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BVY 13-097

November 14, 2013

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

SUBJECT: Technical Specifications Proposed Change No. 306
Eliminate Certain ESF Requirements during Movement of Irradiated Fuel
Vermont Yankee Nuclear Power Station
Docket No. 50-271
License No. DPR-28

REFERENCES:

1. Regulatory Guide 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors," July 2000
2. Letter, USNRC to Entergy Nuclear Operations, Inc., "Pilgrim Nuclear Power Station - Issuance of Amendment RE: Alternative Source Term for the Fuel Handling Accident Dose Consequences (TAC No. MC2705)," dated April 28, 2005

Dear Sir or Madam:

In accordance with 10CFR50.90, Entergy Nuclear Operations, Inc. (ENO) is proposing an amendment to Appendix A, Technical Specifications (TS), of the Vermont Yankee Nuclear Power Station (VY) Renewed Facility Operating License, DPR-28, to change the requirements associated with handling irradiated fuel and performing core alterations. Specifically, the changes would eliminate operability requirements for secondary containment when handling sufficiently decayed irradiated fuel or a fuel cask and while performing core alterations.

The changes proposed in this application are consistent with Technical Specification Task Force Traveler 51, “Revise Containment Requirements During Handling Irradiated Fuel and Core Alterations." In support of these changes, ENO has completed a calculation using a selective implementation of alternate source term guidance for evaluating the potential dose consequences of a fuel handling accident (FHA). This calculation uses the guidelines detailed in Regulatory Guide 1.183 (Reference 1). The calculation demonstrates that radiological doses at the exclusion area boundary, low population zone and in the control room from a FHA after a certain amount of time following shutdown are within allowable limits without crediting secondary containment operability and operation of the standby gas treatment system. The proposed changes are consistent with similar changes approved for Pilgrim Nuclear Station (Reference 2).

ENO has reviewed the proposed amendment in accordance with 10CFR50.92 and concludes it does not involve a significant hazards consideration.

In accordance with 10CFR50.91, a copy of this application, with attachments, is being provided to the State of Vermont, Department of Public Service.

Attachment 1 to this letter provides a detailed description and evaluation of the proposed change. Attachment 2 contains a markup of the current TS and Bases pages. Attachment 3 contains the retyped TS and Bases pages. Bases changes are provided for information only. Attachment 4 contains the new fuel handling accident calculation.

ENO requests NRC review and approval of the proposed changes by December 1, 2014 to support the anticipated fuel handling operations following final shutdown of the VY reactor, which is expected to occur in the fourth quarter of 2014. ENO requests a 60 day implementation period for the approved changes.

New regulatory commitments are described in Attachment 5 of this letter.

If you have any questions on this transmittal, please contact Mr. Coley Chappell at 802-451-3374.

I declare under penalty of perjury that the foregoing is true and correct.

Executed on November 14, 2013.

Sincerely,

for CJW

CJW/plc

Attachments:

1. Description and Evaluation of the Proposed Changes
2. Markup of the Current Technical Specifications and Bases Pages
3. Retyped Technical Specifications and Bases Pages
4. AREVA Document No. 32-9145461-001, "VYNPP - Re-analysis of AST/FHA Radiological Consequences with Open Containment"
5. List of Regulatory Commitments

cc: Mr. William M. Dean
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Attachment 1

Vermont Yankee Nuclear Power Station

Proposed Change 306

Description and Evaluation of Proposed Changes

1. SUMMARY DESCRIPTION

This evaluation supports a request to amend the Renewed Facility Operating License (OL) DPR-28 for Vermont Yankee Nuclear Power Station (VY).

The proposed change would revise the Technical Specifications (TS) contained in Appendix A of the VY OL to change the requirements associated with handling irradiated fuel, or a fuel cask, and while performing core alterations for selected specifications. Specifically, the changes would eliminate operability requirements for secondary containment (TS 3.7.C), the Standby Gas Treatment (SGT) system (TS 3.7.B.4), and reactor building ventilation isolation and SGT system initiation instrumentation (TS Table 3.2.3) when handling sufficiently decayed irradiated fuel, or a fuel cask, and while performing core alterations. The proposed changes would require subject systems to be operable only during movement of "recently" irradiated fuel assemblies in secondary containment and during operations with the potential to drain the reactor vessel (OPDRV). The period of sufficient radioactive decay was determined to be 13 days and will be defined in the TS Bases.

The proposed changes are consistent with the generic changes included in Technical Specification Task Force (TSTF) Traveler 51, "Revise Containment Requirements During Handling Irradiated Fuel and Core Alterations," (Reference 1) and those previously approved for the Pilgrim Nuclear Power Station (Reference 14).

A revised fuel handling accident (FHA) (or refueling accident) analysis has been developed to support the proposed changes using the Alternate Source Term (AST) methodology described in Regulatory Guide 1.183 (Reference 2). Attachment 4 provides the supporting calculation for the revised FHA analysis.

Entergy Nuclear Operations, Inc. (ENO) requests NRC review and approval of the proposed changes by December 1, 2014 to support the anticipated fuel handling operations following final shutdown of the VY reactor, which is expected to occur in the fourth quarter of 2014 (Reference 3).

2. DETAILED DESCRIPTION

The VY TS currently impose restrictions on plant operations when handling irradiated fuel assemblies and while performing core alterations. These restrictions require that certain structures, systems or components (SSCs) be operable. These restrictions assure that the radiological consequences of a fuel handling accident do not exceed those estimated in design-basis analyses.

TSTF-51 removes TS requirements for engineered safeguard features (ESF) (e.g., primary/secondary containment, standby gas treatment, isolation capability) to be operable after sufficient radioactive decay has occurred to ensure off-site doses remain below the NUREG-0800 Standard Review Plan limits. TSTF-51 also deletes operability requirements during core alterations for ESF mitigation features.

The proposed TS changes are discussed below (additions are shown in underline format and deletions are shown in strikethrough format):

a. TS Table 3.2.3, Reactor Building Ventilation Isolation and Standby Gas Treatment System Initiation Instrumentation

Footnotes applicable to the High Reactor Building Ventilation Radiation (Trip Function 3) and High Refuel Floor Zone Radiation (Trip Function 4) trip functions are revised as follows:

Current TS	Proposed TS
(c) During movement of irradiated fuel assemblies or fuel cask in secondary containment.	(c) During movement of recently irradiated fuel assemblies ~~or fuel cask~~ in secondary containment.
(d) During Alteration of the Reactor Core.	(d) ~~During Alteration of the Reactor Core~~ Deleted

b. TS 3.7.B.4, Standby Gas Treatment System

TS 3.7.B.4 is revised as follows:

Current TS 3.7.B.4	Proposed TS 3.7.B.4
4. With two trains of the Standby Gas Treatment System inoperable, or as made applicable by Specification 3.7.B.3:	4. With two trains of the Standby Gas Treatment System inoperable, or as made applicable by Specification 3.7.B.3:
a. With the reactor in the run mode, startup mode, or hot shutdown condition, the reactor shall be placed in hot shutdown within 12 hours and cold shutdown within 36 hours.	a. With the reactor in the run mode, startup mode, or hot shutdown condition, the reactor shall be placed in hot shutdown within 12 hours and cold shutdown within 36 hours.
b. During movement of irradiated fuel assemblies or the fuel cask in the secondary containment, during core alterations, or during operations with the potential for draining the reactor vessel, immediately:	b. During movement of recently irradiated fuel assemblies ~~or the fuel cask~~ in the secondary containment~~, during core alterations,~~ or during operations with the potential for draining the reactor vessel, immediately:
i. Suspend movement of irradiated fuel assemblies and the fuel cask in secondary containment; and	i. Suspend movement of recently irradiated fuel assemblies ~~and the fuel cask~~ in secondary containment; and
ii. Suspend core alterations; and	ii. ~~Suspend core alterations; and~~ Deleted
iii. Initiate action to suspend operations with the potential for draining the reactor vessel.	iii. Initiate action to suspend operations with the potential for draining the reactor vessel.

c. TS 3.7.C, Secondary Containment System

TS 3.7.C.1 is revised as follows:

Current TS 3.7.C.1	Proposed TS 3.7.C.1
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 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.	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 b. During movement of recently irradiated fuel assemblies ~~or the fuel cask~~ in secondary containment; or c. ~~During alteration of the Reactor Core; or~~ Deleted d. During operations with the potential for draining the reactor vessel.

TS 3.7.C.4 is revised as follows:

Current TS 3.7.C.4	Proposed TS 3.7.C.4
4. With Secondary Containment Integrity not maintained during movement of irradiated fuel assemblies or the fuel cask in secondary containment, during alteration of the Reactor Core, or during operations with the potential for draining the reactor vessel, immediately perform the following actions: a. Suspend movement of irradiated fuel assemblies and the fuel cask in secondary containment; and b. Suspend alteration of the Reactor Core; and c. Initiate action to suspend operations with the potential for draining the reactor vessel.	4. With Secondary Containment Integrity not maintained during movement of recently irradiated fuel assemblies ~~or the fuel cask~~ in secondary containment~~,~~ ~~during alteration of the Reactor Core,~~ or during operations with the potential for draining the reactor vessel, immediately perform the following actions: a. Suspend movement of recently irradiated fuel assemblies ~~and the fuel cask~~ in secondary containment; and b. ~~Suspend alteration of the Reactor Core; and~~ Deleted c. Initiate action to suspend operations with the potential for draining the reactor vessel.

The proposed changes would revise the Limiting Conditions for Operation in VY TS to relax secondary containment operability requirements when handling fuel that is not "recently" irradiated. The proposed changes would allow for more efficient performance of reactor defueling and decommissioning activities, including transfer of spent reactor fuel from the spent fuel pool to dry fuel storage casks, while continuing to provide adequate controls against the release of fission product radioactivity to the outside atmosphere during fuel handling activities.

Current TS (Table 3.2.3, Specifications 3.7.B.4 and 3.7.C) require secondary containment, together with the SGT system to be operable:

(1) During movement of irradiated fuel or the fuel cask,
(2) During core alterations, and
(3) During operations with the potential for draining the reactor vessel (OPDRVs).

Changes are proposed to the secondary containment isolation requirements in TS Table 3.2.3 and the SGT system and Secondary Containment operability requirements specified in TS 3.7.B.4 and TS 3.7.C for refueling operations based on the revised FHA. The proposed changes will eliminate operability requirements during fuel handling activities that do not involve "recently" irradiated fuel, movement of a fuel cask, or core alterations. The systems will still be required to be operable during OPDRVs and during fuel handling activities involving recently irradiated fuel.

Based on TSTF-51, changes to the TS Bases define what time period must elapse before fuel is no longer considered "recently" irradiated. Based on the calculation performed to support the revised FHA analysis, the minimum time period that must elapse following reactor shutdown is 13 days. The time period was calculated using the AST methodology in accordance with Regulatory Guide 1.183 and the requirements in 10 CFR 50.67. This is included in the TS Bases to reflect the proposed changes to the TS. Proposed changes to the TS Bases are provided for information in Attachment 2. Upon approval of this amendment, changes to the Bases will be incorporated in accordance with TS 6.7.E, the TS Bases Control Program.

TSTF-51 states that licensees adding the term "recently" must make certain commitments which are consistent with draft NUMARC 93-01, Revision 3 (Reference 4). The following addresses compliance with this condition of the TSTF.

2.2 Administrative Controls

The industry, through the Nuclear Energy Institute (NEI) and its predecessor, the Nuclear Management and Resources Council (NUMARC), has developed guidance to assess and manage the increase in risk that may result during outage activities. This guidance was issued as NUMARC 91-06 "Guidelines for Industry Actions to Assess Shutdown Management" and NUMARC 93-01 "Industry Guideline for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants". Sections 11.3.5 and 11.3.6 of NUMARC 93-01 Rev. 3 address risk assessments for shutdown conditions and include guidance for managing an open containment.

TSTF-51, "Reviewer's Note" states that licensees adding the term "recently" must make the following commitments which are consistent with draft NUMARC 93-01, Revision 3, Section 11.2.6 "Safety Assessment for Removal of Equipment from Service During Shutdown Conditions", subheading "Containment-Primary (PWR)/Secondary(BWR)":

- During fuel handling/core alterations, ventilation system and radiation monitor availability (as defined in NUMARC 91-06) should be assessed, with respect to filtration and monitoring of releases from the fuel. Following shutdown, radioactivity in the fuel decays away fairly rapidly. The basis of the Technical Specification operability amendment is the reduction in doses due to such decay. The goal of maintaining ventilation system and radiation monitor availability is to reduce doses even further below that provided by the natural decay.

- A single normal or contingency method to promptly close primary or secondary containment penetrations should be developed. Such prompt methods need not completely block the penetration or be capable of resisting pressure.

The purpose of the "prompt methods" mentioned above is to enable ventilation systems to draw the release from a postulated fuel handling accident in the proper direction such that it can be treated and monitored."

NUMARC 93-01, Revision 3 was issued in July 2000 and has subsequently been superseded by NUMARC 93-01, Revision 4A (Reference 4). Section 11 of Rev. 4A has been endorsed by Regulatory Guide 1.160, Revision 3 (Reference 5), which superseded Regulatory Guide 1.182, Revision 0. Sections 11.3.5 and 11.3.6 of NUMARC 93-01, Rev. 4A address the scope and methods of assessing shutdown conditions. Section 11.3.6.5 in particular provides guidelines for assessing containment systems to be removed from service including the capability to achieve containment closure in sufficient time to mitigate potential fission product release. The guidelines are the same as those in Section 11.2.6 of draft NUMARC 93-01, Revision 3 as discussed in TSTF-51. ENO will implement these guidelines prior to implementation of the license amendment.

Contingency plans for closure of openings will include the following:

- Equipment and tools needed to facilitate closure will be staged
- Personnel responsible for closure will be knowledgeable and trained in the procedures for establishing building integrity
- The closure response team will be accompanied by a Radiation Protection (RP) technician for radiation protection monitoring
- Hoses and cables routed through openings will employ a means to allow rapid, safe disconnect and removal
- One door in each airlock will be capable of expeditious closure

2.3 Background and Basis for the Proposed Changes

Regulatory Guide 1.183 provides guidance for the implementation of ASTs. 10 CFR 50.67 requires licensees seeking to use ASTs to apply for a license amendment and include an evaluation of the consequences of the affected design-basis accidents. This application addresses these requirements by proposing limited scope application of the AST described in Regulatory Guide 1.183 in evaluating the radiological consequences of an FHA. As part of the implementation of the AST, the total effective dose equivalent (TEDE) acceptance criterion of 10 CFR 50.67(b)(2) is applied.

The NRC has already approved use of AST at VY by License Amendment 223 (Reference 6). The refueling accident described in Updated Final Safety Analysis Report (UFSAR) Section 14.6.4 reflects a FHA analysis that implements the AST methodology.

3. TECHNICAL EVALUATION

3.1 System Descriptions

3.1.1 Secondary Containment

Secondary containment and the SGT system mitigate the potential effects of a FHA and are part of the primary success path for a design-basis FHA. The safety objectives of the secondary containment are to minimize ground level release of airborne radioactive materials and to provide a means for a controlled release of the building atmosphere should a design basis accident occur. There are two principal accidents for which credit is taken for secondary containment integrity. These are the loss-of-coolant accident (LOCA) and the refueling accident. The analysis of these accidents is discussed in the UFSAR Section 14.6. The secondary containment performs a passive function in response to either of these limiting events. Leak tightness is required to ensure that the release of radioactive materials from the primary containment is restricted to those leakage paths and associated leakage rates assumed in the accident analysis and that fission products entrapped in the secondary containment will be treated by the SGT system prior to discharge to the environment. In the COLD SHUTDOWN and Refueling modes (with reactor coolant temperature less than 212°F), the probability of a LOCA event is reduced due to the pressure and temperature limitations in these conditions. Therefore, maintaining secondary containment integrity is not required in the cold shutdown condition or the refuel mode, except for other situations for which significant releases of radioactive material can be postulated.

UFSAR Section 5.3 describes the secondary containment requirements, including the SGT system (Section 5.3.4) and reactor building ventilation system (Section 5.3.5).

3.1.2 Reactor Building Ventilation Isolation and Standby Gas Treatment System Initiation Instrumentation

The reactor building ventilation isolation and SGT system initiation instrumentation automatically initiates closure of the Reactor Building Automatic Ventilation System Isolation Valves (RBAVSIVs) and starts the SGT system. The function of these components and systems, in combination with other accident mitigation systems, is to limit fission product release during and following postulated DBAs. Reactor Building (i.e., secondary containment) isolation and establishment of vacuum with the SGT system ensures that fission products that leak from primary containment following a DBA, or are released outside primary containment, or are released during certain operations when primary containment is not required to be operable, are maintained within applicable limits. The isolation and initiation signals generated by the reactor building ventilation isolation and SGT system initiation instrumentation are implicitly assumed in the safety analyses to initiate closure of the RBAVSIVs and start the SGT system to limit offsite doses.

The High Reactor Building Ventilation Radiation and High Refueling Floor Zone Radiation Trip Functions are required to be operable in RUN, STARTUP/HOT STANDBY, HOT SHUTDOWN, Refuel (with reactor coolant temperature greater than 212°F) where considerable energy exists in the reactor coolant system (RCS); thus, there is a possibility of pipe breaks resulting in significant releases of radioactive steam and gas. In COLD SHUTDOWN and Refuel (with reactor coolant temperature less than 212°F), the probability and consequences of these events are low due to the RCS pressure and temperature limitations of these Modes; thus, these Trip Functions are not required. In addition, the Trip Functions are also required to be operable during OPDRVs, during movement of irradiated fuel assemblies or fuel cask in the secondary containment, and during Alteration of the

Reactor Core, because the capability of detecting radiation releases due to fuel failures must be provided to ensure that offsite dose limits are not exceeded.

3.2 Alternate Source Term

ENO has completed a calculation (Reference 7) following 13 days of decay time evaluating the potential dose consequences of the FHA. A copy of the calculation is included as Attachment 4 of this submittal. This calculation demonstrates that radiological doses at the exclusion area boundary (EAB), low population zone (LPZ) and in the control room (CR) are within allowable limits of 10 CFR 50.67 without crediting secondary containment operability, control room high efficiency air filtration or standby gas treatment systems, after a 13 day fuel decay period following reactor shutdown.

The postulated accident scenarios were based on the AST methodology in Regulatory Guide 1.183, Appendix B. Two main configurations of the Reactor Building during fuel movement were considered, as follows:

- A closed containment, for the purpose of replicating the current licensing basis and thus provide a valid starting point for the re-analyses documented in the present calculation, and
- An open containment, for the purpose of identifying the required decay time prior to fuel movement and the various pre- and post-FHA Main Control Room (CR) ventilation configurations that would support refueling with open containment.

The VY design inputs for the FHA analysis are summarized in Table 3-1 through Table 3-3 of Attachment 4. The assumptions associated with each of the scenarios are described in subsections 5.1.1 (closed containment) and 5.2.1 (open containment) of Attachment 4. It should be noted that alternate scenarios described in the analysis considering upgrading the CR ventilation system with filtration or purging of the CR atmosphere following CR isolation prior to fuel movement are not considered in this license amendment request.

3.2.1 Acceptance Criteria

The basic radiological acceptance criteria associated with the AST methodology are spelled out in 10 CFR 50.67(b)(2), and amount to 25 rem TEDE for offsite receptors and 5 rem for control room personnel. These criteria, however, are for evaluating certain postulated accidents of exceedingly low occurrence probability and low risk of public exposure to radiation. For events with higher probability of occurrence, such as a FHA, the acceptance criteria for the offsite receptors are more stringent, while that for the control room operators remains the same. The applicable AST criteria for an FHA are as follows (Regulatory Guide 1.183 and 10 CFR 50.67(b)(2)):

- Exclusion Area Boundary (EAB): 6.3 rem TEDE
- Low Population Zone (LPZ): 6.3 rem TEDE
- Control Room (CR): 5.0 rem TEDE

The EAB and LPZ criteria are referred to as being "well within" the regulatory limits (i.e., ~25% below limit).

The LPZ doses were not addressed in the calculation since the release occurs within two hours, and as a result they are bounded by the corresponding doses at the EAB.

3.3 Atmospheric Dispersion

Atmospheric dispersion factors (X/Qs) at the normal control room air intake were calculated using the ARCON 96 computer code. The following table (from Table 3-3 of Attachment 4) provides the X/Qs used in the analysis:

No.	Release Point	Receptor Point	Post FHA Interval	χ/Q (sec/m^3)
1	Main stack (Calculation of record with closed containment, for confirmatory analyses)	EAB	Instantaneous release	1.35E-04
2		Control Room Fresh Air Intake	Instantaneous release	6.04E-05
3	Reactor Building blowout panel (Calculation with open containment for present application)	EAB	0 - 2 hrs	1.69E-03
4		Control Room Fresh Air Intake	0 - 2 hrs	5.89E-03
			2 - 8 hrs	1.53E-03
			8 - 24 hrs	6.41E-04
			24 - 96 hrs	6.64E-04
			96 - 720 hrs	5.10E-04

Primary assumptions used in the analysis for two release locations are summarized as follows:

<u>Main Stack Release Point (Closed Containment Scenario)</u>

- The reactor building was assumed to be closed during the refueling operations, such that all releases to the environment would be via the main stack, with no credit for any filtration by the SGT system, or any in-transit decay and plateout.
- Transport of the released radioactivity to the receptors of interest is dictated by the applicable atmospheric dispersion factors in the above table.
- The CR ventilation system was assumed to remain in the normal operating mode during the entire exposure interval (30 days). The air intake flow is 3700 cubic feet per minute (cfm) (includes additional flow from surrounding areas as a result of ingress, egress and leakage) and is unfiltered.

<u>Reactor Building Blowout Panel (Open Containment Scenario)</u>

- The reactor building was assumed to be open during the refueling operations, with all post-FHA releases to the environment assumed to be at ground level, via the RB blowout panels.
- Transport of the released radioactivity to the receptors of interest is dictated by the applicable atmospheric dispersion factors in above table for ground-level releases.
- The CR ventilation system was assumed to remain in the normal operating mode during the entire exposure interval (30 days), with an intake flow of 3700 cfm, unfiltered. For demonstrative purposes, the CR envelope was considered to be fully isolated prior to fuel movement, with an unfiltered inleakage of 50 cfm.

As noted in the introduction of Regulatory Guide 1.194 (Reference 8), many of the positions in the guide represent significant changes. ARCON96 implements an improved building wake dispersion algorithm; assessments of ground level, building vent, elevated and diffuse-source release models; use of hour-by-hour meteorological observations; sector averaging; and directional dependence of dispersion conditions. Therefore, no discussion of the comparison with current licensing basis X/Q values is presented.

3.4 Radiological Consequences of a Design-Basis Fuel Handling Accident

The analysis was performed using the ELISA-2, Version 2.4 computer code. The dose conversion factors in ELISA-2 are from Federal Guidance Reports 11 and 12 (References 9 and 10). Dose rates and cumulative doses are computed for each organ, TEDE, skin and air. Of these, only the TEDE doses are presented in the main body of the calculation for comparing with regulations, which only specify TEDE limits.

For the closed containment scenario, the dose consequences for the FHA scenario described in subsection 5.1 of Attachment 4 are presented in Table 5-2 of Attachment 4. It is seen that there is very good agreement between the calculation of record and the analyses documented in the present calculation. The only noteworthy difference (about 10%, which is still acceptable) is due to the underestimation of the Xe-135m contribution to the dose, resulting from this noble-gas isotope not being identified in the calculation of record as a daughter product of I-135 in the decay correction. The corresponding EAB dose in Case A3 is not impacted due to the long pre-FHA decay time for this isotope. The CR doses in all cases are not affected by the Xe-135m underestimation primarily due to the finite-cloud correction to the submersion dose.

For the open containment scenario, the dose consequences for the FHA scenario described in subsection 5.2 of Attachment 4 are presented in Table 5-3 of Attachment 4 and shown graphically in Figures 5-1 and 5-2 of Attachment 4. It is seen that the EAB TEDE dose with open containment is less than the regulatory limit of 6.3 rem limit even with a 24-hr pre-FHA decay time. On the other hand, a decay time of about 11 to 13 days is needed to achieve a CR dose less than 5 rem, for intake flows ranging between 50 cfm (with the CR pre-isolated) and 3700 cfm (current CR configuration). For this license amendment request, the most conservative value of 13 days is applied throughout this application.

3.4.1 Results of Fuel Handling Accident Dose Consequences

The dose evaluations of the postulated fuel handling accident demonstrates that the calculated TEDE values to the CR, EAB, and LPZ using a 13 day decay period for the reactor fuel with open containment and with the normal, unfiltered, control room ventilation system in operation are less than the regulatory limit of 6.3 rem TEDE even with a 24 hour pre-FHA decay time. The 13 day decay time is needed to achieve a CR dose of less than 5 rem.

3.5 Core Alterations

The accidents postulated to occur during core alterations, in addition to the fuel handling accident, are inadvertent criticality due to control rod removal error and the inadvertent loading of, and subsequent operation with, a fuel assembly in an improper location. These events are not postulated to result in fuel cladding integrity damage. Therefore, the only accident postulated to occur during core alterations that result in significant radioactive release is the FHA. Thus, the consequence of a FHA envelops the consequences of potential accidents postulated to occur during core alterations. Therefore, the proposed changes omitting core alterations are justified.

3.6 Fuel Cask Movement

The operability requirements during movement of a fuel cask for ESF mitigation are deleted as part of this proposed license amendment. The only accident postulated during handling of a fuel cask is the FHA. The FHA assumes an irradiated fuel assembly is dropped onto the reactor core or in the spent fuel pool from the maximum height allowed by the fuel handling equipment. The analysis assumes that the entire amount of potential energy is available for application to the fuel assemblies involved in the accident. Also, none of the energy associated with the dropped fuel assembly is absorbed by the fuel material.

The Reactor Building crane is a 110-ton capacity overhead bridge crane that provides services for the reactor and refueling area. The crane handles the spent fuel shipping cask. The Reactor Building crane was modified in 1976 by replacing the original trolley with one that has a dual load path on the main hoist when used for cask handling operations. The design of the new trolley satisfies the criteria for dual load path or "single-failure-proof' cranes, and, with issuance of Amendment 29 to the VY operating license on January 28, 1977, the NRC staff accepted the crane as single-failure-proof (Reference 15).

In addition to the hardware improvements implemented to modify the crane with single-failure-proof components, VY implemented a number of other improvements to enhance conformance with the guidelines of NUREG-0612, "Control of Heavy Loads at Nuclear Power Plants" (Reference 11). These improvements included:

- revising maintenance procedures to define safe load paths for major loads.
- revising procedures to include training and qualification requirements for crane operators, sling selection criteria, crane inspections prior to use, and supervisory oversight of heavy lift operations.
- procurement of special lifting devices and performance of periodic non-destructive examinations to monitor the condition of lifting devices.

The NRC staff accepted these improvements through a safety evaluation transmitted by letter dated June 27, 1984 (Reference 12).

The single-failure-proof modifications, the procured special lifting devices, and the sling selection procedures provide reasonable assurance that the handling system used for heavy load movement of a fuel cask near the spent fuel pool will have designed-in features to prevent a load drop. As outlined in NUREG-0612, the NRC staff accepts that provision of a single-failure-proof handling system, in conjunction with other actions implemented at VY, provides defense-in-depth against drops of loads heavier than one fuel assembly and its associated handling tool. This would include a fuel cask loaded with spent fuel. Thus, actions and events necessary to result in a heavy load drop from the Reactor Building crane over spent fuel are not sufficiently credible that this event was included among design basis events.

The VY Technical Requirements Manual (TRM) contains the crane functionality and surveillance requirements, as well as requirements to install mechanical rail stops to prevent the movement of a fuel cask over irradiated fuel. License Amendment 239 relocated the TS requirements for the Reactor Building crane to the TRM (Reference 13).

Since the FHA resulting from a dropped fuel cask is shown to not be credible, the proposed TS changes omitting operability requirements during movement of a fuel cask for ESF mitigation is justified.

4. REGULATORY EVALUATION

4.1 Applicable Regulatory Requirements/Criteria

This section describes how the proposed changes and ENO's technical analyses satisfy applicable regulatory requirements and acceptance criteria.

<u>10 CFR 50 Appendix A General Design Criterion 61, "Fuel Storage and Handling and Radiological Control"</u>

The general design criteria (GDC) in place today became effective after the VY construction permit was issued. A September 18, 1992 memorandum (ML003763736) to the NRC Executive Director of Operations from the Secretary of the NRC summarized the results of a Commissioners vote in which the Commissioners instructed the NRC staff not to apply the GDC to plants with construction permits issued prior to May 21, 1971. VY's construction permit was issued on December 11, 1967.

VY's design and licensing basis for fuel storage and handling and radiological controls is detailed in the Updated Final Safety Analysis Report (UFSAR) and other plant-specific licensing basis documents. UFSAR Appendix F provides a comparison of VY with the proposed GDC published by the Atomic Energy Commission (AEC) for public comment in the Federal Register dated July 11, 1967.

<u>10 CFR 50.67 "Accident Source Term"</u>

10 CFR 50.67 permits licensees to voluntarily revise the accident source term used in design-basis radiological consequence analyses. This document is part of a 10 CFR 50.90 license amendment application and evaluates the consequences of a design-basis fuel handling accident previously reported in the safety analysis report.

<u>10 CFR 50.65 "Requirements for monitoring the effectiveness of maintenance at nuclear power plants"</u>

10 CFR 50.65(a)(4) requires licensees to assess and manage changes in risk that result from taking risk-significant systems out-of-service or during certain maintenance activities. The NRC staff, in Regulatory Guide 1.160, endorsed NUMARC 93-01, Revision 4A as providing acceptable methods, with clarifications and provisions described in Part C of the Regulatory Guide, for complying with the requirements of 10 CFR 50.65. Section 11.3.6.5 "Containment - Primary (PWR)/Secondary (BWR)," of NUMARC 93-01, Revision 4A states:

> Maintenance activities involving the need for open containment should include evaluation of the capability to achieve containment closure in sufficient time to mitigate potential fission product release. This time is dependent on a number of factors, including the decay heat level and the amount of RCS inventory available.
>
> For BWRs, technical specifications may require secondary containment to be closed under certain conditions, such as during fuel handling and operations with a potential to drain the vessel.
>
> In addition to the guidance in NUMARC 91-06, for plants which obtain license amendments to utilize shutdown safety administrative controls in lieu of Technical Specification requirements on primary or secondary containment operability and ventilation system operability during fuel handling or core alterations, the following guidelines should be included in the assessment of systems removed from service:

- During fuel handling/core alterations, ventilation system and radiation monitor availability (as defined in NUMARC 91-06) should be assessed, with respect to filtration and monitoring of releases from the fuel. Following shutdown, radioactivity in the RCS decays fairly rapidly. The basis of the Technical Specification operability amendment is the reduction in doses due to such decay. The goal of maintaining ventilation system and radiation monitor availability is to reduce doses even further below that provided by the natural decay, and to avoid unmonitored releases.

- A single normal or contingency method to promptly close primary or secondary containment penetrations should be developed. Such prompt methods need not completely block the penetration or be capable of resisting pressure. The purpose is to enable ventilation systems to draw the release from a postulated fuel handling accident in the proper direction such that it can be treated and monitored.

To further limit the potential radiological consequences of a fuel handling accident at VY, ENO will revise the VY guidelines for assessing systems removed from service during the handling of recently irradiated fuel assemblies or core alterations to implement the provisions of Section 11.3.6.5 of NUMARC 93-01, "Industry Guideline for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants," Revision 4A. These guidelines will address the capabilities to promptly close secondary containment and will be completed prior to the implementation of this license amendment. See Section 2.2 of this application for additional discussion on the administrative controls to be implemented upon approval of this amendment.

<u>10 CFR 100, Paragraph 11, "Determination of Exclusion Area, Low Population Zone and Population Center Distance"</u>

This paragraph provides criteria for evaluating the radiological aspects of reactor sites. A footnote to 10 CFR 100.11 states that the fission product release assumed in these evaluations should be based on a major accident involving substantial meltdown of the core with subsequent release of appreciable quantities of fission products. A similar footnote appears in 10 CFR 50.67.

In accordance with the provisions of 10 CFR 50.67(a), the radiation dose reference values in 10 CFR 50.67(b)(2) were used in these analyses in lieu of those prescribed in 10 CFR 100. (Refer to footnote 5 on page 1.183-7 of Regulatory Guide 1.183, dated July 2000.)

<u>Regulatory Guide 1.25, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Fuel Handling Accident in the Fuel Handling and Storage Facility for Boiling and Pressurized Water Reactors", March 1972</u>

Regulatory Guide 1.25 is not applicable to the application. Regulatory Guide 1.183 supersedes corresponding radiological assumptions provided in other regulatory guides and standard review plan chapters when used in conjunction with an approved alternate source term and the TEDE criteria provided in 10 CFR 50.67.

Regulatory Guide 1.183, "Alternative Radiological Source Terms for evaluating Design Basis Accidents at Nuclear Power Reactors", July 2000

Regulatory Guide 1.183 outlines acceptable applications of ASTs, the scope, nature and documentation of associated analyses and evaluations, consideration of impacts on analyzed risk; and content of submittals. It also establishes acceptable ASTs and identifies the attributes of ASTs acceptable to the NRC staff. This guide also identifies acceptable radiological analysis assumptions for use in conjunction with the AST.

Regulatory Guide 1.183, Appendix B, Sec. 5.3, Footnote 3 states that the following provisions need to be implemented for refueling operations with open containment:

> "The (NRC) staff will generally require that technical specifications allowing such operations (i.e., open containment during fuel handling operations) include administrative controls to close the airlock, hatch, or open penetrations within 30 minutes. Such administrative control will generally require that a dedicated individual be present, with necessary equipment available, to restore containment closure should a fuel handling accident occur. Radiological analyses should generally not credit this manual isolation."

The administrative controls that VY will put in place following approval of this amendment request are described in Section 2.2 of this application.

ENO used this regulatory guide extensively in the preparation of this "limited scope implementation" evaluation, the supported application and the supporting analyses. This application and the supporting analyses comply with this guidance to the extent practical.

NUREG-1465, "Accident Source Terms for Light-Water Nuclear Power Plants"

NUREG-1465 provides more realistic estimates than Technical Information Document 14844 of "source term" releases into containment in terms of timing, nuclide types, quantities, and chemical form, given a severe core melt. NUREG-1465 provides much of the technical basis for the regulatory positions in Regulatory Guide 1.183.

NUREG-0800, Standard Review Plan, Section 15.7.4, "Radiological Consequences of Fuel Handling Accidents"

This SRP section covers the review of the radiological effects of a postulated fuel handling accident. Revision 1 does not reflect the guidance in Regulatory Guide 1.183 or the promulgation of 10 CFR 50.67.

NUREG-0612, "Control of Heavy Loads at Nuclear Power Plants: Resolution of Generic Technical Activity A-36"

NUREG-0612 provides guidance on the control of heavy loads that may be handled in several plant areas. If these loads were to drop in certain locations in the plant, they may impact spent fuel, fuel in the core, or equipment that may be required to achieve safe shutdown and continue decay heat removal. The guidelines proposed include definition of safe load paths, use of load handling procedures, training of crane operators, guidelines on slings and special lifting devices, periodic inspection and maintenance for the crane, as well as various alternatives that include: use of a single failure proof handling system, use of mechanical stops or electrical interlocks to keep heavy loads away from fuel or safe shutdown equipment, or analyzing the consequences of postulated heavy load drops to show these are within acceptable limits.

4.2 Precedent

The proposed VY TS changes follow TSTF-51 and similar changes approved for Pilgrim Nuclear Power Station. Even though VY's current TS appear different in style than the TSTF-51, the proposed changes are consistent with TSTF-51 requirements, and approved Pilgrim TS changes in scope and requirements.

4.3 No Significant Hazards Consideration Determination

Pursuant to 10CFR50.92, Entergy Nuclear Operations, Inc. (ENO) 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 would 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.

ENO proposes changes to the (a) Vermont Yankee Nuclear Power Station (VY) licensing basis of the Fuel Handling Accident (FHA) as described in the VY Updated Final Safety Analysis Report (UFSAR) and (b) VY Technical Specifications (TS). The proposed changes to the TS would eliminate secondary containment operability requirements when handling sufficiently decayed irradiated fuel, or fuel casks, and performing core alterations, and clarify requirements associated with operations with potential to drain the reactor vessel (OPDRVs). These changes revise secondary containment isolation requirements, and standby gas treatment (SGT) system, and secondary containment operability requirements for refueling operations.

The discussion below addresses each of these criteria and demonstrates that the proposed amendment does not constitute a significant hazard.

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

   Response: No.

   The proposed changes do not modify the design or operation of equipment used to move spent fuel or to perform core alterations. The proposed changes cannot increase the probability of any previously analyzed accident because they are based on changes in Source Term, atmospheric dispersion and dose consequence analysis methodology, not in procedures or equipment used for fuel handling.

   The conservative re-analysis of the FHA concludes that the radiological consequences are within the regulatory limits established 10 CFR 50.67. This conclusion is based on the Alternate Source Term and guidance provided in Appendix B of Regulatory Guide 1.183 and analyses of fission product release and transport path that does not take credit for dose mitigation provided by engineered safeguards including secondary containment and the SGT system. The results of the core alteration events, other than the FHA, remain unchanged from the original design-basis that showed these events do not result in fuel cladding damage or radioactive release.

Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

   Response: No.

   The proposed changes do not introduce any new modes of plant operation and do not involve physical modifications to the plant.

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

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

   Response: No.

   Regulation 10 CFR 50.67 permits licensees to voluntarily revise the accident source term used in design-basis radiological consequence analyses. This license amendment application evaluates the consequences of a design-basis fuel handling accident in accordance with this regulation and Regulatory Guide 1.183. The revised analysis concludes that the radiological consequences of the fuel handling accident are less than the regulatory allowable limits. Safety margins and analytical conservatisms are retained to ensure the analysis adequately bounds all postulated event scenarios. The selected assumptions and release models provide an appropriate and prudent safety margin against unpredicted events in the course of an accident and compensates for large uncertainties in facility parameters, accident progression, radioactive material transport and atmospheric dispersion. The proposed TS applicability statements continue to ensure that the total effective dose equivalent (TEDE) at the boundaries of the control room, the exclusion area, and low population zone boundaries are below the corresponding regulatory allowable limits in 10 CFR 50.67(b)(2).

   Therefore, the proposed change does not involve a significant reduction in the margin of safety.

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

4.4 Conclusion

Analyses of the radiological dose of a postulated FHA involving irradiated fuel assembles that have been allowed to decay for at least 13 days show that the calculated TEDE values to the control room occupants and at the exclusion area boundary, without crediting secondary containment and SGT system operation, are below the allowable TEDE limits established in Regulatory Guide 1.183 and 10 CFR 50.67. Therefore, after 13 days, movement of irradiated fuel assemblies can commence and continue without the operability requirements for the SGT system and secondary containment. This conclusion forms the basis for the proposed TS changes, with the exception of the deletion of operability requirements during movement of a fuel cask for ESF mitigation features. The basis for this change is described in Section 3.6 of this request.

5. ENVIRONMENTAL CONSIDERATIONS

This amendment request meets the eligibility criteria for categorical exclusion from environmental review set forth in 10CFR51.22(c)(9) as follows:

(i) The amendment involves no significant hazards determination.

As described in Section 4 of this evaluation, the proposed change involves no significant hazards consideration.

(ii) There is no significant change in the types or significant increase in the amounts of any effluent that may be released offsite.

The proposed amendment does not involve any physical alterations to the plant configuration that could lead to a change in the type or amount of effluent release offsite.

(iii) There is no significant increase in individual or cumulative occupational radiation exposure.

The proposed amendment does not involve a significant increase in individual or cumulative occupational radiation exposure.

Based on the above, ENO concludes that the proposed change meets the eligibility criteria for categorical exclusion as set forth in 10CFR51.22(c)(9). Pursuant to 10CFR51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of this amendment.

6. REFERENCES

1. TSTF-51, Rev. 2, "Revise Containment Requirements During Handling Irradiated Fuel and Core Alterations," Excel Services Corporation

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

3. Letter, Entergy Nuclear Operations, Inc. to USNRC, "Notification of Permanent Cessation of Operations," BVY 13-079, dated September 23, 2013

4. NUMARC 93-01, Revision 4A, "Industry Guideline for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants," April 2011

5. Regulatory Guide 1.160, Revision 3, "Monitoring the Effectiveness of Maintenance at Nuclear Power Plants," May 2012

6. Letter, USNRC to Entergy Nuclear Operations, Inc., "Vermont Yankee Nuclear Power Station Issuance of Amendment RE: Alternative Source Term (TAC No. MC0253)," NVY 05-045, dated March 29, 2005

7. AREVA Document No. 32-9145461-001, "VYNPP - Re-analysis of AST/FHA Radiological Consequences with Open Containment," October 2010

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

9. EPA 520/1-88-020, Federal Guidance Report No. 11, "Limiting Values of Radionuclide Intake and Air Concentration, and Dose Conversion Factors for Inhalation, Submersion, and Ingestion," September 1988

10. EPA 402-R-93-081, Federal Guidance Report No. 12, "External Exposure to Radionuclides in Air, Water, and Soil," September 1993

11. NUREG-0612, " Control of Heavy Loads at Nuclear Power Plants: Resolution of Generic Technical Activity A-36," July 1980

12. Letter, USNRC to Vermont Yankee Nuclear Power Corporation, "Control of Heavy Loads (Phase I)," NVY 84-139, dated June 27, 1984

13. Letter, USNRC to Entergy Nuclear Operations, Inc., "Vermont Yankee Nuclear Power Station Issuance of Amendment RE: Relocation of Reactor Building Crane Technical Specification (TAC No. MD9725)," NVY 09-077, dated July 13, 2009

14. Letter, USNRC to Entergy Nuclear Operations, Inc., "Pilgrim Nuclear Power Station - Issuance of Amendment RE: Alternative Source Term for the Fuel Handling Accident Dose Consequences (TAC No. MC2705)," dated April 28, 2005

15. Letter, USNRC to Vermont Yankee Nuclear Power Corporation, Amendment No. 29, dated January 28, 1977

Attachment 2

Vermont Yankee Nuclear Power Station

Proposed Change 306

Markup of the Current Technical Specifications and Bases Pages

Table 3.2.3 (page 1 of 1)
Reactor Building Ventilation Isolation and Standby Gas Treatment System Initiation Instrumentation

TRIP FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER TRIP SYSTEM	ACTIONS WHEN REQUIRED CHANNELS ARE INOPERABLE	TRIP SETTING
1. Low Reactor Vessel Water Level	RUN, STARTUP/HOT STANDBY, HOT SHUTDOWN, Refuel[(a), (b)]	2	Note 1	≥ 127.0 inches
2. High Drywell Pressure	RUN, STARTUP/HOT STANDBY, HOT SHUTDOWN, Refuel[(a)]	2	Note 1	≤ 2.5 psig
3. High Reactor Building Ventilation Radiation	RUN, STARTUP/HOT STANDBY, HOT SHUTDOWN, Refuel[(a), (b), (c), (d)]	1	Note 1	≤ 14 mR/hr
4. High Refueling Floor Zone Radiation	RUN, STARTUP/HOT STANDBY, HOT SHUTDOWN, Refuel[(a), (b), (c), (d)]	1	Note 1	≤ 100 mR/hr

(a) With reactor coolant temperature > 212 °F.

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

(c) During movement of irradiated fuel assemblies ~~or fuel cask~~ in secondary containment.
recently

(d) ~~During Alteration of the Reactor Core.~~
Deleted

3.7 LIMITING CONDITIONS FOR OPERATION

4.7 SURVEILLANCE REQUIREMENTS

shutdown condition, the actions and completion times of Specification 3.7.B.4.b shall apply. After seven days with an inoperable train of the Standby Gas Treatment System during refueling or cold shutdown conditions requiring secondary containment integrity, the operable train of the Standby Gas Treatment System shall be placed in operation and its associated diesel generator shall be operable, or the actions and completion times of Specification 3.7.B.4.b shall apply.

4. With two trains of the Standby Gas Treatment System inoperable, or as made applicable by Specification 3.7.B.3:

   a. With the reactor in the run mode, startup mode, or hot shutdown condition, the reactor shall be placed in hot shutdown within 12 hours and cold shutdown within 36 hours.

   b. During movement of recently irradiated fuel assemblies ~~or the fuel cask~~ in the secondary containment~~, during core alterations,~~ or during operations with the potential for draining the reactor vessel, immediately:

3.7 LIMITING CONDITIONS FOR OPERATION

4.7 SURVEILLANCE REQUIREMENTS

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

ii. Deleted ~~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

C. Secondary Containment System

1. 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 1,550 cfm, shall be demonstrated at least quarterly.

* 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

recently

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

Deleted

c. ~~During alteration of the Reactor Core; or~~

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

4.7 SURVEILLANCE REQUIREMENTS

3.7 LIMITING CONDITIONS FOR OPERATION	4.7 SURVEILLANCE REQUIREMENTS
2. With Secondary Containment Integrity not maintained with the reactor in the Run Mode, Startup Mode, or Hot Shutdown condition, restore Secondary Containment Integrity within four (4) hours.	2. Intentionally blank.
3. If Specification 3.7.C.2 cannot be met, place the reactor in the Hot Shutdown condition within 12 hours and in the Cold Shutdown condition within the following 24 hours.	3. Intentionally blank.
4. With Secondary Containment Integrity not maintained during movement of recently irradiated fuel assemblies ~~or the fuel cask~~ in secondary containment~~, during alteration~~ of the Reactor Core, or during operations with the potential for draining the reactor vessel, immediately perform the following actions: a. Suspend movement of recently irradiated fuel assemblies ~~and the fuel cask~~ in secondary containment; and b. Deleted ~~Suspend alteration of the Reactor Core; and~~ c. Initiate action to suspend operations with the potential for draining the reactor vessel.	4. Intentionally blank.

BASES: 3.2.C/4.2.C REACTOR BUILDING VENTILATION ISOLATION AND STANDBY GAS TREATMENT SYSTEM INITIATION
APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

instrumentation are implicitly assumed in the safety analyses of References 2, 3, and 4, to initiate closure of the RBAVSIVs and start the SGT System to limit offsite doses.

Reactor building ventilation isolation and Standby Gas Treatment System initiation instrumentation satisfies Criterion 3 of 10 CFR 50.36(c)(2)(ii).

The operability of the reactor building ventilation isolation and Standby Gas Treatment System initiation instrumentation is dependent on the operability of the individual instrumentation channel Trip Functions specified in Table 3.2.3. Each Trip Function must have the required number of operable channels in each trip system, with their trip setpoints within the calculational as-found tolerances specified in plant procedures. Operation with actual trip setpoints within calculational as-found tolerances provides reasonable assurance that, under worst case design basis conditions, the associated trip will occur within the Trip Settings specified in Table 3.2.3. As a result, a channel is considered inoperable if its actual trip setpoint is not within the calculational as-found tolerances specified in plant procedures. The actual trip setpoint is calibrated consistent with applicable setpoint methodology assumptions.

In general, the individual Trip Functions are required to be OPERABLE in RUN, STARTUP/HOT STANDBY, HOT SHUTDOWN, Refuel (with reactor coolant temperature > 212°F), during operations with the potential for draining the reactor vessel (OPDRVs)~~,~~ and during movement of recently irradiated fuel assemblies ~~or fuel cask~~ in secondary containment~~, and during Alteration of the Reactor Core~~; consistent with the Applicability for the SGT System and secondary containment requirements in Specifications 3.7.B and 3.7.C. Trip Functions that have different Applicabilities are discussed below in the individual Trip Functions discussion.

Insert 1

The specific Applicable Safety Analyses, LCO, and Applicability discussions are listed below on a Trip Function by Trip Function basis.

1. Low Reactor Vessel Water Level

Low reactor pressure vessel (RPV) water level indicates that the capability to cool the fuel may be threatened. Should RPV water level decrease too far, fuel damage could result. An isolation of the secondary containment and actuation of the SGT System are initiated in order to minimize the potential of an offsite release. The Low Reactor Vessel Water Level Trip Function is one of the Trip Functions assumed to be operable and capable of providing isolation and initiation signals. The isolation and initiation of systems on Low Reactor Vessel Water Level support actions to ensure that any offsite releases are within the limits calculated in the safety analysis.

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

BASES: 3.2.C/4.2.C REACTOR BUILDING VENTILATION ISOLATION AND STANDBY GAS TREATMENT SYSTEM INITIATION

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

have originated from the primary containment due to a break in the RCPB or the refueling floor due to a fuel handling accident. When High Reactor Building Ventilation Radiation or High Refueling Floor Zone Radiation is detected,

secondary containment isolation and actuation of the SGT System are initiated to support actions to limit the release of fission products as assumed in the UFSAR safety analyses (Ref. 4).

The High Reactor Building Ventilation Radiation and High Refueling Floor Zone Radiation signals are initiated from radiation detectors that are located on the ventilation exhaust duct coming from the reactor building and the refueling floor zones, respectively. Two channels of High Reactor Building Ventilation Radiation Trip Function and two channels of High Refueling Floor Radiation Trip Function are available and are required to be operable to ensure that no single instrument failure can preclude the isolation and initiation function.

The Trip Settings are chosen to promptly detect gross failure of the fuel cladding.

The High Reactor Building Ventilation Radiation and High Refueling Floor Zone Radiation Trip Functions are required to be operable in RUN, STARTUP/HOT STANDBY, HOT SHUTDOWN, Refuel (with reactor coolant temperature > 212°F) where considerable energy exists in the RCS; thus, there is a possibility of pipe breaks resulting in significant releases of radioactive steam and gas. In COLD SHUTDOWN and Refuel (with reactor coolant temperature ≤ 212°F), the probability and consequences of these events are low due to the RCS pressure and temperature limitations of these Modes; thus, these Trip Functions are not required. In addition, the Trip Functions are also required to be operable during OPDRVs~~,~~ and during movement of recently irradiated fuel assemblies ~~or fuel cask~~ in the secondary containment, ~~and during Alteration of the Reactor Core,~~ because the capability of detecting radiation releases due to fuel failures (due to fuel uncovery or dropped fuel assemblies) must be provided to ensure that offsite dose limits are not exceeded. Insert 1

ACTIONS

Table 3.2.3 ACTION Note 1

Because of the diversity of sensors available to provide isolation signals and the redundancy of the isolation design, an allowable out of service time of 12 hours or 24 hours depending on the Trip Function (12 hours for those Trip Functions that have channel components common to RPS instrumentation, i.e., Trip Functions 1 and 2, and 24 hours for those Trip Functions that do not have channel components common to RPS instrumentation, i.e., all other Trip Functions), has been shown to be acceptable (Refs. 5 and 6) to permit restoration of any inoperable channel to operable status. This out of service time is only acceptable provided the associated Trip Function is still maintaining isolation capability (refer to next paragraph). If the inoperable channel cannot be restored to operable status within the allowable out of service time, the channel must be placed in the tripped condition per Table 3.2.3 Note 1.a.1) or 1.a.2), as applicable. Placing the inoperable channel in trip would conservatively compensate for the inoperability, restore capability to accommodate a single failure, and allow operation to continue. Alternately,

BASES: 3.7 (Cont'd)

surveillances such as monthly torus to drywell vacuum breaker tests. Procedurally, when AC-6A is open, AC-6 and AC-7 are closed to prevent overpressurization of the SBGT system or the reactor building ductwork, should a LOCA occur. For this and similar analyses performed, a spurious opening of AC-6 or AC-7 (one of the closed containment isolation valves) is not assumed as a failure simultaneous with a postulated LOCA. Analyses demonstrate that for normal plant operation system alignments, including surveillances such as those described above, that SBGT integrity would be maintained if a LOCA was postulated. Therefore, during normal plant operations, the 90 hour clock does not apply. Accordingly, opening of the 18 inch atmospheric control isolation valves AC-7A, AC-7B, AC-8 and AC-10 will be limited to 90 hours per calendar year (except for performance of the subject valve stroke time surveillances - in which case the appropriate corresponding valves are closed to protect equipment should a LOCA occur). This restriction will apply whenever primary containment integrity is required. The 90 hour clock will apply anytime purge and vent evolutions can not assure the integrity of the SBGT trains or related equipment.

B. and C. Standby Gas Treatment System and Secondary Containment System

The secondary containment is designed to minimize any ground level release of radioactive materials which might result from a serious accident. The Reactor Building provides secondary containment during reactor operation, when the drywell is sealed and in service; the Reactor Building provides primary containment when the reactor is shutdown and the drywell is open, as during refueling. Because the secondary containment is an integral part of the complete containment system, secondary containment is required at all times that primary containment is required except, however, for initial fuel loading and low power physics testing.

In the Cold Shutdown condition or the Refuel Mode, the probability and consequences of the LOCA are reduced due to the pressure and temperature limitations in these conditions. Therefore, maintaining Secondary Containment Integrity is not required in the Cold Shutdown condition or the Refuel Mode, except for other situations for which significant releases of radioactive material can be postulated, such as during operations with a potential for draining the reactor vessel, ~~during alteration of the Reactor Core,~~ or during movement of recently irradiated fuel assemblies ~~or the fuel cask~~ in the secondary containment.

Insert 1

In order for secondary containment integrity to be met, the secondary containment must function properly in conjunction with the operation of the Standby Gas Treatment System to ensure that the required vacuum can be established and maintained. This means that the reactor building is intact with at least one door in each access opening closed, and all reactor building automatic ventilation system isolation valves are operable or the affected penetration flow path is isolated.

With the reactor in the Run Mode, the Startup Mode, or the Hot Shutdown condition, if Secondary Containment Integrity is not maintained, Secondary Containment Integrity must be restored within 4 hours. The 4 hours provides a period of time to correct the problem that is commensurate with the importance of maintaining secondary containment during the Run Mode, the Startup Mode, and the Hot Shutdown condition. This time period also ensures that the probability of an accident (requiring Secondary Containment Integrity) occurring during periods where Secondary Containment Integrity is not maintained, is minimal.

BASES: 3.7 (Cont'd)

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If Secondary Containment Integrity cannot be restored within the required time period, the plant must be brought to a mode or condition in which the LCO does not apply.

Movement of irradiated fuel assemblies ~~or the fuel cask~~ in the secondary containment~~, alteration of the Reactor Core,~~ and operations with the potential for draining the reactor vessel can be postulated to cause fission product release to the secondary containment. In such cases, the secondary containment is the only barrier to release of fission products to the environment. ~~Alteration of the Reactor Core and movement of~~ irradiated fuel assemblies ~~and the fuel cask~~ must be immediately suspended if Secondary Containment Integrity is not maintained. Suspension of these activities shall not preclude completing an action that involves moving a component to a safe position. Also, action must be immediately initiated to suspend operations with the potential for draining the reactor vessel to minimize the probability of a vessel draindown and subsequent potential for fission product release. Actions must continue until operations with the potential for draining the reactor vessel are suspended.

Movement of recently

BASES: 3.7 (Cont'd)

The Standby Gas Treatment System (SGTS) is designed to filter and exhaust the Reactor Building atmosphere to the stack during secondary containment isolation conditions, with a minimum release of radioactive materials from the Reactor Building to the environs. To insure that the standby gas treatment system will be effective in removing radioactive contaminates from the Reactor Building air, the system is tested periodically to meet the intent of ANSI N510-1975. Laboratory charcoal testing will be performed in accordance with ASTM D3803-1989, except, as allowed by GL 99-02, testing can be performed at 70% relative humidity for systems with humidity control. Both standby gas treatment fans are designed to automatically start upon containment isolation and to maintain the Reactor Building pressure to approximately a negative 0.15 inch water gauge pressure; all leakage should be in-leakage. Should the fan fail to start, the redundant alternate fan and filter system is designed to start automatically. Each of the two fans has 100% capacity. This substantiates the availability of the operable train and results in no added risk; thus, reactor operation or refueling operation can continue. If neither train is operable, the plant is brought to a condition where the system is not required.

When the reactor is in cold shutdown or refueling the drywell may be open and the Reactor Building becomes the only containment system. During cold shutdown the probability and consequences of a DBA LOCA are substantially reduced due to the pressure and temperature limitations in this mode. However, for other situations under which significant radioactive release can be postulated, such as during operations with a potential for draining the reactor vessel~~, during core alterations,~~ or during movement of recently irradiated fuel in the secondary containment, operability of standby gas treatment is required.

Both trains of the Standby Gas Treatment System are normally operable when secondary containment integrity is required. However, Specification 3.7.B.3 provides Limiting Conditions for Operation when one train of the Standby Gas Treatment System is inoperable. Provisional, continued operation is permitted since the remaining operable train is adequate to perform the required radioactivity release control function. If the applicable conditions of Specification 3.7.B.3 cannot be met, the plant must be placed in a mode or condition where the Limiting Conditions for Operation do not apply.

Entry into a refueling condition with one train of SBGTS inoperable is acceptable and there is no prohibition on mode or condition entry in this situation. In this case, the requirements of TS 3.7.B.3.b are sufficient to ensure that adequate controls are in place. During refueling conditions, accident risk is significantly reduced, and the primary activities of concern involve ~~core alterations,~~ movement of recently irradiated fuel assemblies~~,~~ and OPDRVs.

During refueling and cold shutdown conditions Specification 3.7.B.3.b provides for the indefinite continuance of refueling operations with one train of the Standby Gas Treatment System inoperable. When the seven-day completion time associated with Specification 3.7.B.3.b is not met and secondary containment integrity is required, the operable train of the Standby Gas Treatment System should immediately be placed into operation. This action ensures that the remaining train is operable, that no failures that could prevent automatic actuation have occurred, and that any other failure would be readily detected. An alternative to placing the operable train of Standby Gas Treatment in operation is to immediately suspend activities that represent a potential for releasing radioactive material to the secondary containment, thus placing the plant in a condition that minimizes risk.

Insert 1

"Recently irradiated" fuel is defined as fuel that has occupied part of a critical reactor core within the previous 13 days, i.e. reactor fuel that has decayed less than 13 days following reactor shutdown. This minimum decay period is enforced to maintain the validity of the Fuel Handling Accident dose consequence analysis.

Attachment 3

Vermont Yankee Nuclear Power Station

Proposed Change 306

Retyped Technical Specifications and Bases Pages

Table 3.2.3 (page 1 of 1)
Reactor Building Ventilation Isolation and Standby Gas Treatment System Initiation Instrumentation

TRIP FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER TRIP SYSTEM	ACTIONS WHEN REQUIRED CHANNELS ARE INOPERABLE	TRIP SETTING
1. Low Reactor Vessel Water Level	RUN, STARTUP/HOT STANDBY, HOT SHUTDOWN, Refuel[(a), (b)]	2	Note 1	≥ 127.0 inches
2. High Drywell Pressure	RUN, STARTUP/HOT STANDBY, HOT SHUTDOWN, Refuel[(a)]	2	Note 1	≤ 2.5 psig
3. High Reactor Building Ventilation Radiation	RUN, STARTUP/HOT STANDBY, HOT SHUTDOWN, Refuel[(a), (b), (c), (d)]	1	Note 1	≤ 14 mR/hr
4. High Refueling Floor Zone Radiation	RUN, STARTUP/HOT STANDBY, HOT SHUTDOWN, Refuel[(a), (b), (c), (d)]	1	Note 1	≤ 100 mR/hr

(a) With reactor coolant temperature > 212 °F.

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

(c) During movement of recently irradiated fuel assemblies in secondary containment.

(d) Deleted

BASES: 3.2.C/4.2.C REACTOR BUILDING VENTILATION ISOLATION AND STANDBY GAS TREATMENT SYSTEM INITIATION

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

instrumentation are implicitly assumed in the safety analyses of References 2, 3, and 4, to initiate closure of the RBAVSIVs and start the SGT System to limit offsite doses.

Reactor building ventilation isolation and Standby Gas Treatment System initiation instrumentation satisfies Criterion 3 of 10 CFR 50.36(c)(2)(ii).

The operability of the reactor building ventilation isolation and Standby Gas Treatment System initiation instrumentation is dependent on the operability of the individual instrumentation channel Trip Functions specified in Table 3.2.3. Each Trip Function must have the required number of operable channels in each trip system, with their trip setpoints within the calculational as-found tolerances specified in plant procedures. Operation with actual trip setpoints within calculational as-found tolerances provides reasonable assurance that, under worst case design basis conditions, the associated trip will occur within the Trip Settings specified in Table 3.2.3. As a result, a channel is considered inoperable if its actual trip setpoint is not within the calculational as-found tolerances specified in plant procedures. The actual trip setpoint is calibrated consistent with applicable setpoint methodology assumptions.

In general, the individual Trip Functions are required to be OPERABLE in RUN, STARTUP/HOT STANDBY, HOT SHUTDOWN, Refuel (with reactor coolant temperature > 212°F), during operations with the potential for draining the reactor vessel (OPDRVs) and during movement of recently irradiated fuel assemblies in secondary containment; consistent with the Applicability for the SGT System and secondary containment requirements in Specifications 3.7.B and 3.7.C. Trip Functions that have different Applicabilities are discussed below in the individual Trip Functions discussion.

"Recently irradiated" fuel is defined as fuel that has occupied part of a critical reactor core within the previous 13 days, i.e. reactor fuel that has decayed less than 13 days following reactor shutdown. This minimum decay period is enforced to maintain the validity of the Fuel Handling Accident dose consequence analysis.

The specific Applicable Safety Analyses, LCO, and Applicability discussions are listed below on a Trip Function by Trip Function basis.

1. Low Reactor Vessel Water Level

Low reactor pressure vessel (RPV) water level indicates that the capability to cool the fuel may be threatened. Should RPV water level decrease too far, fuel damage could result. An isolation of the secondary containment and actuation of the SGT System are initiated in order to minimize the potential of an offsite release. The Low Reactor Vessel Water Level Trip Function is one of the Trip Functions assumed to be operable and capable of providing isolation and initiation signals. The isolation and initiation of systems on Low Reactor Vessel Water Level support actions to ensure that any offsite releases are within the limits calculated in the safety analysis.

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

BASES: 3.2.C/4.2.C REACTOR BUILDING VENTILATION ISOLATION AND STANDBY GAS TREATMENT SYSTEM INITIATION

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

have originated from the primary containment due to a break in the RCPB or the refueling floor due to a fuel handling accident. When High Reactor Building Ventilation Radiation or High Refueling Floor Zone Radiation is detected,

secondary containment isolation and actuation of the SGT System are initiated to support actions to limit the release of fission products as assumed in the UFSAR safety analyses (Ref. 4).

The High Reactor Building Ventilation Radiation and High Refueling Floor Zone Radiation signals are initiated from radiation detectors that are located on the ventilation exhaust duct coming from the reactor building and the refueling floor zones, respectively. Two channels of High Reactor Building Ventilation Radiation Trip Function and two channels of High Refueling Floor Radiation Trip Function are available and are required to be operable to ensure that no single instrument failure can preclude the isolation and initiation function.

The Trip Settings are chosen to promptly detect gross failure of the fuel cladding.

The High Reactor Building Ventilation Radiation and High Refueling Floor Zone Radiation Trip Functions are required to be operable in RUN, STARTUP/HOT STANDBY, HOT SHUTDOWN, Refuel (with reactor coolant temperature > 212°F) where considerable energy exists in the RCS; thus, there is a possibility of pipe breaks resulting in significant releases of radioactive steam and gas. In COLD SHUTDOWN and Refuel (with reactor coolant temperature ≤ 212°F), the probability and consequences of these events are low due to the RCS pressure and temperature limitations of these Modes; thus, these Trip Functions are not required. In addition, the Trip Functions are also required to be operable during OPDRVs and during movement of recently irradiated fuel assemblies in the secondary containment, because the capability of detecting radiation releases due to fuel failures (due to fuel uncovery or dropped fuel assemblies) must be provided to ensure that offsite dose limits are not exceeded.

"Recently irradiated" fuel is defined as fuel that has occupied part of a critical reactor core within the previous 13 days, i.e. reactor fuel that has decayed less than 13 days following reactor shutdown. This minimum decay period is enforced to maintain the validity of the Fuel Handling Accident dose consequence analysis.

ACTIONS

Table 3.2.3 ACTION Note 1

Because of the diversity of sensors available to provide isolation signals and the redundancy of the isolation design, an allowable out of service time of 12 hours or 24 hours depending on the Trip Function (12 hours for those Trip Functions that have channel components common to RPS instrumentation, i.e., Trip Functions 1 and 2, and 24 hours for those Trip Functions that do not have channel components common to RPS instrumentation, i.e., all other Trip Functions), has been shown to be acceptable (Refs. 5 and 6) to permit restoration of any inoperable channel to operable status. This out of service time is only acceptable provided the associated Trip Function is still maintaining isolation capability (refer to next paragraph). If the inoperable channel cannot be restored to operable status within the allowable out of service time, the channel must be placed in the tripped condition per Table 3.2.3 Note 1.a.1) or 1.a.2), as applicable. Placing the inoperable channel in trip would conservatively compensate for the inoperability, restore capability to accommodate a single failure, and allow operation to continue. Alternately,

3.7 LIMITING CONDITIONS FOR OPERATION

4.7 SURVEILLANCE REQUIREMENTS

shutdown condition, the actions and completion times of Specification 3.7.B.4.b shall apply. After seven days with an inoperable train of the Standby Gas Treatment System during refueling or cold shutdown conditions requiring secondary containment integrity, the operable train of the Standby Gas Treatment System shall be placed in operation and its associated diesel generator shall be operable, or the actions and completion times of Specification 3.7.B.4.b shall apply.

4. With two trains of the Standby Gas Treatment System inoperable, or as made applicable by Specification 3.7.B.3:

   a. With the reactor in the run mode, startup mode, or hot shutdown condition, the reactor shall be placed in hot shutdown within 12 hours and cold shutdown within 36 hours.

   b. During movement of recently irradiated fuel assemblies in the secondary containment or during operations with the potential for draining the reactor vessel, immediately:

3.7 LIMITING CONDITIONS FOR OPERATION

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

ii. Deleted

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. 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 1,550 cfm, shall be demonstrated at least quarterly.

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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 recently irradiated fuel assemblies in secondary containment; or

c. Deleted

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

4.7 SURVEILLANCE REQUIREMENTS

3.7 LIMITING CONDITIONS FOR OPERATION	4.7 SURVEILLANCE REQUIREMENTS
2. With Secondary Containment Integrity not maintained with the reactor in the Run Mode, Startup Mode, or Hot Shutdown condition, restore Secondary Containment Integrity within four (4) hours.	2. Intentionally blank.
3. If Specification 3.7.C.2 cannot be met, place the reactor in the Hot Shutdown condition within 12 hours and in the Cold Shutdown condition within the following 24 hours.	3. Intentionally blank.
4. With Secondary Containment Integrity not maintained during movement of recently irradiated fuel assemblies in secondary containment of the Reactor Core, or during operations with the potential for draining the reactor vessel, immediately perform the following actions: a. Suspend movement of recently irradiated fuel assemblies in secondary containment; and b. Deleted c. Initiate action to suspend operations with the potential for draining the reactor vessel.	4. Intentionally blank.

BASES: 3.7 (Cont'd)

surveillances such as monthly torus to drywell vacuum breaker tests. Procedurally, when AC-6A is open, AC-6 and AC-7 are closed to prevent overpressurization of the SBGT system or the reactor building ductwork, should a LOCA occur. For this and similar analyses performed, a spurious opening of AC-6 or AC-7 (one of the closed containment isolation valves) is not assumed as a failure simultaneous with a postulated LOCA. Analyses demonstrate that for normal plant operation system alignments, including surveillances such as those described above, that SBGT integrity would be maintained if a LOCA was postulated. Therefore, during normal plant operations, the 90 hour clock does not apply. Accordingly, opening of the 18 inch atmospheric control isolation valves AC-7A, AC-7B, AC-8 and AC-10 will be limited to 90 hours per calendar year (except for performance of the subject valve stroke time surveillances - in which case the appropriate corresponding valves are closed to protect equipment should a LOCA occur). This restriction will apply whenever primary containment integrity is required. The 90 hour clock will apply anytime purge and vent evolutions can not assure the integrity of the SBGT trains or related equipment.

B. and C. Standby Gas Treatment System and Secondary Containment System

The secondary containment is designed to minimize any ground level release of radioactive materials which might result from a serious accident. The Reactor Building provides secondary containment during reactor operation, when the drywell is sealed and in service; the Reactor Building provides primary containment when the reactor is shutdown and the drywell is open, as during refueling. Because the secondary containment is an integral part of the complete containment system, secondary containment is required at all times that primary containment is required except, however, for initial fuel loading and low power physics testing.

In the Cold Shutdown condition or the Refuel Mode,the probability and consequences of the LOCA are reduced due to the pressure and temperature limitations in these conditions. Therefore, maintaining Secondary Containment Integrity is not required in the Cold Shutdown condition or the Refuel Mode, except for other situations for which significant releases of radioactive material can be postulated, such as during operations with a potential for draining the reactor vessel, or during movement of recently irradiated fuel assemblies in the secondary containment.

"Recently irradiated" fuel is defined as fuel that has occupied part of a critical reactor core within the previous 13 days, i.e. reactor fuel that has decayed less than 13 days following reactor shutdown. This minimum decay period is enforced to maintain the validity of the Fuel Handling Accident dose consequence analysis.

In order for secondary containment integrity to be met, the secondary containment must function properly in conjunction with the operation of the Standby Gas Treatment System to ensure that the required vacuum can be established and maintained. This means that the reactor building is intact with at least one door in each access opening closed, and all reactor building automatic ventilation system isolation valves are operable or the affected penetration flow path is isolated.

With the reactor in the Run Mode, the Startup Mode, or the Hot Shutdown condition, if Secondary Containment Integrity is not maintained, Secondary Containment Integrity must be restored within 4 hours. The 4 hours provides a period of time to correct the problem that is commensurate with the importance of maintaining secondary containment during the Run Mode, the Startup Mode, and the Hot Shutdown condition. This time period also ensures that the probability of an accident (requiring Secondary Containment Integrity) occurring during periods where Secondary Containment Integrity is not maintained, is minimal.

BASES: 3.7 (Cont'd)

If Secondary Containment Integrity cannot be restored within the required time period, the plant must be brought to a mode or condition in which the LCO does not apply.

Movement of recently irradiated fuel assemblies in the secondary containment and operations with the potential for draining the reactor vessel can be postulated to cause fission product release to the secondary containment. In such cases, the secondary containment is the only barrier to release of fission products to the environment. Movement of recently irradiated fuel assemblies must be immediately suspended if Secondary Containment Integrity is not maintained. Suspension of these activities shall not preclude completing an action that involves moving a component to a safe position. Also, action must be immediately initiated to suspend operations with the potential for draining the reactor vessel to minimize the probability of a vessel draindown and subsequent potential for fission product release. Actions must continue until operations with the potential for draining the reactor vessel are suspended.

BASES: 3.7 (Cont'd)

The Standby Gas Treatment System (SGTS) is designed to filter and exhaust the Reactor Building atmosphere to the stack during secondary containment isolation conditions, with a minimum release of radioactive materials from the Reactor Building to the environs. To insure that the standby gas treatment system will be effective in removing radioactive contaminates from the Reactor Building air, the system is tested periodically to meet the intent of ANSI N510-1975. Laboratory charcoal testing will be performed in accordance with ASTM D3803-1989, except, as allowed by GL 99-02, testing can be performed at 70% relative humidity for systems with humidity control. Both standby gas treatment fans are designed to automatically start upon containment isolation and to maintain the Reactor Building pressure to approximately a negative 0.15 inch water gauge pressure; all leakage should be in-leakage. Should the fan fail to start, the redundant alternate fan and filter system is designed to start automatically. Each of the two fans has 100% capacity. This substantiates the availability of the operable train and results in no added risk; thus, reactor operation or refueling operation can continue. If neither train is operable, the plant is brought to a condition where the system is not required.

When the reactor is in cold shutdown or refueling the drywell may be open and the Reactor Building becomes the only containment system. During cold shutdown the probability and consequences of a DBA LOCA are substantially reduced due to the pressure and temperature limitations in this mode. However, for other situations under which significant radioactive release can be postulated, such as during operations with a potential for draining the reactor vessel or during movement of recently irradiated fuel in the secondary containment, operability of standby gas treatment is required.

Both trains of the Standby Gas Treatment System are normally operable when secondary containment integrity is required. However, Specification 3.7.B.3 provides Limiting Conditions for Operation when one train of the Standby Gas Treatment System is inoperable. Provisional, continued operation is permitted since the remaining operable train is adequate to perform the required radioactivity release control function. If the applicable conditions of Specification 3.7.B.3 cannot be met, the plant must be placed in a mode or condition where the Limiting Conditions for Operation do not apply.

Entry into a refueling condition with one train of SBGTS inoperable is acceptable and there is no prohibition on mode or condition entry in this situation. In this case, the requirements of TS 3.7.B.3.b are sufficient to ensure that adequate controls are in place. During refueling conditions, accident risk is significantly reduced, and the primary activities of concern involve movement of recently irradiated fuel assemblies and OPDRVs.

During refueling and cold shutdown conditions Specification 3.7.B.3.b provides for the indefinite continuance of refueling operations with one train of the Standby Gas Treatment System inoperable. When the seven-day completion time associated with Specification 3.7.B.3.b is not met and secondary containment integrity is required, the operable train of the Standby Gas Treatment System should immediately be placed into operation. This action ensures that the remaining train is operable, that no failures that could prevent automatic actuation have occurred, and that any other failure would be readily detected. An alternative to placing the operable train of Standby Gas Treatment in operation is to immediately suspend activities that represent a potential for releasing radioactive material to the secondary containment, thus placing the plant in a condition that minimizes risk.

Attachment 4

Vermont Yankee Nuclear Power Station

Proposed Change 306

AREVA Document No. 32-9145461-001, "VYNPP - Re-analysis of AST/FHA Radiological Consequences with Open Containment"

AREVA

CALCULATION SUMMARY SHEET (CSS)

Document No. 32 - 9145461 - 001

Safety Related: ☒ Yes ☐ No

Title VYNPP - Re-analysis of AST/FHA Radiological Consequences with Open Containment

PURPOSE AND SUMMARY OF RESULTS:

The present calculation deals with the radiological evaluation of a design-basis fuel handling accident (FHA) taking place at the Vermont Yankee Nuclear Power Plant, based on the AST methodology (Ref. [1]), an open containment, and ground-level releases. The objectives in the original analysis (in Rev. 000) were as follows:

- Determination of the minimum required decay time after reactor shutdown that would ensure MCR habitability in its existing configuration (i.e., during normal operation with 3700 cfm of unfiltered air supplied for ventilation), and also under a pre-isolation condition (with an assumed 50 cfm of unfiltered inleakage for demonstrative purposes), and the required decay time for meeting the EAB dose,
- The benefits of upgrading the MCR HVAC system to include emergency intake-flow filtration, and
- The viability of MCR isolation prior to fuel movement, followed by MCR purge a few hours after an FHA.

Other objectives implemented in Rev. 001 include the following:

- Determination of the required decay time at which the EAB TEDE dose would be less than 1 rem (the recommended Protective Action Guide (PAG) limit for Evacuation (Ref. [2], Table 2-1),
- Clarification that the analysis covers a fuel assembly drop in either the reactor cavity pool or in the SFP during a full-core offload, and
- Re-issuance of the calculation as "Non-Proprietary."

The starting point of the analysis was confirmation of the FHA results in the VYNPP calculation of record (Ref. [3]). A summary of the results and conclusions appear in Section 6.0. Points of interest are as follows:

(a) The EAB TEDE dose with open containment is less than the regulatory limit (6.3 rem) even with a 24-hr pre-FHA decay time, and the required decay time for an EAB dose of less than 1 rem TEDE is 17 days.

(b) A decay time of about 11 to 13 days would be needed to achieve an MCR dose less than 5 rem, for intake flows ranging between 50 cfm (with the MCR pre-isolated) and 3700 cfm (current MCR configuration).

(c) Upgrade of the MCR HVAC system to include intake-flow emergency filtration can reduce the required decay time to 5 days.

(d) Pre-isolation of the MCR followed by MCR purge appears to be a viable option.

This work was performed under AREVA's Quality Assurance Program and is appropriate for safety-related design work. The latest revision of AREVA NP Procedure 0402-01, Calculations, was followed.

THE FOLLOWING COMPUTER CODES HAVE BEEN USED IN THIS DOCUMENT:

CODE/VERSION/REV	CODE/VERSION/REV
ELISA-2, Version 2.4	

THE DOCUMENT CONTAINS ASSUMPTIONS THAT SHALL BE VERIFIED PRIOR TO USE

☐ YES

☒ NO

AREVA

VYNPP - Re-analysis of AST/FHA Radiological Consequences with Open Containment

Review Method: ☒ Design Review (Detailed Check)
☐ Alternate Calculation

Signature Block

Name and Title (printed or typed)	Signature	P/R/A and LP/LR	Date	Pages/Sections Prepared/Reviewed/Approved
John N. Hamawi Advisory Engineer	John Hamawi	P	10/29/13	All
Adam Mancini Advisory Engineer	AMancini	R	10/30/13	All Rev. 001 changes
Mark A. Rinckel Technical Manager	Mark A Rinckel	A	11/5/13	Reviewer is independent

Note: P/R/A designates Preparer (P), Reviewer (R), Approver (A);
LP/LR designates Lead Preparer (LP), Lead Reviewer (LR)

Project Manager Approval of Customer References (N/A if not applicable)

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Mentoring Information (not required per 0402-01)

Name (printed or typed)	Title (printed or typed)	Mentor to: (P/R)	Signature	Date
N/A				

Record of Revision

Revision No.	Pages/Sections/Paragraphs Changed	Brief Description / Change Authorization
Rev. 000	N/A	Original issue
Rev. 001	CSS	Updated the cover sheet to include the objectives of the new revision, and the additional results.
	Section 1.1	Inserted comment clarifying that, for either an open or closed containment, the FHA analytical model and associated assumptions are applicable to a fuel assembly drop in either the reactor cavity pool or in the SFP during a full-core offload.
	Section 1.2	Added the PAG limit of 1 rem TEDE as part of the acceptance criteria.
	Table 3-1	Added table Note (a) clarifying the use and implications of the radial peaking factor.
	Table 3-3	Added table note regarding the applicability of the 1995-1999 hourly meteorological data in the current applications.
	Section 4.1	Identified the operating platform for the new ELISA-2 computer run (differs from than in Rev. 000).
	Section 4.2	Updated the ColdStor location name for the Rev. 000 computer files, and included the location for the new files. Updated the title for Table 4-1 to identify it as being solely for the Rev. 000 computer files, and added new Table 4-2 for the Rev. 001 files.
	Section 5.2.2	Updated the documentation to include the EAB doses following the additional decay times of 17 and 19 days, and updated the corresponding entries in Table 5-3 and Figure 5-1.
	Section 6.0	Updated the Summary of Results and Conclusions to address the new objectives added in Rev. 001.
	Section 7.0	Added new Ref. [2], the EPA PAG manual; remaining references were automatically renumbered.
	All	Minor editorial changes, as needed, including identification of the correct RG 1.183 Appendix B section (namely, Section 5.3 instead of 4.1) for the release rate from the containment to the atmosphere. Section 4.1 is for releases from the fuel building, which for BWRs is within the containment.

Table of Contents

Table of Contents
(continued)

List of Tables

List of Figures

1.0 ANALYTICAL METHODOLOGY

1.1 General

The postulated accident scenarios were based on the Alternative Source Term (AST) Methodology in Ref. [1] (RG 1.183, Appendix B). Two main configurations of the reactor building during fuel movement were considered, as follows:

- A closed containment, for the purpose of replicating the current licensing basis (in Ref. [3]) and thus provide a valid starting point for the re-analyses documented in the present calculation, and
- An open containment, for the purpose of identifying the required decay time prior to fuel movement and the various pre- and post-FHA Main Control Room (MCR) ventilation configurations that would support refueling with open containment.

It is noted that, for either configuration, i.e., open or closed containment, the analytical model and associated assumptions are applicable to a fuel assembly drop in either the reactor cavity pool or in the SFP during a full-core offload.

The analysis was based on the ELISA-2 computer code [4]. The dose conversion factors in ELISA-2 are from Federal Guidance Reports 11 and 12 (Refs. [5] and [6]). Dose rates and cumulative doses are computed for each organ, TEDE, skin and air. Of these, only the TEDE doses are presented in the main body of the calculation for comparing with regulations, which only specify TEDE limits; all other doses may be found in the ELISA-2 computer outputs, along with time-dependent dose rates.

Reference [1] (RG 1.183, Appendix B, Sec. 5.3, Footnote 3) states that the following provisions need to be implemented for refueling operations with open containment:

> "The (NRC) staff will generally require that technical specifications allowing such operations (i.e., open containment during fuel handling operations) include administrative controls to close the airlock, hatch, or open penetrations within 30 minutes. Such administrative control will generally require that a dedicated individual be present, with necessary equipment available, to restore containment closure should a fuel handling accident occur. Radiological analyses should generally not credit this manual isolation."

This provision is discussed further in Ref. [7] (NUMARC 93-01, Sec. 11.3.6.5).

1.2 Acceptance Criteria

The basic radiological acceptance criteria associated with the AST methodology are spelled out in 10 CFR 50.67, and amount to 25 rem TEDE for offsite receptors and 5 rem for control room personnel. These criteria, however, are for evaluating potential reactor accidents of exceedingly low occurrence probability and low risk of public exposure to radiation. For events with higher probability of occurrence, such as a fuel handling accident, the acceptance criteria for the offsite receptors are more stringent, while that for the control room operators remains the same. The applicable AST criteria for an FHA are as follows (Ref. 1, and 10 CFR 50.67):

Exclusion Area Boundary (EAB): 6.3 rem TEDE

Low Population Zone (LPZ): 6.3 rem TEDE

Control Room (CR): 5.0 rem TEDE

The EAB and LPZ criteria are referred to as being "well within" the regulatory limits (i.e., 25%).

The LPZ doses were not addressed in the present calculation since the release occurs within two hours, and as a result they are bounded by the corresponding doses at the EAB.

In Rev. 001, an additional objective was added to determine the required decay time that would preclude Evacuation as a protective action following an FHA. The limit for such an action is 1 rem TEDE (Ref. [2], Table 2-1).

1.3 Roundoff Errors

Some of the entries in the various tables in this calculation were generated using a Microsoft Office EXCEL-2003 spreadsheet. As a result, some results may not be exactly duplicated by hand calculation due to round-off of significant figures. Validations of the spreadsheet results are included as footnotes to the tables, where applicable.

2.0 ASSUMPTIONS

2.1 Assumptions Requiring Justification

There are no unjustified assumptions employed in the present calculation.

2.2 Justified Assumptions

Release Rate from Reactor Building

In line with Ref. [1] (RG 1.183, Appendix B, Sec. 4.1 for an FHA in the SFP, and Sec. 5.3 for an FHA in the reactor cavity pool), the radioactive material that escapes the water pool was assumed to get released to the environment over a 2-hour interval. Analytically, this was accomplished by using a building air exchange rate of 2.0 air changes per hour (ACH, equivalent to 48 air changes per day, an ELISA-2 input). This air exchange rate leads to [1.0 - exp{-2.0 (hr^{-1})*15 (min) / 60 (min/hr)}] = 39.3% of the airborne activity within the reactor building getting released within 15 minutes. Other release fractions as a function of post-FHA time are listed below.

Table 2-1: Fractional Release of Radioactivity to Atmosphere at 2 Air Changes per Hour

Post-FHA Time (min)	Activity Released to Atmosphere	Post-FHA Time (min)	Activity Released to Atmosphere
15	39.3%	60	86.5%
30	63.2%	90	95.0%
45	77.7%	120	98.2%

The 2 ACH is the RB release rate that was used in the majority of the computer runs. Four sensitivity cases (ELISA-2 Run Cases K10 through K13 in Section 5.4) make use of 1 and 0.25 ACH release rates to assess the dose impact on MCR purge initiation time.

It is noted that, for the MCR 30-day dose computations, the releases from the RB were assumed to continue for 30 days. Included in the releases beyond 2 hours are the (100 – 98.2) = 1.8% still airborne within the RB at 2 hrs (from Table 2-1), as well as the noble gases generated by the decay of iodines retained by the pool water.

MCR Filtered Intake Flow and Unfiltered Inleakage

For the cases with filtered MCR intake (postulated upgrade of HVAC system), the total exhaust flow from the MCR was assumed to be 3700 cfm, and to consist of both the intake flow and any unfiltered inleakage. For these cases, the filtered intake flow was assumed to be 3700 cfm less the unfiltered inleakage. This is conservative since it increases the fraction of unfiltered flow entering the MCR, while maintaining a fixed clean-up rate.

The unfiltered intake flows were assumed to include 10 cfm due to ingress and egress, in line with Ref. [8].

The atmospheric dispersion factors for the transport of RB releases to the MCR main intake and to the MCR inleakage location were assumed to be the same.

MCR Finite Cloud Correction

Doses to MCR personnel due to the external gamma radiation from airborne radioactivity within the MCR were adjusted using the Murphy/Campe finite-cloud correction model in Ref. 1 (Sec. 4.2.7). Even though this model tends to be non-conservative for nuclides emitting low gamma radiation, it was used in the present application in lieu of the nuclide-specific finite-cloud model in ELISA-2. This selection is immaterial since the MCR doses are primarily due to the inhalation pathway, with the submersion pathway contributing less than about 1% (from the ELISA-2 output files, summary pages, showing the percent contribution of the inhalation pathway to the total dose).

2.3 Modeling Simplifications

There are no modeling simplifications employed in the present calculation. ELISA-2 was modeled to handle the pre-FHA decay correction, the time-release from the RB, and the time-dependence of the MCR flows (as applicable). The ELISA-2 built-in logic accounts for the time-dependent generation and release of noble gases from the decay of halogens retained by the pool water, and on the MCR intake filter where applicable; these releases extend beyond the end of the 2-hr release from the RB.

3.0 DESIGN INPUT

The design input employed in the analyses, and the associated references, are summarized in Table 3-1 through Table 3-3.

Table 3-1: VYNPP – Design Input for FHA

No.	DESCRIPTION		VALUE	REFERENCE / COMMENTS
A - FHA Source Term				
A1	Power level for DBA analysis [Includes 2 % measurement uncertainty]		1950 MWt	Ref. [3]
A2	Number of assemblies in core		368	
A3	Maximum allowed radial peaking factor[(a)]		1.65	
A4	Pin failure fraction		0.571%	
A5	Equivalent number of damaged peak assemblies, and fractions thereof		2.101	Calculated: [368 assemblies * 0.571% failure fraction]
A6	Power level associated with damaged rods in peak assemblies		18.37 MWt	Calculated: [1950 MWt * 0.571% failure fraction * 1.65 Peaking Factor]
A7	Fuel rod gap fractions (AST Methodology) I-131 Kr-85 Other noble gases Other halogens Alkali metals (Cs and Rb)		 0.08 0.10 0.05 0.05 0.12	Ref. [1] (Reg. Guide 1.183, Table 3)
A8	Undecayed core inventory for radionuclides important in the evaluation of FHAs		See Table 3-2	Ref. [9], Table 4.5
A9	Post-shutdown decay time prior to postulated accident		Various	Assumed values
B - Atmospheric Release Resulting from Postulated FHA				
B1	Percent of damaged-fuel rod gap activity release		100 %	Ref. [1] (Reg. Guide 1.183, Appendix B)
B2	Overall pool decontamination factor (DF)	Noble gases	1	
		Halogens	200	
		Alkalis	Infinite	
B3	Required water depth above fuel for above DFs		23 ft	
B4	Halogen composition in airborne release (composition above pool)	Elemental	57 %	
		Organic	43 %	
B5	Reactor building configuration during refueling operations		Open	
B6	Potential release point to the atmosphere		RB blowout panel	Ref. [10]
B7	Release duration to atmosphere, and corresponding air changes per hour (ACH)	Base Case	2 hrs (2 ACH)	Ref. [1] (Reg. Guide 1.183, Appendix B)
		Certain Others	> 2 hr (0.25 and 1 ACH)	Sensitivity analyses

Table 3-1 (Continued)

VYNPP – Design Input for FHA

No.	DESCRIPTION			VALUE	REFERENCE / COMMENTS
C - Control Room Characteristics					
C1	Control room free air volume			$41534\ ft^3$	Ref. [3]
C2	MCR HVAC configuration	Nominal unfiltered intake flow for accident duration (assumed to include fresh air and air from surrounding areas as a result of ingress, egress and inleakage)		3700 cfm	Ref. [3] (Sensitivity analyses include various pre-FHA decay times ranging from 1 to 15 days) [See Sections 5.1 and 5.2]
C3	MCR HVAC configuration	With MCR isolation prior to fuel movement, and unfiltered inleakage, for accident duration		50 cfm	Minimized value for sensitivity analysis, along with various pre-FHA decay times (same as in Case C2) [See Section 5.2]
C4	MCR HVAC configuration	With operator-initiated ESF filtration (4" charcoal beds with 99% halogen removal efficiency, or 2" beds with 95% efficiency), and unfiltered inleakage		Filtration actuation at 20 and 30 minutes post FHA, and 50, 100 and 200 cfm unfiltered inleakage; 3700 cfm total exhaust.	Sensitivity analyses, along with various pre-FHA decay times ranging from 3 to 8 days [See Section 5.3]
C5	MCR HVAC configuration	With MCR isolation prior to fuel movement, followed by post-FHA purge		100 and 150 cfm unfiltered inleakage, various RB release rates, time-shifted atmospheric dispersion factors, and various purge initiation times	Sensitivity analyses, with 5-day pre-FHA decay time [See Section 5.4]
D - Other Variables					
D1	Atmospheric dispersion factors from release point to locations of interest			See Table 3.3	Ref. [3], and Ref. [10], Section 6
D2	Breathing rates	Control Room	0 - 720 hrs	$3.5E\text{-}04\ m^3/sec$	Ref. [1], pg 1.183-18
		EAB	0 - 2 hr	$3.5E\text{-}04\ m^3/sec$	Ref. [1], pg 1.183-16
D3	Control room occupancy factors		0 - 24 hrs	1.0	Ref. [1], pg 1.183-18
			24 - 96 hrs	0.6	
			96 - 720 hrs	0.4	
D4	Exposure Intervals[(b)]	Control room		30 days	Ref. [1], Sections 4.1.3, 4.1.5 and 4.2.6
		EAB		2 hrs	
D5	Regulatory dose limits	Control room TEDE		5 rem	Ref. [1], pg 1.183-19, and Ref. [11] , Sec. (b)(2)(iii)
		EAB TEDE		6.3 rem	Ref. [1], Table 6
		LPZ TEDE		6.3 rem	
D6	PAG Evacuation dose limit (EAB TEDE)			1 rem	Ref. [2], Table 2-1

(a) Item #A3: In line with RG 1.183 (Ref. [1], Sec. 3.1), the radial peaking factor is applied to the average fuel-assembly inventory based on the core inventory in Table 3-2. This is a conservative approach and bounds any potential variations in the FHA source term resulting from variations in the EFPDs and burnup in any given cycle.

(b) Item #D4: Even though all radioactivity is released to the atmosphere within 2 hours following a design-basis FHA, the exposure intervals for the CR personnel was assumed to be 30 days. This provides adequate time for cleanup of the airborne radioactivity still present within the CR after termination of the 2-hr release, and also accounts for the delayed release of noble-gas decay products from the refueling pool water produced upon decay of halogens retained therein.

Table 3-2: VYNPP Undecayed Core Inventory for Radionuclides Important in the Radiological Evaluation of DBAs

(From Ref. [9], Table 4.5, based on 1950 MWt, an enrichment range from 3.0 to 4.65 wt % U-235, and core-average burnup from 5 to 58 GWD/MTU)

Nuclide	Core Ci	Nuclide	Core Ci
Br-83	8.267E+06	I-132	7.900E+07
Kr-83m	8.265E+06	Te-133	6.602E+07
Br-85	1.874E+07	Te-133m	4.493E+07
Kr-85	9.852E+05	I-133	1.130E+08
Kr-85m	1.894E+07	Xe-133	1.128E+08
Rb-86	2.496E+05	Xe-133m	3.428E+06
Kr-87	3.788E+07	Te-134	1.036E+08
Kr-88	5.355E+07	I-134	1.254E+08
Kr-89	6.755E+07	Cs-134	2.971E+07
Sr-89	6.724E+07	I-135	1.051E+08
Sr-90	7.999E+06	Xe-135	4.540E+07
Y-90	8.363E+06	Xe-135m	2.232E+07
Sr-91	8.684E+07	Cs-136	7.602E+06
Y-91	8.270E+07	Xe-137	9.893E+07
Sr-92	8.987E+07	Cs-137	1.186E+07
Y-92	9.008E+07	Ba-137m	1.124E+07
Y-93	9.857E+07	Xe-138	9.851E+07
Zr-95	9.645E+07	Ba-139	1.043E+08
Nb-95	9.673E+07	Ba-140	1.004E+08
Zr-97	9.596E+07	La-140	1.009E+08
Mo-99	1.034E+08	La-141	9.573E+07
Tc-99m	9.051E+07	Ce-141	9.255E+07
Ru-103	9.889E+07	La-142	9.387E+07
Ru-105	7.844E+07	Ce-143	9.228E+07
Rh-105	7.183E+07	Pr-143	9.181E+07
Ru-106	5.554E+07	Ce-144	7.268E+07
Sb-127	7.194E+06	Nd-147	3.736E+07
Te-127	7.151E+06	Np-239	1.496E+09
Te-127m	9.705E+05	Pu-238	7.668E+05
Sb-129	1.976E+07	Pu-239	2.864E+04
Te-129	1.947E+07	Pu-240	6.061E+04
Te-129m	2.890E+06	Pu-241	1.281E+07
Te-131m	8.405E+06	Am-241	1.702E+04
I-131	5.564E+07	Cm-242	6.669E+06
Xe-131m	6.192E+05	Cm-244	2.358E+06
Te-132	7.739E+07		

Table 3-3: Atmospheric Dispersion Factors for the Postulated FHA

(From Ref. [3], and Ref. [10], Section 6)

No.	Release Point	Receptor Point	Post-FHA Interval	χ/Q (sec/m^3)
1	Main stack (Calculation of record with closed containment, for confirmatory analyses)	EAB	Instantaneous release	1.35E-04
2		Control Room Fresh Air Intake	Instantaneous release	6.04E-05
3	RB blowout panel, for present application with open containment	EAB	0 - 2 hrs	1.69E-03
4		Control Room Fresh Air Intake	0 - 2 hrs	5.89E-03
			2 - 8 hrs	1.53E-03
			8 - 24 hrs	6.41E-04
			24 - 96 hrs	6.64E-04
			96 - 720 hrs	5.10E-04

Note: The atmospheric dispersion factors in this table were based on the analysis of a combined 5-years' worth of hourly meteorological data collected on site (1995-1999), and were used in the AST implementation at VYNPP. According to Reg. Guide 1.183 (Ref. [1], Section 5.3, Meteorology Assumptions), *"Atmospheric dispersion values (χ/Q) for the EAB, the LPZ, and the control room that were approved by the staff during initial facility licensing or in subsequent licensing proceedings may be used in performing the radiological analyses identified in this guide."* The FHA analysis is addressed in Appendix B of the guide.

4.0 SOFTWARE AND COMPUTER FILES

4.1 Software

Computation of the EAB and MCR radiological consequences for the postulated FHA were based on the ELISA-2 computer code (Ref. [4], Version 2.4) for all the analyzed scenarios. ELISA-2 is in the AREVA NP Engineering Application Software Index (EASI). There are no software errors that affect its application in the present calculation.

All ELISA-2 runs were carried out on the following platforms:

Rev. 000: HP 9000/785 CPU running HP UX B.10.20.

Rev. 001: HP 9000/800 CPU running HP UX B.11.11

4.2 Computer Files

The input and output files for the computer runs associated with the present document are available on the AREVA ColdStor System, under the following folders:

Rev. 000: \cold\General-Access\32\32-9000000\32-9145461-000\official

Rev. 001: \cold\ General-Access\32\32-9000000\32-9145461-001\official

The list of files in these folders are presented in Table 4-1 and Table 4-2. It is noted that each listed case has a number of stacked sub-cases, bringing the total number of analyzed scenarios to 72 for Rev. 000 and to 3 for Rev. 001.

Table 4-1: List of Computer Files Transferred to the AREVA ColdStor Server for Rev. 000

File Name	Size (Bytes)	Date/Time
ELISA-2 Input (*.el1) and Output (*.el2) Files for Section 5.1 (Confirmatory Analyses - Closed Containment and Elevated Release)		
eli2-VY-FHA-A.el1	6430	Sep 23 2010 16:01:04
eli2-VY-FHA-A.el2	157935	Sep 23 2010 16:01:12
ELISA-2 Input and Output Files for Section 5.2 (FHA with Open Containment, Ground-Level Release and Existing MCR Design)		
eli2-VY-FHA-B.el1	8918	Sep 21 2010 09:22:42
eli2-VY-FHA-B.el2	330662	Sep 21 2010 09:29:58
eli2-VY-FHA-C.el1	10989	Sep 24 2010 09:50:42
eli2-VY-FHA-C.el2	398144	Sep 24 2010 09:51:38
eli2-VY-FHA-D.el1	10980	Sep 24 2010 09:51:16
eli2-VY-FHA-D.el2	398144	Sep 24 2010 09:51:28

Table 4-1 (Continued)

List of Computer Files Transferred to the AREVA ColdStor Server for Rev. 000

File Name	Size (Bytes)	Date/Time
ELISA-2 Input and Output Files for Section 5.3 (FHA with Open Containment, Ground-Level Release and MCR Upgraded Design)		
eli2-VY-FHA-E.el1	9646	Sep 24 2010 11:41:04
eli2-VY-FHA-E.el2	257679	Sep 24 2010 11:41:14
eli2-VY-FHA-F.el1	6417	Sep 24 2010 12:24:02
eli2-VY-FHA-F.el2	174641	Sep 24 2010 12:24:12
eli2-VY-FHA-G.el1	9670	Sep 24 2010 11:43:50
eli2-VY-FHA-G.el2	257679	Sep 24 2010 11:44:06
eli2-VY-FHA-H.el1	6425	Sep 24 2010 12:24:48
eli2-VY-FHA-H.el2	174641	Sep 24 2010 12:24:54
eli2-VY-FHA-I.el1	9670	Sep 24 2010 11:47:26
eli2-VY-FHA-I.el2	257679	Sep 24 2010 11:47:36
eli2-VY-FHA-J.el1	6425	Sep 24 2010 12:25:24
eli2-VY-FHA-J.el2	174641	Sep 24 2010 12:25:32
ELISA-2 Input and Output Files for Section 5.4 (FHA with Open Containment, Ground-Level Release, Pre-FHA MCR Isolation and Post-FHA MCR Purge)		
eli2-VY-FHA-K.el1	19053	Sep 23 2010 13:53:52
eli2-VY-FHA-K.el2	717635	Sep 23 2010 13:54:44

Table 4-2: List of Computer Files Transferred to the AREVA ColdStor Server for Rev. 001

File Name	Size (kBytes)	Date/Time
ELISA-2 Input and Output Files for Section 5.2 (FHA with Open Containment, Ground-Level Release, Additional Decay Times for EAB Dose – Supplements Case B in Table 4-1)		
eli2-VY-FHA-B1.el1*	3346	Oct 23 2013 13:58:00
eli2-VY-FHA-B1.el2*	127291	Oct 23 2013 14:51:44

* Includes Case B8 as a confirmatory subcase duplicating the results in Rev. 000 (files el2-VY-FHA-B.el*)

5.0 ANALYSIS AND RESULTS

Presented in the subsections which follow are the following FHA evaluations:

- Confirmatory analyses for comparison with the calculation of record (with closed containment and elevated release),
- Analyses similar to those in the calculation of record, but for a ground-level release, with extended pre-FHA decay times to determine the acceptable scenario that would meet the dose acceptance criteria at all receptors of interest (EAB and MCR), along with a sensitivity runs with MCR isolation prior to fuel movement,
- Sensitivity analyses under the assumption that the MCR HVAC system will be upgraded to include emergency ESF filtration of the intake flow, and
- Sensitivity analyses with pre-FHA MCR isolation and post-FHA MCR purge (without ESF filtration).

Details follow. It is noted that the description which follows deals with an FHA taking place in the reactor cavity pool. Nonetheless, the analytical model and associated assumptions are applicable to a fuel assembly drop in either the reactor cavity pool or in the SFP during a full-core offload.

5.1 Confirmatory Analyses – Closed Containment and Elevated Release

5.1.1 Accident Scenario

Pertinent assumptions employed in the calculation of record (Ref. [3]) are as follows:

(a) The reactor has been operating at full power (1950 MWt) for an extended period of time.

(b) The reactor is shutdown, refueling operations are initiated and an FHA takes place at either 24 hours after shutdown (all rods in)[1], or at 96 hours.

(c) The accident was assumed to involve the dropping of an assembly onto other assemblies, leading to the equivalent clad failure of the fuel rods in 2.101 assemblies (from Table 3-1, Item #A5). All failed rods are peak powered, with a radial peaking factor of 1.65 (from Table 3-1, Item #A3).

(d) All activity within the gaps of the failed fuel rods is released to the reactor cavity pool. The released activity corresponds to 8% of the entire inventory of I-131 in the rods (i.e., within the fuel matrix and gaps), 10% of the Kr-85, 5% of the remaining halogens and noble gases, and 12% of the alkalis (Cs and Rb), from Table 3-1, Item #A7. The activity released from the damaged fuel rods is presented in Table 5-1.

(e) All the noble gases and $(1/200)^{th}$ of the halogens escape from the pool and are released to the refueling level. All the alkalis are retained by the pool. The halogen composition above the pool is 57 % in elemental form and 43 % in organic form[2], from Table 3-1, Items #B2 and #B4.

[1] Fuel movement initiation at 96 hours after shutdown is more typical (Ref. [3]).

(f) In line with Ref. [1] (Appendix B, Sec. 5.3), the radioactive material that escapes the reactor cavity pool was assumed to get released to the environment over a 2-hour interval. As a simplification, the calculation of record assumed the release to the atmosphere to be instantaneous; the same assumption was made in the confirmatory analyses.

(g) The reactor building was assumed to be closed during the refueling operations, such that all releases to the environment would be via the main stack, with no credit for any filtration by the SGTS system, or any in-transit decay and plateout. A sensitivity analysis assuming a ground-level release was also evaluated in the MCR habitability, in both the calculation of record and the current calculation.

(h) Transport of the released radioactivity to the receptors of interest is dictated by the applicable atmospheric dispersion factors in Table 3-3.

(i) The MCR ventilation system was assumed to remain in the normal operating mode during the entire exposure interval (30 days). The air intake flow is 3700 cfm (includes additional flow from surrounding areas as a result of ingress, egress and leakage) and is unfiltered.

(j) Breathing rates and MCR occupancy factors are as given in Table 3-1, Items #D2 and #D3.

(k) The control room operators were assumed to be located at the base of a hemispherical cloud having a volume equal to the free air volume of the control room. Finite-cloud correction to the submersion dose was based on the Murphy/Campe equation in Reg. Guide 1.183 (Sec. 4.2.7).

2 The halogen composition has no impact in this analysis since the filtration efficiencies in the sensitivity cases with ESF filters was assumed to be the same for all halogen species.

Table 5-1: Undecayed Activity Available for Release from FHA Damaged Fuel Rods

Nuclide	Core Inventory (Ci)	Specific Activity (Ci / MWt)	Activity within Failed Peak-Powered Fuel Rods (Ci)	Gap Fraction[a]	FHA Source Term (Ci Available for Release)
Kr-83m	8.265E+06	4.238E+03[b]	7.786E+04[c]	0.05	3.893E+03[d]
Kr-85	9.852E+05	5.052E+02	9.281E+03	0.10	9.281E+02
Kr-85m	1.894E+07	9.713E+03	1.784E+05	0.05	8.921E+03
Kr-87	3.788E+07	1.943E+04	3.568E+05	0.05	1.784E+04
Kr-88	5.355E+07	2.746E+04	5.045E+05	0.05	2.522E+04
Kr-89	6.755E+07	3.464E+04	6.364E+05	0.05	3.182E+04
Xe-131m	6.192E+05	3.175E+02	5.833E+03	0.05	2.917E+02
Xe-133	1.128E+08	5.785E+04	1.063E+06	0.05	5.313E+04
Xe-133m	3.428E+06	1.758E+03	3.229E+04	0.05	1.615E+03
Xe-135	4.540E+07	2.328E+04	4.277E+05	0.05	2.138E+04
Xe-135m	2.232E+07	1.145E+04	2.103E+05	0.05	1.051E+04
Xe-137	9.893E+07	5.073E+04	9.320E+05	0.05	4.660E+04
Xe-138	9.851E+07	5.052E+04	9.280E+05	0.05	4.640E+04
Br-83	8.267E+06	4.239E+03	7.788E+04	0.05	3.894E+03
Br-85	1.874E+07	9.610E+03	1.765E+05	0.05	8.827E+03
I-131	5.564E+07	2.853E+04	5.242E+05	0.08	4.193E+04
I-132	7.900E+07	4.051E+04	7.442E+05	0.05	3.721E+04
I-133	1.130E+08	5.795E+04	1.065E+06	0.05	5.323E+04
I-134	1.254E+08	6.431E+04	1.181E+06	0.05	5.907E+04
I-135	1.051E+08	5.390E+04	9.901E+05	0.05	4.950E+04
Rb-86	2.496E+05	1.280E+02	2.351E+03	0.12	2.822E+02
Cs-134	2.971E+07	1.524E+04	2.799E+05	0.12	3.359E+04
Cs-136	7.602E+06	3.898E+03	7.161E+04	0.12	8.594E+03
Cs-137	1.186E+07	6.082E+03	1.117E+05	0.12	1.341E+04
Te-131m[e]	8.405E+06	4.310E+03	7.918E+04	0.08	6.334E+03
Te-132[e]	7.739E+07	3.969E+04	7.291E+05	0.05	3.645E+04
Te-133[e]	6.602E+07	3.386E+04	6.219E+05	0.05	3.110E+04
Te-133m[e]	4.493E+07	2.304E+04	4.233E+05	0.05	2.116E+04

(a) The gap fractions are from Table 3-1, Item #A7.

(b) Kr-83m specific activity: 8.265E+06 (Ci) / 1950 (MWt) = 4.238E+03 (Ci/MWt)

(c) Kr-83m activity in failed rods: 4.238E+03 (Ci/MWt) * 18.37 (MWt, from Table 3-1, Item #A6) = 7.786E+04 (Ci)

(d) Kr-83m FHA source term: 7.786E+04 (Ci) * 0.05 (gap fraction) = 3.893E+03 (Ci)

(e) The Te isotopes were used only in the confirmatory analysis, for consistency with the calculation of record (Ref. [3]). In line with Ref. [1], Reg. Guide 1.183, Appendix B, Section 1.2, only the nobles, halogens and alkalis are to be assumed as present in the fuel rod gaps.

5.1.2 Radiological Consequences and Comparison with Calculation of Record

The dose consequences for the FHA scenario described in this subsection are presented in Table 5-2. It is seen that there is very good agreement between the calculation of record and the analyses documented in the present calculation. The only relatively large difference (about 10%, which is still acceptable) is due to the underestimation of the Xe-135m contribution to the dose, resulting from this noble-gas isotope not being identified in the calculation of record as a daughter product of I-135 in the decay correction. The corresponding EAB dose in Case A3 is not impacted due to the long pre-FHA decay time for this isotope. The MCR doses in all cases are not affected by the Xe-135m underestimation primarily due to the finite-cloud correction to the submersion dose.

Table 5-2: FHA Re-Analysis – Comparison with Calculation of Record

[From ELISA-2 output file eli2-VY-FHA-A.el2, last page of each case analyzed]

ELISA-2 Run Case	Receptor	Release Point	Decay Time (days)	TEDE Dose (rem)		
				ELISA-2 (A)	Calc. of Record (B)	Percent Difference[(a)]
A1	EAB	Elevated	1	0.5215	0.472	-10.5
A2	MCR	Elevated	1	0.1555	0.153	-1.6
A3	EAB	Elevated	4	0.2771	0.274	-1.1
A4	MCR	Elevated	4	0.1081	0.107	-1.0
A5	MCR	Ground-Level	14	4.387	<5	N/A

(a) Percent difference = [(B - A) / B]*100

5.2 FHA with Open Containment, Ground-Level Release and Existing MCR Design

5.2.1 Accident Scenario

Assumptions associated with this accident scenario are as follows:

(a) The reactor has been operating at full power (1950 MWt) for an extended period of time.

(b) The reactor is shutdown, refueling operations are initiated and an FHA takes place at various assumed decay times after reactor shutdown, ranging from 1 to 19 days for the EAB dose and from 1 to 15 for the MCR dose.

(c)-(e) See Section 5.1.1 for details on the FHA source term. It is noted, however, that the Te isotopes in Table 5-1.were not included, as clarified in the table notes.

(f) In line with Ref. [1] (Appendix B, Sec. 5.3), the radioactive material that escapes the reactor cavity pool was assumed to get released to the environment over a 2-hour interval, based on 2 air changes per hour. See Section 2.2 for further details.

(g) The reactor building was assumed to be open during the refueling operations, with all post-FHA releases to the environment assumed to be at ground level, via the RB blowout panels.

(h) Transport of the released radioactivity to the receptors of interest is dictated by the applicable atmospheric dispersion factors in Table 3-3 for ground-level releases.

(i) Two MCR ventilation configurations were assumed: (1) Normal operating mode during the entire exposure interval (30 days), with an intake flow of 3700 cfm, unfiltered, and (2) Fully isolated prior to fuel movement, with an unfiltered inleakage of 50 cfm. [Note: The 50-cfm unfiltered inleakage requires a very tight MCR envelope. It was selected solely for demonstrative purposes.]

(j)-(k) See Section 5.1.1 for details on the breathing rate, MCR occupancy factors and MCR finite-cloud correction to the submersion dose.

5.2.2 Radiological Consequences

The dose consequences for the FHA scenario described in this subsection are presented in Table 5-3 and shown graphically in Figure 5-1 and Figure 5-2. It is seen that the EAB TEDE dose with open containment is less than the regulatory limit of 6.3 rem limit even with a 24-hr pre-FHA decay time. Also, a decay time of 17 days is required for this dose to drop below the EPA PAG Evacuation limit of 1 rem TEDE.

On the other hand, a decay time of about 11 to 13 days is needed to achieve an MCR dose less than 5 rem, for intake flows ranging between 50 cfm (with the MCR pre-isolated) and 3700 cfm (current MCR configuration).

It is noted that simple pre-isolation of the MCR does not provide much relief. This is because the MCR cleanup rate is also reduced. Thus even though much less activity enters the MCR envelope, it stays therein for a much longer time. Examples of the time dependence of the MCR cumulative dose are shown in Table 5-4 and Figure

5-3. These results suggest the possibility of reducing the pre-FHA decay time by MCR isolation prior to fuel movement, followed by MCR purge a few hours after the postulated FHA; see Section 5.4 for further details.

Table 5-3: VY FHA - EAB and MCR TEDE Dose vs. Decay Time

[Ground-level release with open containment at 2 air changes per hour]

Decay Time (days)	EAB TEDE Dose (rem)	MCR TEDE Dose (rem) vs. Intake Flow	
		3700 cfm	50 cfm
1	5.895	14.48	12.47
3	3.643	11.14	9.834
5	2.953	9.185	8.152
7	2.451	7.691	6.835
9	2.042	6.464	5.746
11	1.705	5.437	4.834
13	1.424	4.575	4.067
15	1.190*	3.849	3.422
17	0.9957	N/A	N/A
19	0.8333	N/A	N/A
ELISA-2 Run Case (last output page of each case analyzed)	eli2-VY-FHA-B.el2 (Subcases B1-B8) and eli2-VY-FHA-B1.el2 (Subcases B9-B10)	eli2-VY-FHA-C.el2 (Subcases C1-C8)	eli2-VY-FHA-D.el2 (Subcases D1-D8)

(See Figure 5-1 and Figure 5-2 for graphical presentations)

* The EAB dose at 15 days (1.190 rem TEDE, from eli2-VY-FHA-B.el2) was reproduced in eli2-VY-FHA-B1.el2 (for Rev. 001) to confirm consistency in the input files and continuity of results.

Table 5-4: Time Dependence of MCR Cumulative Dose – Sample Cases

Post-FHA Time (hr)	MCR Cumulative TEDE Dose (rem)	
	13-day decay, 3700 cfm intake	11-day decay, 50 cfm inleakage
0.0	0.0	0.0
0.1	0.1956	0.003727
0.25	0.8733	0.02111
0.5	2.106	0.07238
1	3.651	0.2199
1.5	4.271	0.3906
2	4.502	0.5654
4	4.574	1.214
6	4.575	1.773
8	4.575	2.253
12	4.575	3.019
24	4.575	4.312
45	4.575	4.728
96	4.575	4.832
720	4.575	4.834
ELISA-2 Run	eli2-VY-FHA-C.el2, output page 106 (Case C7)	eli2-VY-FHA-D.el2, output page 95 (Case D6)

(See Figure 5-3 for graphical presentations)

Figure 5-1: VY FHA - EAB TEDE Dose vs. Decay Time

(Ground-level release with open containment at 2 air changes per hour)

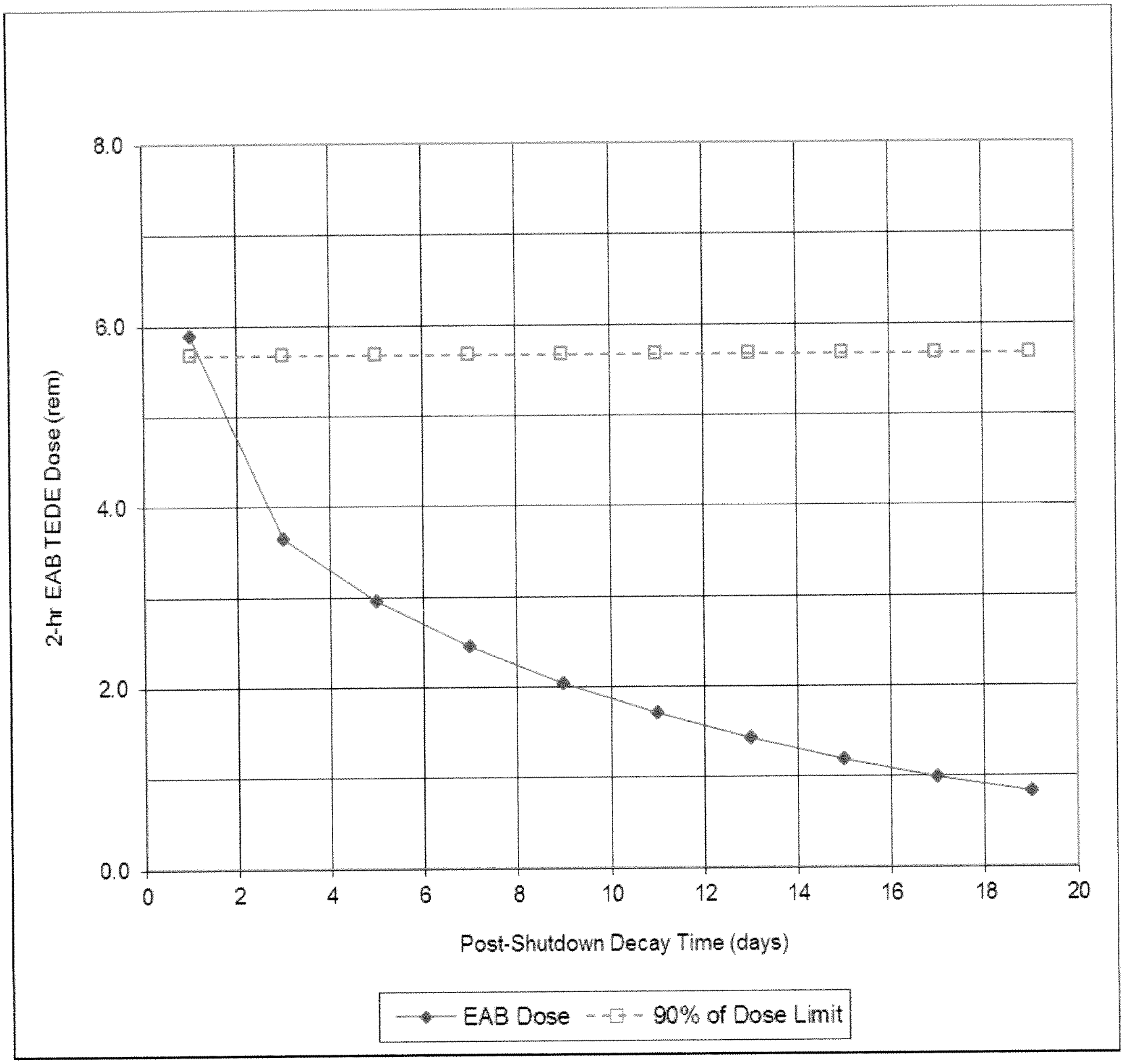


Figure 5-2: VY FHA - MCR TEDE Dose vs. Decay Time and Intake/Inleakage Flow Rate

(Ground-level release with open containment at 2 air changes per hour)

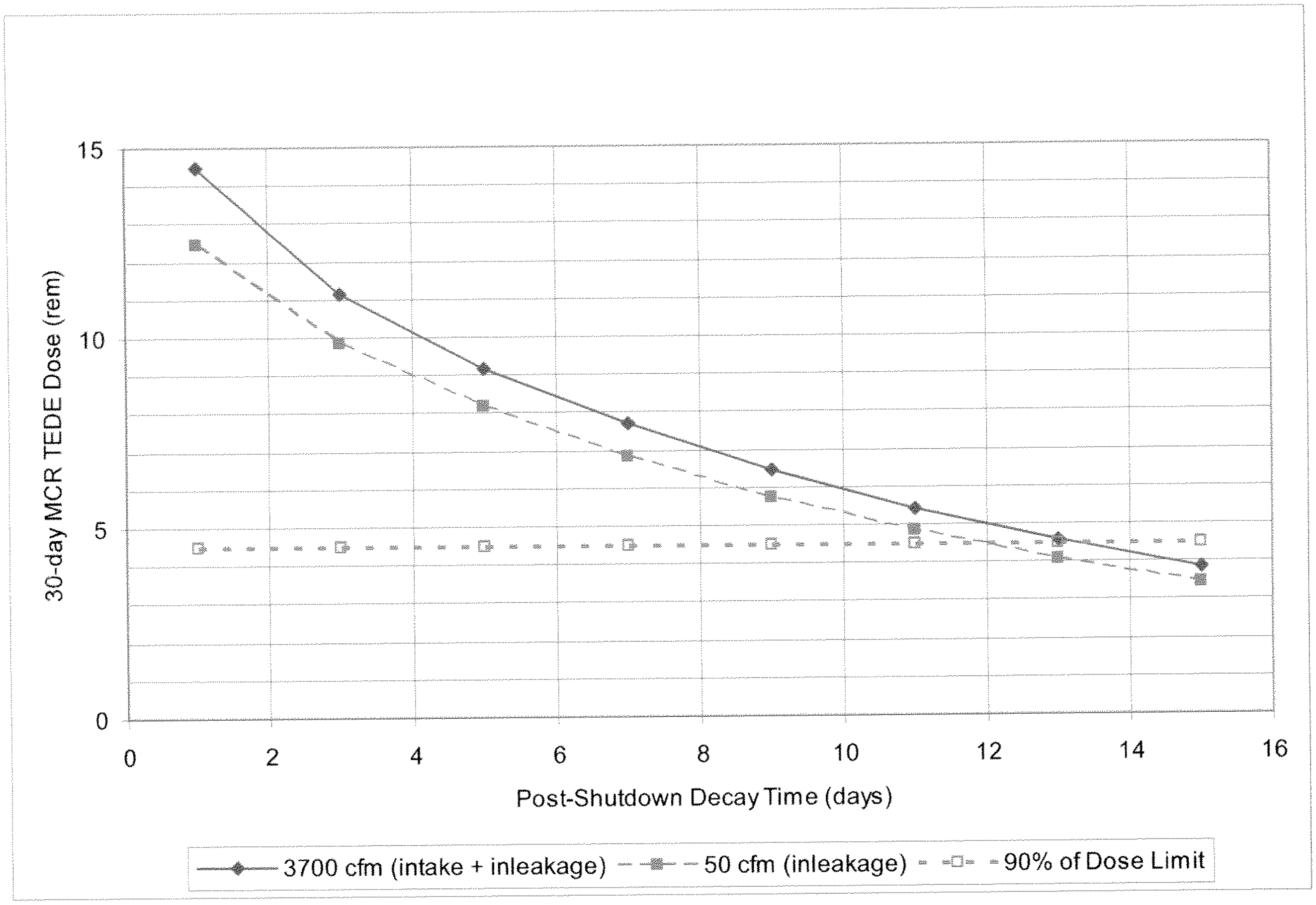
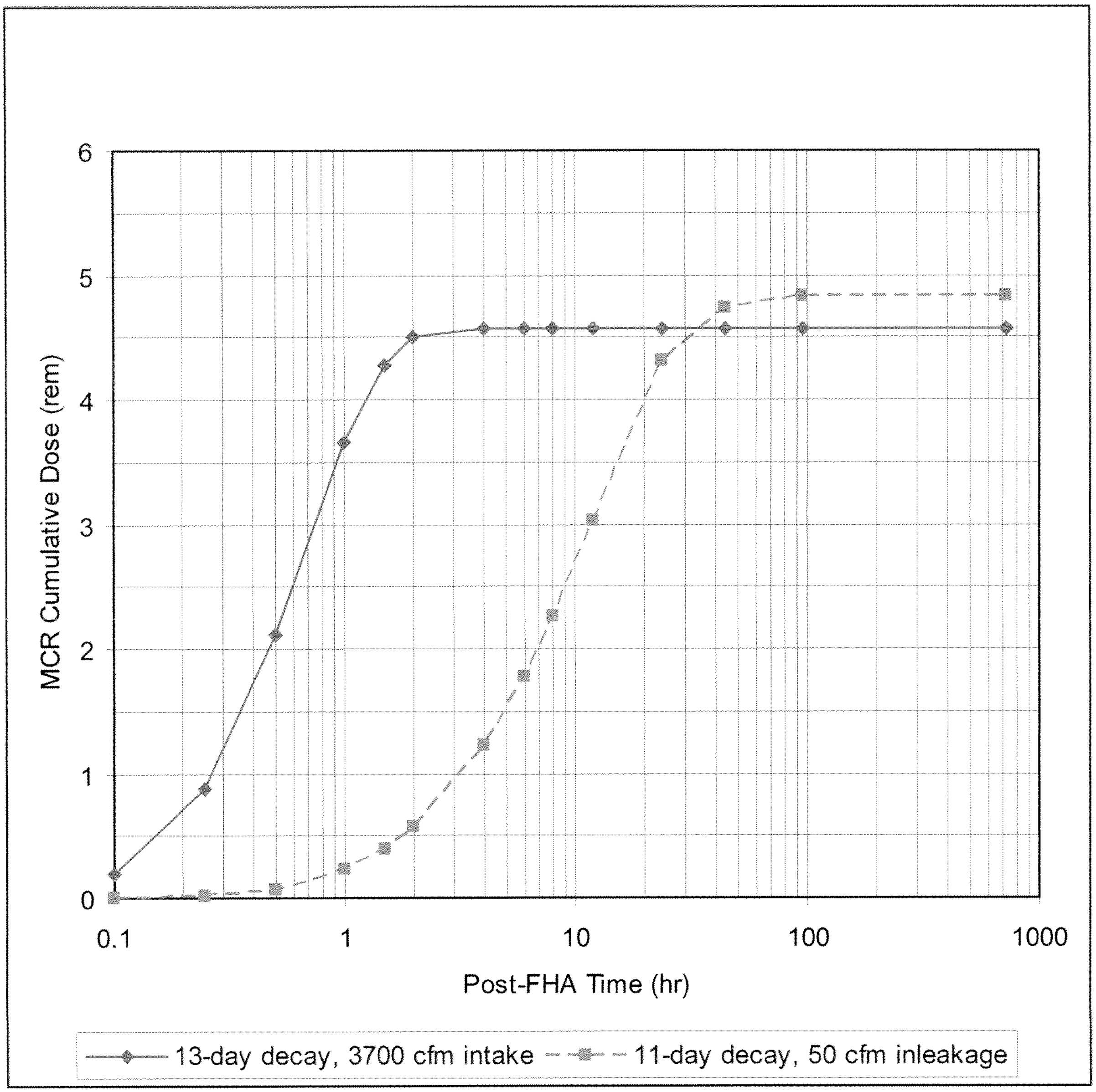


Figure 5-3: Time Dependence of MCR Cumulative TEDE Dose – Sample Cases

[Note the difference in decay times between the two cases.]



5.3 FHA with Open Containment, Ground-Level Release and MCR Upgraded Design

This subsection evaluates the benefits of upgrading the MCR HVAC system to include charcoal intake filters.

5.3.1 Accident Scenario

The associated assumptions are as listed in Section 5.2.1, with the following exceptions:

(1) Based on scoping analyses, the pre-FHA decay times were assumed to range from 4 to 8 days.

(2) The MCR ventilation configuration was assumed to be in the normal mode at the time of the accident, drawing in 3700 cfm of outside air. At either 20 or 30 minutes after the accident (both cases analyzed), emergency filtration is actuated diverting the intake flow through 2-inch or 4-inch charcoal beds (with 95% or 99% efficiency, respectively, for the removal of iodines and particulates). Unfiltered inleakage is also assumed, at 50, 100 or 200 cfm, as part of the total intake flow of 3700 cfm (i.e., the exhaust flow is 3700 cfm).

5.3.2 Radiological Consequences

The FHA dose consequences with the postulated upgrade to the MCR HVAC system are presented in Table 5-5. It is seen that the MCR TEDE doses can be maintained below the 5 rem TEDE limit under the following alternative conditions:

(a)	Decay prior to fuel movement:	5 days
	Unfiltered inleakage:	50 to 200 cfm
	Filtration efficiency:	99% (4" charcoal beds)
	Delay in emergency filtration actuation:	20 minutes
(b)	Decay prior to fuel movement:	5-6 days (depending on unfiltered inleakage)
	Unfiltered inleakage:	50 to 200 cfm
	Filtration efficiency:	95% (2" charcoal beds)
	Delay in emergency filtration actuation:	20 minutes
(c)	Decay prior to fuel movement:	8 days
	Unfiltered inleakage:	50 to 200 cfm
	Filtration efficiency:	99% (4" charcoal beds)
	Delay in emergency filtration actuation:	30 minutes

The 20-minute delay time in the MCR filtration actuation was based on SRP Section 6.4 (Ref. [8]), while the 30-min delay is what is typically allocated for operator action. The 20-min delay is expected to be acceptable for an FHA, and provides a significant reduction in the decay time prior to fuel movement.

Table 5-5: VY FHA - MCR TEDE Dose with ESF Filtration vs. Decay Time and Unfiltered Inleakage

[Ground-level release with open containment at 2 air changes per hour; MCR total exhaust rate of 3700 cfm, continuous; from ELISA-2 output files eli2-VY-FHA-E.el2 through eli2-VY-FHA-J.el2, last page of each case analyzed.]

Pre-FHA Decay Time (days)	MCR Dose (TEDE rem) vs. Unfiltered Inleakage		
	50 cfm	100 cfm	200 cfm
MCR ESF Filtration Actuation in 20 minutes (99% charcoal filters)			
4	**5.114**	**5.182**	**5.318**
5	4.659	4.721	4.845
6	4.259	4.316	4.429
ELISA-2 Run Cases	E1-E3	G1-G3	I1-I3
MCR ESF Filtration Actuation in 20 minutes (95% charcoal filters)			
4	**5.314**	**5.380**	**5.510**
5	4.842	4.901	**5.020**
6	4.426	4.481	4.590
ELISA-2 Run Cases	E4-E6	G4-G6	I4-I6
MCR ESF Filtration Actuation in 30 minutes (99% charcoal filters)			
5	**5.981**	**6.025**	**6.113**
6	**5.468**	**5.508**	**5.588**
7	**5.007**	**5.044**	**5.118**
8	4.589	4.623	4.690
ELISA-2 Run Cases	F1-F4	H1-H4	J1-J4

(Bold entries are for MCR TEDE doses that are above the 5 rem regulatory limit)

5.4 FHA with Open Containment, Ground-Level Release, Pre-FHA MCR Isolation and Post-FHA MCR Purge

This subsection evaluates the potential of ensuring the MCR habitability following an FHA based on isolation of the MCR prior to fuel movement, followed by purge of the MCR atmosphere a few hours after an FHA has taken place. These analyses are to demonstrate feasibility. The capability to identify less contaminated air at the MCR air intake than that within the MCR as a pre-condition for the purge credit would likely be required.

5.4.1 Accident Scenario

The associated assumptions are as listed in Section 5.2.1, with the following exceptions:

(1) Based on scoping analyses, the following scenario (which meets the MCR dose limit of 5 rem TEDE) was selected as a base case:

- Pre-FHA decay of 5 days.
- Refueling operations with open containment, vented at the rate of 2 air changes per hour (the assumed value described in Sec. 2.2), with no delay in the start of the atmospheric release following the accident.
- MCR isolation prior to fuel movement, with a total unfiltered intake flow plus unfiltered inleakage of 150 cfm (where the intake flow would be for maintaining an acceptable CO_2 level within the MCR envelope).
- MCR purge at 3.5 hours after the postulated FHA, at the normal flow rate of 3700 cfm.

(2) Sensitivity analyses were then carried out to determine acceptable variations in the following parameters:

- Purge delay,
- Release delay from the reactor building, along with or without associated time shifting of the accident atmospheric dispersion factor,
- MCR unfiltered inleakage,
- Reduced release rate from the reactor building (with no delay in the release initiation), and
- Time shifting of the atmospheric dispersion factor for a reduced release rate from the Reactor Building.

5.4.2 Radiological Consequences

The dose consequences of the various FHA scenarios evaluated are presented in Table 5-6. The following are noted:

(a) The base-case dose is about half of that without purge. Delaying the purge by 30 minutes increases the dose by about 0.5 rem to a value beyond the acceptance criterion.

(b) Should the radioactivity released from the failed fuel rods get retained within the RB atmosphere for more than 3 hours, the MCR dose rate would exceed the limit. This implies that purge initiation needs to be coupled to the start of the release from containment.

(c) A reduced MCR unfiltered inleakage would result in a lower dose.

(d) A reduced RB release rate would also result in a lower dose.

(e) For a slow RB release rate, time-shifting of the atmospheric dispersion factor does not lead to a significant change in the overall dose.

In general, pre-isolation of the MCR followed by MCR purge appears to be a viable option. However, it may only be acceptable as an interim measure for refueling operations with open containment, with the premise that suitable improvements will be made to the MCR HVAC system to mitigate the radiological consequences of an FHA.

Table 5-6: VY FHA - MCR TEDE Dose with 5-Day Decay, Pre-FHA Isolation and Post-FHA Purge

[Ground-level release with open containment at various air changes per hour, RB release delay, and χ/Q time shifting; from ELISA-2 output file eli2-VY-FHA-K.el2, last page of each case analyzed.]

ELISA-2 Run CASE	RB Air Changes /hr	MCR Unfiltered Inleakage (cfm)	3700-cfm MCR Purge Initiation Time (hr)	RB Release Delay	χ/Q Time Shift	MCR TEDE Dose (rem)
Impact of No Purge or Purge Delay						
K1	2	150	No purge	no	no	**9.014**
K2	2	150	4	no	no	**5.036**
K3 (Base Case)	**2**	**150**	**3.5**	**no**	**no**	4.571
Impact of RB Release Delay and χ/Q Time Shifting						
K5	2	150	3.5	no	yes, 1 hr	1.542
K6	2	150	3.5	yes, 1 hr	yes, 1 hr	3.472
K7	2	150	3.5	yes, 2 hr	yes, 2 hr	2.422
K8	2	150	3.5	yes, 3 hr	yes, 3 hr	3.831
K9[(a)]	2	150	3.5	yes, 3.5 hr	yes, 3.5 hr	**9.064**
Impact of Reduced MCR Unfiltered Inleakage						
K3 (Base Case)	**2**	**150**	**3.5**	**no**	**no**	4.571
K4	2	100	3.5	no	no	3.388
Impact of Reduced Release Rate from Reactor Building						
K3 (Base Case)	**2**	**150**	**3.5**	**no**	**no**	4.571
K10	1	150	3.5	no	no	3.930
K11	0.25	150	3.5	no	no	2.516
Impact of χ/Q Time Shifting with a Slow Release Rate from Reactor Building						
K11	0.25	150	3.5	no	no	2.516
K12	0.25	150	3.5	no	yes, 2 hr	1.887
K13	0.25	150	3.5	no	yes, 3.5 hr	2.422

(a) Case K9 is equivalent to no pre-FHA MCR isolation, and a total decay of 5 days + 3.5 hrs.

6.0 SUMMARY OF RESULTS AND CONCLUSIONS

The present calculation dealt with the radiological evaluation of a design-basis fuel handling accident (FHA) taking place at the Vermont Yankee Nuclear Power Plant , based on the AST methodology (Ref. [1]), an open containment, and ground-level releases. It is noted that, regardless of the containment configuration (open or closed), the FHA analytical model and associated assumptions are applicable to a fuel assembly drop in either the reactor cavity pool or in the SFP during a full-core offload.

The starting point of the analysis was confirmation of the FHA results in the VYNPP calculation of record, as described in Section 5.1. This was followed by different scenario sets to evaluate the following:

- The minimum required decay time after reactor shutdown that would ensure the MCR habitability under its current configuration during normal operation, and also under a pre-isolation condition,
- The benefits of upgrading the MCR HVAC system to include emergency filtration of the intake flow, and
- The viability of MCR isolation prior to fuel movement, and MCR purge following an FHA.

Reference is made to Sections 5.2, 5.3 and 5.4 for details. A summary of the conclusions is present below.

(a) The EAB TEDE dose with open containment is less than the regulatory limit of 6.3 rem TEDE even with only a 24-hr pre-FHA decay time. In addition, a decay time of 17 days is required for this dose to drop below the EPA PAG Evacuation limit of 1 rem TEDE. It is noted that since the release occurs within two hours, the 2-hr doses at the EAB bound the corresponding 30-day doses at the LPZ (which is farther out).

(b) A decay time of about 11 to 13 days would be needed to achieve an MCR dose less than 5 rem, for intake flows ranging between 50 cfm (with the MCR pre-isolated) and 3700 cfm (current MCR configuration).

(c) Simple pre-isolation of the MCR does not provide much relief. This is because the MCR cleanup rate is also reduced, thus extending the exposure interval (see Figure 5-3).

(d) Upgrade of the MCR HVAC system to include intake-flow emergency filtration will result in acceptable MCR doses under the following alternative conditions:

1.	Decay prior to fuel movement:	5 days
	Unfiltered inleakage:	50 to 200 cfm
	Filtration efficiency:	99% (4" charcoal beds)
	Delay in emergency filtration actuation:	20 minutes (SRP Section 6.4 (Ref. [8])
2.	Decay prior to fuel movement:	5-6 days (depending on unfiltered inleakage)
	Unfiltered inleakage:	50 to 200 cfm
	Filtration efficiency:	95% (2" charcoal beds)
	Delay in emergency filtration actuation:	20 minutes
3.	Decay prior to fuel movement:	8 days

Unfiltered inleakage:	50 to 200 cfm
Filtration efficiency:	99% (4" charcoal beds)
Delay in emergency filtration actuation:	30 minutes

(e) Pre-isolation of the MCR followed by MCR purge appears to be a viable option. The base-case scenario analyzed consisted of the following:

1. Pre-FHA decay of 5 days.

2. Refueling operations with open containment, vented at the rate of 2 air changes per hour, with no delay in the start of the atmospheric release following the accident.

3. MCR isolation prior to fuel movement, with a total unfiltered intake flow plus unfiltered inleakage of 150 cfm (where the intake flow would be for maintaining an acceptable CO_2 level within the MCR envelope).

4. MCR purge at 3.5 hours after the postulated FHA, at the normal flow rate of 3700 cfm.

However, such a configuration may only be acceptable as an interim measure for refueling operations with open containment, with the premise that suitable improvements (such as ESF filtration, or relocation of the air intake) will be made to the MCR HVAC system to mitigate the radiological consequences of an FHA.

7.0 REFERENCES

1. US NRC Regulatory Guide 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors" (Rev. 0, July 2000)

2. EPA 400-R-92-001, Manual of Protective Action Guides and Protective Actions for Nuclear Incidents (1991)

3.* ENTERGY Calculation VYC-2299, "Radiological AST Fuel Handling Accident Analysis [PSAT 3019CF.QA.05, Rev. 0]" (Jun. 2003)

4. AREVA NP Document 32-9053350-001, "ELISA-2 - A Software Package for the Radiological Evaluation of Licensing and Severe Accidents at Light-Water Nuclear Power Plants Based on the Classical and Alternative-Source-Term Methodologies" (Aug. 2008) [See also AREVA NP Document 2A4.26-2A4-ELISA2-2.4_Users_Manual-000, "ELISA-2 Version 2.4 User's Manual – Revision 2".]

5. EPA 520/1-88-020, Federal Guidance Report No. 11, "Limiting Values of Radionuclide Intake and Air Concentration, and Dose Conversion Factors for Inhalation, Submersion, and Ingestion" (ORNL, September 1988)

6. EPA 402-R-93-081, Federal Guidance Report No. 12, "External Exposure to Radionuclides in Air, Water, and Soil" (ORNL, September 1993)

7. NUMARC 93-01, "Industry Guidelines for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants"

8. NUREG-0800, Standard Review Plan, Sec. 6.4, "Control Room Habitability System" (Rev. 3, Mar. 2007)

9.* ENTERGY Calculation VYC-2260, "Bounding Core Inventories of Actinides and Fission Products for Design-Basis Applications at 1950 MWt" (Rev. 0, Feb. 2003)

10.* ENTERGY Calculation VYC-2275, "Control Room Air Intake X/Q Due to Release from Reactor Building Blowout Panel Using Arcon96 Methodology" (Rev. 0, April 2003)

11. Code of Federal Regulations Title 10 Part 50.67 (10CFR50.67).

* References identified with an (*) are maintained within the VYNPP Records System and are not retrievable from AREVA Records Management. These are acceptable references per AREVA Administrative Procedure 0402-01, Attachment 8. See page 2 of the present document for Project Manager Approval of customer references.

Attachment 5

Vermont Yankee Nuclear Power Station

Proposed Change 306

List of Regulatory Commitments

List of Regulatory Commitments

This table identifies actions discussed in this letter for which Entergy commits to perform. Any other actions discussed in this submittal are described for the NRC's information and are **not** commitments.

COMMITMENT	TYPE (Check one)		SCHEDULED COMPLETION DATE (If Required)
	ONE-TIME ACTION	CONTINUING COMPLIANCE	
During fuel handling/core alterations, ventilation system and radiation monitor availability (as defined in NUMARC 91-06) will be assessed, with respect to filtration and monitoring of releases from the fuel. The goal of maintaining ventilation system and radiation monitor availability is to reduce doses even further below that provided by the natural decay.		x	Commitment will be implemented prior to use of the amendment
A single normal or contingency method to promptly close primary or secondary containment penetrations will be established. Such prompt methods need not completely block the penetration or be capable of resisting pressure. Contingency plans for prompt closure of openings will include the following: - Equipment and tools needed to facilitate closure will be staged, - Personnel responsible for closure will be knowledgeable and trained in the procedures for establishing building integrity, - The closure response team will be accompanied by a Radiation Protection (RP) technician for radiation protection monitoring, - Hoses and cables routed through openings will employ a means to allow rapid, safe disconnect and removal, and - One door in each airlock will be capable of expeditious closure		x	Commitment will be implemented prior to use of the amendment
Revise VY UFSAR to reflect revised fuel handling accident analysis.	x		Completed in accordance with next scheduled UFSAR update following amendment approval