December 19, 2007

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. 1 - ISSUANCE OF AMENDMENT RE: IMPLEMENTATION OF ALTERNATIVE RADIOLOGICAL SOURCE TERM (TAC NO. MD3896)

Dear Mr. Polson:

The Commission has issued the enclosed Amendment No. 194 to Renewed Facility Operating License No. DPR-63 for the Nine Mile Point Nuclear Station, Unit No. 1 (NMP-1). The amendment consists of changes to the License and to the Technical Specifications in response to your application transmitted by letter dated December 14, 2006, as supplemented by letters dated July 17, 2007, August 1, 2007, and September 19, 2007.

The amendment revises 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 revision replaces the methodology that is based on Technical Information Document (TID)-14844, "Calculation of Distance Factors for Power and Test Reactor Sites," with the alternate 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 next regular biweekly *Federal Register* notice.

Sincerely,

/RA/

Marshall J. David, Project Manager
Plant Licensing Branch I-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-220

Enclosures:
1. Amendment No. 194 to DPR-63
2. Safety Evaluation

cc w/encls: See next page

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cc w/encls: See next page

Package No.: ML073230603
Amendment No.: ML073230597
Tech Spec No.: ML NRR-058

OFFICE	LPLI-1/PM	LPLI-1/LA	AADB/BC*	CSGB/BC*	SCVB/BC*	ITSB/BC	OGC	LPLI-1/BC
NAME	MDavid	SLittle	MHart	AHiser	RDennig	TKobetz	ASilvia	MKowal
DATE	11/23/07	11/20/07	09/19/07	08/02/07	09/28/07	12/13/07	12/12/07	12/19/07

*SE transmitted by memo on date shown.
OFFICIAL RECORD COPY

DATED: December 19, 2007

AMENDMENT NO. 194 TO RENEWED FACILITY OPERATING LICENSE NO. DPR-63
NINE MILE POINT, UNIT NO. 1

PUBLIC	
LPLI-1	
MKowal	RidsNrrDorlLpl-1
SLittle	RidsNrrLASLittle
MDavid	RidsNrrPMMDavid
BSingal	RidsNrrDpr
OGC	RidsOgcMailCenter
GHill (2)	
RDennig	RidsNrrDssScvb
MHart	RidsNrrDraAadb
AHiser	RidsNrrDciCsgb
TKobetz	RidsNrrDirsItsb
ACRS	RidsNrrAcrsAcnwMailCenter
GDentel, RI	RidsRgn1MailCenter

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NINE MILE POINT NUCLEAR STATION, LLC (NMPNS)

DOCKET NO. 50-220

NINE MILE POINT NUCLEAR STATION, UNIT NO. 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 194
Renewed License No. DPR-63

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 December 14, 2006, as supplemented by letters dated July 17, August 1, and September 19, 2007, 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. DPR-63 is hereby amended to read as follows:

(2) Technical Specifications

The Technical Specifications contained in Appendix A, which is attached hereto, as revised through Amendment No. 194, is hereby incorporated into this license. Nine Mile Point Nuclear Station, LLC shall operate the facility in accordance with the Technical Specifications.

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: December 19, 2007

ATTACHMENT TO LICENSE AMENDMENT NO. 194

TO RENEWED FACILITY OPERATING LICENSE NO. DPR-63

DOCKET NO. 50-220

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
3	3

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
6	6
44	44
99	99
---	99a
131	131
164	164
165	165
168	168
170	170
171	171
174	174
175	175
178	178
179	179
240	240
242	242
246	246
247a	247a

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 194 TO RENEWED

FACILITY OPERATING LICENSE NO. DPR-63

NINE MILE POINT NUCLEAR STATION, LLC

NINE MILE POINT NUCLEAR STATION, UNIT NO. 1

DOCKET NO. 50-220

1.0 INTRODUCTION

By letter dated December 14, 2006 (Agencywide Documents Access and Management Systems (ADAMS) Accession No. ML070110221), as supplemented by letters dated July 17, 2007 (ADAMS Accession No. ML072070214), August 1, 2007 (ADAMS Accession No. ML072220146), and September 19, 2007 (ADAMS Accession No. ML072620361), Nine Mile Point Nuclear Station (NMPNS), LLC (the licensee) submitted a license amendment request (LAR) for Nine Mile Point, Unit No. 1 (NMP1). 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."

The supplements dated July 17, 2007, August 1, 2007, and September 19, 2007, 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 March 13, 2007 (72 FR 11390).

In its response to Generic Letter (GL) 2003-01, "CR Habitability," dated January 31, 2005 (ADAMS Accession No. ML050460309), the licensee noted that, as part of Amendment No. 161, issued on May 23, 1998, the NRC staff added a license condition requiring it to submit a license amendment request by December 18, 1998, with proposed methods for complying with 10 CFR

Part 50, Appendix A, General Design Criterion (GDC) 19 dose guidelines without relying on the use of potassium iodide (KI). A revised dose analysis was submitted by the licensee on December 18, 1998, as required. This dose analysis is based on source term methodologies and assumptions derived from TID-14844, and is considered by the licensee to be the current licensing/design basis analysis. Since the license condition remains in effect, KI is currently maintained in the control room (CR) available for operator use even though its use is no longer credited in the revised dose analysis. Also in its response to GL 2003-01, the licensee stated that the AST methodology, as defined in 10 CFR 50.67 and described in RG 1.183, would be used to demonstrate compliance with the guidelines of GDC 19 in a future LAR.

The current NMP1 licensing basis uses a source term that is based on TID-14844 to calculate the radiological consequences of postulated DBAs. The LAR contains the reanalysis and licensing basis alternative method changes necessary to meet GL 2003-01 objectives. Use of the AST methodology would allow an increase in the assumed design-basis unfiltered inleakage into the CR envelope to a value larger than that observed in tracer gas testing and would eliminate the need for KI for the CR operators.

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 50.67. The applicable criteria are 5 rem Total Effective Dose Equivalent (TEDE) in the 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 guidance of NUREG-0737, "Clarification of TMI Action Plan Requirements," and requirements in 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 GDC 19, and the accident dose criteria in 10 CFR 50.67, as supplemented by the regulatory guidance in Regulatory Position 4.4 and Table 6 of RG 1.183, and in NUREG-0800, "Standard Review Plan [SRP] for the Review of Safety Analysis Reports for Nuclear Power Plants," Section 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 NMP1 Updated Final Safety Analysis Report (UFSAR) and TSs, as well as consideration of any applicable alternative documentation the licensee may have provided.

- 10 CFR 50.67, "Accident source term"

- 10 CFR Part 50, Appendix A, "General Design Criteria for Nuclear Power Plants," Criterion 19, "Control room"

- 10 CFR Part 50, Appendix E, "Emergency Planning and Preparedness for Production and Utilization Facilities"

- SRP 2.3.4, "Short-Term Diffusion Estimates for Accidental Atmospheric Releases"

- SRP 6.4, “Control Room Habitability Systems”
- SRP 6.5.2, “Containment Spray as a Fission Product Cleanup System”
- SRP 9.4.1, “Control Room Area Ventilation System”
- SRP 15.0.1, “Radiological Consequence Analyses Using Alternative Source Term”
- NUREG-0737, “Clarification of TMI Action Plan Requirements”
- NUREG-1465, “Accident Source Terms for Light-Water Nuclear Power Plants”
- 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 CR Radiological Habitability Assessments at Nuclear Power Plants”
- RG 1.197, “Demonstrating CR Envelope Integrity at Nuclear Power Reactors”
- Regulatory Issue Summary 2006-04, “Experience With Implementation of Alternative Source Terms”

3.0 TECHNICAL EVALUATION

The NRC staff reviewed the regulatory and technical analyses performed by the licensee in support of its LAR. The NRC staff reviewed the assumptions, inputs, and methods used by the licensee to assess the impacts of the proposed license amendment. In several instances, the NRC staff also performed independent calculations to confirm the conservatism of the licensee’s analyses. 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. NMP1 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 NMP1 χ/Q values for the loss-of-coolant accident (LOCA), fuel-handling accident (FHA) (also called a refueling accident in the licensee's LAR), and control rod drop accident (CRDA). In addition, the licensee assumed an instantaneous ground level puff release modeled in accordance with the methodology discussed in RG 1.194, "Atmospheric Relative Concentrations for CR Radiological Habitability Assessments at Nuclear Power Plants," to calculate CR 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 XV of the NMP1 UFSAR.

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 NRC 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, "Onsite Meteorological Programs."

The NRC staff performed confirmatory and quality assurance evaluations of the meteorological data presented using the methodology described in NUREG-0917, "Nuclear Regulatory Commission Staff Computer Programs for Use with Meteorological Data." 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 similarly from year to year for both measurement levels with the highest occurrence of 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-B stability classes) occurring 20.7%, neutral and slightly stable conditions (C-D 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, NMP1 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 for 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 a request for additional information (RAI) dated June 8, 2007. As a result, the licensee performed sensitivity calculations, which are discussed in more detail in Section 3.1.3.

For the reasons noted above, the meteorological data presented for years 1997 through 2001 were found acceptable by the NRC 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 for the LOCA, FHA, MSLB, and CRDA using guidance provided in RG 1.194. 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,

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[1] According to a presentation delivered at the Nuclear Utility Meteorological Data Users Group 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

release characterization, source-receptor configuration, meteorological regimes, or terrain conditions that preclude use of this model in support of the current LAR for NMP1.

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, and the straight-line distance between the source and intake/receptor, all in meters, the direction between intake to source, in degrees, and the 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 three different source points: NMP1 main stack (MS), NMP1 reactor building (RB) blowout panel, and NMP1 turbine building (TB) blowout panel. The release heights for each of these sources are 106.7 meters, 24.0 meters, and 22.1 meters, respectively. All releases were assumed to occur at ground level for the purpose of atmospheric dispersion analyses. The MS release point was treated as a ground level release pursuant to RG 1.194, which states that the use of stack release mode is acceptable when the release point is greater than 2.5 times the height of the adjacent structure(s). Because the stack height is less than 2.5 times the height of adjacent structures, no credit was taken for an elevated release. The MS is considered an acceptable ground level release by this characterization. The primary onsite receptors used for the NMP1 atmospheric dispersion evaluations, as noted in Table 3.1.1, were the NMP1 CR intake and the NMP1 TSC intake.

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 TB blowout panel to the 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 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 and F-stability class.

The NRC staff qualitatively reviewed the inputs to the ARCON96 calculations and found them generally consistent with the site configuration drawings, Figure A7-1 in Attachment (7) of the December 14, 2006, submittal, and Attachment (4) to the licensee's July 17, 2007, 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 has 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 NMP1.

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 (elevated MS and ground level RB and TB). 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 response to an NRC staff RAI, 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 period, using 60.7 meters wind data, the licensee performed sensitivity calculations using the RADionuclide Transport and Removal And Dose estimation (RADTRAD) computer code (discussed further in Section 3.2) and determined the doses were 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 Section 4.6, "Subroutine ENVLOP" of NUREG/CR-2858. Thus, the 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 (as noted in Table 3.2). Considering the licensee's compliance with dose limits and only a slight differential margin from the staff's comparative χ/Q values, these estimates were found acceptable for use in the dose assessment for the current LAR.

Page 3 of Attachment (7) of the December 14, 2006, submittal stated the distances to the NMP1 EAB and 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 830 meters and non-coastal χ/Q values were used to obtain the bounding atmospheric dispersion factors. The distance to the EAB for the coastal sectors, defined as the W, WNW, NW, NNW, N, NNE, NE, and ENE directions, were replaced with the minimum non-coastal distance of 830 meters.

Per 10 CFR 100.3, a licensee has the authority to determine all activities including exclusion or removal of personnel and property from within the exclusion area. In an NRC staff 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 July 17, 2007, the licensee stated that boating activity is prohibited out to 700 meters from the centerline of NMP1 in accordance with 33 CFR 165.911, “Security Zones; Captain of the Port Buffalo Zone.” Additionally, the licensee stated that the methodology of excluding coastal sectors in evaluating EAB and outer boundary of LPZ χ/Q values was approved for NMP, Unit No. 2 (NMP2) in Safety Evaluation Report NUREG-1047, 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 UFSAR. 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 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 χ/Q values at a distance of 700 meters (the distance to which boating activity is prohibited from the centerline of NMP1). The NRC staff found that the resultant χ/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 830 meter distance. The licensee considered overall site ground level EAB (830 meters) and outer boundary of LPZ (6116 meters overall) χ/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 short-term fumigation χ/Q for the west-southwest sector, LPZ 0-8 hours short-term fumigation χ/Q value in the south sector, and sector dependent non-fumigation χ/Q values in the south (LPZ 8-24 and LPZ 24-96 hours) and east (LPZ 96-720 hours) sectors were determined most limiting using PAVAN.

The licensee’s offsite χ/Q values, listed in Table 3.1.2, represent a change from those used in the current licensing basis. For the reasons noted above (e.g., the comparative staff calculations and meeting dose criteria), the staff finds these χ/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

RG 1.183 states that 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. The wind speed to be assumed is the 1-hour average value that is exceeded only 5% of the total number of hours in the data set. The licensee estimated the relevant NMPNS 95% wind speed as approximately 22 miles per hour (mph). The NRC staff confirmed the estimate of approximately 22 mph from the 1997 through 2001 wind data measured at the 9.4 meter level.

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
- MSLB
- CRDA

As a minimum effort to revise the NMP1 licensing basis to incorporate a full implementation of the AST, RG 1.183, Regulatory Position 1.2.1, “Full Implementation,” 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 NMP1.

The licensee’s letters dated December 14, 2006, and July 17, 2007, provided 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 CR. For full implementation of the AST DBA analysis methodology, the dose acceptance criteria specified in 10 CFR 50.67 provide an alternative to the previous whole body and thyroid dose guidelines stated in 10 CFR 100.11 and GDC 19. The current LAR is considered a full implementation of the AST.

RG 1.183, Regulatory Position 3.1, “Fission Product Inventory”, states:

> “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. The NRC staff finds the licensee’s use 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 6.3 kilowatt per foot (kw/ft) peak rod average power for burnups exceeding 54,000 MWd/MTU. The licensee stated that NMP1 meets these fuel criteria because no NMP1 fuel exceeds a burnup of 34,000 MWd/MTU.

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 December 14, 2006, including all supporting supplements.

3.2.1 LOCA

The current NMP1 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 NMP1 UFSAR, Section XV-C.2.0, "Loss-of-Coolant Accident." To support implementation of the AST, as requested by the subject LAR, the licensee reanalyzed the offsite and CR 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 NMP1 will remain adequate following implementation of the AST.

The licensee submitted the AST-based reanalysis of the LOCA as Attachments (8) and (10) of the December 14, 2006, submittal. Included in this reanalysis are the assumptions, parameters, and newly calculated offsite and CR doses associated with implementing the AST methodology. The licensee cited RG 1.183 as providing the primary radiological analysis assumptions for its 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 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 1,887 megawatts thermal (MWth), or 1.02 times the current licensed thermal power level of 1,850 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 NMP1. 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 assumed 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 reviewed the licensee's assessment of the following potential post-LOCA activity release pathways:

- Primary Containment (PC) Leakage Bypassing Secondary Containment
- Pre-drawdown PC Leakage Bypassing Secondary Containment
- Post-drawdown PC Leakage through Secondary Containment
- ESF Leakage
    - Bypassing Secondary Containment
    - Through Secondary Containment

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

- RB Cloud Shine to the CR
- External Plume Shine to the CR
- CR Filter Shine to the CR

Consistent with regulatory requirements, the licensee assumed a loss of offsite power (LOOP) concurrent with the design-basis LOCA. Subsequently, the licensee 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.

For releases into containment, the licensee assumed that activity released from the reactor coolant system is well-mixed between the drywell and the torus airspace volumes, 2 hours after the onset of gap release following the restoration of ECCS, which is 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 thermalhydraulic response of cooling water quenching the molten core and core debris in the PC will result in the drywell and torus airspace volumes becoming well-mixed. This assumption is acceptable for the "light bulb" and torus design of the Mark I containment at NMP1, as it is configured with downcomers from the drywell that extend below the surface of the torus 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 pool fluid.

By crediting the NMP1 liquid poison system (LPS) capability to introduce sodium pentaborate to act as a buffer into the reactor coolant, the licensee determined that the suppression pool pH remains above 7 for the duration of the accident. Therefore, in analyzing activity transport from containment, it was not necessary 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 with the remaining 5% as iodine and hydrionic acid, with not less than 1% of each as iodine and hydrionic acid. 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, 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." A detailed description of the formation of hydrochloric acid from the decomposed cable insulation is provided by the licensee's July 17, 2007, responses to the NRC staff's RAI. 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 NMP1, the minimum pH will be less than 7. 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 LPS. The primary function of the LPS is to shut down the reactor after control rod failure or during anticipated transient without scram event. However, since sodium pentaborate in the LPS is derived from a strong base and a weak acid, it has buffering properties and can 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 can be kept above 7 for 720 hours (30 days) after a LOCA.

The licensee's calculation consisted of two steps. In the first step, the licensee calculated pH of the suppression pool water without addition of the sodium pentaborate buffer. This calculation produced continuously decreasing values of pH, which became less then 7 for times greater than 9 to 10 hours after a LOCA. In the second step, the licensee buffered the suppression pool water making certain that the whole buffering solution was added and properly dispersed in the suppression pool prior to 9 hours after a LOCA. Before and during addition of the buffer, the presence of cesium hydroxide, will maintain pH higher than 7. The pH for a fully buffered suppression pool at 30 days after LOCA was calculated to be 7.8.

The licensee determined that the time for injection, transport and mixing of sodium pentaborate, before it becomes uniformly distributed in the suppression pool and starts fully exercising its buffering action, is 7.4 hours, which is within the time frame of 9 hours. For the remainder of the 30 day post-LOCA mission time, the pH in the suppression pool will be maintained by the added sodium pentaborate buffer from the LPS. This procedure allows the licensee to maintain suppression pool pH higher than 7 for the entire mission time of 30 days after a LOCA.

The NRC staff reviewed the 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 for 30 days following a LOCA. By maintaining the suppression pool pH above 7, 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 Bypassing Secondary Containment

The licensee modeled this pathway as leakage through the lines which penetrate the PC and the RB. It was postulated that leakage from the PC through the closed containment isolation valves (CIVs) in these penetrations would bypass the RB and RB emergency ventilation system (RBEVS) filters, thereby resulting in an unfiltered ground level release. As a design basis, this release pathway includes leakage through main steam isolation valves (MSIV) and the combined leakage from feedwater, torus vent, drywell vent, and emergency condenser vent and drain line CIVs. A total leakage of 100 standard cubic feet per hour (scfh), with a maximum of 50 scfh per main steam line (MSL), is assumed for MSIV leakage. For combined leakage of the remaining non-MSIV bypass pathways, a total of 41.5 scfh is assumed, which is consistent with the TS 6.5.7 containment leakage rate acceptance criteria. Upon NRC staff acceptance of the requested license amendment, these limits become the new licensing basis for NMP1.

For releases through this pathway, the licensee has taken credit for the mitigation of particulate radionuclide and elemental iodine activity. There are a number of mechanisms and processes used to model the mitigation and removal of the activity associated with these radionuclides, and in its LOCA analysis, the licensee considered the following:

- PC Spray Removal
- Natural Deposition in the PC
- Impaction in MSLs – Compact Streamline Impingement
- Impaction in MSLs – Orifice Plugging
- Sedimentation in MSLs

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, natural deposition, impaction, and sedimentation are all acting on the overall in-containment aerosol (and, indirectly, elemental for sprays and natural deposition) 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, as implemented for the design-basis NMP1 LOCA analysis, a contiguous model, used by the licensee to address this concern, is acceptable, because it accounts for series and parallel effects of each removal process. This is illustrated by the changing particulate geometric mean radius entering the MSLs and the changing particulate mass mean diameter shown, respectively, in the graphs of Figures 2 and 3 provided in the licensee's RAI response of July 17, 2007.

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

3.2.1.2.1.1 PC Spray Removal

The containment spray pumps will automatically start within 60 seconds of the reactor protection system (RPS) receiving both a high drywell pressure and a low-low reactor water level signal. Subsequently, sprays are assumed to begin at 75 seconds, and as a result, the drywell sprays are active prior to the onset of the gap activity release at 2 minutes. The containment spray pumps transfer coolant from the suppression pool into the drywell and torus airspace. The containment spray system is described in NMP1 UFSAR Sections VII-B and XVI-C.2.0. The containment spray system is designated as safety-related with operability requirements defined in NMP1 TS Section 3.3.7 and, therefore, is acceptable for credit in this DBA scenario.

To calculate particulate removal by the containment sprays, the licensee used four system-related parameters as code inputs. These four parameters are droplet size, spray flow rate, spray fall height, and sprayed volume. The licensee determined the droplet size from spray nozzle testing data. For the primary spray subsystem, the mass mean droplet size was determined to be 779.2 μm. For the secondary spray subsystem, the mass mean droplet size was determined to be 813.5 μm.

The spray flow rates are 6449 gallons per minute (gpm) for the primary spray subsystem and 6383 gpm for the secondary spray subsystem. The licensee conservatively credited only the secondary subsystem in the analysis, as it has both a larger mass mean droplet size and lower flow rate, both of which tend to reduce spray removal. To account for “drywell congestion”, the licensee multiplied the secondary spray flow rate by 0.67 for additional conservatism. Also, the fall height of 21.4 feet, used by the licensee, conservatively reflects a one-third reduction to account for “drywell congestion.”

The resulting particulate removal rate, or λ, was calculated and applied to the RADTRAD model. The NRC staff confirmed that the licensee’s calculated λs, which were submitted in this LAR and which characterize spray removal, compare reasonably well with those calculated by other methods found acceptable to the NRC staff and listed in RG 1.183 and SRP 6.5.2.

Though there is no specified maximum elemental iodine decontamination factor (DF) calculated (i.e., DF=200), as to be consistent with the guidance, the licensee does limit the elemental iodine removal rate to 20 hr^{-1}, which meets the elemental iodine spray λ limitation endorsed by the NRC staff guidance in SRP 6.5.2. Therefore, the licensee’s overall credit for spray removal is found acceptable to the NRC staff.

3.2.1.2.1.2 Natural Deposition in the PC

The licensee’s dose analysis assumed that natural deposition of particulate activity occurs in the PC. Rather than using a deterministic formula to calculate the removal associated with natural deposition in the containment (i.e., NUREG/CR-6189 or SRP 6.5.2), the licensee employed a proprietary code to comprehensively model this phenomenon along with the other credited activity removal phenomena due to sprays. The licensee stated, and the NRC staff agrees, that the effects of natural deposition are largely insignificant when compared to the activity removal of containment sprays. To confirm the conservatism of the licensee’s model, the NRC staff used the generally accepted simplified natural deposition model from NUREG/CR-6189, referred to as the Powers natural deposition model, as implemented in the RADTRAD dose consequence computer code. The NRC staff generally accepts use of the 10th percentile confidence interval (90 percent probability) natural deposition removal values implemented in the RADTRAD code, so that is what was used for the confirmatory calculation to support this evaluation. The Powers natural deposition model was derived by correlation of results of Monte Carlo uncertainty analyses of detailed models of aerosol behavior in the containment under accident conditions. The NRC staff modeled the combined removal by sprays and natural deposition in the PC and compared the resulting dose consequence to that calculated by the licensee. Also, the NRC staff assessed the individual activity removal attributable to these two mechanisms. The results confirmed that the effect of natural deposition is largely overshadowed by that of sprays, and that the licensee's model of overall PC activity removal is conservative. Therefore, the NRC staff finds the licensee's calculated removal in the PC to be acceptable.

3.2.1.2.1.3 Impaction in MSLs

In the design-basis LOCA analysis, the licensee reduced aerosol mass and activity, as well as elemental iodine activity, for releases through the MSLs and other lines with closed CIVs, by a DF of 2, attributed to a dynamic particulate phenomenon called impaction. To achieve this activity removal, the licensee credited the effects of a combination of two types of impaction taking place primarily at closed isolation valves, as described in its July 17, 2007, response to the NRC staff's RAI. The two types of impaction can be described as (1) compact streamline impingement and (2) orifice plugging. The licensee described the first, compact streamline impingement, as removal of aerosol as it is entering a passage. The second, orifice plugging, can be described as plugging, or clogging, within and at the entrance to small passages due to an accumulation of removed particulate. The licensee acknowledged that, because there are two closed MSIVs modeled in one MSL at NMP1, the effectiveness of impaction downstream of the MSIVs will be inherently reduced in that line, as the aerosol size distribution is also reduced. The licensee also acknowledged that the compact streamline impingement type of impaction alone may not necessarily result in the credited DF of 2; however, the licensee contended that crediting the orifice plugging impaction alone can result in such removal. The basis for the licensee's contention is that, on a mass basis, only a small fraction of the leaked particulate would plug the postulated leak path, thereby closing the pathway's opening and preventing any additional leakage from being released through that pathway. Considering this, the licensee maintained that a much larger DF, related to the complete clogging and prevention of any additional particulate from being transported through the MSIV, could be credited.

The NRC staff believes that, though there is merit to this plugging phenomenon and impaction in theory, there is not enough empirical evidence, directly related to the unique and hypothetical conditions associated with a design-basis LOCA event, to warrant full credit for such a considerable DF attributable to impaction. Therefore, the NRC staff does not generally endorse taking credit for impaction when modeling removal of particulates in main steam lines following a LOCA. However, the NRC staff does believe that enough evidence exists to verify the conservatism of a DF of 2 in the specific design-basis LOCA model at NMP1. The contribution of this impaction DF to the overall iodine activity decontamination, does not lead to an excessive overall credit for iodine removal in the MSLs. Based on the approximate DF of 4 that the licensee credits for removal by sedimentation (See Section 3.2.1.2.1.4), combined with this DF of 2, the licensee is assuming less than a 90% overall iodine removal efficiency in the steam lines. If this MSIV leakage pathway were modeled using a well-mixed model, as described and previously approved in AEB 98-03, "Assessment of Radiological Consequences for the Perry

Pilot Plant Application using the Revised (NUREG-1465) Source Term," December 9, 1998, the calculated activity removal in the MSLs would be analogous to that calculated by the licensee. Therefore, the NRC staff finds the overall iodine removal credited by the licensee to be acceptable, as modeled for NMP1.

3.2.1.2.1.4 Sedimentation in MSLs

The term sedimentation, or settling, refers to the gravity-driven phenomenon of particulate falling out of a gaseous suspension. The licensee credited sedimentation within the MSLs for the MSIV leakage contribution to PC leakage bypassing secondary containment. In the July 17, 2007, response to an NRC staff RAI, the licensee stated that the aerosol activity removal associated with sedimentation in the MSLs is approximately equal to a DF of 4. Effectively, this indicates that, with regard to sedimentation, the licensee is crediting the MSLs for approximately 75% filtration of aerosol activity. The NRC staff finds that this is a reasonable credit for aerosol activity removal in MSLs and is consistent with the calculation of such removal at various other plants using different models, including, but not limited to, the determination of activity removal efficiency based on the Monte Carlo assessment of aerosol settling velocities described in AEB 98-03. Because the sedimentation removal credited by the licensee's model correlates well with AEB 98-03 and other previously approved models, the NRC staff finds their removal to be acceptable.

3.2.1.2.2 Pre-drawdown PC Leakage Bypassing Secondary Containment

The NMP1 current design-basis containment leak rate (L_a) of 1.5 percent weight per day (% per day) at containment peak pressure, as reflected in the leak rate limit in NMP1 TS 3.3.3, was assumed in the AST LOCA re-analysis. This pathway, as analyzed by the licensee, models leakage from PC that occurs prior to establishing a sustained negative pressure in the RB and, therefore, is assumed to be released directly to the environment from the refueling floor elevation, potentially via the sheet-metal siding.

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 (or 2.0333 hours, as stated in the LAR) after 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 onset of gap release. This is acceptable because the licensee assumed the drywell sprays to be operating from the beginning of the accident. The licensee calculated removal by sprays in the same manner as discussed in Section 3.2.1.2.1.1. This PC leakage during drawdown is assumed to be released to the environment at ground level.

The licensee assumed that the release rate during drawdown corresponds to the PC leak rate of 1.5% per day from the drywell, and also from the torus airspace (after 122 minutes). The licensee conservatively assumed a drawdown time of 6 hours from the start of the DBA-LOCA in its analysis. In Attachment (4) to the December 14, 2006, LAR, the licensee showed that an RB pressure of -0.25 inches water gauge (WG) is actually achieved at approximately 5 hours. In the that same drawdown analysis, the licensee also indicated that, in addition, the RB pressure becomes negative at approximately 26 minutes and that there is a significant period when the pressure is -0.15 inches WG or less; this was shown to occur from approximately 67 minutes to 5 hours. The licensee's analysis is discussed in more detail next.

The licensee provided its analysis of the RB positive pressure period in Attachment (4) to the December 14, 2006, LAR; the positive pressure period is the period when a loss of offsite power causes a loss of RB negative pressure relative to the external atmospheric static pressure. The start of the EDGs followed by the start of the RBEVS returns the RB to a negative pressure. The time of positive pressure relative to the atmospheric static pressure is called the drawdown time.

The licensee's analysis of the RB positive pressure period in Attachment (4) to the LAR is a new licensing basis analysis for NMP1. The plant-specific drawdown calculations were performed using the GOTHIC 7.2a containment analysis software. The calculations utilize the GOTHIC subdivided volume capability to represent each building elevation and to model buoyancy effects, natural circulation flow paths, and building heat sinks.

The model was benchmarked to data collected during the performance of NMP1 TS Surveillance Requirement (SR) 4.4.1, which demonstrates the capability of each RBEVS train to maintain a negative pressure of at least 0.25 inches WG less than atmospheric pressure with a wind speed of zero and a maximum RB inleakage of 1600 cubic feet per minute (cfm). The GOTHIC model response without LOCA heat loads agrees with the RB surveillance test response of approximately 2 to 4 minutes.

The licensee stated that the following conservative conditions were included in the analysis:

- Loss of offsite power and failure of one of the two 100% capacity RBEVS trains to operate.

- Maximum RB inleakage allowed by TS 3.4.1.

- Design-basis post-LOCA RB heat loads, including maximum post-LOCA suppression pool heatup, operation of two core spray pump sets and one containment spray pump set, heat loads from the emergency condensers on the refuel floor elevation and from the spent fuel pool (assumed to be at a constant 90 °F based on manual restart of a spent fuel pool cooling pump), electrical heat loads from equipment required to operate to mitigate the LOCA, and solar heat loads.

- Winter atmospheric conditions based on onsite meteorological data collected for the 5-year period of 1997 through 2001 (consistent with the guidance provided in RG 1.183, Appendix A, Section 4.3). The use of the summer conditions results in a drawdown time that is approximately one-half that of the winter case and thus is less limiting.

The licensee's analysis shows an initial rapid rise in RB pressure following a LOCA event. As shown in Figures A4-1 and A4-2 of Attachment (4) of the LAR, the RB pressure in the area above the refuel floor elevation remains positive for approximately 26 minutes, decreases to -0.15 inches WG at approximately 67 minutes, and reaches -0.25 inches WG at approximately 5 hours. At elevations below the refuel floor, the positive pressure times and the times to achieve -0.25 inches WG are considerably shorter.

The NRC staff reviewed the licensee's analysis for drawdown calculations associated with a LOCA and determined that sufficient conservatism was utilized in the analysis. Based on the information presented in the licensee's submittal, as discussed above, the NRC staff finds the use of GOTHIC 7.2a to perform the containment drawdown calculations for the AST LOCA analysis acceptable.

3.2.1.2.3 Post-drawdown PC Leakage through Secondary Containment

This pathway, as analyzed by the licensee, models airborne releases from the PC to the RB, beginning after the assumed drawdown period of 6 hours.

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 after 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 onset of gap release. This is acceptable because the licensee assumed the drywell sprays to be operating from the beginning of the accident. The licensee calculated removal by sprays in the same manner as discussed in Section 3.2.1.2.1.1.

As stated above, the release rate during drawdown corresponds to the PC leak rate of 1.5 % per day from the drywell, and also from the torus airspace (after 122 minutes). After 24 hours, this PC leak is reduced by a factor of 2 to 0.75 % per day, based on a reduction in containment pressure, as is consistent with allowances in the guidance of RG 1.183. Activity released through this pathway is filtered by the RBEVS. The filtration efficiencies associated with this system are 95% for elemental iodine, 90% for organic iodine, and 95% for particulate forms of radionuclide activity. PC leakage following drawdown was assumed to occur as an elevated release to the environment via the MS.

3.2.1.2.4 ESF Leakage

For this pathway, the licensee modeled ESF activity leakage as a continuous 1200 gallon per hour (gph), or 2.67 cfm, volumetric flow rate from the suppression pool control volume to the RB. The licensee assumed that, during the first 30 minutes of the drawdown period, this activity is released directly to the environment at ground level. After 30 minutes, an increasing portion of the total flow was assumed to be released directly to the environment as an elevated release from the MS. The licensee performed a drawdown analysis to quantify the increasing ratio of leakage that leaves as an elevated release versus a ground level release. The licensee determined that, for the ground level releases, the ESF leak rate of 2.67 cfm was assumed to begin at the initiation of the LOCA, then it was reduced to 1.74 cfm at 0.5 hrs, 0.83 cfm at 0.7 hrs, 0.22 at 1.0 hr, 0.10 cfm at 122 minutes, stopping when drawdown is assumed to be complete at 6 hours. For the elevated release, a leak rate of 0.93 cfm is assumed to begin at 0.5 hrs, this then increases to 1.84 cfm at 0.7 hrs, 2.45 cfm at 1.0 hr, and 2.57 at 122 minutes. By 6 hours, the full 2.67 cfm is released to the RB volume, filtered by the RBEVS, and discharged via the MS.

The licensee provided justification for this ESF leakage model in Appendix A to the LOCA design analysis provided in Attachment (10) to the December 14, 2006, submittal. It is based on calculated capability for the RB to leak through surfaces versus through the RBEVS and out the MS. The licensee contended that, based on limiting atmospheric conditions outside of the NMP1 RB and a calculated 10% iodine re-evolution, the ratio of the volumetric flow that is able to leak from the RB during drawdown, to the total volumetric flow, is the fraction of the ESF leakage iodine re-evolution during drawdown that can be leaked from the RB at ground level. The remainder is assumed to be released, filtered and elevated to the MS, with the RBEVS exhaust flow rate, but with no holdup in the RB. This treatment is conservative and, therefore, acceptable to the NRC staff.

3.2.1.3 Direct Shine Dose

The licensee's evaluation of post-LOCA shine doses to CR personnel from the passing external activity plume, the CR air treatment system (CRATS) filters, and the RB airborne activity cloud was performed using the QADMOD code. QADMOD has been applied previously to analyses of this type for NMP1. The QADMOD results were independently verified by the licensee either manually or with the MicroShield code. Both the QADMOD and MicroShield codes are point-kernel integration codes used for general purpose gamma shielding analyses.

The complex geometries associated with the direct shine dose assessments performed for NMP1 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 models implement sufficient conservatism, such as the assumed source-receiver orientation and inclusion of Rb-88 activity, to compensate for potential non-conservative treatment of the modeled geometries by the chosen point-kernel codes. It is also of note that, at NMP1, the total direct shine dose contribution constitutes a small percentage of the total LOCA dose. Therefore, for the general application of these codes as implemented for the design-basis LOCA analysis at NMP1, the NRC staff finds the licensee's direct shine dose assessment acceptable.

3.2.1.4 TSC Dose Consequence Assessment

In the LAR, the licensee indicated that the TSC 30-day inhalation and immersion doses were 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 stated that this was done because the emergency ventilation and filtration system in the TSC is manually initiated by the first person to arrive; for an off-hours event, actuation could be delayed by up to 1 hour. For calculation of shine dose, the licensee stated that the direct shine doses to the TSC were based on a comparison to the AST shine doses for the CR. In its examination of post-LOCA TSC dose consequences, the licensee found that the 30-day doses do not exceed 5 rem TEDE. The licensee's analyses indicate, and the NRC staff agrees, that they comply with the regulatory guidance and requirements for the TSC as given in NUREG-0737 and Paragraph IV.E.8 of Appendix E to 10 CFR Part 50.

3.2.1.5 Conclusion

The licensee concluded that the radiological consequences at the EAB, LPZ, and CR 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 CR for the duration of the accident. The NRC staff finds that the licensee used sufficiently conservative analysis assumptions and inputs consistent with applicable regulatory requirements and guidance identified in Section 2.0, above, and with those stated in the NMP1 UFSAR 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. Also, the licensee's analyses indicate that they comply with the regulatory guidance and requirements for the TSC as given in NUREG-0737 and Paragraph IV.E.8 of Appendix E to 10 CFR Part 50.

3.2.2 FHA

The current NMP1 design-basis FHA analysis is based on the traditional accident source term described in TID-14844. The NMP1 licensing basis analysis is presented in UFSAR Section XV-C.3.0, "Refueling Accident." To support implementation of the AST, as requested by the subject LAR, the licensee reanalyzed the offsite and CR radiological consequences of the postulated FHA. This reanalysis was performed to demonstrate that the ESFs designed to mitigate the radiological consequences at NMP1 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 this reanalysis are the assumptions, parameters, and newly calculated offsite and CR doses associated with implementing the AST methodology. The licensee cited RG 1.183 as providing the primary radiological analysis assumptions for its reanalysis of the postulated design-basis FHA. Specifically, the NRC staff's guidance for analyses of the FHA is detailed in Appendix B of RG 1.183.

As analyzed for NMP1, the postulated FHA involves a 30' drop of a fuel assembly on top of other fuel assemblies in the 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 NMP1 refueling equipment configuration and is the limiting case because it results in the maximum release of fission products to the RB. Also, the licensee has determined that 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 FHA analysis, the licensee assumed that the core isotopic inventory available for release into containment is based on maximum full power operation of the core assemblies at 1,887 MWth, or 1.02 times the current licensed thermal power level of 1,850 MWth, in order to account for the ECCS evaluation uncertainty. Additionally, the burnup and enrichment parameters that were assumed when determining the core isotopic inventory are within current licensed limits for fuel at NMP1. 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% were assumed, respectively. 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 were assumed to be released. 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 delay before any movement of fuel can be initiated. Therefore, movement of fuel, or fuel handling, before this period of decay would be unanalyzed and not consistent with the assumed design-basis. The licensee assumed that 125 fuel pins, 63 for the dropped assembly and 62 for the struck assembly, will be damaged as the result of the postulated FHA, and will instantaneously release all of their available gap activity to the outside atmosphere, taking no credit for RB closure or isolation. This is a conservative treatment, since RG 1.183 allows for the assumption of a 2-hour release duration. Although NMP1 is currently using General Electric 11 (GE11) 9x9 fuel, the licensee stated that 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 NMP1 licensing basis, as described in NMP1 UFSAR Section XV-C.3.0, and the guidance in RG 1.183 and is, therefore, acceptable to the NRC staff.

3.2.2.2 Transport Methodology and Assumptions

As analyzed for NMP1, the postulated FHA involves a 30' drop of a fuel assembly on top of other fuel assemblies in the core during refueling operations. Even though the most limiting drop height and subsequent fuel damage is associated with a drop over the core, the licensee used the more conservative DF associated with the depth of water in the canal to the spent fuel pool. The minimum water coverage allowed by the NMP1 TS is 22'-9", which is less than the 23' water coverage required to assume an overall DF of 200 in accordance with the guidance of RG 1.183, Appendix B. Therefore, using the e^{Kd} relationship discussed in the licensee's July 17, 2007, RAI response, where "d" is equal to the water depth and "K" is a constant equal to 0.2458, the licensee interpolated an adjusted overall iodine DF of 191. 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 been previously found acceptable to the NRC staff. The licensee discussed the interpolation in the July 17, 2007, response to the NRC staff RAI regarding this topic. Noble gas activity is assumed to be released from the reactor vessel water without experiencing any reduction. The DF calculation by the licensee is conservative and acceptable to the NRC staff because it is consistent with the guidance of RG 1.183.

The licensee assumed that the release to the outside atmosphere is instantaneous and complete, and that transport to the receptor is instantaneous as well. So, no credit is taken for holdup or dilution in the containment volume, and exposure time and the duration of release are equivalent. Also, the licensee took no credit for secondary containment or CR isolation or filtration. In its 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 CR 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, and 5 rem for access to, and occupancy of, the CR for the duration of the accident. The NRC staff finds that the licensee used analysis assumptions and inputs consistent with applicable regulatory requirements and guidance identified in Section 2.0, above, and with those stated in the NMP1 UFSAR 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 MSLB Accident

The current NMP1 design-basis MSLB accident analysis is based on the traditional accident source term described in TID-14844. The NMP1 licensing basis analysis is presented in UFSAR Section XV-C.1.0, "Main Steam Line Break Outside the Drywell." To support implementation of the AST, as requested by the subject LAR, the licensee reanalyzed the offsite and CR radiological consequences of the postulated MSLB. This reanalysis was performed to demonstrate that the ESFs designed to mitigate the radiological consequences at NMP1 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 were the assumptions, parameters, and newly calculated offsite and CR doses associated with implementing the AST methodology. The licensee cited RG 1.183 as providing the primary radiological analysis assumptions for its 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 a double ended guillotine break of one main steam line outside the secondary containment with displacement of the pipe ends such that the maximum blowdown rates are 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 NRC staff agrees that the radiological consequences of an MSLB outside secondary containment will bound the consequences of an MSLB inside containment. Thus, only an MSLB outside of containment was considered with regard to the radiological consequences.

3.2.3.1 Activity Source

For the design-basis MSLB accident, the licensee assumed no fuel failure, consistent with the current NMP1 licensing basis, because the core is not postulated to become uncovered. Therefore, in order to determine the maximum offsite and CR 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 NMP1 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.

For this LAR, the licensee is seeking to reduce the TS 3.2.4.a maximum coolant equilibrium iodine activity concentration from 9.47 microcuries per gram (μCi/gm) total to 0.2 μCi/gm dose equivalent (DE) I-131. So, the maximum coolant equilibrium iodine activity concentration assumed in the MSLB accident analysis was 0.2 μCi/gm 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 this activity meets the lower acceptance criterion for the equilibrium activity case that is suggested in Table 6 of RG 1.183, and 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 because 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. 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 state that cesium activity be considered as a dose contributor.

3.2.3.2 Transport Methodology and Assumptions

As stated above, the licensee has defined the design-basis MSLB accident as a double ended guillotine break of one main steam line outside the secondary containment with displacement of the pipe ends such that the maximum blowdown rates are 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 an MSLB inside containment. Thus, only an MSLB outside of containment was considered with regard to the radiological consequences.

Consistent with the current NMP1 licensing basis, the licensee assumed break isolation in 11 seconds, corresponding to the maximum MSIV closing time of 10 seconds, plus an assumed closure signal delay time of 1 second. The licensee took no credit for reduction in break flow as the valves are closing. The total assumed coolant mass release is 107,150 pounds-mass (lbm), consisting of 80,900 lbm of liquid and 25,250 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 CR and the environment was assumed, and no credit was taken for CR filtration, other iodine removal mechanisms, or decay. The release modeled by the licensee was assumed to waft over the CR intake at a rate of 1 m/s, leaving it resident and contributing to dose for 136 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 its analysis of the dose consequences resulting from this design-basis MSLB accident. This spreadsheet was provided for NRC staff review as part of the MSLB accident design analysis in Attachments (8) and (10) to the licensee's December 14, 2006, submittal.

3.2.3.3 Conclusion

The licensee concluded that the radiological consequences at the EAB, LPZ, and CR 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 CR 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 CR for the duration of the accident. The NRC staff finds that the licensee used analysis assumptions and inputs consistent with the applicable regulatory requirements and guidance identified in Section 2.0, above, and with those stated in the NMP1 UFSAR 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 CRDA

The current NMP1 design-basis CRDA analysis is based on the traditional accident source term described in TID-14844. The NMP1 licensing basis analysis is presented in UFSAR Section XV-C.4.0, "Control Rod Drop Accident." To support implementation of the AST, as requested by the subject LAR, the licensee reanalyzed the offsite and CR radiological consequences of the postulated CRDA. This reanalysis was performed to demonstrate that the ESFs designed to mitigate the radiological consequences at NMP1 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 CR doses associated with implementing the AST methodology. The licensee cited RG 1.183 as providing the primary radiological analysis assumptions for the 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 the LAR, the licensee stated that NMP1 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) to occur.

3.2.4.1 Activity Source

For the CRDA analysis, the licensee assumed that the core isotopic inventory available for release is based on maximum full power operation of the core assemblies at 1,887 MWth, or 1.02 times the current licensed thermal power level of 1,850 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 NMP1. 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 850 fuel pins experience cladding failure following the postulated CRDA. No fuel is assumed to melt, however. The licensee stated that the failure of 850 pins for the GE 8x8 fuel assemblies in the core bounds the GE11 9x9 fuel assemblies. The licensee also stated that, as noted in NMP1 UFSAR Section XV-C.4.2, 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 boiling-water reactor reload package for BPWS plants. Therefore, the assumption of 850 failed fuel pins is acceptable to the NRC staff.

3.2.4.2 Transport Methodology and Assumptions

As stated above, 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 its LAR, the licensee stated that NMP1 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) to occur. The NMP1 AST analysis for the CRDA considered 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 was released 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.

Although the licensee did credit the elevated release associated with the MS for the MVP releases of Case 2, it did not credit secondary containment exhaust, or CR intake, filtration for activity removal in either case. To calculate accident dose consequences, the licensee used a spreadsheet for Case 1 and the RADTRAD code for Case 2. These calculations were provided for NRC staff review as part of the CRDA design analysis in Attachments (8) and (10) to the licensee's December 14, 2006, submittal.

In the July 17, 2007, 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 stated that any steam seal leakage would be small in comparison to the MVP exhaust flow, and thus was not specifically included in its analysis. The licensee also stated that, because the volumetric exhaust flow of the MVPs is more than 20 times greater than that of the steam jet air ejector (2000 cfm versus ~100 cfm maximum), 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 NMP1.

3.2.4.3 Conclusion

The licensee concluded that the radiological consequences at the EAB, LPZ, and CR 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 NMP1 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 CR for the duration of the accident. The NRC staff finds that the licensee used analysis assumptions and inputs consistent with applicable regulatory requirements and guidance identified in Section 2.0, above, and with those stated in the NMP1 UFSAR 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 CR Habitability and Modeling

The current NMP1 DBA analyses, as described in UFSAR Section XV-C, do not calculate CR dose; therefore, the CR dose model provided in the revised DBA analyses that support this AST-based LAR, represents a change in the NMP1 licensing basis.

For the revised analyses where CR isolation and/or filtration is credited, the licensee assumed a CRATS intake flow rate of 2250 cfm ±10%, and assumed 95% filtration efficiency for elemental, a 90% filtration efficiency for organic, and a 95% filtration efficiency for particulate forms of iodine. For conservatism, the lower flow uncertainty value, 2025 cfm, was used for modeling. Where CRATS is credited for CR filtration, the licensee assumed that the CR was automatically isolated on a LOCA signal, and that the filtration associated with CRATS was available from the onset of the accident.

In a letter dated January 31, 2005, from the licensee to the NRC staff (ADAMS Accession No. ML050460309), it is indicated that the highest measured unfiltered inleakage into the NMP1 CR is 45 cfm. For the DBA analyses that model actual NMP1 CR functionality, the licensee assumed an unfiltered inleakage of 100 cfm, to bound the as-tested, worst-case, unfiltered inleakage. This value is conservative and provides margin for future measurements of CR inleakage. The major parameters and assumptions used by the licensee for modeling of the CR, and found acceptable to the NRC staff, are presented in Table 3.2.5.

3.4 TS Changes

3.4.1 TS 1.16, Addition of Definition of "*Dose Equivalent I-131*"

The licensee proposed to add the definition of *Dose Equivalent I-131* to NMP1 TS 1.16. 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 inhalation dose conversion factors.

With the implementation of 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 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 addition of the TS 1.16 definition *Dose Equivalent I-131* acceptable.

3.4.2 TS 1.17, Addition of Definition of "*Recently Irradiated Fuel*"

The licensee proposed to add the definition of *Recently Irradiated Fuel* to NMP1 TS 1.17. Recently irradiated fuel is fuel that has occupied part of a critical reactor core within the previous 24 hours (i.e., reactor fuel that has decayed less than 24 hours following reactor shutdown). The new definition is consistent with the AST analysis for the design-basis FHA, which assumes a post-shutdown 24-hour decay period to determine the release activity inventory. Thus, this proposed definition of *Recently Irradiated Fuel* 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 addition of the TS 1.17 definition *Recently Irradiated Fuel* acceptable with respect to the radiological consequences of design-basis accidents.

Furthermore, the NRC staff finds this change acceptable because it is consistent with the guidance of TS Task Force (TSTF) traveler TSTF-51-A, "Revise Containment Requirements During Handling Irradiated Fuel and Core Alterations," Revision 2. Specifically, TSTF-51 revises containment requirements during handling irradiated fuel and core alterations by removing the TS requirements for ESF to be operable after sufficient radioactive decay has occurred to ensure off-site doses remain below the SRP limits (a small fraction of 10 CFR Part 100). Fuel movement could still proceed prior to the amount of decay occurring but only with the appropriate ESF systems operable. Associated with this change is the deletion of operability requirements during core alterations for ESF mitigation features. This change will allow plants the flexibility to move personnel and equipment and perform work which would affect containment operability during the handling of irradiated fuel.

TSTF-51 states that, following reactor shutdown, decay of short-lived fission products greatly reduces the fission product inventory present in irradiated fuel. The proposed changes are based on performing analyses assuming a longer decay period to take advantage of the reduced radionuclide inventory available for release in the event of an FHA. Following sufficient decay occurring, the primary success path for mitigating the FHA no longer includes the functioning of the active containment systems. Therefore, the operability requirements of the TS are modified to reflect that water level and decay time are the primary success path for mitigating the FHA.

To support this change in requirements during the handling of irradiated fuel, the operability requirements during core alterations for ESF mitigation features are deleted. The accidents postulated to occur during core alterations, in addition to FHAs, are: inadvertent criticality (due to a control rod removal error or continuous control rod withdrawal error during refueling or boron dilution), and the inadvertent loading of, and subsequent operation with, a fuel assembly in an improper location. These events are not postulated to result in fuel cladding integrity damage. Since the only accident postulated to occur during core alterations that results in a significant radioactive release is the FHA, the proposed TS requirements omitting core alterations is justified.

Also, the TS only allow the handling of irradiated fuel in the reactor vessel when the water level in the reactor cavity is at the high water level. Therefore, the proposed changes only affect containment requirements during periods of relatively low shutdown risk during refueling outages. Therefore, the proposed changes do not significantly increase the shutdown risk.

3.4.3 Revision to TS 3.1.2, LIQUID POISON SYSTEM (LPS)

The licensee proposed to revise the operability requirements for the LPS to include both the power operating condition, and whenever the reactor coolant system temperature is greater than 212 °F (except for reactor vessel hydrostatic or leakage testing with the reactor not critical). Also, the Objective statement is revised to reflect the new post-LOCA pH control function of the LPS. This change implements AST assumptions regarding the use of the LPS to control the suppression pool pH following a LOCA involving significant fission product release.

The NRC staff reviewed the LPS with respect to its role in delivery of sodium pentaborate to the suppression pool for pH control. As discussed 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. As such, the new role being assigned to the LPS is a safety-related role. The LPS is designated a safety-related system. The licensee stated that all of the LPS components required for the injection of sodium pentaborate solution into the reactor are classified safety-related.

As a safety-related system, the LPS 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, specifically:

- The system components required for reactivity control and new suppression pool pH control function are seismically qualified.

- The system is provided with emergency power with the capability to supply power from the EDGs.

- The system is inspected and tested in accordance with the American Society of Mechanical Engineers inservice inspection and inservice testing programs, as required by 10 CFR 50.55a, Codes and standards.

- The system is within the scope of the 10 CFR 50.65, Requirements for monitoring the effectiveness of maintenance at nuclear power plants (i.e., the Maintenance Rule).

- Most components (i.e., pumps, explosive valves, etc.) are redundant in parallel trains powered from different electrical busses. The exceptions are the containment isolation check valves and the selector switch in the main control room.

- Procedures will be updated to activate the LPS in 1.5 hours post-LOCA. The emergency operating procedures (EOPs) and the severe accident procedures (SAPs) will be revised, as appropriate, to reflect the post-LOCA function of the LPS and to assure that, once initiated, the entire contents of the LPS storage tank are injected to accomplish the pH control function.

- Training will be provided on the new LPS injection function as part of operator re-qualification training and EOP and SAP training.

The NRC staff considered components that could be subject to single failure. The licensee identified two components, the containment isolation check valves and the main control room selector switch. The containment isolation valves are 2-inch check vales, manufactured by the Chapman Division of the Crane Co., model number 1523V-WE. The LPS containment isolation check valves are tested each refueling outage. In the periodic inspections and testing of these valves, NMP1 has not experienced any failures of these valves to open on demand. A review of the industry databases, EPIX and NPRDS, was performed and no failures of check valves of this manufacturer and type failing to open were identified. Although acknowledging that a single failure to open of one of the two check valves could prevent LPS injection, the NRC staff has determined that the potential for failure is very low based on the quality as established by its procurement, periodic testing and inspection, and historical performance of the component. The NRC staff finds the use of a single penetration of the containment with the identified check valves, as described by the licensee, to be acceptable.

For the selector switch, the licensee's review of the industry databases, EPIX and NPRDS, identified no previously documented failures of this type of switch. The NRC staff acknowledged that the selector switch in the main control room could fail and prevent either train or both trains of injection from functioning. The NRC staff determined that the switch was a highly reliable component at an accessible location. The switch could easily be replaced or bypassed to start one of the LPS trains if the switch were to fail.

The NRC staff reviewed the transport of the sodium pentaborate from the reactor vessel to the suppression pool. The LPS injects the sodium pentaborate to the reactor vessel. The transport of reactor vessel contents including the sodium pentaborate to the suppression pool is by flow through the break (assumed to be a large recirculation pipe break) to the drains that feed the suppression pool. Core spray injection would provide water directly to the core. One train of flow would be 3,600 gpm. The water would flow downward in the core to the bottom of the lower shroud plate, where it joins the LPS flow. This flow would sweep the sodium pentaborate from the vessel, through the drywell to the suppression pool. The licensee provided schematic diagrams of the reactor vessel showing the LPS injection location and the flow path for the core spray injection that mixes with the sodium pentaborate and transports it from the vessel to the suppression pool. The large core spray flow combined with the relatively small LPS injection flow provides good mixing and a significant level of transport from the vessel.

The transporting time of the sodium pentaborate was determined by identifying each of the volumes that will tend to hold up the solution and delay its transport to the suppression pool. The maximum total time for the injection, transport, and mixing of 1,114 gallons of sodium pentaborate solution in the suppression pool is approximately 7.4 hours. Thus, in order to remain within the 9-hour time calculated for the pool pH to drop below 7.0 without buffering, the LPS must be initiated approximately 1.5 hours after the onset of the accident.

The NRC staff concluded that there would be mixing and transport and that it was reasonable to assume the concentration of sodium pentaborate in the core would equalize with the concentration in the suppression pool within an acceptable time after LPS injection. As a consequence, there would be sufficient pH control to deter and prevent iodine re-evolution.

The specific change being made to TS 3.1.2 is to ensure that the LPS is operable for all plant conditions when the reactor coolant system is above 212 °F (other than reactor vessel hydrostatic or leakage testing with the reactor not critical), such that the system is available to maintain the suppression pool pH above 7.0, consistent with the AST methodology and analysis assumptions. The exception for reactor vessel hydrostatic or leakage testing with the reactor not critical is consistent with previous approved analyses considering a large reactor coolant system line break occurring during the subject testing, in which fuel failure was not postulated. Also, the Objective statement in TS 3.1.2 is revised to reflect the new post-LOCA function of the LPS. On the basis of the above discussion, the NRC staff finds these changes acceptable.

3.4.4 Revision to TS 3.2.4, REACTOR COOLANT ACTIVITY

The licensee proposed to revise the reactor coolant radioactivity concentration limit specified in NMP1 TS 3.2.4.a from 9.47 μCi of total iodine per gm of water to 0.2 μCi/gm DE I-131. In addition, the licensee proposed a new TS section, 3.2.4.b, which would allow operation to continue for up to 48 hours if reactor coolant activity exceeds 0.2 μCi/gm DE I-131 but is less than or equal to 4.0 μCi/gm DE I-131. These revisions are consistent with the reactor coolant activity concentration assumed in the design-basis MSLB accident analysis as described in Section 3.2.3.1, above. The licensee's analysis of the MSLB accident used a source term based on the maximum short-term reactor coolant specific activity of 4.0 μCi/gm DE I-131 and resulted in calculated radiological consequences, shown in Table 3.2, which are below the applicable acceptance criteria discussed in Section 2.0. The licensee stated, and the NRC staff agrees, that AST analyses have determined that the design-basis MSLB accident is more limiting than the previously-analyzed small-break LOCA outside of the PC, which was the basis for the current TS reactor coolant activity concentration limit of 9.47 μCi/gm total iodine.

From the TS requirement and safety perspective, this proposed revision will implement a limit that is more conservative than the existing requirement, and the licensee stated that review of recent operating experience indicates that the reactor coolant specific activity remains well below the proposed revised limit. Therefore, the NRC staff finds the proposed revision to the NMP1 TS 3.2.4.a limit, and the addition of TS 3.2.4.b, to be acceptable with respect to the radiological consequences of design-basis accidents.

The licensee also proposed to change the title of TS 3.2.4 to REACTOR COOLANT SPECIFIC ACTIVITY to be more consistent with the content of the revised TS. The NRC staff finds this proposed administrative change acceptable.

3.4.5 Revision to TS 3.2.4/4.2.4, LIMITING CONDITION FOR OPERATION and SURVEILLANCE REQUIREMENT

To compliment the proposed TS changes discussed in Section 3.4.4, above, the licensee proposed to also update the limiting conditions for operation in TS 3.2.4 and the SRs in TS 4.2.4 in a manner similar to that of the Standard Technical Specifications (NUREG-1433). Because these changes are consistent with generic TS language with regard to the applicability of reactor coolant activity concentration limits, the NRC staff also finds these changes acceptable with respect to the radiological consequences of DBAs, as was the case for the propose changes discussed in Section 3.4.4, above.

3.4.6 Revision to TS 3.3.3, LEAKAGE RATE

Under the heading Objective for this TS, reference is made to 10 CFR Part 100 with regard to post-LOCA radiation exposure to the public. The licensee proposed to replace the reference to 10 CFR Part 100 with 10 CFR 50.67. With implementation of the AST, the accident whole body and thyroid dose guidelines of 10 CFR Part 100 are superseded by the TEDE criteria of 10 CFR 50.67. Therefore, upon approval of this LAR, this TS change is applicable and acceptable to the NRC staff.

3.4.7 Revision to TS 3.4.0, REACTOR BUILDING

The licensee proposed revising the operability requirements for maintaining RB (i.e., secondary containment) integrity by deleting the requirement that RB integrity be in effect during the refueling condition, replacing the term “irradiated fuel” with “recently irradiated fuel,” and adding “during operations with a potential for draining the reactor vessel (OPDRVs)” as a condition when RB integrity must be in effect. The proposed change implements AST methodology regarding radiological consequences of the design-basis FHA analysis. Formatting changes were also proposed to improve usability.

After reactor shutdown, decay of the short-lived fission products greatly reduces the fission product inventory present in irradiated fuel. The proposed changes are based on a specific decay period, which takes advantage of the reduced radionuclide inventory available for release in the event of an FHA. The decay period was calculated to be 24 hours. After 24 hours of decay time, movement of irradiated fuel assemblies can commence and continue without secondary containment integrity and without operability of the RBEVS, the RB isolation valves, or the CRATS.

Using the “recently irradiated fuel” concept discussed in Section 3.4.2 provides a mechanism for applying a cutoff in fission product decay to various TS where the concept applies. The duration of 24 hours has been shown by analysis to provide sufficient decay.

A new requirement that RB integrity be in effect during OPDRVs is being proposed, since OPDRVs are refueling activities that can be postulated to cause fission product release different than the FHA. Since secondary containment is the only barrier to the release of fission products into the environment during OPDRVs, RB integrity as defined in TS Definition 1.12 is required. In accordance with this definition, at least one door in each access opening must be closed (per TS 3.4.3), the RBEVS must be operable (per TS 3.4.4), and all RB ventilation automatic isolation valves must be operable or secured in the closed position (per TS 3.4.2). Additionally, the CRATS (TS 3.4.5) is required to be operable to assure control room habitability during OPDRVs. TS Sections 3.4.2, 3.4.3, 3.4.4, and 3.4.5 are revised to add operability requirements for OPDRVs.

By letter dated September 19, 2007, the licensee stated that the NMP1 shutdown safety procedure presently includes guidance on equipment availability during shutdown and contingency planning, as well as the requirements contained in the licensing and design-basis. As a defense-in-depth measure, to assure that actions are taken to limit the potential radiological consequences of the FHA, the licensee will revise the shutdown safety procedure to implement the guidelines of Section 11.3.6.5. of NUMARC 93-01, “Industry Guideline for Monitoring the

Effectiveness of Maintenance at Nuclear Power Plants," Revision 3. The licensee stated that these revisions will be incorporated prior to completing implementation of the AST license amendment, once approved, and will address the following attributes:

- A statement specifying that during fuel handling/core alterations, the ability to filter and monitor any release should be maintained. In particular, the RBEVS and its associated radiation monitors should be available but are not required to be operable.

- A statement specifying that the ability to restore secondary containment capability during fuel handling/core alterations should be maintained. A contingency method to immediately close any external openings in the secondary containment should be developed.

- A statement specifying that, when necessary, the Station Shift Manager will ensure that the necessary actions are taken to close all external openings in the secondary containment.

As stated in Section 3.2.2.2, the closing of these openings is not credited in the FHA analysis, and is not required to meet the dose release limits of the SRP. Nevertheless, for defense-in-depth, the licensee stated that the revisions to the shutdown safety procedure will be incorporated prior to implementation of this amendment.

On the basis of the above discussion, the NRC staff finds the proposed changes to TS 3.4.0, 3.4.2, 3.4.3, 3.4.4, and 3.4.5 and to the shutdown safety procedure acceptable.

3.4.8 Revision to TS 3.4.1, LEAKAGE RATE; TS 3.4.2, REACTOR BUILDING INTEGRITY - ISOLATION VALVES; TS 3.4.3, ACCESS CONTROL; and TS 3.4.4, EMERGENCY VENTILATION SYSTEM

The licensee proposed to revise TSs 3.4.1, 3.4.2, 3.4.3, and 3.4.4, as follows:

- For TS 3.4.1, the proposal changes, "Whenever the reactor is in the refueling or power operating condition or when the reactor coolant system temperature is above 215°F, the reactor building leakage rate as determined by Specification 4.4.1 shall not exceed 1600 cfm." to, "At all times when secondary containment integrity is required, the reactor building leakage rate as determined by Specification 4.4.1 shall not exceed 1600 cfm."

- For TS 3.4.1.a, the proposal replaces, "Suspend immediately irradiated fuel handling, fuel pool and reactor cavity activities, and irradiated fuel cask handling operations in the reactor building." with, "Suspend any of the following activities: 1. Handling of recently irradiated fuel in the reactor building, 2. Irradiated fuel cask operations in the reactor building, 3. Operations with a potential for draining the reactor vessel (OPDRVs)."

- For TS 3.4.2.a, the proposal changes, “The normal Ventilation System isolation valves shall be operable whenever the reactor is in the refueling or power operating conditions, when the reactor coolant system temperature is above 215°F, and whenever irradiated fuel or the irradiated fuel cask is being handled in the reactor building.” to, “The normal Ventilation System isolation valves shall be operable at all times when secondary containment integrity is required.”

- For TS 3.4.2.b, the proposal replaces, “If Specification 3.4.2.a is not met, the reactor shall be in the cold shutdown condition within ten hours and handling of irradiated fuel cask shall cease.” with, “If specification 3.4.2.a is not met, then the actions listed below shall be taken: 1. The reactor shall be in the cold shutdown condition within then hours, 2. Suspend any of the following activities: a. Handling of recently irradiated fuel in the reactor building, b. Irradiated fuel cask handling operations in the reactor building, c. Operations with a potential for draining the reactor vessel (OPDRVs)”

- For TS 3.4.3.a, the proposal changes, “Whenever the reactor is in the power operating condition, or when the reactor coolant system temperature is above 215°F, or when irradiated fuel is being handled in the reactor building, or during core alterations, or during irradiated fuel cask handling operations in the reactor building, the following conditions will be met: …” to, “At all times when secondary containment integrity is required, the following conditions will be met: …”

- For TS 3.4.3.b.2, the proposal deletes, “a. core alterations,”; changes “b.” from, “Handling of irradiated fuel in the reactor building,” to, “Handling of recently irradiated fuel in the reactor building,”; re-letters “b.” and “c.” to “a.” and “b.”; and adds a new “c.” which states, “Operations with a potential for draining the reactor vessel (OPDRVs).”

- For TS 3.4.4.e, the proposal changes the second paragraph from, “During refueling, from and after the date that one circuit of the emergency ventilation system is made or found to be inoperable for any reason, fuel handling is permissible during the succeeding seven days unless such circuit is sooner made operable, provided that during such seven days all active components of the other emergency ventilation circuit shall be operable. Fuel handling may continue beyond seven days provided the operable emergency ventilation circuit is in operation.” to, “During handling of recently irradiated fuel in the reactor building, handling of an irradiated fuel cask in the reactor building, and operations with a potential for draining the reactor vessel (OPDRVs), from and after the date that one circuit of the emergency ventilation system is made or found to be inoperable for any reason, recently irradiated fuel handling in the reactor building, irradiated fuel cask handling in the reactor building, or OPDRVs are permissible during the succeeding seven days unless such circuit is sooner made operable, provided that during such seven days all active components of the other emergency ventilation circuit shall be operable. Recently irradiated fuel handling in the reactor building, irradiated fuel cask handling in the reactor building, or OPDRVs may continue beyond seven days provided the operable emergency ventilation circuit is in operation.”

The NRC staff reviewed these proposed changes and finds them acceptable because they are consistent with TSTF-51, as discussed in Section 3.4.2, and with the AST analysis and proposed TS changes discussed previously in this SE.

3.4.9 Revision to TS 3.4.5, CONTROL ROOM AIR TREATMENT SYSTEM

The license proposed to revise the conditions for which TS Section 3.4.5 is applicable by deleting the refueling condition, replacing the term “irradiated fuel” with “recently irradiated fuel,” and adding “during operations with a potential for draining the reactor vessel (OPDRVs).” In addition, the action statements contained in TS 3.4.5.e and 3.4.5.f are revised to more clearly distinguish the actions to be taken during reactor operation versus those to be taken when handling recently irradiated fuel or an irradiated fuel cask in the RB or during OPDRVs.

Specifically, the licensee proposed that TS 3.4.5 be changed as follows:

- For TS 3.4.5.a, the proposal changes, “Except as specified in Specification 3.4.5e below, the control room air treatment system shall be operable during refueling and power operating conditions and also whenever irradiated fuel or the irradiated fuel cask is being handled in the reactor building.” to, “Except as specified in Specification 3.4.5e below, the control room air treatment system shall be operable for the following conditions: 1. Power operating condition, and whenever the reactor coolant system temperature is greater than 212°F, 2. Whenever recently irradiated fuel or an irradiated fuel cask is being handled in the reactor building, and 3. During operations with a potential for draining the reactor vessel (OPDRVs).”

- For TS 3.4.5.e, the proposal changes, “From and after the date that the control room air treatment system is made or found to be inoperable for any reason, reactor operation or refueling operations is permissible only during the succeeding sever days unless the system is sooner made operable.” to, “From and after the date that the control room air treatment system is made or found to be inoperable for any reason, reactor operation, recently irradiated fuel handling, irradiated fuel cask handling, or OPDRVs are permissible only during the succeeding seven days unless the system is sooner made operable.”

- For TS 3.4.5.f, the proposal changes, “If these conditions cannot be met, reactor shutdown shall be initiated and the reactor shall be in cold shutdown within 36 hours for reactor operations and refueling operations shall be terminated within two hours.” to, “If these conditions cannot be met, then the actions listed below shall be taken: 1. If in the power operating condition, or if the reactor coolant system temperature is greater than 212°F, reactor shutdown shall be initiated and the reactor shall be in cold shutdown within 36 hours. 2. Suspend any of the following activities within two hours: a. Handling of recently irradiated fuel in the reactor building, b. Irradiated fuel cask handling operations in the reactor building, c. Operations with a potential for draining the reactor vessel (OPDRVs).”

The NRC staff reviewed these proposed changes and found that they are consistent with the AST analysis and the proposed TS changes discussed previously in this SE, and are, therefore, acceptable.

3.4.10 Revision to TS Table 3.6.2j, EMERGENCY VENTILATION INITIATION and TS Table 3.6.2l, CONTROL ROOM AIR TREATMENT SYSTEM INITIATION

The licensee proposed to revise note (a) of TS Table 3.6.2j, which applies to the High Radiation Refueling Platform instrumentation, from, “This function shall be operable anytime that irradiated fuel or the irradiated fuel cask is being handled in the reactor building.” to, “This function shall be operable whenever recently irradiated fuel or an irradiated fuel cask is being handled in the reactor building, and during operation with a potential for draining the reactor vessel (OPDRVs).” The licensee also proposed that note (a) be made applicable to the High Radiation Reactor Building Ventilation Duct instrumentation in the refuel mode of operation.

In addition, in accordance with the licensee’s RAI response dated July 17, 2007, the licensee proposed adding to TS Table 3.6.2l a new note (c), “May be bypassed in the cold shutdown condition.” The proposed note will be applicable for Low-Low Reactor Water Level during shutdown and for High Drywall Pressure during shutdown.

The NRC staff’s assessment of these proposed changes found that they are consistent with the guidance of TSTF-51 and with the AST analysis and the proposed TS changes discussed previously in this SE. Therefore, the NRC staff finds the requested changes acceptable.

3.5 Renewed Operating License Change

The licensee proposed to delete Renewed Operating License Condition 2.C.(3). This license condition required the licensee to submit an application for a license amendment, including supporting analyses and evaluations by December 18, 1998, to demonstrate compliance with GDC 19 dose guidelines under accident conditions based upon system design and without reliance upon the use of KI. The licensee submitted the required analysis on December 18, 1998, but the license condition remains in effect. The AST analyses provided in this LAR demonstrate that the NMP1 control room operator dose for the most limiting DBA remains within the acceptance criteria of 10 CFR 50.67 without reliance on the use of KI. The NRC staff agrees with the deletion of the license condition.

3.6 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 regulatory requirements and guidance identified in Section 2.0. The NRC staff compared the doses estimated by the licensee to the applicable criteria identified in Section 2.0. The NRC staff also finds, with reasonable assurance that the licensee’s estimates of the CR, EAB, and LPZ doses will comply with these criteria. The NRC staff further finds reasonable assurance that NMP1, 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 SRs. 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 11390). 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 NMP1. In performing this review, the NRC staff relied upon information placed on the docket by the licensee, NRC staff experience in performing similar reviews and, where deemed necessary, on NRC staff confirmatory calculations. The NRC 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 NRC 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 NRC staff finds the methods and assumptions used by the licensee to be in compliance with applicable requirements. The NRC staff compared the doses estimated by the licensee to the applicable acceptance criteria and to the results estimated by the NRC staff in its confirmatory calculations. The NRC 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 NMP1, 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 NRC 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 NMP1 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 NMP1 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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Table 3.1.1

NMP1 CR and TSC Atmospheric Dispersion Factors

[a]Source / Receptor	χ/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
RB/CR	4.82E-4	2.61E-4	9.25E-5	6.70E-5	4.93E-5
TB/CR	1.03E-3	5.85E-4	2.07E-4	1.75E-4	1.52E-4
MS/CR	2.27E-4	1.26E-4	4.30E-5	3.58E-5	2.59E-5
RB/TSC	7.09E-4	5.60E-4	2.34E-4	1.71E-4	1.41E-4
TB/TSC	5.91E-4	4.26E-4	1.63E-4	1.35E-4	1.16E-4
MS/TSC	3.47E-4	2.42E-4	8.22E-5	6.06E-5	5.00E-5

[b]SOURCE/ RECEPTOR	MSLB χ/Q VALUE (sec/m^3)
TB/CR	9.98E-4
TB/TSC	9.80E-4

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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 CR; TSC = Technical Support Center

b NMP1 assumed an instantaneous ground level puff release and transport χ/Q value for the MSLB event.

Table 3.1.2

NMP1 Offsite Atmospheric Dispersion Factors

Offsite Dose Location		[a]χ/Q Values 0-2 Hours sec/m³	0-8 Hours sec/m³	8-24 Hours sec/m³	24-96 Hours sec/m³	96-720 Hours sec/m³
Ground Release	EAB	1.90E-4	----	----	----	----
	LPZ	------	1.63E-5	1.10E-5	4.67E-6	1.67E-6
[b]Elevated Release	EAB	5.98E-5	----	----	----	----
	LPZ	----	2.12E-5	8.40E-7	3.45E-7	1.11E-7

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 NMP1

	CR		[1]EAB		LPZ	
Design-Basis Accident	**[2]Total Dose**	**Acceptance Criteria**	**[3]Total Dose**	**Acceptance Criteria**	**[4]Total Dose**	**Acceptance Criteria**
	(rem TEDE)	**(rem TEDE)**	**(rem TEDE)**	**(rem TEDE)**	**(rem TEDE)**	**(rem TEDE)**
LOCA	4.81E+00	*5.0*	9.02E+00	*25*	1.60E+00	*25*
FHA	8.47E-01	*5.0*	4.47E-01	*6.3*	3.84E-02	*6.3*
MSLB	1.76E+00	*5.0*	5.30E-01	*25*	4.50E-02	*25*
CRDA						
Case 1	6.10E-01	*5.0*	6.30E-01	*6.3*	5.40E-02	*6.3*
Case 2	1.60E+00	*5.0*	3.40E-01	*6.3*	2.10E-01	*6.3*

1 The licensee calculated the EAB dose for the worst 2-hour period of the accident duration.
2 The licensee's CR dose results have been rounded to three significant digit precision.
3 The licensee's EAB dose results have been rounded to three significant digit precision.
4 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	1887
Primary Containment Volume, ft^3	
Drywell Airspace	180,000
Torus Airspace (minimum)	120,000
Suppression Pool (minimum)	79,700
Secondary Containment Volume, ft^3	2,100,000
Spray Delay time, min	0
Spray Flow Rate, gpm	6383
Drywell and Wetwell Airspace Mixing Initiation, min	122
Primary Containment Leakage Rate, weight % per day	
0 to 24 hrs	1.50
24 hrs to 30 days	0.75
Spray Removal in Containment	
Particulate DF	
0.0 – 0.0309 hrs	52.895
0.0309 – 0.4963 hrs	19.125
0.4963 – 0.5236 hrs	31.616
0.5236 – 0.8003 hrs	22.682
0.8003 – 1.9848 hrs	19.849
1.9848 – 2.7268 hrs	4.936
2.7268 – 4.2805 hrs	2.975
4.2805 – 14.99 hrs	2.01
Elemental DF	
0.0 – 0.0309 hrs	20.0
0.0309 – 0.4963 hrs	19.125
0.4963 – 0.5236 hrs	20.0
0.5236 – 0.8003 hrs	20.0
0.8003 – 1.9848 hrs	19.849
1.9848 – 2.7268 hrs	4.936
2.7268 – 4.2805 hrs	2.975
4.2805 – 14.99 hrs	2.01
MSL Volume (per line), ft^3	82.4

Table 3.2.1 (cont'd)

Parameter	Value
MSIV Leakage Rate, scfh Per MSL Total	 50 100
Total Other CIV Bypass Leakage Rate, scfh	41.5
Aerosol Removal in Steam Lines, DF	~8
ESF Leakage Iodine Re-Evolution, %	10
ESF Leakage Iodine Release Species, % Elemental Organic	 97 3
RBEVS Filter Efficiency, % Aerosol/Particulate Elemental Organic	 95 95 90
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	1887
Peaking Factor	1.8
Number of Failed Fuel Pins	125
Fuel Decay Time, hr	24
Fraction of Core Inventory in Fuel Gap	
Kr-85	0.10
I-131	0.08
Other Noble Gases	0.05
Other Iodines	0.05
Minimum Water Depth Above Damaged Fuel, ft	22.75
Iodine Decontamination Factor	191
Iodine Speciation in Fuel Gap, %	
Elemental	99.85
Organic	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	1887
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	11
Coolant Mass Blowdown, lbm Liquid Steam Total	 80,900 26,250 107,150
Time for Puff to Traverse CR Intake, sec	136
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	1887
Peaking Factor	1.8
Failed Fuel Fuel Melt, pins Cladding Failure, pins Total Failure Fraction	 0 850 0.02577
Fraction of Core Inventory in Fuel Gap Noble Gas Iodine Alkali Metals Other Halogens	 0.10 0.10 0.14 0.05
Iodine Speciation from Failed Fuel, % Elemental Organic Aerosol/Particulate	 4.85 0.15 95.0
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	50,000
Main Condenser Leakage Rate, % per day	1.0
Main Condenser MVP Exhaust Rate, lbm per hr	280,000
Atmospheric Dispersion Factors	Tables 3.1.1 and 3.1.2

Table 3.2.5

Key Parameters Used in Modeling the CR for Design-Basis Radiological Consequence Analyses

Parameter	Value
CR Volume, ft^3	135,000
Normal Intake Rate, cfm	2250 ± 10% (2025 used in analysis)
CRATS Intake Rate, cfm	2250 ± 10% (2025 used in analysis)
CRATS Initiation Delay, min	0
CRATS Filter Efficiency, % Elemental Organic Aerosol/Particulate	 95 90 95
Unfiltered Inleakage, cfm	100
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