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May 22, 2000
BVY 00-25

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

References:
(a) 10CFR50.55a, Codes and Standards, subsection (b)(2)(vi).
(b) 10CFR50.55a, Codes and Standards, subsection (b)(2)(x).
(c) 10CFR50.55a, Codes and Standards, subsection (b)(2)(x)(E).
(d) American Society of Mechanical Engineers (ASME), Section XI, Division 1, Subsection IWE, 1992 Edition and 1992 Addenda.
(e) 10CFR50.55a, Codes and Standards, subsection (g)(6)(ii)(B).
(f) Letter, VYNPC to USNRC, "Response to Generic Letter 98-04," BVY 98-147, dated November 12, 1998.

Subject: Vermont Yankee Nuclear Power Station
License No. DPR-28 (Docket No. 50-271)
Proposed Change No. 222 – Inservice Inspection of Class MC Components

Pursuant to 10CFR50.90, Vermont Yankee (VY) hereby proposes to amend its Facility Operating License, DPR-28, by incorporating the attached proposed change into the VY Technical Specifications (TS). This proposed change will remove the TS 4.7.A.1 surveillance requirement for visual inspection of suppression chamber coating integrity once each refueling outage, and instead impose periodic inspection under VY's ASME Section XI, Subsection IWE Inservice Inspection (ISI) Program in compliance with the schedule set forth in Reference (e). The ISI Program provides more definitive inspection and acceptance criteria for primary containment examinations than those presently being applied under TS 4.7.A.1.

Attachment 1 to this letter contains supporting information and the safety assessment for the proposed change. Attachment 2 contains the determination of no significant hazards consideration. Attachment 3 provides a mark-up of the current Technical Specification pages. Attachment 4 provides the retyped Technical Specification pages.

VY has reviewed the proposed Technical Specification change in accordance with 10CFR50.92 and concludes that the proposed change does not involve a significant hazards consideration.

VY has also reviewed the proposed change against the criteria of 10CFR51.22 for environmental considerations and concludes that the proposed change will not increase the types and amounts of effluents that may be released offsite. Thus, VY believes that the proposed change is eligible for categorical exclusion from the requirements for an environmental impact statement in accordance with 10CFR51.22(c)(9).

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VY requests that a license amendment be issued by October 31, 2000 for implementation within 60 days of the effective date of the amendment. If you have any questions regarding this submittal, please contact Mr. Wayne M. Limberger at (802) 258-4237.

Sincerely,

VERMONT YANKEE NUCLEAR POWER CORPORATION

Samuel L. Newton
Vice President, Operations

Attachments
cc: USNRC Region 1 Administrator
USNRC Resident Inspector – VYNPS
USNRC Project Manager – VYNPS
Vermont Department of Public Service

STATE OF VERMONT)
)ss
WINDHAM COUNTY)

Then personally appeared before me, Samuel L. Newton, who being duly sworn, did state that he is Vice President, Operations of Vermont Yankee Nuclear Power Corporation, that he is duly authorized to execute and file the foregoing document in the name and on the behalf of Vermont Yankee Nuclear Power Corporation, and that the statements therein are true to the best of his knowledge and belief.

Sally A. Sandstrum, Notary Public
My Commission Expires February 10, 2003

Attachment 1

Vermont Yankee Nuclear Power Station

Proposed Technical Specification Change No. 222

Inservice Inspection of Class MC Components

Supporting Information and Safety Assessment for Proposed Change

DESCRIPTION OF CHANGE

VY proposes to change the Technical Specifications to remove the suppression chamber coating inspection criteria from TS 4.7.A.1 and instead apply the containment examination criteria defined in ASME Section XI, Subsection IWE, as required by current NRC regulations. The NRC has amended 10CFR50.55a to establish more definitive examination criteria for the primary containment. The suppression chamber coating visual inspection criteria that were incorporated into the 1972 Technical Specifications will be removed from TS 4.7.A.1 to support consolidation of our primary containment inspections under the criteria and schedule defined in the ISI program required by References (a), (b) and (c). In adopting the NRC-approved inspection criteria of Subsection IWE to meet 10CFR50.55a(b)(2)(vi) and 10CFR50.55a(b)(2)(x), VY also plans to apply the examination frequency imposed in 10CFR50.55a(b)(2)(x)(E) in lieu of the frequency stated in TS 4.7.A.1.

The proposed change is as follows:

> Technical Specification 4.7.A.1 states: "A visual inspection of the suppression chamber interior including water line regions and the interior painted surfaces above the water line shall be made at each refueling outage." VY proposes to remove this statement in its entirety since containment inspection requirements are provided in the applicable regulations – specifically, 10CFR50.55a(b)(2)(vi) and 10CFR50.55a(b)(2)(x).
>
> The corresponding Technical Specification Bases, which presently describe this surveillance as an "inspection of the paint" to assure it is intact, will also be revised to reflect this change.

Under this change, the once-per-refueling-outage (~18-month) visual inspection frequency and general inspection guidance provided in TS 4.7.A.1 will be replaced with the once-per-period (~40-month) general visual inspection schedule of Reference (c) and the more definitive inspection and acceptance criteria of Reference (d); in addition, the Code-required VT-3 examinations will be completed once in each 10-year inservice inspection interval. A different segment of the containment surface will be selected for inspection each refueling outage under the program, such that all required inspections are completed within the specified frequency. This comprehensive program will satisfy the NRC's expectations for both examination rigor and frequency while at the same time reducing redundant work and personnel radiation exposure stemming from the more-frequent inspections presently being conducted under TS 4.7.A.1.

REASON FOR CHANGE

This change is necessary to update Technical Specifications to eliminate a non-Code containment inspection that has been rendered redundant by more recent NRC regulations in this area. Incorporation of this change will: 1) enhance the detection of corrosion that could potentially lead to a reduction in containment integrity; 2) preclude potential conflicts regarding interpretation of the inspection frequency requirements; 3) reduce redundant work and achieve radiation exposure reductions; and 4) provide an equivalent degree of confidence in the integrity of suppression chamber coating.

BASIS FOR CHANGE

The change deals specifically with conforming the TS to agree with the incorporation of Subsection IWE into the VY ISI Program and to permit adoption of the inspection frequency stipulated in the regulation. Moving to Code-based inspections will provide more stringent criteria for inspection and acceptance of coating defects as they pertain to coating adherence and primary containment pressure boundary and structural integrity

In 1996, the NRC amended Reference (a) to incorporate by reference the 1992 Edition and 1992 Addenda of the Code for examination of Class MC components and adopted Reference (e) to establish an expedited examination schedule. Subsection IWE of the Code provides the requirements for inservice inspection (ISI) of Class MC (metal containment) components of light-water-cooled power plants. The inservice examinations of Code Class MC components and items, as specified in Table IWE-2500-1, shall be performed in accordance with Reference (d) as modified and supplemented by the requirements of Reference (b). The amended rule became effective on September 9, 1996. It required licensees to incorporate the new criteria into their ISI plans and to complete the first-period containment inspection no later than September 9, 2001. VY has presently completed approximately two-thirds of the first-period inspection objectives.

The Code-required inspections are specifically designed to detect evidence of primary containment base-metal degradation resulting from corrosion; the physical condition of the containment coating is only one leading indicator evaluated during these inspections. Because suppression chamber visual inspections under TS 4.7.A.1 are not supported by definitive inspection and acceptance criteria, Code-based inspection in accordance with the regulations (including removal of coating as necessary to facilitate inspection) is more appropriate to the corrosion failure mechanisms involved than the coating-adherence inspections being performed under the TS requirements. In addition, the inspection program that would be implemented under Reference (d) would supersede that cited in Reference (f), and would provide an equivalent degree of confidence in suppression chamber coating integrity.

Because no definitive inspection and acceptance criteria comparable to those in the Code are provided in the Technical Specifications, the more frequent but less comprehensive inspections under TS 4.7.A.1 do not add significant value to the assurance of containment pressure-boundary and structural integrity now achieved under the Code.

SAFETY ASSESSMENT

Since 1972, VY has conducted visual inspections of suppression chamber coating integrity in accordance with TS 4.7.A.1, accumulating a considerable history of containment coating performance. Although coating degradation was detected during the course of these inspections, the inspections have detected no areas of significant corrosion penetrating through the primer coat and requiring corrective base-metal repair. Instances of localized corrosion on wetted areas of the suppression chamber were determined to be non-invasive and did not require repair other than repainting with qualified coating products. Visual inspections and/or ultrasonic thickness measurements at the affected locations have shown no impact on shell thickness. Additionally, the VY containment coating system utilizes an inorganic zinc-based primer that resists the aggressive progress of substrate corrosion following a breach of the primer.

Application of the more recent Code as stipulated in 10CFR50.55a(b)(2)(vi) and supplemented by the requirements of 10CFR50.55a(b)(2)(x), will result in a "substantial safety increase" as indicated in the NRC's Statements of Consideration (61 FR 41303) that accompanied publication of the final rule. The Code criteria for investigating and resolving evidence of coating degradation (e.g. thinning, cracking, blistering, scabbing or peeling and the associated impact on base-metal integrity) are more definitive than those presently applied under TS 4.7.A.1 and will improve inspection quality.

The concerns of NRC Generic Letter (GL) 98-04 regarding the potential impact of primary containment coating debris generated during a LOCA on the safety function of emergency core cooling (ECCS) systems were addressed through replacement of the suction strainers for these systems. The new strainers are sized to accommodate complete detachment of all remaining primary containment topcoat material without loss of ECCS suction. These physical modifications are further supported by refined transport analyses predicting that a smaller amount of material would be transported to the vicinity of the suction strainers than previously assumed. Due to these actions, ECCS suction strainer clogging and resultant loss of ECCS safety functions during a LOCA as a result of paint debris accumulation are not considered credible.

Therefore, the slight increased risk to containment integrity of a gradual, localized coating degradation that could potentially exist undetected in some locations for approximately 40 months instead of the previously established refueling outage (~18 month) interval is adequately offset by the historical performance of the coating system, the increase in safety provided by the current Code-based inspection practices, and the modifications and analyses performed in response to GL 98-04.

On these bases, VY concludes that the proposed change will have no adverse impact on plant safety.

Attachment 2

Vermont Yankee Nuclear Power Station

Proposed Technical Specification Change No. 222

Inservice Inspection of Class MC Components

Determination of No Significant Hazards Consideration

Pursuant to 10CFR50.92, VY has reviewed the proposed change and concludes that the change does not involve a significant hazards consideration since it satisfies the criteria in 10CFR50.92(c). This change replaces the surveillance requirements of Technical Specification 4.7.A.1 with the more definitive primary containment examination criteria of ASME Section XI, Subsection IWE, 1992 Edition and Addenda stipulated in 10CFR50.55a.

1. <u>The operation of Vermont Yankee Nuclear Power Station in accordance with the proposed amendment will not involve a significant increase in the probability or consequences of an accident previously evaluated.</u>

   The proposed change conforms the TS to current regulations, credits actions taken under GL 98-04 to address coating delamination concerns, and eliminates redundant surveillance criteria. Since reactor operation under the revised Specification is unchanged, no design or analytical acceptance criteria will be exceeded. As such, this change does not impact initiators of analyzed events or assumed mitigation of accident or transient events. The structural and functional integrity of plant systems is unaffected. Thus, there is no significant increase in the probability or consequences of accidents previously evaluated.

2. <u>The operation of Vermont Yankee Nuclear Power Station in accordance with the proposed amendment will not create the possibility of a new or different kind of accident from any accident previously evaluated.</u>

   The proposed change does not affect any parameters or conditions that could contribute to the initiation of any accident. No new accident modes are created. No safety-related equipment or safety functions are altered as a result of these changes. Because it does not involve any change to the plant or the manner in which it is operated, the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. <u>The operation of Vermont Yankee Nuclear Power Station in accordance with the proposed amendment will not involve a significant reduction in a margin of safety.</u>

   The proposed change does not affect design margins or assumptions used in accident analyses and has no effect on any initial condition. The capability of safety systems to function and limiting safety system settings are similarly unaffected as a result of this change.
   Thus, the margins of safety required for safety analyses are maintained.

Vermont Yankee has also reviewed the NRC examples of license amendments considered not likely to involve significant hazards considerations as provided in the final adoption of 10CFR50.92 published in the Federal Register (FR), Volume 51, No. 44, dated March 6, 1986. Example (7) on FR page 7751 provides a discussion of changes performed "to conform a license to changes in the regulations, where the license change results in very minor changes to facility operations clearly in keeping with the regulations." This proposed changes satisfies this definition, which indicates that it is likely no significant hazards considerations are involved.

On the basis of the above, VY has determined that operation of the facility in accordance with the proposed change does not involve a significant hazards consideration as defined in 10 CFR 50.92(c), in that it:

1) does not involve a significant increase in the probability or consequences of an accident previously evaluated;

2) does not create the possibility of a new or different kind of accident from any previously analyzed accident; and

3) does not involve a significant reduction in a margin of safety.

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Attachment 3

Vermont Yankee Nuclear Power Station

Proposed Technical Specification Change No. 222

Inservice Inspection of Class MC Components

Marked-up Version of the Current Technical Specification Pages

3.7 LIMITING CONDITIONS FOR OPERATION

3.7 STATION CONTAINMENT SYSTEMS

Applicability:

Applies to the operating status of the primary and secondary containment systems.

Objective:

To assure the integrity of the primary and secondary containment systems.

Specification:

A. Primary Containment

1. Whenever primary containment is required, the volume and temperature of the water in the suppression chamber shall be maintained within the following limits:

   a. Maximum Water Temperature during normal operation - 90°F.

   b. Maximum Water Temperature during any test operation which adds heat to the suppression pool - 100°F; however, it shall not remain above 90°F for more than 24 hours.

   c. If Torus Water Temperature exceeds 110°F, initiate an immediate scram of the reactor. Power operation shall not be resumed until the pool temperature is reduced below 90°F.

   d. During reactor isolation conditions, the reactor pressure vessel shall be depressurized to less than 200 psig

4.7 SURVEILLANCE REQUIREMENTS

4.7 STATION CONTAINMENT SYSTEMS

Applicability:

Applies to the primary and secondary containment system integrity.

Objective:

To verify the integrity of the primary and secondary containments.

Specification:

A. Primary Containment

1. Verify daily that the suppression chamber water level and average temperature are within applicable limits.

   ~~A visual inspection of the suppression chamber interior including water line regions and the interior painted surfaces above the water line shall be made at each refueling outage.~~

   Verify suppression pool average temperature is within the applicable limits every 5 minutes when performing testing that adds heat to the suppression pool.

   Whenever there is indication of relief valve operation with the temperature of the suppression pool reaching 160°F or more and the primary coolant system pressure greater than 200 psig, an external visual examination of the suppression chamber shall be conducted before resuming power operation.

4.7 STATION CONTAINMENT SYSTEMS

A. Primary Containment System

~~The interiors of the drywell and suppression chamber are painted to prevent rusting. The inspection of the paint during each major refueling outage assures the paint is intact. Experience with this type of paint at fossil fueled generating stations indicates that the inspection interval is adequate.~~

Because of the large volume and thermal capacity of the suppression pool, the level and temperature normally changes very slowly and monitoring these parameters daily is sufficient to establish any temperature trends.

The average temperature is determined by taking an arithmetic average of OPERABLE suppression pool water temperature channels. The daily frequency has been shown, based on operating experience, to be acceptable. The frequencies are further justified in view of other indications available in the Control Room, including alarms, to alert operators to an abnormal condition.

When heat is being added to the suppression pool by testing, however, it is necessary to monitor suppression pool temperature more frequently. The 5 minute frequency during testing is justified by the rate at which tests will heat up the suppression pool. This has been shown to be acceptable based on operating experience, and provides assurance that allowable pool temperatures are not exceeded.

The requirement for an external visual examination following any event where potentially high loadings could occur provides assurance that no significant damage was encountered. Particular attention should be focused on structural discontinuities in the vicinity of the relief valve discharge since these are expected to be the points of highest stress. Visual inspection of the suppression chamber including water line regions each refueling outage is adequate to detect any changes in the suppression chamber structures.

The interiors of the drywell and suppression chamber are painted with an inorganic zinc primer to prevent rusting that could lead to degradation of the containment pressure boundary. The inspection of the painted surfaces as part of inservice inspection under 10CFR 50.55a(b)(2)(vi) assures that the paint and the underlying base metal have not degraded. Experience with this type of coating during plant operating cycles between 1972 and the present indicates that this inspection methodology and interval are adequate.

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Attachment 4

Vermont Yankee Nuclear Power Station

Proposed Technical Specification Change No. 222

Inservice Inspection of Class MC Components

Retyped Technical Specification Pages

3.7 LIMITING CONDITIONS FOR OPERATION

3.7 STATION CONTAINMENT SYSTEMS

Applicability:

Applies to the operating status of the primary and secondary containment systems.

Objective:

To assure the integrity of the primary and secondary containment systems.

Specification:

A. Primary Containment

1. Whenever primary containment is required, the volume and temperature of the water in the suppression chamber shall be maintained within the following limits:

   a. Maximum Water Temperature during normal operation - 90°F.

   b. Maximum Water Temperature during any test operation which adds heat to the suppression pool - 100°F; however, it shall not remain above 90°F for more than 24 hours.

   c. If Torus Water Temperature exceeds 110°F, initiate an immediate scram of the reactor. Power operation shall not be resumed until the pool temperature is reduced below 90°F.

   d. During reactor isolation conditions, the reactor pressure vessel shall be depressurized to less than 200 psig

4.7 SURVEILLANCE REQUIREMENTS

4.7 STATION CONTAINMENT SYSTEMS

Applicability:

Applies to the primary and secondary containment system integrity.

Objective:

To verify the integrity of the primary and secondary containments.

Specification:

A. Primary Containment

1. Verify daily that the suppression chamber water level and average temperature are within applicable limits.

   Verify suppression pool average temperature is within the applicable limits every 5 minutes when performing testing that adds heat to the suppression pool.

   Whenever there is indication of relief valve operation with the temperature of the suppression pool reaching 160°F or more and the primary coolant system pressure greater than 200 psig, an external visual examination of the suppression chamber shall be conducted before resuming power operation.

4.7 STATION CONTAINMENT SYSTEMS

A. Primary Containment System

The interiors of the drywell and suppression chamber are painted with an inorganic zinc primer to prevent rusting that could lead to degradation of the containment pressure boundary. The inspection of the painted surfaces as part of inservice inspection under 10 CFR 50.55a(b)(2)(vi) assures that the paint and the underlying base metal have not degraded. Experience with this type of coating during plant operating cycles between 1972 and the present indicates that this inspection methodology and interval are adequate.

Because of the large volume and thermal capacity of the suppression pool, the level and temperature normally changes very slowly and monitoring these parameters daily is sufficient to establish any temperature trends.

The average temperature is determined by taking an arithmetic average of OPERABLE suppression pool water temperature channels. The daily frequency has been shown, based on operating experience, to be acceptable. The frequencies are further justified in view of other indications available in the Control Room, including alarms, to alert operators to an abnormal condition.

When heat is being added to the suppression pool by testing, however, it is necessary to monitor suppression pool temperature more frequently. The 5 minute frequency during testing is justified by the rate at which tests will heat up the suppression pool. This has been shown to be acceptable based on operating experience, and provides assurance that allowable pool temperatures are not exceeded.

The requirement for an external visual examination following any event where potentially high loadings could occur provides assurance that no significant damage was encountered. Particular attention should be focused on structural discontinuities in the vicinity of the relief valve discharge since these are expected to be the points of highest stress. Visual inspection of the suppression chamber including water line regions each refueling outage is adequate to detect any changes in the suppression chamber structures.