March 29, 2005

Mr. Michael Kansler
President
Entergy Nuclear Operations, Inc.
440 Hamilton Avenue
White Plains, NY 10601

SUBJECT: VERMONT YANKEE NUCLEAR POWER STATION - ISSUANCE OF AMENDMENT RE: ALTERNATIVE SOURCE TERM (TAC NO. MC0253)

Dear Mr. Kansler:

The Commission has issued the enclosed Amendment No. 223 to Facility Operating License DPR-28 for the Vermont Yankee Nuclear Power Station (VYNPS), in response to your application dated July 31, 2003, as supplemented on October 10, November 7 (2 letters), November 20, December 11 (2 letters), and December 30, 2003, and February 10, February 18, February 25, March 17, May 12, and July 20, 2004.

The amendment revises the VYNPS licensing basis to incorporate a full-scope application of an alternative source term methodology in accordance with Title 10 of the *Code of Federal Regulations*, Section 50.67.

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

Sincerely,

/RA/

Richard B. Ennis, Senior Project Manager, Section 2
Project Directorate I
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket No. 50-271

Enclosures: 1. Amendment No. 223 to
License No. DPR-28
2. Safety Evaluation

cc w/encls: See next page

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President
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Sincerely,
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Richard B. Ennis, Senior Project Manager, Section 2
Project Directorate I
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket No. 50-271
Enclosures: 1. Amendment No. 222 to License No. DPR-28
2. Safety Evaluation
cc w/encls: See next page

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OFFICE	PDI-VY/PM	PDI-2/LA	EMCB/SC	EMEB/SC	SPSB/SC	SPLB/SC
NAME	REnnis	CRaynor	LLund	KManoly	RDennig	DSolorio
DATE	3/17/05	3/14/05	SE Dated 12/22/03	SE Dated 3/12/04	SE Dated 4/6/04	SE Dated 3/10/04

OFFICE	SRXB/SC	IROB/SC	OGC	PDI-2/SC		
NAME	FAkstulewicz	TBoyce	TSmith	DRoberts		
DATE	2/25/05	3/2/05	3/9/05	3/25/05		

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ENTERGY NUCLEAR VERMONT YANKEE, LLC

AND ENTERGY NUCLEAR OPERATIONS, INC.

DOCKET NO. 50-271

VERMONT YANKEE NUCLEAR POWER STATION

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 223
License No. DPR-28

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

   A. The application for amendment filed by Entergy Nuclear Vermont Yankee, LLC and Entergy Nuclear Operations, Inc. (the licensee) dated July 31, 2003, as supplemented on October 10, November 7 (2 letters), November 20, December 11 (2 letters), and December 30, 2003, and February 10, February 18, February 25, March 17, May 12, and July 20, 2004, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

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

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

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

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 3.B of Facility Operating License No. DPR-28 is hereby amended to read as follows:

   (B) Technical Specifications

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 223, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of its date of issuance and shall be implemented within 60 days.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA/

Darrell J. Roberts, Chief, Section 2
Project Directorate I
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Attachment: Changes to the Technical Specifications

Date of Issuance: March 29, 2005

ATTACHMENT TO LICENSE AMENDMENT NO. 223

FACILITY OPERATING LICENSE NO. DPR-28

DOCKET NO. 50-271

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

Remove	Insert
5	5
75	75
76	76
140	140
147	147
155a	155a
156	156
163	163
167	167
170	170
171	171
175	175
265	265
266	266

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 223 TO FACILITY OPERATING LICENSE NO. DPR-28

ENTERGY NUCLEAR VERMONT YANKEE, LLC

AND ENTERGY NUCLEAR OPERATIONS, INC.

VERMONT YANKEE NUCLEAR POWER STATION

DOCKET NO. 50-271

1.0 INTRODUCTION

By letter dated dated July 31, 2003 (Reference 1), as supplemented on October 10 (Reference 2), November 7 (2 letters) (References 3 and 4), November 20 (Reference 5), December 11 (2 letters) (References 6 and 7), and December 30, 2003 (Reference 8), and February 10 (Reference 9), February 18 (Reference 10), February 25 (Reference 11), March 17 (Reference 12), May 12 (Reference 13), and July 20, 2004 (Reference 40), Entergy Nuclear Vermont Yankee, LLC and Entergy Nuclear Operations, Inc. (Entergy or the licensee) submitted a request to amend the Vermont Yankee Nuclear Power Station (VYNPS) Technical Specifications (TSs). The supplements dated October 10, November 7 (2 letters), November 20, December 11 (2 letters), and December 30, 2003, and February 10, February 18, February 25, March 17, May 12, and July 20, 2004, provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the staff's original proposed no significant hazards consideration determination as published in the *Federal Register* on November 25, 2003 (68 FR 66135).

The proposed amendment would revise the VYNPS licensing basis to incorporate a full-scope application of an alternative source term (AST) methodology in accordance with Title 10 of the *Code of Federal Regulations* (10 CFR) Section 50.67.

The licensee's application also included a request for exemptions from 10 CFR Part 50, Appendix J. Specifically, the licensee requested permanent exemption to permit exclusion of the main steam pathway leakage contributions from the overall integrated leakage rate Type A test measurement and from the sum of the leakage rates from local leakage rate from Type B and Type C tests. This request was reviewed concurrently as a separate licensing action.

2.0 REGULATORY EVALUATION

In the past, power reactor licensees have typically used U.S. Atomic Energy Commission Technical Information Document TID-14844, "Calculation of Distance Factors for Power and Test Reactor Sites," dated March 23, 1962 (Reference 14), as the basis for design-basis accident (DBA) analysis source terms. The power reactor siting regulation, which contains offsite dose limits in terms of whole body and thyroid dose, 10 CFR Part 100, Section 11 (10 CFR 100.11), "Determination of Exclusion Area, Low Population Zone, and Population Center Distance," makes reference to TID-14844.

In December 1999, the Nuclear Regulatory Commission (NRC or the Commission) issued 10 CFR 50.67, "Accident Source Term," which provides a mechanism for licensed power reactors to replace the traditional accident source term used in their DBA analyses with an AST. Regulatory guidance for the implementation of these ASTs is provided in Regulatory Guide (RG) 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors" (Reference 15). Section 50.67 of 10 CFR requires a licensee seeking to use an AST to apply for a license amendment and requires that the application contain an evaluation of the consequences of affected DBAs. Entergy's application of July 31, 2003, as supplemented, addresses these requirements in proposing to use the AST described in RG 1.183 as the VYNPS DBA source term used to evaluate the radiological consequences of DBAs. As part of the implementation of the AST, the total effective dose equivalent (TEDE) acceptance criterion of 10 CFR 50.67(b)(2) replaces the previous whole body and thyroid dose guidelines of 10 CFR 100.11 and 10 CFR Part 50, Appendix A, General Design Criterion (GDC) 19, "Control Room," for a loss-of-coolant accident (LOCA), main steamline break (MSLB) accident, fuel handling accident (FHA), and control rod drop accident (CRDA).

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

The NRC staff also considered the following regulatory requirements and guidance in its review:

GDC 19, "Control Room," of Appendix A to 10 CFR Part 50, insofar as it requires that adequate radiation protection be provided to permit access and occupancy of the control room under accident conditions without personnel receiving radiation exposures in excess of 5 rem whole body, or its equivalent to any part of the body, for the duration of the accident. Although VYNPS is a pre-GDC plant, the acceptance criteria of GDC 19 were used for evaluation purposes by both the licensee and the NRC staff.

NUREG-0800, "Standard Review Plan" (SRP), Chapter 6.5.2, "Containment Spray as a Fission Product Cleanup System, Revision 2," (Reference 16), provides guidance for determining the fission product removal effectiveness for the containment spray and the spray additive or pH control systems.

NUREG-0800, SRP 15.0.1, "Radiological Consequence Analyses Using Alternative Source Terms, Revision 0," (Reference 17) provides guidance for the safety review of the radiological consequences of DBAs associated with implementing an AST. SRP 15.0.1 supports the guidance outlined in RG 1.183.

General Electric (GE) Report, NEDC-31858P-A, Revision 2, entitled "BWROG [Boiling Water Reactor Owners Group] Report for Increasing MSIV [Main Steam Isolation Valve] Leakage Rate Limits and Elimination of Leakage Control Systems," (BWROG Report) (Reference 18), was referenced by the licensee as a basis for the acceptability of the proposed changes. The BWROG report summarizes data on the seismic performance of main steam piping and condensers in past strong-motion earthquakes at various facilities, and compares design attributes of the piping and condensers with those in typical GE Mark I, II, and III nuclear plants. The NRC staff, in its safety evaluation report (SER) of the BWROG report dated March 3, 1999 (Reference 19), determined that the BWROG approach of utilizing the earthquake experience-based methodology, supplemented by plant-specific seismic adequacy evaluations is an acceptable basis to demonstrate the seismic ruggedness of non-seismically analyzed main steam system piping and condensers. However, the NRC staff identified certain limitations that required individual licensees to provide plant-specific design information and evaluation when the BWROG approach was elected for resolving the MSIV leakage issue.

3.0 TECHNICAL EVALUATION

3.1 Alternate Leakage Treatment

RG 1.183, Appendix A, provides assumptions that are acceptable for the evaluation of radiological consequences of the design-basis LOCA using AST. For boiling water reactor (BWR) MSIV leakage, RG 1.183 allows credit for reducing MSIV releases due to holdup and deposition in the main steam piping downstream of the MSIVs and in the main condenser, including the treatment of air ejector effluent by off-gas systems, if the components and piping systems used in the release path are capable of performing their safety functions during and following a safe shutdown earthquake (SSE). In the July 31, 2003, submittal, Entergy discussed an Alternate Leakage Treatment (ALT) strategy that credits the reduction in MSIV release due to the holdup and deposition provided by the downstream components. The licensee employed the NRC-approved model presented in NEDC-31858P-A, Revision 2, for evaluating reduction of MSIV releases. The proposed ALT pathway includes the main condenser and piping connected to the main steamlines between the MSIVs and the turbine stop valves along with the associated valves at VYNPS.

The licensee's supplement of November 7, 2003, provided the ALT Pathways and Boundaries Seismic Verification Report and the associated ALT Pathways and Boundaries Walkdown Report used to evaluate the seismic ruggedness of the VYNPS ALT pathway. Even though VYNPS was licensed prior the issuance of Appendix A to 10 CFR 100 and has not committed to Part 100, the licensee, nevertheless, performed reviews to demonstrate that the piping and related supports fell within the bounds of the experience database presented in NEDC-31858P-A, Revision 2. The NRC staff reviewed the seismic adequacy of the proposed ALT against the guidance presented in NEDC-31858P-A, Revision 2, and its associated NRC SER.

3.1.1 Seismic Isolation Boundaries

The NRC staff’s SER approving NEDC-31858P-A identified limitations to be addressed as part of a plant-specific application of the AST methodology. These limitations relate to (1) assuring that the ALT pathways for MSIV leakage are functionally reliable commensurate with their intended safety function, and (2) assuring that the pathways, including the main condenser, are seismically rugged. The licensee has identified the ALT pathways and related seismic isolation boundaries that are credited in support of the AST analysis, using the isolated condenser strategy, for processing MSIV leakage. Post-LOCA access of the boundary isolation valves for the ALT pathways is not required, as the licensee indicated that the valves are remotely operated and should fail in the required position for establishing the boundary isolation consistent with the AST analysis. Consequently, new operator actions outside the control room are not necessary for establishing the isolation boundaries for the ALT pathways.

During review of the seismic isolation boundaries described by the licensee for the ALT pathways, the NRC staff identified an unisolated valve that was not clearly identified by the licensee as being part of the seismic isolation boundaries. In response to the NRC staff's request for additional information (RAI), the licensee stated in its February 10, 2004, supplement, that the manual valve identified by the NRC staff, and several other manual valves, were, in fact, included in the seismic isolation boundaries. The licensee clarified that manual valves that are part of the seismic isolation boundaries that are normally in the required closed position were not explicitly identified. Instead, the focus of the July 31, 2003, submittal was on those seismic boundary isolation valves that would typically require operator action to reposition. The licensee further indicated that the manual boundary isolation valves that are normally closed are verified to be closed prior to plant start-up in accordance with pre-established normal valve line-up procedures. If one of these valves is inadvertently left open, the immediate area around the valve would heatup very quickly due to steam that would be escaping through the valve to the environment. Such a condition would be promptly detected and corrected.

Based on its review of the information presented by the licensee, the NRC staff finds that the licensee has identified suitable ALT pathways, including boundary isolation valves, that can be relied upon in support of the proposed AST analytical methodology.

3.1.2 Seismic Demand

Seismic Ground Motion and Response Spectra: The horizontal earthquake time history used to generate response spectra for VYNPS was based on the 1952 Kern County earthquake recorded at Taft, California, scaled to peak ground acceleration of 0.14g for the SSE. The developed ground response spectrum (GRS) shapes for the operational basis earthquake (OBE) and SSE are shown in the VYNPS Updated Final Safety Analysis Report (UFSAR) (Reference 20). Since the VYNPS GRS is not one of the BWR plant GRS shown in the BWROG report, Entergy compared the VYNPS GRS with the BWR plant GRS to establish applicability of the BWROG experienced methods for demonstrating seismic ruggedness of main steam piping, attached leakage path piping, other ALT pathway and boundary components, and associated supports/anchorages at VYNPS.

Individual plots of the 5 percent damped BWROG GRS compared with the VYNPS 5 percent damped SSE GRS are shown in Figures 3-2 to 3-10 of Reference 3. In general, the earthquake experience database sites have experienced strong ground motions that are in excess of the VYNPS SSE in the frequency range of interest. All the database site ground motions envelop the VYNPS SSE GRS by large factors in frequency bands above 1 Hz. The licensee's use of the BWROG's earthquake experience-based methodology was reviewed by the NRC staff and found acceptable to verify the seismic adequacy of equipment in the alternative MSIV leakage pathway.

Piping: Entergy indicated that since the piping at VYNPS was fabricated and installed using the industry standard practice complying with the American Society of Mechanical Engineers (ASME) USA Standards (USAS) B31.1-1967 criteria (Reference 21), the ALT seismic boundary piping at VYNPS is consistent in design practice and construction with the piping results from the facilities in the earthquake experience database. Table 3.1 of Reference 3 presents a summary of piping data (sizes, schedules, materials, etc.) for the main steam and drain piping at VYNPS, and Table 3.2 of Reference 3 presents similar data for facilities in the earthquake experience database. Table 3-3 of Reference 3 presents a summary comparison of the same data for VYNPS and the facilities in the earthquake experience database. This comparison shows that pipe sizes and ratios (D/t) of pipe diameter (D) to pipe wall nominal thickness (t) for the ALT pathway and boundary piping fall within the limits of the pipe sizes and the ratios of the earthquake experience database piping. However, there is an exception because some of the VYNPS seismic boundary piping is 5" outer-diameter (OD), which is not explicitly represented in the earthquake experience database. Considering that piping of both smaller and larger sizes with comparable and enveloping D/t ratios are adequately represented in the database, as shown in Figure 3-11 of Reference 3, Entergy concluded that the 5" OD piping is adequately enveloped by the experience data and supporting analysis. The NRC staff reviewed the analyses and concurs with the licensee's judgment that the VYNPS ALT piping data are enveloped by the experience database.

Equipment and Other Features: Entergy employed the NRC-approved Seismic Qualification Utility Group's (SQUG) Generic Implementation Procedure, Revision 2 (GIP-2) methodology (Reference 22) to address the seismic adequacy of other equipment such as valves, instruments and tanks. Entergy compared the VYNPS design-basis SSE GRS with the GIP-2 bounding spectrum, and found that the GIP-2 bounding spectrum envelops the VYNPS design-basis SSE GRS. The NRC staff determined that the use of the GIP-2 methodology for evaluating the equipment within the scope of the ALT seismic boundary is a reasonable approach for concluding the seismic adequacy of these components and is, therefore, acceptable.

3.1.3 Seismic Verification Walkdowns

Section 5.5 of the NRC's SER for NEDC-31858P-A, Revision 2, states that walkdowns will be performed by the licensee to identify outlier design features that could constitute potential failure modes. In order to confirm the functional capability of the leakage pathway, Entergy indicated that a Seismic Review Team (SRT) performed a seismic verification walkdown of the MSIV leakage pathway. The SRT consisted of degreed engineers with greater than 20 years experience in structural engineering and/or earthquake experience methodology.

The SRT performing the field walkdown first reviewed the installed scope of equipment, piping and tubing. Evaluations of piping and equipment design were performed to assure that installations are representative of database design practice and that components are free of known seismic vulnerabilities. The review included: (1) piping, pipe support and equipment design attributes; (2) seismic anchor motion and interaction considerations; (3) valve design attributes; and (4) potential external corrosion indications.

Piping, Pipe Support and Equipment Design Attributes: Entergy stated that the SRT reviewed the various piping configurations and tubing supports that make up the ALT paths and boundary. Specific design attributes that were screened include:

a) piping with dead-weight support greatly in excess of USAS B31.1 Code-suggested spans, or tubing with excessive sagging;
b) heavy, unsupported in-line components;
c) piping constructed of non-ductile materials such as cast iron or polyvinyl chloride (PVC);
d) non-standard fittings, or unusual attachments that could cause excessive localized stresses;
e) pipe supports that exhibit non-ductile behavior; and
f) presence of severe corrosion.

Seismic Anchor Movement Issues: Entergy stated that the SRT evaluated potential seismic damage to piping, tubing and supports that were attributed to seismic anchor movement. The SRT considered the damage to be the result of excessive thermal movement of end equipment, differential movement between supports in adjacent buildings, and excessive movements imposed on branch lines by flexible headers. These attributes were evaluated during the piping walkdowns.

Seismic Interaction Issues: Entergy indicated that the seismic interaction review was a visual inspection of structures, piping, or equipment adjacent to the components under evaluation. The interaction review evaluated conditions where postulated seismically-induced failures and displacements of adjacent structures, piping, or equipment could adversely affect the required seismic performance of the system and components.

Valve Design Attributes: Entergy stated that the guidelines used to screen valves that are relied upon to establish the ALT pathway, or are part of the seismic verification boundary, were consistent with the SQUG GIP-2.

Representative Bounding Analysis Review: Entergy stated that the SRT selected representative supports and anchorages for a plant-specific seismic evaluation following the walkdown. The SRT particularly considered heavily loaded supports or those for which anchorage capacity appeared marginal. The SRT also determined if an enveloping analytical assessment would be appropriate and beneficial.

The licensee indicated that a seismic verification walkdown of normally inaccessible piping and equipment was conducted during Refueling Outage 24 (RFO-24) and stated that it resolved all required modifications during the outage. Based on its review of the information presented by the licensee, the NRC staff finds that the licensee has taken reasonable measures to ensure resolution of identified outliers.

3.1.4 Seismic Capacity

Building Seismic Capacity: Entergy indicated that the piping and equipment of the ALT pathways and boundaries are located within two buildings at VYNPS - the Reactor Building and the Turbine Building. The Reactor Building is a Seismic Class I Structure. Its seismic capability has been assured in accordance with existing Seismic Class I design-basis requirements for

VYNPS, as described in the VYNPS UFSAR. However, except for the diesel generators and oil day tank areas, which are designated as Seismic Class I areas, the Turbine Building is a Seismic Class II structure. Entergy performed an assessment of the seismic capability of the Turbine Building to demonstrate that Turbine Building structures do not fail during or after an SSE event in a manner that would adversely impact the condenser and other piping and equipment relied upon to contain leakage through the MSIVs. This assessment was based on a review of the current design, existing calculations and the extent of conformance with Unresolved Safety Issue (USI) A-46 and Individual Plant Examination of External Events (IPEEE) programs. Entergy stated that the results of the evaluation showed that the Turbine Building structure is capable of withstanding the VYNPS SSE without structural damage. In addition, a seismic verification walkdown was performed during RFO-24 to verify the structural integrity of the masonry walls. On these bases, Entergy concluded that the Turbine Building structure will not adversely impact the functionality of the condenser, steam piping, and other components relied upon to contain MSIV leakage during and after the VYNPS SSE.

Based on its review of the information presented by the licensee, the NRC staff concurs with licensee's conclusion that the Reactor Building and the Turbine Building at VYNPS are seismically adequate to support the piping and equipment of the ALT pathways and boundaries.

<u>Experience-Based Condenser and Anchorage Capacity</u>: Using SQUG GIP-2 methods, Entergy performed an evaluation to verify whether the condenser falls within the bounds of the earthquake experience database and to establish the design adequacy of the condenser and associated anchorage for seismic SSE demand applicable to the VYNPS site. Specifically, the condenser shell was evaluated to determine global and local shell stresses. The evaluation followed the recommendations of NEDC-31858P-A, Revision 2, combined with analysis using stress allowables consistent with SQUG GIP-2 recommendations. Entergy stated that the condensers are MSIV alternate leakage path walkdown outliers because they are not specifically included in the SQUG GIP-2 twenty classes of equipment.

Seismic capacity versus demand was evaluated by comparing the VYNPS condenser with the condensers in the seismic experience database that have experienced strong motion earthquakes in excess of the VYNPS SSE. Condenser size, construction, and design characteristics were summarized and compared with parameters relating specifically to the earthquake experience condensers. Based on this comparison, Entergy determined that the VYNPS condenser is similar to those within the earthquake experience database.

Anchorages were evaluated using established procedures from the GIP-2, and concrete structures were evaluated using the criteria of American Concrete Institute (ACI) 318-99, "Building Code Requirement for Reinforced Concrete" (Reference 23). Load factors and allowable stresses were modified to be consistent with SQUG GIP-2 methods. Stresses in the condenser shell were evaluated using the American Institute of Steel Construction (AISC) methods, with allowable stresses modified to be consistent with SQUG GIP-2 values.

Entergy demonstrated that the condenser design is typical of those at the facilities that have experienced earthquakes equivalent to, and in excess of, the VYNPS SSE. Thus, Entergy concluded that the VYNPS condenser satisfies the SQUG capacity versus demand requirement on the basis that they compare favorably with the database condensers. Entergy also indicated that the condenser anchorages meet the criteria of the GIP-2, and the condenser shell stresses are lower than allowable limits under SSE demand. On these bases, Entergy concluded that the condenser is adequate as part of the MSIV ALT path.

Entergy stated that a walkdown of the VYNPS main condenser, which was inaccessible during normal power operation, was performed during RFO-24 to ensure that the construction and installation details conform to plant design drawing details. Based on its review of the information presented by the licensee, the NRC staff finds that the licensee's determination concerning the condenser and condenser anchorage seismic adequacy is reasonable and acceptable.

<u>Turbine Stop and Main Steam Control Valves</u>: Entergy identified the turbine stop valves (SVs) and main steam control valves (CVs) as outliers since these valves are not included in the scope of the USI A-46 or IEEE programs at VYNPS based on the plant developed safe shutdown equipment list. Entergy performed an evaluation for the SVs and CVs using the earthquake experience database and manual calculation methods that follow the rules of SQUG GIP-2. Calculations were performed to address the SVs' operator weak link. Based on review of the operator design drawings and adequacy of the load path to the rigid supports and their anchorages, this weak leak is the yoke legs. Entergy demonstrated that an evaluation of the yoke under a 3g lateral load per the GIP-2 shows that the seismic yoke stresses are small.

Based on a comparison of valve configuration, support load path to the Turbine Building structure, and a comparison of the VYNPS design basis SSE with those of the earthquake experience database, Entergy's evaluation concluded that the existing design of the SVs and CVs can be expected to demonstrate excellent performance under earthquake loading, without breach of pressure boundary or functional failure. Entergy indicated that a walkdown of these components was performed during RFO-24 to confirm this conclusion. Based on its review of the information presented by the licensee, the NRC staff finds that the licensee's determination concerning the turbine SVs and main steam CVs and associated supports seismic adequacy is reasonable and acceptable.

<u>Bypass Valves Steam Chest</u>: Entergy performed an assessment of the steam chest and associated supports using a combination of seismic experience data and SQUG engineering experience from the GIP-2. The steam chest valves are hydraulically actuated spring assist-to-close valves similar to the design of the SVs discussed above. Entergy indicated that the valve bodies for these valves are not of cast iron construction, therefore, the earthquake experience database for the SVs provides assurances for the structural integrity of the steam chest valves.

As indicated above, Entergy identified that the weak link for the valve assembly is the valve yoke and that an evaluation of the yoke under a 3g lateral load per GIP-2 showed that the seismic yoke stresses are small. Entergy's assessment of the vertical and horizontal rigid rods showed that the pipe reactions are within the design capacity of the rods and that adequate load path exists to transfer the support loads to the Turbine Building structure. Entergy indicated that a walkdown of these components was performed during RFO-24 to confirm this conclusion. Based on its review of the information presented by the licensee, the NRC staff finds that the licensee's determination concerning the bypass valves' steam chest CVs and associated supports seismic adequacy is reasonable and acceptable.

3.1.5 Analytical Assessments

Entergy performed analytical assessments for specific piping and components to address potential piping concerns or assess conditions found during the seismic verification walkdown that do not meet the walkdown screening guidelines, or which were judged by the SRT to require further review for outlier resolution.

Entergy selected the analytical criteria for the evaluation of piping, supports and associated components to address the primary concern of ensuring the functionality of the main steam piping downstream of the outboard MSIV, including bypass/drain piping, and the main condenser to remain structurally intact and act as a holdup volume for fission products during and after an SSE. For piping analysis, Entergy used the USAS B31.1 piping code requirements with piping critical damping of 5 percent, and an allowable stress limit of 2.4 S_h, where S_h is defined as material allowable stress at maximum operating temperature as listed in the B31.1 Code. Entergy indicated that seismic SSE demand was based on the VYNPS design basis in-structure response spectra, as utilized and accepted as seismic demand within the VYNPS A-46 program. For the supports and components analysis, Entergy used the criteria of GIP-2 for the seismic adequacy of the components. Allowable stresses were derived from Part 2 of the AISC Code, and allowable loads for the concrete expansion anchors were obtained from Appendix C of GIP-2. Table 8.1 of Reference 3 summarizes the qualification criteria for piping, supports and equipment.

Entergy indicated that the majority of the piping under review is of A106 Grade B carbon steel material, with material allowables of S_h = 15,000 psi, S_y = 35,000 psi, and S_u = 60,000 psi at room temperature and S_h = 15,000 psi, S_y = 26,500 psi, and S_u = 60,000 psi at the maximum operating temperature of the ALT pathways and boundaries piping (S_y is defined as material yield stress at normal operating temperature; S_u is defined as material ultimate strength at temperature). Entergy used the equation in Table 8-1 of Reference 3, and calculated piping stresses, which are less than 1.03 S_y at room temperature and 1.36 S_y at maximum operating temperature. Based on this evaluation, Entergy concluded that limiting the range of applied stress to less than 2 S_y will ensure no significant membrane stress rupture will occur and accumulated cyclic damage will be elastic. In addition, given the limited number of strong motion cycles during a design-basis SSE event, only elastic cycling well below the 2 S_y limit will occur. Therefore, the licensee determined that a fatigue failure from a postulated SSE loading would not occur.

To address the structural capability for the Path 2 piping and systems, Entergy performed an analytical evaluation of the Path 2 piping and supports. Path 2 is defined as an alternate ALT drain path that follows main steam low point drains to the condenser via air operated valve LCV-2-143 as described in Reference 1. Entergy stated that the evaluation methods used are consistent with, or conservative with respect to, database evaluation methods. Entergy indicated that the Path 2 piping and supports are considered seismically rugged. Also, Entergy performed an analytical evaluation of the main steam piping. The criteria utilized in this evaluation were plant design criteria as outlined in the VYNPS UFSAR. The UFSAR criteria are conservative relative to the criteria established for assessment of the ALT pathways and boundaries seismic verification. Entergy stated that the calculated seismic stresses have significant margin relative to established stress allowables. The NRC staff reviewed the analyses and concurs with the licensee's judgment of considering the piping seismically rugged.

3.1.6 Fraction of MSIV Leakage to the High-Pressure Turbine

NEDC-31858P-A, Revision 2, provides an approved method for calculating the total amount of MSIV leakage that is allowed to escape to the atmosphere through the high-pressure (HP) turbine. The approved method involves the ratio of the HP turbine throttle valve disc seating area compared to the ALT pathway flow area to the condenser. The approved methodology limits this ratio to 0.01 (or roughly 1 percent of the MSIV leakage). The NRC staff found inconsistencies with the licensee's discussion of this ratio. The licensee indicated in its February 10, 2004, supplement, that the inconsistency in Reference 1 was due to an editorial error; and that the correct ratio value for VYNPS is 0.008.

While resolving the above inconsistency, the licensee also found that the MSIV leakage ratio was not calculated correctly, and indicated that plant modifications would be made to increase the ALT pathway flow area to the main condenser so that the 0.008 ratio would be achieved. The NRC staff issued another RAI dated February 17, 2004, to obtain additional clarification as to the nature of the calculational error that was made and plant modifications that were planned. In its response dated February 18, 2004, the licensee indicated that the calculational error was due to using the wrong piping schedule in the calculations (schedule 80 instead of schedule 160). The licensee also stated that it would: (a) replace a segment of 1" pipe with 2" pipe (approximately 6 linear feet), and (b) replace a 1" valve with a 2" valve, associated with the main steam drain lines that are part of the primary ALT pathway. The larger pipe size in conjunction with the increased port diameter of the new valve will ensure that the previously calculated ratio of 0.008 is satisfied. These modifications were completed during RFO-24.

Based on its review of the information presented by the licensee, the NRC staff finds that the licensee's determination of the HP turbine-to-ALT pathway area ratio to be consistent with the guidance of NEDC-31858P-A, Revision 2, and is acceptable.

3.1.7 NRC Staff Conclusion Concerning Seismic Adequacy

Based on the above evaluation, the NRC staff concludes that there is reasonable assurance that the VYNPS main condenser and the piping connected to the main steamlines between the MSIVs and the turbine SVs, along with the associated valves, will be seismically adequate for the proposed ALT pathways. The NRC staff's conclusion is based on: (1) the comparison that indicated that the VYNPS SSE in the frequency range of interest is well below the seismic ground motion spectra that was experienced at the facilities in the earthquake experience database; (2) the design attributes of the VYNPS main condenser are generally enveloped by those of the condensers in the earthquake experience database, and that the condenser assembly has sufficient anchorage capacity; (3) the non-seismically analyzed ALT piping is adequately represented by piping in the earthquake experience database that demonstrated good seismic performance; (4) the detailed analyses performed for the non-seismic portion of the main steam drain lines indicated adequate safety margins for piping stresses and support loads; and (5) the Turbine Building has been adequately designed to withstand the SSE loads.

It should be noted that the NRC staff's acceptance of the experience-based and GIP-2 methodology as presented by Entergy is restricted to its application for ensuring the pressure boundary integrity and functionality of the MSIV ALT pathway. The NRC staff's acceptance of the methodology for this application is not an endorsement for the use of the experience-based methodology for other applications at VYNPS.

As stated above, the licensee elected to utilize seismic experience-based methodology based on NEDC-31858P-A, Revision 2. The NRC staff, in its SER of the BWROG report, determined the BWROG approach of utilizing the earthquake experience-based methodology, supplemented by plant-specific seismic adequacy evaluations, an acceptable basis to demonstrate the seismic ruggedness of non-seismically analyzed mainsteam piping and condensers.

On the basis of the information provided by the licensee, the NRC staff concludes that the piping and components which comprise the ALT pathway are seismically rugged and are able to perform the safety function of an MSIV leakage treatment system. Therefore, it is acceptable for the VYNPS AST methodology to credit the ALT pathway as proposed.

3.2 Radiological Consequences of DBAs

The NRC staff reviewed the licensee's analysis methods, assumptions, and inputs, using docketed information provided by the licensee. Because the licensee provided proprietary calculations that use a version of the same computational DBA dose code, (RADTRAD, Reference 24), that the staff uses, the NRC staff did not perform extensive independent dose calculations. Instead, the NRC staff reviewed the licensee’s analyses, and verified appropriate use of the code and appropriate conservatisms in the inputs and assumptions made by the licensee. Although the NRC staff performed some additional calculations to verify acceptability of the licensee’s assumptions, the NRC staff's findings are based on the licensee’s analyses.

The current licensed reactor power level for VYNPS is 1593 megawatts thermal (MWt). Entergy has submitted a separate amendment request for an extended power uprate that would increase the licensed power level to 1912 MWt. Entergy performed the radiological analyses that support this AST amendment assuming a reactor power equal to 1950 MWt (102 percent of 1912 MWt). This is a conservative approach for this amendment request and is acceptable to the NRC staff.

In support of full implementation of an AST, Entergy re-analyzed the following DBAs: LOCA, MSLB, FHA, and CRDA. The NRC staff finds that the licensee generally followed RG 1.183 guidance concerning design-basis radiological analyses. The analyses performed by the licensee show that with an assumed thermal power of 1950 MWt, the radiological consequences of the DBAs met the dose acceptance criteria in RG 1.183, 10 CFR 50.67 and GDC-19. The licensee’s analysis results are listed in Table 1.

3.2.1 LOCA

The objective of analyzing the radiological consequences of a LOCA is to evaluate the performance of various plant safety systems intended to mitigate the postulated release of radioactive materials from the plant to the environment. In accordance with RG 1.183 guidance, Entergy determined the inventory of fission products in the reactor core based on the proposed uprated maximum full power operation of the core using an appropriate isotope generation and depletion computer code. Fission products from the damaged fuel are released into the reactor coolant system (RCS) and then into the primary containment (i.e., drywell and wetwell). With a LOCA, it is anticipated that the initial fission product release to the primary containment will last 30 seconds and will release all of the radioactive materials dissolved or suspended in the RCS liquid. The gap inventory release phase begins two minutes after the event starts and is assumed to continue for 30 minutes. As the core continues to degrade, the gap inventory release phase ends and the in-vessel release phase begins. This phase continues for 1.5 hours. Tables 1, 4, and 5 of RG 1.183 define the source term used for these two phases.

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

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

3.2.1.1 Suppression Pool Post-LOCA pH Control

According to NUREG-1465, "Accident Source Terms for Light-Water Nuclear Power Plants" (Reference 25), the iodine entering the containment of an RCS during an accident would be composed of at least 95 percent cesium iodide (CsI). Upon deposition on interior surfaces and dissolution in the suppression pool of a BWR, the predominant form of the iodine would be the iodide ion (I^-). At pH less than 7.0, a large fraction of the iodide would be converted by irradiation into molecular iodine (I_2) and released into the containment atmosphere. If the pH were maintained above 7.0, however, the fraction of I^- converted to I_2 would be only about $3x10^{-4}$. Since the pH of the suppression pool is not normally controlled, I_2 may be released during a LOCA as the acids, which are produced due to the radiation effects of the LOCA, lower the pH.

One way to minimize this release is to add an alkaline chemical capable of buffering the pH at a value above 7.0. As outlined in Reference 1, Entergy proposes to do this by adding sodium pentaborate ($Na_2B_{10}O_{16} \bullet 10H_20$) from the standby liquid control (SLC) system during a LOCA. Although designed as a backup method to maintain the reactor subcritical after an accident, the SLC system can be used as a pH control injection. Entergy proposes to use the SLC system to inject sodium pentaborate to the reactor pressure vessel (RPV), where it will mix with ECCS flow and spill over to the drywell and then to the suppression pool. Sodium pentaborate, a base, will neutralize acids generated in the post-accident primary containment environment. The licensee used a combination of known parameters and conservative assumptions as inputs to a proprietary computer code to calculate pH at discrete times for 30 days following the postulated accident. Credit for the SLC system in the radiological analyses is based on operation of one SLC pump, initiated within two hours after the event starts. The VYNPS operating procedures will be revised to direct operators, upon detection of symptoms indicating that core damage is occurring, to manually initiate the SLC system.

Nitric Acid

Nitric acid (HNO_3) is produced by the irradiation of water and air following a LOCA. The amount of HNO_3 was calculated by the licensee's contractor using a proprietary water radiolysis model.

HNO_3 production is proportional to the time-integrated dose rate for gamma and beta radiation. The model predicted 668 moles of nitric acid would be produced over the 30-day period.

Hydrochloric Acid

Hydrochloric acid (HCl) is generated by the irradiation and heating of Hypalon and PVC cable insulation. The amount of acid generated is proportional to the amount of beta and gamma radiation absorbed by the insulation. The amount of HCl generated was determined by the licensee using a proprietary cable radiolysis model with guidance from NUREG/CR-5950, "Iodine Evolution and pH Control" (Reference 26), and NUREG-1081, "Post-Accident Gas Generation from Radiolysis of Organic Materials" (Reference 27). The model predicted 2187 moles of HCl would be produced over the 30-day period.

Sodium Pentaborate Buffering

In order to counter the effect of the increasing nitric and hydrochloric acids in the suppression pool, sodium pentaborate solution from the SLC system would be added within a few hours to buffer the pH in the alkaline range. The injection would be accomplished from the main control room with a keylock switch manipulation, which is an existing operator action intended for reactivity control in the core. According to VYNPS TS 3.4/4.4 and the corresponding TS Bases, the borate addition must achieve a concentration of 800 ppm natural boron in the reactor core within 138 minutes at a rate of 35 gallons per minute. This requires a range of 3850 gallons of 10.1 percent sodium pentaborate solution to 4830 gallons of 8.15 wt. percent sodium pentaborate solution. The corresponding minimum boron injection is about 600 pounds-mass (lbm).

The SLC system is designed to inject the borate into the reactor vessel, but under the postulated accident conditions it leaks into the suppression pool through the postulated pipe break. The licensee assumed that the operators could inject the borate to the pool through another pathway if the vessel and pool are not immediately connected as a result of the accident. Although the operators are instructed to initiate the injection upon detection of high drywell radiation, the licensee conservatively assumed that injection of the borate begins within two hours of the onset of the LOCA. The borate buffering was assumed to begin at five hours.

Determination of the Suppression Pool pH

The pH was calculated as a function of time, beginning one hour after the accident and ending after 30 days. The calculations were performed partly by hand and partly by a proprietary computer code using inputs described in Reference 1. During normal operation, the pH in the suppression pool is not controlled, but the licensee estimated the value at 4.8 based on the conductivity, which is controlled. At the beginning of the postulated accident, no effect of borate was considered, but the pH was maintained greater than 8.0 because of the high cesium hydroxide concentration relative to the amount of acid produced. The licensee assumed that the pH in the suppression pool would increase after an accident due to the release of cesium compounds. However, the cesium compounds were considered only for the first five hours of the analysis. Beginning at five hours, the effect of the borate was evident and the pH value was 8.6. Beginning at 12 hours, the calculated pH began a sequential decrease to a final value of 8.1 at 30 days.

SLC System Redundancy

The VYNPS SLC system is classified as a safety-related system, as defined in 10 CFR 50.2, and satisfies the requirements for such systems. However, the SLC system has active components that cannot be considered redundant. These are (1) the two check valves (in series) on the SLC injection line, and (2) the SLC initiation control switch in the main control room. The licensee has provided information on the design, purchasing, performance history, inspection and testing of these components to show that the quality and reliability of these components is such that they will perform as needed to provide SLC injection for post-LOCA suppression pool pH control.

To show that the SLC system can perform its LOCA suppression pool pH control safety function, the licensee demonstrated sufficient reliability of the two containment isolation check valves. In response to NRC staff questions on the proposed use of the SLC system for pH control for a DBA LOCA, in a letter dated February 25, 2004, the licensee stated that no failures to open on demand (the pertinent action to fulfill the safety function) have been observed at VYNPS for the specific valves in question. It determined this through a review of VYNPS maintenance and surveillance records. The SLC containment isolation check valves are tested each RFO. The licensee further stated that for the specific model of check valve, five failures to open on demand have been found in industry databases, but that the failure modes were not applicable to VYNPS. The recorded failures were due to corrosion products, solidified boric acid, or debris in the valve. VYNPS has corrosion resistant stainless steel valves which are maintained and tested in a demineralized water system which does not have boric acid present, is not as corrosive, and does not have debris in the system.

Through a series of three telephone conferences on July 6, 7 and 16, 2004, the NRC staff asked further clarifying questions of the licensee on how it performed the search of the industry component performance databases. Entergy provided a written response to the NRC staff by letter dated July 20, 2004. In the response, Entergy stated that it searched the databases for all Rockwell-Edwards check valve failures for model numbers that included “3674.” This broad search included check valves in both pressurized water reactor (PWR) and BWR applications. The results of this broad search were then narrowed to only count instances of “failure to open,” the type of failure that would prevent the SLC system from fulfilling the pH control function. The final step was to further narrow the results from the “failure to open” search to identify BWR SLC system valves with that failure. Entergy affirmed that no failures to open on demand were recorded in the industry databases for the Rockwell-Edwards model 3674 valve (or its improved model number 36274) installed in BWR SLC systems. All failures to open on demand of the check valve occurred in PWR plants, and all except one in systems that include boric acid. The remaining valve was installed on a gas containing system (not a fluid containing system like SLC) and was a carbon steel valve (not corrosion resistant) welded to stainless steel piping. The failure mode for this valve was corrosion enhanced by the dissimilar metals. All these valves are not considered by the staff to be installed in an environment sufficiently similar to the VYNPS SLC containment isolation check valves to be considered in evaluating the VYNPS SLC check valves. The staff further finds that reasons these valves failed to open are not applicable to VYNPS SLC containment check valves. The staff finds that the results of the industry component performance database search as discussed by Entergy are too broad and are not pertinent to evaluating the VYNPS SLC containment check valves.

In its July 20, 2004 letter, Entergy provided additional information on the performance history of the Rockwell-Edwards model 3674 check valves installed in the VYNPS SLC system. There are a total of four Rockwell-Edwards model 3674 check valves installed in the VYNPS SLC system: the two containment isolation valves, installed in series, and two others installed one each at the discharge of each SLC pump. The containment isolation check valves are full flow tested once each RFO. The SLC pump discharge valves are tested quarterly. The tests are performed with demineralized water and the check valves are left containing demineralized water. None of the four SLC check valves has failed to open on demand at VYNPS.

Although acknowledging that a single failure-to-open of one of the two containment isolation check valves would prevent SLC injection, the NRC staff has determined that the potential for failure is very low based upon the quality as established by the procurement, periodic testing, and historical performance of the component. The staff finds that the VYNPS SLC system containment isolation check valves are sufficiently reliable to allow injection of sodium pentaborate from the SLC system to control pH in the suppression pool for a DBA LOCA.

For the control switch, the licensee’s review of industry operating experience shows that although several modes of failure have been reported, the licensee believes the switch to be highly reliable. Regardless of the reliability of the switch itself, the licensee will revise plant procedures, prior to implementation of this amendment, to provide instruction to jumper the switch contacts in the event of failure of the switch. Because the SLC injection is not needed immediately after the LOCA occurs, and the SLC is assumed to inject two hours after the LOCA for pH control of the suppression pool, operators would have adequate time to take this action.

Based on its review of the information presented by the licensee, the NRC staff finds that the non-redundant components of the VYNPS SLC system are of sufficient quality and reliability, or compensatory measures can be taken, to ensure that SLC injection will occur when required.

SLC System Operating Procedures

The licensee’s procedure for the operator to manually initiate SLC injection is contained in a controlled copy, staged at the control room panel near the SLC system controls and indications. Initiation of SLC injection for suppression pool pH control will be based on high drywell radiation levels. The annunciator response procedures will be revised to require SLC injection before the drywell reaches 4,000 R/hr. The revised procedures will be issued before the implementation of the AST license amendment. Licensed personnel and shift technical advisors will be initially trained before the implementation of the AST amendment and receive periodic refresher training on these procedures as part of the Licensed Operator Requalification Program.

The SLC system could also be initiated post-LOCA through two other means. The “RPV Control” Emergency Operating Procedure lists the SLC system as an alternate injection sub-system. Because the DBA LOCA event for the AST is assumed to occur due to loss of all other normal and emergency injection sub-systems, the SLC injection would also be initiated on low RPV water level. The LOCA event assumed in the AST (loss of injection and core melting) would result in entry into the Severe Accident Guidelines, which currently contain instructions to inject SLC upon entry.

The containment high-range radiation monitors would indicate high drywell radiation, which would direct (per procedure) the operators to initiate SLC injection. The containment high-range radiation monitors are required to be operable by VYNPS TS 3.2.G, “Post-Accident Instrumentation.” The containment high-range radiation monitors meet the quality criteria for a Type E variable as defined in RG 1.97, "Instrumentation for Light-Water-Cooled Nuclear Power Plants to Access Plant and Environs Conditions During and Following an Accident," Revision 3, Tables 1 and 2 (Reference 28). The reactor water level instruments are used as another means to indicate that SLC injection is needed for suppression pool pH control. These instruments meet the category 1 quality criteria for a Type B variable as defined in RG 1.97, Tables 1 and 2.

Based on its review of the information presented by the licensee, the NRC staff finds that the licensee’s procedures and planned training provides reasonable assurance that operators will initiate post-LOCA SLC injection for suppression pool pH control.

NRC Staff Conclusion Concerning Suppression Pool pH Adequacy

The licensee proposed adding sodium pentaborate from the SLC system as the pH buffer, and provided supporting documentation from a proprietary computer model and manual calculations. According to the licensee's analysis, the suppression pool pH would be greater than 8.0 for at least 30 days. In a letter dated December 11, 2003, the licensee answered questions from the NRC staff about the methodology and input parameters. The staff reviewed the licensee’s methodology for determining the pH and performed an independent evaluation of the licensee’s calculations. The VYNPS UFSAR and TSs were reviewed to verify the volume of the containment system and the SLC system volume and concentration. Hand calculations were performed to check the amount of cable insulation exposed to the radiation, the amount of acid produced, the concentration of cesium in the suppression pool, the amount of boron added from the SLC, and the resulting pH values at times 0, 5 hours, and 30 days. Consistent with the licensee’s analysis, the NRC staff’s calculations indicated that the suppression pool pH would remain above 8.0 for at least 30 days after a LOCA starts.

Based on the above evaluation, the NRC staff finds that the pH of the suppression pool will be maintained at a level above 7.0 following a LOCA, thus, preventing re-evolution of elemental iodine dissolved in the suppression pool water.

3.2.1.2 Containment Leakage Pathway

The licensee’s AST analyses assume that the primary containment leaks at its design leakage rate of 0.8 percent of its contents by weight per day for the first 24 hours of a LOCA and then at 0.4 percent for the remainder of the 30-day accident duration. RG 1.183, Appendix A, Section 3.7 states that for BWRs, primary containment leakage may be reduced after the first 24 hours, if supported by plant configuration and analysis, to a value not less than 50 percent of the TS leak rate. In Reference 1, the licensee stated that it performed an evaluation that justifies the reduction in assumed containment leakage after 24 hours and provided a summary of the justification. For VYNPS's containment peak pressure of approximately 44 psig, the licensee’s evaluation showed that a reduction of 50 percent in the containment volumetric leak rate is not achieved until the containment reaches a pressure of 5.5 psig. The licensee stated that the use of drywell sprays reduces the VYNPS drywell pressure to a value of approximately 5.3 psig at 24 hours from a peak value of 44 psig. This is a pressure decrease of a factor of eight. Based on its review of the information presented by the licensee, the NRC staff finds that the licensee's justification that the VYNPS containment leak rate is reduced to half its TS value at 24 hours is reasonable and acceptable.

The licensee’s evaluation took credit for removal of iodine by the drywell sprays. The VYNPS drywell spray system is designated safety-related and has been designed to work in post-LOCA containment conditions. Its availability is governed by the VYNPS TSs. Drywell spray manual initiation is called for in the plant procedures based on the drywell high-range radiation monitors response. The drywell high-range radiation monitoring system is safety-related and has been designed to work in post-LOCA containment conditions. Its availability is also governed by the VYNPS TSs. Entergy performed an evaluation to show that the radiation level would be reached within five minutes into the gap release phase of the LOCA. The licensee’s dose analyses assume that spray operation is initiated 13 minutes after the start of the gap release. Entergy determined the drywell spray removal for elemental iodine and particulate in accordance with the recommendations in SRP 6.5.2. The NRC staff reviewed the licensee’s calculation of the spray removal coefficients included in the proprietary version of the evaluation entitled "Radiological Evaluation of DBA-Loss of Coolant Accident," which was included in the October 10, 2003, supplement. A non-proprietary description of the licensee’s evaluation is in Attachment 5 of Reference 1. Based on its review of the information presented by the licensee, the NRC staff finds that the licensee’s calculation of spray removal coefficients and credit of the drywell spray system in their dose analyses is consistent with SRP 6.5.2, Revision 2 and, therefore, is acceptable.

Entergy asserts that the drywell and wetwell air spaces become well mixed following the restoration of core cooling because the thermal-hydraulic conditions in the primary containment are expected to be quite active with steaming and condensing. The licensee assumed that the radioactivity release is diluted into the larger volume of the wetwell plus drywell air spaces after 122 minutes. Before this time, the radioactivity is only assumed to be released into the drywell net free volume. Based on its review of the information presented by the licensee, the NRC staff finds this formulation and the modeling of the primary containment volume reasonable and acceptable.

Leakage from the primary containment will collect in the free volume of the secondary containment and be released to the environment via ventilation system exhaust or leakage. Following a LOCA, the standby gas treatment system (SGTS) fans start and draw down the secondary containment to create a negative pressure with reference to the environment. This pressure differential ensures that leakage from the primary containment is collected and processed by the SGTS. SGTS exhaust is processed through charcoal filter media prior to release to the environment via the site’s elevated stack. Entergy does not credit dilution or holdup of leakage in the secondary containment. In addition, Entergy conservatively assumes that a positive pressure exists in the secondary containment for the first 10 minutes after the accident and that the resulting leakage is released directly to the environment as a ground-level release. This positive pressure period is caused by the plant response for loss of offsite power.

3.2.1.3 MSIV Leakage

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

The NRC staff finds assumptions on credit for holdup and plate-out in the condenser and main steamlines acceptable based on the NRC staff's conclusion above in Section 3.1.7 of this safety evaluation (SE) that main steamline and components that comprise the ALT pathway are seismically rugged and are able to perform the safety function of an MSIV leakage treatment system. Entergy credits aerosol and elemental iodine removal due to deposition in the main steamlines and the main condenser. One main steamline is assumed to have a failure of either the inboard or outboard MSIV. For the other three main steamlines, sedimentation is credited in the inboard-to-outboard MSIV volumes and in the horizontal piping section volumes from the outboard MSIV to the point where the drainlines tap off. Sedimentation is also credited in the main condenser.

The aerosol removal efficiencies for the main steamlines and main condenser were determined based on the methodology in AEB-98-03, "Assessment of Radiological Consequences for the Perry Pilot Plant Application Using the Revised (NUREG-1465) Source Term," Appendix A (Reference 29). The NRC staff finds the licensee's use of the AEB-98-03 methodology and the resulting aerosol removal efficiencies reasonable and acceptable.

Entergy used the guidance in SRP 6.5.2 to estimate the elemental iodine removal coefficients for the main condenser. The licensee stated that it believes the conditions in the main condenser are more closely approximated by the containment conditions than those in the main steam piping. The main condenser elemental iodine removal was calculated using conservative inputs and assumptions to give a 99.8 percent removal efficiency. As a point of reference, other BWR Mark I plants with AST have calculated similar elemental iodine removal efficiencies for the main condenser with other methodologies. Based on its review of the information presented by the licensee, the NRC staff finds the licensee's calculation of the main condenser elemental iodine removal efficiencies acceptable.

For elemental iodine removal in the main steamlines, Entergy uses the Brockmann-Bixler pipe deposition model incorporated in the NRC-sponsored RADTRAD computer code (Reference 24). Entergy modeled two of the four main steamlines with a total MSIV leakage of 124 scfh, which is equivalent to the maximum leakage that would be allowed by the proposed changes to the VYNPS TSs. One of the two steamlines was designated as being faulted within the containment with its inboard MSIV assumed to have failed open. As a result, Entergy conservatively did not credit deposition between the RPV and the outboard MSIV in this faulted steamline. Entergy credited deposition downstream of the outboard MSIVs in both steamlines. The NRC staff considers this approach conservative because it postulates multiple failures, thereby exceeding minimum regulatory guidance.

Appendix A to RG 1.183, Section 6.3, states that steamline deposition models should be based on the assumption of well-mixed volumes, but allows other models, such as slug flow, to be used if justified. Entergy used models that assume the main steamline is well mixed. Additionally, the licensee used plant-specific piping geometry data and included only horizontal main steamlines in determining the total volume and interior surface area used in the deposition models. Based on its review of the information presented by the licensee, the NRC staff finds that the licensee followed the guidance in RG 1.183 in determining the aerosol and elemental iodine deposition efficiencies, and that the input values to the dose analysis are acceptable.

3.2.1.4 Leakage from ECCS

During the progression of a LOCA, some fission products released from the fuel will be carried to the suppression pool via spillage from the RCS and by spray removal processes. Post-LOCA, the suppression pool is a source of water for ECCS. Since portions of these systems are located outside of the primary containment, leakage from these systems is evaluated as a potential radiation exposure pathway. For the purposes of assessing the consequences of leakage from the ECCS, Entergy assumes that all of the radioiodines released from the fuel are instantaneously moved to the suppression pool. This source term assumption is conservative in that all of the radioiodine released from the fuel is assumed to be in both the primary containment atmosphere leakage and the ECCS leakage, concurrently. In a mechanistic treatment, the radioiodines in the primary containment atmosphere would relocate to the suppression pool over time. Noble gases released from the fuel are assumed to remain in the drywell atmosphere. Since aerosols and particulate radionuclides are not expected to become airborne on release from the ECCS, they are not included in the ECCS source term. These assumptions follow the guidance in RG 1.183.

The analysis considers the equivalent of 1 gallon per minute (gpm) ECCS leakage starting at the onset of a LOCA. This leakage rate includes a factor of 2 multiplier over the VYNPS TS limit, in accordance with guidance in RG 1.183, to address increases in the leakage due to normal material degradation between surveillance tests. Entergy assumes 10 percent of the iodine in the ECCS leakage becomes airborne and is available for release. No credit was assumed for holdup and dilution in the secondary containment. As was assumed for the primary containment leakage pathway, the leakage enters the environment as an unfiltered ground-level release for the first 10 minutes after the event starts. After this 10-minute positive pressure period, the leakage enters the environment via the SGTS as a filtered elevated release, with a percentage that bypasses the filter.

3.2.1.5 Offsite Doses

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

3.2.1.6 Control Room Doses

Entergy evaluated the dose to operators in the control room. It was assumed that the control room would not be isolated during the event. The control room ventilation system draws in 3700 cubic feet per minute (cfm) of unfiltered outside air. Entergy analyzed the control room dose over a 30-day period. The resulting 30-day TEDE to an individual in the control room is less than the 10 CFR 50.67 criteria.

3.2.1.7 LOCA Conclusion

Based on the review discussed above, the NRC staff concluded that the licensee's application of the AST to the VYNPS LOCA analysis is acceptable. Table 1 provides the doses projected by Entergy. Table 2 provides the LOCA analysis assumptions found acceptable by the NRC staff.

3.2.2 MSLB

The MSLB accident considered is the complete severance of a main steamline outside the primary containment with the reactor at hot-standby conditions to maximize the mass of coolant released through the break. The radiological consequences of a break outside containment will bound the results from a break inside containment. The MSIVs are assumed to isolate the leak within 6.8 seconds. This assumed time is based on the allowed MSIV closure time and the response time for the isolation logic. There is no fuel damage projected for the design-basis MSLB. The analysis is performed for two activity release cases, based on the maximum equilibrium and pre-accident iodine spike concentrations of 1.1 microcuries per gram (μCi/gm) and 4 μCi/gm dose equivalent I-131, respectively. All of the accident activity was assumed released within 6.8 seconds following the accident as a ground-level release, with no credit for Turbine Building holdup or dilution. These assumptions are in accordance with RG 1.183.

Entergy evaluated the maximum two-hour TEDE to an individual located at the EAB and the 30-day TEDE to an individual at the outer boundary of the LPZ. The resulting doses are less than the RG 1.183 dose acceptance criteria and are less than the 10 CFR 50.67 criteria.

Entergy evaluated the dose to operators in the control room, using a puff release control room atmospheric dispersion factor. The NRC staff finds the use of the puff release control room atmospheric dispersion factor acceptable because of the very short duration of the MSLB release (6.8 seconds). A further discussion of the NRC staff's review of the licensee's atmospheric dispersion factors may be found in Section 3.2.5.2, below. It was assumed that the control room would not be isolated during the event. The control room ventilation system draws in 3700 cfm of unfiltered outside air. Entergy analyzed the control room dose over a 30-day period. The resulting 30-day TEDE to an individual in the control room is less than the 10 CFR 50.67 criteria.

Based on this review, the NRC staff concluded that the licensee's application of the AST to the VYNPS MSLB analysis is acceptable. Table 1 provides the doses projected by Entergy. Table 3 provides the analysis assumptions found acceptable by the NRC staff.

3.2.3 FHA

The FHA analysis postulates that a spent fuel assembly is dropped during refueling 24 hours after shutdown. The kinetic energy developed in this drop is conservatively assumed to be dissipated in damage to the cladding on 193 fuel rods. The fission product inventory in the fuel rod gap of the damaged fuel rods is assumed to be instantaneously released because of the accident. This activity is assumed to be released from the damaged fuel and the overlying fuel pool to the secondary containment building, from where it is assumed to be released to the environment within two hours. Although radiation monitors in the exhaust ducts from the refueling floor would automatically actuate the SGTS, Entergy assumed no credit for filtration by the SGTS. Credit was taken for containment and collection by the SGTS and elevated release through the plant stack.

Fission products released from the damaged fuel are decontaminated by passage through the pool water, with the degree of decontamination depending on their physical and chemical form. Entergy assumed no decontamination for noble gases, a factor of 200 decontamination of radioiodines, and retention of all aerosol and particulate fission products.

Entergy evaluated the maximum two-hour TEDE to an individual located at the EAB and the 30-day TEDE to an individual at the outer boundary of the LPZ. The resulting doses are less than the RG 1.183 dose acceptance criteria and are less than the 10 CFR 50.67 criteria.

Entergy evaluated the dose to operators in the control room. It was assumed that the control room would not be isolated during the event. The control room ventilation system draws in 3700 cfm of unfiltered outside air. Entergy analyzed the control room dose over a 30-day period. The resulting 30-day TEDE to an individual in the control room is less than the 10 CFR 50.67 criteria.

Based on this review, the NRC staff concluded that the licensee's application of the AST to the VYNPS FHA analysis is acceptable. Table 1 provides the doses projected by Entergy. Table 4 provides the analysis assumptions found acceptable by the NRC staff.

3.2.4 CRDA

The CRDA analysis postulates a sequence of mechanical failures that results in the rapid removal (i.e., drop) of a control rod. Localized damage to fuel cladding and a limited amount of fuel melt are projected. A reactor trip would occur. The MSIVs are assumed to remain open for the duration of the event. Entergy has projected that 850 fuel rods would be breached by the event, and of these damaged rods, none would exceed the threshold for melting. Entergy analyzed three cases for radioactivity release from the CRDA. Case 1 (condenser leakage) assumes manual isolation of the MSIVs prior to any release to the atmosphere via the Advanced Off Gas (AOG) system. Case 2 (AOG release) assumes the MSIVs remain open after the CRDA and the AOG system remains operational. All releases to the environment are via the AOG system and stack and include only noble gases. Case 3 (RCS recirculation sampling line release) assumes the sampling lines remain open for 30 days after the CRDA with a constant leak rate of 32 gallons per hour. Release pathways for Cases 1 and 2 are mutually exclusive, while the release modeled in Case 3 is additive to either other case.

The CRDA analysis was performed using the gap fractions and fuel melt fractions from Appendix C of RG 1.183. Ten percent of the core inventory of noble gases and iodines and 12 percent of the alkali metals are assumed to be in the fuel gap. For Case 1, Entergy assumed that 100 percent of the noble gases, 10 percent of the iodines and 1 percent of the alkali metals released reach the main condenser due to plate-out in the RPV and main steamlines. Of the iodine that enters the main condenser, 90 percent plates out. There is no reduction in noble gases. The fission product gases in the main condenser are released at a rate of 1 percent by volume over 24 hours as a ground-level release. In all three cases, the control room ventilation system was not assumed to isolate.

Entergy evaluated the maximum two-hour TEDE to an individual located at the EAB and the 30-day TEDE to an individual at the outer boundary of the LPZ. The resulting doses for are less than the RG 1.183 dose acceptance criteria and are less than the 10 CFR 50.67 criteria.

Entergy evaluated the dose to operators in the control room. It was assumed that the control room would not be isolated during the event. The control room ventilation system draws in 3700 cfm of unfiltered outside air. Entergy analyzed the control room dose over a 30-day period. The resulting 30-day TEDE to an individual in the control room is less than the 10 CFR 50.67 criteria.

Based on this review, the NRC staff concluded that the licensee's application of the AST to the VYNPS CRDA analysis is acceptable. Table 1 provides the doses projected by Entergy. Table 5 provides the analysis assumptions found acceptable by the NRC staff.

3.2.5 Atmospheric Relative Concentration Estimates

As discussed above, the licensee performed a dose assessment for four postulated DBAs: LOCA, MSLB, FHA, and CRDA. In its dose assessment, the licensee used previously calculated relative concentration (χ/Q) values for the following postulated cases:

- LOCA release from the MSIV in the turbine building, MSLB, FHA and CRDA ground level release and LOCA, FHA and CRDA stack release to the EAB;
- LOCA and CRDA stack release to the LPZ;
- FHA stack release to the control room; and
- CRDA ground level and stack releases to the control room.

These values are discussed in the SER associated with VYNPS Amendment 212, dated September 18, 2002 (Reference 30). As part of this amendment request, the license calculated new χ/Q values for:

- postulated LOCA ground-level release from the reactor building bypass and siding to the EAB;
- LOCA and CRDA ground-level releases to the LPZ;
- LOCA ground-level and stack releases to the control room;
- MSLB puff release to the control room; and
- FHA ground-level release to the control room.

The licensee provided detailed information on the inputs and assumptions used in its calculations in Attachment 1 to a letter dated December 30, 2003.

3.2.5.1 Meteorological Data

Entergy calculated the new χ/Q values for the LOCA, MSLB, FHA, and CRDA dose assessments described above using meteorological data collected during calendar years 1995 through 1999. Wind direction and wind speed data were measured at 10.7 and 90.5 meters above grade. Delta-temperature was measured between approximately 60 and 10 meters. The NRC staff performed a review of the five years of meteorological data using the methodology described in NUREG-0917, "Nuclear Regulatory Commission Staff Computer Programs for Use with Meteorological Data" (Reference 31). Data recovery during the five-year period was greater than 90 percent during each of the years for each measured parameter and generally in the upper 90 percentiles. This meets the recommendation of RG 1.23, "Onsite Meteorological Programs" (Reference 32). There were several outages of relatively long duration (e.g., greater than a week's duration), but the outages are not judged to have a significant effect on the calculated χ/Q values. With respect to atmospheric stability measurements, the length and time of occurrence of stable and unstable atmospheric conditions appeared reasonable with respect to expected meteorological conditions. Stable and neutral conditions were consistently reported to occur at night and unstable and neutral conditions during the day. The longest reported continuous occurrence of a single unstable category was 8 consecutive hours, which is consistent with expected meteorological conditions. Wind direction frequency occurrence at both the 10.7 and 90.5 meter levels were very similar from year to year throughout the five-year period. While the 90.5 meter level showed more distinctive bimodal flow, winds at both heights were predominately from the north northwest and generally from the south sectors. The lower level experienced secondary winds generally from the westerly quadrants. Based on this review, the NRC staff finds that the data provides a suitable base for calculation of the new χ/Q estimates used in dose assessments.

3.2.5.2 Control Room Atmospheric Dispersion Factors

As stated above, the licensee calculated new control room χ/Q values for postulated LOCA, MSLB, and ground-level FHA releases. The NRC staff qualitatively reviewed the inputs to the calculations and found them generally consistent with site configuration drawings and staff practice.

LOCA Control Room χ/Q Values

New χ/Q values for the LOCA were modeled as ground-level releases from the main steamline isolation valve in the Turbine Building and from the Reactor Building bypass and siding. Calculations were based upon the ARCON96 methodology discussed in RG 1.194, "Atmospheric Relative Concentrations for Control Room Radiological Habitability Assessments at Nuclear Power Plants" (Reference 33). The Reactor Building siding release was assumed to occur at a minimum straight-line distance of approximately 9.8 meters from the control room intake. This release was assumed to occur as a diffuse release only from the top part of the containment building, rather than the full face of the wall. RG 1.194 states that the ARCON96 methodology should not be used at distances less than 10 meters. However, since the distance is only slightly less than 10 meters and the licensee has used the noted conservatism, the NRC staff judges that use of the ARCON96 methodology is acceptable in this case.

The licensee states that calculations for postulated releases from the plant stack were performed in accordance with guidance in RG 1.194 by comparing and combining results from ARCON96 computer code and a licensee computer code similar to the PAVAN code, which is based on methodology described in RG 1.145, "Atmospheric Dispersion Models for Potential Accident Consequence Assessments at Nuclear Power Plants," Revision 1 (Reference 34). The NRC staff finds this methodology acceptable.

The NRC staff performed limited confirmatory calculations and obtained results similar to the licensee's estimates. Based on its review, the NRC staff finds the resultant χ/Q estimates acceptable.

MSLB Control Room χ/Q Values

The MSLB accident was modeled as a hemispherical puff assuming an instantaneous release that flashed to steam at atmospheric pressure and saturation temperature. All of the radioactive material was assumed to remain within the bubble and to be transported directly past the control room air intake. The licensee used a procedure for instantaneous puff releases similar to that described in RG 1.194 and provided a discussion comparing the procedures that it had used with the procedures described in RG 1.194. The NRC staff finds this methodology acceptable. Additionally, the NRC staff performed comparative calculations and finds the χ/Q values calculated by the licensee acceptable.

FHA Control Room χ/Q Values

The FHA was modeled as a ground-level release from the Reactor Building siding in the same manner as described above for the LOCA calculation. Based on its review, the NRC staff finds the resultant χ/Q estimates acceptable.

3.2.5.3 Offsite Atmospheric Dispersion Factors

The licensee calculated new χ/Q values for a postulated ground-level release from the Reactor Building bypass and siding to the EAB and a ground-level release to the LPZ. The licensee used a computer code similar to the PAVAN computer code which is based on methodology described in RG 1.145. The NRC staff made comparative calculations using the PAVAN methodology and found the χ/Q values calculated by the licensee acceptable.

3.2.5.4 Conclusions Regarding Atmospheric Relative Concentration Estimates

The NRC staff reviewed the licensee's calculations for the new χ/Q values used in the dose assessment described above. The staff qualitatively reviewed the inputs to the licensee's calculations and found them to be consistent with site configuration drawings and staff practice. Table 6 provides the new χ/Q values calculated by Entergy. Based on this review, the NRC staff finds the new χ/Q values acceptable.

3.2.6 Post-Accident Access to Vital Areas

TID-14844 includes a radiological source term that was used in the original licensing of nuclear power reactors. Item II.B.2. of NUREG-0737, "Clarification of TMI Action Plan Requirements" (Reference 35), recommends that the licensee demonstrate by calculations, that plant radiation shielding design is sufficient to allow personnel access to vital areas of the plant in the post-accident environment, assuming that the design basis source term, as detailed in TID-14844, is contained in reactor fluid (liquid and gas) bearing plant systems. The AST defined in RG 1.183 is an alternative to this earlier radiological source term.

By letter dated October 10, 2003, Entergy submitted an analysis addressing this post-accident personnel access issue. By comparing the time-dependent gamma radiation emission characteristics of the isotopic mix assumed in the AST to the isotopic mix of the TID-14844 source term, the licensee demonstrated that the current design basis calculations using the TID source term is conservative and bounding.

Although the AST assumes a significantly higher release of cesium from the core (25 percent (AST) versus 1 percent (TID)), this is offset by the substantially smaller fraction of radioactive iodine assumed released (30 percent (AST) versus 50 percent (TID)). In addition, the 1 percent of all other particulates assumed released in TID-14844 substantially exceeds the amount of these nuclides assumed released by the AST.

Based on its review of the information presented by the licensee, the NRC staff finds that the licensee's evaluation concerning post-accident access to vital areas is consistent with the guidance of RG 1.183 and is acceptable.

3.3 Proposed TS Changes

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

a. In Section 1.0, the definition of “Dose Equivalent Iodine-131,” would be changed to add reference to Federal Guidance Report (FGR) 11, “Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion, and Injection,” 1988, and FGR 12, “External Exposure to Radionuclides in Air, Water, and Soil,” 1993. The word “thyroid,” would be deleted.

   The licensee’s revised accident analyses used dose conversion factors from FGR 11 and FGR 12. Both of these references are cited in the guidance in RG 1.183. Therefore, this change conforms with the implementation of AST and is acceptable.

   As discussed above in SE Section 2.0, as part of the implementation of the AST, the TEDE acceptance criterion of 10 CFR 50.67(b)(2) replaces the previous whole body and thyroid dose guidelines of 10 CFR 100.11 and 10 CFR Part 50, Appendix A, GDC 19. Therefore, the deletion of the word “thyroid,” conforms with the implementation of AST and is acceptable.

b. TS 3.7.A.4, would be revised and subdivided into TS 3.7.A.4.a, b, and c. TS 3.7.A.4 would be changed to:

4. Whenever primary containment integrity is required:

a. The leakage rate from any one main steam isolation valve (MSIV) shall not exceed 62 scfh at 44 psig (Pa);

a. The combined leakage rate from the main steam pathways shall not exceed 124 scfh at 44 psig (Pa); and

b. The combined leakage rate from the secondary containment bypass pathways shall not exceed 5 scfh at 44 psig (Pa).

Surveillance Requirement (SR) TS 4.7.A.4 would also be revised and subdivided into TS 4.7.A.4.a, b, and c. The specific test criteria of TS 4.7.A.4 would be relocated to the Primary Containment Leakage Rate Testing Program (PCLRTP). TS 4.7.A.4 would be changed to:

3. In accordance with the PCLRTP, verify that the following leakage rates are within acceptable limits:

a. The leakage rate through each MSIV;

b. The combined leakage rate for the main steam pathways; and

c. The combined leakage rate for the secondary containment bypass pathways.

Entergy analyzed the main steam leakage pathway (with an increase in leakage from 62 scfh to 124 scfh at Pa), the secondary containment bypass leakage pathways, and the containment leakage pathway (La) separately in its dose consequence analyses. The licensee determined the MSIV leakage rates using the methodology described in NEDC-31858P-A, Revision 2. The calculated radiological consequences of these leakages rates are within the criteria of 10 CFR 50.67 and are, therefore, acceptable.

The specific test criteria are relocated to the PCLRTP. The PCLRTP is a formal program, required by TS 6.7.C, and under the controls of 10 CFR 50.59, "Changes, tests, and experiments." The NRC staff finds that there is reasonable assurance that the licensee will employ maintenance of proper SRs and, therefore, relocation of implementing details to the PCLRTP is acceptable.

c. In TSs 4.7.C.1.a and 4.7.C.1.c, the specified SGTS flow rate of 1500 cfm would be changed to 1550 cfm.

The increase in SGTS flow rate test criteria is due to an increase in the assumed minimum analytical SGTS flow rate in the secondary containment drawdown analysis. Based on the information provided by the licensee, the proposed change does not alter the manner in which the facility is operated or maintained, is consistent with the AST analyses and conforms with the performance capability and requirements of ensuring SGTS and secondary containment operability. The change is, therefore, acceptable.

d. TS 6.7.C, which provides the details for the PCLRTP, would be revised as follows:

1. The word “leak” would be changed to “leakage” in several places. This change is editorial in nature and, therefore, is acceptable.

2. The first paragraph of TS 6.7.C and the leakage rate acceptance criteria in TS 6.7.C.3 and 6.7.C.4 would be revised to reflect that the leakage contributions from the main steam pathways are excluded from both the sum of the leakage rates from Type B and Type C tests and the overall integrated leakage rate from Type A tests.

   The main steam leakage effluent has a different pathway to the environment, when compared to a typical containment penetration. It is not directed into the secondary containment and filtered through the SGTS as is other containment leakage. Instead, the main steam leakage is collected and treated via the ALT path having different mitigation characteristics.

   In performing accident analyses, it is appropriate to group various leakage effluents according to the treatment they receive before being released to the environment (e.g., from main steam pathways). The proposed change would more appropriately permit ALT pathway leakage to be independently grouped with its unique leakage limits. In this manner, the VYNPS PCLRTP will be made more consistent with the limiting assumptions used in the associated accident consequence analyses. As previously noted, the calculated radiological consequences of the combined leakages are within the criteria of 10 CFR 50.67.

   The NRC staff concludes that the proposed changes are acceptable because: (1) the leakage rates for the subject pathways will be contained in TS 3.7.A.4.a, 3.7.A.4.b, and 3.7.A.4.c; (2) the leakage rate acceptance criteria from all measured pathways are consistent with the leakage rates assumed in the AST analyses; and (3) the sum of the limiting leakage rates from all leakage pathways does not result in radiological doses exceeding the limits specified in 10 CFR 50.67. In addition, the proposed TS changes conform to the associated exemption from 10 CFR Part 50, Appendix J, which was granted by the NRC on March 17, 2005 (Reference 36).

3. TS 6.7.C would be revised to correct a typographical error such that the acceptance criterion is changed from "< 1.0 La" to "≤ 1.0 La." This typographical error was inadvertently introduced in the final TS pages for Amendment No. 215.

However, this change was not proposed as part of that amendment request. As such, the NRC staff concludes that the proposed change is administrative in nature and, therefore, is acceptable.

e. Changes would also be made to the TS Bases for clarity and to conform with the changes being made to the associated TSs. The NRC staff has no objection to these changes.

3.4 NRC Conclusions Concerning Full Implementation of AST

In Reference 1, as supplemented, Entergy proposed a full-scope implementation of the AST. Based on the above evaluation, the NRC staff finds that the licensee has met the requirements of 10 CFR 50.67 and the guidance of RG 1.183 for a full-scope implentation.

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

Since these analyses were performed at a power level of 1950 MWt (102 percent of 1912 MWt), the NRC staff finds that the radiological consequences of these DBAs would remain bounding up to a rated thermal power of 1912 MWt. However, the approval of this amendment does not constitute authority to operate above the current licensed rated thermal power of 1593 MWt.

This licensing action is considered a full implementation of the AST. With this approval, the previous accident source term in the VYPNS design basis is superceded by the AST proposed by Entergy in its application of July 31, 2003, as supplemented. The previous offsite and control room accident dose criteria expressed in terms of whole body, thyroid and skin doses are superceded by the TEDE criteria of 10 CFR 50.67 or small fractions thereof, as defined in RG 1.183. All future radiological analyses performed to demonstrate compliance with regulatory requirements shall address all characteristics of the AST and the TEDE criteria as described in the VYNPS design basis.

4.0 STATE CONSULTATION

In accordance with the Commission's regulations, the Vermont State official was notified of the proposed issuance of the amendment. The State of Vermont provided comments on the proposed amendment by letter dated June 9, 2004 (Reference 37). Previous to this notification, in a letter dated August 8, 2003 (Reference 38), the State of Vermont had posed several questions to the NRC staff regarding the proposed amendment. The NRC staff responded to these questions in a letter dated December 16, 2003 (Reference 39).

In its letter dated June 9, 2004, the State of Vermont stated that the AST proposal should not be approved without modifications. The State provided three comments as the reasons for its position. The NRC staff's resolution of these comments is as follows:

Comment 1: The SLC system does not appear to meet the single failure criteria appropriate for a system used to mitigate the consequences of a DBA.

Response: As background to the above comment, the State of Vermont's letter noted that the VYNPS SLC system was not designated as an engineered safety features (ESF) system in its original design. However, since the proposed AST amendment credits the SLC system for a DBA mitigation function (i.e., pH control of the suppression pool), the State believes that the system should now be evaluated as an ESF system. The letter references VYNPS UFSAR Section 1.5.6 which states, in part, that essential safety actions shall be carried out by redundant and independent equipment so that no single failure of an active component can prevent the required actions.

Reference 41 provides guidance used by the NRC staff to assess the acceptability of reliance on the SLC system to control the pH of the water in a BWR suppression pool following a LOCA. As noted in this guidance document, BWR SLC systems were originally designed as backup systems to shut down the reactor if the control rods failed to function. SLC systems are also used to shut down the reactor during an anticipated transient without scram (ATWS) event. At some facilities, such as VYNPS, a new function has been proposed for the SLC system as part of AST amendment requests. This new function is the control of the pH in the suppression pool following a design-basis LOCA. Use of the SLC system for pH control would mitigate the release of radioactive iodine in the containment atmosphere following a LOCA.

Since the proposed pH control function would be used for accident mitigation, the system was evaluated by the NRC staff as an ESF system. As discussed in Reference 41, associated with the performance of an accident mitigation function are: (1) high reliability, usually demonstrated by the capability to overcome a single "active" failure and (2) an expected quality (safety-related) for the system and components designated to perform such a function. In developing the guidelines for reviewing the SLC reliability and quality for AST, the NRC staff considered the following guidance from RG 1.183:

> 5.1.2 Credit for Engineered Safeguard Features
>
> 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. (Emphasis added.)

Reference 41 discusses alternative approaches for demonstrating reliability and quality when the RG 1.183 guidance is not met. Based on the Reference 41 guidance, the NRC staff provided the following request for additional information question to the licensee with respect to SLC and the single failure criterion:

> The SLC system should not be rendered incapable of performing its AST function due to a single failure of an active component. For this purpose the check valve is considered an active device for AST since the check valve must open to inject sodium pentaborate for suppression pool pH control.
>
> If the SLC system can not be considered redundant with respect to its active components, this lack of redundancy may be offset by providing information in (a) or (b) or (c) below:
>
> (a) Show acceptable quality and reliability of the non-redundant active components and/or compensatory actions in the event of failure of the non-redundant active components.
>
> If you choose this option, provide the following information to justify the lack of redundancy of active components in the SLC system:
>
> (1) Identify the non-redundant active components in the SLC system and provide their make, manufacturer, and model number.
>
> (2) Provide the design-basis conditions for the component and the environmental and seismic conditions under which the component may be required to operate during a design-basis accident. Environmental conditions include design-basis pressure, temperature, relative humidity and radiation fields.
>
> (3) Indicate whether the component was purchased in accordance with Appendix B to 10 CFR Part 50. If the component was not purchased in accordance with Appendix B, provide information on the quality standards under which it was purchased.

(4) Provide the performance history of the component both at the licensee's facility and in industry databases such as EPIX and NPRDS.

(5) Provide a description of the component's inspection and testing program, including standards, frequency, and acceptance criteria.

(6) Indicate potential compensating actions that could be taken within an acceptable time period to address the failure of the component. An example of a compensating action might be the ability to jumper a switch in the control room to overcome its failure. The staff reviewer will consider the availability of compensating actions and the likelihood of successful injection of the sodium pentaborate where non-redundant active components fail to perform their intended functions.

OR

(b) Provide for an alternative success path for injecting chemicals into the suppression pool.

If you choose to address the SLC system's susceptibility to single failure by selecting an alternative injection path, the alternative path must be capable of performing the AST function noted above and all components which make up the alternative path should meet the same quality characteristics required of the SLC system (described in Items 1(a)-1(e), 2 and 3 above). Provide a description of the alternative injection path, its capabilities, and quality characteristics.

If the use of an alternate path is part of the Emergency Operating Procedures (EOPs), then the license amendment request needs to address the following items: (1) Does the alternate injection path require actions in areas outside the control room? (2) How accessible will these areas be? (3) What additional personnel will be required?

OR

(c) Show that 10 CFR 50.67 and Appendix A, GDC 19 doses are met even if pH is not controlled.

You may choose to demonstrate, through dose calculations, that 10 CFR 50.67 and GDC 19 (or equivalent used in original licensing) doses are met even if pH is not controlled. The re-evolution of iodine in the particulate form from the water in the suppression pool to the elemental form for airborne iodine must be incorporated into the calculation. The calculation may take credit for the mitigating capabilities of other equipment, for example the standby gas treatment system (SGTS), if such equipment would be available. If you choose this option, please provide the dose calculations (including all inputs and assumptions) and any supporting calculations on re-evolution of iodine.

The licensee provided its response to the above question in a letter dated February 25, 2004 (Reference 11). Further information related to this question was provided in a letter dated July 20, 2004 (Reference 40). The licensee responses addressed the considerations in option (a). As discussed in SE Section 3.2.1.1 (sub-section titled "SLC System Redundancy") the NRC staff determined that the non-redundant components of the VYNPS SLC system are of sufficient quality and reliability, or compensatory measures can be taken, to ensure that SLC injection will occur when required.

Since the SLC system will have an ESF function following implementation of this proposed amendment, the NRC staff expects that the licensee will revise the affected UFSAR sections, including Section 1.5.6, to indicate that VYNPS meets the intent of the single failure criteria for this new SLC function through the quality and reliability of the non-redundant components, and by compensatory measures.

Comment 2: The MSIV ALT pathway does not appear to meet the quality standards appropriate for a system used to mitigate the consequences of a DBA.

Response: As discussed in the NRC staff's SE (Reference 19) for BWROG Topical Report NEDC-31858P-A, Revision 2 (Reference 18), the staff has accepted, on a generic basis, the use of the ALT pathway to mitigate the consequences of leakage past the MSIVs. In Reference 19, the staff acknowledges that requiring the non-seismically analyzed portions of the ALT to meet Seismic Category 1 requirements cannot be justified from a cost-benefit standpoint. The staff determined that the BWROG's approach of utilizing the earthquake experience-based methodology, supplemented by plant-specific seismic walkdowns and analytical evaluations, provides a viable alternative for demonstrating the seismic ruggedness of the non-seismically analyzed ALT. VYNPS has evaluated the ALT in accordance with the approved topical report (Reference 18), including the limitations identified in the staff's SE (Reference 19).

As discussed above in SE Section 3.1.7, on the basis of the information provided by the licensee, the NRC staff concludes that the piping and components which comprise the ALT pathway are seismically rugged and are able to perform the safety function of an MSIV leakage treatment system.

Comment 3: There is no reason to reduce safety margins for Vermonters by doubling the amount of allowed leakage from MSIVs from the leakage levels Vermont Yankee has met for the past 32 years.

Response: The licensee determined the MSIV leakage rates using the methodology described in NEDC-31858P-A, Revision 2. The calculated radiological consequences of this leakage rate are within the criteria of 10 CFR 50.67, and are, therefore, acceptable. Maintaining radiological consequences within the regulatory limits, ensures that adequate safety margins exist and, therefore, the NRC staff has reasonable assurance that public health and safety will not be endangered by operation at the proposed leakage rates.

5.0 ENVIRONMENTAL CONSIDERATION

The amendment changed a requirement with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 and changed SRs. The NRC staff has determined that the amendment involves no significant increase in amounts, and no significant change in the types of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendment involves no significant hazards consideration, and there has been no public comment on such finding (68 FR 66135). Accordingly, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9) and (10). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendment.

6.0 CONCLUSION

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

7.0 REFERENCES

1. Letter from J. K. Thayer, Entergy, to NRC Document Control Desk, "Vermont Yankee Nuclear Power Station, License No. DPR-28 (Docket No. 50-271), Technical Specification Proposed Change No. 262, Alternative Source Term," July 31, 2003.

2. Letter from J. K. Thayer, Entergy, to NRC Document Control Desk, "Vermont Yankee Nuclear Power Station, License No. DPR-28 (Docket No. 50-271), Technical Specification Proposed Change No. 262 - Supplement No. 1, Alternative Source Term," October 10, 2003.

3. Letter from J. K. Thayer, Entergy, to NRC Document Control Desk, "Vermont Yankee Nuclear Power Station, License No. DPR-28 (Docket No. 50-271), Technical Specification Proposed Change No. 262 - Supplement No. 2, Alternative Source Term - Seismic Verification Reports," November 7, 2003.

4. Letter from J. K. Thayer, Entergy, to NRC Document Control Desk, "Vermont Yankee Nuclear Power Station, License No. DPR-28 (Docket No. 50-271), Technical

Specification Proposed Change No. 262 - Supplement No. 3, Alternative Source Term - Modified Exemption Request," November 7, 2003.

5. Letter from J. K. Thayer, Entergy, to NRC Document Control Desk, "Vermont Yankee Nuclear Power Station, License No. DPR-28 (Docket No. 50-271), Technical Specification Proposed Change No. 262 - Additional Information, Alternative Source Term - Copyright Release," November 20, 2003.

6. Letter from J. K. Thayer, Entergy, to NRC Document Control Desk, "Vermont Yankee Nuclear Power Station, License No. DPR-28 (Docket No. 50-271), Technical Specification Proposed Change No. 262 - Alternative Source Term - Response to Request for Additional Information," December 11, 2003.

7. Letter from J. K. Thayer, Entergy, to NRC Document Control Desk, "Vermont Yankee Nuclear Power Station, License No. DPR-28 (Docket No. 50-271), Technical Specification Proposed Change No. 262 - Supplement No. 4, Alternative Source Term - Calculations," December 11, 2003.

8. Letter from J. K. Thayer, Entergy, to NRC Document Control Desk, "Vermont Yankee Nuclear Power Station, License No. DPR-28 (Docket No. 50-271), Technical Specification Proposed Change No. 262 - Alternative Source Term - Second Response to Request for Additional Information," December 30, 2003.

9. Letter from J. K. Thayer, Entergy, to NRC Document Control Desk, "Vermont Yankee Nuclear Power Station, License No. DPR-28 (Docket No. 50-271), Technical Specification Proposed Change No. 262 - Supplement 8, Alternative Source Term - Request for Additional Information," February 10, 2004.

10. Letter from J. K. Thayer, Entergy, to NRC Document Control Desk, "Vermont Yankee Nuclear Power Station, License No. DPR-28 (Docket No. 50-271), Technical Specification Proposed Change No. 262 - Supplement 9, Alternative Source Term - Response to Request for Additional Information," February 18, 2004.

11. Letter from J. K. Thayer, Entergy, to NRC Document Control Desk, "Vermont Yankee Nuclear Power Station, License No. DPR-28 (Docket No. 50-271), Technical Specification Proposed Change No. 262 - Supplement 10, Alternative Source Term - Response to Request for Additional Information," February 25, 2004.

12. Letter from J. K. Thayer, Entergy, to NRC Document Control Desk, "Vermont Yankee Nuclear Power Station, License No. DPR-28 (Docket No. 50-271), Technical Specification Proposed Change No. 262 - Supplement 11, Alternative Source Term - Meteorological Database for Ground-Level Releases," March 17, 2004.

13. Letter from J. K. Thayer, Entergy, to NRC Document Control Desk, "Vermont Yankee Nuclear Power Station, License No. DPR-28 (Docket No. 50-271), Technical Specification Proposed Change No. 262 - Supplement 12, Alternative Source Term - Response to Request for Additional Information," May 12, 2004.

14. U.S. Atomic Energy Commission, TID-14844, "Calculation of Distance Factors for Power and Test Reactor Sites," March 23, 1962.

15. NRC RG 1.183, "Alternative Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors," July 2000.

16. NRC SRP 6.5.2, "Containment Spray as a Fission Product Cleanup System," Revision 2, 1988.

17. NRC SRP 15.0.1, "Radiological Consequences Analyses Using Alternative Source Terms," Revision 0, July 2000.

18. GE Topical Report, NEDC-31858P-A, Revision 2, "BWROG Report for Increasing MSIV Leakage Rate Limits and Elimination of Leakage Control Systems," August 1999.

19. NRC, "Safety Evaluation of GE Topical Report, NEDC-31858P-A, Revision 2, BWROG Report for Increasing MSIV Leakage Rate Limits and Elimination of Leakage Control Systems," March 3, 1999.

20. Entergy, VYNPNS USFAR, Revision 19.

21. American Society of Mechanical Engineers, "Power Piping," USA Standard Code for Pressure Piping USAS B31.1-1967.

22. SQUG, "Generic Implementation Procedure (GIP) for Seismic Verification of Nuclear Plant Equipment," Revision 2 corrected, February 14, 1992.

23. American Concrete Institute, "Building Code Requirement for Reinforced Concrete," ACI 318-99.

24. NRC NUREG/CR-6604, RADTRAD: A simplified Model for RADionuclide Transport and Removal And Dose Estimate," April 1998, and Supplement 1, June 1999.

25. NRC NUREG-1465, “Accident Source Terms for Light Water Reactors for Light-Water Nuclear Power Plants," February 1995.

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

27. NRC NUREG-1081, "Post-Accident Gas Generation from Radiolysis of Organic Materials," September 1984.

28. NRC RG 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.

29. AEB-98-03, "Assessment of Radiological Consequences for the Perry Pilot Plant Application Using the Revised (NUREG-1465) Source Term," December 9, 1998.

30. NRC, "Issuance of Amendment Re: Main Steamline Radiation Monitor (TAC No. MB4610)," September 18, 2002.

31. NRC NUREG-0917, "Nuclear Regulatory Commission Staff Computer Programs for Use with Meteorological Data," July 1982.

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

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

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

36. NRC, "Issuance of Exemption from 10 CFR Part 50, Appendix J (TAC No. MC0253)," March 17, 2005.

37. Letter from David O’Brien, Vermont Department of Public Service, to Richard Ennis, NRC, “Alternate Source Term - State of Vermont Comments,” June 9, 2004.

38. Letter from William K. Sherman, Vermont Department of Public Service, to Robert Pulsifer, NRC, “Alternate Source Term - State of Vermont Questions,” August 8, 2003.

39. Letter from Richard B. Ennis, NRC, to William K. Sherman, Vermont Department of Public Service, December 16, 2003.

40. Letter from J. K. Thayer, Entergy, to NRC Document Control Desk, "Vermont Yankee Nuclear Power Station, License No. DPR-28 (Docket No. 50-271), Technical Specification Proposed Change No. 262 - Supplement No. 13, Alternative Source Term," July 20, 2004.

41. “Guidance on the Assessment of a BWR SLC System for pH Control,” February 12, 2004.

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Date: March 29, 2005

TABLE 1
Licensee Calculated Radiological Consequences of DBAs (TEDE rem)

DBA	EAB	LPZ	Control Room
LOCA			
w/ SGTS Failure	3.14	0.53	3.40
w/ MSIV Failure	2.44	0.50	2.00
MSLB			
1.1 µCi/gm DE I-131	0.98	<0.98	0.55
4.0 µCi/gm DE I-131	3.57	<3.57	2.00
FHA	0.47	<0.47	0.15
CRDA			
Case 1	0.27	0.018	0.35
Case 2	0.17	0.021	0.0013
Case 3	0.11	0.060	0.048
Case 1 + Case 3	0.38	0.078	0.40
Case 2 + Case 3	0.28	0.081	0.049

Dose Acceptance Criteria, TEDE (rem)

DBA	EAB	LPZ	Control Room
LOCA	25	25	5
MSLB			
Equilibrium Activity	2.5	2.5	5
Pre-incident Spike	25	25	5
FHA	6.3	6.3	5
CRDA	6.3	6.3	5

TABLE 2
Assumptions for LOCA Analysis

Parameter	Value
Core Power, MWt (102 percent of 1912 MWt)	1950
Core inventory	Calculated by ORIGEN
Core release fractions and timing	RG 1.183, Tables 1 and 4
Iodine species fraction	
Particulate/aerosol	95
Elemental	4.85
Organic	0.15
Drywell volume, ft^3	128,370
Torus airspace volume, ft^3	103,932
Suppression pool liquid volume, ft^3	68,000
Containment leakage, volume percent/day	
0 - 24 hours	0.8
> 24 hours	0.4
Secondary containment bypass leakage, scfh	5
MSIV leakage (total), scfh	124
MSIV leakage that bypasses main condenser, scfh	5
ECCS leak rate, gpm	1
ECCS leakage release fraction	0.1
Secondary containment drawdown with 1 train, minutes	10
SGTS filter efficiency, percent	
Particulates	95
Elemental	95
Organic	95
Drywell spray initiation time, minutes	15
Drywell spray removal coefficients for particulates, hr^{-1}	
0.25 - 2.033 hrs	20
2.033 - 2.068 hrs	11.3
> 2.068 hrs	1.13
Drywell spray removal coefficients for elemental iodine, hr^{-1}	20
Elemental iodine decontamination factor limit	200
Particulate deposition efficiency in piping, percent	
Steamline leakage between MSIVs	38
Alternate leakage treatment (ALT) pathway	71
Combined steamlines and ALT	82
as above with one MSIV failed	77
Main condenser	95.1
Elemental iodine deposition efficiency in piping, percent	
Alternate leakage treatment (ALT) pathway	58
Combined steamlines and ALT	58
as above with one MSIV failed	58
Main condenser	99.8
Dose conversion factors	FGR11 and FGR12
Atmospheric dispersion factors	Table 6
Control room modeling	Table 7

TABLE 3
Assumptions for MSLB Analysis

Parameter	Value
RCS activity	
Equilibrium iodine case	1.1 µCi/gm D.E.I-131
Pre-incident iodine spike case	4.0 µCi/gm D.E.I-131
Iodine species release fraction to environment	
Elemental	0.97
Organic	0.03
Mass release	
Steam, lbm	21,798
Liquid, lbm	37,702
Break isolation time, sec	6.8
Dose conversion factors	FGR11 and FGR12
Atmospheric dispersion factors	Table 6
Control room modeling	Table 7

TABLE 4
Assumptions for FHA Analysis

Parameter	Value
Reactor power, MWt,	1950
Radial peaking factor	1.65
Fuel decay period, hours	24
Number of damaged fuel rods	193
Equivalent to damaged fuel assemblies	2.1
Total number of fuel assemblies in core	368
Fraction of gap activity released from damaged rods	1.0
Fraction of core inventory in gap	
I-131	0.08
Kr-85	0.10
Other halogens and noble gases	0.05
Pool decontamination factor, effective	200
Iodine species fraction above pool water	
Elemental	0.57
Organic	0.43
Release duration, hours	
From fuel and pool	Instantaneous
From secondary containment	2
Collection and filtration by SGTS	None
Assumed release point, 80 percent of release	Plant stack
Assumed release point, 20 percent of release	Ground level
Dose conversion factors	FGR11 and FGR12
Atmospheric dispersion factors	Table 6
Control room modeling	Table 7

TABLE 5
Assumptions for CRDA Analysis

Parameter	Value
Reactor power, MWt	1950
Core inventory	Calculated by ORIGEN
Radial peaking factor	1.5
Fuel assemblies in core	368
Fuel rods in assembly	60
Rods that exceed DNB	850
Fraction of rods that exceed DNB that experience melt	0
Gap fraction	
Noble gas and iodine	0.10
Dose conversion factors	FGR11 and FGR12
Atmospheric dispersion factors	Table 6
Control room modeling	Table 7
Case 1: Leakage from main condenser based on manual isolation of MSIVs	
Fraction of core release that enters condenser	
Noble gases	1.0
Iodine	0.1
Iodine retention in condenser	0.9
Iodine species release fraction to environment	
Elemental	0.97
Organic	0.03
Condenser leakage, percent/day	1.0
Release duration, hours	24
Case 2: Release from AOG system when MSIVs remain open	
AOG charcoal delay times	
Iodines	Infinite
Kryptons	24 hours
Xenons	16.6 days
Case 3: Release from RCS recirculation lines in the reactor building (added to Case 1 or Case 2)	
RCS sampling line flow rate, gph	32
Coolant mass assumed to mix with iodine, lbm	393,187

TABLE 6
Vermont Yankee Relative Concentration (X/Q) Values (sec/m^3)

Receptor/Source	Accidents	0-0.5 hr	0.5-1 hr	1-2 hr	2-8 hr	8-24 hr	1-4
EAB							
Ground*	LOCA, MSLB FHA, CRDA	1.69 E-03			N/A	N/A	N
Ground (RB bypass/siding)	LOCA	1.476 E-03			N/A	N/A	N
Stack*	LOCA, FHA CRDA	2.03 E-04	1.54 E-04	9.17 E-05	N/A[†]	N/A[†]	N
LPZ							
Ground	LOCA, CRDA	5.25 E-05			2.23 E-05	1.47 E-05	5.95
Stack*	LOCA, CRDA	2.55 E-05	2.55 E-05	1.87 E-05	1.01 E-05	1.09 E-06	6.90
Control Room/TSC							
Ground (MSIV - turbine bldg)	LOCA	4.66 E-03			3.46 E-03	1.45 E-03	1.09
Ground (RB bypass)	LOCA	2.25 E-03			8.18 E-04	3.53 E-04	2.77
Ground (RB siding)	LOCA	2.98 E-03	N/A	N/A	N/A	N/A	N
Stack	LOCA	1.92 E-05			8.28 E-07	3.36 E-07	3.08
Puff	MSLB	1.44 E-03			N/A	N/A	N
Ground	FHA	5.89 E-03			N/A	N/A	N
Stack*	FHA	2.39 E-04	1.05 E-06	8.70 E-07	N/A	N/A	N
Ground*	CRDA	3.67 E-03	3.67 E-03	2.19 E-03	7.57 E-04	3.93 E-04	2.71
Stack*	CRDA	2.39 E-04	1.05 E-06	8.70 E-07	4.79 E-07	2.34 E-07	1.23

* X/Q values for these receptor/source pairs are discussed in the safety evaluation report associated with Amendment 212 dated September 18, 2002.

[†] Licensee provided values for these time periods, but staff does not approve the values because EAB calculations should use the X/Q values that apply to the two-hour time period having the most limiting dispersion.

TABLE 7
Control Room Modeling Assumptions

Parameter	Value
No control room isolation assumed for any DBA	
Control room volume, ft^3	41,534
Normal mode ventilation, scfm	>9100
Fresh air intake, scfm	3700
Assumed unfiltered inleakage, scfm	3700
Breathing rate, m^3/sec	3.5E-4
Control room occupancy factors	
0 - 1 day	1.0
1 - 4 days	0.6
4 - 30 days	0.4
Control room atmospheric dispersion factors	Table 6