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Constellation Energy
Nine Mile Point Nuclear Station

May 31, 2007

U. S. Nuclear Regulatory Commission
Washington, DC 20555-0001

ATTENTION: Document Control Desk

SUBJECT: Nine Mile Point Nuclear Station
Unit No. 2; Docket No. 50-410

License Amendment Request Pursuant to 10 CFR 50.90:
Application of Alternative Source Term

REFERENCES:

(a) J. J. DiNunno et al., Technical Information Document (TID)-14844, "Calculation of Distance Factors for Power and Test Reactor Sites," U.S. Atomic Energy Commission (now USNRC), 1962

(b) NRC Regulatory Guide 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors," July 2000

(c) Letter from W. C. Holston (NMPNS) to Document Control Desk (NRC), dated January 31, 2005, Response to Generic Letter 2003-01, Control Room Habitability (TAC Nos. MB9825, MB9826)

(d) Letter from T. J. O'Connor (NMPNS) to Document Control Desk (NRC), dated January 27, 2006, Response to NRC Generic Letter 2003-01, Control Room Habitability – Commitment Completion Date Change (TAC Nos. MB9825 and MB9826)

Pursuant to 10 CFR 50.90, Nine Mile Point Nuclear Station, LLC, (NMPNS) hereby requests an amendment to Nine Mile Point Unit 2 (NMP2) Renewed Operating License NPF-69. The proposed amendment would revise the accident source term in the design basis radiological consequence analyses in accordance with 10 CFR 50.67, which requires that a licensee who seeks to revise its current accident source term apply for a license amendment under 10 CFR 50.90. The proposed accident source term revision replaces the current methodology that is based on TID-14844 (Reference a) with the alternative source term methodology described in Regulatory Guide 1.183 (Reference b). This submittal fulfills the

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NMPNS commitment for completing and submitting the analysis needed to meet Generic Letter 2003-01 objectives (References c and d).

This license amendment request is for full implementation of the alternative source term (AST) as described in Reference (b), with the exception that TID-14844 (Reference a) will continue to be used as the radiation dose basis for equipment qualification and vital area access. Proposed changes in the licensing basis for NMP2 resulting from AST application include the following:

- Revision of the Technical Specification (TS) definition of Dose Equivalent I-131 to be consistent with the AST analyses.

- TS changes that reflect revised design requirements regarding the use of the Standby Liquid Control System to buffer the suppression pool pH to prevent iodine re-evolution following a postulated design basis loss of coolant accident (LOCA).

- Revisions to the TS operability requirements for the Control Room Envelope Filtration System and the Control Room Envelope Air Conditioning System, consistent with the assumptions contained in the AST Fuel Handling Accident (FHA) analysis. The AST FHA analysis does not take credit for operation of these systems during the movement of irradiated fuel and during core alterations.

- Credit for operation of the residual heat removal system in the drywell spray mode for the post-LOCA removal of airborne elemental iodine and particulates from the drywell atmosphere.

The renewed operating license currently allows NMP2 to operate at a maximum reactor core power level of 3,467 megawatts thermal (MWt). NMPNS is considering an extended power uprate (EPU) project that would increase the maximum licensed reactor core power level to 3,988 MWt. Therefore, the AST analyses have been performed using a bounding core isotopic inventory that is based on operation at 3,988 MWt.

The description and technical basis of the proposed change are contained in Attachment (1) and the other attachments referenced therein. The proposed TS changes are shown in the markup in Attachment (2). Associated TS Bases page markups are shown in Attachment (3). The TS Bases changes are provided for information only and will be processed in accordance with the NMP2 TS Bases Control Program (TS 5.5.10). The detailed calculations that contain input data, assumptions, and analysis methodologies are provided in Attachment (7).

Attachment (1), Section A1-9, provides a list of regulatory commitments contained in this submittal. Following NRC approval, the NMP2 Updated Safety Analysis Report (USAR) will be updated to reflect the AST analyses in accordance with 10 CFR 50.71(e) as part of the regular USAR update process.

NMPNS requests approval of this request in a timely manner, with implementation within 120 days of receipt of the approved amendment. This implementation period will provide adequate time to complete implementation activities using the appropriate change control processes.

Pursuant to 10 CFR 50.91(b)(1), NMPNS has provided a copy of this license amendment request, with attachments, to the appropriate state representative.

Should you have any questions regarding the information in this submittal, please contact T. F. Syrell, Acting Licensing Director, at (315) 349-7198.

Very truly yours,

Kevin J. Nietmann
Acting Vice President Nine Mile Point

STATE OF NEW YORK :
: **TO WIT:**
COUNTY OF OSWEGO :

I, Kevin J. Nietmann, being duly sworn, state that I am Acting Vice President Nine Mile Point, and that I am duly authorized to execute and file this request on behalf of Nine Mile Point Nuclear Station, LLC. To the best of my knowledge and belief, the statements contained in this document are true and correct. To the extent that these statements are not based on my personal knowledge, they are based upon information provided by other Nine Mile Point employees and/or consultants. Such information has been reviewed in accordance with company practice and I believe it to be reliable.

Subscribed and sworn before me, a Notary Public in and for the State of New York and County of Oswego, this 31st day of May, 2007.

WITNESS my Hand and Notarial Seal:

Sandra A. Oswald
Notary Public

My Commission Expires:

5/31/07
Date

KJN/DEV/kms

SANDRA A. OSWALD
Notary Public, State of New York
No. 01OS6032276
Qualified in Oswego County
Commission Expires 10-25-09

Attachments:

(1) Technical Basis and No Significant Hazards Determination
(2) Proposed Technical Specification Changes (Mark-up)
(3) Changes to Technical Specification Bases (Mark-up)
(4) Suppression Pool pH Control in the Event of a Design Basis LOCA
(5) Evaluation of SLC System Injection Flow Transport and Mixing
(6) Calculation of New Atmospheric Dispersion Factors
(7) Enclosed Calculations for Alternative Source Term

cc:

S. J. Collins, NRC (without Attachment 7)
M. J. David, NRC
Resident Inspector, NRC (without Attachment 7)
J. P. Spath, NYSERDA (without Attachment 7)

ATTACHMENT (1)

TECHNICAL BASIS AND

NO SIGNIFICANT HAZARDS DETERMINATION

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A1-1. DESCRIPTION

The proposed amendment revises the accident source term in design basis radiological consequence analyses for Nine Mile Point Unit 2 (NMP2). The proposed revisions to NMP2 Renewed Operating License NPF-69 are supported by the results of the revised design basis accident (DBA) analyses that have been performed to implement the revised accident source term. This submittal fulfills our commitment in References A1-8.1 and A1-8.2 for completing and submitting the radiological analysis needed to meet Generic Letter 2003-01, "Control Room Habitability," objectives.

This application is submitted, in part, pursuant to Nuclear Regulatory Commission (NRC) regulation 10 CFR 50.67 that states: "A licensee who seeks to revise its current accident source term in design basis radiological consequence analyses shall apply for a license amendment under § 50.90." Section 50.67 further states: "The application shall contain an evaluation of the consequences of applicable design basis accidents previously analyzed in the safety analysis report." Additionally, 10 CFR 50.67 sets new acceptance criteria for radiological consequences based on total effective dose equivalent (TEDE), replacing the traditional whole body and thyroid dose guidelines stated in 10 CFR Part 50, Appendix A, General Design Criterion (GDC) 19 and 10 CFR 100.11. For NMP2, the following four bounding DBAs were re-analyzed for this application:

1. Loss of Coolant Accident (LOCA)
2. Main Steam Line Break (MSLB) accident,
3. Fuel Handling Accident (FHA), and
4. Control Rod Drop Accident (CRDA).

The proposed accident source term revision follows the guidance in Regulatory Guide (RG) 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors" (Reference A1-8.3). The accident source term discussed in RG 1.183 is herein referred to as the Alternative Source Term (AST). RG 1.183 permits full or selective implementation of the AST characteristics. This license amendment request is for full implementation of the AST as described in RG 1.183, with the exception that the current methodology of Technical Information Document (TID)-14844, "Calculation of Distance Factors for Power and Test Reactor Sites" (Reference A1-8.4) will continue to be used as the radiation dose basis for equipment qualification and vital area access. Full implementation of the AST is a modification of the facility design basis that addresses all characteristics of the AST; that is, composition and magnitude of the radioactive material, its chemical and physical form, and the timing of its release. Full implementation revises the plant licensing basis to specify the AST in place of the previous accident source term and establishes the TEDE dose as the new acceptance criteria. This applies not only to the analyses performed in the application (which may only include a subset of the plant analyses), but also to all future design basis analyses.

Approval of this proposed change will provide a source term for NMP2 that will result in a more accurate assessment of the DBA radiological doses. The improved dose assessment results in revisions to some current licensing basis requirements. The proposed changes to the NMP2 Technical Specifications (TS) are described in the following section.

A1-2. PROPOSED CHANGE

The license amendment request revises the NMP2 licensing basis to fully implement the RG 1.183 AST. As indicated in Section A1-1 above, implementation of AST for NMP2 consists of reevaluation of the applicable DBAs (LOCA, MSLB accident, FHA, and CRDA) using the AST and the 10 CFR 50.67

TEDE acceptance criteria. The proposed license amendment also revises certain TS requirements that are associated with and justified by the analyses performed to support the AST. The proposed TS changes are described below and are indicated on the mark-up pages provided in Attachment (2). Associated TS Bases page markups are shown in Attachment (3). The TS Bases changes are provided for information only and will be processed in accordance with the NMP2 TS Bases Control Program (TS 5.5.10).

A1-2.1 Technical Specification Changes

A1-2.1.1 TS 1.0, Definitions

The current definition for DOSE EQUIVALENT I-131 is revised to delete the word "thyroid" and to replace the references to TID-14844, Regulatory Guide 1.109, Rev. 1, and ICRP 30 with a reference to Federal Guidance Report No. 11, 1988. The proposed revised definition is as follows:

> DOSE EQUIVALENT I-131 shall be that concentration of I-131 (microcuries/gram) that alone would produce the same dose as the quantity and isotopic mixture of I-131, I-132, I-133, I-134, and I-135 actually present. The dose conversion factors used for this calculation shall be the Committed Effective Dose Equivalent dose conversion factors listed in Table 2.1 of Federal Guidance Report No. 11, EPA, "Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion," 1988.

A1-2.1.2 TS 3.1.7, Standby Liquid Control (SLC) System

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

A1-2.1.3 TS 3.3.7.1, Control Room Envelope Filtration (CREF) System Instrumentation

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

A1-2.1.4 TS 3.7.2, Control Room Envelope Filtration (CREF) System

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

A1-2.1.5 TS 3.7.3, Control Room Envelope Air Conditioning (AC) System

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

A1-3. BACKGROUND

The current NMP2 licensing basis utilizes a source term that is based on TID-14844 (Reference A1-8.4) to calculate the radiological consequences of postulated design basis accidents. In response to NRC Generic Letter 2003-01 (References A1-8.1 and A1-8.2), Nine Mile Point Nuclear Station, LLC (NMPNS) indicated that reanalysis of applicable accident scenarios in Chapter 15 of the NMP2 Updated Safety Analysis Report (USAR), using AST methodology, would be used to demonstrate control room habitability. To that end, this submittal contains the reanalysis and licensing basis changes necessary to meet Generic Letter 2003-01 objectives. Use of AST methodology increases the design basis unfiltered inleakage into the control room envelope to a value larger than that previously observed in the tracer gas testing.

The fission product release from the reactor core into primary containment following a DBA is referred to as the "source term." The source term is characterized by the composition and magnitude of the radioactive material, the chemical and physical properties of the material, and the timing of the release from the reactor core. Since the publication of TID-14844, significant advances have been made in understanding the composition and magnitude, chemical form, and timing of fission product releases from severe nuclear power plant accidents. Many of these insights developed out of the major research efforts started by the NRC and the nuclear industry after the accident at Three Mile Island.

In 1995, NUREG-1465 (Reference A1-8.6) was published with revised ASTs for use in the licensing of future Light Water Reactors (LWRs). This NUREG represents the result of decades of research on fission product release and transport in LWRs under accident conditions. On December 23, 1999, the NRC issued the final rule on "Use of Alternative Source Terms at Operating Reactors." The final rule, issued as 10 CFR 50.67, "Accident Source Term," allows holders of operating licenses issued prior to January 10, 1997, to voluntarily replace the traditional source term used in DBA analyses with alternative source terms such as the one described in NUREG-1465. One of the major insights summarized in NUREG-1465 involves the timing and duration of fission product releases.

The five release phases describing the progression of a severe accident in a LWR are listed in NUREG-1465 and are given below:

1. Coolant Activity Release
2. Gap Activity Release
3. Early In-vessel Release
4. Ex-vessel Release
5. Late In-vessel Release

Phases 1, 2, and 3 are considered in current (i.e., pre-AST) DBA evaluations; however, they are all assumed to occur instantaneously. Phases 4 and 5 are related to severe accident evaluations. Under the AST methodology, only the coolant activity release (i.e., Phase 1) is assumed to occur instantaneously and ends with the onset of the gap activity release (i.e., Phase 2). This approach represents a more realistic time sequence for activity release. The insights from NUREG-1465 were subsequently incorporated into RG 1.183.

A1-4. TECHNICAL ANALYSIS

A1-4.1 Radiological Consequence Analyses

NMPNS has performed radiological consequence analyses of the DBAs documented in Chapter 15 of the NMP2 USAR that potentially result in the most significant control room and offsite exposures. These analyses were performed to support full scope implementation of AST. The AST analyses have been performed in accordance with the guidance in RG 1.183 and Standard Review Plan (SRP) Section 15.0.1 (Reference A1-8.7). Acceptance criteria consistent with those required by 10 CFR 50.67 and RG 1.183, Table 6, were used to replace the current design basis source term acceptance criteria. The following NMP2 DBAs were addressed:

- Loss of Coolant Accident (LOCA), USAR Section 15.6.5
- Main Steam Line Break (MSLB) Accident, USAR Section 15.6.4
- Fuel Handling Accident (FHA), USAR Section 15.7.4
- Control Rod Drop Accident (CRDA), USAR Section 15.4.9

The AST analyses included the following:

1. Identification of the core source term based on plant specific analysis of core fission product inventory.
2. Determination of the release fractions.
3. Analysis of new atmospheric dispersion factors (X/Q values) for the radiological propagation pathways.
4. Calculation of fission product deposition rates and removal efficiencies.
5. Calculation of offsite and control room personnel TEDE doses.
6. Evaluation of suppression pool pH to ensure that the iodine deposited into the suppression pool during a DBA LOCA does not re-evolve and become airborne as elemental iodine.
7. Evaluation of other related design and licensing bases such as NUREG-0737, "Clarification of TMI Action Plan Requirements" (Reference A1-8.9).

In addition, the doses in the NMP2 control room due to Nine Mile Point Unit 1 (NMP1) accidents, and the doses in the NMP1 control room due to NMP2 accidents, have been evaluated.

The analysis methodology, assumptions, and inputs for radionuclide release, transport, and removal for each of the analyzed DBAs are described in the following sections. An assessment of conformance with the guidance provided in RG 1.183 is provided in Tables A1-1 through A1-5.

A1-4.1.1 Evaluation Methodology

A summary of the computer codes used in the AST analyses is provided in Table A1-6. Summary descriptions of these codes are provided below or in the section describing the specific DBA where the code is utilized.

A1-4.1.1.1 Fission Product Inventory

The ORIGEN2 code (Reference A1-8.10) was used to calculate plant-specific fission product inventories which bound the effect of 24-month fuel cycles. The current licensed thermal power is 3,467 MWt. The DBA radiological analyses in this submittal have been performed for an assumed power level of 3,988 MWt (plus the current accident analysis design basis allowance of 2% for instrument uncertainty), in order to accommodate a potential future extended power uprate. Currently, the core is General Electric GE-11 and GE-14 fuel (see Table A1-7 for fuel data). Bounding values of fission product activity were determined for each radionuclide in the DBA radiological analyses by considering enrichment and exposure. Fission product activities were calculated for immediately after shutdown and decayed for the required times. The shutdown values are shown in Table A1-8.

A1-4.1.1.2 Dose Assessment

The RADTRAD computer code Version 3.03 (Reference A1-8.11) was used for the LOCA and CRDA calculations. Due to simplifying and conservative assumptions, a spreadsheet was used to calculate doses for the MSLB accident, the FHA, and one of the two CRDA cases. The computer code STARDOSE (Reference A1-8.12) was used to check the RADTRAD results. The RADTRAD and STARDOSE programs are radiological consequence analysis codes used to determine post-accident doses at offsite and control room locations. The STARDOSE code is the proprietary property of Polestar Applied Technology, Inc. The NRC has previously reviewed results obtained from the application of the STARDOSE code as part of the Vermont Yankee, Browns Ferry, and Columbia Generating Station AST applications (References A1-8.23, A1-8.24, and A1-8.25, respectively).

The evaluation of post-LOCA shine doses to control room personnel from the passing plume, the CREF system filters, and the reactor building airborne activity was performed using the QADMOD code (Reference A1-8.14). The shine dose results are those calculated as part of the current licensing basis analyses. The applicability of the current licensing basis shine doses has been demonstrated by comparing the current licensing basis integrated gamma energy (MeV) for each photon energy group with that calculated for the AST using the MicroShield code (Reference A1-8.13). The MicroShield code is a point kernel integration code used for general purpose gamma shielding analysis. Direct shine from the Standby Gas Treatment System (SGTS) filters to the control room is neglected due to distance and shielding, as the SGTS filters are located on the opposite side of the reactor building (see Attachment 6, Figure A6-1).

A1-4.1.1.3 Containment Activity Removal

Credit is taken for the reduction of airborne activity in the primary containment due to drywell sprays (RG 1.183, Appendix A, Sections 3.2 and 3.3). No credit is taken for natural deposition of activity in the primary containment.

For spray removal (applied to activity in the drywell only), the methods of SRP Section 6.5.2 (Reference A1-8.27) are used. It is assumed that elemental iodine is removed at the same rate as particulate. Since particulate is removed at a rate less than the 20 per hour rate permitted by SRP Section 6.5.2 for removal of elemental iodine, this assumption is conservative. Elemental iodine removal is assumed to stop when the average airborne elemental iodine concentration in the primary containment corresponds to the amount released divided by 200, the decontamination factor (DF) limit from SRP Section 6.5.2.

Particulate removal by sprays as described in SRP Section 6.5.2 is 1.5 times the product of two ratios: QH/V (the volumetric flow rate times the spray fall height divided by the volume being sprayed) and e/D (the spray efficiency divided by the droplet diameter). The e/D ratio is given in SRP Section 6.5.2 as 10 per meter until 98% of the particulate is removed, and one per meter thereafter. Therefore, only QH/V needs to be determined.

The values used are as follows:

Q = 5237.5 gpm (approximately 85% of actual spray flow based on walkdown of spray headers to assess near-field blockage),
H = 31.5 feet (50% of theoretical fall height to account for drywell internal structures), and
V = 306,200 ft^3

The resulting spray removal rate is 19.8 per hour until 98% of the particulate has been removed and 1.98 per hour thereafter.

A1-4.1.1.4 Secondary Containment Bypass Line Activity Removal

Credit is taken for the reduction of airborne activity due to natural deposition (RG 1.183, Appendix A, Section 6.3) for steam lines with both main steam isolation valves (MSIVs) closed and for the other secondary containment bypass pathways. Only the space between the closed containment isolation valves is credited. No credit is taken for deposition in the steam line with one MSIV assumed to be stuck open.

The methods of AEB-98-03 (Reference A1-8.22) are used for the determination of this natural removal of activity. The particulate deposition velocity used is equal to the third percentile value of 6.6E-5 m/s from Appendix A of AEB-98-03. This is conservatively low and reflects the effectiveness of spray removal in the drywell.

As discussed in Section A1-4.1.3.1, for the MSIV failure scenario, the primary containment (PC) leak rate (including that for bypass pathways) is assumed to decrease by a factor of two at 24 hours. However, no credit is taken for the leak rate reduction in terms of increasing removal efficiency of activity in the bypass pathways. This is conservative for the MSIV failure scenario.

A DF of two for elemental iodine is credited. This is consistent with the assumption of elemental iodine being plated-out on aerosol as is used in the drywell spray calculation. The DF of two for elemental iodine is conservative with respect to the calculated removal efficiencies for aerosol removal in the steam lines and the other secondary containment bypass pathways; i.e., the calculated aerosol removal efficiencies exceed 50%. No credit is taken for the removal of organic iodine.

It is assumed that the normal operating temperature of the main steam lines (558°F) is unchanged for the duration of the dose analysis. This maximizes flow and minimizes residence time in the steam lines. Only the steam lines have operating temperatures (in the portion between isolation valves) significantly greater than the assumed 340°F temperature of the post-accident drywell gas. Therefore, the temperature correction (which results in an increase in flow rate) is applied only to the steam lines.

A1-4.1.1.5 Conversion of Allowable Leakage to Volumetric Flow Rate

The maximum allowable MSIV leakage rate of 24 scfh at a test pressure of 40 psig is converted to a true volumetric flow rate (cfh or cfm) for the appropriate conditions, with the following results:

Drywell to space between MSIVs:	24 scfh = 9.3 cfh = 0.155 cfm
Steam lines to environment:	24 scfh = 10.5 cfh = 0.175 cfm

The methodology used in performing this conversion is as follows:

- Calculate the cfh corresponding to scfh measured at 40 psig as cfh = scfh x [14.7/(40 +14.7)] = scfh x 0.269
- Recognize that the MSIV test pressure of 40 psig (which is also approximately equal to the primary containment leak rate test pressure, P_a) is above critical pressure.
- Recognize that the volumetric flow will be determined by the sonic velocity in the leak path.
- Recognize that the sonic velocity varies as follows:

$$\frac{v'_{sonic}}{v_{sonic}} = \frac{\sqrt{(k'-1)\frac{c'_p}{M'}T'}}{\sqrt{(k-1)\frac{c_p}{M}T_{\text{standard}}}} = \sqrt{\frac{k'-1}{k-1}\frac{c'_p}{c_p}\frac{M}{M'}\frac{T'}{T}} = \left(\frac{(0.3)(8)(29)(800R)}{(0.4)(7)(18)(530R)}\right)^{0.5} = 1.444$$

Where: Unprimed values represent test conditions,
Primed values represent accident conditions, and
Accident conditions are steam at 340°F

- Calculate the volumetric flow multiplier for scfh at test conditions to obtain cfh at accident conditions as 0.269 x 1.444 = 0.388.
- Calculate the volumetric flow out of the drywell as 24 scfh x 0.388 or 9.3 cfh.
- Recognize that for flow out of the space between the MSIVs (with the conservative assumption that the pressure in that space is equal to the drywell), the sonic velocity ratio is the square root of the assumed temperature in the steam line space to the square root of the peak drywell temperature; i.e., $(1018\ R/800\ R)^{1/2} = 1.128$.
- Calculate the volumetric flow out of the space between the MSIVs as 9.3 cfh x 1.128 or 10.5 cfh.

The same approach is used for other secondary containment bypass pathways, except for the temperature correction in the space between the isolation valves (see Section A1-4.1.1.4).

A1-4.1.1.6 Delay of Activity Releases

The current licensing basis analysis (USAR Sections 6.2.3 and 15.6.5) includes credit for delay of activity releases via the main steam lines and the other secondary containment bypass pathways. For the purpose of the AST analysis, three groups of bypass pathways (other than the main steam lines) have been defined, as follows:

Group 1: Bypass from the drywell, delays neglected (all bypass pathways originating in the drywell except those listed below in the third group).

Group 2: Bypass from the wetwell, delays neglected.

Group 3: Bypass from the drywell (feedwater, 14" containment purge, and reactor water cleanup (RWCU)), delays considered and conservatively combined.

For each of these three groups, the current leakage limits have been combined and adjusted to determine an effective leak rate using the worst X/Q values for the associated release points.

The release delays are simplified in this analysis so that credit for delay is taken only for the steam line with one MSIV assumed to be failed open (5.26 hours), the other three steam lines (7.11 hours), and the Group 3 bypass lines. Individually, the Group 3 bypass pathways have the following delay times: feedwater lines - 9.61 hours, minimum; 14" containment purge line from the drywell - 10.58 hours; and RWCU line - 13.09 hours. By assuming that all of the leakage for the Group 3 pathways occurs in the RWCU line (with the leak rate in that line limited to be equivalent to the product of the volumetric flow and the penetration (defined as 1 - removal efficiency) for all four lines; i.e., 13.36 scfh or 5.18 cfh), the minimum delay for the Group 3 pathways is determined to be 2.45 hours. The 2.45-hour delay is used in the dose analysis. This delay is conservative for the MSIV failure scenario because it assumes only one isolation valve in the RWCU line is closed even though it would be appropriate to consider both the RWCU isolation valves to be closed. If both isolation valves were closed, the delay would be approximately 30% greater.

A1-4.1.1.7 Containment Pressure Reduction

The current licensing basis analyses for post-LOCA primary containment pressure response are described in USAR Section 6.2.1. Containment pressure transients for both a reactor recirculation suction line break and a main steam line break have been performed for the following three cases:

Case A: Offsite power available, all emergency core cooling system (ECCS) equipment and containment spray operating (USAR Figures 6.2-5 and 6.2-15),

Case B: Loss of offsite power, minimum diesel generator power available for ECCS and containment spray mode (only Division 2 available) (USAR Figures 6.2-3 and 6.2-16), and

Case C: Loss of offsite power, minimum diesel generator power available for ECCS and residual heat removal (RHR) shutdown cooling mode (only Division 2 available) (USAR Figures 6.2-4 and 6.2-17).

In the LOCA radiation dose analyses supporting AST implementation, it is assumed that containment sprays will be operated to control containment pressure, temperature, and radiation levels consistent with the current licensing basis analyses and as directed by the Emergency Operating Procedures (EOPs) and

Severe Accident Procedures (SAPs). The analyses in USAR Section 6.2.1 demonstrate that if both RHR systems (Division 1 and Division 2) are operating (i.e., Case A above), then the containment pressure decreases sufficiently at 24 hours to allow credit for a factor of two reduction in the primary containment leak rate, in accordance with RG 1.183. However, if one electrical division fails (e.g., the emergency diesel generator (EDG) fails to start following the loss of offsite power), then only one RHR division is operating (Case B with sprays above) and containment pressure does not decrease sufficiently at 24 hours to allow credit for a factor of two reduction in the primary containment leak rate. Thus, the AST analysis has considered two single failure scenarios:

> Scenario 1 – One MSIV fails to close. Both divisions of the RHR system operate, with one division operating in the containment spray mode. The containment leak rate is reduced by 50% after 24 hours. No credit is taken for activity removal in the piping between the MSIVs for the affected main steam line.
>
> Scenario 2 – One electrical division fails. One division of the RHR system operates in the containment spray mode, but there is no reduction in the containment leak rate at 24 hours and there are no MSIV failures.

A1-4.1.2 Inputs and Assumptions

General inputs for the DBA radiological consequence analyses are listed in Table A1-9. Event-specific inputs and assumptions are further discussed in the following sections. New atmospheric dispersion factors (X/Q values) for the control room intake, Technical Support Center (TSC) intake, and offsite (exclusion area boundary (EAB) and low population zone (LPZ)) have been calculated. Section A1-4.3 and Attachment (6) provide additional information regarding X/Q values.

A1-4.1.3 Loss of Coolant Accident (LOCA)

The radiological consequences of the DBA LOCA were analyzed using the RADTRAD code and verified with the STARDOSE code, with the inputs and assumptions defined in Section A1-4.1.3.1 below. The LOCA analysis is fully documented in Calculation H21C-106 (see Attachment 7).

A1-4.1.3.1 Inputs and Assumptions

The key inputs used in the AST LOCA analysis are included in Tables A1-9 through A1-12. These inputs and assumptions fall into three categories: Radionuclide Release Inputs and Timing, Radionuclide Transport Inputs, and Radionuclide Removal Inputs. The LOCA analysis is fully documented in Calculation H21C-106 (see Attachment 7). The analysis includes the following release pathways (illustrated schematically on Figure A1-1):

Pathway 1: Leakage from the Primary Containment (PC) to the Reactor Building (RB; i.e., secondary containment) at the TS leak rate limit of 1.1% of PC air weight per day. During RB drawdown (0 to 60 minutes), this leakage is assumed to be released from the RB to the environment unfiltered at ground level. Subsequent to re-establishing RB negative pressure, this pathway is assumed to be filtered by the SGTS and released via the main stack.

Pathway 2: Traversing in-core probe (TIP) leakage from the PC to the RB at the rate of 0.12% per day, which assumes that a TIP is inserted when the LOCA occurs and the guide tube fails, the TIP fails to withdraw, and the shear valve fails to close. Inclusion of this pathway is consistent with the current licensing basis analysis described in USAR Section 15.6.5. This pathway is modeled the same as Pathway 1 above.

Pathway 3: MSIV leakage from the PC to the environment. There are 4 main steam lines. Each MSIV is assumed to leak at 24 scfh for a total of 96 scfh. Deposition in the piping between the inboard and outboard MSIVs is assumed (when both valves are closed), and delay of activity releases is credited in the analysis, consistent with the current licensing basis analysis described in USAR Sections 6.2.3 and 15.6.5.

Pathway 4: Secondary containment (SC) bypass leakage directly from the PC to the environment (similar to MSIV leakage). This includes multiple piping pathways in the following systems: main steam drains, reactor water cleanup, feedwater, drywell floor and equipment drains and vents, post-accident sampling system (PASS), instrument air and nitrogen supply, and PC purge. Deposition in the piping between the inboard and outboard isolation valves is assumed, and delay of activity releases is credited for the Group 3 bypass pathways (feedwater, 14" PC purge, and RWCU), consistent with the current licensing basis analysis described in USAR Sections 6.2.3 and 15.6.5.

Pathway 5: Engineered safety feature (ESF) leakage from the PC into the RB and subsequent release to the environment. During RB drawdown (0 to 60 minutes), this leakage is assumed to be released from the RB to the environment unfiltered at ground level. Subsequent to re-establishing RB negative pressure, this pathway is assumed to be filtered by the SGTS and released via the main stack.

Pathway 6: PC purge coincident with the LOCA. A containment purge in the pressure control mode is assumed to be in progress when the LOCA occurs. This is a short release from the PC via the SGTS filters and the main stack to the environment. Inclusion of this pathway is consistent with the current licensing basis analysis described in USAR Section 15.6.5. In Appendix F of calculation H21C-106, this dose contribution is shown to be negligible.

The LOCA analysis assumes a concurrent loss of offsite power. Two single failure scenarios were analyzed: one in which an MSIV fails to close (affecting Pathway 3) and one in which one electrical division fails (affecting Pathways 1 through 4). The limiting scenario is the assumed failure of an MSIV to close.

Radionuclide Release Inputs and Timing

The Pathways 1, 2 and 5 releases are from the RB unfiltered at ground level during RB drawdown, and from the plant stack via the SGTS filters after RB negative pressure is re-established. The Pathways 3 and 4 releases are secondary containment bypass pathways (including MSIV leakage and leakage via other identified bypass piping) that provide pathways from the primary containment. They are treated as ground level releases from either the radwaste/reactor building vent, SGTS building, PASS panel, or the main steam tunnel. Event timing is as follows:

- LOCA occurs at time zero. Degraded core cooling leads to core damage. Reactor coolant activity is assumed to be immediately released to the PC.
- Release from core to PC begins at 2 minutes.
- Drywell sprays are manually initiated and spray begins at 20 minutes.

- The SGTS starts automatically within a few minutes and RB drawdown is achieved in 60 minutes (for a single SGTS train operating).

- Further core damage and associated activity releases are terminated at 122 minutes by assumed restoration of core cooling. Drywell and suppression chamber airspace become well-mixed at that time.

- For the single MSIV failure to close scenario, the PC pressure has decreased to less than 5 psig by 24 hours, and the PC leak rate (including SC bypass pathways) has become a factor of two less than the maximum PC leak rate (except for ESF liquid leakage). For the single electrical division failure scenario, the PC leak rate (including SC bypass pathways) is constant for the duration of the event.

- By 720 hours, essentially all particulate activity has been leaked or deposited and gaseous I-131 (the principal dose contributor excluding particulate I-131) has gone through nearly four half-lives. The dose calculation is terminated in accordance with RG 1.183.

The timing of these events is based on RG 1.183 and as further discussed below.

Reactor Building Drawdown Time

Prior to establishing a sustained negative pressure in the RB, PC leakage to the RB is assumed to be released to the RB and thence to the environment unfiltered at ground level. The reactor building drawdown analysis is described in USAR Section 6.2.3. This drawdown analysis was reviewed and accepted by the NRC as part of License Amendment No. 56, issued by NRC letter dated August 30, 1994 (Reference A1-8.21). The 60-minute secondary containment drawdown time is based on operation of a single SGTS train (assuming failure of an electrical division). The 60-minute drawdown time is conservative for the MSIV failure scenario since both SGTS trains would be operating, resulting in a shorter drawdown time.

Drywell Spray Initiation

Containment spray is an operating mode of the RHR system. The RHR pumps are initiated automatically in the low pressure coolant injection (LPCI) mode of operation on a LOCA signal (reactor vessel low water level and/or high drywell pressure). Consistent with the analyses for ECCS performance and containment pressure response described in USAR Sections 6.2.1 and 6.3, one of the RHR pumps is assumed to be manually transferred to the containment spray mode of operation after 20 minutes. Reasonable assurance of the timeliness of this manual action is provided by the following existing procedures:

- The EOPs direct the operator to initiate drywell sprays for containment pressure control when the suppression chamber pressure exceeds 10 psig. The peak containment pressure for a design basis LOCA would rapidly exceed this threshold.

- The SAPs direct the operator to initiate drywell spray when the drywell radiation level exceeds a value that is indicative of significant core damage (such as that postulated to occur for a design basis LOCA).

The containment spray system is described in USAR Section 6.2.2 and is safety-related, required to be operable by TS 3.6.1.6 and TS 3.6.2.4, and supplied with emergency power.

Drywell and Suppression Chamber Mixing

RG 1.183 establishes that only the drywell volume should be credited for diluting the activity release from the core for a BWR. For plants with Mark II containment designs, no specific guidance on how to treat mixing between the drywell and the remainder of the containment is provided. Instead, the general guidance is that the suppression chamber airspace "...may be included provided there is a mechanism to ensure mixing..." The NMP2 analysis credits mixing of the drywell and suppression chamber airspace volumes beyond 122 minutes, following the assumed restoration of core/core debris cooling. At this time, considerable thermal-hydraulic activity in the PC will result in the drywell and suppression chamber airspace volumes becoming well-mixed.

Standby Liquid Control System Injection

The analysis credits the pH buffering effect of sodium pentaborate solution introduced into the suppression pool post-LOCA by operation of the SLC system. The SLC injection will maintain the suppression pool pH above 7.0 for the 30-day duration of the accident; therefore, radioiodine re-evolution does not need to be considered.

The SLC system is described in USAR Section 9.3.5 and is safety-related, required to be operable by TS 3.1.7, and supplied with emergency power. Suitability of the SLC system to perform the post-LOCA pH control function, details of the AST analysis for suppression pool pH control, and a discussion of procedural guidance for post-LOCA injection of the sodium pentaborate solution using the SLC system are addressed in Attachment (4).

Primary Containment Leakage and Leak Rate Reduction Justification

The maximum allowable primary containment leakage rate is 1.1% PC air weight per day, per TS 5.5.12.c. This leakage rate, plus 0.12% per day for TIP leakage, was assumed in the AST analyses for the first 24 hours. RG 1.183 requires justification for implementing a factor of two decrease in PC leakage rate at 24 hours after the start of the accident. For the MSIV failure scenario, both containment spray system divisions operate and reduce the drywell pressure from its peak value of 39.75 psig to approximately 5 psig at 24 hours (a factor of seven reduction based on the gauge pressure). Thus, for the MSIV failure scenario, a factor of two reduction in PC leakage rate at 24 hours is justified. However, for the electrical division failure scenario, only one containment spray system division operates. The containment pressure reduction for this scenario is not sufficient to justify full credit for PC leakage rate reduction by a factor of two at 24 hours; therefore, PC leakage rate reduction is not assumed for the electrical division failure scenario. Calculation H21C-106 (see Attachment 7) provides additional details.

Engineered Safety Feature Leakage

Leakage from ESF components outside primary containment was reviewed. NMP2 has implemented a program in accordance with TS 5.5.2 to minimize leakage from those portions of systems outside containment that could contain highly radioactive fluids during a serious transient or accident to levels as low as practicable. The program includes the following:

- Preventive maintenance and periodic visual inspection requirements; and

- System leak test requirements for each system at 24 month intervals.

The NMP2 program effectively eliminates ESF leakage. However, the AST LOCA analysis assumed an ESF leakage rate of 62 gpm into the reactor building starting at the onset of the event. This leakage rate is comprised two contributors:

1. The sum of the simultaneous leakage from all ESF components (such as pump seals, valve stem packing, flanges, etc.) that is allowed by the program specified in TS 5.5.2. This allowed leakage is 1 gpm. In accordance with RG 1.183, the 1 gpm value is increased by a factor of two for purposes of the AST dose analysis.

2. A leakage rate 60 gpm due to the assumed failure of two RHR system sample lines in the reactor building and allowed leakage past the two isolation valves in an RHR line to the liquid radwaste system. The 60 gpm value is not doubled. Inclusion of this leakage is consistent with the current licensing basis analysis described in USAR Section 15.6.5.

MSIV Leakage Rate

The total MSIV leakage rate of 96 scfh (maximum of 24 scfh in each of the 4 lines) was assumed in the analysis for the first 24 hours. For the MSIV failure scenario, the MSIV leakage rate was reduced by a factor of two at 24 hours, consistent with the PC containment leakage rate reduction. For the electrical division failure scenario, no leak rate reduction is credited. The maximum allowable MSIV leakage value is specified in TS Surveillance Requirement (SR) 3.6.1.3.12. The allowable leakage was converted to a true volumetric flow rate for the appropriate conditions, as described in Section A1-4.1.1.5 and in Calculation H21C-106.

Secondary Containment Bypass Leakage

Primary containment leakage via the lines which penetrate the RB is taken into account. In addition to the four main steam lines, these include multiple piping pathways in the following systems: main steam drains, reactor water cleanup, feedwater, drywell floor and equipment drains and vents, PASS, instrument air and nitrogen supply, and PC purge. Leakage from the PC through the closed primary containment isolation valves in these systems could bypass the RB and the SGTS filters and could also result in a ground-level release. These lines are divided into three groups as discussed in Section A1-4.1.1.6, according to whether they originate in the drywell or the wetwell, and whether or not they have sufficient delay time for holdup credit. A total effective leakage rate for each group is used in the analysis. For the MSIV failure scenario, the bypass leakage rates were reduced by a factor of two at 24 hours, consistent with the PC containment leakage rate reduction. For the electrical division failure scenario, no leak rate reduction is credited. Activity removal efficiencies are based on the maximum leak rates, so for the MSIV failure scenario, there is additional conservatism in these efficiencies after 24 hours. The maximum allowable bypass leakage values are specified in TS Table 3.6.1.3-1 and are controlled by the 10 CFR 50 Appendix J Testing Program Plan that is described in TS 5.5.12.

Radionuclide Transport Inputs

Pathways 1 and 2 – Leakage from Primary Containment Atmosphere to the Reactor Building

Pathways 1 and 2 are combined in the model. This consists of leakage from the PC to the RB at 1.1% per day plus TIP leakage at 0.12% per day.

Assumptions

The release from the core is assumed to enter the drywell only. Mixing within the entire PC is not assumed to occur until after the end of the release from the core (at 2.033 hours). The drywell sprays are assumed to begin operating at 20 minutes.

The release rate from the PC to the RB corresponds to the TS leak rate of 1.1% air weight per day, plus 0.12% per day for TIP leakage. A RB drawdown time of 60 minutes from the start of the DBA-LOCA is used in the analysis. During drawdown (0 to 60 minutes) the release is assumed to be at ground level, unfiltered. After RB negative pressure is re-established at 60 minutes, this leakage is filtered by the SGTS and released to the environment via the main stack. SGTS filter efficiencies of 99% for particulates, elemental iodine, and organic iodine are assumed. A 50% mixing credit is taken for dilution/mixing in the secondary containment. This credit is consistent with the current licensing basis for NMP2 and was included in the supporting information that was reviewed and accepted by the NRC in License Amendment No. 56 issued by NRC letter dated August 30, 1994 (Reference A1-8.21). Mixing in the secondary containment is provided by the reactor building emergency recirculation system unit coolers together with local area unit coolers. A factor of two reduction in the PC and TIP leak rates is assumed to occur at 24 hours, based on containment pressure reduction, for the MSIV failure scenario only.

Pathways 3 and 4 – Leakage from Primary Containment Directly to the Environment (Secondary Containment Bypass Pathways)

These pathways model the leakage from the lines which penetrate the PC and then penetrate the RB. Leakage from the PC through the closed primary containment isolation valves (PCIVs) in these systems could bypass the RB and the SGTS filters and could also result in a ground-level release. This includes MSIV leakage and the leakage from the other secondary containment bypass pathways.

Assumptions

The release from the core is assumed to enter the drywell only. Mixing within the entire PC is not assumed to occur until after the end of the release from the core (at 2.033 hours), and no credit is taken for natural deposition in the containment. The drywell sprays are assumed to begin operating at 20 minutes. A factor of two reduction in leak rates is assumed to occur at 24 hours, based on containment pressure reduction, for the MSIV failure scenario only. These releases are all assumed to be released at ground level.

The MSIV leakage pathway includes credit for activity removal between the inboard and outboard MSIVs only. The model includes four parallel main steam line flow paths to the environment. For the lines with both MSIVs closed, the model credits deposition in the volume between the MSIVs using the methods of AEB-98-03 (Reference A1-8.22). For the line with an MSIV that is assumed not to close, the volume between the MSIVs is ignored and no credit for deposition is taken. In addition, the piping upstream of the inboard MSIV and downstream of the outboard MSIV is neglected for deposition. A delay of activity releases via the main steam lines is taken into account consistent with the current licensing basis analysis described in USAR Sections 6.2.3 and 15.6.5. As discussed in the USAR, the main steam piping downstream of the MSIVs is seismically rugged and would remain intact during and following a design basis earthquake.

The non-MSIV secondary containment bypass leakage is treated in the same manner as the MSIV pathway. As discussed in Section A1-4.1.1.6, the non-MSIV bypass pathways have been combined into three groups, each with a total effective leak rate. The first group includes lines from the drywell with no credit for delay time, the second group is the lines from the wetwell which do not credit delay, and the third group includes the lines from the drywell with credit for delay time. Only the length of the line between the PCIVs is credited for activity removal.

Pathway 5 – ESF Leakage from the Suppression Pool to the Reactor Building

ESF leakage is modeled as a continuous 62 gpm volumetric flow from the suppression pool control volume to the RB. During the drawdown period, the release is assumed to be released to the environment at ground level, unfiltered. After RB negative pressure is re-established at 60 minutes, this leakage is filtered by the SGTS and released to the environment via the main stack.

Assumptions

The ESF leak rate of 62 gpm is assumed to begin at the initiation of the accident. Ten percent of the iodine in the ESF leakage is assumed to become airborne. A drawdown time of 60 minutes from the start of the DBA-LOCA is used in the analysis. During the 60 minute drawdown period, all of the elemental and organic iodine that becomes airborne is released unfiltered at ground level. After RB negative pressure is re-established at 60 minutes, this leakage is filtered by the SGTS and released to the environment via the main stack. A 50% mixing credit is taken for dilution/mixing in the secondary containment, as noted for Pathways 1 and 2 above.

Pathway 6 – PC Purge Coincident with the LOCA

A PC purge through the 2-inch pressure control line is assumed to be in progress when the LOCA occurs. This release is filtered by the SGTS and released from the main stack. The release is terminated within 5 seconds by closure of the primary containment purge isolation valves.

Radionuclide Removal Inputs

LOCA activity release is partially removed by spray in the drywell, natural deposition in the main steam lines, and by removal by the SGTS filters.

In the Drywell

The drywell spray removal rate development applies to both the MSIV leakage pathway and the RB/SGTS/main stack pathway, as well as to the secondary containment bypass leakage pathways.

Drywell spray removal for particulates is determined using the SRP Section 6.5.2 methodology (Reference A1-8.27). The spray flow rates are 6,559.9 gpm for spray loop A and 6,143.4 gpm for spray loop B. The spray flow rate credited in the analysis is based on the smaller loop B flow. To account for drywell congestion, the spray flow rate is reduced by approximately 15% to 5,237.5 gpm, and one half of the 63 ft fall height from the spray header to the drywell floor (i.e., 31.5 feet) is used.

The particulate removal rate, λ, was calculated in Calculation H21C-106 (see Attachment 7) and applied to the RADTRAD model. The removal rate for elemental iodine is assumed to be the same as that of the particulate which is less than the 20 per hour limit for elemental iodine from SRP Section 6.5.2. The DF for elemental iodine is limited to a maximum value of 200 based on SRP Section 6.5.2.

In the Main Steam Lines and Other Bypass Lines

For the NMP2 AST analyses, the particulate settling rates were calculated by using the 3rd percentile values from AEB-98-03, taking into account the drywell spray as discussed in Section A1-4.1.1.4.

A1-4.1.3.2 Technical Support Center (TSC) LOCA 30-Day Dose

An analysis for the TSC 30-day inhalation and immersion doses was performed. Two scenarios were considered, one with the TSC occupied at the initiation of the event and the other assuming that the TSC is not activated for 1 hour. This was done because the emergency ventilation (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. Three external shine dose contributions were considered: (1) shine from the RB (calculated based on a comparison of AST source strength to that for the current licensing basis), (2) shine from the plume, and (3) shine from the TSC filters (also calculated based on a comparison of AST source strength to that for the current licensing basis).

A1-4.1.3.3 Results

The LOCA doses are the result of the following activity contributions:

1. Primary to secondary containment (reactor building) leakage. This leakage is directly released into the RB. During the drawdown period, this leakage is assumed to be released without filtration to the environment as a ground level release. Following drawdown, it is filtered by the SGTS prior to release through the plant stack.

2. MSIV leakage from the primary containment directly to the environment. Credit is taken for deposition in the main steam piping between the inboard and outboard MSIVs (if both valves are closed) and delay of the activity release. Delay is also credited if one valve is assumed to be failed open, but the delay is not as great as when both valves are assumed to be closed.

3. Secondary containment bypass leakage (other than through the MSIVs), assumed to be released at ground level. Credit is taken for deposition in piping between the inboard and outboard isolation valves and delay of activity release for the Group 3 bypass leakage pathways (feedwater lines, the 14" containment purge line from the drywell, and the RWCU line).

4. ESF leakage into the secondary containment. This leakage is directly released from the suppression pool into the RB environment. During the drawdown period, the activity in this ESF leakage is assumed to be released unfiltered to the environment at ground level. After RB negative pressure has been re-established, the airborne portion of the activity in the ESF leakage is filtered by the SGTS prior to release through the plant stack.

5. Containment purge coincident with the LOCA. This is a short-duration release via the SGTS filters and main stack. The dose from this contributor is negligible.

6. Post-DBA LOCA radiation shine dose to personnel within the control room from airborne activity in the RB, activity collected on the CREF system filters, and the external radioactive plume.

The limiting scenario is an assumed loss of offsite power concurrent with the LOCA and failure of an MSIV to close. This scenario maximizes the calculated doses in comparison to other single failures that could be postulated.

The radiological consequences for the postulated LOCA are given in Table A1-13, along with the results from the current licensing basis source term analysis. As indicated, the EAB, LPZ, and control room calculated doses remain within the regulatory limits.

The analysis for the TSC demonstrates that 30-day inhalation, immersion, and external shine doses do not exceed 5 rem TEDE.

A1-4.1.4 Main Steam Line Break (MSLB) Accident

Section 15.6.4 of the NMP2 USAR describes the design basis MSLB accident. The postulated MSLB accident assumes a double ended guillotine break of one main steam line outside the primary containment, downstream of the outboard MSIV. The break flow is terminated by closure of the MSIVs. The break mass released includes the mass of steam in the broken line and connecting lines, plus the steam and coolant that passes through the MSIVs prior to closure. The MSLB accident analysis is fully documented in Calculation H21C-101 (see Attachment 7).

Fuel damage is not predicted for this event, as the core is not uncovered. The case evaluated corresponds to the maximum iodine concentration that is allowed in the primary coolant, as specified in TS 3.4.8, "RCS Specific Activity;" i.e., a pre-accident spike of 4 μCi/gm Dose Equivalent I-131.

A1-4.1.4.1 Inputs and Assumptions

The key inputs and assumptions used in the AST MSLB accident analysis are shown in Table A1-14. The radiological consequences of the design basis MSLB accident were analyzed using a spreadsheet and followed the guidance of RG 1.183. The following conservative assumptions were used in the analysis:

- Break isolation is assumed in 5.5 seconds, corresponding to the maximum MSIV closing time of 5 seconds plus a closure signal delay time of 0.5 second.
- Following accident initiation, the radionuclide inventory from the released coolant is assumed to reach the environment instantaneously. No holdup in the turbine building is credited.
- The entire released coolant mass is conservatively used (rather than just the liquid mass) in the calculation of the activity released.
- An infinite exchange rate between the control room and the environment is assumed. No credit is taken for filtration of the control room intake air.
- No credit is taken for other iodine removal mechanisms, such as plate-out, sedimentation, condensation, or decay.

The MSLB analysis included continuous release X/Q values for the EAB and LPZ and an instantaneous ground level puff release X/Q for the control room. The inputs shown in Table A1-15 were used to calculate the puff release X/Q, and the complete calculation (H21C-094) is provided in Attachment (7). The resulting X/Q values that were used for the MSLB radiological dose calculations are shown in Table

A1-16. Additional information concerning the calculation of new atmospheric dispersion factors (X/Q values) is provided in Attachment (6).

This event only credits closure of the MSIVs to terminate the reactor blowdown. Since the MSIVs are redundant, the release is not impacted by a single failure. No other safety systems are credited in the determination of releases and consequences. Therefore, single failures have no adverse effects on the analysis results.

A1-4.1.4.2 Results

The radiological consequences for the postulated MSLB accident are given in Table A1-17, along with the results from the current licensing basis source term analysis. As indicated, the EAB, LPZ, and control room calculated doses remain within the regulatory limits.

A1-4.1.5 Fuel Handling Accident

Section 15.7.4 of the NMP2 USAR describes the design basis FHA. The postulated FHA involves a 32.95 foot drop of a fuel assembly on top of other fuel assemblies in the reactor core during refueling operations. The drop distance bounds the maximum height that is allowed by the NMP2 fuel handling equipment and is the limiting case since it results in the maximum release of fission products to the reactor building. Damage due to a fuel assembly drop 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. The FHA analysis is fully documented in Calculation H21C-102 (see Attachment 7).

A1-4.1.5.1 Inputs and Assumptions

The key inputs and assumptions used in the AST FHA analysis are shown in Table A1-18. The X/Q values used for the analysis are summarized in Table A1-20. Additional information concerning the calculation of new atmospheric dispersion factors (X/Q values) is provided in Attachment (6). Because of the simplifying, conservative assumptions used, the radiological consequences of the design basis FHA were analyzed using a spreadsheet. The analysis followed the guidance of RG 1.183. The following assumptions were used in the analysis:

- The accident is assumed to occur at 24 hours after shutdown. Consequently, release activity inventories were calculated that correspond to this post-shutdown decay time. Fuel handling would not begin before 24 hours after shutdown.

- The activity inventory from two full fuel assemblies is released. This is bounding for the 124 damaged rods for GE 8x8 fuel assemblies, the 140 damaged rods for GE11 9x9 fuel assemblies, and the 172 damaged rods for GE14 10x10 fuel assemblies determined for the current licensing basis analysis (described in USAR Section 15.7.4).

- A core radial peaking factor of 1.8 is applied to the assembly inventory.

- The radionuclide inventory from the damaged fuel rods is assumed to be released to the environment instantaneously (even though this release could be assumed to occur over a two-hour period per RG 1.183). Thus, radioactive decay that would occur during a two-hour release period is neglected.

- The release to the environment is modeled as a ground level release, with no credit taken for secondary containment or release via the main stack.

- Even though the maximum fuel damage is for a drop in the refueling cavity onto the reactor core, a more conservative spent fuel pool DF for elemental iodine is used in the analysis. The minimum depth of water in the canal to the spent fuel pool is 22'-3". An adjusted DF was calculated as follows:

  With 23 feet of water, the DF for inorganic iodine is 285. With organic iodine assumed to be 0.0015 of the total release, application of a DF of 285 to the inorganic forms results in an overall DF of 200 for all iodine for 23 feet of water. Assuming that the relationship between the inorganic iodine DF and the depth of water (d) is exponential (i.e., $DF_{iI} = e^{-c \times d}$), for DF_{iI} =285 and d = 23', c = -0.2458. Thus, with 22'-3" of water, DF_{iI} = 237. This results in an overall iodine DF of 175 for the actual water depth of 22'-3" (vs. an overall iodine DF of 200 for the reference water depth of 23').

- No DF is applied to noble gases.

- The DF for other radionuclides is assumed to be infinite, per RG 1.183.

- Filtration by the SGTS and the CREF system is not credited.

Since this event does not credit any safety systems in the determination of releases and consequences, single failures have no adverse effects on the analysis results.

The core inventories at 24 hours after shutdown were calculated by the RADDECAY code (Reference A1-8.16). The gap activity of noble gas and iodine (set at 99.85% elemental, 0.15% organic per RG 1.183) was added from the core to the gap. The RADDECAY calculation starts with time zero inventories for the noble gas and iodine isotopes. Given the activity (Ci or Ci/MWt) of an isotope at time zero, RADDECAY calculates the curies or Ci/MWt of that isotope and its daughters at any subsequent time. To obtain the total curies of the isotope of interest, the curies resulting from its direct decay plus the curies resulting from decay in chains in which it is a daughter product must be added together. This adjustment has been made to the isotopes of interest, and the resulting fission product inventory is summarized in Table A1-19.

A1-4.1.5.2 Results

The radiological consequences for the postulated fuel handling accident are given in Table A1-21, along with the results from the current licensing basis source term analysis. As indicated, the EAB, LPZ, and control room calculated doses remain within the regulatory limits.

A1-4.1.6 <u>Control Rod Drop Accident (CRDA)</u>

Section 15.4.9 of the NMP2 USAR describes the design basis CRDA. This accident involves 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. NMP2 is a banked position withdrawal sequence (BPWS) plant and the GESTAR generic CRDA analysis demonstrates that the accident does not result in fuel melting for BPWS plants (References A1-8.17 and A1-8.18). However, for the purpose of this analysis, fuel damage (i.e., cladding perforation and fuel melting) is assumed to occur. The NMP2 AST analysis for the CRDA considers two scenarios with regard to the activity release pathways, as follows:

Case 1: The activity that reaches the turbine/condenser is released via leakage to the environment.

Case 2: The activity that reaches the condenser is released via the mechanical vacuum pumps.

The control rod drop accident analysis is fully documented in Calculation H21C-103 (see Attachment 7).

A1-4.1.6.1 Inputs and Assumptions

The key inputs and assumptions used in the AST CRDA analysis are shown in Table A1-22. A core radial peaking factor of 1.8 was used in the analysis. The X/Q values used for the analysis are summarized in Table A1-23.

For Case 1, leakage from the turbine/condenser at a rate of 1% per day for a period of 24 hours is assumed, at which time the leakage is assumed to terminate.

For Case 2, the maximum activity concentration that will not cause isolation of the mechanical vacuum pumps on a high main steam line radiation signal is assumed to be released via the main stack at the mechanical vacuum pump flow rate, and retention by the charcoal delay beds in the offgas system is neglected.

The radiological consequences were analyzed using a spreadsheet for Case 1 and the RADTRAD code for Case 2. The RADTRAD results were verified with the STARDOSE code. No credit was taken for operation of the CREF system or any other safety systems to mitigate the consequences of the event, and no single failures were considered.

A1-4.1.6.2 Results

The radiological consequences for the postulated CRDA are given in Table A1-24, along with the results from the current licensing basis source term analysis. As indicated, the EAB, LPZ, and control room calculated doses remain within the regulatory limits.

A1-4.1.7 Control Room Doses for Accident at Adjacent Unit

The habitability of the NMP2 control room due to a DBA at NMP1, and the habitability of the NMP1 control room due to a DBA at NMP2, have been evaluated using AST methodology and the appropriate atmospheric dispersion factors. The calculations listed in Attachment (7) provide additional details regarding these evaluations. The resultant control room doses at each unit have been determined to be within the regulatory limits.

A1-4.2 Suppression Pool pH Control

The AST LOCA analysis takes credit for minimization of re-evolution of elemental iodine from the suppression pool. Re-evolution is strongly dependent on suppression pool pH. An analysis determined that sodium pentaborate solution injection via the SLC system must commence within approximately 11 days of the onset of a LOCA. Using the assumptions of a minimum quantity and concentration of available sodium pentaborate solution (as specified in TS 3.1.7 and TS Figure 3.1.7-1) and conservative modeling of acids and bases that could be added to the suppression pool post-LOCA, the minimum pool pH at 30 days post-LOCA remains above 7.0. This pH satisfies the conditions for inhibiting the release of the chemical form of elemental iodine from the suppression pool water.

Details of the AST analysis for suppression pool pH control are provided in Attachment (4). Based on the results of this analysis, the SLC system will be credited for limiting radiological dose following LOCAs involving fuel damage.

A1-4.3 Atmospheric Dispersion Factors

New atmospheric dispersion factors (X/Q values) are calculated for use in evaluating the radiological consequences of the design basis accidents. Offsite EAB and LPZ X/Q values are calculated using the guidance of RG 1.145 (Reference A1-8.19) and the PAVAN computer code. Conservative estimates of X/Q values for accident releases (except the MSLB accident) to the NMP2 control room air intakes and to the TSC air intake are calculated using the ARCON96 computer code, consistent with the procedures given in RG 1.194 (Reference A1-8.20). These calculations use meteorological data collected by the Nine Mile Point onsite meteorological measurements program for the five-year period from 1997 through 2001. For the MSLB accident, the X/Q values for the control room air intakes are determined using an instantaneous ground-level puff release model, as described in RG 1.194.

Additional information regarding the onsite meteorological measurement program, the X/Q calculation methodology, and the results of the new X/Q calculations is provided in Attachment (6). All input files for ARCON96 and PAVAN, including the meteorological data input files, are provided in Calculation H21C076 (see Attachment 7).

A1-4.4 NUREG-0737 Evaluation

An evaluation was performed to identify potential impacts of applying AST methodologies on the following NUREG-0737 (Reference A1-8.9) items:

<u>Item II.B.2, Post-Accident Vital Area Access</u>

The source terms (airborne activity in the reactor building and activity in the suppression pool water) for the doses in areas where access is required post-accident were evaluated to assess the impact of AST. The evaluation determined that the existing TID-14844 based analyses are conservative and bounding. Given compliance with the GDC-19 limit of 5 Rem when dose is based on TID-14844 source terms, compliance with 10 CFR 50.67 control room dose limits can be expected with the AST-based analysis. Therefore, the historically analyzed cases are sufficient, and no additional analysis of vital areas is necessary. In addition, no new post-accident access requirements have been identified as a result of implementing AST.

<u>Items III.A.1.2 and III.D.3.4, Control Room and Technical Support Center Habitability</u>

The control room radiological dose impact of AST has been specifically calculated for each of the four DBAs analyzed for AST implementation, and TSC habitability has been analyzed for the DBA LOCA. The results of these analyses are presented in Section A1-4.1 above.

Item III.D.1.1, Primary Coolant Outside Containment

The contributions to the radiological dose consequences resulting from piping shine and post-LOCA ESF leakage were considered as part of the radiological dose analysis for the LOCA. The LOCA analysis methodology and results are presented in Section A1-4.1.3 above.

A1-4.5 Proposed Revisions to the Technical Specifications

This section provides the justification for the proposed revisions to the TS that are associated with the licensing basis revision to implement the AST. The AST analyses described in the preceding discussions and the enclosed calculations support these changes. Attachment (2) provides the existing TS pages marked-up to show the proposed changes.

A1-4.5.1 TS 1.0, Definitions

The definition for DOSE EQUIVALENT I-131 is revised to conform to the implementation of the AST. The revised accident analyses use committed effective dose equivalent dose conversion factors from Table 2.1 of Federal Guidance Report (FGR) 11. This reference is cited in RG 1.183.

With the implementation of AST, the previous whole body and thyroid dose guidelines of 10 CFR 100.11 and 10 CFR 50, Appendix A, GDC 19, are replaced by the TEDE criteria of 10 CFR 50.67(b)(2). The analyses performed in support of this license amendment request determined radiological consequences in terms of the TEDE dose quantity and were shown to be in compliance with the dose criteria of 10 CFR 50.67. This new definition is acceptable since it reflects adoption of the dose conversion factors and dose consequences of the revised radiological analyses.

A1-4.5.2 TS 3.1.7, Standby Liquid Control (SLC) System

The Applicability of TS 3.1.7, "Standby Liquid Control (SLC) System," is revised to include Mode 3 (Hot Shutdown), and Required Action C has been revised to add an additional action (C.2) to be in Mode 4 within 36 hours. These changes support the use of the SLC system for buffering the suppression pool pH following a LOCA involving fuel damage, consistent with the AST methodology and analysis assumptions.

A1-4.5.3 TS 3.3.7.1, Control Room Envelope Filtration (CREF) System Instrumentation

The operability requirements for the "Main Control Room Ventilation Radiation Monitor – High" function (Function 3) are revised by deleting the requirement that this function be operable during core alterations, and by replacing the term "irradiated fuel assemblies" with "recently irradiated fuel assemblies." Analysis of the radiological consequences of the design basis FHA using AST methodology involving irradiated fuel assemblies that have been allowed to decay for 24 hours shows that the calculated TEDE values both offsite (EAB and LPZ) and to control room occupants are below the applicable acceptance values (see Section A1-4.1.5 above). This analysis does not credit operation of the CREF system or the instrumentation that initiates CREF system operation. The proposed change is consistent with the revised CREF system operability requirements discussed in Section A1-4.5.4 below.

A1-4.5.4 TS 3.7.2, Control Room Envelope Filtration (CREF) System

The operability requirements for the CREF system are revised by deleting the requirement that this system be operable during core alterations and to replace the term "irradiated fuel assemblies" with "recently irradiated fuel assemblies." Analysis of the radiological consequences of the design basis FHA using AST methodology involving irradiated fuel assemblies that have been allowed to decay for 24 hours shows that the calculated TEDE values both offsite (EAB and LPZ) and to control room occupants are below the applicable acceptance values (see Section A1-4.1.5 above). This analysis does not credit operation of the CREF system. Thus, after 24 hours of decay time, movement of irradiated fuel assemblies can commence and continue without the CREF system being operable. This change is consistent with the scope and intent of Technical Specification Task Force (TSTF) traveler TSTF-51-A, Revise Containment Requirements during Handling Irradiated Fuel and Core Alterations," Revision 2 (Reference A1-8.5), which was approved by the NRC on November 1, 1999. NMP2 has previously incorporated the TSTF-51-A changes for secondary containment systems in License Amendment No. 101, issued by the NRC on February 11, 2002 (Reference A1-8.8).

A1-4.5.5 TS 3.7.3, Control Room Envelope Air Conditioning (AC) System

The operability requirements for the Control Room Envelope AC system are revised by deleting the requirement that this system be operable during core alterations and to replace the term "irradiated fuel assemblies" with "recently irradiated fuel assemblies." Analysis of the radiological consequences of the design basis FHA using AST methodology involving irradiated fuel assemblies that have been allowed to decay for 24 hours shows that the calculated TEDE values both offsite (EAB and LPZ) and to control room occupants are below the applicable acceptance values (see Section A1-4.1.5 above). This analysis does not credit operation of the Control Room Envelope AC system. Thus, after 24 hours of decay time, movement of irradiated fuel assemblies can commence and continue without the Control Room Envelope AC system being operable. This change is consistent with the scope and intent of TSTF-51-A. NMP2 has previously incorporated the TSTF-51-A changes for secondary containment systems in License Amendment No. 101, issued by the NRC on February 11, 2002 (Reference A1-8.8).

A1-4.5.6 TS Bases Changes

With implementation of the AST, the accident whole body and thyroid dose guidelines of 10 CFR Part 50, Appendix A, GDC 19 and 10 CFR 100 are replaced by the TEDE criteria of 10 CFR 50.67. Thus, references to GDC 19 and 10 CFR 100 are replaced with 10 CFR 50.67. In addition, the definition of "recently irradiated fuel" that is currently contained in the Bases for various secondary containment-related TS sections is revised from "the previous 2 days" to "the previous 24 hours," consistent with the AST analysis of the FHA.

Other changes are being made to the TS Bases for clarity and to conform to the changes being made to the associated TS sections. The revisions to the TS Bases incorporate supporting information for the proposed TS changes. The Bases do not establish actual requirements, and as such do not change technical requirements of the TS. The Bases changes are therefore acceptable, since they administratively document the reasons and provide additional understanding for the associated TS requirements. The TS Bases changes will be processed in accordance with the NMP2 TS Bases Control Program (TS 5.5.10).

A1-4.5 Conclusions

Implementation of the AST as the plant radiological consequences analysis licensing basis requires a license amendment pursuant to the requirements of 10 CFR 50.67. The analyses described above demonstrate that the offsite and control room post-accident doses will not exceed the values specified in 10 CFR 50.67 following AST implementation. It has also been determined that continued compliance with NUREG-0737, Item II.B.2, will be maintained and that vital areas remain accessible post-accident. Implementation of the AST provides the basis for proposed changes to the Technical Specifications described herein. This submittal also fulfills the NMPNS commitment for completing and submitting the analysis needed to meet Generic Letter 2003-01 objectives.

The habitability of the NMP2 control room due to a DBA at NMP1, and the habitability of the NMP1 control room due to a DBA at NMP2, have been evaluated using AST methodology. The resultant control room doses have been determined to be within the regulatory limits.

Based on the considerations discussed above and detailed in the attachments and enclosures to this submittal, (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 requested license amendment will not be inimical to the common defense and security or to the health and safety of the public.

A1-5. NO SIGNIFICANT HAZARDS DETERMINATION

Nine Mile Point Nuclear Station, LLC (NMPNS) is requesting a revision to Renewed Operating License No. NPF-69 for Nine Mile Point Unit 2 (NMP2). The proposed amendment would revise the accident source term used in the NMP2 design basis radiological consequence analyses in accordance with 10 CFR 50.67. The proposed accident source term revision replaces the current methodology that is based on Technical Information Document (TID)-14844, "Calculation of Distance Factors for Power and Test Reactor Sites," with the alternative source term (AST) methodology described in Regulatory Guide 1.183, "Alternative Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors." The proposed license amendment request is for full implementation of the AST as described in Regulatory Guide 1.183, with the exception that TID-14844 will continue to be used as the radiation dose basis for equipment qualification and vital area access.

The AST analyses were performed using the guidance provided in Regulatory Guide 1.183 and Standard Review Plan Section 15.0.1, "Radiological Consequence Analyses Using Alternative Source Terms." The four limiting design basis accidents (DBAs) considered were the Loss of Coolant Accident (LOCA), the Main Steam Line Break Accident, the Fuel Handling Accident, and the Control Rod Drop Accident.

NMPNS has evaluated whether or not a significant hazards consideration is involved with the proposed amendment by focusing on the three standards set forth in 10 CFR 50.92, "Issuance of amendment," as discussed below:

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

   Response: No.

   Adoption of the AST and those plant systems affected by implementing AST do not initiate DBAs. The AST does not affect the design or manner in which the facility is operated; rather, for postulated accidents, the AST is an input to calculations that evaluate the radiological consequences. The AST does not by itself affect the post-accident plant response or the actual pathway of the radiation released from the fuel. It does, however, better represent the physical characteristics of the release, so that appropriate mitigation techniques may be applied. Implementation of the AST has been incorporated in the analyses for the limiting DBAs at NMP2.

   The structures, systems and components affected by the proposed change mitigate the consequences of accidents after the accident has been initiated. Application of the AST does result in changes to NMP2 Updated Safety Analysis Report (USAR) functions (e.g., Standby Liquid Control system). As a condition of application of AST, NMPNS is proposing to use the Standby Liquid Control system to control the suppression pool pH following a LOCA. These changes do not require any physical modifications to the plant. As a result, the proposed changes do not involve a revision to the parameters or conditions that could contribute to the initiation of a DBA discussed in Chapter 15 of the NMP2 USAR. Since design basis accident initiators are not being altered by adoption of the AST, the probability of an accident previously evaluated is not affected.

   Plant-specific AST radiological analyses have been performed and, based on the results of these analyses, it has been demonstrated that the dose consequences of the limiting events considered in the analyses are within the acceptance criteria provided by the NRC for use with the AST. These criteria are presented in 10 CFR 50.67 and Regulatory Guide 1.183. Even though the AST dose limits are not directly comparable to the previously specified whole body and thyroid dose guidelines of General Design Criterion 19 and 10 CFR 100.11, the results of the AST analyses have demonstrated that the 10 CFR 50.67 limits are satisfied. Therefore, it is concluded that adoption of the AST does not involve a significant increase in the consequences of an accident previously evaluated.

   Based on the above discussion, it is concluded that the proposed changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

   Response: No.

   Implementation of AST and the proposed changes does not alter or involve any design basis accident initiators. These changes do not involve any physical changes to the plant and do not affect the design function or mode of operations of systems, structures, or components in the facility prior to a postulated accident. Since systems, structures, and components are operated essentially no differently after the AST implementation, no new failure modes are created by this proposed change.

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

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

Response: No.

The changes proposed are associated with a new licensing basis for analysis of NMP2 DBAs. Approval of the licensing basis change from the original source term to the AST is being requested. The results of the accident analyses performed in support of the proposed changes are subject to revised acceptance criteria. The limiting DBAs have been analyzed using conservative methodologies, in accordance with the guidance contained in Regulatory Guide 1.183, to ensure that analyzed events are bounding and that safety margin has not been reduced. The dose consequences of these limiting events are within the acceptance criteria presented in 10 CFR 50.67 and Regulatory Guide 1.183. Thus, the proposed changes continue to ensure that the doses at the exclusion area boundary and low population zone boundary, as well as in the control room, are within corresponding regulatory criteria.

Therefore, by meeting the applicable regulatory criteria for AST, it is concluded that the proposed changes do not involve a significant reduction in a margin of safety.

Based on the above, NMPNS concludes that the proposed amendment presents no significant hazards considerations under the standards set forth in 10 CFR 50.92(c), and, accordingly, a finding of "no significant hazards consideration" is justified.

A1-6. ENVIRONMENTAL ASSESSMENT

A review has determined that the proposed amendment would change a requirement with respect to installation or use of a facility component located within the restricted area, as defined in 10 CFR 20, or would change an inspection or surveillance requirement. However, the proposed amendment does not involve (i) a significant hazards consideration, (ii) a significant change in the types or significant increase in the amounts of any effluent that may be released offsite, or (iii) a significant increase in individual or cumulative occupational radiation exposure. Accordingly, the proposed amendment meets the eligibility criterion for categorical exclusion set forth in 10 CFR 51.22(c)(9). Therefore, pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the proposed amendment.

A1-7. PRECEDENT

Other boiling water reactor plants have previously submitted, and the NRC has approved, applications for the use of AST using approaches similar to those described in this submittal for NMP2. These include Vermont Yankee Nuclear Power Station (TAC No. MC0253, approved March 29, 2005), Browns Ferry Nuclear Plant (TAC Nos. MB5733, MB5734, MB5735, approved September 27, 2004), Limerick Generating Station (TAC Nos. MC2295 and MC2296, approved August 23, 2006), and Columbia Generating Station (TAC No. MC4570, approved November 27, 2006).

A1-8. REFERENCES

1. Letter from W. C. Holston (NMPNS) to Document Control Desk (NRC), dated January 31, 2005, Response to Generic Letter 2003-01, Control Room Habitability (TAC Nos. MB9825, MB9826)

2. Letter from T. J. O'Connor (NMPNS) to Document Control Desk (NRC), dated January 27, 2006, Response to NRC Generic Letter 2003-01, Control Room Habitability – Commitment Completion Date Change (TAC Nos. MB9825 and MB9826)

3. NRC Regulatory Guide 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors," July 2000

4. J. J. DiNunno et al., Technical Information Document (TID)-14844, "Calculation of Distance Factors for Power and Test Reactor Sites," U.S. Atomic Energy Commission (now NRC), 1962

5. Technical Specification Task Force Improved Standard Technical Specifications Change Traveler TSTF-51-A, "Revise Containment Requirements During Handling Irradiated Fuel and Core Alterations," Revision 2

6. L. Soffer et al., NUREG-1465, "Accident Source Terms for Light-Water Nuclear Power Plants," USNRC, February 1995

7. NUREG-0800, Standard Review Plan, Section 15.0.1, "Radiological Consequence Analyses Using Alternative Source Terms," Rev. 0, July 2000

8. Letter from P. S. Tam (NRC) to J. T. Conway (NMPNS), dated February 11, 2002, Nine Mile Point Nuclear Station, Unit No. 2 – Issuance of Amendment Re: Ventilation Requirements during Irradiated Fuel Handling (TAC No. MB1479)

9. NUREG-0737, "Clarification of TMI Action Plan Requirements," November 1980

10. ORIGEN2 Computer Code, Oak Ridge National Laboratory

11. NUREG/CR-6604, "RADTRAD: A simplified Model for Radionuclide Transport and Removal and Dose Estimation," April 1998, and Supplement 1, June 8, 1999

12. STARDOSE Model Report, Polestar Applied Technology, Inc., January 31, 1997

13. MicroShield, Version 7.02, Grove Engineering Inc.

14. QADMOD, Version 0, Level 3, Oak Ridge National Laboratory

15. STARNAUA Code, Polestar Applied Technology, Inc.

16. RADDECAY, Version 3, Grove Engineering, Inc.

17. NEDE-24011-P-A-14, "General Electric Standard Application for Reactor Fuel (GESTAR II)," U. S. Supplement, June 2000

18. NEDE-24011-P-A (GESTAR-II), Amendment 22

19. NRC Regulatory Guide 1.145, "Atmospheric Dispersion Models for Potential Accident Consequence Assessments at Nuclear Power Plants," Revision 1, November 1982

20. NRC Regulatory Guide 1.194, "Atmospheric Relative Concentrations for Control Room Radiological Habitability Assessments at Nuclear Power Plants," June 2003

21. Letter from D. S. Brinkman (NRC) to B. R. Sylvia (NMPC) dated August 30, 1994, Issuance of Amendment for Nine Mile Point Nuclear Station, Unit 2 (TAC No. M89785)

22. Scaperow, J., et. al., "Assessment of Radiological Consequences for the Perry Pilot Plant Application using the Revised (NUREG-1465) Source Term," AEB-98-03, December 9, 1998

23. Letter from J. K. Thayer (Entergy Nuclear Vermont Yankee, LLC) to Document Control Desk (NRC) dated July 31, 2003, Technical Specification Proposed Change No. 262, Alternative Source Term (TAC No. MC0253)

24. Letter from T. E. Abney (Tennessee Valley Authority) to Document Control Desk (NRC), dated July 31, 2002, Browns Ferry Nuclear Plant (BFN) – Units 1, 2, and 3 – License Amendment - Alternative Source Term (TAC Nos. MB5733, MB5734, MB5735, MC0156, MC0157, MC0158)

25. Letter from D. K. Atkinson (Northwest Energy) to Document Control Desk (NRC) dated September 30, 2004, Columbia Generating Station, Docket No. 50-397, License Amendment Request – Alternative Source Term (TAC No. MC4570)

26. Letter from J. H. Mueller (NMPC) to Document Control Desk (NRC) dated March 29, 2001, Proposed Technical Specification Changes – Incorporation of NRC-Approved Generic Changes TSTF-51, TSTF-204, and TSTF-287

27. NUREG-0800, Standard Review Plan, Section 6.5.2, "Containment Spray as a Fission Product Cleanup System," Revision 2, December 1988

28. Regulatory Issue Summary (RIS) 2006-04, "Experience with Implementation of Alternative Source Terms," March 7, 2006

A1-9. REGULATORY COMMITMENTS

The following table identifies those actions committed to by NMPNS in this submittal. Any other statements in this submittal are provided for information purposes and are not considered to be regulatory commitments.

REGULATORY COMMITMENTS	DUE DATE
Environmental qualification of SLC system components for the post-LOCA environment associated with the new suppression pool pH control function will be established in accordance with the station design change process prior to completing implementation of the AST license amendment.	120 days following NRC approval of the license amendment request.
The Emergency Operating Procedures (EOPs) and Severe Accident Procedures (SAPs) will be revised, as appropriate, to reflect the post-LOCA function of the SLC system, include instructions to manually actuate the SLC system based on high drywell radiation levels, and assure that, once initiated, the entire contents of the SLC system storage tank are injected to accomplish the pH control function.	120 days following NRC approval of the license amendment request.
Training will be provided to licensed operators and shift technical advisors (STAs) for the procedure revisions that specifically address sodium pentaborate solution injection for pH control following a LOCA.	120 days following NRC approval of the license amendment request.

Table A1-1
Conformance with Regulatory Guide (RG) 1.183
Section C, Regulatory Position

RG Section	RG Position	NMP2 Analysis	Comments
3.1	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 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 ORIGEN 2 or ORIGEN-ARP. Core inventory factors (Ci/MWt) provided in TID 14844 and used in some analysis computer codes were derived for low burnup, low enrichment fuel and should not be used with higher burnup and higher enrichment fuels.	Conforms	The ORIGEN2 code was used to determine core isotopic inventory, based on a 24-month fuel cycle, 1,400 effective full power days (EFPD) per cycle, and 4.1% average enrichment. The inventory of fission products in the core is based on the current licensed reactor core thermal power of 3,467 MWt plus 2% (i.e., 3,536 MWt), and then increased using a simple ratio to accommodate an assumed future extended power uprate to 3,988 MWt +2% (i.e., 4,067 MWt).
3.1	For the DBA LOCA, all fuel assemblies in the core are assumed to be affected and the core average inventory should be used. For DBA events that do not involve the entire core, the fission product inventory of each of the damaged fuel rods is determined by dividing the total core inventory by the number of fuel rods in the core. To account for differences in power level across the core, radial peaking factors from the facility's core operating limits report (COLR) or technical specifications should be applied in determining the inventory of the damaged rods.	Conforms	A bounding peaking factor of 1.8 is used for DBA events that do not involve the entire core, with fission product inventories for damaged fuel rods determined by dividing the total core inventory by the number of fuel rods in the core.

Table A1-1
Conformance with Regulatory Guide (RG) 1.183
Section C, Regulatory Position

RG Section	RG Position	NMP2 Analysis	Comments
3.1	No adjustment to the fission product inventory should be made for events postulated to occur during power operations at less than full rated power or those postulated to occur at the beginning of core life. For events postulated to occur while the facility is shutdown, e.g., a fuel handling accident, radioactive decay from the time of shutdown may be modeled.	Conforms	No adjustments for less than full power are made in any analyses. The fuel handling accident (FHA) models radioactive decay from the time of shutdown.
3.2	The core inventory release fractions, by radionuclide groups, for the gap release and early in-vessel damage phases for DBA LOCAs are listed in Table 1 for BWRs and Table 2 for PWRs. These fractions are applied to the equilibrium core inventory described in Regulatory Position 3.1. Footnote 10 to Position 3.2: The release fractions listed here have been determined to be acceptable for use with currently approved LWR fuel with a peak burnup up to 62,000 MWD/MTU. The data in this section may not be applicable to cores containing mixed oxide (MOX) fuel.	Conforms	The fractions from Regulatory Position 3.2, Table 1 are used. The criteria of Footnote 10 to Position 3.2 are met.

Table A1-1
Conformance with Regulatory Guide (RG) 1.183
Section C, Regulatory Position

RG Section	RG Position	NMP2 Analysis	Comments
3.2	For non-LOCA events, the fractions of the core inventory assumed to be in the gap for the various radionuclides are given in Table 3. The release fractions from Table 3 are used in conjunction with the fission product inventory calculated with the maximum core radial peaking factor. Footnote 11 to Table 3 of Position 3.2: The release fractions listed here have been determined to be acceptable for use with currently approved LWR fuel with a peak burnup up to 62,000 MWD/MTU provided that the maximum linear heat generation rate does not exceed 6.3 kw/ft peak rod average power for burnups exceeding 54 GWD/MTU. As an alternative, fission gas release calculations performed using NRC approved methodologies may be considered on a case-by-case basis. To be acceptable, these calculations must use a projected power history that will bound the limiting projected plant-specific power history for the specific fuel load. For the BWR rod drop accident and the PWR rod ejection accident, the gap fractions are assumed to be 10% for iodines and noble gases.	Conforms	Conforms to Footnote 11 of Table 3 of Position 3.2. A bounding peaking factor of 1.8 is used for DBA events that do not involve the entire core.
3.3	Table 4 tabulates the onset and duration of each sequential release phase for DBA LOCAs at PWRs and BWRs. The specified onset is the time following the initiation of the accident (i.e., time = 0). The early in-vessel phase immediately follows the gap release phase. The activity released from the core during each release phase should be modeled as increasing in a linear fashion over the duration of the phase. For non-LOCA DBAs in which fuel damage is projected, the release from the fuel gap and the fuel pellet should be assumed to occur instantaneously with the onset of the projected damage.	Conforms	The BWR durations from Table 4 of Position 3.3 are used. The LOCA is modeled in a linear fashion. Non-LOCA events are modeled as an instantaneous release.

Table A1-1
Conformance with Regulatory Guide (RG) 1.183
Section C, Regulatory Position

RG Section	RG Position	NMP2 Analysis	Comments
3.3	For facilities licensed with leak-before-break methodology, the onset of the gap release phase may be assumed to be 10 minutes. A licensee may propose an alternative time for the onset of the gap release phase, based on facility-specific calculations using suitable analysis codes or on an accepted topical report shown to be applicable to the specific facility. In the absence of approved alternatives, the gap release phase onsets in Table 4 should be used.	N/A	NMP2 does not use leak-before-break methodology for the DBA analyses.
3.4	Table 5 lists the elements in each radionuclide group that should be considered in design basis analyses.	Conforms	This guidance is applied in the analyses, as supplemented by RIS 2006-04 (Reference A1-8.28).
3.5	Of the radioiodine released from the reactor coolant system (RCS) to the containment in a postulated accident, 95 percent of the iodine released should be assumed to be cesium iodide (CsI), 4.85 percent elemental iodine, and 0.15 percent organic iodide. This includes releases from the gap and the fuel pellets. With the exception of elemental and organic iodine and noble gases, fission products should be assumed to be in particulate form. The same chemical form is assumed in releases from fuel pins in FHAs and from releases from the fuel pins through the RCS in DBAs other than FHAs or LOCAs. However, the transport of these iodine species following release from the fuel may affect these assumed fractions. The accident-specific appendices to this regulatory guide provide additional details.	Conforms	This guidance is applied in the analyses.

Table A1-1
Conformance with Regulatory Guide (RG) 1.183
Section C, Regulatory Position

RG Section	RG Position	NMP2 Analysis	Comments
3.6	The amount of fuel damage caused by non-LOCA design basis events should be analyzed to determine, for the case resulting in the highest radioactivity release, the fraction of the fuel that reaches or exceeds the initiation temperature of fuel melt and the fraction of fuel elements for which the fuel clad is breached. Although the NRC staff has traditionally relied upon the departure from nucleate boiling ratio (DNBR) as a fuel damage criterion, licensees may propose other methods to the NRC staff, such as those based upon enthalpy deposition, for estimating fuel damage for the purpose of establishing radioactivity releases.	Conforms	Fuel damage assessment for the CRDA is based on GESTAR standard analyses to estimate fuel damage.
4.1.1	The dose calculations should determine the TEDE. TEDE is the sum of the committed effective dose equivalent (CEDE) from inhalation and the deep dose equivalent (DDE) from external exposure. The calculation of these two components of the TEDE should consider all radionuclides, including progeny from the decay of parent radionuclides, that are significant with regard to dose consequences and the released radioactivity.	Conforms	This guidance is applied in the analyses, as supplemented by RIS 2006-04. TEDE doses are calculated by RADTRAD, with decay and daughter products enabled. Additional noble gases and Rb88 are also included.
4.1.2	The exposure-to-CEDE factors for inhalation of radioactive material should be derived from the data provided in ICRP Publication 30, "Limits for Intakes of Radionuclides by Workers." Table 2.1 of Federal Guidance Report 11, "Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion," provides tables of conversion factors acceptable to the NRC staff. The factors in the column headed "effective" yield doses corresponding to the CEDE.	Conforms	Federal Guidance Report (FGR) 11 dose conversion factors are used.

Table A1-1
Conformance with Regulatory Guide (RG) 1.183
Section C, Regulatory Position

RG Section	RG Position	NMP2 Analysis	Comments
4.1.3	For the first 8 hours, the breathing rate of persons offsite should be assumed to be 3.5 x 10^{-4} cubic meters per second. From 8 to 24 hours following the accident, the breathing rate should be assumed to be 1.8 x 10^{-4} cubic meters per second. After that and until the end of the accident, the rate should be assumed to be 2.3 x 10^{-4} cubic meters per second.	Conforms	This guidance is applied in the analyses.
4.1.4	The DDE should be calculated assuming submergence in semi-infinite cloud assumptions with appropriate credit for attenuation by body tissue. The DDE is nominally equivalent to the effective dose equivalent (EDE) from external exposure if the whole body is irradiated uniformly. Since this is a reasonable assumption for submergence exposure situations, EDE may be used in lieu of DDE in determining the contribution of external dose to the TEDE. Table III.1 of Federal Guidance Report 12, "External Exposure to Radionuclides in Air, Water, and Soil," provides external EDE conversion factors acceptable to the NRC staff. The factors in the column headed "effective" yield doses corresponding to the EDE.	Conforms	FGR 12 conversion factors are used.

Table A1-1
Conformance with Regulatory Guide (RG) 1.183
Section C, Regulatory Position

RG Section	RG Position	NMP2 Analysis	Comments
4.1.5	The TEDE should be determined for the most limiting person at the EAB. The maximum EAB TEDE for any two-hour period following the start of the radioactivity release should be determined and used in determining compliance with the dose criteria in 10 CFR 50.67. The maximum two-hour TEDE should be determined by calculating the postulated dose for a series of small time increments and performing a "sliding" sum over the increments for successive two-hour periods. The maximum TEDE obtained is submitted. The time increments should appropriately reflect the progression of the accident to capture the peak dose interval between the start of the event and the end of radioactivity release (see also Table 6).	Conforms	The maximum two-hour LOCA EAB doses have been calculated.
4.1.6	TEDE should be determined for the most limiting receptor at the outer boundary of the low population zone (LPZ) and should be used in determining compliance with the dose criteria in 10 CFR 50.67.	Conforms	This guidance is applied in the analyses.
4.1.7	No correction should be made for depletion of the effluent plume by deposition on the ground.	Conforms	No such corrections are made in the analyses.

Table A1-1
Conformance with Regulatory Guide (RG) 1.183
Section C, Regulatory Position

RG Section	RG Position	NMP2 Analysis	Comments
4.2.1	The TEDE analysis should consider all sources of radiation that will cause exposure to control room personnel. The applicable sources will vary from facility to facility, but typically will include: • Contamination of the control room atmosphere by the intake or infiltration of the radioactive material contained in the radioactive plume released from the facility, • Contamination of the control room atmosphere by the intake or infiltration of airborne radioactive material from areas and structures adjacent to the control room envelope, • Radiation shine from the external radioactive plume released from the facility, • Radiation shine from radioactive material in the reactor containment, • Radiation shine from radioactive material in systems and components inside or external to the control room envelope, e.g., radioactive material buildup in recirculation filters.	Conforms	All sources of radiation that will cause exposure to control room personnel have been considered in the analyses.
4.2.2	The radioactive material releases and radiation levels used in the control room dose analysis should be determined using the same source term, transport, and release assumptions used for determining the EAB and the LPZ TEDE values, unless these assumptions would result in non-conservative results for the control room.	Conforms	The source term, transport, and release methodology are the same for both the control room and offsite locations.

Table A1-1
Conformance with Regulatory Guide (RG) 1.183
Section C, Regulatory Position

RG Section	RG Position	NMP2 Analysis	Comments
4.2.3	The models used to transport radioactive material into and through the control room, and the shielding models used to determine radiation dose rates from external sources, should be structured to provide suitably conservative estimates of the exposure to control room personnel.	Conforms	This guidance is applied in the analyses. The models used in the AST analyses are described in Section A1-4 and are suitably conservative.
4.2.4	Credit for engineered safety features that mitigate airborne radioactive material within the control room may be assumed. Such features may include control room isolation or pressurization, or intake or recirculation filtration. Refer to Section 6.5.1, "ESF Atmospheric Cleanup System," of the SRP and Regulatory Guide 1.52, "Design, Testing, and Maintenance Criteria for Postaccident Engineered-Safety-Feature Atmosphere Cleanup System Air Filtration and Adsorption Units of Light-Water-Cooled Nuclear Power Plants," for guidance. The control room design is often optimized for the DBA LOCA and the protection afforded for other accident sequences may not be as advantageous. In most designs, control room isolation is actuated by engineered safeguards feature (ESF) signals or radiation monitors (RMs). In some cases, the ESF signal is effective only for selected accidents, placing reliance on the RMs for the remaining accidents. Several aspects of RMs can delay the control room isolation, including the delay for activity to build up to concentrations equivalent to the alarm setpoint and the effects of different radionuclide accident isotopic mixes on monitor response.	Conforms	Filtration of intake air by the CREF system is credited in the LOCA analysis. The CREF system is automatically initiated upon a LOCA signal or upon a high radiation signal from either of the two control room air intake radiation monitors. No credit for filtration by the CREF system is taken in the MSLB accident, FHA, or CRDA analyses.
4.2.5	Credit should generally not be taken for the use of personal protective equipment or prophylactic drugs. Deviations may be considered on a case-by-case basis.	Conforms	Such credits are not taken.

Table A1-1
Conformance with Regulatory Guide (RG) 1.183
Section C, Regulatory Position

RG Section	RG Position	NMP2 Analysis	Comments
4.2.6	The dose receptor for these analyses is the hypothetical maximum exposed individual who is present in the control room for 100% of the time during the first 24 hours after the event, 60% of the time between 1 and 4 days, and 40% of the time from 4 days to 30 days. For the duration of the event, the breathing rate of this individual should be assumed to be 3.5×10^{-4} cubic meters per second.	Conforms	This guidance is applied in the analyses.
4.2.7	Control room doses should be calculated using dose conversion factors identified in Regulatory Position 4.1 above for use in offsite dose analyses. The DDE from photons may be corrected for the difference between finite cloud geometry in the control room and the semi-infinite cloud assumption used in calculating the dose conversion factors. The following expression may be used to correct the semi-infinite cloud dose, DDE_{∞}, to a finite cloud dose, DDE_{finite}, where the control room is modeled as a hemisphere that has a volume, V, in cubic feet, equivalent to that of the control room. $$DDE_{finite} = \frac{DDE_{\infty} V^{0.338}}{1173}$$	Conforms	This guidance is applied in the analyses.
4.3	The guidance provided in Regulatory Positions 4.1 and 4.2 should be used, as applicable, in re-assessing the radiological analyses identified in Regulatory Position 1.3.1, such as those in NUREG-0737. Design envelope source terms provided in NUREG-0737 should be updated for consistency with the AST. In general, radiation exposures to plant personnel identified in Regulatory Position 1.3.1 should be expressed in terms of TEDE. Integrated radiation exposure of plant equipment should be determined using the guidance of Appendix I of this guide.	Conforms	Based on an evaluation, the existing TID-14844 based analyses are shown to be conservative and bounding. Given compliance with the GDC-19 limit of 5 Rem when dose is based on TID-14844 source terms, compliance with 10 CFR 50.67 control room dose limits can be expected with the AST-based analysis. Therefore, the historically analyzed cases are sufficient and no additional analysis of vital areas is necessary.

Table A1-1
Conformance with Regulatory Guide (RG) 1.183
Section C, Regulatory Position

RG Section	RG Position	NMP2 Analysis	Comments
4.4	The radiological criteria for the EAB, the outer boundary of the LPZ, and for the control room are in 10 CFR 50.67. These criteria are stated for evaluating reactor accidents of exceedingly low probability of occurrence and low risk of public exposure to radiation, e.g., a large-break LOCA. The control room criterion applies to all accidents. For events with a higher probability of occurrence, postulated EAB and LPZ doses should not exceed the criteria tabulated in Table 6. The acceptance criteria for the various NUREG-0737 (Ref. 2) items generally reference General Design Criteria 19 (GDC 19) from Appendix A to 10 CFR Part 50 or specify criteria derived from GDC-19. These criteria are generally specified in terms of whole body dose, or its equivalent to any body organ. For facilities applying for, or having received, approval for the use of an AST, the applicable criteria should be updated for consistency with the TEDE criterion in 10 CFR 50.67(b)(2)(iii).	Conforms	This guidance is applied in the analyses of design basis accidents. See RG Section 4.3 above regarding NUREG-0737 items.
5.1.1	The evaluations required by 10 CFR 50.67 are re-analyses of the design basis safety analyses and evaluations required by 10 CFR 50.34; they are considered to be a significant input to the evaluations required by 10 CFR 50.92 or 10 CFR 50.59. These analyses should be prepared, reviewed, and maintained in accordance with quality assurance programs that comply with Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants," to 10 CFR Part 50.	Conforms	The AST analyses were prepared as specified in the guidance.

Table A1-1
Conformance with Regulatory Guide (RG) 1.183
Section C, Regulatory Position

RG Section	RG Position	NMP2 Analysis	Comments
5.1.2	Credit may be taken for accident mitigation features that are classified as safety-related, are required to be operable by technical specifications, are powered by emergency power sources, and are either automatically actuated or, in limited cases, have actuation requirements explicitly addressed in emergency operating procedures. The single active component failure that results in the most limiting radiological consequences should be assumed. Assumptions regarding the occurrence and timing of a loss of offsite power should be selected with the objective of maximizing the postulated radiological consequences.	Conforms, based on results of evaluation	Systems credited for accident mitigation include the drywell spray system, the SGTS, and the CREF system. These systems are classified as safety-related, are required to be operable by technical specifications, are powered by emergency power sources, and, except for the drywell spray system, are automatically actuated. The drywell spray system is manually initiated from the control room according to emergency operating and severe accident procedures. The analyses also take credit for SLC system operation for post-LOCA suppression pool pH control. The SLC system is safety-related, required to be operable by technical specifications, and supplied with emergency power. The SLC system is manually initiated from the control room according to emergency operating and severe accident procedures. Suitability of the SLC system to perform the post-LOCA pH control function is addressed in Attachment (4).
5.1.3	The numeric values that are chosen as inputs to the analyses required by 10 CFR 50.67 should be selected with the objective of determining a conservative postulated dose. In some instances, a particular parameter may be conservative in one portion of an analysis but be nonconservative in another portion of the same analysis.	Conforms	Conservative assumptions were used in the analyses.
5.1.4	Licensees should ensure that analysis assumptions and methods are compatible with the ASTs and the TEDE criteria.	Conforms	Analysis assumptions and methods were in accordance with this guidance.

Table A1-1
Conformance with Regulatory Guide (RG) 1.183
Section C, Regulatory Position

RG Section	RG Position	NMP2 Analysis	Comments
5.3	Atmospheric dispersion values (X/Q) for the EAB, the LPZ, and the control room that were approved by the staff during initial facility licensing or in subsequent licensing proceedings may be used in performing the radiological analyses identified by this guide. Methodologies that have been used for determining X/Q values are documented in Regulatory Guides 1.3 and 1.4, Regulatory Guide 1.145, "Atmospheric Dispersion Models for Potential Accident Consequence Assessments at Nuclear Power Plants," and the paper, "Nuclear Power Plant Control Room Ventilation System Design for Meeting General Criterion 19." References 22 (Murphy and Campe paper, August 1974) and 28 (RG 1.145) should be used if the FSAR X/Q values are to be revised or if values are to be determined for new release points or receptor distances. Fumigation should be considered where applicable for the EAB and LPZ. For the EAB, the assumed fumigation period should be timed to be included in the worst 2-hour exposure period. The NRC computer code PAVAN implements Regulatory Guide 1.145 and its use is acceptable to the NRC staff. The methodology of the NRC computer code ARCON96 is generally acceptable to the NRC staff for use in determining control room X/Q values. Meteorological data collected in accordance with the site-specific meteorological measurements program described in the facility FSAR should be used in generating accident X/Q values. Additional guidance is provided in Regulatory Guide 1.23, "Onsite Meteorological Programs." All changes in X/Q analysis methodology should be reviewed by the NRC staff.	Conforms	New atmospheric dispersion values (X/Q) for the EAB, the LPZ, and the control room have been calculated using the ARCON96 and PAVAN computer codes and meteorological data for the five-year period from 1997 through 2001. See Attachment (6).

Table A1-2 Conformance with Regulatory Guide (RG) 1.183 Appendix A, Assumptions for Evaluating the Radiological Consequences of a LWR Loss-of-Coolant Accident			
RG Section	**RG Position**	**NMP2 Analysis**	**Comments**
1	Acceptable assumptions regarding core inventory and the release of radionuclides from the fuel are provided in Regulatory Position 3 of this guide.	Conforms	See RG Section 3 below.
2	If the sump or suppression pool pH is controlled at values of 7 or greater, the chemical form of radioiodine released to the containment should be assumed to be 95% cesium iodide (CsI), 4.85 percent elemental iodine, and 0.15 percent organic iodide. Iodine species, including those from iodine re-evolution, for sump or suppression pool pH values less than 7 will be evaluated on a case-by-case basis. Evaluations of pH should consider the effect of acids and bases created during the LOCA event, e.g., radiolysis products. With the exception of elemental and organic iodine and noble gases, fission products should be assumed to be in particulate form.	Conforms	The stated distributions of iodine chemical forms are used. The post-LOCA suppression pool pH has been evaluated. The pH remains above 7 for at least 30 days by injection of sodium pentaborate solution by the SLC system. See Attachments (4) and (5).
3.1	The radioactivity released from the fuel should be assumed to mix instantaneously and homogeneously throughout the free air volume of the primary containment in PWRs or the drywell in BWRs as it is released. This distribution should be adjusted if there are internal compartments that have limited ventilation exchange. The suppression pool free air volume may be included provided there is a mechanism to ensure mixing between the drywell to the wetwell. The release into the containment or drywell should be assumed to terminate at the end of the early in-vessel phase.	Conforms	The radioactivity released from the fuel is assumed to instantaneously and homogenously mix throughout the drywell air space. Mixing with the wetwell air space is assumed to occur after the release from the core has ended (at 2.033 hours). At this time, considerable thermal-hydraulic activity in the PC will result in the drywell and wetwell air space volumes becoming well-mixed.

Table A1-2
Conformance with Regulatory Guide (RG) 1.183
Appendix A, Assumptions for Evaluating the Radiological Consequences of a LWR Loss-of-Coolant Accident

RG Section	RG Position	NMP2 Analysis	Comments
3.2	Reduction in airborne radioactivity in the containment by natural deposition within the containment may be credited. Acceptable models for removal of iodine and aerosols are described in Chapter 6.5.2, "Containment Spray as a Fission Product Cleanup System," of the Standard Review Plan (SRP), NUREG-0800 and in NUREG/CR-6189, "A Simplified Model of Aerosol Removal by Natural Processes in Reactor Containments." The latter model is incorporated into the analysis code RADTRAD. The prior practice of deterministically assuming that a 50% plateout of iodine is released from the fuel is no longer acceptable to the NRC staff as it is inconsistent with the characteristics of the revised source terms.	Conforms	No credit for natural deposition within the containment is taken.

Table A1-2
Conformance with Regulatory Guide (RG) 1.183
Appendix A, Assumptions for Evaluating the Radiological Consequences of a LWR Loss-of-Coolant Accident

RG Section	RG Position	NMP2 Analysis	Comments
3.3	Reduction in airborne radioactivity in the containment by containment spray systems that have been designed and are maintained in accordance with Chapter 6.5.2 of the SRP may be credited. Acceptable models for the removal of iodine and aerosols are described in Chapter 6.5.2 of the SRP and NUREG/CR-5966, "A Simplified Model of Aerosol Removal by Containment Sprays." This simplified model is incorporated into the analysis code RADTRAD. The evaluation of the containment sprays should address areas within the primary containment that are not covered by the spray drops. The mixing rate attributed to natural convection between sprayed and unsprayed regions of the containment building, provided that adequate flow exists between these regions, is assumed to be two turnovers of the unsprayed regions per hour, unless other rates are justified. The containment building atmosphere may be considered a single, well-mixed volume if the spray covers at least 90% of the volume and if adequate mixing of unsprayed compartments can be shown. The SRP sets forth a maximum decontamination factor (DF) for elemental iodine based on the maximum iodine activity in the primary containment atmosphere when the sprays actuate, divided by the activity of iodine remaining at some time after decontamination. The SRP also states that the particulate iodine removal rate should be reduced by a factor of 10 when a DF of 50 is reached. The reduction in the removal rate is not required if the removal rate is based on the calculated time-dependent airborne aerosol mass. There is no specified maximum DF for aerosol removal by sprays. The maximum activity to be used in determining the DF is defined as the iodine activity in the columns labeled "Total" in Tables 1 and 2 of this guide multiplied by 0.05 for elemental iodine and by 0.95 for particulate iodine (i.e., aerosol treated as particulate in SRP methodology).	Conforms	Credit is taken for reduction in airborne activity in the containment by the drywell spray system as determined using the methodology of SRP Section 6.5.2.

Table A1-2
Conformance with Regulatory Guide (RG) 1.183
Appendix A, Assumptions for Evaluating the Radiological Consequences of a LWR Loss-of-Coolant Accident

RG Section	RG Position	NMP2 Analysis	Comments
3.4	Reduction in airborne radioactivity in the containment by in-containment recirculation filter systems may be credited if these systems meet the guidance of Regulatory Guide 1.52 and Generic Letter 99-02. The filter media loading caused by the increased aerosol release associated with the revised source term should be addressed.	N/A	No in-containment recirculation filter systems exist at NMP2.
3.5	Reduction in airborne radioactivity in the containment by suppression pool scrubbing in BWRs should generally not be credited. However, the staff may consider such reduction on an individual case basis. The evaluation should consider the relative timing of the blowdown and the fission product release from the fuel, the force driving the release through the pool, and the potential for any bypass of the suppression pool. Analyses should consider iodine re-evolution if the suppression pool liquid pH is not maintained greater than 7.	Conforms	No credit is taken for suppression pool scrubbing in the LOCA AST re-analysis. Analyses have been performed that determined that the suppression pool pH is maintained greater than 7; therefore, iodine re-evolution is not expected. See Attachment (4).
3.6	Reduction in airborne radioactivity in the containment by retention in ice condensers, or other engineering safety features not addressed above, should be evaluated on an individual case basis. See Section 6.5.4 of the SRP.	N/A	NMP2 does not have ice condensers. Other than the containment spray system, NMP2 does not have any other systems for the reduction of airborne radioactivity in the containment.

Table A1-2
Conformance with Regulatory Guide (RG) 1.183
Appendix A, Assumptions for Evaluating the Radiological Consequences of a LWR Loss-of-Coolant Accident

RG Section	RG Position	NMP2 Analysis	Comments
3.7	The primary containment (i.e., drywell for Mark I and II containment designs) should be assumed to leak at the peak pressure technical specification leak rate for the first 24 hours. For PWRs, the leak rate may be reduced after the first 24 hours to 50% of the technical specification leak rate. For BWRs, leakage may be reduced after the first 24 hours, if supported by plant configuration and analyses, to a value not less than 50% of the technical specification leak rate. Leakage from subatmospheric containments is assumed to terminate when the containment is brought to and maintained at a subatmospheric condition as defined by technical specifications. For BWRs with Mark III containments, the leakage from the drywell into the primary containment should be based on the steaming rate of the heated reactor core, with no credit for core debris relocation. This leakage should be assumed during the two-hour period between the initial blowdown and termination of the fuel radioactivity release (gap and early in-vessel release phases). After two hours, the radioactivity is assumed to be uniformly distributed throughout the drywell and the primary containment.	Conforms	NMP2 has a Mark II containment. The primary containment leakage is assumed to be 1.1% of containment air weight per day, in accordance with TS SR 3.6.1.1.1 and the 10 CFR 50 Appendix J Testing Program Plan (TS 5.5.12). An additional 0.12% per day is included for traversing in-core probe (TIP) leakage, assuming a TIP is inserted when the LOCA event occurs, the guide tube fails, the TIP fails to withdraw, and the shear valve fails to close. Therefore, the analysis assumes a total of 1.22% of containment air weight per day for 24 hours, and 0.61% per day from 24 hours to 720 hours based on containment pressure reductions (for the MSIV failure scenario only).

Table A1-2
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Appendix A, Assumptions for Evaluating the Radiological Consequences of a LWR Loss-of-Coolant Accident

RG Section	RG Position	NMP2 Analysis	Comments
3.8	If the primary containment is routinely purged during power operations, releases via the purge system prior to containment isolation should be analyzed and the resulting doses summed with the postulated doses from other release paths. The purge release evaluation should assume that 100% of the radionuclide inventory in the reactor coolant system liquid is released to the containment at the initiation of the LOCA. This inventory should be based on the technical specification reactor coolant system equilibrium activity. Iodine spikes need not be considered. If the purge system is not isolated before the onset of the gap release phase, the release fractions associated with the gap release and early in-vessel phases should be considered as applicable.	Conforms	The NMP2 primary containment is not routinely purged during power operation. Purging is limited to inerting, de-inerting, and occasional short pressure control activities. However, for the AST analysis, purging through the 2-inch pressure control line is assumed to be in progress when the LOCA occurs. These releases have been evaluated and have been shown to be negligible.
4.1	Leakage from the primary containment should be considered to be collected, processed by engineered safety feature (ESF) filters, if any, and released to the environment via the secondary containment exhaust system during periods in which the secondary containment has a negative pressure as defined in technical specifications. Credit for an elevated release should be assumed only if the point of physical release is more than two and one-half times the height of any adjacent structure.	Conforms	This guidance is applied in the analyses. Since the NMP2 main stack height is greater than 2.5 times the height of adjacent structures, releases from the main stack are considered elevated releases.
4.2	Leakage from the primary containment is assumed to be released directly to the environment as a ground-level release during any period in which the secondary containment does not have a negative pressure as defined in technical specifications.	Conforms	This guidance is applied in the analyses.

Table A1-2
Conformance with Regulatory Guide (RG) 1.183
Appendix A, Assumptions for Evaluating the Radiological Consequences of a LWR Loss-of-Coolant Accident

RG Section	RG Position	NMP2 Analysis	Comments
4.3	The effect of high wind speeds 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. Ambient temperatures used in these assessments should be the 1-hour average value that is exceeded only 5% or 95% of the total numbers of hours in the data set, whichever is conservative for the intended use (e.g., if high temperatures are limiting, use those exceeded only 5%).	Conforms	Wind speed is not factored into the secondary containment drawdown analysis. However, a conservative bounding outdoor temperature of -20°F is used rather than the 5th percentile value of 19.5°F (the coldest recorded temperature for 1997-2001 was -4.5°F), along with an indoor temperature of 105°F. The current NMP2 drawdown analysis, described in USAR Section 6.2.3, was reviewed and accepted by the NRC as part of License Amendment No. 56 (Reference A1-8.21).
4.4	Credit for dilution in the secondary containment may be allowed when adequate means to cause mixing can be demonstrated. Otherwise, the leakage from the primary containment should be assumed to be transported directly to exhaust systems without mixing. Credit for mixing, if found to be appropriate, should generally be limited to 50%. This evaluation should consider the magnitude of the containment leakage in relation to contiguous building volume or exhaust rate, the location of exhaust plenums relative to projected release locations, the recirculation ventilation systems, and internal walls and floors that impede stream flow between the release and the exhaust.	Conforms	A 50% mixing credit is taken for dilution/mixing in secondary containment. This credit is consistent with the current licensing basis for NMP2 and was included in the supporting information that was reviewed and accepted by the NRC in License Amendment No. 56 issued by NRC letter dated August 30, 1994 (Reference A1-8.21). Mixing in the secondary containment is provided by the reactor building emergency recirculation system unit coolers together with local area unit coolers.

Table A1-2 Conformance with Regulatory Guide (RG) 1.183 Appendix A, Assumptions for Evaluating the Radiological Consequences of a LWR Loss-of-Coolant Accident			
RG Section	**RG Position**	**NMP2 Analysis**	**Comments**
4.5	Primary containment leakage that bypasses the secondary containment should be evaluated at the bypass leak rate incorporated in the technical specifications. If the bypass leakage is through water, e.g., via a filled piping run that is maintained full, credit for retention of iodine and aerosols may be considered on a case-by-case basis. Similarly, deposition of aerosol radioactivity in gas-filled lines may be considered on a case-by-case basis.	Conforms	Reactor building (secondary containment) bypass leakage rates are included in the analysis. The maximum allowable bypass leakage values are specified in TS Table 3.6.1.3-1. No credit is taken for retention in water filled piping. Deposition in gas-filled lines is considered only in the main steam piping between the main steam isolation valves and in the piping between the isolation valves in the other secondary containment bypass lines. In addition, delay of activity releases via the main steam lines and other selected secondary containment bypass pathways is credited in the analysis, as discussed in Section A1-4.1.1.6. Credit for delay time is consistent with the current NMP2 licensing basis.
4.6	Reduction in the amount of radioactive material released from the secondary containment because of ESF filter systems may be taken into account provided that these systems meet the guidance of Regulatory Guide 1.52 and Generic Letter 99-02.	Conforms	The SGTS HEPA and charcoal adsorber filters are credited in the evaluation of a LOCA for onsite and offsite dose consequences. The SGTS is a safety related system and is described in USAR Section 6.5.1. Filter testing is in accordance with the Ventilation Filter Testing Program (TS 5.5.7). The SGTS meets the guidance of Generic Letter 99-02 and Regulatory Guide 1.52, Rev. 2.

Table A1-2 Conformance with Regulatory Guide (RG) 1.183 Appendix A, Assumptions for Evaluating the Radiological Consequences of a LWR Loss-of-Coolant Accident			
RG Section	**RG Position**	**NMP2 Analysis**	**Comments**
5.1	With the exception of noble gases, all the fission products released from the fuel to the containment (as defined in Tables 1 and 2 of this guide) should be assumed to instantaneously and homogeneously mix in the primary containment sump water (in PWRs) or suppression pool (in BWRs) at the time of release from the core. In lieu of this deterministic approach, suitably conservative mechanistic models for the transport of airborne activity in containment to the sump water may be used. Note that many of the parameters that make spray and deposition models conservative with regard to containment airborne leakage are nonconservative with regard to the buildup of sump activity.	Conforms	With the exception of noble gases, all the fission products released from the fuel to the containment are assumed to instantaneously and homogeneously mix in the suppression pool at the time of release from the core.

Table A1-2
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Appendix A, Assumptions for Evaluating the Radiological Consequences of a LWR Loss-of-Coolant Accident

RG Section	RG Position	NMP2 Analysis	Comments
5.2	The leakage should be taken as two times the sum of the simultaneous leakage from all components in the ESF recirculation systems above which the technical specifications, or licensee commitments to item III.D.1.1 of NUREG-0737, would require declaring such systems inoperable. The leakage should be assumed to start at the earliest time the recirculation flow occurs in these systems and end at the latest time the releases from these systems are terminated. Consideration should also be given to design leakage through valves isolating ESF recirculation systems from tanks vented to atmosphere, e.g., emergency core cooling system (ECCS) pump miniflow return to the refueling water storage tank.	Conforms	ESF leakage is minimized at NMP2 through implementation of the program specified in TS 5.5.2, "Primary Coolant Sources Outside Containment." The leak rate of 62 gpm assumed in the AST analyses is comprised of two contributors: (1) Two times the sum of the allowed simultaneous leakage from all ECCS components (2 x 1 gpm = 2 gpm); and (2) 60 gpm due to the assumed failure of two RHR system sample lines in the reactor building and allowed leakage past the two isolation valves in an RHR line to the liquid radwaste system (this value is not doubled). The high pressure core spray system (HPCS) initially takes suction from the condensate storage tank (CST). Leakage to atmospheric tanks is credible only for lines connecting from ECCS pump discharges to such a tank, due to relative elevations. The only applicable leakage paths are the HPCS and Reactor Core Isolation Cooling (RCIC) test lines that discharge to the CST. These lines are isolated by two normally closed valves. Evaluations have shown that the dose contribution due to leakage past these valves to the CST is negligible. ESF leakage is conservatively assumed to begin at the time of the accident.

Table A1-2
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Appendix A, Assumptions for Evaluating the Radiological Consequences of a LWR Loss-of-Coolant Accident

RG Section	RG Position	NMP2 Analysis	Comments
5.3	With the exception of iodine, all radioactive materials in the recirculating liquid should be assumed to be retained in the liquid phase.	Conforms	This guidance is applied in the analyses.
5.4	If the temperature of the leakage exceeds 212°F, the fraction of total iodine in the liquid that becomes airborne should be assumed equal to the fraction of the leakage that flashes to vapor. This flash fraction, FF, should be determined using a constant enthalpy, h, process, based on the maximum time-dependent temperature of the sump water circulating outside the containment: $$FF = \frac{h_{f1} - h_{f2}}{h_{fg}}$$ *Where:* h_{f1} is the enthalpy of liquid at system design temperature and pressure; h_{f2} is the enthalpy of liquid at saturation conditions (14.7 psia, 212°F); and h_{fg} is the heat of vaporization at 212°F.	N/A	The temperature of the leakage does not exceed 212°F.
5.5	If the temperature of the leakage is less than 212°F or the calculated flash fraction is less than 10%, the amount of iodine that becomes airborne should be assumed to be 10% of the total iodine activity in the leaked fluid, unless a smaller amount can be justified based on the actual sump pH history and area ventilation rates.	Conforms	The temperature of the leakage does not exceed 212°F. A release fraction of 10% is used.

Table A1-2
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Appendix A, Assumptions for Evaluating the Radiological Consequences of a LWR Loss-of-Coolant Accident

RG Section	RG Position	NMP2 Analysis	Comments
5.6	The radioiodine that is postulated to be available for release to the environment is assumed to be 97% elemental and 3% organic. Reduction in release activity by dilution or holdup within buildings, or by ESF ventilation filtration systems, may be credited where applicable. Filter systems used in these applications should be evaluated against the guidance of Regulatory Guide 1.52 and Generic Letter 99-02.	Conforms	This guidance is applied in the analyses. Reduction in release activity by the SGTS HEPA and charcoal adsorber filters is credited after negative pressure is re-established in the secondary containment. The SGTS is a safety related system and is described in USAR Section 6.5.1. Filter testing is in accordance with the Ventilation Filter Testing Program (TS 5.5.7). The SGTS meets the guidance of Generic Letter 99-02 and Regulatory Guide 1.52, Rev. 2.
6.1	For the purpose of this analysis, the activity available for release via MSIV leakage should be assumed to be that activity determined to be in the drywell for evaluating containment leakage (see Regulatory Position 3). No credit should be assumed for activity reduction by the steam separators or by iodine partitioning in the reactor vessel.	Conforms	This guidance is applied in the analyses.
6.2	All the MSIVs should be assumed to leak at the maximum leak rate above which the technical specifications would require declaring the MSIVs inoperable. The leakage should be assumed to continue for the duration of the accident. Postulated leakage may be reduced after the first 24 hours, if supported by site-specific analyses, to a value not less than 50% of the maximum leak rate.	Conforms	MSIV leakage assumed in this accident analysis is 24 scfh for any one line and 96 scfh for all four steam lines when tested at $\geq$ 40 psig. The maximum allowable MSIV leakage values are specified in TS Surveillance Requirement (SR) 3.6.1.3.12 and are controlled by the 10 CFR 50 Appendix J Testing Program Plan (TS 5.5.12). Reduction in leakage rates after 24 hours is based on post-accident containment pressure reductions (where credited). No credit is taken for leakage rate reductions below 50% of the MSIV leakage limit.

Table A1-2
Conformance with Regulatory Guide (RG) 1.183
Appendix A, Assumptions for Evaluating the Radiological Consequences of a LWR Loss-of-Coolant Accident

RG Section	RG Position	NMP2 Analysis	Comments
6.3	Reduction of the amount of released radioactivity by deposition and plateout on steam system piping upstream of the outboard MSIVs may be credited, but the amount of reduction in concentration allowed will be evaluated on an individual case basis. Generally, the model should be based on the assumption of well-mixed volumes, but other models such as slug flow may be used if justified.	Conforms	Credit is taken for deposition on steam system piping between the inboard and outboard MSIVs using the guidance of AEB-98-03 (see Section A1-4.1.1.4).
6.4	In the absence of collection and treatment of releases by ESFs such as the MSIV leakage control system, or as described in paragraph 6.5 below, the MSIV leakage should be assumed to be released to the environment as an unprocessed, ground-level release. Holdup and dilution in the turbine building should not be assumed.	Conforms	This guidance is applied in the analyses. NMP2 does not have a MSIV leakage control system.
6.5	A reduction in MSIV releases that is due to holdup and deposition in main steam piping downstream of the MSIVs and in the main condenser, including the treatment of air ejector effluent by offgas systems, may be credited if the components and piping systems used in the release path are capable of performing their safety function during and following a safe shutdown earthquake (SSE). The amount of reduction allowed will be evaluated on an individual case basis. References A-9 and A-10 provide guidance on acceptable models.	Conforms	Delay of activity of releases via the main steam lines and other selected secondary containment bypass pathways is credited in the analysis, as discussed in Section A1-4.1.1.6. Credit for delay time is consistent with the current NMP2 licensing basis. Deposition in the main steam piping downstream of the outboard MSIVs and in the main condenser is not credited in the analysis, even though the piping is seismically rugged and would remain intact during and after a design basis earthquake.

Table A1-2
Conformance with Regulatory Guide (RG) 1.183
Appendix A, Assumptions for Evaluating the Radiological Consequences of a LWR Loss-of-Coolant Accident

RG Section	RG Position	NMP2 Analysis	Comments
7.0	The radiological consequences from post-LOCA primary containment purging as a combustible gas or pressure control measure should be analyzed. If the installed containment purging capabilities are maintained for purposes of severe accident management and are not credited in any design basis analysis, radiological consequences need not be evaluated. If the primary containment purging is required within 30 days of the LOCA, the results of this analysis should be combined with consequences postulated for other fission product release paths to determine the total calculated radiological consequences from the LOCA. Reduction in the amount of radioactive material released via ESF filter systems may be taken into account provided that these systems meet the guidance in Regulatory Guide 1.52 and Generic Letter 99-02.	Conforms	Containment purging as a combustible gas or pressure control measure is not required nor credited in any design basis analysis for 30 days following a design basis LOCA at NMP2. For the AST analysis, purging through the 2-inch pressure control line is assumed to be in progress when the LOCA occurs.

Table A1-3 Conformance with Regulatory Guide (RG) 1.183 Appendix B, Assumptions for Evaluating the Radiological Consequences of a Fuel Handling Accident			
RG Section	**RG Position**	**NMP2 Analysis**	**Comments**
1.1	The number of fuel rods damaged during the accident should be based on a conservative analysis that considers the most limiting case. This analysis should consider parameters such as the weight of the dropped heavy load or the weight of a dropped fuel assembly (plus any attached handling grapples), the height of the drop, and the compression, torsion, and shear stresses on the irradiated fuel rods. Damage to adjacent fuel assemblies, if applicable (e.g., events over the reactor vessel), should be considered.	Conforms	The analysis assumes that the activity inventory from two full fuel assemblies is released. This is bounding for the 124 damaged rods for GE 8x8 fuel assemblies, the 140 damaged rods for GE11 9x9 fuel assemblies, and the 172 damaged rods for GE14 (10x10) fuel assemblies determined for the current licensing basis (described in USAR Section 15.7.4). The number of fuel rods damaged is based on a 32.95-ft drop onto the reactor core and includes the weight of the grapple. Damage due to a fuel assembly drop into the reactor vessel bounds a drop in the spent fuel pool.
1.2	The fission product release from the breached fuel is based on Regulatory Position 3.2 of this guide and the estimate of the number of fuel rods breached. All the gap activity in the damaged rods is assumed to be instantaneously released. Radionuclides that should be considered include xenons, kryptons, halogens, cesiums, and rubidiums.	Conforms	This guidance is applied in the analyses.
1.3	The chemical form of radioiodine released from the fuel to the spent fuel pool should be assumed to be 95% cesium iodide (CsI), 4.85 percent elemental iodine, and 0.15 percent organic iodide. The CsI released from the fuel is assumed to completely dissociate in the pool water. Because of the low pH of the pool water, the iodine re-evolves as elemental iodine. This is assumed to occur instantaneously. The NRC staff will consider, on a case-by-case basis, justifiable mechanistic treatment of the iodine release from the pool.	Conforms	This guidance is applied in the analyses.

Table A1-3 Conformance with Regulatory Guide (RG) 1.183 Appendix B, Assumptions for Evaluating the Radiological Consequences of a Fuel Handling Accident			
RG Section	**RG Position**	**NMP2 Analysis**	**Comments**
2	If the depth of water above the damaged fuel is 23 feet or greater, the decontamination factors for the elemental and organic species are 500 and 1, respectively, giving an overall effective decontamination factor of 200 (i.e., 99.5% of the total iodine released from the damaged rods is retained by the water). This difference in decontamination factors for elemental (99.85%) and organic iodine (0.15%) species results in the iodine above the water being composed of 57% elemental and 43% organic species. If the depth of water is not 23 feet, the decontamination factor will have to be determined on a case-by-case method.	Conforms	Even though the maximum fuel damage is for a drop in the refueling cavity onto the reactor core, a more conservative spent fuel pool DF for inorganic iodine is used in the analysis. Based on the minimum depth of water in the canal to the spent fuel pool of 22'-3", an overall iodine DF of 175 was calculated. See Section A1-4.1.5.
3	The retention of noble gases in the water in the fuel pool or reactor cavity is negligible (i.e., decontamination factor of 1). Particulate radionuclides are assumed to be retained by the water in the fuel pool or reactor cavity (i.e., infinite decontamination factor).	Conforms	This guidance is applied in the analyses.
4.1	The radioactive material that escapes from the fuel pool to the fuel building is assumed to be released to the environment over a 2-hour time period.	Alternate approach	The radioactive material that escapes from the fuel pool to the reactor building is conservatively assumed to be released instantaneously to the environment.
4.2	A reduction in the amount of radioactive material released from the fuel pool by engineered safety feature (ESF) filter systems may be taken into account provided these systems meet the guidance of Regulatory Guide 1.52 and Generic Letter 99-02. Delays in radiation detection, actuation of the ESF filtration system, or diversion of ventilation flow to the ESF filtration system should be determined and accounted for in the radioactivity release analyses.	Conforms	The radioactive material is assumed to be released directly to the environment. No credit is taken for filtration of the release from the reactor building by the SGTS.

Table A1-3
Conformance with Regulatory Guide (RG) 1.183
Appendix B, Assumptions for Evaluating the Radiological Consequences of a Fuel Handling Accident

RG Section	RG Position	NMP2 Analysis	Comments
4.3	The radioactivity release from the fuel pool should be assumed to be drawn into the ESF filtration system without mixing or dilution in the fuel building. If mixing can be demonstrated, credit for mixing and dilution may be considered on a case-by-case basis. This evaluation should consider the magnitude of the building volume and exhaust rate, the potential for bypass to the environment, the location of exhaust plenums relative to the surface of the pool, recirculation ventilation systems, and internal walls and floors that impede stream flow between the surface of the pool and the exhaust plenums.	N/A	No credit is taken for mixing in the reactor building or filtration of the release from the reactor building by the SGTS.
5.1	If the containment is isolated during fuel handling operations, no radiological consequences need to be analyzed.	N/A	Secondary containment isolation is not credited. The radioactive material is assumed to be released directly to the environment.
5.2	If the containment is open during fuel handling operations, but designed to automatically isolate in the event of a fuel handling accident, the release duration should be based on delays in radiation detection and completion of containment isolation. If it can be shown that containment isolation occurs before radioactivity is released to the environment, no radiological consequences need to be analyzed.	N/A	Secondary containment isolation is not credited. The radioactive material is assumed to be released directly to the environment.

Table A1-3
Conformance with Regulatory Guide (RG) 1.183
Appendix B, Assumptions for Evaluating the Radiological Consequences of a Fuel Handling Accident

RG Section	RG Position	NMP2 Analysis	Comments
5.3	If the containment is open during fuel handling operations (e.g., personnel air lock or equipment hatch is open), the radioactive material that escapes from the reactor cavity pool to the containment is released to the environment over a 2-hour time period. Note 3: The staff will generally require that technical specifications allowing such operations include administrative controls to close the airlock, hatch, or open penetrations within 30 minutes. Such administrative controls will generally require that a dedicated individual be present, with necessary equipment available, to restore containment closure should a fuel handling accident occur. Radiological analyses should generally not credit this manual isolation.	Conforms	Secondary containment is assumed to be open during fuel handling operations, and secondary containment isolation is not credited. An instantaneous release to the environment is assumed. Administrative controls are in place to assure that actions are taken to reduce the potential radiological consequences of a fuel handling accident. These controls were described in Reference A1-8.26, were accepted by the NRC in their safety evaluation for NMP2 License Amendment No. 101 (Reference A1-8.8), and have been incorporated into plant procedures.
5.4	A reduction in the amount of radioactive material released from the containment by ESF filter systems may be taken into account provided that these systems meet the guidance of Regulatory Guide 1.52 and Generic Letter 99-02. Delays in radiation detection, actuation of the ESF filtration system, or diversion of ventilation flow to the ESF filtration system should be determined and accounted for in the radioactivity release analyses.	N/A	No credit is taken for filtration of the release from the reactor building by the SGTS.

Table A1-3
Conformance with Regulatory Guide (RG) 1.183
Appendix B, Assumptions for Evaluating the Radiological Consequences of a Fuel Handling Accident

RG Section	RG Position	NMP2 Analysis	Comments
5.5	Credit for dilution or mixing of the activity released from the reactor cavity by natural or forced convection inside the containment may be considered on a case-by-case basis. Such credit is generally limited to 50% of the containment free volume. This evaluation should consider the magnitude of the containment volume and exhaust rate, the potential for bypass to the environment, the location of exhaust plenums relative to the surface of the reactor cavity, recirculation ventilation systems, and internal walls and floors that impede stream flow between the surface of the reactor cavity and the exhaust plenums.	N/A	The radioactive material is assumed to be released directly to the environment without any credit for dilution or mixing inside the secondary containment.

Table A1-4
Conformance with Regulatory Guide (RG) 1.183
Appendix C, Assumptions for Evaluating the Radiological Consequences of a BWR Rod Drop Accident

RG Section	RG Position	NMP2 Analysis	Comments
1	Assumptions acceptable to the NRC staff regarding core inventory are provided in Regulatory Position 3 of this guide. For the rod drop accident, the release from the breached fuel is based on the estimate of the number of fuel rods breached and the assumption that 10% of the core inventory of the noble gases and iodines is in the fuel gap. The release attributed to fuel melting is based on the fraction of the fuel that reaches or exceeds the initiation temperature for fuel melting and on the assumption that 100% of the noble gases and 50% of the iodines contained in that fraction are released to the reactor coolant.	Conforms	Release fractions in accordance with this guidance are used. Releases are based on the fuel damage postulated to occur (see Section A1-4.1.6), which includes both fuel cladding perforation and fuel melting. A conservative radial peaking factor of 1.8 is used.
2	If no or minimal fuel damage is postulated for the limiting event, the released activity should be the maximum coolant activity (typically 4 μCi/gm DE I-131) allowed by the technical specifications. Note 1: The activity assumed in the analysis should be based on the activity associated with the projected fuel damage or the maximum technical specification values, whichever maximizes the radiological consequences. In determining the dose equivalent I-131 (DE I-131), only the radioiodine associated with normal operations or iodine spikes should be included. Activity from projected fuel damage should not be included.	N/A	Fuel damage is postulated, consistent with the current licensing basis analysis described in USAR Section 15.4.9. The projected fuel damage is the limiting case.
3.1	The activity released from the fuel from either the gap or from fuel pellets is assumed to be instantaneously mixed in the reactor coolant within the pressure vessel.	Conforms	This guidance is applied in the analyses.
3.2	Credit should not be assumed for partitioning in the pressure vessel or for removal by the steam separators.	Conforms	No partitioning is assumed.

Table A1-4
Conformance with Regulatory Guide (RG) 1.183
Appendix C, Assumptions for Evaluating the Radiological Consequences of a BWR Rod Drop Accident

RG Section	RG Position	NMP2 Analysis	Comments
3.3	Of the activity released from the reactor coolant within the pressure vessel, 100% of the noble gases, 10% of the iodine, and 1% of the remaining radionuclides are assumed to reach the turbine and condensers.	Conforms	This guidance is applied in the analyses.
3.4	Of the activity that reaches the turbine and condenser, 100% of the noble gases, 10% of the iodine, and 1% of the particulate radionuclides are available for release to the environment. The turbine and condensers leak to the atmosphere as a ground- level release at a rate of 1% per day for a period of 24 hours, at which time the leakage is assumed to terminate. No credit should be assumed for dilution or holdup within the turbine building. Radioactive decay during holdup in the turbine and condenser may be assumed. Note 2: If there are forced flow paths from the turbine or condenser, such as unisolated motor vacuum pumps or unprocessed air ejectors, the leakage rate should be assumed to be the flow rate associated with the most limiting of these paths. Credit for collection and processing of releases, such as by off gas or standby gas treatment, will be considered on a case-by-case basis.	Conforms	This guidance is applied in the analyses for the two cases analyzed. These cases are (1) release via leakage from the main condenser; and (2) forced flow via the mechanical vacuum pumps. This second case is based on the maximum activity concentration that will not cause isolation of the mechanical vacuum pumps on a high main steam line radiation signal. Retention by the charcoal delay beds in the offgas system is neglected.
3.5	In lieu of the transport assumptions provided in paragraphs 3.2 through 3.4 above, a more mechanistic analysis may be used on a case-by-case basis. Such analyses account for the quantity of contaminated steam carried from the pressure vessel to the turbine and condensers based on a review of the minimum transport time from the pressure vessel to the first main steam isolation (MSIV) and considers MSIV closure time.	N/A	Paragraphs 3.2 through 3.4 (see above) are used in the analysis.

Table A1-4
Conformance with Regulatory Guide (RG) 1.183
Appendix C, Assumptions for Evaluating the Radiological Consequences of a BWR Rod Drop Accident

RG Section	RG Position	NMP2 Analysis	Comments
3.6	The iodine species released from the reactor coolant within the pressure vessel should be assumed to be 95% CsI as an aerosol, 4.85% elemental, and 0.15% organic. The release from the turbine and condenser should be assumed to be 97% elemental and 3% organic.	Conforms	This guidance is applied in the analyses.

Table A1-5
Conformance with Regulatory Guide (RG) 1.183
Appendix D, Assumptions for Evaluating the Radiological Consequences of a BWR Main Steam Line Break Accident

RG Section	RG Position	NMP2 Analysis	Comments
1	Assumptions acceptable to the NRC staff regarding core inventory and the release of radionuclides from the fuel are provided in Regulatory Position 3 of this guide. The release from the breached fuel is based on Regulatory Position 3.2 of this guide and the estimate of the number of fuel rods breached.	N/A	No fuel damage is projected for this event. The release estimate is based on coolant activity.
2	If no or minimal fuel damage is postulated for the limiting event, the released activity should be the maximum coolant activity allowed by technical specification. The iodine concentration in the primary coolant is assumed to correspond to the following two cases in the nuclear steam supply system vendor's standard technical specifications.	Conforms	No fuel damage is expected. There is no concern of uncovering the core, as the swell that results from reactor depressurization will maintain adequate core coverage until MSIV isolation. The released activity corresponds to the maximum primary coolant activity allowed by TS 3.4.8.
2.1	The concentration that is the maximum value (typically 4.0 μCi/gm DE I-131) permitted and corresponds to the conditions of an assumed pre-accident spike, and	Conforms	The analysis assumes coolant activity of 4.0 μCi/gm DE I-131 corresponding to an assumed pre-accident spike.
2.2	The concentration that is the maximum equilibrium value (typically 0.2 μCi/gm DE I-131) permitted for continued full power operation.	N/A	This case was not analyzed since the calculated offsite dose results for the pre-accident spike case are less than 2.5 rem TEDE (the acceptance criterion given in RG 1.183, Table 6, for the equilibrium iodine activity case).
3	The activity released from the fuel should be assumed to mix instantaneously and homogeneously in the reactor coolant. Noble gases should be assumed to enter the steam phase instantaneously.	N/A	No fuel damage is projected for this event. The release estimate is based on coolant activity.

Table A1-5
Conformance with Regulatory Guide (RG) 1.183
Appendix D, Assumptions for Evaluating the Radiological Consequences of a BWR Main Steam Line Break Accident

RG Section	RG Position	NMP2 Analysis	Comments
4.1	The main steam line isolation valves (MSIV) should be assumed to close in the maximum time allowed by technical specifications.	Conforms	Break isolation is assumed in 5.5 seconds, corresponding to the maximum MSIV closing time of 5 seconds plus a closure signal delay time of 0.5 second. This is unchanged from the existing analysis described in USAR Section 15.6.4. The maximum allowed MSIV closure time is specified in TS SR 3.6.1.3.7.
4.2	The total mass of coolant released should be assumed to be that amount in the steam line and connecting lines at the time of the break plus the amount that passes through the valves prior to closure.	Conforms	Mass of coolant released is per this guidance.
4.3	All the radioactivity in the released coolant should be assumed to be released to the atmosphere instantaneously as a ground-level release. No credit should be assumed for plateout, holdup, or dilution within facility buildings.	Conforms	This guidance was used in the analysis.
4.4	The iodine species released from the main steam line should be assumed to be 95% CsI as an aerosol, 4.85% elemental, and 0.15% organic.	Conforms	This guidance was used in the analysis.

Table A1-6
Computer Codes Used in AST
Design Basis Radiological Analyses

Task	Computer Code	Version or Revision	Comments
Determination of X/Q values for on site receptors near building structures.	ARCON96	---	NUREG/CR - 6331, Rev. 1, May 1997
Calculate doses due to MSLB, FHA, and CRDA.	Excel	---	Spreadsheet
General purpose gamma shielding analysis.	MicroShield	7.02	Point Kernel Integration code. Developed by Grove Engineering.
Calculate fission product inventories.	ORIGEN	ORIGEN2	The code is referenced in RG 1.183 and consistent with NRC recommendation. ORNL/TM-7175
Determination of X/Q values for the EAB and LPZ.	PAVAN	2.0	NUREG/CR-2858, Nov. 1982
Perform radioactive decay of the source term.	RADDECAY	Version 3	Developed by Grove Engineering
Calculate both on-site and off-site doses.	RADTRAD	3.03	Referenced by RG 1.183 NUREG/CR-6604, USNRC April 1998
Perform independent check of dose calculations.	STARDOSE	03/01/1997	Polestar Applied Technology code
Evaluate aerosol removal in containment and the main steam lines as a function of time.	STARNAUA	---	Developed by Polestar Applied Technology, Inc. Utilized in other utility AST submittals.
Perform an independent check of MicroShield results.	QADMOD	Version 0, Level 3	Stone & Webster Point Kernel Gamma-Ray Shielding Code

Table A1-7
Fuel Data

Fuel Vendor	General Electric
Fuel Type	GE11 & GE14
Initial Bundle Mass of Uranium (kg)	169.7
Initial Core Average Enrichment (U-235 wt%)	4.1
Core Average Bundle Power (MWt/bundle)	4.538
End of Cycle Core Wide Exposure (MWd/ST)	34,000

Table A1-8
Core Fission Product Inventory

Isotope	Ci/MWt t = 0		Isotope	Ci/MWt t = 0
Kr83M	3.27E+03		I132	3.92E+04
Kr85	3.93E+02		I133	5.51E+04
Kr85M	6.82E+03		I134	6.03E+04
Kr87	1.30E+04		I135	5.16E+04
Kr88	1.83E+04		Xe131M	3.04E+02
Kr89	2.22E+04		Xe133	5.27E+04
Rb86	7.29E+01		Xe133M	1.63E+03
Sr89	2.45E+04		Xe135	1.91E+04
Sr90	3.14E+03		Xe135M	1.09E+04
Sr91	3.10E+04		Xe137	4.80E+04
Sr92	3.38E+04		Xe138	4.50E+04
Y90	3.24E+03		Cs134	7.29E+03
Y91	3.18E+04		Cs136	2.28E+03
Y92	3.40E+04		Cs137	4.35E+03
Y93	3.96E+04		Ba137M	4.12E+03
Zr95	4.46E+04		Ba139	4.89E+04
Zr97	4.51E+04		Ba140	4.71E+04
Nb95	4.48E+04		La140	5.12E+04
Mo99	5.13E+04		La141	4.45E+04
Tc99M	4.49E+04		La142	4.29E+04
Ru103	4.29E+04		Ce141	4.47E+04
Ru105	3.01E+04		Ce143	4.11E+04
Ru106	1.76E+04		Ce144	3.70E+04
Rh105	2.84E+04		Pr143	3.97E+04
Sb127	3.01E+03		Nd147	1.80E+04
Sb129	8.91E+03		Np239	5.78E+05
Te127	3.00E+03		Pu238	1.45E+02
Te127M	4.05E+02		Pu239	1.34E+01
Te129	8.76E+03		Pu240	1.89E+01
Te129M	1.30E+03		Pu241	5.49E+03
Te131M	3.97E+03		Am241	7.48E+00
Te132	3.85E+04		Cm242	1.85E+03
I131	2.71E+04		Cm244	1.23E+02

Table A1-9 Accident Radiological Consequence Analyses Inputs	
Input/Assumption	**Value**
No. of Fuel Assemblies in Core	764
Fuel Type	Current – GE11 (9x9) and GE14 (10x10) Past – GE 8x8
Control Room (CR) Volume	3.81E+05 ft^3
CR Normal Mode Ventilation	Not Used
CR Emergency Mode Ventilation (LOCA only)	1,500 scfm plus 10%
Assumed CR Unfiltered In-leakage Rate	250 scfm*
CR Filtered Recirculation	750 cfm minus 10%
Control Room Envelope Filtration (CREF) System Filter Efficiencies	Particulates & Elemental I – 99% Organic I – 99%
Reactor Building Free Air Volume	3.88E+06 ft^3
Reactor Building Drawdown Time	60 minutes with one SGTS fan
Standby Gas Treatment System (SGTS) Flow Rate	4,000 cfm with one SGTS fan
SGTS Filter Efficiency	Particulates & Elemental I – 99% Organic I – 99%
Environment Breathing Rate (Regulatory Guide 1.183)	0-8 hours: 3.5E-04 m^3/sec 8-24 hours: 1.8E-04 m^3/sec 1-30 days: 2.3E-04 m^3/sec
Control Room Breathing Rate (Regulatory Guide 1.183)	3.5E-04 m^3/sec
Control Room Occupancy Factors (Regulatory Guide 1.183)	0-1 day: 1.0 1-4 days: 0.6 4-30 days: 0.4

*Bounds the highest measured inleakage value of 174 scfm (documented in Reference A1-8.1).

Table A1-10
LOCA Inputs

Input/Assumption	Value
Fission Products Release Fractions (Regulatory Guide 1.183, Table 1)	**BWR Core Inventory Fraction Released Into Containment** Group / Gap Release Phase / Early In-vessel Phase / Total Noble Gases: 0.05 / 0.95 / 1.0 Halogens: 0.05 / 0.25 / 0.3 Alkali Metals: 0.05 / 0.20 / 0.25 Tellurium Metals: 0.00 / 0.05 / 0.05 Ba, Sr: 0.00 / 0.02 / 0.02 Noble Metals: 0.00 / 0.0025 / 0.0025 Cerium Group: 0.00 / 0.0005 / 0.0005 Lanthanides: 0.00 / 0.0002 / 0.0002
Fission Product Release Timing (Regulatory Guide 1.183, Table 4)	**LOCA Release Phases (BWR)** Phase / Onset / Duration Gap release: 2 min / 0.5 hr Early In-Vessel: 0.5333 hr / 1.5 hr
Fission Product Iodine Chemical Form (Regulatory Guide 1.183, App. A)	Particulate 95% Elemental 4.85% Organic 0.15%
Control Room Isolation	None Assumed
ESF Leakage Release Fractions	10% of the radioiodine in the leaked coolant is assumed to become airborne in the reactor building (secondary containment).
Leakage Rates	
Primary Containment (PC) Leak Rate	1.1% containment air weight/day (Technical Specification limit)
Traversing In-core Probe (TIP) Leakage (PC to SC)	0.12% /day
Secondary Containment (SC) Bypass Leak Rate (beginning at t = 0 hours)	See TS Table 3.6.1.3-1 and the discussion in Section A1-4.1.1.6)
Assumed ESF Leak Rate	62 gpm (See discussion in Section A1-4.1.3.1)

Table A1-10
LOCA Inputs (Cont'd)

Input/Assumption	Value
ESF Leakage Temperature	<212°F
MSIV Leak Rate at Test Pressure of 40 psig	96 scfh total; 24 scfh maximum for one line
Volumes	
Drywell Airspace	306,200 ft^3
Suppression Chamber Airspace	190,800 ft^3 (Minimum)
Suppression Pool	145,000 ft^3 (Minimum)
Reactor Building (Secondary Containment) Free Volume	3,880,000 ft^3
Removal Inputs	
Drywell Spray Flow Rate	5,237.5 gpm*
Drywell Accident Conditions (Max. pressure bounds DBA LOCA and temperature bounds small steam line break)	P = 40 psig, T = 340°F
Steam Line Removal Efficiencies:	
Steam Line Conditions	Saturated Conditions at 1,050 psia
Steam Line Volume: Inboard to Outboard MSIV (each line)	59.27 ft^3 to 65.69 ft^3

* The smaller of the flow rates for the two drywell spray loops.

Table A1-11
LOCA Release Fractions as Release Rates Over the Accident Duration

Time Period (seconds)	Fraction of core inventory*	
0 - 120	No Release	
120 - 1920	Gases	Xe, Kr – 0.1/hr (0.05 total) Elemental I – 4.9E-3/hr (2.4E-3 total) Organic I – 1.5E-4/hr (7.5E-5 total)
	Aerosols	I, Br – 0.095/hr (0.0475 total) Cs, Rb – 0.1/hr (0.05 total)
1920 - 7320	Gases	Xe, Kr – 0.63/hr (0.95 total) Elemental I – 8.1E-3/hr (1.2E-2 total) Organic I – 2.5E-4/hr (3.8E-4 total)
	Aerosols	I, Br – 0.158/hr (0.2375 total) Cs, Rb – 0.133/hr (0.2 total) Te Group – 0.033/hr (0.05 total) Ba, Sr – 0.013/hr (0.02 total) Noble Metals – 1.7E-3/hr (2.5E-3 total) La Group – 1.3E-4/hr (2E-4 total) Ce Group – 3.3E-4/hr (5E-4 total)

*Release fractions and rates are from RG 1.183, Table 1 considering the chemical form described in RG 1.183, Section 3.5 (Reference A1-8.3).

Table A1-12
X/Q Values for LOCA Radiological Dose Calculations
(sec/m^3)

Release Location	Release Timing				
	0-2 hrs	2-8 hrs	8-24 hrs	1-4 days	4-30 days
EAB*					
Rw/Rx Bldg Vent	1.19E-04	—NA—	—NA—	—NA—	—NA—
Main Steam Tunnel	1.19E-04	—NA—	—NA—	—NA—	—NA—
SGT Bldg	1.19E-04	—NA—	—NA—	—NA—	—NA—
PASS Panel	1.19E-04	—NA—	—NA—	—NA—	—NA—
NMP2 Main Stack	2.96E-05	—NA—	—NA—	—NA—	—NA—
LPZ					
Rw/Rx Bldg Vent	1.62E-05	1.62E-05	1.09E-05	4.59E-06	1.33E-06
Main Steam Tunnel	1.62E-05	1.62E-05	1.09E-05	4.59E-06	1.33E-06
SGT Bldg	1.62E-05	1.62E-05	1.09E-05	4.59E-06	1.33E-06
PASS Panel	1.62E-05	1.62E-05	1.09E-05	4.59E-06	1.33E-06
NMP2 Main Stack	1.42E-05	1.42E-05	5.41E-07	2.31E-07	7.65E-08
NMP2 Control Room					
Rw/Rx Bldg Vent	1.09E-03	7.23E-04	2.46E-04	1.92E-04	1.47E-04
Main Steam Tunnel	1.47E-03	8.80E-04	3.32E-04	2.26E-04	1.68E-04
SGT Bldg	5.31E-04	3.70E-04	1.35E-04	9.16E-05	6.70E-05
PASS Panel	3.74E-04	2.05E-04	7.31E-05	5.53E-05	4.04E-05
NMP2 Main Stack	8.03E-05	4.48E-05	1.68E-05	1.20E-05	8.83E-06
Technical Support Center					
Rw/Rx Bldg Vent	2.70E-04	1.64E-04	5.41E-05	3.86E-05	2.86E-05
Main Steam Tunnel	3.27E-04	2.41E-04	8.38E-05	5.95E-05	4.76E-05
SGT Bldg	1.62E-04	1.19E-04	4.28E-05	2.72E-05	2.24E-05
PASS Panel	2.69E-04	1.91E-04	7.19E-05	4.22E-05	3.40E-05
NMP2 Main Stack	4.95E-5	2.69E-05	1.03E-05	6.67E-06	4.85E-06

NA – Not Applicable

* Worst 2 hours

Table A1-13
LOCA Radiological Consequence Analysis
(rem TEDE)

Dose Component	Offsite Dose		Control Room Dose
	EAB	LPZ	
Limiting Case	0.66	0.77	1.65
Regulatory Limit	25	25	5
Current Analysis* (Regulatory Limit) - rem	4.3 (25) Whole Body 22.4 (300) Thyroid	2.6 (25) Whole Body 58.3 (300) Thyroid	1.27 (5) Whole Body 29.4 (30) Thyroid

* EAB, LPZ, and Control Room doses are from USAR Section 15.6.5 (Table 15.6-16b).

Table A1-14
MSLB Accident Inputs

Input/Assumption	Value
Mass Release	4.86E+07 gm total*, of which 4.1E+07 gm is liquid (1.58E+07 gm flashes and is released as steam); and 7.1E+06 gm is steam
MSIV Isolation Time	5.5 seconds*
DE I-131 Equilibrium Value**	0.2 µCi/gm
DE I-131 Pre-Accident Spike	4 µCi/gm

* Unchanged from existing USAR analysis. See USAR Table 15.6-6.

** This case not analyzed, since the EAB and LPZ dose values for the pre-accident spike case are less than 2.5 rem TEDE (the acceptance criterion given in RG 1.183, Table 6 for the equilibrium iodine activity case).

Table A1-15
MSLB Accident Puff Release X/Q Inputs

Input/Assumption	Value
Mass Release	4.86E+07 gm
Bubble Transverse Time, Main Steam Tunnel to Control Room Fresh Air Intake (1 m/s wind speed)	124 seconds

Table A1-16
X/Q Values for MSLB Accident Radiological Dose Calculations
(sec/m^3)

Time Period	Control Room Puff	EAB	LPZ
0 – 2 hrs	1.47E-03	1.19E-04	1.62E-05

Table A1-17
MSLB Calculated Radiological Consequences

(rem TEDE)

Case	Offsite Dose		Control Room Dose
	EAB	**LPZ**	(Puff Release)
4.0 μCi/gm DE I-131**	0.39	0.053	3.0
Regulatory Limit	25	25	5
Current Analysis* (Regulatory Limit) - rem	0.061 (25) Whole Body 6.9 (300) Thyroid	0.0057 (25) Whole Body 0.65 (300) Thyroid	0.0077 (5) Whole Body 26 (30) Thyroid

* Doses are from USAR Section 15.6.4 (Table 15.6-9).

** Since the EAB and LPZ values are less than 2.5 rem TEDE (the acceptance criterion given in RG 1.183, Table 6 for the equilibrium iodine activity case), the equilibrium iodine activity case did not need to be analyzed.

Table A1-18 Fuel Handling Accident Inputs	
Input/Assumption	**Value**
Number of Failed Rods	2 full fuel assemblies*
Radial Peaking Factor	1.8
Fuel Decay Period	24 hrs
Minimum Depth of Water Above Top of Fuel	22 ft 3 inches
Pool Water Iodine Decontamination Factors (DF)	Elemental Iodine – 237 Organic Iodine – 1 Overall – 175
Release Period	Instantaneous
Release Location	Radwaste/Reactor Building Vent
Release Fractions	Noble Gases Excluding Kr-85 5 % Kr-85 10 % I-131 8 % Iodines except I-131 5 %

* The activity inventory from two full fuel assemblies is released. This is bounding for the 124 damaged rods for GE 8x8 fuel assemblies, the 140 damaged rods for GE11 9x9 fuel assemblies, and the 172 damaged rods for GE14 10x10 fuel assemblies determined for the current licensing basis analysis described in USAR Section 15.7.4.

Table A1-19
Fuel Handling Accident Fission Product Inventory

Isotope	Ci/MWt t = 0	Ci/MWt Adjusted	Ci/MWt t = 24 hr
Kr-83m	3.27E+03	same	negligible
Kr-85m	6.82E+03	same	1.66E+02
Kr-85	3.93E+02	7.86E+02	7.86E+02
Kr-87	1.30E+04	same	2.81E-02
Kr-88	1.83E+04	same	5.23E+01
Kr-89	2.22E+04	same	negligible
Te-131m	3.97E+03	*	*
I-131	2.71E+04	4.34E+04	4.00E+04
Xe-131m	3.04E+02	same	3.03E+02
Te-132	3.85E+04	*	*
I-132	3.92E+04	same	3.21E+04
I-133	5.51E+04	same	2.48E+04
Xe-133m	1.63E+03	same	1.48E+03
Xe-133	5.27E+04	same	5.09E+04
I-134	6.03E+04	same	negligible
I-135	5.16E+04	same	4.18E+03
Xe-135m	1.09E+04	same	negligible
Xe-135	1.91E+04	same	1.23E+04
Xe-137	4.80E+04	same	negligible
Xe-138	4.50E+04	same	negligible

* Considered as parent only.

Table A1-20
X/Q Values for Fuel Handling Accident Radiological Dose Calculations
(sec/m^3)

Release Location	Release Timing				
	0-2 hrs	2-8 hrs	8–24 hr	1-4 days	4-30 days
EAB					
Ground	1.19E-04	—NA—	—NA—	—NA—	—NA—
LPZ					
Ground	1.62E-05		—NA—	—NA—	—NA—
Control Room					
Ground	1.09E-03	—NA—	—NA—	—NA—	—NA—

Table A1-21
Fuel Handling Accident Calculated Radiological Consequences
(rem TEDE)

Case	Offsite Dose		Control Room Dose
	EAB	LPZ	
24 Hours after shutdown	0.45	0.061	3.2
Regulatory Limit	6.3	6.3	5
Current Analysis* (Regulatory Limit) - rem	0.65 (25) Whole Body 51 (300) Thyroid	0.061 (25) Whole Body 4.8 (300) Thyroid	0.11 (5) Whole Body 11 (30) Thyroid

* Doses are from USAR Section 15.7.4 (Table 15.7-12).

Table A1-22
CRDA Inputs

Input/Assumption	Value
Number of Failed Rods	770*
Percent Fuel Melt for Failed Rods	0.77%
Radial Peaking Factor	1.8
Release Period (Case 1)	24 hours
Main Condenser Leakage Rate (Case 1)	1% per day for 24 hours
Main Condenser Volume	97,000 ft^3
Main Condenser Mechanical Vacuum Pump Flow Rate (Case 2)	2,500 cfm
Gap Release Fractions	Noble Gas 10% Iodine 10% Br 5% Cs, Rb 12%
Melted Fuel Release Fractions	Noble Gas 100% Iodine 50%
Activity that Reaches the Condenser	Noble Gas 100% Iodine 10% Cs, Rb 1%
Airborne Activity Available for Release from the Condenser	Noble Gas 100% Iodine 10% Cs, Rb 1%

* The current licensing basis analysis (USAR Section 15.4.9) is based on the failure of 770 rods for GE 8x8 fuel assemblies. The resulting activity release is bounding for GE11 9x9 and GE14 10x10 fuel assemblies. CRDA results for banked position withdrawal sequence (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 (References A1-8.17 and A1-8.18). Thus, the CRDA has been deleted from the standard GE BWR reload package for BPWS plants.

Table A1-23
X/Q Values for CRDA Radiological Dose Calculations

(sec/m^3)

Release Location	Release Timing				
	0-2 hrs	2-8 hrs	8-24 hrs	1-4 days	4-30 days
EAB					
Ground	1.19E-04	—NA—	—NA—	—NA—	—NA—
Stack Normal	—NA—	—NA—	—NA—	—NA—	—NA—
Stack Fumigation	2.96E-05	—NA—	—NA—	—NA—	—NA—
LPZ					
Ground	1.62E-05		1.09E-05	4.59E-06	1.33E-06
Stack	1.42E-05		5.41E-07	2.31E-07	7.65E-08
Control Room					
Ground	1.09E-03	7.23E-04	2.46E-04	1.92E-04	1.52E-04
Stack Normal	—NA—	4.48E-05	1.68E-05	1.20E-05	8.83E-06
Stack Fumigation	8.03E-05	—NA—	—NA—	—NA—	—NA—

Table A1-24
CRDA Calculated Radiological Consequences

(rem TEDE)

Case	**Offsite Dose**		**Control Room Dose** (Puff Release)
	EAB	**LPZ**	
Condenser Leakage	0.57	0.077	1.26
Mech. Vacuum Pump	1.0	1.2	2.3
Regulatory Limit	6.3	6.3	5
Current Analysis* (Regulatory Limit) - rem	0.021 (25) Whole Body 0.33 (300) Thyroid	0.006 (25) Whole Body 0.18 (300) Thyroid	0.0019 (5) Whole Body 0.022 (30) Thyroid

* Doses are from USAR Section 15.4.9 (Table 15.4-13).

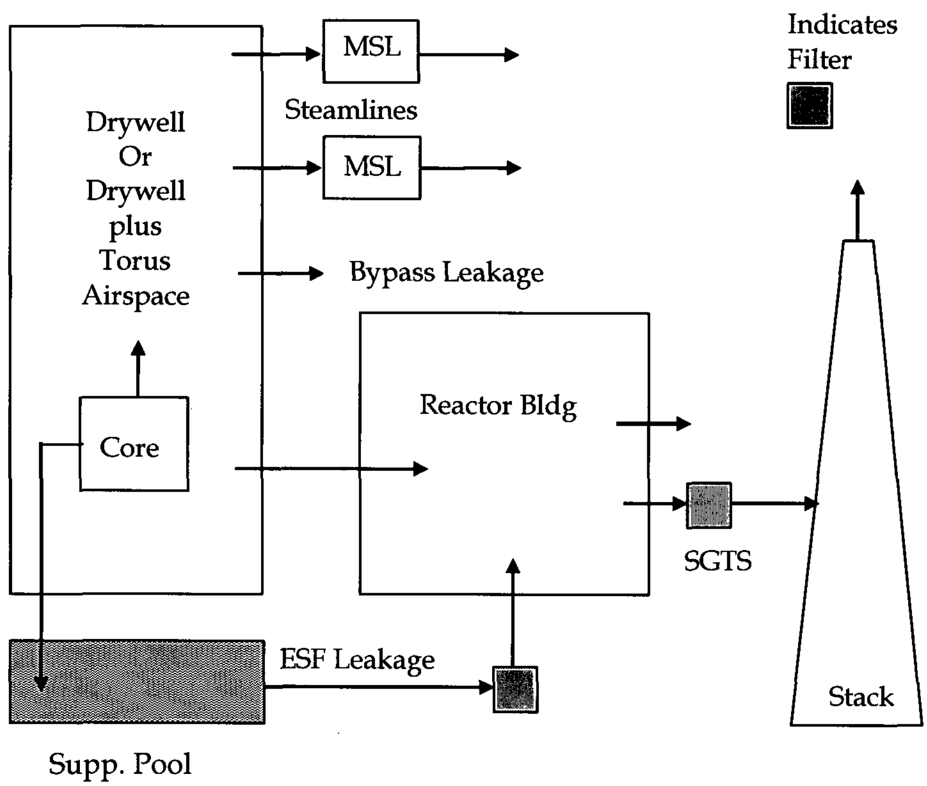

Figure A1-1

RADTRAD Modeling of LOCA Release Pathways

ATTACHMENT (2)

PROPOSED TECHNICAL SPECIFICATION CHANGES (MARK-UP)

The current versions of the following Technical Specifications pages have been marked-up by hand to reflect the proposed changes.

1.1-2
1.1-3
3.1.7-1
3.3.7.1-4
3.7.2-1
3.7.2-2
3.7.2-3
3.7.3-1
3.7.3-2
3.7.3-3

1.1 Definitions (continued)

CHANNEL FUNCTIONAL TEST	A CHANNEL FUNCTIONAL TEST shall be the injection of a simulated or actual signal into the channel as close to the sensor as practicable to verify OPERABILITY, including required alarm, interlock, display, and trip functions, and channel failure trips. The CHANNEL FUNCTIONAL TEST may be performed by means of any series of sequential, overlapping, or total channel steps so that the entire channel is tested.
CORE ALTERATION	CORE ALTERATION shall be the movement of any fuel, sources, or reactivity control components, within the reactor vessel with the vessel head removed and fuel in the vessel. The following exceptions are not considered to be CORE ALTERATIONS: a. Movement of source range monitors, local power range monitors, intermediate range monitors, traversing incore probes, or special movable detectors (including undervessel replacement); and b. Control rod movement, provided there are no fuel assemblies in the associated core cell. Suspension of CORE ALTERATIONS shall not preclude completion of movement of a component to a safe position.
CORE OPERATING LIMITS REPORT (COLR)	The COLR is the unit specific document that provides cycle specific parameter limits for the current reload cycle. These cycle specific limits shall be determined for each reload cycle in accordance with Specification 5.6.5. Plant operation within these limits is addressed in individual Specifications.
DOSE EQUIVALENT I-131	DOSE EQUIVALENT I-131 shall be that concentration of I-131 (microcuries/gram) that alone would produce the same ~~thyroid~~ dose as the quantity and isotopic mixture of I-131, I-132, I-133, I-134, and I-135 actually present. The ~~thyroid~~ dose conversion factors used for this calculation shall be ~~those listed in Table III of TID-14844, AEC, 1962, "Calculation of Distance Factors for Power and Test Reactor Sites;" Table E-7 of~~ Insert 1

(continued)

<u>INSERT 1</u> (for TS page 1.1-2; Definition for DOSE EQUIVALENT I-131)

the Committed Effective Dose Equivalent dose conversion factors listed in Table 2.1 of Federal Guidance Report No. 11, EPA, "Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion," 1988.

1.1 Definitions

DOSE EQUIVALENT I-131 (continued)	~~Regulatory Guide 1.109, Rev. 1, NRC, 1977; or ICRP 30, Supplement to Part 1, page 192-212, Table titled, "Committed Dose Equivalent in Target Organs or Tissues per Intake of Unit Activity."~~
EMERGENCY CORE COOLING SYSTEM (ECCS) RESPONSE TIME	The ECCS RESPONSE TIME shall be that time interval from when the monitored parameter exceeds its ECCS initiation setpoint at the channel sensor until the ECCS equipment is capable of performing its safety function (i.e., the valves travel to their required positions, pump discharge pressures reach their required values, etc.). Times shall include diesel generator starting and sequence loading delays, where applicable. The response time may be measured by means of any series of sequential, overlapping, or total steps so that the entire response time is measured.
END OF CYCLE RECIRCULATION PUMP TRIP (EOC-RPT) SYSTEM RESPONSE TIME	The EOC-RPT SYSTEM RESPONSE TIME shall be that time interval from initial movement of the associated turbine stop valves or turbine control valves to complete suppression of the electric arc between the fully open contacts of the recirculation pump circuit breaker. The response time may be measured by means of any series of sequential, overlapping, or total steps so that the entire response time is measured.
ISOLATION SYSTEM RESPONSE TIME	The ISOLATION SYSTEM RESPONSE TIME shall be that time interval from when the monitored parameter exceeds its isolation initiation setpoint at the channel sensor until the isolation valves travel to their required positions. The response time may be measured by means of any series of sequential, overlapping, or total steps so that the entire response time is measured.
LEAKAGE	LEAKAGE shall be: a. <u>Identified LEAKAGE</u> 1. LEAKAGE into the drywell such as that from pump seals or valve packing, that is captured and conducted to a sump or collecting tank; or

(continued)

3.1 REACTIVITY CONTROL SYSTEMS

3.1.7 Standby Liquid Control (SLC) System

LCO 3.1.7 Two SLC subsystems shall be OPERABLE.

APPLICABILITY: MODES 1 ~~and 2.~~ , 2, and 3.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One SLC subsystem inoperable.	A.1 Restore SLC subsystem to OPERABLE status.	7 days
B. Two SLC subsystems inoperable.	B.1 Restore one SLC subsystem to OPERABLE status.	8 hours
C. Required Action and associated Completion Time not met.	C.1 Be in MODE 3. AND C.2 Be in MODE 4.	12 hours 36 hours

Add

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.1.7.1 Verify available volume of sodium pentaborate solution is within the limits of Figure 3.1.7-1.	24 hours

(continued)

Table 3.3.7.1-1 (page 1 of 1)
Control Room Envelope Filtration System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER TRIP SYSTEM	CONDITIONS REFERENCED FROM REQUIRED ACTION A.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
1. Reactor Vessel Water Level – Low Low, Level 2	1,2,3, (a)	2	B	SR 3.3.7.1.1 SR 3.3.7.1.2 SR 3.3.7.1.3 SR 3.3.7.1.4 SR 3.3.7.1.5	≥ 101.8 inches
2. Drywell Pressure – High	1,2,3	2	C	SR 3.3.7.1.1 SR 3.3.7.1.2 SR 3.3.7.1.3 SR 3.3.7.1.4 SR 3.3.7.1.5	≤ 1.88 psig
3. Main Control Room Ventilation Radiation Monitor – High	1,2,3, (a),(b)	2	B	SR 3.3.7.1.1 SR 3.3.7.1.2 SR 3.3.7.1.4 SR 3.3.7.1.5	≤ 5.92 x 10^{-6} µCi/cc

(a) During operations with a potential for draining the reactor vessel.

(b) During ~~CORE ALTERATIONS, and during~~ movement of recently irradiated fuel assemblies in the secondary containment.

3.7 PLANT SYSTEMS

3.7.2 Control Room Envelope Filtration (CREF) System

LCO 3.7.2 Two CREF subsystems shall be OPERABLE.

---NOTE---
The control room envelope boundary may be opened intermittently under administrative control.

APPLICABILITY: MODES 1, 2, and 3,
During movement of recently irradiated fuel assemblies in the secondary containment,
~~During CORE ALTERATIONS,~~
During operations with a potential for draining the reactor vessel (OPDRVs).

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One CREF subsystem inoperable. OR Two CREF subsystems inoperable with safety function maintained.	A.1 Restore CREF subsystem(s) to OPERABLE status.	7 days
B. Two CREF subsystems inoperable due to inoperable control room envelope boundary in MODES 1, 2, and 3.	B.1 Restore control room envelope boundary to OPERABLE status.	24 hours
C. Required Action and Associated Completion Time of Condition A or B not met in MODE 1, 2, or 3.	C.1 Be in MODE 3. AND C.2 Be in MODE 4.	12 hours 36 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
D. Required Action and associated Completion Time of Condition A not met during movement of recently irradiated fuel assemblies in the secondary containment, ~~during CORE ALTERATIONS,~~ or during OPDRVs.	------------NOTE------------ LCO 3.0.3 is not applicable. ---------------------------------	
	D.1 Place OPERABLE components of CREF subsystem(s) equivalent to a single CREF subsystem in emergency pressurization mode.	Immediately
	<u>OR</u>	
	D.2.1 Suspend movement of recently irradiated fuel assemblies in the secondary containment.	Immediately
	~~<u>AND</u>~~	
	~~D.2.2 Suspend CORE ALTERATIONS.~~	~~Immediately~~
	<u>AND</u>	
	D.2.~~3~~ 2 Initiate action to suspend OPDRVs.	Immediately
E. Two CREF subsystems inoperable with safety function not maintained in MODE 1, 2, or 3 for reasons other than Condition B.	E.1 Enter LCO 3.0.3.	Immediately

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
F. Two CREF subsystems inoperable with safety function not maintained during movement of recently irradiated fuel assemblies in the secondary containment, ~~during CORE ALTERATIONS,~~ or during OPDRVs.	------------NOTE------------ LCO 3.0.3 is not applicable. -------------------------------	
	F.1 Suspend movement of recently irradiated fuel assemblies in the secondary containment.	Immediately
	~~AND~~	
	~~F.2 Suspend CORE ALTERATIONS.~~	~~Immediately~~
	AND	
	F.~~3~~2 Initiate action to suspend OPDRVs.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.2.1	Operate each CREF subsystem for $\geq$1 continuous hour.	31 days
SR 3.7.2.2	Perform required CREF System filter testing in accordance with the Ventilation Filter Testing Program (VFTP).	In accordance with the VFTP
SR 3.7.2.3	Verify each CREF subsystem actuates on an actual or simulated initiation signal.	24 months

(continued)

3.7 PLANT SYSTEMS

3.7.3 Control Room Envelope Air Conditioning (AC) System

LCO 3.7.3 Two control room envelope AC subsystems shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3,
During movement of recently irradiated fuel assemblies in the secondary containment,
~~During CORE ALTERATIONS,~~
During operations with a potential for draining the reactor vessel (OPDRVs).

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One control room envelope AC subsystem inoperable. OR Two control room envelope AC subsystems inoperable with safety function maintained.	A.1 Restore control room envelope AC subsystem(s) to OPERABLE status.	30 days
B. Required Action and Associated Completion Time of Condition A not met in MODE 1, 2, or 3.	B.1 Be in MODE 3. AND B.2 Be in MODE 4.	12 hours 36 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
C. Required Action and associated Completion Time of Condition A not met during movement of recently irradiated fuel assemblies in the secondary containment, ~~during CORE ALTERATIONS,~~ or during OPDRVs.	------------NOTE------------ LCO 3.0.3 is not applicable. ----------------------------		
	C.1	Place OPERABLE components of control room envelope AC subsystem(s) equivalent to a single control room envelope AC subsystem in operation.	Immediately
	OR		
	C.2.1	Suspend movement of recently irradiated fuel assemblies in the secondary containment.	Immediately
	~~AND~~		
	~~C.2.2~~	~~Suspend CORE ALTERATIONS.~~	~~Immediately~~
	AND		
	C.2.~~3~~ 2	Initiate action to suspend OPDRVs.	Immediately
D. Two control room envelope AC subsystems inoperable with safety function not maintained in MODE 1, 2, or 3.	D.1	Enter LCO 3.0.3.	Immediately

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
E. Two control room envelope AC subsystems inoperable with safety function not maintained during movement of recently irradiated fuel assemblies in the secondary containment, ~~during CORE ALTERATIONS,~~ or during OPDRVs.	------------NOTE------------ LCO 3.0.3 is not applicable. ---------------------------- E.1 Suspend movement of recently irradiated fuel assemblies in the secondary containment.	Immediately
	~~AND~~ ~~E.2 Suspend CORE ALTERATIONS.~~	~~Immediately~~
	AND E.~~3~~ 2 Initiate action to suspend OPDRVs.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.7.3.1 Verify each control room envelope AC subsystem has the capability to remove the assumed heat load.	24 months

ATTACHMENT (3)

CHANGES TO TECHNICAL SPECIFICATION BASES (MARK-UP)

The current versions of the following Technical Specifications Bases pages have been marked-up by hand to reflect the proposed changes. These Bases pages are provided for information only and do not require NRC approval.

B 2.0-4	B 3.3.6.2-1	B 3.7.2-1
B 2.0-5	B 3.3.6.2-6	B 3.7.2-2
B 2.0-6	B 3.3.7.1-2	B 3.7.2-3
B 2.0-7	B 3.3.7.1-4	B 3.7.2-4
B 2.0-8	B 3.3.7.1-5	B 3.7.2-6
B 3.1.6-5	B 3.4.8-1	B 3.7.2-7
B 3.1.7-1	B 3.4.8-2	B 3.7.2-9
B 3.1.7-2	B 3.4.8-3	B 3.7.3-2
B 3.1.7-3	B 3.4.8-4	B 3.7.3-3
B 3.1.7-6	B 3.6.1.6-1	B 3.7.3-4
B 3.1.8-1	B 3.6.1.6-2	B 3.7.3-5
B 3.1.8-5	B 3.6.1.6-3	B 3.7.3-6
B 3.2.3-1	B 3.6.1.6-4	B 3.7.4-1
B 3.3.6.1-8	B 3.6.4.1-1	B 3.7.6-1
B 3.3.6.1-9	B 3.6.4.1-2	B 3.7.6-3
B 3.3.6.1-13	B 3.6.4.2-1	B 3.9.6-1
B 3.3.6.1-14	B 3.6.4.2-2	B 3.9.6-3
B 3.3.6.1-15	B 3.6.4.3-2	B 3.9.7-1
	B 3.6.4.3-3	B 3.9.7-3

BASES

APPLICABLE SAFETY ANALYSES (continued)

2.1.1.3 Reactor Vessel Water Level

During MODES 1 and 2, the reactor vessel water level is required to be above the top of the active irradiated fuel to provide core cooling capability. With fuel in the reactor vessel during periods when the reactor is shut down, consideration must be given to water level requirements due to the effect of decay heat. If the water level should drop below the top of the active irradiated fuel during this period, the ability to remove decay heat is reduced. This reduction in cooling capability could lead to elevated cladding temperatures and clad perforation in the event that the water level becomes < 2/3 of the core height. The reactor vessel water level SL has been established at the top of the active irradiated fuel to provide a point that can be monitored and to also provide adequate margin for effective action.

SAFETY LIMITS

The reactor core SLs are established to protect the integrity of the fuel clad barrier to prevent the release of radioactive materials to the environs. SL 2.1.1.1 and SL 2.1.1.2 ensure that the core operates within the fuel design criteria. SL 2.1.1.3 ensures that the reactor vessel water level is greater than the top of the active irradiated fuel in order to prevent elevated clad temperatures and resultant clad perforations.

APPLICABILITY

SLs 2.1.1.1, 2.1.1.2, and 2.1.1.3 are applicable in all MODES.

SAFETY LIMIT VIOLATIONS

2.2

Exceeding an SL may cause fuel damage and create a potential for radioactive releases in excess of 10 CFR ~~100~~ 50.67 ~~"Reactor Site Criteria,"~~ limits (Ref. 5). Therefore, it is required to insert all insertable control rods and restore compliance with the SL within 2 hours. The 2 hour Completion Time ensures that the operators take prompt remedial action and the probability of an accident occurring during this period is minimal.

(continued)

BASES (continued)

REFERENCES

1. 10 CFR 50, Appendix A, GDC 10.

2. GE Service Information Letter No. 516, Supplement 2, "Core Flow Indication in the Low-Flow Region," January 19, 1996.

3. NEDE-24011-P-A, "GE Standard Application for Reactor Fuel," (revision specified in the COLR).

4. Supplemental Reload Licensing Report for Nine Mile Point Nuclear Station Unit 2 (revision specified in the COLR).

5. 10 CFR ~~100~~ 50.67, "Accident Source Term."

B 2.0 SAFETY LIMITS (SLs)

B 2.1.2 Reactor Coolant System (RCS) Pressure SL

BASES

BACKGROUND

The SL on reactor steam dome pressure protects the RCS against overpressurization. In the event of fuel cladding failure, fission products are released into the reactor coolant. The RCS then serves as the primary barrier in preventing the release of fission products into the atmosphere. Establishing an upper limit on reactor steam dome pressure ensures continued RCS integrity. According to 10 CFR 50, Appendix A, GDC 14, "Reactor Coolant Pressure Boundary," and GDC 15, "Reactor Coolant System Design" (Ref. 1), the reactor coolant pressure boundary (RCPB) shall be designed with sufficient margin to ensure that the design conditions are not exceeded during normal operation and anticipated operational occurrences (AOOs).

During normal operation and AOOs, RCS pressure is limited from exceeding the design pressure by more than 10%, in accordance with Section III of the ASME Code (Ref. 2). To ensure system integrity, all RCS components are hydrostatically tested at 125% of design pressure, in accordance with ASME Code requirements, prior to initial operation when there is no fuel in the core. Any further hydrostatic testing with fuel in the core may be done under LCO 3.10.1, "Inservice Leak and Hydrostatic Testing Operation." Following inception of unit operation, RCS components shall be pressure tested in accordance with the requirements of ASME Code, Section XI (Ref. 3).

Overpressurization of the RCS could result in a breach of the RCPB, reducing the number of protective barriers designed to prevent radioactive releases from exceeding the limits specified in 10 CFR ~~100~~ 50.67 ~~"Reactor Site Criteria"~~ (Ref. 4). If this occurred in conjunction with a fuel cladding failure, the number of protective barriers designed to prevent radioactive releases from exceeding the limits would be reduced.

APPLICABLE SAFETY ANALYSES

The RCS safety/relief valves and the Reactor Protection System Reactor Vessel Steam Dome Pressure—High Function have settings established to ensure that the RCS pressure SL will not be exceeded.

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

The RCS pressure SL has been selected such that it is at a pressure below which it can be shown that the integrity of the system is not endangered. The reactor pressure vessel is designed to ASME, Boiler and Pressure Vessel Code, Section III, 1971 Edition, including Addenda through the winter of 1972 (Ref. 5), which permits a maximum pressure transient of 110%, 1375 psig, of design pressure 1250 psig. The SL of 1325 psig, as measured in the reactor steam dome, is equivalent to 1375 psig at the lowest elevation of the RCS. The RCS is designed to ASME Code, Section III, 1977 Edition, including Addenda through the summer of 1977 (Ref. 6), for the reactor recirculation piping, which permits a maximum pressure transient of 110% of design pressures of 1250 psig for suction piping up to the reactor recirculation pump, 1650 psig for discharge piping up to and including the discharge blocking valve, and 1550 psig for the piping after the discharge blocking valve. The RCS pressure SL is selected to be the lowest transient overpressure allowed by the applicable codes.

SAFETY LIMITS

The maximum transient pressure allowable in the RCS pressure vessel under the ASME Code, Section III, is 110% of design pressure. The maximum transient pressure allowable in the RCS piping, valves, and fittings is 110% of design pressures of 1250 psig for suction piping up to the reactor recirculation pump, 1650 psig for discharge piping up to and including the discharge blocking valve, and 1550 psig for the piping after the discharge blocking valve. The most limiting of these allowances is the 110% of the reactor vessel and the suction piping up to the reactor recirculation pump design pressures; therefore, the SL on maximum allowable RCS pressure is established at 1325 psig as measured at the reactor steam dome.

APPLICABILITY

SL 2.1.2 applies in all MODES.

SAFETY LIMIT VIOLATIONS

2.2

Exceeding the RCS pressure SL may cause RCS failure and create a potential for radioactive releases in excess of 10 CFR ~~100, "Reactor Site Criteria,"~~ 50.67 limits (Ref. 4). Therefore, it is required to insert all insertable control

(continued)

BASES

SAFETY LIMIT VIOLATIONS

2.2 (continued)

rods and restore compliance with the SL within 2 hours. The 2 hour Completion Time ensures that the operators take prompt remedial action and also assures that the probability of an accident occurring during this period is minimal.

REFERENCES

1. 10 CFR 50, Appendix A, GDC 14 and GDC 15.

2. ASME, Boiler and Pressure Vessel Code, Section III, Article NB-7000.

3. ASME, Boiler and Pressure Vessel Code, Section XI, Article IWA-5000.

4. 10 CFR ~~100~~ 50.67, "Accident Source Term."

5. ASME, Boiler and Pressure Vessel Code, Section III, 1971 Edition, Addenda, winter of 1972.

6. ASME, Boiler and Pressure Vessel Code, Section III, 1977 Edition, Addenda, summer of 1977.

BASES

REFERENCES (continued)

5. ~~NUREG-0800, "Standard Review Plan," Section 15.4.9, "Radiological Consequences of Control Rod Drop Accident (BWR)," Revision 2, July 1981.~~ Regulatory Guide 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors," July 2000

6. 10 CFR ~~100.11, "Determination of Exclusion Area Low Population Zone and Population Center Distance."~~ 50.67, "Accident Source Term."

7. NEDO-10527, "Rod Drop Accident Analysis for Large BWRs," (including Supplements 1 and 2), March 1972.

8. NEDO-21778-A, "Transient Pressure Rises Affected Fracture Toughness Requirements for Boiling Water Reactors," December 1978.

9. ASME, Boiler and Pressure Vessel Code.

10. NEDO-21231, "Banked Position Withdrawal Sequence," January 1977.

11. 10 CFR 50.36(c)(2)(ii).

B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.7 Standby Liquid Control (SLC) System

BASES

BACKGROUND

The SLC System is designed to provide the capability of bringing the reactor, at any time in a fuel cycle, from full power and minimum control rod inventory (which is at the peak of the xenon transient) to a subcritical condition with the reactor in the most reactive xenon free state without taking credit for control rod movement. The SLC System satisfies the requirements of 10 CFR 50.62 (Ref. 1) on anticipated transient without scram (ATWS).

Insert A

The SLC System consists of a boron solution storage tank, two positive displacement pumps, two explosive valves, which are provided in parallel for redundancy, and associated piping and valves used to transfer borated water from the storage tank to the reactor pressure vessel (RPV). The borated solution is discharged through the high pressure core spray system sparger.

APPLICABLE SAFETY ANALYSES

The SLC System is manually initiated from the main control room, as directed by the emergency operating procedures, if the operator believes the reactor cannot be shut down, or kept shut down, with the control rods. The SLC System can also be automatically initiated as required by Reference 1; however, this is not necessary for SLC System OPERABILITY. The SLC System is used in the event that not enough control rods can be inserted to accomplish shutdown and cooldown in the normal manner. The SLC System injects borated water into the reactor core to compensate for all of the various reactivity effects that could occur during plant operation. To meet this objective, it is necessary to inject, using both SLC pumps, a quantity of boron that produces a concentration equivalent to 780 ppm of natural boron in the reactor core, including recirculation loops, at 68°F and reactor water level at level 8. To allow for potential leakage and imperfect mixing in the reactor system, an additional amount of boron equal to 25% of the amount cited above is added (Ref. 2). An additional amount is provided to accommodate dilution in the RPV by the residual heat removal shutdown cooling piping. The volume versus concentration limits in Figure 3.1.7-1 are calculated such that the required concentration is achieved. This quantity of borated solution is the amount that is above the pump suction

(continued)

INSERT A (for TS Bases page B 3.1.7-1)

The SLC System is also used to maintain the suppression pool pH at or above 7.0 following a design basis loss of coolant accident (LOCA) involving significant fuel damage. Maintaining the bulk suppression pool pH above 7.0 following an accident ensures that iodine will be retained in the suppression pool water (Ref. 4), as assumed in the Alternative Source Term analysis methodology.

BASES

APPLICABLE SAFETY ANALYSES (continued)

shutoff level in the boron solution storage tank. No credit is taken for the portion of the tank volume that cannot be injected.

Insert B

The SLC System satisfies Criter~~ion~~a ~~4~~ 3 and 4 of Reference 3.

LCO

The OPERABILITY of the SLC System provides backup capability for reactivity control, independent of normal reactivity control provisions provided by the control rods. Insert C The OPERABILITY of the SLC System is based on the conditions of the borated solution in the storage tank and the availability of a flow path to the RPV, including the OPERABILITY of the pumps and valves. Two SLC subsystems are required to be OPERABLE, each containing an OPERABLE pump, an explosive valve and associated piping, valves, and instruments and controls to ensure an OPERABLE flow path.

APPLICABILITY

In MODES 1 and 2, shutdown capability is required. In MODES 3 and 4, control rods are not able to be withdrawn since the reactor mode switch is in shutdown and a control rod block is applied. This provides adequate controls to ensure the reactor remains subcritical. In MODE 5, only a single control rod can be withdrawn from a core cell containing fuel assemblies. Demonstration of adequate SDM (LCO 3.1.1, "SHUTDOWN MARGIN (SDM)") ensures that the reactor will not become critical. Therefore, the SLC System is not required to ~~be OPERABLE during these conditions, when only a single control rod can be withdrawn.~~ perform its ATWS function during MODES 3, 4, or 5.

Insert D

ACTIONS

A.1

If one SLC System subsystem is inoperable, the inoperable subsystem must be restored to OPERABLE status within 7 days. In this condition, the remaining OPERABLE subsystem is adequate to shutdown the unit. However, the overall capability is reduced since the remaining OPERABLE subsystem cannot meet the requirements of Reference 1. The 7 day Completion Time is based on the availability of an OPERABLE subsystem capable of shutting down the unit and the low probability of a Design Basis Accident (DBA) or severe transient occurring concurrent with the failure of the Control Rod Drive System to shut down the plant.

(continued)

INSERT B (for TS Bases page B 3.1.7-2)

Following a LOCA, the radiological consequences from the accident will remain within the limits of 10 CFR 50.67 (Ref. 5) provided sufficient iodine activity is retained in the suppression pool water. Credit for iodine retention in the suppression pool is allowed (Ref. 4) as long as the bulk suppression pool pH is maintained at or above 7.0. The Alternative Source Term analysis methodology credits the use of the SLC System for injecting the sodium pentaborate solution into the reactor pressure vessel following a LOCA to maintain the pH of the suppression pool water at or above 7.0.

INSERT C (for TS Bases page B 3.1.7-2)

Additionally, an OPERABLE SLC System has the ability to inject borated solution under post-LOCA conditions to maintain the bulk suppression pool pH at or above 7.0.

INSERT D (for TS Bases page B 3.1.7-2)

In MODES 1, 2, and 3, the SLC System must be OPERABLE to ensure that the radiological consequences of a LOCA involving significant fuel damage remain within the limits of 10 CFR 50.67 (Ref. 5). The SLC System is used to maintain the bulk suppression pool pH at or above 7.0 following a LOCA to ensure that iodine will be retained in the suppression pool water (Ref. 4), as assumed in the Alternative Source Term analyses.

BASES

ACTIONS (continued)

B.1

If both SLC subsystems are inoperable, at least one subsystem must be restored to OPERABLE status within 8 hours. The allowed Completion Time of 8 hours is considered acceptable, given the low probability of a DBA or transient occurring concurrent with the failure of the control rods to shut down the reactor.

C.1 and C.2 Add

and to MODE 4 within 36 hours

If any Required Action and associated Completion Time is not met, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 12 hours. The allowed Completion Times are ~~Time of 12 hours is~~ reasonable, based on operating experience, to reach ~~MODE 3~~ the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.1.7.1, SR 3.1.7.2, and SR 3.1.7.3

SR 3.1.7.1 through SR 3.1.7.3 are 24 hour Surveillances, verifying certain characteristics of the SLC System (e.g., the volume and temperature of the borated solution in the storage tank), thereby ensuring the SLC System OPERABILITY without disturbing normal plant operation. These Surveillances ensure the proper borated solution and temperature, including the temperature of the pump suction piping, are maintained. Maintaining a minimum specified borated solution temperature is important in ensuring that the boron remains in solution and does not precipitate out in the storage tank or in the pump suction piping. The 24 hour Frequency of these SRs is based on operating experience that has shown there are relatively slow variations in the measured parameters of volume and temperature.

SR 3.1.7.4 and SR 3.1.7.6

SR 3.1.7.4 verifies the continuity of the explosive charges in the injection valves to ensure proper operation will occur if required. Other administrative controls, such as those that limit the shelf life of the explosive charges,

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.1.7.8 and SR 3.1.7.9 (continued)

path for injecting the sodium pentaborate solution. An acceptable method for verifying that the suction piping up to the suction valve is unblocked is to pump from the storage tank to the test tank. Upon completion of this verification, the pump suction piping between the pump suction valve and pump suction must be drained and flushed with demineralized water, since this piping is not heat traced. The 24 month Frequency is acceptable since there is a low probability that the subject piping will be blocked due to precipitation of the boron from solution in the heat traced piping. This is especially true in light of the daily temperature verification of this piping required by SR 3.1.7.3. However, if, in performing SR 3.1.7.3, it is determined that the temperature of this piping has fallen below the specified minimum, SR 3.1.7.9 must be performed once within 24 hours after the piping temperature is restored within the limits of SR 3.1.7.3.

SR 3.1.7.10

Enriched sodium pentaborate solution is made by mixing granular, enriched sodium pentaborate with water. Isotopic tests on the granular sodium pentaborate to verify the actual B-10 enrichment must be performed prior to addition to the SLC tank in order to ensure that the proper B-10 atom percentage is being used.

REFERENCES

1. 10 CFR 50.62.
2. USAR, Section 9.3.5.3.
3. 10 CFR 50.36(c)(2)(ii).
4. NUREG-1465, "Accident Source Terms for Light-Water Nuclear Power Plants," USNRC, February 1995.
5. 10 CFR 50.67, "Accident Source Term."

B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.8 Scram Discharge Volume (SDV) Vent and Drain Valves

BASES

BACKGROUND

The SDV vent and drain valves are normally open and discharge any accumulated water in the SDV to ensure that sufficient volume is available at all times to allow a complete scram. During a scram, the SDV vent and drain valves close to contain reactor water. The SDV consists of header piping that connects to each hydraulic control unit (HCU) and drains into an instrument volume. There are two headers and two instrument volumes, each receiving approximately one half of the control rod drive (CRD) discharges. The two instrument volumes are connected to a common drain line with two valves in series. Each header is connected to a common vent line with two valves in series. The header piping is sized to receive and contain all the water discharged by the CRDs during a scram. The design and functions of the SDV are described in Reference 1.

APPLICABLE SAFETY ANALYSES

The Design Basis Accident and transient analyses assume all the control rods are capable of scramming. The primary function of the SDV is to limit the amount of reactor coolant discharged during a scram. The acceptance criteria for the SDV vent and drain valves are that they operate automatically to:

a. Close during scram to limit the amount of reactor coolant discharged so that adequate core cooling is maintained and offsite doses remain within the limits of 10 CFR ~~100~~ 50.67 (Ref. 2); and

b. Open on scram reset to maintain the SDV vent and drain path open so there is sufficient volume to accept the reactor coolant discharged during a scram.

Isolation of the SDV can also be accomplished by manual closure of the SDV valves. Additionally, the discharge of reactor coolant to the SDV can be terminated by scram reset or closure of the HCU manual isolation valves. For a bounding leakage case, the offsite doses are ~~well~~ within the limits of 10 CFR ~~100~~ 50.67 (Ref. 2) and adequate core cooling is maintained (Ref. 3). The SDV vent and drain valves also allow continuous drainage of the SDV during normal plant

(continued)

BASES

SURVEILLANCE REQUIREMENTS	SR 3.1.8.3 (continued) the reactor at power. Operating experience has shown these components usually pass the Surveillance when performed at the 24 month Frequency; therefore, the Frequency was concluded to be acceptable from a reliability standpoint.
REFERENCES	1. USAR, Section 4.6.1.1.2. 2. 10 CFR ~~100~~ 50.67, "Accident Source Term." 3. NUREG-0803, "Generic Safety Evaluation Report Regarding Integrity of BWR Scram System Piping," August 1981. 4. 10 CFR 50.36(c)(2)(ii).

B 3.2 POWER DISTRIBUTION LIMITS

B 3.2.3 LINEAR HEAT GENERATION RATE (LHGR)

BASES

BACKGROUND

The LHGR is a measure of the heat generation rate of a fuel rod in a fuel assembly at any axial location. Limits on the LHGR are specified to ensure that fuel design limits are not exceeded anywhere in the core during normal operation, including anticipated operational occurrences (AOOs). Exceeding the LHGR limit could potentially result in fuel damage and subsequent release of radioactive materials. Fuel design limits are specified to ensure that fuel system damage, fuel rod failure or inability to cool the fuel does not occur during the anticipated operating conditions identified in Reference 1.

APPLICABLE SAFETY ANALYSES

The analytical methods and assumptions used in evaluating the fuel system design are presented in References 1 and 2. The fuel assembly is designed to ensure (in conjunction with the core nuclear and thermal hydraulic design, plant equipment, instrumentation, and protection system) that fuel damage will not result in the release of radioactive materials in excess of the guidelines of 10 CFR, Parts 20, 50, and ~~100~~ 50.67. The mechanisms that could cause fuel damage during operational transients and that are considered in fuel evaluations are:

a. Rupture of the fuel rod cladding caused by strain from the relative expansion of the UO_2 pellet; and

b. Severe overheating of the fuel rod cladding caused by inadequate cooling.

A value of 1% plastic strain of the fuel cladding has been defined as the limit below which fuel damage caused by overstraining of the fuel cladding is not expected to occur (Ref. 3).

Fuel design evaluations have been performed and demonstrate that the 1% fuel cladding plastic strain design limit is not exceeded during continuous operation with LHGRs up to the operating limit specified in the COLR. The analysis also includes allowances for short term transient operation above the operating limit to account for AOOs, plus an allowance for densification power spiking.

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

1.a. Reactor Vessel Water Level—Low Low Low, Level 1 (continued)

Therefore, isolation of the MSIVs and other interfaces with the reactor vessel occurs to prevent offsite dose limits from being exceeded. The Reactor Vessel Water Level—Low Low Low, Level 1 Function is one of the many Functions assumed to be OPERABLE and capable of providing isolation signals. The Reactor Vessel Water Level—Low Low Low, Level 1 Function associated with isolation is assumed in the analysis of the recirculation line break (Ref. 2). The isolation of the MSL on Level 1 supports actions to ensure that offsite dose limits are not exceeded for a DBA.

Reactor vessel water level signals are initiated from four differential pressure transmitters that sense the difference between the pressure due to a constant column of water (reference leg) and the pressure due to the actual water level (variable leg) in the vessel. Four channels of Reactor Vessel Water Level—Low Low Low, Level 1 Function are available and are required to be OPERABLE to ensure that no single instrument failure can preclude the isolation function.

The Reactor Vessel Water Level—Low Low Low, Level 1 Allowable Value is chosen to be the same as the ECCS Level 1 Allowable Value (LCO 3.3.5.1) to ensure that the MSLs isolate on a potential loss of coolant accident (LOCA) to prevent offsite doses from exceeding 10 CFR ~~100~~ 50.67 limits.

This Function isolates the Group 1 valves.

1.b. Main Steam Line Pressure—Low

Low MSL pressure indicates that there may be a problem with the turbine pressure regulation, which could result in a low reactor vessel water level condition and the RPV cooling down more than 100°F/hour if the pressure loss is allowed to continue. The Main Steam Line Pressure—Low Function is directly assumed in the analysis of the pressure regulator failure (Ref. 5). For this event, the closure of the MSIVs ensures that the RPV temperature change limit (100°F/hour) is not reached. In addition, this Function supports actions to ensure that Safety Limit 2.1.1.1 is not exceeded. (This

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

1.b. Main Steam Line Pressure—Low (continued)

Function closes the MSIVs prior to pressure decreasing below 766 psig, which results in a scram due to MSIV closure, thus reducing reactor power to < 25% RTP.)

The MSL low pressure signals are initiated from four pressure transmitters that are connected to the MSL header. The transmitters are arranged such that, even though physically separated from each other, each transmitter is able to detect low MSL pressure. Four channels of Main Steam Line Pressure—Low Function are available and are required to be OPERABLE to ensure that no single instrument failure can preclude the isolation function.

The Allowable Value was selected to be high enough to prevent excessive RPV depressurization.

The Main Steam Line Pressure—Low Function is only required to be OPERABLE in MODE 1 since this is when the assumed transient can occur (Ref. 4).

This Function isolates the Group 1 valves.

1.c. Main Steam Line Flow—High

Main Steam Line Flow—High is provided to detect a break of the MSL and to initiate closure of the MSIVs. If the steam were allowed to continue flowing out of the break, the reactor would depressurize and the core could uncover. If the RPV water level decreases too far, fuel damage could occur. Therefore, the isolation is initiated on high flow to prevent or minimize core damage. The Main Steam Line Flow—High Function is directly assumed in the analysis of the main steam line break (MSLB) accident (Ref. 6). The isolation action, along with the scram function of the RPS, ensures that the fuel peak cladding temperature remains below the limits of 10 CFR 50.46 and offsite doses do not exceed the 10 CFR ~~100~~ 50.67 limits.

The MSL flow signals are initiated from 16 differential pressure transmitters that are connected to the four MSLs (the differential pressure transmitters sense differential pressure across a flow venturi). The transmitters are arranged such that, even though physically separated from

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

1.h Manual Initiation (continued)

Initiation Function are available and are required to be OPERABLE in MODES 1, 2, and 3, since these are the MODES in which the MSL Isolation automatic Functions are required to be OPERABLE.

There is no Allowable Value for this Function since the channels are mechanically actuated based solely on the position of the switch and push buttons.

This Function isolates the Group 1 valves.

2. Primary Containment Isolation

2.a. Reactor Vessel Water Level—Low Low, Level 2

Low RPV water level indicates the capability to cool the fuel may be threatened. The valves whose penetrations communicate with the primary containment are isolated to limit the release of fission products. The isolation of the primary containment on Level 2 supports actions to ensure that offsite dose limits of 10 CFR ~~100~~ 50.67 are not exceeded. The Reactor Vessel Water Level—Low Low, Level 2 Function associated with isolation is implicitly assumed in the USAR analysis as these leakage paths are assumed to be isolated post LOCA.

Reactor Vessel Water Level—Low Low, Level 2 signals are initiated from differential pressure transmitters that sense the difference between the pressure due to a constant column of water (reference leg) and the pressure due to the actual water level (variable leg) in the vessel. Four channels of Reactor Vessel Water Level—Low Low, Level 2 Function are available and are required to be OPERABLE to ensure no single instrument failure can preclude the isolation function.

The Reactor Vessel Water Level—Low Low, Level 2 Allowable Value was chosen to be the same as the ECCS Reactor Vessel Water Level—Low Low, Level 2 Allowable Value (LCO 3.3.5.1), since isolation of these valves is not critical to orderly plant shutdown.

This Function isolates the Group 2, 3, 8, and 9 valves.

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

2.b. Drywell Pressure—High

High drywell pressure can indicate a break in the RCPB inside the drywell. The isolation of some of the PCIVs on high drywell pressure supports actions to ensure that offsite dose limits of 10 CFR ~~100~~ 50.67 are not exceeded. The Drywell Pressure—High Function associated with isolation of the primary containment is implicitly assumed in the USAR accident analysis as these leakage paths are assumed to be isolated post LOCA.

High drywell pressure signals are initiated from pressure transmitters that sense the pressure in the drywell. Four channels of Drywell Pressure—High Function are available and are required to be OPERABLE to ensure that no single instrument failure can preclude the isolation function.

The Allowable Value was selected to be the same as the RPS Drywell Pressure—High Allowable Value (LCO 3.3.1.1), since this may be indicative of a LOCA inside primary containment.

This Function isolates the Group 3, 8, and 9 valves.

2.c. Standby Gas Treatment (SGT) System Exhaust Radiation—High

High ventilation exhaust radiation is an indication of possible gross failure of the fuel cladding. The release may have originated from the primary containment due to a break in the RCPB. When Exhaust Radiation—High is detected, valves whose penetrations communicate with the primary containment atmosphere are isolated to limit the release of fission products.

The Exhaust Radiation—High signals are initiated from a radiation detector that is located on the ventilation exhaust piping of the SGT System. The signal from the detector is input to an individual monitor whose trip output, after a preselected time delay, is assigned to both isolation channels. Two channels of SGT Exhaust—High Function are available and are required to be OPERABLE to ensure that no single instrument failure, other than the sensor/trip output, can preclude the isolation function.

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

2.c. Standby Gas Treatment (SGT) System Exhaust Radiation—High (continued)

The Allowable Values are chosen to promptly detect gross failure of the fuel cladding and to ensure offsite doses remain below 10 CFR 20 and 10 CFR ~~100~~ 50.67 limits.

This Function isolates the Group 9 valves.

2.d. Manual Initiation

The Manual Initiation switch and push button channels introduce signals into the primary containment isolation logic that are redundant to the automatic protective instrumentation and provide manual isolation capability. There is no specific USAR safety analysis that takes credit for this Function. It is retained for overall redundancy and diversity of the isolation function as required by the NRC in the plant licensing basis.

There are four switch and push buttons (with two channels per switch and push button) for the logic, with two switch and push buttons per trip system. Eight channels of the Manual Initiation Function are available and are required to be OPERABLE in MODES 1, 2, and 3, since these are the MODES in which the Primary Containment Isolation automatic Functions are required to be OPERABLE.

There is no Allowable Value for this Function since the channels are mechanically actuated based solely on the position of the switch and push buttons.

This Function isolates the Group 2, 3, 8, and 9 valves.

3. Reactor Core Isolation Cooling System Isolation

3.a. RCIC Steam Line Flow—High

RCIC Steam Line Flow—High Function is provided to detect a break of the RCIC steam lines and initiates closure of the steam line isolation valves. If the steam is allowed to continue flowing out of the break, the reactor will depressurize and core uncovery can occur. Therefore, the isolation is initiated on high flow to prevent or minimize

(continued)

B 3.3 INSTRUMENTATION

B 3.3.6.2 Secondary Containment Isolation Instrumentation

BASES

BACKGROUND

The secondary containment isolation instrumentation automatically initiates closure of appropriate secondary containment isolation valves (SCIVs) and starts the Standby Gas Treatment (SGT) System. The function of these systems, in combination with other accident mitigation systems, is to limit fission product release during and following postulated Design Basis Accidents (DBAs) (Ref. 1), such that offsite radiation exposures are maintained within the requirements of 10 CFR ~~100~~ 50.67 that are part of the NRC staff approved licensing basis. Secondary containment isolation and establishment of vacuum with the SGT System within the assumed time limits ensures that fission products that are released during certain operations that take place inside primary containment when primary containment is not required to be OPERABLE, or that take place outside primary containment, are maintained within applicable limits.

The isolation instrumentation includes the sensors, relays, and switches that are necessary to cause initiation of secondary containment isolation. Most channels include electronic equipment (e.g., trip units) that compares measured input signals with pre-established setpoints. When the setpoint is exceeded, the channel output relay actuates, which then outputs a secondary containment isolation signal to the isolation logic. Functional diversity is provided by monitoring a wide range of independent parameters. The input parameters to the isolation logic are (a) reactor vessel water level, (b) drywell pressure, (c) reactor building above the refuel floor exhaust radiation, and (d) reactor building below the refuel floor exhaust radiation. Redundant sensor input signals from each parameter are provided for initiation of isolation parameters. In addition, manual initiation of the logic, while not required to be OPERABLE by this Specification, is also provided.

For both the Reactor Vessel Water Level—Low Low, Level 2 and Drywell Pressure—High Functions, the secondary containment isolation instrumentation logic receives input from four channels. The output from these channels are arranged into two two-out-of-two trip systems. For both the Reactor Building Above the Refuel Floor Exhaust Radiation—

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

3.4. Reactor Building Above the Refuel Floor and Reactor Building Below the Refuel Floor Exhaust Radiation – High
(continued)

Reactor Building Above the Refuel Floor Exhaust Radiation – High signals are initiated from gaseous radiation detectors that are located on the ventilation exhaust ducting coming from the refuel floor. Reactor Building Below the Refuel Floor Exhaust Radiation – High signals are initiated from gaseous radiation detectors that are located on the ventilation exhaust ducting coming from the different areas of the secondary containment below the refuel floor. The signal from each detector is input to an individual monitor whose trip outputs are assigned to an isolation channel. Two channels of Reactor Building Above the Refuel Floor Exhaust Radiation – High Function and two channels of Reactor Building Below the Refuel Floor Exhaust Radiation – High Function are available and are required to be OPERABLE to ensure that no single instrument failure can preclude the isolation function.

The Allowable Values are chosen to promptly detect gross failure of the fuel cladding.

The Exhaust Radiation – High Functions are required to be OPERABLE in MODES 1, 2, and 3 where considerable energy exists; thus, there is a probability of pipe breaks resulting in significant releases of radioactive steam and gas. In MODES 4 and 5, the probability and consequences of these events are low due to the RCS pressure and temperature limitations of these MODES; thus, these Functions are not required. In addition, the Functions are required to be OPERABLE during OPDRVs and movement of recently irradiated fuel assemblies in the secondary containment because the capability of detecting radiation releases due to fuel failures (due to fuel uncovery or dropped fuel assemblies) must be provided to ensure that offsite dose limits are not exceeded. Due to radioactive decay, this Function is only required to isolate secondary containment during fuel handling accidents involving handling recently irradiated fuel (i.e., fuel that has occupied part of a critical reactor core within the previous ~~2 days~~ 24 hours).

ACTIONS

A Note has been provided to modify the ACTIONS related to secondary containment isolation instrumentation channels. Section 1.3, Completion Times, specifies that once a Condition has been entered, subsequent divisions, subsystems, components, or variables expressed in the Condition discovered to be inoperable or not within limits

(continued)

BASES (continued)

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

The ability of the CREF System to maintain the habitability of the control room envelope is explicitly assumed for certain accidents as discussed in the USAR safety analyses (Refs. 2 and 3). CREF System operation ensures that the radiation exposure of control room personnel, through the duration of any one of the postulated accidents, does not exceed the limits set by ~~GDC 19 of 10 CFR 50, Appendix A~~ 10 CFR 50.67.

CREF System instrumentation satisfies Criterion 3 of Reference 4.

The OPERABILITY of the CREF System instrumentation is dependent upon the OPERABILITY of the individual instrumentation channel Functions specified in Table 3.3.7.1-1. Each Function must have a required number of OPERABLE channels, with their setpoints within the specified Allowable Values, where appropriate. The actual setpoint is calibrated consistent with applicable setpoint methodology assumptions.

Allowable Values are specified for each CREF System Function specified in the Table. Nominal trip setpoints are specified in the setpoint calculations. These nominal setpoints are selected to ensure that the setpoints do not exceed the Allowable Value between successive CHANNEL CALIBRATIONS. Operation with a trip setpoint that is less conservative than the nominal trip setpoint, but within its Allowable Value, is acceptable. A channel is inoperable if its actual trip setpoint is not within its required Allowable Value.

Trip setpoints are those predetermined values of output at which an action should take place. The setpoints are compared to the actual process parameter (e.g., reactor vessel water level), and when the measured output value of the process parameter exceeds the setpoint, the associated device (e.g., trip unit) changes state. The analytic limits are derived from the limiting values of the process parameters obtained from the safety analysis. The Allowable Values are derived from the analytic limits by accounting for calibration uncertainty, process measurement uncertainty, primary element uncertainty, instrument uncertainty, and applicable environmental effects. The trip setpoints are derived from the analytical limits by accounting for calibration uncertainty, process measurement uncertainty, primary element uncertainty, instrument uncertainty, applicable environmental effects, and drift.

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

2. Drywell Pressure—High

High pressure in the drywell could indicate a break in the reactor coolant pressure boundary (RCPB). A high drywell pressure signal could indicate a LOCA and will automatically initiate the CREF System, since this could be a precursor to a potential radiation release and subsequent radiation exposure to control room personnel.

Drywell Pressure—High signals are initiated from four pressure transmitters that sense drywell pressure. Four channels of Drywell Pressure—High Function are available (two channels per trip system) and are required to be OPERABLE to ensure that no single instrument failure can preclude CREF System initiation.

The Drywell Pressure—High Allowable Value was chosen to be the same as the Secondary Containment Isolation Drywell Pressure—High Allowable Value (LCO 3.3.6.2).

The Drywell Pressure—High Function is required to be OPERABLE in MODES 1, 2, and 3 to ensure that control room personnel are protected during a LOCA. In MODES 4 and 5, the Drywell Pressure—High Function is not required since there is insufficient energy in the reactor to pressurize the drywell to the Drywell Pressure—High setpoint.

3. Main Control Room Ventilation Radiation Monitor—High

involving handling recently irradiated fuel (i.e., fuel that has occupied part of a critical reactor core within the previous 24 hours)

High radiation within the common intake duct of the main control room outside air intakes is an indication of possible gross failure of the fuel cladding. The release may have originated from the primary containment due to a break in the RCPB or the refueling floor due to a fuel handling accident. When main control room ventilation high radiation is detected (above measured background), the CREF System is automatically initiated in the emergency pressurization mode since this radiation release could result in radiation exposure to control room personnel.

The Main Control Room Ventilation Radiation Monitor—High Function consists of four independent monitors. Four channels of Main Control Room Ventilation Radiation Monitor—High Function are available and are required to be

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

3. Main Control Room Ventilation Radiation Monitor—High (continued)

OPERABLE to ensure that no single instrument failure can preclude CREF System initiation. The Allowable Value was selected to ensure protection of the control room personnel.

The Main Control Room Ventilation Radiation Monitor—High Function is required to be OPERABLE in MODES 1, 2, and 3, and during ~~CORE ALTERATIONS,~~ OPDRVs~~,~~ and movement of recently irradiated fuel in the secondary containment to ensure that control room personnel are protected during a LOCA, fuel handling event, or a vessel draindown event. During MODES 4 and 5, when these specified conditions are not in progress (e.g., OPDRVs ~~CORE ALTERATIONS~~), the probability of a LOCA ~~or fuel damage~~ is low; thus, the Function is not required. Insert E

ACTIONS

A Note has been provided to modify the ACTIONS related to CREF System instrumentation channels. Section 1.3, Completion Times, specifies that once a Condition has been entered, subsequent divisions, subsystems, components, or variables expressed in the Condition discovered to be inoperable or not within limits will not result in separate entry into the Condition. Section 1.3 also specifies that Required Actions of the Condition continue to apply for each additional failure, with Completion Times based on initial entry into the Condition. However, the Required Actions for inoperable CREF System instrumentation channels provide appropriate compensatory measures for separate inoperable channels. As such, a Note has been provided that allows separate Condition entry for each inoperable CREF System instrumentation channel.

A.1

Required Action A.1 directs entry into the appropriate Condition referenced in Table 3.3.7.1-1. The applicable Condition specified in the Table is Function dependent. Each time an inoperable channel is discovered, Condition A is entered for that channel and provides for transfer to the appropriate subsequent Condition.

(continued)

INSERT E (for TS Bases page B 3.3.7.1-5)

Also, due to radioactive decay, this Function is only required to initiate the CREF System during fuel handling accidents involving handling recently irradiated fuel (i.e., fuel that has occupied part of a critical reactor core within the previous 24 hours).

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.8 RCS Specific Activity

BASES

BACKGROUND

During circulation, the reactor coolant acquires radioactive materials due to release of fission products from fuel leaks into the coolant and activation of corrosion products in the reactor coolant. These radioactive materials in the coolant can plate out in the RCS, and, at times, an accumulation will break away to spike the normal level of radioactivity. The release of coolant during a Design Basis Accident (DBA) could send radioactive materials into the environment.

Limits on the maximum allowable level of radioactivity in the reactor coolant are established to ensure, in the event of a release of any radioactive material to the environment during a DBA, radiation doses are maintained within the limits of 10 CFR ~~100~~ 50.67 (Ref. 1).

This LCO contains iodine specific activity limits. The iodine isotopic activities per gram of reactor coolant are expressed in terms of a DOSE EQUIVALENT I-131. The allowable levels are intended to limit the 2 hour radiation dose to an individual at the site boundary to ~~a small fraction~~ 10% of the 10 CFR ~~100~~ 50.67 limit.

APPLICABLE SAFETY ANALYSES

Analytical methods and assumptions involving radioactive material in the primary coolant are presented in the USAR (Ref. 2). The specific activity in the reactor coolant (the source term) is an initial condition for evaluation of the consequences of an accident due to a main steam line break (MSLB) outside containment. No fuel damage is postulated in the MSLB accident, and the release of radioactive material to the environment is assumed to end when the main steam isolation valves (MSIVs) close completely.

This MSLB release forms the basis for determining offsite doses (Ref. 2). The limits on the specific activity of the primary coolant ensure that the 2 hour ~~thyroid and whole body doses~~ TEDE dose at the site boundary, resulting from an MSLB outside containment during steady state operation, will not exceed 10% of the dose ~~guidelines~~ limits of 10 CFR ~~100~~ 50.67.

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

The limit on specific activity is a value from a parametric evaluation of typical site locations. This limit is conservative because the evaluation considered more restrictive parameters than for a specific site, such as the location of the site boundary and the meteorological conditions of the site.

RCS specific activity satisfies Criterion 2 of Reference 3.

LCO

The specific iodine activity is limited to ≤ 0.2 μCi/gm DOSE EQUIVALENT I-131. This limit ensures the source term assumed in the safety analysis for the MSLB is not exceeded, so any release of radioactivity to the environment during an MSLB is less than a small fraction of the 10 CFR ~~100~~ 50.67 limits.

APPLICABILITY

In MODE 1, and MODES 2 and 3 with any main steam line not isolated, limits on the primary coolant radioactivity are applicable since there is an escape path for release of radioactive material from the primary coolant to the environment in the event of an MSLB outside of primary containment.

In MODES 2 and 3 with the main steam lines isolated, such limits do not apply since an escape path does not exist. In MODES 4 and 5, no limits are required since the reactor is not pressurized and the potential for leakage is reduced.

ACTIONS

A.1 and A.2

When the reactor coolant specific activity exceeds the LCO DOSE EQUIVALENT I-131 limit, but is ≤ 4.0 μCi/gm, samples must be analyzed for DOSE EQUIVALENT I-131 at least once every 4 hours. In addition, the specific activity must be restored to the LCO limit within 48 hours. The Completion Time of once every 4 hours is based on the time needed to take and analyze a sample. The 48 hour Completion Time to restore the activity level provides a reasonable time for temporary coolant activity increases (iodine spikes or crud bursts) to be cleaned up with the normal processing systems.

A Note permits the use of the provisions of LCO 3.0.4.c. This allowance permits entry into the applicable MODE(S) while relying on the ACTIONS.

(continued)

BASES

ACTIONS

A.1 and A.2 (continued)

This allowance is acceptable due to the significant conservatism incorporated into the specific activity limit, the low probability of an event which is limiting due to exceeding this limit, and the ability to restore transient specific activity excursions while the plant remains at, or proceeds to power operation.

B.1, B.2.1, B.2.2.1, and B.2.2.2

If the DOSE EQUIVALENT I-131 cannot be restored to ≤ 0.2 μCi/gm within 48 hours, or if at any time it is > 4.0 μCi/gm, it must be determined at least every 4 hours and all the main steam lines must be isolated within 12 hours. Isolating the main steam lines precludes the possibility of releasing radioactive material to the environment in an amount that is more than ~~a small fraction of~~ the requirements of 10 CFR ~~100~~ 50.67 during a postulated MSLB accident.

Alternately, the plant can be brought to MODE 3 within 12 hours and to MODE 4 within 36 hours. This option is provided for those instances when isolation of main steam lines is not desired (e.g., due to the decay heat loads). In MODE 4, the requirements of the LCO are no longer applicable.

The Completion Time of once every 4 hours is the time needed to take and analyze a sample. The 12 hour Completion Time is reasonable, based on operating experience, to isolate the main steam lines in an orderly manner and without challenging plant systems. Also, the allowed Completion Times for Required Actions B.2.2.1 and B.2.2.2 for bringing the plant to MODES 3 and 4 are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS	SR 3.4.8.1 This Surveillance is performed to ensure iodine remains within limit during normal operation. The 7 day Frequency is adequate to trend changes in the iodine activity level. This SR is modified by a Note that requires this Surveillance to be performed only in MODE 1 because the level of fission products generated in other MODES is much less.
REFERENCES	1. 10 CFR ~~100.11.~~ 50.67, "Accident Source Term." 2. USAR, Section 15.6.4. 3. 10 CFR 50.36(c)(2)(ii).

B 3.6 CONTAINMENT SYSTEMS

B 3.6.1.6 Residual Heat Removal (RHR) Drywell Spray

BASES

BACKGROUND

The primary containment is designed with a suppression pool so that, in the event of a loss of coolant accident (LOCA), steam released from the primary system is channeled through the suppression pool water and condensed without producing significant pressurization of the primary containment. The primary containment is designed so that with the pool initially at the minimum water volume and the worst single failure of the primary containment heat removal systems, suppression pool energy absorption combined with subsequent operator controlled pool cooling will prevent the primary containment pressure from exceeding its design value. However, the primary containment must also withstand a postulated bypass leakage pathway that allows the passage of steam from the drywell directly into the suppression pool airspace, bypassing the suppression pool. The RHR Drywell Spray System is designed to mitigate the effects of bypass leakage. In addition, credit is taken for the turbulence induced by the sprays to ensure a well-mixed primary containment atmosphere during post-LOCA, which reduces the potential for a nonuniform hydrogen and oxygen concentration within the primary containment.

Insert F

There are two redundant, 100% capacity RHR drywell spray subsystems. Each subsystem consists of a suction line from the suppression pool, an RHR pump, and one spray sparger inside the drywell. Dispersion of the spray water is accomplished by spray nozzles in each subsystem.

The RHR drywell spray mode will be manually initiated, if required, following a LOCA, according to emergency procedures.

APPLICABLE SAFETY ANALYSES

Reference 1 contains the results of analyses that predict the primary containment pressure response for a LOCA with the maximum allowable bypass leakage area.

The equivalent flow path area for bypass leakage has been specified to be 0.054 ft^2. The analysis demonstrates that with drywell spray operation (in conjunction with RHR suppression pool spray operation) the primary containment

(continued)

<u>INSERT F</u> (for TS Bases page B 3.6.1.6-1)

The RHR Drywell Spray System is also operated post-LOCA to remove fission products from the drywell atmosphere and to reduce primary containment pressure.

BASES

APPLICABLE SAFETY ANALYSES (continued)

pressure remains within design limits. The RHR drywell spray mode also helps promote a uniform mixing of hydrogen and oxygen in the containment following a LOCA (Ref. 2).

Insert G

The RHR drywell spray satisfies Criterion 3 of Reference 3.

LCO

In the event of a Design Basis Accident (DBA), a minimum of one RHR drywell spray subsystem is required to mitigate potential bypass leakage paths, maintain the primary containment peak pressure below design limits, and ensures adequate mixing of the containment atmosphere, and remove fission products from the drywell atmosphere. To ensure that these requirements are met, two RHR drywell spray subsystems must be OPERABLE. Therefore, in the event of an accident, at least one subsystem is OPERABLE assuming the worst case single active failure. An RHR drywell spray subsystem is OPERABLE when the pump and associated piping, valves, instrumentation, and controls are OPERABLE.

APPLICABILITY

In MODES 1, 2, and 3, a DBA could cause pressurization of primary containment. In MODES 4 and 5, the probability and consequences of these events are reduced due to the pressure and temperature limitations in these MODES. Therefore, maintaining RHR drywell spray subsystems OPERABLE is not required in MODE 4 or 5.

ACTIONS

A.1

With one RHR drywell spray subsystem inoperable, the inoperable subsystem must be restored to OPERABLE status within 7 days. In this condition, the remaining OPERABLE RHR drywell spray subsystem is adequate to perform the primary containment bypass leakage mitigation, and mixing, and fission product removal functions. However, the overall reliability is reduced because a single failure in the OPERABLE subsystem could result in reduced primary containment bypass leakage mitigation, and mixing, and fission product removal capability. The 7 day Completion Time was chosen in light of the redundant RHR drywell spray capabilities afforded by the OPERABLE subsystem and the low probability of a DBA occurring during this period.

(continued)

INSERT G (for TS Bases page B 3.6.1.6-2)

Following a LOCA, the radiological consequences from the accident will remain within the limits of 10 CFR 50.67 (Ref. 4) provided sufficient fission products are removed from the drywell atmosphere. The Alternative Source Term analysis methodology credits the RHR Drywell Spray System for the removal of fission products from the drywell atmosphere, as allowed by NUREG-0800, Section 6.5.2 (Ref. 5). The Drywell Spray System is also credited for reducing the post-LOCA primary containment pressure, which reduces the leak rate of airborne activity from primary containment.

BASES

ACTIONS (continued)

B.1

and fission product removal

With two RHR drywell spray subsystems inoperable, one subsystem must be restored to OPERABLE status within 8 hours. In this condition, there is a substantial loss of the primary containment bypass leakage mitigation function(s). The 8 hour Completion Time is based on this loss of function and is considered acceptable due to the low probability of a DBA and because alternative methods to reduce primary containment pressure and ensure adequate mixing in the primary containment are available.

C.1 and C.2

If any Required Action and associated Completion Time cannot be met, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 12 hours and to MODE 4 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.6.1.6.1

Verifying the correct alignment for manual and power operated valves in the RHR containment spray mode flow path provides assurance that the proper flow paths will exist for system operation. This SR does not apply to valves that are locked, sealed, or otherwise secured in position, since these were verified to be in the correct position prior to locking, sealing, or securing. A valve is also allowed to be in the nonaccident position provided it can be aligned to the accident position within the time assumed in the accident analysis. This is acceptable, since the RHR drywell spray mode is manually initiated. This SR does not require any testing or valve manipulation; rather, it involves verification that those valves capable of being mispositioned are in the correct position. This SR does not apply to valves that cannot be inadvertently misaligned, such as check valves.

The 31 day Frequency of this SR is justified because the valves are operated under procedural control, improper valve position would affect only a single subsystem, the

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.6.1.6.1 (continued)

probability of an event requiring initiation of the system is low, and the system is a manually initiated system. This Frequency has been shown to be acceptable based on operating experience.

SR 3.6.1.6.2

Verifying, by administrative means, that each required RHR pump is OPERABLE ensures that the RHR pump is capable of performing its intended function (i.e., capable of developing the assumed drywell spray flow rate) when in the drywell spray mode. This Surveillance is met by verifying that another required Surveillance, which demonstrated the RHR pump OPERABILITY, was performed within the required Frequency. The verification can be performed by examining logs or other information, to determine if a required RHR pump is out of service for maintenance or other reasons. It is not necessary to perform an additional Surveillance needed to demonstrate the OPERABILITY of the required RHR pumps. The Frequency of 92 days is consistent with the normal RHR pump flow rate Surveillance Frequency ("in accordance with the Inservice Testing Program") in other Surveillances.

SR 3.6.1.6.3

This Surveillance is performed every 10 years to verify by performance of an air flow test that the spray nozzles in the drywell spray spargers are not obstructed and that flow will be provided when required. The 10 year Frequency is adequate to detect degradation in performance due to the passive nozzle design and its normally dry state and has been shown to be acceptable through operating experience.

REFERENCES

1. USAR, Section 6.2.1.1.3.
2. USAR, Section 6.2.5.2.1.
3. 10 CFR 50.36(c)(2)(ii).
4. 10 CFR 50.67, "Accident Source Term."
5. NUREG-0800, Section 6.5.2, Revision 2, December 1988

B 3.6 CONTAINMENT SYSTEMS

B 3.6.4.1 Secondary Containment

BASES

BACKGROUND

The function of the secondary containment is to contain, dilute, and hold up fission products that may leak from primary containment following a Design Basis Accident (DBA). In conjunction with operation of the Standby Gas Treatment (SGT) System and closure of certain valves whose lines penetrate the secondary containment, the secondary containment is designed to reduce the activity level of the fission products prior to release to the environment and to isolate and contain fission products that are released during certain operations that take place inside primary containment, when primary containment is not required to be OPERABLE, or that take place outside primary containment.

The secondary containment (consisting of the reactor building and auxiliary bay structures) is a structure that completely encloses the primary containment and those components that may be postulated to contain primary system fluid, with the exception of the ASME III Code Class 1 piping and valves in the steam tunnel (Ref. 1). This structure forms a control volume that serves to hold up and dilute the fission products. It is possible for the pressure in the control volume to rise relative to the environmental pressure (e.g., due to pump/motor heat load additions). To prevent ground level exfiltration while allowing the secondary containment to be designed as a conventional structure, the secondary containment requires support systems to maintain the control volume pressure at less than the external pressure. Requirements for these systems are specified separately in LCO 3.6.4.2, "Secondary Containment Isolation Valves (SCIVs)," and LCO 3.6.4.3, "Standby Gas Treatment (SGT) System."

APPLICABLE SAFETY ANALYSES

There are two principal accidents for which credit is taken for secondary containment OPERABILITY. These are a loss of coolant accident (LOCA) (Ref. 2), and a fuel handling accident involving handling recently irradiated fuel (i.e., fuel that has occupied part of a critical reactor core within the previous 24 hours ~~2 days~~) (Ref. 3). The secondary containment performs no active function in response to each of these limiting events; however, its leak tightness is required to ensure that the release of radioactive materials from the primary containment is restricted to those leakage paths and associated leakage rates assumed in the accident analysis,

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

and that fission products entrapped within the secondary containment structure will be treated by the SGT System prior to discharge to the environment.

Secondary containment satisfies Criterion 3 of Reference 4.

LCO

An OPERABLE secondary containment provides a control volume into which fission products that bypass or leak from primary containment, or are released from the reactor coolant pressure boundary components located in secondary containment, can be diluted and processed prior to release to the environment. For the secondary containment to be considered OPERABLE, it must have adequate leak tightness to ensure that the required vacuum can be established and maintained.

APPLICABILITY

In MODES 1, 2, and 3, a LOCA could lead to a fission product release to primary containment and leaks to secondary containment. Therefore, secondary containment OPERABILITY is required during the same operating conditions that require primary containment OPERABILITY.

In MODES 4 and 5, the probability and consequences of the LOCA are reduced due to the pressure and temperature limitations in these MODES. Therefore, maintaining secondary containment OPERABLE is not required in MODE 4 or 5 to ensure a control volume, except for other situations for which significant releases of radioactive material can be postulated, such as during operations with a potential for draining the reactor vessel (OPDRVs) or during movement of recently irradiated fuel assemblies in the secondary containment.

Due to radioactive decay, secondary containment is only required to be OPERABLE during fuel handling involving handling recently irradiated fuel (i.e., fuel that has occupied part of a critical reactor core within the previous ~~2~~ 24 hours ~~days~~).

ACTIONS

A.1

If secondary containment is inoperable, it must be restored to OPERABLE status within 4 hours. The 4 hour Completion Time provides a period of time to correct the problem that is commensurate with the important of maintaining secondary

(continued)

B 3.6 CONTAINMENT SYSTEMS

B 3.6.4.2 Secondary Containment Isolation Valves (SCIVs)

BASES

BACKGROUND

The function of the SCIVs, in the combination with other accident mitigation systems, is to limit fission product release during and following postulated Design Basis Accidents (DBAs) (Refs. 1 and 2). Secondary containment isolation within the time limits specified for those isolation valves designed to close automatically ensures that fission products that leak from primary containment following a DBA, that are released during certain operations when primary containment is not required to be OPERABLE, or that take place outside primary containment, are maintained within the secondary containment boundary.

The OPERABILITY requirement for SCIVs help ensure that an adequate secondary containment boundary is maintained during and after an accident by minimizing potential paths to the environment. These isolation devices are either passive or active (automatic). Manual valves, de-activated automatic valves secured in their closed position (including check valves with flow through the valve secured), and blind flanges are considered passive devices.

Automatic SCIVs (i.e., dampers) close on a secondary containment isolation signal to establish a boundary for untreated radioactive material within secondary containment following a DBA or other accidents.

Other penetrations are isolated by the use of valves in the closed position or blind flanges (which includes plugs and caps as listed in Reference 3).

APPLICABLE SAFETY ANALYSES

The SCIVs must be OPERABLE to ensure the secondary containment barrier to fission product releases is established. The principal accidents for which the secondary boundary is required are a loss of coolant accident (Ref. 1) and a fuel handling accident involving handling recently irradiated fuel (i.e., fuel that has occupied part of a critical reactor core within the previous 2 days 24 hours (Ref. 2). The secondary containment performs no active function in response to each of these limiting events, but the boundary established by SCIVs is required to ensure that leakage from the primary containment is processed by the Standby Gas Treatment (SGT) System before being released to the environment.

(continued)

APPLICABLE SAFETY ANALYSES (continued)

Maintaining SCIVs OPERABLE with isolation times within limits ensures that fission products will remain trapped inside secondary containment so that they can be treated by the SGT System prior to discharge to the environment.

SCIVs satisfy Criterion 3 of Reference 4.

LCO

SCIVs form a part of the secondary containment boundary. The SCIV safety function is related to control of offsite radiation releases resulting from DBAs.

The power operated, automatic isolation valves are considered OPERABLE when their isolation times are within limits and the valves actuate on an automatic isolation signal. The valves covered by this LCO, along with their associated stroke times, are listed in Reference 3.

The normally closed manual SCIVs are considered OPERABLE when the valves are closed and blind flanges in place, or open under administrative controls. These passive isolation valves or devices are listed in Reference 3.

APPLICABILITY

In MODES 1, 2, and 3, a DBA could lead to a fission product release to the primary containment that leaks to the secondary containment. Therefore, OPERABILITY of SCIVs is required.

In MODES 4 and 5, the probability and consequences of these events are reduced due to pressure and temperature limitations in these MODES. Therefore, maintaining SCIVs OPERABLE is not required in MODE 4 or 5, except for other situations under which significant releases of radioactive material can be postulated, such as during operations with a potential for draining the reactor vessel (OPDRVs) or during movement of recently irradiated fuel assemblies in the secondary containment. Due to radioactive decay, SCIVs are only required to be OPERABLE during fuel handling involving handling recently irradiated fuel (i.e., fuel that has occupied part of a critical reactor core within the previous ~~2 days~~ 24 hours).

ACTIONS

The ACTIONS are modified by three Notes. The first Note allows penetration flow paths to be unisolated intermittently under administrative controls. These controls consist of stationing a dedicated operator, who is

(continued)

BACKGROUND (continued)

protect the charcoal from fouling. The charcoal adsorber removes gaseous elemental iodine and organic iodides, and the final HEPA filter is provided to collect any carbon fines exhausted from the charcoal adsorber.

The SGT System automatically starts and operates in response to actuation signals indicative of conditions or an accident that could require operation of the system. Following initiation, both fans will start and the associated train inlet and fan discharge valves will open. Negative pressure in the reactor building is automatically controlled by the SGT System filter train recirculation line pressure control valves.

APPLICABLE SAFETY ANALYSES

24 hours

The design basis for the SGT System is to mitigate the consequences of a loss of coolant accident and fuel handling accidents. Due to radioactive decay, the SGT System is only required to be OPERABLE during fuel handling involving handling recently irradiated fuel (i.e., fuel that has occupied part of a critical reactor core within the previous ~~2 days~~) (Refs. 3 and 4). For all events analyzed, the SGT System is shown to be automatically initiated to reduce, via filtration and adsorption, the radioactive material released to the environment.

The SGT System satisfies Criterion 3 of Reference 5.

LCO

Following a DBA, a minimum of one SGT subsystem is required to maintain the secondary containment at a negative pressure with respect to the environment and to process gaseous releases. Meeting the LCO requirements for two OPERABLE subsystems ensures operation of at least one SGT subsystem in the event of a single active failure.

APPLICABILITY

In MODES 1, 2, and 3, a DBA could lead to a fission product release to primary containment that leaks to secondary containment. Therefore, SGT System OPERABILITY is required during these MODES.

In MODES 4 and 5, the probability and consequences of these events are reduced due to the pressure and temperature limitations in these MODES. Therefore, maintaining the SGT System OPERABLE is not required in MODE 4 or 5, except for other situations under which significant releases of radioactive material can be postulated, such as during operations with a potential for draining the reactor vessel (OPDRVs) or during movement of recently irradiated fuel assemblies in the secondary containment. Due to radioactive

(continued)

BASES

APPLICABILITY (continued)	decay, the SGT System is only required to be OPERABLE during fuel handling involving handling recently irradiated fuel (i.e., fuel that has occupied part of a critical reactor core within the previous ~~2 days~~ 24 hours).

ACTIONS

A.1

With one SGT subsystem inoperable, the inoperable subsystem must be restored to OPERABLE status within 7 days. In this condition, the remaining OPERABLE SGT subsystem is adequate to perform the required radioactivity release control function. However, the overall system reliability is reduced because a single failure in the OPERABLE subsystem could result in the radioactivity release control function not being adequately performed. The 7 day Completion Time is based on consideration of such factors as the availability of the OPERABLE redundant SGT subsystem and the low probability of a DBA occurring during this period.

B.1 and B.2

If the SGT subsystem cannot be restored to OPERABLE status within the required Completion Time in MODE 1, 2, or 3, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 12 hours and to MODE 4 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

C.1, C.2.1, and C.2.2

During movement of recently irradiated fuel assemblies in the secondary containment or during OPDRVs, when Required Action A.1 cannot be completed within the required Completion Time, the OPERABLE SGT subsystem should be immediately placed in operation. This Required Action ensures that the remaining subsystem is OPERABLE, that no failures that could prevent automatic actuation will occur, and that any other failure would be readily detected.

An alternative to Required Action C.1 is to immediately suspend activities that represent a potential for releasing a significant amount of radioactive material to the secondary containment, thus placing the unit in a condition that minimizes risk. If applicable, movement of recently irradiated fuel assemblies must be

(continued)

B 3.7 PLANT SYSTEMS

B 3.7.2 Control Room Envelope Filtration (CREF) System

BASES

BACKGROUND

The CREF System provides a radiologically controlled environment from which the unit can be safely operated following a Design Basis Accident (DBA). The control room envelope consists of all rooms and areas located in the main control room and relay room of the control building. Included in the envelope are the main control room, relay room, instrument shop, training room, shift supervisor's office, lunch room, toilets, corridors, work release room, and HVAC equipment rooms (Ref. 1).

The safety related function of the CREF System used to control radiation exposure consists of two independent and redundant high efficiency air filtration subsystems for treatment of recirculated air and outside supply air. Each subsystem includes a control room outdoor air special filter train (CROASFT), which consists of an electric heater, a prefilter, a high efficiency particulate air (HEPA) filter, an activated charcoal adsorber section, a second HEPA filter, a filter booster fan, and the associated ductwork and dampers. The electric heater is used to reduce the relative humidity of the air entering the filter train but is not required for CROASFT OPERABILITY. Prefilters and HEPA filters remove particulate matter that may be radioactive. The charcoal adsorbers provide a holdup period for gaseous iodine, allowing time for decay. Each subsystem also includes the necessary outside air intake(s) and two air conditioning units (fan portion only), one for the control room and one for the relay room. Each outside air intake is capable of providing 100% of the necessary makeup flow. Therefore, ~~normally~~ only one outside air intake is required ~~necessary~~. ~~However, when the unit is in MODE 1, 2, or 3 with MSIV leakage > 15 scfh for any MSIV, both outside air intakes, including the capability to isolate the intakes, are necessary. Both outside air intakes are required in these conditions since the accident analysis assumes the most contaminated outside air intake is isolated 8 hours after the accident to ensure the dose to control room envelope personnel does not exceed the limit. The outside air intake that is not isolated continues to be capable of providing 100% of the necessary makeup flow.~~ The two ~~required~~ outside air intakes are allowed to be common to both subsystems (since there are only two outside air

(continued)

BASES

BACKGROUND (continued)

intakes for the CREF System). ~~Alternately, if MSIV leakage is > 15 scfh for any MSIV, an additional analysis may be performed to determine the "effective" MSIV leakage. The "effective" MSIV leakage is the individual MSIV leak rate when all four main steam lines are assumed to leak at the same rate, and the doses in the control room envelope are equivalent to those when the individual "as-left" valve leak rates are used. If the "effective" MSIV leakage is ≤ 15 scfh, then only one outside air intake is necessary.~~

The CROASFT portion of the safety related CREF System is normally in standby, but the remaining portions of the CREF System (the outside air intakes and fan portion of the air conditioning units) are operated to maintain the control room envelope environment during normal operation. Upon receipt of the initiation signal(s) (indicative of conditions that could result in radiation exposure to control room envelope personnel), the CREF System automatically switches to the emergency pressurization mode of operation to ~~prevent~~ minimize infiltration of contaminated air into the control room envelope. A system of valves and dampers redirects all control room envelope outside air flow through the two CROASFTs. In addition, a portion of the control room air is recirculated through the CROASFTs. The air conditioning units (fan portion only) maintain the 1/8 inch positive pressure; the CROASFT booster fan only provides the motive force to overcome the added resistance of the CROASFT being in service.

The CREF System is designed to maintain the control room envelope environment for a 30 day continuous occupancy (i.e., considering the occupancy factors of ~~NUREG-0800, Table 6.4-1,~~ Regulatory Guide 1.183, Regulatory Position 4.2.6, Ref. 2) after a DBA, while limiting the dosage to personnel to not more than 5 rem ~~whole body or its equivalent to any part of the body.~~ TEDE. CREF System operation in maintaining the control room envelope habitability is discussed in the USAR, Sections 6.4.1 and 9.4.1 (Refs. 3 and 4, respectively).

APPLICABLE SAFETY ANALYSES

The ability of the CREF System to maintain the habitability of the control room envelope is an explicit assumption for the safety analyses presented in the USAR, Chapters 6 and 15 (Refs. 5 and 6, respectively). The emergency pressurization mode of the CREF System is assumed to operate following a loss of coolant accident ~~, main steam line break,~~ and a fuel handling accident involving handling recently irradiated fuel (i.e, fuel that has occupied part of a critical reactor core within the previous 24 hours. ~~, and control rod drop accident.~~ The radiological doses to control room envelope personnel as a

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(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

result of the various DBAs are summarized in Reference 6. No single active failure will cause the loss of outside or recirculated air from the control room envelope.

The CREF System satisfies Criterion 3 of Reference 7.

LCO

Two redundant subsystems of the CREF System are required to be OPERABLE to ensure that at least one is available, assuming a single failure disables the other subsystem. Total system failure could result in exceeding a dose of 5 rem TEDE to the control room operators in the event of a DBA.

The CREF System is considered OPERABLE when the individual components necessary to control operator exposure are OPERABLE in both subsystems. A subsystem is considered OPERABLE when its associated:

a. CROASFT is OPERABLE;

b. Air conditioning units (fan portion only) are OPERABLE (one for the control room and one for the relay room), including the ductwork, to maintain air circulation to and from the control room envelope; and

c. One of the two ~~Necessary~~ outside air intake(s) is ~~are~~ OPERABLE. ~~When the unit is not in MODES 1, 2, and 3, or when the unit is in MODE 1, 2, or 3 with MSIV leakage ≤ 15 scfh for each MSIV, only one outside air intake is necessary. When the unit is in MODE 1, 2, or 3 with MSIV leakage > 15 scfh for any MSIV, both outside air intakes, including the capability to isolate the intakes, are necessary and are allowed to be common to both subsystems. Alternately, if MSIV leakage is > 15 scfh for any MSIV, an additional analysis may be performed to determine the "effective" MSIV leakage. If the "effective" MSIV leakage is ≤ 15 scfh, then only one outside air intake is necessary.~~

A CROASFT is considered OPERABLE when its associated filter booster fan is OPERABLE; HEPA filter and charcoal adsorber are not excessively restricting flow and are capable of performing their filtration functions; and ductwork, valves, and dampers are OPERABLE, and air circulation through the filter train can be maintained.

In addition, the control room envelope boundary must be maintained, including the integrity of the walls, floors,

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(continued)

BASES

LCO (continued)

ceilings, ductwork, and access doors, such that the pressurization limit of SR 3.7.2.4 can be met. However, it is acceptable for access doors to be open for normal control room envelope entry and exit and not consider it to be a failure to meet the LCO.

The LCO is modified by a Note allowing the control room envelope boundary to be opened intermittently under administrative controls. For entry and exit through the doors, the administrative control of the opening is performed by the person(s) entering or exiting the area. For other openings, these controls consist of stationing a dedicated individual at the opening who is in continuous communication with the control room. This individual will have a method to rapidly close the opening when a need for control room isolation is indicated.

APPLICABILITY

In MODES 1, 2, and 3, the CREF System must be OPERABLE to control operator exposure during and following a DBA, since the DBA could lead to a fission product release.

In MODES 4 and 5, the probability and consequences of a DBA are reduced due to the pressure and temperature limitations in these MODES. Therefore, maintaining the CREF System OPERABLE is not required in MODE 4 or 5, except for the following situations under which significant radioactive releases can be postulated:

b ~~a.~~ During movement of recently irradiated fuel assemblies in the secondary containment~~;~~. Insert H

~~b. During CORE ALTERATIONS; and~~

a ~~c.~~ During operations with a potential for draining the reactor vessel (OPDRVs)~~.~~; and

ACTIONS

A.1

With one CREF subsystem inoperable, or with both CREF subsystems inoperable but the CREF System safety function maintained, the inoperable CREF subsystem(s) must be restored to OPERABLE status within 7 days. The CREF System safety function is maintained when the CREF System components equivalent to one CREF subsystem are

(continued)

<u>INSERT H</u> (for TS Bases page B 3.7.2-4)

Due to radioactive decay, the CREF System is only required to be OPERABLE during fuel handling involving handling recently irradiated fuel (i.e., fuel that has occupied part of a critical reactor core within the previous 24 hours).

BASES

ACTIONS (continued)

D.1, D.2.1, and D.2.2~~, and D.2.3~~

Insert J

~~LCO 3.0.3 is not applicable while in MODE 4 or 5. However, since irradiated fuel assembly movement can occur in MODE 1, 2, or 3, the Required Actions of Condition D are modified by a Note indicating that LCO 3.0.3 does not apply. If moving irradiated fuel assemblies while in MODE 1, 2, or 3, the fuel movement is independent of reactor operations. Entering LCO 3.0.3 while in MODE 1, 2, or 3 would require the unit to be shutdown, but would not require immediate suspension of movement of irradiated fuel assemblies. The Note to the ACTIONS, "LCO 3.0.3 is not applicable," ensures that the actions for immediate suspension of irradiated fuel assembly movement are not postponed due to entry into LCO 3.0.3.~~

During movement of recently irradiated fuel assemblies in the secondary containment ~~during CORE ALTERATIONS,~~ or during OPDRVs, if the inoperable CREF subsystem(s) cannot be restored to OPERABLE status within the required Completion Time, the OPERABLE components of the CREF subsystem(s) equivalent to a single CREF subsystem (e.g., the CROASFT and fan portion of the air conditioning units do not have to be powered from the same electrical division) may be placed in the emergency pressurization mode. This action ensures that the remaining subsystem (or components in both subsystems equivalent to a single CREF subsystem) is OPERABLE, that no failures that would prevent automatic actuation will occur, and that any active failure will be readily detected.

An alternative to Required Action D.1 is to immediately suspend activities that present a potential for releasing radioactivity that might require isolation of the control room envelope. This places the unit in a condition that minimizes risk.

If applicable, ~~CORE ALTERATIONS and~~ movement of recently irradiated fuel assemblies in the secondary containment must be suspended immediately. Suspension of these activities shall not preclude completion of movement of a component to a safe position. Also, if applicable, action must be initiated immediately to suspend OPDRVs to minimize the probability of a vessel draindown and subsequent potential for fission product release. Action must continue until the OPDRVs are suspended.

(continued)

<u>INSERT J</u> (for TS Bases page B 3.7.2-6)

The Required Actions of Condition D are modified by a Note stating that LCO 3.0.3 is not applicable. If moving recently irradiated fuel assemblies while in MODES 4 or 5, LCO 3.0.3 would not specify any action. If moving recently irradiated fuel assemblies while in MODES 1, 2, or 3, the fuel movement is independent of reactor operations. Therefore, in either case, inability to suspend movement of recently irradiated fuel assemblies would not be a sufficient reason to require a reactor shutdown.

BASES

ACTIONS (continued)

E.1

If both CREF subsystems are inoperable with the CREF System safety function not maintained in MODE 1, 2, or 3 for reasons other than an inoperable control room envelope boundary (i.e., Condition B), the CREF System may not be capable of performing the intended function and the unit is in a condition outside of the accident analyses. Therefore, LCO 3.0.3 must be entered immediately.

F.1 and F.2~~, and F.3~~

Insert K

~~LCO 3.0.3 is not applicable while in MODE 4 or 5. However, since irradiated fuel assembly movement can occur in MODE 1, 2, or 3, the Required Actions of Condition F are modified by a Note indicating that LCO 3.0.3 does not apply. If moving irradiated fuel assemblies while in MODE 1, 2, or 3, the fuel movement is independent of reactor operations. Entering LCO 3.0.3 while in MODE 1, 2, or 3 would require the unit to be shutdown, but would not require immediate suspension of movement of irradiated fuel assemblies. The Note to the ACTIONS, "LCO 3.0.3 is not applicable," ensures that the actions for immediate suspension of irradiated fuel assembly movement are not postponed due to entry into LCO 3.0.3.~~

During movement of recently irradiated fuel assemblies in the secondary containment~~, during CORE ALTERATIONS,~~ or during OPDRVs, with two CREF subsystems inoperable with the CREF System safety function not maintained, action must be taken immediately to suspend activities that present a potential for releasing radioactivity that might require isolation of the control room envelope. This places the unit in a condition that minimizes risk.

If applicable, ~~CORE ALTERATIONS and~~ movement of recently irradiated fuel assemblies in the secondary containment must be suspended immediately. Suspension of these activities shall not preclude completion of movement of a component to a safe position. If applicable, actions must be initiated immediately to suspend OPDRVs to minimize the probability of a vessel draindown and subsequent potential for fission product release. Actions must continue until the OPDRVs are suspended.

<u>INSERT K</u> (for TS Bases page B 3.7.2-7)

The Required Actions of Condition F are modified by a Note stating that LCO 3.0.3 is not applicable. If moving recently irradiated fuel assemblies while in MODES 4 or 5, LCO 3.0.3 would not specify any action. If moving recently irradiated fuel assemblies while in MODES 1, 2, or 3, the fuel movement is independent of reactor operations. Therefore, in either case, inability to suspend movement of recently irradiated fuel assemblies would not be a sufficient reason to require a reactor shutdown.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.7.2.4

This SR verifies the integrity of the control room envelope and the assumed inleakage rates of potentially contaminated air. The control room envelope positive pressure, with respect to potentially contaminated adjacent areas, is periodically tested to verify proper function of the CREF System. The SR requires all combinations of the CREF System to be verified. This can be met by determining (by test) the worst combination of the air conditioning units (fan portion only), then testing the worst combination of the air conditioning units (fan portion only) with each CROASFT. During the emergency pressurization mode of operation, the CREF System is designed to slightly pressurize the control room envelope to ≥ 0.125 inches water gauge positive pressure with respect to outside atmosphere to prevent unfiltered inleakage. The CREF System is designed to maintain this positive pressure at an outside air intake flow rate of ≤ 1500 cfm to the control room envelope in the emergency pressurization mode. Compliance with this SR is demonstrated by measurement of the pressure in the control room and relay room, which are representative of adequate positive pressure in both elevations of the control room envelope. The Frequency of 24 months on a STAGGERED TEST BASIS is consistent with industry practice and other filtration system SRs.

REFERENCES

1. USAR, Section 6.4.2.1.
2. ~~NUREG-0800, Table 6.4-1~~ Regulatory Guide 1.183, July 2000.
3. USAR, Section 6.4.1.
4. USAR, Section 9.4.1.
5. USAR, Chapter 6.
6. USAR, Chapter 15.
7. 10 CFR 50.36(c)(2)(ii).
8. Regulatory Guide 1.52, Revision 2, March 1978.

BASES

APPLICABLE SAFETY ANALYSES (continued)

components in the control room envelope. A single active failure of a component of the Control Room Envelope AC System, assuming a loss of offsite power, does not impair the ability of the system to perform its design function. Redundant detectors and controls are provided for control room envelope temperature control. The Control Room Envelope AC System is designed in accordance with Seismic Category I requirements. The Control Room Envelope AC System is capable of removing sensible and latent heat loads from the control room envelope, including consideration of equipment heat loads and personnel occupancy requirements to ensure equipment OPERABILITY.

The Control Room Envelope AC System satisfies Criterion 3 of Reference 3.

LCO

Two independent and redundant subsystems of the Control Room Envelope AC System are required to be OPERABLE to ensure that at least one is available, assuming a single failure disables the other subsystem. Total system failure could result in the equipment operating temperature exceeding limits.

The Control Room Envelope AC System is considered OPERABLE when the individual components necessary to maintain the control room envelope temperature are OPERABLE in both subsystems. These components include the control room and relay room air conditioning units (cooling coils and fans only), the control building chilled water subsystems, ductwork, dampers, and associated instrumentation and controls. In addition, during conditions in MODES other than MODES 1, 2, and 3 when the Control Room Envelope AC System is required to be OPERABLE (e.g., during ~~CORE ALTERATIONS~~ operations with a potential for draining the reactor vessel (OPDRVs)), the necessary portions of the SW System and Ultimate Heat Sink capable of providing cooling to the hermetic centrifugal water chillers are part of the OPERABILITY requirements covered by this LCO.

APPLICABILITY

In MODE 1, 2, or 3, the Control Room Envelope AC System must be OPERABLE to ensure that the control room envelope temperature will not exceed equipment OPERABILITY limits following control room envelope isolation.

In MODES 4 and 5, the probability and consequences of a Design Basis Accident are reduced due to the pressure and

(continued)

BASES

APPLICABILITY (continued)

temperature limitations in these MODES. Therefore, maintaining the Control Room Envelope AC System OPERABLE is not required in MODE 4 or 5, except for the following situations under which significant radioactive releases can be postulated:

b ~~a~~. During movement of recently irradiated fuel assemblies in the secondary containment~~;~~. Insert L

~~b. During CORE ALTERATIONS; and~~

a ~~c~~. During operations with a potential for draining the reactor vessel (OPDRVs); and

ACTIONS

A.1

With one control room envelope AC subsystem inoperable, or with both control room envelope AC subsystems inoperable but the Control Room Envelope AC System safety function maintained, the inoperable control room envelope AC subsystem(s) must be restored to OPERABLE status within 30 days. The Control Room Envelope AC System safety function is maintained when the Control Room Envelope AC System components equivalent to one control room envelope AC subsystem are OPERABLE. With the unit in this condition, the remaining OPERABLE control room envelope AC subsystem (or OPERABLE components in both subsystems) is adequate to perform the control room envelope air conditioning function. However, the overall reliability is reduced because a single failure in the OPERABLE subsystem (or remaining OPERABLE portions of the subsystems, as applicable) could result in loss of the control room envelope air conditioning function. The 30 day Completion Time is based on the low probability of an event occurring requiring control room envelope isolation, the consideration that the remaining subsystem (or components in both subsystems) can provide the required protection, and the availability of alternate cooling methods.

B.1 and B.2

In MODE 1, 2, or 3, if the inoperable control room envelope AC subsystem(s) cannot be restored to OPERABLE status within the associated Completion Time, the unit must be placed in a MODE that minimizes risk. To achieve this status the unit

(continued)

INSERT L (for TS Bases page B 3.7.3-3)

Due to radioactive decay, the Control Room Envelope AC System is only required to be OPERABLE during fuel handling involving handling recently irradiated fuel (i.e., fuel that has occupied part of a critical reactor core within the previous 24 hours).

BASES

ACTIONS

B.1 and B.2 (continued)

must be placed in at least MODE 3 within 12 hours and in MODE 4 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

C.1, C.2.1, and C.2.2~~, and C.2.3~~

Insert M

~~LCO 3.0.3 is not applicable while in MODE 4 or 5. However, since irradiated fuel assembly movement can occur in MODE 1, 2, or 3, the Required Actions of Condition C are modified by a Note indicating that LCO 3.0.3 does not apply. If moving irradiated fuel assemblies while in MODE 1, 2, or 3, the fuel movement is independent of reactor operations. Entering LCO 3.0.3 while in MODE 1, 2, or 3 would require the unit to be shutdown, but would not require immediate suspension of movement of irradiated fuel assemblies. The Note to the ACTIONS, "LCO 3.0.3 is not applicable," ensures that the actions for immediate suspension of irradiated fuel assembly movement are not postponed due to entry into LCO 3.0.3.~~

During movement of recently irradiated fuel assemblies in the secondary containment~~, during CORE ALTERATIONS,~~ or during OPDRVs, if Required Action A.1 cannot be completed within the required Completion Time, the OPERABLE components of the control room envelope AC subsystem(s) equivalent to a single control room envelope AC subsystem (e.g., the control building chilled water subsystem and air conditioning units do not have to be powered from the same electrical division) may be placed immediately in operation. This action ensures that the remaining subsystem (or components in both subsystems equivalent to a single control room envelope AC subsystem) is OPERABLE, that no failures that would prevent actuation will occur, and that any active failure will be readily detected.

An alternative to Required Action C.1 is to immediately suspend activities that present a potential for releasing radioactivity that might require isolation of the control room envelope. This places the unit in a condition that minimizes risk.

(continued)

INSERT M (for TS Bases page B 3.7.3-4)

The Required Actions of Condition C are modified by a Note stating that LCO 3.0.3 is not applicable. If moving recently irradiated fuel assemblies while in MODES 4 or 5, LCO 3.0.3 would not specify any action. If moving recently irradiated fuel assemblies while in MODES 1, 2, or 3, the fuel movement is independent of reactor operations. Therefore, in either case, inability to suspend movement of recently irradiated fuel assemblies would not be a sufficient reason to require a reactor shutdown.

BASES

and

ACTIONS

C.1, C.2.1, C.2.2~~, and C.2.3~~ (continued)

recently

If applicable, ~~CORE ALTERATIONS and~~ movement of recently irradiated fuel assemblies in the secondary containment must be suspended immediately. Suspension of these activities shall not preclude completion of movement of a component to a safe position. Also, if applicable, action must be initiated immediately to suspend OPDRVs to minimize the probability of a vessel draindown and subsequent potential for fission product release. Action must continue until the OPDRVs are suspended.

D.1

If both control room envelope AC subsystems are inoperable with the Control Room Envelope AC System safety function not maintained in MODE 1, 2, or 3, the Control Room Envelope AC System may not be capable of performing the intended function. Therefore, LCO 3.0.3 must be entered immediately.

and

E.1 and E.2~~, and E.3~~

Insert N

~~LCO 3.0.3 is not applicable while in MODE 4 or 5. However, since irradiated fuel assembly movement can occur in MODE 1, 2, or 3, the Required Actions of Condition E are modified by a Note indicating that LCO 3.0.3 does not apply. If moving irradiated fuel assemblies while in MODE 1, 2, or 3, the fuel movement is independent of reactor operations. Entering LCO 3.0.3 while in MODE 1, 2, or 3 would require the unit to be shutdown, but would not require immediate suspension of movement of irradiated fuel assemblies. The Note to the ACTIONS, "LCO 3.0.3 is not applicable," ensures that the actions for immediate suspension of irradiated fuel assembly movement are not postponed due to entry into LCO 3.0.3.~~

recently

During movement of recently irradiated fuel assemblies in the secondary containment~~, during CORE ALTERATIONS,~~ or during OPDRVs with two control room envelope AC subsystems inoperable with the Control Room Envelope AC System safety function not maintained, action must be taken to immediately suspend activities that present a potential for releasing radioactivity that might require isolation of the control room envelope. This places the unit in a condition that minimizes risk.

(continued)

INSERT N (for TS Bases page B 3.7.3-5)

The Required Actions of Condition E are modified by a Note stating that LCO 3.0.3 is not applicable. If moving recently irradiated fuel assemblies while in MODES 4 or 5, LCO 3.0.3 would not specify any action. If moving recently irradiated fuel assemblies while in MODES 1, 2, or 3, the fuel movement is independent of reactor operations. Therefore, in either case, inability to suspend movement of recently irradiated fuel assemblies would not be a sufficient reason to require a reactor shutdown.

BASES

ACTIONS

E.1 and E.2~~, and E.3~~ (continued)

If applicable, ~~CORE ALTERATIONS and~~ handling of recently irradiated fuel in the secondary containment must be suspended immediately. Suspension of these activities shall not preclude completion of movement of a component to a safe position. Also, if applicable, action must be initiated immediately to suspend OPDRVs to minimize the probability of a vessel draindown and subsequent potential for fission product release. Action must continue until the OPDRVs are suspended.

SURVEILLANCE REQUIREMENTS

SR 3.7.3.1

This SR verifies that the heat removal capability of the system is sufficient to remove the control room envelope heat load assumed in the safety analyses. The SR consists of a combination of testing and calculation. The 24 month Frequency is appropriate since significant degradation of the Control Room Envelope AC System is not expected over this time period.

REFERENCES

1. USAR, Section 6.4.
2. USAR, Section 9.4.1.
3. 10 CFR 50.36(c)(2)(ii).

B 3.7 PLANT SYSTEMS

B 3.7.4 Main Condenser Offgas

BASES

BACKGROUND	During unit operation, steam from the low pressure turbine is exhausted directly into the main condenser. Air and noncondensible gases are collected in the main condenser, then exhausted through the steam jet air ejectors (SJAEs) to the Main Condenser Offgas System. The offgas from the main condenser normally includes radioactive gases. The Main Condenser Offgas System has been incorporated into the unit design to reduce the gaseous radwaste emission. This system uses a catalytic recombiner to recombine radiolytically dissociated hydrogen and oxygen. The gaseous mixture is cooled by the offgas condenser; the water and condensibles are stripped out by the offgas condenser and dryers. The radioactivity of the remaining gaseous mixture (i.e., the offgas recombiner effluent) is monitored downstream of the offgas dryers prior to entering the charcoal adsorbers.
APPLICABLE SAFETY ANALYSES	The main condenser offgas gross gamma activity rate is an initial condition of the Main Condenser Offgas System failure event as discussed in the USAR, Section 15.7.1 (Ref. 1). The analysis assumes a gross failure in the Main Condenser Offgas System that results in the rupture of the Main Condenser Offgas System pressure boundary. The gross gamma activity rate is controlled to ensure that during the event, the calculated offsite doses will be well within the limits (NUREG-1047, Ref. 2) of 10 CFR 100 (Ref. 3). The main condenser offgas limits satisfy Criterion 2 of Reference 4.
LCO	To ensure compliance with the assumptions of the Main Condenser Offgas System failure event (Ref. 1), the fission product release rate should be consistent with a noble gas release to the reactor coolant of 100 μCi/Mwt-second after decay of 30 minutes. The LCO is established consistent with this requirement (3536 Mwt x 100 μCi/Mwt-second $\approx$ 350,000 μCi/second).

(continued)

This event has not been re-analyzed using alternative source term methodology.

B 3.7 PLANT SYSTEMS

B 3.7.6 Spent Fuel Storage Pool Water Level

BASES

BACKGROUND

The minimum water level in the spent fuel storage pool meets the assumptions of iodine decontamination factors following a fuel handling accident.

A general description of the spent fuel storage pool design is found in the USAR, Section 9.1.2 (Ref. 1). The assumptions of the fuel handling accident are found in the USAR, Section 15.7.4 (Ref. 2).

APPLICABLE SAFETY ANALYSES

The water level above the irradiated fuel assemblies is an explicit assumption of the fuel handling accident (Ref. 2). A fuel handling accident is evaluated to ensure that the radiological consequences (calculated ~~whole body and thyroid~~ doses at the exclusion area and low population zone boundaries) are ≤ 25% (NUREG-0800, Section ~~15.7.4~~ 15.0.1, Ref. 3) of the 10 CFR ~~100~~ 50.67 (Ref. 4) exposure guidelines. A fuel handling accident could release a fraction of the fission product inventory by breaching the fuel rod cladding as discussed in the Regulatory Guide ~~1.25~~ 1.183 (Ref. 5).

The fuel handling accident is evaluated for the dropping of an irradiated fuel assembly onto the reactor core. The consequences of a fuel handling accident over the spent fuel storage pool are less severe than those of the fuel handling accident over the reactor core (Ref. 2). The water level in the spent fuel storage pool provides for absorption of water soluble fission product gases and transport delays of soluble and insoluble gases that must pass through the water before being released to the secondary containment atmosphere. This absorption and transport delay reduces the potential radioactivity of the release during a fuel handling accident.

The spent fuel storage pool water level satisfies Criterion 2 of Reference 6.

LCO

The specified water level preserves the assumption of the fuel handling accident analysis (Ref. 2). As such, it is the minimum required for fuel movement within the spent fuel storage pool.

(continued)

BASES

REFERENCES (continued)

3. NUREG-0800, Section ~~15.7.4~~ 15.0.1, Revision ~~0~~ 0, ~~July 1981~~ July, 2000.
4. 10 CFR ~~100~~ 50.67, "Accident Source Term."
5. Regulatory Guide ~~1.25, March 1972~~ 1.183, July 2000.
6. 10 CFR 50.36(c)(2)(ii).

B 3.9 REFUELING OPERATIONS

B 3.9.6 Reactor Pressure Vessel (RPV) Water Level—Irradiated Fuel

BASES

BACKGROUND

The movement of irradiated fuel assemblies within the RPV requires a minimum water level of 22 ft 3 inches above the top of the RPV flange. During refueling, this maintains a sufficient water level in the reactor vessel cavity and spent fuel storage pool. Sufficient water is necessary to retain iodine fission product activity in the water in the event of a fuel handling accident (Refs. 1 and 2). Sufficient iodine activity would be retained to limit offsite doses from the accident to ~~≤~~ 25% of 10 CFR ~~100~~ limits, as provided by the guidance of Reference 3.

50.67

≤

APPLICABLE SAFETY ANALYSES

During movement of irradiated fuel assemblies the water level in the RPV is an initial condition design parameter in the analysis of a fuel handling accident in containment postulated by Regulatory Guide ~~1.25~~ (Ref. 1). ~~A minimum water level of 23 ft (Regulatory Position c.1.c of Ref. 1) allows a decontamination factor of 100 (Regulatory Position c.1.g of Ref. 1) to be used in the accident analysis for iodine. This relates to the assumption that 99% of the total iodine released from the pellet to cladding gap of all the dropped fuel assembly rods is retained by the refueling cavity water. The fuel pellet to cladding gap is assumed to contain 10% of the total fuel rod iodine inventory (Ref. 1), except I-131 which is 12% (Ref. 4).~~

1.183

Insert P

Analysis of the fuel handling accident inside containment is described in Reference 2. With a minimum water level of 22 ft 3 inches, ~~(a decontamination factor of 100 is still expected at a water level as low as 22 ft 3 inches) and a~~ minimum decay time of 24 hours prior to fuel handling, the analysis and test programs demonstrate that the iodine release due to a postulated fuel handling accident is adequately captured by the water, and that offsite doses are maintained within allowable limits (Ref. 5). While the worst case assumptions include the dropping of the irradiated fuel assembly being handled onto the reactor core, the possibility exists of the dropped assembly striking the RPV flange and releasing fission products. Therefore, the minimum depth for water coverage to ensure

the adjusted elemental iodine decontamination factor is 175.

Assuming

(continued)

INSERT P (for TS Bases pages B 3.9.6-1 and B 3.9.7-1)

which states: "If the depth of water above the damaged fuel is 23 feet or greater, the decontamination factors for the elemental and organic species are 500 [subsequently corrected to 285 by RIS 2006-04 (Ref. 7)] and 1, respectively, giving an overall effective decontamination factor of 200 (i.e., 99.5% of the total iodine released from the damaged rods is retained by the water). This difference in decontamination factors for elemental (99.85%) and organic (0.15%) species results in the iodine above the water being composed of 57% elemental and 43% organic species. If the depth of water is not 23 feet, the decontamination factor will have to be determined on a case by case method." In accordance with Regulatory Guide 1.183, the fraction of fission product inventory in the fuel pellet to cladding gap is 10% for Kr-85, 8% for I-131, and 5% for other noble gases and halogens.

BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.9.6.1

Verification of a minimum water level of 22 ft 3 inches above the top of the RPV flange ensures that the design basis for the postulated fuel handling accident analysis during refueling operations is met. Water at the required level limits the consequences of damaged fuel rods, which are postulated to result from a fuel handling accident in containment (Ref. 2).

The Frequency of 24 hours is based on engineering judgment and is considered adequate in view of the large volume of water and the normal procedural controls on valve positions, which make significant unplanned level changes unlikely.

REFERENCES

1. Regulatory Guide ~~1.25, March 23, 1972~~ 1.183, July 2000.
2. USAR, Section 15.7.4.
3. NUREG-0800, Section ~~15.7.4~~ 15.0.1.
4. USAR, Table 15.7-9.
5. 10 CFR ~~100.11~~ 50.67, "Accident Source Term."
6. 10 CFR 50.36(c)(2)(ii).
7. Regulatory Issue Summary (RIS) 2006-04, March 7, 2006

B 3.9 REFUELING OPERATIONS

B 3.9.7 Reactor Pressure Vessel (RPV) Water Level—New Fuel or Control Rods

BASES

BACKGROUND

The movement of new fuel assemblies or handling of control rods within the RPV when fuel assemblies seated within the reactor vessel are irradiated requires a minimum water level of 22 ft 3 inches above the top of irradiated fuel assemblies seated within the RPV. During refueling, this maintains a sufficient water level above the irradiated fuel. Sufficient water is necessary to retain iodine fission product activity in the water in the event of a fuel handling accident (Refs. 1 and 2). Sufficient iodine activity would be retained to limit offsite doses from the accident to ≤ ~~<~~ 25% of 10 CFR ~~100~~ 50.67 limits, as provided by the guidance of Reference 3.

APPLICABLE SAFETY ANALYSES

During movement of new fuel assemblies or handling of control rods over irradiated fuel assemblies, the water level in the RPV is an initial condition design parameter in the analysis of a fuel handling accident in containment postulated by Regulatory Guide ~~1.25~~ 1.183 (Ref. 1). ~~A minimum water level of 23 ft (Regulatory Position C.1.c of Ref. 1) allows a decontamination factor of 100 (Regulatory Position C.1.g of Ref. 1) to be used in the accident analysis for iodine. This relates to the assumption that 99% of the total iodine released from the pellet to cladding gap of all the dropped fuel assembly rods is retained by the refueling cavity water. The fuel pellet to cladding gap is assumed to contain 10% of the total fuel rod iodine inventory (Ref. 1), except I-131 which is 12% (Ref. 4).~~

Insert P

Analysis of the fuel handling accident inside containment is described in Reference 2. With a minimum water level of 22 ft 3 inches, ~~(a decontamination factor of 100 is still expected at a water level as low as 22 ft 3 inches) and~~ Assuming a minimum decay time of 24 hours prior to fuel handling, the analysis and test programs demonstrate that the iodine release due to a postulated fuel handling accident is adequately captured by the water, and that offsite doses are maintained within allowable limits (Ref. 5). The related assumptions include the worst case dropping of an irradiated fuel assembly onto the reactor core loaded with irradiated fuel assemblies.

the adjusted elemental iodine decontamination factor is 175.

(continued)

<u>INSERT P</u> (for TS Bases pages B 3.9.6-1 and B 3.9.7-1)

which states: "If the depth of water above the damaged fuel is 23 feet or greater, the decontamination factors for the elemental and organic species are 500 [subsequently corrected to 285 by RIS 2006-04 (Ref. 7)] and 1, respectively, giving an overall effective decontamination factor of 200 (i.e., 99.5% of the total iodine released from the damaged rods is retained by the water). This difference in decontamination factors for elemental (99.85%) and organic (0.15%) species results in the iodine above the water being composed of 57% elemental and 43% organic species. If the depth of water is not 23 feet, the decontamination factor will have to be determined on a case by case method." In accordance with Regulatory Guide 1.183, the fraction of fission product inventory in the fuel pellet to cladding gap is 10% for Kr-85, 8% for I-131, and 5% for other noble gases and halogens.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.9.7.1 (continued)

operations is met. Water at the required level limits the consequences of damaged fuel rods, which are postulated to result from a fuel handling accident in containment (Ref. 2).

The Frequency of 24 hours is based on engineering judgment and is considered adequate in view of the large volume of water and the normal procedural controls on valve positions, which make significant unplanned level changes unlikely.

REFERENCES

1. Regulatory Guide ~~1.25, March 23, 1972~~ 1.183, July 2000.
2. USAR, Section 15.7.4.
3. NUREG-0800, Section ~~15.7.4~~ 15.0.1.
4. USAR, Table 15.7-9.
5. 10 CFR ~~100.11~~ 50.67, "Accident Source Term."
6. 10 CFR 50.36(c)(2)(ii).
7. Regulatory Issue Summary (RIS) 2006-04, March 7, 2006

ATTACHMENT (4)

SUPPRESSION POOL pH CONTROL
IN THE EVENT OF A DESIGN BASIS LOCA

A4-1. INTRODUCTION

The alternative source term (AST) loss of coolant accident (LOCA) analysis takes credit for minimization of re-evolution of elemental iodine from the suppression pool. Re-evolution is strongly dependent on suppression pool pH. The analysis credits the pH buffering effect of sodium pentaborate solution introduced into the suppression pool post-LOCA by operation of the Standby Liquid Control (SLC) system to maintain the pH above 7.0. This pH satisfies the conditions for inhibiting the release of the chemical form of elemental iodine from the suppression pool water.

The purpose of this attachment is to (1) provide the details of the AST analysis for suppression pool pH control; (2) evaluate suitability of the SLC system to perform the post-LOCA suppression pool pH control function; and (3) address procedural guidance for post-LOCA injection of the sodium pentaborate solution using the SLC system.

A4-2. SUPPRESSION POOL POST-LOCA pH ANALYSIS

A4-2.1 Analysis Summary

Analyses have been performed to demonstrate that the pH of the suppression pool remains continuously above 7.0 following a LOCA for the 30-day duration of the accident. The impacts of severe accident management response actions are not considered in the analyses. A complete description of the analysis methods, assumptions, inputs, and results is provided in calculation H21C-097 (see Attachment 7).

Calculation H21C-097 determines the suppression pool pH values as a function of time without addition of the sodium pentaborate solution in the SLC system. The effect on the final pH of adding sodium pentaborate to the suppression pool water via the SLC system is subsequently determined to verify that the suppression pool water pH can be maintained above 7.0 based on Technical Specification (TS) requirements for the SLC system.

The suppression pool water pH is calculated using the methodology described in NUREG/CR-5950 (Reference A4-5.1) and as developed for the equivalent calculation done for Grand Gulf Nuclear Station (GGNS). The accuracy of translation of the equations in these documents into spreadsheet cell formulas for the Nine Mile Point Unit 2 (NMP2) calculation was verified by performing benchmarking calculations using the GGNS design input data. The benchmarking results, described in Section 5.8 of calculation H21C-097, demonstrate that the GGNS and NMP2 analyses yield very similar results.

The design inputs for the NMP2 calculations were conservatively established to maximize the post-LOCA production of acids and to minimize the post-LOCA production and/or addition of bases. Other design input values were selected to minimize the calculated pH. Significant inputs to the suppression pool pH analysis are provided in Table A4-1.

Calculation H21C-097 assumes that a total of 4,288 gallons of sodium pentaborate solution are added via the SLC system at a rate of 41.2 gpm to buffer the suppression pool pH, based on the minimum requirements of TS 3.1.7. SLC system flow mixing and transport of the sodium pentaborate solution to the suppression pool have been evaluated to determine the time at which manual initiation of the SLC system must occur to assure that the suppression pool pH remains above 7.0 throughout the duration of the accident.

A4-2.2 Results and Conclusions

The calculated post-accident suppression pool pH as a function of time after accident initiation is shown on Figure 4-1 of calculation H21C-097. Without addition of sodium pentaborate solution from the SLC system, the pH in the suppression pool falls below 7.0 between approximately 12 and 14 days. Therefore, injection of sodium pentaborate solution by the SLC system is required to prevent iodine re-evolution.

Calculation H21C-097 shows that addition of 4,288 gallons of sodium pentaborate solution via the SLC system buffers the suppression pool water and results in a final pH at 30 days of 8.3, thereby satisfying the conditions for inhibiting the release of iodine in the elemental form from the suppression pool water. As discussed in Section A4-2.4 below, injection, transport, and mixing of the sodium pentaborate solution takes less than 5 hours to complete. Thus, there is ample time after the start of the LOCA (approximately 11 days) to manually initiate the SLC system such that the suppression pool pH remains above 7.0. Manual initiation of the SLC system is expected early in a design basis LOCA as a result of emergency operating procedures and severe accident guidelines, particularly for events resulting in fuel damage that would be consistent with AST source terms.

A4-2.4 Analysis of Transport and Mixing of the Injected Sodium Pentaborate Solution

As described in NMP2 Updated Safety Analysis Report (USAR) Section 9.3.5, the sodium pentaborate solution is pumped into the high pressure core spray (HPCS) line downstream of the inboard containment isolation valve and is discharged radially over the top of the core through the HPCS sparger.

Detailed discussion of mixing and transport of the sodium pentaborate solution to the suppression pool is provided in Attachment (5), replacing in its entirety the discussion that appears in Calculation H21C-097 under Design Input 4.14. Schematic diagrams illustrating the flow paths within the reactor vessel and from the vessel to the suppression pool are provided in Attachment (5). As determined in Attachment (5), the maximum total time for the injection, transport, and mixing of 4,288 gallons of sodium pentaborate solution in the suppression pool is less than 5 hours. Thus, in order to remain within the 12 to 14 day time calculated for the suppression pool pH to drop below 7.0 without buffering, the SLC system must be initiated no later than approximately 11 days after the onset of the accident.

A sensitivity analysis was performed to determine the minimum quantity of sodium pentaborate solution required to achieve a pH of 7.0. The sodium pentaborate solution volume in the calculation spreadsheets used in Calculation H21C-097 was adjusted to reach a final (30-day) pH value of 7.0, and the volume of solution remaining in the SLC system piping and in the HPCS system piping and spargers after injection ceases was also taken into account. The minimum required sodium pentaborate solution volume was calculated to be about 200 gallons. This value is less than 5% of the minimum TS-required volume of 4,288 gallons and provides sufficient margin to account for any potential sodium pentaborate hold-up or hideout not accounted for in the evaluation.

Based on the above, adequate transport of the sodium pentaborate to the suppression pool as well as suppression pool recirculation mixing will occur prior to the time that credit is needed for the buffering effect of the sodium pentaborate for pH control.

A4-3. EVALUATION OF SUITABILITY OF THE SLC SYSTEM TO PERFORM THE POST-LOCA pH CONTROL FUNCTION

A4-3.1 SLC System Design Description

The NMP2 SLC system is described in USAR Section 9.3.5 and is required to be operable in accordance with TS 3.1.7. The SLC system consists of an ambient pressure boron solution storage tank with immersion heater for low-temperature sodium pentaborate solution storage, two high-pressure positive displacement pumps for injecting the sodium pentaborate solution into the reactor core, two motor-operated pump suction valves, two explosive-actuated injection valves for isolating the sodium pentaborate solution from the reactor until required, two injection line parallel motor-operated stop-check valves, an injection line isolation check valve , a test water tank, additional valves, piping, and associated instrumentation. A piping and instrumentation diagram for the SLC system is shown on USAR Figure 9.3-17. A simplified diagram of the SLC system is shown on Figure A4-1 of this attachment.

The two positive displacement pumps (one in standby) take suction from the storage tank and discharge through the two explosive-actuated valves connected in parallel to a common discharge header. The sodium pentaborate solution is pumped into the high pressure core spray (HPCS) line downstream of the inboard containment isolation valve and is discharged radially over the top of the core through the HPCS sparger. The pumps are each designed to deliver a minimum of 41.2 gpm of sodium pentaborate solution to the reactor.

A4-3.2 SLC System Design Criteria and Applicable Program Requirements

All of the SLC system components required for the injection of sodium pentaborate solution into the reactor are classified as safety related. Commensurate with the high degree of reliability required for safety-related service, the SLC system equipment and components are designed, tested, and maintained in accordance with the governing design criteria and program requirements outlined below.

A4-3.2.1 Seismic Qualification

All of the SLC system components required for the injection of sodium pentaborate solution into the reactor are classified as seismic Category I and are designed to withstand specified earthquake loadings, as described in USAR Sections 3.7 and 3.9.

A4-3.2.2 AC Power

All electrical components required for operation of the SLC system are classified as electrical Class 1E and are powered from Class 1E emergency power sources that are backed up by the emergency diesel generators upon a loss of offsite power. The common storage tank heater is powered from a source that can be connected to an emergency diesel generator in the event of a loss of offsite power.

A4-3.2.3 Inservice Inspection (ISI) and Inservice Testing (IST) Programs

The applicable components of the SLC system are inspected and tested in accordance with the NMP2 ASME Boiler and Pressure Vessel Code ISI and IST programs as required by 10 CFR 50.55a.

A4-3.2.4 Maintenance Rule Program

The SLC system is included in the scope of the NMP2 Maintenance Rule program consistent with 10 CFR 50.65.

A4-3.2.5 Environmental Qualification

The electrical components required to function to perform the post-LOCA SLC system sodium pentaborate injection safety function are the SLC pumps, pump suction valves, explosive-actuated injection valves, and the electrical power supplies and associated controls supporting them. These SLC system electrical components are currently included in the NMP2 environmental qualification (EQ) program and are qualified for conditions associated with the current functions of the SLC system. Environmental qualification of these SLC system components for the post-LOCA environment associated with the new suppression pool pH control function will be established in accordance with the station design change process prior to completing implementation of the AST license amendment.

A4-3.3 SLC System Reliability

The SLC system has suitable redundancy in components and features to assure that for onsite or offsite power operation, its safety function of injecting sodium pentaborate solution into the reactor for the purpose of suppression pool pH control can be accomplished. The SLC system is composed of two separate, 100% capacity subsystems that are each separately capable of performing the suppression pool pH control safety function. Each subsystem is manually initiated from the main control room by turning its own keylocked switch. The system is also initiated automatically by a signal from the redundant reactivity control system (RRCS). The common liquid poison storage tank and the common sodium pentaborate injection line are passive components that are not subject to active failures. However, there is a check valve in the common injection line that is not redundant. This limited lack of redundancy is offset by an evaluation provided in Section A4-3.3.1 below demonstrating that the check valve has acceptable quality and reliability.

A4-3.3.1 Injection Line Check Valve

The common SLC system injection line contains two parallel motor-operated stop-check valves (2SLS*MOV5A and 2SLS*MOV5B) that are in series with an injection check valve (2SLS*V10). These are containment isolation valves, with the two stop-check valves located outside primary containment and the check valve located inside primary containment. A complete failure of one of the stop-check valves to open would not prevent the SLC system from performing its safety function. Only a failure of the injection check valve (2SLS*V10) to open could prevent the system from performing its safety function.

The injection check valve is a nominal 2-inch valve manufactured by the Velan Engineering Co., Velan Valve Corp.; model number B08-3036Z-14MS. The existing valve was procured as part of the original system supply in accordance with the quality requirements applicable to safety related equipment at the time of purchase. It is a mechanical component with no non-metallic parts. As such, environmental qualification in accordance with 10 CFR 50.49 is not required. The valve is classified as seismic Category I as discussed in Section A4-3.2.1 above.

A search of the Equipment Performance Information and Exchange System (EPIX) and Nuclear Plant Reliability Data System (NPRDS) databases identified no failures of any common injection line check valves to open. The only identified failure to open of any BWR liquid poison system check valve occurred on a pump discharge check valve (upstream of the explosive valve on the redundant pump trains). The valve manufacturer was not the same as for NMP2, and the valve failed partially open. One failure of a Velan check valve was also identified, but the model number was not the same as the NMP2 SLC system injection check valve. This valve was also identified as being partially open.

Pump flow through the check valve is verified every 24 months on a staggered test basis using the system pumps, explosive valves, and demineralized water, as required by TS 3.1.7. No failures of this surveillance test have been identified. Satisfactory operation of the check valve is verified by pump discharge pressure greater than reactor pressure and flow greater than or equal to the required value. The maximum pressure drop for the existing system safety function (Anticipated Transients Without Scram (ATWS) mitigation) would be approximately 300 psid (difference between SLC pump discharge relief valve setpoint and reactor pressure). For the post-LOCA pH control function, the maximum pressure drop could approach 1,400 psid due to the conditions in the vessel (i.e., faulted line followed by rapid depressurization of the reactor). The high differential pressure increases the likelihood that the check valve will open under LOCA conditions.

Based on the check valve design attributes, EPIX and NPRDS reviews, NMP2 performance history, and the high differential pressures expected under post-accident conditions, the injection line check valve is of acceptable quality and reliability.

A4-4. PROCEDURAL GUIDANCE FOR POST-LOCA INJECTION OF THE SODIUM PENTABORATE SOLUTION USING THE SLC SYSTEM

The SLC system will be credited for limiting radiological dose following a LOCA involving fuel damage in accordance with the AST analyses for suppression pool pH control. The AST analysis provides for SLC system actuation by no later than approximately 11 days following accident initiation and completion of injection of an adequate volume and content of sodium pentaborate solution within 5 hours (see Attachment 5), which will ensure the suppression pool pH remains at or above 7.0 for 30 days.

Initiation of the SLC system is accomplished from the main control room with simple keylock switch manipulation. Actuation of the keylock switch (one for each SLC subsystem) is the only action necessary to initiate injection of the sodium pentaborate solution into the reactor vessel. The new SLC system function to control suppression pool pH does not involve any change to the actions needed to be performed to initiate SLC system injection. Operators are familiar with operation of the SLC system due to previous training for ATWS events and loss of emergency core cooling capability scenarios. Indications of proper SLC system operation are provided in the control room as described in USAR Sections 9.3.5 and 7.4.

Plant emergency operating procedures (EOPs) presently provide instructions to initiate the SLC system as well as other sources\of water for emergency core cooling. Specifically, procedure N2-EOP-RPV, "RPV Control," is entered with reactor pressure vessel (RPV) water level below the scram setpoint, RPV pressure above the high pressure scram setpoint, or drywell pressure above the scram setpoint. These EOP entry conditions are indicative of a plant condition that could degrade to imminent or actual core damage. The RPV low level and drywell high pressure entry conditions ensure that N2-EOP-RPV is entered for a LOCA. When conditions defined in the EOPs indicate that adequate core cooling cannot be restored and maintained, for any reason, then entry into the Severe Accident Procedures (SAPs) is directed. Specifically, procedure N2-SAP-2, "RPV, Containment, and Radioactivity Release Control," requires SLC system injection to prevent core re-criticality, regardless of whether or not an ATWS condition exists.

In addition, a core damage event large enough to release substantial quantities of fission products into the drywell will result in high drywell radiation alarms. Two drywell high range radiation monitors provide independent and redundant indication, recording, and alarm functions in the control room and are used in the Nine Mile Point Site Emergency Plan implementing procedures to estimate core damage; thus, their use in an AST capacity is consistent with current use. Prior to completing implementation of the AST license amendment, the EOPs and SAPs will be revised, as appropriate, to reflect the post-LOCA function of the SLC system, include instructions to manually actuate the SLC system based on high drywell radiation levels, and assure that, once initiated, the entire contents of the SLC system storage tank are injected to accomplish the pH control function.

The reactor water level, reactor pressure, and drywell pressure instruments used to measure conditions for EOP and SAP entry meet the quality requirements for a Type A, Category I variable as defined in Regulatory Guide 1.97 (Reference A4-5.2), and as discussed in NMP2 USAR Section 7.5.2. The drywell high range radiation monitors meet the quality requirements for a Type E, Category I variable. This monitoring instrumentation is required to be operable by TS 3.3.3.1, "Post Accident Monitoring (PAM) Instrumentation."

Procedure N2-EOP-C5, "Failure to Scram," currently calls for termination of SLC system injection as a reactivity control measure if an ATWS event was in progress and it was subsequently determined that the reactor would remain shutdown without SLC system injection. Since the AST LOCA scenario does not assume that an ATWS event has occurred, this EOP does not require revision.

Licensed operators and shift technical advisors (STAs) have received initial training on the EOPs and SAPs, and will continue to receive periodic refresher training. Additionally, prior to completing implementation of the AST license amendment, training will be provided to licensed operators and STAs for the procedure revisions that specifically address sodium pentaborate solution injection for pH control following a LOCA.

The procedures that will implement SLC system injection of the sodium pentaborate solution for post-LOCA suppression pool pH control are controlled procedures that are prepared, reviewed, and approved in accordance with the Quality Assurance program.

A4-5. REFERENCES

1. NUREG/CR-5950, "Iodine Evolution and pH Control," December 1992

2. Regulatory Guide 1.97, "Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant and Environs Conditions During and Following an Accident," Revision 3, May 1983

Table A4-1
Suppression Pool pH Control Inputs

Input/Assumption	Value
Suppression Pool Liquid Volume (Maximum)	154,400 ft^3
Reactor Coolant System Inventory Excluding Suppression Pool	669,175 lbm
Volume of 9.423% Sodium Pentaborate ($Na_2O*5B_2O_3*10H_2O$) Solution	4558.6 gallons (13.6 weight %) 4,288 gallons (14.4 weight %)
Initial Suppression Pool pH	5.3
Inventory of Chloride Bearing Cable in Primary Containment	See Calculation H21C-097, Attachment 3 and Table 4-4

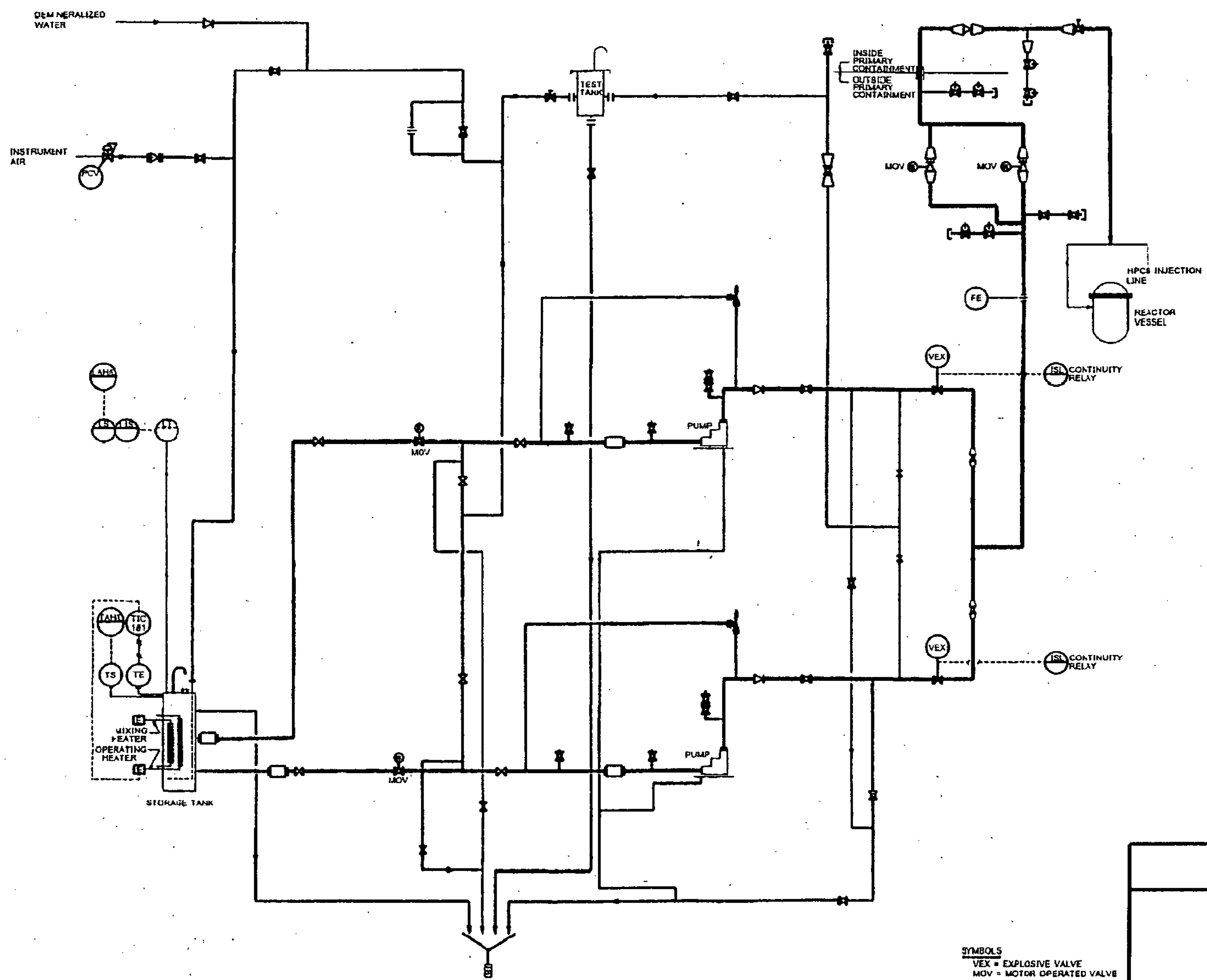


FIGURE A4-1

SLC SYSTEM CONFIGURATION

ATTACHMENT (5)

EVALUATION OF SLC SYSTEM INJECTION FLOW TRANSPORT AND MIXING

A5-1. INTRODUCTION

This attachment provides an evaluation of the transport and mixing of the sodium pentaborate solution through the reactor and containment systems to demonstrate that the amount and timing of the sodium pentaborate solution injected by the Standby Liquid Control (SLC) system supports the conclusion of the suppression pool post-LOCA pH evaluation (i.e., that the suppression pool pH is maintained greater than or equal to 7.0 for the 30-day duration of the design basis LOCA).

A5-2. SUPPRESSION POOL MIXING

Calculation H21C-097, Design Input 4.14, states the following:

> *"The limiting Design Basis Accident (DBA) LOCA is identified in UFSAR Section 6.2.1.1.5 (Ref. 7.11.1) as Case C of UFSAR Section 6.2.1.1.3 (Ref. 7.11.2), which corresponds to Case C of Reference 7.6.5. For this case, a minimum of one Low Pressure Core Injection (LPCI) pump is operable throughout the accident (Ref. 7.6.5, p. 21). Given that the reactor vessel depressurizes reasonably quickly for a large break LOCA (see Ref. 7.6.5, Tables 6.2-9 and 6.2-10), a minimum LPCI flow rate of 6,000 to 7,000 gpm can be expected per Figure 6.2-3 of Reference 7.6.5. This flow rate equates to approximately 0.3 complete exchanges of the water in the suppression pool per hour (1 complete exchange in approximately 3 hours)."*

This design input identifies the low pressure coolant injection (LPCI) flow as between 6,000 and 7,000 gpm, based on one LPCI pump operating throughout the accident. For this evaluation, only the flow available after thirty minutes following the event is of interest since the SLC system is not assumed to be initiated prior to 30 minutes. From 30 minutes after the LOCA until the end of the event, the high pressure core spray (HPCS) pump, one LPCI pump for injection, and one LPCI (residual heat removal) pump for containment spray or suppression pool cooling will be operating. All three pumps would contribute to suppression pool mixing, but only the HPCS pump and one LPCI pump would provide flow through the downcomers. Pump flows are obtained from the Updated Safety Analysis Report (USAR), Table 6.3-1. The table indicates that the flows are based on the differential pressure between the reactor and the containment. For a large-break LOCA, the reactor pressure drops to a value of approximately 20 psia in less than 100 seconds. Therefore, the reactor pressure will be approximately the same as the containment pressure. For a 0-psig differential pressure, USAR Table 6.3-1 indicates that the total pump flow with one HPCS pump and one LPCI pump operating will be 12,910 gpm. Using this value instead of the 6,000 to 7,000 gpm value used in Calculation H21C-097 results in approximately 0.6 complete exchanges of the water in the suppression pool per hour, which is judged to provide adequate mixing of the suppression pool.

A5-3. FLOW PATH EVALUATION

A5-3.1 Flow Inside the Reactor Vessel

Figure A5-1 is a simplified diagram of the flow path for the HPCS, LPCI, and SLC systems inside the reactor vessel. The HPCS flow enters the reactor vessel through spray headers (spargers) located just above the reactor core, and the LPCI flow enters the inside of the core shroud at the top of the core. The SLC system sodium pentaborate solution enters the vessel through the HPCS spargers, as described in USAR Section 9.3.5. USAR Figure 6.3-15 indicates that a post-LOCA steady state level of approximately 35 feet above the vessel bottom (4.5 feet above top of active fuel) is attained. The HPCS/LPCI/SLC solution mixture flows along/through the fuel rods, channels, fuel support pieces, core bypass regions, and the control rod and nuclear instrument guide tubes to the lower plenum of the vessel. From the lower plenum the mixture flows upward through the jet pump mixer/diffuser sections and into the vessel downcomer region. The water level in the vessel downcomer region depends on the recirculation line break location. The maximum water level will occur for a break at or near the recirculation inlet lines. Based on reactor vessel water volume information provided in USAR Table 6.2-9 and Figure 5.1-1b, the volume of water in the vessel is approximately 11,835 ft^3. This value is conservatively high, since the value used for the downcomer region includes the volume up to normal reactor water level.

SLC system flow enters the vessel through the HPCS sparger. Since the sparger is below the post-LOCA water level, the SLC system flow will be entrained with the emergency core cooling flow moving downward through the core. This entrainment will ensure adequate mixing of the solution with the emergency core cooling flow, even if the HPCS system is not operating.

The only potential holdup volume in the vessel is the unbroken recirculation loop. This volume is filled with water before SLC system injection begins and it is not associated with the active flow path. Thus, diversion/entrapment of significant amounts of sodium pentaborate solution during the approximate 2-hour injection period (Section A5-3.4 below) is not expected.

A5-3.2 <u>Flow Inside the Drywell</u>

The combined SLC/emergency core cooling flows will flow out the outlet nozzle of the broken recirculation loop and spill to the drywell floor (see Figure A5-2). At the time of SLC system injection, a pool of water will already exist on the floor. The level of the drywell pool will be at the top of the downcomer pipes which extend above the drywell floor and lead to the suppression pool. The volume of the drywell pool is 2,320 ft^3. The only holdup volumes in the drywell are the equipment and floor drain sumps. These sumps are recessed into the drywell floor and are covered with steel plate, preventing free communication with the flow through the pool on the drywell floor. Since these volumes are filled with water before SLC system injection begins and they are not associated with the active flow path, diversion/entrapment of significant amounts of sodium pentaborate solution during the injection and transport period is not expected.

A5-3.3 <u>Flow Into the Suppression Pool</u>

The eight (8) downcomers internal to the vessel pedestal area will not experience flow, since the openings in the pedestal support are above the drywell pool elevation. Flow from the remaining 113 downcomers travels vertically downward to the suppression pool, resulting in no additional holdup volumes. The 113 downcomers are uniformly spaced around the drywell/suppression pool.

A5-3.4 <u>Transport Time</u>

As the SLC system injects the sodium pentaborate solution, the solution will mix with the emergency core cooling flow and travel through the reactor vessel and drywell to the suppression pool. Each of the volumes identified above will tend to hold up the solution and delay its transport to the suppression pool.

The first holdup volume is the reactor vessel. The volume of liquid remaining in the vessel was estimated to be 11,835 ft^3, or 88,538 gallons (including the volume of the active (broken) recirculation loop). Considering only the flow from the HPCS pump and one LPCI pump (12,910 gpm total), the vessel volume will undergo one complete exchange every 6.9 minutes. Assuming that SLC system injection stops after 4,558.6 gallons of sodium pentaborate solution are injected, the SLC system injection takes

111 minutes (4558.6 gallons / 41.2 gpm = 111 minutes), or 1.85 hours. The larger minimum SLC volume of 4558.6 gallons specified on TS Figure 3.1.7-1 is used to maximize the injection time.

By the end of SLC system injection, the volume in the vessel has undergone 16.1 exchanges and is expected to have reached an equilibrium concentration determined by the SLC system to emergency core cooling flow ratio. The percent of sodium pentaborate solution remaining in the transport pool is approximated by the relationship

$$P = (1/2^N) \bullet 100$$

where P = percent of sodium pentaborate remaining in the transport pool
N = number of complete transport pool volume exchanges

Using this relationship, after seven additional pool exchanges less than 1% of the remaining injected sodium pentaborate will remain in the reactor. Therefore, after another 48 minutes, essentially all of the sodium pentaborate solution will have been cleared from the vessel and moved to the pool on the drywell floor. The total elapsed time for this phase of the transport is 159.3 minutes, or 2.7 hours (injection plus clearing time).

The volume of the pool on the drywell floor is calculated to be 17,356 gallons (2,320 ft^3), and undergoes one complete exchange every 1.4 minutes. The start of this phase occurs after all of the sodium pentaborate solution has entered the drywell pool. By this time, however, the drywell pool has undergone 119 complete exchanges, and most of the solution has already moved into the suppression pool. Seven drywell pool exchanges will move more than 99% of the remaining solution to the suppression pool. The time to accomplish 7 exchanges is less than 10 minutes.

The final volume is the suppression pool. Since the sodium pentaborate solution is introduced uniformly within the suppression pool through the downcomer pipes, one complete exchange of the pool volume is judged to be sufficient to ensure adequate initial mixing. Assuming that the HPCS pump and one LPCI pump are providing flow through the downcomers, one complete exchange will be accomplished in approximately 1.6 hours (0.6 exchanges per hour). This exchange rate is judged to be sufficient to maintain mixing of the pool, as discussed in Section A5-2 above.

The sum of these time periods is the total time required to transport the sodium pentaborate to the suppression pool and ensure adequate mixing. The total time is 4.4 hours, which is well within the 12 to 14 days calculated for the pool pH to drop below 7.0 without buffering.

A5-4. SENSITIVITY ANALYSIS – INJECTED VOLUME OF SODIUM PENTABORATE SOLUTION

The preceding evaluation was based on the injection, mixing, and transport of 4,558.6 gallons of sodium pentaborate solution from the SLC system storage tank to the suppression pool. As discussed in Attachment (4), Section A4-2, the post-LOCA suppression pool pH following addition of 4,288 gallons of 14.4 weight percent sodium pentaborate solution has been calculated to be 8.3 (reference Calculation H21C-097). The injection, mixing, and transport of 4,288 gallons of sodium pentaborate solution would take slightly less total time than that for 4,558.6 gallons.

The acceptance criterion, however, is that the suppression pool pH remain greater than or equal to 7.0. A sensitivity analysis was performed to determine the minimum quantity of sodium pentaborate solution required to achieve a pH of 7.0. Using the minimum concentration (13.6 weight percent) and the resulting specific gravity (1.068), the solution volume in the spreadsheets used in Calculation H21C-097 was adjusted to reach a final (30-day) pH value of 7.0. The minimum required solution volume is calculated to be 117.1 gallons. Also considered is the volume of sodium pentaborate solution that would remain in the SLC system piping and in the HPCS system piping and spargers (assuming the HPCS system is not operating) and thus would not be injected into the reactor vessel. This volume is estimated to be 82.6 gallons. Therefore, the total minimum quantity of solution required to achieve a pH of 7.0 is about 200 gallons, which is less than 5% of the minimum TS-required volume of 4,288 gallons and provides sufficient margin to account for any potential sodium pentaborate hold-up or hideout not accounted for in this evaluation.

Note that the reduction in required injection volume reduces the total injection time by approximately 106 minutes but has no significant impact on transport time. Thus, the total time for the injection, mixing, and transport of 200 gallons of sodium pentaborate solution would be reduced by at least 1.8 hours, to approximately 2.6 hours.

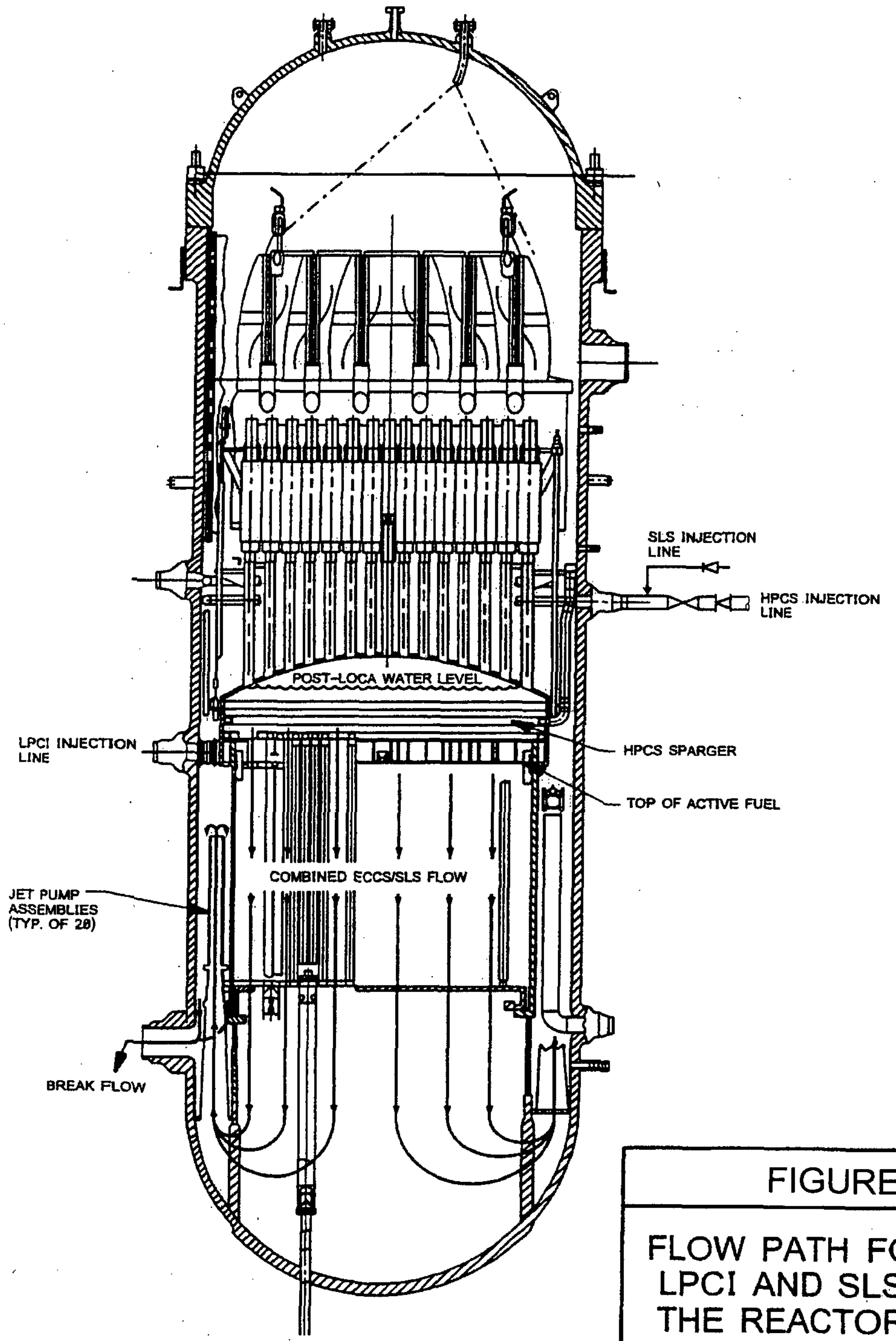


FIGURE A5-1

FLOW PATH FOR HPCS, LPCI AND SLS INSIDE THE REACTOR VESSEL

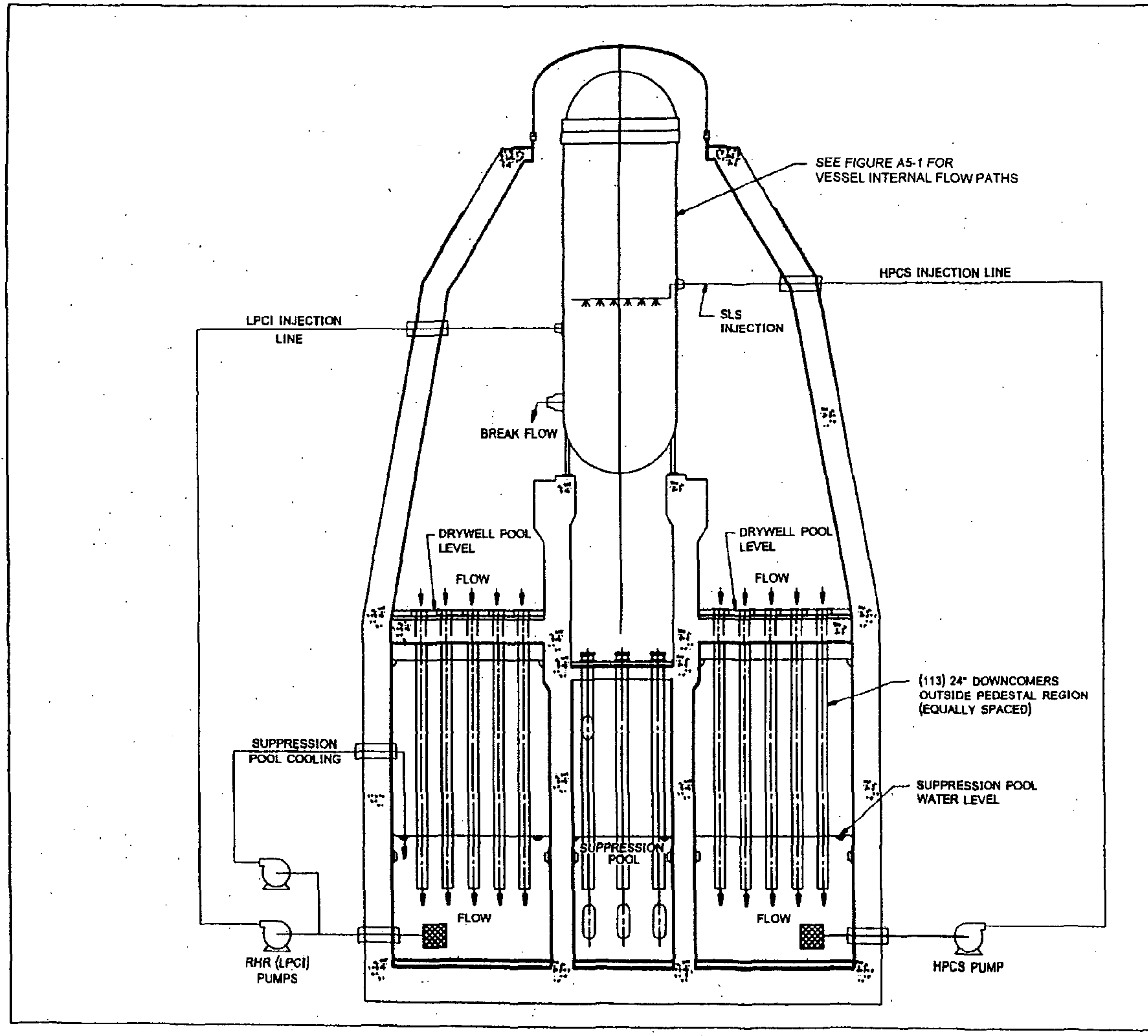


FIGURE A5-2

MIXED FLOW RETURN TO THE SUPPRESSION POOL

ATTACHMENT (6)

CALCULATION OF NEW ATMOSPHERIC DISPERSION FACTORS

A6-1. INTRODUCTION

New atmospheric dispersion factors (X/Q values) are calculated for use in evaluating the radiological consequences of the Nine Mile Point Unit 2 (NMP2) design basis accidents. These calculations use meteorological data collected by the Nine Mile Point onsite meteorological measurements program for the five-year period from 1997 through 2001.This attachment provides information regarding the onsite meteorological measurement program and the X/Q calculation methodology, and summarizes the results of the calculations.

A6-2. ONSITE METEOROLOGICAL MEASUREMENTS PROGRAM

The Nine Mile Point Nuclear Station (NMPNS) meteorological measurement program is described in NMP2 Updated Safety Analysis Report (USAR) Section 2.3.3.2. The program meets the intent and recommendations of Regulatory Guide (RG) 1.23 (Reference A6-5.1) and NUREG-0654 (Reference A6-5.2) for the operational measurements program. The program consists of monitoring wind speed, wind direction, ambient temperature, and precipitation. The operability of the meteorological monitoring instrumentation ensures that sufficient meteorological data are available for estimating potential radiation doses to the public as a result of routine or accidental release of radioactive materials to the atmosphere. This capability is required to evaluate the need for initiating protective measures to protect the health and safety of the public.

The primary meteorological tower is of steel, open-lattice construction and is located approximately 0.5 miles west-southwest of the station near the shore of Lake Ontario. The primary tower is 61-m (200-ft) high and is instrumented with wind direction and speed sensors at three levels: 9 m (30 ft), 30 m (100 ft), and 61 m (200 ft). Sigma theta is derived for each of the three wind levels. Ambient temperature is measured at the 9 m (30 ft) level, and temperature differences are determined between the 9 m (30 ft) and 61 m (200 ft) levels. This is the primary method used to determine atmospheric stability. Dew point temperature is obtained at the 9 m (30 ft) level. Near the base of the tower, precipitation and barometric pressure are also measured. The primary tower is located in terrain that is characteristic of the area and at approximately the same elevation as finished plant grade. The terrain is predominately flat throughout the area and in the vicinity of the tower.

The backup wind direction and speed instrumentation is located east of the J. A. FitzPatrick plant on a 27 m (90 ft) utility pole. Data collected coincidentally from the primary tower and backup tower over the same three-year period have been analyzed. Based upon this analysis and an earlier study by Meteorological Environmental Services, Inc., the backup tower measurements are in general agreement with the primary tower and are adequate for use during emergency situations.

Meteorological instrumentation calibration schedules are specified to conform to RG 1.23 recommendations. Meters and other equipment used in calibrations are, in turn, calibrated at scheduled intervals. Inspection and maintenance of equipment is accomplished in accordance with procedures in the instrument manufacturer's manuals. Inspection is implemented by qualified technicians who are capable of performing the maintenance, if required.

Digital data processing at each meteorological tower is accomplished by a remote data acquisition system (RDAS) computer. These RDAS computers sample each sensor's analog processor at a rate of once per second and process the data into 1-, 15-, and 60-min averages. Averaged data are transmitted via modem to a central processing system (CPS) computer for access and storage. Each RDAS computer is housed in an environmentally-controlled instrument cabinet at the meteorological towers. The CPS computer is housed in an environmentally-controlled meteorological computer building. Better than 90 percent data recovery is attained from each measuring and recording system.

A6-3. ATMOSPHERIC DISPERSION FACTOR CALCULATIONS

Meteorological data utilized for calculation of new atmospheric dispersion coefficients (X/Qs) were selected from the historical record of the NMPNS meteorological monitoring program. The period 1997-2001 was selected because it represents a complete and accurate data set that is representative of the site meteorological data. The data was reviewed to ensure instrumentation problems and missing or anomalous observations did not affect the validity of the data. This is consistent with the guidance in RG 1.194 (Reference A6-5.3) that considers five years of hourly observations to be representative of long-term trends.

Recorded meteorological hourly average data were used to generate joint frequency distributions of wind direction, wind speed, and atmospheric stability class, in accordance with the RG 1.23 and RG 1.145 (Reference A6-5.4). Wind roses and joint frequency distributions were reviewed for meteorological and climatological reasonableness and found to be acceptable prior to use. A review was also conducted on specific hourly data prior to the execution of the atmospheric calculations in the PAVAN and ARCON96 computer programs. This consisted of manual spot checks of the spreadsheet reformatted data in comparison with the raw data.

The five possible NMP2 locations where accident radionuclide releases are assumed to occur are the radwaste/reactor building vent, the main steam tunnel, the standby gas treatment building, the post-accident sampling system (PASS) panel, and the main stack. Information regarding these release points and their proximity to receptor locations is provided in Tables A6-1 and A6-2. Figure A6-1 is a site plan showing the relative locations of the release points and receptors. In addition, in order to evaluate the impact of an accident at Nine Mile Point Unit 1 (NMP1) on the NMP2 control room, information for the three NMP1 release points (the reactor building blowout panel, the turbine building blowout panel, and the main stack) is also provided.

A6-3.1 Control Room and Technical Support Center (Excluding Main Steam Line Break)

Control room and Technical Support Center (TSC) X/Q values were calculated using ARCON96 for various source/receptor scenarios using the guidance contained in RG 1.194. The scenarios were analyzed using the hourly-averaged meteorological joint wind and stability database for the five-year period from 1997 through 2001. All of the assumed release points listed in Tables A6-1 and A6-2, except the NMP2 main stack, were modeled as ground-level (vent) releases in accordance with RG 1.145 because their height is less than 2.5 times the highest adjacent structure. Conservative building wake areas, calculated considering the complexity of the geometry of the NMP1 and NMP2 structures, were input into ARCON96 to account for wake effects.

A6-3.2 Offsite – Exclusion Area Boundary (EAB) and Low Population Zone (LPZ)

The computer program PAVAN is used to determine X/Q values used in the assessment of dose consequences of design basis accidents in nuclear power stations. PAVAN is a straight line Gaussian dispersion model. The program implements the NRC guidance provided in RG 1.145. Utilizing joint frequency of occurrence distributions of wind direction, wind speed, and Pasquill atmospheric stability class, PAVAN calculates X/Q values as a function of direction for various time-averaging periods at the EAB and the outer boundary of the LPZ. Calculations are made from assumed ground-level (i.e. nonelevated) releases (such as vents and building penetrations), which are less than 2.5 times the height of adjacent solid structures, and from elevated releases (i.e. stacks). Three procedures are utilized for calculating X/Q: a direction-dependent approach, a direction-independent approach, and an overall site X/Q approach.

The PAVAN model contains certain model options for executing the program. The following table summarizes the options invoked for the EAB and LPZ X/Q calculations for NMP2.

Option No.	Description	Option Invoked?	
		Main Stack Release	Blowout Panel Release
1	Calculate σ_y and σ_x based on desert diffusion.	No	No
2	X/Q values include evaluation for no building wake.	No	No
3	ENVLOP calculations printed which describe upper envelope curve.	No	No
4	Print points used in upper envelope curve and calculation.	No	No
5	Null	---	---
6	Joint frequency distribution in % frequency format	No	No
7	Print X/Q calculation details	Yes	Yes
8	Distribute calm winds observations into first wind speed category.	No	No
9	Use site-specific terrain adjustment factors for the annual average calculations.	Yes*	No
10	Assume a default terrain adjustment factor for the annual average calculations. Option 10 is applied, which together with application of Option 9 means that site specific terrain factors will be used.	Yes	Yes

* Since there are no severe terrain features, such as deep valleys or mountains, in the vicinity of NMPNS to affect the diffusion of radionuclides from the evaluated main stack, the default terrain adjustment factors (TAF-1) were applied.

The radwaste/reactor building vent, the main steam tunnel, the standby gas treatment building, the PASS panel, and the main stack are the assumed accident release points. Only the NMP2 main stack qualifies as an elevated release per RG 1.145. Therefore, the other four NMP2 release points were executed by PAVAN as ground type releases. The NMP2 main stack was executed as an elevated release. Source-to-receptor horizontal distances are: 1,615 m (5,299 ft) from the NMP2 main stack to the EAB; 1,381 m (4,531 ft) from the radwaste/reactor building vent to the EAB; and 6,116 m (20,060 ft) to the LPZ. Due to the close proximity of the ground level release points, identical distances to the EAB and LPZ were used.

NMPNS meteorological data from the five-year period from 1997 through 2001 was used in the PAVAN analysis. The format of PAVAN meteorological input consists of a joint wind direction (based on sixteen 22.5 degree sectors), wind speed (12 intervals), and stability class (7 classes) occurrence frequency distribution. Since the NMPNS meteorological data fails to provide a maximum wind speed for category 12 winds, a conservative value of 60.5 m/s was selected. Maximum wind speed is required input for each wind speed category in PAVAN. The coastal sectors were not considered in determining the X/Q values for the EAB and LPZ.

A6-3.3 Control Room – Main Steam Line Break (MSLB) Puff Release

The MSLB accident evaluation utilizes an instantaneous "puff" release X/Q. The puff release is modeled in accordance with RG 1.194, Section C.5, with the following assumed site meteorological conditions:

- Wind Speed: 1 meter/second toward the receptor; and
- Stability Class F.

The horizontal distance from the main steam tunnel (the assumed MSLB release point) to the NMP2 control room intakes ranges from 63.67 m (209 ft) to 73.37 m (241 ft), as shown in Table A6-2. The X/Q is calculated for the closest intake. It takes approximately 124 seconds for the puff to pass completely over the NMP2 control room air intake. Following initiation of the Control Room Envelope Filtration (CREF) system, the control room outside air intake flow rate varies with time as the control room ventilation system transitions from the normal to emergency mode of operation. As such, the control room outside air intake flow rate is modeled parametrically, and the highest X/Q value is used in the dose analysis.

A6-4. SUMMARY OF RESULTS

The X/Q values resulting from the ARCON96 modeling analysis of each release point and meteorological database scenario for the required time intervals are shown in Table A6-3 for the NMP2 control room, Table A6-4 for the TSC, and Table A6-5 for the NMP1 control room.

The X/Q values for the EAB and LPZ calculated by the PAVAN modeling analysis of each release scenario are presented in Tables A6-6 and A6-7 for each of the time intervals required by RG 1.145.

For the MSLB instantaneous puff release, the integrated X/Q value calculated for the NMP2 control room air intake is 1.204E-04 sec/m^3.

All input files for ARCON96 and PAVAN, including the meteorological data input files, are provided in Calculation H21C076 (see Attachment 7).

A6-5. REFERENCES

1. Regulatory Guide 1.23 (Safety Guide 23), "Onsite Meteorological Programs," February 1972

2. NUREG-0654, "Criteria for Preparation and Evaluation of Radiological Emergency Response Plans and Preparedness in Support of Nuclear Power Plants," Revision 1, November 1980

3. Regulatory Guide 1.194, "Atmospheric Relative Concentrations for Control Room Radiological Habitability Assessments at Nuclear Power Plants," June 2003

4. Regulatory Guide 1.145, "Atmospheric Dispersion Models for Potential Accident Consequence Assessments at Nuclear Power Plants," Revision 1, November 1982

Table A6-1
Release/Intake Elevations

Point of Interest	Elevation (ft)	Elevation (m)
NMP2 Main Stack	429	130.8
NMP2 Radwaste/Reactor Building Vent	187	57
NMP2 Main Steam Tunnel	45.08	13.7
NMP2 Standby Gas Treatment Building	23.5	7.2
NMP2 PASS Panel	82	25
NMP2 Control Room Intake West-High	36	11
NMP2 Control Room Intake West-Low	15.5	4.7
NMP2 Control Room Intake East-High	52.75	16.1
NMP2 Control Room Intake East-Low	19	5.8
NMP1 Main Stack	350	106.7
NMP1 Reactor Building Blowout Panel (relative to bottom of panel)	78.9	24
NMP1 Turbine Building Blowout Panel (relative to bottom of panel)	72.4	22.1
NMP1 Control Room Intake (height equal to roof elevation)	72	21.95
Technical Support Center Intake	21	6.4

Table A6-2
Release/Intake Distances and Directions

Release/Intake	Horizontal Distance (ft)	Horizontal Distance (m)	Sector Bearing Relative to True North
NMP2 Releases to the NMP2 Control Room (CR)			
NMP2 Main Stack / NMP2 CR West-High	937	286	225° SW
NMP2 Main Stack / NMP2 CR West-Low	919	280	225° SW
NMP2 Main Stack / NMP2 CR East-High	843	257	202.5° SSW
NMP2 Main Stack / NMP2 CR East-Low	846	258	202.5° SSW
NMP2 Rw/Rx Building Vent / NMP2 CR West-High	250.21	76.26	186.34° S
NMP2 Rw/Rx Building Vent / NMP2 CR West-Low	218.91	66.72	188.39° S
NMP2 Rw/Rx Building Vent / NMP2 CR East-High	210.86	64.27	161.63° SSE
NMP2 Rw/Rx Building Vent / NMP2 CR East-Low	210.86	64.27	161.63° SSE
NMP2 Main Steam Tunnel / NMP2 CR West-High	240.72	73.37	182.23° S
NMP2 Main Steam Tunnel / NMP2 CR West-Low	208.88	63.67	183.75° S
NMP2 Main Steam Tunnel / NMP2 CR East-High	209.85	63.96	156.19° SSE
NMP2 Main Steam Tunnel / NMP2 CR East-Low	209.85	63.96	156.19° SSE
NMP2 SGT Building / NMP2 CR West-High	411.83	125.52	204.71° SSW
NMP2 SGT Building / NMP2 CR West-Low	384.85	117.30	207.29° SSW
NMP2 SGT Building / NMP2 CR East-High	334.79	102.04	193.49° SSW
NMP2 SGT Building / NMP2 CR East-Low	334.79	102.04	193.49° SSW
NMP2 PASS Panel / NMP2 CR West-High	423.74	129.15	176.13° S
NMP2 PASS Panel / NMP2 CR West-Low	391.42	119.31	176.44° S
NMP2 PASS Panel / NMP2 CR East-High	393.81	120.03	161.85° SSE
NMP2 PASS Panel / NMP2 CR East-Low	393.81	120.03	161.85° SSE
NMP2 Releases to the NMP1 Control Room			
NMP2 Main Stack / NMP1 CR	1355	413	233° SW
NMP2 Rw/Rx Building Vent / NMP1 CR	615	187	246° WSW
NMP2 Main Steam Tunnel / NMP1 CR	595	181	246° WSW
NMP2 SGT Building / NMP1 CR	800	244	242° WSW
NMP2 PASS Panel / NMP1 CR	660	201	230° SW

Table A6-2 (Cont'd)

Release/Intake	Horizontal Distance (ft)	Horizontal Distance (m)	Sector Bearing Relative to True North
NMP2 Releases to the Technical Support Center			
NMP2 Main Stack / TSC	1182	360.29	54.87° NE
NMP2 Rw/Rx Building Vent / TSC	461.01	140.51	255.58° WSW
NMP2 Main Steam Tunnel / TSC	441.30	134.51	256.01° WSW
NMP2 SGT Building / TSC	637.97	194.45	247.88° WSW
NMP2 PASS Panel / TSC	485.55	148.00	233.49° SW
NMP1 Releases to the NMP2 Control Room			
NMP1 Main Stack / NMP2 CR West-High	740	225.6	121° ESE
NMP1 Main Stack / NMP2 CR West-Low	720	219.5	119.5° ESE
NMP1 Main Stack / NMP2 CR East-High	800	243.8	115° ESE
NMP1 Main Stack / NMP2 CR East-Low	800	243.8	115° ESE
NMP1 Rx Bldg Blowout Panel / NMP2 CR West-High	766.28	233.56	112.24° ESE
NMP1 Rx Bldg Blowout Panel / NMP2 CR West-Low	750.68	228.81	110.09° ESE
NMP1 Rx Bldg Blowout Panel / NMP2 CR East-High	838.88	255.69	106.73° ESE
NMP1 Rx Bldg Blowout Panel / NMP2 CR East-Low	838.88	255.69	106.73° ESE
NMP1 Turb Bldg Blowout Panel / NMP2 CR West-High	751.92	229.19	98.08° E
NMP1 Turb Bldg Blowout Panel / NMP2 CR West-Low	743.81	226.71	95.68° E
NMP1 Turb Bldg Blowout Panel / NMP2 CR East-High	840.50	256.18	93.90° E
NMP1 Turb Bldg Blowout Panel / NMP2 CR East-Low	840.50	256.18	93.90° E

Table A6-3
ARCON96 Results - X/Q Values for the NMP2 Control Room

Release Point	X/Q Dispersion Coefficients (s/m³)				
	0-2 hrs	2 – 8 hrs	8 – 24 hrs	1–4 days	4–30 days
West-Upper Intake					
NMP2 Main Stack	7.04E-05	3.95E-05	1.49E-05	9.96E-06	7.46E-06
NMP2 Rw/Rx Building Vent	8.24E-04	6.29E-04	2.28E-04	1.56E-04	1.25E-04
NMP2 Main Steam Tunnel	1.13E-03	7.49E-04	2.76E-04	1.90E-04	1.49E-04
NMP2 SGT Building	3.62E-04	2.59E-04	9.48E-05	6.16E-05	4.42E-05
NMP2 PASS Panel	3.36E-04	2.00E-04	7.31E-05	5.53E-05	4.04E-05
NMP1 Main Stack	1.06E-04	5.90E-05	2.23E-05	1.73E-05	1.43E-05
NMP1 Rx Building Blowout Panel	1.23E-04	7.21E-05	2.57E-05	2.28E-05	2.05E-05
NMP1 Turb Building Blowout Panel	1.30E-04	9.03E-05	3.45E-05	2.92E-05	2.56E-05
East-Upper Intake					
NMP2 Main Stack	8.03E-05	4.48E-05	1.68E-05	1.20E-05	8.83E-06
NMP2 Rw/Rx Building Vent	1.09E-03	7.23E-04	2.46E-04	1.92E-04	1.47E-04
NMP2 Main Steam Tunnel	1.47E-03	8.80E-04	3.32E-04	2.26E-04	1.68E-04
NMP2 SGT Building	5.31E-04	3.70E-04	1.35E-04	9.16E-05	6.70E-05
NMP2 PASS Panel	3.74E-04	2.05E-04	7.08E-05	5.41E-05	3.88E-05
NMP1 Main Stack	9.83E-05	5.81E-05	2.22E-05	1.83E-05	1.57E-05
NMP1 Rx Building Blowout Panel	1.06E-04	6.70E-05	2.39E-05	2.17E-05	1.96E-05
NMP1 Turb Building Blowout Panel	1.09E-04	7.73E-05	2.95E-05	2.45E-05	2.14E-05
West-Lower Intake					
NMP2 Main Stack	7.15E-05	4.01E-05	1.52E-05	1.01E-05	7.55E-06
NMP2 Rw/Rx Building Vent	9.03E-04	6.93E-04	2.50E-04	1.71E-04	1.36E-04
NMP2 Main Steam Tunnel	1.46E-03	9.74E-04	3.63E-04	2.45E-04	1.90E-04
NMP2 SGT Building	4.05E-04	2.95E-04	1.08E-04	6.98E-05	5.00E-05
NMP2 PASS Panel	3.84E-04	2.28E-04	8.23E-05	6.28E-05	4.57E-05
NMP1 Main Stack	1.10E-04	6.16E-05	2.31E-05	1.85E-05	1.54E-05
NMP1 Rx Building Blowout Panel	1.26E-04	7.73E-05	2.74E-05	2.45E-05	2.23E-05
NMP1 Turb Building Blowout Panel	1.31E-04	9.42E-05	3.59E-05	3.01E-05	2.63E-05
East-Lower Intake					
NMP2 Main Stack	7.78E-05	4.31E-05	1.64E-05	1.16E-05	8.61E-06
NMP2 Rw/Rx Building Vent	9.43E-04	6.34E-04	2.13E-04	1.67E-04	1.29E-04
NMP2 Main Steam Tunnel	1.46E-03	8.70E-04	3.32E-04	2.23E-04	1.68E-04
NMP2 SGT Building	5.33E-04	3.72E-04	1.36E-04	9.17E-05	6.72E-05
NMP2 PASS Panel	3.67E-04	2.01E-04	6.95E-05	5.32E-05	3.83E-05
NMP1 Main Stack	9.54E-05	5.68E-05	2.15E-05	1.79E-05	1.54E-05
NMP1 Rx Building Blowout Panel	1.06E-04	6.68E-05	2.39E-05	2.16E-05	1.95E-05
NMP1 Turb Building Blowout Panel	1.08E-04	7.69E-05	2.96E-05	2.44E-05	2.14E-05

Table A6-4
ARCON96 Results - X/Q Values for the TSC

Release Point	X/Q Dispersion Coefficients (s/m^3)				
	0-2 hrs	2 – 8 hrs	8 – 24 hrs	1–4 days	4–30 days
NMP2 Main Stack	4.95E-05	2.69E-05	1.03E-05	6.67E-06	4.85E-06
NMP2 Rw/Rx Building Vent	2.70E-04	1.64E-04	5.41E-05	3.86E-05	2.86E-05
NMP2 Main Steam Tunnel	3.27E-04	2.41E-04	8.38E-05	5.95E-05	4.76E-05
NMP2 SGT Building	1.62E-04	1.19E-04	4.28E-05	2.72E-05	2.24E-05
NMP2 PASS Panel	2.69E-04	1.91E-04	7.19E-05	4.22E-05	3.40E-05

Table A6-5
ARCON96 Results - X/Q Values for the NMP1 Control Room

Release Point	X/Q Dispersion Coefficients (s/m^3)				
	0-2 hrs	2 – 8 hrs	8 – 24 hrs	1–4 days	4–30 days
NMP2 Main Stack	4.18E-05	2.30E-05	8.94E-06	5.62E-06	4.31E-06
NMP2 Rw/Rx Building Vent	1.77E-04	1.09E-04	3.92E-05	2.48E-05	1.85E-05
NMP2 Main Steam Tunnel	1.90E-04	1.37E-04	4.93E-05	3.12E-05	2.56E-05
NMP2 SGT Building	1.11E-04	8.09E-05	2.91E-05	1.82E-05	1.45E-05
NMP2 PASS Panel	1.59E-04	1.13E-04	4.19E-05	2.48E-05	2.00E-05

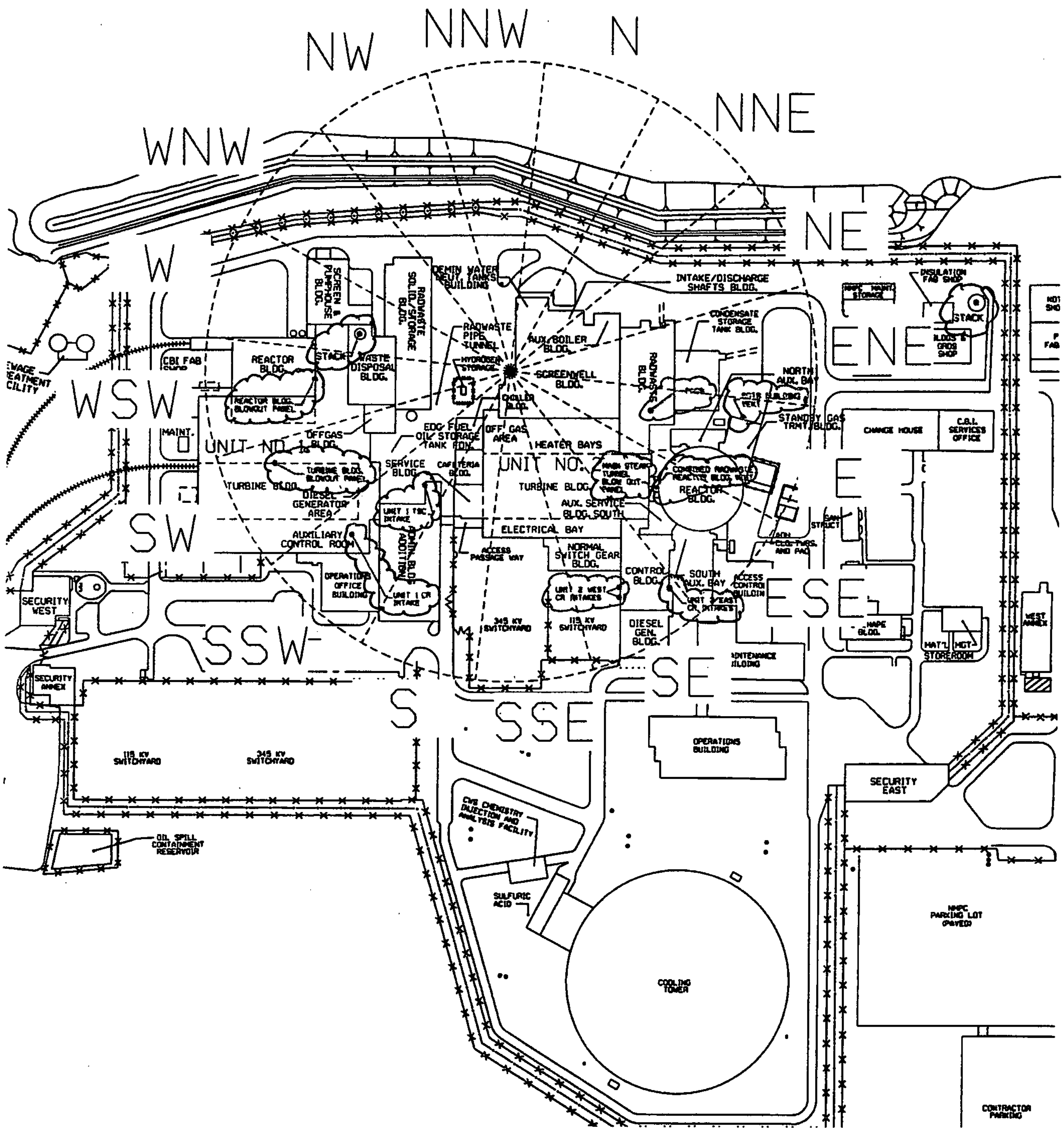
Table A6-6
PAVAN Results – NMP2 Ground Level Release X/Q Values

Boundary	X/Q Dispersion Coefficients (s/m^3)				
	0 – 2 hours	0 – 8 hours	8 – 24 hours	1 – 4 days	4 – 30 days
EAB	1.19E-04	---	---	---	---
LPZ	---	1.62E-05	1.09E-05	4.59E-06	1.33E-06

Table A6-7
PAVAN Results – NMP2 Main Stack Release X/Q Values

Boundary	X/Q Dispersion Coefficients (s/m^3)				
	0 – 2 hours	0 – 8 hours	8 – 24 hours	1 – 4 days	4 – 30 days
EAB	2.96E-05	---	---	---	---
LPZ	---	1.42E-05	5.41E-07	2.31E-07	7.65E-08

Figure A6-1
Site Plan Showing Relative Locations of Release and Intake Points



ATTACHMENT (7)

ENCLOSED CALCULATIONS FOR ALTERNATIVE SOURCE TERM

The following calculations are provided in this attachment:

1. H21C-106, "Unit 2 LOCA w/LOOP, AST Methodology"
2. H21C-093, "LOCA Bypass Piping Models for Alternative Source Term Methodology (AST)"
3. H21C-101, "U2 MSLB, AST Methodology"
4. H21C-102, "U2 FHA, AST Methodology"
5. H21C-103, "U2 CRDA, AST Methodology"
6. H21C-097, "Post-LOCA Suppression Pool pH Analysis"
7. H21C076, "X/Qs for Releases from NMP Units 1 & 2 (CNS Calcs NMPAST-01-001 & NMPAST-02-001)"
8. H21C-094, "Calculation of Atmospheric Dispersion Parameter for MSLB Release to Unit 2 Control Room"