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BVY 03-70

U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, DC 20555

Subject: **Vermont Yankee Nuclear Power Station**
License No. DPR-28 (Docket No. 50-271)
Technical Specification Proposed Change No. 262
Alternative Source Term

Pursuant to 10CFR50.90 and 10CFR50.67, Vermont Yankee[1] (VY) hereby proposes to amend its Facility Operating License, DPR-28, by incorporating a revision to the licensing basis of the Vermont Yankee Nuclear Power Station (VYNPS) that supports a full scope application of an Alternative Source Term (AST) methodology. Associated, proposed Technical Specification (TS) changes, which are supported by the AST analyses, are included in this application for a license amendment. In addition, VY is requesting a specific exemption from 10CFR50.54(o) and the requirements of Sections III.A and III.B of 10CFR50, Appendix J, Option B.

10CFR50.67, "Accident Source Term," provides a mechanism for currently licensed nuclear power reactors to replace the traditional source term used in design basis accident analyses with an alternative source term. Under this provision, licensees who seek to revise the accident source term in design basis radiological consequence analyses must apply for a license amendment under 10CFR50.90.

Full Scope AST analyses were performed by VY in accordance with the guidance in Regulatory Guide 1.183[2], and Section 15.0.1 of the Standard Review Plan[3]. VY performed AST analyses for the four design basis accidents that could potentially result in significant control room and offsite doses. These include the loss of coolant accident, the main steam line break accident, the refueling accident, and the control rod drop accident. The analyses demonstrate that using AST methodologies, post-accident control room and offsite doses remain within regulatory acceptance limits.

VY proposes implementation of this Proposed Change through a change to the VYNPS licensing basis, including the TS and associated Bases. Upon approval, conforming changes will be made to the VYNPS Updated Final Safety Analysis Report (UFSAR) and submitted to the NRC staff in accordance with 10CFR50.71 as part of the regular UFSAR update process.

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[1] Entergy Nuclear Vermont Yankee, LLC and Entergy Nuclear Operations, Inc. are the licensees of the Vermont Yankee Nuclear Power Station.

[2] U.S. Nuclear Regulatory Commission, Regulatory Guide 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors," July 2000.

[3] U.S. Nuclear Regulatory Commission , NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants – LWR Edition," Section 15.0.1, "Radiological Consequence Analyses Using Alternative Source Terms," Rev. 0, July 2000.

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Proposed changes in the licensing basis for VYNPS resulting from application of the AST include the following:

- Revisions to the primary containment leakage rate testing program, including changes to the TS and a proposed exemption to Sections III.A and III.B of 10CFR50, Appendix J, Option B.
- Revised test criteria for periodic TS surveillances of the secondary containment.
- Credit for use of the standby liquid control (SLC) system to buffer suppression pool pH to prevent iodine re-evolution following a postulated design basis loss-of-coolant accident (LOCA).
- New offsite atmospheric dispersion factors (X/Qs) for ground level releases.
- Revised TS definition of Dose Equivalent Iodine I-131.
- Various references to 10CFR100 in the TS Bases are being changed to 10CFR50.67 to reflect adoption of the Alternative Source Term.

Table 6 of Attachment 1 provides a description of each proposed TS change.

The current operating license allows VYNPS to operate at a maximum steady-state power level of 1593 megawatts thermal (MWt). VY is currently engaged in an Extended Power Uprate (EPU) project to increase the maximum licensed thermal power to 1912 MWt. Therefore, the AST analyses which have been performed consider the core isotopic values at EPU conditions and this application for license amendment is based on a bounding core isotopic inventory. The analyses are also applicable to operation in the maximum extended load line limit (MELLLA) power-flow condition as proposed by VY[4].

The use of an AST results in changes in the design basis accident radiological consequences; however, the AST methodology has no direct impact on the probability or initiation of the evaluated design basis accidents. Application of AST methodology and the other changes requested by this application for a license amendment do not increase the core damage frequency or the large early release frequency. Therefore, this request for a revision to VYNPS's licensing basis is not being submitted as a "risk-informed licensing action" as defined by Regulatory Guide 1.174.[5]

Several domestic boiling water reactors (Duane Arnold, Brunswick Units 1 and 2, Grand Gulf, Hope Creek, Clinton, and Perry) have previously provided justification for the use of AST methodology utilizing a similar approach. These applications of AST methodology have been approved by NRC.

Attachment 1 to this letter contains a description and summary safety assessment of each proposed TS change. Also, included in Attachment 1 is a request for a regulatory exemption that VY requests the NRC staff grant concurrently with the license amendment. Attachment 2 contains the determination of no significant hazards consideration. Attachment 3 provides a mark-up of the current TS and TS Bases pages indicating the proposed changes. Attachment 4 provides the retyped TS and TS Bases pages. Attachment 5 provides the AST Safety Assessment for VYNPS, and Attachment 6 consists of eight calculations that support the Safety Assessment. Three of the calculations are considered proprietary information to Polestar Applied Technology, Inc. The three calculations are clearly marked as proprietary

[4] Vermont Yankee letter to U.S. Nuclear Regulatory Commission, "Technical Specification Proposed Change No. 257, Implementation of ARTS/MELLLA at Vermont Yankee (BVY 03-23)," March 20, 2003.
[5] U.S. Nuclear Regulatory Commission, Regulatory Guide 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis," Revision 0, July 1998.

and should be withheld from public disclosure in accordance with 10CFR2.790. Polestar's affidavit for proprietary information is contained within Attachment 6. Also, included in Attachment 6 are non-proprietary versions of the same three calculations.

VY has reviewed the proposed change to the current licensing basis in accordance with 10CFR50.92 and concludes that the proposed change does not involve a significant hazards consideration. VY has also determined that the proposed change satisfies the criteria for a categorical exclusion in accordance with 10CFR51.22(c)(9) and does not require an environmental review. Therefore, pursuant to 10CFR51.22(b), no environmental impact statement or environmental assessment needs to be prepared for this change.

VY requests that this application for a license amendment be approved by March 2004 to support activities planned for the next, scheduled refueling outage.

If you have any questions, please contact Mr. Len Gucwa at (802) 258-4225.

Sincerely,

Jay K. Thayer
Site Vice President

STATE OF VERMONT)
)ss
WINDHAM COUNTY)

Then personally appeared before me, Jay K. Thayer, who, being duly sworn, did state that he is Site Vice President of the Vermont Yankee Nuclear Power Station, that he is duly authorized to execute and file the foregoing document, and that the statements therein are true to the best of his knowledge and belief.

Sally A. Sandstrum, Notary Public
My Commission Expires February 10, 2007

Attachments

cc: USNRC Region 1 Administrator
USNRC Resident Inspector - VYNPS
USNRC Project Manager - VYNPS
Vermont Department of Public Service (w/o proprietary information)

Attachment 1

Vermont Yankee Nuclear Power Station

Proposed Technical Specification Change No. 262

Alternative Source Term

Supporting Information and Safety Assessment of Proposed Change

INTRODUCTION

Vermont Yankee (VY) hereby proposes to amend the licensing basis of the Vermont Yankee Nuclear Power Station (VYNPS) through the full scope application of an Alternative Source Term (AST) methodology. Associated, proposed Technical Specification (TS) and TS Bases changes, which are justified by the AST analyses, are included in this application for a license amendment. In addition, VY is requesting that the NRC concurrently grant an exemption to certain provisions of Sections III.A and III.B of 10CFR50, Appendix J, Option B.

Regulatory Basis

This full implementation of AST analyses will modify VYNPS' licensing bases by adopting the AST methodology which replaces the current accident source term with an alternative source term as prescribed in 10CFR50.67 and establishes the 10CFR50.67 total effective dose equivalent (TEDE) dose limits as a new acceptance criterion. The current TID-14844[1] accident source term will remain the licensing basis for equipment qualification purposes.

The current operating license allows VYNPS to operate at a maximum steady-state power level of 1593 megawatts thermal (MWt). VY is currently engaged in an Extended Power Uprate (EPU) project to increase the maximum licensed thermal power to 1912 MWt. Therefore, the AST analyses which have been performed consider the core isotopic values at EPU conditions and this application for license amendment is based on a bounding core isotopic inventory. The analyses also include operation in the maximum extended load line limit (MELLLA) power-flow condition as proposed by VY[2].

Regulatory Guide 1.183 recommends that changes to the UFSAR that reflect the revised analyses be submitted to the NRC staff. Upon issuance of a license amendment, conforming UFSAR changes will be completed as required by VY procedures and submitted to the NRC staff in accordance with the regular UFSAR update process as required by 10CFR50.71. In lieu of providing the NRC staff with proposed UFSAR changes at this time, the supporting DBA calculations are being provided in Attachment 6.

The license amendment would revise VYNPS' licensing basis as follows:

- Revisions to the primary containment leakage rate testing program, including changes to the TS and a proposed exemption to Sections III.A and III.B of 10CFR50, Appendix J, Option B.
- Revised test criteria for periodic TS surveillances of the secondary containment.
- Credit for use of the standby liquid control (SLC) system to buffer suppression pool pH to prevent iodine re-evolution following a postulated design basis loss-of-coolant accident (LOCA).
- New offsite atmospheric dispersion factors (X/Qs) for ground level releases.
- Revised TS definition of Dose Equivalent Iodine I-131.
- Various references to 10CFR100 in the TS Bases are being changed to 10CFR50.67 to reflect adoption of the Alternative Source Term.

[1] J.J. DiNunno et al., "Calculation of Distance Factors for Power and Test Reactor Sites," Technical Information Document (TID)-14844, U.S. Atomic Energy Commission, 1962.
[2] Vermont Yankee letter to U.S. Nuclear Regulatory Commission, "Technical Specification Proposed Change No. 257, Implementation of ARTS/MELLLA at Vermont Yankee (BVY 03-23)," March 20, 2003.

BACKGROUND

On December 23, 1999, the NRC published 10CFR50.67, "Accident Source Term," in the Federal Register. This regulation provides a mechanism for licensed power reactors to replace the current accident source term used in design basis accident (DBA) analyses with an alternative source term. The direction provided in 10CFR50.67 is that licensees who seek to revise their current accident source term in design basis radiological consequence analyses must apply for a license amendment under 10CFR50.90.

Regulatory Guide (RG) 1.183[3] and Standard Review Plan (SRP) Section 15.0.1[4] were used by VY in preparing the AST analyses. These documents were prepared by the NRC staff to address the use of ASTs at current operating power reactors. The RG establishes the parameters of an acceptable AST and identifies the significant attributes of an AST acceptable to the NRC staff. In this regard, the RG provides guidance to licensees for operating power reactors on acceptable applications for an AST; the scope, nature, and documentation of associated analyses and evaluations; consideration of impacts on risk; and acceptable radiological analysis assumptions. The SRP provides guidance to the staff on the review of AST submittals.

Acceptance criteria consistent with that required by 10CFR50.67 were used to replace VYNPS' current design basis source term acceptance criteria. The AST analyses were performed for four DBAs that could potentially result in control room and offsite doses. These include the LOCA, the main steam line break accident, the refueling accident, and the control rod drop accident.

SAFETY ASSESSMENT

Vermont Yankee has performed a full scope implementation of the AST as defined in RG 1.183. A detailed description of the AST analyses is provided in Attachment 5 and the methods and results of the analyses are summarized in this section. Calculations of DBAs using AST methodology are included in Attachment 6. The analyses included the following:

- Identification of the core source term based on plant specific analysis of core fission product inventory.
- Determination of the release fractions for the four DBAs that could potentially result in significant control room and offsite doses.
- Calculation of fission product deposition rates and removal efficiencies.
- Calculation of offsite and control room personnel TEDE.

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[3] U.S. Nuclear Regulatory Commission, Regulatory Guide 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Plants," July 2000.
[4] U.S. Nuclear Regulatory Commission, NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants – LWR Edition," Section 15.0.1, "Radiological Consequence Analyses Using Alternative Source Terms," Rev. 0, July 2000.

- Evaluation of suppression pool pH to ensure that the particulate iodine deposited into the suppression pool during a DBA LOCA does not re-evolve and become airborne as elemental iodine.
- Calculation of new control room atmospheric dispersion factors (X/Q) for a main steam line break accident instantaneous ground level puff release.
- Evaluation of other related design and licensing bases such as NUREG-0737, "Clarification of TMI Action Plan Requirements."
- Calculation of new control room and exclusion area boundary (EAB) atmospheric dispersion factors (X/Q) for reactor building leakage.

The radiological dose analyses for AST have been performed assuming reactor operation at Extended Power Uprate conditions, including 2% calorimetric uncertainty (1950 MWt). This results in a conservative estimate of fission product releases for the current licensed power level.

The AST analyses were performed in accordance with RG 1.183. The results were evaluated to confirm compliance with the acceptance criteria presented in 10CFR50.67 and General Design Criterion (GDC) 19 of 10CFR50, Appendix A. Although VY is a pre-GDC plant, the acceptance criteria of GDC-19 were used for evaluation purposes.

Evaluation

DBAs that potentially result in significant control room and offsite doses were addressed using methods and input assumptions consistent with the AST methodology. The following DBAs were addressed:

- Loss of Coolant Accident (LOCA);
- Main Steam Line Break Accident (MSLB);
- Refueling Accident (RA);
- Control Rod Drop Accident (CRDA);

Results

Loss of Coolant Accident

The radiological consequences of the DBA LOCA were analyzed. The post-accident doses are the result of the following activity considerations:

- Primary to secondary containment leakage. This leakage is directly released into secondary containment and filtered by the standby gas treatment (SGT) system prior to elevated release (after the reactor building drawdown time; during the drawdown time this leakage is assumed to be directly to the environment).
- ECCS leakage into the secondary containment. This leakage is directly released into the secondary containment environment and the airborne portion is filtered by the SGT system prior to elevated release through the plant stack (after the reactor building drawdown time; during the drawdown time this leakage is assumed to be directly to the environment).

- Main steam isolation valve (MSIV) leakage from the primary containment into the main condenser (with a fraction that bypasses the main condenser directly to the atmosphere). Leakage passes through the main steam leakage pathway to the main condenser with credit for deposition before it is released to the environment as a ground level release.

- Reactor building (i.e., secondary containment) bypass leakage. This leakage is an unfiltered ground-level release.

- Post-DBA LOCA radiation shine dose to operators within the control room from activity released to the reactor building.

Loss of Coolant Accident

For the AST LOCA analysis, EAB, low population zone (LPZ), and control room calculated radiological doses are within the regulatory limits of 10CFR50.67. These results are summarized in the following Table 1 along with results for the LOCA analysis using the current source term.

Table 1

LOCA Radiological Consequence Analysis (rem TEDE)			
Dose Component	**Offsite Dose**		**Control Room Dose**
	EAB	**LPZ**	
SGT System Single Failure (limiting case)			
Direct Primary Containment Leakage[5]	1.8	0.08	2.8
Release via RB and Plant Stack	1.3	0.44	0.036
Release via Main Steam Lines and Condenser	0.035	0.0016	0.53
TOTAL	**3.14**	**0.52**	**3.40**
Regulatory Limit	25	25	5
Current Analyses[6] (Regulatory Limit) – rem	4.30E-01 (25) Gamma 9.4E+01 (300) Thyroid	2.80E-01 (25) Gamma 8.4E+00 (300) Thyroid	3.0E-03 (5) Gamma 2.02E+01 (30) Thyroid

[5] Primary containment leakage directly to the environment includes reactor building bypass and reactor building siding pathways.
[6] Current analysis two-hour doses were evaluated at the maximum offsite distance of 1900 m due to the topographical considerations, since there is no effective stack height at this distance. 30-day doses at 8050 m.

Main Steam Line Break Accident

For the main steam line break analysis, the EAB, LPZ, and control room calculated radiological doses are within the regulatory limits of 10CFR50.67 as shown in Table 2. The control room doses were determined using the new X/Q value for an instantaneous ground level release that is conservative relative to the puff methodology in RG 1.194[7]. These results are summarized in Table 2 below along with the results from the current source term analysis.

Table 2

Main Steam Line Break Accident Radiological Consequence Analysis (rem TEDE)			
	Offsite Dose		**Control Room Dose**
Case	**EAB**	**LPZ**	**(puff release)**
1.1 μCi/gm DE I-131	0.981	< 0.981	0.55
4.0 μCi/gm DE I-131	3.57	< 3.57	2.00
Regulatory Limit	2.5/25	2.5/25	5
Current Analyses[8] (Regulatory Limit) - rem		1.2E-02 (25) Gamma 1.4E+01 (300) Thyroid	

[7] U.S. Nuclear Regulatory Commission, Regulatory Guide 1.194, "Atmospheric Relative Concentrations for Control Room Radiological Habitability Assessments at Nuclear Power Plants," June 2003.
[8] The current licensing basis 2-hour dose corresponds to an offsite receptor at 1900 m where the receptor elevation is equal to the stack release elevation.

Refueling Accident

For the AST design basis refueling accident the EAB, LPZ, and control room calculated radiological doses are within the regulatory limits of 10CFR50.67. The results are summarized in Table 3 below along with the results of the current source term analyses.

Table 3

Refueling Accident Radiological Consequence Analysis (rem TEDE)			
	Offsite Dose		**Control Room Dose**
Case	**EAB**	**LPZ**	
24-Hours after shutdown (elevated release)	0.472	< 0.472	0.153
Regulatory Limit	6.3	6.3	5
Current Analyses[9] (Regulatory Limit) - rem	2.7E-02 (25) Gamma 3.2E+01 (300) Thyroid	8.4E-04 (25) Gamma 3.4E+00 (300) Thyroid	

[9] The current licensing basis 2-hour dose corresponds to an offsite receptor at 1900 m where the receptor elevation is equal to the stack release elevation.

Control Rod Drop Accident

The radiological consequences of the design basis control rod drop accident using AST methodology were analyzed. The EAB, LPZ, and control room calculated radiological doses are within the regulatory limits of 10CFR50.67 after AST implementation. The results are summarized in Table 4 below along with the results of the current source term analyses.

Table 4

Control Rod Drop Accident Radiological Consequence Analysis (rem TEDE)			
	Offsite Dose		**Control Room Dose**
Case	**EAB**	**LPZ**	
Limiting Results	3.8E-01	8.1E-02	4.0E-01
Regulatory Limit	6.3	6.3	5
Current Analyses[10] (Regulatory Limit) - rem	1.5E-02 (25) Gamma 2.3E-02 (300) Beta 3.0E+00 (300) Thyroid	7.4E-03 (25) Gamma 1.2E-02 (300) Beta 1.8E+00 (300) Thyroid	9.7E-03 (5) Gamma 3.7E-01 (30) Beta 28 (30) Thyroid

Suppression Pool pH Control

The AST LOCA analysis takes credit for the minimization of re-evolution of elemental iodine from the suppression pool over the long-term post-accident. Re-evolution is strongly dependent on suppression pool pH. The analysis assumes that injection of sodium pentaborate (SPB) via SLC commences within two hours of the onset of a LOCA. Using the assumptions of a minimum TS quantity/concentration (TS Figure 3.4.1) of available SPB solution, the minimum suppression pool pH at 30 days post-LOCA remains above 7.0. This pH satisfies the conditions for inhibiting the release of the chemical form of elemental iodine from the containment. As discussed in the accompanying Safety Assessment of the AST analyses, the SLC system will be credited for suppression pool pH control, thus limiting the radiological dose following LOCAs involving fuel damage.

NUREG-0737 Evaluation

The revised analyses include consideration of the impacts of AST methodology for several NUREG-0737 items. These are summarized below.

- Post-accident vital area access and sampling: Post-accident personnel missions resulting in mission doses (including post-accident sampling) have been previously identified. The implementation of the AST methodology with the alternative leakage treatment (ALT) does not result in any new operator missions. ALT is discussed in Appendix A of the Safety

[10] See Attachment 5 for other cases examined.

Assessment (Attachment 5). Plant calculations used in support of plant post-accident vital area access (prepared in accordance with NUREG-0737, Items II.B.2 and II.B.3) were evaluated for impact by AST. The evaluation considered the comparative radiation levels from AST and the existing TID-14844 methodology source terms (such as airborne activity in the reactor building and turbine building, and also as activity in the suppression pool water).

- Post-accident radiation monitor: Post-accident containment high range radiation monitoring calculations were revised for impact by AST (NUREG-0737, Item II.F.1).
- Control room radiation protection: The control room radiological dose impact of AST has been specifically calculated for each of the four DBAs analyzed for AST implementation (NUREG-0737 item III.D.3.4).
- Radioactive sources outside the primary containment: The DBA LOCA control room dose analysis, as well as that for offsite doses, includes the effects of coolant leakage outside the primary containment and (for the control room and TSC dose analyses only) the shine contribution from the reactor building and other source term bearing systems and/or components (NUREG-0737, Item III.D.1.1).

Primary Containment Leakage Rate Testing Program

In accordance with the requirements of TS 6.7.C, VY has implemented leakage rate testing of the primary containment as required by 10CFR50.54(o) and 10CFR50, Appendix J – Option B, as modified by approved exemptions. Option B of Appendix J to 10CFR50 was initially implemented at VYNPS upon the issuance of license amendment no. 152[11].

Background – Primary Containment Leakage Rate Testing Program

With application of the AST, the proposed changes to primary containment leakage testing requirements will:

- Provide an exemption to 10CFR50 Appendix J, to exclude the measured leakage from the secondary containment bypass pathways and the main steam pathways from the combined local leakage rates (Type B and Type C tests) and the overall integrated leakage rate (Type A tests);
- Provide additional exceptions to Regulatory Guide 1.163, NEI 94-01, Rev. 0, and ANSI/ANS 56.8-1994;
- Revise the current provisions of TS 3/4.7.A.4 by (1) increasing the allowable leakage rate for individual MSIVs; (2) establishing the main steam pathway that incorporates five pathways (i.e., the four main steam lines and the main steam drain line); (3) increasing the combined leakage rate from the MSIVs to an aggregate leakage rate from the main steam pathways; (4) adding a specific provision that limits allowable secondary containment bypass leakage; and (5) require testing for the specific pathways in accordance with the PCLRTP. The PCLRTP establishes the test frequency and acceptance criteria for testing; and
- Clarify and correct certain omissions to TS 6.7.C.

[11] U.S. Nuclear Regulatory Commission letter to Vermont Yankee Nuclear Power Corporation, "Issuance of Amendment No. 152 to Facility Operating License No. DPR-28, VYNPS (TAC No. M99264)," February 26, 1998.

The proposed changes to the primary containment leakage rate testing program are based on the revised design basis radiological consequences analyses (i.e., the AST analyses) that consider the primary containment leakage rate at the maximum allowable leakage rate (La), the main steam pathways that exhaust to the main condenser, and the secondary containment bypass pathways (excluding the main steam) as individual factors of the radiological consequence analysis. The results of the revised design basis radiological consequences analyses are included herein.

The primary containment La at the calculated peak containment internal pressure (Pa) is unchanged at 0.8% of the containment air weight per day. Pa is 44 psig for VYNPS' design basis LOCA.

Licensing Topical Report NEDC-31858P

NEDC-31858P[12] established a means for demonstrating that alternative leakage treatment (ALT) pathways using the main steam system piping and the main condenser are capable of performing a post-accident dose mitigation function for MSIV leakage under assumed conditions. Appendix A to Attachment 5 of this submittal provides additional details of VYNPS' ALT strategy. The Boiling Water Reactor Owners Group (BWROG) evaluated increasing the TS limit for MSIV leakage in NEDC-31858P, which concludes among other things, that MSIV leakage could be increased to 200 scfh per main steam line without inhibiting the safety function of the MSIV. NEDC-31858 also found that a leakage rate of 200 scfh for an MSIV does not represent abnormal or excessive leakage for a valve of this size and type. The BWROG report further found that disassembly and refurbishment of MSIVs to meet low leakage limits frequently contribute to repeated failures from maintenance-induced defects such as seat cracks, excessive pilot valve seat machining, and mechanical defects induced by assembly and disassembly.

VY proposes to utilize the main steam drain line to preferentially direct MSIV leakage to the main condenser. This drain path takes advantage of the large volume of the steam lines and condenser to provide holdup and plate-out of fission products that may leak through the closed MSIVs. In this manner, the main steam lines, main steam drain piping, and the main condenser are used to mitigate the consequences of an accident to limit potential offsite doses to meet 10CFR50.67 and GDC-19 limits.

MSIV Leakage

Based on the revised AST analyses, the proposed changes increase the individual main steam isolation valve leakage limit and combined main steam pathways leakage limit. The proposed change revises current TS 3.7.A.4 and 4.7.A.4.

Individual MSIV Leakage Rate (Proposed TS 3.7.A.4.a)

The proposed change increases the individual MSIV leakage limit to 62 scfh. An increase in the allowable individual valve leakage limit is warranted since retaining the present leakage limit of 31 scfh would cause unnecessary maintenance on the valves simply to maintain the low leakage limit with no corresponding increase in safety at the expense of maintenance personnel radiation exposure and other burdens. Reducing the frequency of MSIV rebuilds during outages would also extend the service life of the MSIVs. New Specification 3.7.A.4.a will be added with the increased individual valve leakage limit.

[12] Boiling Water Reactor Owners Group, NEDC-31858P, "BWROG Report for Increasing MSIV Leakage Rate Limits and Elimination of Leakage Control Systems," Revision 2, September 1993.

Combined MSIV Leakage (Proposed TS 3.7.A.4.b)

Currently, the aggregate main steam leakage limit includes only the four main steam lines. The proposed specification will incorporate the main steam drain line as an additional pathway originating in the primary containment and exhausting into the main condenser. The aggregate main steam pathways' leakage calculation method is being changed from a sum of the maximum pathways to a new methodology. As presently calculated, the sum of the maximum pathways assumes multiple failures of the main steam isolation valves. The present assumption of multiple active failures is overly conservative.

The new methodology is based on the revised design basis radiological accident analyses that postulate: one main steam isolation valve fails to close, as a single active failure, and its companion valve leaks at the proposed main steam isolation valve limit of 62 scfh at Pa (44 psig). The remaining lines are assumed to leak at minimum pathway conditions. Therefore, the calculation method for ensuring the analyses assumptions remain valid is: the most limiting main steam line maximum pathway value will be added to the minimum pathway values of the four remaining main steam pathways to provide an aggregate that will be compared to the analyzed specification limit of 124 scfh at Pa (44 psig). New Specification 3.7.A.4.b will be added with the increased combined leakage limit.

Secondary Containment Bypass Leakage (Proposed TS 3.7.A.4.c)

In the VYNPS plant configuration, leakage from the primary containment is contained within secondary containment (the reactor building), collected and filtered through the standby gas treatment (SGT) system and discharged to the environment via the plant stack. The main steam lines and other flow penetrations provide certain pathways that could permit radioactivity in primary containment to bypass the secondary containment. As a result, such pathway leakage would not be collected and processed through the SGT system, but would enter the environment as an unfiltered ground release.

The pathways (not including main steam) that originate in the primary containment that have been conservatively postulated to bypass the secondary containment are addressed as a single input to the radiological consequences analyses. The proposed change will address these secondary containment bypass (SCB) pathways by summing the pathways to provide an aggregate that will be compared to the specification limit of 5 scfh at Pa. The calculation method will be similar to the method described for the main steam pathways. New Specification 3.7.A.4.c will be added.

Surveillance Requirements (Proposed TS 4.7.A.4)

The surveillance requirements and specific acceptance criteria of TS 4.7.A.4 will be relocated to the Primary Containment Leakage Rate Testing Program Plan (PCLRTP), where PCLRTP surveillance requirements are best controlled. The PCLRTP implements the requirements of TS 6.7.C as an administrative program of VY and is controlled under the provisions of 10CFR50.59.

Main steam and SCB pathway leakage will continue to be controlled in accordance with the PCLRTP. The PCLRTP Plan contains the surveillance requirements and the acceptance criteria for the assets tested in accordance with the Type A and Types B and C Testing Programs. Accordingly, assets that are performance-based are assigned an administrative limit (for a determination of testing interval) and a corrective action limit (for a determination of operability and repair/replacement); assets that are prescriptive-based (e.g., MSIV, etc.) are assigned only a corrective action limit. The corrective action limit is the acceptance criterion. A group of assets (e.g., a pathway, group of pathways, etc.) may be assigned an additional acceptance criterion such as the pathway summation method for the main steam

and SCB pathways to ensure the design bases radiological consequences analyses assumptions remain valid.

Presently, the main steam and SCB pathway Types B and C test results are added to the results from other Types B and C Testing: (1) to establish a combined total primary containment leakage rate which is evaluated against the limit for total primary containment leakage rate; and (2) for computation of the combined leakage rate for penetrations and valves subject to Types B and C tests to be less than La with margin.

Proposed Exemption to 10CFR50, Appendix J

10CFR50.54(o) requires that primary reactor containments be subject to the requirements of Appendix J to 10CFR50. Appendix J specifies the leakage rate test requirements, schedules, and acceptance criteria for tests of the leak-tight integrity of the primary reactor containment and systems and components which penetrate the containment. Option B, Section III.A requires that the overall integrated leakage rate must not exceed the allowable leakage (La) with margin, as specified in the TS. The overall integrated leakage rate, as specified in the 10CFR50, Appendix J definitions, includes the contribution from main steam and secondary containment bypass leakage. (Main steam leakage includes leakage through four main steam lines and the main steam drain line.) Option B, Section III.B of 10CFR50, Appendix J requires that the sum of the leakage rates of Type B and Type C local leakage rate tests be less than the performance criterion (La) with margin, as specified in the TS. Concurrent with the request for license amendment, VY hereby requests an exemption from 10CFR50.54(o) and the requirements of 10CFR50, Appendix J, Option B, Sections III.A and III.B to permit exclusion of the main steam and secondary containment bypass pathway leakage contributions from the overall integrated leakage rate Type A test measurement and from the sum of the leakage rates from Type B and Type C tests. This request for exemption is similar to an exemption granted from the requirements of Sections III.A and III.B of Option B for the Browns Ferry nuclear plant on March 14, 2000.[13]

10CFR50.12 – Specific Exemptions

10CFR50.12 states that the Commission will not consider granting an exemption unless special circumstances are present. VY believes this request meets the criterion of a special circumstance as defined in 50.12(a)(2)(ii), which states: "Application of the regulation in the particular circumstances would not serve the underlying purpose of the rule or is not necessary to achieve the underlying purpose of the rule." The underlying purpose of the rule is to establish test requirements that ensure that (1) leakage through these containments or systems and components penetrating these containments do not exceed allowable leakage rates, and (2) integrity of the containment structure is maintained during its service life.

The above cited requirements of Appendix J require that the main steam and secondary containment bypass leakage measurements are included in the leakage measurements of other containment penetrations when containment leakage tests are performed. With the VYNPS AST analyses, these requirements are inconsistent with the design of VYNPS and the analytical models used to calculate the radiological consequences of design basis accidents. At VYNPS, the leakage from primary containment penetrations, under accident conditions, is collected and treated by the secondary containment system, or would bypass the secondary containment. However, the main steam effluent has a different pathway to

[13] U.S. Nuclear Regulatory Commission letter to Tennessee Valley Authority, "Browns Ferry Nuclear Plant, Units 2 and 3 – Issuance of Exemption From 10 CFR Part 50, Appendix J (TAC Nos. MA6815 and MA6816)," March 14, 2000.

the environment and is not directed into the secondary containment and filtered by the standby gas treatment system as is other containment leakage. Instead, the main steam leakage is collected and treated via an alternative leakage treatment (ALT) path having different mitigation characteristics (see Appendix A to the AST Safety Assessment in Attachment 5).

In performing accident analyses, it is appropriate to group various leakage effluents according to the treatment they receive before being released to the environment, i.e., secondary containment bypass leakage is grouped, leakage into secondary containment is grouped, and ALT leakage (i.e., from main steam pathways) is grouped, with specific limits for each group defined in the TS. The proposed exemption would more appropriately permit ALT and secondary containment pathway leakage to be independently grouped with their unique leakage limits. In this manner, the VYNPS containment leakage testing program will be made more consistent with the limiting assumptions used in the associated accident consequence analyses. Corresponding changes to the TS, which implement the requested exemption, are also proposed.

VY has analyzed the main steam leakage pathway for 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 the dose consequence analyses in Attachment 5. The calculated radiological consequences of the combined leakages are within the criteria of 10CFR50.67, and are therefore acceptable.

It is anticipated that the revised limits on main steam isolation valve leakage will potentially result in a reduction of unnecessary maintenance on these valves simply to maintain the low leakage rate and support reducing maintaining worker exposure to as low as reasonably achievable.

Based on the foregoing, the removal of the SCB and main steam pathways from Specifications 6.7.C.3 and 6.7.C.4 is warranted since a separate specification has been provided for these pathways. The revised design basis radiological consequences analyses address these pathways as individual factors, exclusive of the Primary Containment leakage.

Description and Safety Assessment for Specific Changes to TS and TS Bases

A description of each proposed TS change and the associated basis/safety assessment are included in Table 6. The preceding discussion, the Safety Assessment in Attachment 5, and the calculations in Attachment 6 support these changes. The change numbers in Table 6 correspond to the change numbers (in boxes) on the marked-up TS pages provided in Attachment 3. Attachment 4 provides the proposed TS and TS Bases incorporated into re-typed TS and TS Bases pages.

Table 6

<table>
<tr><th colspan="3">Description and Safety Assessment for Specific Changes to TS and TS Bases</th></tr>
<tr><td rowspan="2">Change #1</td><td>Current Technical Specification:

Current TS 1.0.CC provides a definition of “Dose Equivalent I-131,” and includes reference to dose conversion factors listed in Regulatory Guide 1.109, Revision 1, for calculating thyroid dose equivalent I-131.</td><td>Proposed Change:

The current TS 1.0.CC definition of “Dose Equivalent I-131” is revised to remove the word “thyroid” and 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 Ingestion,” 1988; and FGR 12, “External Exposure to Radionuclides In Air, Water, and Soil,” 1993.

The proposed revised TS 1.0.CC definition is:

The dose equivalent I-131 shall be that concentration of I-131 (microcurie/gram) which alone would produce the same dose as the quantity and isotopic mixture of I-131, I-132, I-133, I-134 and I-135 actually present. The dose conversion factors used for this calculation shall be those listed in Federal Guidance Report (FGR) 11, “Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion,” 1988; FGR 12, “External Exposure to Radionuclides In Air, Water, and Soil,” 1993; or NRC Regulatory Guide 1.109, Revision 1, October 1977.</td></tr>
<tr><td colspan="2">Basis / Safety Assessment:

The existing definition is revised to conform to the implementation of AST. The new citations are as cited in RG 1.183, which was found to be acceptable by the NRC for AST</td></tr>
</table>

Table 6
(continued)

Change #1	**Basis / Safety Assessment:** (continued)
	applications. These revised values were used in the re-analyses of the design basis accidents using AST methodology. The revised accident analyses use inhalation committed effective dose equivalent dose conversion factors from FGR 11 and external committed effective dose equivalent dose conversion factors from FGR 12. Dose conversion factors from Regulatory Guide 1.109, Revision 1, are used in other calculations of dose equivalency. With the implementation of AST, the accident dose guidelines of 10CFR100 are superseded by the dose criteria of 10CFR50.67. The whole body and thyroid doses of 10CFR100 are replaced by the total effective dose equivalent (TEDE) criteria of 10CFR50.67. A conforming change to the definition is to delete the word "thyroid" from the definition. The analyses performed in support of this amendment request determined radiological consequences in terms of the TEDE dose quantity and were shown to be in compliance with the dose criteria of 10CFR50.67. These changes to the definition are acceptable because they reflect adoption of the dose conversion factors and dose consequences of the revised radiological analyses.

Table 6
(continued)

Change #2	Current Technical Specification:	Proposed Change:
	Current TS 4.7.A.2 refers to the "Primary Containment Leak [sic] Rate Testing Program." Current TS 3.7.A.4 specifies individual valve and combined MSIV leakage rate limits at the calculated peak accident containment pressure of 44 psig. Current TS 4.7.A.4 specifies individual valve and combined MSIV leakage rate limits when MSIVs are tested at a test pressure ≥ 24 psig.	TS 4.7.A.2 revises the title of the primary containment leakage rate testing program to the "Primary Containment Leakage Rate Testing Program." TS 3.7.A.4 is revised and subdivided into TS 3.7.A.4.a, b, and c. TS 3.7.A.4 is 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);* b. *The combined leakage rate from the main steam pathways shall not exceed 124 scfh at 44 psig (Pa); and* c. *The combined leakage rate from the secondary containment bypass pathways shall not exceed 5 scfh at 44 psig (Pa).* TS 4.7.A.4 is revised and subdivided into TS 4.7.A.4.a, b, and c. The specific test criteria of TS 4.7.A.4 are being relocated to the PCLRTP. TS 4.7.A.4 is changed to: 4. *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.*

Table 6
(continued)

Change #2	**Basis / Safety Assessment:** (See preceding discussion on Primary Containment Leakage Rate Testing Program for additional bases.) The leakage rate assumptions in the revised safety analyses form the basis for the revised TS limits. Editorial changes are made to TS 4.7.A.2 to clarify the meaning and understanding of the TS to avoid confusion and potential error. Changing the word "leak" to "leakage" in the TS 4.7.A.2 is acceptable because it does not change the intent or technical meaning of the specification and is made for clarity and consistency purposes. In proposed TS 3.7.A.4.a, the individual MSIV allowable leakage rate is proposed to be one-half the total main steam leakage rate allowable value. The proposed increase in individual MSIV leakage rate (from 31 scfh at Pa in TS 3.7.C.4 to 62 scfh at Pa) will not inhibit the safety function of the MSIVs. The disadvantages of increased maintenance and higher worker radiation exposure associated with maintaining relatively low individual MSIV leakage rates are not justified by any additional conservatism the individual limits might provide. The change in MSIV leakage rates is based on the utilization of the methodology described in NEDC-31858P. The NRC staff has previously determined that the methodology of NEDC-31858P is acceptable for such applications. In proposed TS 3.7.A.4.b, the combined leakage rate from the main steam pathways (four main steam lines and a main steam drain line) is increased from 62 scfh at Pa for all four steam lines to 124 scfh at Pa for the main steam pathways. Consistent with VY's modeling of main steam leakages, the main steam drain line contribution is conservatively included. The proposed main steam leakage rate limit of 124 scfh at Pa is acceptable since this value was assumed in the revised accident analyses and is coupled with other leakage rates to achieve acceptable, calculated radiological doses for the assumed DBAs. Because calculated doses are below the regulatory criteria of 10CFR50.67, additional leakage rate margin exists. Proposed TS 3.7.A.4.c ensures that the leakage rate of secondary containment bypass leakage paths (excluding main steam) is less than the specified leakage rate. This provides assurance that the assumptions in the radiological analyses are met. The specific test criteria in TS 4.7.A.4 are being relocated to the PCLRTP. The PCLRTP is a formal program, required by TS 6.7.C, and under of the controls of 10CFR50.59. It is appropriate that the details of primary containment testing, including MSIVs, should reside in the PCLRTP. The relocated surveillance requirement details ensure that the performance of the primary containment, including MSIVs, meets the revised leakage rate testing acceptance criteria in TS 6.7.C. As such, these surveillance requirement details are implementing details of the PCLRTP and more appropriately reside in the PCLRTP. Changes to the PCLRTP may be made in accordance with the provisions of 10CFR50.59, provided the requirements of TS 6.7.C and applicable regulations are met. These controls are adequate to ensure maintenance

Table 6
(continued)

<table>
<tr><td rowspan="1">Change
#2</td><td colspan="2">Basis / Safety Assessment: (continued)

of proper surveillance requirements in the PCLRTP, and therefore relocation of implementing details from TS 4.7.A.4 to the PCLRTP is acceptable.

The overall change to the primary containment leakage rate test program is acceptable because the individual MSIV and combined maximum leakage rates from all sources are consistent with the accident analyses that use primary containment leakage rates as an input assumption.</td></tr>
<tr><td rowspan="2">Change
#3</td><td>Current Technical Specification:

Current TS 4.7.C.1.a and 4.7.C.1.c both specify a standby gas treatment (SGT) system “filter train flow rate of not more than 1500 cfm” when conducting certain surveillance testing.</td><td>Proposed Change:

The specified SGT system flow rate of “1500 cfm” is changed to “1550 cfm” in TS 4.7.C.1.a and 4.7.C.1.c.</td></tr>
<tr><td colspan="2">Basis / Safety Assessment:

The increase in SGT system flow rate test criteria is necessary due to an increase in the assumed minimum, analytical SGT system flow rate to 1450 cfm in the secondary containment drawdown analysis. Considering inaccuracies in the test methods associated with the subject surveillance requirements, together with the increased flow assumed in analyses, the proposed change is warranted and is within the performance capability of the SGT system.

A review of the historical basis for the current (1500 cfm) TS value indicates that it was selected in or about calendar year 1972 as a “nominal” value, consistent with actual plant testing that established the TS test conditions.

This change is acceptable because it does not change the manner in which the facility is operated or maintained, is consistent with the AST analyses and conforms with the performance capability and requirements for ensuring SGT system and secondary containment operability.</td></tr>
</table>

Table 6
(continued)

Change #4	**Current TS Bases:**	**Proposed Change:**
	Current TS 6.7.C specifies that the Primary Containment Leak Rate Test Program (PCLRTP) shall be in accordance with Regulatory Guide (RG) 1.163 as modified by an exception to NEI 94-01. (RG 1.163 endorses, with certain exceptions, NEI 94-01, and ANSI/ANS-56.8-1994.) The structure of TS 6.7.C could be improved for usability. The title heading of TS 6.7.C is "Primary Containment Leak [sic] Rate Testing Program." A typographical error exists in TS 6.7.C.1 in that the primary containment leakage rate acceptance criterion is given as "< 1.0 La."	TS 6.7.C is revised to incorporate the exemption to Sections III.A and III.B of 10CFR50, Appendix J, Option B that is requested herein. TS 6.7.C is revised such that the leakage contributions from the secondary containment bypass pathways and 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. Editorial changes are made to restructure the format and clarify TS 6.7.C. Revised terminology is implemented for consistency and correctness (e.g., "leak" is changed to "leakage"). The first paragraph of TS 6.7.C is revised to state: *A program shall be established to implement the leakage rate testing of the primary containment as required by 10 CFR 50.54(o) and 10 CFR 50, Appendix J, Option B as modified by approved exemptions. This program shall be in accordance with the guidelines contained in Regulatory Guide 1.163, entitled "Performance Based Containment Leak-Test Program," dated September 1995, as modified by the following:* • *The first Type A test after the April 1995 Type A test shall be performed no later than November 2005. (This is an exception to Section 9.2.3 of NEI 94-01, Rev. 0, "Industry Guideline for Implementing Performance-Based Option of 10CFR50, Appendix J.")* • *The leakage contributions from the secondary containment bypass pathways and the main steam pathways are excluded from the sum of the leakage rates from Type B and Type C tests specified in (1) Section*

Table 6
(continued)

<table>
<tr><td rowspan="2">Change
#4</td><td></td><td>Proposed Change: (continued)

III.B of 10CFR50, Appendix J – Option B; (2) Section 6.4.4 of ANSI/ANS 56.8-1994; and (3) Section 10.2 of NEI 94-01, Rev. 0.

• The leakage contributions from the secondary containment bypass pathways and the main steam pathways are excluded from the overall integrated leakage rate from Type A tests specified in (1) Section III.A of 10CFR50, Appendix J – Option B; (2) Section 3.2 of ANSI/ANS 56.8-1994; and (3) Sections 8.0 and 9.0 of NEI 94-01, Rev. 0.

TS 6.7.C.1 is changed to correct a typographical error, such that the acceptance criterion is now stated as "≤ 1.0 La."</td></tr>
<tr><td colspan="2">Basis / Safety Assessment:

As discussed previously in this request for a change in the licensing basis for VYNPS, VY is requesting an exemption from the requirements of Sections III.A and III.B of 10CFR50, Appendix J, Option B to exclude the leakage contributions of secondary containment bypass pathways and main steam pathways from the overall integrated leakage rates from Type A tests and from the sum of the leakage rates from Type B and Type C tests. Because TS 6.7.C invokes compliance to RG 1.163, which endorses, with certain exceptions, NEI 94-01, and ANSI/ANS-56.8-1994, certain exceptions are also need to these associated guidelines. For Type A tests, in addition to the exemption to Section III.A of Appendix J, Option B, exceptions are needed to Section 3.2 of ANSI/ANS 56.8-1994 and Sections 8.0 and 9.0 of NEI 94-01, Revision 0. Compliance with ANSI/ANS 56.8-1994 is required as a condition of compliance with RG 1.163. For Type B and Type C tests, in addition to the exemption to Section III.B of Appendix J, Option B, exceptions are needed to Section 6.4.4 of ANSI/ANS 56.8-1994 and Section 10.2 of NEI 94-01, Revision 0. These exceptions are acceptable because they conform to the exemption requested.

Upon granting the requested exemption, the exceptions for including secondary containment bypass pathways and main steam pathways leakage rates in the introductory paragraph of TS</td></tr>
</table>

Table 6
(continued)

<table>
<tr><td>Change
#4</td><td colspan="2">Basis / Safety Assessment: (continued)

6.7.C are acceptable because the leakage rates for the subject pathways will be contained in separate specifications (i.e., proposed TS 3.7.A.4. a, 3.7.A.4.b and 3.7.A.c.) and the leakage rate acceptance criteria from all measured pathways are consistent with the leakage rates assumed in the AST analyses. The sum of the limiting leakage rates from all leakage pathways does not result in radiological doses exceeding the limits specified in 10CFR50.67.

Editorial changes are made to TS 6.7.C to clarify the meaning and understanding of the TS to avoid confusion and potential error. These changes are acceptable because they do not change the technical meaning or intent of the specification. These changes include slight reformatting of the specification and changing the word "leak" to "leakage" in TS 6.7.C. These changes are acceptable because they do not change the intent or technical meaning of the specifications and are made for clarity and consistency purposes. The change in terminology from "leak rate" to "leakage rate" is also consistent with the definition of this term in ANSI/ANS 56.8-1994.

TS 6.7.C.1 is changed to correct a typographical error, such that the acceptance criterion is now properly stated as "≤ 1.0 La." When Option B of Appendix J was adopted through license amendment 152, the acceptance criterion expression for primary containment leakage rate properly stated "≤ 1.0 La." VY's request for license amendment that became Amendment no. 215[14] inadvertently modified the "≤" symbol. This change, therefore, corrects that administrative error and is acceptable because the inadvertent change in Amendment 215 was unintended and not within the scope of changes incorporated by that license amendment.</td></tr>
<tr><td>Change
#5</td><td>Current TS Bases:

Current TS 6.7.C.3 and 6.7.C.4 provide conditions and acceptance criteria for primary containment leakage rate testing. The leakage contributions from secondary containment bypass pathways and main steam pathways are currently included in determining the sum of the leakage rates from Type B and Type C tests.</td><td>Proposed Change:

The acceptance criteria specified in TS 6.7.C.3 and 6.7.C.4 for allowable Type B and Type C leakage rate testing are revised to exclude the contributions from both secondary containment bypass pathways and main steam pathways.

TS 6.7.C.3 is also clarified by stating that requirement is applicable where <u>primary</u> containment integrity is required (addition of the word, "primary").

TS 6.7.C.3 and 6.7.C.4 are revised to state:

3. The combined local leakage rate test acceptance criterion for Type B and Type C tests (excluding the leakage</td></tr>
</table>

[14] U.S. Nuclear Regulatory Commission letter to Vermont Yankee, "Vermont Yankee Nuclear Power Station – Issuance of Amendment Re: One-Time Extension of Appendix J Type A Integrated Leakage Rate Test Interval (TAC No. MB6507)," (License Amendment No. 215), June 2, 2003.

Table 6
(continued)

<table>
<tr><td rowspan="2">Change
#5</td><td></td><td>Proposed Change: (continued)

contributions from both the secondary containment bypass pathways and the main steam pathways) is ≤ 0.6 La, calculated on a maximum pathway basis, prior to entering a mode of operation where primary containment integrity is required.

4. The combined local leakage rate test acceptance criterion for Type B and Type C tests (excluding the leakage contributions from both the secondary containment bypass pathways and the main steam pathways) is ≤ 0.6 La, calculated on a minimum pathway basis, at all times when primary containment integrity is required.</td></tr>
<tr><td colspan="2">Basis / Safety Assessment:

Upon granting the requested exemption, the exclusion of secondary containment bypass pathway and main steam pathway leakage rates in 6.7.C.3 and 6.7.C.4 are acceptable because the leakage rates for the subject pathways will be contained in separate specifications (i.e., proposed TS 3.7.A.4. a, 3.7.A.4.b, and 3.7.A.4.c). The leakage rate acceptance criteria from all measured pathways are consistent with the leakage rates assumed in the AST analyses. The sum of the limiting leakage rates from all leakage pathways does not result in radiological doses exceeding the limits specified in 10CFR50.67. This is an administrative change to incorporate the provisions of the requested exemption.</td></tr>
</table>

Table 6
(continued)

<table>
<tr><td rowspan="2">Change
#6</td><td>Current TS Bases:

The TS Bases provide explanation and rationale for associated TS requirements, and in some cases, how they are to be implemented.</td><td>Proposed Change:

Associated changes to the TS Bases are being made to conform to the changed TS (e.g., changing 10CFR100 to 10CFR50.67) and to add clarity to existing requirements.</td></tr>
<tr><td colspan="2">Basis / Safety Assessment:

The accident dose guidelines of 10CFR100 are superseded by the dose criteria of 10CFR50.67. The whole body and thyroid doses of 10CFR100 are replaced by the total effective dose equivalent (TEDE) criteria of 10CFR50.67, and references to 10CFR100 are replaced with 10CFR50.67. This is a conforming change.

Other changes were made to the TS Bases for clarity and to conform to the changes being made to the associated Specifications. The revisions to the TS Bases incorporate supporting information for the proposed TS changes. Bases do not establish actual requirements, and as such do not change technical requirements of the TS. The Bases changes are therefore acceptable, since they administratively document the reasons and provide additional understanding for the associated TS requirements.</td></tr>
</table>

Conclusion

In conclusion, based on the considerations discussed above and detailed in Attachment 5, (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner; (2) such activities will be conducted in compliance with the Commission's regulations; and (3) the issuance of the requested license amendment will not be inimical to the common defense and security or to the health and safety of the public.

Attachment 2

Vermont Yankee Nuclear Power Station

Proposed Technical Specification Change No. 262

Alternative Source Term

Determination of No Significant Hazards Consideration

Description of amendment request:

Vermont Yankee (VY) is submitting a request for amendment to the VY Technical Specifications (TS). The proposed amendment is a full implementation of an alternative source term (AST) adopting AST methodology by revising the current accident source term and replacing it with an accident source term as prescribed in 10CFR50.67.

AST analyses were performed using the guidance provided by Regulatory Guide 1.183, "Alternative Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors," dated July 2000, and Standard Review Plan Section 15.0.1, "Radiological Consequences Analyses Using Alternative Source Terms." The four limiting design basis accidents (DBAs) considered were the Control Rod Drop Accident, the Refueling Accident, the Loss of Coolant Accident, and the Main Steam Line Break Accident. As a result of the application of a revised accident source term, changes to the TS which revise the definition of dose equivalent I-131, revise the requirements of the primary containment leakage rate test program, and revise the standby gas treatment system required flow rate are proposed. Conforming changes would also be made to TS Bases. In addition, VY is concurrently requesting an exemption to certain requirements of Sections III.A and III.B of 10CFR50, Appendix J, Option B regarding the primary containment leakage rate testing program.

Basis for No Significant Hazards Determination:

Pursuant to 10CFR50.92, Vermont Yankee (VY) has reviewed the proposed change and concludes that the change does not involve a significant hazards consideration since the proposed change satisfies the criteria in 10CFR50.92(c). These criteria require that operation of the facility in accordance with the proposed amendment will not: (1) involve a significant increase in the probability or consequences of an accident previously evaluated, (2) create the possibility of a new or different kind of accident from any accident previously evaluated, or (3) involve a significant reduction in a margin of safety. The discussion below addresses each of these criteria and demonstrates that the proposed amendment does not constitute a significant hazard.

The proposed change does not involve a significant hazards consideration because:

A. **The proposed amendment does not involve a significant increase in the probability or consequences of an accident previously evaluated.**

Adoption of the AST and those plant systems affected by implementing AST do not initiate DBAs. The AST does not affect the design or manner in which the facility is operated; rather, once the occurrence of an accident has been postulated, the new accident source term is an input to analyses that evaluate the radiological consequences. The implementation of the AST and changed TS have been incorporated in the analyses for the limiting DBAs at VYNPS.

The structures, systems and components affected by the proposed change act as mitigators to the consequences of accidents. Based on the revised analyses, the proposed changes to the TS (including revised leakage limits) impose certain performance criteria which do not increase accident consequences. The proposed changes do not involve a revision to the parameters or conditions that could contribute to the initiation of a design basis accident discussed in Chapter 14 of the Updated Final Safety Analysis Report.

Plant specific AST radiological analyses have been performed and, based on the results of these analyses, it has been demonstrated that the dose consequences of the limiting events considered in the analyses are within the regulatory guidance provided by the Nuclear Regulatory Commission for use with the AST. This guidance is presented in 10CFR50.67, Regulatory Guide 1.183, and Standard Review Plan (SRP) Section 15.0.1. Therefore, the proposed amendment does not result in a significant increase in the consequences or increase the probability of any previously evaluated accident.

B. **The proposed amendment does not create the possibility of a new or different kind of accident from any accident previously evaluated.**

Implementation of AST and the proposed changes does not alter or involve any design basis accident initiators. These changes do not affect the design function or mode of operations of systems, structures, or components in the facility prior to a postulated accident. Since systems, structures, and components are operated essentially no differently after the AST implementation, no new failure modes are created by this proposed change. Therefore, the proposed license amendments will not create the possibility of a new or different kind of accident from any accident previously evaluated.

C. **The proposed amendment does not involve a significant reduction in a margin of safety.**

The changes proposed are associated with a revision to the licensing basis for VYNPS. Approval of the licensing basis change from the original source term to the alternative source term is requested by this application for a license amendment. The results of the accident analyses revised in support of the proposed change are subject to the acceptance criteria in 10CFR50.67. The analyzed events have been carefully selected, and the analyses supporting these changes have been performed using approved methodologies to ensure that analyzed events are bounding and safety margin has been retained. The dose consequences of these limiting events are within the acceptance criteria presented in 10CFR50.67, Regulatory Guide 1.183, and SRP 15.0.1. Thus, by meeting the applicable regulatory limits for AST, there is no significant reduction in a margin of safety.

Therefore, because the proposed changes continue to result in dose consequences within the applicable regulatory limits, the changes are considered to not result in a significant reduction in a margin of safety.

Conclusion

On the basis of the above, VY has determined that operation of the facility in accordance with the proposed change does not involve a significant hazards consideration as defined in 10CFR50.92(c), in that it: (1) does not involve a significant increase in the probability or consequences of an accident previously evaluated; (2) does not create the possibility of a new or different kind of accident from any accident previously evaluated; and (3) does not involve a significant reduction in a margin of safety.

Attachment 3

Vermont Yankee Nuclear Power Station

Proposed Technical Specification Change No. 262

Alternative Source Term

Mark-up of the Current Technical Specifications

1.0 DEFINITIONS

Z. Surveillance Interval - The surveillance interval is the calendar time between surveillance tests, checks, calibrations, and examinations to be performed upon an instrument or component when it is required to be operable. These tests unless otherwise stated in these specifications may be waived when the instrument, component, or system is not required to be operable, but these tests shall be performed on the instrument, component, or system prior to being required to be operable.

AA. Deleted

BB. Source Check - The qualitative assessment of channel response when the channel sensor is exposed to a radioactive source.

CC. Dose Equivalent I-131 - The dose equivalent I-131 shall be that concentration of I-131 (microcurie/gram) which alone would produce the same ~~thyroid~~ dose as the quantity and isotopic mixture of I-131, I-132, I-133, I-134 and I-135 actually present. The ~~thyroid~~ dose conversion factors used for this calculation shall be those listed in NRC Regulatory Guide 1.109, Revision 1, October 1977.

Federal Guidance Report (FGR) 11, "Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion," 1988; FGR 12, "External Exposure to Radionuclides In Air, Water, and Soil," 1993; or

1

DD. Deleted

EE. Deleted

FF. Deleted

GG. Deleted

HH. Deleted

II. Deleted

JJ. Deleted

KK. Deleted

LL. Deleted

MM. Deleted

NN. Core Operating Limits Report - The Core Operating Limits Report is the unit-specific document that provides core operating limits for the current operating reload cycle. These cycle-specific core operating limits shall be determined for each reload cycle in accordance with Specification 6.6.C. Plant operation within these operating limits is addressed in individual specifications.

BASES:

3.2 PROTECTIVE INSTRUMENTATION

In addition to reactor protection instrumentation which initiates a reactor scram, station protective instrumentation has been provided which initiates action to mitigate the consequences of accidents which are beyond the reactor operator's ability to control, or terminate a single operator error before it results in serious consequences. This set of Specifications provides the limiting conditions of operation for the primary system isolation function and initiation of the core standby cooling and standby gas treatment systems. The objectives of the Specifications are (i) to assure the effectiveness of any component of such systems even during periods when portions of such systems are out of service for maintenance, testing, or calibration, and (ii) to prescribe the trip settings required to assure adequate performance. This set of Specifications also provides the limiting conditions of operation for the control rod block system and surveillance instrumentation.

Isolation valves are installed in those lines that penetrate the primary containment and must be isolated during a loss-of-coolant accident so that the radiation dose limits are not exceeded during an accident condition. Actuation of these valves is initiated by protective instrumentation shown in Table 3.2.2 which senses the conditions for which isolation is required. Such instrumentation must be available whenever primary containment integrity is required. The objective is to isolate the primary containment so that the limits of ~~10 CFR 100~~ are not exceeded during an accident. The objective of the low turbine condenser vacuum trip is to minimize the radioactive effluent releases to as low as practical in case of a main condenser failure. Subsequent releases would continue until operator action was taken to isolate the main condenser unless the main steam line isolation valves were closed automatically on low condenser vacuum. The manual bypass is required to permit initial startup of the reactor during low power

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10CFR50.67

[illegible] ation which initiates primary system isolation is connected in a dual channel arrangement. Thus, the discussion given in the bases for Specification 3.1 is applicable here.

The low reactor water level instrumentation is set to trip when reactor water level is 127" above the top of the enriched fuel. This trip initiates closure of Group 2 and 3 primary containment isolation valves. For a trip setting of 127" above the top of the enriched fuel, the valves will be closed before perforation of the clad occurs even for the maximum break and, therefore, the setting is adequate.

The top of the enriched fuel (351.5" from vessel bottom) is designated as a common reference level for all reactor water level instrumentation. The intent is to minimize the potential for operator confusion which may result from different scale references.

The low-low reactor water level instrumentation is set to trip when reactor water level is 82.5" H_2O indicated on the reactor water level instrumentation above the top of the enriched fuel. This trip initiates closure of the Group 1 primary containment isolation valves and also activates the ECCS and RCIC System and starts the standby diesel generator system. This trip setting level was chosen to be low enough to prevent spurious operation, but high enough to initiate ECCS operation and primary system isolation so that no melting of the fuel cladding will occur, and so that post-accident cooling can be accomplished and the limits of ~~10CFR100~~ will not be violated.

BASES: 3.2 (Cont'd)

For the complete circumferential break of 28-inch recirculation line and with the trip setting given above, ECCS initiation and primary system isolation are initiated in time to meet the above criteria. The instrumentation also covers the full range of spectrum breaks and meets the above criteria.

The high drywell pressure instrumentation is a backup to the water level instrumentation, and in addition to initiating ECCS, it causes isolation of Group 2, 3, and 4 isolation valves. For the complete circumferential break discussed above, this instrumentation will initiate ECCS operation at about the same time as the low-low water level instrumentation, thus, the results given above are applicable here also. Certain isolation valves including the TIP blocking valves, CAD inlet and outlet, drywell vent, purge and sump valves are isolated on high drywell pressure. However, since high drywell pressure could occur as the result of non-safety-related causes, such as not venting the drywell during startup, complete system isolation is not desirable for these conditions and only certain valves are required to close. The water level instrumentation initiates protection for the full spectrum of loss of coolant accidents and causes a trip of certain primary system isolation valves.

Venturis are provided in the main steam lines as a means of measuring steam flow and also limiting the loss of mass inventory from the vessel during a steam line break accident. In addition to monitoring steam flow, instrumentation is provided which causes a trip of Group 1 isolation valves. The primary function of the instrumentation is to detect a break in the main steam line, thus only Group 1 valves are closed. For the worst case accident, main steam line break outside the drywell, this trip setting of 140 percent of rated steam flow in conjunction with the flow limiters and main steam line valve closure limit the mass inventory loss such that fuel is not uncovered, cladding temperatures remain less than 1295°F and release of radioactivity to the environs is well below 10CFR~~100~~.

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Temperature monitoring instrumentation is provided in the main steam line tunnel to detect leaks in this area. Trips are provided on this instrumentation and when exceeded cause closure of Group 1 isolation valves. Its setting of ambient plus 95°F is low enough to detect leaks of the order of 5 to 10 gpm; thus, it is capable of covering the entire spectrum of breaks. For large breaks, it is a backup to high steam flow instrumentation discussed above, and for small breaks, with the resultant small release of radioactivity, gives isolation before the limits of 10CFR~~100~~ are exceeded.

Isolation of the condenser mechanical vacuum pump (MVP) is assumed in the safety analysis for the control rod drop accident (CRDA). The MVP isolation instrumentation initiates closure of the MVP suction isolation valve following events in which main steam line radiation monitors exceed a predetermined value. A High Main Steam Line Radiation Monitor trip setting for MVP isolation of ≤ 3 times background at rated thermal power (RTP) is as low as practicable without consideration of spurious trips from nitrogen-16 spikes, instrument instabilities and other operational occurrences. Isolating the condenser MVP limits the release of fission products in the event of a CRDA.

Pressure instrumentation is provided which trips when main steam line pressure drops below 800 psig. A trip of this instrumentation results in closure of Group 1 isolation valves. In the refuel, shutdown, and startup modes, this trip function is provided when main steam line flow exceeds 40% of rated capacity. This function is provided primarily to provide protection against a pressure regulator malfunction which would cause the

BASES: 3.6 and 4.6 (Cont'd)

The actual shift in RT_{NDT} of ... material in the core region will be e... by removing and eva... el material irr... wall of th... at the i... identic... with ... Batte... provi... data belt... appro...

Limits on the maximum allowable level of radioactivity in the reactor coolant are established to ensure that in the event of a release of any radioactive material to the environment during a design basis accident, radiation doses are maintained within the limits of 10CFR50.67.

The Limiting Conditions for Operation contain iodine specific activity limits. The iodine isotopic activities per gram of reactor coolant are expressed in terms of DOSE EQUIVALENT I-131. The allowable levels are intended to limit the 2-hour radiation dose to an individual at the site boundary to within 10CFR50.67 dose guidelines.

In orde... region, 50°F of eac...

The number of ... the frequencies for re... to assure compliance with ... CFR Part 50.

B. Coolant Chemistry

A steady-state radioiodine concentration limit of 1.1 µCi of I-131 dose equivalent per gram of water in the Reactor Coolant System can be reached if the gross radioactivity in the gaseous effluents is near the limit, as set forth in the Offsite Dose Calculation Manual, or if there is a failure or prolonged shutdown of the cleanup demineralizer. ~~In the event of a steam line rupture outside the drywell, the NRC staff calculations show the resultant radiological dose at the site boundary to be less than 30 Rem to the thyroid. This dose was calculated on the basis of the radioiodine concentration limit of 1.1 µCi of I-131 dose equivalent per gram of water, atmospheric diffusion from an equivalent elevated release of 10 meters at the nearest site boundary (190 m) for a X/Q = 3.9 x 10^{-3} sec/m^3 (Pasquill D and 0.33 m/sec equivalent), and a steam line isolation valve closure time of five seconds with a steam/water mass release of 30,000 pounds.~~

The iodine spike limit of four (4) microcuries of I-131 dose equivalent per gram of water provides an iodine peak or spike limit for the reactor coolant concentration to assure that the radiological consequences of a postulated LOCA are within 10CFR ~~Part 100~~ 50.67 dose guidelines.

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The reactor coolant sample will be used to assure that the li Specification 3.6.B.1 is not exceeded. The radioiodine concentration would not be expected to change rapidly during steady-state operation over a period of 96 hours. In addition, the trend of the radioactive gaseous effluents, which is continuously monitored, is a good indicator of the trend of the radioiodine concentration in the reactor coolant. When a significant increase in radioactive gaseous effluents is indicated, as specified, an additional reactor coolant sample shall be taken and analyzed for radioactive iodine.

3.7 LIMITING CONDITIONS FOR OPERATION

at normal cooldown rates if the torus water temperature exceeds 120°F.

e. Minimum Water Volume - 68,000 cubic feet

f. Maximum Water Volume - 70,000 cubic feet

2. Primary containment integrity shall be maintained at all times when the reactor is critical or when the reactor water temperature is above 212°F and fuel is in the reactor vessel except while performing low power physics tests at atmospheric pressure at power levels not to exceed 5 Mw(t).

3. If a portion of a system that is considered to be an extension of primary containment is to be opened, isolate the affected penetration flow path by use of at least one closed and deactivated automatic valve, closed manual valve or blind flange.

4. ~~Whenever primary containment integrity is required, the leakage from any one main steam line isolation valve shall not exceed 31 scf/hr at 44 psig (P_a), and the combined leakage from all four main steam lines shall not exceed 62 scf/hr at 44 psig (P_a).~~

< INSERT 3.7.A.4 >

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4.7 SURVEILLANCE REQUIREMENTS

2. The primary containment integrity shall be demonstrated as required by the Primary Containment ~~Leak~~ Rate Testing Program (PCLRTP).

Leakage

2

3. (Blank)

4. ~~Verify leakage rate through each main steam line isolation valve is ≤23 scf/hr and that the combined maximum pathway leakage rate for all four main steam lines is ≤46 scf/hr when tested at ≥24 psig (P_t).~~

< INSERT 4.7.A.4 >

Insert to TS 3.7.A.4

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);

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

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

Insert to TS 4.7.A.4

4. 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.

3.7 LIMITING CONDITIONS FOR OPERATION

i. Suspend movement of irradiated fuel assemblies and the fuel cask in secondary containment; and

ii. Suspend core alterations; and

iii. Initiate action to suspend operations with the potential for draining the reactor vessel.

C. Secondary Containment System

1. Secondary Containment Integrity shall be maintained during the following modes or conditions:

a. Whenever the reactor is in the Run Mode, Startup Mode, or Hot Shutdown condition*; or

4.7 SURVEILLANCE REQUIREMENTS

C. Secondary Containment System

1. Surveillance of secondary containment shall be performed as follows:

a. A preoperational secondary containment capability test shall be conducted after isolating the Reactor Building and placing either Standby Gas Treatment System filter train in operation. Such tests shall demonstrate the capability to maintain a 0.15 inch of water vacuum under calm wind ($2 < u < 5$ mph) condition with a filter train flow rate of not more than ~~1500~~ cfm.

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* NOTE: The reactor mode switch may be changed to either the Run or Startup/Hot Standby position, and operation not considered to be in the Run Mode or Startup Mode, to allow testing of instrumentation associated with the reactor mode switch interlock functions, provided:

1. Reactor coolant temperature is $\leq$ 212°F;
2. All control rods remain fully inserted in core cells containing one or more fuel assemblies; and
3. No core alterations are in progress.

3.7 LIMITING CONDITIONS FOR OPERATION

b. During movement of irradiated fuel assemblies or the fuel cask in secondary containment; or

c. During alteration of the Reactor Core; or

d. During operations with the potential for draining the reactor vessel.

4.7 SURVEILLANCE REQUIREMENTS

b. Additional tests shall be performed during the first operating cycle under an adequate number of different environmental wind conditions to enable valid extrapolation of the test results.

c. Secondary containment capability to maintain a 0.15 inch of water vacuum under calm wind ($2<\bar{u}<5$ mph) conditions with a filter train flow rate of not more than ~~1,500~~ cfm, shall be demonstrated at least quarterly and at each refueling outage prior to refueling.

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BASES:

3.7 STATION CONTAINMENT SYSTEMS

A. Primary Containment

The integrity of the primary containment and operation of the core standby cooling systems in combination limit the off-site doses to values less than to those suggested in ~~10 CFR 100~~ 10CFR50.67 in the event of a break in the primary system piping. Thus, containment integrity is specified whenever the potential for violation of the primary reactor system integrity exists. Concern about such a violation exists whenever the reactor is critical, above atmospheric pressure and temperature above 212°F. An exception is made to this requirement during initial core loading and while a low power test program is being conducted and ready access to the reactor vessel is required. The reactor may be taken critical during the period; however, restrictive operating procedures will be in effect again to minimize the probability of an accident occurring. Procedures and the Rod Worth Minimizer would limit control worth to less than 1.30% delta k.

The pressure suppression pool water provides the heat sink for the reactor primary system energy release following postulated rupture of the system. The pressure suppression chamber water volume must absorb the associated decay and structural sensible heat released during primary system blowdown from normal operating pressure.

Since all the gases in the drywell are purged into the pressure suppression chamber air space during a loss-of-coolant accident, the pressure resulting from isothermal compression plus the vapor pressure of the liquid must not exceed 62 psig, the allowable internal design pressure for the pressure suppression chamber. The design volume of the suppression chamber (water and air) was obtained by considering that the total volume of reactor coolant to be condensed is discharged to the suppression chamber and that the drywell volume is purged to the suppression chamber (Reference Section 5.2 FSAR).

Using the minimum or maximum water volumes given in the specification, the calculated peak accident containment pressure is approximately 44 psig, which is below the ASME design pressure of 56 psig.[3] The minimum volume of 68,000 ft^3 results in a submergency of approximately four feet. The majority of the Bodega tests[2] were run with a submerged length of four feet and with complete condensation. Thus, with respect to downcomer submergence, this specification is adequate.

The maximum temperature at the end of blowdown tested during the Humbolt Bay[1] and Bodega Bay tests was 170°F and this is conservatively taken to be the limit for complete condensation of the reactor coolant, although condensation would occur for temperature above 170°F.

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(1) Robbins, C. H., "Tests on a Full Scale 1/48 Segment of the Humbolt Bay Pressure Suppression Containment", GEAP-3596, November 17, 1960.

(2) Bodega Bay Preliminary Hazards Summary Report, Appendix 1, Docket 50-205, December 28, 1962.

(3) Internal design pressure is 62 psig.

BASES: 4.7 (Cont'd)

The primary containment preoperational test pressures are based upon the calculated primary containment pressure response in the event of a loss-of-coolant accident. The calculated peak accident containment pressure would be about 44 psig, which would reduce to 27 psig within about 20 seconds following the pipe break. The suppression chamber pressure rises to about 25 psig within 30 seconds, equalizes with drywell pressure, and then decays with drywell pressure.[1]

The ASME design pressure of the drywell and absorption chamber is 56 psig.[2] The design leak rate is 0.5%/day at this pressure. As pointed out above, the pressure response of the drywell and suppression chamber following an accident would be the same after about 10 seconds. Based on the primary containment pressure response and the fact that the drywell and suppression chamber function as a unit, the primary containment will be tested as a unit rather than the individual components separately.

~~The design basis loss-of-coolant accident was evaluated at the primary containment maximum allowable accident leak rate of 1.5%/day at 44 psig. The analysis showed that with this leak rate and a standby gas treatment system filter efficiency of 90% for halogens, 95% for particulates, and assuming the fission product release fractions stated in TID-14844, the maximum total whole body passing cloud dose is about 1.65 rem and the maximum total thyroid dose is about 280 rem at the site boundary over an exposure duration of two hours. The resultant dose that would occur for the duration of the accident at the low population distance of 5 miles is lower than those stated due to the variability of meteorological conditions that would be expected to occur over a 30-day period. Thus, these doses are the maximum that would be expected in the unlikely event of a design basis loss-of-coolant accident. These doses are also based on the assumption of no holdup in the secondary containment, resulting in a direct release of fission products from the primary containment through the filters and stack to the environs. Therefore, the specified primary containment leak rate and filter efficiency are conservative and provide margin between expected off-site doses and 10 CFR 100 guidelines. An additional factor of two for conservatism is added to the above doses by limiting the test leak rate (L a) to a value of 0.80 wt.%/day.~~

< INSERT Bases 4.7 >

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(1) Section 14.6 of the FSAR.

(2) 62 psig is the maximum internal design pressure for this ASME design (56 psig) pressure.

INSERT to Bases 4.7

Maintaining the primary containment OPERABLE requires compliance with the visual examinations and leakage rate test requirements of the Primary Containment Leakage Rate Testing Program (PCLRTP) required by Specification 6.7.C. The PCLRTP specifies the leakage rate test requirements, schedules, and acceptance criteria for tests of the leak tight integrity of the primary reactor containment and systems and components which penetrate the containment.

The PCLRTP implements the leakage rate testing of the primary containment as required by 10CFR50.54(o) and 10CFR50, Appendix J, Option B as modified by approved exemptions. The leakage limits prescribed by the PCLRTP are consistent with the design of VYNPS and the analytical models used to calculate the radiological consequences of design basis accidents described in the Updated Final Safety Analysis Report.

Consistent with the limiting assumptions used in the associated accident consequence analyses, the PCLRTP differentiates three leakage pathways to the environment: (1) primary containment leakage to secondary containment, which is filtered through the standby gas treatment system before being released via the plant stack; (2) main steam pathways; and (3) secondary containment bypass pathways. Leakage effluent from the main steam and secondary containment bypass pathways have different pathways (ground level) to the environment than the leakage into secondary containment. These pathways are defined in the PCLRTP.

BASES: 4.7 (Cont'd)

The test frequencies are adequate to detect equipment deterioration prior to significant defects, but the tests are not frequent enough to load the filters, thus reducing their reserve capacity too quickly. That the testing frequency is adequate to detect deterioration was demonstrated by the tests which showed no loss of filter efficiency after 2 years of operation in the rugged shipboard environment on the NS Savannah (ORNL 3726). Pressure drop tests across filter sections are performed to detect gross plugging of the filter media. Considering the relatively short time that the fans may be run for test purposes, plugging is unlikely, and the test interval is reasonable. Such heater tests will be conducted once during each operating cycle. Considering the simplicity of the heating circuit, the test frequency is sufficient. Air distribution tests will be conducted once during each operating cycle.

The in-place testing of charcoal filters is performed using a halogenated hydrocarbon, which is injected into the system upstream of the charcoal filters. Measurements of the challenge gas concentration upstream and downstream of the charcoal filters is made. The ratio of the inlet and outlet concentrations gives an overall indication of the leak tightness of the system. Although this is basically a leak test, since the filters have charcoal of known efficiency and holding capacity for elemental iodine and/or methyl iodine, the test also gives an indication of the relative efficiency of the installed system.

High-efficiency particulate air filters are installed before and after the charcoal filter to minimize potential release of particulates to the environment and to prevent clogging of the iodine filters. An efficiency of 99% is adequate to retain particulates that may be released to the Reactor Building following an accident. This will be demonstrated by testing with DOP as the test medium.

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The efficiencies of the particulate and charcoal filters are sufficient to prevent exceeding 10CFR100 limits for the accidents analyzed. The analysis of post-accident hydrogen purge assumed a charcoal filter efficiency of 95%. Hence requiring in-place test efficiencies of 99% for these filters and a laboratory methyl iodide test of 97.5% for the charcoal provides adequate margin.

The test interval for filter efficiency was selected to minimize plugging of the filters. In addition, testing for methyl iodide removal efficiency will be demonstrated. This will be done either by removal of a charcoal sample cartridge which contains charcoal equivalent to the bed thickness or removing one adsorber tray from the system and using the charcoal therein, after mixing, to obtain at least two samples equivalent to the bed thickness. Any HEPA filters found defective should be replaced with filters qualified according to Regulatory Position C.3.d of Regulatory Guide 1.52. If laboratory test results are unacceptable, all charcoal adsorbent in the system should be replaced with charcoal adsorbent qualified according to Regulatory Guide 1.52.

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BASES: 4.7 (Cont'd)

D. Primary Containment Isolation Valves

Those large pipes comprising a portion of the reactor coolant system whose failure could result in uncovering the reactor core are supplied with automatic isolation valves (except those lines needed for emergency core cooling system operation or containment cooling). The closure times specified herein and per Specification 4.6.E are adequate to prevent loss of more cooling from the circumferential rupture of any of these lines outside the containment than from a steam line rupture. Therefore, the isolation valve closure times are sufficient to prevent uncovering the core.

Purge and vent valve testing performed by Allis-Chalmers has demonstrated that all butterfly purge and vent valves installed at Vermont Yankee can close from full open conditions at design basis containment pressure. However, as an additional conservative measure, limit stops have been added to valves 16-19-7/7A. limiting the opening of these valves to 50° open while operating, [illegible]ed by NRC in their letter of May 22, 1984. (NVY 84-108) 50.67 6

In order to assure that the doses that may re[illegible] a steam line break do not exceed the 10CFR100 guidelines, it is necessary that no fuel rod perforation resulting from the accident occur prior to closure of the main steam line isolation valves. Analyses indicate the fuel rod cladding perforations would be avoided for the main steam valve closure times, including instrument delay, as long as 10.5 seconds. The test closure time limit of five seconds for these main steam isolation valves provides sufficient margin to assure that cladding perforations are avoided and 10CFR100 limits are not exceeded. Redundant valves in each line ensure that isolation will be effected applying the single failure criteria.

The main steam isolation valves are primary containment isolation valves and are tested in accordance with the requirements of the Inservice Testing program.

The containment is penetrated by a large number of small diameter instrument lines. The flow check valves in these lines are tested for operability in accordance with Specification 4.6.E.

E. Reactor Building Automatic Ventilation System Isolation Valves (RBAVSIVs)

In the event that there are one or more RBAVSIVs inoperable when secondary containment integrity is required, the affected penetrations that have been isolated must be verified to be isolated on a periodic basis. This is necessary to ensure that those penetrations required to be isolated following an accident, but no longer capable of being automatically isolated, will be in the isolated position should an event occur. The verification frequency of once per 31 days is appropriate because the valves are operated under administrative controls and the probability of their misalignment is low. Verification of isolation does not require any testing or device manipulation. Rather, it involves verification that the affected penetration remains isolated.

The RBAVSIVs covered by this surveillance requirement, along with their test requirements, are included in the Inservice Testing Program.

BASES:

3.8 RADIOACTIVE EFFLUENTS

A. Deleted

B. Deleted

C. Deleted

D. Liquid Holdup Tanks

The tanks listed in this Specification include all outdoor tanks that contain radioactivity that are not surrounded by liners, dikes, or walls capable of holding the tank contents, or that do not have tank overflows and surrounding area drains connected to the liquid radwaste treatment system.

Restricting the quantity of radioactive material contained in the specified tanks provides assurance that in the event of an uncontrolled release of the tanks' contents, the resulting concentrations would be less than the limits of 10CFR Part 20.1001-20.2402, Appendix B, Table 2, Column 2, at the nearest potable water supply and in the nearest surface water supply in an Unrestricted Area.

E. Deleted

F. Deleted

G. Deleted

H. Deleted

I. Deleted

J. Explosive Gas Mixture

The hydrogen monitors are used to detect possible hydrogen buildups which could result in a possible hydrogen explosion. Automatic isolation of the off-gas flow would prevent the hydrogen explosion and possible damage to the augmented off-gas system. Maintaining the concentration of hydrogen below its flammability limit provides assurance that the releases of radioactive materials will be controlled.

total effective dose equivalent

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10CFR50.67

K. Steam Jet Air Ejector (SJAE)

Restricting the gross radioactivity release rate of gases from the main condenser SJAE provides reasonable assurance that the ~~total body exposure~~ to an individual at the exclusion area boundary will ~~not~~ exceed ~~a small fraction of~~ the limits of ~~10CFR Part 100~~ in the event this effluent is inadvertently discharged directly to the environment without treatment. This specification implements the requirements of General Design Criteria 60 and 64 of Appendix A to 10CFR Part 50.

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4

LEAKAGE

< INSERT #1 to 6.7.C >

Report for the p... ...rt in which any change ...DCM wa... shall be identified ...fected pages, clearly ...that was changed, and ...the date (e.g., month/year) the change ...mented.

C. PRIMARY CONTAINMENT ~~LEAK~~ RATE TESTING PROGRAM

A program shall be established to implement the ~~Leak~~ rate testing of the primary containment as required by 10 CFR 50.54(o) and 10 CFR 50, Appendix J, Option B as modified by approved exemptions. This program shall be in accordance with the guidelines contained in Regulatory Guide 1.163, entitled "Performance Based Containment Leak-Test Program," dated September 1995, ~~as modified by the following exception to NEI 94-01, Rev. 0, "Industry Guideline for Implementing Performance-Based Option of 10CFR50, Appendix J":~~

~~Section 9.2.3: The first Type A test after the April 1995 Type A test shall be performed no later than November 2005.~~

The peak calculated containment internal pressure for the design basis loss of coolant accident, Pa, is 44 psig.

The maximum allowable primary containment ~~leak~~ rate, La, at Pa, shall be 0.8% of primary containment air weight per day.

4

≤

~~Leak rate acceptance criteria are:~~

1. Primary containment ~~leak~~ rate acceptance criterion ~~<~~ 1.0 La.

2. The as-left primary containment ~~integrated leak~~ rate test (Type A test) acceptance criterion is < 0.75 La.

~~3. The combined local leak rate test (Type B and C tests) acceptance criterion is < 0.60 La, calculated on a maximum pathway basis, prior to entering a mode of operation where containment integrity is required.~~

~~4. The combined local leak rate test (Type B and C tests) acceptance criterion is < 0.60 La, calculated on a minimum pathway basis, at all times when primary containment integrity is required.~~

5. Airlock overall ~~leak rate~~ acceptance criterion is < 0.10 La when tested at > Pa.

The provision of the Definition (1.0.Y) for Surveillance Frequency does not apply to the test frequencies specified in the Primary Containment ~~Leak Rate~~ Testing Program.

D. Radioactive Effluent Controls Program

This program conforming to 10 CFR 50.36a provides for the control of radioactive effluents and for maintaining the doses to members of the public from radioactive effluents as low as reasonably

< INSERT #2 to 6.7.C >

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INSERT #1 to TS 6.7.C

as modified by the following:

- The first Type A test after the April 1995 Type A test shall be performed no later than November 2005. (This is an exception to Section 9.2.3 of NEI 94-01, Rev. 0, "Industry Guideline for Implementing Performance-Based Option of 10CFR50, Appendix J.")

- The leakage contributions from the secondary containment bypass pathways and the main steam pathways are excluded from the sum of the leakage rates from Type B and Type C tests specified in (1) Section III.B of 10CFR50, Appendix J – Option B; (2) Section 6.4.4 of ANSI/ANS 56.8-1994; and (3) Section 10.2 of NEI 94-01, Rev. 0.

- The leakage contributions from the secondary containment bypass pathways and the main steam pathways are excluded from the overall integrated leakage rate from Type A tests specified in (1) Section III.A of 10CFR50, Appendix J – Option B; (2) Section 3.2 of ANSI/ANS 56.8-1994; and (3) Sections 8.0 and 9.0 of NEI 94-01, Rev. 0.

INSERT #2 to TS 6.7.C

3. The combined local leakage rate test acceptance criterion for Type B and Type C tests (excluding the leakage contributions from both the secondary containment bypass pathways and the main steam pathways) is $\leq$ 0.6 La, calculated on a maximum pathway basis, prior to entering a mode of operation where primary containment integrity is required.

4. The combined local leakage rate test acceptance criterion for Type B and Type C tests (excluding the leakage contributions from both the secondary containment bypass pathways and the main steam pathways) is $\leq$ 0.6 La, calculated on a minimum pathway basis, at all times when primary containment integrity is required.

Attachment 4

Vermont Yankee Nuclear Power Station

Proposed Technical Specification Change No. 262

Alternative Source Term

Retyped Technical Specification Pages

Listing of Affected Technical Specifications Pages

Replace the Vermont Yankee Nuclear Power Station Technical Specifications pages listed below with the revised pages included herein. The revised pages contain vertical lines in the margin indicating the areas of change.

Current Page	New Page
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 *
267 *	267 *

* - for text overrun

1.0 DEFINITIONS

Z. Surveillance Interval - The surveillance interval is the calendar time between surveillance tests, checks, calibrations, and examinations to be performed upon an instrument or component when it is required to be operable. These tests unless otherwise stated in these specifications may be waived when the instrument, component, or system is not required to be operable, but these tests shall be performed on the instrument, component, or system prior to being required to be operable.

AA. Deleted

BB. Source Check - The qualitative assessment of channel response when the channel sensor is exposed to a radioactive source.

CC. Dose Equivalent I-131 - The dose equivalent I-131 shall be that concentration of I-131 (microcurie/gram) which alone would produce the same dose as the quantity and isotopic mixture of I-131, I-132, I-133, I-134 and I-135 actually present. The dose conversion factors used for this calculation shall be those listed in Federal Guidance Report (FGR) 11, "Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion," 1988; FGR 12, "External Exposure to Radionuclides In Air, Water, and Soil," 1993; or NRC Regulatory Guide 1.109, Revision 1, October 1977.

DD. Deleted

EE. Deleted

FF. Deleted

GG. Deleted

HH. Deleted

II. Deleted

JJ. Deleted

KK. Deleted

LL. Deleted

MM. Deleted

NN. Core Operating Limits Report - The Core Operating Limits Report is the unit-specific document that provides core operating limits for the current operating reload cycle. These cycle-specific core operating limits shall be determined for each reload cycle in accordance with Specification 6.6.C. Plant operation within these operating limits is addressed in individual specifications.

BASES:

3.2 PROTECTIVE INSTRUMENTATION

In addition to reactor protection instrumentation which initiates a reactor scram, station protective instrumentation has been provided which initiates action to mitigate the consequences of accidents which are beyond the reactor operator's ability to control, or terminate a single operator error before it results in serious consequences. This set of Specifications provides the limiting conditions of operation for the primary system isolation function and initiation of the core standby cooling and standby gas treatment systems. The objectives of the Specifications are (i) to assure the effectiveness of any component of such systems even during periods when portions of such systems are out of service for maintenance, testing, or calibration, and (ii) to prescribe the trip settings required to assure adequate performance. This set of Specifications also provides the limiting conditions of operation for the control rod block system and surveillance instrumentation.

Isolation valves are installed in those lines that penetrate the primary containment and must be isolated during a loss-of-coolant accident so that the radiation dose limits are not exceeded during an accident condition. Actuation of these valves is initiated by protective instrumentation shown in Table 3.2.2 which senses the conditions for which isolation is required. Such instrumentation must be available whenever primary containment integrity is required. The objective is to isolate the primary containment so that the limits of 10CFR50.67 are not exceeded during an accident. The objective of the low turbine condenser vacuum trip is to minimize the radioactive effluent releases to as low as practical in case of a main condenser failure. Subsequent releases would continue until operator action was taken to isolate the main condenser unless the main steam line isolation valves were closed automatically on low condenser vacuum. The manual bypass is required to permit initial startup of the reactor during low power operation.

The instrumentation which initiates primary system isolation is connected in a dual channel arrangement. Thus, the discussion given in the bases for Specification 3.1 is applicable here.

The low reactor water level instrumentation is set to trip when reactor water level is 127" above the top of the enriched fuel. This trip initiates closure of Group 2 and 3 primary containment isolation valves. For a trip setting of 127" above the top of the enriched fuel, the valves will be closed before perforation of the clad occurs even for the maximum break and, therefore, the setting is adequate.

The top of the enriched fuel (351.5" from vessel bottom) is designated as a common reference level for all reactor water level instrumentation. The intent is to minimize the potential for operator confusion which may result from different scale references.

The low-low reactor water level instrumentation is set to trip when reactor water level is 82.5" H_2O indicated on the reactor water level instrumentation above the top of the enriched fuel. This trip initiates closure of the Group 1 primary containment isolation valves and also activates the ECCS and RCIC System and starts the standby diesel generator system. This trip setting level was chosen to be low enough to prevent spurious operation, but high enough to initiate ECCS operation and primary system isolation so that no melting of the fuel cladding will occur, and so that post-accident cooling can be accomplished and the limits of 10CFR50.67 will not be violated.

BASES: 3.2 (Cont'd)

For the complete circumferential break of 28-inch recirculation line and with the trip setting given above, ECCS initiation and primary system isolation are initiated in time to meet the above criteria. The instrumentation also covers the full range of spectrum breaks and meets the above criteria.

The high drywell pressure instrumentation is a backup to the water level instrumentation, and in addition to initiating ECCS, it causes isolation of Group 2, 3, and 4 isolation valves. For the complete circumferential break discussed above, this instrumentation will initiate ECCS operation at about the same time as the low-low water level instrumentation, thus, the results given above are applicable here also. Certain isolation valves including the TIP blocking valves, CAD inlet and outlet, drywell vent, purge and sump valves are isolated on high drywell pressure. However, since high drywell pressure could occur as the result of non-safety-related causes, such as not venting the drywell during startup, complete system isolation is not desirable for these conditions and only certain valves are required to close. The water level instrumentation initiates protection for the full spectrum of loss of coolant accidents and causes a trip of certain primary system isolation valves.

Venturis are provided in the main steam lines as a means of measuring steam flow and also limiting the loss of mass inventory from the vessel during a steam line break accident. In addition to monitoring steam flow, instrumentation is provided which causes a trip of Group 1 isolation valves. The primary function of the instrumentation is to detect a break in the main steam line, thus only Group 1 valves are closed. For the worst case accident, main steam line break outside the drywell, this trip setting of 140 percent of rated steam flow in conjunction with the flow limiters and main steam line valve closure limit the mass inventory loss such that fuel is not uncovered, cladding temperatures remain less than 1295°F and release of radioactivity to the environs is well below 10CFR50.67.

Temperature monitoring instrumentation is provided in the main steam line tunnel to detect leaks in this area. Trips are provided on this instrumentation and when exceeded cause closure of Group 1 isolation valves. Its setting of ambient plus 95°F is low enough to detect leaks of the order of 5 to 10 gpm; thus, it is capable of covering the entire spectrum of breaks. For large breaks, it is a backup to high steam flow instrumentation discussed above, and for small breaks, with the resultant small release of radioactivity, gives isolation before the limits of 10CFR50.67 are exceeded.

Isolation of the condenser mechanical vacuum pump (MVP) is assumed in the safety analysis for the control rod drop accident (CRDA). The MVP isolation instrumentation initiates closure of the MVP suction isolation valve following events in which main steam line radiation monitors exceed a predetermined value. A High Main Steam Line Radiation Monitor trip setting for MVP isolation of ≤ 3 times background at rated thermal power (RTP) is as low as practicable without consideration of spurious trips from nitrogen-16 spikes, instrument instabilities and other operational occurrences. Isolating the condenser MVP limits the release of fission products in the event of a CRDA.

Pressure instrumentation is provided which trips when main steam line pressure drops below 800 psig. A trip of this instrumentation results in closure of Group 1 isolation valves. In the refuel, shutdown, and startup modes, this trip function is provided when main steam line flow exceeds 40% of rated capacity. This function is provided primarily to provide protection against a pressure regulator malfunction which would cause the

BASES: 3.6 and 4.6 (Cont'd)

The actual shift in RT_{NDT} of the critical plate and weld material in the core region will be established periodically during operation by removing and evaluating, in accordance with ASTM E185, reactor vessel material irradiation surveillance specimens installed near the inside wall of the reactor vessel in the core area. Since the neutron spectra at the irradiation samples and vessel inside radius are essentially identical, the measured transition shift for a sample can be applied with confidence to the adjacent section of the reactor vessel. Battelle Columbus Laboratory Report BCL-585-84-3, dated May 15, 1984, provides this information for the ten-year surveillance capsule. When data from the next surveillance capsule is available, the predicted beltline ART_{NDT} will be re-assessed and the P/T curves revised as appropriate.

In order to prevent undue stress on the vessel nozzles and bottom head region, the recirculation loop temperatures will be maintained within 50°F of each other prior to startup of an idle loop.

The number of reactor vessel irradiation surveillance specimens and the frequencies for removing and testing these specimens are provided to assure compliance with the requirements of Appendix H to 10CFR Part 50.

B. Coolant Chemistry

A steady-state radioiodine concentration limit of 1.1 µCi of I-131 dose equivalent per gram of water in the Reactor Coolant System can be reached if the gross radioactivity in the gaseous effluents is near the limit, as set forth in the Offsite Dose Calculation Manual, or if there is a failure or prolonged shutdown of the cleanup demineralizer. Limits on the maximum allowable level of radioactivity in the reactor coolant are established to ensure that in the event of a release of any radioactive material to the environment during a design basis accident, radiation doses are maintained within the limits of 10CFR50.67.

The Limiting Conditions for Operation contain iodine specific activity limits. The iodine isotopic activities per gram of reactor coolant are expressed in terms of DOSE EQUIVALENT I-131. The allowable levels are intended to limit the 2-hour radiation dose to an individual at the site boundary to within 10CFR50.67 dose guidelines.

The iodine spike limit of four (4) microcuries of I-131 dose equivalent per gram of water provides an iodine peak or spike limit for the reactor coolant concentration to assure that the radiological consequences of a postulated LOCA are within 10CFR50.67 dose guidelines.

The reactor coolant sample will be used to assure that the limit of Specification 3.6.B.1 is not exceeded. The radioiodine concentration would not be expected to change rapidly during steady-state operation over a period of 96 hours. In addition, the trend of the radioactive gaseous effluents, which is continuously monitored, is a good indicator of the trend of the radioiodine concentration in the reactor coolant. When a significant increase in radioactive gaseous effluents is indicated, as specified, an additional reactor coolant sample shall be taken and analyzed for radioactive iodine.

3.7 LIMITING CONDITIONS FOR OPERATION

at normal cooldown rates if the torus water temperature exceeds 120°F.

e. Minimum Water Volume - 68,000 cubic feet

f. Maximum Water Volume - 70,000 cubic feet

2. Primary containment integrity shall be maintained at all times when the reactor is critical or when the reactor water temperature is above 212°F and fuel is in the reactor vessel except while performing low power physics tests at atmospheric pressure at power levels not to exceed 5 Mw(t).

3. If a portion of a system that is considered to be an extension of primary containment is to be opened, isolate the affected penetration flow path by use of at least one closed and deactivated automatic valve, closed manual valve or blind flange.

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);

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

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

4.7 SURVEILLANCE REQUIREMENTS

2. The primary containment integrity shall be demonstrated as required by the Primary Containment Leakage Rate Testing Program (PCLRTP).

3. (Blank)

4. 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.

3.7 LIMITING CONDITIONS FOR OPERATION

i. Suspend movement of irradiated fuel assemblies and the fuel cask in secondary containment; and

ii. Suspend core alterations; and

iii. Initiate action to suspend operations with the potential for draining the reactor vessel.

C. Secondary Containment System

1. Secondary Containment Integrity shall be maintained during the following modes or conditions:

a. Whenever the reactor is in the Run Mode, Startup Mode, or Hot Shutdown condition*; or

4.7 SURVEILLANCE REQUIREMENTS

C. Secondary Containment System

1. Surveillance of secondary containment shall be performed as follows:

a. A preoperational secondary containment capability test shall be conducted after isolating the Reactor Building and placing either Standby Gas Treatment System filter train in operation. Such tests shall demonstrate the capability to maintain a 0.15 inch of water vacuum under calm wind ($2 < u < 5$ mph) condition with a filter train flow rate of not more than 1550 cfm.

* NOTE: The reactor mode switch may be changed to either the Run or Startup/Hot Standby position, and operation not considered to be in the Run Mode or Startup Mode, to allow testing of instrumentation associated with the reactor mode switch interlock functions, provided:

1. Reactor coolant temperature is $\leq$ 212°F;
2. All control rods remain fully inserted in core cells containing one or more fuel assemblies; and
3. No core alterations are in progress.

3.7 LIMITING CONDITIONS FOR OPERATION

b. During movement of irradiated fuel assemblies or the fuel cask in secondary containment; or

c. During alteration of the Reactor Core; or

d. During operations with the potential for draining the reactor vessel.

4.7 SURVEILLANCE REQUIREMENTS

b. Additional tests shall be performed during the first operating cycle under an adequate number of different environmental wind conditions to enable valid extrapolation of the test results.

c. Secondary containment capability to maintain a 0.15 inch of water vacuum under calm wind ($2<\bar{u}<5$ mph) conditions with a filter train flow rate of not more than 1550 cfm, shall be demonstrated at least quarterly and at each refueling outage prior to refueling.

BASES:

3.7 STATION CONTAINMENT SYSTEMS

A. Primary Containment

The integrity of the primary containment and operation of the core standby cooling systems in combination limit the off-site doses to values less than to those suggested in 10CFR50.67 in the event of a break in the primary system piping. Thus, containment integrity is specified whenever the potential for violation of the primary reactor system integrity exists. Concern about such a violation exists whenever the reactor is critical, above atmospheric pressure and temperature above 212°F. An exception is made to this requirement during initial core loading and while a low power test program is being conducted and ready access to the reactor vessel is required. The reactor may be taken critical during the period; however, restrictive operating procedures will be in effect again to minimize the probability of an accident occurring. Procedures and the Rod Worth Minimizer would limit control worth to less than 1.30% delta k.

The pressure suppression pool water provides the heat sink for the reactor primary system energy release following postulated rupture of the system. The pressure suppression chamber water volume must absorb the associated decay and structural sensible heat released during primary system blowdown for normal operating pressure.

Since all the gases in the drywell are purged into the pressure suppression chamber air space during a loss-of-coolant accident, the pressure resulting from isothermal compression plus the vapor pressure of the liquid must not exceed 62 psig, the allowable internal design pressure for the pressure suppression chamber. The design volume of the suppression chamber (water and air) was obtained by considering that the total volume of reactor coolant to be condensed is discharged to the suppression chamber and that the drywell volume is purged to the suppression chamber (Reference Section 5.2 FSAR).

Using the minimum or maximum water volumes given in the specification, the calculated peak accident containment pressure is approximately 44 psig, which is below the ASME design pressure of 56 psig.[3] The minimum volume of 68,000 ft^3 results in a submergency of approximately four feet. The majority of the Bodega tests[2] were run with a submerged length of four feet and with complete condensation. Thus, with respect to downcomer submergence, this specification is adequate.

The maximum temperature at the end of blowdown tested during the Humbolt Bay[1] and Bodega Bay tests was 170°F and this is conservatively taken to be the limit for complete condensation of the reactor coolant, although condensation would occur for temperature above 170°F.

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(1) Robbins, C. H., "Tests on a Full Scale 1/48 Segment of the Humbolt Bay Pressure Suppression Containment", GEAP-3596, November 17, 1960.

(2) Bodega Bay Preliminary Hazards Summary Report, Appendix 1, Docket 50-205, December 28, 1962.

(3) Internal design pressure is 62 psig.

BASES: 4.7 (Cont'd)

The primary containment preoperational test pressures are based upon the calculated primary containment pressure response in the event of a loss-of-coolant accident. The calculated peak accident containment pressure would be about 44 psig, which would reduce to 27 psig within about 20 seconds following the pipe break. The suppression chamber pressure rises to about 25 psig within 30 seconds, equalizes with drywell pressure, and then decays with drywell pressure.(1)

The ASME design pressure of the drywell and absorption chamber is 56 psig.(2) The design leak rate is 0.5%/day at this pressure. As pointed out above, the pressure response of the drywell and suppression chamber following an accident would be the same after about 10 seconds. Based on the primary containment pressure response and the fact that the drywell and suppression chamber function as a unit, the primary containment will be tested as a unit rather than the individual components separately.

Maintaining the primary containment OPERABLE requires compliance with the visual examinations and leakage rate test requirements of the Primary Containment Leakage Rate Testing Program (PCLRTP) required by Specification 6.7.C. The PCLRTP specifies the leakage rate test requirements, schedules, and acceptance criteria for tests of the leak tight integrity of the primary reactor containment and systems and components which penetrate the containment.

The PCLRTP implements the leakage rate testing of the primary containment as required by 10CFR50.54(o) and 10CFR50, Appendix J, Option B as modified by approved exemptions. The leakage limits prescribed by the PCLRTP are consistent with the design of VYNPS and the analytical models used to calculate the radiological consequences of design basis accidents described in the Updated Final Safety Analysis Report.

Consistent with the limiting assumptions used in the associated accident consequence analyses, the PCLRTP differentiates three leakage pathways to the environment: (1) primary containment leakage to secondary containment, which is filtered through the standby gas treatment system before being released via the plant stack; (2) main steam pathways; and (3) secondary containment bypass pathways. Leakage effluent from the main steam and secondary containment bypass pathways have different pathways (ground level) to the environment than the leakage into secondary containment. These pathways are defined in the PCLRTP.

(1) Section 14.6 of the FSAR.

(2) 62 psig is the maximum internal design pressure for this ASME design (56 psig) pressure.

BASES: 4.7 (Cont'd)

The test frequencies are adequate to detect equipment deterioration prior to significant defects, but the tests are not frequent enough to load the filters, thus reducing their reserve capacity too quickly. That the testing frequency is adequate to detect deterioration was demonstrated by the tests which showed no loss of filter efficiency after 2 years of operation in the rugged shipboard environment on the NS Savannah (ORNL 3726). Pressure drop tests across filter sections are performed to detect gross plugging of the filter media. Considering the relatively short time that the fans may be run for test purposes, plugging is unlikely, and the test interval is reasonable. Such heater tests will be conducted once during each operating cycle. Considering the simplicity of the heating circuit, the test frequency is sufficient. Air distribution tests will be conducted once during each operating cycle.

The in-place testing of charcoal filters is performed using a halogenated hydrocarbon, which is injected into the system upstream of the charcoal filters. Measurements of the challenge gas concentration upstream and downstream of the charcoal filters is made. The ratio of the inlet and outlet concentrations gives an overall indication of the leak tightness of the system. Although this is basically a leak test, since the filters have charcoal of known efficiency and holding capacity for elemental iodine and/or methyl iodine, the test also gives an indication of the relative efficiency of the installed system.

High-efficiency particulate air filters are installed before and after the charcoal filter to minimize potential release of particulates to the environment and to prevent clogging of the iodine filters. An efficiency of 99% is adequate to retain particulates that may be released to the Reactor Building following an accident. This will be demonstrated by testing with DOP as testing medium.

The efficiencies of the particulate and charcoal filters are sufficient to prevent exceeding 10CFR50.67 limits for the accidents analyzed. The analysis of post-accident hydrogen purge assumed a charcoal filter efficiency of 95%. Hence requiring in-place test efficiencies of 99% for these filters and a laboratory methyl iodide test of 97.5% for the charcoal provides adequate margin.

The test interval for filter efficiency was selected to minimize plugging of the filters. In addition, testing for methyl iodide removal efficiency will be demonstrated. This will be done either by removal of a charcoal sample cartridge which contains charcoal equivalent to the bed thickness or removing one adsorber tray from the system and using the charcoal therein, after mixing, to obtain at least two samples equivalent to the bed thickness. Any HEPA filters found defective should be replaced with filters qualified according to Regulatory Position C.3.d of Regulatory Guide 1.52. If laboratory test results are unacceptable, all charcoal adsorbent in the system should be replaced with charcoal adsorbent qualified according to Regulatory Guide 1.52.

BASES: 4.7 (Cont'd)

D. Primary Containment Isolation Valves

Those large pipes comprising a portion of the reactor coolant system whose failure could result in uncovering the reactor core are supplied with automatic isolation valves (except those lines needed for emergency core cooling system operation or containment cooling). The closure times specified herein and per Specification 4.6.E are adequate to prevent loss of more cooling from the circumferential rupture of any of these lines outside the containment than from a steam line rupture. Therefore, the isolation valve closure times are sufficient to prevent uncovering the core.

Purge and vent valve testing performed by Allis-Chalmers has demonstrated that all butterfly purge and vent valves installed at Vermont Yankee can close from full open conditions at design basis containment pressure. However, as an additional conservative measure, limit stops have been added to valves 16-19-7/7A, limiting the opening of these valves to 50° open while operating, as requested by NRC in their letter of May 22, 1984. (NVY 84-108)

In order to assure that the doses that may result from a steam line break do not exceed the 10CFR50.67 guidelines, it is necessary that no fuel rod perforation resulting from the accident occur prior to closure of the main steam line isolation valves. Analyses indicate the fuel rod cladding perforations would be avoided for the main steam valve closure times, including instrument delay, as long as 10.5 seconds. The test closure time limit of five seconds for these main steam isolation valves provides sufficient margin to assure that cladding perforations are avoided and 10CFR50.67 limits are not exceeded. Redundant valves in each line ensure that isolation will be effected applying the single failure criteria.

The main steam isolation valves are primary containment isolation valves and are tested in accordance with the requirements of the Inservice Testing program.

The containment is penetrated by a large number of small diameter instrument lines. The flow check valves in these lines are tested for operability in accordance with Specification 4.6.E.

E. Reactor Building Automatic Ventilation System Isolation Valves (RBAVSIVs)

In the event that there are one or more RBAVSIVs inoperable when secondary containment integrity is required, the affected penetrations that have been isolated must be verified to be isolated on a periodic basis. This is necessary to ensure that those penetrations required to be isolated following an accident, but no longer capable of being automatically isolated, will be in the isolated position should an event occur. The verification frequency of once per 31 days is appropriate because the valves are operated under administrative controls and the probability of their misalignment is low. Verification of isolation does not require any testing or device manipulation. Rather, it involves verification that the affected penetration remains isolated.

The RBAVSIVs covered by this surveillance requirement, along with their test requirements, are included in the Inservice Testing Program.

BASES:

3.8 RADIOACTIVE EFFLUENTS

A. Deleted

B. Deleted

C. Deleted

D. Liquid Holdup Tanks

The tanks listed in this Specification include all outdoor tanks that contain radioactivity that are not surrounded by liners, dikes, or walls capable of holding the tank contents, or that do not have tank overflows and surrounding area drains connected to the liquid radwaste treatment system.

Restricting the quantity of radioactive material contained in the specified tanks provides assurance that in the event of an uncontrolled release of the tanks' contents, the resulting concentrations would be less than the limits of 10CFR Part 20.1001-20.2402, Appendix B, Table 2, Column 2, at the nearest potable water supply and in the nearest surface water supply in an Unrestricted Area.

E. Deleted

F. Deleted

G. Deleted

H. Deleted

I. Deleted

J. Explosive Gas Mixture

The hydrogen monitors are used to detect possible hydrogen buildups which could result in a possible hydrogen explosion. Automatic isolation of the off-gas flow would prevent the hydrogen explosion and possible damage to the augmented off-gas system. Maintaining the concentration of hydrogen below its flammability limit provides assurance that the releases of radioactive materials will be controlled.

K. Steam Jet Air Ejector (SJAE)

Restricting the gross radioactivity release rate of gases from the main condenser SJAE provides reasonable assurance that the total effective dose equivalent to an individual at the exclusion area boundary will not exceed the limits of 10CFR50.67 in the event this effluent is inadvertently discharged directly to the environment without treatment. This specification implements the requirements of General Design Criteria 60 and 64 of Appendix A to 10CFR Part 50.

Report for the period of the report in which any change to the ODCM was made. Each change shall be identified by markings in the margin of the affected pages, clearly indicating the area of the page that was changed, and shall indicate the date (e.g., month/year) the change was implemented.

C. PRIMARY CONTAINMENT LEAKAGE RATE TESTING PROGRAM

A program shall be established to implement the leakage rate testing of the primary containment as required by 10 CFR 50.54(o) and 10 CFR 50, Appendix J, Option B as modified by approved exemptions. This program shall be in accordance with the guidelines contained in Regulatory Guide 1.163, entitled "Performance Based Containment Leak-Test Program," dated September 1995, as modified by the following:

- The first Type A test after the April 1995 Type A test shall be performed no later than November 2005. (This is an exception to Section 9.2.3 of NEI 94-01, Rev. 0, "Industry Guideline for Implementing Performance-Based Option of 10CFR50, Appendix J.")
- The leakage contributions from the secondary containment bypass pathways and the main steam pathways are excluded from the sum of the leakage rates from Type B and C tests specified in (1) Section III.B of 10CFR50, Appendix J - Option B; (2) Section 6.4.4 of ANSI/ANS 56.8-1994; and (3) Section 10.2 of NEI 94-01, Rev. 0.
- The leakage contributions from the secondary containment bypass pathways and the main steam pathways are excluded from the overall integrated leakage rate from Type A tests specified in (1) Section III.A of 10CFR50, Appendix J - Option B; (2) Section 3.2 of ANSI/ANS 56.8-1994; and (3) Sections 8.0 and 9.0 of NEI 94-01, Rev. 0.

The peak calculated containment internal pressure for the design basis loss of coolant accident, Pa, is 44 psig.

The maximum allowable primary containment leakage rate, La, at Pa, shall be 0.8% of primary containment air weight per day.

Leakage rate acceptance criteria are:

1. Primary containment leakage rate acceptance criterion ≤ 1.0 La.
2. The as-left primary containment integrated leakage rate test (Type A test) acceptance criterion is ≤ 0.75 La.
3. The combined local leakage rate test acceptance criterion for Type B and Type C tests (excluding the leakage contributions from both the secondary containment bypass pathways and the main steam pathways) is ≤ 0.6 La, calculated on a maximum pathway basis, prior to entering a mode of operation where primary containment integrity is required.
4. The combined local leakage rate test acceptance criterion for Type B and Type C tests (excluding the leakage contributions from both the secondary containment bypass pathways and the main steam pathways) is ≤ 0.6 La, calculated on a minimum pathway basis, at all times when primary containment integrity is required.

5. Airlock overall leakage rate acceptance criterion is $\leq$ 0.10 La when tested at $\geq$ Pa.

The provision of the Definition (1.0.Y) for Surveillance Frequency does not apply to the test frequencies specified in the Primary Containment Leakage Rate Testing Program.

D. Radioactive Effluent Controls Program

This program conforming to 10 CFR 50.36a provides for the control of radioactive effluents and for maintaining the doses to members of the public from radioactive effluents as low as reasonably achievable. The program shall be contained in the ODCM, shall be implemented by operating procedures, and shall include remedial actions to be taken whenever the program limits are exceeded. The program shall include the following elements:

a. Limitations on the functional capability of radioactive liquid and gaseous monitoring instrumentation including surveillance tests and setpoint determination in accordance with the methodology in the ODCM;

b. Limitations on the concentrations of radioactive material released in liquid effluents from the site to unrestricted areas, conforming to 10 times the concentration values in Appendix B, Table 2, Column 2, to 10 CFR 20.1001 - 20.2402;

c. Monitoring, sampling, and analysis of radioactive liquid and gaseous effluents pursuant to 10 CFR 20.1302 and with the methodology and parameters in the ODCM;

d. Limitations on the annual and quarterly doses or dose commitment to a member of the public from radioactive materials in liquid effluents released from the unit to unrestricted areas, conforming to 10 CFR 50, Appendix I;

e. Determination of cumulative and projected dose contributions from radioactive effluents for the current calendar quarter and current calendar year in accordance with the methodology and parameters in the ODCM at least every 31 days;

f. Limitations on the functional capability and use of the liquid and gaseous effluent treatment systems to ensure that appropriate portions of these systems are used to reduce releases of radioactivity when the projected doses in a period of 31 days would exceed 2 percent of the guidelines for the annual dose or dose commitment, conforming to 10 CFR 50, Appendix I;

g. Limitations on the dose rate resulting from radioactive material released in gaseous effluents from the site to areas at or beyond the site boundary shall be limited to the following:

   1. For noble gases: less than or equal to a dose rate of 500 mrems/yr to the total body and less than or equal to a dose rate of 3000 mrems/yr to the skin, and

   2. For iodine-131, iodine-133, tritium, and for all radionuclides in particulate form with half lives greater than 8 days: less than or equal to a dose rate of 1500 mrems/yr to any organ;

h. Limitations on the annual and quarterly air doses resulting from noble gases released in gaseous effluents from the unit to areas at or beyond the site boundary, conforming to 10 CFR 50, Appendix I;

i. Limitations on the annual and quarterly doses to a member of the public from iodine-131, iodine-133, tritium, and all radionuclides in particulate form with half lives greater than 8 days in gaseous effluents released from the unit to areas beyond the site boundary, conforming to 10 CFR 50, Appendix I; and

j. Limitations on the annual dose or dose commitment to any member of the public, beyond the site boundary, due to releases of radioactivity and to radiation from uranium fuel cycle sources, conforming to 40 CFR 190.

Attachment 5

Vermont Yankee Nuclear Power Station

Proposed Technical Specification Change No. 262

Alternative Source Term

Safety Assessment

VERMONT YANKEE NUCLEAR POWER STATION (VYNPS)

APPLICATION FOR LICENSE AMENDMENT
ALTERNATIVE SOURCE TERM

SAFETY ASSESSMENT

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ACRONYMS AND ABBREVIATIONS

λ	Removal coefficient (hr^{-1})
$\mu Ci/gm$	micro-curies per gram
η	Removal efficiency
χ/Q	Atmospheric Dispersion Factor
" HG	inches of mercury
ALT	Alternative Leakage Treatment
AOG	Advanced Off Gas
AST	Alternative Source Term
BPWS	Banked Position Withdrawl Sequence
BWR	Boiling Water Reactor
BWROG	Boiling Water Reactor Owners Group
CAD	Containment Air Dilution
Cfm	cubic feet per minute
CsI	Cesium Iodine
CRDA	Control Rod Drop Accident
DBA	Design Basis Accident
DE	Dose Equivalent
DW	Drywell (Volume designation in RADTRAD)
EAB	Exclusion Area Boundary
ECCS	Emergency Core Cooling System
EDG	Emergency Diesel Generator
EPU	Extended Power Uprate
Ft	feet
GDC	General Design Criterion
GE	General Electric
gpm	gallons per minute
H+	Hydrogen Ion
HEPA	High Efficiency Particulate Air
HNO_3	Nitric Acid
Hrs	hours

ACRONYMS AND ABBREVIATIONS

In	inch
IST	In Service Teststing
Lbm	pounds-mass
LOCA	Loss of Coolant Accident
LPZ	Low Population Zone
MC	Main Condenser (Volume designation in RADTRAD)
MHE	Maximum Hypothetical Earthquake (same as SSE)
m/s	meters per second
MSIV	Main Steam Isolation Valve
MSL	Main Steam Line
MSLB	Main Steam Line Break
MWt	Megawatt thermal
OH-	Hydroxyl Ions
PC	Primary Containment (Volume designation in RADTRAD)
PCIS	Primary Containment Isolation Signal
pH	Hydrogen Ion Concentration
Psid	pounds per square inch differential
RB	Reactor Building, same as SC
Rem	roentgen equivalent man
RG	Regulatory Guide
SC	Secondary Containment, same as RB (Volume designation in RADTRAD)
Scfh	standard cubic feet per hour
SGTS	Standby Gas Treatment System
Secs	Seconds
SER	Safety Evaluation Report
SLC	Standby Liquid Control
Sodium-Pentaborate	$Na_2O\text{-}5B_2O_3\text{-}10H_2O$
SR	Surveillance Requirement
SRP	Standard Review Plan
SSE	Safe Shutdown Earthquake

ACRONYMS AND ABBREVIATIONS

TEDE	Total Effective Dose Equivalent
TS	Technical Specification
TSC	Technical Support Center
UFSAR	Updated Final Safety Analysis Report
UHS	Ultimate Heat Sink
VYNPS	Vermont Yankee Nuclear Power Station
WW	Wet Well, same as Torus (Volume designation in RADTRAD)

1. INTRODUCTION

1.1 Evaluation Overview and Objective

The objective of this safety assessment is to document implementation of the Alternative Source Term (AST) for VYNPS. The implementation of AST is governed by 10 CFR 50.67, the guidelines of the Standard Review Plan (SRP) Section 15.0.1 (Reference 1), and Regulatory Guide (RG) 1.183 (Reference 2).

VY has elected to perform a full scope implementation of the AST as defined in RG 1.183. The implementation consists of the following:

1. Identification of the core source term based on plant specific analysis of core fission product inventory.

2. Determination of the release fractions for the four Updated Final Safety Analysis Report (UFSAR) Boiling Water Reactor (BWR) Design Basis Accidents (DBAs) that could potentially result in control room and offsite doses. These are the loss of coolant accident (LOCA), the main steam line break accident, the refueling accident, and the control rod drop accident.

3. Calculation of fission product deposition rates and removal efficiencies.

4. Calculation of offsite and control room personnel Total Effective Dose Equivalent (TEDE).

5. Evaluation of suppression pool pH to ensure that the particulate iodine deposited into the suppression pool during a DBA LOCA does not re-evolve and become airborne as elemental iodine.

6. Calculation of new control room and EAB atmospheric dispersion factors (χ/Q) for Reactor Building leakage.

7. Calculation of a new Control Room atmospheric dispersion factors (χ/Q) for a main steam line break accident instantaneous ground level puff release.

8. Evaluation of other related design and licensing bases such as NUREG-0737 (Reference 3).

The radiological dose analyses have been performed assuming reactor operation at the Extended Power Uprate thermal power of 1950 MWt (102% of 1912 MWt). This results in a conservative estimate of fission product releases for operation at current licensed power of 1593 MWt.

1.2 Major Aspects of AST Analyses

Implementation of AST includes changes to the methodology presently used at VYNPS. These include:

1. Development of a bounding plant-specific core fission product inventory.
2. Establishing an Alternative Leakage Treatment pathway to the condenser.
3. Seismic ruggedness evaluation of the ALT boundary.
4. Evaluating the Reactor Building (Secondary Containment) Drawdown time.
5. Analysis of a new χ/Q for reactor building leakage and for an instantaneous ground level puff release to the atmosphere resulting from the main steam line break accident.
6. New requirements were developed for post-LOCA standby liquid control (SLC) system operation for suppression pool pH control.

1.3 ALT Pathway Seismic Ruggedness

Regulatory Guide 1.183, Appendix A (Reference 2) allows credit for a reduction in MSIV leakage rate due to hold up and retention in the main steam line piping downstream of the MSIV and in the condenser. This credit is based, in part, on the piping and components on the alternative leakage treatment (ALT) release path being capable of performing their safety functions during and after a safe shutdown earthquake. The VY AST implementation credits the ALT pathway. Appendix A of this Safety Assessment describes the ALT application.

1.4 Reactor Building Drawdown Time

The Alternative Source Term LOCA analysis considers the reactor building positive pressure period. This is the period when a loss of off-site power causes a loss of reactor building negative pressure relative to the environment. The start of the Emergency Diesel Generator (EDG) followed by the start of the Standby Gas Treatment System (SGTS) returns the reactor building to sub-atmospheric conditions. The time of positive pressure is called the "drawdown" time. The primary containment leakage rate in the reactor building is assumed to be released directly to the environment during the drawdown period. A plant specific calculation was performed to determine a conservative drawdown time. Appendix B of this Safety Assessment describes the drawdown time evaluation.

1.5 Summary

Implementation of the AST as the plant radiological consequence analyses licensing basis requires a license amendment per the requirements of 10 CFR 50.67. The enclosed AST analyses (References 33 through 39) demonstrate the offsite and control room post-accident radiological doses remain within regulatory limits.

2. EVALUATION

2.1 Scope

2.1.1 Accident Radiological Consequence Analyses

The DBA accident analyses documented in Chapter 14 of the VYNPS UFSAR (Reference 4) that could potentially result in control room and offsite doses were addressed using methods and input assumptions consistent with the AST. The following DBAs were addressed:

- LOCA, UFSAR Section 14.6.3
- Main Steam Line Break Accident, UFSAR Section 14.6.5
- Refueling Accident, UFSAR Section 14.6.4
- Control Rod Drop Accident, UFSAR Section 14.6.2

The analysis was performed per RG 1.183. The results were evaluated to confirm compliance with the acceptance criteria presented in 10 CFR 50.67 and GDC 19[1] of 10 CFR 50, Appendix A. Computer codes used in the design basis accident analyses results are listed in Table 2-1.

2.1.2 Suppression Pool pH Control

A calculation was performed to evaluate the suppression pool pH in the event of a DBA LOCA. The objective of the analysis was to demonstrate that the suppression pool pH remains at or above 7.0, thus ensuring that the particulate iodine (cesium iodide - CsI) deposited into the suppression pool during this event does not re-evolve and become airborne as elemental iodine. The analysis credits the pH buffering effect of sodium pentaborate introduced into the suppression pool post-LOCA by SLC operation to maintain the pH above 7.0.

SLC at VYNPS is a safety related system and its availability is governed by the Technical Specifications. SLC is suitably redundant in components and features to assure that its safety function can be accomplished assuming a single active failure. VYNPS has addressed one active and one passive potential failure that could impact the SLC system. The active failure is in the single control room keylocked-switch and associated logic that actuates SLC. The passive failure is one of the two check valves in series on the injection line that are credited to change state to inject the SLC solution. The mean failure frequency for check valves failing to open is in the range of 2.7E-4 per demand. The AST application would put even additional differential opening force on the check valves due to the depressurized reactor and an even lower failure rate would be expected. The mean failure frequency for manual switches

[1] Note that VYNPS is not a GDC plant and that compliance to GDCs are for reference purposes only and do not change the VYNPS original licensing basis.

(toggle, rotary and push button) is in the range of 2.4E-05 per demand. In addition, a failure of the keyed-switch could be addressed by repairs in the control room considering that SLC injection is really not necessary for the first two hours. Considering the failure rates, the VYNPS SLC system is considered suitably redundant and reliable for application as a pH control function.

2.1.3 Main Steam Line Break Accident Puff Release Dispersion Factor

A new control room χ/Q was determined for use in the main steam line break accident analysis. This χ/Q reflects an instantaneous ground level "puff" release to the atmosphere and is conservative to the puff methodology in RG 1.194. No buoyancy is considered.

2.1.4 NUREG-0737 Evaluation

An evaluation was performed to identify potential impacts of applying AST methodologies in accordance with NUREG-0737. This evaluation included the following:

- Revision of the current radiological dose analyses for post-accident vital area access and post-accident sampling (NUREG-0737, Item II.B.2 and Item II.B.3),
- Revision of the current radiological dose analyses for the post-accident containment high range radiation monitors (NUREG-0737, Item II.F.1),
- Revision of control room post-accident radiological dose analyses for emergency support facility upgrades and control room habitability (NUREG-0737, Items III.A.1.2 and III.D.3.4), and
- Consideration of post-accident sources of radiation and radioactivity outside the primary containment in terms of impact on dose analysis related to integrity of systems outside containment likely to contain radioactive material (NUREG-0737, Item III.D.1.1).

2.1.5 Environmental Qualification

The radiation doses used for the environmental qualification analyses at the original licensed thermal power conditions were calculated using source terms determined by TID-14844 (Reference 5) methodology. The radiation doses used for the environmental qualification analyses at the current licensed thermal power are adjusted upward from the original values based on the determined source term of the ORIGEN computer code for the and Extended Power Uprate (EPU) condition.

2.2 Method of Evaluation

2.2.1 Accident Radiological Consequence Analyses

Analyses were prepared for the simulation of the radionuclide release, transport, removal, and doses estimated for the postulated accidents listed in Section 2.1.1.

The ORIGEN code (Reference 6) was used to calculate plant-specific fission product inventories which bound the effect of eighteen-month fuel cycles, power operation at EPU conditions (1950 MWt (102% of 1912 MWt)), and using current and anticipated fuel lattice designs. The fission product inventory for the General Electric GE-14 fuel design was evaluated. Bounding values of fission product activity were determined for each radionuclide in the DBA radiological analyses by considering enrichment and exposure. Fission product activities were calculated for immediately after shutdown and decayed for the required times. The shutdown values are shown in Table 2-2. The calculation is fully documented in Reference 38.

The RADTRAD computer code Version 3.02(a) (Reference 7) was used for the DBA dose calculations except for the FHA and MSLB. Due to simplifying and conservative assumptions for the FHA and MSLB, a spreadsheet was used to calculate the control room, EAB and LPZ doses. The computer code STARDOSE (Reference 8) was used to check the RADTRAD and spreadsheet results, except for the MSLB results which were verified by manual calculations. The RADTRAD and STARDOSE programs are radiological consequence analysis codes used to determine post-accident doses at offsite and control room locations. The STARDOSE code is the proprietary property of Polestar Applied Technology, Inc., and the NRC has previously reviewed results obtained from the application of this code[2].

The existing UFSAR χ/Q values were developed prior to and used in support of an earlier license amendment request (References 9 and 10) for elimination of the main steam line high radiation containment isolation requirement following the BWROG methodology (Reference 23). The χ/Q values for elevated releases to other receptors were evaluated using the methods of Regulatory Guides 1.111 (Reference 13) and/or 1.145 (Reference 14). The meteorological data used for generating the χ/Q values was reviewed by the NRC in the Safety Evaluation for Amendments 212 (References 9 and 10) and found to be of high quality (Reference 10). New χ/Q sets have been developed supplementing the UFSAR sets for the AST implementation. Updated Control Room χ/Q values for reactor building releases were calculated using the computer code ARCON96 (Reference 11) using the methods of Regulatory Guide 1.194 (Reference 12). The χ/Q values applicable to the time periods, distances, and geometric relationships are shown in Tables 2-3 through 2-6. New control room χ/Q values associated with an instantaneous ground level puff release were also generated for the case of a main steam line break accident (see Section 2.2.3).

[2] Perry AST application

The post-LOCA shine dose to personnel in the TSC includes the radiation shine from the secondary containment airborne activity. This evaluation was performed using the MicroShield code, Version 5.03 (Reference 15). MicroShield is a point kernel integration code used for general purpose gamma shielding analysis. MicroShield has been used in safety-related applications by many nuclear power plants. The MicroShield results were independently verified with the QADMOD code. The QADMOD code is a Point Kernel gamma-ray shielding code with Geometric Progression Building Factors (Reference 26).

2.2.2 Suppression Pool pH Control Calculation

The calculation methodology for suppression pool pH control was based on the approach outlined in NUREG-1465 (Reference 16) and NUREG/CR-5950, (Reference 17). Specifically, credit was taken for sodium pentaborate addition to the suppression pool water as a result of SLC operation. The pH of the suppression pool water was then calculated using the STARpH code (Reference 18). This same methodology and code for calculation of transient suppression pool pH (including the formation of acids by radiation effects on Drywell components) was applied to the Hope Creek AST application (Reference 19).

Calculations were performed to verify sufficient sodium pentaborate solution is available to maintain the suppression pool pH at or above 7.0 for 30 days post accident. The design inputs were conservatively established to maximize the post-LOCA production of acids and to minimize the post-LOCA production and/or addition of bases. Other design input values such as initial suppression pool volume and pH were selected to minimize the calculated pH. It was determined that the calculated required quantity of sodium pentaborate was met by the current TS limit (Reference 37).

2.2.3 Main Steam Line Break Accident Instantaneous Ground Level Puff Release Dispersion Factor

To meet the criteria of RG 1.183, assuming an instantaneous puff release for the main steam line break, a new χ/Q for a puff release was calculated. The calculation of the main steam line break accident ground level puff release dispersion factor uses plant parameters for the main steam line break accident (e.g., mass of liquid-steam mixture released, timing of release, temperature of the liquid-steam mixture) to obtain the initial conditions of the released steam puff. The steam puff is treated as a hemispherical "bubble" with a given transit time up to and across the control room intake. Air entrainment is not considered (i.e., minimum dilution). No credit is taken for concentration gradients within the bubble. In particular, no credit is taken for a vertical concentration gradient; (i.e., the concentration at the elevation of the control room air intake is assumed to be the same as that of the leading edge of the bubble). The VY "puff" dispersion factor was compared to the dispersion factor calculated by the RG 1.194 method and found to be conservative.

The bubble is assumed to be released from the turbine stop valve corresponding to the shortest distance to the control room intake. No credit is taken for wind direction; (i.e., it is assumed that the centerline of the bubble trajectory always passes over the control room intake). The puff χ/Q calculation is described in Section 2.3.1.2.

2.2.4 NUREG-0737 Evaluation

- Post Accident Vital Area Access and Sampling - Post-accident personnel missions resulting in mission doses (including post-accident sampling) have been previously identified. The implementation of the AST methodology with the ALT treatment does not result in any new operator missions. Plant calculations used in support of plant post-accident vital area access (prepared in accordance with NUREG-0737, Items II.B.2 and II.B.3) were evaluated for impact by AST. The evaluation considered the comparative radiation levels from AST and the existing TID-14844 methodology source terms (such as airborne activity in the reactor building and also as activity in the suppression pool water).

- Post-Accident Radiation Monitor - Post-accident containment high range radiation monitoring calculations were revised for impact by AST (NUREG-0737, Item II.F.1).

- Control Room Radiation Protection - The control room radiological dose impact of AST has been specifically calculated for each of the four DBAs analyzed for AST implementation (NUREG-0737, Item III.D.3.4).

- Radioactive Sources Outside the Primary Containment - The DBA LOCA control room dose analysis, as well as that for offsite doses, considers the effects of coolant leakage outside the primary containment and (for the control room and TSC dose analyses only) the shine contribution from the reactor building and other source term bearing systems and/or components (NUREG - 0737, Item III.D.1.1).

2.3 Inputs and Assumptions

2.3.1 Accident Radiological Consequence Analyses

For AST accident radiological consequences, analyses were performed for the four DBAs that could potentially result in control room and offsite doses. These are the LOCA, main steam line break accident, refueling accident, and control rod drop accident.

Plant-specific fuel design parameters were used in the fission product and transuranic nuclide inventories for the accident analyses. Table 2-7 summarizes key fuel cycle parameters.

The reactor core inventory for the AST dose analyses is based on a parametric approach that included the fuel design and exposures experienced and projected for

EPU. Enrichment and average core burn-up included 3.0 and 4.65 w/o U-235 and exposure steps from 5 to 58 GWd/MT respectively. The AST source term is a composite from the parametric ORIGEN cases based on the maximum nuclide concentration and bounds any combination of enrichment and exposure on the range that is expected for VYNPS (Reference 38). For the refueling accident analyses, a RG 1.183 minimum core radial peaking factor of 1.65 was used along with the core isotopic inventory after 24 hours of decay.

The release source term is developed using the radionuclide isotopes listed in Table 2-2 and the release fractions from Table 1 of RG 1.183. The radionuclides that are included are those identified as being potentially important contributors to TEDE in NUREG/CR-6604 (Reference 7). Release fractions for LOCA as release rates are shown in Table 2-9.

Credit taken for the SGTS and the system functions modeled in the AST radiological dose analyses is presented in Table 2-8. The Advanced Off Gas system is credited as a pathway in one of the three CRDA cases evaluated. The CRDA analysis for AST was carried out with the same scenarios and key assumptions as were submitted and reviewed under Amendment 212 (References 9 and 10).

An assumed unfiltered in-leakage rate of 3700 cubic feet per minute into the control room was used; that is, the post-isolation unfiltered in-leakage rate is the same as the pre-isolation unfiltered fresh air intake. This in-leakage rate was acknowledged by NRC in the Safety Evaluation for Amendments 212 (Reference 10).

The control room ventilation system is manually placed in recirculation mode by the control room operators when measured airborne contamination in the turbine building is greater than 0.3 DAC values of 10CFR20, Appendix B, Table 1, Column 3 (Reference 30). The adequacy of the radiation monitoring set-point was reviewed as part of the AST NUREG-0737 evaluation. However, this manual action is not credited in the analysis since the post-isolation unfiltered in-leakage assumed is the same as the pre-isolation unfiltered fresh air intake.

The VYNPS Emergency Core Cooling Systems are designed, maintained, and tested to minimize the radiological consequences following a postulated DBA. The AST analyses inputs and assumptions are consistent with the design and licensing bases for these systems.

The standard breathing rates specified in RG 1.183 have been used. The key accident radiological consequence analyses inputs are summarized in Table 2-10.

2.3.1.1 LOCA Inputs and Assumptions

The key inputs used in this analysis are included in Tables 2-3 and 2-11. These inputs and assumptions fall into three categories: Radionuclide Release Inputs and Timing, Radionuclide Transport Inputs, Radionuclide and Removal Inputs. The calculation is

documented in Reference 33. The analysis includes three release pathways (or cases) as follows:

Case 1: Leakage from Primary Containment (PC) directly to the environment (Secondary Containment (SC) or Reactor Building (RB) bypass);

Case 2: Leakage from the PC into the RB and subsequent release to the environment via the Standby Gas Treatment System (SGTS) and the plant stack;

Case 3: Leakage from the PC via the Main Steam Isolation Valves (MSIVs) to the Main Condenser (MC) and subsequent release to the environment.

All of these pathways are analyzed for two accident scenarios: one in which the failure of an SGTS train delays drawdown of the SC (affecting Cases 1 and 2) and one in which an MSIV fails to close (affecting Case 3). The results of these two scenarios determine the limiting single failure. Assumptions are then scenario specific. Summaries of the results are presented in Table 3-1.

The analysis assumes; the main steam lines and the main condenser are seismically rugged, and will remain intact during and after a design basis Maximum Hypothetical Earthquake (MHE), and that the MSIV leakage eventually collects in the main condenser (except for a small portion that is assumed to bypass the main condenser).

Radionuclide Release Inputs and Timing

The Case 1 and Case 3 releases are from either the RB (Case 1) or the MC/turbine stop valves (Case 3) both at ground level. The Case 2 releases are from the plant stack. The exact leak location for the release from the MC is not known, but it is assumed to be at the location of the turbine stop valves where the leakage bypassing the MC is also assumed to occur. The RB bypass is also treated as a ground-level release. It may occur from two locations: the RB siding on the refueling elevation during drawdown (i.e., the establishing of a stable negative pressure in the RB at the beginning of SGTS operation) and at the RB penetration for the nitrogen system.

Event Timing

- LOCA occurs at time 0 minutes. Degraded core cooling leads to core damage.
- Release from core to PC begins at 2 minutes. A drain-line pathway is established from the main steam lines to the MC.
- SGTS starts automatically and RB drawdown is achieved by 10 minutes.
- Drywell sprays are initiated at 15 minutes.
- Further core damage and associated activity releases are terminated at 122 minutes by assumed restoration of core cooling. Drywell and Torus airspace become well-mixed at that time.

- Within a few hours, Standby Liquid Control (SLC) is initiated and the contents of the SLC system begin to mix with the suppression pool water.
- By 24 hours, the containment pressure has decreased to less than 5.5 psig, and the PC leak rate has become a factor of two less than the maximum PC leak rate (except for Engineered Safety Feature (ESF) liquid leakage).
- By 720 hours, essentially all particulate activity has been leaked or deposited and gaseous I-131 (the principal dose contributor excluding particulate I-131) has gone through nearly four half-lives. The dose calculation is terminated in accordance with Reference 2.

The timing of all of these events is based on Reference 2 except for establishing the drain-line pathway, drawdown time, Drywell spray initiation, Drywell and Torus mixing, SLC injection, and containment leak rate reduction justification. These are covered in the following justifications.

Drain Line Pathway

The drain-line pathway to the MC is expected to be established very early in the accident response. However, if such a response were delayed for half an hour, the dose impact would be minimal (less than two percent of the CR dose limit). Therefore, the exact timing of this action is not considered critical. The ALT pathway is described in Appendix A of this Safety Assessment.

Drawdown Time

This is leakage from the PC that occurs prior to establishing a sustained negative pressure in the SC; and, therefore, it is assumed to leak directly to the environment from the refueling elevation via sheet-metal siding. The time at which the SC pressure becomes sufficiently low to justify no further out leakage is an important parameter of the DBA-LOCA analysis. The value used is that specified in Appendix B of this Safety Assessment.

Drywell Spray Initiation

Drywell spray initiation is called for in the plant procedures. For example, an accident involving the degree of core damage postulated in Reference 2 for the DBA LOCA (and used herein), the plant procedures would be called upon to guide operator actions. This guidance calls for Drywell spray operation before the radiation level in the Drywell exceeds 4,000 rads/hour.

The Drywell high radiation monitor response due to the release of gap activity (5% of the entire core noble gases, halogens and alkali metals in the first 32 minutes) to the containment using shutdown core inventory (i.e., early in the accident) will yield an indication on the containment high-range monitor from the noble gases alone of about 4,500 rads/hr in five (5) minutes. Assuming two sources (noble gases and halogens)

are released from the gap and remain airborne, the monitor reading at this time would be exceed 20,000 rads/hr. The following table provides the Drywell high range monitor response as a function of time corresponding to the gap inventory with the assumption that only the noble gases are released and become airborne. This assumption minimizes the monitor response per Reference 39.

Time	Monitor Response (rads/hr)
5	4,500
10	8,300
15	11,000

Therefore, the radiation level calling for spray operation will be reached well before the assumed spray actuation time of 13 minutes after the start of the gap activity release. The VY sprays are designated Safety-Related and their availability is governed by the Technical Specifications.

Drywell and Torus Mixing

Reference 2 establishes that only the Drywell volume should be credited for diluting the activity release from the core for a BWR. For Mark III containment designs, specific instructions are then provided as to how to subsequently treat mixing between the Drywell and the remainder of the containment. For Mark I and Mark II plants, however, no specific guidance is provided. Instead, the general guidance is that the Torus airspace "...may be included provided there is a mechanism to ensure mixing...".

AST applications have been accepted by the NRC in which the full containment volume (Drywell + Torus airspace) has been credited from time zero. The VY analysis credits the two volumes beyond 122 minutes following the restoration of core/core debris cooling when considerable thermal-hydraulic activity in the PC will result in the Drywell and Torus airspace volumes becoming well-mixed.

SLC Injection

The injection of the SLC sodium pentaborate is justified by plant procedures (as with Drywell sprays). If core damage is expected or identified as a result of normal and emergency core cooling not being available or sufficient, the plant procedures provide guidance for injecting all available water sources into the reactor vessel. This would include SLC injection. Therefore, SLC injection is expected for this event.

The VY SLC system is designated Safety-Related and its availability is governed by the Technical Specifications and was discussed in Section 2.1.2.

The SLC injection will maintain the suppression pool pH above 7.0 for 30 days, and radioiodine re-evolution does not need to be considered (Reference 37).

Primary Containment Leakage

The maximum allowable primary containment leakage rate is eight-tenths percent (0.8%) primary containment air weight per day. This leakage rate was assumed in the AST analyses for the first 24 hours.

Containment Leak Rate Reduction Justification

Reference 2 requires justification for implementing a factor of two decrease in PC leak rate at 24 hours after the start of the accident. Reference 33 provides the full justification and a summary is included in this section.

The use of sprays reduces the VY Drywell pressure to ~20 psia (5.3 psig) at 24 hours from a peak value of 58.7 psia (44 psig), a ratio of 0.12 based on the gauge pressure.

If the leak path is sufficiently restrictive so that choked flow is not occurring and the problem may be treated as incompressible flow, a factor of 3.33 reduction in containment pressure will yield a reduction in volumetric flow of about 1.8 (approximately a factor of two) if the density is assumed constant. Since the containment is a closed system, the density of the non-condensables will not change during depressurization (the pressure decrease being the result of a temperature reduction) except for steam condensation. However, the steam condensation effect is not neglected in this evaluation. For VY's peak pressure of about 44 psig, the factor of two reduction in volumetric leak rate is not achieved until a pressure of about 5.5 psig is attained, about a factor of eight reduction in containment pressure. NRC has previously given credit for a factor of two reduction in containment leak rate at 24 hours in some BWR AST applications with as little as a factor of two reduction in containment pressure. VY meets this basis at approximately 24 hours since the pressure reduction for VY (with spray credit) is more than a factor of eight; i.e., it is a factor of 44/5.3 or 8.3. The VY pressure decrease of a factor of eight is a sound technical basis for the containment leakage rate reduction.

ESF Leakage

Leakage from Engineered Safety Features (ESF) was reviewed (Reference 20). Vermont Yankee has implemented a program to reduce leakage from systems outside containment that would or could contain radioactive fluids during an accident to as low as practical levels. The program includes the following (Reference 24):

- Provisions establishing preventative maintenance and periodic visual inspection requirements,
- System leakage inspections, to the extent permitted by system design and radiological conditions for each system at a frequency not to exceed refueling cycle intervals. The following systems are subject to this testing.
    - o Residual Heat Removal

- o Core Spray
- o Reactor Water Clean-up
- o HPCI and RCIC
- o Sampling systems

The Vermont Yankee program effectively eliminates ESF leakage. However, the LOCA analysis assumed an Emergency Core Cooling System leakage rate of one-half (0.5) gallon per minute (gpm) into the reactor building analyzed as one (1) gpm starting at the onset of the event.

MSIV Leakage Rate

The total MSIV leakage rate of 124 scfh (maximum of 62 scfh in any two lines is the limiting case) was assumed in the analyses. The steam line filtration efficiencies calculation does not include the factor of two reduction in MSIV (and other) leak rates that is assumed at 24 hours. This is a conservative assumption since a reduction in these leak rates would increase the filtration efficiencies.

Secondary Containment Bypass Leakage

Primary containment leakage via the nitrogen supply lines which penetrate the RB from the outside on the RB South wall and then penetrates the PC via a Drywell penetration. Leakage from the PC back through this system's closed primary containment isolation valves (PCIVs) could bypass the SC and the SGTS filters and could also result in a ground-level release. A leakage rate of 5 scfh is conservatively assumed to begin at the start of the event.

<u>Radionuclide Transport Inputs</u>

Case 1 – Leakage from Primary Containment Directly to the Environment (Bypass Pathway)

This is the first pathway that makes a significant contribution to the DBA-LOCA doses. There are two components of this pathway. The first is pre-drawdown PC leakage (0.8 %/day). This is leakage from the PC that occurs prior to establishing a sustained negative pressure in the SC; and, therefore, it is assumed to leak directly to the environment from the refueling elevation via sheet-metal siding.

The second component is the nitrogen supply which penetrates the PC and then penetrates the RB on the RB's south side. Leakage from the PC through this system's closed containment isolation valves (CIVs) could bypass the SC and the SGTS filters and could also result in a ground-level release.

Pathway Assumptions – Case 1

The drawdown bypass occurs during the first 10 minutes of the DBA-LOCA, accident time. Even though there is a period during the 10 minutes when the RB pressure is actually sub-atmospheric, the full 10 minutes is used.

The release from the core is assumed to enter the Drywell only. Mixing within the entire PC is not assumed to occur until after the end of the release.

No credit is taken for natural deposition in the Drywell during the drawdown period; credit for Drywell deposition does not begin until Drywell sprays start at 15 minutes. No credit is taken for deposition in the unspecified leak path(s) that lead to this bypass.

The sustained bypass through the nitrogen system is treated very conservatively. No credit is taken for deposition in piping or components (either inside or outside the PC), and this includes the nitrogen heater. Both this release and the drawdown bypass are assumed to be released at ground level.

The drawdown bypass corresponds to the PC leak rate of 0.8%/day. The sustained bypass via the nitrogen supply pathway has an assumed leak rate of 5 scfh which equates to 0.035%/day. The leakage rate is reduced to 0.019%/day at the end of the release (2.033 hours) after establishing well mixed conditions. Then at 24 hours, this leakage rate reduces to 0.010%/day after a reduction in containment pressure.

The Case 1 model includes two parallel main steam line flow paths to the ALT volume as well as the pathway to the RB. These are discussed in more detail for the MSIV leakage pathway RADTRAD model and the RB/SGTS/plant stack pathway RADTRAD model, respectively. They are included in this model to properly account for the associated leakage out of the PC.

Single-Failure Considerations

If there is not a single-failure of a SGTS train, there will not be a positive pressure period for the RB and there will not be any drawdown bypass. There will continue to be a RB bypass associated with the nitrogen system.

Case 2 – Leakage from Primary Containment to the Environment via the Reactor Building, SGTS, and Plant Stack (RB/SGTS/Plant Stack Pathway)

For this pathway, a single junction is provided from the "Drywell" control volume (before 2.033 hours) and a single junction is provided from the "DW and WW" control volume (after 2.033 hours) to represent the 0.8%/day PC leakage to the RB. Added to this is the ESF leakage which is modeled as a continuous 1 gpm (0.134 cfm) volumetric flow from the "Pool" control volume to the RB.

Pathway Assumptions – Case 2

Airborne releases from the PC to the RB begin after the drawdown period. ESF leakage is assumed to begin immediately.

Since the "Pool" control volume receives the full release in parallel with the "Drywell" and the "DW and WW" control volumes, five percent of the iodine (total) is in elemental and organic form. If the particulate were filtered out entirely in the junction from the "Pool" to the RB, only 5% of the iodine would be released to the RB. Ten percent is required. Therefore, the particulate filter is set at 94.74% permitting another 5% of the iodine to become airborne. This iodine does not have the correct chemical form; but since the SGTS filter efficiencies are all 95% and since the CR has no incoming air filtration, the dose calculation for radioiodine is correct.

This approach to ESF leakage also "inadvertently" permits 100% of the noble gas and slightly more than five percent of the particulate in the one gpm "Pool" control volume leakage to be released to the RB along with the intended 10% of the radioiodine. This is conservative.

Single Failure Considerations

If there is not a single-failure of a SGTS train, there will not be a positive pressure period for the RB, and there will not be any drawdown bypass

Case 3 – Leakage from Primary Containment to the Environment via the Main Steam Lines and the Main Condenser (MSIV Pathway)

For this pathway, two junctions are provided from the "Drywell" control volume and two from the "DW and WW" control volume to represent the two leaking steam lines. These junctions all terminate in the "ALT" control volume. The "ALT" control volume represents the isolated main steam lines out to the turbine stop valves. This control volume can leak directly to the environment (representing main condenser bypass), and it can leak to the main condenser (drain line connection). The main condenser can then leak to the environment.

The RADTRAD model for this pathway also includes leakage from the PC to the RB so that the PC activities are determined correctly. However, no leakage to the environment is permitted other than that through the MSIVs. Drywell sprays are modeled in an identical manner as described for the bypass pathways.

Pathway Assumptions – Case 3

The details for developing the RADTRAD modeling of the MSIV leakage pathway are covered in Appendix A. The removal efficiency summary is provided in Table 2-11.

The containment pressure reduction would result in factor of two reduction in MSIV (and other) leak rates that is assumed to occur at 24 hours. Even though this

reduction in MSIV leak rate would increase the filtration efficiencies, that benefit is conservatively omitted.

Single Failure Considerations

To consider a single failure of an MSIV to close, the analysis in Reference 33 considered two MSIV leakage pathway models. The first (using the terminology of Reference 33) is "A" in which the space between the MSIVs is ignored. This would correspond to a failure of one MSIV to close. Under that condition, the space between the MSIVs could be considered part of the drywell (inboard MSIV fails to close) or part of the control volume defined by the closed inboard MSIV (outboard MSIV fails to close) and the turbine stop valves. The former is the more conservative assumption, and it is on that basis that the "A" removal efficiencies were calculated; i.e., they were kept the same as "B2" (see next paragraph).

The second pathway model considered in Reference 33 consists of control volumes "B1" and "B2" in series. This pathway model is for lines with both MSIVs closed. To model a single failure of an MSIV, it is only necessary (1) to use the average particulate DF for the two Appendix A models (instead of that for the B1/B2 models alone) for the RADTRAD input for the pathways from the ALT volume to the main condenser and to the environment and (2) to reduce the ALT volume by the volume of one line between the MSIVs corrected for the expanded flow in the ALT as compared to that in the space between the two MSIVs. This is explained more fully in Reference 33.

<u>Radionuclide Removal Inputs</u>

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

In the Drywell

The Drywell spray removal rate development applies to both the MSIV leakage pathway and the RB/SGTS/plant stack pathway, as well as to the RB bypass leakage pathway described above.

Drywell spray removal for particulate is determined in accordance with Reference 21. There are three system-related parameters that are needed to employ the Reference 21 particulate removal model: spray flow rate, spray fall height, and the volume sprayed.

The spray flow rate is 6650 gpm for one of two redundant loops. Only one loop is credited. A reduction is taken in the spray flow and in the fall height to account for Drywell congestion. The particulate removal rate, λ, was calculated in Reference 33 and rounded down to 20 hr^{-1} as applied to the RADTRAD model. To properly reflect the effective increase in volume that occurs as a result of Drywell/Torus mixing at the

end of the release phase, the removal lambda is decreased from 20 hr^{-1} to 11.3 hr^{-1} at 2.033 hours.

The removal rate for elemental iodine is limited to 20 hr^{-1} and the elemental iodine spray λ limitation of Reference 21 is complied with.

In the Steam Lines

The AEB-98-03 (Reference 22) model is used as the basis for obtaining the deposition velocity for particulate. The AEB-98-03 model assumes a well mixed control volume. For the VY application, this particulate settling velocity distribution needs to be adjusted to account for the DW spray.

Modification of the AEB-98-03 settling velocity distribution because of the Drywell spray credit is fully described in Reference 33. The median value of the modified settling velocity distribution calculated in Reference 33 is 5E-5 $m \cdot s^{-1}$ and is applied to the main steam lines and condenser control volumes.

Aerosol Removal in the Main Steam Lines and Condenser

It is appropriate (and somewhat conservative) to assume an aerosol settling velocity of 5E-5 $m \cdot s^{-1}$ in the main steam lines and main condenser when the Drywell sprays are operating. For any other time (e.g., the first 15 minutes of the accident), it is even more conservative to do so.

The condenser removal is also treated conservatively by using only four percent (4%) of that projected surface area available for sedimentation. In the condenser, the calculated volumetric flow rate is 123 cfh or 2.05 cfm. This is conservative in that no steam condensation in the main condenser is credited, only a decrease in the temperature of the leakage. The leakage of 2.05 cfm is about three percent per day of the 107,000 ft^3 main condenser volume or 1.15E-3 volumes per hour.

The aerosol sedimentation λ_{sed} and the removal efficiency, η_{sed}, were calculated in Reference 33 and are summarized in the following table.

Volume	λ sed (hr^{-1})	η sed
Steam Line Outboard MSIV to Turbine Stop Valves	0.56	71%
Steam Line Between MSIV	0.56	38%
Condenser	0.0225	95.1

Elemental Iodine Removal in the Main Steam Lines and Condenser

The model used in the steam lines is the Bixler Model from NUREG/CR-6604 and the application is discussed in Reference 33. The analysis documented in Reference 33 fully documents the elemental iodine removal λ_{ei} in the steam lines.

The model used in Reference 33 for the Main Condenser is taken from SRP 6.5.2 (Reference 21). The elemental iodine removal coefficient, λ_{w}, is a function of the deposition velocity which, from Reference 21, is 4.9 m hr^{-1} and the condenser surface area for deposition which is conservatively calculated to be 4078 ft^2. The λ_{w} becomes 0.61 hr^{-1}.

The removal efficiency is obtained from the λ_{w} and the condenser leakage the λ_{leak} of 1.1553E-03 hr^{-1}.

Volume	λ_{w} (hr^{-1})	η_{w}
Condenser	0.61	99.8

One may notice that the elemental iodine removal efficiency in the condenser is greater than the corresponding removal efficiency for particles; i.e., 99.8% > 95.1%. In this regard, it is important to note that very small particles are actually removed more readily by diffusion than by sedimentation and that when the removal process becomes dependent on diffusion, the smaller the particle, the better the removal. In the limit, gases diffuse more readily than particles; and, therefore, it is not inconsistent that gases would be removed more readily than very small particles in the main condenser.

<u>Technical Support Center (TSC) LOCA 30-day Dose</u>

The Technical Support Center (TSC) 30-day dose analysis is based on the analysis of the Control Room 30-day dose. The control room dose analysis used the STARDOSE code only for independent verification; but in the TSC analysis, the code is used to generate the activity releases used for assessment of both the inhalation and the shine pathways.

The same fresh air intake location which serves the CR also serves the TSC; and as with the CR, there is no filtration of the makeup air supply. Therefore, the TSC dose from activity brought into the ventilation system will be the same as that for the CR.

The assessment of TSC habitability differs from the assessment of CR habitability in the area of external radiation effects. The TSC is not heavily shielded in the same manner as the CR. Therefore, some conservatisms in the RB drawdown bypass modeling and in the nitrogen supply RB bypass modeling that were included for the CR have been relaxed for the TSC. The TSC calculation has two parts: (1) a recalculation of the CR/TSC doses from activity brought in through the common fresh

air intake using the STARDOSE model (but with some of the bypass conservatisms relaxed and including holdup within the RB for consistency with the TSC external shine calculation), and (2) a calculation of the TSC external shine using the RB activities as a function of time. Reference 40 documents the comprehensive NUREG 0737 evaluation.

The base case analyzed for the TSC is the limiting case for the CR dose; i.e., the failure of one SGTS train. This failure also maximizes the activity within the RB.

The dose from activity entering the CR/TSC ventilation supply for the DBA-LOCA includes the contribution from the reactor bypass pathways. The bypass refinements are as follows:

- Credit is taken for the fact that during the nominal 10-minute drawdown time, the RB is actually at a positive pressure for only six minutes. The six minutes of positive pressure are assumed to be at the end of the 10-minute period.
- Credit is taken for particulate deposition in a portion of the nitrogen supply leakage pathway leading from the PC to the South wall of the RB. This credit makes use of the same modeling as that used for the MSIVs.

2.3.1.2 Main Steam Line Break Accident Inputs and Assumptions

The MSLB accident is initiated from hot stand-by conditions in order to conservatively maximize the mass of coolant released from the break and thus maximizing the activity released. The main steam line break accident assumes a double ended break of one main steam line outside the secondary containment with displacement of the pipe ends that permits maximum blow-down rates. Following accident initiation, the radionuclide inventory from the released coolant is assumed to reach the environment instantaneously. The main steam line break accident analysis is fully document in Reference 34.

The radiological consequences of the design basis main steam line break accident were analyzed using a spreadsheet. Two cases were evaluated that corresponded to the iodine concentration in the primary coolant:

- Pre-accident spike of 4 µCi/gm Dose Equivalent (DE) I-131 (TS maximum allowed value).
- A value of 1.1 µCi/gm DE I-131 corresponding to the maximum TS value allowed for continued operation.

The break mass released includes the line inventory plus the system mass released through the break prior to isolation. Break isolation was assumed in 6.8 seconds. This assumption bounds the maximum isolation time for an MSIV of 5.5 seconds including isolation instrumentation response time. This results in the maximum radiological release for analysis. Other assumptions in the analysis are the following:

- There is no holdup in the Turbine Building.
- The entire released coolant mass is conservatively used (rather than just the liquid mass) in the calculation of the activity released.
- There is no fuel damage.
- Pre-accident noble gas release rate to the atmosphere set to TS limit of 0.16Ci/sec.
- An infinite exchange rate between the Control Room and the environment is assumed.
- The analysis assumes an instantaneous ground level puff release.

The MSLB evaluation considered both a quasi-steady state release χ/Q and an instantaneous "puff" release χ/Q. The accident scenario is a turbine building release that lasts only 6.8 seconds. It is unlikely that the control room dose will result from a slow release of activity from the turbine building following a very rapid release from the main steam line. Therefore, the slow release quasi-steady state χ/Q is not considered valid and is not consistent with other more realistic release characteristics. The instantaneous "puff" release χ/Q is used in the MSLB analysis.

The analysis considered four steps to the puff release χ/Q calculation: (1) evaluating the initial conditions of the steam release, (2) determining the volume and the diameter of the assumed hemispherical steam bubble, (3) evaluating the transit time for the bubble across the Control Room air intake and the associated χ/Q, and (4) comparing the VYNPS calculation to the NRC modeling described in Reference 12.

The Regulatory Guide 1.194 puff model (Reference 12) produced a χ/Q that is about 20% less than the hemispherical model developed for the VYNPS application. Therefore, the hemispherical model is conservative with respect to RG 1.194 model.

RG 1.183, in Section 4.4 of Appendix D, indicates that the iodine species released from the main steam line should be assumed to be 95 percent CsI as an aerosol, 4.85 percent elemental, and 0.15 percent organic. This difference is inconsequential for the VYNPS MSLB AST analysis since no credit is taken for filtration or other removal mechanisms of iodine, such as plate-out, sedimentation, condensation, or decay.

The key inputs used in this analysis are included in Tables 2-4 and 2-12.

2.3.1.3 Refueling Accident Inputs and Assumptions

This postulated refueling accident involves a 30 foot drop of a fuel assembly on top of other fuel assemblies during refueling operations. The drop distance bounds the maximum height that is allowed by the VYNPS refueling equipment and is the

limiting case since the kinetic energy for the drop produces the largest number of damaged fuel pins on impact. The refueling accident analysis is fully document in Reference 35.

The analysis is fully compliant with Regulatory Guide 1.183 (Reference 2). The analysis was performed for 24 and 96 hours after shutdown and included both elevated and ground releases. The 24 hour elevated calculation will be discussed. Following accident initiation at 24 hours after shutdown, the radionuclide inventory from the damaged fuel pins is assumed to leak out to the environment instantaneously (even though releases to the environment could be assumed to occur over a 2-hour period according to Reference 1). The RG 1.183 minimum core radial peaking factor used for the VY analysis is 1.65.

The analysis was performed with credit for the plant stack in achieving an elevated release for the activity released from the damaged fuel. SGTS filtration is not credited. Due to these simplifying, conservative assumptions, a spreadsheet was used to calculate the control room, EAB and LPZ doses. The spreadsheet results were verified with the STARDOSE code. The elevated release case corresponds to 24 hour decay time are presented in this safety assessment. Assumptions with justifications are the following:

- The accident is assumed to occur at 24 hours after shutdown. Consequently, core inventories were calculated that correspond to this decay time. Fuel handling would not begin before 24 hours after shutdown.
- The release occurs within two hours (Reference 2).
- No DF is applied to noble gases.
- The refueling pool DF for elemental iodine was calculated as follows:
    - Assume an effective DF of 200.
    - The fraction of the iodine inventory released from the pool is 1/200 = 0.005. Of this, 0.0015 is for organics, so the elemental iodine release fraction is 0.0035.
    - $DF_{eI} = 1.0/0.0035 \approx 285$.
- The DF for other radionuclides is assumed to be infinite (Reference 2).
- Credit is taken for containment, collection, and elevated release of the activity escaping the fuel pool. No credit is needed (or taken) for SGTS filters.

The core inventories at 24 hours after shutdown were calculated by the RADDECAY Code (Reference 27). The gap activity of noble gas and iodine (set at 99.85% elemental, 0.15% organic per Reference 2) was added from the core to the gap. The starting point of the calculation was the t = 0 shutdown inventories (Ci/MWt). The

RADDECAY calculation starts with time zero inventories for the noble gas and iodine isotopes. Given the activity (Ci or Ci/MWt) of an isotope at time zero, RADDECAY calculates the curies or Ci/MWt at any subsequent time of that isotope and its daughters. To obtain the total curies of the isotope of interest one must add the curies resulting from its direct decay plus the curies resulting from decay in chains in which it is a daughter product. This adjustment is made to the isotopes of interest and summarized in Table 2-14.

The on-site and off-site χ/Qs are provided in Table 2-5.

All fuel types currently stored in the fuel pools are bounded by this analysis. The key inputs used in this analysis are included in Table 2-13.

2.3.1.4 Control Rod Drop Accident Inputs and Assumptions

The VYNPS AST analysis for the control rod drop accident considers the three scenarios that have been previously reviewed by the NRC (References 9 and 10). The condenser leakage (Case 1) analysis assumes manual isolation of the MSIV prior to any release of activity to the atmosphere via the Advanced Off Gas (AOG) system. As a result, the activity released from the damaged fuel that reaches the turbine and the condenser is retained within these systems and the AOG lines. Retention by the AOG charcoal beds is neglected.

The AOG release analysis (Case 2) assumes the MSIVs remain open after the CRDA and the AOG remains operational. All releases to the environment in this case are via the AOG and stack and include only krypton and xenon noble gases.

The RCS recirculation sampling line (Case 3) analysis assumes the sampling lines remain open for 30 days after the CRDA with a constant leak rate of 32 gph. Release pathways for Cases 1 and 2 are mutually exclusive, while release pathway for Case 3 is additive to both pathways 1 and 2. The χ/Q values used for the analysis are summarized in Table 2-6. The RG 1.183 minimum core radial peaking factor used was 1.5.

VY is a BPWS plant and the GESTAR generic CRDA analysis demonstrates the accident does not result in fuel melting for BPWS plants (References 31 and 32). The control rod drop accident analysis is fully document in Reference 36.

2.3.2 Suppression Pool pH Control

NUREG-1465 notes that SRP 6.5.5, (Reference 25) allows credit for fission product scrubbing in the suppression pool. Although fission product removal by suppression pool scrubbing is not credited in the VYNPS analyses, removal by Drywell sprays is credited; and this will lead to a large fraction of activity being deposited in the pool water. The pool water will also retain soluble gaseous and soluble fission products such as iodides and cesium, but not noble gases. Once deposited the iodine will

remain in solution as long as the suppression pool pH is maintained at or above 7.0. The pH analysis is fully document in Reference 37.

It is expected that the initial effects on post-accident suppression pool pH will come from rapid fission product transport and the formation of cesium compounds, which would result in increasing the suppression pool pH. However, cesium compounds are not credited in the long-term pH analyses and the determination of the final (30 day) pH value. As radiolytic production of nitric acid and hydrochloric acid proceeds and these acids are transported to the pool over the first days of the event, the pH would become more acidic.

Upon detection of high Drywell radiation associated with the postulated activity release, plant procedures will be revised as necessary to require manual initiation of SLC injection for a LOCA. The buffering effect of SLC injection within a few hours is sufficient to offset the effects of these acids that are transported to the pool and maintain suppression pool pH at or above 7.0.

The current design function of the SLC System is to provide a backup method, independent of control rods, to make the reactor subcritical over a full range of operating conditions. The system actuation requirements for reactivity control are explicitly addressed in the VYNPS Emergency Operating Procedures. The SLC system is designed as a safety related system as described in the UFSAR Section 3.8. The operability requirements are specified in the TS Section 3.4.

The SLC System will be credited for limiting radiological dose following LOCAs involving fuel damage in accordance with the AST analyses for suppression pool pH control. A core damage event large enough to release substantial quantities of fission products into the Drywell will result in high Drywell radiation alarms. The operational response procedures will be revised as necessary to include instructions to manually actuate the SLC System. The AST analysis provides for SLC System actuation a few hours following accident initiation and completion of injection of an adequate volume and content of sodium pentaborate within several hours, which will ensure the suppression pool pH remains at or above 7.0 for 30 days.

Initiation of the SLC system for a loss of core cooling is not a new operator action. Plant procedures presently provide instruction to initiate the SLC system as well as other sources water for emergency core cooling.

Initiation of the SLC system will be accomplished from the main control room with a simple keylock switch manipulation. This switch is located on the main control room console and actuation of this switch is the only action necessary to initiate injection of the sodium pentaborate into the reactor vessel. The new SLC System function to control suppression pool pH does not involve any change to the actions needed to be performed to initiate SLC system injection. Indication of proper SLC System operation is provided in the control room as described in UFSAR Section 3.8.

During this postulated event, plant operators will be responding to the event as directed by the plant procedures. Adequate time is available for SLC system initiation during these events. Immediate initiation of the SLC System is not vital since the analysis allows for a few hours before initiation. Operators are familiar with operation of the SLC system due to previous training for Anticipated Transients Without Scram (ATWS) events and loss of emergency core cooling capability.

With certain post LOCA conditions, existing VYNPS procedures direct the operations of systems to accomplish a total flood-up of the primary containment. This floodup uses the ultimate heat sink (UHS) (Connecticut River) as the preferential source of makeup water since it is the only safety related makeup water source capable of accomplishing flood-up. A review of the past five years of data from the recirculation water system concludes that the minimum river pH has been above a pH of 7.0 for most of the time reviewed. On very few occasions, the pH was less than 7.0, but at no times was the pH less than 6.8. Although the condensate storage tank (CST) is safety related and could be used as a makeup source, it does not have sufficient volume to flood containment without repeated refilling. Consequently, the addition of a large amount of water from the UHS to the suppression pool and containment inventory will not result in a pH below 7.0.

2.3.3 Main Steam Line Break Accident Puff Release Dispersion Factor

A new control room χ/Q value for an instantaneous ground level puff release to the atmosphere was determined for use in the main steam line break accident radiological dose analysis. The inputs used in the determination of the χ/Q value are provided in Section 2.3.1.2 and Table 2-17.

2.3.4 NUREG-0737 Evaluation

The inputs and assumptions utilized in the NUREG-0737 evaluation include the AST plant-specific fission products inventories and other applicable inputs as described in Section 2.3.1.

Table 2-1
Computer Codes Used in AST
Design Basis Radiological Analyses

Task	Computer Code	Version or Revision	Comments
Determination of χ/Q and deposition of nuclear power plant effluents during continuous, intermittent and accident conditions in open-terrain sites.	AEOLUS3	MOD 1	Framatome-ANP Computer Code [Developed by ENTECH Engineering, Inc.] Applies the guidance in Reg. Guide 1.145 also has the capability for routine and intermittent releases. See Note[1]
Determination of χ/Q's for on site receptors near building structures	ARCON96	1996	NUREG/CR - 6331, Rev. 1 May, 1997 NRC Sponsored
Point Kernel Integration code used for general purpose gamma shielding analysis.	MicroShield	5.03	Code used in nuclear radiological analyses. Developed by Grove Engineering. Used in safety-related applications by many nuclear plants in the U.S.
Used to calculate fission product inventories Used to develop photon spectrum for DW Monitor Response	ORIGEN	ORIGEN2	The code is referenced in RG 1.183 and consistent with NRC recommendation. ORNL/TM-7175
Used to perform radioactive decay of the source term	RADDECAY	Version 3	Developed by Grove Engineering
Used for both on-site and off-site Dose Calculations	RADTRAD	3.02a	Referenced by RG 1.183 NUREG/CR-6604 USNRC April 1998

[1] Results were reviewed in Safety Evaluation for VYNPS Amendment No. 212 (References 9 & 10).

Table 2-1
Computer Codes Used in AST
Design Basis Radiological Analyses

Task	Computer Code	Version or Revision	Comments
Determination of χ/Q for Potential Accident Consequence Assessments at Nuclear Power Plants	SKIRON-II	Version 1	Framatome-ANP Computer Code [Developed by ENTECH Engineering, Inc.] Implements Reg. Guide 1.145, with one difference: it uses the sliding-window approach to obtain averages greater than 1 hour. See Note[1]
Used to perform independent check of dose calculations.	STARDOSE	03/01/1997	Polestar Applied Technology code (Reference 8)
Used to evaluate Suppression Pool Water pH as a function of time	STARpH	1.04	Utilized in other AST Submittals & Developed by Polestar. NRC reviewed and approved for use of STARpH for Hope Creek. (Reference 19)
Used to perform an independent check of MicroShield.	QADMOD	Version 5.03	Point Kernel Gamma-Ray Shielding Code with Geometric Progression Building Factors

[1] Results were reviewed in Safety Evaluation for VYNPS Amendment No. 212 (References 9 & 10).

Table 2-2
Bounding Fission Product Inventory

Isotope	Ci/MWt t = 0		Isotope	Ci/MWt t = 0
Kr83M	4.24E+03		Te132	3.97E+04
Kr85	5.05E+02		I132	4.05E+04
Kr85M	9.71E+03		I133	5.79E+04
Rb86	1.28E+02		Xe133	5.78E+04
Kr87	1.94E+04		Xe133M	1.76E+03
Kr88	2.75E+04		I134	6.43E+04
Kr89	3.46E+04		Cs134	1.52E+04
Sr89	3.45E+04		I135	5.39E+04
Sr90	4.10E+03		Xe135	2.33.E+04
Y90	4.29E+03		Xe135M	1.14E+04
Sr91	4.45E+04		Cs136	3.90E+03
Y91	4.24E+04		Xe137	5.07E+04
Sr92	4.61E+04		Cs137	6.08E+03
Y92	4.62.E+04		Ba137M	5.76E+03
Y93	5.05E+04		Xe138	5.05E+04
Zr95	4.95E+04		Ba139	5.35E+04
Nb95	4.96E+04		Ba140	5.15E+04
Zr97	4.92E+04		La140	5.17E+04
Mo99	5.30E+04		La141	4.91E+04
Tc99M	4.64E+04		Ce141	4.75E+04
Ru103	5.07E+04		La142	4.81E+04
Ru105	4.02E+04		Ce143	4.73E+04
RH105	3.68E+04		Pr143	4.71E+04
Ru106	2.85E+04		Ce144	3.73E+04
Sb127	3.69E+03		Nd147	1.92E+04
Te127	3.67E+03		Np239	7.67E+05
Te127M	4.98E+02		Pu238	3.93E+02
Sb129	1.01E+04		Pu239	1.47E+01
Te129	9.98E+03		Pu240	3.11E+01
Te129M	1.48E+03		Pu241	6.57E+03
Te131M	4.31E+03		Am241	8.73E+00
I131	2.85E+04		Cm242	3.42E+03
Xe131M	3.18E+02		Cm244	1.21E+03

Table 2-3
χ/Q Values for Radiological Dose Calculations - LOCA
(sec/m^3)

Release Location	Release Timing						
	0-0.5 hr	0.5-1 hr	1-2 hrs	2-8 hrs	8-24 hrs	1-4 days	4-30 days
EAB							
Ground [1] MSIV	1.70E-03			—NA—	—NA—	—NA—	—NA—
Ground RB Bypass [2]	1.476E-03			—NA—	—NA—	—NA—	—NA—
Ground RB Siding [2]	1.476E-03			—NA—	—NA—	—NA—	—NA—
Stack Normal [3]	—	1.54E-04	9.17E-05	4.04E-5	5.26E-6	—NA—	—NA—
Stack Fumig. [3]	2.03E-4	—	—	—	—	—NA—	—NA—
LPZ							
Ground MSIV [4]	2.74E-05		1.75E-05	8.01E-06	1.00E-06	5.80E-07	3.37E-7
Ground RB Bypass [5]	5.25E-5			2.23E-5	1.47E-5	5.95E-6	1.63E-6
Ground RB Siding [5]	5.25E-5			2.23E-5	1.47E-5	5.95E-6	1.63E-6
Stack [6]	2.55E-05		1.87E-05	1.01E-05	1.09E-06	6.90E-07	4.61E-07
CONTROL ROOM and TSC							
Ground RB Bypass [7]	2.25E-3			8.18E-4	3.53E-4	2.77E-4	2.23E-4
Ground RB Siding [8]	2.98E-3	—NA—	—NA—	—NA—	—NA—	—NA—	—NA—
Ground MSIV [7]	4.66E-3			3.46E-3	1.45E-3	1.09E-3	9.92E-4
Stack Normal [9]	—	1.92E-5	1.92E-05	8.28E-7	3.36E-7	3.08E-7	1.79E-07
Stack Fumig. [9]	1.92E-5	—	—	—	—	—	—

Table 2-3
χ/Q Values for Radiological Dose Calculations – LOCA

NOTES

[1] SKIRON-II. Max. 0-2 hr. Previously calculated and 0-2 hr NRC reviewed (Ref. 9 and 10).
[2] AEOLUS-3 generated. RB Siding χ/Q bounds the RB Bypass. RB Siding release is used for both.
[3] SKIRON-II. Stack release previously calculated. 0-2 hr values NRC reviewed (Ref. 9 and 10).
[4] SKIRON-II. Previously calculated and applied in CRDA. Documented in UFSAR (Ref.4).
[5] AEOLUS-3. In view of the 5 mile distance, the RB Bypass and Siding LPZ χ/Q are the same.
[6] SKIRON-II. Stack to LPZ have been previously reviewed (References 9 and 10).
[7] ARCON96. Based on RG 1.194 point source
[8] ARCON96. Based on RG 1.194 area source. Applicable only during drawdown time.
[9] ARCON96 and AEOLUS. Based on RG 1.194 for habitability assessments.
NA – Not Applicable

Table 2-4 χ/Q Values for Radiological Dose Calculations - MSLB (sec/m^3)			
Time Period	Control Room Puff	EAB [1]	LPZ
0 – 2 hrs	1.44E-03	1.70E-03	*

[1] SKIRON-II. Max. 0-2 hr. Previously calculated and 0-2 hr NRC reviewed (Ref. 9 and 10

* LPZ dose not necessary since release is limited to 2 hrs and EAB is more limiting

Table 2-5 χ/Q Values for Radiological Dose Calculations Refueling Accident (sec/m^3)			
Release Location	Release Timing		
	0-0.5 hr	0.5 – 1 hr	1 – 2 hrs
EAB			
Ground	1.70E-03		
Stack	2.03E-04	1.54E-06	9.17E-05
CONTROL ROOM			
Ground [1]	5.89E-03		
Stack	2.39E-04	1.05E-06	8.70E-07
LPZ *			

*LPZ dose not necessary since release is limited to two hours and EAB is more limiting

[1] RB siding facing F.A.I. treated conservatively as point source following RG 1.194

Table 2-6
χ/Q Values for Radiological Dose Calculations - CRDA
(sec/m^3)

Release Location	Release Timing						
	0-0.5 hr	0.5-1 hr	1-2 hrs	2-8 hrs	8-24 hrs	1-4 days	4-30 days
EAB							
Ground [1]	1.70E-03			–NA–	–NA–	–NA–	–NA–
Stack Normal [2]	—	1.54E-4	9.17E-5	–NA–	–NA–	–NA–	–NA–
Stack Fumig. [2]	2.03E-04	—	—	–NA–	–NA–	–NA–	–NA–
LPZ							
Ground [3]	2.74E-05		1.75E-05	8.01E-06	1.00E-06	5.80E-07	3.37E-7
Stack [4]	2.55E-05		1.87E-05	1.01E-05	1.09E-06	6.90E-07	4.61E-07
CONTROL ROOM							
Ground [5]	3.67E-03		2.19E-03	7.57E-04	3.93E-03	2.71E-04	2.04E-04
Stack Normal [6]	—	1.05E-06	8.70E-07	4.79E-7	2.34E-7	1.23E-7	6.90E-08
Stack Fumig. [6]	2.39E-04	—	—	—	—	—	—

[1] SKIRON-II. Max. 0-2 hr. Previously calculated and 0-2 hr NRC reviewed (Ref. 9 and 10).
[2] SKIRON-II. Stack release previously calculated. 0-2 hr values NRC reviewed (Ref. 9 and 10).
[3] SKIRON-II. Previously calculated and applied in CRDA. Documented in UFSAR (Ref.4).
[4] SKIRON-II. Previously calculated and applied in CRDA. NRC reviewed (Ref. 9 and 10).
[5] Murphy-Campe based. Previously calculated and applied in CRDA. NRC reviewed (Ref.9 and 10). This values have been preserved in order to assess the impact of AST on the current licensing basis results.
[6] Stack release. NRC reviewed. (Ref. 9 and 10)

Table 2-7
Fuel Data

Fuel Data	General Electric
Fuel Type	GE14
Initial Bundle Mass of Uranium (kg)	182.0
Initial Core Average Enrichment (U-235 wt%)	3.0 and 4.65 [(1)]
Core Average Bundle Power (MWt/bundle)	5.30
End of Cycle Core Wide Exposure (GWd/MT)	5 - 58.0 [(1)]

[1] Range considered in developing a composite bounding source term

Table 2-8
SGTS Functions Modeled in Dose Analyses

DBA Dose Analysis	Flow/ Secondary Containment	HEPA Particulate Removal	Charcoal Adsorber
LOCA	Y	Y	Y
Main Steam Line Break Accident	N[1]	N[1]	N[1]
Refueling Accident	N[2]	N[2]	N[2]
Control Rod Drop Accident	N[3]	N[3]	N[3]

[1] No release to secondary containment.
[2] No credit taken for holdup or filtering in secondary containment.
[3] CRDA Cases 1 & 3 take no credit for SGTS filtration, Cases 1 and 3 are ground release and Case 2 is an elevated releases via AOG without credit for charcoal beds.

Table 2-9
LOCA Release Fractions as Release Rates Over the Duration

Time Period (seconds)	Fraction of core inventory [1]	
0 - 120	No Release	
120 - 1920	Gases	Xe, Kr – 0.1/hr (0.05 total) Elemental I – 4.9E-3/hr (2.4E-3 total) Organic I – 1.5E-4/hr (7.5E-5 total)
	Aerosols	I, Br – 0.095/hr (0.0475 total) Cs, Rb – 0.1/hr (0.05 total)
1920 - 7320	Gases	Xe, Kr – 0.63/hr (0.95 total) Elemental I – 8.1E-3/hr (1.2E-2 total) Organic I – 2.5E-4/hr (3.8E-4 total)
	Aerosols	I, Br – 0.158/hr (0.2375 total) Cs, Rb – 0.133/hr (0.2 total) Te Group – 0.033/hr (0.05 total) Ba, Sr – 0.013/hr (0.02 total) Noble Metals – 1.7E-3/hr (2.5E-3 total) La Group – 1.3E-4/hr (2E-4 total) Ce Group – 3.3E-4/hr (5E-4 total)

[1] Release fractions and rates are from RG 1.183 Table I considering the chemical form described in RG 1.183, Section 3.5 (Reference 2).

Table 2-10
Accident Radiological Consequence Analyses Inputs

Input/Assumption	Value
CR Normal Mode Ventilation	> 9100 scfm
Fresh air intake	3700 scfm
Assumed CR unfiltered In-leakage Rate	3700 scfm
Reactor Building Drawdown Time	
LOCA – Control Room	10 minutes
LOCA – Technical Support Center	6 minutes
Control Room volume	41,534 ft^3
SGTS Flow Rate	≤ 1450 cfm
SGTS Filter Efficiency	
Particulate Iodine, Cesium and other Aerosols	95%
Elemental and Organic Iodine	95%
Noble Gases	0%
AOG Charcoal Delay Times (CRDA Case 2 Only)	Iodines – Infinite Kryptons – 24 hours Xenons – 16.6 days
Environment Breathing Rate (Regulatory Guide 1.183)	0-8 hours: 3.5E-04 m^3/sec 8-24 hours: 1.8E-04 m^3/sec 1-30 days: 2.3E-04 m^3/sec
Control Room Breathing Rate (Regulatory Guide 1.183)	3.5E-04 m^3/sec
Control Room Occupancy Factors (Regulatory Guide 1.183)	0-1 day: 1.0 1-4 days: 0.6 4-30 days: 0.4

Table 2-11
LOCA Inputs

Input/Assumption	Value
Fission Products Release Fractions (Regulatory Guide 1.183 Table 1)	**BWR Core Inventory Fraction Released Into Containment** Group / Gap Release Phase / Early In-vessel Phase / Total Noble Gases: 0.05 / 0.95 / 1.0 Halogens: 0.05 / 0.25 / 0.3 Alkali Metals: 0.05 / 0.20 / 0.25 Tellurium Metals: 0.00 / 0.05 / 0.05 Ba, Sr: 0.00 / 0.02 / 0.02 Noble Metals: 0.00 / 0.0025 / 0.0025 Cerium Group: 0.00 / 0.0005 / 0.0005 Lanthanides: 0.00 / 0.0002 / 0.0002
Fission Product Release Timing (Regulatory Guide 1.183 Table 4)	**LOCA Release Phases BWR** Phase / Onset / Duration Gap release: 2 min / 0.5 hr Early In-Vessel: 0.5 hr / 1.5 hr
Fission Product Iodine Chemical Form (Regulatory Guide 1.183, App. A)	Particulate 95% Elemental 4.85% Organic 0.15%
Control Room Isolation	None Assumed
ECCS Leakage Release Fractions	Ten percent of the radioiodine in the leaked coolant is assumed to become airborne in the reactor building (secondary containment).
Flow Rates	
Primary Containment Leak Rate (30 days)	0.8% containment air weight/day
Secondary Containment Bypass Leak Rate (30 Days)	5 scfh beginning at t=0 hours
Assumed ECCS Leak Rate (30 days)	0.5 gpm analyzed as 1.0 gpm
ECCS Leakage Temperature	<212°F

Table 2-11
LOCA Inputs

Input/Assumption	Value
MSIV Leak Rate at test pressure of 24 psig	124 scfh total 62 scfh maximum for one line
RB (SC) Bypass Leakage rate	5 scfh
MSIV Leakage that Bypasses Main Condenser	0.8% (percentage of total MSIV leakage)
Volumes	
Drywell Airspace	128,370 ft^3 (Min value used for dose calculation)
Torus Airspace	103,932 ft^3 (Minimum)
Suppression Pool	68,000 ft^3 (Minimum)
RB (SC) Free Volume	1,786,000 ft^3 (No dilution or hold-up credit taken for this volume)
High Pressure Turbine	(No credit taken)
Low Pressure Turbine	35,000 ft^3
Condenser Volume	72,000 ft^3
Removal Inputs	
Drywell Sprays Flow Rate	6650 gpm
Drywell Accident Conditions (Max Pressure bounds DBA LOCA and temperature bounds small steam line break. Both are from EPU containment response analysis.	P = 44 psig, T = 338°F
Steam Line and Main Condenser Removal Efficiencies:	
Condenser Volume	107,000 ft^3
Condenser Settling Area	4078 ft^2 (4% of projected tube area)
Steam Line Conditions	Saturated Conditions at 1050 psia
Steam Line Volume: Inboard to Outboard MSIV	26 ft^3
Steam Line Volume: Outboard	263 ft^3

Table 2-11 LOCA Inputs		
Input/Assumption	**Value**	
MSIV to Stop Valves (per line)		
MC Sedimentation Height	26.2 ft	
	Removal Efficiency for Aerosol Particles	Removal Efficiency for Elemental Iodine
Steam Line Leakage (62 scfh/line) (Between MSIV)	38%	Assumed Negligible
As above, with one MSIV failed	0%	0%
ALT Volume (Remainder of steam lines to turbine stop valves)	71%	58%
As above, with one MSIV failed	Assumed no change	Assumed no change
Combined Steam Lines and ALT	82%	58%
As above, with one MSIV failed	77%	Assumed no change
Main Condenser	95.1%	99.8%

Table 2-12
Main Steam Line Break Accident Inputs

Input/Assumption	Value
Mass Release	21,798 lbm steam 37,702 lbm water (saturated @ 1045psia)
MSIV Isolation Time	6.8 seconds
DE I-131 Equilibrium Value	1.1 µCi/gm
DE-I-131 Pre-Accident Spike	4 µCi/gm

Table 2-13
Refueling Accident Inputs

Input/Assumption	Value	
Number of Failed Rods (Equivalent Assemblies)	193 (2.1)	
Radial Peaking Factor	1.65	
Fuel Decay Period	24 hrs	
Pool Water Iodine Decontamination Factor Iodine	200	
Release Period	Instantaneous	
Release Location	Stack (No credit for holdup or SGTS operation)	
Release Fractions	Noble Gases	
	excluding Kr-85	5 percent
	Kr-85	10 percent
	I-131	8 percent
	Iodines except I-131	5 percent

Table 2-14
Fission Product Inventory
(Refueling Accident)

Isotope	Ci/MWt t = 0	Ci/MWt Adjusted [1]	Ci/MWt t = 24 hr
Br-83	4.24E+03	*	*
Kr-83M	4.24E+03	same	15.6
Br-85	9.61E+03	*	*
Kr-85M	9.71E+03	same	239
Kr-85	1.28E+01	1.01E+03	1010
Kr-87	1.94E+04	same	0.038
Kr-88	2.75E+04	same	72.3
Kr-89	3.46E+04	same	negligible
Te-131M	4.31E+03	*	*
I-131	2.85E+04	4.56E+04	42105
Xe-131M	3.18E+02	same	327
Te-132	3.97E+04	*	*
I-132	4.05E+04	same	33065
Te-133M	2.30E+04	*	*
Te-133	3.39E+04	*	*
I-133	5.79E+04	same	26656
Xe-133M	1.76E+03	same	1594
Xe-133	5.78E+04	same	55528
Te-134	5.31E+04	*	*
I-134	6.43E+04	same	negligible
I-135	5.39E+04	same	4351
Xe-135M	1.14E+04	same	negligible
Xe-135	2.33E+04	same	15285
Xe-137	5.07E+04	same	negligible
Xe-138	5.05E+04	same	negligible

[1] Adjusted for direct decay and decay chains in which the radionuclide is a daughter product

* Considered as parent only

Table 2-15
Control Rod Drop Accident Inputs

Input/Assumption	Value
Number of Failed Rods	850
Percent Fuel Melt for Failed Rods	No Melting
Radial Peaking Factor	1.50
Release Period (Case 1)	24 hours
Condenser/LP Turbine Leakage rate (Case 1)	1%/day for 24 hours
AOG Charcoal Delay Times (Case 2)	Iodines – Infinite Kryptons 24 hours Xenons 16.6 days
RCS Sampling Line Flowrate (Case 3)	32 gph
Coolant Mass Assumed to Mix with Iodine (Case 3)	393,187 lbm
Gap Release Fractions	Noble Gas 10% Iodine 10%% Cs, Rb 12%
Activity that reaches the condenser (Case 1)	Noble Gas 100% Iodine 10% Cs, Rb 1%
Activity released from the condenser (Case 1)	Noble Gas 100% Iodine 10% Cs, Rb 1%

Table 2-16
Suppression Pool pH Control Inputs

Input/Assumption	Value
Maximum Suppression Pool (SP) Liquid Volume (Maximum/Minimum)	68,000 ft^3 / 70,000 ft^3
Reactor Coolant System Inventory Excluding SP (Maximum/Minimum)	397,989 lbm / 559,828 lbm
Drywell Volume	128,370 ft^3
Suppression Pool Air Space Volume	103,932 ft^3
Sodium Pentaborate ($Na_2O*5B_2O_3*10H_2O$) Mass	600 lbm
Initial Suppression Pool Conductivity for Initial pH	5 μmho/cm
Length of PVC Jacketed Cable in the Drywell	25,000 ft
Length of Hypalon Jacketed Cable in the Drywell	6,000 ft
Average Cable Outside Diameter	1.0 inches
Average Cable Jacket Thickness	0.080 inch
Percent of Drywell Cable in Conduit	100%

Table 2-17
Main Steam Line Break Accident Puff Release χ/Q Inputs

Input/Assumption	Value
Mass Release	21,798 lbm steam 37,702 lbm water (saturated @ 1045psia)
Bubble Geometry	Hemispherical
Turbine Building Bubble Transverse Time to CR Fresh Air Intake (1 m/s wind speed)	51.7 seconds

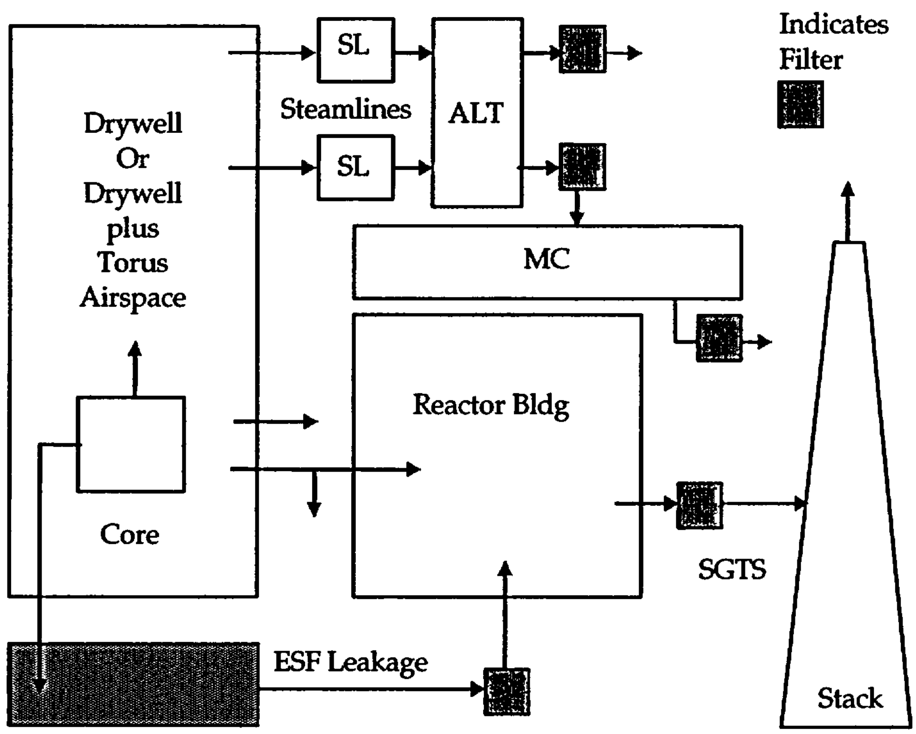

Figure 2-1
VYNPS RADTRAD Modeling

3. RESULTS

3.1 Evaluation Results

3.1.1 Accident Radiological Consequence Analyses

The postulated accident radiological consequence analyses were updated for AST implementation impact. Comparison of updated AST doses to existing licensing basis doses considers impact from the assumed operation at Extended Power Uprate conditions (1950 MWt (102% of 1912 MWt)) as well as the change in analysis methodology.

3.1.1.1 LOCA

Control Room, EAB and LPZ

The radiological consequences of the DBA LOCA were analyzed using the RADTRAD code and verified with STARDOSE with the inputs/assumptions defined in Section 2.3.1.1 of this report (Reference 33). The post accident doses are the result of the following activity considerations:

1. Primary to secondary containment leakage. This leakage is directly released into secondary containment and filtered by SGTS System prior to elevated release through the plant stack. The leakage to the reactor building during the draw-down period is assumed to be directly released to the environment.

2. ECCS leakage into the secondary containment. This leakage is directly released into the secondary containment environment and the airborne portion is filtered by SGTS System prior to elevated release through the plant stack. The leakage to the reactor building during the draw-down period is assumed to be directly released to the environment.

3. MSIV leakage from the primary containment into the main condenser (with a fraction that bypasses the main condenser directly to the atmosphere). Leakage passes through the ALT MSIV leakage pathway to the main condenser with credit for deposition before it is released to the environment.

4. Bypass leakage from the secondary containment through the nitrogen supply line. Released from the reactor building South wall.

5. Post-DBA LOCA radiation shine dose to personnel within the TSC from activity released to the reactor building. (Control Room contribution is negligible).

6. Secondary containment leakage is released directly to the environment and is not isolated for the duration of the event.

7. Evaluation of the worst single failure between a SGTS fan failing to start and an MSIV failing to close.

   The analysis in Reference 33 demonstrated the failure of one SGTS train maximizes the dose in comparison to one MSIV failing to close. This is due to the direct release of radionuclides to the environment during the drawdown time of 10 minutes. If the two SGTS trains are in operation, there would be no time period at the start of the accident during which the secondary containment can become pressurized relative to the outside environment. Accordingly, one train operation of SGTS is the most limiting case. As a result, all eight MSIVs are assumed close in the limiting analysis.

The EAB, LPZ, control room and TSC calculated doses are within the regulatory limits. Table 3-1 presents the results of the LOCA radiological consequence analysis.

Technical Support Center

The Technical Support Center (TSC) 30-day dose analysis is based on the analysis of the Control Room 30-day dose. The assessment of TSC habitability differs from the assessment of CR habitability in the area of external radiation effects. The TSC is not heavily shielded in the same manner as the CR. Therefore, some conservatisms in the RB drawdown bypass modeling and in the nitrogen supply RB bypass modeling that were included for the CR were relaxed for the TSC. The TSC shine calculation includes shine from activity within the RB and contribution from the external cloud. The contribution from the external cloud is determined to be 15 times the whole body dose from sources internal to the CR.

Reference 40 documents the comprehensive NUREG 0737 evaluation. The TSC 30 day results are summarized in Table 3-5.

3.1.1.2 Main Steam Line Break Accident.

(Reference 34)

The Main Steam Line Break accident EAB, LPZ and control room calculated doses are within the regulatory limits for the cases analyzed. The control room dose was determined using the new χ/Q values. Table 3-2 presents the results of the main steam line break accident radiological consequence analysis.

3.1.1.3 Refueling Accident

(Reference 35)

The radiological consequences of the design basis refueling accident were analyzed using the simplified and conservative assumptions described in Section 2.3.1.3. A spreadsheet calculation was carried out to obtain the results for two

cases (0% and 20% ground release for 24 hours decay). The spreadsheet results were verified with STARDOSE code. The dose agreement between the spreadsheet and STARDOSE was excellent for the four cases. The results presented in safety assessment are for the 24 hour elevated release.

The spreadsheet results for 24 hour decay before the fuel handling accident are shown in Tables 3-3. Control room and EAB doses are explicitly calculated; and because the release occurs within two hours, the EAB doses are bounding for the LPZ.

3.1.1.4 Control Rod Drop Accident

(Reference 36)

The radiological consequences of the design basis control rod drop accident were analyzed using the RADTRAD code and the inputs / assumptions defined in Section 2.3.1.4 of this report. The RADTRAD results were verified with the STARDOSE code. The EAB, LPZ, control room and calculated doses are within the regulatory limits. Table 3-4 presents the results of the control rod drop accident analysis.

3.1.2 Suppression Pool pH Control

The re-evolution of elemental iodine from the suppression pool is strongly dependent on suppression pool pH. The analysis assumed that sodium pentaborate was injected via SLC within a few hours of core damage. The modeling of the primary containment cabling results in the production of a large amount of hydrochloric acid. The minimum suppression pool pH at 30 days post-LOCA remains above 7.0, which satisfies the conditions for inhibiting the release of iodine in the elemental form from the suppression pool water. The analysis is fully described in Reference 37.

The quantity of SLC calculated as necessary to meet AST requirements is within the current TS requirements.

3.1.3 Main Steam Line Break Accident Instantaneous Ground Level Puff Release Dispersion Factor

(Reference 34)

The new control room χ/Q value for an instantaneous ground level puff release to the atmosphere was calculated for use in the main steam line break accident radiological dose analysis. The χ/Q value is shown in Table 2-4.

3.1.4 NUREG-0737 Evaluation

The results of the NUREG-0737 evaluation are summarized below.

- Post-Accident Vital Area Access and Sampling - The results of the revision of post-accident mission doses demonstrate that the current calculated doses (based on TID-14844 source terms) bound the doses that would be calculated based on AST source terms. The evaluated mission doses for VYNPS remain less than 5 rem TEDE.
- Post-Accident Radiation Monitor - The containment high range radiation monitors used to monitor post-accident primary containment radiation levels were evaluated for the impact of AST. The monitors continue to provide their design function and envelop the projected radiation exposure rates.
- Control Room and TSC Radiation Protection - The resultant doses to the control room for each of the four DBAs analyzed for AST have been determined. The results of these analyses are presented in Section 3.1.1.
- Radioactive Sources Outside the Primary Containment - The contribution of radiological dose consequences as a result of piping shine and ECCS leakage was determined as part of the radiological dose analysis for the LOCA. The results of this analysis are presented in Section 3.1.1.1.

3.2 Summary

Implementation of the AST as the plant radiological consequence analyses licensing basis requires a license amendment pursuant to the requirements of 10 CFR 50.67. Radiological dose analyses were performed for the four DBAs with a potential for offsite/control room dose. Doses calculated with the AST for accidents involving damaged fuel reflect delayed and/or reduced activity releases (relative to those of TID-14844 and RG 1.3) to the primary containment, reactor building, and or/or steam lines, as applicable. Offsite and control room doses remain within regulatory requirements.

Table 3-1
LOCA Radiological Consequence Analysis
(rem TEDE)

Dose Component	**Offsite Dose**		**Control Room Dose**
	EAB	**LPZ**	
SGTS Single Failure Case			
Direct Primary Containment Leakage [1]	1.8	0.08	2.8
Release Via RB and Plant Stack	1.3	0.44	0.036
Release Via Main Steam Lines and MC	0.035	0.0016	0.53
TOTAL SGTS Failure	**3.14**	**0.52**	**3.40**
MSIV Single Failure			
Direct Primary Containment Leakage [1]	1.1	0.053	1.4
Release Via RB and Plant Stack	1.3	0.44	0.036
Release Via Main Steam Lines and MC	0.039	0.0017	0.56
TOTAL MSIV Failure	**2.44**	**0.49**	**2.00**
Regulatory Limit	25	25	5
Current Analysis (Regulatory Limit) -[2] rem	4.30E-01 (25) Gamma 9.4E+01 (300) Thyroid	2.80E-01 (25) Gamma 8.4E+00 (300) Thyroid	3.0E-03 (5) Gamma 2.02E+01 (30) Thyroid

1 Primary leakage direct to the environment includes the reactor building bypass and reactor building siding pathways.

2 Current analysis two hour doses were evaluated at the maximum off site distance of 1900 meters due to the topographical considerations since there is no effective stack height at this distance. Thirty day doses at 8050 meters. (Reference 4, Table 14.9.4)

Table 3-2
Main Steam Line Break Accident Radiological Consequence Analysis
(rem TEDE)

Case	Offsite Dose		Control Room Dose
	EAB	LPZ	(Puff Release)
1.1 μCi/gm DE I-131	0.981	< 0.981	0.55
4.0 μCi/gm DE I-131	3.57	< 3.57	2.00
Regulatory Limit	2.5 / 25	2.5 / 25	5
Current Analysis (Regulatory Limit) [1] - rem		1.2E-02 (25) Gamma 1.4E+01 (300) Thyroid	

[1] Current analysis two hour doses were evaluated at the maximum off site distance of 1900 meters due to the topographical considerations since there is no effective stack height at this distance. Thirty day doses at 8050 meters. (Reference 4, Table 14.9.4)

Table 3-3
Refueling Accident Radiological Consequence Analysis
(rem TEDE)

Case	Offsite Dose		Control Room Dose
	EAB	LPZ	
24 Hours after shutdown Elevated Release	0.47194	< 0.47194	0.15305
Regulatory Limit	6.3	6.3	5
Current Analysis (Regulatory Limit) [1] - rem	2.7E-02 (25) Gamma 3.2E+01 (300) Thyroid	8.4E-04 (25) Gamma 3.4E+00 (300) Thyroid	

[1] Current analysis two hour doses were evaluated at the maximum off site distance of 1900 meters due to the topographical considerations since there is no effective stack height at this distance. Thirty day doses at 8050 meters. (Reference 4, Table 14.9.4)

Table 3-4 Control Rod Drop Accident Radiological Consequence Analysis (rem TEDE)			
Case	Offsite Dose		Control Room Dose
	EAB	LPZ	
Case 1	2.7E-01	6.1E-03	3.5E-01
Case 2	1.7E-01	2.1E-02	1.3E-03
Case 3	1.1E-01	6.0E-02	4.8E-02
Case 1 + Case 3	3.8E-01	6.6E-02	4.0E-01
Case 2 + Case 3	2.8E-01	8.1E-02	4.9E-02
Regulatory Limit	6.3	6.3	5
Current Analysis (Regulatory Limit) [1]- rem	1.5E-02 (25) Gamma 2.3E-02 (300) Beta 3.0E+00 (300) Thyroid	7.4E-03 (25) Gamma 1.2E-02 (300) Beta 1.8E+00 (300) Thyroid	9.7E-03 (5) Gamma 3.7E-01 (30) Beta 28 (30) Thyroid

[1] The current analysis values provided for the control room correspond to Case 1 + Case 3 and for the EAB and LPZ to Case 3 (References 9 and 10)

Table 3-5 Technical Support Center DBA LOCA Dose (rem TEDE)			
Case	Internal	External	Total
TSC	2.3	1.2	3.5
Regulatory Limit			5
Current Analysis (Regulatory Limit) [1]- rem			4.7 (5) Gamma 15 (30) Thyroid

[1] The current analysis values are for the limiting location in the TSD and does not include MSIV leakage per BYV 94-02

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29. BWROG, NEDC-31858P-A, Volume 1, "BWROG Report for Increasing MSIV Leakage Rate Limits and Elimination of Leakage Control Systems", August 1999.

30. VY Off Normal Procedure ON-3153, "Excessive Radiation Levels", Revision 12

31. NEDC-24011-P-A-14-U (GESTAR-II) Supplement for United States

32. NEDC-24011-P-A (GESTAR-II) Amendment 22

33. POLESTAR Calculation, PSAT-3019CF.QA.08, "Radiological Evaluation of a DBA-Loss of Coolant Accident"

34. POLESTAR Calculation, PSAT-3019CF.QA.06, "MSLB Dose for VY"

35. POLESTAR Calculation, PSAT-3019CF.QA.05, "FHA Dose for VY"

36. POLESTAR Calculation, PSAT-3019CF.QA.07, "Radiological Evaluation of a CRDA"

37. POLESTAR Proprietary Calculation, PSAT 3019CF.QA.04 "Suppression Pool pH"

38. VYC-2066, "Bounding Core Inventories of Actinides and Fission Products for Design-Basis Applications at 1950 MWt"

39. VYC-2312, "VY Post-LOCA Drywell High Range Monitor Response for Core Damage Assessment at 1912 MWth".

40. POLESTAR Proprietary Calculation, PSAT 3019CF.QA.09, Alternative Source Term Impact on NUREG-0737 Equipment Qualification, Vital Area Access, and Areas of Continuous Occupancy.

Appendix A Alternative Leakage Treatment Isolated Condenser Strategy

Regulatory Guide 1.183, Appendix A, provides assumptions acceptable to the NRC for evaluation of radiological consequences of LOCA using AST. For BWR MSIV leakage, Regulatory Guide 1.183 allows credit for reduction in 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 function during and following a safe shutdown earthquake (SSE). Regulatory Guide, Appendix A states that an acceptable model for evaluating reduction of MSIV releases is provided in General Electric Topical Report NEDC-31858P-A, Revision 2, "BWROG Report for Increasing MSIV Leakage Limits and Elimination of leakage Control Systems" (Reference 29).

The NRC Safety Evaluation for NEDC-31858P-A, Revision 2, identified limitations to be addressed as part of a plant specific application of the Alternative Leakage Treatment (ALT) methodology. These limitations relate to assuring the ALT pathway for MSIV leakage is functionally reliable commensurate with its intended safety function and assuring the pathway, including the main condenser, is seismically rugged.

Vermont Yankee has identified the ALT drain paths and seismic isolation boundaries required to process the MSIV leakage for the Alternative Source Term (AST) analysis using the isolated main condenser strategy. Figure A-1 shows the ALT pathways that include the main condenser and the piping connected to the main steam lines between the (MSIVs) and the turbine stop valves along with associated valves when the plant is at 100% power. Figure A-2 is the same as Figure A-1 with the valves in the Post Accident drain pathway lineup. It should be noted that the normal position of several of the valves (e.g. PRV-OG-834B, FCV-101-35 and 36A) may be different from what is shown in Figure A-1. This is due to the required position at various power levels or operating philosophy. However, this has no effect on the required failure position.

The ALT boundary includes a primary ALT drain path, a backup ALT drain path, and an additional ALT drain path that are readily available to provide additional redundancy if they are required.

The ALT drain paths and the active boundary end points require valves that are reliable and the valves will be included in the IST Program. These valves can be operated from the Control Room and fail to the required position on loss of power or process air. Local operation of equipment in radiological areas is not necessary following an accident. As a result, there are no new access requirements to vital areas created as a result of implementing the ALT strategy.

A primary and backup ALT drain path is required to assure controlled leakage considering a failure. Valves required to open must have high reliability. The reliability is assured by having power from a reliable source or failure to the required position on loss of power or air along with the ability to operate the valve from the Control Room. In addition to being ALT drain paths, the paths are part of the seismic boundary.

Air operated valves LCVs-101-38A, B, C and D, LCV-2-143, and LCV-101-39 fail open on loss of air or power. Failure of an EDG does not compromise the ability of these valves to open.

The primary ALT drain path is via the MS low point drain valves, LCVs-101-38A,B,C and D which as previously stated are air operated valves that fail open on loss of air or power. Any of the 4 valves provides adequate drainage. These valves are operated at CRP 9-23.

The backup ALT drain path is via the MS low point drain air operated valve located just downstream of the MSIVs, LCV-2-143, that fails open on loss of air or power. This valve is located downstream of normally open manual valve V60-24 which serves as an orifice. LCV-2-143 is operated at CRP 9-23.

A third ALT drain path is via the SJAE supply line low point drain air operated valve, LCV-101-39 that fails open on loss of air or power. This valve is operated at CRP 9-23.

Since the radiological analysis accounts for leakage through the turbine stop valves (0.5% of total MSIV leakage), it is not necessary to meet the flow area fraction ratio described in Section 4.0 to Appendix C of NEDC-31858P Rev. 2.

ALT Seismic Boundary

The ALT seismic boundary includes the main condenser and all piping and tubing located off the MS lines between the MSIVs and the turbine stop valves which could result in steam leakage. In addition to the above the following leakage paths are within the ALT seismic boundary:

AOG steam supply
MS sample line
Steam to turbine steam seal system
Steam to SJAEs
Steam to turbine bypass valves
Steam to EPR, MPR and miscellaneous instruments
Stop valve and stop valve drains

Valves required to close must have high reliability. High reliability infers power from a reliable source or failure to the required position on loss of power or air along with the ability to operate the valve in the required time frame from the control room. The turbine stop valves have high reliability and fail closed on turbine trip.

Air operated valves PRV-OG-834A&B, FCV-101-37 and PCV-101-35 and the MSIVs fail closed on loss of air or power. Failure of an EDG does not compromise the ability of these valves to close. These valves operate independently of each other. Failure of a MSIV does not cause failure of the other valves and vice versa.

The AOG boundary is at valves PRV-OG-834A & B, which are air operated valves arranged in parallel that fail closed on loss of air or power. These valves are operated at CRP 9-50.

The SJAE boundary is at valves FCV-101-37and PCV-101-35 which are air operated valves that fail closed on loss of air or power. These valves are operated at CRP 9-6.

The turbine steam seal system boundary is at MOV 60-6 and normally closed MOV 60-10 which is arranged in parallel. MOV 60-6 is closed by procedure at 70% power, thus at 100% power both sources of steam are already isolated. MOVs fail as is on loss of power. Therefore no modifications are required to isolate the steam seal regulator at power greater than 70%.

The sample lines, Electronic Pressure Regulator (EPR), Mechanical Pressure Regulator (MPR) and miscellaneous instruments do not require active isolation since they are closed systems and entirely within the seismic boundary.

The turbine bypass valves and the stop valve drains also do not require active isolation since the valves are normally closed valves that fail as is on loss of power.

A new ¾" check valve (OG-779) will be installed on the drain line downstream of the trap off the AOG steam supply line to the AOG building in line No. ¾"- MS-189-D3 near the condenser.

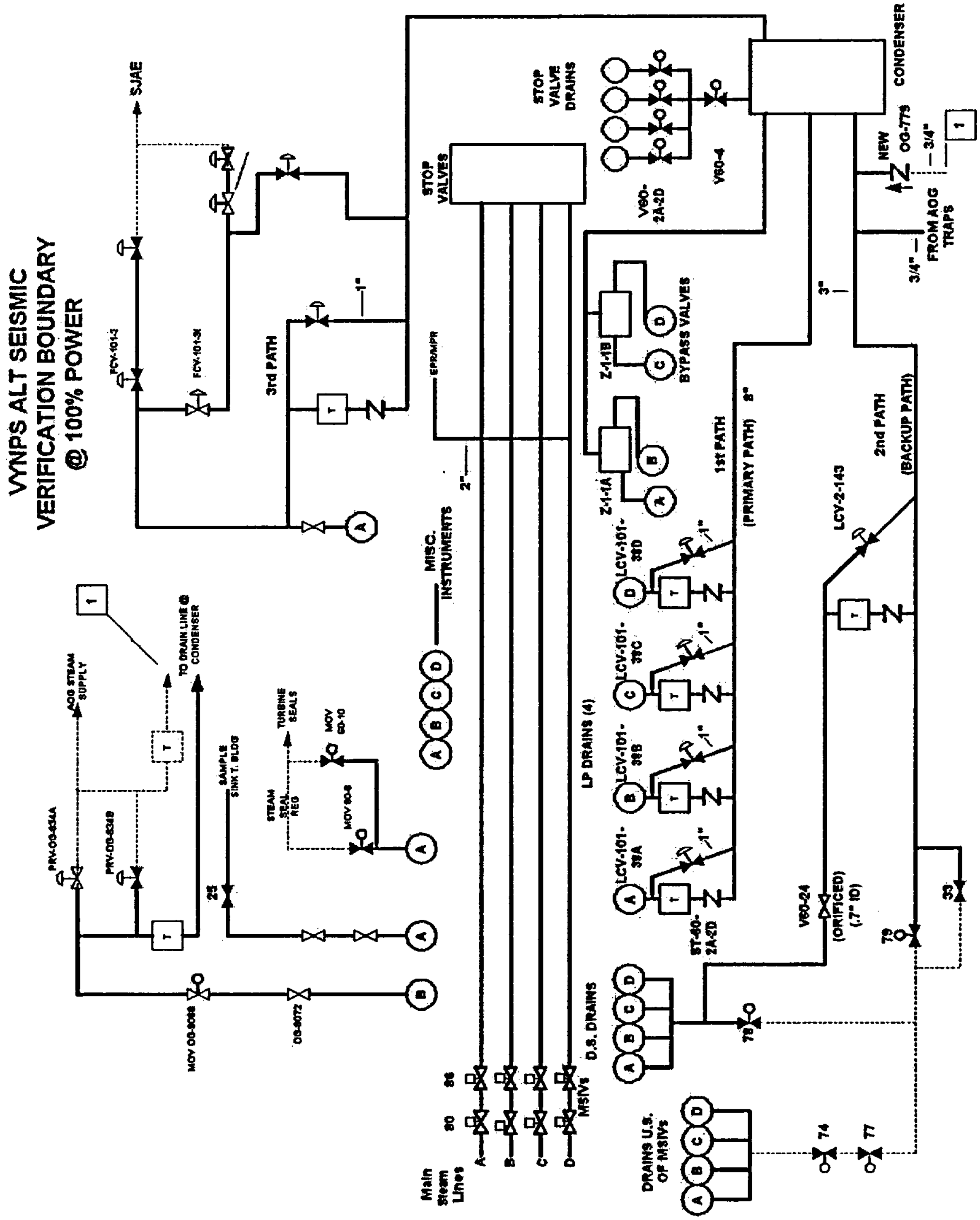
ALT Boundary Seismic Evaluation

A separate evaluation was performed to evaluate the seismic ruggedness of the Vermont Yankee ALT pathway (Reference 28). Seismic assessment of significant elements and boundary components was addressed with the determinations made in all cases that the applicable items satisfy the associated seismic ruggedness criteria. Walkdowns of all normally accessible areas during power operation have been performed. A walkthrough of normally inaccessible areas was performed which confirmed in general that the construction of piping, supports and components is similar to the piping which has been walked down to date.

The results of the walkdown of the accessible areas verified the seismic ruggedness of the piping and equipment. Walkdowns and any necessary follow-on analytical assessments for remaining ALT pathway and boundary piping, equipment and supports will be conducted during RFO 24. Any components and piping configuration identified during the scheduled walkdowns that may be associated with poor piping and/or component seismic performance will be identified as an outlier. All outliers will be evaluated and where necessary, modifications will be performed.

The work performed to date and to be completed during RFO24 is consistent with the recommendations of the BWROG approved topical report (Reference 29). Based on the partial results, there are no indications that the ALT pathways and boundary piping, equipment and supports will not satisfy necessary seismic criteria for VYNPS. Appropriate planning and measures to address potential outlier issues have been implemented. Based on previous ALT implementations, it is anticipated that a discrete number of passive (pipe support type) modifications may be required.

The seismic evaluation and any required modifications will be completed during Refueling Outage 24 (RFO 24) in April 2004.



FigureA-1
VYNPS ALT Seismic Verification Boundary
100% Power

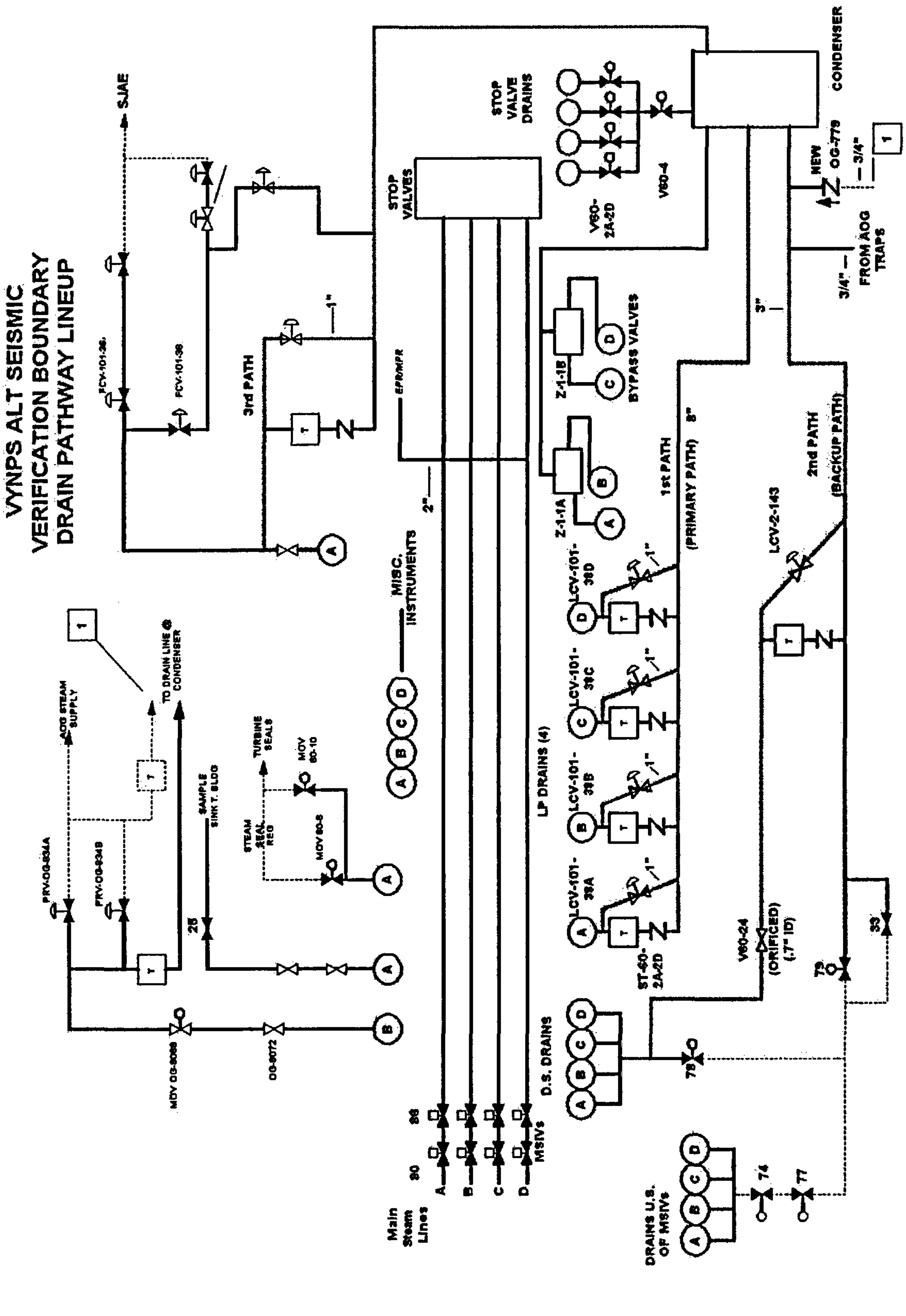


Figure A-2
VYNPS ALT Seismic Verification Boundary
Drain Pathway Analysis

Appendix B Reactor Building Positive Pressure Period

The Alternative Source Term LOCA analysis considers the reactor building positive pressure period. This is the period when a loss of off-site power causes a loss of reactor building negative pressure relative to the environment. The start of the EDG followed by the start of the SGTS returns the reactor building to sub-atmospheric conditions. The time of positive pressure relative to the atmospheric pressure is called the "drawdown" time. The primary containment leakage in the reactor building is assumed to be released directly to the environment during the drawdown period. A plant specific calculation was performed to determine the drawdown time. This is a new licensing basis analysis for Vermont Yankee.

The drawdown calculation was performed with the GOTHIC code. Two models were developed to evaluate reactor building pressurization following a LOCA, coincident with a Loss of Off-site Power (LOOP). The two scenarios differ in that one assumes two Diesel Generators start upon the LOOP, the other assumes that only one Diesel Generator starts upon LOOP. In both cases, only one train of the Standby Gas Treatment (SBGT) System is assumed to operate. This is assumed the worst single failure since having both trains available would not result in a positive pressure period. In addition, the transient is initialized with the reactor building at pressure corresponding to SGTS technical specification minimum pressure differential. This is conservative since the normal reactor differential pressure is about an order of magnitude greater.

Two Diesel Generator Scenario

The two diesels reach full speed and are loaded at 13 seconds and one train of the Standby Gas Treatment System, at 1450 cfm, is available at 15 seconds. The Reactor Building includes heat addition by the solar loads, post LOCA suppression pool temperature, Drywell temperature, Spent Fuel Pool (SFP) heat load, RB Corner Room ECCS heat loads, and additional RB equipment heat loads (including passive heat structures).

There is one core spray pump, one RHRSW Pump (on at 10 minutes), and 2 LPCI pumps (RHR Pumps) operating in each of the corner rooms. At 10 minutes, one RHR pump is assumed tripped. RRUS 7/8 are available in each corner room to remove heat. The heat removal capability of RRUs 5/6 is available but neglected; however, the fan heat is added to the corner room as a source of heat.

Additional heat loads, representing the loads added to diesel generators 1A and 1B, per VY UFSAR Section 8.5 were considered. In addition to these diesel generator loads, some other sources of heat are added inside the Reactor Building, at their respective elevations.

The results show that the reactor building pressure remains negative for 2.73 minutes, becomes positive for 5.59 minutes and returns negative in 8.32 minutes.

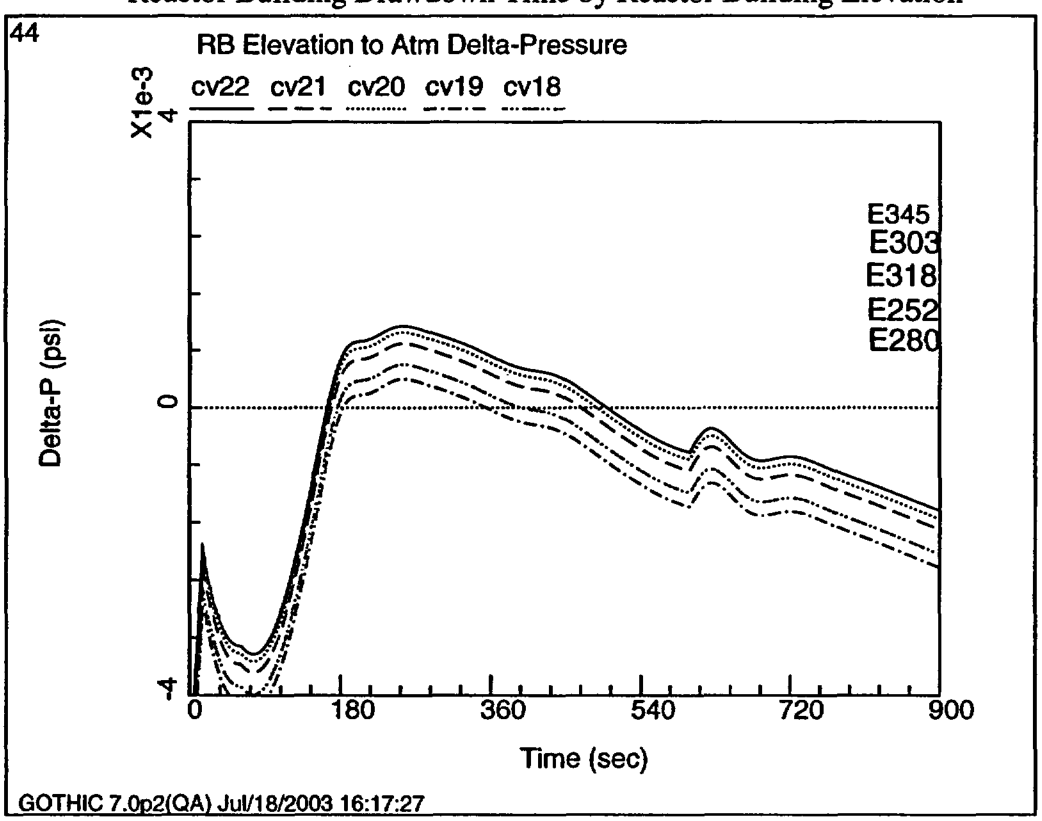
Single Diesel Generator Scenario

Only one diesel reaches full speed and is loaded at 13 seconds and the Standby Gas Treatment System, at 1450 cfm, is available at 15 seconds into the event. As in the first scenario, the Reactor Building includes heat addition by the same loads. The diesel that is assumed to fail is Diesel Generator 1B. This is a conservative assumption since the Reactor Building loads on the operating Diesel Generator 1A are higher and thus lead to longer drawdown times.

Heat addition from the corner rooms to the Reactor Building is calculated as part of this analysis (Section 5.3). In one Corner Room, one Core Spray Pump, one RHRSW Pump (on at 10 minutes), and one RHR pump are operating, with RRU 7/8 removing heat. The heat removal capability of RRUs 5/6 is neglected, however the fan heat is added to the corner room as a source of heat. In the other corner room, one RHR pump is operating with no active heat removal, except to the Reactor Building through the stairs and the corner room walls. Additional heat loads, representing the loads added by a single diesel generator.

The results show that the reactor building pressure remains negative for 2.78 minutes, becomes positive for 5.39 minutes and returns negative in 8.17 minutes.

Figure B-1
Reactor Building Drawdown Time by Reactor Building Elevation



Polestar Applied Technology, Inc.

AFFIDAVIT

I, David E.W. Leaver, being duly sworn, depose and state as follows:

(1) I am a Principal and an Officer of Polestar Applied Technology, Inc. ("Polestar") and am responsible for the function of reviewing the information described in paragraph (2) which is sought to be withheld, and have been authorized to apply for its withholding.

(2) The information sought to be withheld is contained in portions of Polestar reports PSAT 3019CF.QA.04, PSAT 3019CF.QA.08, PSAT 3019CF.QA.09. These reports are being prepared for Entergy Nuclear Operations, Inc. in support of an Entergy submittal to NRC on alternate source term (AST). The Polestar reports address post-accident sump pH, DBA-LOCA dose, and vital area access at the Vermont Yankee Nuclear Power Station.

(3) In making this application for withholding of proprietary information of which it is the owner, Polestar relies upon the exemption from disclosure set forth in the NRC regulations 10 CFR 9.17(a)(4), 2.790(a)(4), and 2.790(d)(1) for "trade secrets and commercial or financial information obtained from a person and privileged or confidential" (Exemption 2.790(a)(4)). The material for which exemption from disclosure is here sought is all "confidential commercial information".

(4) Some examples of categories of information which fit into the definition of proprietary information are:

a. Information that discloses a process or method, including supporting data and analyses, where prevention of its use by Polestar's competitors without license from Polestar constitutes a competitive economic advantage over other companies.

b. Information which, if used by a competitor, would significantly reduce his expenditure of resources or improve his competitive position in the analysis, design, assurance of quality, or licensing of a similar product;

c. Information which reveals cost or price information, production capacities, budget levels, or commercial strategies of Polestar, its customers, or its suppliers;

d. Information which reveals aspects of past, present, or future Polestar customer-funded development plans and programs, of potential commercial value to Polestar;

e. Information which discloses patentable subject matter for which it may be desirable to obtain patent protection.

The information sought to be withheld is considered to be proprietary for the reasons set forth in both paragraphs (4)a and (4)b, above.

(5) The information sought to be withheld is being submitted to Entergy (and, we trust, to NRC) in confidence. The information is of a sort customarily held in confidence by Polestar, and is in fact so held. The information sought to be withheld has, to the best of my knowledge and belief, consistently been held in confidence by Polestar, no public disclosure has been made, and it is not available in public sources. All disclosures to third parties including any required transmittals to NRC, have been made, or must be made, pursuant to regulatory provisions or proprietary agreements which provide for maintenance of the information in confidence. Its initial designation as proprietary information, and the subsequent steps taken to prevent its unauthorized disclosure, are as set forth in paragraphs (6) and (7) following.

(6) Initial approval of proprietary treatment of a document is made by the manager of the originating component, the person most likely to be acquainted with the value and sensitivity of the information in relation to industry knowledge. Distribution of such documents within Polestar is limited to those with a need to know.

(7) The approval of external release of such a document typically requires review by the project manager, and the Polestar Principal closest to the work, for technical content, competitive effect, and determination of the accuracy of the proprietary designation. Disclosures outside Polestar are limited to regulatory bodies, customers, and potential customers, and their agents, suppliers, and licensees, and others with a legitimate need for the information, and then only in accordance with appropriate regulatory provisions or proprietary agreements.

(8) The information identified in paragraph (2), above, is classified as proprietary because it contains detailed information on and results from methodologies developed by Polestar and applied under the Polestar 10 CFR 50, Appendix B Quality Assurance Program. The methodologies in PSAT 3019CF.QA.04 address acid generation due to radiolysis of water and cable in BWRs, buffer effect on pH, and Polestar's STARpH computer code for post-accident sump pH calculations. The methodologies in PSAT 3019CF.QA.08 address the design basis LOCA dose calculation including mechanistic fission product removal phenomena. The methodologies in PSAT 3019CF.QA.09 address the use of generic arguments and insights with respect to vital area access of AST vs. the

TID 14844 source term. This more detailed, mechanistic treatment of post-accident pH and fission product removal phenomena were not traditionally considered in USNRC licensing design basis calculations prior to AST, and thus new methods development was required. The work on vital area access represents new insights which were required to adapt TID 14844 approaches for the AST.

The methodologies used in this Vermont Yankee work are several of a number of Polestar developed methods, models, and codes. Development of these methods, models, and codes was achieved at a significant cost to Polestar, well over $100,000, which is a significant fraction of internal research and development resources available to a company the size of Polestar.

The development of the methods, models and codes, along with the interpretation and application of the results, is derived from the extensive experience database that constitutes a major Polestar asset.

(9) Public disclosure of the information sought to be withheld is likely to cause substantial harm to Polestar's competitive position and foreclose or reduce the availability of profit-making opportunities. The information is part of Polestar's comprehensive technology base on application of the AST to operating plants and advanced light water reactors, and its commercial value extends beyond the original development cost. The value of the technology base goes beyond the extensive physical database and analytical methodology and includes development of the expertise to determine and apply the appropriate evaluation process. In addition, the technology base includes the value derived from providing analyses done with methods which have been developed and are being maintained in accordance with 10 CFR 50, Appendix B requirements.

The research, development, engineering, analytical and review costs comprise a substantial investment of time and money by Polestar.

The precise value of the expertise to devise an evaluation process and apply the correct analytical methodology is difficult to quantify, but it clearly is substantial.

Polestar's competitive advantage will be lost if its competitors are able to use the results of the Polestar experience to normalize or verify their own process or if they are able to claim an equivalent understanding by demonstrating that they can arrive at the same or similar conclusions.

The value of this information to Polestar would be lost if the information were disclosed to the public. Making such information available to competitors without their having been required to undertake a similar expenditure of

resources would unfairly provide competitors with a windfall, and deprive Polestar of the opportunity to exercise its competitive advantage to seek an adequate return on its relatively large investment in developing these very valuable analytical tools.

STATE OF CALIFORNIA)
) ss:
COUNTY OF SANTA CLARA)

David E.W. Leaver, is being duly sworn, deposes and says:

That he has read the foregoing affidavit and the matters stated therein are true and correct to the best of his knowledge, information, and belief.

Executed at Los Altos, California, this 29th day of July 2003.

David E W Leaver

David E.W. Leaver
Polestar Applied Technology, Inc.

Subscribed and sworn before me this 29th day of July 2002.

RUTH TUBBS
Commission # 1234269
Notary Public - California
Santa Clara County
My Comm. Expires Oct 8, 2003

Ruth T Tubbs

Notary Public, State of California

Attachment 6

Vermont Yankee Nuclear Power Station

Proposed Technical Specification Change No. 262

DBA Calculations for Alternative Source Term

DESIGN BASIS ACCIDENT CALCULATIONS for ALTERNATIVE SOURCE TERM

Table of Contents