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November 15, 1996
BVY 96-144

United States Nuclear Regulatory Commission
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References: (a) License No. DPR-28 (Docket No. 50-271)
(b) Letter, VYNPC to USNRC, " Core Spray and Residual Heat Removal Systems Containment Isolation Valves at Vermont Yankee," BVY 96-123, dated October 15, 1996
(c) Telecon, USNRC to VYNPC, dated November 13, 1996

Subject: 10CFR50.59 Evaluation Regarding Core Spray and RHR Containment Isolation Valves at Vermont Yankee

In Reference (b) Vermont Yankee provided information regarding redesignation of several containment isolation valves at Vermont Yankee Nuclear Power Station. In Reference (c), the NRC requested a copy of the evaluation performed by Vermont Yankee pursuant to 10CFR50.59(a)(2) in support of this redesignation. The evaluation is attached.

We trust that the information provided is acceptable. However, should you have questions or require additional information, please contact this office.

Sincerely,

VERMONT YANKEE NUCLEAR POWER CORPORATION

James J. Duffy

James J. Duffy
Licensing Engineer

Attachment: 10CFR50.59 evaluation

A001/1

c: USNRC Region I Administrator
USNRC Project Manager - VYNPS
USNRC Resident Inspector - VYNPS

10CFR50.59 REVIEW AND SAFETY EVALUATION

Title: Alternate RHR and Core Spray Containment Isolation Valve Isolation Basis

1.0. Description

The Residual Heat Removal System (RHR) and Core Spray System (CS) at Vermont Yankee contain Containment Isolation Valves (CIV) in each line penetrating the primary containment. The original design criteria follows the General Design Criteria of 10CFR50, Appendix A (GDC 55 & 56) and is discussed in FSAR Appendix F. Vermont Yankee FSAR Section 5.2.3.4 repeats the design requirements which include two isolation valves in series which automatically close and are capable of remote manual actuation from the Control Room. Automatic closure of RHR and CS injection lines is not required following a Loss of Coolant Accident (LOCA) since they provide a needed ECCS function, but inboard check valves are relied upon for automatic isolation in response to outside containment system breaks, if needed.

Two motor operated valves are located just outside of the containment boundary. In the CS system, the first opens to provide ECCS injection following an accident; The second motor operated valve provides a pressure barrier to allow operability testing of the injection valve during normal plant operation. In the RHR system the second valve opens to provide ECCS injection and RHR flow control while the first is used for operability testing.

To assure that the containment integrity is maintained following design basis accidents, 10CFR50, Appendix J requires leak rate testing of containment isolation valves. "Type C" tests are required of valves which 1) connect the inside and outside of containment during normal operation (e.g. vent valves), 2) close automatically due to a containment isolation signal, 3) operate intermittently under post accident conditions or 4) are associated with BWR mainsteam or feedwater penetrations.

The Vermont Yankee technical specification (Table 4.7.2a) lists the various Containment Isolation Valves (CIV) which are required to have Type C Local Leak Rate Tests (LLRT) performed per 10CFR50 Appendix J. None of the subject RHR or CS valves are identified on this table. However, because they are identified in the FSAR as containment isolation valves, Vermont Yankee has considered that Type C leak testing is required of the RHR and CS injection line CIVs and has included them in their Appendix J testing program.

The technical specification basis references the detailed containment isolation valve discussion of FSAR section 5.2. Table 5.2.2 of the FSAR identifies the inboard check valves and outboard MOVs associated with the RHR and CS injection lines; Table 7.3.1 further identifies the two outboard isolation valves as having remote manual closure

capability in addition to the check valve. The RHR/CS logic prevents closure of the test isolation valve when an injection signal is present.

This change provides an accepted alternate to GDC 55 & 56 isolation requirements such that it is possible to eliminate the CIV designation to certain valves. This justification, in turn, will result in reduced outage time, reduced occupational radiation exposure, and reduced costs associated with testing of the reclassified valves.

This change reflects aspects of similar changes contained in references 1 and 2.

The valves addressed by this evaluation are:

Valve	System	Description	Current CIV	Proposed CIV
RHR-27A/B	LPCI	System Isolation/Throttle Valve	Yes	Yes
RHR-25 A/B	LPCI	Test Isolation Valve	Yes	No
RHR-210A/B	LPCI	RHR-25A/B Bonnet relief	N/A	No
RHR-46A/B	LPCI	Injection check Valve	Yes	No
CS-12A/B	Core Spray	Injection Gate Valve	Yes	Yes
CS-11A/B	Core Spray	Test Isolation Valve	Yes	No
CS-13A/B	Core Spray	Injection check Valve	Yes	No
CS-30A/B	Core Spray	Injection check valve bypass	Yes	No

Justification for the change

The RHR and CS Systems are among the ESF Systems included in the Vermont Yankee design. The RHR system includes various normal and post-accident operational modes, including Shutdown Cooling, Low Pressure Coolant Injection (LPCI), and Containment Spray. The RHR and CS Systems deliver cooling water to the reactor following postulated accidents when the reactor is at low pressure conditions. The RHR System also delivers cooling water in the form of containment spray, to reduce containment pressure post-accident.

In performing these functions, the RHR and CS Systems penetrate the Primary Containment Pressure Boundary and the Reactor Coolant Pressure Boundary (RCPB), as necessary. In order to allow the normal and emergency passage of water through the

containment while preserving the ability to prevent or limit the escape of fission products following a postulated accident, containment isolation provisions are included in the system designs.

The RHR and CS Systems are designed to satisfy the appropriate containment isolation requirements including those contained in 10CFR50 Appendix A General Design Criteria (GDC) 55 and 56. The current licensing basis satisfies these GDCs through the use of redundant CIVs. FSAR Appendix F, "Conformance to AEC General Design Criteria", states that the CS and RHR lines "have check valves inside and two isolation valves outside...".

The use of closed, extended containment boundary system piping outside containment, in conjunction with the use of a single, remote manual CIV outside containment provides a dual isolation barrier which satisfies the "other defined basis" requirements of 10CFR50 Appendix A General Design Criteria (GDC) 55 and 56. Thus, GDC 55 and 56 can be met for these RHR and CS lines without the use of two CIVs in each line.

Reference 2 describes the basis for acceptance of this basis as follows:

> *One such "other defined basis" is the case where a qualified "closed system outside containment" may be credited as a substitute for one of the two CIVs. Paragraph 3.6.7 of N271-1976/ANS-56.2, "American National Standard Containment Isolation Provisions for Fluid Systems", which is endorsed by Regulatory Guide 1.141, specifies that design criteria applicable to a closed system outside containment. Under the terms of these criteria, a single CIV in conjunction with a qualified closed system constitutes an acceptable containment piping penetration arrangement...*

In order for a piping system loop located outside containment to be considered "closed", specific design criteria must be met. Per paragraph 3.6.7 of ANSI/ANS 56.2-1984 (revision to N271-1976/ANS-56.2) a closed system outside containment must:

1. Not communicate with the outside atmosphere

The subject RHR and CS lines are ESF systems that circulate water from the suppression pool to the reactor pressure vessel following postulated accidents. The RHR and CS Systems are designed as closed systems outside of the containment and are physically located in the Reactor Building. These piping systems, in conjunction with the CIVs outside containment, minimize any direct communication between the post-accident containment atmosphere and the Reactor Building under the various post-accident scenarios, including postulated single failures.

Additionally, lines that communicate with areas outside the reactor building are isolated during accidents by normally closed manual valves, automatic acting check valves or automatic isolation valves. These secondary containment isolation valves operate so that the reactor building can be maintained at a negative pressure relative to the outside atmosphere to assure only filtered communication with the outside atmosphere.

2. Meet Safety Class 2 design requirements

Safety classification for systems at Vermont Yankee are performed in accordance with the Vermont Yankee Safety Class Manual which is primarily based on ANS-22, Draft 4, rev 1, May 1973. The RHR and CS systems are classified and were designed to safety class 2 standards. Additionally, RHR and CS instrumentation that performs an electrical safety function also is classified as Safety Class 2.

Instrumentation and the instrument tubing that does not perform an electrical safety function is classified as NNS beyond the root isolation valve (which is Safety Class 2). This classification is per the Safety Class Manual and is consistent with more recent regulatory guidance provided in Regulatory Guide 1.151 (Section C.2.a). As part of this assessment these instruments and the instrument tubing have been verified to be seismic category 1 (see criteria 5) to ensure that they will maintain their integrity following a design basis seismic event.

3. Withstand temperature and pressure equal to the containment design conditions

The RHR and CS systems are designed to be operating under post accident conditions. The system design temperatures and pressures are consistent with expected analyzed post accident containment conditions.

4. Withstand loss-of-coolant accident transient

The RHR and CS systems are designed to be operating following a design basis loss of coolant accident. An assessment of the ECCS systems capability to withstand the LOCA transient is contained in the FSAR.

5. Meet seismic Category I design requirements

The primary piping for the RHR and CS meets seismic Category 1 criteria. This piping was upgraded following issuance of IEB 79-02 and 79-14 to address concerns about the adequacy of as-built piping systems.

Instrument tubing connected to instruments that do not have an electrical safety function is classified as NNS per the Vermont Yankee Safety Classification Manual. The instrumentation on the RHR and CS system was field run. To support this safety evaluation, an assessment of the field configuration was performed and supports were added, where needed, to provide added assurance that the tubing will maintain its integrity following a seismic event.

6. Be protected against overpressure from thermal expansion of contained fluid when isolated, if required

The piping and components of the RHR and CS systems are equipped with relief valves as required by the applicable design codes to protect system piping and component integrity. Relief devices that perform a safety related function are tested per the station In-service Testing program. For the piping between the tested CIV and the Reactor, overpressure protection is provided by the safety/relief valves that protect the Reactor Pressure Vessel.

7. Be protected against a high energy line break unless it can be demonstrated that the high energy line break will not result in the need for containment isolation

The RHR and CS systems are located such that the containment isolation function or piping integrity will not be affected by a high energy line break outside the containment.

An assessment of the impact of high energy line breaks on safety related equipment outside containment has been performed for Vermont Yankee and based on this assessment the ECCS systems are considered protected from the effects of potential high energy line breaks.

No additoinal passive failure beyond the initiating LOCA event is required to be assumed in the evaluation of design basis accidents. Therefore, the closed-loop integrity of the RHR and CS system water pumping loops is assumed, and any leakage into these systems can be considered to be contained within these extensions of containment. Furthermore, high energy line breaks (HELBs) outside containment do not create a need for containment isolation beyond isolation of the effected line.

8. Be protected against loss of function from missiles

General Design Criteria 4 states that components important to safety "be appropriately protected against ...the effects of missiles... that may result from equipment failures...". The intent of this design criteria as it applies to maintaining the RHR and CS system as a closed system is that missiles should not cause a loss of integrity of the system piping. Only the main turbine and reactor recirculation pump motors are specifically addressed (Regulatory Guides 1.14 and 1.115) by specific design criteria or regulatory guides.

The design of Vermont Yankee accounts for protection from reasonably postulated missiles. There are no high energy components in proximity to the RHR or CS piping that represent missile sources. The design of rotating equipment associated with these systems also is such that external missiles are not reasonably postulated. Therefore the integrity of the RHR and CS systems is not reasonably challenged by the effect of missiles and this criteria can be considered to be satisfied

9. Be capable of being leak tested

Paragraph 3.6.4 of ANSI/ANS 56.2-1984 allows exemption from leakage testing if "system integrity is being maintained for those systems operating during normal plant operation at a pressure equal to or greater than the containment design pressure". The RHR and CS systems are maintained in a pressurized state exceeding containment design pressure during normal plant operation. Leakage monitoring (see discussion below) assures that system leakage is identified and addressed.

The following sections provide further justification for this position.

Leakage Monitoring

Provisions for leak detection and removal are detailed in FSAR section 4.10 "Nuclear System Leakage Rate Limits and Leakage Detection Systems" and Section 10.16 "Station Equipment and Floor Drainage Systems". These systems provide information to control room personnel on potential leaks in the primary and secondary containments such that action can be taken should a leak in a system be detected. Additional system specific monitoring of potential leaks in the RHR and CS systems is provided by:

1. The RHR and CS systems are pressurized by a keep fill system at approximately 80 to 90 psig. Normal rounds performed by station personnel (operators, rad protection, management, etc.) will identify leakage from either system and corrective action would be initiated.

2. Technical Specification section 6.10, "Integrity of systems Outside Containment" requires a detailed inspection by VT2 qualified operations personnel once per cycle to identify and correct system leakage. These inspection are performed at system operating condition which is greater than maximum post-LOCA containment pressure.

3. Station personnel are present during system surveillance testing when the system is subject to a change in system pressure from the Standby keep fill pressure to

operating conditions. Leakage that occurs during system surveillance testing would be identified and corrected.

4. During normal plant operation, valve leakage past the pump discharge check valves and other system isolation valves could impact the ability of the systems to remain water filled following an accident. Because the leakage path would be to the torus, this situation will result in the operational need to periodically reduce torus water level. Therefore the systems ability to remain water filled is monitored based on the need to reduce torus water level. If excessive leakage is noted maintenance of the valves would be planned.

Based on the above monitoring it is concluded that the applicable portions of the RHR and CS systems are monitored during operation at a pressure greater than the expected peak accident pressure (44 psig) and leakage is maintained at extremely low levels.

Valve functions

Although two of the valves subject to this evaluation will no longer be considered CIVs, the proposed change does not affect the other design, functions, and operation of these lines. The valves will remain fully operational and capable of performing all of their other required system and safety-related functions.

The valves are designed to be opened post-accident, as necessary, in order to properly align the RHR and CS Systems for the performance of their required safety functions and they are designed to circulate radioactive water containing a deterministic post accident source term from the suppression pool following postulated accidents. These considerations bound consideration of any post-accident leakage past the CIVs from a containment isolation or qualification perspective since the radioactive material will be contained within the system piping. The ability of the subject valves to perform required functions will be demonstrated during normal plant operation and via surveillance testing in accordance with the applicable Technical Specification requirements.

The subject valves are all Reactor Coolant System Pressure Isolation Valves and subject to operational leakage control per Technical Specification 3.6.C. The leak-tight integrity of these valves as high-low pressure interface boundaries will continue to be demonstrated by their performance during normal operation. In addition, these valves will continue to be inspected and tested per the applicable requirements of Section XI of the ASME Boiler and Pressure Vessel Code.

2.0 Unresolved Safety Question (USQ) Determination [10CFR50.59 (a)(2)]

1. May the proposed activity increase the probability of an accident previously evaluated in the FSAR including those enveloped by any of the following accidents:

Chap 14.6.2 - Control Rod Drop Accident
Chap 14.6.3 - Loss of Coolant Accident
Chap 14.6.4 - Refueling Accident
Chap 14.6.5 - Main Steam Line Break Accident

Applicable FSAR Sections: 14.6.3

YES ________ NO ___X___

Discussion

The proposed change does not affect the probability of any accident since it only is a reclassification of existing equipment.

The proposed change does not affect the design, functions, and operation of any valves in the RHR or CS systems. The valves will remain fully operational and capable of performing all of their required system and safety-related functions. Other applicable requirements, pertaining either to the valves or the systems they serve which are imposed by Technical Specifications, are not affected by this change.

The subject valves which are Reactor Coolant System Pressure Isolation Valves remain subject to Technical Specification 3.6.C (reactor coolant system leakage). In addition, all of the valves will continue to be inspected and tested per the applicable requirements of Section XI of the ASME Boiler and Pressure Vessel Code.

2. May the proposal increase the radiological consequences, (analyzed in Chapter 14.9, Table 14.9.4) of an accident previously evaluated in the FSAR including those enveloped by any of the following accidents:

Chap 14.6.2 - Control Rod Drop Accident
Chap 14.6.3 - Loss of Coolant Accident
Chap 14.6.4 - Refueling Accident
Chap 14.6.5 - Main Steam Line Break Accident

Applicable FSAR Sections: 14.6.3

YES ________ NO ___X___

Discussion

The proposed change does not alter the physical plant or the manner in which it is operated. Only the declassification of certain valves as non-CIV valves is addressed. The change eliminates conservative Appendix J testing requirements for these reclassified valves while retaining those tests needed to assure required containment isolation capabilities. The new bases for satisfying GDC 55 and 56 provide equivalent levels of protection against off-site radiation releases by assuring the dual barriers of piping integrity and a CIV.

Consideration of the passive failure of piping in addition to the design basis LOCA is not required since it is of extreme low probability. However, if the piping integrity (one barrier) is not intact for some reason, radiological releases and their consequences due to the leakage of the subject valves will be within the existing plant licensing basis as confirmed by 10CFR50, Appendix J testing of the CIVs.

For single active failures of mechanical and electrical components which result in the CIV failure to close, the integrity of the closed system provides equivalent protection. Furthermore, the lines have a high probability of remaining water filled which further limits any potential release. Alternate valves, even though not leakage tested, also can be closed to isolate a non-operating system.

Since the valves, piping and the systems they serve are located in the Reactor Building, any leakage (e.g., packing gland leakage) which escapes the confines of the closed system piping will be contained within the containment or in the Reactor Building. These areas are radiologically controlled and monitored. Any releases to the Reactor Building would be processed by the Standby Gas Treatment System, as required. This assures that all radioactive releases to the environment are within the existing plant licensing bases.

The elimination of conservative 10CFR50 Appendix J Type C testing for the reclassified valves will not affect the existing radiological release evaluations currently described in the FSAR. The valves which are listed on Table 7.3.1 which are reclassified will be identified by addition of a note clarifying the containment isolation function and the lack of testing requirements.

Therefore radiological consequences are not increased.

3. May the proposed activity increase the probability of a malfunction occurring which initiates a FSAR 14.5 abnormal operational transient and causes:

Nuclear System Pressure Increases
Reactor Vessel Moderator Temperature Decreases
Positive Reactivity Increase
Reactor Vessel Coolant Inventory Decreases
Reactor Core Coolant Flow Increases
Core Coolant Temperature Increases
An Excess of Coolant Inventory
Loss of Habitability Of Main Control Room

or - Impacts station blackout, anticipated transient without scram, Appendix R or Alternate Shutdown.

or - Would prevent the successful performance of a safety function of any structure, system or component.

Applicable FSAR Sections: 14.5

YES ________ NO ____X___

Discussion

The proposed change does not alter the physical plant or the manner in which it is operated. The change eliminates conservative Appendix J testing requirements for certain valves while retaining those tests needed to assure required containment isolation capabilities. The new bases for satisfying GDC 55 and 56 provide equivalent levels of protection against off-site radiation releases. Therefore, clarifying the testing requirements associated with the subject valves does not increase the probability of occurrence of a malfunction of any equipment.

4. May the proposed activity increase the radiological consequences, above that analyzed in Chapter 14.2, resulting from a malfunction which initiates a FSAR 14.5 abnormal operational transient and causes:

Nuclear System Pressure Increases
Reactor Vessel Moderator Temperature Decreases
Positive Reactivity Increase
Reactor Vessel Coolant Inventory Decreases
Reactor Core Coolant Flow Increases
Core Coolant Temperature Increases
An Excess of Coolant Inventory
Loss of Habitability Of Main Control Room

or - Impacts station blackout, anticipated transient without scram, Appendix R or Alternate Shutdown.

Applicable FSAR Sections: 14.2, 14.5

YES ________ NO ___X___

Discussion

The proposed change does not alter the physical plant or the manner in which it is operated. The change eliminates conservative Appendix J testing requirements for certain valves while retaining those tests needed to assure required containment isolation capabilities. The new bases for satisfying GDC 55 and 56 provide equivalent levels of protection against off-site radiation releases. No new failure modes for any equipment are generated.

No redesign of these components is involved in this change. Any failure mode which exists with the current components still exists following this change. Therefore, reclassification of certain subject valves from Table 7.3.1 does not increase the consequences of occurrence of a malfunction of any equipment.

The dual barriers provided by the CIVs and closed system piping outside the containment satisfy the "other defined basis" requirements of GDC 55 and 56, as clarified in RG 1.141 and the ANSI/ANS Standard referenced therein. These dual barriers prevent post-accident containment leakage and provide protection against postulated single active and passive failures. Therefore, the post-accident integrity of the containment is assured following the clarification of the testing requirements associated with the subject valves on Table 7.3.1.

Reclassification of these valves results in reduced outage time, reduced occupational radiation exposure, and reduced costs associated with testing of the reclassified valves without increasing the consequence of any design basis event.

5. May the proposed activity create the possibility of an accident occurring which is different from

FSAR 14.4.3 - Mech Failure Leading to Rad Material Boundary Breach
FSAR 14.4.3 - Overheating of the Fuel Barrier
FSAR 14.4.3 - Arbitrary Single Pipe Rupture

Applicable FSAR Sections: 14.4.3

YES ________ NO ___X___

Discussion

This proposal does not involve any hardware or logic changes, nor does it alter the way in which any plant systems operate. Post accident containment isolation features, boundaries, and system interfaces are not affected by the changes. The change eliminates conservative Appendix J tests for certain valves while retaining those tests needed to assure that the containment isolation features perform as required. Therefore, the possibility of an accident of a different type is not created.

6. May the proposed activity create the possibility of a different type of malfunction of equipment different than any previously evaluated in the FSAR?

Applicable FSAR Sections: 14.0

YES ________ NO ___X___

Discussion

This proposal does not involve any hardware or logic changes, nor does it alter the way in which any plant systems operate. Post accident containment isolation features, boundaries, and system interfaces are not affected by the changes. The change eliminates conservative Appendix J tests for certain valves while retaining those tests needed to assure that the containment isolation features perform as required. Therefore, no malfunction of a different type than previously evaluated is created.

7. Does the proposed activity reduce the difference between a system failure point and accepted safety limit or reduce the margin of safety as defined in the basis for any Technical Specification?

Applicable FSAR Sections: None

YES ________ NO ___X___

Discussion

The Vermont Yankee Technical Specifications sections 3/4.7 and the Appendix J testing program identify valves subject to Appendix J Type C testing. The proposed change will not affect the functional capability of any plant safety-related structures, systems, or components. Therefore system safety margins are unaffected.

3.0 FSAR Changes

Attachment A provides FSAR changes necessary to reflect the modification to the Containment Isolation valve basis for meeting GDC 55 & 56.

4.0 Technical Specification Changes

Attachment B provides changes to the Technical Specification bases necessary to reflect the modification to the Containment Isolation valve basis for meeting GDC 55 & 56 and to be consistent with the FSAR.

5.0 Conclusion

This proposal makes no physical change to the facility. The only change is a clarification of the leak testing requirements associated with certain valves and clarification of the applicability of GDC 55 and 56 to the affected lines. No Unresolved Safety Questions are involved since 1) the probability or consequences of an accident previously evaluated in the FSAR are not increased, 2) the probability or consequences of a malfunction of equipment important to safety previously evaluated in the FSAR are not increased, 3) the possibility of an accident or malfunction of a different type than any previously evaluated in the FSAR are not created, and 4) the margin of safety as defined in the basis for any Technical Specification is not reduced.

While some changes are made to the Appendix J testing program, they do not reduce the margin of safety in the plant design or operation and prior NRC approval should not be required.

6.0 References

1. License Amendment "Issuance of amendment and exemption re: Type C testing of primary containment isolation valves," issued by the NRC on April 22, 1994 for Detroit Edison Company's Fermi-2 Nuclear Plant (Docket No. 50-341)

2. Safety Evaluation by the Office of Nuclear Reactor Regulation related to Amendment No. 93 to Facility Operating License No NPD-57, Public Service Electric and Gas Company, Atlantic City Electric Company, Hope Creek Generating Station, Docket No. 50-354., February 22, 1996