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March 20, 2003
BVY 03-027

U.S. Nuclear Regulatory Commission
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References: (a) Letter, USNRC to VYNPC, "TMI Action Plan Item II.K.3.3, Reporting of Relief Valve and Safety 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)
Vermont Yankee Cycle 22 10CFR50.59 Report

In accordance with 10CFR50.59, attached is a copy of the Vermont Yankee (VY) Cycle 22 10CFR50.59 Report. This report contains a brief description of the 50.59 evaluations that supported changes, tests and experiments made between May 21, 2001 and October 5, 2002.

Additionally, in accordance with Reference (a), VY reports that there were no Relief Valve or Safety Valve failures or challenges during this period.

We trust that the information provided is adequate; however, should you have questions or require additional information, please contact Ronda Daflucas at (802) 258-4232.

Sincerely,

Gautam Sen
Manager, Licensing

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

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SUMMARY OF VERMONT YANKEE COMMITMENTS

BVY NO.: 03-027 Vermont Yankee Cycle 22 10CFR50.59 Report

The following table identifies commitments made in this document by Vermont Yankee. Any other actions discussed in the submittal represent intended or planned actions by Vermont Yankee. They are described to the NRC for the NRC's information and are not regulatory commitments. Please notify the Licensing Manager of any questions regarding this document or any associated commitments.

COMMITMENT	COMMITTED DATE OR "OUTAGE"
None	N/A

Docket No. 50-271
BVY 03-027

Attachment 1

Vermont Yankee Nuclear Power Station

Cycle 22 10CFR50.59 Report

Vermont Yankee Cycle 22 10CFR50.59 Report

Between May 21, 2001 and October 5, 2002, Vermont Yankee (VY) implemented a number of changes requiring evaluation in accordance with 10CFR50.59. This report includes the 10CFR50.59 evaluation summaries for Vermont Yankee Design Changes (VYDCs), Minor Modifications (MMs), Temporary Modifications (TMs), Installation and Test Procedures (I&Ts), Basis for Maintaining Operability (BMOs), Special Test Procedures (STPs), procedure changes, Updated Final Safety Analysis Report (UFSAR) Changes and Setpoint Changes.

The following changes did not require prior Nuclear Regulatory Commission approval. They were reviewed by the Plant Operations Review Committee and approved by the General Manager.

Vermont Yankee Design Change (VYDC) 99-006, "Main Steam Line Break (MSLB) at Low Power Analysis"

General Summary

VYDC 99-006 was written to revise the plant design basis to document acceptable plant performance in the event of a main steam line break at low power.

50.59 Evaluation Summary **(50.59 Evaluation 2000-04)**

This design change incorporated a new Main Steam Line Break (MSLB) analysis at the hot standby condition into the plant design basis. The analysis evaluated the MSLB at low power. The affect on post MSLB environmental profiles in the reactor building and the differential pressure across required structures, including block walls was also evaluated. UFSAR Section 14.6.5 was updated to reflect the analysis. This change did not modify any physical plant system, structure or component or the way any system, structure or component functioned.

There was no increase in the probability of occurrence or consequences of an accident or malfunction as previously evaluated in the UFSAR as a result of this modification. This change did not present significant hazards not described or implicit in the UFSAR and there was reasonable assurance that the health and safety of the public was not endangered.

Vermont Yankee Design Change (VYDC) 99-007 Rev. 1, "Containment Response Analysis"

General Summary

This design documented and incorporated the results of a series of containment response analyses performed by General Electric (GE). The original design change incorporated peak pressure aspects of the analyses and was supported by Safety Evaluation (SE) 2000-035.

This revision added the containment response to smaller breaks including Small Break Analysis (SBA), Intermediate Break Analysis (IBA) and Small Steam Line Break Analysis (SSLB). The results show that the peak containment pressures and temperatures are bounded by the results in

the PULD for the IBA and SBA. For the SSLB the drywell shell temperature remains below the design value. The peak drywell airspace temperature increased to 336.1 degrees F. For bounding purposes, the changes used 340 degrees F.

GE used the LAMB, M3CPT and SHEX computer codes for the analyses. The use of LAMB and M3CPT were covered previously in SE 2000-035. For the smaller break analyses, GE used the SHEX computer code. GE has used this code to model many BWRs.

In support of this change, the Environmental Qualification (EQ) Program was revised to incorporate the higher drywell airspace temperature.

50.59 Evaluation Summary **(50.59 Evaluation 2002-01)**

VYDC 99-007 documented the results of Containment System response analyses. This 50.59 evaluation concluded that the results of the analyses had no affect on the accidents and malfunctions previously evaluated in the UFSAR. Further, this change had no potential for creation of a new type of event not previously evaluated in the UFSAR. It also did not have a deleterious impact on the fission product barriers described in the UFSAR. The change in methods was benchmarked and found to produce results either the same or conservative with respect to previous analyses used in the UFSAR.

Vermont Yankee Design Change (VYDC) 2000-001, "Drywell Floodup Level Switch"

VYDC 2000-001 replaced the Drywell Flood Level Alarm Switch (LSH-16-19-26) with a multi-level float switch which provides the required indication with minimal drift. Two new cables were also installed, one which penetrated the boundary between the Control Room and the Cable Vault.

50.59 Evaluation Summary **(50.59 Evaluation 2001-016)**

This evaluation was written because the switch vessel and sensing line are part of the primary containment pressure boundary. Both the switch vessel and sensing line were qualified and seismically mounted to provide boundary integrity. The level switch is classified as safety class 2/N (SC2 mechanical, NNS electrical). Cable routing was in accordance with Vermont Yankee's Separation Criteria. This design change did not involve breaching secondary containment because isolation valves in the sensing line outside the containment were closed while the work was performed.

There was no increase in the probability of occurrence or consequences of an accident or malfunction as previously evaluated in the UFSAR as a result of this modification. This change did not present significant hazards not described or implicit in the UFSAR and there was reasonable assurance that the health and safety of the public was not endangered.

Vermont Yankee Design Change (VYDC) 2000-007, Major Change 2, "Service Water/Fire Pump Flow Test Loop"

General Summary

VYDC 2000-007 was a change to the design that reduced the maximum flow rate to be achieved during pump testing. The reduced flow remains in excess of the previous maximum demonstrated flow rate – a flow rate that was sufficient to demonstrate Technical Specification compliance.

50.59 Evaluation Summary **(50.59 Evaluation 2000-023, Revision 2)**

The reduction of the maximum flow rate during pump testing did not increase the probability of occurrence or consequences of an accident or malfunction as previously evaluated in the UFSAR. This change did not present significant hazards not described or implicit in the UFSAR and there was reasonable assurance that the health and safety of the public was not endangered.

Vermont Yankee Design Change (VYDC) 2000-009, "Cycle 22 Reload"

General Summary

VYDC 2000-009 provided information that supported the operation of the Vermont Yankee Nuclear Power Station through Reload 21/Cycle 22. The Reload 21/Cycle 22 refueling replaced 88 fuel assemblies, 4 control blades and 7 Local Power Range Monitors (LPRM's). The 88 fuel bundles were GE-13 type fuel bundles and replaced 88 irradiated GE-9B fuel bundles. The GE-13 type fuel bundles were the same type used in the previous two cycles. The control rod blades being replaced met or exceeded the requirements for the current design, therefore there was no impact on the safety analysis.

Seven LPRM's were also replaced in various core locations with an equivalent replacement. LPRMs do not affect or control core reactivity and are not considered in the reload transient or accident analysis. Since the LPRMs are not used in the licensing analysis, replacing the LPRMs with an equivalent component did not have an impact on the design change or reload safety analysis.

50.59 Evaluation Summary **(50.59 Evaluation 2001-019)**

VYDC 2000-009 provided a detailed evaluation of the impact on the plant design bases as a result of the Reload 21/Cycle 22 core design. An evaluation of the power generation design bases and the safety design bases was performed for each system that was potentially impacted. In all cases, the designs presented in the VYDC maintained the design bases for that respective system. The reload methodologies employed for Cycle 22 are those of the General Electric Company and are approved for VY. These methods are identified in the Technical Specifications.

There was no increase in the probability of occurrence or consequences of an accident or malfunction as previously evaluated in the UFSAR as a result of this modification. This change did not present significant hazards not described or implicit in the UFSAR and there was reasonable assurance that the health and safety of the public was not endangered.

Vermont Yankee Design Change (VYDC) 2000-017, "CRP 9-21 Recorder Replacement"

General Summary

VYDC 2000-017 replaced six Leeds & Northrup multipoint temperature recorders in Control Room Panel (CRP) 9-21 with an Allen-Bradley Programmable Logic Controller (PLC) based temperature display system. Both the existing system and the new PLC system were classified as Non-Nuclear Safety (NNS). The existing recorders provided various temperature indications of the Recirculation Pump/Motor-Generators, High Pressure Coolant Injection pump, RHR system, Fuel Pool Cooling system, Safety/Relief valve positions, and Reactor Vessel. The PLC system will accept the thermocouple and RTD temperature inputs from these systems, convert those outputs to a digital value corresponding to temperature and display the temperatures on a multiple page touch screen video display. The PLC system will also generate alarms in the Control Room if selected inputs exceed preset setpoints. Concurrent with these alarms, the associated input display on the touchscreen monitor will change color to allow plant operators to determine the parameter in alarm.

50.59 Evaluation Summary **(50.59 Evaluation 2001-035)**

Specific operator inputs and operational needs were factored into the video display screen development for the PLC system to ensure that the video displays met operator requirements. Additionally, the displays were reviewed to verify that no human factor concerns were created.
A Validation and Verification Program was performed prior to installation of the PLC system to ensure that interaction between the PLC system and Emergency Response Facility Information System (ERFIS) was correct, accurate and controlled.

There was no increase in the probability of occurrence or consequences of an accident or malfunction as previously evaluated in the UFSAR as a result of this design change. This change did not present significant hazards not described or implicit in the UFSAR and there was reasonable assurance that the health and safety of the public was not endangered.

Vermont Yankee Design Change (VYDC) 2000-027, "Main Turbine Electronic Pressure Regulator (EPR) Replacement"

General Summary

VYDC 2000-027 replaced the existing Electronic Pressure Regulator (EPR) with a new EPR. The new EPR operation is based around a PLC that performs main steam pressure regulating functions. Since the original EPR was a totally analog electronic pressure regulating system, and the new EPR is a hybrid system consisting of analog and digital electronics, it represents an analog-to-digital upgrade, and was evaluated against the guidance of Generic Letter (GL) 95-02.

50.59 Evaluation Summary **(50.59 Evaluation 2002-03)**

GL 95-02 provides guidance such that a determination of whether or not an unreviewed safety question exists with respect to 10CFR50.59, and is primarily focused on Analog-to-Digital upgrades used in safety related applications. The EPR replacement governed by this design change was classified as Non-Nuclear Safety (NNS) related. However, there were concerns stemming from the design characteristics specific to the new digital electronics that could result in failure modes and system malfunctions that either were not considered during the initial plant design, or may not have been evaluated in sufficient detail in the safety analysis report. These concerns included potential common mode failures due to:

1. The use of common software in redundant channels.
2. Increased sensitivity to the effects of electromagnetic interference.
3. The improper use and control of equipment used to control and modify software and hardware configurations.
4. The effect that some digital designs have on diverse trip functions.
5. Improper system integration.
6. Inappropriate commercial dedication of digital electronics.

The above concerns were evaluated against the guidance contained in GL 95-02 and addressed.

All control board equipment installed by this design was reviewed to ensure that no seismic II/I concerns were created, and to ensure that no human factors concerns were created by the design change. All fire and EQ barrier penetrations affected by this design change were restored to their original ratings such that no Fire or EQ concerns were created. The control functions provided by the new EPR mimicked the original EPR such that all assumptions in VY analysis relative to turbine pressure response remained unchanged.

There was no increase in the probability of occurrence or consequences of an accident or malfunction as previously evaluated in the UFSAR as a result of this design change. This change did not present significant hazards not described or implicit in the UFSAR and there was reasonable assurance that the health and safety of the public was not endangered.

Vermont Yankee Design Change (VYDC) 2000-030, Major Change #14, "Replacement of 24VDC ECCS Batteries with DC Power Supplies"

General Summary

The original design change replaced the Safety Class Electrical (SCE) ECCS System A and B battery chargers with new SCE DC/DC converters. The design also installed an Appendix R DC/DC converter which opened in parallel with the ECCS System B converter. Major Change #14 to the design change eliminated the parallel operation of the ECCS B converter and the Appendix R converter.

50.59 Evaluation Summary **(50.59 Evaluation 2001-004 Revision 1)**

This 50.59 evaluation was written to address elimination of the parallel operation of the ECCS System B converter and the Appendix R converter as normally open circuit breaker now provides the isolation between the Appendix R converter and the SCE (Division SII) equipment. The converter, fuses and diodes previously were credited as performing the isolation function.

There was no increase in the probability of occurrence or consequences of an accident or malfunction as previously evaluated in the UFSAR as a result of this design change. This change did not present significant hazards not described or implicit in the UFSAR and there was reasonable assurance that the health and safety of the public was not endangered.

Vermont Yankee Design Change (VYDC) 2001-002, Rev. 0, "Feedwater Control System Replacement – Phase 1"

General Summary

VYDC 2001-002 upgraded several components that are part of the Feedwater Control (FWC) System. The change involved replacement of three old analog controller units with three new digital controller units that will be integrated with the remaining devices that make up the FWC system. The Feedwater System Flow Control Valve (FCV) electro-pneumatic (I/P) transducers and mechanical valve positioners were replaced on each FCV with a new digital electro-pneumatic valve positioner. The upgraded FWC system will continue to perform its original design objective of regulation of feedwater flow to maintain proper water level in the reactor vessel during planned operations.

50.59 Evaluation Summary **(50.59 Evaluation 2002-06)**

This evaluation identified that the design change should be evaluated with respect to the guidance contained in GL 95-02 for analog-to-digital upgrade concerns. The evaluation of the six areas of concern under GL 95-02 determined that this design change did not require prior NRC approval.

The 50.59 evaluation considered the effect of this design change on the causes of accidents and malfunctions previously evaluated, and the consequences of those accidents and malfunctions, with no increase in the potential for cause of or the consequences of the events considered. In addition, the evaluation concluded the design changes as described above, would not provide the potential for creation of an accident or malfunction of a different type than previously evaluated. It was determined that the proposed design change would have no impact on fission product barriers as described in the UFSAR and that it would not impact the methodologies used in establishing the design bases or in the safety analyses.

Minor Modification (MM) 99-038, "Replacement of Magnetic Only HFB327OML Circuit Breaker With Thermal Magnetic HFD3015L Circuit Breaker in MCC-9A-2D"

General Summary

MM 99-038 replaced the original magnetic only breaker in MCC-9A-2D feeding standby gas treatment electric heater EUH-2 with a thermal magnetic breaker which will provide the appropriate heater circuit protection. The original design specified a magnetic only breaker. A thermal magnetic breaker with a contactor is the proper breaker to use for a non-motor load which requires remote control.

50.59 Evaluation Summary **(50.59 Evaluation 2000-053)**

This 50.59 evaluation was required because the existing configuration rendered the system different in design than that described in the UFSAR. UFSAR drawing G-191301, Sheet 1, showed the breaker as a magnetic only breaker. The drawing was updated to reflect the installation of the thermal magnetic breaker.

There was no increase in the probability of occurrence or consequences of an accident or malfunction as previously evaluated in the UFSAR as a result of this minor modification. This change did not present significant hazards not described or implicit in the UFSAR and there was reasonable assurance that the health and safety of the public was not endangered.

Minor Modification (MM) 2000-038, Replacement of Molded Case Circuit Breakers on DC-3 Distribution Panel"

General Summary

MM 2000-038 replaced all molded case circuit breakers on the DC-3 distribution panel. Most of these breakers were original plant equipment and were two-pole breakers. These breakers were replaced with three-pole breakers. The change from 2-pole to 3-pole breakers did not impact the breaker physical size, weight or performance. The breakers and DC-3 distribution panel were added to the Preventive Maintenance (PM) Program after completion of the Minor Modification.

50.59 Evaluation Summary **(50.59 Evaluation 2001-014)**

All breakers replaced under MM 2000-038 were classified Safety Class Electrical (SCE). Those breakers previously designated "magnetic" were replaced with "thermal magnetic" breakers. This was a result of a recommendation to improve cable protection.

This modification did not increase the probability of occurrence or consequences of an accident or malfunction as previously evaluated in the UFSAR nor did it present significant hazards not described or implicit in the UFSAR. There was reasonable assurance that the health and safety of the public was not endangered.

Minor Modification (MM) 2000-048, "P-37-1A/B Obsolete Pump Switch Removal from CRP 9-4"

General Summary

The function of the Radwaste decontamination shop sump was to gather chemical decontamination solution wastewater from a decontamination room located in the Tool Crib mezzanine located in the Turbine Building. The decontamination shop sump pumps, through manual or automatic operation, transfer this fluid from the sump to the Chemical Waste Tank (TK-19-1A), for subsequent processing. The decontamination shop sump pumps were previously electrically removed from service (power cables disconnected at MCC 10B compartments 3K and 3M).

MM 2000-048 removed the control circuitry, switches and indication lamps from panel CRP 9-4 of both decontamination shop sump pumps (P-37-1A & P-37-1B). This was accomplished by retiring/abandoning in place the associated control /interlock level switch (20-481) and the associated cabling (C11004D and C11004E) and conduits. This MM also removed the internal wiring from the terminal block (TB JJ) to the indication lights (20A-DS8A, DS-8B, DS10A and DS-10B) and control switches (20A S3A and S3B), removed the switches and indication lights, and filled in the switch and indication light holes. The sump, sump pumps and associated cables, conduit and piping, located below the RCA Tool Crib, were retired/abandoned in place.

50.59 Evaluation Summary **(50.59 Evaluation 2001-005)**

The decontamination shop sump pump is part of the Radioactive Waste Processing System and is not described in the Vermont Yankee Technical Specifications. A review of the UFSAR did not identify any text that describes the operation of the decontamination shop sump pumps. UFSAR Section 9.2.4 referenced Drawing G-191177, Sheets 1 through 4. The control switches RMS S3A and S3B were removed from G-191177 Sheet 1 and a note was added to all of the drawings stating that "The sump, sump pumps, float level switch functions (including associated cables and conduits) and piping are abandoned in place".

This change did not increase the probability of occurrence or consequences of an accident or malfunction as previously evaluated in the UFSAR. This change did not present significant hazards not described or implicit in the UFSAR and there was reasonable assurance that the health and safety of the public was not endangered.

Minor Modification (MM) 2001-022, "Core Spray System Component Safety Class Re-Classification"

General Summary

MM 2001-022 was an administrative change only; there were no physical or functional changes to the plant. The MM revised UFSAR drawing G-191168, "Core Spray System" to correct the safety classification of 6 components. The safety classification of orifices (RO-31A/B) and associated piping of the core spray piping D/P instrument line was changed from SC2 to SC1.

The orifice plates (RO-42A/B) in the full flow test line were changed from NNS to SC2 and the safety classification of the core spray pump casing drain valves V14-28A/B were changed from NNS to SC2.

50.59 Evaluation Summary **(50.59 Evaluation 2001-036)**

The above listed components were erroneously classified during the original classification process. All components involved were original plant equipment. A review of all plant records including the maintenance history records determined that no work orders were performed on these components that would adversely affect their re-classification. There were no program changes required and all seismic requirements were unaffected by the corrections.

This change did not increase the probability of occurrence or consequences of an accident or malfunction as previously evaluated in the UFSAR. This change did not present significant hazards not described or implicit in the UFSAR and there was reasonable assurance that the health and safety of the public was not endangered.

Minor Modification (MM) 2001-035, "Replacement of RATS-1A Diesel Room Temperature Controller"

General Summary

MM 2001-035 replaced the existing obsolete analog RATS-1A Diesel Room Temperature Controller with a new Safety Class Electrical (SCE) Honeywell digital controller. The existing circuit uses two temperature switches and a seal-in relay to start and stop the Diesel Room Temperature Controller. The new controller has an adjustable deadband so that the controller output contact can be used to control the fan directly without a seal-in relay. The seal-in relay and the Auto/Manual selector switch in the existing controller were eliminated.

50.59 Evaluation Summary **(50.59 Evaluation 2001-038)**

The new controller resulted in improved control and higher reliability. It did not increase the frequency or consequence of an accident or malfunction. It did not create the possibility of a different accident or malfunction than previously evaluated. It did not result in a design limit or fission product barrier being exceeded or altered. It did not change any method of evaluation. It has been qualified for its application. Therefore, this change did not require prior NRC approval.

Minor Modification (MM) 2001-038, "Stator Leak Monitoring System"

General Summary

MM 2001-038 installed the GE supplied Stator Leak Monitoring System (SLMS). The system is composed of a sample pump; H2 analyzer; data acquisition and display unit; and an air treatment system that removes moisture, CO2 and particulate from the air supply and measures air flow. The unit provides air injection to the bottom of the Stator Water Storage Tank through the tank drain to provide vent flow and aerate the water. The unit has a high H2 alarm contact that

indicates high leakage. It is connected to a spare annunciator point in the H2 & Stator Water Cooling Panel which in turn causes a group alarm on CRP 9-7. A 110 VAC power feed was provided from the associated H2 & Stator Water Cooling Panel and a 6.5 CFM, 85 psi Instrument Air supply was provided.

Summary of 50.59 Evaluation **(50.59 Evaluation 2002-04)**

The SLMS provides diagnostic information to monitor stator leakage. It did not increase the frequency or consequence of an accident or malfunction. It did not create the possibility of a different accident or malfunction than previously evaluated. It did not result in a design limit or fission product barrier being exceeded or altered. It did not change any method of evaluation. Therefore, this change did not require prior NRC approval.

Minor Modification (MM) 2001-046, "Replacement of RATS-1B Diesel Room Temperature Controller"

General Summary

MM 2001-046 replaced the existing analog RATS-1B Diesel Room Temperature Controller with a new SCE Honeywell digital controller. The existing controller had a non-adjustable deadband. The existing circuit used two temperature switches and a seal-in relay to start and stop the Diesel Room Fan. The new controller has an adjustable deadband so that the controller output contact can be used to control the fan directly without a seal-in relay. The seal-in relay was eliminated but the function remained the same. The Auto/Manual selector switch was also eliminated since the preferred method of overriding the thermostat is the local key lock Auto/Run switch. Several cables were physically relabeled in the plant as SCE (SI) to reflect their safety class division.

50.59 Evaluation Summary **(50.59 Evaluation 2002-02)**

The new controller results in improved control and higher reliability. It does not increase the frequency or consequence of an accident or malfunction. It does not create the possibility of a different accident or malfunction than previously evaluated. It does not result in a design limit or fission product barrier being exceeded or altered. It does not change any method of evaluation. It has been qualified for its application. Therefore, this change did not require prior NRC approval.

Temporary Modification (TM) 2001-018, "Connection of Recorder to Monitor Electric Pressure Regulator (EPR) Operation"

General Summary

TM 2001-018 connected a recorder to the EPR power supply and control signals to monitor EPR operation for step changes and other instabilities. Changes and instabilities were noted which caused reactor pressure perturbations. This monitoring did not affect the operation of the EPR and did not result in a significant load change to the circuit. There were no seismic issues associated with any safety related equipment.

50.59 Evaluation Summary **(50.59 Evaluation 2001-033)**

The connection of this recorder to the EPR power supply did not add a significant load to the EPR circuit. There were no seismic interaction concerns with safety related equipment. This recorder did not perform any active safety function and was classified as NNS.

This change did not increase the probability of occurrence or consequences of an accident or malfunction as previously evaluated in the UFSAR. This change did not present significant hazards not described or implicit in the UFSAR and there was reasonable assurance that the health and safety of the public was not endangered.

Temporary Modification (TM) 2001-019, "Connection of Recorder for Temporary Monitoring of the Feedwater Regulating Valve Control Loop for Troubleshooting"

General Summary

TM 2001-019 authorized the long-term installation of a recorder for monitoring various analog points which make up the inputs and control scheme to the feedwater regulating valve control loop. The TM was required to allow monitoring of the control signal at various points to attempt to determine if any were acting on the control loop in an adverse manner. There was no change to system operation as a result of this TM.

50.59 Evaluation Summary **(50.59 Evaluation 2001-034)**

The connection of this recorder did not add a significant load to the affected circuit. There were no seismic interaction concerns with safety related equipment. This recorder did not perform any active safety function and was classified as NNS.

This change did not increase the probability of occurrence or consequences of an accident or malfunction as previously evaluated in the UFSAR. This change did not present significant hazards not described or implicit in the UFSAR and there was reasonable assurance that the health and safety of the public was not endangered.

Installation and Test (I&T) Procedure 2002-002, "Electronic Pressure Regulator (EPR) and Feedwater Control (FWC) System Tuning and Dynamic Response Verification

General Summary

I&T procedure 2002-002 provided the steps necessary to tune/verify the dynamic response of the EPR and FWC component replacements. Verifying the EPR system assures that the turbine control/bypass valve response closely mimics the response of the replaced system. Testing the EPR included changing reactor pressure settings and verifying the expected response. These pressure perturbations are larger than used in normal surveillances, but necessary to ensure response of the controller over a wider pressure range. The perturbation steps range from 1 psi to 10 psi and are based on 1972 plant startup tests and GE recommendations for tuning the EPR.

The procedure cautioned the operator to closely observe plant response when the perturbation is initiated. The expected controller response is to dampen the pressure perturbation and return reactor pressure to the EPR setpoint value.

The Mechanical Pressure Regulator (MPR) was used as the backup regulator in the event the EPR failed. The MPR was available at all times to dampen a pressure perturbation in the unlikely event that the EPR did not fulfill that function.

Tuning the FWC system included adjusting the control board mounted level controllers and local feedwater valve control components to adequately provide predictable and stable reactor water level control. Testing the FWC system included changing reactor level settings and verifying the expected response. The basis for the level of the perturbations is the 1972 startup tests. The expected FWC response was to restore normal level. Based on the first few tests, controller adjustments were made to speed or slow the response characteristics.

The effects of reactor water level dropping to 155" during the FWC portion of the test was evaluated and the UFSAR analysis remained bounding. The combination of lower decay heat and power level and performance of the testing during power ascension from RFO 23 more than compensate for lower reactor vessel level in the Appendix R, LOCA and FW Controller high flow UFSAR analyses. Additionally, the operators closely monitored plant responses and were ready to terminate testing in the event a failure occurred.

The I&T also required monitoring and data analysis during normal turbine valve testing, a typical turbine overspeed test and a feedwater pump swap to verify control system response. The procedure controlled temporary instrumentation and data recorders for tuning and testing.

<u>50.59 Evaluation Summary</u> **(50.59 Evaluation 2002-08)**

This evaluation concluded that the proposed tests did not invalidate results or assumptions stated in the UFSAR Chapter 14 Accident and Abnormal Operational Transients. This conclusion was based on a review of the test procedure and nature of the test performed. Each of the tests were performed within the plant's operating envelope, the replacement controls function to mitigate system perturbations to be performed in the tests and the range of the test perturbations that represented reasonable changes to plant conditions that were not expected to initiate a plant transient. Several analyses use a nominal reactor water level of 160". The effects of lower reactor water level was evaluated to ensure that the UFSAR analysis remained bounding. Test termination criteria was provided in the test procedure. In the unlikely event that a replacement controller malfunctioned to cause a plant transient, all protective systems were available to safely shut the plant down and consequences were within analyzed plant limits.

This change did not increase the probability of occurrence or consequences of an accident or malfunction as previously evaluated in the UFSAR. This change did not present significant hazards not described or implicit in the UFSAR and there was reasonable assurance that the health and safety of the public was not endangered.

Basis for Maintaining Operability (BMO) 2000-07, Rev. 1, "Potential Over-Pressurization of HPCI/RCIC on Turbine Over-Speed Trip – Closure Memo"

General Summary

BMO 2000-07 was written in response to GE Service Information Letter (SIL) 0623 that identified a potential non-conforming condition that could occur at Vermont Yankee. This condition identified the possibility that peak pressure could exceed pump, piping and attached components design pressure by more than code allowable when a turbine over-speed trip occurs due to a failure of the turbine speed controller when in the test or minimum flow modes of operation.

Evaluations conducted by the engineering staff indicated that the only identified potential failure would be leakage of the local flow indicator at the RCIC control panel and subsequent loss of function. There were no hardware changes to the RCIC or HPCI systems.

50.59 Evaluation Summary **(50.59 Evaluation 2001-029)**

Potential failures of the discharge piping, associated instruments, and the pumps were evaluated. The only identified potential failure was a local flow indicator in the RCIC System. The consequences of failure of the local flow indicator were leakage at the RCIC local control panel and loss of function (local flow indication). These consequences were not considered significant. Precautions were added to the operating procedures for the RCIC and HPCI systems to alert the operators that a mechanical overspeed trip has the potential to exceed allowable pressures.

There was no increase in the probability of occurrence or consequences of an accident or malfunction as previously evaluated in the UFSAR. This BMO did not present significant hazards not described or implicit in the UFSAR and there was reasonable assurance that the health and safety of the public was not endangered.

AP 0077, Rev. 0, "Barrier Control Process – Appendix H

AP 0077 provides the administrative requirements for breaching barriers, blocking vent or flood paths or impacting other special features at Vermont Yankee. There were two 50.59 evaluations associated with this procedure. The first 50.59 evaluation, SE 2001-008, was written for Appendix H, "Flooding Barriers". This Appendix deals with internal and external flooding of the plant. The second 50.59 evaluation, SE 2001-009, was written for Appendix E, "Control Room Envelope Barriers". This Appendix addressed special considerations and compensatory measures for breaching the Control Room Envelope (CRE).

50.59 Evaluation Summary **(50.59 Evaluation 2001-008)**

This 50.59 evaluation was written to address the compensatory measures for Flooding Barriers and those actions taken when a flooding barrier is breached or a flood relief path is blocked.

An assessment of INFO Notice 97-98 and NRC Inspection Manual Part 9900 was performed. With the exception of closing and opening doors and re-sealing an opening, all compensatory actions to address flooding were already contained within Vermont Yankee's licensing basis.

Therefore, this revision to AP 0077 did not increase the probability of occurrence or consequences of an accident or malfunction as previously evaluated in the UFSAR nor did it present significant hazards not described or implicit in the UFSAR. There was reasonable assurance that the health and safety of the public was not endangered.

50.59 Evaluation Summary **(50.59 Evaluation 2001-009)**

This 50.59 evaluation was performed to evaluate the impact of breaching the Control Room envelope and performance of compensatory actions. Per AP 0077, compensatory measures will be initiated that will allow for the immediate sealing of all open penetrations to the Control Room envelope within ten minutes. The ten-minute limitation was based on guidance provided in Amendment No. 18 to the UFSAR which estimated a ten minute transport time from the Reactor Building to the Stack and subsequently to the Control Room ventilation intake.

The compensatory measures in Appendix E ensured that there would be no increase in the radiological dose to Control Room personnel in the event of a DBA LOCA during barrier breaches. This revision did not increase the probability of occurrence or consequences of an accident or malfunction as previously evaluated in the UFSAR nor did it present significant hazards not described or implicit in the UFSAR. There was reasonable assurance that the health and safety of the public was not endangered.

FSAR Change 18/029, "Revise UFSAR Section 10.5.3 and OP 1101, OP 1102 and OP 2184"

General Summary

UFSAR Section 10.3.5 was revised to refine restrictions on movement of irradiated fuel and/or non-fuel, irradiated materials when the Fuel Pool gate is removed and the Normal Fuel Pool Cooling System (NFPCS) is aligned to the Reactor Well and/or Dryer/Separator pit. Prior to this revision, movement of irradiated fuel and irradiated non-fuel materials was not permitted when the NFPCS was aligned as described. Revision of this section allowed fuel movement to occur during periods of filter demineralizer operation for clarification of the Reactor Well.

50.59 Evaluation Summary **(50.59 Evaluation 2002-05)**

Existing UFSAR Section 10.3.5 indicated that specific restrictions exist for irradiated fuel movement during refueling operations with the Fuel Pool gate removed and the NFPCS aligned to the Reactor Well and/or Dryer Separator pit. These restrictions required that the fuel preparation machine be fully lowered, if it contained irradiated fuel, and that no fuel movement be permitted. These restrictions were supported by analysis which assumed that multiple failures of non-seismic piping occur causing a rapid decrease in Fuel Pool level. The acceptance criterion for this analysis limited the loss of Fuel Pool level so as not to uncover the suction of

Standby Fuel Pool Cooling System (SFPCS) and to ensure that 10 feet of water remained above any fuel stored in the spent fuel racks or fuel preparation machine.

The changes described are the result of additional analysis of the extent and consequence of the failure of non-seismic piping within the NFPCS. The new analysis credits the single, worst case, guillotine break of non-seismic piping and determines the impact on water level within the Fuel Pool prior to imposition of clearly defined, procedurally directed Operator actions. The crediting of a guillotine break is considered conservative in relation to guidance provided in NRC Branch Technical Position MEB-3-1 on moderate energy piping and also consistent with other Licensing basis failure analyses regarding postulated breaks in piping, (i.e. VY flooding analyses). The required Operator actions remain identical to those previously considered and credited within the UFSAR, and approved plant procedures. Operator diagnosis and response time are consistent with those promulgated within ANSI N58.8.

Therefore, this change did not increase the probability of occurrence or consequences of an accident or malfunction as previously evaluated in the UFSAR nor did it present significant hazards not described or implicit in the UFSAR. There was reasonable assurance that the health and safety of the public was not endangered.

Setpoint Change 2001C-043, "RCIC Turbine Exhaust High Pressure Trip

General Summary

Setpoint Change 2001C-043 increased the setpoint for the RCIC turbine exhaust line high-pressure trip switches (PSH-13-72A/B) from 27.7 PSIG to 42.3 PSIG. The increase was required to provide an operating margin against an inadvertent RCIC turbine trip caused by the installation of stronger springs in the RCIC turbine exhaust check valves, V13-6 and V13-7. The stronger springs were installed via Minor Modification 2001-020 to ensure that the valves would seal with acceptable leakage rates for containment isolation valves.

50.59 Evaluation Summary **(50.59 Evaluation 2001-032)**

The safety objective of the RCIC system is to provide makeup water to the reactor vessel during shutdown and isolation in order to prevent the release of radioactive materials to the environs as a result of inadequate core cooling. The change to the high exhaust pressure trip setpoint will provide additional margin against spurious trips, thus improving system availability in order to meet its safety objective. This setpoint change will not affect the ability of the valves to perform their containment isolation function.

There was no increase in the probability of occurrence or consequences of an accident or malfunction as previously evaluated in the UFSAR as a result of this setpoint change. This change did not present significant hazards not described or implicit in the Vermont Yankee UFSAR and there was reasonable assurance that the health and safety of the public was not endangered.