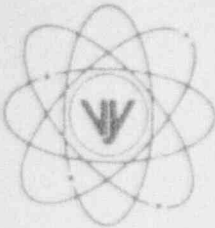


VERMONT YANKEE NUCLEAR POWER CORPORATION



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March 24, 1993
BVY 93 - 24

United States Nuclear Regulatory Commission
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References: a. License No. DPR-28 (Docket No. 50-271)

Subject: Vermont Yankee 1992 Annual Operating Report

Dear Sir:

Enclosed please find one copy of the Vermont Yankee Nuclear Power Corporation Annual Operating Report submitted in accordance with 10CFR50.59(b)(2). This report describes the facility changes, tests, and experiments conducted without prior NRC approval during the year 1992.

We trust this information is acceptable; however, should you have any questions, please contact this office.

Very truly yours,

VERMONT YANKEE NUCLEAR POWER CORPORATION

Leonard A. Tremblay, Jr.
Leonard A. Tremblay, Jr.
Senior Licensing Engineer

cc: USNRC Region I Administrator
USNRC Resident Inspector - VYNPS
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VERMONT YANKEE

1992 ANNUAL OPERATING REPORT

OPERATIONS SUMMARY

Between January 1 and December 31 of 1992, Vermont Yankee implemented a number of changes. The following report describes those changes which constituted a change in the facility as described in the Final Safety Analysis Report (FSAR). The report includes nine (9) Engineering Design Change Requests (EDCRs), one (1) EDCR Installation and Test Procedure, fifteen (15) Plant Design Change Requests (PDCRs), eight (8) Temporary Modifications (TMs), one (1) Special Test Procedure, one (1) evaluation of the Standby Gas Treatment (SBGT) System temperature sensing element, one (1) evaluation of Reactor Building Closed Cooling Water (RBCCW) - Service Water (SW) discharge line work, the permanent removal of the Circulating Water Spray Pond Chlorine Monitor, and the core performance analysis for Cycle 16. There were no Valve Lineup Deviations, or Setpoint Changes performed during the year and no Safety Relief Valve Failures that required a safety evaluation in accordance 10CFR50.59.

A. Changes in Facility Design

1. During 1992 there were no changes made which required authorization from the Commission.
2. The following changes did not require Commission approval. They were reviewed by the Plant Operations Review Committee (PORC) and approved by the Plant Manager and the Vice President, Engineering. It was determined that these changes did not involve unreviewed safety questions as defined in 10CFR50.59(a)(2).
 - a. EDCR 90-402 ECN1 "Control Rod Blade Storage Rack Modification" was completed 2/21/92.

General Summary:

Design Change 90-402, implemented 8/31/90, modified the existing 30 cell rod rack assembly and added one additional 30 cell assembly to the spent fuel pool. In Engineering Change Notice (ECN) 1, counterweights were added to the racks to improve seismic stability and provide relief from loading restrictions imposed with the original design. ECN1 qualified the racks for storage for the following: control rod blades (CRBs), single

blade guides, double blade guides, dunking chambers, cut and folded CRBs, 5 velocity limiters, and folded LPRM strings; also qualified was the temporary storage of fuel in the CRB rack only during fuel sipping evolutions.

Safety Evaluation Summary:

The safety evaluation for this ECN addressed the thermal hydraulic effects of loading a fuel bundle into a sipping can and sipping the bundle. The sipping can system is connected to the plant systems through the 115v AC electrical distribution system and demineralizer water systems. The equipment is operated on the refueling floor and in the spent fuel pool and does not affect any equipment important to safety.

There is 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 is not endangered.

- b. EDCR 90-403 "Feedwater Check Valve Replacement" was completed 4/10/92.

General Summary:

To facilitate hydrostatic testing of the feedwater system, this design change installed a 3/4" vent and a 1-1/2" fill connection on the Non-Nuclear Safety (NNS) portion of the feedwater piping. The vent connection was installed on line 16-FDW-14 in the steam tunnel. The fill connection was installed on line 10-FDW-11 in the turbine building. This fill connection is supported from the 10-FDW-11 line to minimize both vibration of the cantilevered fill line, and stresses on the connection to the 10 inch line.

Safety Evaluation Summary:

The 3/4" vent and 1-1/2" fill connections on the NNS feedwater piping consist of piping, valves, and fittings. The valves are normally closed. During normal feedwater system operation, the connections are passive pressure boundary components. The only malfunction important to safety would be pressure boundary failure. Both connections are small bore. The volume of feedwater lost due to a break on one or both of these lines is enveloped by previously evaluated high energy line breaks in large

bore feedwater and steam piping at or near the locations of the connections. These components were designed to standards equal to or better than the original equipment.

There is 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 is not endangered.

- c. EDCR 90-404 "Generator Step-up Transformer Replacement"
was completed 4/17/92.

General Summary:

This design change replaced the Main Step-Up Transformer. Concerns were raised regarding potential degradation of the transformer to the point of failure, and severe failures of General Electric main step-up transformers with similar designs. The new transformer includes improvements in design, efficiency and reliability; is physically and electrically interchangeable with the old transformer; and has a greater output rating: 650 vs 616 MVA. This will allow it to run cooler and should extend the life of the transformer.

Safety Evaluation Summary:

Vermont Yankee's Technical Specifications require the availability of at least one offsite transmission line, one start-up transformer and one delayed access source. The new transformer offers improvements in design and reliability when compared with the old; thus availability of the delayed access source through the main step-up transformer is not reduced. The margin of safety as defined in the basis of technical specifications is not reduced. The new transformer is functionally a direct replacement for the old transformer. All supporting systems such as the deluge system and auxiliary power supplies were evaluated to ensure their adequacy.

There is 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 is not endangered.

- d. EDCR 90-405 "Drywell RRU Replacement/Upgrade" was completed 4/10/92.

General Summary:

This design change replaced the Drywell Reactor Recirculation Units, RRUs 1 through 4, as a result of numerous maintenance issues encountered with the equipment since commercial operation began. Blank flanges were included on the new units to allow the use of portable refrigeration during future outages. The discharge throttling dampers were replaced with gravity exhaust dampers, and the inlet dampers were replaced with throttling dampers. An additional ventilation supply point was added to provide supplemental under vessel cooling during plant shutdowns. The Reactor Building Closed Cooling Water (RBCCW) piping spool pieces were also replaced. The RRU load shed scheme was modified to ensure proper diesel generator loading. Cooling capability was improved with 100% backup. There are now eight fans and eight motors (Two fans and two motors in each of the RRUs).

Safety Evaluation Summary:

The purpose for replacing the units was to improve the reliability of the Drywell Atmospheric Cooling System. The Drywell Atmospheric Cooling System is classified as Non-Nuclear Safety (NNS) because it plays no part in the initiation of any accidents evaluated in the FSAR nor is it required to support any accident function or accident response. The control circuit, including all components, associated tubing and power to the RRU motors, does not provide a safety function and is considered NNS. The circuit breaker within the motor control center isolates NNS equipment from the safety class bus.

The portion of the RBCCW system located inside containment is Safety Class 2 because it forms part of the containment pressure boundary. The Vermont Yankee Safety Class Manual requires that all piping penetrations into containment have two means of isolation: one is provided by the isolation valve located outside containment, the second is provided by the closed loop formed by the RBCCW system.

The affected Emergency Response Facility Information System (ERFIS) inputs are NNS and are suitably isolated from safety class inputs.

There is no increase in the probability of occurrence or consequences of an accident or malfunction as previously

evaluated in the FSAR. This 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 is not endangered.

- e. EDCR 90-406 "Torus Vent Line" was completed 4/17/92.

General Summary:

The Torus Vent System (TVS) was installed in response to NRC Generic Letter 89-16, which recommended all Mark I containment BWRs to install a hardened wetwell vent for mitigating a multi-failure beyond design basis event that results in a loss of containment cooling. (Identified as the "TW" sequence in a Probabilistic Risk Assessment) The sequence is a mechanical failure of all pumps and cooling supplies to the Torus. This results in the Torus overheating and becoming over-pressurized. Electrical power is available at all times during the event.

The TVS discharges into the Standby Gas Treatment System (SBGTS) outlet piping to the stack. The vent is a passive system. A rupture disk was installed in the line with a setpoint such that the Primary Containment Pressure Limit (PCPL) will not be exceeded. A normally open Motor Operated Valve (MOV) is downstream of the rupture disk for isolating the vent after initiation. Operation of the MOV is from a keylock switch, located in the control room. Open and closed valve position indication is also provided in the control room.

Each backdraft damper in the SBGTS was replaced with a check valve for isolating the SBGTS from the vented steam. The net increase in system pressure losses due to the check valves does not adversely affect the operation of the SBGTS. Further modifications included adjustments in the fan performance to eliminate vibrations, slower fan speeds to optimize their performance and adjustments to the manual suction dampers to compensate for the fan speed reduction.

Safety Evaluation Summary:

For primary containment pressures less than the rupture disk setpoint, vent path isolation is ensured by the presence of the disk. The Torus Vent System does not interface with any of the systems, structures, or components considered as accident initiators in the FSAR. Because the vent is installed within the design limits of the primary containment, the system will withstand the

peak transient pressure and temperature following the DBA LOCA, accommodate the effects of a potential metal-water reaction following a LOCA, and maintain the primary containment leakage barrier during all normal operating conditions as well as during all previously evaluated accidents. The design requirements imposed on the rupture disk itself ensure its integrity within the design limits of the primary containment.

All components installed in the SBGTS were designed to standards that are equal to or better than those of the existing components. The check valves added to the SBGTS were designed to fully open under the rated conditions of the exhausted air from the SBGTS. They are also normally closed and can therefore perform the isolation function of the original dampers. Thus, they do not inhibit the operation of the SBGTS during any design basis accident or abnormal operational transient previously evaluated in the FSAR. The fan motor re-sizing also does not inhibit the operation of the SBGTS. The design air flow rate of the system is not impacted, nor is the ability of the system to maintain the required negative pressure inside the reactor building. The additional piping and components from the TVS rupture disk up to the SBGTS discharge line is an isolated extension of the SBGTS and therefore does not have any effect on system operation.

There is 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 is not endangered.

- f. EDCR 90-408 "Reactor Building Closed Cooling Water Isolation Valve Power" was completed 4/2/92.

General Summary:

This design was implemented to correct the condition of the Reactor Building Closed Cooling Water (RBCCW) isolation/return valve V70-117 from being powered from a Non-Nuclear Safety (NNS) electrical bus, which was contrary to FSAR requirements, and to correct the exemption of MOV-70-117 from the Environmental Qualification (EQ) Program. This design change replaced the Non-Nuclear Safety (NNS) electrical supply of MOV-70-117 with a Safety Class Electrical (SCE) source, upgraded the components of the motor operator to environmentally qualified components, replaced the MOV-70-117 0.66 hp

motor with a 0.33 hp, 5 ft/lb motor, and replaced the two-rotor limit switch with a four-rotor limit switch to enhance the MOV circuitry.

Safety Evaluation Summary:

The upgrade of the motor operator with EQ components provides added assurance for valve V70-117 to perform its safety function. The replacement motor, although it has a lower hp rating, meets V70-117 5ft/lb requirements and EQ requirements assuring proper operation. The motor operator torque switches provide protection for the valve during both opening and closing operations. The RBCCW system piping was analyzed to take into account the weight of the valve and motor operator and was unaffected by this design change. The change to environmentally qualified MOV components and the change from a two-rotor to a four-rotor limit switch circuit design are enhancements for assuring proper valve operations.

There is 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 is not endangered.

- g. EDCR 90-409 "Cable Separation/Reroute" was completed 4/14/92.

General Summary:

This design change replaced and rerouted existing Safety Class Electrical (SCE) instrumentation and control cables as a result of the upgrading of several existing instrument loops from Non-Nuclear Safety (NNS) to SCE and replaced and rerouted one Standby Liquid Control (SLC) cable to ensure physical separation of the SLC pump control circuits. The affected instrumentation cables supply signals to and from the control room and reactor building, and provide interconnections between various control room panels. The affected control cables provide interconnections between equipment in the reactor building. The power cable provides power to Reactor Core Isolation Cooling (RCIC) Flow Control Loop. The instrumentation loops have cabling which was replaced and routed through the divisional low-level cable trays. The control cable was replaced and routed through dedicated conduit. The power cable was replaced and routed through an existing divisional raceway.

Safety Evaluation Summary:

The loops affected by this design change provide various SCE control and indication functions. This design change only rerouted and reinstalled the cables associated with these loops in accordance with the Vermont Yankee (VY) separation criteria and requirements for safety class equipment. Existing Environmentally Qualified (EQ) cables were replaced with cables that are environmentally qualified. This design did not alter any function previously provided and the loops are operationally unchanged. The implementation of this design in conjunction with the recent upgrade in safety classification of some of the loops provides an enhancement over the original installations. This design improved the reliability of the affected systems. These cables, routed in low-level signal trays, have less electrostatic and electromagnetic interference from control and power circuits.

There is 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 is not endangered.

- h. EDCR 90-410 "Vernon Station Line Breaker 3V4 Modification" was completed 4/4/92.

General Summary:

The Station Blackout Rule (10CFR50.63) specifies that a failure of both Emergency Diesel Generators (EDGs) to start on demand must be considered. The postulated failures include a loss of control power of one EDG, concurrent with a mechanical failure of the other EDG. The alternate AC source for Vermont Yankee (VY) is the Vernon Hydroelectric Station, which can energize either Emergency Bus 3 or 4 by closing the supply line breaker, 3V4, in conjunction with either breaker 3V or 4V. The Vermont Yankee (VY) EDGs and breakers are configured such that EDG 1B and breaker 3V are associated with Bus 3, while EDG 1A and breaker 4V are associated with Bus 4.

Previously, the 125v DC control power for breaker 3V4 was associated with the control power source for breaker 3V and EDG 1B. In that configuration, a loss of EDG 1B control power would render that EDG, and breakers 3V and 3V4 inoperable, thereby removing the ability to energize

Bus 3 from the tie line to the Vernon Hydroelectric Station. Additionally, the ability for Bus 4 to be energized from the tie line would be negated as breaker 3V4 would be inoperable.

This design change provided an independent 125v DC control power source for breaker 3V4, which is diverse from the normal/alternate control power sources that powered the Bus 3 and 4 breakers. Breaker 3V4 control power supply was removed from the original source, and powered from the alternate shutdown battery DC-1AS. In effect, the existing normal/alternate power from DC-1/DC-2 became two alternate control power sources for breaker 3V4.

A box mounted on the rear of Bus 3, Cubicle 2, in the switchgear room, contains the Bus 3 normal/alternate control voltage transfer knife switch. A designation of SW-1 has been assigned to the Bus 3 normal/alternate transfer switch. Additionally, a second knife switch, SW-2, was installed in this box. The output of the switch SW-2 was wired to breaker 3V4. The normal side of the switch was connected to the feeder cable from the DC-1AS supply. The alternate side of the switch was wired to SW-1 control power output contacts. In this configuration, Battery DC-1AS acts as the normal supply, and the Bus 3 normal/alternate supply is the alternate supply. This feature allows maintenance on Battery DC-1AS as required.

Safety Evaluation Summary:

Breaker 3V4 is a 4kv breaker that, when closed in conjunction with breakers 3V or 4V, allows either Bus 3 or 4 respectively to be energized from the Vernon Hydroelectric Station tie line. The classification of Breaker 3V4, and Battery DC-1AS is Non-Nuclear Safety (NNS). However, the breakers associated with Bus 3 are Safety Class; for this reason, the new normal/alternate transfer switch was installed as a Safety Class Electrical (SCE) and Seismic device to ensure that it cannot fail, preventing the Bus 3 safety class breakers from performing their intended functions. Because breaker 3V4 does not interface with any plant safety related equipment, it cannot fail in a manner that prevents any safety related equipment from performing their intended functions.

There is 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 is not endangered.

- i. EDCR 92-401 "Reactor Building Closed Cooling Water Heat Exchanger Alternate Service Water Line" was completed 4/14/92.

General Summary:

This design change provided a 4" Service Water (SW) System outlet on the discharge side of both Reactor Building Closed Cooling Water (RBCCW) heat exchangers, E8-1A and E8-1B, to support replacement of the SW-92A & B heat exchanger discharge valves by allowing the utilization of one discharge valve for isolation during the replacement of the opposite heat exchanger's discharge valve. Two inch plugs were added which provide connection points to be used for draining the RBCCW heat exchanger during normal service.

Safety Evaluation Summary:

The Service Water System pressure boundary modification does not affect a radiological boundary evaluated in the FSAR accident analysis. The Service Water System provides cooling to the RBCCW heat exchangers, which in turn provide safety-related cooling to systems such as the Fuel Pool Cooling and RHR pump seals. The modification does not affect the integrity of the system or impact the design heat load or flow rate of the existing cooling system.

There is 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 is not endangered.

- j. "Installation and Test Procedure SPN-70828-707 for EDCR 89-408 Standby Fuel Pool Cooling System" was completed 2/26/92.

General Summary:

EDCR 89-408 is in the process of being implemented to install the new Standby Fuel Pool Cooling (FPC) System, and includes 12 separate Installation and Test (I&T)

procedures.

This I&T procedure, SPN-70828-707, involves the installation of two motor operated isolation valves. These MOV's are used to isolate the existing Seismic Class II FPC System from the new Seismic Class I Standby FPC System upon line break and low fuel pool level.

Because the SC valves were installed in the existing FPC System's common suction line, the cooling function to the fuel pool needed to be secured to accommodate the installation of the new MOVs. This MOV installation was accomplished prior to the 1992 refueling outage, when the heat load in the fuel pool was at its lowest level. The FSAR transient analysis was not affected since the RHR system was not actively used for fuel pool cooling, but was available for contingency purposes only. Sufficient temperature margin existed, with alternative measures for temperature control within the I&T.

Safety Evaluation Summary

The affected portion of the existing Fuel Pool Cooling System is Safety Class 3. The MOV installation required the isolation of the existing FPC System during which time a piping segment was removed from suction line FPC 1B and replaced with a piping assembly consisting of two MOVs connected by a new 8 foot section of pipe. The installation process required a breach of the piping connected to and located below the fuel pool. The possibility for an uncontrolled leak and a potential release due to inadvertent fuel pool inventory loss was nullified by the redundant isolation provided by the existing system valve and specially designed pipe plugs. The installation was restricted to the FPC System. The I&T procedure provided for continual emphasis of the existing system's safety design basis functions with no impact on other plant systems important to safety.

There is no increase in the probability of occurrence or consequences of an accident or malfunction as previously evaluated in the FSAR. This I&T procedure 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 is not endangered.

- k. PDCR 90-08 "Analog Trip Modification Phase III" was completed 4/6/92.

General Summary:

This change replaced the existing Barton Differential Pressure Indicating Switches, used to sense Main Steam Line High Flow, with Rosemount Differential Pressure analog instrumentation. This increases the accuracy and reliability of the Primary Containment Isolation System Main Steam Line High Steam Flow Isolation Sensors. The Rosemount transmitters and associated trip units have proven to be more reliable than the Barton type pressure switches, and the use of the Rosemount equipment will simplify the calibration and functional checks.

Due to the proven significant reduction in setpoint drift experienced with analog instrumentation loops of the type installed with this design, fewer man-hours are being spent calibrating the sensors, which results in less chance of an error which could cause a spurious plant trip. The time required to perform the monthly functional test is shortened and simplified, and the complexity of the systems and maintenance requirements of the trip sensors have been minimized as a result of this change.

Safety Evaluation Summary:

Installation of the new equipment resulted in minimization of set point drift, less susceptibility to mechanical shock and vibration resulting in a reduction in the potential for an inadvertent plant trip caused by vibration, and an overall improvement in instrument accuracy. The equipment was mounted in a manner that would prevent a failure during a seismic event.

There is 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 is not endangered.

- l. PDCR 90-10 "V10-89B Control Power and Bus 8E/9D Load Shedding" was completed 4/13/92.

General Summary:

This design change permanently supplies Residual Heat

Removal (RHR) System Heat Exchanger Service Water Outlet Valve V10-89B instrumentation with an alternate power source from the "SI" division of the Emergency Power System. Power was previously supplied from Motor Control Center (MCC) 9D, which is powered by the "SII" division of the Emergency Power System. The SII division also supplies control and 480 volt power to the "A" train RHR Valve V10-89A. Subsequently, a failure of the "A" emergency diesel generator to start upon loss of off-site power could have rendered both RHR Service Water trains inoperable.

The single swing bus designated MCC 89, which was part of the original plant design, was subsequently split into two buses, MCC 8E and MCC 9D; the load shed relay "BR", left over from the original MCC 89, shed three loads off MCC 8E and one load from MCC 9D. The old load shedding and blocking function of the "BR" relay was replaced in this design change with two safety class electrical relays. These relays were mounted seismically in panel CRP9-25. These relays are classified as Safety Class Electrical (SCE) and are actuated by the existing load shedding scheme.

Safety Evaluation Summary:

For the RHR valve instrumentation, the power was changed from a "SII" division source to a "SI" division source of power from the Emergency Power System. This does not alter the way the valve or related circuitry operates. The power remains a safety class power source and is now powered from the appropriate division.

The separation of loads shed from MCCs 8E and 9D was changed so that loss of power on MCC 9D will not initiate shedding of loads on MCC 8E. The same loads will still be shed and blocked from restart until the load shed circuit is manually reset.

There is 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 is not endangered.

- m. PDCR 91-01 "Switchyard Breaker Failure Relay Test Switches" (Non-Nuclear Safety) was completed 4/13/92.

General Summary:

This design installed test switches in the current transformer (CT) circuits of the switchyard breaker failure relays. These relays had not previously been calibrated because of the inability to safely isolate them from the CT circuits. Functional testing had been done in the past, but the failure to calibrate the relays raised a concern as to whether the relays would function correctly if a breaker failed. The new test switches installed in the CT circuits of the breaker failure relays provide a safe method of relay isolation. Closing the switch will first short out the CT circuit prior to isolating the breaker failure relay, preventing any accidental opening of an energized CT circuit.

Safety Evaluation Summary:

The protective relay functions were not changed with the installation of the test switches. These switches facilitate calibration of the breaker failure relays and reinforce the reliability of the relays should a breaker fail to open. This design change involved Non-Nuclear Safety (NNS) and Seismic Class II equipment only. The availability of the safety class electrical buses to provide power was not affected as the protective relaying in the 345 KV Switchyard was not affected.

There is 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 is not endangered.

- n. PDCR 91-02 "Containment Spray System Valve Improvements" was completed 4/18/92.

General Summary:

This design change replaced the 12" Containment Spray Isolation Valves V10-26A/B and 31B with double disc, motor-operated gate valves, installed a low-point drain on the Lower Containment Spray Header, removed pipe support RHR-HD-194B at Penetration X39B, and changed support SS-HD-87B (X39A) to a lateral support. Valves V10-26A and 31A are the containment isolation valves for

the Lower Containment Spray Header and were replaced to eliminate leakage, as was V10-31B, the containment isolation valve for the Upper Containment Spray Header. The installation of a drain at the low point of the Lower Containment Spray Header minimizes corrosion in that spray header by allowing water to drain from the header after periodic surveillance cycling of the spray header isolation valves. The drain piping directs the draining water into the nearest Torus Downcomer.

Safety Evaluation Summary:

The safety classification of the supply piping and the spray header is Safety Class 2. The safety classification of the 3/8" drain line is Non-Nuclear Safety (NNS). The new valves perform the same function as the original valves, and were manufactured in accordance with a design code that is equal to or better than the original valves. The addition of the new drain nozzle has been analyzed to demonstrate that the ability of the Containment Spray Header to perform its design function is insignificantly affected. Thus, the ability of the Containment Spray System to remove heat from containment will not be affected.

There is 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 is not endangered.

- o. PDCR 91-03 "Condensate Demineralizer Valve Improvements" was completed 4/16/92.

General Summary:

This design change modified the Condensate Demineralizer system by installing manual butterfly valves on Condensate Demineralizer Vessels B, C, and D to improve vessel isolation capabilities, installing bypass lines around the manual outlet vessel isolation valves to equalize the system pressure, installing normal/bypass switches to provide an electrical override for the "I" valves, to allow manual pressurization of the vessels, and installing differential pressure switches across the "A" valves to prevent the "A" valves from opening with a differential pressure greater than 25 psid. The function of the Condensate Demineralizer System is to remove impurities and solids from the condensate prior to it

being fed back into the reactor vessel via the Feedwater System. The Condensate Demineralizer System is classified as Non-Nuclear Safety (NNS).

Safety Evaluation Summary:

These enhancements to the Condensate Demineralizers do not affect any equipment important to safety. The Condensate Demineralizer System is passive in nature, is used to maintain the required purity of the Feedwater supplied to the reactor, and is isolated from any safety-related system. The butterfly valves do not restrict flow in any way that could cause a feed pump trip.

There is 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 is not endangered.

- p. PD CR 91-05 "Emergency Core Cooling System Test Switch/Undervoltage Input" was completed 4/16/92.

General Summary:

This design change installed keylock test switches between 4KV buses 3 and 4 grid undervoltage relays and their voltage inputs to simplify the procedure for Emergency Core Cooling System (ECCS) testing and to eliminate possible malfunctions associated with testing the undervoltage condition on the vital 4KV switchgear buses. The new test switches help to alleviate test set-up problems caused by the former location and position of the wires into the affected relays.

Safety Evaluation Summary:

The addition of the keylock test switch to the 4KV buses enhances the safety of Vermont Yankee in that the switch prevents unnecessary opening of the switchgear door while circuits are energized, eliminating a hazard to personnel and preventing a possible short circuit of the secondary side of the potential transformers. Also, the elimination of the process for disconnecting and reconnecting the wires to the undervoltage relay prevents possible connection errors and prevents possible damage to the wires from repeated disconnection. Addition of the keylock test switch does not change the system function or response. The switches are only operated

during ECCS testing and are keylocked to prevent accidental operation. The keylock test switch is Safety Class Electrical.

There is 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 is not endangered.

- g. PDCR 91-08 "Chilled Water Supply to Heating, Ventilation & Air Conditioning Units 4, 5 and 6" was completed 6/22/92.

General Summary:

This design change converted the chilled water supply to HVAC units 4, 5, and 6 from a Temporary Modification (TM 89-043) to permanent status. TM 89-043 changed the heat sink for Air Conditioners 4, 5, and 6 in the Chemistry office area from the Potable Water System to a chilled water supply. The chilled water supply comes from SCH-2, a Non-Nuclear Safety (NNS) water chiller. The reason for the temporary modification was that an excessive amount of potable water was being used as the heat sink and then discharged into the main septic system.

Safety Evaluation Summary:

The HVAC system is not an initiator of any previously evaluated accidents or operational transients. The portion of the HVAC system associated with this design change is not utilized to mitigate the consequences of an accident or operational transient. The Non-Nuclear Safety (NNS) portion of the HVAC system does not interface with any system that could create an accident of a different type than previously evaluated.

There is 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 is not endangered.

- r. PDCR 91-09 "Liquid Radwaste Containment Valve Replacement" was completed 4/17/92.

General Summary:

This design change replaced the 3" Liquid Radioactive Waste (LRW) Primary Containment Isolation Valves V20-82, 83, 94 and 95 with air operated ball valves. Valves V20-82 and 83 are in the discharge piping from the Drywell Floor Drain Sump. Valves V20-94 and 95 are in the discharge piping for the Drywell Equipment Drain Sump. New air operators, ASCO NP-1 series solenoid valves and new NAMCO limit switches were installed with the ball valves.

Safety Evaluation Summary:

The LRW system is not an accident initiating system. The ability of the LRW system to perform its design function is not affected by this valve replacement. The new valves perform the same function as the original valves and were manufactured to a design code equal to or better than the original valves. The new valve design ensures that the valve body will not leak and that the orientation of the stem cannot affect closing of the valve. The valves are classified as Safety Class 2, Seismic Class I.

There is 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 is not endangered.

- s. PDCR 91-11 "RHR Head Spray Removal" was completed 4/17/92.

General Summary:

This design change removed the remaining components associated with the head spray portion of the Residual Heat Removal (RHR) System (the elimination of the Head Spray function was accomplished under PDCR 81-12). All piping and supports inside containment from the containment penetration to the first weld below the Drywell bulkhead were removed. The piping penetrating the bulkhead and the piping and flange above the bulkhead were plugged and capped, providing a leaktight seal. All unused RHR system piping and components, upstream to the

main system header and RHR/CST cross connect, were removed. The containment penetration was capped only on the outside. The process piping within the penetration remained in place to minimize disposal costs and provide for potential future needs; the inside opening was sealed to minimize the spread of contamination. Unused cables and conduit were removed. The Head Spray MOV indicator lamps on the CRP 9-3 graphic display were removed, as were the applicable mimic sections. The affected MCC cubicle compartments were redesignated as spares with internal components retired in place for future use.

Safety Evaluation Summary

Removal of the inactive portions of the RHR Head Spray System does not impact any other equipment in a way that could cause an increase in the probability of an equipment malfunction. After component removal, the remaining piping system was still within the allowable seismic stress limits. The RHR piping is classified as Safety Class 2 and the RHR/CST cross connect is classified as Non-Nuclear Safety (NNS).

There is 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 is not endangered.

- t. PDCR 91-16 "Reactor Protection System Motor Generator Panel Upgrade" was completed 4/3/92.

General Summary:

This design change replaced the six GE Electrical Protection Assemblies (EPAs) of the Reactor Protection System (RPS) with six new ASEA assembly units. RPS motor generator power protection panels are designed to provide redundant protection for the RPS buses against transitory overvoltage, undervoltage, and underfrequency conditions. The old GE design combined all protective functions on one card; the ASEA design dedicates each monitoring and trip function to an individual relay. The GE EPAs had been a source of maintenance and calibration problems. The new ASEA equipment requires less maintenance and calibration time for plant surveillance procedures. The new equipment can either be calibrated in place, removed and bench calibrated (either the existing relay or a spare), or any combination of the above.

Safety Evaluation Summary:

The equipment affected by this PDCR are classified as Safety Class Electrical and Seismic Class I, except the indicating lights and test jacks. The new equipment monitors the same parameters and provides isolation to the RPS buses in the same manner as the existing EPA equipment. The ASEA design, like the GE design, has two redundant power protection panels connected in series to satisfy the necessary single failure criteria.

There is 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 is not endangered.

- u. PDCR 91-18 "Main Steam Isolation Valve Switch Bar Design" was completed 4/23/92.

General Summary:

This design change replaced the Main Steam Isolation Valve (MSIV) switch bars, which allows quick replacement of the switches each outage, limits the radiation exposure during replacement, and mechanically supports the four limit switches and electrical terminal box. These limit switches provide indication of valve fully closed and open positions as well as a Reactor Protection System (RPS) scram logic signal when the valves are less than 90% open. The old copper tubing was replaced with stainless steel flex conduit. The terminal boxes and blocks were replaced with quick disconnect connectors.

Safety Evaluation Summary:

The function of the MSIV position limit switches has not been altered. The electrical operation and locations of the MSIV limit switches have not been affected by this change. No safety function has been altered or modified by the implementation of this design. All components and materials used in the fabrication of the switch bars are fully qualified for Safety Class Electrical, Seismic Class I and EQ applications.

There is 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 is not endangered.

- v. PDCR 91-19 "Feedwater Heater 3A and 3B Replacement" (Non-Nuclear Safety) was completed 4/17/92.

General Summary:

This design change replaced the Westinghouse low pressure feedwater heaters E-3-1A and E-3-1B with new heaters manufactured by YUBA Heat Transfer. The significant difference between the old and new heaters is the use of stainless steel for the shell and shell side components to mitigate the problem of erosion/corrosion. Overall length was increased to include an internal flash chamber located beyond the tubing U-bends. Due to the extra length, modifications were made to the heater drain piping 10" HD-3A and 10" HD-3B geometry for the drain inlet connection at the top of the flash chamber of each respective heater.

Safety Evaluation Summary:

The feedwater heaters, piping, systems, instrument and valve components affected by this design change are classified as Non-Nuclear Safety (NNS). The impact of thermal transients from a loss of feedwater heater (LOFWH) event on the reactor/fuel integrity has been analyzed in the FSAR. Licensing basis assumptions for LOFWH analysis are not changed by the heater replacements.

There is 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 is not endangered.

- w. PDCR 91-21 "Safety Valve Acoustic Accelerometer Instrument Relocation" was completed 4/4/92.

General Summary:

This change relocated the Safety Valve Acoustic Accelerometer Instrument Rack to a spare slot in the bottom of Control Room Panel (CRP) 9-18, and permanently removed the Vibration and Loose Parts Monitoring Cabinet.

These changes were completed in preparation for the installation of a new Turbine Vibration Instrumentation Cabinet, to be completed under PDCR 91-12 (Turbine Trip System Modifications).

Safety Evaluation Summary:

The relocation of the Safety Valve (SV) Acoustic Accelerometer Instrumentation Rack does not add, delete or replace any instruments in this system. The SV Acoustic Accelerometer System provides a monitoring function (safety valve position). This is a post-accident instrument system which provides valve position indication only; its function remains the same in the new location.

There is 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 is not endangered.

- x. PDCR 92-02 "Local Leak Rate Test Connection Improvements" was completed 4/13/92.

General Summary:

This change installed new 3/4" Local Leak Rate Test (LLRT) Appendix J test connections on the 8" CS-4A and CS-4B Core Spray injection lines at penetrations X-16B and X-16A respectively, and the 2" CUW-19 Reactor Water Cleanup (RWCU) Suction line. The Core Spray test connections are utilized to allow air testing of the V14-12A and V14-12B valves in accordance with the Vermont Yankee (VY) Appendix J LLRT program. Previously, there had been no test connection between V14-13A/B and V14-12A/B. The third leak test connection, on the Reactor Water Cleanup suction line, allows air testing of the V12-15 valve in accordance with 10CFR50 Appendix J requirements. Previously there had been no test connection between V12-46 and V12-15.

Safety Evaluation Summary:

The test connections are Seismic Class I and Safety Class 1 and 2 material which ensures the integrity of the systems. The RWCU test connection is within primary containment and the effects of breaks or leaks in the system have been previously analyzed. Materials used on

the RWCU test connection in the drywell meet FSAR requirements in that they will not fail by decomposition or corrosion and affect vital systems. These test connections will not hinder the performance of any system that is relied upon to mitigate the consequences of an accident, in that the test connections meet the same design criteria as the Core Spray and RWCU Systems, respectively.

There is 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 is not endangered.

- y. PDCR 92-015 "Intake Boom Installation" was completed 12/15/92.

General Summary

This design change installed a floating, continuous, skirted boom off-shore across the intake structure for the purpose of diverting floating debris and submerged debris (approximately 24" below the surface) away from the intake trash racks, thus reducing accumulation on the racks, improving the circulating water intake trash rack cleaning process, and alleviating personnel safety concerns caused by the manual method of cleaning the trash racks. The diverted debris floats downstream with the river current. A self-floating structural steel gate was installed in the boom line to allow passage of a boat. The boom may remain installed throughout the year except during the time the river freezes solid or during heavy ice flow.

Safety Evaluation Summary

The floating, continuous, skirted boom is Non-Nuclear Safety (NNS), acting as an offshore barrier to divert debris away from the circulating water and service water intake structure. The boom does not impact the design basis of either system. The boom does not perform, support, or inhibit any safety function. The only significant consideration is a failure mode of the boom in which all or a portion would break loose and be drawn across the circulating water intake racks or service water intake, with the boom representing a condition of excessive debris accumulation. However, normal intake operation would be possible since the overall area for

intake volume is not appreciably reduced.

There is no increase in the probability of occurrence or consequences of an accident or malfunction as previously evaluated in the FSAR. This 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 is not endangered.

z. Temporary Modification 91-064 was installed 6/24/92.

General Summary:

This Temporary Modification installed temporary shielding on safety class portions of the Fuel Pool Cooling (FPC) System heat exchanger suction lines, and installed shielding curtains/racks around the heat exchangers. This shielding was required to perform other work activities.

Safety Evaluation Summary:

The shielding affected the Fuel Pool Cooling (FPC) System, adjacent structures, and the FPC heat exchangers, which play no role in the initiation or mitigation of any design basis accident in the FSAR. Shielding analysis ensured safe plant operation such that an operating event would not disable or cause a failure of other safety systems.

There is no increase in the probability of occurrence or consequences of an accident or malfunction as previously evaluated in the FSAR. This 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 is not endangered.

aa. Temporary Modification 91-067 was installed 3/11/92.

General Summary:

This Temporary Modification rerouted the Reactor Water Cleanup (RWCU) return line to the Residual Heat Removal (RHR) System at Valve V10-3 via the shutdown cooling loop. The line was rerouted to maintain RWCU operability during implementation of EDCR 90-403, replacement of Feedwater Check Valves V2-27B and V2-96B in the Main Steam Tunnel. This modification was removed on 4/4/92.

Safety Evaluation Summary:

The plant was shutdown for refueling while this Temporary Modification was in progress. There was no direct interface with systems that initiate any of the design basis accidents. A seismic analysis performed on the modification and its comparable pressure retaining ability provided assurance that the pressure boundary would not be breached as a result of a seismic event.

There is no increase in the probability of occurrence or consequences of an accident or malfunction as previously evaluated in the FSAR. This 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 is not endangered.

- bb. Temporary Modification 92-19 was installed 3/6/92.

General Summary:

During plant shutdown, this Temporary Modification installed a High Efficiency Particulate Air Filter (HEPA) unit to take a suction from the High Pressure Coolant Injection (HPCI) steam supply line and exhaust it to the Reactor Building HVAC System to facilitate ventilation of radioactive gases from the reactor vessel and subsequently allow quicker access to the turbine generator. This modification was removed on 4/10/92.

Safety Evaluation Summary:

HPCI was not required to be operable during the installation of this modification. Installation of the HEPA unit only extended the boundary of the Reactor Building HVAC System inside secondary containment by processing gases and air from the HPCI steam line through the Reactor Building HVAC System rather than through duct work on the refuel floor or by use of the Main Condenser Vacuum Pump.

There is no increase in the probability of occurrence or consequences of an accident or malfunction as previously evaluated in the FSAR. This 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 is not endangered.

- cc. Temporary Modification 92-023 was installed 3/21/92.

General Summary:

This modification installed a temporary work platform on top of the Condensate Storage Tank (CST) to support work being performed inside the tank. This modification was removed on 7/1/92.

Safety Evaluation Summary:

The temporary platform did not affect the functional capability of the CST. The platform was within the design load (40 psf) for the CST and did not compromise the structural integrity of the tank. The CST is classified as Safety Class 2, Seismic Class I.

There is no increase in the probability of occurrence or consequences of an accident or malfunction as previously evaluated in the FSAR. This 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 is not endangered.

- dd. Temporary Modification 92-024 was installed 3/27/92.

General Summary:

This modification installed temporary personnel protection screens on the High Pressure Coolant Injection (HPCI), Reactor Core Isolation Cooling (RCIC), and Condensate Storage Tank (CST) transfer suctions located inside the CST to provide protection to divers in the CST in the event of a HPCI or RCIC initiation and to provide protection from the CST transfer suction while in the CST. This modification was removed on 3/27/92.

Safety Evaluation Summary:

The personnel protection screens were designed not to impede nor prevent suction flow through the HPCI, RCIC or CST transfer nozzles. The barriers were analyzed for seismic affects on the suction nozzles, and were within design loads. The HPCI and RCIC Systems and the CST are classified as Safety Class 2.

There is no increase in the probability of occurrence or consequences of an accident or malfunction as previously

evaluated in the FSAR. This 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 is not endangered.

- ee. Temporary Modification 92-027 was installed 3/13/92.

General Summary:

This modification provided a temporary path for the non-isolable service water flow from RBCCW Heat Exchanger E8-1B to allow replacement of Valve V70-92A. This change provided a temporary interface at the flanged 4" connection of the RBCCW Heat Exchanger to serve as an alternate flow path for Service Water. This modification was removed on 3/20/92. The Service Water System is classified as Safety Class 3.

Safety Evaluation Summary:

There were two potential issues identified for this Temporary Modification: 1) Penetration of secondary containment was made in accordance with procedure OP2116, and 2) Effluent discharge of Service Water from the RBCCW Heat Exchanger was through an alternate path via the Roof Drain System, which is not radiologically monitored. This effluent was monitored by grab samples at least once every 24 hours. The effluent was monitored in accordance with Vermont Yankee (VY) Tech Specs.

There is no increase in the probability of occurrence or consequences of an accident or malfunction as previously evaluated in the FSAR. This 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 is not endangered.

- ff. Temporary Modification 92-028 was installed 3/20/92.

General Summary

This modification provided a temporary path for the non-isolable service water flow from RBCCW Heat Exchanger E8-1A to allow replacement of Valve V70-92B. This change provided a temporary interface at the flanged 4" connection of the RBCCW Heat Exchanger to serve as an alternate flow path for Service Water. This modification was removed on 3/31/92. The Service Water System is

classified as Safety Class 3.

Safety Evaluation Summary:

There were two potential issues identified for this Temporary Modification: 1) Penetration of secondary containment was made in accordance with procedure OP2116, and 2) Effluent discharge of Service Water from the RBCCW Heat Exchanger was through an alternate path via the Roof Drain System, which is not radiologically monitored. This effluent was monitored by grab samples at least once every 24 hours. The effluent was monitored in accordance with Vermont Yankee (VY) Tech Specs.

There is no increase in the probability of occurrence or consequences of an accident or malfunction as previously evaluated in the FSAR. This 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 is not endangered.

gg. Temporary Modification 92-056 was installed 4/5/92.

General Summary

To achieve the required operator thrust for Core Spray Valve V14-5B, the original 2 ft.lb., 0.13 horsepower motor was replaced with a 5 ft.lb., 0.33 horsepower motor. The larger motor accommodates the new thrust value incorporated into the valve setting, based on the conservative thrust data obtained from the MOVATS database, as part of the plant's scheduled updating of Motor Operated Valves (MOVs). (EDCR 92-402 was approved on 1/8/93 which made this a permanent change.)

Safety Evaluation Summary

The function and configuration of this valve and the Core Spray System has not been changed. The new motor meets all the functional requirements of the old motor, except for the increased horsepower, to increase valve stem thrust. The electrical effect of the higher horsepower has been evaluated. Emergency Diesel Generator (EDG) loading, cable size, thermal overload size, circuit breaker size and circuit breaker setting are all within the capability of the installed hardware. This valve and the Core Spray System are classified as Safety Class 2, Seismic Class I. The effect of the increase in motor weight was evaluated and it was determined that the increased weight is acceptable since stresses due to the

added weight are well below the allowable stress limits.

There is no increase in the probability of occurrence or consequences of an accident or malfunction as previously evaluated in the FSAR. This 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 is not endangered.

hh. Reactor Building Closed Cooling Water - Service Water (RBCCW-SW) Discharge Line Work

General Summary

Service Water (SW) valves SW-92 A and B were replaced during the March 92 refueling outage. This was accomplished while the SW System was online, since cooling to the fuel pool, reactor vessel, and key components is required even with the reactor shutdown. The following actions were accomplished in preparation for this valve replacement:

1. EDCR 92-401 (previously summarized) describes the installation of the alternate discharge line/drain line and includes a safety evaluation addressing 50.59(a)(2) concerns.
2. Temporary Modification 92-27 (previously summarized) describes the temporary path from the RBCCW Heat Exchanger to the roof drain and includes a safety evaluation addressing 50.59(a)(2) concerns.
3. An engineering evaluation showed that SW ultimate cooling capability would not be affected with respect to #1 and #2 above, and that if SW were lost, adequate time would exist for the operator to take compensatory actions.
4. An engineering evaluation described how secondary containment would be provided by the proposed SW system lineup with the creation of a 9 ft. loop seal and the avoidance of the known SW ejector/eductor mechanism. Additionally, because of low segregated flows, flooding of the 303' level was predicted not to occur.

Safety Evaluation Summary

The SW system was evaluated and would function as intended, ensuring that the ECCS/support systems would function properly, with no increase in radioactive

material release occurring from overheating of equipment/piping/fuel (and subsequent failure) due to lack of SW. The potential to increase radioactive material release from secondary containment from an open 18 inch discharge line was evaluated, and it was determined that it would perform normally during the valve replacement. Flooding at the 303' level would not increase radioactive material release since none of the radioactive material barriers or nuclear safety-engineered safeguard systems would be damaged by minor flooding. The Service Water System is classified as Safety Class 3, Seismic Class I.

There is no increase in the probability of occurrence or consequences of an accident or malfunction as previously evaluated in the FSAR. This valve replacement 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 is not endangered.

ii. SBGT System Temperature Sensing Element Installation Anomaly

General Summary

During maintenance on the "B" train of the Standby Gas Treatment (SBGT) System, it was discovered that the temperature sensing element TE-1-125-4B was installed incorrectly. Per the manufacturer's recommendations, the temperature element which is installed in the side of the charcoal canister, should penetrate ~6" into the charcoal; however, because the center of the fitting which provides the breach of the canister for the thermocouple was not drilled out, the element remained ~1/2" from the outside edge of the canister and the charcoal, degrading the indication provided by the temperature element. After analysis, it was determined to leave the installation unchanged until the next scheduled maintenance on the system.

The charcoal temperature indication provides information to the operator of the internal temperature of the charcoal bed; the device measures charcoal filter decay heat during and following an accident. TE-1-125-4B in its present location still provides a general indication of the presence of decay heat within the charcoal. Fission products will be spread throughout the charcoal filter, producing decay heat. Increased temperature inside the charcoal will result in increased temperature of the charcoal canister, which will be indicated by the

existing temperature element. The temperature indicator is one of four confirming sources of information which alert the operator to initiate decay heat cooling. This condition was evaluated and determined to be acceptable in its present condition on 11/19/92.

Safety Evaluation Summary

FSAR analysis shows that under worst case conditions, minimum flow through the SBGT filters is sufficient to prevent overheating of the charcoal. Compensatory actions in place insure that minimum flow is maintained at all times after operation under accident conditions. The temperature element still provides indication of increasing temperatures within the charcoal. Compensatory actions in place insure that minimum flow is maintained through the charcoal which prevent excessive temperatures from decay heat buildup.

There is no increase in the probability of occurrence or consequences of an accident or malfunction as previously evaluated in the FSAR. Continuation of the present condition of the "B" SBGT charcoal filter temperature element 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 is not endangered.

jj. Permanent Removal of the Spray Pond Chlorine Monitor

General Summary:

The Spray Pond Chlorine Monitor was placed at the effluent of the Spray Pond to monitor the concentration of Total Residual Chlorine in the effluent of the Spray Pond during periods when chlorination of the Circulating Water was in progress. This was a permitted release under the 1986 National Pollutant Discharge Elimination System (NPDES) permit. The current 1991 permit does not require continuous monitoring of the Total Residual Chlorine in the Circulating Water Effluent during chlorination; therefore, the monitor was no longer required and was removed.

Safety Evaluation Summary:

No accident involving chlorine addition to the Circulating Water and the resultant discharge to the Connecticut River is evaluated in the FSAR. The monitor has no control function for the injection of liquid

sodium hypochlorite (bleach) into the Circulating Water System. The current NPDES permit requires a grab sample and analysis for total residual chlorine concentration; therefore, this monitor is no longer required. The concentration of total residual chlorine in the Spray Pond effluent has no effect on any equipment important to safety.

There is no increase in the probability of occurrence or consequences of an accident or malfunction as previously evaluated in the FSAR. 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 is not endangered.

kk. Cycle 16 Core Performance Analysis was completed 2/20/92.

General Summary

There were three major changes made to the Vermont Yankee Cycle 16 core. These changes included:

- 1) 128 General Electric GE-9B bundles,
- 2) Repositioning of 4 Siemens Nuclear Power (SNP) Qualification Fuel Assemblies (QFAs), and
- 3) Insertion of 1 GE Control Rod.

The Cycle 16 Core retained 240 irradiated fuel bundles, and 88 irradiated control rods.

Safety Evaluation Summary:

The Cycle 16 core design does not increase the probability of an accident because the bundles, channels and control rods used in the Cycle 16 core have been approved by the NRC to be used in all BWRs. The bundles, channels and control rods were designed to meet the same licensing criteria as previously used at Vermont Yankee. The core is being operated in the same power and flow regions as previously assumed in the design analysis. The thermal-hydraulic physics, transient and accident analyses for the Cycle 16 core were performed using NRC-approved methods. These analyses show that no safety limit would be violated nor would there be any increase in the consequences of an accident previously evaluated. The core changes do not affect any assumptions used in evaluating the radiological consequences of a malfunction of equipment important to safety.

There is no increase in the probability of occurrence or consequences of an accident or malfunction as previously evaluated in the FSAR. These core changes 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 is not endangered.

B. Tests and Experiments

1. Moisture Separator Performance Testing - completed on 5/6/92. (Special Test Procedure 92-01)

General Summary

This test determined the moisture content in the four crossaround pipes and evaluated the separation capability of the four moisture separators. The testing was performed eight times - one test for each injection point. Each test injected a chemical tracer, lithium nitrate, at the injection point and collected a sample at the corresponding moisture separator drain line. Condensate demineralizers removed the majority of the chemical tracer; the remainder was removed by the Reactor Water Cleanup (RWCU) System.

Safety Evaluation Summary

The testing was performed on Non Nuclear Safety (NNS) and Seismic Class II systems. Increasing reactor coolant conductivity below any administrative action levels would not initiate any accident. This testing did not affect the operation of any equipment important to safety.

There is no increase in the probability of occurrence or consequences of an accident or malfunction as previously evaluated in the FSAR. This 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 is not endangered.

C. Valve Line-up Deviations

1. None

D. Safety and Relief Valve Challenges and/or Failures

1. None

E. Special Test Procedures

1. Special Test Procedure 92-01 Moisture Separator
Performance Testing

See General Summary and Safety Evaluation under Section
B. above.