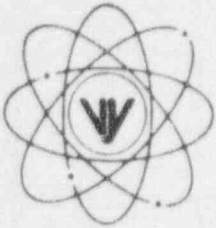


VERMONT YANKEE NUCLEAR POWER CORPORATION



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November 6, 1995
BVY 95 - 110

United States Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, DC 20555

References: a. License No. DPR-28 (Docket No. 50-271)

Subject: Vermont Yankee Operating Report - Missed Safety Evaluation Summaries

While preparing the Cycle 17 Operating Report, Vermont Yankee determined that summaries of several 50.59(a)(2) safety evaluations written during Cycle 16 were not reported to the NRC in the Cycle 16 Operating Report as required by 50.59(b)(2). Using our new Event Reporting process, we reviewed previously submitted operating reports and identified additional summaries of safety evaluations not reported. The safety evaluation summaries not previously reported are attached.

Since discovering the oversight, Vermont Yankee has made improvements to its safety evaluation tracking program to ensure that all safety evaluation summaries are included in future operating reports. These improvements include assigning a tracking number to each 50.59(a)(2) safety evaluation and requiring the Plant Operations Review Committee secretary to establish and maintain a safety evaluation tracking number database.

Should you have any questions regarding the enclosed material, please contact this office.

Sincerely,

VERMONT YANKEE NUCLEAR POWER CORPORATION

James J. Duffy
Licensing Engineer

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United States Nuclear Regulatory Commission
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Page 2

Attachment

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

CYCLE 16

The following items were inadvertently omitted from the Cycle 16 Operating Report: Safety Evaluation for Removal of Biological Shield Wall Nozzle Shield Blocks, and seven Temporary Modifications (TMs).

1. Removal of Biological Shield Wall Nozzle Shield Blocks

General Summary

Prior to 1984, to facilitate in-service inspection (ISI) of reactor vessel nozzles, Vermont Yankee removed and re-installed the reactor vessel nozzle shield blocks each refueling outage. In 1984 Vermont Yankee determined that overall personnel radiation exposure would be reduced by permanent shield block removal. Therefore, the shield blocks were removed; however, the Final Safety Analysis Report (FSAR) was not updated to reflect this change.

In 1992, Vermont Yankee performed a 50.59(a)(2) safety evaluation supporting removal of the reactor vessel nozzle shield blocks, updated the FSAR and included this change in the 1992 FSAR update submittal. A summary of the safety evaluation supporting this change is provided below.

Safety Evaluation Summary

As a result of the block removal, radiation streaming occurs, subjecting equipment to an increased radiation dose per unit time. The consequence of that increase is a possible reduction in life of equipment sensitive to radiation. Evaluation of equipment adversely affected has determined that removal of the shield blocks will not result in additional EQ equipment replacement at least through the current licensed life of Vermont Yankee. An elevated neutron dose was detected via radiation surveys at the core spray piping penetrations in the reactor building; these areas have been suitably posted in accordance with the plant radiation protection requirements.

The nozzle shield blocks played no role in preventing the occurrence of any accidents, nor in mitigating the consequences of any accidents; nor did this action increase the probability of occurrence of a malfunction of equipment important to safety. The only effect of the shield block removal is increased radiation dose, and all affected components have been evaluated for the increased dose. No additional equipment replacements are required due to shield block removal through at least the year 2012.

There was no increase in the probability of occurrence or consequences of an accident or malfunction as previously evaluated in the FSAR. The removal of nozzle shield blocks did not present significant hazards not described or implicit in the

Vermont Yankee FSAR, and there is reasonable assurance that the health and safety of the public was not endangered.

2. **Temporary Modification 93-020** was installed 8/29/93 and removed 10/15/93.

General Summary

To maintain condensate transfer system operability while the condensate storage tank (CST) was drained for repair, this temporary modification provided a suction path from condensate hotwell "A" to the condensate transfer pumps suction header. Normal supply to the condensate transfer pumps is from the condensate storage tank. This TM supplied suction pressure to the CST pumps via two sump pumps installed in the hotwell. The two pumps and filters were completely separate and capable of operating independently to supply water to the condensate transfer pumps.

Safety Evaluation Summary

This temporary modification was operated with the reactor in cold shutdown to support system operation for the condensate transfer system and to maintain water inventory. The components used in this modification had pressure ratings well in excess of the shutoff head for the sump pumps utilized. No pressure boundary failure or pump operational failure would have increased the consequences of a condensate transfer system malfunction such that safety equipment or systems could have been affected. The worst case possibility would have been minor flooding in the condenser pit with low contamination levels. Due to the continuous outage work activity in this area, the leakage would have been quickly identified and secured in minutes. No EQ instrumentation would have been affected.

There was no increase in the probability of occurrence or consequences of an accident or malfunction as previously evaluated in the FSAR. This temporary modification did not present significant hazards not described or implicit in the Vermont Yankee FSAR, and there is reasonable assurance that the health and safety of the public was not endangered.

3. **Temporary Modification 93-027** was installed 9/1/93 and removed 9/29/93.

General Summary

This temporary modification installed temporary lead shielding on portions of the recirculation system piping. This shielding was required to reduce radiation exposure to personnel to support the 1993 refueling outage. This modification restricted the amount of shielding to limit loading on the piping. This modification had no effect on the system function.

Safety Evaluation Summary

Piping stresses remained within allowable design limits with the applicable shielding weight restrictions during all modes of operation while the shielding was in place. The addition of temporary shielding had no detrimental effect on the recirculation piping, nor on any equipment important to safety. Restrictions were established to provide protection to ensure that all activities were performed safely and that the system design bases were met.

There was no increase in the probability of occurrence or consequences of an accident or malfunction as previously evaluated in the FSAR. This temporary modification did not present significant hazards not described or implicit in the Vermont Yankee FSAR, and there is reasonable assurance that the health and safety of the public was not endangered.

4. **Temporary Modification 93-029** was installed 8/31/93 and removed 10/6/93.

General Summary

This modification rerouted the Reactor Water Cleanup (RWCU) return Bline to the Residual Heat Removal (RHR) system at valve V10-3 via the shutdown cooling loop in support of the maintenance activities on feedwater check valves V2-27B and V2-28B during the 1993 refueling outage.

Safety Evaluation Summary

Retention of the RWCU line break isolation circuitry ensured that the consequences of a small break Loss of Coolant Accident (LOCA) were within the original design basis envelope of the FSAR LOCA analyses. The material specifications were equal to the original specifications with the exception of the stainless steel flex pipe. This pipe was of adequate pressure rating to meet the pressure requirements of the system during shutdown. This modification did not introduce any new pipe break/leakage path which was not already analyzed for pipe breaks in the affected areas. The low energy of the contained fluid would have resulted in a lower environmental effect from a line break than the existing EQ high energy line break analysis.

There was no increase in the probability of occurrence or consequences of an accident or malfunction as previously evaluated in the FSAR. This temporary modification did not present significant hazards not described or implicit in the Vermont Yankee FSAR, and there is reasonable assurance that the health and safety of the public was not endangered.

5. **Temporary Modification 93-049** was installed and removed 9/1/93.

General Summary

This temporary modification installed a jumper to provide the full-in indication for Control Rod 26-39, and imposed administrative controls on the tag-outs used to secure the control rod drive (CRD) for rod 26-39. By isolating this rod in the full-in position and then supplying the full-in signal to the refueling circuitry, the logic for this rod was defeated for the single rod permissive, service platform jib crane hoist lock permissive, and permissive for refueling bridge in reverse direction. However, the refueling interlock circuitry for all other rods operated normally.

This jumper was installed to provide the appropriate rod position indication to the refuel interlock circuitry such that refueling could proceed while the repair of the position probe connector was being made.

Safety Evaluation Summary

Refuel interlocks are designed to prevent any condition which could lead to inadvertent criticality due to control rod withdrawal error during refueling. The refueling interlocks function to prevent inadvertent criticality. Rod 26-39 was disabled in a known position (full-in) such that no movement of this rod could occur. This temporary modification only provided the refueling interlock circuit indication of this rod's known/verified position. The failure mode for failure of the installed jumper would have been the same as a failure for the magnetically-operated reed switch, ie: fail open and initiate a rod or refueling block. Rod movement of Rod 26-39 was prevented such that its position could not have been changed from that being provided by the jumper.

There was no increase in the probability of occurrence or consequences of an accident or malfunction as previously evaluated in the FSAR. This temporary modification did not present significant hazards not described or implicit in the Vermont Yankee FSAR, and there is reasonable assurance that the health and safety of the public was not endangered.

6. **Temporary Modification 93-053** was installed and incorporated into PDCR 92-011 on 9/6/93.

General Summary

This modification added an electrical block to the Refueling Platform upward hoist movement control circuit to prevent the fuel grapple hoist from being able to travel in the upward direction when the hoist load is > 485 lbs and a full closure of the Grapple J-hook is not sensed by the control logic (ie. grapple limit switches). This interlock was provided to ensure a positive closure on the fuel bale handle prior to allowing any significant upward movement of the hoist and respective fuel assembly.

Safety Evaluation Summary

This modification decreases the probability of a fuel bundle being dropped by ensuring a positive closure on the fuel bundle bale handle prior to allowing significant upward movement of the hoist and fuel assembly. With respect to the Refueling Accident analyzed in the FSAR, this control circuit modification did not result in an increase of the maximum height allowed by the fuel handling equipment. The only FSAR transient potentially affected by this modification is "Fuel Assembly Insertion Error During Refueling", which states that "the refueling interlocks require that all control rods must be fully inserted before a fuel bundle may be inserted into the core". This modification only affects the ability of the Refuel Hoist to be raised. It in no way interfaces with the portion of the bridge control logic that blocks insertion of a fuel bundle into the core.

There was no increase in the probability of occurrence or consequences of an accident or malfunction as previously evaluated in the FSAR. This temporary modification did not present significant hazards not described or implicit in the Vermont Yankee FSAR, and there is reasonable assurance that the health and safety of the public was not endangered.

7. **Temporary Modification 93-063** was installed 10/7/93 and removed 10/8/93.

General Summary

This temporary modification installed a jumper to defeat the interlock preventing the movement of the refueling bridge in the reverse (northerly) direction over the core when the reactor mode switch was not in REFUEL. Without the jumper installed during plant shutdown, refueling platform travel toward the core was prevented with the mode switch in NOT IN REFUEL. This modification allowed for in-vessel inspections to be performed from the refueling bridge while the mode switch was in SHUTDOWN.

Safety Evaluation Summary

Administrative controls were in place to prevent fuel moves while the mode switch was in SHUTDOWN and in-vessel inspections were being performed from the refueling bridge. Controls included controlling fuel movement by way of the Fuel Loading Schedule with SRO supervision and notification to the control room of all inspection activities. The main fuel grapple was White Tagged out of service. A Caution Tag was placed on the reactor mode switch to keep the mode switch in SHUTDOWN while the temporary modification was in effect.

There was no increase in the probability of occurrence or consequences of an accident or malfunction as previously evaluated in the FSAR. This temporary modification did not present significant hazards not described or implicit in the

Vermont Yankee FSAR, and there is reasonable assurance that the health and safety of the public was not endangered.

8. **Temporary Modification 93-065** was installed 10/23/93 and is still open.

General Summary

This temporary modification provides a lower pressure source of cooling water to the Residual Heat Removal (RHR) Service Water (SW) motor cooling coils, supplying water at a pressure of 100 psig.

The strainer, solenoid valve and pressure control valve are rated at less than or equal to 300 psig. During shutdown cooling, pump discharge pressure can be greater than 340 psig. Thus, this modification provides cooling water at a low enough pressure for the RHRSW motor cooling system components.

Cooling water had previously been provided to the motor cooling coils from the discharge of each RHRSW pump. This supply was capped and a new tee was installed at instrument root valves V70-817A(B). During normal and emergency operation, the cooling coils are supplied by service water at 100 psig. During alternate cooling operation, water is supplied by the RHRSW pumps after alternate cooling is lined up (also at 100 psig).

Safety Evaluation Summary

This modification does not revise how the RHRSW pumps operate during the Loss of Coolant Accident and Steam Line Break Accident. This modification provides a lower pressure source of water to the motor coolers, thereby assuring the motor cooling components are operating within their design pressure ratings. During all modes of operation, except alternate cooling, the water comes from the same source as the suction supply to the RHRSW pumps. During alternate cooling the water is supplied from the discharge of the RHRSW pumps via valve PCV-70-69A/B. This modification is bounded by the Vermont Yankee Flooding Study. This modification did not introduce any new or change any mechanical or electrical system interactions. No active failure mechanisms have been introduced.

There was no increase in the probability of occurrence or consequences of an accident or malfunction as previously evaluated in the FSAR. This temporary modification did not present significant hazards not described or implicit in the Vermont Yankee FSAR, and there is reasonable assurance that the health and safety of the public was not endangered.

1992

The following items were inadvertently omitted from the 1992 Annual Operating Report: Safety Evaluation for procedure OP2190, EDCR 87-409 ECN3, PDCR 91-017, and PDCR 92-003.

1. Procedure 2190, Service and Instrument Air, revision 18

General Summary

Following Plant Operations Review Committee review of the 4/23/91 reactor scram, a new section was added to this procedure for the installation of a temporary diesel driven air compressor and its subsequent removal when no longer required.

Safety Evaluation Summary

The procedure change provides an alternate source of air to the service and instrument air system which meets the design requirements of the system. The temporary diesel driven air compressor was properly grounded and a fire control area was established. Installation of the temporary air compressor did not change, degrade or prevent actions described or assumed in the FSAR. System function and response did not change. The installation of a temporary air compressor did not introduce any different system responses to any failure modes.

There was no increase in the probability of occurrence or consequences of an accident or malfunction as previously evaluated in the FSAR. This procedure revision did not present significant hazards not described or implicit in the Vermont Yankee FSAR, and there is reasonable assurance that the health and safety of the public was not endangered.

2. EDCR 87-409 ECN 3 Modifications of High Speed Valve Actuators was completed 6/18/92.

General Summary

This design change eliminated the electric brakes on all high speed Limitorque actuators. The reason for this action was that EQ qualified brakes and brake parts were no longer available. Analysis indicated that several of the valves did not require brakes and that others would still require damping to limit the seating forces. Limitorque's current high speed valve designs utilized a double spring pack in place of the electric brakes; Vermont Yankee was supplied with kits to convert actuators to this new style. This modification added a second spring pack to the stem of the actuators to replace the braking function of the electric motor brakes. The new kit spring packs absorb the extra motor inertia once the limit switches turn off the motor. Also, motor thermal overloads for Recirculation Valves V2-53A and V2-53B were resized to be consistent with other MOVs whose control was modified by bypass

torque switch protection.

Safety Evaluation Summary

Valves requiring a locking mechanism to prevent reopening following an analyzed accident are equipped with locking operator gear sets, locking valve stems, or motor brakes. Any one of these provides an approved locking method. Valves with locking operator gear sets or locking valve stems had their unnecessary motor brakes removed. Two valves which had none of the approved methods had motor brakes installed. Valves with the clutch tripper removed from the operator still function as intended. Each safety-related valve was modified to either maintain or increase the probability that its safety function be completed as required.

Conversion and removal of the brakes that did not perform a safety function increased the ability of the affected valves to actuate when required. The addition of safety-related brakes to two valves ensured that the valves stay closed when required. The removal of the clutch trippers from valves 2-43A/B, 2-53A/B, 10-18, 14-11A/B and 14-12A/B eliminated a failure mechanism and helped ensure that the valves perform as required under accident conditions. Replacement of the motor thermal overload heaters prevents the motors from burning up, thereby maintaining valve reliability. Therefore, this design change results in an overall increase in plant safety.

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

3. **PDCR 91-017 Chem Lab/Detergent Waste Lines Replacement** was completed 8/24/92.

General Summary

This design change installed a Chemistry lab and detergent waste transfer system to replace the original Chemistry lab and detergent waste lines. The waste system pumps waste from the Chemistry lab sinks, Radiation Protection (RP) decon showers and RP decon sink to the Chemistry waste tank in Rad Waste. The waste being pumped is the same type and quantity of waste that previously was sent to Rad Waste via the Chemistry lab and detergent waste drain lines; therefore, the offsite dose rate is unaffected. The system pumps waste from holding tanks under two of the Chemistry lab sinks and the RP decon sink to the chemical waste tank in Rad Waste. The piping is routed from the RP decon shower room to the Chemistry lab and then through Switch Gear to Rad Waste.

The pumps, holding tanks, piping and valves that were utilized were selected for their chemical resistance characteristics. The pipe was routed through the switchgear room so that no electrical cabinets are directly below the pipe. Piping routed through the switchgear wall has fire rated penetrations. Pipe flanges outside the switchgear room were not installed directly above electrical cables or components. These flanges were installed with Vue-Guards to collect any leakage. The original Chemistry lab and detergent waste lines were abandoned in place. All accessible piping was filled with concrete mix to ensure that the lines are not used.

Safety Evaluation Summary

The modifications of this design change do not interact with any system or component that could possibly initiate an accident or operational transient previously evaluated. This change was designed and installed to ensure that all equipment important to safety located in the switchgear room was not affected. These modifications are not associated with, connected to, or cannot adversely affect any systems or components that are utilized to mitigate the consequences of an accident or operational transient. The pumping system is only used to transport chemical and low level radioactive liquid to Radwaste, is classified as Non Nuclear Safety and does not have the potential to create a malfunction, accident or operational transient of a different type than previously analyzed.

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

4. **PDCR 92-003 Moisture Separator Performance Testing** was completed 9/16/92.

General Summary

This design change installed injection and sample point test connections to facilitate Moisture Separator performance testing. The testing involves injection and subsequent measurement of non-radioactive tracer, lithium-7, in the steam and drain piping of all four Moisture Separators. Two 1/2" injection point test connections were located on each of the four 36" cross under piping lines, between the high pressure turbine outlet and the moisture separators. One 3/4" injection test connection was located on the 24" expansion piping below each moisture separator. The 3/4" sample point test connections was located further downstream on each of the 6" drain lines. Stainless steel turbine runs from the piping taps to the feedwater pump room where the required testing apparatus is located. The injection and retrieval of the tracer is conducted from this location. Isolation valves and caps are located on the turbine at the test location.

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

1991

The following items were inadvertently omitted from the 1991 Annual Operating Report: Safety Evaluation for OP4627, revision 19; EDCR 86-415; PDCR 89-17; the Use of Service Water for Deep Basin Deicing; and two temporary modifications.

1. **Procedure OP4627 revision 19 and the Permanent Removal of the Spray Pond Chlorine Monitor**

General Summary

The spray pond chlorine monitor was removed from the spray pond effluent because the National Pollution Discharge Elimination System (NPDES) permit issued to Vermont Yankee in January 1991 did not require continuous monitoring of total residual chlorine in the circulating water effluent during chlorination; therefore, the spray pond chlorine monitor was no longer required. Procedure OP4627, "Sampling and Treatment of the Circulating and Service Water System" was revised to delete the reference to the chlorine analyzer.

Safety Evaluation Summary

The spray pond chlorine monitor monitored the concentration of total residual chlorine present in the effluent of the spray pond during periods of chlorination of the circulating water system and did not have a control function; therefore, no equipment important to safety could have been impacted. Under the NPDES permit issued in January 1991, there was no limit for total residual chlorine in the effluents discharge. The concentration of total residual chlorine in the spray pond effluent has no effect on any equipment important to safety. The removal of the monitor did not affect the limit applied to the concentration of total residual chlorine present "in the immediate vicinity of the plant discharge..." (VY Technical Specifications).

There was no increase in the probability of occurrence or consequences of an accident or malfunction as previously evaluated in the FSAR. The removal of the spray pond chlorine monitor did not present significant hazards not described or implicit in the Vermont Yankee FSAR, and there is reasonable assurance that the health and safety of the public was not endangered.

2. **EDCR 86-415 Emergency Response Facility Information System (ERFIS) Installation** was completed 1/7/91.

General Summary

This design change installed the Emergency Response Facility Information System (ERFIS) in response to NRC requirements following the Three Mile Island accident, as detailed in NUREG-0696 and NUREG-0737. ERFIS gives operators archival and

real-time graphic displays of critical safety parameters; provides meteorological, plant, dose rate, and plume arrival data for normal operation and to emergency planning centers in the event of an accident; improves storage, retrieval, calculation, representation, and dissemination of important operational data; and incorporates nuclear software, 3D Monicore, capable of supporting 18-month station operating cycles utilizing high energy fuel bundles and nuclear energy performance predictors. This design change also isolated the safety class signals as necessary to maintain separation between safety class and nonsafety class signals.

Safety Evaluation Summary

This design change did not alter any of the control or safety functions available in the Control Room. This change provided the Control Room with additional indications of plant conditions presented in a format eliminating nonessential information and displaying key plant parameters. These displays aid in the avoidance of plant safety challenges and the mitigation of off-normal conditions.

All Safety Class Electrical instrument loops and electrical components tapped for input into ERFIS have Safety Class Electrical isolation which is accomplished by the use of Safety Class Electrical digital or analog isolators, or by contact-to-contact or coil-to-contact isolation provided by Safety Class Electrical equipment. This change did not involve any change in Technical Specification setpoints, protective function, or design basis of the plant. No existing safety functions were affected, and all new equipment was installed such that there was no detrimental effect on any Safety Class Electrical systems or components. The margin of safety, as defined in the basis for any Technical Specification, was not reduced.

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

3. **PDCR 89-017 Coolidge Line Primary Relaying Upgrade (NNS: Non Nuclear Safety)** was installed 3/4/91.

General Summary

This design change replaced the primary protective relay scheme for the 340 Coolidge line - which had been difficult to work on, prone to misoperation due to its age, and difficult to conduct tests on - to a Permissive Overreach scheme. The new relaying package is a solid state static distance relay, operating as a high speed permissive scheme which uses the microwave system for its permissive communication medium. The new scheme provides redundancy in having two high

speed schemes with different communications mediums. Additional benefits of the new equipment include self diagnostics, fault location capability, fault data recording, and remote fault data acquisition over the microwave system.

Safety Evaluation Summary

The modification of the Primary Phase Comparison scheme to a Permissive Overreach scheme did not change the Protective Relay functions. The new system improved the protection of the 340 Coolidge line by ensuring that failures at the Coolidge Substation do not adversely affect the Vermont Yankee Switchyard. Also, relocation of the Primary Protection communications means from the power line carrier to the microwave communications system ensures redundancy with the backup protection communications, which remains on the power line carrier. This change did not affect the availability of the safety class electrical buses to provide power, because there were no changes to the protective relaying in the 345 KV switchyard.

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

4. Use of Service Water (SW) for Deep Basin Deicing

General Summary

During plant recovery from the 4/23/91 Loss of Normal Power (LNP) event, reduced SW flow to the emergency diesel generators and reverse flow to the instrument air compressors were identified. It was determined that the apparent cause of the flow problems was the valve lineup in use at the time of the event; this lineup had SW-11 open, using the deep basin as the discharge path. To eliminate the high backpressure, the discharge was redirected to the condenser discharge block via valve SW-1. Revised methods for deicing and providing cooling tower water makeup included the following: SW discharge be directed to the deep basin when river temperature falls below 40 degrees F; before the river temperature exceeds 45 degrees F, the discharge be redirected to the condenser discharge block; and thermal relief valves and PV-51 series valves be reset as close to 75 psig as practical.

Safety Evaluation Summary

This evaluation addressed returning the discharge flow path from the condenser discharge block to the deep basin when river temperatures are at or below 45 degrees F. This valve lineup could have affected the emergency diesel generators (DGs), the Residual Heat Removal (RHR) heat exchangers, the Emergency Core Cooling

System (ECCS) corner room coolers and the Reactor Building Closed Cooling Water (RBCCW) heat exchangers, which are used to mitigate accidents evaluated in the FSAR. Although the DGs may have potentially received less than design flow through their coolers during a postulated event, adequate cooling would have been provided with the river water temperature at or below 45 degrees F. The discharge path did not affect the cooling capability of the RHR heat exchangers. In spite of reduced flow, the ECCS corner rooms would have remained sufficiently cooled with the SW temperatures at or below 45 degrees F. Any reduction in cooling to the RBCCW system would not have been an immediate concern. The cooling water temperature would have been significantly below that assumed in the analyses for the affected component, the fuel pool, and manual actions would have been taken by operators to improve cooling to the fuel pool to ensure the technical specification temperature limit was not reached.

Although instrument air compressors are not relied upon for accident mitigation, they are utilized if available. The total flow passing through all four air compressors would be approximately 40 gpm. Because the magnitude of this flow is a very small percentage of the overall system flow, particularly if all four SW pumps are in use (as was the case during the LNP event), the effect of this reduction in flow to the essential components cooled by the SW system was not a concern.

There was no increase in the probability of occurrence or consequences of an accident or malfunction as previously evaluated in the FSAR. This temporary modification did not present significant hazards not described or implicit in the Vermont Yankee FSAR, and there is reasonable assurance that the health and safety of the public was not endangered.

5. **Temporary Modification 90-026** was installed 7/16/90 and removed 5/11/94.

General Summary

This temporary modification isolated hypochlorite flow to the service water (SW) chlorination system to address mechanical problems and safety hazards due to the deterioration of the plumbing. A new, upgraded circulation water/service water chemical treatment system was installed under design change PDCR 93-019.

Safety Evaluation Summary

Prior to this temporary modification, the service water chlorination system (SWCS) had not been used for eight years. The SWCS is considered Non Nuclear Safety (NNS) and its use is not described in the FSAR. No equipment important to safety was impacted by the isolation of the hypochlorite supply to the SWCS since this system is NNS, was not capable of being operated, and had no control functions. This modification prevented accidental injection of hypochlorite into the circulating

water system as a result of tubing rupture, and resolved safety concerns of employees in the hypochlorite storage area.

There was no increase in the probability of occurrence or consequences of an accident or malfunction as previously evaluated in the FSAR. This temporary modification did not present significant hazards not described or implicit in the Vermont Yankee FSAR, and there is reasonable assurance that the health and safety of the public was not endangered.

6. **Temporary Modification 91-018** was installed 6/21/91 and removed 1/3/92.

General Summary

This temporary modification provided a drainage path for chemical waste by replacing the Chemistry Lab floor drain with a system that pumped chemical waste from a holding tank under a Chemistry Lab sink to the chemical waste tank in Radwaste. The pump, holding tank, piping and valves were selected for their chemical resistance characteristics. The piping was routed from the Chemistry Lab through the Switchgear Room to Radwaste. No electrical cabinets were directly below the pipe. Fire rated penetrations were installed where the piping was routed through the Switchgear wall.

Safety Evaluation Summary

This modification was designed and installed to ensure that all equipment important to safety located in the Switchgear room was not affected. The portions of Radwaste affected by this temporary modification are not relied upon to mitigate the consequences of an equipment malfunction. The pipe was routed in a way that no electrical cabinets or cables were directly below the pipe. All flanges located in the Switchgear room were installed with Vue-Gards to collect any unforeseen leakage. This modification was not associated with, or connected to, any systems or components utilized to mitigate the consequences of an accident or operational transient.

There was no increase in the probability of occurrence or consequences of an accident or malfunction as previously evaluated in the FSAR. This temporary modification did not present significant hazards not described or implicit in the Vermont Yankee FSAR, and there is reasonable assurance that the health and safety of the public was not endangered.

1990

The following items were inadvertently omitted from the 1990 Annual Operating Report: Lineup Deviations 90-07, 90-10, 90-12 and 90-13, GE vacuum fuel sipping equipment, closing of Core Spray Valve 11B, Diesel Generator Preventative Maintenance Plan, four temporary modifications, and the Vermont Yankee response to NRC concerns regarding Residual Heat Removal Service Water motor cooling piping changes.

1. **Lineup Deviation 90-7** was initiated 4/21/90 and restored 4/23/90.

General Summary

This lineup deviation closed High Pressure Coolant Injection HPCI-20, a normally open valve, and opened HPCI-19, a normally closed valve, due to a 125 VDC ground on DC-1 caused by HPCI-19.

Safety Evaluation Summary

HPCI-20 and HPCI-19 are functionally identical. The valves and valve operators are of the same construction and operate in an identical fashion. Both valves receive the same open signal, on low reactor pressure vessel level and high drywell pressure, from the same devices, to provide the injection pathway on HPCI system initiation. Failure of a HPCI system discharge valve to open, either HPCI-19 or HPCI-20, would result in the same consequence.

There was no increase in the probability of occurrence or consequences of an accident or malfunction as previously evaluated in the FSAR. This lineup deviation did not present significant hazards not described or implicit in the Vermont Yankee FSAR, and there is reasonable assurance that the health and safety of the public was not endangered.

2. **Lineup Deviation 90-10** was initiated 5/23/90 and restored 5/26/90.

General Summary

This lineup deviation isolated the Turbine Building Cooling Water (TBCCW) Surge Tank (TK-32-1A) automatic fill line by closing valve TCW-143A. This valve was closed as a temporary measure due to excessive leakage past automatic makeup valve LCV-104-6.

Safety Evaluation Summary

Isolation of the automatic fill line did not affect any of the systems whose failures result in a design bases accident. The TBCCW is not relied upon to mitigate the consequences of design bases accidents. Also, the TBCCW system is isolable from station safety cooling water systems that are relied upon for accident mitigation.

There was no increase in the probability of occurrence or consequences of an accident or malfunction as previously evaluated in the FSAR. This lineup deviation did not present significant hazards not described or implicit in the Vermont Yankee FSAR, and there is reasonable assurance that the health and safety of the public was not endangered.

3. **Lineup Deviation 90-12** was initiated 5/23/90 and restored 7/12/90.

General Summary

This lineup deviation closed Control Rod Drive (CRD) valve CRD-39 due to a leaking seal line on the "B" CRD pump. This lineup deviation, along with the closure of suction valve CRD-159B described in Lineup Deviation 90-13, isolated the positive pressure seal line to the CRD drive pump P-38-1B, thus effectively isolating one of the two CRD drive water pumps.

Safety Evaluation Summary

Isolation of the CRD pump could not have resulted in a design basis accident, since the station is designed assuming a complete loss of CRD flow. The CRD pumps are not required to mitigate the consequences of any design basis accidents. CRD pumps do not impact any other equipment in a way that could have caused an increase in the probability of equipment malfunction.

There was no increase in the probability of occurrence or consequences of an accident or malfunction as previously evaluated in the FSAR. This lineup deviation did not present significant hazards not described or implicit in the Vermont Yankee FSAR, and there is reasonable assurance that the health and safety of the public was not endangered.

4. **Lineup Deviation 90-13** was initiated 5/23/90 and restored 7/12/90.

General Summary

This lineup deviation closed Control Rod Drive (CRD) suction valve CRD-159B due to a leaking seal line on the "B" CRD pump. This lineup deviation, along with the closure of valve CRD-39 described in Lineup Deviation 90-12, isolated the positive pressure seal line to the CRD drive pump P-38-1B, thus effectively isolating one of the two CRD drive water pumps.

Safety Evaluation Summary

Isolation of the CRD pump could not have resulted in a design basis accident, since the station is designed assuming a complete loss of CRD flow. The CRD pumps are not required to mitigate the consequences of any design basis accidents. CRD pumps

do not impact any other equipment in a way that could have caused an increase in the probability of equipment malfunction.

There was no increase in the probability of occurrence or consequences of an accident or malfunction as previously evaluated in the FSAR. This lineup deviation did not present significant hazards not described or implicit in the Vermont Yankee FSAR, and there is reasonable assurance that the health and safety of the public was not endangered.

5. **General Electric Vacuum Fuel Sipping Equipment** safety evaluation was written 8/29/90.

General Summary

A question was raised during review of EDCR 90-402, "Control Rod Blade Storage Rack Modifications", regarding the thermal hydraulic effects of loading a fuel bundle into a sipping can and "sipping" the bundle. The concern was whether use of the sipping can could create the possibility for the fuel bundle to be uncovered of water, heat up, and potentially fail.

The sipping can is equipped with a temperature monitoring thermistor which provides an audible alarm and a visual alarm upon reaching a preset temperature. Procedure requires reflooding and opening of the sipping can if the can temperature reaches 140 degrees F, the alarm point. This ensures a wide margin between the highest allowable temperature and the boiling point of the water. Conservative calculation determined that the can would heat up at 4.4 degrees F per minute. It takes less than one minute to reflood the can. Therefore, upon reaching the 140 degree alarm, the can will be reflooded and opened well before the 160 degree saturation temperature.

Safety Evaluation Summary

The sipping can is designed so that the water outlet line from the can is above the fueled portion of the bundle. Therefore, when air is introduced into the can, it forms a bubble at the top of the can but cannot push water level lower than the water outlet line before blowing out through the line. This eliminates the concern of uncovering the fueled portion of the fuel bundle during the sipping process.

Movement of fuel to and from the sipping cans is performed in a similar manner to movement of fuel to both the core and spent fuel pool; no special or different lifting hardware or procedures are used. Any problem occurring with the fuel sipping process would be bounded by the accident analysis described in FSAR section 14.6.4, Refueling Accident. The sipping can equipment is operated on the refueling floor and in the spent fuel pool and does not affect any equipment important to safety.

There was no increase in the probability of occurrence or consequences of an accident or malfunction as previously evaluated in the FSAR. Use of vacuum fuel sipping equipment did not present significant hazards not described or implicit in the Vermont Yankee FSAR, and there is reasonable assurance that the health and safety of the public was not endangered.

6. **Closing Core Spray Valve 11B** safety evaluation was written 2/21/90.

General Summary

Leakage past Core Spray Valve CS-12B increased the pressure in the downstream low pressure piping. To resolve this, MOV valve V14-11B in pathway CS-11B was closed. CS-11B is designed to meet pressure and temperature requirements. CS-12B, in the degraded condition that allowed for the observed condition, was operable to satisfy the required operational and safety design bases. The closing of CS-11B was consistent with the actions required by Technical Specification section 3.7.D.2 which provides the required action in the event that there is an inoperable containment isolation valve. Given that CS-12B was not inoperable but was degraded, this condition was within the bounds previously reviewed and approved by the NRC.

Both Core Spray valves on each line receive open signals on low pressure to provide an injection pathway. Two redundant injection paths were provided to ensure an injection pathway existed. Plant design bases require that the plant be designed to accommodate the worst possible single failure of an active component. The closure of an additional valve, even though it provides an additional valve that could fail closed, does not require an additional failure be assumed. Based on this, closing CS-11B did not impact the design basis availability of the core spray system.

Safety Evaluation Summary

Valve 11B was still operational and supplied with an open signal upon a core spray logic initiation signal. If this valve had failed to open, many other high pressure and low pressure injection paths were still available for keeping the core flooded and ensuring long term cooling. All equipment directly related to the 11B's function is independent of its position. Changing position of this valve did not increase the probability of it to malfunction. Although 12B was considered degraded (due to its leakage), its availability remained the same. The core spray system, if called upon, would have operated in the manner in which it was intended. Closing 11B to ensure containment was a conservative action within the intent and rules set forth by Technical Specifications.

There was no increase in the probability of occurrence or consequences of an

accident or malfunction as previously evaluated in the FSAR. Closing core spray valve 11B did not present significant hazards not described or implicit in the Vermont Yankee FSAR, and there is reasonable assurance that the health and safety of the public was not endangered.

7. **Emergency Diesel Generator Preventative Maintenance Program Changes**

General Summary

The Emergency Diesel Generator preventive maintenance program was revised to schedule major electrical and mechanical overhauls and preventative maintenance work (inspection/overhaul) to be performed every 24 months $\pm 0/-2$ months rather than at 12 to 18 months; and to perform the 24 month inspection/overhaul during 7-day LCOs on alternate diesels in alternate years. Based on the high reliability experienced by the engines with no failures to start on demand and only one series of problems associated with start times, an enhanced preventative maintenance program was developed that allowed an extended interval between overhauls.

Program changes include the increase of diesel oil sampling and analysis from semiannually to quarterly; continued monthly trending of engine operating data; quarterly vibration analysis of the diesel engines during normal surveillance operations; performance of a minimum of one mini-inspection at between 16-18 months following a major overhaul, to provide a check of the conditions of the engine and to enhance predictive and preventative actions to identify and correct unusual wear or deterioration; performance of routine major overhaul/preventative maintenance on a 22-24 month basis during operations and within 7-day LCOs, with alternate engines scheduled 12 months apart.

Safety Evaluation Summary

These maintenance practices are unrelated to the accidents evaluated in the FSAR, and they neither increase nor decrease the probability of occurrence. These changes are limited to improvements of the preventative maintenance of the diesel and should result in increased diesel reliability. The margin of safety as defined in the bases of Technical Specifications was not reduced because each diesel is capable of supplying 100 percent of the necessary emergency loads required under the postulated design basis accident and the changes were implemented to increase the available diesel generator's reliability.

There was no increase in the probability of occurrence or consequence of an accident or malfunction as previously evaluated in the FSAR. Revising the emergency diesel generator preventive maintenance plan did not present significant hazards not described or implicit in the Vermont Yankee FSAR, and there is reasonable assurance that the health and safety of the public was not endangered.

8. **Temporary Modification 90-025** was installed 8/17/90 and restored 11/27/90.

General Summary

This temporary modification removed the Reactor Building Closed Cooling Water (RBCCW) chemical feeder tank TK-50-1A, and installed in its place a portable skid consisting of a mechanical filter, carbon filters and mixed bed deionizers. This filtration system was used to clean the RBCCW system, to remove all the corrosion inhibitor Drewgard 100 from the system, reduce filterable solids, and reduce system chloride levels to < 5.0 ppm.

Safety Evaluation Summary

The filtration system did not interfere with normal operation of the RBCCW system, nor with the ability of the RBCCW system to support the safety design bases. The portions of the RBCCW system required to mitigate the consequences of an equipment malfunction were not affected by the installation of this temporary system.

There was no increase in the probability of occurrence or consequences of an accident or malfunction as previously evaluated in the FSAR. This temporary modification did not present significant hazards not described or implicit in the Vermont Yankee FSAR, and there is reasonable assurance that the health and safety of the public was not endangered.

9. **Temporary Modification 90-044** was installed 8/22/90 and restored 12/18/90.

General Summary

This modification removed solenoid valve SE-72-2B, which was inoperative, from the "B" railroad airlock door seal supply, and replaced it with 3/8 " stainless steel tubing. This modification also removed, from Door B, the interlock that ensures that the door is shut prior to repressurizing the seal. An operating procedure was included in this temporary modification to ensure that the door would be fully closed prior to repressurizing the seal.

Safety Evaluation Summary

The function of the reactor building airlock door seals was not changed by this modification. The components that make up the door seal system and the reactor building penetration itself do not act as an initiator for any design basis accidents. In removing solenoid valve SE-72-2B from the flow path, this modification removed the only electrical component that could have depressurized the seal; the system was then completely mechanical. Solenoid valve SE-72-2B is a normally de-energized, normally open valve with the safety function of acting as a pressure boundary; the replacement tubing performed the same function. This temporary modification did

not affect the interlocks for Door A. If Door B was not closed and its seal pressurized, the seal for Door A would not have been able to be depressurized or its door opened. The gauge test connection was intended to be used to manually vent the seal.

There was no increase in the probability of occurrence or consequences of an accident or malfunction as previously evaluated in the FSAR. This temporary modification did not present significant hazards not described or implicit in the Vermont Yankee FSAR, and there is reasonable assurance that the health and safety of the public was not endangered.

10. **Vermont Yankee Response to NRC Concerns Regarding Residual Heat Removal Service Water Motor Cooling Piping Changes**

During 1990, the NRC performed a Safety System Functional Inspection (SSFI) on the residual heat removal service water (RHRSW) and core spray systems. One question that arose was Vermont Yankee's use of a copper tie line, which connected the four RHRSW motor cooler water discharges together at one common point. Vermont Yankee:

1. Developed a 50.59(a)(2) safety evaluation which noted that the configuration was consistent with 10CFR50 Appendix A guidelines for single mechanical failure.
2. Committed to upgrade the configuration. This upgrade was completed via a series of design changes which now provide for two stainless tie-ins (i.e., stronger material, elimination of single failure).

Since subsequent design changes have been installed, the 50.59(a)(2) evaluation described in number "1" above is no longer needed, nor is it applicable.

1989

The following items were inadvertently omitted from the 1989 Annual Operating Report: One Mechanical Bypass, Special Test Procedure 89-03, Diesel Generator Air Start Check Valves, Revision 3 to the On-Site Low Level Waste Storage Facility, and six temporary modifications.

1. **Mechanical Bypass 89-0018** was installed 3/22/89 and superceded by Temporary Modification 89-015, which was installed 5/4/89 and removed 10/11/90.

General Summary

This mechanical bypass and safety evaluation were initiated to address the portion of EDCR 86-412, "High Density Spent Fuel Racks" which included moving three racks into the reactor building through the Equipment Airlock, and necessitated moving the temporary mechanical seal from one airlock door to the other a minimum of seven times. (For example, to bring the first rack into the reactor building: the mechanical seal was moved from the outer door to the inner door; the outer door was opened and the truck driven into the airlock, and the outer door was closed; the seal was moved from the inner door to the outer door, and the truck was driven into the reactor building; the truck was unloaded and driven back into the airlock; the seal was moved from the outer door to the inner door; the outer door was opened and the truck driven out.)

Safety Evaluation Summary

The probability of occurrence of an accident was not increased in that the seal was designed and tested to ensure that secondary containment be maintained and that the system required to support post accident consequences not be affected. Failure of the seal, which was highly unlikely, would not have negatively impacted any equipment, i.e. the seal material was softer than any material in the reactor building airlock door system.

There was no increase in the probability of occurrence or consequences of an accident or malfunction as previously evaluated in the FSAR. This mechanical bypass did not present significant hazards not described or implicit in the Vermont Yankee FSAR, and there is reasonable assurance that the health and safety of the public was not endangered.

2. **Removal of Air Start Check Valve Internals**

General Summary

During the NRC Safety System Functional Inspection (SSFI) performed on the Diesel Generator (DG) system, a concern was raised on the DG air starting system/sensing line check valves. A safety evaluation was written to ensure that all

performance aspects of these check valves are maintained and that the removal of the check valve internals is appropriate. The pressure switch sensing lines check valves were required to flow in both directions for proper operation; they needed to "leak" flow to the pressure switch for the pressure sensing attribute. These check valves were not specifically designed to leak; if cleaned and carefully aligned, they could have become leak tight. If completely leak tight, the sensing lines would have become pressure bound and inoperable. Based on this, the valve internals were removed.

Safety Evaluation Summary

One for one evaluations do not require safety evaluations. However, for this repair, the design basis was complex and the safety evaluation was provided for information.

The probability of an analyzed accident or malfunction was not increased; the repair ensured that the pressure sensing attribute of the receiver tank check valves was maintained. The maintenance attribute did not impact accident performance. The repair did not create a different type of accident than previously analyzed. The two receiver tanks on the compressor/receiver skid are not considered single failure proof between each other (ie. there are two tanks but only one compressor and one pressure switch) but rather single failure proof between skids. This repair did not affect skid independence. The FSAR also recognized that the failure of a single active component in a diesel starting air system could result in the failure of the associated diesel to start. Based on this, each diesel is provided with a dedicated start system. Removal of the check valve internals does not impact this redundancy.

There was no increase in the probability of occurrence or consequences of an accident or malfunction as previously evaluated in the FSAR. The removal of air start check valve internals did not present significant hazards not described or implicit in the Vermont Yankee FSAR, and there is reasonable assurance that the health and safety of the public was not endangered.

3. Low Level Waste Storage Pad Facility, revision 3

General Summary

Vermont Yankee's low level waste storage pad facility was constructed with concrete storage modules for the interim storage of low level waste which consists of dewatered resin and filter media, and dry active waste (DAW). The facility consists of a prepared area approximately 180' x 176' outside of/adjacent to the north protected area. The area is surrounded by an 8' high chain link fence for security and radiation protection purposes. A minimum of 21' is maintained between the fenced area and the existing plant security fence to comply with security regulations.

The area was excavated to a depth of approximately 1' and backfilled with approximately 2' of processed gravel; this provided a base elevation of approximately 252' and the necessary footing for the concrete storage modules. The area was compacted to a 4-6 ton/square foot load bearing rating. Prefabricated, reinforced concrete storage modules were provided on an as-needed basis for waste storage.

Safety Evaluation Summary

The probability or consequences of a previously evaluated accident or malfunction was not increased since there was no change to the fundamental approach of transferring radioactive waste from the plant to a container whether that container be used for immediate off-site transport or for interim storage prior to transport. The radiological consequences of a postulated drop of a waste container while being transferred into a storage module are bounded by the dose consequences calculated in the FSAR for a similar accident involving the drop of a fuel assembly while being transferred to storage. Design basis events such as flood, fire, tornado and earthquake have been assessed and found not to present radiological concerns more severe than the postulated waste liner drop accident.

For tornado and seismic events, it was assumed for analysis that failure of concrete storage modules would occur allowing the release of a fraction of the stored waste to the environment. The radiological consequences were found to be below those of the postulated liner drop accident and therefore not a limiting condition as long as the curie content in the stored waste did not exceed certain values. Limits on the amount of radioactivity allowed in storage at any time are: for the rectangular modules storing DAW and condensate resins - 770 curies of principal gamma emitters; and for the cylindrical modules storage high activity cleanup resins - 2800 curies of principal gamma emitter. The dose from the controlling design basis event does not exceed the criteria of limiting offsite exposures to less than a small fraction (10 percent) of 10 CFR Part 100.

There was no increase in the probability of occurrence or consequences of an accident or malfunction as previously evaluated in the FSAR. Construction of the Low Level Waste Storage Pad Facility did not present significant hazards not described or implicit in the Vermont Yankee FSAR, and there is reasonable assurance that the health and safety of the public was not endangered.

4. **Temporary Modification 89-036** was installed and removed 6/21/89.

General Summary

To allow removal and testing of the Reactor Water Clean Up (RWCU) system high demineralizer inlet temperature switch, this temporary modification installed an electrical jumper to allow the other RWCU protection features to remain operational.

The installation of this jumper on CRP 9-41 between terminal points AA-14 and AA-15 removed the high demineralizer inlet temperature protection feature from the RWCU system. Normally, a high temperature signal from this switch results in a RWCU isolation from valves V-12-15, 18, and 68. With the jumper installed, the demineralizers were not automatically protected from high inlet temperature.

Safety Evaluation Summary

Because this isolation is described in the FSAR and pictorially shown in the FSAR, a safety evaluation was performed. The only purpose of the RWCU high demineralizer inlet temperature isolation is to protect the demineralizer resins from damage. This isolation function is not used to mitigate the consequences of an accident. The loss of RWCU demineralizer resins from an overheated condition, or loss of the entire RWCU system, is already an analyzed condition that requires plant shutdown when reactor water chemistry degrades beyond predetermined setpoints. The RWCU system receives a PCIS (Primary Containment Isolation System) Group 5 Isolation Signal on low reactor water level which isolates Valves 12-15, 18, and 68; the introduction of a jumper in the circuitry discussed above did not affect the ability of this PCIS signal to operate as designed.

There was no increase in the probability of occurrence or consequences of an accident or malfunction as previously evaluated in the FSAR. This temporary modification did not present significant hazards not described or implicit in the Vermont Yankee FSAR, and there is reasonable assurance that the health and safety of the public was not endangered.

5. **Temporary Modification 89-039** was installed 8/10/89 and removed 7/31/91.

General Summary

This modification removed flow switch FS-104-31 from the process radiation monitor RM-17-351 circuitry. This was done to increase the operability of RM-17-351. Due to the tight tolerances in FS-104-31, there had been problems with the flow switch sticking due to silt build-up from the service water (SW) system. RM-17-351 had been rendered inoperable.

The function of the flow switch was to provide an immediate indication of loss of flow through the SW radiation monitor, which effectively disables the monitor. To compensate for removal of the flow switch, flow through the radiation monitor was verified by the auxiliary operator once per shift. In the worst case, the radiation monitor could have been lost for the better part of a shift before detection; this was not considered a problem since the compensatory measure for inoperability of the radiation monitor is daily sampling of the service water.

Safety Evaluation Summary

The SW radiation monitor performed a monitoring function only. There was no connection to any safety system. The only function lost was the alarm feature for loss of flow; the radiation monitor was fully operable to monitor any potential release through the SW system. Loss of flow to the radiation monitor would not have resulted in damage to the monitor upon loss of flow. Removal of the flow switch from the circuit reduced the potential for loss of the radiation monitor due to a faulty flow switch.

There was no increase in the probability of occurrence or consequences of an accident or malfunction as previously evaluated in the FSAR. This temporary modification did not present significant hazards not described or implicit in the Vermont Yankee FSAR, and there is reasonable assurance that the health and safety of the public was not endangered.

6. **Temporary Modification 89-043** was installed 7/28/89 and removed 10/25/91.

General Summary

This modification replaced the potable water supply for the Chemistry office area air conditioners 4, 5, and 6 with chilled water. This change was made because of the great demand for potable water. This change also reduced the amount of water going to the septic system.

Safety Evaluation Summary

The Non-Nuclear Safety chilled water system normally has no effect on any safety class systems. However, the ventilation system procedure allows for the cross-connection of the Control Room chilled water system (Safety Class 3) to the Administration Building chilled water system (Non-Nuclear Safety). Air conditioners 4, 5, and 6 were valved out prior to cross connecting the two systems; this ensured that the configuration would not be changed in any way.

There was no increase in the probability of occurrence or consequences of an accident or malfunction as previously evaluated in the FSAR. This temporary modification did not present significant hazards not described or implicit in the Vermont Yankee FSAR, and there is reasonable assurance that the health and safety of the public was not endangered.

7. **Temporary Modification 89-045** was installed 8/8/89 and removed 9/3/90.

General Summary

This temporary modification installed a universal disturbance analyzer on the

Uninterruptible Power Supply (UPS) "A" Inverter to monitor logic for spurious trip signals. Temporary Modification 89-045 was removed during the 1990 refueling outage upon implementation of EDCR 89-407, UPS Replacement.

Safety Evaluation Summary

The test leads were soldered to the circuit board pins to provide a reliable hookup. Safety class fuses with a capacity not to exceed the signal magnitude were installed to isolate the analyzer and protect the UPS in the event of an analyzer or test lead failure. Due to the protection of the fuses, the probability of an occurrence of an accident was not increased above that already analyzed.

There was no increase in the probability of occurrence or consequences of an accident or malfunction as previously evaluated in the FSAR. This temporary modification did not present significant hazards not described or implicit in the Vermont Yankee FSAR, and there is reasonable assurance that the health and safety of the public was not endangered.

8. **Temporary Modification 89-050** was installed 9/13/89 and removed 4/2/92.

General Summary

This modification changed the Reactor Building Closed Cooling Water (RBCCW) Motor Operated Valve (MOV) V70-117 power supply from a non-safety motor-control center MCC 7A, to AC emergency power bus 8B. The motive and control cables were relocated to MCC 8B, compartment 10G, a "B" diesel generator load. The valve is rated at 2/3 of a horsepower with a full load amperage of 2.3. The addition of this valve to MCC 8B resulted in an approximate increase of 0.83 KW to the "B" diesel generator. This relates to 0.028 percent of total diesel generator capacity, or 0.346 percent of the remaining capacity.

Safety Evaluation Summary

The placement of this valve on a safety related bus was previously evaluated in the FSAR and should have been originally located there. The FSAR assumed this valve would be provided with safety class power to ensure closure following an accident; this assumption was supported with the implementation of this modification. There was no change in operation in either the V70-117 valve or the loads supplied from MCC 8B.

There was no increase in the probability of occurrence or consequences of an accident or malfunction as previously evaluated in the FSAR. This temporary modification did not present significant hazards not described or implicit in the Vermont Yankee FSAR, and there is reasonable assurance that the health and safety of the public was not endangered.

9. **Temporary Modification 89-055** was installed 12/12/89 and removed 4/28/91.

General Summary

This modification removed the mechanical interlock for the two DC output breakers in the relay house battery charger 4A-5A. This allowed distribution panel DC-4A to be tied to distribution panel DC-5A, thus allowing removal of the 4A and 5A batteries for maintenance. Only one battery at a time was out of service.

Safety Evaluation Summary

None of the affected equipment was considered important to safety, nor did they interact with any safety equipment. The function of the affected components is to trip breakers in the switchyard; the loss of offsite power is already evaluated in the FSAR. Loss of these busses and subsequent fault in the switchyard would have lead to a generator failure, which was already analyzed in the FSAR.

There was no increase in the probability of occurrence or consequences of an accident or malfunction as previously evaluated in the FSAR. This temporary modification did not present significant hazards not described or implicit in the Vermont Yankee FSAR, and there is reasonable assurance that the health and safety of the public was not endangered.