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August 1, 2007

BVY 07-051

ATTN: Document Control Desk
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

Reference: Letter USNRC to VYNPC, "TMI Action Plan Item II.K.3.3, Reporting of Relief Valve Failures and Challenges", NVY 82-44, dated March 30, 1982

Subject: **Vermont Yankee Nuclear Power Station**
License No. DPR-28 (Docket No. 50-271)
Cycle 25 10CFR50.59 Report

Dear Sir or Madam:

In accordance with 10CFR50.59, attached is the Vermont Yankee Cycle 25 10CFR50.59 Report. This report contains a brief description of the 50.59 Evaluations that were performed between November 12, 2005 and June 6, 2007.

Additionally, in accordance with the referenced letter, Vermont Yankee reports that there were no Main Steam Relief Valve or Safety Valve failures or challenges during this period.

There are no new commitments contained in this submittal.

If you have any questions or require additional information, please contact Mr. David Mannai at (802) 258-5422.

Sincerely,

Ted A. Sullivan
Site Vice President
Vermont Yankee Nuclear Power Station

Attachment: Vermont Yankee Cycle 25 10CFR50.59 Report

cc: (next page)

IE47

NRR

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Docket No. 50-271

Attachment

Vermont Yankee Nuclear Power Station

Vermont Yankee Cycle 25 10CFR50.59 Report

Vermont Yankee Cycle 25 10CFR50.59 Report

Between November 12, 2005 and June 6, 2007, Vermont Yankee implemented six changes requiring evaluation in accordance with 10CFR50.59. This report provides a summary of the evaluation performed for each change. The evaluations were reviewed by the On-Site Safety Review Committee (OSRC), approved by the OSRC Chairman and concluded that prior Nuclear Regulatory Commission review and approval was not required.

10CFR50.59 Evaluation No. 2005-02 Rev. 0, Engineering Request (ER) 04-1272, "Reactor Core Isolation Cooling (RCIC) System Suction Valve Isolation Upgrade"

This ER changed the RCIC system auto isolation logic to include closing the RCIC Torus Suction valves on an auto isolation signal, overriding the open signal from the Condensate Storage Tank low level signal.

This change did not increase the probability of occurrence of any previously analyzed accident or malfunction because neither the RCIC system nor RCIC auto isolation circuit have any impact on accident and malfunction initiators. This change did not increase the consequences of any previously analyzed accident or malfunction because the change will not degrade the functioning of the RCIC auto isolation logic to deal with accidents or malfunctions. This change did not create the possibility of an accident or malfunction of a different type that previously analyzed because no new failure modes are being introduced. This change did not result in a design basis limit being exceeded or altered because it does not involve structures, systems or components with the potential to affect fission product barriers and there is no adverse impact on any other systems. This change did not result in a departure from any method of evaluation because the change does not involve any methods of evaluation described in the UFSAR.

10CFR50.59 Evaluation No. 2006-01 Rev. 0, Special Test Instruction (STI) 05-VY1-0003-000, "Residual Heat Removal Service Water (RHRSW) System Pump High Flow Test "

This test was implemented to demonstrate that running two RHRSW system pumps in parallel will achieve the desired flow rate of 4900-5000 gpm with the RHRSW system aligned with the supply from the Service Water (SW) system and the RHR Heat Exchanger in service. This test was required to demonstrate the capability of the RHRSW system to support mitigation of events such as Station Blackout (SBO), Loss of Auxiliary Power, Relief Valve Discharge Transient and Appendix R Fires.

This change did not increase the probability of occurrence of any previously analyzed accident or malfunction because the RHRSW system is not an initiator of an accident and no new failure modes are introduced. Operation of the RHRSW system with two pumps in operation at the established flow rates has been evaluated to be within the capabilities and design parameters of the RHR system. This change did not increase the consequences of any previously analyzed accident or malfunction because the mitigation capability of the RHRSW system is not

impacted. This change did not create the possibility of an accident or malfunction of a different type than previously analyzed because the test is not the initiator of any new malfunction and no new failure modes are being introduced. This change did not result in a design basis limit being exceeded or altered because the test will not result in operation of the system or its components outside their design capabilities. This change did not result in a departure from any method of evaluation because the test does not involve any methods of evaluation described in the UFSAR.

10CFR50.59 Evaluation No. 2006-02 Rev.0, ER 04-1273, "High Pressure Coolant Injection (HPCI) Manual Initiation of Automatic Isolation"

This ER changed the HPCI auto isolation circuit to add a manual switch to allow the operator the capability to manually initiate an automatic isolation.

This change did not increase the probability of occurrence of any previously analyzed accident or malfunction because the HPCI auto isolation circuit has no impact on accident and malfunction initiators. This change did not increase the consequences of any previously analyzed accident or malfunction because the change will not degrade the functioning of the HPCI auto isolation circuit to deal with accidents or malfunctions. This change did not create the possibility of an accident or malfunction of a different type than previously analyzed because no new failure modes are being introduced. This change did not result in a design basis limit being exceeded or altered because it does not involve structures, systems or components with the potential to affect fission product barriers and there is no adverse impact on any other systems. This change did not result in a departure from any method of evaluation because the change does not involve any methods of evaluation described in the UFSAR.

10CFR50.59 Evaluation No. 2006-03 Rev. 0, Temporary Modification (TM) 2006-011, "Remove 'Ones' Position from Control Rod Drive (CRD) Position Indication (PIP) Probe 26-07 to the Rod Position Indicating System (RPIS)"

This TM addressed the PIP "01" switch problem by disabling the "ones" input line from the PIP for rod 26-07 to its associated RPIS probe buffer card. With the TM installed, intermediate rod positions 01,11,21,31 and 41 will not be indicated on the Control Room Panel 9-5 full core or 4-rod displays for rod 26-07. All other rod indications will remain unchanged.

This change did not increase the probability of occurrence of any previously analyzed accident or malfunction because the RPIS has no impact on accident and malfunction initiators. This change did not increase the consequences of any previously analyzed accident or malfunction because the RPIS is not relied upon to deal with accidents or malfunctions. This change did not create the possibility of an accident or malfunction of a different type than previously analyzed because no new failure modes are being introduced. This change did not result in a design basis limit being exceeded or altered because it does not involve structures, systems or components with the potential to affect fission product barriers and there is no adverse impact on any other systems. This change did not result in a departure from any method of evaluation because the change does not involve any methods of evaluation described in the UFSAR.

10CFR50.59 Evaluation No. 2007-01 Rev. 0, Special Test Procedure (STP) 2007-01 "Hydraulic Performance Test of the Alternate Cooling System (ACS)"

This STP is the second 10 year test of the ACS. The test was performed to demonstrate that the RHRSW system in the ACS mode can meet its hydraulic performance requirements. This test satisfies commitments made in response to Generic Letter 89-13 "Service Water System Problems Affecting Safety Related Equipment" and to implement In-service Testing Program requirements.

This change did not increase the probability of occurrence of any previously analyzed accident or malfunction because the test was performed with the reactor shutdown and in the refuel mode and had no impact on accident initiators. USFAR Malfunctions involving the RHRSW and ACS systems were reviewed and it was concluded that there was no more than a minimal increase in the likelihood. This change did not increase the consequences of any previously analyzed accident or malfunction because the reactor will be shutdown and in the refuel mode with the reactor cavity flooded and the spent fuel pool gates removed. Based on this, the system was still available to perform accident mitigation if required. This change did not create the possibility of an accident or malfunction of a different type than previously analyzed because the test does not alter any equipment, system performance or operator actions and UFSAR malfunctions and accident types remain bounding. This change did not result in a design basis limit being exceeded or altered because it does not involve structures, systems or components with the potential to affect fission product barriers and there will be no adverse impact on any other systems. This change did not result in a departure from any method of evaluation because the change does not involve any methods of evaluation described in the UFSAR.

10CFR50.59 Evaluation No. 2007-02 Rev. 0, Changes to Procedures OP1403 "Fuel Bundle Non-Destructive Testing and Reconstitution," Rev. 25, OP1405 "Water Submersible Gamma Spectrometer," Rev. 1, and changes to Technical Specification (TS) Bases section 3.7.C.

This 10CFR50.59 evaluation supported changes to the above procedures and the referenced TS Bases to restrict the movement of multiple irradiated fuel rods and allow the movement of an individual irradiated fuel rod when secondary containment is not operable. This supported performance of gamma scanning activities during the refueling outage, for periods of time when secondary containment was not operable.

This change did not increase the probability of occurrence of any previously analyzed accident or malfunction because the movement of a single fuel pin does not introduce the possibility of a change in the frequency of occurrence of an accident. A refueling accident (RA) is already considered in the UFSAR and this would be considered a subset of the RA. The basis for the RA is that an irradiated fuel assembly is dropped within the reactor core. The handling of an individual fuel pin within the spent fuel pool does not increase the probability of the RA or other malfunction as defined in the UFSAR. This change did not increase the consequences of any previously analyzed accident or malfunction because the impact of a dropped fuel pin without secondary containment was evaluated and the consequences were determined to be bounded by the UFSAR RA. This change did not create the possibility of an accident or malfunction of a

different type than previously analyzed because a RA is already analyzed in the UFSAR. This change did not result in a design basis limit being exceeded or altered because the movement will be performed in the spent fuel pool, with the plant shutdown, the reactor head removed, the spent fuel pool gates removed and the cavity flooded. This change did not result in a departure from any method of evaluation because the change does not involve any methods of evaluation described in the UFSAR.