VERMONT YANKEE

CYCLE 17 OPERATING REPORT

Between October 25, 1993, and May 2, 1995, 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). This report includes eight Engineering Design Change Requests (EDCRs), sixteen Plant Design Change Requests (PDCRs), twenty-eight Temporary Modifications (TMs), ten Special Test Procedures, two Bases for Maintaining Operation (BMO), three valve lineup deviations, one instrument setpoint change, one core reload, one connection of chart recorder to master trip card test jacks, a modification to the refuel floor auxiliary bridge, a modification to the Low Level Waste Storage Pad, and a fuse addition to the Main Steam Relief Valve Bellows leakage detection circuitry performed under the minor modification process. There were also one safety class reclassification and seven procedures requiring safety evaluations. There were no safety relief valve failures or service water lineup changes.

I. Changes in Facility Design

A. Between October 25, 1993, and May 2, 1995, there were no changes made which required authorization from the commission.

B. The following changes did not require Commission approval. They were reviewed by the Plant Operations Review Committee and approved by the Plant Manager. It was determined that these changes did not involve unreviewed safety questions as defined in 10CFR50.59(a)(2).

1. **EDCR 92-405 Emergency Diesel Generator (EDG) Relay Replacement** was completed 4/20/95.

General Summary
This design change replaced the four EDG relays, 25DG (synchronization check), 51V (voltage controlled overcurrent), 59DG (breaker voltage permissive) and 60DG (voltage balance/blown fuse), due to the unavailability of replacement parts. The new 60DG relay was connected to a time delay relay to ensure simultaneous sensing of the generator and bus potentials. With the exception of the 59DG relay, the replacement relays were installed in approximately the same locations as the previous relays. The 59DG relay was relocated to the relay cabinet door due to its need to be panel mounted.

Safety Evaluation Summary
This design change does not alter any automatic operation of equipment required to perform a safety function; nor does it alter the operators' actions. Therefore it does not impact the operators' ability to safely shut down the plant. The replacement

relays are functionally identical and satisfy all the design and functional requirements of the previously installed relays. The relays are seismically mounted to ensure that the seismic integrity of the relays and relay cabinet doors are maintained.

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.

2. **EDCR 92-407 Turbine Building Roof Ventilation Re-Route** was completed 5/3/94.

General Summary
This design change disabled and sealed the turbine building roof ventilators (TRV-5 and TRV-6), which discharged air directly into the environs; and rerouted exhaust air to the main stack, thus providing radiation monitoring of turbine building exhaust air by using the existing sampling system in the stack. Monitoring capability is provided for normal plant operating conditions as well as accident and post-accident conditions in accordance with the requirements of NUREG 0737.

The new exhaust system consists of two 25,000 cfm centrifugal fans configured to run in parallel for a total design capacity of 50,000 cfm. A single suction is taken from the east wall of the turbine building and routed outside, where it enters a plenum common to both fans in the HVAC room. Each fan discharge duct ties into the existing HVAC duct inside the fan room which continues to the main stack. A manually operated isolation damper is installed on the suction side of each fan. A gravity operated backdraft damper on the discharge side prevents backflow when the respective fan is not operating.

Safety Evaluation Summary
One of the design basis accidents described in the FSAR is a main steam line break; because the pipe break is assumed to occur in the turbine building, the consequences of this could be affected by the new exhaust system. The old TRV's would immediately be shut down if high radiation levels were detected in the turbine building. However, the new exhaust system would not be shut down as quickly because the emergency stop switch, for the fans, is disabled. The additional 50,000 cfm of air exhausting to the environment would not increase the radiological consequences of the accident because: 1) The description of the main steam line break in the FSAR assumes that the pressure buildup inside the turbine building causes the blowout panels to blowout; the release path, therefore, is independent of either the old roof ventilators or the new exhaust system, and 2) the accident description in the FSAR assumes that all the activity released from the reactor vessel

to the turbine building escapes to the environment. The maximum possible amount of radiological material is, therefore, already assumed to be released.

Due to the increase in flow rate out the stack, there will be a small (less than a factor of two) increase in the reported consequences of the control rod drop accident as presented in section 14.6 of the FSAR. However, the Control Rod Drop accident is also evaluated in FSAR section 14.9; the reported consequences in section 14.6 are 3 to 5 orders of magnitude below the consequences discussed in section 14.9, which assumes a ground level release of fission products. The control rod drop accident analyzed in section 14.9 is the licensing basis event.

The new exhaust system does not interact with, or in any way affect, any of the safety systems defined in the FSAR. Other than the main steam line break addressed above, there are no safety systems, design basis accidents, or abnormal operational transients described in the FSAR which result in a radiological release to the turbine building. Additionally, the system and its components are virtually identical to existing station HVAC systems. It does not interfere with the function of any system 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. 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. **EDCR 94-401 ATWS Diversity Upgrades** was completed 4/27/95.

General Summary

The regulatory requirements of 10CFR50.62, Anticipated Transient Without Scram (ATWS) Rule, state in part that "each boiling water reactor must have an Alternate Rod Injection (ARI) system that is diverse (from the reactor trip system) from sensor output to the actuation device." Previous to this design change, both the ARI and Reactor Protection System (RPS) used Rosemount trip cards and Agastat relays; the ARI system did not adequately comply with the ATWS Rule Equipment Diversity requirement of different manufacturers as defined by the NRC.

This design change installed new ARI/RPT level and pressure transmitters, modified two Reactor Water Level instrument loops, installed impulse tubing for new transmitters off existing instrument tubing, installed new instrument cable and conduit, used existing power supplies in each Containment Air Dilution (CAD) panel to power transmitter instrument loops, and installed new diverse alarm relay modules and relays in card file/relay racks in the CAD panels. New instrument tubing, valves, fittings and transmitter pressure boundary portions are Safety Class 2. The new ARI/RPT instrumentation is Safety Class Electrical.

Safety Evaluation Summary
The ARI/RPT system provides a means to scram the reactor and to mitigate the consequences of a failure of control rods to insert. The new equipment meets or exceeds the safety and design criteria of the previous equipment. This design change did not degrade the integrity of any radioactive material barriers. This change maintained the independence of the ATWS system from the Reactor Protection System (RPS) such that malfunctions in the ATWS mitigation system will not affect the RPS or any other safety related system. None of the malfunctions identified in the FSAR were made more probable by the implementation of this design change, and no new failure modes of safety related systems were introduced by this design.

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. **EDCR 94-403 Degraded Grid Under Voltage Relay Replacement** was completed 4/25/95.

General Summary
In response to NRC concerns identified in the Electrical Distribution System Functional Inspection (EDSFI), this design change installed new degraded grid undervoltage relays to prevent inadvertent separation of a Safety Class Electrical (SCE) Bus from the preferred (offsite) power source. The new relays are more accurate and have a smaller reset ratio. This design change also improves the accuracy of the instrument loop and significantly reduces the potential for setpoint drift outside Technical Specification limits. Also included were improvements to the diesel generator breaker circuitry with the implementation of a minor wiring change to prevent uncontrolled closure of the breaker.

Safety Evaluation Summary
The operation and safety function of the degraded grid protection system and diesel generator response to a Loss of Normal Power (LNP) were not impacted by this modification. Operation of the degraded grid protection system did not change. This design change did not introduce any new signals into the LNP/Diesel breaker control logic, nor did it interfere with the safety function of equipment or components. The automatic start of the diesel generator and closure of its output breaker during a LNP scenario, with or without the presence of and accident, did not change.

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.

5. **EDCR 94-404 Main Steam Isolation Valve (MSIV) Solenoid Valve Replacement** was completed 4/27/95.

General Summary
Due to the manufacturer's discontinuation of specific solenoid valves used for Vermont Yankee MSIVs, this design change replaced eight MSIV solenoid valve assemblies with new solenoid valve clusters, one on each MSIV. The cluster arrangement of three solenoids mounted onto a single valve manifold block provides sufficient space between the DC and AC solenoid valve coils to reduce any potential synergistic aging interaction between the two coils. The clusters have been designed and specifically tested to ensure they meet all specifications. Consequently, installation of these clusters can operationally be considered exact replacements.

Safety Evaluation Summary
MSIVs are used as mitigators of accidents listed in the FSAR. The replacement MSIV solenoid valve clusters were seismically and environmentally tested and qualified for both normal and accident conditions; the testing verified that the clusters will not fail in a manner to prevent the MSIVs from performing their intended safety function. The testing also verified that the solenoid valve clusters are not susceptible to a failure that would initiate inadvertent closure of one or all of the MSIVs under normal operating conditions.

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.

6. **EDCR 94-406 Motor Operated Valve (MOV) Improvements** was completed 5/2/95.

General Summary

NRC Generic Letter 89-10 requested that licensees develop a comprehensive program to ensure that MOVs perform their safety function for design basis conditions. As part of this program, design basis reviews were performed to evaluate valve operability for worst case accident conditions and degraded bus voltage conditions. Certain MOVs were found to require hardware modification to ensure that the required thrust for worst case accident and degraded bus voltage will be achieved so that the valves will perform their safety function.

The Close Torque Switch bypass range was reduced on Recirculation Pump Discharge Valves V2-53A and B, Recirculation Pump Discharge Bypass Valves V2-54A and B, and Residual Heat Removal Discharge to Radwaste valve V10-66. On Reactor Water Cleanup Unit (RWCU) Inboard Isolation Valve V12-15, the SMB-000 actuator and 5 ft-lb AC motor were replaced with a SMB-00 actuator and a 10 ft-lb AC motor. On RWCU Outboard Isolation Valve V12-18, the SMB-000 actuator and 5 ft-lb DC motor were replaced with a SMB-00 actuator and a 15 ft-lb DC motor. On HPCI Inboard Isolation valve V23-15, the SMB-0 actuator and 40 ft-lb AC motor were replaced with an SMB-1 actuator and 60 ft-lb AC motor.

Safety Evaluation Summary
This design change resolved Generic Letter 89-10 concerns for the affected valves, thereby reducing the chance of a malfunction of equipment by ensuring that valve motor-operator switch settings are set and maintained to assure valve operability. These modifications did not introduce any previously unreviewed interactions or delete designed interaction between structures, systems or components important to safety. Automatic and manual operation of the valves and related Emergency Core Cooling System components were not affected. The modifications did not affect the system operational requirements or maximum valve operating times, did not reduce the difference between a system failure point and accepted safety limit, and did not in any way reduce the margin of safety.

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.

7. **EDCR 94-408 Moisture Separator Drain System Modifications** was completed 4/26/95.

General Summary
This design change upgraded the Moisture Separator (MS) Drain System to correct full power stability issues. The normal and emergency pneumatic systems were replaced with electronic/digital level control systems. The normal MS drain valves were replaced with new valves that are more appropriately sized for the MS drain flow and easier to maintain. The new digital controllers provide valve position feedback to facilitate system troubleshooting and testing. This design change also installed new electronic positioners on the MS emergency dump valves that were also

replaced during the 1995 refuel outage.

Safety Evaluation Summary
The components affected by the equipment installed with this design change are neither accident initiators nor accident mitigators. Failure of these level control systems would not result in transient more severe than the 100 degrees F step reduction in feedwater temperature with subsequent failure to scram as assumed in the FSAR. The moisture separator drain tank level control systems cannot fail in a manner that would either increase the Turbine Stop valve closure time, or prevent the bypass valves from opening following a Turbine Trip.

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.

8. **EDCR 94-410 SW-92 Valve Replacement** had 99 percent of the work completed on 4/28/95; the remaining details were completed 6/15/95.

General Summary
This design change modified the seismically supported service water (SW) discharge lines 12"SW-16A and 16B from each Reactor Building Closed Cooling Water (RBCCW) heat exchanger; removed the common discharge line, including the connecting tee to the overall system return header 18"SW-12; installed new 8-inch drag valves in each discharge line; added a new common discharge header, which includes a spectacle flange and tee into the overall system return header 18"SW-12 to allow for easy isolation of the SW piping associated with the RBCCW heat exchangers from the SW system discharge header; and added instrumentation that allows for monitoring SW pressures and flow, using delta-p, through the heat exchangers. These modifications provide the operators with better SW flow control capability across the RBCCW heat exchangers.

Safety Evaluation
The SW system interfaces with other systems that have the potential for containing radioactive material. The SW system absorbs heat from these systems and discharges it to the Connecticut River. This design change did not affect the pressures of the systems cooled by the SW system, and there was no appreciable pressure change in the SW system; thus, there is no increased potential for discharge of process water to the Connecticut River. Although this design change reduced the number of flow paths available to cool each heat exchanger (one 10-inch line versus one 12-inch line and one 4-inch line), the new line includes a valve that is designed to control flow over the complete range of flows. The previous configuration did not

have that capability. Also, the heat exchangers have 100 percent redundancy and provide no safety-related cooling function. Consequences of a failure of the SW system piping associated with the RBCCW heat exchangers are the same as for the previous installation, actions assumed in the accident analysis were not degraded or compromised, and the accident mitigation process was not degraded by the modifications of this design change.
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.

9. **PDCR 92-006 Closed Circuit Television Modifications**, completed 11/24/9. Classified as Safeguards information.

10. **PDCR 92-009 Switchyard Battery Charger Alarms** was completed 8/6/93; a 10 CFR 50.59(a)(2) safety evaluation was subsequently written on 6/29/94.

PDCR 92-009 included a written 50.59 (a)(1) screening assessment which evaluated the change made under the PDCR and concluded that the changes did not impact safety and had no potential for an unreviewed safety question. The assessment was performed in accordance with the methodology which was in place at the time and had been presented at training.

In October 1993, as a result of questions raised on the adequacy of Temporary Modification Screening criteria which was in place at the time, an interim policy was established that Temporary Modifications or Design Changes involving safety class or Technical Specification structures, systems or components be accompanied by a 50.59 a(2) safety evaluation. As a result of that policy, it was determined that a 50.59 a(2) safety evaluation should be prepared for PDCR 92-009 since the Switchyard Battery Chargers are Technical Specification equipment.

A 50.59 a(2) safety evaluation was prepared; it documented that the change did not result in an unreviewed safety question. This confirmed the conclusion in the screening assessment documented in section 11.0 of the PDCR.

General Summary
This design change added an alarm bypass switch to each of the switchyard battery chargers BC-5A-A, BC-5A-B, BC-4A-5A, and BC-4A. The switch allows bypass of the alarms for chargers removed from service. The switch has "Normal" and "Bypass" positions and is wired to the field side of the charger alarm terminal block to minimize the impact to the charger. The switch bypassed the annunciator input

to preclude the need to lift leads via the Temporary Modification process when necessary to prevent alarms from chargers removed from service.

Safety Evaluation Summary

Adding a bypass switch to the Non Nuclear Safety switchyard battery chargers alarm circuitry has no potential to impact primary systems or any safety related equipment. There is no possibility of any accident occurring as a result of this modification. There was no possibility of malfunction which could cause or threaten failure of any radioactive material barriers, nuclear safety systems, or engineered safeguard systems.

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.

11. **PDCR 93-011 North Warehouse Oil Burner** was completed 5/1/94.

General Summary

This design change installed, in the north warehouse, a waste oil burner, which burns radioactively contaminated waste oil, and a 500-gallon holding tank. Thermal fire detectors were installed. A 12' by 12' by 6" berm was added around the holding tank to provide a volumetric dike for the maximum amount of oil which could be stored in the tank. An alarm was installed on the outside of the north warehouse.

This design change additionally installed local thermal detectors in the Containment Access Building to enhance the fire detection/shutoff capabilities of the waste oil burner already in place.

Safety Evaluation Summary

All systems important to safety are isolated from the north warehouse waste oil burner and are not affected by its operation. Due to the alarm, automatic shutdown feature, and routine security tours, any fire would be quickly detected. An evaluation was performed regarding the consequences of a fire burning the entire stored inventory of radioactive material in the North Warehouse; it was determined that all site boundary dose criteria would be met. Radioactively contaminated waste oil is burned in accordance with effluent release limitation of 10CFR Part 20 and Part 50, Appendix I, as well as Technical Specification gaseous pathway dose and dose rate limits.

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.

12. **PDCR 93-019 Circulation Water/Service Water Chemical Treatment System** was completed 7/19/94.

General Summary

A study conducted in 1991 concluded that a mixture of approximately equal percentages by volume of sodium bromide and sodium hypochlorite would improve the effectiveness of the chemical treatment of the circulating water and service water systems. The study also recommended the addition of an injection system for sodium bromide and replacement of major portions of the hypochlorite injection system and the pH control system. This design change installed new and replacement equipment to incorporate these improvements for the chemical treatment of circulating water and service water.

Safety Evaluation Summary

The only safety system which could be affected in any manner by this modification is service water. Loss of Residual Heat Removal Service Water has been evaluated in the FSAR (14.5.7); this evaluation concludes that loss of service water will not result in conditions which are adverse to the health and safety of the general public. The injection of sodium bromide and sodium hypochlorite into the service water system reduces bio-fouling within the service water piping and heat exchangers. Changes made by this design change do not adversely impact the performance of the circulating water or service water systems.

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.

13. **PDCR 94-001 Low Pressure Turbine Retrofit** was completed 4/26/95.

General Summary

This design change replaced the low pressure turbines, which had been damaged by erosion/corrosion. The new turbines are more efficient, with an increase of gross megawatt output from the main generator. This increase is primarily attributed to the improved steam path design of the replacement low pressure turbines. Some of the steam path efficiency improvements consist of plugging the seventh stage wheel hole, reducing radial tip spill strip and diaphragm packing clearances, installing integral covered buckets on stages seven and eight, and contouring the sidewalls of the double flow inlet tub.

Safety Evaluation Summary

Of the four design basis accidents (DBA) discussed in the FSAR, the Main Steam Line Break is the only DBA which the turbine generator system interfaces with. The main steam line DBA assumes that a steam line is instantly severed outside containment. This design change does not physically alter the main steam lines in any way. All changes take place downstream of the turbine stop and control valves, and high pressure turbine. Also, the Main Steam flow rate and turbine control valve inlet pressure remain the same. This ensures that the probability of a previo[Busly evaluated accident will not increase. The only new failure mode added by this modification is the potential for a gasket failure on the turbine extraction line expansion bellows joints. This failure potential is minimized by the design of the gaskets, which consist of both an inner and outer 304L stainless steel ring, and the Quality Control imposed during torquing of the flanged joints.

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.

14. **PDCR 94-002 Turbine Performance Monitoring Instrumentation Installation Phase II** was completed 4/26/95.

General Summary

This design change installed three performance monitoring transmitters to monitor the Steam Seal Regulator (SSR) Header flow associated with the high and low pressure turbine packings, and one performance monitoring transmitter to monitor the steam jet air ejector (SJAE) steam supply flow. The transmitter indications are used for turbine plant performance analysis purposes. The SJAE flow transmitter required the installation of a new flow element in the associated steam piping. All the transmitters and associated flow elements are classified as non-nuclear safety (NNS). The piping in which the associated flow elements are installed is [Bclassified as NNS related and non-seismic.

Safety Evaluation Summary

The piping and associated components affected by the instrumentation installed by this design change are neither initiators nor mitigators of any FSAR Chapter 14 accidents. The instrumentation installed by this design change provides indication of the associated SJAE and SSR header steam line flows only, is used for operational analysis of the associated systems, and does not initiate nay automatic functions;

therefore, the instrumentation cannot initiate any abnormal transients. This instrumentation was designed and installed to have no effect on the operation of the SJAE or SSR systems, thus ensuring that these systems will accomplish their intended functions.

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.

15. **PDCR 94-004 Scram Test Panel Upgrades** was completed 4/28/95.

General Summary

This design change implemented long term solutions to issues associated with scram timing and Reactor Protection System (RPS) circuit enhancements. The (no longer in use) Average Power Range Monitor (APRM) Setdown and Select Rod Insert circuitry in the Control Room was disconnected and removed. The equipment and wiring in control room panels CRP 9-16 and CRP 9-28, no longer utilized for either full or single rod scram timing, was disconnected and removed. The tap locations for the Emergency Response Information System (ERFIS) computer points M548 through M555 were relocated to the other side of the scram solenoid group "white" light current limiting resistors to provide better isolation between the wiring for the scram solenoids of different rod groups. Disconnect type terminal blocks were installed in the sensor circuits for the RPS relays that initiate a scram on Main Steam Isolation Valve (MSIV) or Stop Valve closure to provide a means for these relays to be individually de-energized to support I&C monthly Technical Specification surveillance testing.

Safety Evaluation Summary

This design change supports the bases of the RPS by providing individual rod scram time test data which is more representative of the actual scram time by allowing the individual scram switches to be toggled to the full down position during surveillance testing and providing a more appropriate means of de-energizing the RPS sensor circuits that initiate a scram on MSIV and stop valve closure to satisfy Technical Specification 4.1.A surveillance requirements.

The RPS performs a mitigating function for accidents described in the FSAR. This design change does not directly or indirectly increase the radioactive material release because this change had no impact on the Safety Objective or Safety Design Bases of the RPS. The only change in equipment operation is that when the rod scram switch is moved to the full down (previously "select") position, the rod will scram as the 117 and 118 scram solenoids from the RPS neutral bus are disconnected. This action is consistent with the design bases of the system and more clearly satisfies RPS Safety Design Bases 7.f. There is no increase in the probability of an inadvertent scram caused by the open circuit failure of a multiple of the new disconnects because the disconnect type terminal blocks are qualified Safety Class Electric and seismically mounted within the respective RPS Division cabinets. This design change did not alter the required functions of safety related equipment, nor does it have the potential to impact the Primary System either directly or indirectly.

16. **<u>PDCR 94-005 Low Pressure Turbine Replacement - Office Facility</u>** was completed 3/8/95.

<u>General Summary</u>

This design change constructed a permanent modular office facility at the north end of the turbine building operating floor at elevation 272'6", adjacent to the viewing gallery wall and approximately centered between the viewing windows. This facility is a two level structure consisting of a 14' by 32' break area for craft personnel on the lower level and two offices for supervisors on the second level, and is intended for use only during outages. A heating, ventilation and air conditioning (HVAC) system was installed; the HVAC equipment is located on the Administration Building roof, and the duct work is routed to the facility across the top of the viewing gallery. The HVAC unit has an internal damper which is opened by a motor and closed by a spring. This eliminates the possibility of air flow through the facility upon a loss of power to the HVAC unit.

<u>Safety Evaluation Summary</u>

This structure has no direct ties to a safety-related system. Failure of this modular office facility will not result in a challenge to a safety system. This facility does not form or directly connect to the primary barriers which limit the consequences of an accident. Operation or failure of this office facility will not directly or indirectly result in a challenge to or failure of the barriers which could increase radioactive material release.

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.

17. **<u>PDCR 94-006 Low Pressure Turbine Storage Facility</u>** was completed 12/1/94.

<u>General Summary</u>

This design change constructed a weather tight storage facility for the Low Pressure

Turbine components which were replaced during the 1995 outage. Storage of the turbine components meets the criteria established for maintaining power generation through the license period. The storage facility is located adjacent to the 345 KV switchyard and consists of four fiberglass-reinforced enclosures. Four concrete slabs, on engineered structural fill, were installed for each enclosure.

Safety Evaluation Summary

The enclosures are physically separated from all other plant buildings, are not Fire Control areas, and are not vital to plant safety or operation. Failure of these structures will not result in a challenge to a safety system. This change does not directly or indirectly impact the barriers credited in the FSAR for limiting the consequences associated with equipment malfunctions of safety-related components. As this change did not alter the function or performance of a safety system, it therefore did not alter the challenges the barriers were designed to mitigate.

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.

18. **PDCR 94-007 Containment Air Improvements** was completed 4/28/95.

General Summary

This design change installed a filtered pneumatic supply through the Primary Containment check valves for the Containment Air supply. Stainless steel piping was installed from the original tee, assuring that no particulate contamination can be introduced from the piping. This assures that gas going through the check valves is clean and should not be contaminated by particulate entrainment. The filters were installed to allow on-line maintenance. This design change did not change the method that supplies nitrogen to the containment during normal and off-normal operations while the containment is inerted.

Safety Evaluation Summary

The configuration of the piping for containment Penetration X-22 was revised. However, the original concept for Primary Containment Isolation was not changed. This design ensures that the valves perform their safety class function and mitigate the consequences of an accident. The Primary Containment Isolation valves are identical as before the design change. The outboard valve was relocated closer to the inboard valve. Relocating an identical valve closer to Containment cannot increase the potential to initiate an Abnormal Operating Transient. Installing a new test connection for testing V72-89C did not increase the potential to initiate an Abnormal

Operating Transient because the test connection is a passive, small branch connection that is valved out during operation. Installing filters upstream of the Containment Isolation valves in the Non Nuclear Safety portion of the line does not increase the potential of the NNS/non-seismic nitrogen or instrument air system for causing a transient. The plugging of one of the parallel filters has the same potential to cause a loss of pneumatic pressure as the original Containment Air Compressor filter. Redesignating the Containment Air Compressor as the backup source of pneumatic pressure cannot increase the potential to initiate an Abnormal Operating Transient. All the current sources, Nitrogen makeup, Instrument Air and the Containment Air Compressor are Non Nuclear Safety.

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.

19. **PDCR 94-013 Stack Flow Monitor Upgrade** was completed 7/7/94.

General Summary

This design change upgraded the Stack Air Flow Monitor. The old system was no longer able to be effectively maintained and calibrated, in part due to its age, unique design, and lack of vendor support. Two new solid state thermal sensing probes were installed at the north and south azimuths of the off-gas stack at elevation 461'. A new electronics control and display enclosure, which performs analysis of sensor inputs and liquid crystal display (LCD) readouts, was installed in the Victoreen Room at the plant stack. System operation with the new instrumentation essentially remains unchanged. The new system employs the same air flow measurement and totalizer features as that of the previous instrumentation. However, the new system is configured to indicate in volumetric flow units, which differs from the original system which was configured to display velocity. The volumetric flow readout provides Control Room operators with data that can be directly utilized as a diagnostic tool in comparing stack flow to the summation of operating exhaust fan air flows.

Safety Evaluation Summary

The plant stack and associated air flow monitoring system do not communicate with or provide control input into any nuclear safety or engineered safeguard systems. Therefore, the plant stack and its associated air flow monitor system are neither initiators nor contributors to any accidents described in the FSAR. The physical modifications of this design change have been evaluated and found not to impact the stack's structural integrity. There would be no increase in the volume of radioactivity released during any of the accidents described in the FSAR, and the plant stack's function as an elevated radiological release point would be maintained. The failure mode of the plant stack is to not provide a point of elevated release following an accident. The installation of this design change does not change this, or create a new failure mode.

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.

20. **PDCR 94-014 Security Fence and Lighting** was completed 12/27/94 and is classified as Safeguards information.

21. **PDCR 94-015 Construction Storage Building Modifications** was completed 4/14/95.

General Summary
This design modified the interior of the Construction Storage Building and added a 2000 square foot extension to the west side. The extension contains an air compressor room, electric utility room, gas manifold room and craft staging area. The craft staging area provides housing for the number of craft personnel that would require four trailers; this addition thus eliminates the need for four outage trailers, and provides an environment for positive control over craft personnel. The modification to the existing building provides a clean work shop for all work outside the RCA, clean tool room, storage for calibrated electrical equipment, insulators work area, electrical work area, carpentry work area, welding and machining.

Safety Evaluation Summary
There is no interface with any safety class plant systems which could increase the possibility of an accident. The Construction Storage Building is not adjacent to any safety related structures which could be impaired by collapse of the Construction Storage Building, nor is it located over any safety piping or conduit. Adequate fire protection is maintained by secured cables, fire control areas, physical separation, hydrant access and safety flash arresters on the fuel gas piping. Neither the addition nor the shop area is used for bulk storage of combustibles or flammable gases or liquids.

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.

22. **PDCR 94-017 Service Water Operational Performance Inspection (SWOPI) Electrical Modifications** was completed 4/28/95.

General Summary
In response to the NRC SWOPI results, this design installed the following modifications:

To isolate the turbine building service water (SW) loads whenever SW header pressure drops below a predetermined value, two Safety Class Electrical pressure switches were connected to the SW header in each Residual Heat Removal (RHR) corner room. The new switches are mounted adjacent to Instrument Racks 25-7A and 25-23. The pressure switches are connected in a one out of two, taken twice logic. The logic energizes time delay relays; after the appropriate time delay, the relays energize the closing coils in the motor starters for V70-19A, V70-19B and V70-20. The relays also inhibit energization of the opening coil in the motor starters for these valves. The arrangement of the pressure switches and the logic is such that a loss of pressure in the SW header in either RHR corner room will cause all three valves to close.

When the turbine building SW loads are isolated by closing valves V70-19A and B, the service water header pressure indication provided by PT-104-20 will be disabled since the transmitter is downstream of these valves. The instrument line to PT-104-20 was capped. Two new Safety Class 2 pressure transmitters were installed to monitor the SW header piping. These transmitters each provide a signal to a dual indicator which replaced indicator PI-104-20 on control room panel 9-6.

The Safety Class motor control circuit which controls the safety related Fan 2-1 of the Cooling Towers was relocated from two feet below the Maximum Probable Flood (MPF) water level, to a separate cabinet, with all Safety Class components located above the MPF level. A concrete pad was poured and an enclosure was mounted on the pad; installed in this enclosure are contactors, overload heaters, timing relays, auxiliary relays and new control power transformers.

The old circuit breakers which supplied the cooling tower fans had inadequate interrupting capability to effectively clear the maximum calculated short circuit current. These were replaced with now motor protectors with an interrupting capacity significantly higher than the calculated fault current. These motor protectors are similar in size to the old breakers and were mounted in the original location. Also replaced were circuit breakers providing the bus tie and breakers which feed lighting panels, the freeze protection panel and power receptacles.

Safety Evaluation Summary
Automatic isolation of non-essential SW system loads assures the ability of the SW system to perform its safety function by automatically closing valves to the turbine building non-essential loads. This eliminates reliance on manual operator action. This change provides added assurance that the SW system will provide cooling to safety related loads following an accident or event. Automatic isolation of nonessential SW system loads improves the ability of the SW to provide cooling to accident mitigation equipment. The only credible failure from this modification is an inadvertent isolation of the non-essential service water loads. This failure would not affect the ability of the SW system to perform its safety function.

All equipment added is of equal or better quality than the original. The failure modes of this equipment have been analyzed, and are the same failure modes for other plant protection actuation systems. The addition of a small quantity of safety class 3 instrument tubing, four Safety Class 3 pressure switches and two Safety Class 3 transmitters does not increase the probability of a failure of the SW pressure boundary. There are no additional failure modes introduced as a result of this modification. None of the failure modes are made more probable as a result of this modification. The only new type of component added by this change are the pressure switches. Due to the design of the circuitry, failure of a pressure switch will not result in any new failure modes at the system level.

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.

23. **PDCR 94-021 Appendix J Improvements** was completed 4/28/95.

General Summary
The vacuum breaker check valves on High Pressure Coolant Injection (HPCI) and Reactor Core Isolation Cooling (RCIC) steam exhaust lines inside the Torus (X-221 and X-212) were removed and replaced with welded caps. This eliminated the need for 10CFR50, Appendix J, Type B leak rate testing of the vacuum breakers, and reduced the total number of containment atmosphere leakage paths. The vacuum breaker check valves, which were removed, performed no function; they had been replaced with vacuum breaker lines outside the Torus under EDCR 73-32.

The blind flange at penetration X-35D, Traversing Incore Probe (TIP) Spare, was replaced with a testable flange containing double O-rings. This made it possible to perform 10CFR50, Appendix J, Type B leak rate testing of penetration X-35D.

Double isolation valve test connections were added at penetrations X-40B-B and X-40D-B. The safety classification boundary of Post-Accident Sampling (PAS) system lines were moved to the second valve capable of automatic closure or administratively controlled in the closed position. This modification changed the function of valves PAS-101, 102, 103, 104, 108C and 109C to containment isolation valves, and provided a means to test the valves in accordance with 10CFR50, Appendix J. Valves PAS-114 and 115 were replaced with new Safety Class 2 valves.

The line from the Instrument Air (IA) system to the Drywell was cut and capped at penetration X-26. A section of the line including containment isolation valves (V16-19-51, 52) and associated test connection valves was removed, and the remaining pipe ends were capped. Flow indicator FI-1-158-4 was retired in place. This modification was based on the assumption that FI-1-158-4 and the line containing V16-19-34 perform no function; this assumption was reached after a review of past performances of 10CFR50, Appendix J Type A ILRT, and a search of all Vermont Yankee procedures, showed that FI-1-158-4 was no longer being used.

Permanent junction boxes were installed to connect temporary instruments and computer for performance of 10CFR50, Appendix J, Type A Integrated Leak Rate Test (ILRT). The new junction boxes are located inside the Drywell and Torus and in the reactor building. Plant instruments previously used for this function were retired in place or removed to allow their electrical penetrations through primary containment (X-105C and X-214) to be used for connecting the new junction boxes. Plant process computer systems (CVAX and ERFIS) programs that use inputs from the instruments being retired in place or removed were revised. This design allows state of the art instrumentation and computers to be used for ILRTs for the remaining life of the plant.

An isolation valve was installed between steam trap ST-60-3 and V60-20. This eliminates the need to remove steam trap ST-60-3 and install a blind flange every time 10CFR50 Appendix J local leak rate testing is performed on the main steam isolation valves.

Safety Evaluation Summary

This modification satisfied the design criteria; and inclusion of the Post Accident Sampling (PAS) containment isolation valves in the Appendix J and In-Service Testing (IST) programs ensured the valves do not degrade beyond acceptable limits. The affected portion of the Instrument Air (IA) system is neither an initiator nor a mitigator of any postulated accidents evaluated in the FSAR. Any failure is bounded by FSAR Section 14.6.5, "Main Steam Line Break Accident". Integrity of primary containment penetrations X-105C and X-214 was not affected by this modification, although cables were terminated in their junction boxes. Primary containment was not adversely affected by any part of this modification. No new failure mechanisms were introduced by this design change.

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.

24. **PDCR 94-022 House Heating Boilers Fuel Oil Storage System** was completed 1/15/95.

General Summary
This design change replaced the 5,000 gallon underground fuel oil day tank with a 12,000 gallon above ground storage tank to serve the House Heating Boilers. Until this design change, the 75,000 gallon oil storage tank had been utilized to serve both the House Heating Boilers and the Station Diesel Generators. This change allowed the 75,000 gallon tank to serve the diesel generator exclusively, with the new 12,000 tank handling the supply for the house heating boilers.

The existing supply and return lines to the house heating boilers were connected to the new storage tank rather than the day tank. The existing supply line from the 75,000 gallon diesel generator fuel oil storage tank that was previously used to fill the day tank, was routed to the new storage tank to be used to provide a backup source of fuel. The existing control loop which filled the day tank from the 75,000 gallon tank was modified so that transfer from the 75,000 gallon tank to the new 12,000 gallon tank must be manually initiated under administrative control. A local fill connection for the 12,000 tank was installed, as were the level instrumentation and controls.

Safety Evaluation Summary
This change involved non-safety related components which have no direct ties to any safety related system. Failure of these components will not result in a challenge to a safety system, nor are the components initiators of any accidents evaluated in the FSAR. The fuel oil storage tank does not form or directly connect to the primary barriers (fuel cladding, primary pressure boundary or containment) which limit the consequences of an accident. This design change does not impact the operation of Station Emergency Diesel Generators. This change does not alter the function or performance of any safety system. There were no changes to the emergency diesel generator fuel supply system that would affect any technical specification safety margins.

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.

25. **Temporary Modification 93-058** was installed 11/18/93 and removed 3/31/95.

General Summary
This Temporary Modification replaced Safety Class 3 valve V70-512A, which was flanged to the Reactor Building Closed Cooling Water (RBCCW) heat exchanger discharge, with the flange and a plug assembly previously installed on the outlet side of the valve. After removal, V70-512A replaced valve V70-92D, which was severely corroded. Valve V70-512A was the only Safety Class 3 valve available at the time for replacement.

Safety Evaluation Summary
The alternate service water (SW) outlet from the RBCCW heat exchanger is a passive pressure boundary in the SW system. The flange assembly now functions as a passive pressure boundary, just as valve 512A had since it was installed. This boundary modification does not affect the radiological boundary evaluated in the FSAR, and does not affect system integrity or design heat loads or flow rates.

As this connection is a passive pressure boundary, the only possible malfunction is pressure boundary failure. Section 10.6.6 of the FSAR discusses a SW pipe failure. In the unlikely event that either one of the flange assemblies failed during operation, the leak could be isolated and the second 100% capacity RBCCW heat exchanger could be used to carry the entire cooling load. The consequences of a 12 inch SW line break (such as in the inlet or outlet to the heat exchanger) far exceed the consequences of a break in this 4 inch line. Based on these facts, the consequences of a malfunction that was previously analyzed in the FSAR are not increased.

There was 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 was not endangered.

26. **Temporary Modification 93-059** was installed 11/18/93 and removed 1/25/95.

General Summary
This Temporary Modification replaced Safety Class 3 valve V70-512B, which was flanged to the Reactor Building Closed Cooling Water (RBCCW) heat exchanger discharge, with the flange and plug assembly previously installed on the outlet side of the valve. After removal, V70-512B replaced valve V7(-92C, which was severely corroded. Valve V70-512B was the only Safety Class 3 valve available at the time

for replacement.

Safety Evaluation Summary
The alternate service water (SW) outlet from the RBCCW heat exchanger is a passive pressure boundary in the SW system. The flange assembly now functions as a passive pressure boundary, just as valve 512B had since it was installed. This boundary modification does not affect the radiological boundary evaluated in the FSAR and does not affect system integrity or design heat loads or flow rates.

As this connection is a passive pressure boundary, the only possible malfunction is pressure boundary failure. Section 10.6.6 of the FSAR discusses a SW pipe failure. In the unlikely event that either one of the flange assemblies failed during operation, the leak could be isolated and the second 100% capacity RBCCW heat exchanger could be used to carry the entire cooling load. The consequences of a 12 inch SW line break (such as in the inlet or outlet to the heat exchanger) far exceed the consequences of a break in this 4 inch line. Based on these two facts, the consequences of a malfunction that was previously analyzed in the FSAR are not increased.

There was 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 was not endangered.

27. **Temporary Modification 93-067** was installed 11/23/93 and removed 11/30/93.

General Summary
This temporary modification installed a freeze seal downstream of valve V70-111A. V70-111A is the outlet isolation valve for the Reactor building Closed Cooling Water (RBCCW) supply to Fuel Pool Cooling (FPC) Heat Exchanger (HX) E-9-1A. The valve had a broken stem, and no other method of isolating this valve was available.

Safety Evaluation Summary
The isolation of V70-111A and E-9-1A did not increase heat loads on the RBCCW system, change design flow rates or degrade the radiological boundary between the fuel pool cooling system and service water system. The Standby FPC System in conjunction with the "B" FPC HX maintained fuel pool temperature within required specifications. Therefore, the ability of the systems affected by this modification to mitigate the consequences of an accident was not degraded.

There was 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 was not endangered.

28. **Temporary Modification 94-003** was installed 2/24/94 and is currently in place.

General Summary
Diesel fire pumps in standby applications, which are required to fast start immediately and come up to full speed with a load (fire pump) on the engine, need to have lube oil and coolant temperatures maintained at an elevated level. This modification installed a tank heater and lube oil heater on the diesel fire pump. The tank heater maintains the jacket coolant temperature of the engine between 100 degrees F and 120 degrees F. The lube oil heater was added to the lube oil sump to maintain temperature between 100 degrees F and 120 degrees F. This modification ensures that the jacket coolant temperature and lube oil temperature remain at acceptable levels for standby operations.

Safety Evaluation Summary
This modification does not interface with any safety related systems. The jacket and lube oil heaters only interact with the diesel fire pump engine. The interaction of this pump with the Fire Protection System is no different than previously evaluated. This Temporary Modification supports the bases of the fire protection system by improving the reliability of the diesel fire pump during cold weather conditions.

There was 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 was not endangered.

29. **Temporary Modification 94-004** was installed 3/25/94 and removed 2/14/95.

General Summary
This temporary modification installed two pressure gages in the service water (SW) system to monitor SW pressure during diesel generator surveillance testing. One gage was installed in the Non Nuclear Safety (NNS) portion of the SW system in the alternate cooling discharge cooling line. The other gage was installed on the diesel generator cooling water outlet at a point specifically designed for a pressure test connection which has a Safety Class 3 isolation valve. The connection downstream of the isolation valve is classified as NNS. Installation of the NNS pressure gage at this test point is within the design bases of the service water system. Failure of either

gage has no adverse effect on the capability of equipment important to safety.

Safety Evaluation Summary
This modification did not affect the functioning of the service water system or alternate cooling system. FSAR section 1.5 requires the design of the service water system to include allowances for environmental phenomena at the site. The addition of these instrument valves in accordance with this temporary modification does not adversely affect the ability of the piping to withstand a seismic event or any other environmental phenomena.

There was 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 was not endangered.

30. **Temporary Modification 94-005** was installed 4/20/94 and removed 4/24/94.

General Summary
To investigate the cause of a control circuit power loss, this temporary modification installed an electrical jumper to provide the ability to open the outer reactor building equipment door (railroad entrance). The door OPEN logic circuitry does not function normally for the outer door because there is no power available to the control circuitry for the inner door; the two doors' logic circuitry is interlocked such that one door cannot be opened unless the other door is closed and the seal inflated. The temporary jumper in the outer door OPEN logic bypassed the interlock. Administrative controls verified that the seal was intact prior to installation of this temporary modification, and eliminated the chance of the inner seal deflating while the repair activity was being performed.

Safety Evaluation Summary
Installation of the jumper in the outer railroad airlock OPEN circuit did not reduce the ability of the equipment airlock system to maintain secondary containment integrity. The inner door remained closed and the seal inflated during the repair. The existing failure mode or possible malfunction of the railroad airlock access doors is to open simultaneously when they would be required to provide secondary containment integrity. The installation of this temporary modification to the airlock door circuitry did not change this failure mode or create a new one that would increase the radiological consequences of an accident.

There was 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 was not endangered.

31. **Temporary Modification 94-006** was installed 5/9/94 and removed 4/22/95.

General Summary

This temporary modification provided a short term resolution to NRC Inspection Report 92-81 item 04, which identified a concern with the design of the existing degraded grid undervoltage protection system and setpoints of the Division II (Bus 4) Degraded Grid Undervoltage relays. During maximum plant loading (Emergency Core Cooling System, and Cooling Tower Operation), with the plant supplied from the Startup Transformers and the minimum grid voltage available, the voltage on Safety Class Electrical (SCE) Buses 3 and 4 could have dropped below the degraded grid relay setpoints, due to transient voltage dips, on pump starts.

The voltage would have recovered to a level sufficient to operate all SCE loads and reset the Degraded Grid relays on Division I, but might not have been high enough to reset the Division II relays. As a result, approximately ten seconds into the event, the Division II emergency loads would have been disconnected from the "preferred power supply" and a Service Water and RHR pump would have been immediately connected to the "A" Diesel Generator. Since the swap of Bus 4 from the offsite power source to the diesel generator would occur in only a few cycles, an out-of-phase breaker closure could have resulted, since residual voltage would still be present on the bus.

Temporary Modification 94-006 removed non-essential loads from Startup Transformer T-3-1B, during accident conditions which results in ECCS initiation, to ensure the voltage on Bus 4 would recover to a sufficient level to prevent inadvertent separation of the SII Division from the "Preferred Power Source" in the event an accident would have been coincident with the minimum expected grid voltage at the Vermont Yankee 115KV Switchyard. A spare set of Normally Open relay contacts from "A" Residual Heat Removal Control Logic Relays 10A-K9A and 10A-K70A were wired in parallel with the control logic trip circuit for 4KV BKR 53. Actuation of either of these relays would have tripped BKR 53 which effectively sheds cooling tower loads from the T-3-1B Startup Transformer in the event a Drywell High Pressure or Reactor Lo-Lo Level accident signal had been received.

The long term solution was provided in EDCR 94-403, which installed higher accuracy relays equipped with a significantly smaller reset voltage bandwidth which prevents inadvertent separation and improves accuracy of the instrument loop, and eliminates the potential for setpoint drift; thus allowing this Temporary Modification to be removed.

Safety Evaluation Summary
This temporary modification ensured that adequate voltage would be available at Emergency Bus 4 to prevent inadvertent separation of the SII Division from the "Preferred Power Source" in the event of an accident signal coincident with the minimum expected grid voltage at the Vermont Yankee 115 KV Switchyard. The only safety class equipment that this temporary modification interfaced with were two control logic relays in the "A" Residual Heat Removal subsystem. The performance of the associated FSAR 1.6.2.11 "Core Standby Cooling Systems" equipment would not be impacted because the separation of the Non Nuclear Safety BKR 53 trip circuit from the safety class electrical circuits connected to these relays was reviewed and found to be acceptable.

There was 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 was not endangered.

32. **Temporary Modification 94-009** was installed 5/16/94 and removed 6/10/94.

General Summary
This temporary modification installed a remote bulb/ capillary tube thermometer at the West Cooling Tower, Cell. No. 1, to permit temperature monitoring of the Deep Basin water to verify that the temperature remains below the operability limit as described in Basis for Maintaining Operation (BMO) 94-02.

Safety Evaluation Summary
This modification did not affect the operability of the Alternate Cooling system. The local temperature monitor performed a passive function in relation to the operation of this system. The probable failure mode associated with this modification was the separation of the copper-sheathed capillary tube together with the stainless steel encased bulb. The weight of this assembly was such that, coupled with the location of the temperature sensor approximately 30' horizontally from the Alternate Cooling system's suction located in the deep basin, make it extremely unlikely that it would enter the suction piping. The velocity in the vicinity is extremely low (<1m/sec) and is judged to be insufficient to cause this assembly to be swept into the suction piping. Therefore, there was reasonable justification to conclude that the nuclear safety/ engineered safeguard systems cooled by the Alternate Cooling system would not be impacted.

There was 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 was not endangered.

33. **Temporary Modification 94-010** was installed 5/13/94 and removed 10/31/94.

General Summary

This temporary modification defeated the interlocks that prevent movement of the Refuel Bridge over the core when the Reactor Mode Switch (RMS) is in a position other than "Refuel". Moving the Refuel Bridge over the core allowed for inspections and assessment of the Refuel Bridge drive system and track (rail) condition, which was necessary to support design development of a replacement bridge drive system. This temporary modification also aided Plant Housekeeping requirements by significantly reducing the potential for tools or materials to be dropped into the Spent Fuel Pool during the inspection.

Safety Evaluation Summary

During plant operation with the reactor critical and the mode switch in RUN, the refuel interlocks are not required to perform or prevent any system response, nor do they perform a mitigating function for any accidents evaluated in the FSAR. This temporary modification did not result in the addition of any new equipment. Defeating the refuel interlocks in a plant configuration for which they are not required could not result in an accident or transient which has not already been analyzed. There were no new failure modes for refueling interlocks introduced by this temporary modification. The failure mode for the limit switches would be to reinstate the refueling interlocks which prevent movement of the refueling bridge (when the mode switch is not in refuel) and also initiate a Rod Withdraw Block.

There was 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 was not endangered.

34. **Temporary Modification 94-012** was installed 5/24/94 and removed 3/19/95.

General Summary

This temporary modification connected a Heise gauge, vent container with isolation valve and stainless steel tubing and fittings to pressure switch PS-2-71C via the normally closed PS-2-71C test valve. This provided the ability to read the pressure on the PS-2-71C sensing line and also to vent pressure from the sensing line, as addressed in the Basis for Maintaining Operation (BMO) 94-03. The BMO addressed a small leak in the Safety Relief Valve RV-2-71C valve bellows. During the 1995 refueling outage the bellows was repaired and the leak was eliminated.

Safety Evaluation Summary

The safety function of PS-2-71C and the associated tubing, fittings, and valves is to maintain primary containment integrity following an accident. This temporary modification was attached to the Non Nuclear Safety side of the PS-2-71C test connection and was seismically isolated from the test connection. This ensured that there would be no effect on the normal operation of PS-2-71C while the test valve was closed and left unattended. The PS-2-71C test valve was opened in conjunction with the isolation valve only when either an Operator or Technician was stationed at the valve. The installation and operation of this temporary modification did not affect the ability of PS-2-71C to perform its intended safety functions of passive indication of SRV bellows pressure and primary containment.
There was 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 was not endangered.

35. **Temporary Modification 94-014** was installed 6/10/94 and removed 3/29/95.

General Summary

This temporary modification removed the improper "31" position from Rod 06-15. This measure was taken to allow for clarity for all other rod positions for Rod 06-15. This modification only affected the Rod Position Information System (RPIS) acknowledgement and display of intermediate positions 01, 11, 21, 31, and 41 for Rod 06-15. This temporary modification allowed the rod drift alarm indication and the annunciator to be reset so as not to mask a valid rod drift if received on any rod, including Rod 06-15.

Safety Evaluation Summary

This temporary modification did not affect the Scram Insertion Times, nor did it reduce the difference between a system failure point and accepted safety limit. Regarding Scram Insertion Times, the only ERFIS feature that may have been lost by installation of the temporary modification would have been the ability to obtain an insertion time for Rod 06-15 during ERFIS full scram data collection. Although this feature may have been lost for this one rod, single rod scram time testing via the two pen recorder would have remained available.

There was 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 was

not endangered.

36. **Temporary Modification 94-017** was installed 07/20/94 and removed 03/31/95.

General Summary
Temporary Modification 94-017 opened bypass switches for a channel of the Bentley-Nevada vibration monitoring system for the recirc pumps and motors. This was done to bypass the input from failed components in order to preclude masking the remaining recirc pump and motor vibration alarms.

Safety Evaluation Summary
The recirc pump shaft vibration monitoring system and related alarms are not used to protect any radioactive material barrier, nor do they potentially impact nuclear safety or engineered safeguard systems. The change involved bypassing a failed vibration channel and deenergizing the failed components. This change had no potential to impact the primary systems or any Safety Related Equipment. Bypass and deenergization of the failed channel did not create the possibility of an accident, but enabled the operable channels of equipment to provide alarms.

There was 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 was not endangered.

37. **Temporary Modification 94-018** was installed 7/26/94 and is still open.

General Summary
During Emergency Diesel Generator (EDG) operation, lube oil accumulates in the upper piston skirts. It is necessary to air roll a diesel after operation to remove the lube oil to prevent it from seeping past the piston rings and into the cylinders, causing excess smoke and potential exhaust manifold fires. To allow air rolling of the EDG after operation, this temporary modification installed a ball valve in the supply tubing for the #14 bearing oil booster and the Woodward Governor oil booster.

Safety Evaluation Summary
This modification provided the ability to roll over an EDG after operation. It does not affect how the EDG operates after starting. It does prevent the diesel from responding to an automatic start demand while the engine is being air rolled. The length of time the diesel cannot respond to an automatic start signal is 1 to 3 minutes during the performance of air rolling. This modification increases the chance of a

malfunction, but such a malfunction would not initiate an abnormal operating transient as described in FSAR 14.5. This temporary modification does not affect the ability of the EDG to automatically respond to a start demand when the fuel rack is tripped. Considering the reduction in the probability of a diesel engine exhaust fire and the improved reliability of the diesel, the air rolling modification is considered to have a net safety benefit.

There was 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 was not endangered.

38. **Temporary Modification 94-019** was installed 7/27/94 and is still open.

General Summary and Safety Evaluation Summary
This temporary modification is identical to Temporary Modification 94-018 described above, except that 94-019 installed two ball valves on the "B" EDG in the Starting Air System (Temporary Modification 94-018 dealt with the "A" EDG).

39. **Temporary Modification 94-021** was installed 8/12/94 and removed 10/29/94.

General Summary
This temporary modification opened all six louvers in the diesel penthouse by opening circuit #12 on LP-1AJ, the power feed to the solenoid supply valves. This was done to maintain operability of the "B" Diesel Generator by supplying air to the "B" Diesel Generator Room. The operability of the diesel generator with the louvers open is only permitted during warm weather conditions; the louvers were redesigned and functional prior to cold weather operation.

Safety Evaluation Summary
This modification supported the bases of the diesel generator by ensuring adequate ventilation was available in the event the diesel generator had been required to perform its intended safety function. This modification did not directly or indirectly increase the radioactive material release from the four accidents as evaluated in FSAR section 14.6 because these louvers were already in their accident response position. Had there been an accident in which a diesel generator (DG) start had been initiated, the louvers were positioned to allow the design air flow requirements on a DG initiation signal. The open louvers would not have become an unmonitored release path because the normal Turbine building ventilation system ensures air inflow through the louvers at all times.

There was 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 was not endangered.

40. **<u>Temporary Modification 94-023</u>** was installed 9/2/94 and removed 4/4/95.

<u>General Summary</u>
To allow operation of the single rod scram switches during single rod scram testing without the concern of overshooting, this temporary modification removed the neutral return for the Reactor Protection System (RPS) select bus in Control Room Panel CRP 9-16.

<u>Safety Evaluation Summary</u>
The RPS performs a mitigating function of accidents described in the FSAR. This modification could not directly or indirectly increase the radioactive material release from the accidents because the disconnection of the Neutral return line from the Select Rod Insert Bus did not impact the Safety Objective or Safety Design Bases of the RPS. Also, implementation of this modification did not result in any change in equipment or sensor setpoints. The only change in equipment operation was that when an individual rod scram switch is moved to the full down position, the rod would scram due to the disconnection of the respective 117 and 118 scram solenoids from the RPS Neutral Bus. This action is consistent with the design bases of the system.

There was 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 was not endangered.

41. **<u>Temporary Modification 94-024</u>** was installed 10/4/94 and removed 10/5/94.

<u>General Summary</u>
This temporary modification installed a bypass around the Vital Bus transfer switch alternate supply contacts to allow the safe removal of the vital bus transfer switch for replacement while maintaining a reliable source of power to the vital bus. The bypass was installed in Control Room Panel CRP 9-45 from the alternate supply leads of the transfer switch to the 100 A Circuit Breaker #14 on the Vital Distribution Panel.

Safety Evaluation Summary
Initiation of the Emergency Core Cooling System (ECCS) systems which are required to function to mitigate the consequences of an accident, are designed to be initiated from DC power, independent of Vital AC. All potential failures of Vital AC were evaluated and would not have degraded the performance of required mitigation systems. Operation of loads fed from the Vital AC bus did not change due to the use of the maintenance tie. While operating in this configuration, all instrumentation required for Engineered Safety Factor system operability remained fully functional.

There was 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 was not endangered.

42. **Temporary Modification 94-027** was installed 10/15/94 and removed 4/27/95.

General Summary
This modification removed valve V70-92C from the 4" service water (SW) line and installed two 4" 150# blank raised face flanges. Valve V70-92C had been stuck in the open position and would not close. A replacement valve was not available. The blank flanges provided isolation and allowed for repairs to the heat exchanger. A loop seal assured that secondary containment integrity was preserved while the SW system was opened to the Reactor Building atmosphere, during installation of the modification.

Safety Evaluation Summary
Two safety evaluations were written in support of this temporary modification: one addressing the installation of flanges to isolate the "B" RBCCW heat exchanger and to operate with one RBCCW heat exchanger out of service; and one to assess the operational configuration needed to support installation of the temporary modification.

The replacement of normally closed valve SW-92C with two blind flanges is functionally equivalent; either configuration prevents flow. The structural and pressure retaining integrity of the system was unaffected. Since both configurations are equivalent, the modification had no effect on any plant systems or equipment which prevent or mitigate failures of radioactive material barriers. Plant operation with the redundant heat exchanger E-8-1B valved out could not create any different accident than when the redundant heat exchanger is in service.

This modification was performed during a hot shutdown condition during which no fuel was moved. The Residual Heat Removal (RHR) Heat Exchangers (Hx) are used for containment cooling which helps maintain containment integrity during a Loss of Coolant Accident. Since the blind flanges were installed downstream of the discharge throttle valves of the "A" RBCCW Hx and therefore on the common discharge header for the entire Service Water System, there would have been little, if any, impact on the flow distribution to the remaining essential equipment if the flange had not been installed.

The heat removal capability of the Service Water System was not affected by this temporary modification since the installation of the blind flanges was on the discharge side of the components. If a problem had occurred during the installation of this temporary modification, cooling water could still have been supplied to the safety related loads.

There was 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 was not endangered.

43. **Temporary Modification 94-028** was installed 10/16/94 and removed 4/29/95.

General Summary

This modification removed the eight test solenoid valves for the extraction steam reverse current valves (RCVs). This was done following the discovery that the reverse current valves required eight minutes to close, as opposed to the expected two seconds. The test solenoid valves were not capable of passing sufficient air to rapidly vent the reverse current valve actuator to allow the spring assist to close. Removing the test solenoids allowed the turbine oil trip valve to properly vent the reverse current valve operators when required.

This modification installed flexible hose in place of the test solenoid valves. A single manual vent valve was installed to allow all current valve operators to be vented off and tested. A manual valve was provided upstream of the turbine oil trip valve to allow all test functions to be performed manually and locally in the lube oil room, without entry into the heater bay.

Safety Evaluation Summary

The only items potentially impacted by this temporary modification were the reactor moderator temperature decreases and reactor vessel moderator pressure increases. FSAR section 14.5.2.1 indicates that loss of a feedwater heater can occur as the result of shutting an extraction steam line or bypassing a heater or string of heaters. Testing of the RCV's was accomplished while above 95% power; at this power level there was sufficient steam flow, in the individual extraction steam lines, to ensure the

RCV's spring pressure would be overcome and that adequate steam would continue to be supplied to the individual heaters for the short duration of the RCV tests.

There was 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 was not endangered.

44. **Temporary Modification 94-029** was installed 11/17/94 and removed 4/2/95.

General Summary
To perform leak rate testing on core spray valve CS-13A, this temporary modification seismically attached the loop seal section of the test rig to the 8" CS-3A header. The seismic mounting of the test rig tubing ensured the integrity of the Core Spray CS-24A/25A drain valve connection during the leak rate test.

Safety Evaluation Summary
The Core Spray System performs a mitigating function of accidents described in the FSAR. The Core Spray A Subsystem is connected to primary containment penetration X-16B and is required to provide isolation following a Loss of Coolant Accident. Performance of the CS-13A Leak Rate Surveillance Test did not affect the ability of the Core Spray discharge MOV's, CS-11A or CS-12A, to be operated for the Core Spray function or for containment isolation function.

There was 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 was not endangered.

45. **Temporary Modification 94-031** was installed 1/4/95 and is still open.

General Summary
This temporary modification installed a fire retardant heatable fabric shelter with an external electric heater bank and the necessary ducting and appurtenances to provide a warm environment around the diesel oil storage tank. This modification provides the capability to heat the air surrounding the tank to help maintain oil temperature above Technical Specifications limits.

Safety Evaluation Summary
This temporary modification enhances the operation of the Fuel Oil Tank by ensuring

that the ambient temperature of the oil is maintained above the pour point temperature under all weather conditions. Fire prevention requirements such as control of combustibles near the heaters, exclusion zones from fire control areas, etc., have been incorporated into the system materials and layout.
Failure of the modification's ability to heat the oil in the tank would result in a very slow temperature decrease due to the heat capacity of the large mass of oil. Operator action would be taken to mitigate the consequences of loss of diesel operability if the oil were to cool below the Technical Specification requirements for the pour point. No potential failure of this modification will increase the probability that the diesels will be unavailable to mitigate consequences of accidents 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 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.

46. **Temporary Modification 95-002** Reactor Recirculation Unit (RRU) 5 modification was installed and removed 2/10/95; RRU 7 modification was installed and removed 2/13/95; RRU's 6 and 8 modifications were installed and removed 2/27/95.

General Summary

Service water flow through the RRU coils deposited silt and bio-fouling within the RRU, which increased the service water differential pressure across the RRU coils. The differential pressure increase indicated that the heat transfer surfaces were becoming fouled and in need of cleaning. This temporary modification connected station service air to the inlet piping of RRU 5, 6, 7 and 8 for cleaning. The introduction of air into the RRU coils created a very turbulent flow and helped remove foreign material within the RRU coils. The RRUs were considered operable during the cleaning.

Safety Evaluation Summary

The injected air would not have caused air binding of the applicable RRU or any other Service Water (SW) equipment, nor result in air pockets which could cause a water hammer. The air was entrained in the flow stream and would not have migrated upstream of the RRU to the supply header or other equipment. The pressure integrity of the SW system was not compromised. Although a Non Nuclear Safety (NNS) air source was installed, the Safety Class 3 isolation valve could have been closed rapidly if conditions had warranted. The air supply was regulated to less than or equal to 125 psig, the pressure rating of the SW system. This ensured that overpressurization would not occur.

There was 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 was not endangered.

47. **Temporary Modification 95-003** was installed 3/1/95 and is still open.

General Summary
This temporary modification installed a temperature monitoring system to determine changes in DG storage tank bulk temperature during the winter season; this was done to address NRC concerns regarding adequacy of the diesel generator (DG) fuel oil storage system. A thermocouple was installed through the 3" pipe cap at the access hatch at the top of the tank to record bulk tank oil temperature. The thermocouple was located approximately four feet below minimum oil level. An additional thermocouple was fixed to the outside of the tank wall within the pump house to record tank skin temperature. A third thermocouple was installed to monitor air space temperature within the pit. All three thermocouples were connected to a recorder in the fuel oil pump house.

Thermocouples are bi-metallic devices that internally generate a voltage signal proportional to the sensed temperature from the dissimilar metal junction at the tip of the thermocouple assembly. As such, there is no possible source of ignition that is in direct contact with the fuel oil as a result of this installation.

Safety Evaluation Summary
This modification does not affect the operation of the diesel fuel oil system or any other plant system. Technical Specification bases required a minimum quantity of 25,000 gallons of fuel oil, that meets specific quality requirements, be available in the fuel oil storage tank. The installation of temperature monitoring does not impact the quantity or quality of the fuel oil being stored. The monitoring system helps assure that the fuel oil meets the Technical Specifications requirements.

There was 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 was not endangered.

48. **Temporary Modification 95-006** was installed 3/21/95 and removed 4/21/95.

General Summary
This temporary modification provided a temporary path for the Service Water

discharge from the Reactor Building Closed Cooling Water (RBCCW) Heat Exchanger E-8-1A and E-8-1B to the roof drain at elevation 303'0" on the west side of the Reactor Building. This allowed work to be performed on common discharge line SW-12 and allowed replacement of the SW-92 valves while maintaining the RBCCW operational; this in turn ensured that normal fuel pool cooling remained operational through the RBCCW/Fuel Pool Cooling Heat Exchanger.

Safety Evaluation Summary
During the time this temporary modification was installed, the reactor was in shutdown/refuel. This modification did not bypass Secondary Containment during refuel activities due to the loop seal installed where the modification tied into the roof drain. Therefore, the Standby Gas Treatment (SBGT) system remained operable and stack release was not impacted. RBCCW remained available to cool the components in use; RBCCW heat loads were at a minimum.

There was 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 was not endangered.

49. **Temporary Modification 95-019** was installed 3/16/95 and removed 6/1/95.

General Summary
In response to new Condensate Storage Tank (CST) requirements which disallow the discharge of water through the hotwell level control system to the CST during outages, this temporary modification provided an alternate flow path. The internals and actuator of directional control valve V20-342 were removed, which allows the valve to act as a "tee" allowing flow from the Reactor Water Cleanup System (RWCU) directly to the waste sample tanks instead of sending the water to the hotwell. Water from the RWCU is diverted around the radwaste process system with a valve installed in the inlet of the waste surge tank TK-11-1A. This modification allows the filtration and demineralization process of the radwaste system to be bypassed; however, there is no threat of releasing unacceptable water to the CST because a Chemistry sample of the water is required to generate a permit to release water from the sample tanks to the CST.

Safety Evaluation Summary
This modification does not affect the probability of a malfunction which initiates any of the transients listed in the FSAR. It does not increase the radiological consequences above that analyzed in the FSAR. The liquid radwaste system cannot fail in a manner which will affect or initiate any transient, and is not used to mitigate the radiological effects of those transients.

There was 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 was not endangered.

50. **Temporary Modification 95-024** was installed 3/28/95 and removed 4/9/95.

General Summary
This modification defeated the interlock preventing the movement of the refueling bridge in the reverse (north) direction over the core when the reactor mode switch was not in refuel. This allowed performance of in-vessel inspections, shroud inspections and feedwater nozzle inspections, as well as LPRM and dry-tube replacement with the mode switch in shutdown,

Safety Evaluation Summary
Failure of the refueling interlocks is not an initiator for any accidents. The interlocks do not mitigate any of the accidents, nor do they make up any part of the barriers to radioactive release. No fuel movement was permitted during the implementation of this modification. A caution tag was placed on the mode switch to keep the mode switch in SHUTDOWN for the duration of this temporary modification. With the mode switch in SHUTDOWN, it is impossible to withdraw a control rod. Control blade removal from the top requires the mode switch to be in REFUEL and a Senior Reactor Operator to be present. In addition, the design of the control rod, incorporating the velocity limiter, does not physically permit the upward removal of the control rod without simultaneous or prior removal of the four adjacent fuel bundles or blade guide, thus eliminating any hazardous condition.

There was 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 was not endangered.

51. **Temporary Modification 95-029** was installed 4/11/95 and removed 4/15/95.

General Summary
This modification installed a freeze seal on the 1" PSE jet pump high pressure instrument line in order to provide adequate isolation for maintenance on leaking valve V2-3-20B. The function of this line during normal operation is to provide an input to jet pump #1 monitoring instrumentation. The function of this line during shutdown/refueling is to provide a boundary to maintain coolant inventory in the

reactor vessel. Implementation of this temporary modification changed the method of coolant containment from a mechanical valve to an ice plug.

Safety Evaluation Summary
The line being modified does not play a part in the mitigation of any of the malfunctions evaluated in the FSAR. Any leakage from this line, if the freeze seal had totally failed, would have been minor and could not have threatened nuclear safety or released significant quantities of radioactive material. Implementation of this temporary modification did not reduce the margin of safety, failure point, or accepted safety limit of the jet pumps or associated jet pump instrumentation.

There was 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 was not endangered.

52. **Temporary Modification 95-031** was installed 4/19/95 and removed 4/20/95.

General Summary
This modification installed monitoring instrumentation to determine transient performance of the emergency diesel generators during the Integrated Emergency Core Cooling System (ECCS) test. This modification was implemented in response to questions raised during the NRC Emergency Diesel Generator Systematic Functional Inspection (EDSFI) over differences between the strip chart recordings of diesel generator voltage from the integrated ECCS test and the predicted voltages contained in the analytical model in the diesel generator loading calculation.

Safety Evaluation Summary
This temporary modification did not affect operability of the diesel generators during the integrated ECCS test, when the plant was in cold shutdown. During the test the diesel generators were considered operable, but the accident mitigation systems that the diesel generator supports were not required to be operable, and the accidents described in FSAR Chapter 14 were not postulated to occur. The diesel generators' safety function was not jeopardized by the implementation of this modification.

There was 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 was not endangered.

53. **Special Test Procedure 94-001 Turbine Performance Testing** was completed 06/29/94.

General Summary
Special Test Procedure 94-001 was performed prior to installation of the new low pressure turbines. This test was established the corrected turbine/generator performance/capacity, and was a "Dry Run" test for the final performance/capacity test that was performed after the 1995 Low Pressure Turbine Replacement Project.

Safety Evaluation Summary
This test, for the collection of data, was conducted with the plant operating at 100% power. Temporary instrumentation consisted of a multi-port pressure measuring device connected to existing plant pressure test points with vacuum rated instrument turbine. In the event of a tubing or instrument leak, all leakage would have been into the condenser and would have been tracked by monitoring condenser in-leakage.

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

54. **Special Test Procedure 94-002 Hydraulic/Thermal Performance Testing of ACS-Cell #1** was completed 11/10/94.

General Summary
Special Test Procedure 94-002 tested the performance capability of the Alternate Cooling Tower Cell (Cell #1 or CT-2-1) using the normal heat load on the Service Water (SW) System. This test used the normal SW system alignment except that the SW discharge was via the Alternate Cooling cell inlet valve (SW-17) and the normal SW discharge to the circulating water system (SW-1).

Safety Evaluation Summary
The analyses performed in support of this test ensured adequate service water flow to each diesel generator in all operating conditions, including postulated design basis accident conditions. This provided additional assurance that cooling water flow would not prevent the diesel generators from supplying 100% of the emergency loads required under the postulated design basis accident conditions. The test did not affect the readiness of the diesel generators.

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

55. **Special Test Procedure 95-002 Hydraulic Performance Testing of the Alternate Cooling System** was completed 4/5/95.

General Summary

Special Test Procedure 95-002 tested the hydraulic performance capability of the Alternate Cooling Tower Cell (CT-2-1 or Cell #1) during the 1995 refueling outage. One Residual Heat Removal Service Water (RHRSW) loop was used to demonstrate RHRSW pump hydraulic and suction capability at maximum loop flow in the Alternate Cooling Mode. This test also demonstrated that a total loop flow of approximately 4500 gpm with an Residual Heat Removal Heat Exchanger cooling water flow of approximately 3400 gpm is achievable and that an RHRSW pump can operate at up to 3000 gpm without cavitation concerns.

Safety Evaluation Summary

This test was performed when the reactor was in cold shutdown. The non-associated service water (SW) "B" loop was operating in the normal lineup to provide required plant cooling water for shutdown cooling, Reactor Building Closed Cooling Water cooling and standby fuel pool cooling. Having one SW loop operable provided for decay heat removal at a positive RHRSW/RHR pressure differential and provided assurance that "B" Emergency Diesel Generator would automatically receive required cooling water if a Loss of Normal Power event had occurred during testing.

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

56. **Special Test Procedure 95-003 In-Situ Differential Pressure Testing of Valves RHR-34A and RHR-39A** was completed 3/16/95.

General Summary

NRC Generic Letter 89-10, "Safety-Related Motor Operated Valve (MOV) Testing and Surveillance" requested licensees to develop a comprehensive program to ensure that safety-related MOVs will function when subjected to the conditions expected during both normal plant operations and design basis events. The Vermont Yankee Motor Operated Valve Program Plan established design basis review, testing, inspection, and maintenance requirements for certain Safety Class 1, 2, and 3 MOVs. Special Test Procedure 95-003 was written to gather information on MOV performance when operated under specific test conditions of differential pressure and flow rate.

Valve RHR-34A (V10-34A) is the Suppression Chamber Spray Bypass isolation/throttle valve for the Residual Heat Removal (RHR) system. The valve is normally closed and is manually opened to return water to the Torus during either RHR pump surveillance or primary containment cooling operations. Valve RHR-39A (V10-39A) is the Suppression Chamber Spray/Test isolation valve for the RHR system. The valve is also normally closed and manually opened to return water to the Torus during either RHR pump surveillance or primary containment cooling operations. Both valves have the safety function to close, when open, upon receipt of an RHR initiation or Primary Containment Isolation System Group II isolation signal to properly direct RHR System flow for core cooling and to assure primary containment integrity.

Safety Evaluation Summary

To preclude inadvertent coolant decreases, RHR valves in lines connected to the primary coolant system were verified to be shut prior to testing. However, if a decrease in water level had occurred, the leak path could have been manually closed. If the water level had continued to decrease such that adequate core cooling was no longer assured, the remaining ECCS subsystems would have provided sufficient capability for cooling.

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

57. **Special Test Procedure 95-004 In-Situ Differential Pressure Testing of Valves RHR-34A and RHR-39A** was completed 3/29/95.

General Summary

The purpose of this Special Test was to gather information on motor operated valve (MOV) performance when operated under specific test conditions of differential pressure and flow rate. See General Summary of Special Test Procedure 95-003 for reference to NRC Generic Letter 89-10.

Valves CS-11B and CS-12B (V14-11B and V14-12B) are the injection isolation valves for the "B" Core Spray Subsystem. Valve CS-11B is normally open and valve CS-12B is normally closed. Both valves have the safety function to open, when closed, upon receipt of a Core Spray System initiation signal to properly direct Core Spray System flow to the reactor pressure vessel.

Safety Evaluation Summary
During the testing, water level was administratively controlled between 500 and 540 inches above Top of Active Fuel (TAF). This ensured that the core submergence was maintained and flooding would not occur. If a malfunction had occurred in either the "B" Core Spray Subsystem or the Reactor Water Cleanup System such that the water level in the Reactor Well and the Dryer/ Separator Pit would decrease below the administrative limit of 500 inches above TAF, the isolation valves could have been closed to terminate the level decrease. The Refuel Floor Radiation Monitors' high trip function was bypassed to ensure that isolation of the Reactor Building normal ventilation would not occur due to shine from the Moisture Separator and Steam Dryer during such an event.

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

58. **Special Test Procedure 95-006 In-Situ Pressure Lock Testing of Valve CS-12B** was completed 3/29/95.

General Summary
The purpose of this Special Test was to gather information on core spray motor operated valve (MOV) performance when operated under specific test conditions of bonnet pressure locking. This Special Test Procedure provided supporting information for the analytical methods used to determine if certain MOVs would open if bonnet pressure locking conditions exist. The procedure also provided specific information on the Core Spray Injection Valves and demonstrated the ability of these valves to open under potential bonnet pressure locking conditions. See General Summary of Special Test Procedure 95-003 for reference to NRC Generic Letter 89-10.

Safety Evaluation Summary
The margin of safety for the removal of the "B" Core Spray Subsystem from operation during plant cold shutdown conditions is defined in Technical Specifications 3.5 and the associated bases. Technical Specifications 3.5.H.3 permits removal of all Core and Containment Cooling Subsystems from service provided no work is permitted which has the potential for draining the reactor vessel. The Core Spray spargers are located above the reactor core. As such, a failure of the Core Spray System during the testing would not have resulted in draining the reactor pressure vessel.

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

accident or malfunction as previously evaluated in the FSAR. This special test 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 was not endangered.

59. **Special Test Procedure 95-007 In-Situ Differential Pressure Testing of Valve SW-19A** was completed 4/22/95.

General Summary
The purpose of this Special Test was to gather information on motor operated valve (MOV) performance when operated under specific test conditions of differential pressure and flow rate. See General Summary of Special Test Procedure 95-003 for reference to NRC Generic Letter 89-10.

Valve SW-19A (V70-19A), in conjunction with valves SW-19B and SW-20, provides isolation for the nonessential Turbine Building loads of the Service Water System. Valve SW-19A is normally open and has the safety function to automatically close on low service water supply header pressure and to be manually opened for Alternate Cooling Water System operation.

Safety Evaluation Summary
With the reactor in cold shutdown condition, the only accident parameter variation which could have occurred related to the testing would have been a core coolant temperature increase due to loss of Station Service Water flow to the Residual Heat Removal Service Water System. Additionally, if loss of off-site power had occurred, Station Service Water flow to the Emergency Diesel Generators (EDGs) would have been required. Prior to the start of testing and removal of valve SW-19A from service, alternate testing of valves SW-19B and SW-20 was performed to ensure operability during the time period when valve SW-19A was considered inoperable.

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

60. **Special Test Procedure 95-008 In-Situ Differential Pressure Testing of Valves RHR-25A and RHR-27A** was completed 4/24/95.

General Summary
The purpose of this Special Test was to gather information on motor operated valve (MOV) performance when operated under specific test conditions of differential

pressure and flow rate. See General Summary of Special Test Procedure 95-003 for reference to NRC Generic Letter 89-10.

Valves RHR-25A and RHR-27A are Residual Heat Removal (RHR) Inboard and Outboard Injection Valves, respectively. Both valves have the safety function to open automatically upon receipt of a reactor low-low level and low pressure to initiate Low Pressure Coolant Injection (LPCI) and properly direct RHR System flow for core cooling.

Safety Evaluation Summary
To preclude inadvertent coolant inventory decreases, the companion RHR injection valve was verified to be shut prior to static testing. The only changes to a normal valve lineup consist of the position changes to RHR-25A and RHR-27A. However, if a decrease in water level had occurred, the leak path could have been manually closed. If the water level had continued to decrease such that adequate core cooling was no longer assured, the remaining Emergency Core Cooling Subsystems would have provided sufficient capability for cooling. This ensured that any accidents that would have occurred during the time when the "A" RHR Subsystem was out of service would have been bounded by the previous design basis analyses.

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

61. **Special Test Procedure 95-009 In-Situ Differential Pressure Testing of Valves RHR-25B and RHR-27B** was completed 4/25/95.

General Summary and Safety Evaluation Summary
Special Test Procedure 95-009 is identical to Special Test Procedure 95-008 described above, and valves RHR-25B and RHR-27B are identical to valves RHR-25A and RHR-27A tested in Special Test Procedure 95-008. The tests and safety evaluations were identical. See Special Test Procedure 95-008 for description.

62. **Special Test Procedure 95-010 In-Situ Differential Pressure Testing of Valve RCW-117** was completed 4/25/95.

General Summary
The purpose of this Special Test was to gather information on motor operated valve (MOV) performance when operated under specific test conditions of differential pressure and flow rates. See General Summary of Special Test Procedure 95-003 for

reference to NRC Generic Letter 89-10.

Valve RCW-117 provides primary containment isolation for the Reactor Building Closed Cooling Water System return line from the Drywell. Valve RCW-117 is normally open and has the safety function to be manually closed from the Control Room if a failure of the seismically qualified, closed loop inside the primary containment should occur following a Loss of Coolant Accident.

Safety Evaluation Summary
The testing was performed in cold shutdown, when primary containment was not required. Starting and stopping flow to the Drywell (Primary Containment) loads could not have resulted in malfunctions in other portions of the Reactor Building Closed Cooling Water, as the system was designed for such a configuration. If a malfunction of valve RCW-117 had occurred during the testing, the consequences of a malfunction of equipment important to safety would have been unchanged; valve RCW-117 would have remained inoperable. If a complete loss of Reactor Building Closed Cooling Water had occurred, Station Service Water would have been provided to the Residual Heat Removal and Control Rod Drive pump motor coolers.

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

63. **Basis for Maintaining Operability (BMO) 94-05 "A" Core Spray Subsystem** was initiated 4/5/94.

General Summary
Since startup from the 1993 Refueling Outage, the "A" Core Spray Subsystem pressure, as read at the pump discharge, had exceeded the static pressure provided by the Condensate System. The leakage rate past the CS-12A valve was not considered abnormal and BMO 94-05 concluded that there was reasonable assurance that the "A" Core Spray Subsystem would provide the required primary system pressure isolation, primary containment isolation, and operate properly when required during design basis accident conditions. On 5/17/94 the leakage rate increased. Valve CS-12A was cycled open and valve CS-11A was closed. Valve CS-841A was White Tagged closed. Valve CS-841A is the Safety Class 2 root isolation valve for pressure switch PS-14-47A. Leakage past valve CS-841A was observed and valve CS-842A, the Non Nuclear Safety second root isolation valve, was also closed. To compensate for the loss of the continuous pressure monitoring provided by PS-14-47A, twice per shift monitoring of PI-14-49A was initiated. With both valve CS-11A and CS-12A closed, questions were raised as to whether, in light of the 1000 psig pressure

between theses valves, they would open during design basis accident conditions. This BMO and Safety Evaluation determined the acceptability of declaring the "A" Core Spray System operable with valve CS-11A closed and CS-12A open and continuing plant operations with this configuration.

Safety Evaluation Summary
Core Spray Valve 11A, normally open, and 12A, normally closed, are the outboard injection valves for the Core Spray Subsystem "A". These valves function as injection valves during an accident and primary containment pressure boundaries, depending on which valve is closed, during normal operations. As the 12A valve allowed leakage past its seat and created a pressure increase between the valves, the normal valve lineup was changed such that the 11A valve was closed and the 12A valve was opened. Additionally, the pressure switch installed between the valves, used to detect leakage, was isolated to prevent damage to the instrument.

This configuration change retains the function of an injection valve to backup the Core Spray check valve and prevent loss of reactor coolant outside containment, and provides an acceptable safety class boundary with the new configuration. Temporary pressure monitoring is provided to replace the isolated pressure switch. These changes will not affect the ability or reliability of the "A" Core Spray Subsystem to perform its required functions.

There was no increase in the probability of occurrence or consequences of an accident or malfunction as previously evaluated in the FSAR. This BMO 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.

64. **Basis for Maintaining Operation (BMO) 94-11 Vital AC Operation on Alternate Source** was initiated 8/3/94.

General Summary
On 7/20/94, the normal power source from the Safety Class Electrical (SCE) Vital AC Motor Generator (MG) Set to the 120/240 volt uninterruptible Vital AC distribution panel transferred to the maintenance tie, Transformer DT-1. Transformer DT-1 is classified as SCE, and is supplied from SCE MCC 9A, which is powered from Emergency Diesel Generator (EDG) DG-1-1A. An attempt to transfer the distribution panel back to its normal supply resulted in tripping the Vital MG set output breaker and excitation system. Troubleshooting efforts indicated that the source of the problem was not the MG set, but the transfer switch. The Vital AC switch was subsequently removed and a new ASCO transfer switch was installed. A safety evaluation was prepared to demonstrate that there was not an unreviewed safety question created by operation with Vital AC powered by its alternate power

supply.

Safety Evaluation Summary
Emergency Core Cooling Systems (ECCS) required to mitigate the consequences of an accident are designed to be initiated from DC power, independent of Vital AC. All potential failures of Vital AC were evaluated and would not degrade the performance of required mitigation systems. During a Loss of Coolant Accident and High Energy Line Break, the vital AC bus is assumed to be powered via the alternate source since the Vital MG is not EQ qualified. Automatic operation of systems, structures and components required to mitigate the consequences of a control rod drop or refueling accident are not impacted by operation of the Vital AC bus from its alternate source. All required transient mitigation systems were available and would function 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 BMO 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.

65. **Valve Lineup Deviation 94-06**

This lineup deviation described the closure of normally-open core spray valve CS-11A. This was done to protect the "A" Core Spray piping from reactor pressure. This deviation had an impact on the safe operation of the facility; a full description of the issue and safety evaluation are summarized under section "BMO 94-05".

66. **Valve Lineup Deviation 94-07**

This lineup deviation described the closure of normally-open core spray valve CS-11A. This was done to protect the "A" Core Spray piping from reactor pressure. This deviation had an impact on the safe operation of the facility; a full description of the issue and safety evaluation are summarized under section "BMO 94-05".

67. **Valve Lineup Deviation 94-08**

This lineup deviation described the opening of normally-closed core spray valve CS-12A. This was done to protect the "A" core spray piping from reactor pressure. This deviation had an impact on the safe operation of the facility; a full description of the issue and safety evaluation are summarized under section "BMO 94-05".

68. **Instrument Setpoint Change Request 93-043** was initiated 11/16/93.

General Summary
This Instrument Setpoint Change Request reflects the two possible modes of operation which can be used to provide containment instruments with nitrogen gas during operation with the containment inerted. The normal method of operation is with the Nitrogen Supply System adjusted to provide nitrogen at 110 psi and the Containment Air Compressor set to start at 95 psi and shut off at 105 psi. In this mode of operation, the Nitrogen Supply System is the primary supply for the containment instruments with the Containment Air Compressor used as the backup supply. The alternate mode of operation, already described in the FSAR, is to adjust the Nitrogen Supply System for a supply pressure of 90 psi. In this mode, the normal supply of nitrogen for the containment instruments is from the Containment Air Compressor which will cycle between 95 to 105 psi with the Nitrogen Supply System providing the backup function.

Safety Evaluation Summary
There is no Safety Objective nor Safety Design Basis for the Containment Instrument N2 Subsystem as described in FSAR Section 10.14, "Station Instrument and Service Air Systems". The only equipment parameter that this subsystem could affect which could potentially impact the radioactive material release from the Control Rod Drop Accident, Loss of Coolant Accident, or Main Steam Line Break Accident is the closure time of the inboard Main Steam Isolation Valves (MSIVs). However, these referenced closure times will not be challenged because containment nitrogen system pressure will be maintained within the MSIV actuator design operating pressure range of 90 to 110 psig. The ability of the actuator solenoid valves to initiate MSIV closure will not be jeopardized because the Maximum Operating Pressure Differential Specification for these solenoids, 125 psid, remains above the Containment Instrument N2 Subsystem upper pressure setting of 110 psid.

There was no increase in the probability of occurrence or consequences of an accident or malfunction as previously evaluated in the FSAR. This FSAR 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.

69. **Cycle 18 Core Operating Limits Report**

This report provided the cycle-specific limits for operation of the Vermont Yankee Power Station for Cycle 18. The limits included the maximum average planar linear heat generation rate, maximum linear heat generation rate, and minimum critical power ratio.

The Cycle 18 core contains 248 irradiated GE-9B bundles and 120 fresh GE-9B bundles, manufactured by General Electric. The average initial enrichment for the irradiated bundles is 3.11 and 3.35 weight percent U-235, and the average initial enrichment for the new bundles is 3.35 weight percent U-235. All bundles have Zr-2 channels. Cycle 18 also contains 79 irradiated control rods, 10 new control rods, 18 irradiated Local Power Range Monitor (LPRM) strings, and 2 new LPRM strings. The new channels, new control rods and new LPRMs are one-for-one replacements for the previous equipment.

The new bundles are mechanically equivalent to the GE-9B bundles in Vermont Yankee. The mechanical evaluations included bounding assumptions relative to operation out to the maximum allowable planar exposure of the fuel. The maximum planar exposure is based on the peak pellet exposure of 60 GWd/MTU.

Safety Evaluation Summary
None of the changes made for Cycle 18 increased the probability of an accident or transient previously evaluated in the FSAR. No plant hardware modifications affect the safety analysis assumptions. The new GE-9B bundles differ from the irradiated bundles in the average initial enrichment, which will not cause an increase in the probability of an accident or transient because the initiating event does not depend on the fuel characteristics.

The core changes do not increase the probability of a thermal-hydraulic instability. The control rods will perform their function of bringing the core subcritical, even with the highest worth rod withdrawn. The Loss of Cooling Accident/Emergency Core Cooling System analysis, assuming the operating limits on the COLR, shows that Cycle 18 meets the acceptance criteria of 10CFR50.46. Therefore, the operation of Cycle 18 within the operating limits in the COLR will not reduce the margin of safety.

There was no increase in the probability of occurrence or consequences of an accident or malfunction as previously evaluated in the FSAR. The Cycle 18 reload 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.

70. **Significant Corrective Action Report 95-005: Connection of Omega Model 585 Chart Recorder to Rosemount Master Trip Card 2-3-72B(M) Test Jacks** was installed on 10/4/94 and removed 11/10/94.

General Summary
A chart recorder was connected to Master Trip Card 2-3-72B(M) via test jacks located on the front of the card file. This was done in response to downspiking in the

Reactor Water Level Instrument Loop LT-2-3-72B. One channel monitored the 1 to 5 VDC input signal to Master Trip Card 2-3-72B(M) at test jack J1, with the other channel monitoring the 1 to 5 VDC output from 2-3-72B(M) at test jack J2. Both measurements were taken with respect to Signal Ground Test jack J2 located on the front of the associated calibration unit.

Safety Evaluation Summary
The Rosemount 710DU Master Trip Unit that was monitored is part of the accident mitigation equipment described in FSAR Section 7.4, "Core Standby Cooling Systems Control and Instrumentation". The Failure Modes and Effects Analysis for this connection concluded that the credible failures of the monitoring equipment would not affect the response time or ability of the trip system to initiate the appropriate emergency core cooling system responses, in the event of a Low-Low Reactor Water Level condition.

In addition to initiating Core Cooling systems, the card file also secures the High Pressure Coolant Injection (HPCI) and Reactor Core Isolation Cooling (RCIC) systems on High Reactor Water Level to prevent gross carryover of moisture into the respective turbine. In the event of a short circuit failure of the recorder sensing input connection, this high water level trip feature would be effectively disabled because the logic is arranged in a "two out of two taken once" scheme. However, the loss of this feature is evaluated under FSAR 7.4.3.2.4, "HPCI Turbine and Turbine Auxiliaries Control".

There was no increase in the probability of occurrence or consequences of an accident or malfunction as previously evaluated in the FSAR. This connection 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.

71. **Refuel Floor Auxiliary Bridge** was installed and used during the 1995 Refueling Outage.

General Summary
The refuel floor auxiliary bridge was placed over the vessel cavity to facilitate In Vessel Visual Inspection, In Service Inspection requirements during refueling outages. The bridge weighs approximately 6,000 pounds and consists of three sections bolted together at two splice joints. Transportation of the bridge is accomplished using the refuel floor "single failure proof" crane and a redundant system of lifting lugs welded to the top face of the auxiliary bridge girders. This mode of transportation meets the "heavy" load lift requirements of NUREG-0612. Although the auxiliary bridge is a non-safety related piece of equipment not described in the FSAR, it is positioned directly over equipment important to safety.

Safety Evaluation Summary
This auxiliary bridge is not a safety system, nor does it directly interface with a system important to safety. The bridge was designed to the same degree of safety as the refueling platform. A seismic evaluation determined that the bridge could not fail, slip off the tracks or overturn due to a seismic event. Measures have been taken to ensure that no material used in the construction of the bridge could inadvertently fall into the open reactor vessel. As a result, the positioning of the auxiliary bridge over the vessel cavity does not degrade or challenge the performance of any safety system below it to function in an accident.

There was no increase in the probability of occurrence or consequences of an accident or malfunction as previously evaluated in the FSAR. The installation of the refuel floor auxiliary bridge 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.

72. **Low Level Waste (LLW) Storage Pad Safety Evaluation Revision 4** was issued in July 1994.

General Summary
The Vermont Yankee LLW storage pad had not been used since 1991 due to the availability of offsite disposal at Barnwell, South Carolina. On 7/1/94, Barnwell was closed to all out of region waste generators. On-site storage of low level radioactive waste was again necessary on the Vermont Yankee LLW storage pad. The Vermont Yankee LLW Safety Evaluation was updated to reflect changes in waste handling operations. The updated safety evaluation provides for higher Curie limits in packaged waste (primarily expected in condensate polishing resins) to reflect higher concentrations of activity in various waste streams than previously accounted for. This revision also recognized that dry active waste (DAW) can be volume reduced off-site through such means as incineration, with repackaged waste returned to Vermont Yankee for storage.

Safety Evaluation Summary
Revision 4 noted that DAW processed on-site is packed in noncombustible steel boxes. Off-site processing of DAW by incineration produces a waste form (non-airborne dispersible ash) that will no longer support combustion. The margins of safety as defined in the bases for the technical specifications are not reduced. The storage pad facility located outside the plant protected area will have no impact on in-plant safety-related systems. But the facility is within the realm of accident event consequences evaluated in the FSAR. Thus, the margin of safety defined in the Technical Specifications is not compromised.

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

accident or malfunction as previously evaluated in the FSAR. This revision to the LLW Storage Pad Safety Evaluation 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.

73. **Addition of Fuse to Main Steam (MS) Relief Valve Bellows Leakage Detection Circuitry** was performed under Minor Modification 95-029. The NRC subsequently expressed concern that although a safety review had been performed, a full 50.59(a)(2) safety evaluation had not. To address this concern, a 50.59(a)(2) safety evaluation was prepared and confirmed that no unreviewed safety question existed as a result of the implementation of this Minor Modification. The evaluation was presented to the Plant Operations Review Committee prior to startup from the 1995 outage.

General Summary

It was determined that a short to ground in the PS-2-71A,B,C,D pressure switch circuitry due to the development of a harsh environment could potentially render the MS relief valves inoperable. To prevent the loss of the automatic/manual operation of MS relief valves due to a fault in the pressure switch circuitry, safety class fuses have been installed to isolate the pressure switch circuitry from the remaining MS relief valve control circuitry. The installation of these fuses isolated the non-environmentally qualified pressure switch bellows leakage detection system circuitry from MS relief valve control circuitry.

Following this modification, if ground faults occur at the pressure switch circuitry concurrent with other substantial ground faults in the respective 125 VDC bus, the fuses will blow and isolate the MS relief valve control circuitry.

Safety Evaluation Summary

The MS relief valve bellows leakage detection pressure switches are used to confirm the integrity of RV bellows to assure operability of MS relief valves. The mitigating functions of the MS relief valves is unaffected by the addition of fuses to isolate pressure switch circuitry from RV control circuitry. Installation of these fuses ensures the continued operation of the MS relief valve automatic/manual circuitry in the event of a ground fault in the pressure switch circuitry concurrent with a substantial ground on the negative leg of the respective 125 VDC system. The operation of the pressure switch bellows leakage detection system and the MS relief valves is the same following the fuse addition as it was before the fuse addition.

There was no increase in the probability of occurrence or consequences of an accident or malfunction as previously evaluated in the FSAR. The fuse addition to MS relief valve bellows leakage detection circuitry 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.

74. **Safety Class Reclassification: Pressure Transmitter PT-1-156-2 and Tubing**

General Summary
Pressure transmitter PT-1-156-3 and tubing, located downstream of valve V16-19-1 (AC-1) have been upgraded to Safety Class 2 and Seismic Class I. Valve AC-1 is attached to Primary Containment Penetration X-52 connection "F". Valve AC-1 is normally open to allow PT-1-156-3 to sense drywell pressure. 3/8" diameter stainless steel instrument tubing runs from the valve to PT-1-156-3, which is mounted on the 280' elevation. A failure of the tubing and/or transmitter while valve AC-1 is open could compromise primary containment integrity. Therefore, PT-1-156-3 and the tubing have been reclassified as Safety Class 2. PT-1-156-3 was replaced with a Safety Class 2 transmitter. The tubing, fittings and supports were evaluated and found suitable for upgrade. Seismic adequacy has been confirmed.

Safety Evaluation Summary
The safety function of PT-1-156-3 and the associated tubing, fittings and valves is to maintain primary containment integrity following an accident. This upgrade ensures that future maintenance and testing of these components will be done to more stringent QA requirements. The upgrade did not adversely affect the capability of these components or any other equipment or systems important to safety to perform their safety function.

There was no increase in the probability of occurrence or consequences of an accident or malfunction as previously evaluated in the FSAR. This safety class reclassification 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.

75. **Procedure OP1408, revision 13, Local Power Range Monitor (LPRM) Removal and Replacement** was issued 3/26/95.

General Summary
This procedure was revised to allow LPRM replacement to be performed in either the Refuel or Shutdown modes. A prerequisite was added identifying the need for a Temporary Modification to allow the bridge over the core when in the Shutdown mode. Changes were also made to accommodate LPRM replacement without secondary containment. Rigging guidelines were established to ensure the lifting of the LPRM bender meets the requirements of NUREG 0612, "Control of Heavy Loads at Nuclear Power Plants". All other tools lifted per this procedure meet the NUREG requirements with no changes to the lifting methods. A prerequisite was added to

ensure all other items lifted over fuel during performance of this procedure have been analyzed.

Safety Evaluation Summary
One of the accidents listed in the FSAR is a "mechanical failure of various components leading to the release of radioactive material from one or more barriers". The potential for dropping an LPRM on the core causing fuel damage falls into this category. The radiological consequences associated with the use of this procedure and the potential to drop an LPRM is bounded by the Refueling Accident, FSAR section 14.6.4; and safety design basis limits for accident, section 14.3. The LPRMs and associated lifting devices are not accident initiators, nor are they designed to mitigate accidents or make up any part of the barriers to radioactive release.

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.

76. **Procedure OP2124, revision 36, Residual Heat Removal System** was issued 8/31/94.

General Summary
This procedure revision added a flush of the Residual Heat Removal System (RHRS) to the vessel with condensate prior to placing the RHRS in service in the shutdown cooling mode. The purpose of the flush is to eliminate Group IV isolation signals due to pressure spikes that have occurred in the past while placing the RHRS in the shutdown cooling mode. The pressure spikes are attributed to high initial RHR flow interacting with a steam/vapor void in the piping high points downstream of the Low Pressure Coolant Injection (LPCI) valve. The function of the flush is to displace/ collapse any existing voids in the injection piping by slowly filling the injection header with the condensate system prior to starting the RHR pump and initiating shutdown cooling flow.

Safety Evaluation Summary
Administrative controls require the Control Room to monitor the primary pressure and level during the flush and for an operator to be available in the area to close the manual isolation valves if directed by the control room. In the event that a loss of RHR system integrity occurs during this evolution, the control room also has the capability to isolate the RHR system remotely to mitigate any potential transfer of radioactive material release/loss of primary coolant inventory. Eliminating potential for Group IV isolation signals improves the reliability of the shutdown cooling mode of RHRS operation.

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.

77. **Procedure ON3159, revision 0, Loss of DC-1** was issued 1/19/95.

General Summary

This procedure responds to a complete loss of DC-1. Guidance is provided for deenergizing Recirc Motor Generator (MG) Set "B" as soon as possible to prevent catastrophic damage due to loss of lube oil. This is accomplished by manually tripping the drive motor breaker, provided that a locked rotor current is not present. In the event the rotor has seized up, steps are taken to deenergize Bus 1 by scramming the plant and/or deenergizing the Startup Transformers. Control power for Buses 1, 3, and 8, and DG-1B is transferred to DC-2; this precludes the possibility of cross connecting a faulted bus. Subsequent actions are taken to restore DC-1 to service and return control power for the above equipment to the normal supply.

Safety Evaluation Summary

This procedure places the plant in single loop operation, transfers control power for Buses 1, 3, and 8, and DG-1B to their alternate source, and takes action to restore DC-1 to service. Single loop operation has been previously evaluated. Transferring control power makes the equipment on Buses 1, 3, and 8, and DG-1 available. By doing so, it provides the operators with increased flexibility in dealing with any transients that may occur during the time that DC-1 is out of service. This increased availability could prevent and/or mitigate the consequences of an operational transient. The additional loading and loss of separation will not adversely affect the ability to mitigate the consequences of any operational transients.

Transferring control power places an additional load on DC-2. The current calculation does not address this minor load. However, there is some excess conservatism in the current calculation that will be removed in the next revision. By comparison, the control power for the switchgear and DG-1B is a considerably smaller load. Therefore, performance of this procedure will not cause an overloading of DC-2 and will not affect the results of any accidents 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 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.

78. **Procedure ON3160, revision 0, Loss of DC-2 and DC-3** was issued 1/19/95.

General Summary

This procedure responds to a complete loss of DC-2. Operators are instructed to immediately restore power to DC-3A by manual transfer to DC-1. This is done to restore Main and Auxiliary Transformer cooling. It is assumed that a problem on DC-3A cannot be the cause of the loss of DC-2 by applying single failure criteria. In the event power cannot be restored immediately, a scram is initiated to remove the heat load from the transformers. Guidance is provided for deenergizing Recirc MG Set "A" as soon as possible to prevent catastrophic damage due to loss of lube oil. This is accomplished by manually tripping the drive motor breaker, provided that a locked rotor current is not present. In the event the rotor has seized up, steps are taken to deenergize Bus 1 by scramming the plant and/or deenergizing the Startup Transformers. Control power for Buses 4 and 9 is transferred to DC-2AS. Control power for Bus 2 is transferred to DC-1. This precludes the possibility of cross connecting a faulted bus. Subsequent actions are taken to restore DC-2 and DC-3 to service and return control power for the above equipment to the normal supply.

Safety Evaluation Summary

This procedure restores power to DC-3A from DC-1, places the plant in single loop operation, transfers control power for Buses 2, 4, and 9 to their alternate source, and then takes action to restore DC-2 and DC-3 to service. Single loop operation has been previously evaluated. Transferring control power makes the equipment on Buses 2, 4, and 9 available. By doing so, it provides the operators with increased flexibility in dealing with any transients that may occur during the time that DC-2 and DC-3 are out of service. This increased availability could prevent and/or mitigate the consequences of an operational transient. The additional loading and loss of separation will not adversely affect the ability to mitigate the consequences of any operational transients.

Transferring control power for Bus 2 and repowering DC-3A places an additional load on DC-1. The current calculations do not address this minor load. However, there is some excess conservativism in the current calculations that will be removed in the next revision. By comparison, the control power for the switchgear and DC-3A is a considerably smaller load. The current calculations on DC-2AS include buses 4 and 9. Therefore, performance of this procedure will not cause an overloading of DC-1 or DC-2AS and will not affect the results of any of the accidents 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 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.

79. **Procedure AP3532, revision 4, Emergency Preparedness Organization (Implementing Procedure to the VY E-Plan)** was issued 3/30/95.

General Summary

This procedure was revised to reflect the change in management responsibility for emergency planning form the Director of External Affairs to the Operations Support Manager. The relocation of this function provides a more direct access to additional technical resources to enhance the level of planning and maintenance for implementation of the Emergency Plan. The Vermont Yankee Emergency Plan Coordinator will continue to be responsible for all Emergency Plan activities.

Safety Evaluation Summary

This change alters an administrative procedure contained in the FSAR. The change alters responsibilities from those previously documented in submittals to the NRC. Responsibility has been shifted to an individual of equal or higher rank and qualifications, or to a lower level individual with upper level supervision. The Vermont Yankee Emergency Plan, Section 12.5, will be revised to reflect the new management structure as defined in AP3532, revision 4.

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.

80. **Procedure OP4123, revision 27, Core Spray System Surveillance** was issued 4/17/95.

General Summary

Valve CS-12 and CS-13A, in combination, were leaking. This resulted in an increase in pressure in the low pressure piping outboard of valve CS-12A. During the 1995 outage, valves CS-12A and 13A were repaired under Work Order 94-04530. As an interim measure, procedure OP4123 was revised to add CS-13A Leak Rate Surveillance Test as a quarterly test. This was done in conjunction with Temporary Modification 94-029, under which a test rig was seismically mounted to allow for the leak rate testing.

Safety Evaluation Summary

The Core Spray System performs a mitigating function of the accidents described in the FSAR. The Core Spray A Subsystem is connected to primary containment penetration X-16B and is required to provide isolation following a Loss of Coolant Accident. Performance of the CS-13A Leak Rate Surveillance Test did not affect the ability of the MOV's, CS-11A, or CS-12A to be operated for the Core Spray function or for containment isolation function.

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.

81. **Procedure OP4195, revision 21, Fuel Oil Transfer System Surveillance** was issued 1/4/95.

General Summary

OP4195 was revised to add section "Obtaining Fuel Oil Storage Tank (FOST) oil temperature" as a daily surveillance. Obtaining FOST oil temperature is performed during normal plant operation, verifies that the oil temperature is maintained above the oil Cloud Point temperature, and determines the effectiveness of the FOST heating system installed under Temporary Modification 94-031. The Cloud Point generally relates to the temperature at which wax crystals precipitate from the fuel during use, and results in plugging of filtration media with wax precipitates. If the oil temperature is not maintained above the Cloud Point, the continued operability of the Fuel Oil Transfer System to supply the Emergency Diesel Generators (EDGs) is uncertain.

The surveillance consists of opening FO-9, Recirc Valve, then starting the Fuel Oil Transfer Pump which is selected to supply the heating boiler day tank. The heating boiler day tank flow path is verified isolated during this evolution. FO-9 is a manually operated valve which is normally locked closed. FO-9 is also the SC3/NNS boundary isolation. During the surveillance an operator is stationed at the valve area to secure the evolution and return the system to normal standby if directed by the Control Room.

Safety Evaluation Summary

This surveillance changes how the fuel oil transfer system functions during accidents in that the recirculation lineup established during this evolution would prevent EDG day tank automatic refill at the design rate. However, the equipment operated during this surveillance is administratively controlled by procedure, the evolution takes a short time to perform, and the system can easily be restored to the normal standby lineup. The interim system lineup deviation is not expected to be established for more than 15 minutes. Considering that each EDG day tank is sized to provide 3 hours of EDG operation at full load per FSAR section 8.5.4, there is sufficient time for operator actions to restore the system to standby is required for plant conditions; thus there would be no loss of standby EDG capability.

The Fuel Oil Transfer System supports equipment used in mitigating accidents described in the FSAR. The diesel generators provide the electrical source if off-site electrical power is not available. This power source is for the low pressure ECCS equipment and SBGT for long term mitigation for the consequences of all four accidents. This surveillance does not change how the EDG operates during these accidents.

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.