July 31, 2001

Gary Van Middlesworth
Site Vice President
Duane Arnold Energy Center
Nuclear Management Company, LLC
3277 DAEC Road
Palo, IA 52324-0351

SUBJECT: DUANE ARNOLD ENERGY CENTER - ISSUANCE OF AMENDMENT REGARDING ALTERNATIVE SOURCE TERM (TAC NO. MB0347)

Dear Mr. Van Middlesworth:

The U.S. Nuclear Regulatory Commission has issued the enclosed Amendment No. 240 to Facility Operating License No. DPR-49 for the Duane Arnold Energy Center. This amendment consists of changes to the Technical Specifications (TS) in response to your application dated October 19, 2000, as supplemented March 23, April 9, and June 27, 2001. In your March 23 supplement, you requested the Nuclear Regulatory Commission (NRC) staff process certain proposed TS changes related to the fuel handling accident as a separate licensing action. Subsequently, the NRC approved TS changes in Amendment No. 237, dated April 16, 2001, associated with (1) secondary containment operability during core alterations, and (2) selective implementation of the alternative source term to the fuel handling accident. The enclosed amendment addresses the remainder of your October application.

The amendment revises the licensing basis to utilize the full scope of an alternative radiological source term for accidents as described in NUREG-1465, “Accident Source Terms for Light-Water Nuclear Power Plants,” and revises the TSs implementing various assumptions in the alternative source term analyses.

A copy of the Safety Evaluation is also enclosed. The Notice of Issuance will be included in the Commission's next biweekly *Federal Register* notice.

Sincerely,

/RA/

Brenda L. Mozafari, Project Manager, Section 1
Project Directorate III
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket No. 50-331

Enclosures: 1. Amendment No. 240 to License No. DPR-49
2. Safety Evaluation

cc w/encls: See next page

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PUBLIC	OGC	PWilson	
PDIII-1 Reading	ACRS	ESullivan	
CCraig	WBeckner	SLaVie	
FLyon	GHill(2)	CLauron	
THarris	GGrant, RGN-III	BMozafari	*No significant changes to SE

OFFICE	PDIII-1/PM	PDIII-1/LA	EMCB/SC	SPSB/SC	OGC	PDIII-1/SC
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DATE	7/3/01	7/5/01	5/30/01	4/30/01	7/12/01	7/16/01

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Duane Arnold Energy Center

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NUCLEAR MANAGEMENT COMPANY, LLC

DOCKET NO. 50-331

DUANE ARNOLD ENERGY CENTER

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 240
License No. DPR-49

1. The U.S. Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by Nuclear Management Company, LLC (the licensee) dated October 19, 2000, as supplemented March 23, April 9, and June 27, 2001, complies with the standards and requirements of the Atomic Energy Act of l954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with l0 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment and paragraph 2.C.(2) of Facility Operating License No. DPR-49 is hereby amended to read as follows:

(2) Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 240, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. The license amendment is effective as of the date of issuance and shall be implemented within 30 days of the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA/

Claudia M. Craig, Chief, Section 1
Project Directorate III
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Attachment: Changes to the Technical Specifications

Date of Issuance: July 31, 2001

ATTACHMENT TO LICENSE AMENDMENT NO. 240

FACILITY OPERATING LICENSE NO. DPR-49

DOCKET NO. 50-331

Replace the following pages of the Appendix A Technical Specifications with the attached revised pages. The revised areas are identified by amendment number and contain marginal lines indicating the areas of change.

Remove	Insert
1.1-3	1.1-3
3.3-71	3.3-71
3.4-13	3.4-13
3.4-14	3.4-14

The following Technical Specification Bases pages are provided for information only:

B 3.4-33
B 3.4-34
B 3.4-35
B 3.4-36
B 3.6-90
B 3.6-91
B 3.7-18
B 3.7-30
B 3.7-32
B 3.7-37
B 3.7-39
B 3.9-19
B 3.9-21

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 240 TO FACILITY OPERATING LICENSE NO. DPR-49

NUCLEAR MANAGEMENT COMPANY, LLC

DUANE ARNOLD ENERGY CENTER

DOCKET NO. 50-331

1.0 INTRODUCTION

By application dated October 19, 2000, as supplemented March 23, April 9, and June 27, 2001, Nuclear Management Company, LLC (NMC, or the licensee) requested a license amendment for the Duane Arnold Energy Center (DAEC). The licensee proposed replacing the current accident source term used in design basis radiological analyses with an alternative source term (AST) pursuant to 10 CFR 50.67, "Accident Source Term." Several other changes to the licensing basis and some technical specification (TS) changes were requested. In a teleconference between the Nuclear Regulatory Commission (NRC) staff and NMC on March 20, 2001, the staff identified additional information necessary to complete its review of the request. NMC responded to this request by letter dated March 23, 2001. In that response, NMC requested the staff process certain TS changes and the fuel handling accident (FHA) design-basis accident (DBA) as a separate licensing action. The April 9 and June 27, 2001, supplements provided "clean typed" TS pages to correspond to the "marked-up" TS pages provided in the March 23 supplement. The supplemental information contained clarifying information and did not change the initial no significant hazards consideration determination and did not expand the scope of the original *Federal Register* notice.

The NRC approved the TS changes associated with secondary containment operability during refueling operations and the selective implementation of the AST to the FHA in Amendment No. 237, dated April 16, 2001. The safety evaluation (SE) herein addresses the staff review of the DBA analyses and TS changes remaining in the October application:

1. Extending the selective AST implementation approved by Amendment No. 237 on April 16, 2001, to a full AST implementation by review of the remaining DBA accident analyses.

2. Revising atmospheric dispersion factors related to release points and receptors associated with the DBAs.

3. Revising several DAEC TSs for conformance with the AST implementation and various assumptions in the supporting DBA analyses:

a. Changing the reference in the Section 1.1 definition of "Dose Equivalent Iodine-131," from the existing reference to Federal Guidance Report (FGR) 11, "Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors," 1989, and FGR-12, "External Exposure to Radionuclides in Air, Water, and Soil," 1993. Also, the word "thyroid" was deleted.

b. Reducing the TS surveillance requirement (SR) 3.3.7.1.3 instrument setpoint for the control building air intake radiation monitor from ≤50 mR/hr to ≤5 mR/hr.

c. Reducing the dose equivalent I-131 specific activity limits stated in TS 3.4.6 from 1.2 μCi/ml to 0.2 μCi/gm and from 12 μCi/ml to 2 μCi/gm.

d. Various references to 10 CFR Part 100, in both the TSs and their Bases, will be changed to 10 CFR 50.67, to conform to the AST implementation.

2.0 EVALUATION

2.1 Alternative Source Term

In December 1999, the NRC issued a new regulation, 10 CFR 50.67, "Accident Source Term," which provided a mechanism for licensed power reactors to replace the traditional accident source term used in their DBA analyses with ASTs. Regulatory guidance for the implementation of these ASTs is provided in Regulatory Guide (RG) 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors." Section 50.67 requires a licensee seeking to use an AST to apply for a license amendment and requires that the application contain an evaluation of the consequences of affected DBAs. NMC's application of October 19, 2000, addresses these requirements in proposing to use the AST described in RG 1.183 as the DAEC DBA source term used to evaluate the radiological consequences of DBAs. As part of the implementation of the AST, the total effective dose equivalent (TEDE) acceptance criterion of 10 CFR 50.67(b)(2) replaces the previous whole body and thyroid dose guidelines of 10 CFR 100.11 and 10 CFR Part 50, Appendix A, General Design Criterion (GDC)-19, for the loss-of-coolant accident (LOCA), the main steamline break (MSLB) accident, and the control rod drop accident (CRDA).

The accident source term is intended to be representative of a major accident involving significant core damage and is typically postulated to occur in conjunction with a large LOCA. As a result of significant core damage, fission products are available for release into the containment environment. An AST is an accident source term that is different from the accident source term used in the original design and licensing of the facility and has been approved for use under 10 CFR 50.67. Although an acceptable AST is not set forth in the regulations, RG 1.183 identifies an AST that is acceptable to the staff for use at operating reactors.

The staff reviewed the NMC implementation of the AST and TEDE criteria and found it to meet the requirements of 10 CFR 50.67 and the guidance provided in RG 1.183. It is, therefore, acceptable. With this determination and the prior selective AST implementation for the FHA analysis approved in Amendment No. 237, DAEC has satisfied the requirements for full AST implementation as described in RG 1.183.

2.2 Radiological Consequences of DBAs

The staff reviewed the licensee's analysis methods, assumptions, and inputs using docketed information provided by the licensee. Although the staff performed independent calculations as a means of confirming the licensee's results, the staff's acceptance is based on the licensee's analysis. Table 1 provides the MSLB, CRDA, and LOCA analysis assumptions found acceptable by the staff. Table 2 provides the doses projected by NMC.

NMC performed the radiological analyses supporting this amendment assuming a reactor power equal to 102 percent of 1912 MWt. This power level exceeds the current licensed reactor power for DAEC. The increased power level was used by NMC in support of a future power uprate application. This is a conservative approach for this amendment and is acceptable to the staff. However, the staff notes that this amendment does not revise the current licensed reactor power, since the power uprate request is being pursued in a separate licensing action. However, this SE and that prepared for Amendment No. 237 will provide a basis for the staff's conclusion in the subsequent licensing action that the radiological consequences of the proposed power uprate are acceptable.

2.2.1 Loss-of-Coolant Accident (LOCA)

The objective of analyzing the radiological consequences of a LOCA is to evaluate the performance of various plant safety systems intended to mitigate the postulated release of radioactive materials from the plant to the environment. NMC assumes an abrupt failure of a large reactor coolant pipe and assumes that substantial core damage occurs due to this event. The assumption of core damage is conservative in that DBA thermo-hydraulic analyses in the DAEC Updated Final Safety Analysis Report (UFSAR) conclude the fuel damage thresholds are not exceeded.

2.2.1.a. Source Term

Fission products from the damaged fuel are released into the reactor coolant system (RCS) and then into the primary containment (i.e., drywell and wetwell). With the LOCA, it is anticipated that the initial fission product release to the primary containment will last 30 seconds and will release all of the radioactive materials dissolved or suspended in the RCS liquid. The gap inventory release phase begins two minutes after the event starts and is assumed to continue for 30 minutes. As the core continues to degrade, the gap inventory release phase ends and the in-vessel release phase begins. This phase continues for 1.5 hours. Tables 1, 4, and 5 of RG 1.183 define the source term used for these two phases. These data are summarized below.

Table 2.2.1-1 Release Fractions as a Function of Release Period

Radionuclide Group	Gap Release *(0.5 hrs)*	Early In-Vessel *(1.5 hrs)*
Noble Gases (Xe, Kr)	0.05	0.95
Halogens (I, Br)	0.05	0.25

Alkaline Metals (Cs, Rb)	0.05	0.20
Tellurium Group (Te, Sb, Se)	0	0.05
Ba, Sr	0	0.02
Noble Metals (Ru, Rh, Pd, Mo, Tc, Co)	0	0.0025
Cerium Group (Ce, Pu, Np)	0	0.0005
Lanthanides (La, Zr, Nd, Eu, Nb, Pm, Pr, Sm, Y, Cm, Am)	0	0.0002

The inventory in each release phase is released at a constant rate over the duration of the phase and starting at the onset of the phase. Once dispersed in the primary containment, the release to the environment is assumed to occur through three pathways:

- Leakage of primary containment atmosphere (i.e., design leakage).
- Leakage of primary containment atmosphere via design leakage through main steam isolation valves (MSIVs).
- Leakage from emergency core cooling systems (ECCS) that recirculate suppression pool water outside of the primary containment (i.e., design leakage).

The LOCA considered in this evaluation is a complete and instantaneous severance of one of the recirculation loops. The pipe break results in a blowdown of the reactor pressure vessel (RPV) liquid and steam to the drywell via the severed recirculation pipe. The resulting pressure buildup drives the mixture of steam, water, and other gases down through vents to the downcomers and into the suppression pool water thereby condensing the steam and reducing the pressure. Due to the postulated loss of core cooling, the fuel heats up, resulting in the release of fission products. Under the TID14844 (J.J. DiNunno, et al., “Calculation of Distance Factors for Power and Test Reactor Sites,” USAEC TID-14844, U.S. Atomic Energy Commission, 1962) assumption of instantaneous core damage, this initial blowdown would also include fission products, a fraction of which would be retained by the suppression pool water. Under the AST, the fission product release occurs in phases over a two-hour period. Significant quantities of fission products would not be part of the initial blowdown to the suppression pool. Subsequent recirculation of suppression pool water by the ECCS would cause some transport of fission products between the drywell and the wetwell, and some scrubbing effect. NMC has conservatively assumed no credit for suppression pool scrubbing of fission products. In addition, NMC has conservatively assumed that the fission product release from the RPV is homogeneously dispersed within the drywell free volume only, ignoring the free volume of the wetwell.

NMC assumes that a portion of the fission products released from the RPV will plate out due to natural deposition processes. NMC models this deposition using the 10-percentile model described in the staff-accepted NUREG/CR-6189, “A Simplified Model of Aerosol Removal by Natural Processes in Reactor Containments” (i.e., the “Powers Model”).

The AST assumes that the iodine released to the containment includes 95 percent cesium iodide (CsI), 4.85 percent elemental iodine, and 0.15 percent organic iodide. The assumption

of this iodine specification is predicated on maintaining the containment sump water at pH 7.0 or higher. NMC proposes to use the standby liquid control system to inject 2500 gallons of sodium pentaborate to the RPV, where it will mix with ECCS flow and spill over to the drywell and then to the suppression pool. Sodium pentaborate, a base, will neutralize acids generated in the post-accident primary containment environment.

2.2.1.b. Evaluation of pH Analysis

The fission products which result from significant core damage can be limited from escaping the containment environment through chemical means. In particular, elemental iodine formed during the accident can be held in the liquid phase and collected into the suppression pool if the suppression pool is maintained at a pH of 7.0 or greater.

The licensee states that the calculation methodology is based on the approach provided in the following: NUREG-1465, “Accident Source Terms for Light-Water Nuclear Power Plants,” USNRC Draft Regulatory Guide DG-1081, “Alternate Radiological Source Terms for Evaluating the Radiological Consequences of Design-Basis Accidents at Boiling and Pressurized Water Reactors,” and NUREG-5950, “Iodine Evolution and pH Control.”

The licensee’s analyses conservatively maximized the post-LOCA production of acids (Hydroiodic Acid, Hydrochloric Acid, and Nitric Acid) and minimized the post-LOCA production and/or addition of bases (Cesium Hydroxide and Sodium Pentaborate).

Hydroiodic acid is a strong acid. However, the production of this acid during an accident is not enough to significantly affect the large water volume in the suppression pool. Therefore, this acid is not considered a significant contributor in affecting the suppression pool pH.

Hydrochloric acid contributes to the decrease of the suppression pool pH and is assumed to be generated by the irradiation of electrical cable (Hypalon) found in the primary containment drywell. Specifically, the analyses assumed 50 percent of the identified cable mass as being “free drop” and exposed to both beta and gamma radiation. The remaining cable mass is assumed to undergo beta radiation dose equal to 50 percent of the incident dose due to self and structural shielding. These assumptions are conservative and consistent with the staff’s position in NUREG-5950.

Nitric acid, a product of the irradiation of water and air, is initially assumed to decrease the pH of the reactor water. However, as the suppression pool volume increases, the amount of nitric acid formed proportionally decreases. This results from the decrease in specific radiation activity since the total radiation source term remains constant while the volume of water available for irradiation increases. The staff finds this assumption appropriate in that nitric acid production is independent of the suppression pool volume and does not significantly decrease the suppression pool’s pH.

Cesium that is not in the form of cesium iodide (CsI) is assumed in the analyses to exit the RCS in the form of cesium hydroxide (CsOH) and deposit into the suppression pool. CsOH is a base and can result in an increase of the pool’s pH. Two cases were analyzed: (1) all CsOH reaching the suppression pool and (2) no credit taken for CsOH deposition into the suppression pool but with addition of sodium pentaborate. The licensee’s analyses show that dependence solely on CsOH deposition (Case 1) is not sufficient in maintaining pH below 7.0. This is

consistent with the iodine source term behavior and its transport being independent of iodine re-evolution and pH control. For Case 2, the licensee assumed 2500 gallons of 11.8 wt% of sodium pentaborate is injected at 26 gpm within 2 hours of the accident. In addition to the injection of this basic solution, ECCS core spray and low pressure core injection pumps are available to draw suction from the suppression pool and one complete exchange is obtained within 1.2 hours. The staff finds these assumptions appropriate in determining that complete mixing of the sodium pentaborate and reactor water has occurred within 2 hours of the accident.

The results of the licensee's time dependent analyses are found in Table 3-8, "Acid Generation and Calculated Pool pH Values" which is further illustrated in Figure 3-1, "Pool pH Response" of the submittal. The results show that the suppression pool pH will be maintained above 7.0 throughout the duration of the accident.

Based on the staff's evaluation of the above discussion of the analytical assumptions applied in the licensee's calculations, the staff finds that the pH of the suppression pool will be maintained at a level above 7.0, thus, preventing re-evolution of elemental iodine dissolved in the suppression pool water.

2.2.1.c. Containment Leakage Pathway

The Mark I primary containment is projected to leak at its design leakage of 2.0 percent of its contents by weight per day for the first 24 hours and then at 1.0 percent for the remainder of the 30-day accident duration. Leakage from the drywell/wetwell will collect in the free volume of the secondary containment and be released to the environment via ventilation system exhaust or leakage. Following a LOCA, the standby gas treatment system (SGTS) fans start and draw down the secondary containment to create a negative pressure with reference to the environment. This pressure differential ensures that leakage from the drywell/wetwell is collected and processed by the SGTS. SGTS exhaust is processed through charcoal filter media prior to release to the environment via the site's elevated stack. NMC does not credit dilution or holdup of leakage in the secondary containment. In addition, NMC assumes that a positive pressure exists in the secondary containment for the first five minutes after the accident and that this leakage is released directly to the environment as a ground level release.

The original DAEC licensing basis did not consider a positive pressure and there are no secondary containment drawdown time SRs. NMC has stated that existing SRs for secondary leakage (SR 3.6.4.1.3) or SGTS (SR 3.6.3.1) could not be performed satisfactorily in the presence of leakage that could prevent establishing a negative pressure within 5 minutes. DAEC also noted that their assumption of 5 minutes is larger than the 2 minute drawdown time typical at boiling-water reactors having this SR. NMC has taken the position that the existing DAEC TSs are adequate to ensure this analysis assumption is met. While an explicit drawdown time specification would be preferable, the staff has decided to accept the position proffered by NMC. The staff has reasonable assurance that the existing SRs will identify leakage that prevents achieving the requisite negative pressure and will adequately serve as an indication that the analysis assumption can not be met. The staff has reasonable assurance that NMC's decisions not to credit holdup and dilution in the secondary containment, and not to credit wetwell free volume in establishing containment release rates, largely compensate for any uncertainty in the drawdown time assumption.

2.2.1.d. Main Steam Isolation Valve Leakage

The four main steamlines, which penetrate the primary containment, are automatically isolated by the MSIVs in the event of a LOCA. There are two MSIVs on each steamline, one inside containment and one outside containment. The MSIVs are functionally part of the primary containment boundary and design leakage through these valves provides a leakage path for fission products to bypass the secondary containment and enter the environment as a ground level release. DAEC conservatively assumes that the fission products released from the core are dispersed equally throughout the drywell via the severed recirculation line. This analysis crediting of holdup and dilution is consistent with the postulated progression of the accident. Following the initial blowdown of the RPV, the fuel heats up and fuel melt begins, and the steaming in the RPV carries fission products to the containment. When core cooling is restored, steam is rapidly generated in the core. This steam and the ECCS flow carry fission products from the core to the primary containment via the severed recirculation line, resulting in well-mixed RPV dome and containment fission product concentrations. Once the rapid steaming stops, the containment contents can flow back into the RPV through the severed line and would be available for release via the MSIVs.

License Amendment No. 207, issued in February 1995, increased the allowable MSIV leakage from 11.5 standard cubic feet per hour (scfh) for any one MSIV to 100 scfh for any one MSIV with a total leakage of 200 scfh through all four main steamlines. The analysis that supported this change was based, in part, on the Boiling Water Reactor Owners' Group (BWROG) proposed topical report NEDC-31858P, Revision 2, "BWROG Report for Increasing Main Steam Isolation Valve Leakage Rate Limits and Elimination of Leakage Control Systems," dated September 1993. This topical report, now generically accepted, provided a methodology for establishing seismic performance of the steam piping and the main condenser and a methodology of assessing the radiological consequences. This latter methodology takes advantage of the large volume in the main steamlines and the main condenser to provide hold-up and plate-out of fission products that leak past the MSIVs. In the supporting SE for Amendment No. 207, the staff concluded that the DAEC main steamlines, main steam drain lines, condenser, and associated interconnecting piping and supports will be seismically adequate to support crediting hold-up and plate-out in the steamlines and main condenser.

NMC assumes that the elemental iodine and particulate filter removal efficiency of the main condenser is 99.6 percent for the turbine stop valve drain path and 99.7 percent for the MSIV drain path, consistent with this approved prior design basis. These efficiencies were determined using a methodology based on the TID14844 iodine species fractions of 91 percent elemental, 4 percent organic and 5 percent particulate. The staff believes that the efficiencies determined using that methodology are bounding for the AST with its 95 percent aerosol (particulate) iodine species and 0.15 percent organic iodine species.

For deposition in the main steamlines, NMC uses the Brockmann-Bixler pipe deposition model incorporated in the NRC-sponsored RADTRAD computer code. NMC conservatively modeled two of the four main steamlines with 100-scfh leakage into each line. The sum of the leakage, 200 scfh, is equivalent to the maximum leakage allowed by DAEC TSs. These flows are reduced by 50 percent at 24 hours to reflect the decrease in containment pressure, which is the driving force for the steam release. One of the two steamlines was designated as being faulted within the containment with its inboard MSIV assumed to have failed open. As a result, NMC conservatively did not credit deposition between the RPV and the outboard MSIV in this faulted

steamline. NMC credited deposition downstream of the outboard MSIVs in both steamlines. The staff considers this approach conservative in that it exceeds minimum regulatory guidance since multiple failures are postulated.

Two aspects of NMC's use of the Brockmann-Bixler deposition model warrant discussion:

- Section 2.2.6.1 of NUREG/CR-6604, "RADTRAD: A Simplified Model for RADionuclide Transport and Removal And Dose Estimation," briefly describes the bases of the Brockmann-Bixler model. The deposition efficiency is defined in the context of "…the horizontally projecting lower surface…" NMC included both horizontal and vertical main steamlines in determining the total volume and interior surface area used in the model.

- Appendix A to RG 1.183, Section 6.3, provides the guidance that steamline deposition models should be based on the assumption of well-mixed volumes, but other models such as plug flow may be used if justified. Brockmann-Bixler is a plug flow model.

In response to a staff request for additional information, NMC identified that the MSIV leakage model did include vertical pipe runs. NMC provided a description of a sensitivity analysis that indicated that the inclusion of these lines changes the projected dose rates due to MSIV leakage by less than about 2 percent for offsite receptors and about 3.2 percent for control room operators. NMC identified their analysis assumption of gas pressure equal to 3.11 atmospheres and gas temperature of 550 degrees Fahrenheit for the entire duration of the accident as a compensating conservatism. (Deposition would improve with the reduced pressure and temperature from plant cool down.) The staff agrees that the difference in this particular analysis is insignificant. However, the staff cannot endorse the inclusion of vertical runs in the DAEC design basis where the assumption might be used in future re-analyses. In response to this concern, NMC provided a commitment in its March 23, 2001, letter that, "The design calculations using the Brockman-Bixler pipe deposition model will be updated to only credit horizontal runs of piping in the next revision of these calculations used to support a DAEC licensing action."

The staff considered main steamline deposition credit in its review of the BWROG NEDC-31858P topical report. This model was based on plug flow in the main steamlines and main condenser. Use of this model did require the applicant to seismically qualify the steamlines and the main condenser. During a DBA LOCA, the flow pattern in the main steamline could be plug flow, well-mixed, or some combination of the two. Plug flow effectively results in a longer fission product transport time and more deposition in the steamline. The staff considered main steamline deposition in its review of the Perry Nuclear Plant pilot AST application, dated August 27, 1996. The staff's analysis is documented in the SE for Perry Amendment No. 103, dated March 26, 1999, and in the staff technical report AEB-98-03, "Assessment of Radiological Consequences for the Perry Pilot Plant Application using the Revised (NUREG-1465) Source Term," dated December 9, 1998. The conclusions of this report formed part of the basis of the cited RG-1.183 guidance. Appendix A of AEB-98-03 addresses the use of plug-flow and well-mixed models. In the Perry analysis, the steamlines and the main condenser downstream of the outboard MSIVs had not been qualified for purposes of MSIV holdup and dilution. The staff noted:

> Given the availability of the condenser and piping downstream of the third main steam valve in the earlier cases, it was likely to be less important whether the

> flow was modeled as plug flow or well-mixed. However, we believed that having only the main steamline [upstream of the third MSIV] available for deposition in Perry warranted further consideration of the flow conditions in the main steamline.

The staff finds the use of plug flow to be an acceptable deviation from the guidance of RG 1.183, based on the approval of BWROG NEDC-31858P at DAEC in Amendment No. 207 and on the existence of the seismically adequate alternative leakage paths and main condenser.

2.2.1.e. Leakage from Emergency Core Cooling Systems

During the progression of a LOCA, some fission products released from the fuel will be carried to the suppression pool via spillage from the RCS and by natural processes such as deposition and plateout. Post-LOCA, the suppression pool is a source of water for ECCS. Since portions of these systems are located outside of the primary containment, leakage from these systems is evaluated as a potential radiation exposure pathway. For the purposes of assessing the consequences of leakage from the ECCS, NMC assumes that all of the radioiodines released from the fuel are instantaneously moved to the suppression pool. Noble gases are assumed to remain in the drywell atmosphere. Since aerosols and particulate radionuclides are not expected to become airborne on release from the ECCS, they are not included in the ECCS source term. This source term assumption is conservative in that all of the radioiodine released from the fuel is credited in both the primary containment atmosphere leakage and the ECCS leakage. In a mechanistic treatment, the radioiodines in the primary containment atmosphere would relocate to the suppression pool over time.

The analysis considers the equivalent of 1.5 gpm unfiltered ECCS leakage starting at the onset of the LOCA. This leakage includes a factor-of-two multiplier to address increases in the leakage due to normal material degradation between surveillance tests. NMC assumes the 10 percent of the iodine in the ECCS leakage becomes airborne and is available for release as 97 percent elemental and 3 percent organic iodine. No credit was assumed for hold-up and dilution in the secondary containment. As was assumed for the primary containment leakage pathway, the leakage enters the environment as an unfiltered ground level release for the first five minutes after the event starts. After this five-minute positive pressure period, the leakage enters the environment via the SGTS as a filtered elevated release.

2.2.1.f. Offsite Doses

NMC evaluated the maximum 2-hour TEDE to an individual located at the exclusion area boundary (EAB) and the 30-day TEDE to an individual at the outer boundary of the low populations zone (LPZ). The resulting doses are less than the 10 CFR 50.67 criteria.

2.2.1.g. Control Room Doses

NMC evaluated the dose to operators in the control room and to personnel in the technical support center (TSC). It was assumed that the control room isolation would initiate with an alarm on radiation monitors located in the control room ventilation intake, and the isolation would be complete in four minutes following the start of the event. The TSC is isolated manually, at 30 minutes after the start of the event.

NMC analyzed the control room and TSC doses over a thirty-day period. Although the TSC and control room are both designed to be pressurized during an accident event, NMC assumes that unfiltered inleakage occurs. Since this inleakage has not been quantified, NMC analyzed several cases, including 1000, 500, and 67.5 cfm unfiltered inleakage, and 0 cfm inleakage. The results of this sensitivity analysis indicated that the higher doses are associated with the lower unfiltered inleakage rates. This is not an unexpected result for the LOCA and can be explained by the purging effect of the higher flow rate. This unfiltered infiltration is assumed to commence at the onset of the event and continue for 30 days for both the control room and the TSC.

Prior to the event, the control room ventilation system draws in 3150 cfm of outside air. Once isolation occurs, this makeup flow is stopped and a filtered 1000 cfm makeup is started. The intake filters have efficiencies of 99 percent for particulates, 90 percent for elemental iodine, and 30 percent for organic iodine. The TSC normal outside air makeup is 900 cfm. Once the TSC ventilation is isolated, there is 200 cfm of outside air that mixes with 800 cfm of recirculated air and passes through the TSC filters. The recirculation filter has efficiencies of 99 percent for particulates, 90 percent for elemental iodine, and 30 percent for organic iodine.

The staff is currently developing regulatory guidance regarding control room habitability, including surveillance testing of unfiltered inleakage. In addition, the Nuclear Energy Institute (NEI) is developing an industry initiative document (NEI 99-03) on control room habitability. The staff's acceptance of NMC unfiltered inleakage assumption here does not preclude any future generic regulatory actions that may become applicable to DAEC in this regard.

Based on this review, the staff concluded that the licensee's application of the AST to the DAEC LOCA analysis is acceptable. Figure 1 illustrates the model used by the staff. Table 1 provides the analysis assumptions found acceptable by the staff. Table 2 provides the doses projected by NMC.

2.2.2 Main Steamline Break (MSLB)

The accident considered is the complete severance of a main steamline outside the primary containment with the reactor operating at 1950 MWt. The radiological consequences of a break outside containment will bound the results from a break inside containment. The MSIVs are assumed to isolate the leak within 10.5 seconds. This assumed time is conservative in that the expected time, based on MSIV TSs and testing, is 3 to 5 seconds. The analysis is performed for two activity release cases, based on the maximum equilibrium and pre-accident iodine spike concentrations of 2 uCi/gm and 0.2 uCi/gm, respectively. These activities reflect the proposed changes to the RCS specific activity TSs requested as part of this amendment. All of the accident activity was assumed released within 10.5 seconds following the accident as a ground level release with no credit for turbine building holdup or dilution.

The control room and TSC ventilation system were modeled as described above for the LOCA, with the exception that the control room is not assumed to isolate until the release is complete, e.g., 10.5 seconds. This maximizes the estimated doses and is, therefore, conservative.

Based on this review, the staff concluded that the licensee's application of the AST to the DAEC MSLB analysis is acceptable. Table 1 provides the analysis assumptions found acceptable by the staff. Table 2 provides the doses projected by NMC.

2.2.3 Control Rod Drop Accident (CRDA)

This accident analysis postulates a sequence of mechanical failures that results in the rapid removal (i.e., drop) of a control rod. Localized damage to fuel cladding and a limited amount of fuel melt are projected. A reactor trip will occur. The MSIVs are assumed to remain open for the duration of the event. NMC has projected that 1200 fuel rods would be breached by the event, and, of these damaged rods, 0.77 percent would exceed the threshold for melting.

In its October 19, 2000, application, NMC identified that the CRDA analysis had been performed using the gap fractions and fuel melt fractions from Section 3 of the draft regulatory guide DG-1081, which were revised in the final RG 1.183. Although the gap fractions were generally reduced, the iodine melt release was increased from 0.3 in DG-1081 to 0.5 in Appendix C of RG 1.183. In the application, NMC recognized the difference in gap fractions and indicated that results obtained using the DG-1081 gap fractions would be bounding. NMC also stated that since their results were conservative that they would defer conforming analyses. NMC committed, "However, at such time these analyses are reperformed to support future projects, these analyses will be brought into conformance." In its letter dated March 23, 2001, NMC considered the impact of the increased melt fraction, and concluded that the difference would have a negligible impact on doses. The staff performed its confirming analysis using the guidance of Appendix C of RG 1.183. The staff finds the NMC position acceptable.

It is assumed that 100 percent of the noble gases but only 10 percent of the iodines released reach the main condenser due to plateout in the RPV and main steamlines. Of the iodine that enters the main condenser, 90 percent plates out. There is no reduction in noble gases. The fission product gases in the main condenser are released at a rate of 1 percent by volume over 24 hours as a ground level release.

NMC did not evaluate the doses to the control room and TSC for the CRDA event. NMC stated that CRDA doses would be bounded by the consequences from the other DBAs. To confirm this conclusion, the staff assessed the control room dose using the control room model as described above for the LOCA, with the exception that the control room is not assumed to be isolated for 30 minutes. In the analyses for the other accidents, NMC performed a sensitivity analysis for unfiltered inleakage values of 67.5, 500, and 1000 cfm. For the accidents with a short release duration or those with an elevated release, the 67.5 cfm infiltration was limiting. However, the CRDA involves a 24-hour ground level release. The emergency mode ventilation flow rate is 1000 cfm. NMC asserts that control building positive pressure surveillance tests per SR 3.7.4.4 would be unsatisfactory if such a large leakage pathway existed. NMC states that no other inflow or pressurization source exists for the control building that would mask the performance of the standby filter units during SR 3.7.4.4. For these reasons, the staff assumed an unfiltered infiltration rate of 1000 cfm for its analysis. The staff's analysis indicates that the CRDA control room and TSC doses are not limiting with a large margin, confirming the licensee's position.

Based on this review, the staff concluded that the licensee's application of the AST to the DAEC CRDA analysis is acceptable. Table 1 provides the analysis assumptions found acceptable by the staff. Table 2 provides the doses projected by NMC.

2.2.4 Other Radiological Consequence Analyses

NMC considered the impact of the changes proposed in this amendment and the future power uprate in several additional DAEC radiological consequence analyses. NMC considered the impact on the equipment qualification (EQ) in radiation. Consistent with staff practice (SECY-99-240), NMC used the TID14844 source term in examining these potential impacts. The evaluation showed that the post-accident EQ dose rates and 30 day gamma integrated doses are bounded by those described in the DAEC EQ program design documentation. NMC did use the AST source term for EQ evaluations where the predominant accident dose source is from exposure to the radiological plume. These included: control room; control building heating, ventilation, and air conditioning (HVAC); pump house; and river intake structure. While minor increases in doses were identified, none of these changes resulted in re-designating these areas as harsh environments.

NMC qualitatively evaluated the potential impacts on compliance with various DAEC commitments to NUREG-0737 and performed some calculations to update the mission doses for NUREG-0737 item II.B.2 and the control room doses. NMC evaluated the impact on the post-accident vital mission doses associated with post-accident sampling (PASS) and revised selected calculations as necessary. The PASS mission previously had the highest dose to personnel of all the post-accident missions and, as such, is the limiting case. The evaluation results indicate that the DAEC plant shielding is adequate to maintain all post-accident vital mission doses within regulatory criteria.

NMC also evaluated the impact of the proposed changes on radiation monitor alarm setpoints and determined that the existing setpoints would remain bounding.

Based on the information provided by the licensee, the staff finds these evaluations meet the guidance of RG 1.183 and are, therefore, acceptable.

2.3 Atmospheric Dispersion (χ/Q) Changes

NMC assumed revised χ/Q values in the performance of the DBA analyses. NMC states that the historical χ/Q data did not meet their expectations for level of documentation with regard to the DAEC design basis and, for this reason, new values for various combinations of release points and receptors were generated using the PAVAN-PC code and the ARCON96 code. These were tabulated in the October 19, 2000, application. The hourly observation meteorology data used in these analyses were obtained from the DAEC meteorological program over the period of January 1, 1997, to December 31, 1999. The DAEC program, which is described in UFSAR Section 2.3.3, meets the guidance of RG 1.23 and is subject to the DAEC 10 CFR Part 50, Appendix B, quality assurance program. Data recoverability exceeded 90 percent during this period.

The PAVAN-PC and ARCON96 codes are acceptable methodologies. In response to staff requests, NMC provided copies of the input data printouts from PAVAN-PC and ARCON96. The staff qualitatively reviewed the inputs to the codes and found them to be consistent with UFSAR site configuration drawings and staff practice. The staff also performed a confirmatory calculation of the χ/Q values for a subset of source-receptor combinations. Based on this review, the staff finds the revised χ/Q values acceptable.

2.4 Proposed TS Changes

NMC requested changes to some TSs, and some conforming changes to the TS Bases. The following TSs are affected by the proposed changes:

a. In Section 1.1, the licensee proposes to change the definition of "Dose Equivalent Iodine-131," to substitute Federal Guidance Report (FGR) 11, "Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors," 1989, and FGR-12, "External Exposure to Radionuclides in Air, Water, and Soil," 1993, for the existing reference. Also, the word "thyroid," was deleted.

   The existing definition is based on the dose conversion factors provided in TID-14844. This change conforms to the implementation of the AST. The new citations are as cited in RG 1.183 and are acceptable. These revised values were used in the re-analysis of the MSLB accident, found acceptable above. No other DBA analysis, including the FHA approved in Amendment No. 237, is affected by this change. The change is acceptable.

b. In SR 3.3.7.1.3, the licensee proposes to reduce the instrument setpoint for the control building air intake radiation monitor from ≤50 mR/hr to ≤5 mR/hr.

   This change decreases the control room air intake radiation monitor reading necessary to initiate control room isolation. The effect of this change is to cause control room isolation earlier during the accident progression, a conservative situation. The revised value was used in establishing the control room isolation time for the DBA LOCA. Since automatic isolation was not credited for other DBA analysis, including the FHA approved in Amendment No. 237, these events are not affected by this change. The change is acceptable.

c. In TS 3.4.6 action statements, the licensee proposes to reduce the dose equivalent I-131 specific activity limits from 1.2 μCi/ml to 0.2 μCi/gm and from 12 μCi/ml to 2 μCi/gm.

   This change is a reduction in the radioactive material concentrations in the RCS during normal plant operations. The effect of this change is to reduce the activity available for release in the event of an accident. These revised values were used in the re-analysis of the MSLB accident, found acceptable above. No other DBA analysis, including the FHA approved in Amendment No. 237, is affected by this change. The change is acceptable.

d. The licensee proposes to change various references to 10 CFR Part 100 in the TS Bases to 10 CFR 50.67, to conform to the AST implementation.

   With the implementation of the AST, the accident dose guidelines of 10 CFR Part 100 are superseded by the dose criteria in 10 CFR 50.67. The whole body and thyroid doses of 10 CFR Part 100 are replaced by the TEDE criteria of 10 CFR 50.67. This is a conforming change. The analyses performed in support of this amendment (and that for the FHA in Amendment No. 237) determined radiological consequences in terms of the

TEDE dose quantity, and were shown to be in compliance with the dose criteria in 10 CFR 50.67. The staff has no objection to the changes in the Bases.

3.0 STATE CONSULTATION

In accordance with the Commission's regulations, the Iowa State official was notified of the proposed issuance of the amendment. The State official had no comments.

4.0 ENVIRONMENTAL CONSIDERATIONS

This amendment changes a requirement with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 or changes a surveillance requirement. The staff has determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluent that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendment involves no significant hazards consideration and there has been no public comment on such finding (66 FR 34285). Accordingly, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendment.

5.0 CONCLUSION

In its application dated October 19, 2000, NMC had proposed a full scope implementation of the AST. By letter dated March 23, 2001, NMC asked the staff to treat the FHA as a selective implementation of the AST. Amendment No. 237, addressing that request, was issued on April 16, 2001. With this SE, the staff has completed its review of the remaining DBAs and other radiological consequences of an AST implementation and has determined that NMC has met the requirements of 10 CFR 50.67 and the guidance of RG 1.183 for a full scope implementation.

The staff reviewed the assumptions, inputs, and methods used by NMC to assess the radiological impacts of the proposed changes. In doing this review, the staff relied upon information placed on the docket by NMC, staff experience in doing similar reviews and, where deemed necessary, on staff confirmatory calculations. The staff finds that NMC used analysis methods and assumptions consistent with the conservative guidance of RG 1.183, the proposed TS changes, and the future power uprate. The staff compared the doses estimated by NMC to the applicable criteria and to the results of confirmatory analyses by the staff. The staff finds, with reasonable assurance, that the licensee’s estimates of the EAB, LPZ, control room, and TSC total effective dose equivalent due to postulated DBAs at DAEC will comply with the requirements of 10 CFR 50.67 and the guidance of RG 1.183. The staff finds reasonable assurance that DAEC AST implementation will continue to provide sufficient safety margins with adequate defense-in-depth to address unanticipated events and to compensate for uncertainties in accident progression and in analysis assumptions and parameters.

Since these analyses were performed at a power level of 1950 MWt (102 percent of 1912 MWt), the staff finds that the radiological consequences of these DBAs would remain bounding

up to a rated thermal power of 1912 MWt. However, the approval of this amendment does not constitute authority to operate above the current licensed rated thermal power of 1658 MWt.

This licensing action is considered a full implementation of the AST. With this approval, the previous accident source term in the DAEC design basis is superseded by the AST proposed by NMC in its application of October 19, 2000. The previous offsite, control room, and TSC accident dose criteria expressed in terms of whole body, thyroid, and skin doses are superseded by the TEDE criteria of 10 CFR 50.67 or small fractions thereof, as defined in RG 1.183. All future radiological analyses performed to demonstrate compliance with regulatory requirements will address all characteristics of the AST and the TEDE criteria as described in the now-updated DAEC design basis.

The staff has concluded, based on the considerations discussed above, that: (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.

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Date: July 31, 2001

Figure 1

LOCA Model

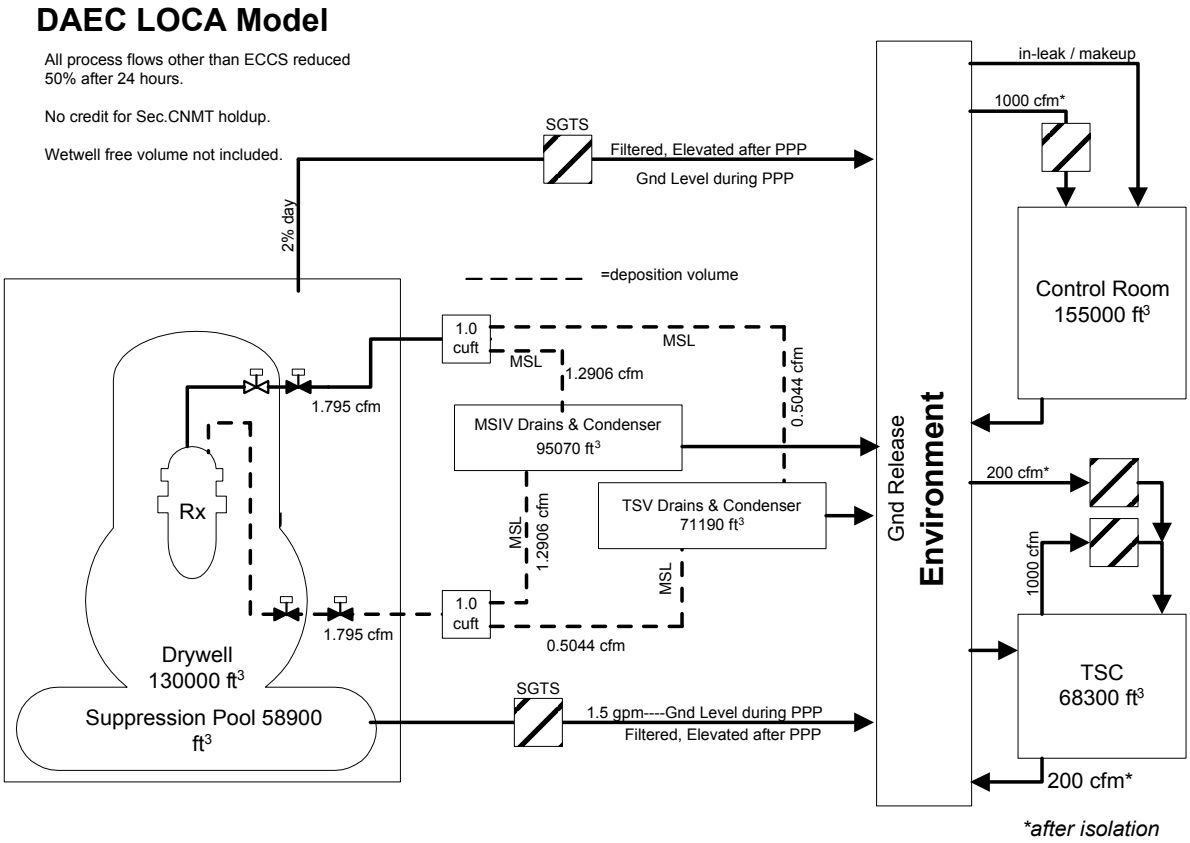

TABLE 1

RADIOLOGICAL ANALYSIS ASSUMPTIONS

Assumptions Common to One or More Analyses

Reactor power, MWt (102% of 1912)	1950
RCS specific activity, equilibrium µCi/gm dose equivalent I-131	0.2
RCS specific activity, spike µCi/gm dose equivalent I-131	2.0
Dose conversion factors	FGR11 and FGR12
Control room volume, ft^3	155,000
Normal ventilation makeup flow, cfm	3150
Control Room filtered makeup flow, cfm	1000
Control Room Filter Efficiency %	
Aerosol	99
Elemental	90
Organic	30
Control Room unfiltered Inleakage, cfm	0, 67.5, 500, 1000
Control room breathing rate, m^3/sec	3.47E-4
Control room occupancy factors	
0-24 hours	1.0
1-4 days	0.6
4-30 days	0.4

Control room χ/Q, sec/m^3

Period	TB Exhaust	Condenser	RB Wall	Elv Stack	RB Vent
0-2 hrs	9.23E-4	1.48E-3	1.33E-2	3.93E-7*	2.85E-3
2-8 hrs	7.96E-4	1.27E-3	1.12E-2	3.75E-7	2.29E-3
8-24 hrs	3.57E-4	5.56E-4	5.21E-3	1.33E-7	1.02E-3
1-4 days	2.47E-4	3.40E-4	3.77E-3	1.04E-7	3.64E-4
4-30 days	1.88E-4	2.65E-4	2.87E-3	9.37E-8	1.80E-4

* 2.62E-4 for 0-30 minute fumigation period.

TSC Isolation, minutes	30
TSC Normal Ventilation, cfm	900
TSC filtered makeup, cfm	200
TSC filtered recirculation, cfm	800

TSC Filter Efficiency %

Aerosol	99
Elemental	90
Organic	30

TSC occupancy factors

0-24 hours	1.0
1-4 days	0.6
4-30 days	0.4

TSC χ/Q, sec/m^3

Period	Condenser	RB Wall	Elv Stack	RB Vent
0-2 hrs	2.14E-3	8.52E-3	2.32E-7*	2.66E-3
2-8 hrs	1.86E-3	7.09E-3	2.16E-7	2.25E-3
8-24 hrs	8.84E-4	3.28E-3	8.00E-8	1.03E-3
1-4 days	6.10E-4	2.36E-3	6.15E-8	4.34E-4
4-30 days	4.69E-4	1.86E-3	5.39E-8	2.30E-4

* 2.38E-4 for 0-30 minute fumigation period.

EAB / LPZ χ/Q, sec/m^3

Period	RB	Elv Stack
0-2 hrs (EAB)	5.57E-4	6.95E-6*
0-8 hrs (LPZ)	6.43E-5	3.58E-6
8-24 hrs	4.46E-5	2.61E-6
1-4 days	2.01E-5	1.32E-6
4-30 days	6.43E-6	4.99E-7

* For 0-30 minute fumigation period: 7.03E-5 @EAB, 3.15E-5 @LPZ.

Assumptions for LOCA Analyses

Core Inventory	Calculated by RADTRAD
Onset of Gap Release Phase, min	2.0

Core release fractions and timing–CNMT atmosphere

Duration, hrs	0.5000E+00	0.1500E+01
Noble Gases:	0.5000E-01	0.9500E+00
Iodine:	0.5000E-01	0.2500E+00
Cesium:	0.5000E-01	0.2000E+00
Tellurium:	0.0000E+00	0.5000E-01
Strontium:	0.0000E+00	0.2000E-01
Barium:	0.0000E+00	0.2000E-01
Ruthenium:	0.0000E+00	0.2500E-02
Cerium:	0.0000E+00	0.5000E-03
Lanthanum:	0.0000E+00	0.2000E-03

Core release fractions and timing–ECCS leakage

Duration, hrs	0.5000E+00	0.1500E+01
Iodine:	0.5000E-01	0.2500E+00
Cesium:	0.5000E-01	0.2000E+00

Iodine species fraction	Atmosphere	Sump
Particulate/aerosol	95	0
Elemental	4.85	100
Organic	0.15	0

Drywell volume, ft^3	130,000
Containment release, %/day	2
After 24 hours	1
MSIV Leakage (total) scfh	
0-24 hours	200
24-720 hours	100
Duration of release, days	30
Reactor Building Positive Pressure, minutes	5
Drywell Natural Deposition	10% Powers Model
Main Steamline Deposition Model	Brockmann-Bixler
Pressure (0-30 days), atm	3.11
Temperature (0-30 days), degreesF	550
Condenser Deposition Elemental & Particulate efficiency, %	
Main Steam Stop Drain Path	99.6
MSIV Drain Path	99.7
SBGT Filter Efficiency, all species, percent	99
ECCS leak rate, gpm (includes 2x multiplier)	1.5
Duration of release, days	30
Control Room Isolation, minutes	4
Containment sump volume, gals	58,900

Assumptions for MSLB Analyses

Mass Release	
Steam, lbm	15,000
Liquid, lbm	80,000
Break Isolation Time, sec	10.5
RCS activity	
Equilibrium iodine spike case	0.2 µCi/gm d.e.I-131
Pre-incident iodine spike case	2.0 µCi/gm d.e.I-131

Iodine species release fraction to environment	Atmosphere
Elemental	0.97
Organic	0.03
Control room HVAC switchover, sec	10.5

Assumptions for Control Rod Drop Accident Analyses

Core Inventory	Calculated by RADTRAD
Radial peaking factor	1.46
Fuel Bundles in Core	368
Fuel Rods in Bundle (effective)	87.3
Rods that exceed DNB	1200
Fraction of rods that exceed DNB that experience melt	0.0077
Gap fraction, Noble gas and iodine	0.01
Melt isotopic composition	
Noble gases	1.0
Iodine	0.3
Fraction of core release that enters condenser	
Noble Gases	1.0
Iodine	0.1
Iodine retention in condenser	0.9
Iodine species release fraction to environment	
Elemental	0.97
Organic	0.03
Condenser leakage, %/day	1.0
Release duration, hours	24
Control room isolation, min	30
TSC isolation, min	30
Control room and TSC unfiltered inleakage, cfm	1000

TABLE 2

RADIOLOGICAL ANALYSIS RESULTS[1], REM TEDE

Event	0-2 hr EAB	30-day LPZ	30-day CR	30-day TSC
Loss-of-Coolant Accident	0.25	0.6	4.2	4.4
Main Steamline Break				
Equilibrium Activity	0.067	0.019	0.24	(2)
Pre-incident Spike	0.67	0.19	2.6	(2)
Control Rod Drop Accident	0.06	0.04	(3)	(3)

Notes

1. Determined by NMC and confirmed by staff.

2. Not calculated. Control room doses are limiting based on location of release in relation to control room and TSC intakes.

3 Not calculated. Bounded by analysis for MSLB.