May 29, 2008

Mr. Keith J. Polson
Vice President Nine Mile Point
Nine Mile Point Nuclear Station, LLC
P. O. Box 63
Lycoming, NY 13093

SUBJECT: NINE MILE POINT NUCLEAR STATION, UNIT NO. 2 - ISSUANCE OF AMENDMENT RE: IMPLEMENTATION OF ALTERNATIVE RADIOLOGICAL SOURCE TERM (TAC NO. MD5758)

Dear Mr. Polson:

The Commission has issued the enclosed Amendment No. 125 to Renewed Facility Operating License No. NPF-69 for the Nine Mile Point Nuclear Station, Unit No. 2 (NMP-2). The amendment consists of changes to the License and to the Technical Specifications (TSs) in response to your application transmitted by letter dated May 31, 2007, as supplemented by letter dated January 7, 2008.

The amendment changes the NMP-2 TSs by revising the accident source term in the design basis radiological consequence analyses in accordance with Title 10 of the *Code of Federal Regulations* (10 CFR) Section 50.67, which requires licensees who seek to revise their accident source term to apply for a license amendment under 10 CFR 50.90. The revised accident source term replaces the methodology that is based on Technical Information Document (TID)-14844, "Calculation of Distance Factors for Power and Test Reactor Sites," with the alternative source term methodology described in Regulatory Guide 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors," with the exception that TID-14844 will continue to be used as the radiation dose basis for equipment qualification and vital area access.

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

Sincerely,

/RA/

Richard V. Guzman, Senior Project Manager
Plant Licensing Branch I-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-410

Enclosures:
1. Amendment No. 125 to NPF-69
2. Safety Evaluation

cc w/encls: See next page

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cc w/encls: See next page

DISTRIBUTION:
(See next page)

Package No.: ML
Amendment No.: ML081230439
Tech Spec No.: ML081560633

* SE provided by memo. No substantial changes made.

OFFICE	LPLI-1/PM	LPLI-1/LA	CSGB/BC	SCVB/BC	AADB/BC
NAME	RGuzman	SLittle	AHiser*	RDennig*	RTaylor*
DATE	5/15/08	5/15/08	8/9/07 SE DTD	3/4/08 SE DTD	3/7/08 SE DTD
OFFICE	RERB/BC	ITSB/BC	OGC	LPLI-1/BC	
NAME	EBenner	RElliott	MSimon	MKowal	
DATE	5/12/08	5/12/08	5/27/08	5/29/08	

OFFICIAL RECORD COPY

DATED: May 29, 2008

AMENDMENT NO. 125 TO RENEWED FACILITY OPERATING LICENSE NO. NPF-69 NINE MILE POINT, UNIT NO. 2

PUBLIC	
LPLI-1	
MKowal	RidsNrrDorlLpl-1
SLittle	RidsNrrLASLittle
RGuzman	RidsNrrPMRGuzman
BSingal	RidsNrrDpr
OGC	RidsOgcMailCenter
GHill (2)	
RDennig	RidsNrrDssScvb
RTaylor	RidsNrrDraAadb
AHiser	RidsNrrDciCsgb
TKobetz	RidsNrrDirsItsb
ACRS	RidsNrrAcrsAcnw&mMailCenter
GDentel, RI	RidsRgn1MailCenter

cc: Plant Service list

Nine Mile Point Nuclear Station, Unit No. 2

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NINE MILE POINT NUCLEAR STATION, LLC (NMPNS)

DOCKET NO. 50-410

NINE MILE POINT NUCLEAR STATION, UNIT NO. 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 125
Renewed License No. NPF-69

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

   A. The application for amendment by Nine Mile Point Nuclear Station, LLC (the licensee) dated May 31, 2007, as supplemented on January 7, 2008, complies with the standards and requirements of the Atomic Energy Act of 1954, 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 10 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 Renewed Facility Operating License No. NPF-69 is hereby amended to read as follows:

(2) Technical Specifications

The Technical Specifications contained in Appendix A and the Environmental Protection Plan contained in Appendix B, both of which are attached hereto, as revised through Amendment No. 125, are hereby incorporated into this license. Nine Mile Point Nuclear Station, LLC shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. This license amendment is effective as of the date of its issuance and shall be implemented within 120 days.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA/

Mark G. Kowal, Chief
Plant Licensing Branch I-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment:
Changes to the License and Technical Specifications

Date of Issuance: May 29,2008

ATTACHMENT TO LICENSE AMENDMENT NO. 125

TO RENEWED FACILITY OPERATING LICENSE NO. NPF-69

DOCKET NO. 50-410

Replace the following page of the Renewed Facility Operating License with the attached revised page. The revised page is identified by amendment number and contains marginal lines indicating the areas of change.

Remove Page	Insert Page
4	4

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

Remove Pages	Insert Pages
1.1-2	1.1-2
1.1-3	1.1-3
3.1.7-1	3.1.7-1
3.3.7.1-4	3.3.7.1-4
3.7.2-1	3.7.2-1
3.7.2-2	3.7.2-2
3.7.2-3	3.7.2-3
3.7.3-1	3.7.3-1
3.7.3-2	3.7.3-2
3.7.3-3	3.7.3-3

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 125 TO

RENEWED FACILITY OPERATING LICENSE NO. NPF-69

NINE MILE POINT NUCLEAR STATION, LLC

NINE MILE POINT NUCLEAR STATION, UNIT NO. 2

DOCKET NO. 50-410

1.0 INTRODUCTION

By letter dated May 31, 2007 (Agencywide Documents Access and Management Systems (ADAMS) Accession No. ML071580314), as supplemented by letter dated January 7, 2008 (ADAMS Accession No. ML080140133), Nine Mile Point Nuclear Station (NMPNS), LLC (the licensee) submitted a license amendment request (LAR) for Nine Mile Point, Unit No. 2 (NMP2). The proposed amendment will fully implement the alternative source term (AST) methodology for analyzing design-basis accident (DBA) radiological consequences, thereby replacing the existing accident radiological source term that is described in TID-14844, "Calculation of Distance Factors for Power and Test Reactor Sites," with the exception that TID-14844 will continue to be used as the radiation dose basis for equipment qualification and vital area access. The LAR provided the Technical Specification (TS) changes and DBA radiological consequence analyses associated with a full-scope implementation of an AST, pursuant to Title 10 of the *Code of Federal Regulations* (10 CFR), Section 50.67 (10 CFR 50.67), "Accident source term," and using the guidance described in Regulatory Guide (RG) 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors." In addition, the dose consequences determined in the revised DBA analyses performed for NMPNS's submittal were calculated assuming a core thermal power approximately 15% higher than the currently licensed core thermal power. This was done so that the DBA analyses could accommodate a potential future extended power uprate (EPU).

The supplement dated January 7, 2008, provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the Nuclear Regulatory Commission (NRC) staff's initial proposed no significant hazards consideration determination as published in the *Federal Register* on July 31, 2007 (72 FR 41786).

2.0 REGULATORY EVALUATION

The NRC staff evaluated the licensee's analysis of the radiological consequences of postulated DBAs against the dose criteria specified in 10 CFR Part 50.67. The applicable criteria are 5 rem Total Effective Dose Equivalent (TEDE) in the control room (CR) for the duration of the event, 25 rem TEDE at the exclusion area boundary (EAB) for the worst 2 hours, and 25 rem TEDE at the outer boundary of the low population zone (LPZ) for the duration of the event. The dose acceptance criterion in the Technical Support Center (TSC) is accepted to be 5 rem TEDE for the duration of the accident to show compliance with the regulatory requirements of NUREG-0737, "Clarification of TMI Action Plan Requirements" and Paragraph IV.E.8 of Appendix E to 10 CFR Part 50.

The regulatory requirements upon which the NRC staff based its acceptance are those in General Design Criteria (GDC) 19, and the accident dose criteria in 10 CFR 50.67, as supplemented in Regulatory Position 4.4 and Table 6 of RG 1.183 and Standard Review Plan (SRP) 15.0.1. The licensee has not proposed any deviation or departure from the guidance provided in RG 1.183. The NRC staff's evaluation is based upon the following regulatory codes, guides, and standards, in addition to relevant information in the NMP2 Updated Safety Analysis Report (USAR) and TSs, as well as consideration for any applicable alternative documentation the licensee may have provided:

- 10 CFR Part 50.67, "Accident source term."
- 10 CFR Part 50, Appendix A, "General Design Criteria for Nuclear Power Plants": Criterion 19, "Control room."
- NUREG-0800 SRP Section 2.3.4, "Short-Term Diffusion Estimates for Accidental Atmospheric Releases."
- NUREG-0800 SRP Section 6.4, "Control Room Habitability Systems."
- NUREG-0800 SRP Section 15.0.1, "Radiological Consequence Analyses Using Alternative Source Term."
- NUREG-0737, "Clarification of TMI Action Plan Requirements."
- NUREG-0917, "Nuclear Regulatory Commission NRC staff Computer Programs for Use with Meteorological Data."
- TID-14844, "Calculation of Distance Factors for Power and Test Reactor Sites."
- RG 1.23, "Onsite Meteorological Programs."
- RG 1.145, "Atmospheric Dispersion Models for Potential Accident Consequence Assessments at Nuclear Power Plants."
- RG 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors."
- RG 1.194, "Atmospheric Relative Concentrations for Control Room Radiological Habitability Assessments at Nuclear Power Plants."

- RG 1.197, “Demonstrating Control Room Envelope Integrity at Nuclear Power Reactors.”
- Regulatory Issue Summary 2006-04, “Experience With Implementation of Alternative Source Terms.”
- Technical Specification (TS) Task Force Traveler TSTF-51, Revision 2, “Revise Containment Requirements During Handling Irradiated Fuel and Core Alterations,” (ML040400343) approved by the NRC on October 13, 1999, which provides for the relaxation of some TS requirements during refueling after a sufficient decay period has occurred.
- NUREG-1433 Rev. 3.0 “Standard Technical Specifications General Electric Plants, BWR/4.”
- Review Guideline “Guidance on the Assessment of a BWR SLC System for pH Control”, February 12, 2004 (ML040640364).

3.0 TECHNICAL EVALUATION

The NRC staff reviewed the regulatory and technical analyses performed by the licensee in support of its proposed license amendment, as they relate to the radiological consequences of DBA analyses. Information regarding these analyses was provided in Attachment 7 of the May 31, 2007, submittal (or the LAR). Additionally, analyses performed to determine the new atmospheric dispersion characteristics of NMP2, as they relate to the evaluation of DBA radiological consequences, are provided in Attachment 6 of the submittal. The NRC staff reviewed the assumptions, inputs, and methods used by the licensee to assess the impacts of the proposed license amendment. The NRC staff also performed independent calculations to confirm the conservatism of the licensee’s analyses. However, the findings of this safety evaluation (SE) are based on the descriptions and results of the licensee’s analyses and other supporting information docketed by the licensee.

3.1 Atmospheric Dispersion Estimates

The licensee generated new atmospheric dispersion factors (χ/Q values) for use in evaluating the radiological consequences of four limiting DBAs at onsite and offsite dose locations. NMP2 is located 6 miles northeast of Oswego, NY. The licensee used onsite meteorological data for calendar years 1997 through 2001 as an input to the ARCON96 (NUREG/CR-6331, Revision 1, “Atmospheric Relative Concentrations in Building Wakes”) and PAVAN (NUREG/CR-2858, “PAVAN: An Atmospheric Dispersion Program for Evaluating Design Basis Accidental Releases of Radioactive Materials from Nuclear Power Stations”) atmospheric dispersion computer models to calculate NMP2 χ/Q values for the Loss-of-Coolant Accident (LOCA), Fuel-Handling Accident (FHA), and Control Rod Drop Accident (CRDA). The licensee also used these models to calculate χ/Q values for input to the dose assessment for Nine Mile Point Unit 1 (NMP1) releases on the NMP2 CR, and vice-versa, for these DBAs. In addition, the licensee assumed an instantaneous ground level puff release modeled in accordance with the methodology discussed in RG 1.194 to calculate CR (for both Units 1 and 2) and TSC χ/Q values and the PAVAN computer model to calculate the EAB and outer LPZ boundary χ/Q values for the Main

Steam Line Break (MSLB) Accident analysis. The resulting χ/Q values represent a change from those currently presented in Chapter 15 of the NMP2 USAR.

3.1.1 Meteorological Data

The licensee used 5 consecutive years of onsite hourly meteorological data collected during calendar years 1997 through 2001. These data were applied to generate new ground level CR and TSC χ/Q values and offsite ground level and elevated release χ/Q values for use in the current LAR. The data were provided for staff review in the form of hourly meteorological data files suitable for input into the ARCON96 CR atmospheric dispersion computer code. A joint wind speed, wind direction, and atmospheric stability frequency distribution (joint frequency distribution or JFD) was developed using the 1997 through 2001 data for use in the PAVAN offsite atmospheric dispersion computer code.

The set of meteorological data (1997 through 2001) used in the current LAR atmospheric dispersion analyses was selected from the historical record of the NMPNS meteorological monitoring program based on a review of the data set quality (i.e., completeness and accuracy of the data). Wind speed and wind direction data used in the atmospheric dispersion analyses were measured on the NMPNS onsite primary meteorological tower at heights of 9.4 meters (~ 30 feet) and 60.7 meters (~ 200 feet) above ground level (AGL). Temperature sensors provided atmospheric stability data (via temperature difference) as well. The combined data recovery of the wind speed, wind direction, and atmospheric stability data was in the upper 90th percentile during each year of the full data set for measurement levels of 9.4 meters and 60.7 meters. The NRC staff determined there was an overall data recovery rate of 98.5%. The licensee noted that the data collection process was based on the guidance provided by RG 1.23, Rev. 0.

The NRC staff performed confirmatory and quality assurance evaluations of the meteorological data presented using the methodology described in NUREG-0917. Further review was performed using computer spreadsheets. Assessment of the wind speed and wind direction data showed similar results from year to year. There was an average wind speed of 3.9 meters per second (m/s) and 6.6 m/s at the 9.4 meter and 60.7 meter heights AGL, respectively, for the 5 consecutive years of meteorological data presented. Winds predominantly blew from the southeast direction at the lower measurement level and from the west southwest direction at the upper level during each of the 5 years. Meteorological trends at NMPNS, representative of calendar years 1985 through 2005, show an average ground level (9.1 meters AGL) wind speed of 4.0 m/s with winds generally blowing from the southeast direction and an average upper height (61.0 meters AGL) wind speed of 6.6 m/s with winds generally blowing from the west southwest direction.[1] Thus, the 1997 through 2001 data are consistent with historical data for NMPNS.

Wind direction frequency distributions for each measurement channel were reasonably similar from year to year between both measurement heights. Wind speed frequency distributions also showed similarity from year to year for both measurement levels with the highest occurrence of

[1] According to a presentation delivered at the Nuclear Utility Meteorological Data Users Group (NUMUG) 2005 10th Annual Conference by onsite meteorologists Thomas Galletta and Anthony Fabrizio entitled "Meteorological Trends at Nine Mile Point." Website: http://hps.ne.uiuc.edu/numug/parchive.htm

wind data in the 3 to 5 m/s range (31.9%) at the 9.4 meter level and a broader modal distribution in the 3 to 10 m/s range at the 60.7 meter level.

Regarding atmospheric stability, measured as the temperature difference between the 60.7 meter and 9.4 meter levels, the time of occurrence and duration of reported stability conditions were generally consistent with expected meteorological conditions (e.g., neutral and slightly stable conditions predominated during the year with stable and neutral conditions occurring at night and unstable and neutral conditions occurring during the day). This resulted in unstable conditions (A-C stability classes) occurring 20.7%, neutral and slightly stable conditions (D-E stability classes) occurring 65.4%, and stable conditions (F-G stability classes) occurring 13.9% of the 5-year period. There was a relatively higher occurrence of stable conditions during the day and unstable conditions at night than normally observed at sites having homogenous terrain. However, NMP2 is sited on the shoreline of Lake Ontario. The NRC staff reviewed these apparent stability anomalies and noted a strong seasonal trend that would correspond to typical seasonal differences between lake and land temperatures.

A comparison of the JFD derived by the NRC staff from the 60.7 meter ARCON96 formatted hourly data with the JFD developed by the licensee for input into the PAVAN atmospheric dispersion model showed good agreement. The licensee did not provide a JFD for measurements at the 9.4 meter level. Consequently, the NRC staff requested a justification of using the 60.7 meter JFD for generation of both the elevated and ground level release χ/Q values used in the dose assessment via an NRC staff's request for additional information (RAI) by letter dated November 8, 2007 (ML073060222). As a result, the licensee performed sensitivity calculations which are discussed in more detail in Section 3.1.3, "Offsite Atmospheric Dispersion Factors."

For the reasons noted above, the meteorological data presented for years 1997 through 2001 were found acceptable by staff evaluation and are considered adequate for use in making atmospheric dispersion estimates used in the LOCA, FHA, MSLB, and CRDA dose assessments performed in support of the current LAR.

3.1.2 Onsite Atmospheric Dispersion Factors

The licensee generated new CR and TSC χ/Q values for postulated ground level releases from NMP1 and NMP2 for the LOCA, FHA, MSLB, and CRDA using guidance provided in RG 1.194. Therefore, χ/Q values for the NMP2 CR due to NMP1 releases and χ/Q values for the NMP1 CR due to NMP2 releases were all evaluated. These new atmospheric dispersion estimates were calculated using ARCON96 for the LOCA, FHA, and CRDA. RG 1.194 states that ARCON96 is an acceptable methodology for assessing onsite χ/Q values for use in DBA radiological analyses. The NRC staff evaluated the applicability of the ARCON96 model and determined that there are no unusual siting, building arrangements, release characterization, source-receptor configuration, meteorological regimes, or terrain conditions that preclude use of this model in support of the current LAR for NMP2.

The wind speed, wind direction, and atmospheric stability measured at the 9.4 meter and 60.7 meter heights AGL served as input for the CR χ/Q calculations. Other inputs included the release/source height, the CR and TSC receptor heights, the straight-line distance between the source and intake/receptor, all in meters, the direction between intake to source, in degrees, and modified default values of 0.2 meters for surface roughness, 0.5 m/s for minimum wind speed, and sector averaging constant of 4.3 (found in Table A-2 of RG 1.194). No diffuse area sources were used in the estimated χ/Q analysis for the purpose of dose assessment.

Radioactive releases from the LOCA, FHA, and CRDA events were assumed to discharge to the environment via five different source points: NMP2 main stack (MS2), NMP2 radwaste/reactor building (RB2) vent, NMP2 main steam tunnel, NMP2 standby gas treatment building (SGTB), and NMP2 post accident sampling system (PASS) panel (PP). The release heights for each of these sources are 130.8 meters, 57.0 meters, 13.7 meters, 7.2 meters, and 25.0 meters, respectively. Releases from NMP1 were also considered. There were three different NMP1 source points from which releases were assumed to discharge to the NMP2 CR: NMP1 main stack (MS1), NMP1 reactor building (RB1) blowout panel, and NMP1 turbine building (TB1) blowout panel. The release heights for each of these sources are 106.7 meters, 24.0 meters, and 22.1 meters, respectively. Essentially, all eight releases were assumed to occur at ground level for the purpose of atmospheric dispersion analyses. The MS2 and MS1 release points were each treated as a ground level release pursuant to RG 1.194, which states that the use of stack release mode is appropriate when the release point is greater than 2.5 times the height of the adjacent structure(s). In this case, the adjacent Unit 2 reactor building has a height of 51.55 meters; thus, the difference in their heights is a factor of 2.54 (approximately 2.5). The use of a ground level release for this layout was justified via confirmatory analysis performed by the NRC staff. As typically found, the ground level release scenario resulted in more conservative χ/Q values than those values using elevated release mode. Similarly, MS1 resulted in a factor of 2.07 times the height of its adjacent building. The primary onsite receptors used for onsite atmospheric dispersion evaluations, as noted in Table 3.1.1 and Table 3.1.2, were the NMP1 CR intake, the TSC, NMP2 CR intake - Upper West, NMP2 CR intake - Upper East, NMP2 CR intake - Lower West, and NMP2 CR intake - Lower East.

The licensee used guidance in RG 1.194 to assess the atmospheric dispersion of the MSLB event, assuming a complete instantaneous ground level puff release and transport from the main steam tunnel to the NMP2 CR. Pursuant to RG 1.194, an accident qualifies as an instantaneous release when 100% of the radionuclides are released directly to the environment over a period no longer than about 1 minute. The MSLB release was modeled by a single χ/Q value throughout the event. As recommended in RG 1.194, the licensee assumed a wind speed of 1 m/s toward the receptor and F-stability class.

The NRC staff qualitatively reviewed the inputs to the ARCON96 calculations and found them generally consistent with site configuration drawings, Figure A6-1 in Attachment 6 of the LAR and Attachment 3 to the licensee's January 7, 2008, responses to the NRC staff's RAI, and NRC staff practice. Additionally, the NRC staff performed a random confirmatory analysis of the licensee's assessments of CR post-accident dispersion conditions generated using the 1997 through 2001 meteorological data and the ARCON96 and RG 1.194 puff model. The NRC staff concluded that the resulting χ/Q values generated by the licensee are acceptable for use in the LOCA, FHA, CRDA, and MSLB dose assessments at the NMP2 CR, TSC, and NMP1 CR.

3.1.3 Offsite Atmospheric Dispersion Factors

The licensee used a JFD derived from the 1997 through 2001 wind data measured on the primary meteorological tower at the 60.7 meter elevation height as input to PAVAN's EAB and outer LPZ boundary calculations for all postulated releases. The atmospheric stability class was calculated using the temperature difference between the 60.7 meter and 9.4 meter heights on the primary tower. The licensee did not use the 9.4 meter wind data measurements for ground level releases. In NMP2's RAI response, the licensee stated that the methodology outlined in PAVAN's equation 1 was properly implemented in the offsite χ/Q calculations and was used to adjust the wind speed data accordingly. The NRC staff acknowledges that PAVAN extrapolates winds to other heights, but uses generic assumptions that may not always closely approximate actual measurements at a site. The licensee performed calculations that determined the 60.7 meter wind data resulted in conservative χ/Q values in comparison to the 9.4 meter wind data input for the 0-2 hour EAB and the 0-24 hour outer LPZ boundary time periods. To address the non-conservative χ/Q values for the 1-30 days outer LPZ boundary time periods, using 60.7 meters wind data, the licensee performed sensitivity calculations using their LOCA models, and determined the doses were essentially bounded by those using the 60.7 meter wind data. Thus, the licensee concluded that the use of the 60.7 meter wind data as input to PAVAN for estimating offsite χ/Q values is acceptable. The NRC staff also made comparative calculations using the 60.7 meter and 9.4 meter wind data and found similar results as the sensitivity analyses performed by the licensee.

In the PAVAN analysis, the licensee assumed 12 classes of wind speed in its evaluation ranging from 0.44 m/s to 60.5 m/s (0.44, 1.0, 2.0, 3.0, 4.0, 5.0, 6.0, 8.0, 10.0, 13.0, 18.0, 60.5 m/s). NRC Regulatory Issue Summary 2006-04, "Experience with Implementation of Alternative Source Terms," suggests that the JFD used as input to PAVAN should have a large number of lower wind speed categories in order to produce the best results for offsite atmospheric dispersion analysis. This guidance suggests that the PAVAN user generates JFDs for the following wind speed categories: calm, 0.5, 0.75, 1.0, 1.25, 1.5, 2.0, 3.0, 4.0, 5.0, 6.0, 8.0 and 10.0 m/s. Primarily, these categories are recommended to generate more χ/Q estimates at the lower values of the cumulative frequency since the 0.5% value is required for proper enveloping of data points as outlined in NUREG/CR-2858, Section 4.6, "Subroutine ENVLOP." Thus, the NRC staff conducted comparative PAVAN calculations using the 9.4 meter data and the suggested wind speed categories which resulted in some χ/Q values being slightly higher than those presented by the licensee. However, the licensee's χ/Q values used in the applicable dose assessment generated values that met dose criteria pursuant to 10 CFR 50.67. Considering the licensee's compliance with dose limits and only a slight differential margin from the NRC staff's comparative χ/Q values, these estimates were found acceptable for use in the dose assessment for the current LAR.

Page 4 of Attachment 6 to the LAR stated the distances to the NMP2 EAB and outer boundary of the LPZ in the coastal sectors were not considered in determining the direction-dependent χ/Q values. The EAB distances in the over water coastal sectors were not clearly defined and in the licensee's analysis, the minimum land sector EAB distance of 1555 meters from the MS2 and 1381 meters from RB2 for the non-coastal χ/Q values were used to obtain the bounding atmospheric dispersion factors. The distances to the EAB for the coastal sectors, defined as the W, WNW, NW, NNW, N, NNE, NE, and ENE directions, were replaced with these (i.e., 1381 meters at ground level and 1555 meters at elevated level) minimum non-coastal distances.

Per 10 CFR 100.3, “Definitions,” the licensee has the authority to determine all activities including exclusion or removal of personnel and property from within the exclusion area. In the NRC staff’s RAI to the licensee, the NRC staff attempted to better determine the licensee’s ability to exclude or remove personnel and property in the coastal sectors. In its response dated January 7, 2008, the licensee stated that boating activity is prohibited out to 700 meters from the centerline of the NMP2 reactor in accordance with 33 CFR 165.911, “Security Zones; Captain of the Port Buffalo Zone.” Additionally, the licensee states that the methodology of excluding coastal sectors in evaluating EAB and outer boundary of LPZ $^{\chi}/_{Q}$ values was approved for NMP2 in NUREG-1047, “Safety Evaluation Report Related to the Operation of Nine Mile Point Nuclear,” including Supplements 1 through 6. The NRC staff notes that the rationale for this acceptance is not clearly specified in these documents, but can be inferred by reference to the NMP2 USAR. The NRC staff reviewed the Oswego County Radiological Emergency Preparedness Plan, Chapter C.8(a)(4), and found that one protective action for an emergency response of NMPNS is to clear all water emergency response planning areas (ERPAs) of all commercial and recreational boat traffic. These ERPAs, noted as 26, 27, 28, and 29 on the site schematic of the NMPNS Radiological Emergency Plans and Procedures document, extend out to 10 miles from the NMPNS (i.e., plume exposure pathway EPZ). This emergency action is implemented in the event of classifying an Alert, Site Area Emergency, or General Emergency.

To further resolve concerns with the distances used in the analysis for the coastal sectors, the NRC staff performed comparative calculations to evaluate $^{\chi}/_{Q}$ values at a distance of 700 meters (the distance to which boating activity is prohibited from the centerline of the NMP2 reactor). The NRC staff found that the resultant $^{\chi}/_{Q}$ values did not result in a significant change in dose estimates for this specific dose assessment when compared with the values generated using the 1555 meter or 1381 meter distance. The licensee considered overall site ground level EAB (1381 meters) and outer boundary of LPZ (6116 meters overall) $^{\chi}/_{Q}$ values for the land sectors as bounding for the DBAs (LOCA, FHA, MSLB, and CRDA) using the PAVAN offsite atmospheric dispersion computer code. In the elevated offsite analysis, EAB (1555 meters) short-term fumigation $^{\chi}/_{Q}$ value for the east-southeast sector, LPZ 0-8 hours short-term fumigation $^{\chi}/_{Q}$ value in the south sector, and sector dependent non-fumigation $^{\chi}/_{Q}$ values in the southwest (LPZ 8-24 hours and LPZ 24-96 hours) and east (LPZ 96-720 hours) sectors were determined most limiting using PAVAN.

The licensee’s offsite $^{\chi}/_{Q}$ values, listed in Table 3.1.3, represent a change from those used in the current licensing basis. For the reasons noted above (e.g., the comparative licensee and NRC staff calculations and meeting of the dose criteria), the NRC staff finds these $^{\chi}/_{Q}$ values acceptable for use in the analysis of the postulated DBAs and their associated dose estimates performed for the current LAR.

3.1.4 Secondary Containment Drawdown – Meteorology

The licensee used a previously approved secondary containment drawdown analysis for NMP2. According to RG 1.183, the effect of high winds on the ability of the secondary containment to maintain a negative pressure should be evaluated on an individual case basis. In this particular case, wind speed was not considered as a factor in the secondary containment drawdown analysis. However, a conservative bounding outdoor temperature of -20 ˚F and indoor temperature of 105 ˚F were used. Guidance found in RG 1.183 states that ambient temperatures used in these assessments should be the 1-hour average value that is exceeded only 5% or 95% of the total numbers of hours in the data set, whichever is considered conservative for the intended use (e.g., if high temperatures are limiting, use those exceeded only 5%). Thus, this analysis was found acceptable by NRC staff as noted in the approval for License Amendment No. 56, which was issued by letter dated August 30, 1994 (ML011130368). This secondary containment drawdown analysis is further described in Section 6.2.3 of NMP2 USAR.

3.2 Radiological Consequences of Design-Basis Accidents

To support the proposed implementation of an AST, the licensee analyzed the radiological dose consequences and provided all major inputs and assumptions for the following DBAs:

- LOCA
- FHA
- MSLBA
- CRDA

As a minimum effort to revise the NMP2 licensing basis to incorporate a full implementation of the AST, RG 1.183 Position 1.2.1 specifies that the DBA LOCA must be reanalyzed using the appropriate guidance therein. In accordance with this RG 1.183 guidance, the licensee re-analyzed the four DBAs listed above, which includes the design-basis LOCA at NMP2.

The licensee's submittal reports the results of the radiological consequence analyses for the above DBAs to show compliance with 10 CFR 50.67 dose acceptance criteria, or fractions thereof, as defined in SRP 15.0.1 and RG 1.183, for doses offsite and in the control room. For full implementation of the AST DBA analysis methodology, the dose acceptance criteria specified in 10 CFR 50.67 provides an alternative to the previous whole body and thyroid dose guidelines stated in 10 CFR 100.11, "Determination of exclusion area, low population zone, and population center distance," and GDC 19. The subject LAR is considered a full implementation of the AST.

RG 1.183, Regulatory Position 3.1, "Fission Product Inventory," states that:

> The inventory of fission products in the reactor core and available for release to the containment should be based on the maximum full power operation of the core with, as a minimum, current licensed values for fuel enrichment, fuel burnup, and an assumed core power equal to the current licensed rated thermal power times the Emergency Core Cooling System (ECCS) evaluation uncertainty. The period of irradiation should be of sufficient duration to allow the activity of dose-significant radionuclides to reach equilibrium or to reach maximum values. The core inventory should be determined using an appropriate isotope generation and depletion computer code such as ORIGEN2 or ORIGEN-ARP.

For accident analyses postulating fuel damage, and in accordance with the guidance of RG 1.183, the licensee calculated the core isotopic inventory available for release using the ORIGEN2 isotope generation and depletion computer code, and then multiplied the isotopic specific activities by the relevant power level and release fractions. However, because the licensee is interested in performing an EPU on NMP2 at some point in the future, the licensee has assumed a core thermal power approximately 15% higher than that which is currently licensed. The licensee has accounted for ECCS uncertainty by adding 2% to this potential EPU power as well. The NRC staff finds the licensee's implementation of the cited isotope generation and depletion computer code to be acceptable for establishing the core inventory for AST accident analyses.

As stated in RG 1.183, the release fractions associated with the light-water reactor (LWR) core inventory released into containment for the design-basis LOCA and non-LOCA events have been determined to be acceptable for use with currently approved LWR fuel with a peak burnup of 62,000 megawatt days per metric ton of uranium (MWd/MTU), provided that the maximum linear heat generation rate does not exceed a 6.3 kilowatt per foot (kw/ft) peak rod average power for burnups exceeding 54,000 MWd/MTU. The licensee states that all NMP2 fuel conforms to these criteria.

To perform independent confirmatory dose calculations for the DBAs, the NRC staff used the NRC-sponsored radiological consequence computer code, "RADTRAD: Simplified Model for RADionuclide Transport and Removal And Dose Estimation," Version 3.03, as described in NUREG/CR-6604. The RADTRAD code, developed by the Sandia National Laboratories for the NRC, estimates transport and removal of radionuclides and the resulting radiological consequences at selected receptors.

The following sections discuss the NRC staff's review of the DBA dose assessment performed by the licensee to support the LAR submittal of May 31, 2007, including supporting supplements.

3.2.1 LOCA

The current NMP2 design-basis LOCA analysis is based on the traditional accident source term described in TID-14844. The current licensing basis radiological consequence analysis for the postulated LOCA is provided in the NMP2 USAR Section 15.6.5, "Loss-of-Coolant Accident." To support implementation of the AST, as requested by the subject LAR, the licensee reanalyzed the offsite and control room radiological consequences of the postulated LOCA. This reanalysis was performed to demonstrate that the engineered safety features (ESFs) designed to mitigate the radiological consequences at NMP2 will remain adequate following implementation of the AST.

The licensee submitted the AST-based reanalysis of the LOCA as an attachment to the LAR. Included in this reanalysis are the assumptions, parameters, and newly calculated offsite and control room doses associated with implementing the AST methodology. The licensee cites RG 1.183 as providing the primary radiological analysis assumptions for their reanalysis of the postulated design-basis LOCA. Specifically, the NRC staff's guidance for analyses of the LOCA is detailed in Appendix A of RG 1.183.

3.2.1.1 Activity Source

For the design-basis LOCA analysis, the licensee assumed that the core isotopic inventory available for release into the containment, is based on maximum full power operation of the core at 4,067 megawatts thermal (MWth), or 1.02 times a 15% uprate of the current licensed thermal power level of 3,467 MWth, in order to account for the ECCS evaluation uncertainty. Additionally, the burnup and enrichment parameters assumed when determining the core isotopic inventory are within current licensed limits for fuel at NMP2. The licensee assumed a 24-month cycle at 1400 effective full-power days (EFPD) per cycle and a 4.1% average enrichment.

The core inventory release fractions and release timing for the gap and early in-vessel release phases of the DBA LOCA were taken from RG 1.183, Tables 1 and 4, respectively. Also consistent with RG 1.183 guidance, the licensee assumes that the speciation of radioactive iodine released from failed fuel is 95% aerosol (particulate), 4.85% elemental, and 0.15% organic. The speciation of radioactive iodine for coolant releases, such as from ESFs, is 97% elemental and 3% organic.

3.2.1.2 Transport Methodology and Assumptions

The licensee calculated the onsite and offsite dose consequences of the design-basis LOCA by modeling the transport of activity released from the core to the environment, while accounting for appropriate activity dilution, holdup, and removal mechanisms. The NRC staff has reviewed the licensee's assessment of the following potential post-LOCA activity release pathways:

- Primary Containment (PC) Leakage to Secondary Containment
    - Traversing In-core Probe (TIP) Leakage from the PC to Secondary Containment
- PC Leakage Bypassing Secondary Containment
    - Main Steam Isolation Valve (MSIV) Leakage Pathway
    - Other Leakage Pathways
- Engineered Safety Feature (ESF) Leakage
- PC Purge Leakage

Also, the NRC staff has reviewed the licensee's assessment of the following potential post-LOCA shine dose pathways:

- Reactor Building (RB) Cloud Shine to the Control Room
- External Plume Shine to the Control Room
- Control Room Filter Shine to the Control Room

Consistent with regulatory requirements, the licensee assumed a loss of offsite power (LOOP) concurrent with the design-basis LOCA. Subsequently, the licensee has assumed that, as a worst case, the single failure of an emergency diesel generator (EDG) delays the startup of ECCS for 2 hours after the onset of gap release. Additionally, to conservatively limit credit for deposition of activity in piping, the licensee assumed that an outboard MSIV fails in the stuck open position, as discussed in Section 3.2.1.2.5.2 of this SE.

For releases into containment, the licensee assumed that activity released from the reactor coolant system is well-mixed between the drywell and the suppression chamber airspace volumes 2 hours after the onset of gap release following the restoration of ECCS, which is postulated to occur 122 minutes after accident initiation. Before this time, the releases are only mixed in the drywell airspace. The licensee assumed that, at the time the ECCS is restored, the thermohydraulic response of cooling water quenching the molten core and core debris in the PC will result in the drywell and suppression chamber airspace volumes becoming well-mixed. This assumption is acceptable for the Mark II containment design of NMP2, as it is configured with downcomers from the drywell that extend below the surface of the suppression pool coolant (wetwell). The licensee's assumption of a one wetwell volume per minute rate of mixing is also acceptable. The licensee takes no credit for the activity decontamination, or scrubbing, associated with such activity releases into the suppression chamber fluid.

By crediting the NMP2 Standby Liquid Control (SLC) System capability to introduce sodium pentaborate to act as a buffer into the reactor coolant, the licensee has determined that the suppression pool pH remains above 7.0 for the duration of the accident. Therefore, in analyzing activity transport from containment, it was unnecessary for the licensee to consider re-evolution of iodine dissolved in the coolant. The licensee's analysis of post-LOCA suppression pool pH was reviewed by the NRC staff as discussed next.

NUREG-1465, "Accident Source Terms for Light-Water Nuclear Power Plants," specifies that the iodine entering the containment after a LOCA is composed of at least 95% cesium iodide (CsI) with the remaining 5% as iodine (I) and hydriodic acid (HI), with not less than 1% of each as I and HI. Once the iodine enters containment, however, additional reactions are likely to occur. The iodine is expected to dissolve in the suppression pool water or plate out on wet surfaces in ionic form. The subsequent iodine behavior within the suppression pool will depend to a great extent on the pH of the suppression pool water, which will change with time. This change is caused by the addition of basic and acidic chemicals generated in the core or formed in the containment. The basic chemical consists of cesium hydroxide (CsOH) which is formed from the cesium released from the damaged core. Its presence causes an increase in the suppression pool water pH. The acidic chemicals generated inside the containment consist of hydrochloric and nitric acids and to a smaller extent hydriodic acid. Hydrochloric acid is generated from decomposition of cable insulation containing chlorine, and nitric acid is formed in the radiation environment existing in the post-LOCA containment.

The licensee calculated the amounts of these chemicals using the methodology described in NUREG/CR-5950, "Iodine Evolution and pH Control." The licensee's post-LOCA suppression pool pH calculations are provided in Attachment 7 of its application, "Calculation H21C-097." Since the concentration of acidic chemicals in the suppression pool increases with time, after an initial period when pH is controlled by cesium hydroxide, the pH of the suppression pool water will be continuously decreasing, reaching it lowest value at 30 days after a LOCA. In the unbuffered suppression pool water of the Nine Mile Point Unit 2 plant, the minimum pH will be less than 7.0. Therefore, if no preventive action is taken, formation of molecular iodine will occur, causing an increase of radiation doses. The preventive action taken by the licensee is to buffer the suppression pool water using sodium pentaborate from the standby liquid control system (SLCS). The primary function of the SLCS is to scram the reactor after a control rod failure or during an anticipated transient without scram (ATWS) event. However, since sodium pentaborate in the SLCS is derived from a strong base and a weak acid, it has buffering properties and could be used as a pH controlling agent in the suppression pool water. The licensee has demonstrated that by using this buffer the pH of the suppression pool could be kept above 7.0 for 720 hours (30 days) after a LOCA.

The licensee calculated the pH of the suppression pool water without the addition of sodium pentaborate buffer. This pH initially rises due to the influence of cesium hydroxide addition at the beginning of the LOCA, and then falls to below a pH of 7.0 between approximately 12 to 14 days after a LOCA. The final pH at 30 days without buffering was calculated to be 4.4. Addition of sodium pentaborate via the SLC system buffers the suppression pool and results in a final pH at 30 days of 8.3. Therefore, the suppression pool pH will satisfy the acceptance criterion of a pH greater than 7.0 with the use of the SLC system.

The NRC staff reviewed the calculations and technical evaluations presented by the licensee and finds them acceptable. The staff reviewed the analyses and justifications provided by the licensee and concludes that the licensee's proposed actions will maintain the suppression pool pH at levels of at least 7.0 for 30 days following a LOCA. By maintaining the suppression pool pH above 7.0, the fraction of radioactive iodine released into containment atmosphere following a LOCA will be minimized.

The following sections detail the NRC staff's review of the licensee's analysis of the post-accident activity release paths and contributors to both CR and offsite dose, as mentioned above.

3.2.1.2.1 PC Leakage to Secondary Containment

The NMP2 current design basis containment leak rate (L_a) of 1.10% weight per day (% per day) at containment peak pressure, as reflected in the leak rate limit in NMP2 TS 5.5.12.c, is assumed in the AST LOCA re-analysis. This pathway was modeled by the licensee as the leakage from the PC that occurs prior to, and after, a sustained negative pressure in the RB is established at 60 minutes after the initiation of the LOCA. This 60 minute time is referred to as the drawdown period. Therefore, the licensee assumed that all activity released from the PC prior to the 60-minute drawdown period is released directly to the environment unfiltered, and at an assumed ground level. Also prior to the completion of drawdown, no credit is taken for holdup in or filtration by the secondary containment/reactor building.

Consistent with RG 1.183 guidance, the licensee assumed that the release from the core enters the drywell only. Only after the end of the release, 122 minutes (i.e., 2.0333 hours) following accident initiation, did the licensee assume that this activity is mixed within the entire PC. Credit for activity removal in the drywell is taken from the time that sprays are initiated at 20 minutes into the accident. This timing represents the acceptable current licensing basis of NMP2. The licensee calculated removal by sprays in the manner discussed in Section 3.2.1.2.5.1.

In their analysis, the licensee assumed a drawdown time of 60 minutes from the start of the DBA-LOCA. The analysis that determined this drawdown time is described in NMP2 USAR Section 6.2.3. This analysis was accepted by the NRC staff as part of License Amendment No. 56, dated August 30, 1994 (ML011130368). The 60-minute time is conservative as it does not account for the operation of both Standby Gas Treatment System (SGTS) trains being in operation.

As stated above, the release rate during drawdown corresponds to the design basis PC leak rate of 1.10% per day from the drywell and suppression chamber airspace (after 122 minutes). After 24 hours this PC leak is reduced by a factor of 2 to 0.55% per day, based on a reduction in containment pressure, as is consistent with allowances in the guidance of RG 1.183. Activity reaching the environment through this pathway, following drawdown, is filtered by the RB SGTS and released via the main stack as an elevated release. The filtration efficiencies associated with this system are 99% for elemental iodine, organic iodine, and particulate forms of radionuclide activity. PC leakage following drawdown is assumed to occur as an elevated release to the environment via the main stack.

3.2.1.2.1.1 TIP Leakage from the PC to Secondary Containment

The inclusion of this activity leakage pathway, which contributes to the total PC leakage to the secondary containment, is consistent with the current licensing basis analysis, as described in USAR Section 15.6.5. As described by the licensee, this pathway was modeled assuming a TIP is inserted when the LOCA occurs and the guide tube fails. Then, the TIP fails to withdraw, and the shear valve fails to close. The resulting leak rate from the PC to the RB was assumed to be 0.12% per day, which was added to the TS containment leakage of 1.10% per day. Like the TS L_a leak rate discussed above, this leakage rate was also reduced by a factor of 2 after 24 hours.

The licensee justified the factor of 2 reductions by citing that the calculated containment pressure has dropped from a peak of 39.75 psig to 5 psig by 24 hours. The NRC staff believes that this approximate factor of 8 decrease in pressure is sufficient to justify the leak rate reduction taken by the licensee.

3.2.1.2.2 PC Leakage Bypassing Secondary Containment

The licensee models this pathway as leakage to the environment from systems characterized by lines that penetrate the PC and the RB. It is postulated that leakage from the PC through the closed containment isolation valves (CIVs) in these penetrations would bypass the RB and SGTS filters, thereby resulting in an unfiltered ground-level release. As a design basis, this release pathway includes leakage through main steam isolation valves (MSIV), as well as the combined leakage through the piping of the main steam drains, reactor water cleanup system, feedwater system, drywell floor and equipment drains and vents, post-accident sampling system (PASS), instrument air and nitrogen supply system, and PC purge system. The licensee divided these bypass pathways into the following three (3) groups, excluding the main steam lines (MSLs):

- Group 1: Bypass from the drywell, delays neglected (all bypass pathways originating in the drywell except those listed below in the third group)
- Group 2: Bypass from the wetwell, delays neglected
- Group 3: Bypass from the drywell (feedwater, 14" containment purge, and reactor water cleanup (RWCU)), delays considered and conservatively combined

To simplify the modeling of 25 lines (excluding the 4 MSLs) that characterize the three defined groups, the licensee conservatively combined these bypass pathways. The method the licensee used to combine these pathways was based on a normalization of χ/Q's. The licensee identified four release points: the main steam tunnel, the SGTS Building, the PASS panel, and the radwaste/reactor building vent. Each of these release points is characterized by specific χ/Q values, as discussed in Section 3.1 above. The licensee normalized each of these χ/Q values to that of the main steam tunnel χ/Q. Then, the appropriate resulting ratio was applied to the effective leakage rates calculated for each of the bypass pathways. (Note that the NMP2 TS leakage rate limit for each of these bypass pathways was used as a design basis for this accident release model; however, effective leakage rates were calculated to reflect activity deposition in the lines, as discussed in Section 3.2.1.2.5.2 of this SE). Following the normalization, the resulting effective leakage rates were combined, and only the main steam tunnel $^{\chi}/_{Q}$ needed to be applied for dose calculation. The licensee performed this normalization consistent with first principles of mathematics and made conservative simplifications in comparing averaged $^{\chi}/_{Q}$ values. Therefore, the NRC staff finds that combining the calculated leakage values is conservative and acceptable.

For MSIV leakage, a total leakage of 96 standard cubic feet per hour (scfh), with a maximum of 24 scfh per MSL, was assumed. These values are consistent with the MSIV leakage surveillance requirements of NMP2 TS 3.6.1.3. These leakage rates were also reduced to reflect deposition, which was calculated as discussed in Section 3.2.1.2.5.2, and adjusted to account for accident temperature and pressure conditions.

The current NMP2 licensing basis provides for mitigation of the permanent secondary containment bypass by an assumed release delay, directly associated with the time it takes the PC leakage to transit the bypass pathway. The licensee has brought forward the applicable delay times from the current licensing basis of NMP2.

The NRC staff does not generally agree with the use of a delayed release assumption in concert with a well-mixed leakage model. However, because the current licensing basis of NMP2 specifies a release delay, and because the likely post-LOCA bypass piping leakage conditions will be a combination of plug flow and well-mixed flow, the NRC staff accepts the licensee's hybridized model.

3.2.1.2.3 ESF Leakage

For this pathway, the licensee modeled ESF activity leakage as a continuous 62 gallon per minute (gpm), or 8.29 cfm, volumetric flow rate from the suppression pool control volume to the RB. The 62 gpm value is equal to 60 gpm, attributed to the assumed failure of two Residual Heat Removal (RHR) system sample lines in the reactor building and allowed leakage past the two isolation valves in an RHR line to the liquid radwaste system, plus a design-basis leakage limit of 1 gpm multiplied by 2 in accordance with the regulatory guidance of RG 1.183. The inclusion of the 60 gpm leakage value is consistent with the current licensing basis analysis described in NMP2 USAR Section 15.6.5. The licensee assumed that 10% of the ESF leakage becomes airborne. A 10% flashing fraction is conservative with respect to applicable guidance. This flashing fraction effectively partitions all particulate radionuclides out of the release. Therefore, all released ESF activity was assumed to be characterized by 97% elemental and 3% organic iodine.

The licensee assumed that, during the 60-minute drawdown period, this activity is released directly to the environment at ground level, without credit for holdup in the RB volume. After the 60-minute drawdown period, the 62 gpm is released to the RB volume, filtered by the SGTS, and discharged to the environment via the main stack. The licensee credited only 50% of the RB volume for mixing of the released ESF leakage. The NRC staff finds the licensee's model of

ESF leakage to be acceptably conservative and consistent with the current licensing basis of NMP2.

3.2.1.2.4 PC Purge Leakage

The licensee assumed a PC purge in the NMP2 pressure control mode to be in progress coincident with the initiation of the design-basis LOCA, consistent with the current licensing basis. For NMP2, the purge is a short release to the environment from the PC via the SGTS filters and main stack. The licensee stated, in their January 7, 2008, RAI response (ML080140133), that the purge through the 2-inch pressure control line of this pathway is terminated within 5 seconds by closure of the PC purge isolation valves. Because this 5-second purge is assumed to occur from the initiation of the accident, it is clear that the purge will cease well before the onset of gap release from the fuel. Therefore, the NRC staff finds the licensee's calculation of a negligible dose contribution from this path to be acceptable.

3.2.1.2.5 Credited Activity Removal Mechanisms

For reduction of airborne activity in the PC, the licensee credits the drywell sprays, but takes no credit for the natural deposition phenomenon. However, for the reduction of activity in the secondary containment bypass pathways, the licensee does credit natural deposition. This credit was limited to only include piping between closed CIVs of the given bypass line. Further, no credit was taken for deposition in the MSL with one MSIV assumed to be stuck open.

The modeling of various mechanisms for radioactive particulate iodine removal, when more than one is used simultaneously for the same activity release in a dose analysis, should consider the effect of one model on the others. Although containment sprays and natural deposition are all acting on the overall in-containment aerosol iodine source term, the total effect from each of these removal mechanisms is not the same as would be found by simply adding the removal coefficients for each model during a given time period. Therefore, a contiguous model should be used by the licensee to address this concern, as it would account for series and parallel effects of each removal process. For the NMP2 design basis, the licensee conservatively limited the credited gravitational settling effects on airborne activity in the bypass piping pathways downstream of the drywell sprays, as discussed in Section 3.2.1.2.5.2 of this SE.

The following subsections discuss the licensee's treatment of and credit taken for each of the aforementioned activity removal mechanisms, as applicable to their design-basis LOCA analysis, and the NRC staff's evaluation of the licensee's model.

3.2.1.2.5.1 Primary Containment Spray Removal

The drywell sprays, which are fed by the RHR pumps, will be manually initiated within 20 minutes following the initiation of the design-basis LOCA, based on existing NMP2 Emergency Operating Procedures. The containment spray pumps transfer coolant from the suppression pool into the drywell airspace. The containment spray system is described in NMP2 USAR Section 6.2.2. The drywell and suppression pool spray systems are designated as safety-related with operability requirements defined in NMP2 TS Sections 3.6.1.6 and 3.6.2.4, respectively, and therefore are acceptable for credit in this DBA scenario.

To calculate particulate removal by the containment sprays, the licensee used the guidance of NUREG-0800, SRP 6.5.2, Rev. 4, and 4 system-related parameters are used as input to calculate the activity removal rate, or λ. The four parameters that are input to the SRP equation are droplet diameter, spray flow rate, spray fall height, and sprayed volume.

For NMP2, the licensee reviewed both Loop A and Loop B of the drywell sprays, and determined that the spray removal by Loop B was the most limiting. The effective spray flow for Loop A is 5519 gpm. For Loop B, the effective spray flow is 5237 gpm. These spray flow values are "effective," because they are corrected for the nozzle flow fraction and elevation above the drywell floor. This correction was calculated by the licensee because not every nozzle receives the same flow or is at the same height. Even with a slightly higher elevation and larger average droplet fall height, the licensee still calculated a lower removal rate, or λ, for Loop B, as the lower effective spray flow is the controlling variable. The calculated λ for the limiting Loop B spray is 19.8 hr^{-1}. This value is conservatively applied to both aerosol and elemental iodine removal; organic iodine removal is ignored. The 19.8 hr^{-1} λ credited for elemental iodine removal is less than the NUREG-0800, SRP 6.5.2 suggested limit of 20 hr^{-1}. Also consistent with the guidance of SRP 6.5.2, the licensee limits the total elemental iodine removal to a DF of 200. In modeling the activity removal in containment credited to the drywell sprays, the licensee conservatively applied the guidance of SRP 6.5.2, and therefore the model and credited activity removal is acceptable to the NRC staff.

3.2.1.2.5.2 Natural Deposition in Secondary Containment Bypass Lines

As mentioned above, the licensee's model for natural deposition was limited to only include the piping of each bypass pathway, as identified in the licensee's LOCA design analysis included as Attachment 7 to the LAR. Only pipe segments between closed CIVs of the given bypass line were credited for natural deposition, and specifically for the MSL with one MSIV assumed to be stuck open, no deposition credit was assumed.

The licensee used the methods described in approved guidance document, AEB 98-03, "Assessment of Radiological Consequences for the Perry Pilot Plant Application Using the Revised (NUREG-1465) Source Term," to determine the extent to which activity is naturally deposited in piping. Consistent with AEB 98-03, the licensee calculated an activity removal efficiency using various plant-specific parameters including the volume of the pipe segment wherein the deposition occurs, the concentration of released airborne activity, and the rate of leakage. The licensee, in turn, used this efficiency to reduce the standard TS limited leakage rate associated with a given bypass pathway to reflect deposition. In addition to the parameters listed, the equations of AEB 98-03 require that a velocity, at which particulate activity will gravitationally settle on the pipe surfaces, be determined. The term "settling," refers to the gravity-driven phenomenon of particulate falling out of a gaseous suspension. A Monte-Carlo analysis, based on parameters that are each defined to be within an empirically determined range, was used in AEB 98-03 to establish a probabilistically characterized range of settling velocities. For the Perry AST pilot plant, for which AEB 98-03 was performed and where drywell sprays were credited, the median (50th percentile) settling velocity value of 0.00117 m/sec was used. However, for additional conservatism, and to address concerns historically documented by the NRC staff, the licensee used the 3rd percentile settling velocity of 0.000066 m/sec. The NRC staff agrees that this 3rd percentile settling velocity value is sufficiently conservative to reflect the effectiveness of drywell spray activity removal in containment upstream of this pathway. Though, as discussed earlier, the LOCA activity leak rates are reduced by a factor of 2 after 24 hours, based on decreasing containment pressure, the licensee conservatively does not credit this reduction to increase removal efficiency by natural deposition in the bypass lines.

For elemental iodine, the licensee assumed that a DF of 2 applies for natural deposition in the bypass piping. This is consistent with the licensee's assumption of elemental iodine plate-out on aerosol particulate, as was used in their drywell spray calculation. The NRC staff notes that the conservatively calculated aerosol activity removal by settling in the piping exceeds a DF of 2, and finds that the DF of 2 for elemental iodine is acceptable. The licensee took no credit for organic iodine removal.

3.2.1.3 Direct Shine Dose

The licensee's evaluation of post-LOCA shine doses to control room personnel from the RB airborne activity cloud, the passing external activity plume, and the activity loaded control room filters was based on the historical NMP2 design basis. The historical external shine doses for NMP2 were calculated using the release characteristics associated with a TID-14844 source term and model based on RG 1.3, "Assumptions for Evaluating the Potential Radiological Consequences of a Loss of Coolant Accident for Boiling Water Reactors." The licensee compared AST-calculated activity releases at NMP2, based on NUREG-1465 and RG 1.183 methodology, to those historically calculated, and showed that the historically calculated values will bound. The NRC staff agrees that, compared to a TID-14844 source term activity release, the AST methodology is more conservative, based on the following three (3) reasons, as documented in the licensee's RAI response of January 7, 2008 (ML080140133):

- The TID-14844 activity releases are instantaneous, whereas NUREG-1465 allows for a release linearly distributed over a period of 2 hours.
- Removal of iodine by sprays and deposition is credible in AST-based models, where no such removal mechanisms were historically applied.
- RG 1.183 allows for a leak rate reduction of 50% at 24 hours, but analyses performed consistent with RG 1.3 take no such reduction.

Though it is agreed that the activity released and available to contribute the NMP2 control room shine dose is bounded by the historical analysis, the NRC staff does note that the historical shine dose calculation implemented the QADMOD point-kernel code. Also, verification of the historically calculated doses was performed using the MicroShield point-kernel code. Both the QADMOD and MicroShield codes are point-kernel integration codes used for general purpose gamma shielding analyses. The potentially complex geometries associated with the direct shine dose assessments, such as those performed for NMP2, are generally more effectively modeled using more powerful particle transport codes. Specifically, MicroShield sacrifices accuracy in lieu of simplicity when modeling complex multidimensional systems of sources, shields, and receivers. However, though it also uses a point-kernel method that implements buildup factors and is subject to mistreatment of albedo effects, QADMOD does allow for the modeling of complex geometries using combinatorial geometry.

It is the NRC staff's judgment that the licensee's direct shine dose model implements sufficient and substantial conservatism that compensates for potential non-conservative treatment of the modeled geometries by the chosen point-kernel codes. Therefore, for the general application of these codes as implemented for design-basis LOCA analysis at NMP2, the NRC staff finds the licensee's direct shine dose assessment acceptable.

3.2.1.4 TSC Dose Consequence Assessment

In the LAR submittal, the licensee indicates that the TSC 30-day inhalation and immersion dose was analyzed, and that two scenarios were considered. The first with the TSC occupied at the initiation of the LOCA, and the other assuming that the TSC is not activated for 1 hour after accident initiation. The licensee states that this was done because the emergency ventilation and filtration system in the TSC is manually initiated by the first person to arrive, and for an off-hours event, actuation could be delayed by up to 1 hour. For calculation of shine dose, the licensee states that the direct shine doses to the TSC were based on a comparison to the AST shine doses for the control room. In their examination of post-LOCA TSC dose consequences, the licensee finds that the 30-day doses do not exceed 5 rem TEDE. The licensee's analyses indicate that they comply with the regulatory requirements for the TSC as given in NUREG-0737 and Paragraph IV.E.8 of Appendix E to 10 CFR Part 50. Therefore, the licensee has sufficiently examined the DBA dose consequences to the TSC to a degree acceptable to the NRC staff.

3.2.1.5 Conclusion

The licensee concluded that the radiological consequences at the EAB, LPZ, and control room are within the dose criteria specified in 10 CFR 50.67 and the accident-specific dose acceptance criteria specified in SRP 15.0.1 and RG 1.183. These accident-specific dose acceptance criteria for the LOCA are a TEDE of 25 rem at the EAB for any 2 hours, 25 rem at the outer boundary of the LPZ for the duration of the accident, and 5 rem for access to, and occupancy of, the control room for the duration of the accident. The NRC staff finds that the licensee used sufficiently conservative analysis assumptions and inputs consistent with applicable regulatory guidance identified in Section 2.0 of this SE and with those stated in the NMP2 USAR as design bases. The NRC staff also performed independent calculations of the dose consequences of the postulated LOCA releases, using the licensee's assumptions for input to the RADTRAD computer code. The NRC staff's calculations confirmed the licensee's dose results. The major parameters and assumptions used by the licensee, and found acceptable to the NRC staff, are presented in Table 3.2.1. The results of the licensee's design-basis radiological consequence calculation are provided in Table 3.2. The EAB, LPZ, and CR doses estimated by the licensee for the LOCA were found to meet the applicable accident dose acceptance criteria and are therefore acceptable.

3.2.2 Fuel-Handling Accident (FHA)

The current NMP2 design basis FHA analysis is based on the traditional accident source term described in TID-14844, "Calculation of Distance Factor for Power and Test Reactor Sites." The NMP2 licensing basis analysis is presented in USAR Section 15.7.4, "Fuel Handling Accident." To support implementation of the AST, as requested by the subject LAR, the licensee reanalyzed the offsite and control room radiological consequences of the postulated FHA. This reanalysis was performed to demonstrate that the ESFs designed to mitigate the radiological consequences at NMP2 will remain adequate after implementation of the AST.

The licensee submitted the AST-based reanalysis of the FHA as part of the LAR. Included in the reanalysis are the assumptions, parameters, and newly calculated offsite and control room doses associated with implementing the AST methodology. The licensee cites RG 1.183 as providing the primary radiological analysis assumptions for their reanalysis of the postulated design-basis FHA. Specifically, the NRC staff's guidance for analysis of the FHA is detailed in Appendix B of RG 1.183.

As analyzed for NMP2, the postulated FHA involves a 32.95-foot drop of a fuel assembly on top of other fuel assemblies in the reactor core during refueling operations. The licensee has determined that the drop distance associated with this location bounds the maximum height that is allowed by the NMP2 refueling equipment configuration and that is the limiting case because it results in the maximum release of fission products to the reactor building. Also, the licensee has determined damage due to a fuel assembly drop over the core into the reactor vessel bounds a drop in the spent fuel pool. All fuel types currently stored in the spent fuel pool are bounded by this analysis.

3.2.2.1 Activity Source

For the design-basis FHA analysis, the licensee assumed that the core isotopic inventory available for release into the containment, is based on maximum full power operation of the core at 4,067 megawatts thermal (MWth), or 1.02 times a 15% uprate of the current licensed thermal power level of 3,467 MWth, in order to account for the ECCS evaluation uncertainty. Additionally, the burnup and enrichment parameters that are assumed when determining the core isotopic inventory are within current licensed limits for fuel at NMP2. The licensee assumed that the assemblies underwent a 24-month cycle at 1400 EFPD per cycle and a 4.1% average enrichment.

A conservative radial peaking factor of 1.80 was applied to the isotopic activity for the damaged fuel assemblies. The fraction of core isotopic activity assumed to be available for release from the gap of failed fuel (i.e., fuel experiencing cladding failure as a result of the drop) is provided in Table 3 of RG 1.183, and was used in the licensee's DBA analysis. These gap fractions are 5% for noble gases and iodines, except for Kr-85 and I-131, where 10% and 8% are assumed, respectively. Consistent with the guidance of RG 1.183, all iodine activity released from the coolant in the reactor vessel was assumed to be of elemental and organic chemical form, in the ratio of 99.85% and 0.15%, respectively. No particulate forms of activity are assumed to be released. All particulate is assumed to be retained by the water in the pool or reactor cavity (i.e., an infinite decontamination factor). Also, the licensee assumed that there is no fuel exceeding the burnup limit assumption expressed in Footnote 11 of RG 1.183. Therefore, the total assumed activity in the fuel gap, and available for release following the postulated FHA drop is found to be acceptable to the NRC staff.

As a design basis, the licensee assumed 24 hours of decay for the accident analysis, corresponding to the time before any movement of fuel can be initiated; the movement of fuel, or fuel handling, before this period would be unanalyzed and not consistent with the assumed design basis. Therefore, activity available for release was calculated to correspond to this post-shutdown decay time. The licensee assumed the 124 fuel pins, 62 for the dropped assembly and 62 for the stuck assembly, will be damaged as a result of the postulated FHA, and will thus instantaneously release all of their available gap activity to the environment, taking no credit for reactor building closure or isolation. This is a conservative treatment, since the RG 1.183 allows for the assumption of a 2-hour release duration. Although NMP2 is currently using GE11 9x9 and GE14 10x10 fuel, the licensee has determined that assuming the postulated FHA with 8x8 fuel is bounding with respect to the fuel failure assumptions. The assumed fuel damage following the DBA is consistent with the current NMP2 licensing basis, as described in NMP2 USAR Section 15.7.4, and the guidance expressed in RG 1.183.

3.2.2.2 Transport Methodology and Assumptions

As analyzed for NMP2, the postulated FHA involves a 32.95-foot drop of a fuel assembly on top of other fuel assemblies in the core, including the weight of the grapple, during refueling operations. Even though the most limiting drop height and subsequent fuel damage is associated with a drop over the core, for conservatism, the licensee assumes the water coverage and DF associated with the spent fuel pool. The minimum water coverage allowed by NMP2 TS 3.9.8 is 22'-3", which is less than the 23' water coverage required to assume an overall DF of 200 in accordance with RG 1.183, Appendix B. So, using the e^{-cd} relationship discussed in the licensee's May 31, 2007 LAR, where d is equal to the water depth and c is a constant equal to -0.2458, the licensee interpolated an adjusted overall DF of 175. This relationship is calculated from the first principles assessment of an inorganic iodine DF of 285 being associated with a water coverage depth of 23 feet, and has previously been found acceptable by the NRC staff. Noble gas activity is assumed to be released from the reactor vessel water without experiencing any reduction. The decontamination factor calculation by the licensee is conservative and acceptable to the NRC staff, as it is consistent with the guidance of RG 1.183.

The licensee assumed that the release to the environment is instantaneous and complete, and that transport to the receptor occurs instantaneously as well. Therefore, no credit is taken for holdup or dilution in the containment volume, and exposure time and the duration of release are equivalent. Also, secondary containment is assumed to be open during fuel handling operations. So, in addition to taking no credit for control room isolation or filtration, the licensee also takes no credit for secondary containment isolation or filtration. In their analysis, the licensee used a spreadsheet to calculate the resulting dose consequences of the design basis FHA. The activity transport model is conservative and, therefore, acceptable to the NRC staff.

3.2.2.3 Conclusion

The licensee concluded that the radiological consequences at the EAB, LPZ, and control room are within the dose criteria specified in 10 CFR 50.67 and the accident-specific dose acceptance criteria specified in SRP 15.0.1 and RG 1.183. These accident-specific dose acceptance criteria for the FHA are a TEDE of 6.3 rem at the EAB for any 2 hours, 6.3 rem at the outer boundary of the LPZ for the duration of the accident, and 5 rem for access to, and occupancy of, the control room for the duration of the accident. The NRC staff finds that the licensee used analysis assumptions and inputs consistent with applicable regulatory guidance identified in Section 2.0 of this SE and with those stated in the NMP2 USAR as design bases. The major parameters and assumptions used by the licensee, and found acceptable to the NRC staff, are presented in Table 3.2.2. The results of the licensee's design-basis radiological consequence calculation are provided in Table 3.2. The EAB, LPZ, and CR doses estimated by the licensee for the FHA accident were found to meet the applicable accident dose acceptance criteria and are, therefore, acceptable.

3.2.3 Main Steam Line Break (MSLB) Accident

The current NMP2 design-basis MSLB accident analysis is based on the traditional accident source term described in TID-14844, "Calculation of Distance Factors for Power and Test Reactor Sites." The NMP2 licensing basis analysis is presented in USAR Section 15.6.4, "Steam System Piping Break Outside Containment." To support implementation of the AST, as requested by the subject LAR, the licensee reanalyzed the offsite and control room radiological consequences of the postulated MSLB. This reanalysis was performed to demonstrate that the ESFs designed to mitigate the radiological consequences at NMP2 will remain adequate after implementation of the AST.

The licensee submitted the AST-based reanalysis of the MSLB accident as part of the LAR. Included in this reanalysis are the assumptions, parameters, and newly calculated offsite and control room doses associated with implementing the AST methodology. The licensee cites RG 1.183 as providing the primary radiological analysis assumptions for their reanalysis of the postulated design-basis MSLB accident. Specifically, the NRC staff's guidance for analyses of the MSLB accident is detailed in Appendix D of RG 1.183.

The licensee has defined the design-basis MSLB accident as an instantaneous circumferential break of one main steam line outside the secondary containment, downstream of the outside isolation valve. It is assumed that pipe end displacement due to this double ended guillotine break is such that the maximum blowdown rate is permitted to occur. The licensee assumed that the break flow is terminated by closure of the MSIVs, and that the coolant mass released through the break includes the line inventory plus the system mass released through the break prior to isolation. The radiological consequences of an MSLB outside secondary containment will bound the consequences of a break inside containment. Thus, only an MSLB outside of containment was considered with regard to the radiological consequences. In addition, this accident is postulated to occur non-mechanistically and without an identified cause in order to evaluate consequences of a hypothetical large steam line rupture.

3.2.3.1 Activity Source

For the design-basis MSLB, the licensee assumed no fuel failure, consistent with the current NMP2 licensing basis, because the core is not postulated to become uncovered. Also, because the accident is assumed to occur at zero power hot standby, the planned future power uprate does not affect dose consequences. To determine the maximum offsite and control room dose, the licensee assumed that a reactor transient, or iodine spike, has occurred prior to the postulated MSLB and has raised the coolant iodine concentration to 20 times the NMP2 TS maximum coolant equilibrium iodine concentration. This is done as a design basis, and in accordance with the guidance of RG 1.183, when no fuel failure is assumed.

The NMP2 TS 3.4.8 maximum coolant equilibrium iodine activity concentration assumed in the MSLB analysis is 0.2 μCi/gm Dose Equivalent (DE) I-131, and the postulated pre-accident iodine spike raises this value to 4.0 μCi/gm DE I-131. Only the spiked activity case was analyzed for dose consequences, because, as shown in Table 3.2, the dose consequence resulting from the spiked activity meets the lower acceptance criterion for the equilibrium activity case that is suggested in Table 6 of RG 1.183, so would therefore meet the higher acceptance criterion for the iodine spike case in RG 1.183 and SRP 15.0.1. Also, because the radiological consequences are directly related to the coolant activity released, and since the equilibrium concentration case has a lower coolant activity release than the iodine spike case, the equilibrium concentration case would meet the equilibrium concentration acceptance criterion.

Consistent with RG 1.183 guidance, the licensee assumed that the speciation of radioactive iodine released from failed fuel is 95% aerosol (particulate), 4.85% elemental, and 0.15% organic. However, the speciation of radioactive iodine released by coolant blowdown is 97% elemental and 3% organic. Because no fuel failure was assumed, the coolant iodine speciation was used for this DBA analysis. The licensee also considered the maximum TS noble gas and cesium activity to be available for release from the steam blowdown and coolant, respectively. This treatment is conservative, with respect to the RG 1.183 guidance, and acceptable to the NRC staff, because the guidance does not explicitly suggest that cesium activity be considered as a dose contributor.

3.2.3.2 Transport Methodology and Assumptions

The licensee has defined the design-basis MSLB accident as an instantaneous circumferential break of one main steam line outside the secondary containment, downstream of the outside isolation valve. It is assumed that pipe end displacement due to this double ended guillotine break is such that the maximum blowdown rate is permitted to occur. The licensee assumed that the break flow is terminated by closure of the MSIVs, and that the coolant mass released through the break includes the line inventory plus the system mass released through the break prior to isolation. The radiological consequences of an MSLB outside secondary containment will bound the consequences of a break inside containment. Thus, only an MSLB outside of containment was considered with regard to the radiological consequences.

Consistent with the current NMP2 licensing basis, the licensee assumed break isolation in 5.5 seconds, corresponding to the maximum MSIV closing time of 5 seconds, plus an assumed closure signal delay time of 0.5 seconds. The licensee took no credit for reduction in break flow as the valves are closing. In their LAR (ML071580324), the licensee stated a total assumed coolant mass release is 4.85 E+07 lbm, consisting of 2.56 E+07 lbm of liquid, 1.58 E+07 lbm of flashed liquid, and 7.10 E+06 lbm of steam. The licensee also assumed that, following accident initiation, the radionuclide inventory from the released coolant reaches the environment instantaneously, taking no credit for holdup in the turbine building. An infinite exchange rate between the control room and the environment was assumed, and no credit was taken for control room filtration, other iodine removal mechanisms, or decay. The release modeled by the licensee was assumed to waft over the control room intake at a rate of 1 m/s, leaving it resident and contributing to dose for 124 seconds, which is based on the size of the activity “puff” that results from the released mass of coolant. The NRC staff finds the use of this puff release model to be acceptable because of the very short duration of the MSLB release and inherent conservatism of the instantaneous release and intake assumed by the licensee.

The licensee used a spreadsheet to perform the calculations for their analysis of the dose consequences resulting from this design-basis MSLB. This spreadsheet was provided for NRC staff review as part of the enclosed MSLB design analysis of Attachment 7 to the May 31, 2007, LAR submittal.

3.2.3.3 Conclusion

The licensee concluded that the radiological consequences at the EAB, LPZ, and control room are within the dose criteria specified in 10 CFR 50.67 and the accident-specific dose acceptance criteria specified in SRP 15.0.1 and RG 1.183. These accident-specific dose acceptance criteria for the MSLB accident, assuming a pre-accident iodine activity spike and no fuel failure, are a TEDE of 25 rem at the EAB for any 2 hours, 25 rem at the outer boundary of the LPZ, and 5 rem for access to, and occupancy of, the control room for the duration of the accident. For an MSLB assuming the equilibrium iodine concentration, the acceptance criteria are a TEDE of 2.5 rem at the EAB for any 2 hours, 2.5 rem at the outer boundary of the LPZ, and 5 rem for access to, and occupancy of, the control room for the duration of the accident. The NRC staff finds that the licensee used analysis assumptions and inputs consistent with applicable regulatory guidance identified in Section 2.0 of this SE and with those stated in the NMP2 USAR as design bases. The major parameters and assumptions used by the licensee, and found acceptable to the NRC staff, are presented in Table 3.2.3. The results of the licensee's design-basis radiological consequence calculation are provided in Table 3.2. The EAB, LPZ, and CR doses estimated by the licensee for the MSLB accident were found to meet the applicable accident dose acceptance criteria and are therefore acceptable.

3.2.4 Control Rod Drop Accident (CRDA)

The current NMP2 design-basis CRDA analysis is based on the traditional accident source term described in TID-14844, "Calculation of Distance Factors for Power and Test Reactor Sites." The NMP2 licensing basis analysis is presented in USAR Section 15.4.9, "Control Rod Drop Accident." To support implementation of the AST, as requested by the subject LAR, the licensee reanalyzed the offsite and control room radiological consequences of the postulated CRDA. This reanalysis was performed to demonstrate that the ESFs designed to mitigate the radiological consequences at NMP2 will remain adequate after implementation of the AST.

The licensee submitted the AST-based reanalysis of the CRDA as part of the LAR. Included in this reanalysis are the assumptions, parameters, and newly calculated offsite and control room doses associated with implementing the AST methodology. The licensee cites RG 1.183 as providing the primary radiological analysis assumptions for their reanalysis of the postulated design-basis CRDA. Specifically, the NRC staff's guidance is detailed in Appendix C of that document.

The licensee has defined the design-basis CRDA as the rapid removal of the highest worth control rod from the core resulting in a reactivity excursion that encompasses the consequences of other postulated CRDAs. In their LAR, the licensee states that NMP2 is a banked position withdrawal sequence (BPWS) plant and that the GESTAR generic CRDA analysis demonstrates that the accident does not result in fuel melting for BPWS plants. However, for the purpose of this analysis, the licensee did conservatively assume some fuel damage (i.e., cladding perforation and fuel melting) to occur.

3.2.4.1 Activity Source

For the design-basis CRDA analysis, the licensee assumed that the core isotopic inventory available for release into the containment is based on maximum full power operation of the core at 4,067 megawatts thermal (MWth), or 1.02 times a 15% uprate of the current licensed thermal power level of 3,467 MWth, in order to account for the ECCS evaluation uncertainty. Additionally, the burnup and enrichment parameters that are assumed when determining the core isotopic inventory are within current licensed limits for fuel at NMP2. The licensee assumed that the assemblies underwent a 24-month cycle at 1400 EFPD per cycle and a 4.1% average enrichment.

A conservative radial peaking factor of 1.80 was applied to the isotopic activity for the failed fuel assemblies. The fraction of core isotopic activity assumed to be available for release from the gap of failed fuel (i.e., fuel with cladding perforation) is provided in RG 1.183, Table 3 and Appendix C, Section 1. These gap fractions were used in the licensee's DBA analysis. These gap fractions are 10% for noble gases and iodines, 12% for alkali metals, and 5% for other halogen isotopes. Consistent with RG 1.183 guidance, the licensee assumed that the speciation of radioactive iodine released from failed fuel is 95% aerosol (particulate), 4.85% elemental, and 0.15% organic, and that all other non-noble gas isotopes are released in 100% particulate form. Also, the licensee assumed that there is no fuel exceeding the burnup limit assumption expressed in Footnote 11 of RG 1.183. Therefore, the total assumed activity in the fuel gap and available for release following the postulated CRDA is found to be acceptable to the NRC staff.

The failed fuel activity release for the design-basis CRDA was characterized by the licensee's estimation that 770 fuel pins experience cladding failure, and that 0.77% of these pins melt, following the postulated CRDA. The licensee asserts that the failure of 770 pins for the GE 8x8 fuel assemblies in the core bounds other fuel assembly types. The licensee states that, as noted in NMP2 USAR Section 15.4.9, CRDA analysis results for BPWS plants have been statistically analyzed and show that, in all cases, the peak fuel enthalpy in a CRDA would be much less than the 280 cal/gm design limit, and thus, the CRDA has been deleted from the standard GE BWR reload package for BPWS plants. Therefore, the NRC staff agrees that the assumption of 770 failed fuel pins, with a melt fraction of 0.0077, is conservative.

3.2.4.2 Transport Methodology and Assumptions

The licensee has defined the design-basis CRDA as the rapid removal of the highest worth control rod from the core resulting in a reactivity excursion that encompasses the consequences of other postulated CRDAs. In their LAR, the licensee states that NMP2 is a BPWS plant and that the GESTAR generic CRDA analysis demonstrates that the accident does not result in fuel melting for BPWS plants. However, for the purpose of this analysis, the licensee did assume fuel damage (i.e., cladding perforation and fuel melting) to occur. The NMP2 AST analysis for the CRDA considers two scenarios with regard to the activity release and transport pathways, as follows:

- Case 1 assumed that activity reaches the turbine/condenser and is released via leakage to the environment at a design-basis leakage rate. The licensee postulated that the MSIVs are manually isolated following accident initiation, and that the turbine/condenser leakage occurs at the design-basis rate of 1% per day for a period of 24 hours, at which time the leakage is assumed to terminate, in accordance with RG 1.183 guidance.

- Case 2 assumed that activity reaches the condenser before it has been isolated and is released via the mechanical vacuum pumps (MVPs). The licensee postulated that maximum activity concentration that will not cause isolation of the MVPs on a high MSL radiation signal would be released to the condenser at the maximum associated release rate, then to the environment via the main stack at the MVP flow rate. No credit was taken for retention by the charcoal delay beds in the offgas system.

Consistent with the guidance of RG 1.183, the licensee assumed that 100% of the noble gas, 10% of the iodine, and 1% of the remaining radionuclides reach the turbine and condensers, and of that activity, 100% of the noble gas, 10% of the iodine, and 1% of the particulate radionuclides are available for release to the environment.

Though the licensee does credit the elevated release associated with the main stack for the MVP releases of Case 2, they do not credit secondary containment exhaust filtration or control room intake filtration for activity removal in either case. To calculate accident dose consequences, the licensee uses a spreadsheet for Case 1 and the RADTRAD code for Case 2. These calculations were provided for NRC staff review as part of the enclosed CRDA design analysis of Attachment 7 to the May 31, 2007, LAR submittal.

In the January 7, 2008, response to an NRC staff RAI asking about the potential of post-CRDA releases from the steam jet air ejector or gland seal steam condenser, the licensee states that any steam seal leakage would be small in comparison to the MVP exhaust flow, and thus was not specifically included in their analysis. The licensee also states that, because the volumetric exhaust flow of the MVPs is larger than that of the steam jet air ejector, and because the release path for the steam jet air ejector would be via the offgas system and the stack (the scenario described in GE Report NEDO-31400A), the MVP case is bounding. The NRC staff agrees with this assessment, and finds the exclusion of these potential leakage paths acceptable for the design-basis CRDA analysis at NMP2.

3.2.4.3 Conclusion

The licensee concluded that the radiological consequences at the EAB, LPZ, and control room are within the dose criteria specified in 10 CFR 50.67 and the accident-specific dose acceptance criteria specified in SRP 15.0.1 and RG 1.183. The accident-specific dose acceptance criteria for the CRDA at NMP2 are a TEDE of 6.3 rem at the EAB for any 2 hours, 6.3 rem at the outer boundary of the LPZ, and 5 rem for access to, and occupancy of, the control room for the duration of the accident. The NRC staff finds that the licensee used analysis assumptions and inputs consistent with applicable regulatory guidance identified in Section 2.0 of this SE and with those stated in the NMP2 USAR as design bases. The NRC staff also performed an independent calculation of the dose consequences of the licensee's CRDA Case 2 using the licensee's assumptions for input to the RADTRAD computer code. The NRC staff's calculation confirmed the licensee's dose results. The major parameters and assumptions used by the licensee, and found acceptable to the NRC staff, are presented in Table 3.2.4. The results of the licensee's design basis radiological consequence calculation are provided in Table 3.2. The EAB, LPZ, and CR doses estimated by the licensee for the CRDA were found to meet the applicable accident dose acceptance criteria and are, therefore, acceptable.

3.3 Control Room Habitability and Modeling

The current NMP2 DBA analyses, as described in USAR Chapter 15, do not calculate control room dose; therefore, the control room dose model provided in the revised DBA accident analyses that support this AST-based LAR represents a change in the NMP2 licensing basis.

For their revised analyses where control room isolation and/or filtration is credited, the licensee assumed an emergency mode control room intake flow rate of 2500 cfm ±10%, and assumed 99% filtration efficiency for elemental iodine, organic iodine, and particulate forms of radionuclide activity. For conservatism, the upper flow uncertainty value, 2750 cfm, is used for modeling, then, as a design basis, reduced to 1650 cfm at 20 minutes. Where control room filtration is credited, the licensee assumed that the control room was automatically isolated on a LOCA signal, and that filtration was delayed for 80 seconds.

In a letter dated January 31, 2005, from the licensee to the NRC staff (ML050460309), it is indicated that the highest measured unfiltered inleakage into the NMP2 control room is 147 cfm. For the DBA analyses that model actual NMP2 control room functionality, the licensee assumed an unfiltered inleakage of 250 cfm, to bound the worst-case unfiltered inleakage as tested. This value is conservative and provides margin for future measurements of control room inleakage. The major parameters and assumptions used by the licensee for modeling of the control room, and found acceptable to the NRC staff, are presented in Table 3.3.

3.4 Technical Specification Changes

3.4.1 Revision to the TS 1.0 Definition of "*Dose Equivalent I-131*"

The licensee has proposed to add the definition of *Dose Equivalent I-131* to NMP2 TS Section 1.0. The licensee's revised DBA dose consequence analyses use committed effective dose equivalent dose conversion factors from Table 2.1 of Federal Guidance Report (FGR) 11, ORNL, 1988, "Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion," as the source of thyroid dose conversion factors instead of the current TID-14844, RG 1.109, Rev. 1, and ICRP 30 referenced dose conversion factors.

With the implementation of the AST, the previous whole body and thyroid dose guidelines of 10 CFR 100.11 and 10 CFR Part 50, Appendix A, GDC 19, are replaced by the TEDE criteria of 10 CFR 50.67(b)(2). This new definition reflects adoption of the dose conversion factors and dose consequences of the revised radiological analyses. Thus, this proposed revision to the definition of *Dose Equivalent I-131* is supported by the justification for the proposed licensing basis revision to implement the AST, and conforms to the implementation of the AST and the TEDE criteria in 10 CFR 50.67. Therefore, the NRC staff finds the proposed revision to the TS 1.0 definition *Dose Equivalent I-131* acceptable.

3.4.2 TS 3.1.7, Standby Liquid Control (SLC) System

The Applicability statement for TS 3.1.7 is revised to include Mode 3 (Hot Shutdown), and Required Action C is revised to add an additional action (C.2) to be in Mode 4 within 36 hours.

This change is consistent with the AST methodology and the LOCA analysis assumptions for this license amendment and is needed to allow reliance on the SLC system for buffering suppression pool pH.

The NRC staff reviewed the quantity of sodium pentaborate available with respect to the quantity of acid producing debris and radiolytic acid production to confirm adequate pH control. In addition, the NRC staff reviewed the SLC system with respect to SLC roles in delivery of sodium pentaborate to the suppression pool for pH control. As described in Section 3.2.1.2, the control of pH in the suppression pool is required to mitigate the consequences of a DBA in which fuel is damaged by maintaining radioactive iodine in solution and not available for release via containment atmosphere leakage. As such, the new role being assigned to the SLC is a safety-related role. The licensee stated that all SLC System components needed for this role are already classified in the USAR as safety related.

The NRC staff concludes that the SLC System as designed and installed is a high-quality system that provides reasonable assurance that the sodium pentaborate will be injected into the core upon activation. This is based on the information below as provided by the licensee in its LAR:

- SLC System components required for the injection of sodium pentaborate solution into the reactor are classified as safety related.

- SLC System components required for the injection of sodium pentaborate solution into the reactor are classified as seismic Category I.

- SLC System electrical components required for the injection of sodium pentaborate solution into the reactor are classified as electrical Class 1E and are powered from Class 1E emergency power sources backed up by emergency diesel generators on a loss of offsite power.

- SLC System components are inspected and tested in accordance with applicable provisions of the NMP2 ASME Code ISI and IST Programs as required by 10 CFR 50.55a.

- SLC System is included in the scope of the NMP2 Maintenance Rule Program consistent with 10 CFR 50.65.

- SLC System electrical components required to function for post-LOCA sodium pentaborate injection are included in the NMP2 Environmental Qualification (EQ) Program. The licensee stated that the SLC System components will have their EQ established for the post-LOCA environment associated with the new suppression pool pH control function in accordance with the station design change process during implementation of the requested AST license amendment.

- SLC system has two separate 100 percent capacity subsystems of the active components required for the injection of sodium pentaborate solution into the reactor. The single common active component is the injection line inboard containment isolation valve (2SLS*V10). The existing valve, an original system safety-related purchase, is a 2-inch, 1500# pressure class, stainless steel, inclined piston check valve manufactured by Velan Valve Corporation, Model B08-3036Z-14MS all metallic construction. Failure of this valve to open would prevent SLC System injection of the sodium pentaborate solution and effective suppression pool pH control. The licensee stated that their search of the industry Equipment Performance Information and Exchange System (EPIX) and Nuclear Plant Reliability Data System (NPRDS) databases identified no failures at any plant of any common SLC System injection line check valves to open. One failure of a [different model] Velan check valve to close was identified in the search. At NMP2 this check valve is tested in the open direction at least once every 24 months with demineralized water by operation of one of the subsystems and verifying that the required system flow rate is met or exceeded. During this test, the SLC pump discharge and reactor pressures are noted. For the existing system safety function of Anticipated Transients Without Scram (ATWS) mitigation the maximum check valve differential pressure would be approximately 300 psid while for the post-LOCA pH control function it would be approximately the pump maximum differential pressure of 1400 psid. The high differential pressure available post-LOCA makes it more likely that the valve will open. Based on the procurement, deterioration-resistant design, lack of relevant failure history, periodic open exercise testing and high differential pressure available post-LOCA, the staff finds this check valve to be of acceptable quality and reliability and that it provides reasonable assurance that the SLC System will be able to inject the sodium pentaborate solution when actuated.

- Initiation of the SLC System is accomplished from the main control room at NMP2 by operators manipulating a simple keylock switch. Plant emergency operating procedures (EOPs) presently provide instructions to initiate SLC System as well as other sources of water for emergency core cooling based on reactor pressure vessel (RPV) water level, RPV pressure, or drywell pressure. Plant severe accident procedures (SAPs) also have provisions for initiating SLC System injection flow. The licensee stated that during implementation of the requested AST license amendment the plant EOPs and SAPs would be revised to reflect the new post-LOCA function of the SLC System, including instructions to manually actuate the SLC system based on high drywell radiation levels and to ensure that once initiated, the entire contents of the SLC System sodium pentaborate storage tank are injected to accomplish the pH control function.

- The licensee stated that during implementation of the proposed license amendment, licensed operators and shift technical advisors (STAs) would be trained on the procedure revisions addressing sodium pentaborate solution injection for suppression pool pH control following a LOCA.

The NRC staff considered the transport of the sodium pentaborate from the reactor vessel to the suppression pool. The SLC system injects the sodium pentaborate to the reactor vessel via the high-pressure core spray (HPCS) injection line. The SLC injection flow mixes with the HPCS flow and is discharged into the vessel via the HPCS sparger above the core. It further mixes there with low-pressure core/coolant injection (LPCI) flow, passes down through the core and up through the jet pump diffusers and out into the downcomer annulus and then on out through the recirculation line break. From the recirculation line break, water will spill to the drywell floor where it will pool until the level is above the top of the downcomer pipes, at which time the water will drain down into the suppression pool. The licensee stated that in about 4.4 hours after the LOCA, the water in the vessel, the drywell floor pool, and the suppression pool will have reached a mixing equilibrium, which is much less than their calculated time of about 12 days for the suppression pool pH to drop below 7.0 without the sodium pentaborate solution injection. As a consequence, the staff finds that there would be adequate mixing for pH control to minimize iodine re-evolution throughout the 30-day post-accident dose determination period.

3.4.3 TS 3.3.7.1, Control Room Envelope Filtration (CREF) System Instrumentation, TS 3.7.2, Control Room Envelope Filtration (CREF) System, TS3.7.3, Control Room Envelope Air Conditioning (AC) System

Footnote (b) of TS Table 3.3.7.1-1 specifies operability requirements for the "Main Control Room Ventilation Radiation Monitor - High" function (Function 3) during core alterations and during movement of irradiated fuel assemblies in the secondary containment. The footnote is revised by deleting "core alterations," and by replacing the term "irradiated fuel assemblies" with "recently irradiated fuel assemblies."

The operability requirements for the CREF system are revised to delete "During CORE ALTERATIONS" and to replace the term "irradiated fuel assemblies" with "recently irradiated fuel assemblies." These changes affect the Applicability statement and portions of Actions D and F of TS 3.7.2.

The operability requirements for the Control Room Envelope AC system are revised to delete "During CORE ALTERATIONS" and to replace the term "irradiated fuel assemblies" with "recently irradiated fuel assemblies." These changes affect the Applicability statement and portions of Actions C and E of TS 3.7.3.

The NRC staff reviewed these proposed changes and found that they are consistent with the AST methodology for determining radiological consequences of the design basis FHA and analysis assumptions. The proposed changes are also consistent with the guidance of TSTF-51. Therefore, the NRC staff finds the requested changes acceptable.

3.5 Summary

As described above, the NRC staff reviewed the assumptions, inputs, and methods used by the licensee to assess the radiological consequences of the postulated DBA analyses with the proposed TS changes. The NRC staff finds that the licensee used analysis methods and assumptions consistent with the conservative regulatory requirements and guidance identified in Section 2.0. The NRC staff further finds reasonable assurance that NMP2, as modified by this approved license amendment, will continue to provide sufficient safety margins, with adequate defense-in-depth, to address unanticipated events and to compensate for uncertainties in accident progression, analysis assumptions, and input parameters. Therefore, the proposed license amendment is acceptable with respect to the radiological consequences of the DBAs.

4.0 STATE CONSULTATION

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

5.0 ENVIRONMENTAL CONSIDERATION

The amendment changes a requirement with respect to use of a facility component located within the restricted area as defined in 10 CFR Part 20 and changes surveillance requirements. The NRC staff has determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents 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 (72 FR 41786). 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.

6.0 CONCLUSION

The NRC staff has reviewed the AST implementation proposed by the licensee for NMP2. In performing this review, the staff relied upon information placed on the docket by the licensee, precedent AST reviews, and, where deemed necessary, on NRC staff confirmatory calculations. The staff reviewed the assumptions, inputs, and methods used by the licensee to assess the radiological impacts of the proposed plant modifications in the context of the proposed AST. The staff finds that the licensee used analysis methods and assumptions consistent with the conservative guidance of RG 1.183, with the exceptions discussed and accepted earlier in this SE. The staff finds the methods and assumptions used by the licensee to be in compliance with applicable requirements. The staff compared the doses estimated by the licensee to the applicable acceptance criteria and to the results estimated by the staff in its confirmatory calculations. The staff finds with reasonable assurance that the licensee's estimates of the TEDE due to DBAs will comply with the requirements of 10 CFR 50.67 and the guidance of RG 1.183.

The NRC staff finds reasonable assurance that NMP2 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 analysis assumptions and parameters. The staff concludes that the proposed AST implementation is acceptable.

This licensing action is considered a full implementation of the AST, where TID-14844 will continue to be used as the radiation dose basis for equipment qualification and vital area access. With this approval, the previous accident source term in the NMP2 design basis is superseded by the AST proposed by the licensee, except for equipment qualification and vital area access where TID-14844 will continue to be used as the radiation dose basis. The previous offsite and control room 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. The characteristics of the AST and the TEDE criteria may not be extended to equipment qualification and vital area access without prior NRC review under 10 CFR 50.67. All future radiological analyses performed to demonstrate compliance with regulatory requirements shall address all characteristics of the AST and the TEDE criteria as described in the NMP2 design basis.

The Commission 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 amendments will not be inimical to the common defense and security or to the health and safety of the public.

Attachment:
Tables

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Date: May 29, 2008

Table 3.1.1

NMP2 Control Room and TSC Atmospheric Dispersion Factors

[a]Source / Receptor	$^{\chi}/_{Q}$ Values				
	0-2 Hours	2-8 Hours	8-24 Hours	24-96 Hours	96-720 Hours
	sec/m^3	sec/m^3	sec/m^3	sec/m^3	sec/m^3
NMP2 Control Room Upper West Intake					
MS2/CR	7.04E-5	3.95E-5	1.49E-5	9.96E-6	7.46E-6
RB2/CR	8.24E-4	6.29E-4	2.28E-4	1.56E-4	1.25E-4
MST/CR	1.13E-3	7.49E-4	2.76E-4	1.90E-4	1.49E-4
SGTB/CR	3.62E-4	2.59E-4	9.48E-5	6.16E-5	4.42E-5
PP/CR	3.36E-4	2.00E-4	7.31E-5	5.53E-5	4.04E-5
MS1/CR	1.06E-4	5.90E-5	2.23E-5	1.73E-5	1.43E-5
RB1/CR	1.23E-4	7.21E-5	2.57E-5	2.28E-5	2.05E-5
TB1/CR	1.30E-4	9.03E-5	3.45E-5	2.92E-5	2.56E-5
NMP2 Control Room Upper East Intake					
MS2/CR	8.03E-5	4.48E-5	1.68E-5	1.20E-5	8.83E-6
RB2/CR	1.09E-3	7.23E-4	2.46E-4	1.92E-4	1.47E-4
MST/CR	1.47E-3	8.80E-4	3.32E-4	2.26E-4	1.68E-4
SGTB/CR	5.31E-4	3.70E-4	1.35E-4	9.16E-5	6.70E-5
PP/CR	3.74E-4	2.05E-4	7.08E-5	5.41E-5	3.88E-5
MS1/CR	9.83E-5	5.81E-5	2.22E-5	1.83E-5	1.57E-5
RB1/CR	1.06E-4	6.70E-5	2.39E-5	2.17E-5	1.96E-5
TB1/CR	1.09E-4	7.73E-5	2.95E-5	2.45E-5	2.14E-5

[a] Source/Receptor pairs are for the LOCA, FHA, and CRDA events. Releases to the TSC are evaluated only in the event of a LOCA. Key: RB = Reactor Building Blowout Panel; TB=Turbine Building Blowout Panel; MS = Main Stack; CR = Unit 1 Control Room; TSC = Technical Support Center

Table 3.1.1 cont'd

NMP2 Control Room Lower West Intake					
MS2/CR	7.15E-5	4.01E-5	1.52E-5	1.01E-5	7.55E-6
RB2/CR	9.03E-4	6.93E-4	2.50E-4	1.71E-4	1.36E-4
MST/CR	1.46E-3	9.74E-4	3.63E-4	2.45E-4	1.90E-4
SGTB/CR	4.05E-4	2.95E-4	1.08E-4	6.98E-5	5.00E-5
PP/CR	3.84E-4	2.28E-4	8.23E-5	6.28E-5	4.57E-5
MS1/CR	1.10E-4	6.16E-5	2.31E-5	1.85E-5	1.54E-5
RB1/CR	1.26E-4	7.73E-5	2.74E-5	2.45E-5	2.23E-5
TB1/CR	1.31E-4	9.42E-5	3.59E-5	3.01E-5	2.63E-5
NMP2 Control Room Lower East Intake					
MS2/CR	7.78E-5	4.31E-5	1.64E-5	1.16E-5	8.61E-6
RB2/CR	9.43E-4	6.34E-4	2.13E-4	1.67E-4	1.29E-4
MST/CR	1.46E-3	8.70E-4	3.32E-4	2.23E-4	1.68E-4
SGTB/CR	5.33E-4	3.72E-4	1.36E-4	9.17E-5	6.72E-5
PP/CR	3.67E-4	2.01E-4	6.95E-5	5.32E-5	3.83E-5
MS1/CR	9.54E-5	5.68E-5	2.15E-5	1.79E-5	1.54E-5
RB1/CR	1.06E-4	6.68E-5	2.39E-5	2.16E-5	1.95E-5
TB1/CR	1.08E-4	7.69E-5	2.96E-5	2.44E-5	2.14E-5
Technical Support Center					
MS2/TSC	4.95E-5	2.69E-5	1.03E-5	6.67E-6	4.85E-6
RB2/TSC	2.70E-4	1.64E-4	5.41E-5	3.86E-5	2.86E-5
MST/TSC	3.27E-4	2.41E-4	8.38E-5	5.95E-5	4.76E-5
SGTB/TSC	1.62E-4	1.19E-4	4.28E-5	2.72E-5	2.24E-5
PP/TSC	2.69E-4	1.91E-4	7.19E-5	4.22E-5	3.40E-5

Table 3.1.1 cont'd

[b]SOURCE/ RECEPTOR	MSLB $\chi/_Q$ VALUE (sec/m^3)
MST/CR	1.47E-3

b NMP1 assumed an instantaneous ground level puff release and transport $\chi/_Q$ value for the MSLB event.

Table 3.1.2

NMP1 Onsite Atmospheric Dispersion Factors via NMP2 Releases

[a, b] SOURCE/ RECEPTOR	$^{\chi}/_{Q}$ Values				
	0-2 Hours	2-8 Hours	8-24 Hours	24-96 Hours	96-720 Hours
	sec/m^3	sec/m^3	sec/m^3	sec/m^3	sec/m^3
MS2/CR1	4.18E-5	2.30E-5	8.94E-6	5.62E-6	4.31E-6
RB2/CR1	1.77E-5	1.09E-4	3.92E-5	2.48E-5	1.85E-5
MST/CR1	1.90E-4	1.37E-4	4.93E-5	3.12E-5	2.56E-5
SGTB/CR1	1.11E-4	8.09E-5	2.91E-5	1.82E-5	1.45E-5
PP/CR1	1.59E-4	1.13E-4	4.19E-5	2.48E-5	2.00E-5

a Source/Receptor pairs are for the LOCA, FHA, and CRDA events. Key: MS2 = Unit 2 Main Stack; RB2 = Unit 2 Combined Radwaste/Reactor Building Vent; MST = Unit 2 Main Steam Tunnel; SGTB = Unit 2 Standby Gas Treatment Building; PP = Unit 2 Pass Panel; CR1 = Unit 1 Control Room

b All releases were modeled as ground level releases pursuant to guidelines in RG 1.194.

Table 3.1.3

NMP2 Offsite Atmospheric Dispersion Factors

Offsite Dose Location		[a]χ/Q Values				
		0-2 Hours	0-8 Hours	8-24 Hours	24-96 Hours	96-720 Hours
		sec/m^3	sec/m^3	sec/m^3	sec/m^3	sec/m^3
Ground Release	EAB	1.19E-4	----	----	----	----
	LPZ	------	1.62E-5	1.09E-5	4.59E-6	1.33E-6
[b]Elevated Release	EAB	2.96E-5	----	----	----	----
	LPZ	----	1.42E-5	5.41E-7	2.31E-7	7.65E-8

[a] These χ/Q values were applied to all postulated accidents evaluated. The 0-2 hour EAB and 0-8 hour outer LPZ boundary ground level χ/Q values are assumed to apply throughout for the MSLB event.

[b] The initial time period EAB and outer LPZ boundary elevated χ/Q values (i.e., fumigation) are assumed to exist throughout the DBA for the EAB dose assessment and for the first 8 hours of the LPZ dose assessment.

Table 3.2

Licensee Calculated Radiological Consequences of Design Basis Accidents at NMP2

Design Basis Accident	Control Room		[a]EAB		LPZ	
	[b]Total Dose (rem TEDE)	Acceptance Criteria (rem TEDE)	[c]Total Dose (rem TEDE)	Acceptance Criteria (rem TEDE)	[d]Total Dose (rem TEDE)	Acceptance Criteria (rem TEDE)
LOCA	1.65E+00	*5.0*	6.57E-01	*25*	7.69E-01	*25*
FHA	3.15E+00	*5.0*	4.50E-01	*6.3*	6.13E-02	*6.3*
MSLB	2.96E+00	*5.0*	4.92E-01	*25*	5.34E-02	*25*
CRDA						
Case 1	1.26E+00	*5.0*	5.68E-01	*6.3*	7.73E-02	*6.3*
Case 2	2.31E+00	*5.0*	1.03E+00	*6.3*	1.17E+00	*6.3*

[a] The licensee calculated the EAB dose for the worst 2-hour period of the accident duration.
[b] The licensee's control room dose results have been rounded to three significant digit precision.
[c] The licensee's EAB dose results have been rounded to three significant digit precision.
[d] The licensee's LPZ dose results have been rounded to three significant digit precision.

Table 3.2.1

Key Parameters Used in Radiological Consequence Analysis of Loss of Coolant Accident

Parameter	Value
Reactor Core Power, MWth	4067
Primary Containment Volume, ft^3 Drywell Airspace Suppression Chamber Airspace (minimum) Suppression Pool (minimum)	 306,200 190,800 145,000
Secondary Containment Volume, ft^3	3,880,000
Spray Delay time, min	20
Spray Flow Rate, gpm Actual Effective (used in analysis)	 6143 5237
Drywell and Wetwell Airspace Mixing Initiation, hrs	2.0333
Primary Containment Leakage Rate, weight % per day 0 to 24 hrs 24 hrs to 30 days	 1.10 0.55
Spray Removal in Containment Particulate λ 0.0 – 0.3 hrs 0.3 – 6.0 hrs Elemental λ 0.0 – 0.3 hrs 0.3 – 2.017 hrs 2.017 – 3.157 hrs	 0.0 19.8 0.0 19.8 1.98
Secondary Containment Bypass Pathway Delay, hrs MSL – MSIV Failed Open MSL – MISV Closed Other Drywell Bypass (Feedwater A, Feedwater B, 14” CPS Line in Drywell)	 5.26 7.11 2.45
MSL Volume (per line), ft^3 Inboard Outboard	 59.27 65.69

Table 3.2.1 cont'd

MSIV Leakage Rate, scfh Per MSL Total	 24 96
Reactor Building Leak Rate During Drawdown, cfm	2670
Reactor Building SGTS Flow Rate, cfm	4000
Reactor Building SGTS Filter Efficiency, % Aerosol/Particulate Elemental Organic	 99 99 99
ESF Leakage Iodine Re-Evolution, %	10
ESF Leakage Iodine Release Speciation, % Elemental Organic	 97 3
Atmospheric Dispersion Factors	Tables 3.1.1 and 3.1.2

Table 3.2.2

Key Parameters Used in Radiological Consequence Analysis of Fuel Handling Accident

Parameter	Value
Reactor Core Power, MWth	4067
Peaking Factor	1.8
Number of Failed Fuel Pins	124
Fuel Decay Time, hr	24
Fraction of Core Inventory in Fuel Gap Kr-85 I-131 Other Noble Gases Other Iodines	 0.10 0.08 0.05 0.05
Minimum Water Depth Above Damaged Fuel, ft	22.25
Iodine Decontamination Factor	175
Iodine Speciation in Fuel Gap, % Elemental Organic	 99.85 0.15
Fuel Activity Release Duration	Instantaneous
Atmospheric Dispersion Factors	Tables 3.1.1 and 3.1.2

Table 3.2.3

Key Parameters Used in Radiological Consequence Analysis of Main Steam Line Break Accident

Parameter	Value
Reactor Core Power, MWth	4067
Failed Fuel, %	0
Reactor Coolant Activity, µCi/gm DE I-131 Equilibrium Iodine Activity Pre-accident Iodine Spike Activity	 0.2 4.0
Iodine-131 DCF, rem/Ci	3.29E+04
Iodine Speciation from Coolant, % Elemental Organic	 97 3
Time Until MSIV Isolation, sec	5.5
Coolant Mass Blowdown, lbm Liquid Steam Total	 4.1E+07 7.1E+06 4.9E+07
Time for Puff to Traverse Control Room Intake, sec	124
Atmospheric Dispersion Factors	Tables 3.1.1 and 3.1.2

Table 3.2.4

Key Parameters Used in Radiological Consequence Analysis of Control Rod Drop Accident

Parameter	Value
Reactor Core Power, MWth	4067
Peaking Factor	1.8
Failed Fuel Cladding Failure, pins Melted Fraction Total Failure Fraction	 770 0.0077 0.0163
Fraction of Core Inventory in Fuel Gap Noble Gas Iodine Alkali Metals Other Halogens	 0.10 0.10 0.12 0.05
Iodine Speciation in Environment after Partitioning, % Elemental Organic Aerosol/Particulate	 97 3 0
Isotopic Fractions Reaching the Turbine/Condenser Noble Gas Iodine Other Radionuclides	 1.0 0.1 0.01
Isotopic Fractions Available for Environmental Release Noble Gas Iodine Other Radionuclides	 1.0 0.1 0.01
Main Condenser Volume, ft^3	97,000
Main Condenser Leakage Rate, % per day	1.0
Main Condenser MVP Exhaust Rate, cfm	5000
Atmospheric Dispersion Factors	Tables 3.1.1 and 3.1.2

Table 3.3

Key Parameters Used in Modeling the Control Room for Design Basis Radiological Consequence Analyses

Parameter	Value
Control Room Volume, ft^3	381,000
Normal Intake Rate, cfm	Not Used
Filtered Emergency Mode Intake Rate, cfm 0 – 20 minutes 20 minutes – 720 hours	 2500 ± 10% (2750 used in analysis) 1650
Recirculation Flow Rate, cfm 0 – 80 seconds 80 seconds – 720 hours	 0 750 ± 10% (675 used in analysis)
Filtration Initiation Delay, sec	50
Filter Efficiency, % Elemental Organic Aerosol/Particulate	 99 99 99
Unfiltered Inleakage Rate, cfm	250
Occupancy Factors 0 – 24 hours 24 – 96 hours 96 – 720 hours	 1.0 0.6 0.4
Breathing Rate, m^3/sec	3.5E-04
Atmospheric Dispersion Factors	Table 3.1.1