in accordance with the requirements of the EGC Quality Assurance Program.

EGC requests approval of the proposed license amendment by January 9, 2019. Once approved, the amendment will be implemented within 60 days.

In accordance with 10 CFR 50.91, "Notice for public comment; State consultation," Paragraph (b), EGC is notifying the State of Illinois of this application for license amendment by transmitting a copy of this letter and its attachments to the designated State Official.

There are no regulatory commitments contained in this letter. Should you have any questions concerning this letter, please contact Mr. Mitchel A. Mathews at (630) 657-2819.

I declare under penalty of perjury that the foregoing is true and correct. Executed on the 9th day of January 2018.

Respectfully,

Patrick R. Simpson

Patrick R. Simpson
Manager – Licensing
Exelon Generation Company, LLC

Attachment: Evaluation of Proposed Change

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

ATTACHMENT 1 - EVALUATION OF PROPOSED CHANGE

Subject: License Amendment Request to Incorporate Revised Alternative Source Term Dose Calculation

1.0 SUMMARY DESCRIPTION

In accordance with 10 CFR 50.90, Exelon Generation Company, LLC (EGC) requests an amendment to Facility Operating License No. NPF- 62 for Clinton Power Station (CPS), Unit 1 to revise the current licensing basis described in the CPS Updated Safety Analysis Report (USAR), Sections 6.4, "Habitability Systems," 9.4, "Heating Ventilation and Air Conditioning (HVAC) Systems," 15.4.9, "Control Rod Drop Accident (CRDA)," 15.6.5, "Loss-of-Coolant Accidents (Resulting from Spectrum of Postulated Piping Break Within the Reactor Coolant Pressure Boundary) - Inside Containment," and 15.7.4, "Fuel Handling Accident." In addition, the proposed amendment will revise CPS Technical Specification (TS) 3.6.1.3, "Primary Containment Isolation Valves (PCIVs)."

The proposed change will revise the Loss-of-Coolant Accident (LOCA) dose calculation and the subsequent calculation results. The initial LOCA dose calculation methodology was submitted and approved by the NRC in Amendment 167 to NPF-62 (i.e., Reference 6.7), which implemented an alternative source term (AST) methodology in accordance with 10 CFR 50.67, "Accident source term." An additional modification to the LOCA dose calculation methodology was submitted on January 29, 2016. This modification revised the post-loss-of-coolant-accident drawdown time for secondary containment from 12 minutes to 19 minutes. This was approved by the NRC in Amendment 210 to NPF-62 (i.e., Reference 6.8).

This proposed change is in response to an NRC violation, and is necessary to restore compliance with existing regulatory requirements. The change removes a reduction factor credit for dual remote Control Room air intakes that had been previously misapplied. This change results in an increase in the Control Room atmospheric dispersion values (X/Qs) based on the elimination of a factor of four reduction factor from the dose analyses. This results in an increase in the post-accident dose for the Control Room. Although the resultant Control Room dose is below the current guideline value defined in NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition," (SRP) Section 15.0.1, "Radiological Consequences Analyses Using Alternative Source Terms," Revision 0, the increase in consequences is more than minimal, and as such requires prior NRC approval in accordance with 10 CFR 50.59(c)(1)(iii). An operability evaluation that assessed the current configuration and determined that Control Room Ventilation system remains operable when considering the non-conforming condition described above will remain in-place until this issue is resolved. The proposed change also incorporates the results of sensitivity studies based on fuel cycle lengths from 12 up to 24 months.

EGC has conducted evaluations to validate that the proposed configuration complies with the applicable 10 CFR 50, Appendix A, General Design Criteria (GDC), as well as the requirements of 10 CFR 50.49, "Environmental qualification of electric equipment important to safety for nuclear power plants," 10 CFR 50.67, "Accident source term," Regulatory Guide (RG) 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors," Revision 0, and RG 1.194, "Atmospheric Relative Concentrations for Control Room Radiological Habitability Assessments at Nuclear Power Plants," dated June 2003.

2.0 DETAILED DESCRIPTION

Because the change in Control Room atmospheric dispersion (X/Qs) is an input value into the post-accident dose calculations described in USAR Sections 15.6.5 and 15.7.4, EGC has re-evaluated the impact of the increased X/Qs on post-accident radiological consequences for the Control Room. The revised fuel handling accident (FHA) and LOCA dose calculations were performed in accordance with the guidance in RG 1.183 and SRP Section 15.0.1, "Radiological Consequences Analyses Using Alternative Source Terms." The FHA was evaluated under 10 CFR 50.59. This evaluation determined that changes to the FHA analyses do not require NRC approval.

RG 1.194 requires that the operator have the ability to select the remote intake with the lowest activity to take credit for dual Control Room remote intakes. Following a loss of divisional power (i.e., a single failure), the ability to select the more favorable remote intake from a Control Room radiological dose perspective would not be possible. Consequently, the previous credit taken for a dual Control Room remote intake can no longer be taken due to single failure issues with the current system design. Based on this, the previous reduction in the atmospheric dispersion by a factor of four to account for selection of the most favorable remote intake is no longer applied.

In addition to the change in Control Room atmospheric dispersion values, additional changes to the current LOCA dose calculation were implemented. These include: 1) reduction in the Feedwater Isolation Valve liquid leakage from 2.0 gallons per minute (gpm) to 1.5 gpm, 2) reduction in the Feedwater Isolation Valve air leakage from 10.98 cubic feet per minute (cfm) to 8.64 cfm, 3) reduction in the Control Room filtered inleakage from 1,100 cfm to 1,000 cfm, 4) credit taken for mixing in 50% of the secondary containment volume, 5) incorporation of a new core inventory, and 6) inclusion of a sensitivity study for a bounding 12 to 24-month fuel cycle core inventory. The new core inventory documents a bounding annual fuel cycle core inventory based on any combination of the core design parameters and enrichment to bound future core designs. As a result of the potential to implement 24-month fuel cycle operation, the Beginning of Cycle (BOC) burnup evaluated for annual cycles was not bounding (i.e., 24-month cycle has a lower BOC burnup). Therefore, a revised core inventory based on a revised BOC burnup was developed. This inventory was used in sensitivity studies to determine whether the dose consequences using the revised core inventory or the bounding annual fuel cycle core inventory is more limiting. By evaluating the dose consequences for both annual cycle core inventories and the revised core inventory the analyses cover operation for any cycle between 12 and 24 months in length.

For LOCA, the revised radiological consequence evaluation identified that the post-accident (i.e., post-LOCA) doses for the Exclusion Area Boundary (EAB) and Low Population Zone (LPZ) decreased, while the value for the Control Room post-accident dose increased. The change in the current post-accident dose for the EAB and LPZ, which is documented in USAR Table 15.6.5-6, "Loss-of-Coolant Accident (Design Basis Analysis) Radiological Effects," was not more than minimal, and therefore could be changed in accordance with 10 CFR 50.59.

However, the increase in post-accident Control Room dose, which is documented in USAR Subsection 15.6.5.5.2, "Control Room," while within the 10 CFR 50.67 limit, was more than minimal, and as such requires prior NRC approval. Specifically, the postulated post-LOCA Control Room dose will increase from the current value of 4.84 roentgen equivalent man (REM)

to 4.89 REM Total Effective Dose Equivalent (TEDE), relative to the RG 1.183, 10 CFR 50.67, and 10 CFR 50, Appendix A, "General Design Criteria for Nuclear Power Plants," Criterion (GDC) 19, "Control Room," limit of 5 REM TEDE. The sensitivity study results for a 12 to 24-month fuel cycle demonstrated that the dose consequences based on an annual fuel cycle core inventory are bounding.

In addition, the Feedwater Isolation Valve (FWIV) liquid leakage is reduced from 2.0 gallons per minute (gpm) to 1.5 gpm. This leakage limit is described in Technical Specification 3.6.1.3, "Primary Containment Isolation Valves (PCIVs)," Surveillance Requirement (SR) 3.6.1.3.11 as shown in Figure 1 below and cannot be revised without prior NRC approval. Enclosure 1 provides a marked-up version of the current TS for NRC review, and Enclosure 2 provides a marked-up version of the affected USAR and TS Bases pages, for information only.

SR 3.6.1.3.11	----------------NOTE-------------------- Only required to be met in MODES 1, 2, and 3. -- Verify that the combined leakage rate for both primary containment feedwater penetrations is ≤ ~~2 gpm~~ 1.5 gpm when pressurized to ≥ 1.1 P_a.	In accordance with the Primary Containment Leakage Rate Testing Program.

Figure 1: *Proposed Revision to SR 3.6.1.3.11*

Other design basis analyses are also impacted by these changes. These analyses are the Fuel Handling Accident (FHA), the Control Rod Drop Accident (CRDA), and the suppression pool pH analysis.

For the FHA, the revised analysis: 1) removes the previous credit for the dual Control Room remote air intake due to the single failure issues, 2) takes credit for the operation and filtration of the Control Room emergency ventilation system after 20 minutes into the event, 3) the Control Room filtered inleakage changes from 650 cfm to 1,000 cfm, 4) radial peaking factor changes from 1.7 to 1.8 (to provide margin to the cycle to cycle variation in core design), 5) updates the source term for a 12-month fuel cycle, and 6) incorporates a sensitivity study for a bounding 12 to 24-month fuel cycle core inventory. This analysis continues using the same AST methodology described in the USAR. The results of the revised analysis indicate an increase in the dose consequences for the EAB, the LPZ, and the Control Room. The increase in dose consequences does not exceed the minimal increase and remains below the RG 1.183 acceptance criterion (i.e., see Enclosure 5). The sensitivity study results demonstrated that the dose consequences based on an annual fuel cycle core inventory are bounding.

The CRDA analysis was also revised to: 1) incorporate the latest 12-month fuel cycle source term, 2) update the radial peaking factor from 1.7 to 1.8, 3) increase the flow rate into the Control Room to 5,410 cfm (i.e., normal Control Room flow of 4,000 cfm plus 10% (4,400 cfm) combined with an inleakage of 1,000 cfm, and 10 cfm for ingress/egress to the Control Room), and 4) incorporate a sensitivity study for a bounding 12 to 24-month fuel cycle core inventory. This

analysis was performed using AST methodology and is based on conservative assumptions considered to be acceptable to the NRC for the purpose of determining the adequacy of the plant design to meet 10 CFR 50.67 limits. USAR Section 15.4.9.5.1, "Analysis," states that, "The design basis analysis is based on the NRC's Standard Review Plan 15.4.9 and Regulatory Guide 1.183." This analysis continues to use the same AST methodology described in the USAR. The results of the revised analysis indicate an increase in the dose consequences for the EAB, the LPZ, and the Control Room. For this analysis, the dose consequences based on an annual fuel cycle core inventory were not bounding for all receptor locations. Therefore, the dose consequences for a CRDA will be based on the revised 12 to 24-month fuel cycle core inventory for the EAB and LPZ doses and the annual fuel cycle core inventory for the Control Room. The increase in dose consequences does not exceed the minimal increase and remains below the RG 1.183 acceptance criterion (i.e., see Enclosure 5).

The suppression pool pH analysis is also impacted by the new 12-month fuel cycle source term. USAR Section 15.6.5.5.1.1, "Fission Product Release from Fuel," describes the fission product release from fuel after a LOCA accident and states, "the suppression pool pH is controlled at values above 7 following the core release period." The revised suppression pool pH calculation verifies that the suppression pool pH remains above 7.0 when the source term is updated to the latest 12-month fuel cycle source term or the core inventory based on a 12 to 24-month fuel cycle.

3.0 TECHNICAL EVALUATION

3.1 System Description

The Control Room Heating Ventilation and Air-Conditioning (HVAC) system is designed to ensure that Main Control Room personnel can remain inside all spaces served by the Control Room HVAC system during normal station conditions and accident conditions in compliance with GDC 19. The Control Room HVAC system is an engineered safety feature and is designed with sufficient redundancy to ensure operation and habitability under any accident condition, and to ensure operation under normal or abnormal station conditions. The system is designed to meet single-failure criteria with the exception of heating and humidification equipment and the remote outside air intakes. The heating and humidification provided in the Control Room HVAC system are not essential for the safety of operating personnel or the function of the safety-related equipment.

The system is designed to maintain a positive pressure within the Control Room Envelope with respect to the adjacent areas to preclude infiltration of unconditioned air, during all the operating modes except when the system is in recirculation mode or when the system is in the maximum outside air purge mode. Maintaining positive pressure during normal mode is not a safety function.

The outside air is normally brought in through one of two minimum outside air intakes and supplied to the operating return air fan suction. These two intakes are physically separated by over 375 feet, and are called "minimum" outside air intakes since one of them is used to supply the minimum required makeup air during normal and abnormal conditions. Wall openings for these intake ducts are missile protected. A third missile-protected air intake is provided to

introduce 100% outside air to permit purging of the Control Room, if required. The minimum quantity of outside air required to provide makeup air for expected leakages and locker room exhaust fan operation and still maintain not less than 0.125 inches of H_20 positive pressure in the Control Room with respect to the adjacent areas is introduced under all station operating conditions. This positive pressure is not maintained when the system is in the maximum outside air purge mode. The Control Room makeup air maintains Control Room pressure at a positive 0.125 inches of H_2O or greater relative to the adjacent areas. During normal operating conditions, the makeup flow rate (i.e., up to approximately 4000 cfm design, 1000 cfm of which is exhausted to atmosphere by the locker room exhaust fan) will vary as doors are opened and closed. Maintaining positive pressure during normal mode is not a safety function.

The two minimum outside air intakes are located on the east and west sides of the plant respectively such that advantages can be derived from the outside wind direction in the event contaminants are present in one outside air intake. High radiation measured at either minimum outside air intake automatically closes the normal intake damper (i.e., by sending closure signals to dampers 0VC03YA, "Control Room Train A Minimum Outside Air Damper," 0VC115YA, "Control Room Isolation Damper," 0VC03YB, "Control Room Train B Minimum Outside Air Damper," and 0VC115YB "Control Room Isolation Damper") and initiates operation of one of two 100% standby makeup air filter trains. This depends upon which HVAC system is operating. This in turn sends a signal to open the appropriate makeup filter train inlet and outlet dampers (inlet: Control Room Isolation Damper, 0VC02YA or 0VC02YB; outlet: Control Room Isolation Damper 0VC06YA or 0VC06YB). The recirculation air filter trains are automatically placed in service upon receiving the same high radiation signal. This is accomplished by sending closure signals to Control Room Isolation Dampers 0VC10YA and 0VC10YB, and by sending a signal to open the appropriate recirculation filter train inlet and outlet dampers (inlet: Control Room Isolation Damper 0VC09YA or 0VC09YB; outlet: Control Room Isolation Damper 0VC11YA or 0VC11YB). For the removal of radioactive contaminants, minimum outside air is thus introduced through the demister, electric heater, medium filter, HEPA filter, iodine adsorbing beds, and downstream HEPA filter. The prefilter limits large particulate loading of the HEPA filter, and the single-stage electric heater assures air entering the charcoal has a relative humidity that is no higher than 70%. The makeup air filter trains are capable of removing 99.95% of all particulate matter larger than 0.3 microns and no less than 99% of all forms of iodine. The recirculation charcoal filter trains are capable of removing no less than 70% of all forms of iodine. The high radiation signal also closes the locker room exhaust dampers, which trips the exhaust fan.

By letter dated April 3, 2003, (Reference 6.1), as supplemented by letters dated December 23, 2003, December 9, 2004, December 17, 2004, March 30, 2005, and August 19, 2005 (i.e., References 6.2 through 6.6, respectively), AmerGen Energy Company, LLC(AmerGen) (i.e., the CPS licensee prior to EGC), requested a license amendment to support Alternative Source Term (AST) methodology for design basis accidents, in accordance with 10 CFR 50.67, with the exception that Technical Information Document (TID) 14844, "Calculation of Distance Factors for Power and Test Reactor Sites," would continue to be used as the radiation dose basis for equipment environmental qualification.

The LOCA analysis supporting the Reference 6.1 license amendment request assumed credit for dual, remote Control Room air intakes and, as allowed by RG 1.194 (i.e., a reduction in the atmospheric dispersion by a factor of four). By letter and Safety Evaluation dated September

19, 2005 (i.e., Reference 6.7), the NRC approved Amendment 167 to NPF-62.

In May 2015, EGC implemented a design change for CPS that revised the operating cycle to the current value of 12 months. This design change revised an input parameter to the original AST post-LOCA dose calculation. As a result, EGC reanalyzed the original AST dose calculations with a 12-month operating cycle. The resultant change to the EAB, LPZ, and Control Room post-LOCA dose values remained below the 10 CFR 50.67 limits, and were implemented in accordance with the requirements of 10 CFR 50.59 (i.e., the resultant change to the values was not more than minimal). An additional modification to the LOCA dose calculation methodology was submitted on January 29, 2016. This modification revised the post-LOCA drawdown time for Secondary Containment from 12 minutes to 19 minutes. This was approved by the NRC in Amendment 210 to NPF-62 (i.e., Reference 6.9).

The Control Room operator can operate hand switches in the Control Room to close the minimum outside air intake damper, which was being used, and to open the other intake damper, to take advantage of the separation of the two air intakes and minimize radioactivity intake.

RG 1.194 requires that the operator have the ability to select the remote intake with the lowest activity to take credit for dual Control Room remote intakes. EGC recently determined that, following a loss of divisional power (i.e., a single failure), the ability to select the more favorable remote intake would not be possible. Consequently, the previous dose reduction credit taken for a dual Control Room remote intake is no longer taken due to the possibility of a single failure hindering Control Room intake realignment with the current system design. Based on this, the previous reduction in the atmospheric dispersion by a factor of four to account for selection of the most favorable remote intake is no longer applied.

3.2 Evaluation of Proposed Change

EGC has evaluated the impact of the revised Control Room atmospheric dispersion (X/Q) values on post-LOCA radiation doses (i.e., TEDE) in the Control Room. The CRDA and FHA were evaluated under 10 CFR 50.59 and did not require prior NRC approval; however, the following aspects of the proposed change require NRC approval: 1) A TS amendment because of the proposed change to feedwater penetration leakage acceptance criterion in SR 3.6.1.3.11, 2) a change regarding the mixing of the secondary containment volume that is considered a change in an input parameter that constitutes non-conservative change to an element of the methodology for determining post-LOCA doses, and 3) A more than a minimal increase in Control Room post-LOCA dose consequences as shown in Table 1 below. No other aspects of this evaluation require prior NRC review and approval.

The results of the evaluations for the 12-month operating cycle with increased Control Room X/Qs are provided in Table 1 below relative to the current AST dose calculation values. The dose consequences associated with a 12-month fuel cycle, given below, bound the dose consequences based on a 12 to 24-month fuel cycle core inventory. Enclosure 3 provides the physical input parameters and assumptions that were used for the revised post-LOCA dose calculations, as well as the corresponding parameters used in the current calculation (i.e., in

support of Amendment 210). Enclosure 4 provides a discussion of both quantitative and qualitative uncertainties in the post- LOCA dose calculations, relative to the 10 CFR 50.67 limits.

Table 1: Post-LOCA Radiation Doses (REM TEDE)

	Current USAR Chapter 15.6.5 Results A	Results with Revised Control Room X/Qs B	10 CFR 50.67 Limits C	Increase in Dose B - A	More Than Minimal Increase in Dose Consequences Thresholds 10% of C - A
EAB	17.31	16.18	25	N/A	0.769
LPZ Boundary	7.42	6.81	25	N/A	1.758
Control Room	4.84	4.89	5	0.05	0.016

4.0 REGULATORY EVALUATION

4.1 Applicable Regulatory Requirements/Criteria

The proposed change will revise an input parameter for the post-accident dose calculation and the subsequent calculation results. The following NRC requirements and guidance documents are applicable to the review of the proposed change:

10 CFR 50.49, "Environmental qualification of electric equipment important to safety for nuclear power plants," requires licensees to establish a program for qualifying the safety-related electric equipment that is relied upon to remain functional during and following design basis events to ensure:

(i) The integrity of the reactor coolant pressure boundary;

(ii) The capability to shut down the reactor and maintain it in a safe shutdown condition; or

(iii) The capability to prevent or mitigate the consequences of accidents that could result in potential offsite exposures comparable to the guidelines in 10 CFR 50.67.

10 CFR 50.67, "Accident source term," requires those licensees with an approved alternative source term to ensure, with reasonable assurance, that:

(i) An individual located at any point on the boundary of the exclusion area for any two-hour period following the onset of the postulated fission product release, would not receive a radiation dose in excess of 0.25 sieverts (Sv) (i.e., 25 REM) total effective dose equivalent (TEDE).

(ii) An individual located at any point on the outer boundary of the low population zone (LPZ), who is exposed to the radioactive cloud resulting from the postulated fission product release (i.e., during the entire period of its passage), would not receive a radiation dose in excess of 0.25 Sv (i.e., 25 REM) total effective dose equivalent (TEDE).

(iii) Adequate radiation protection is provided to permit access to and occupancy of the Control Room under accident conditions without personnel receiving radiation exposures in excess of 0.05 Sv (i.e., 5 REM) total effective dose equivalent (TEDE) for the duration of the accident.

10 CFR 50, Appendix A, General Design Criterion (GDC) 19, "Control Room" requires licensees to provide adequate radiation protection to permit access and occupancy of the Control Room under accident conditions without personnel receiving radiation exposures in excess of 5 REM whole body, or its equivalent to any part of the body, for the duration of the accident.

Regulatory Guide (RG) 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors," provides guidance to licensees of operating power reactors on acceptable applications of alternative source terms (ASTs); the scope, nature, and documentation of associated analyses and evaluations; consideration of impacts on analyzed risk; acceptable radiological analysis assumptions for use in conjunction with the accepted AST, and content of submittals.

RG 1.194, "Atmospheric Relative Concentrations for Control Room Radiological Habitability Assessments at Nuclear Power Plants," dated June 2003, provides guidance to licensees of operating power reactors on acceptable methods for determining atmospheric relative concentration (X/Q) values in support of design basis control room radiological habitability assessments at nuclear power plants. This guide describes methods acceptable to the NRC staff for determining X/Q values that will be used in control room radiological habitability assessments performed in support of applications for licenses and license amendment requests.

EGC has validated that the proposed change complies with the applicable regulations, requirements, and guidance.

4.2 <u>No Significant Hazards Consideration</u>

Exelon Generation Company, LLC (EGC) has determined that the revised Clinton Power Station (CPS), Unit 1 Control Room atmospheric dispersion (X/Q) values will impact the post-loss of coolant accident (LOCA) dose consequences. EGC has evaluated the impact of the revised

X/Qs on post-accident radiological consequences, utilizing the NRC- approved alternative source term (AST) methodology. The Control Rod Drop Accident (CRDA) and Fuel Handling Accident (FHA) were also impacted by the change in source term and control room X/Q but can be implemented without prior NRC approval under 10 CFR 50.59, "Changes, tests and experiments." This radiological consequence evaluation resulted in a projected increase in the post-accident Control Room dose. Although the increased value for post-LOCA Control Room dose remains within the 10 CFR 50.67, "Accident source term," regulatory limit, the increase is more than minimal, and as such requires prior NRC approval.

The revised post-LOCA Control Room dose calculation was performed in accordance with the guidance in Regulatory Guide (RG) 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors" and NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition," (SRP), Section 15.0.1, "Radiological Consequences Analyses Using Alternative Source Terms."

EGC has evaluated whether a significant hazards consideration is involved with the proposed amendments by focusing on the three standards set forth in 10 CFR 50.92, "Issuance of amendment," as discussed below:

1. **Does the proposed amendment involve a significant increase in the probability or consequences of an accident previously evaluated?**

   Response: No

   The proposed change results in higher Control Room X/Qs which are equivalent to reduced atmospheric dispersion. The increased Control Room X/Qs, in turn, result in higher post-accident Control Room doses. Neither the higher X/Qs, nor the resultant increase in the Control Room doses affect any initiator or precursor of any accident previously evaluated. Therefore, the proposed change does not involve a significant increase in the probability of an accident previously evaluated.

   The proposed change results in an increase in the post-LOCA radiological dose to a Control Room occupant. However, the resultant post-LOCA Control Room dose remains within the regulatory limits of 10 CFR 50.67 and 10 CFR 50, Appendix A, "General Design Criteria for Nuclear Power Plants," Criterion 19, "Control Room." Therefore, the proposed change does not involve a significant increase in the consequences of an accident previously evaluated.

   In summary, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. **Does the proposed amendment create the possibility of a new or different kind of accident from any accident previously evaluated?**

   Response: No

   The proposed change does not alter the design function or operation of the Control Room heating, ventilation, and air-conditioning (HVAC) system, or the ability of this system to perform its design function. The only change is the removal of the Control Room dose

reduction factor credit taken for providing a dual remote Control Room air intake. The proposed change does not alter the safety limits, or safety analysis associated with the operation of the plant. Accordingly, the change does not introduce any new accident initiators. Rather, this proposed change is the result of an evaluation of the Control Room doses following the most limiting LOCA that can occur at CPS. The proposed change does not introduce any new modes of plant operation. As a result, no new failure modes are introduced.

Therefore, the proposed change does not create the possibility of a new or different kind of accident from any previously evaluated.

3. **Does the proposed amendment involve a significant reduction in a margin of safety?**

   Response: No

   The revised post-LOCA dose consequences to a Control Room occupant were calculated in accordance with the requirements of 10 CFR 50.67, RG 1.183, and NRC SRP Section 15.0.1, "Radiological Consequences Analyses Using Alternative Source Terms."

   The margin of safety is considered to be that provided by meeting the applicable regulatory limits. The increased Control Room X/Qs result in an increase in Control Room dose following the design basis LOCA; however, since the Control Room dose following the design basis accident remains within the regulatory limits, there is not a significant reduction in a margin of safety.

   Therefore, operation of CPS in accordance with the proposed change will not involve a significant reduction in a margin of safety.

   Based on the above, EGC concludes that the proposed amendment does not involve a significant hazards consideration under the standards set forth in 10 CFR 50.92(c), and, accordingly, a finding of no significant hazards consideration is justified.

4.3 Conclusions

Based on the considerations discussed above: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

5.0 ENVIRONMENTAL CONSIDERATION

EGC has evaluated the proposed amendment for environmental considerations. The review has resulted in the determination that the proposed amendment would change a requirement with respect to installation or use of a facility component located within the restricted area, as defined in 10 CFR 20, "Standards for Protection Against Radiation," or would change an inspection or surveillance requirement. However, the proposed amendment does not involve:

(i) a significant hazards consideration, (ii) a significant change in the types or significant increase in the amounts of any effluent that may be released offsite, or (iii) a significant increase in individual or cumulative occupational radiation exposure. Accordingly, the proposed amendment meets the eligibility criterion for categorical exclusion set forth in 10 CFR 51.22(c)(9). Therefore, pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the proposed amendment.

6.0 REFERENCES

6.1 Letter from M. J. Pacilio (AmerGen Energy Company, LLC(AmerGen)) to NRC, "Request for License Amendment Related to Application of Alternative Source Term," dated April 3, 2003 (NRC Accession No. ML031040096)

6.2 Letter from K. R. Jury (AmerGen) to NRC, "Additional Information Supporting the Request for License Amendment Related to Application of Alternative Source Term," dated December 23, 2003 (NRC Accession No. ML040020026)

6.3 Letter from K. R. Jury to NRC, "Additional Information Supporting the Request for License Amendment Related to Application of Alternative Source Term," dated December 9, 2004 (NRC Accession No. ML043450106)

6.4 Letter from K. R. Jury (AmerGen) to NRC, "Additional Information Supporting the Request for License Amendment Related to Application of Alternative Source Term," dated December 17, 2004 (NRC Accession No. ML043520326)

6.5 Letter from K. R. Jury (AmerGen) to NRC, "Additional Information Supporting the Request for License Amendment Related to Application of Alternative Source Term," dated March 30, 2005 (NRC Accession No. ML050900165)

6.6 Letter from K. R. Jury (AmerGen) to NRC, "Additional Information Supporting the Request for License Amendment Related to Application of Alternative Source Term," dated August 19, 2005 (NRC Accession No. ML052430196)

6.7 Letter from K. N. Jabbour (NRC) to C. M. Crane (AmerGen), "Clinton Power Station, Unit 1 – Issuance of an Amendment – re: Application of an Alternative Source Term Methodology (TAC No. MB8365)," dated September 19, 2005 (NRC Accession No. ML052570461)

6.8 Letter from E. A. Brown (NRC) to B. C. Hanson (EGC), "Clinton Power Station, Unit 1 – Issuance of an Amendment Concerning Application of Revised Alternative Source Term (CAC No. MF7336)," August 17, 2016 (NRC Accession No. ML16217A332)

Clinton Power Station, Unit 1

Application to Incorporate Revised Alternative Source Term Dose Calculation

ENCLOSURE 1 – PROPOSED TECHNICAL SPECIFICATIONS CHANGE (MARK-UP)

3.6-19a

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.6.1.3.11	-----------------NOTE--------------------- Only required to be met in MODES 1, 2, and 3. -- Verify that the combined leakage rate for both primary containment feedwater penetrations is ≤ ~~2 gpm~~ 1.5 gpm when pressurized to ≥ 1.1 P_a.	In accordance with the Primary Containment Leakage Rate Testing Program.
SR 3.6.1.3.12	Verify each instrumentation line excess flow check primary containment isolation valve actuates within the required range.	In accordance with the Surveillance Frequency Control program

Clinton Power Station, Unit 1

Application to Incorporate Revised Alternative Source Term Dose Calculation

ENCLOSURE 2 – REVISED UPDATED SAFETY ANALYSIS REPORT (USAR) AND TECHNICAL SPECIFICATIONS BASES PAGES (MARK-UPS)

USAR Pages

6.4-4
6.4-5
9.4-2
9.4-4
9.4-6
9.4-7
15.4-19
15.4-36
15.6-10
15.6-12
15.6-13
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15.6-34
Figure 15.6.5-1, Page 1 of 2
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15.7-12
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15.7-29
15.7-32

TS Bases Pages

B 3.6-28a

No credit is taken for this dual, separated outside air intake design feature for evaluating the radiological consequences of design basis accidents.

An isometric view of the control room is shown in Figure 6.4-3. Shielding details are provided in Table 6.4-2 and Drawings M01-1524 and M01-1526. Labyrinths at the entrances and shielding slabs over the ceiling penetrations are provided to preclude potential radiation streaming.

6.4.3 System Operating Procedures

During normal operation, one of the two 100% capacity trains of the main control room HVAC system continuously processes the recirculated and makeup air to maintain desired air temperature and quality. The mixture of recirculated and makeup air is filtered by high efficiency (85% by NBS dust spot method), waterproof and fire retardant fiberglass filters. Each control room HVAC system train, with the exception of the chiller, may be manually started through its respective control switch located in the main control room. The system chiller must be manually started from the chiller control panel. Local control of system train components is provided by allowing transfer of system control to the local control panel using the remote/local selector switch. The sequence of operation of the controls is described in Subsection 7.3.1.1.6.

In the event of high radiation detection at the minimum outside air intake(s), the radiation monitoring system will activate an alarm in the main control room, automatically start the makeup filter train, route the supply air stream through the charcoal beds in the recirculation air filter associated with the operating HVAC air handling system and trip the locker room exhaust fan. The 3000 cfm makeup air is automatically routed through the makeup air filter train, for removal of radioactive and nonradioactive particulates and iodine, before going into the control room. The control room operator can operate handswitches to close the minimum outside air intake damper, which was being used, and to open the other intake damper, to take advantage of the separation of the two air intakes and minimize radioactivity intake.

Areas above and below those spaces served by the main control room HVAC system will be maintained at a negative pressure, with respect to those spaces served by the control room HVAC system, by the auxiliary building HVAC system.

6.4.4 Design Evaluations

The control room HVAC system is designed to maintain a habitable environment compatible with prolonged service life of safety-related components in the control room under all station operating conditions. The system is provided with redundant equipment trains to meet the single failure criteria. The equipment trains are powered from redundant ESF buses and are operable during loss of offsite power. All of the control room HVAC system equipment except recirculation, heating and humidification equipment, and exhaust fan is designed for Seismic Category I loads.

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6.4.4.1 Radiological Protection

The two outside minimum air intakes are separated from each other by being located on the east and west sides of the plant respectively. In the post-LOCA environment, air can be supplied to the control room from the intake where the airborne contamination is lowest.

Radiation monitors near the minimum outside air intakes are designed to initiate an alarm in the control room and a control signal to isolate the normal ventilation path, start up one of the makeup air filter trains and divert the outside air through it, route the supply air stream through the recirculation air charcoal adsorber and trip the locker room exhaust fan. Subsequently, readings from these monitors are used by the operator to select the minimum outside air intake

No credit is taken for this dual, separated outside air intake design feature for evaluating the radiological consequences of design basis accidents.

if both minimum outside air intakes are available.

with the lower airborne contamination. The makeup air filter trains are provided with HEPA and charcoal filters to remove particulates and iodine from the makeup air. The recirculation filter units are provided with high efficiency air and charcoal filters to further remove particulates and iodine from the supply air.

The control room is maintained at a positive pressure with respect to adjacent areas when operated in the radiation mode to minimize the ingress of unfiltered outside air. Air locks are also provided at the entrances for the same purpose.

The calculated dose to personnel inside the control room following accidents is reported in Chapter 15.

6.4.4.2 Toxic Gas Protection

Transportation and traffic surveys have shown that the frequency of transportation or delivery of chlorine does not dictate design for a potential chlorine hazard. Additionally, gaseous chlorine is no longer allowed on site by plant procedure and there are no other significant depots of chlorine within a five mile radius of the site. Therefore, no automatic initiation of the control room ventilation chlorine mode and no chlorine detectors are required.

A breathing air system is provided for control room operators. Protection of control room operators against other hazardous chemicals as required by Regulatory Guide 1.78 is provided.

There are three control room ventilation intakes. The first is located on the northwest corner of the Auxiliary Building. The second intake is located on the north wall of the Control Building. The third intake is located on the east wall of the Control Building.

In the highly unlikely event of a chlorine gas problem, the control room operator may manually initiate the chlorine mode of operation for the Control Room HVAC System. (Q&R 450.1)

For refrigerant vapors (freon) or their decomposition products, protection is provided by:

a. Locating refrigerant-using equipment (chillers) with the most refrigerant far from the control room, on the lowest elevation of the control and fuel buildings. These areas are ventilated independently from the control room system.

b. Providing pressure relief valves on each chiller piped directly to the outdoors, except for the drywell water chillers relief line which is connected to the fuel building exhaust system.

c. Pressurizing the control room to preclude infiltration.

d. Ventilating the control room HVAC equipment room independently from the control room HVAC system.

e. Utilizing chilled water cooling coils instead of direct expansion coils.

outside air intake dampers,

9.4.1.1.1 Safety Design Bases

a. The control room HVAC system is an engineered safety feature and is designed with sufficient redundancy to ensure operation and habitability under any accident conditions, and to ensure operation under normal or abnormal station conditions. The system is designed to meet single-failure criteria with the exception of heating and humidification equipment. The heating and humidification provided in the control room HVAC system are not essential for the safety of operating personnel of the function of the safety related equipment.

b. The control room HVAC system is designed to maintain the following temperature and humidity ranges:

Area	Temperature Range	Humidity Range
Main Control Room	71°F to 75°F	35% to 55%
Computer Room	68°F to 72°F	35% to 55%
TMI Panel Room	71°F to 75°F	35% to 55%
Techinical Support Center	71°F to 75°F	35% to 55%
Control Panel Area	71°F to 75°F	No humidity control
Misc. offices & locker rooms	73°F to 77°F	No humidity control
Operations Support Area	55°F to 104°F	No humidity control

c. The system is designed to maintain a positive pressure within the control room envelope with respect to the adjacent areas to preclude infiltration of unconditioned air, during all the operating modes except when the system is in recirculation mode or when the system is in the maximum outside air purge mode. Maintaining positive pressure during normal mode is not a safety function.

d. The system is provided with radiation monitors to monitor radiation levels at the minimum outside air intakes and upon detecting a radiation level exceeding a preset value automatically limits the introduction of contaminants into the system by filtering the contaminated air.

This ventilation system design coupled with the control room shielding assures that the dose to the operators inside the control room is within the limits specified by Criterion 19 of 10 CFR 50 Appendix A for the duration of a design-basis accident.

e. There is no chlorine detection system. The chlorine detectors are retired in place.

f. Provision is made in the system to clean up the inside environs upon smoke detection in the return air or at the minimum outside air intakes.

Due to susceptibility to single failure of outside air intake dampers, No credit is taken for this dual, separated outside air intake design feature for evaluating the radiological consequences of design basis accidents.

c. The outside air is normally brought in through one of two minimum outside air intakes and supplied to the operating return air fan suction. These two intakes are physically separated by over 375 feet, and are called "minimum" outside air intakes since one of them is used to supply the minimum required makeup air during normal and abnormal conditions. Wall openings for these intake ducts are missile protected. A third missile-protected air intake is provided to introduce 100% outside air to permit purging of the control room, if required. The minimum quantity of outside air required to provide makeup air for expected leakages (Table 6.4-1) and locker room exhaust fan operation and still maintain not less than 0.125 inch H_20 positive pressure in the control room with respect to the adjacent areas is introduced under all station operating conditions. This positive pressure is not maintained during running of the chlorine mode during which all intakes are closed or when the system is in the maximum outside air purge mode. The control room makeup air maintains control room pressure at a positive 0.125 inch H_2O or greater relative to the adjacent areas. During normal operating conditions, the makeup flow rate (up to approximately 4000 cfm design, 1000 cfm of which is exhausted to atmosphere by the locker room exhaust fan) will vary as doors are opened and closed. Maintaining positive pressure during normal mode is not a safety function.

d. The two minimum outside air intakes are located on the east and west sides of the plant respectively such that advantages can be derived from the outside wind direction in the event contaminants are present in one outside air intake.

e. Chilled water is supplied to the cooling coils in each air-handling equipment train from one of two separate water circuits from the control room chilled water system which is discussed in Subsection 9.2.8.1. Each is comprised of one 100% capacity refrigeration unit, one 100% capacity chilled water pump, associated piping, and specialties. Condensers are cooled by plant service water or shutdown service water as discussed in Subsections 9.2.1.1 and 9.2.1.2.

f. All 100% redundant equipment is physically separated by missile walls.

g. High radiation measured at either minimum outside air intake automatically closes the normal intake damper (by sending closure signals to dampers 0VC03YA, 0VC115YA, 0VC03YB, and 0VC115YB) and initiates operation of one of two 100% standby makeup air filter trains. This depends upon which HVAC system is operating. This in turn sends a signal to open the appropriate makeup filter train inlet and outlet dampers (inlet: 0VC02YA or 0VC02YB; outlet: 0VC06YA or 0VC06YB). The recirculation air filter trains are automatically placed in service upon receiving the same high radiation signal. This is accomplished by sending closure signals to dampers 0VC10YA and 0VC10YB, and by sending a signal to open the appropriate recirculation filter train inlet and outlet dampers (inlet: 0VC09YA or 0VC09YB; outlet: 0VC11YA or 0VC11YB). For the removal of radioactive contaminants, minimum outside air is thus introduced through demister, electric heater, medium filter, HEPA filter, iodine adsorbing beds, and downstream HEPA filter. The prefilter limits large particulate loading of the HEPA filter, and the single-stage electric heater assures no higher than 70% relative humidity air entering the charcoal. The makeup air filter trains are capable of removing 99.95% of all particulate matter larger than 0.3 microns and no less than 99% of all forms of iodine. The recirculation

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protection against radiation and toxic gases. The system is provided with redundant equipment to meet the single failure criteria. The power for the redundant equipment is supplied from separate essential power sources and is therefore operable during loss-of-offsite power. The power supply, control, and instrumentation meets the criteria of IEEE 279, IEEE 308, and IEEE 323. All of the HVAC equipment and surrounding structure is designed for Seismic Category I.

All control equipment in the control room are rated for continuous operation at 86°F and 104°F temperatures, nevertheless, the control room ambient temperature is maintained at less than 86° F by the control room HVAC system. Total loss of control room HVAC system is anticipated during a Station Blockout event and is expected to raise the control room temperature to higher than 104° F but less than 120° F. This transient condition is not considered to affect the integrity of the control equipment.

A failure analysis is presented in Table 9.4-2.

b. An equipment fire in the control room will not cause the abandonment of the control room and will not prevent a safe shutdown of the station because early ionization detection is assured, fire fighting apparatus is available, and filtration and purging capability are provided (see Subsection 9.5.1).

c. In the event of smoke or products of combustion in the control room return air, ionization detectors will annunciate in the main control room and air will be automatically routed through the recirculation filter units.

d. A radiation monitoring system is provided to detect high radiation at the two minimum outside air intakes and in the control room area. These monitors alarm in the control room upon detection of high radiation conditions. One of the two full-capacity standby makeup air filter trains and one of the two recirculation air filters designed to remove particulates and absorb iodine from the minimum quantity of outside air, are valved-in and the locker room exhaust fan is tripped upon receiving either of the following signals:

1. high radiation at either of two minimum outside air intake louvers, or

2. manual handswitch operation.

When one of these signals is initiated, the makeup train associated with the operating HVAC train automatically initiates. The normal air intake dampers 0VC03YA(B) and 0VC115YA(B) are given closure signals (as well as the maximum outside air purge dampers), the makeup air filter train isolation dampers 0VC02YA(B) and 0VC06YA(B) are given open signals and the makeup air fan starts. The locker room exhaust fan is also tripped. Provision is made for the operator to select either one of the two minimum outside air intakes for processing through the active makeup filter train. Additionally, there is a normally bypassed recirculation charcoal filter train through which all of the control room supply air is filtered upon receipt of a high radiation signal.

No credit is taken for this dual, separated outside air intake design feature for evaluating the radiological consequences of design basis accidents.

No credit is taken for this dual, separated outside air intake design feature for evaluating the radiological consequences of design basis accidents.

e. The makeup filter trains and the recirculation filter, in conjunction with other plant features, are designed and maintained to limit the occupational dose below levels required by Criterion 19 of 10 CFR 50, Appendix A.

f. The introduction of the minimum quantity of outside air maintains the control room, and other areas served by the control room HVAC system, at a positive pressure (with respect to adjacent areas) during all plant operating conditions and therefore precludes infiltration of unfiltered air into the control room. The only exceptions to this are during chlorine mode operation when no makeup air is used and when the system is in the maximum outside air purge mode. Maintaining positive pressure during normal mode is not a safety function.

g. The physical separation of two minimum outside air intakes (i.e., on the east and west sides of the plant) allows the operator to choose the intake with the lowest radiation level.

h. The control room HVAC system operates year-round on a minimum outdoor air cycle, that is, only a maximum of 4000 ft^3/min of air (1000 cfm of which is exhausted to the atmosphere by the locker room exhaust fan) is introduced from outdoors during normal operation. During radiation mode operation, locker room exhaust fan is tripped and makeup air is reduced to 3000 ft^3/min. In the chlorine mode, dampers 0VC01YA, 0VC01YB, 0VC02YA, 0VC02YB, 0VC03YA, 0VC03YB, 0VC115YA, and 0VC115YB (shown in Drawing M05-1102) will be closed to prohibit outside air makeup and the locker room exhaust fan will be tripped.

   If the system is in the maximum outside air purge mode, the purge air dampers will also close. The opposed blade isolation dampers listed above are supplied with tight blade seals and were factory tested to verify acceptable leakage rates. Any damper leakage which would occur passes through the recirculation charcoal adsorber after an accident, therefore there is no deleterious effect on the control room habitability.

   Dampers 0VC01YA, 0VC03YA, 0VC03YB, and 0VC01YB are able to close in 2 seconds after receiving an isolation signal.

i. Each of the two gas operated control room station blackout cooling fans are 100% capacity designed to exhaust a minimum of 5000 CFM. The fans are portable and are stored in the Turbine Bldg. at EL. 800'-0".

9.4.1.4 Testing and Inspection

The following paragraph is considered historical:

All equipment is factory inspected and tested in accordance with the applicable equipment specifications, quality assurance requirements, and codes. System ductwork and erection of equipment is inspected during various construction stages to assure compliance with all applicable standards. Construction tests are performed on all mechanical components and the system will be balanced for the design air and water flows and system operating pressures. Controls, interlocks, and safety devices on each system are cold checked, adjusted, and tested to ensure the proper sequence of operation. A final integrated preoperational test was conducted with all equipment and controls operational to verify the system performance. A

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Of the activity reaching the condenser, 100% of the noble gases and 10% of the iodines (due to partitioning and plateout) remain airborne. The activity airborne in the condenser is assumed to leak directly to the environment a rate of 1.0% per day, with no credit for the Standby Gas Treatment System.

The activity airborne in the condenser is presented in Table 15.4.9-6. The cumulative release of activity to the environment is presented in Table 15.4.9-7.

15.4.9.5.1.3 Results

The calculated exposures from the design basis analysis are presented in Table 15.4.9-8 and are well within the guidelines of 10 CFR 50.67 and Regulatory Guide 1.183.

15.4.9.5.1.4 Main Control Room

The radiological dose for this accident with no credit for Control Room filtration, over a 30-day period after a CRDA is ~~0.438~~ rem TEDE.

0.480

TABLE 15.4.9-8
CONTROL ROD DROP ACCIDENT (DESIGN BASIS ANALYSIS)
RADIOLOGICAL EFFECTS*

	TEDE (REM)
Exclusion Area Boundry (975 Meters)	~~0.041~~ 0.054
Low Population Zone (4018 Meters)	~~0.016~~ 0.020
Control Room	~~0.438~~ 0.480

* For licensed power level. See Appendix 15D, Reload Analysis, for current cycle analysis.

15.6.5.5.1.1 <u>Fission Product Release from Fuel</u>

It is assumed that 100% of the noble gases, 30% of the iodines and other halogens, and smaller fractions of other core isotopes as specified in Table 1 of RG 1.183 are released from an equilibrium core operating at a power level of 3543 MWt prior to the accident. While not specifically stated in RG 1.183, this assumed release implies fuel damage applicable to melt conditions. Even though this condition is inconsistent with operation of the ECCS (see Section 6.3), it is conservatively assumed applicable for the evaluation of this accident. Of this release, 100% of the noble gases become airborne. Due in part to mixing into the suppression pool of the Standby Liquid Control System solution injected into the core, the suppression pool pH is controlled at values above 7 following the core release period. Therefore as per RG 1.183 the chemical form of the iodine released to the containment is assumed to be 95% particulate (aerosol), 4.85% elemental, and 0.15% organic. Before leaving containment, natural deposition of aerosols is assumed, reducing the aerosol availability for airborne release to the environment.

15.6.5.5.1.2 <u>Fission Product Transport to the Environment</u>

The transport pathway consists of leakage from the containment to the secondary containment*-like structures by several different mechanisms and discharge to the environment through the Standby Gas Treatment System (SGTS)

* The secondary containment herein after referred to as "the gas control boundary".

(1) Containment leakage.

The design basis leak rate of the primary containment and its penetrations (excluding the main steam lines feedwater lines, and purge penetrations) is 0.65% per day for the first 24 hours and ~~0.413%~~ 0.403% per day for the duration of the accident. Of this leakage, 92% is to the secondary containment and from there to the environment via a 99% SGTS.

(2) Leakage from the Main Steam Isolation Valves (MSIVs) to the SGTS. It is assumed the MSIVs leak 100 SCFH per valve, 200 SCFH total for the first 24 hours, and ~~63.6~~ 62 SCFH for the duration of the accident. The airborne fission products are assumed to be instantaneously uniformly mixed in the drywell and containment net free volume.

(3) 100% of the containment leakage during the first 19 minutes and 8% of the containment leakage after this time bypasses containment.

(4) Containment atmosphere leakage from the feedwater penetrations is released to the environment at a leak rate equivalent to ~~10.98~~ 8.64 cfm for the first (1) hour. After 1 hour, suppression pool water supplied by the RHR to the feedwater leakage control system (FWLC) leaks at ~~2.0~~ 1.5 gpm outside of secondary containment and iodine is released to the environment for the duration of the accident period.

(5) Containment atmosphere leakage from the purge penetrations is released to the environment at a rate of ~~0.3386~~ 0.3381 cfm for the first 24 hours, and ~~0.2153~~ 0.2096 cfm for the duration of the accident.

Figure 15.6.5-1 provides schematic of LOCA transport pathways.

(6) Credit is taken for mixing primary containment leakage in 50% of the secondary containment volume in accordance with RG 1.183, Appendix A, Section 4.4.

15.6.5.5.2 Control Room

A radiological analysis has been performed (Reference 1) to determine if the ventilation system satisfies the radiation protection guidelines of the NRC Standard Review Plan 6.4 (Reference 4) and, for AST, 10 CFR 50.67 and Regulatory Guide 1.183. The results of the analysis shown below are within these guidelines. A schematic of the control room intake vents is shown in Drawing M01-1115.

The dose received during the 30-day period after a Loss-of-Coolant Accident is:

	DOSE (REM)	NRC LIMIT (REM)
TEDE	~~4.84~~ 4.89	5

* Includes the contribution of direct radiation from external sources.

A list of assumptions and input data follow: The assumptions and inputs listed are nominal base values. Different levels of conservatism may be used in radiological analysis performed to support plant operation provided that USAR dose values are not exceeded.

(1) Source Terms

The source terms used in this analysis are consistent with the guidance found in R.G, 1.183.

(2) Leakage parameters

Primary containment leak rate (%/day)	0.65
Feedwater Penetration Leak Rate (gpm)	~~2.00~~ 1.5
Bypass leak rate (% of containment leak rate) (Exfiltration assumed for first 19 min.)	8
MSIV leak rate (SCFH/line, Total)	100, 200 0-24 hours ~~63.6,~~ 62.0 127 24-720 hours

(3)* Ventilation parameters

Intake flow rate (cfm) (filtered)	3,000 ± 10%
Intake filter efficiency for iodines (%)	99
Recirculation flow rate (cfm) (filtered)	61,000 ± 10%
Total in-leakage (cfm)	~~1100~~ 1000
Recirculation filter efficiency for iodines (%)	70
Control room free volume (ft^3) includes old Technical Support Center)	324,000

* The calculated post-LOCA control room doses account for the most conservative single failure of the ventilation system.

(4) Meteorological Data

Clinton site data from 2000 through 2002 and methodology of Reference 4 were employed for the dose calculations. The following χ/Q values were calculated and used in the control room dose assessment:

INTERVAL (hrs.)	χ/Q VALUES (sec/m^3)	Unfiltered χ/Q (Values (sec/m^3)
2	~~2.36×10^{-4}~~ 9.45×10^{-4}	1.54×10^{-3}
6	7.58×10^{-4} ~~1.77×10^{-4}~~	1.09×10^{-3}
16	~~7.33×10^{-5}~~ 3.28×10^{-4}	4.67×10^{-4}
72	2.61×10^{-4} ~~5.33×10^{-5}~~	3.21×10^{-4}
624	~~4.48×10^{-5}~~ 1.85×10^{-4}	2.64×10^{-4}

~~χ/Q values include credit of a factor of 4 reduction for the effects of Dual Separated Air Intakes, in accordance with Reference 4.~~

15.6.5.6 References

1. CPS Calculation C-20.

2. Deleted

3. Deleted

4. USNRC Standard Review Plan, NUREG-0800 Section 6.4, Rev. 2, July 1981.

5. Deleted

A. Regulatory Guide 1.183

GENERAL COMPLIANCE OR ALTERNATE APPROACH ASSESSMENT:

For commitment, revision number, and scope, see Section 1.8.

This guide provides assumptions acceptable to the NRC that may be utilized in evaluating the radiological consequences of a loss-of-coolant accidents using AST.

The key implementation assumptions used are as follows:

(1) Containment Leak Rate 0.65% per day.

(2) SGTS Filter Efficiency 99% for all iodine forms.

(3) Leakage to Unprocessed Area is 8% of the containment leak rate.

(4) Leakage to Processed and Mixed Area is 92% of the containment leak rate.

Doses evaluated are within regulatory limits specified.

15.6.6 Feedwater Line Break-Outside Containment

In order to evaluate large liquid process line pipe breaks outside containment, the failure of a feedwater line is assumed to evaluate the response of the plant design to this postulated event. The postulated break of the feedwater line, representing the largest liquid line outside the containment, provides the envelope evaluation relative to this type of occurrence. The break is assumed to be instantaneous, circumferential, and downstream of the outermost isolation valve.

A more limiting event from a core performance evaluation standpoint (Feedwater Line Break Inside Containment) has been quantitatively analyzed in Section 6.3, "Emergency Core Cooling

(5) Credit is taken for mixing of primary to secondary containment leakage in 50% of the secondary containment volume

TABLE 15.6.5-6
LOSS-OF-COOLANT ACCIDENT (DESIGN BASIS ANALYSIS)
RADIOLOGICAL EFFECTS*

	TEDE
EXCLUSION AREA BOUNDARY (975 Meters)	~~17.31~~ 16.18
LOW POPULATION ZONE (4018 Meters)	~~7.42~~ 6.81

* Included doses from Bypass leakage, Feedwater Penetration leakage, MSIV leakage and Purge Penetration leakage.

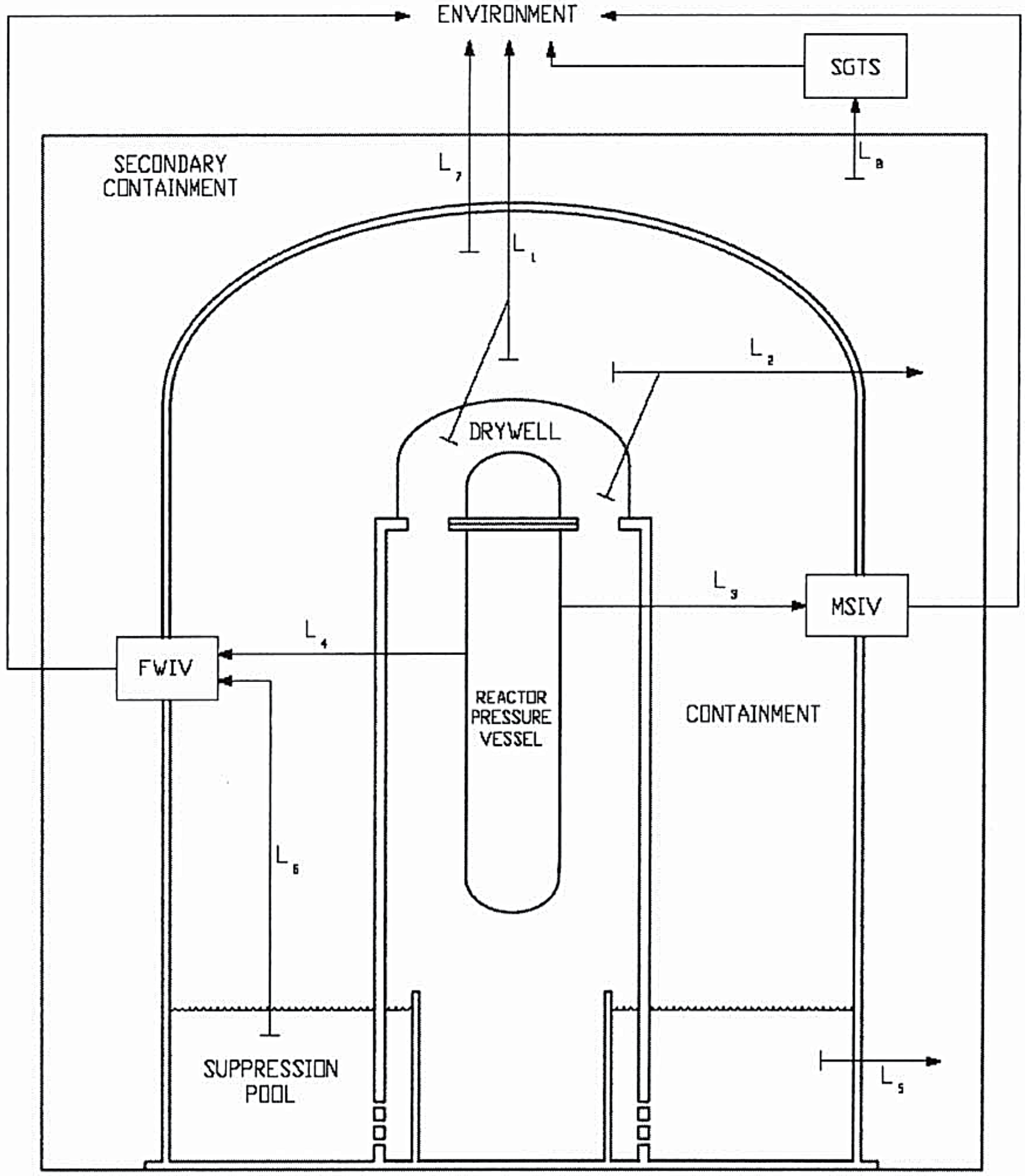

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FIGURE 15.6.5-1
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Schematic of LOCA Transport Pathways

Leakage Rates and Secondary Containment Mixing Parameters		
Path	**Description**	**Parameters & Values**
L_1	Primary Containment Leakage Bypassing Secondary Containment to the Environment	Leak Rate: $0.08*L_a$ = 0.052%/day from 0 to 24 hours = 0.033%/day from 1 to 30 days
L_2	Primary Containment Leakage to Secondary Containment	Leak Rate: $0.92*L_a$ = 0.598%/day from 0 to 19 min Unfiltered during drawdown period and includes the two minute gap release time. $0.92*L_a$ = 0.598%/day from 19 min to 24 hrs SGTS filtered = 0.380%/day from 1 to 30 days SGTS filtered
L_3	MSIV Leakage to Environment	Leak Rate: 200 scfh for all main steam lines, 100 scfh for maximum for any one MS line; reduced to ~~63.6~~ 62.0% of these rates after 1 day
L_4	FWIV Containment Air Leakage to Environment	Leak Rate: ~~10.98~~ 8.64 cfm total, for the one hour before FWIV LCS fills the lines
L_5	ECCS Leakage to Secondary Containment	Leak Rate: 5 gpm from 0 to 30 days
L_6	FWIV LCS Leakage of ECCS Liquid to the Environment	Leak Rate: ~~2~~ 1.5 gpm from 0 to 1 days 1 gpm from 1 to 30 days [Conservatively includes fill time]
L_7	Purge Penetrations 101 and 102 Leakage to the Environment	Leak Rate (for each of two penetrations): $0.02*L_a$ = 0.013%/day from 0 to 1 day = 0.0083%/day from 1 to 30 days
L_8	Release of Secondary Containment Atmosphere through SGTS to the Environment	~~No Secondary Containment mixing credit~~ Mixing in 50% of the Secondary Containment volume is credited. Modeled as: Volume = ~~1~~ 8.50E+05 cu.ft. Outflow = ~~1000~~ 4400 cfm

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FIGURE 15.6.5-1
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Schematic of LOCA Transport Pathways

The RADTRAD computer code developed for and endorsed by the NRC for AST analyses was used in the calculations. The RADTRAD program is a radiological consequence analysis code used to estimate post-accident doses at plant offsite locations and in the control room. The AST FHA analyses take no credit for SGTS operation, secondary containment isolation, or control room air intake or recirculation filtration for the full duration of the accident event.

The fission product inventories for the damaged fuel in the RADTRAD analysis were determined based on the licensed core power level following extended power uprate (EPU) of 3473 megawatts thermal (MWt) and further adjusted to 102% (~~3542~~ 3543 MWt).

The atmospheric dispersion factors (X/Q) utilized are as found in Subsections 2.3.4 and 15.6.5.5. Assumptions used in the AST analysis for the FHA and the results are summarized below.

This design basis AST FHA justifies the changes to Technical Specification (TS) requirements for the operability of certain Engineered Safety Feature (ESF) systems while fuel is being handled and during core alterations. In the most conservative FHA scenario, an irradiated fuel assembly is postulated to drop 34 feet onto the reactor core and cause the release of fission products from inside the fuel cladding that fails because of the impact. This scenario results in the largest number of fuel rod cladding failures and occurs in the primary containment building. This is because the fuel bundle drop in containment is from 34 feet and the corresponding FHA in the fuel building is from six feet.

This bounding FHA was analyzed based on the AST provision for the condition when primary containment, secondary containment, and Standby Gas Treatment Systems (SGTS) are inoperable. This analysis demonstrates that the offsite and control room dose limits as specified in 10CFR50.67 would not be exceeded for irradiated fuel that had been allowed to decay for 24 hours. The 24 hour decay time is based on the minimum requirement for reactor sub-criticality prior to fuel movement as defined in the Operational Requirements Manual (ORM). By demonstrating that the dose limits will be met after a 24 hour decay period, fuel assemblies can be handled without primary and secondary containment systems operable and without the SGTS operable. No credit is taken for administrative actions to manually close doors, hatches, and penetrations.

The key assumptions for the analysis are as follows:

1. The analysis is based on guidance in RG 1.183 and Standard Review Plan 15.0.1.
2. The damaged fuel is assumed to be decayed for 24 hours.
3. The limiting high burnup fuel type, GNF 2, that has a radial peaking factor of ~~1.7~~ 1.8, was used to derive the source term.
4. The reactor power level is consevatively rated at 102% of the licensed power level (~~3542~~ 3543 MWt) which accounts for uncertainty.
5. The ~~dual inlet~~ control room ventilation is used including the radiation monitors at the inlets consistent with the requirements of the control room ventilation system TS.

6. The existing atmospheric dispersion factors for the EAB, the LPZ, and the control room in the USAR continue to apply.

7. For this analysis, the limiting FHA occurs in the primary containment.

8. The FHA is as described in GE document NEDE-24011-P-A-14-US, dated June 2000, which results in 172 failed fuel rods that is caused by a 34 foot drop of a GE 14 fuel bundle that has a 10 x 10 array of fuel onto the reactor core. The NF-500 fuel handling mast is assumed to fall onto the core along with the fuel bundle.

9. Although the depth of the water in the containment assumed for this analysis is over 34 feet, the decontamination factors are conservatively based on a 23 foot depth of water. The TS minimum required water level above the top of the reactor pressure vessel flange is 22 feet and 8 inches.

10. The offsite dose limit is 6.3 Rem Total Effective Dose Equivalent (TEDE) and the control room dose limit is 5 Rem TEDE for the worst case two hour release period.

11. Primary and secondary containment isolations were not credited.

12. The SGTS filtration was not credited.

13. The release of radioactive material was assumed to be directly into the wake of the containment and the ventilation flow into the control room enters through one of the two normal inlets that have radiation monitoring systems.

design basis

14. For the ~~limiting~~ scenario, filtration for the control room ventilation intake flow and recirculation flow is ~~inoperable.~~

credited 20 minutes after the initiation of the event.

15. The calculated EAB dose bounds the LPZ dose. Because the dose limits are equal, EAB dose was used in the analysis.

16. No credit is taken for atmospheric dilution or mixing in the containment.

A summary of the analysis assumptions is provided in Table 15.7.4-1.

15.7.4.3 Radiological Consequences

The resulting dose consequences as calculated using AST releases and the other AST assumptions as described above for offsite and control room doses are provided in Table 15.7.4-4. As indicated, all results are within the AST acceptance criteria.

The results indicate that the calculated consequences of a design basis FHA at or after 24 hours of shutdown will be within regulatory limits without the requirement of containment closure, the need to have SGTS filtration, or the control room emergency ventilation system operational during fuel movement.

15.7.4.4 References

1. Calculation C-022, “Site Boundary and Control Room Dose following a FHA in Containment using Alternative Source Term.

TABLE 15.7.4-1
<u>FUEL HANDLING ACCIDENT PARAMETERS TABULATED FOR POSTULATED ACCIDENT ANALYSIS</u>

I.	Data and assumptions used to estimate radioactive source from postulated accidents			
	A.	Power Level		~~3542~~ 3543 MWt
	B.	Radial peaking factor		~~1.7~~ 1.8
	C.	Fuel Damaged (in a 10 X 10 array)		172 rods
	D.	Release of Activity from fuel by Nuclide		
		(1)	I-131	8%
		(2)	Other Halogens	5%
		(3)	Kr-85	10%
		(4)	Other Noble Gases	5%
	E.	Iodine fractions from fuel		
		(1)	Organic	0.15%
		(2)	Elemental	4.85%
		(3)	Particulate	95%
II.	Data and assumptions used to estimate activity released			
	A.	Release period		2 hours
	B.	Overall pool decontamination factor for iodine isotopes		200
	C.	Overall pool decontamination factor For noble gases		1
	D.	Filtration (SGTS ~~and Control Room~~) efficiencies (%)		0
	E.	All other pertinent data and assumptions		None
III	Dispersion Data			
	A.	Boundary and LPZ distances (m)		975 / 4018
	B.	X/Q for time intervals of		
		(1)	0-2 hr - EAB/LPZ	~~1.8E-4 / 4.2E-5~~ 2.46E-4/5.62E-5
		(2)	2-8 hr - LPZ	1.3E-5
		(3)	8-24 hr - LPZ	8.2E-6
		(4)	1-4 days - LPZ	3.3E-6
		(5)	4-30 days - LPZ	1.6E-6

TABLE 15.7.4-4
FUEL HANDLING ACCIDENT RADIOLOGICAL EFFECTS

Location and Time	Dose (REM TEDE)	Reg. Limit (REM TEDE)
EAB 0 – 2 hours	~~0.48~~ 0.72	6.3
LPZ 0 – 2 hours	~~0.11~~ 0.17	6.3
Control Room 30 days (worst case)	~~0.8~~ 1.18	5.0

SURVEILLANCE REQUIREMENTS (continued)

SR 3.6.1.3.11

This SR ensures that the combined leakage rate of the primary containment feedwater penetrations is less than the specified leakage rate. The leakage rate is based on water as the test medium since these penetrations are designed to be sealed by the FWLCS. The ~~2~~ 1.5 gpm leakage limit has been shown by testing and analysis to bound the condition following a DBA LOCA where, for a limited time, both air and water are postulated to leak through this pathway. The leakage rate of each primary containment feedwater penetration is assumed to be the maximum pathway leakage, i.e., the leakage through the worst of the two isolation valves [either 1B21-F032A(B) or 1B21-F065A(B)] in each penetration. This provides assurance that the assumptions in the radiological evaluations of References 1 and 2 are met (Ref. 15).

Dose associated with leakage (both air and water) through the primary containment feedwater penetrations is considered to be in addition to the dose associated with all other secondary containment bypass leakage paths.

The Frequency is in accordance with the Primary Containment Leakage Rate Testing Program.

A Note is added to this SR which states that the primary containment feedwater penetrations are only required to meet this leakage limit in Modes 1, 2, and 3. In other conditions, the Reactor Coolant System is not pressurized and specific primary containment leakage limits are not required.

SR 3.6.1.3.12

This SR requires a demonstration that each instrumentation line excess flow check valve (EFCV) which communicates to the reactor coolant pressure boundary (Ref. 16) is OPERABLE by verifying that the valve activates within the required flow range. For instrument lines connected to reactor coolant pressure boundary, the EFCVs serve as an additional flow restrictor to the orifices that are installed inside the drywell (Ref. 14). The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

(continued)

Physical Input Parameters and Assumptions Post-LOCA Dose Calculations

The following information provides the physical input parameters and assumptions that were used for the revised post-loss of coolant accident (LOCA) dose calculations (i.e., 12-month operating cycle with 19-minute drawdown time), as well as the corresponding parameters used in both the original calculations (i.e., in support of Amendment 167) and the 12-month operating cycle calculations. The core inventory utilized in the 12 to 24-month operating cycle sensitivity study is also provided.

Table 1 provides the core radionuclide inventories used in the current post-LOCA dose calculation analysis, the revised inventories used for the current 12-month operating cycle, and the Beginning of Cycle (BOC) core inventories used in the 12 to 24-month operating cycle sensitivity study. The core source terms for the 12-month operating cycle calculations were used in the revised post-LOCA dose calculations supporting this license amendment request. The 12 to 24-month fuel cycle sensitivity study demonstrated that the dose consequences associated with an annual cycle are bounding.

Table 2 provides the general parameters that are applicable to post-accident alternative source term (AST) dose calculations for multiple accidents, and Table 3 provides the parameters and data that are applicable to the post-LOCA dose calculation.

All source terms were based on the extended power uprate (EPU) power level of 3,473 Megawatts-thermal (MW_t). Analyses use 3,543 MW_t (i.e., 102% of 3,473 MW_t) in accordance with Regulatory Guide 1.49, "Power Levels of Nuclear Power Plants," Revision 1. The revised core inventories were calculated using the ORIGEN-ARP computer code in accordance with the guidance in RG 1.183. The differences in the core source term are due mainly to differences in fuel burnup.

Table 1: Core Radionuclide Inventories

Isotope	Current Analysis of Record - (Annual Fuel Cycle) - Ci/MW_t	Revised Annual Fuel Cycle Core Inventory - Ci/MW_t	12 to 24-Month Fuel Cycle - BOC Core Inventory - Ci/MW_t
CO-58	2.59E+02	2.59E+02	2.59E+02
CO-60	4.45E+02	4.79E+02	4.79E+02
KR-85	3.32E+02	4.94E+02	1.30E+02
KR-85M	6.80E+03	8.18E+03	9.12E+03
KR-87	1.30E+04	1.64E+04	1.85E+04
KR-88	1.82E+04	2.21E+04	2.50E+04
RB-86	6.66E+01	8.63E+01	1.47E+01
SR-89	2.44E+04	3.07E+04	3.43E+04
SR-90	2.62E+03	3.98E+03	9.89E+02
SR-91	3.09E+04	3.81E+04	4.24E+04
SR-92	3.37E+04	4.00E+04	4.39E+04
Y-90	2.81E+03	4.16E+03	1.01E+03
Y-91	3.17E+04	3.93E+04	4.29E+04
Y-92	3.39E+04	4.04E+04	4.45E+04
Y-93	3.95E+04	4.46E+04	4.81E+04
ZR-95	4.74E+04	5.15E+04	5.29E+04

Table 1: Core Radionuclide Inventories

Isotope	Current Analysis of Record - (Annual Fuel Cycle) - Ci/MW$_t$	Revised Annual Fuel Cycle Core Inventory - Ci/MW$_t$	12 to 24-Month Fuel Cycle - BOC Core Inventory - Ci/MW$_t$
ZR-97	4.97E+04	5.18E+04	5.30E+04
NB-95	4.76E+04	5.18E+04	5.16E+04
MO-99	5.13E+04	5.13E+04	5.15E+04
TC-99M	4.47E+04	4.65E+04	4.53E+04
RU-103	4.30E+04	4.66E+04	3.58E+04
RU-105	3.01E+04	3.46E+04	2.16E+04
RU-106	1.68E+04	2.10E+04	5.07E+03
RH-105	2.82E+04	3.17E+04	2.01E+04
SB-127	3.02E+03	2.67E+03	2.09E+03
SB-129	8.89E+03	8.19E+03	6.73E+03
TE-127	3.00E+03	2.63E+03	2.00E+03
TE-127M	4.06E+02	4.50E+02	2.88E+02
TE-129	8.75E+03	7.67E+03	6.28E+03
TE-129M	1.31E+03	1.48E+03	1.18E+03
TE-131M	3.95E+03	5.54E+03	4.70E+03
TE-132	3.85E+04	3.88E+04	3.82E+04
I-131	2.71E+04	2.74E+04	2.63E+04
I-132	3.91E+04	3.99E+04	3.89E+04
I-133	5.50E+04	5.59E+04	5.65E+04
I-134	6.03E+04	6.39E+04	6.47E+04
I-135	5.15E+04	5.33E+04	5.33E+04
XE-133	5.30E+04	5.59E+04	5.64E+04
XE-135	1.77E+04	1.95E+04	2.23E+04
CS-134	6.07E+03	8.62E+03	6.10E+02
CS-136	2.01E+03	2.46E+03	5.34E+02
CS-137	3.66E+03	5.61E+03	1.11E+03
BA-139	4.88E+04	5.09E+04	5.20E+04
BA-140	4.72E+04	4.90E+04	5.07E+04
LA-140	5.01E+04	4.97E+04	5.13E+04
LA-141	4.45E+04	4.64E+04	4.75E+04
LA-142	4.29E+04	4.52E+04	4.67E+04
CE-141	4.47E+04	4.64E+04	4.78E+04
CE-143	4.10E+04	4.44E+04	4.66E+04
CE-144	3.61E+04	3.68E+04	2.47E+04
PR-143	3.97E+04	4.36E+04	4.55E+04
ND-147	1.80E+04	1.81E+04	1.86E+04
NP-239	5.91E+05	5.90E+05	4.98E+05
PU-238	1.09E+02	1.88E+02	5.50E+00
PU-239	1.19E+01	1.07E+01	7.50E+00
PU-240	1.60E+01	2.15E+01	4.52E+00
PU-241	5.05E+03	5.13E+03	9.65E+02
AM-241	5.87E+00	6.82E+00	3.61E-01
CM-242	1.53E+03	2.38E+03	3.79E+01
CM-244	9.55E+01	2.68E+02	3.34E-01

Table 2: General Parameters and Data Applicable to Post-LOCA Dose Calculation

General AST Analysis Design Inputs for CPS	Current Design Basis	Revised Design Basis
Core Power Level	3543 MWth	3543 MWth
Core Source Terms	34.3 GWD/MTU annual cycle for the 60 isotopes forming the standard RADTRAD library	Bounding source term for burnups of 10 to 53 GWD/MTU for the 60 isotopes forming the standard RADTRAD library
Dose Conversion Factors Inhalation and Whole Body Dose	FGR 11 and 12 for inhalation CEDE and cloud submersion EDE. Values are built into RADTRAD file FGR11&12.INP for a total of 60 isotopes.	FGR 11 and 12 for inhalation CEDE and cloud submersion EDE. Values are built into RADTRAD file FGR11&12.INP for a total of 60 isotopes.
EAB X/Qs Distance to EAB Dispersion Factors 0-2 hr.	 975 meters 2.46E-04 (sec/m^3)	 975 meters 2.46E-04 (sec/m^3)
LPZ – X/Qs Distance to LPZ <u>Dispersion Factors</u> 0 - 8 hr. 8 - 24 hr. 24 - 96 hr. 96 - 720 hr.	 4018 meters 2.48E-05 (sec/m^3) 1.65E-05 (sec/m^3) 6.81E-06 (sec/m^3) 1.91E-06 (sec/m^3)	 4018 meters 2.48E-05 (sec/m^3) 1.65E-05 (sec/m^3) 6.81E-06 (sec/m^3) 1.91E-06 (sec/m^3)

Table 2: General Parameters and Data Applicable to Post-LOCA Dose Calculation

General AST Analysis Design Inputs for CPS	Current Design Basis	Revised Design Basis
CR - X/Qs **Filtered CR Intake:** Dispersion Factors		
0 - 2 hr.	2.36E-04 (sec/m^3)	9.45E-04 (sec/m^3)
2 - 8 hr.	1.77E-04 (sec/m^3)	7.58E-04 (sec/m^3)
8 - 24 hr.	7.33E-05 (sec/m^3)	3.28E-04 (sec/m^3)
24 - 96 hr.	5.33E-05 (sec/m^3)	2.61E-04 (sec/m^3)
96 - 720 hr.	4.48E-05 (sec/m^3)	1.85E-04 (sec/m^3)
Unfiltered CR Intake: Dispersion Factors		
0 - 2 hr.	1.54E-03 (sec/m^3)	1.54E-03 (sec/m^3)
2 - 8 hr.	1.09E-03 (sec/m^3)	1.09E-03 (sec/m^3)
8 - 24 hr.	4.67E-04 (sec/m^3)	4.67E-04 (sec/m^3)
24 - 96 hr.	3.21E-04 (sec/m^3)	3.21E-04 (sec/m^3)
96 - 720 hr.	2.64E-04 (sec/m^3)	2.64E-04 (sec/m^3)

Table 2: General Parameters and Data Applicable to Post-LOCA Dose Calculation

General AST Analysis Design Inputs for CPS	Current Design Basis	Revised Design Basis
Control Room		
Free Volume	324,000 cu. ft. (includes TSC)	324,000 cu. ft. (includes TSC)
Filtered Intake Rate - CREF Mode	3,000 cfm - 10% = 2,700 cfm	3,000 cfm - 10% = 2,700 cfm
Filtered Recirc Rate	61,000 cfm - 10% = 54,900 cfm	61,000 cfm - 10% = 54,900 cfm
Intake Filter Efficiency	99.0% for Aerosols 99.0% for Elemental and Organic Iodine	99.0% for Aerosols 99.0% for Elemental and Organic Iodine
Effective Intake Filter Penetration	1.0% for Aerosols 1.0% for Elemental and Organic Iodine	1.0% for Aerosols 1.0% for Elemental and Organic Iodine
Intake Filter Bypass Allowance	0.05% for all Iodines	0.05% for all Iodines
Recirc Filter Efficiency	70% for all Iodines	70% for all Iodines
Effective Recirc Filter Penetration	30.0% for all Iodines	30.0% for all Iodines
Recirc Filter Bypass Allowance	2.0% for all Iodines	2.0% for all Iodines
Combined CR Filter Efficiency	99.664% for Aerosols 99.664% for Elemental and Organic Iodine	99.664% for Aerosols 99.664% for Elemental and Organic Iodine
Total Filtered Inleakage	1,100 cfm	1,000 cfm
Unfiltered Inleakage	0 cfm	10 cfm
Minimum Suppression Pool volume	146,400 ft^3 (1,095,148 gal)	146,400 ft^3 (1,095,148 gal)

Table 3: Parameters and Data Applicable to Post-LOCA Dose Calculation

General AST Analysis Design Input	Current Design Basis	Revised Design Basis
SGTS Flow Rate (cfm)	4000 cfm + 10% = 4400 cfm Because credit is not taken for mixing in secondary containment, an artificially low secondary containment volume (1 ft^3 and an artificially high SGTS flow rate (10^3 cfm) are used for RADTRAD analyses	4000 cfm + 10% = 4400 cfm
SGTS Iodine Filter Efficiency	99% for Aerosols 99% for Elemental and Organic Iodine	99% for Aerosols 99% for Elemental and Organic Iodine
Credited Containment Removal Mechanisms	Credit Natural Deposition based on Powers' Algorithm built into RADTRAD. (10^{th} percentile used for conservatism)	Credit for Natural Deposition based on Powers' Algorithm built into RADTRAD. (10^{th} percentile used for conservatism)
Secondary Containment Volume	N/A	1,700,000 ft^3 (Rounded down from 1,704.995 ft^3) Credit is taken for mixing of primary containment leakage in 50% of the secondary containment volume in accordance with Regulatory Guide 1.183, Appendix A, Section 4.4. (this is a volume of 8.50E+05 ft^3)
Primary Containment Mixing	A drywell bypass leakage rate of 3,000 cfm for the first two hours is used, followed by an assumption of well-mixed drywell-containment conditions thereafter.	A drywell bypass leakage rate of 3,000 cfm for the first two hours is used, followed by an assumption of well-mixed drywell-containment conditions thereafter.

Table 3: Parameters and Data Applicable to Post-LOCA Dose Calculation

General AST Analysis Design Input	Current Design Basis	Revised Design Basis
Primary Containment Leak Rate (SGTS Filtered and Secondary Containment Bypass)	0.65% per day for first 24 hours (La); 0.413% per day thereafter	0.65% per day. Reduced to 0.403% per day after 24 hours.
Total MSIV Leak Rate Limits: Post-24 hours analytical values	200 scfh total for four lines 100 scfh for any one line 127 scfh total, 63.6 scfh for any one line	200 scfh total for four lines 100 scfh for any one line Reduced to 62.0% of these values after 24 hours in accordance RG 1.183
Purge Penetrations 101 and 102 Leakage to the Environment	**Leak Rate (for each of two penetrations):** 0.02*La = 0.013%/day from 0 to 1 day =0.0083%/day from 1 to 30 days	**Leak Rate (for each of two penetrations):** 0.02*La = 0.013%/day from 0 to 1 day =0.0083%/day from 1 to 30 days
ECCS Leakage into Secondary Containment: Leak Rate Fraction Flashed Filtered by SGTS	 5 gpm plus 2 gpm RCIC[1] backleakage 10% Yes - after drawdown	 5 gpm plus 2 gpm RCIC[1] backleakage 10% Yes - after drawdown
FWIV leak rate: Air (Containment atmosphere): Water (ECCS): FWIV Leakage Flashing Fraction:	**Total for two penetrations:** 10.98 cfm for 1 hour (before FWIV LCS fill the lines) 2 gpm (constant) from 0 to 30 days 10%	 Limit of 8.64 cfm total, (before FWIV LCS fill the lines) Limit of 1.5 gpm total 10%
Fraction of Containment Leakage that Bypasses SGTS	8% after first 19 minutes post-accident, 100% before.	8% after first 19 minutes post-accident, 100% before.

Table 3: Parameters and Data Applicable to Post-LOCA Dose Calculation

General AST Analysis Design Input	Current Design Basis	Revised Design Basis
Aerosol Natural Deposition Coefficients Used in the Containment	Power's model built into RADTRAD as natural deposition time dependent deposition lambdas.	Power's model built into RADTRAD as natural deposition time dependent deposition lambdas.
Suppression pool scrubbing	Not credited	Not credited
Main Steam Line Deposition/Plate-out (where credited)	AEB 98-03 well-mixed flow is assumed. Aerosol settling in horizontal lines only; elemental deposition in all credited lines. Aerosol Settling and Elemental Iodine Deposition in Piping are based on: • Pipe parameters such as volume, aerosol settling area, and elemental iodine deposition areas • containment leak rates as a function of leak acceptance criteria; • inboard and outboard flow rates	AEB 98-03 well-mixed flow is assumed. Aerosol settling in horizontal lines only; elemental deposition in all credited lines. Aerosol Settling and Elemental Iodine Deposition in Piping are based on: • Pipe parameters such as volume, aerosol settling area, and elemental iodine deposition areas • containment leak rates as a function of leak acceptance criteria; • inboard and outboard flow rates
Main Steam Line and Condenser Holdup Credit for MSIV Leakage	No credit is taken for holdup and plate-out downstream of seismically qualified main steam piping or in the condenser since these components have not been evaluated for seismic ruggedness.	No credit is taken for holdup and plate-out downstream of seismically qualified main steam piping or in the condenser since these components have not been evaluated for seismic ruggedness.
Containment Spray Removal Mechanism	Not credited	Not credited
Minimum Suppression Pool volume (ft^3)	146,400	146,400
Releases to Containment	No Core Activity Release for first 120 seconds. (Release Fractions and Timing per RG 1.183)	No Core Activity Release for first 120 seconds. (Release Fractions and Timing per RG 1.183)

Table 3: Parameters and Data Applicable to Post-LOCA Dose Calculation

General AST Analysis Design Input	Current Design Basis	Revised Design Basis
Containment Volume: Drywell (ft^3)	241,699	241,699
Containment (outside Drywell) (ft^3)	1,512,341	1,512,341
Primary Containment Total where applicable (ft^3)	1.754E+06	1.754E+06
Secondary Containment Drawdown Time	19 minutes (includes two-minute gap release time)	19 minutes (includes two minute gap release time)

[1] In 2007, EGC identified that the original AST dose calculations did not include potential releases from the RCIC and HPCS full flow test lines to the RCIC storage tank. EGC implemented a minor revision to add the release pathway associated with an assumed two gallons per minute backleakage into the RCIC storage tank and subsequent release to the environment through the tank vent.

To address the impact of revised X/Qs and revised source term on Chapter 15 Accident Analyses, Exelon Generation Company, LLC (EGC) performed the Clinton Power Station, Unit 1 (CPS) post-LOCA dose calculations, for the proposed change, in accordance with Regulatory Guide (RG) 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors." RG 1.183 requires a number of assumptions that result in conservative onsite and offsite dose values.

In order to provide reasonable assurance of adequate margin to minimize the impact of potential uncertainties in both the dose calculations and the input parameters, EGC evaluated the impact of additional conservative assumptions that are related to actual operating conditions or testing requirements.

The information below provides a sensitivity study of these additional conservatisms, which provides a qualitative estimate for the margin available in the current LOCA dose calculation, relative to the 10 CFR 50.67, "Accident source term," limit. That is, the conservatisms were relaxed, on an individual basis, and the calculation results were compared with the results obtained with the conservatism. All calculations were performed using the RADTRAD computer code, Version 3.03.

1. Leak-Before-Break Credit

As stated in RG 1.183 (i.e., in Section 3.3, "Timing of Release Phases"), for facilities licensed with leak-before-break (LBB) methodology, the onset of gap release may be assumed to be 10 minutes. This assumption has been used by other licensees in alternative source term (AST) calculations; however, since CPS is not licensed with an LBB methodology, EGC assumed that gap release started at two minutes, in accordance with RG 1.183.

In order to quantify the margin that this would provide, EGC revised the LOCA analysis to start the gap release at 10 minutes post-accident. In the reanalysis, decay during this 10-minute interval was ignored and the secondary containment bypass was terminated at nine minutes in the model. This is equivalent to a total positive pressure period of 19 minutes.

The results of this reanalysis indicate that, with credit for LBB, there would be approximately a 2.5% reduction in the postulated Control Room dose.

2. Containment Spray Credit

RG 1.183 allows licensees to take credit for aerosol removal by containment sprays. EGC did not credit spray removal in the CPS LOCA analysis, nor has it been modeled in the CPS LOCA analysis. As an alternative, EGC utilized a surrogate LOCA model, based on the simplified spray model provided in NUREG/CR-5966, "A Simplified Model of Aerosol Removal by Containment Sprays," to quantify the potential dose impact of containment sprays. In a Safety Evaluation dated March 30, 2015 (i.e., ADAMS Accession No. ML15075A139), the NRC approved the use of this model in a full AST license amendment for the Perry Nuclear Power Plant.

The results of the reanalysis using the surrogate LOCA model, with credit for Containment Spray, indicates that there would be approximately a 20% reduction in the postulated Control Room dose. Although the margin from this surrogate LOCA model is not directly applicable to the CPS dose calculations, it does provide additional qualitative margin to the post-LOCA Control Room dose value for CPS.

Impact of Revised X/Q and Revised Source Term on Chapter 15 Accident Analyses

The impact of the revised Control Room atmospheric dispersion (X/Q) value and the revised source term on Chapter 15 Accident Analyses is addressed in this section.

The Chapter 15 accident scenarios that were part of the full AST submittal are listed in Table 1 below. The fuel handling accident (FHA) analysis assumes operation of the Control Room intake and recirculation filtration after 20-minutes post-accident. All of the other non-LOCA AST analyses take no credit for operation of the Standby Gas Treatment System, secondary containment isolation, or Control Room air intake or recirculation filtration for the full duration of the accident event.

Table 1: AST Accident Scenarios and Impact of the Proposed Changes

AST Accident Scenario	Impacted by Control Room X/Q and/or revised source term	Justification
Fuel Handling Accident (FHA)	Yes	The FHA inside containment is limiting and assumes that containment integrity is not available or necessary. This analysis required use of a revised Control Room X/Q and is discussed further below.
Control Rod Drop Accident (CRDA)	Yes	All activity released from the fuel is transported to the turbine/condenser prior to release to the atmosphere. This analysis is not impacted by the change in Control Room X/Q. However, the calculation is impacted by the revised source term and is discussed further below.
Main Steam Line Break (MSLB) accident outside containment	No	All activity released from the fuel is transported outside the primary and secondary containment to the environment. This analysis is not impacted by the change in Control Room X/Q. Because the radionuclide release is based on the normal coolant activity, there is no impact due to the revised source term.
Suppression Pool pH	Yes	The suppression pool pH calculation is impacted by the revised source term and is discussed further below.

The remaining accidents presented in the Updated Safety Analysis Report (USAR) as listed below, are bounded by the LOCA analysis dose consequences and are not impacted by the change in Control Room X/Q or the revised source term.

15.3.3	Seizure of One Recirculation Pump
15.3.4	Recirculation Pump Shaft Break
15.6.2	Instrument Line Break
15.6.6	Feedwater Line Break
15.7.1.1	Main Condenser Gas Treatment System Failure
15.7.3	Liquid Radwaste Tank Failure
15.7.5	Cask Drop Accident
15.8	ATWS

Fuel Handling Accident (FHA)

The revised FHA calculation addresses the NRC concern that the Control Room remote intake dampers do not meet the single criteria. To address this concern, the revised analysis: 1) removes the previous credit for the dual Control Room remote air intake due to the single failure issues, 2) takes credit for the operation and filtration of the Control Room emergency ventilation system after 20 minutes of the event with an activity release from Secondary Containment that does not credit the Primary Containment or Secondary Containment operability, 3) the Control Room filtered inleakage changes from 650 cubic feet per minute (cfm) to 1,000 cfm, 4) radial peaking factor changes from 1.7 to 1.8 (i.e., to provide margin to the cycle to cycle variation in core design), 5) updates the radionuclide inventory for an annual fuel cycle, and 6) incorporates a sensitivity study for a bounding 12 to 24-month fuel cycle core inventory. These changes are not initiators for an accident previously evaluated in the USAR. Thus, the inclusion of these changes in the revised calculation does not result in more than a minimal increase in the frequency of occurrence of an accident previously evaluated in the USAR.

These changes in the revised analysis result in an increase in the dose consequences for the exclusion area boundary (EAB), the low population zone (LPZ) and the Control Room. The increase in dose consequences for the EAB, the LPZ, and the Control Room is less than 10% of the difference between the current calculated values and the regulatory guideline values and the increased doses do not exceed the current SRP guideline value.

For the EAB, the current calculated value is 0.48 roentgen equivalent man (REM) Total Effective Dose Equivalent (TEDE) and the 10 CFR 50.67 value is 25 REM TEDE. Ten percent (i.e., 10%) of 25-0.48 is 2.452 REM TEDE. The new calculated EAB value is 0.7243 REM TEDE based on the 12-month fuel cycle core inventory which bounds the dose consequences based on the 12 to 24-month fuel cycle core inventory. The difference between the new EAB dose and the

current EAB dose is 0.7243-0.48 = 0.2443 REM TEDE. Because the dose increase (i.e., 0.2443 REM TEDE) is less than 10% of the difference between the current value and the regulatory value (i.e., 2.452 REM TEDE) and remains below the current Standard Review Plan (SRP) guideline value of 6.3 REM TEDE, the increase in EAB dose is not considered a more than a minimal increase in the consequences of an accident evaluated in the USAR.

For the LPZ, the current calculated value is 0.11 TEDE and the 10 CFR 50.67 value is 25 REM TEDE. Ten percent (10%) of 25-0.11 is 2.489 REM TEDE. The new calculated LPZ value is 0.1655 REM TEDE based on the 12-month fuel cycle core inventory which bounds the dose consequences based on the 12 to 24-month fuel cycle core inventory. The difference between the new LPZ dose and the current LPZ dose is 0.1655-0.11 = 0.0555 REM TEDE. Because the dose increase (i.e., 0.0555 REM TEDE) is less than 10% of the difference between the current value and the regulatory value (2.489 REM TEDE) and remains below the current SRP guideline value of 6.3 REM TEDE, the increase in LPZ dose is not considered a more than a minimal increase in the consequences of an accident evaluated in the USAR.

For the Control Room, the current calculated value is 0.8 TEDE and the 10 CFR 50.67 value is 5 REM TEDE. Ten percent (10%) of 5-0.8 is 0.420 REM TEDE. The new calculated Control Room value is 1.1826 REM TEDE based on the 12-month fuel cycle core inventory which bounds the dose consequences based on the 12 to 24-month fuel cycle core inventory. The difference between the new Control Room dose and the current Control Room dose is 1.1826-0.8 = 0.3826 REM TEDE. Because the dose increase (i.e., 0.3826 REM TEDE) is less than 10% of the difference between the current value and the regulatory value (0.420) and remains below the current SRP guideline value of 5 REM TEDE, the increase in Control Room dose is not considered a more than a minimal increase in the consequences of an accident evaluated in the USAR.

Table 2: Updated FHA Doses Compared to Minimal Increase Dose Margins and SRP Limits (REM TEDE)

	Current Total Dose A	**Updated Total Dose B**	**10 CFR 50.67 Dose Limits C**	**Dose Increase D=B-A**	**Minimal Dose Increase E= 0.1 (C-A)**	**SRP Dose Limit F**
Control Room	0.8	1.1826	5	0.3826	0.42	5
Exclusion Area Boundary	0.48	0.7243	25	0.2443	2.452	6.3
Low Population Zone	0.11	0.1655	25	0.0555	2.489	6.3

Control Rod Drop Accident (CRDA)

The revised CRDA calculation documents the changes in the input to the CRDA analysis due to the following changes: 1) incorporation of an updated radionuclide inventory, 2) updating the radial peaking factor from 1.7 to 1.8, 3) increasing the normal intake flow rate into the Control

Room to 5,410 cfm (i.e., normal Control Room flow of 4,000 cfm plus 10% (4,400 cfm) combined with an unfiltered inleakage of 1,000 cfm, and 10 cfm for ingress/egress to the Control Room), and 4) incorporation of a sensitivity study based on a 12 to 24-month fuel cycle core inventory. These changes are not initiators for an accident previously evaluated in the USAR. Thus, the inclusion of these changes in the revised calculation does not result in more than a minimal increase in the frequency of occurrence of an accident previously evaluated in the USAR.

These changes in the revised analysis result in an increase in the dose consequences for the exclusion area boundary (EAB), the low population zone (LPZ) and the Control Room. The increase in dose consequences for the EAB, the LPZ and the Control Room is less than 10% the difference between the current calculated values and the regulatory guideline values.

For the EAB, the current calculated value is 0.041 REM TEDE and the 10 CFR 50.67 value is 25 REM TEDE. Ten percent (10%) of 25-0.041 is 2.4959 REM TEDE. The new calculated EAB value is 0.054 REM TEDE based on the sensitivity study using a 12 to 24-month fuel cycle core inventory. The difference between the new EAB dose and the current EAB dose is 0.054-0.041 = 0.013 REM TEDE. Because the dose increase (i.e., 0.013 REM TEDE) is less than 10% of the difference between the current value and the regulatory value (i.e., 2.4959 REM TEDE) and remains below the current SRP guideline value of 6.3 REM TEDE, the increase in EAB dose is not considered a more than a minimal increase in the consequences of an accident evaluated in the USAR.

For the LPZ, the current calculated value is 0.016 TEDE and the 10 CFR 50.67 value is 25 REM TEDE. Ten percent (10%) of 25-0.016 is 2.4984 REM TEDE. The new calculated LPZ value is 0.020 REM TEDE based on the sensitivity study using a 12 to 24-month fuel cycle core inventory. The difference between the new LPZ dose and the current LPZ dose is 0.020-0.016 = 0.004 REM TEDE. Because the dose increase (i.e., 0.004 REM TEDE) is less than 10% of the difference between the current value and the regulatory value (i.e., 2.4984 REM TEDE) and remains below the current SRP guideline value of 6.3 REM TEDE, the increase in LPZ dose is not considered a more than a minimal increase in the consequences of an accident evaluated in the USAR.

For the Control Room, the current calculated value is 0.438 REM TEDE and the 10 CFR 50.67 value is 5 REM TEDE. Ten percent (10%) of 5-0.438 is 0.4562 REM TEDE. The new calculated Control Room value is 0.48 REM TEDE. The difference between the new Control Room dose and the current Control Room dose is 0.48-0.438 = 0.042 REM TEDE. Because the dose increase (i.e., 0.042 REM TEDE) is less than 10% of the difference between the current value and the regulatory value (i.e., 0.4562 REM TEDE) and remains below the current SRP guideline value of 5 REM TEDE, the increase in Control Room dose is not considered a more than a minimal increase in the consequences of an accident evaluated in the USAR.

Table 3: Updated CRDA Doses Compared to Minimal Increase Dose Margins and SRP Limits (REM TEDE)

	Current Total Dose A	Updated Total Dose B	10 CFR 50.67 Dose Limits C	Dose Increase D=B-A	Minimal Dose Increase E= 0.1 (C-A)	SRP Dose Limit F
Control Room	0.438	0.48	5	0.042	0.4562	5
Exclusion Area Boundary	0.041	0.054	25	0.013	2.4959	6.3
Low Population Zone	0.016	0.020	25	0.004	2.4984	6.3

Suppression Pool pH

The suppression pool pH analysis is also impacted by the new 12-month fuel cycle source term. USAR Section 15.6.5.5.1.1, "Fission Product Release from Fuel," describes the fission product release from fuel after a LOCA accident and states, "the suppression pool pH is controlled at values above 7 following the core release period." The revised suppression pool pH calculation verifies that the suppression pool pH remains above 7.0 when the source term is updated to the latest 12-month fuel cycle source term or the core inventory associated with a 12 to 24-month operating cycle.

The impact of the new annual fuel cycle core source term and the 12 to 24-month fuel cycle source term on the qualification of safety related equipment has been assessed. The evaluation identified several Environmental Qualification (EQ) Zones and EQ dose calculations which would have slightly higher radiological dose (i.e., total integrated dose (TID)) than currently credited in the equipment environmental qualification program. The impacted EQ Binders associated with these EQ Zones were then evaluated for the increased dose. These evaluations were consistent with the dose evaluation methodology as described in the Updated Safety Analysis Report.

Equipment EQ Binders identified to have equipment in affected EQ zones were evaluated for impact on qualified life and accident and post-accident qualification of equipment. The environmental qualification of the safety related equipment is still maintained and bounded either by the existing qualification test and analyses established in the associated EQ Binders or with additional supporting qualification test reports for the affected equipment. Based on these evaluations there is no impact on the qualification of any safety-related equipment due to the change in the radiological conditions resulting from the revised accident source term.