VERMONT YANKEE

CYCLE 16 OPERATING REPORT

OPERATIONS SUMMARY

The frequency of submittal for the Operating Report has been changed from an annual basis to be concurrent with the submittal of the FSAR update (cycle specific - 10CFR50.59(b)(2), as amended). Changes to the Vermont Yankee facility which occurred in the 1992 portion of Cycle 16 were covered in the 1992 Annual Operating Report. Between January 1, 1993 and October 24, 1993, (the end of operating cycle 16, by Technical Specifications definition), Vermont Yankee implemented a number of changes. The following report describes those changes which constituted a change in the facility as described in the Final Safety Analysis Report (FSAR). The report includes eleven Engineering Design Change Requests (EDCRs), five Plant Design Change Requests (PDCRs), thirteen Temporary Modifications (TMs), eight Special Test Procedures, one Service Water Lineup Change, one maintenance activity (burning non-contaminated waste oil in the Containment Access Building), a Core Reload, a Bundle (LYV667) Replacement, and a decision to use Reflective 3D Monicore. There were no Safety Relief Valve Failures, Valve Lineup Deviations requiring 50.59 safety evaluations, or Setpoint Changes requiring 50.59 safety evaluations performed between January 1, 1993 and October 24, 1993.

A. Changes in Facility Design

1. Between January 1, 1993 and October 24, 1993, there were no changes made which required prior authorization from the Commission.

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

   a. EDCR 89-408 Spent Fuel Pool Cooling System Enhancement was completed 8/27/93.

   General Summary

   This design added the standby fuel pool cooling subsystem (SFPCS) to the fuel pool cooling demineralizer system (FPCDS) to increase the capability to mitigate abnormal spent fuel pool heat load conditions. This system provides sufficient heat removal capacity to preclude any impact on plant operation due to insufficient spent fuel pool cooling.

The SFPCS is a Safety Class 3, Seismic Class I, two train system designed to prevent a single active failure or common event from disabling both trains such that the ability of the system to remove decay heat is compromised. Each train contains one pump, one heat exchanger and associated valves, instrumentation and controls. The pump circulates the pool water in a closed loop, taking suction from the spent fuel pool through the heat exchanger and returning it back to the pool.

The SFPCS is separated from the normal fuel pool cooling subsystem (NFPCS) by a combination of motor operated valves (MOVs) and check valves. These valves provide isolation of the nonseismic NFPCS from the seismic SFPCS.

Safety Evaluation Summary

The addition of the SFPCS provides spent fuel pool cooling capability beyond that of the NFPCS. No electrical or mechanical single active failure will disable both trains of the SFPCS or cause a failure of a system essential to plant safety. No failure of the SFPCS equipment will adversely affect any accident mitigation system. This ensures no impact on the radiological effects of the design basis accidents identified in the FSAR. The addition of the SFPCS subsystem to the NFPCS provides additional capability to deal with the consequences of NFPCS equipment failure; thus the consequences of a malfunction are actually reduced by the addition of the SFPCS. A differential pressure is maintained between the fuel pool side and the service water side of the heat exchangers; the service water pressure will be kept above the fuel pool water pressure to provide additional assurance against any leakage of fuel pool water into service water.

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.

b. EDCR 90-411 Replacement of PCB Transformers T-6 through T-10 was completed 10/19/93.

General Summary

This design change replaced PCB transformers T-6, T-7, T-8, T-9 and T-10, which were liquid filled, forced air cooled transformers, used for station service and filled with Pyranol, a PCB contaminated insulating fluid. Replacing these transformers relieved Vermont Yankee from all risks and liabilities associated with maintaining them, including high cleanup costs, lost revenue and settlement of third party claims that may have resulted from a PCB fire.

Transformers T-6, T-7, and T-10 are located in the Turbine Building and are classified as Non-Nuclear Safety (NNS). Transformers T-8 and T-9 are located in the Switchgear Room and are classified as Safety Class Electrical (SCE). The new transformers T-6, T-7, T-8, T-9, and T-10 are filled with silicone fluid, suitable for indoor use, and provide a one-for-one replacement for the previous units. The new transformers have essentially the same ratings and performance characteristics as the old units; therefore, there is no impact on the operation of the electrical distribution system at Vermont Yankee.

Safety Evaluation Summary

No technical bases were adversely affected as a result of this modification since this change provided a one-for-one replacement of the transformers. The new transformers were installed in the same location as the old units and a berm around each transformer was added to confine an oil spill and/or fire of the transformer fluid. This modification did not introduce any unwanted or previously unreviewed interactions between structures, systems or components important to safety. Transformers T-8 and T-9 are Class 1E, seismically qualified components and the NNS transformers are mounted to ensure that their failure will not impact operation of safety related equipment and components.

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.

c. EDCR 90-412 Vernon Tie Line Improvements was completed 8/21/93.

General Summary

This design change modified the offsite power source from the Vernon Hydro Station to Vermont Yankee Nuclear Power Station; this source is referred to as the Vernon Tie Line. The line will be used as part of an alternate ac source to ensure ac power is available to Vermont Yankee in the event of a "loss of offsite electric power system concurrent with turbine trip and unavailability of the onsite emergency power system". Also, New England Power Company is upgrading the Vernon Hydro Station, and the old tie line was not compatible with the new configuration.

The new line originates at the hydro station switchyard instead of being fed directly from the generator bus as was the old line. As opposed to the old above ground cable, the new 15 kV cable runs underground to a new 13.2 kV to 4.16 kV transformer located on Vermont Yankee property by the cooling towers. A 5 kV cable runs from the secondary of the transformer, in new ductbank, and is spliced into the 5 kV cable at Manhole P8, outside the protected area.

Safety Evaluation Summary

The reason for modifying the Vernon Tie Line was to improve its reliability in the event of a loss of all ac power to Vermont Yankee. The probability for a malfunction of this line or of the emergency ac power system which it supplies is reduced with the implementation of this modification. The feed from the Vernon switchyard, while electrically more distant from the hydro generators, is comparable in reliability, since the switchyard is also connected to five additional transmission lines from the 69 kV system. This system is independent from the 115 kV and 345 kV systems supplying offsite power to Vermont Yankee. The new underground cable is more reliable than the old overhead line because it is not subject to weather-induced failure.

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.

d. EDCR 92-402 Motor Operated Valve Temporary Modifications Conversions was completed 1/8/93.

General Summary

This design change converted the following five Temporary Modifications (TMs) to permanent changes:

TM 91-54 Core Spray Minimum Flow Bypass Valve V14-5A

TM 92-56 Core Spray Minimum Flow Bypass Valve V14-5B

TM 92-54 Recirculation Pipe Discharge Bypass Valves V2-54A/B
TM 92-57 High Pressure Coolant Injection Steam Supply Line Isolation Valve V23-15
TM 92-61 Residual Heat Removal Discharge to Radwaste Isolation Valve V10-66

TMs 91-54 and 92-56 replaced the Motor Operated Valve 2 ft-lb motors with 5 ft-lb motors, increasing the motor horsepower from 0.13 hp to 0.33 hp, ensuring that the valve motors develop sufficient torque to achieve the higher thrust settings as determined from review of the MOVATS database and NRC Generic Letter 89-10 issues.

TMs 92-54, 92-57 and 92-61 rewired the limit switch compartments of the valves listed above, to bypass the torque switch for 97% - 99% of the close stroke, which ensures that the valves will close to perform their intended safety function. Previously, the higher thrust required to close the valves against high differential pressure would actuate the torque switch and prevent the valve from fully closing. In addition, the motor thermal overloads were replaced, providing protection to prevent the motors from burning up due to a sustained overload condition.

Safety Evaluation Summary:

There is no impact on the requirements imposed by Technical Specifications on the valves modified in this design change. This change does not affect any design limits imposed by the electrical distribution system, seismic and environmental qualification, or Appendix R. This modification does not change any automatic or manual functions required by the valves either during normal or accident conditions; and therefore does not impact the integrity of the radioactive material barriers and/or nuclear safety/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 the health and safety of the public was not endangered.

e. EDCR 92-404 Residual Heat Removal Service Water (RHRSW) System Modifications Associated with Valves V10-89A and V10-89B was completed 10/2/93.

General Summary:

This design change modified the associated mechanical and electrical components of the RHRSW heat exchanger flow and pressure control system. The V10-89A(B) motor operated valves (MOVs) were replaced. The manual isolation valves V10-191A(B) and V10-192A(B) were replaced. The automatic instrument and control logic was revised to allow remote-manual process control. The process instrument taps and impulse lines were modified. The Control Room Panel 9-3 was repaired to "as new" condition. Portions of the carbon steel pipe and components were replaced with stainless steel pipe and components. The safety class boundary on valve SW-500A was modified.

In the past there had been a high incidence of failure for the V10-89A(B) valves due to high vibration of the valves and associated equipment due to high differential pressure across the valve disc in conjunction with the disc being close to the valve seat. Attempts to address these problems had not been successful. This modification was designed to ensure the process can be adequately controlled without damage to the system or components.

Safety Evaluation Summary

The RHRSW system supports the RHR system in performing several safety related functions: the removal of core decay heat during normal and post-accident conditions, alternate cooling, and containment and torus cooling. This design change has not diminished the availability of these functions. There is no new safety concern or new interface with any existing analyzed safety function.

The margin of safety as defined in the Vermont Yankee Technical Specifications was not affected. Directly, the technical specification requires the RHRSW system to be capable of flowing 2700 gpm at 70 psig at the discharge of the heat exchanger. This design change supports this criteria. Indirectly, the RHRSW supports alternate cooling, torus cooling, and containment cooling. The technical specification limits are not adversely affected by this design change. Also, limits placed on the diesel generator capacity have been evaluated; these limits are not adversely impacted 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.

f. <u>EDCR 92-406 Station Service and Instrument Air Compressor Replacement</u> was installed and made operational on 8/20/93.

<u>General Summary</u>

This design change replaced the station service air compressors C-1-1A through C-1-1D. The old units were replaced as a result of the significant maintenance required to maintain availability. The station service air compressors provide oil-free compressed air to the station instrument and service air systems, which provide the station with the compressed air requirements for pneumatic instruments and controls and general station services.

The new air-cooled air compressors negate the need for the cooling water arrangement associated with the old compressors. Air compressor cooling during all operating modes, an LNP, or alternate cooling would be independent of the service water system. This also eliminates compressor overheating concerns due to service water back pressure following an LNP.

The new compressors are located in a new room at the west end of the boiler room and west end of the water treatment area; the abandoned acid storage and caustic storage tanks were removed. The building was modified to facilitate compressor installation and new piping from the compressors to the existing air receivers. Also included was the removal of the existing cooling water piping/valve arrangement and old air compressors. Some of the old compressor controls/instrumentation was retained to interface with new compressor controls to provide the desired compressor operation.

<u>Safety Evaluation Summary</u>

The station instrument and service air systems are classified NNS. They play no part in the initiation of any accident evaluated in the FSAR, nor are they required to support any accident function or accident response. The relocation of the safety class boundary from valve V70-84A(B) to existing Safety Class 3 valve V70-81C(D), which is normally closed, does not increase the probability of a malfunction of any equipment important to safety.

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

g. EDCR 92-408, Generic Letter 89-10 Motor Operated Valve (MOV) Improvements was completed 10/7/93.

General Summary

This design change modified four motor operated valves 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.

Valve V10-16A, Residual heat removal (RHR) pump minimum flow bypass: 2 ft-lb motor was replaced with 10 ft-lb motor.

Valve V10-16B, Residual heat removal (RHR) pump minimum flow bypass: 2 ft-lb motor was replaced with 10 ft-lb motor.

Valve V13-27, Reactor core isolation cooling (RCIC) min flow bypass: 10 ft-lb motor was replaced with 15 ft-lb motor.

Valve V23-16, High pressure coolant injection (HPCI) outboard isolation: 40 ft-lb motor was replaced with 60 ft-lb motor. Gear set change was completed.

Safety Evaluation Summary

Since the required safety function of the valves is not affected due to this modification, this change cannot directly or indirectly increase the chance of an accident. This modification does not degrade or prevent any automatic operation of equipment required to mitigate the consequences of the accidents evaluated in the FSAR. There are no new accidents created by implementation of this design change since there are no major equipment changes or additions.

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.

h. EDCR 93-401 Vermont Yankee Fire Seal Penetration Modification was completed 4/23/93.

General Summary

As a result of an inspection that revealed degraded fire barrier seals, Vermont Yankee contracted Brand Engineering to design replacement fire barrier seals; in certain cases, rather than install new seals, Vermont Yankee elected to grout the penetration, with the grout acting as the fire seal material. Since both the Brand seal design and the grouting method differs from the original seal design, this design change was issued to address the installation of these new fire seals. This design change is limited to piping penetrations only.

Safety Evaluation Summary

The fire seals are passive components that have been evaluated to ensure that they impose no unacceptable loads on the equipment or components that they interface with. Fire barrier penetration seals are not involved in the initiation or mitigation of any accidents or accidents evaluated in the FSAR. The replacement seals have been designed to withstand the environment resulting from equipment failures that could affect the seals (e.g. 40 year normal operation radiation and high energy line breaks).

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.

i. EDCR 93-403 Diesel Generator Service Water Piping Modification was completed 10/19/93.

General Summary

This design change added a new diesel generator service water discharge header, including piping and associated valves; and modified existing discharge lines. This change also allowed for various alignments of the diesel generator discharge, i.e., both through the existing discharge path, both through the new discharge path, DG-1A through the new path and DG-1B through the old path, and each individually through either common discharge line. This line was added to provide additional flow margin to the diesel generators to ensure they are adequately cooled in the event of various accident scenarios.

Safety Evaluation Summary

The service water system and the diesel generators are used in mitigating the effects of a LOCA and a main steamline break. This design change enhances the ability of the diesel generators to perform their intended safety function. The service water system supports the RHR, core spray, RHRSW and RBCCW systems in performing several safety related functions, which continue to be available with the implementation 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.

j. EDCR 93-404 Reactor Vessel Water Level Reference Leg Back Fill Modification was completed 10/21/93.

General Summary

To satisfy a regulatory concern described in NRC Bulletin 93-03 and NRC Generic Letter 92-04 pertaining to the accumulation of non-condensible gases in the reactor vessel water level reference legs, this design change installed a back-fill modification. This design provides a source of relatively gas free water from the control rod drive (CRD) system, and provides a means to isolate the reference legs from the CRD system.

Tubing, fittings, and valves were installed to connect the CRD system to the reactor water level reference legs. A new rack and new valve station were added. Also installed were a means of venting any air entrained in the back fill system, and a means of monitoring the flow rate to the reference legs.

Safety Evaluation Summary

The reference legs and associated instrumentation are not accident initiators. The back fill system is manually initiated, receiving no safety inputs for automatic start or isolation. The back fill system minimizes the potential for non-condensible gases to saturate the reference legs. The reference legs are used with safety related instrumentation to initiate accident signals (RPS/ECCS/PCIS/PAM). This design change enhanced the ability of the reference legs to perform their intended safety functions.

The back fill system contains CRD system water, which is a potentially radioactive process fluid. However, the design is located entirely within the 252-foot and 280-foot elevations of the reactor building; therefore, this design change did not create any new effluent release paths.

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.

k. EDCR 93-406 Core Spray Suction Strainer Replacement was completed 10/14/93.

General Summary

This design change increased the net positive suction head (NPSH) margin for the core pray pumps during design basis accident conditions. During a postulated design basis loss of cooling accident (LOCA), debris from inside the primary containment can migrate to the strainers, partially clogging them. This would increase the pressure drop across the strainers and decrease available NPSH to the pumps. The old core spray strainers were of inadequate size to provide the required NPSH.

This design replaced the four suction strainers for the core spray system with strainers having greater surface area. There are two strainers on each suction tee for the two core spray loops. The design of the new strainers is similar to that of the old RHR system suction strainers.

Safety Evaluation Summary

The strainers are a passive component of the core spray system. The function, operation, and design bases of the core spray system were not changed with this design change. The probability of malfunction of the strainers, or any other part of the core spray system, is not changed by the implementation 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.

1. PDCR 92-001 Control Room HVAC Condenser (SACC-1B) Replacement was completed 3/22/93.

General Summary

This design change replaced one of the Control Room HVAC roof condensers, SACC-1B. Water had entered the old condenser due to a leak in the SCH-1 cooler, and could not be completely evacuated from the condenser; excessive moisture renders the unit unusable.

The replacement unit has a Motor Master Control Package instead of the previous Head Pressure Control Assembly. Both mechanisms perform the function of controlling system pressure by regulating air flow in response to the saturated condensing temperature. The new package performs this operation by varying the speed of the primary fan motor due to heat loads, while the previous assembly throttled the inlet air flow to the primary fan. This change did not require any electrical changes to the plant's electrical distribution system.

Because the new unit is longer, wider and heavier than the previous unit, the mounting frame was modified. The compressor discharge check valve was relocated and the refrigerant lines were reinsulated with a new type of insulation. Over pressure protection was installed to the SACC-1B refrigerant loop. TM 91-045, which installed two relief valves to the SACC-1A refrigeration loop, was converted to a permanent change. A seismic analysis was performed to ensure that the additional weight of the relief valves did not impact the seismic integrity of the refrigeration lines.

Safety Evaluation Summary

The Control Room HVAC system is defined in the Vermont Yankee Safety Class Manual as one of the Emergency Equipment Area Cooling Systems, the function of which is "to maintain temperatures within areas containing ECCS equipment within specified limits. These limits are determined such that reliable operation of the ECCS equipment can be maintained." The Emergency Equipment Area Cooling Systems are defined in the Safety Class Manual as Safety Class 3 for mechanical components and Safety Class Electrical for electrical components.

The Control Room HVAC system is not an initiator of any accidents. The components of this system that are affected by this design change are not utilized to mitigate the consequences of an accident or operational transient, nor do they have an effect on radiological consequence of an equipment malfunction. The changes that this PDCR make to the facility as it is described in the FSAR only illustrate the slightly reduced electrical load of SACC-1B and the installation of two relief valves on SACC-1A. A reduction in electrical loading is conservative and could not initiate an accident or operation transient of a different type.

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

m. PDCR 92-014 High Pressure Coolant Injection (HPCI) Loop Seal Temporary Modification (TM) Conversion was completed 10/19/93.

General Summary

This design change converted to permanent TM 91-025, which was installed to provide a drain path for condensate that collects in the HPCI Gland Seal Exhauster housing. Before the TM was installed, condensate collected in the housing and caused the exhauster breaker to trip on over current. The TM assured that the drain would not become a suction path by replacing a pipe plug in the bottom of the exhauster housing with a vented loop seal.

This change replaced the tubing with stainless steel tubing, installed a sight glass with the connections approximately 4" below the tee's centerline and 9" above the floor, installed a cage around the new tubing and sight glass, and routed the discharge of the tee to the HPCI Condensate pump drip tray.

Safety Evaluation Summary

The Gland Seal Exhauster operates when the HPCI turbine is operating. This design change is a mechanical change only and does not involve any of the logic that starts the HPCI Turbine. The loop seal is not electrically or mechanically connected to any other systems. The Gland Seal Exhauster is part of the HPCI system; the HPCI system is an accident mitigator.

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.

n. PDCR 92-017 Feedwater Heater E-4-1A and E-4-1B Replacement was completed 10/19/93.

General Summary

This design change replaced low pressure feedwater heaters E4-1A and E4-1B. The new heaters have stainless steel shells and shell side components to mitigate the problem of erosion/corrosion. Overall length was increased from 48'2" to 52'3" to include an internal flash chamber, which is a bolted flange removable section. Due to the extra length, the heater drain piping 14" HD-5A and 14" HD-5B geometry was modified at the drain inlet connection at the top of the flash chamber of each heater. Also modified were the pneumatic tubing and instrument cable to LCV-103-3A-1 and LCV-103-3B-1 and the respective heater drain inlet nozzle.

To allow removal/replacement of the 4A & B heaters, the condensate flow control station including FCV-102-4 and Line C-21 was removed and re-installed. FCV-102-4 was replaced with an equivalent flow control valve. The mounting of the process controls of the valve were modified.

Safety Evaluation Summary

The feedwater heaters, piping systems, instrument and valve components affected by this design change are classified as non-nuclear safety (NNS). The replacement heaters were designed in accordance with the original heater code requirements. The replacement improves system reliability, thus reducing the probability of malfunction which could create a Loss of Feedwater Heater event as described in FSAR chapter 14.

Level control changes to use the emergency drain valves to maintain heater drain system levels at power ranges to 80% will stabilize the extreme high and low fluctuating levels which can cause damage to heater internals. The operational arrangement changes do not incapacitate or remove from operation any of the associated LCVs or either of the drain pumps P3-1A/1B.

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.

o. PDCR 93-009 Alternate Cooling Deep Basin Suction Line Cleaning Access Port Installation was completed 10/19/93.

General Summary

This design change installed a cleaning access port on the service water alternate cooling system suction line (24"SW-17). The access port has a 24" tee fitting, a weld neck flange, and a blind flange. This change supports the pipe pig cleaning process. The 24" tee and flange provide an access/connection point for the polymer pig launching station. The access port was permanently installed so that cleaning can be performed in the future if required.

Safety Evaluation Summary

This modification has no detrimental effect on the service water system and alternate cooling system piping. The 24" tee and flange fittings are passive in nature. Additionally, pressure boundary materials were purchased to safety class 3 requirements. This ensures that his modification has no detrimental effect on the service water system, the alternate cooling system, the accident analyses, or the abnormal operational transient analysis.

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.

p. PDCR 93-010 Raising #5 Feedwater Heater Level Transmitters was completed 10/20/93.

General Summary

This design change raised the normal and emergency level transmitters (LT-103-5A/B-1/2) on feedwater heaters E-5-1A and 1B five inches. This change allows the emergency dump valve to operate over an acceptable span, and helps to ensure that the heaters will operate safety and effectively until the 1996 refueling outage.

Minor changes were made to the piping to allow the transmitters to be raised 5". The elbows above the transmitters were replaced by tees with the top legs vented by valves which are normally plugged; this will be used for future calibrations. Minor changes were made to the instrument air and air control lines. Short lengths of 1/4" and 3/8" copper tubing were used to hook up the normal level transmitters. The flex conduit going into the emergency level transmitter was of sufficient length to allow its routing to the new location with no new parts.

Safety Evaluation Summary

Loss of a feedwater heater is analyzed in the FSAR. This design change does not affect the assumptions or end results of the loss of feedwater heater transient analysis, and this change does not increase the chance of a loss of feedwater heater transient or create more severe consequences of the loss of feedwater heater transient.

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.

q. Temporary Modification 92-041 was installed 3/20/92 and is still open. As a result of subsequent review of this tempo ary modification during Cycle 16, a safety evaluation was written.

General Sum ary

This Tempora y Modification installed jumpers to bypass contacts of elays 76A-K14 and 76A-K24 in Control Room Panel 9-25 w ich were formerly used to isolate control room ventilat on upon Toxic Gas Monitor initiation. The Toxic Gas onitor System (TGMS) was no longer functional, as it was disabled by TM 92-01 with the intent to be permanently removed at a later date. The relays were considered part of the Toxic Gas Monitor system but the contacts were wired into a portion of the control room and cable vault heating and ventilation system.

Safety Evaluation Summary

FSAR transients were reviewed for applicability. The only transient potentially affected is Loss of Habitability of the Main Control Room. Criteria for this transient provides for the ability to bring the reactor to hot or cold shutdown by using controls and equipment outside the main control room. This ability is unaffected by this TM. Implementation of this TM does not affect the operation of any equipment or systems.

There was no change to circuit operation relative to control room ventilation or isolation of dampers by the fire protection system. The consequences of the installed jumper failing would be no different than those of the relay contacts that they replaced failing. This would result in dampers going to their fail safe (close) position.

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

r. <u>Temporary Modification 92-089</u> was installed 12/18/92 and is still open. As a result of subsequent review of this temporary modification during Cycle 16, a safety evaluation was written.

<u>General Summary</u>

This temporary modification installed an instrument root valve on PI-104-98A, which is located on the service water inlet line to RRU-8. The fittings and valves are identical to the outlet pressure gage arrangement. The pressure gages on the service water inlet and discharge on RRUs 5, 6, and 7 all have isolation valves. This modification makes the pressure indication and isolation on all four RRUs consistent. The purpose of this modification was to allow calibration of the pressure indicator without taking the RRU out of service.

<u>Safety Evaluation Summary</u>

The addition of a temporary isolation valve had no detrimental effect on the service water system and alternate cooling system piping. This modification had no detrimental effect on any equipment important to safety, nor on the seismic capability of this system.

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

s. Temporary Modification 93-010 was installed 3/22/93 and removed 12/7/93.

General Summary

This temporary modification installed four pressure gages to connections in the service water system. The gages were required to monitor service water pressure during diesel generator surveillance testing. Two of the gages were installed in the NNS portion of the service water system. One of these was in the line to the rad monitor in the lower level turbine building near the rad monitor. The other was in the section of the alternate cooling discharge cooling line from the IR air compressors near the common diesel discharge cooling line at El. 232'6" in the turbine building.

The two other gages were installed on the diesel generator cooling water outlet (one per diesel) 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 NNS pressure gages at these test points was within the design bases of the service water system.

Safety Evaluation Summary

Failure of the test connections or gages would have had no adverse effect on the capability of equipment important to safety. The installation had no detrimental effect on the service water system, the alternate cooling system, the accident analyses, or the abnormal operations transient analysis. Restrictions were established to provide protection to ensure that all activities were performed safely to ensure that the system design bases were met.

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

t. Temporary Modification 93-026 was installed 9/2/93 and removed 9/28/93.

General Summary

This temporary modification installed temporary lead shielding on portions of the recirculation system piping. The temporary shielding was required to reduce radiation exposure to personnel during the 1993 refueling outage. The recirculation system pumps were off during the period the shielding was installed.

Safety Evaluation Summary

The addition of this shielding had no effect on the recirculation system's ability to perform its function. The addition of this shielding had no detrimental effect on the seismic capability of the recirculation system. During the time this shielding was installed, the reactor was shutdown and depressurized. This modification had no detrimental effect on any equipment important to safety.

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

u. Temporary Modification 93-028 was installed 9/10/93 and removed 9/16/93.

General Summary

This temporary modification installed temporary lead shielding on portions of the reactor water cleanup system piping. The affected piping was in the reactor water clean up (RWCU) suction piping, downstream on the first isolation valve (V-15) between the valve and the drywell penetration. The temporary shielding was required to reduce radiation exposure to personnel during the 1993 refueling outage. This modification had no effect on system function. The only possible effect was on the system supports and system response to a seismic event.

Safety Evaluation Summary

The addition of temporary shielding on the reactor water cleanup piping had no detrimental effect on the reactor water cleanup piping. This modification had no detrimental effect on any equipment important to safety. While the shielding was in place, the piping stresses remained within allowable design limits with the applicable shielding weight restrictions during all modes of operation.

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

v. Temporary Modification 93-032 was installed 8/31/93 and removed 10/19/93.

General Summary

This temporary modification installed temporary lead shielding on and around portions of the residual heat removal (RHR) system piping and RHR heat exchanger. The temporary shielding was required to reduce radiation exposure to personnel to support the modifications to the RHR service water (SW) system per EDCR 92-404. The temporary modification had no effect on the system function; the only possible effect was on the system supports and system response to a seismic event.

Safety Evaluation Summary

The shielding was only in place when the RHR and RHRSW systems were out of service. While the shielding was in place, the piping stresses and floor loading remained within allowable design limits with the applicable shielding weight restrictions during all modes of operation. If the shielding had fallen off during a seismic event, it would have had no detrimental impact to other systems or equipment in the area.

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

w. __Temporary Modification 93-034__ was installed 9/22/93 and removed 10/2/93.

__General Summary__

This temporary modification installed temporary lead shielding on and around portions of the residual heat removal (RHR) system piping and RHR heat exchanger. The temporary shielding was required to reduce radiation exposure to personnel to support the modification to the RHR Service Water (RHRSW) system per EDCR 92-404.

__Safety Evaluation Summary__

The shielding was only in place when the RHR and RHRSW systems were out of service. The addition of shielding in accordance with this temporary modification did not adversely affect the ability of the piping to withstand a seismic event. This modification had no detrimental effect on any equipment important to safety.

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

x. __Temporary Modification 93-037__ was installed 8/25/93 and removed 10/20/93.

__General Summary__

This temporary modification installed temporary lead brick shielding on the Refuel Floor (el. 345) to shield the Beta Scintillation Detectors which are used for vacuum fuel sipping. The detectors and associated equipment are not important to safety. The temporary shielding was required to reduce background radiation at the detectors.

__Safety Evaluation Summary__

The addition of temporary shielding on the refuel floor had no detrimental effect on the reactor building/ secondary containment. This modification had no detrimental effect on any equipment important to safety. The addition of shielding in accordance with this temporary modification does not adversely affect the ability of the refuel floor and reactor building 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 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.

y. Temporary Modification 93-039 was installed 9/10/93 and removed 9/23/93.

General Summary

This Temporary Modification installed a jumper to defeat the interlock preventing the movement of the refueling bridge in the reverse (northerly) direction over the core when the reactor mode switch is not in Refuel. Without the jumper installed during plant shutdown, refueling platform travel toward the core is prevented when the mode switch is not in Refuel. This Temporary Modification allowed in-vessel inspections to be performed from the refueling bridge while the mode switch was in Shutdown.

Safety Evaluation Summary

A failure of the refueling interlocks is not an initiator for any accident. The only potentially applicable Design Basis Accident is the Refueling Accident. The initiating cause to this event is the dropping of a fuel bundle. This TM does not increase the chances of a fuel bundle being dropped; fuel will not be moved during the time that this TM is in effect. The refueling interlocks do not perform a mitigating function for any accidents evaluated in the FSAR. The failure mode for failure of the installed jumper would have been to re-instate the refueling interlock which prevents movement of the refueling bridge when the mode switch is not in Refuel; this does not result in any safety concerns.

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

z. <u>Temporary Modification 93-040</u> was installed 9/16/93 and removed 9/29/93.

<u>General Summary</u>

This Temporary Modification removed the discharge flexible connection on the "B" Emergency Diesel Generator (EDG), and installed a reducing elbow and fire hose for the alternate water supply from fire water. This hose was routed to a temporary sleeve installed in a hole drilled through the east wall of the "B" EDG room. The fire hose was connected to the sleeve with a grab sample connection and was routed to the nearest storm drain. This allowed the "B" EDG to remain functional while the common discharge line was modified to incorporate new isolation valves and add an additional discharge line per EDCR 93-403.

<u>Safety Evaluation Summary</u>

During the time this TM was installed, the reactor was in shutdown. The service water system was evaluated to function as intended, thus ensuring that ECCS/support systems would function as intended, and no increase in radioactive material release would have occurred from overheating of equipment/piping/fuel (and subsequent failure) due to lack of SW. No ECCS equipment was required to be operable during the time period this TM was relied upon to make the "B" EDG functional. During the time frame that this TM was installed, workers were performing work in the "A" diesel generator room and in the hallway outside the EDG rooms; any possible flooding incident would have been immediately noticed, and operator action initiated to shut the manual isolation valve and terminate the flooding. Also, any leakage into the building would have been detected immediately by the floor drain and equipment drain sump alarms.

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

aa. <u>Temporary Modification 93-042</u> was installed 9/10/93 and removed 10/11/93.

<u>General Summary</u>

This Temporary Modification provided a temporary path for the service water (SW) discharge from the reactor building closed cooling water (RBCCW) heat exchanger to the roof drain, to provide cooling when the service water discharge main was drained to facilitate service water related design changes and maintenance work. A loop seal provided protection for secondary containment.

FSAR states that "a process radiation monitor is located in the station service water discharge header" to monitor the system discharge before it enters the river. For this configuration, a grab sample was taken every 24 hours per the Technical Specifications.

Safety Evaluation Summary

During the time this TM was installed and used, the reactor was in Shutdown/Refuel. This TM resulted in an increase in the cavity/fuel pool temperature; the rate and amount of change had been evaluated and found to be acceptable by YNSD. An administrative time limit of 8 hours had been established for the Service Water return header to be out of service to replace the V10-192A/B valves. Thus, sufficient margin existed to replace the valves without significant reactor cavity/fuel pool heat up.

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

bb. Temporary Modification 93-050 was installed 9/3/93 and removed 9/23/93.

General Summary

This Temporary Modification installed a temporary power feed to Security Power Panel SSD1 from AC-DP-5, circuit #21, to maintain security access control during BUS 9 maintenance.

Safety Evaluation Summary

The only equipment important to safety that could have been affected by this change were the electrical loads upstream of the AD-DP-5, circuit 21 breaker. This breaker forms the break between the Safety Class Electrical Equipment and the Non-Nuclear Safety Security Lighting panel. This isolation methodology was consistent with the VY Separation Criteria. The change in loading to the B EDG did not create a new failure mode for the B EDG.

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

cc. Temporary Modification 93-064 was installed 10/17/93 and removed 10/22/93.

General Summary

This temporary modification electrically disconnected the normally closed contacts (1AT1-1AT2) of Reactor Protection System (RPS) relays 5A-K27A,B in Control Room Panel (CRP) 9-16. These contacts were originally wired into the select rod insert circuitry as part of the RPS. These normally closed relay contacts were left wired in place in the select rod neutral bus following implementation of design change EDCR 74-19. This was subsequently determined to have a potential effect on individual control rod scram timing and possible false indication of longer scram times.

Safety Evaluation Summary

Although the RPS performs a mitigating function for accidents as evaluated in the FSAR, the portion of the RPS to which the subject relays were associated, generator load reject scram, does not perform a mitigating function for any accidents evaluated in the FSAR. The relay contacts which were removed from the select rod neutral bus of the rod scram solenoids had no effect on the solenoids accomplishing their scram functions under all design bases. The only change in equipment operation is that when an individual rod scram and select switch is moved to the "select" position, that selected rod will scram, which is more consistent with the original design basis of this equipment.

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

dd. Special Test Procedure 93-001 Reactor Vessel Time to Boil was completed 10/13/93.

General Summary

This Special Test Procedure was initiated to take temperature measurements during heatup conditions in the reactor cavity, spent fuel pool, and dryer separator pit. The temperature measurements were used to validate assumptions contained in the Time to Boil (TTB) model related to thermal stratification of the water adjacent to the reactor cavity in the dryer separator pit and the spent fuel pool. The amount of thermal stratification affects the heatup in the reactor cavity if cooling is lost and subsequently the time to boil.

Safety Evaluation Summary

Temperature measuring mandrels were installed in the reactor cavity, spent fuel pool, and dryer separator pit in such a way that they did not interfere with other ongoing work or the operability of any installed plant equipment. The actions taken for this special test procedure did not degrade the level of confidence in the integrity of any barriers, nor affect the proximity to any safety limits.

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 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.

ee. Special Test Procedure 93-003 In-Situ Differential Pressure Testing of Valve CS-26A was completed 8/18/93.

General Summary

NRC Generic Letter 89-10, "Safety-Related Motor Operated Valve 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. In response, VYNPC developed the Vermont Yankee Motor Operated Valve Program Plan. This plan established design basis review, testing, inspection, and maintenance requirements for certain Safety Class 1, 2, and 3 MOVs. Special Test Procedure 93-003 gathered information on Core Spray full flow test valve CS-26A performance when operated under specific test conditions of differential pressure and flow rate.

Safety Evaluation Summary

The Core Spray System is an Emergency Core Cooling system (ECCS) which, in conjunction with the primary and secondary containments, has the safety objective of limiting the release of radioactive materials to the environs following a Loss of Coolant Accident (LOCA). Prior to removal of the "A" Core Spray Subsystem from service, alternate testing of the active components of the "B" Core Spray Subsystem, and "A" and "B" LPCI Subsystems and the "A" and "B" Emergency Diesel Generators were performed to ensure operability of these Subsystems during the time period when the "A" Core Spray Subsystem was considered inoperable. This ensured provision of sufficient capability for cooling over the entire spectrum of break sizes.

The Core Spray Systems are not accident or transient initiators. Since the Core spray systems are normally in standby mode, rendering one loop inoperable did not affect plant operations.

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 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.

ff. Special Test Procedure 93-004 In-Situ Differential Pressure Testing of Valve CS-26B was complete 08/17/93.

General Summary

Special Test Procedure 93-004 was conducted to address NRC Generic Letter 89-10 (see Special Test Procedure 93-003, General Summary, above). Valve CS-26B is the full flow test valve for Core Spray Pump P46-1b. Valve CS-26B is normally closed and has the safety function to close, when open, upon receipt of a Core Spray System initiation signal in order to properly direct Core Spray System flow to the reactor pressure vessel. This test sequence provided data on the effects of multiple strokes on valve performance and correlations between static and dynamic test results.

Safety Evaluation Summary

The Core Spray System is an Emergency Core Cooling system (ECCS) which, in conjunction with the primary and secondary containments, has the safety objective of limiting the release of radioactive materials to the environs following a Loss of Coolant Accident (LOCA). Prior to removal of the "B" Core Spray Subsystem from service, alternate testing of the active components of the "A" Core Spray Subsystem, and "A" and "B" LPCI Subsystems and the "A" and "B" Emergency Diesel Generators were performed to ensure operability of these Subsystems during the time period when the "B" Core Spray Subsystem was considered inoperable. This ensured provision of sufficient capability for cooling over the entire spectrum of break sizes.

The Core Spray Systems are not accident or transient initiators. Since the Core spray systems are normally in standby mode, rendering one loop inoperable did not affect plant operations.

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 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.

gg. <u>Special Test Procedure 93-005 Final Testing and Adjusting Procedure for the Alterrex Excitation System</u> was complete 10/26/93.

<u>General Summary</u>

During the 1993 refueling outage, the Alterrex Excitation Control system underwent a complete readjustment and enhancement to restore and improve the exciter's voltage regulation capability. This special test procedure conducted final functional testing and adjustment of the Alterrex Excitation Control System. This was a post-maintenance dynamic test to allow fine tuning.

<u>Safety Evaluation Summary</u>

The Alterrex Excitation System is not relied upon to mitigate any accident, and an Excitation System failure does not affect any safety equipment. The exciter testing was performed with the generator disconnected from the grid and running at the synchronous speed, 1800 rpm. This condition is no different from that existing during startup, prior to the generator synchronization to the grid.

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

hh. Special Test Procedure 93-006 Primary Containment Temperature and Relative Humidity Surveys was complete 10/14/93.

General Summary

This Special Test Procedure documented the data obtained during the Primary Containment Drybulb Temperature and Relative Humidity Surveys, and maintained the proper status of plant components.

Safety Evaluation Summary

Prior to the primary containment drybulb temperature and relative humidity surveys, the valve and component lineups placed PCIS valves in the isolated positions, so active operation was not required. The surveys were performed using local, handheld instruments which were passive with respect to the equipment located within and outside the primary containment. The positions of valves and components within the CRD, RWCU, Shutdown Cooling, and Emergency Core Cooling Systems were not changed, nor was their operation affected in any way.

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

ii. Special Test Procedure 93-007 In-Situ Differential Pressure Testing of Valve SW-20 was complete 10/18/93.

General Summary

Special Test Procedure 93-009 was conducted to address NRC Generic Letter 89-10 (see Special Test Procedure 93-003, General Summary, above). 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.

Valve SW-20 is the isolation valve for the non-essential Turbine Building loads of the Service Water System. Valve SW-20 is normally open and has the safety function to be closed by manual actuation to isolate the non-essential systems and equipment in the Service Water System, thus providing adequate cooling water flow to essential services.

Safety Evaluation Summary

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

jj. Special Test Procedure 93-008 In-Situ Differential Pressure Testing of Valves RHR-57 and RHR-66 was complete 10/15/93.

General Summary

Special Test Procedure 93-008 was conducted to address NRC Generic Letter 89-10 (see Special Test Procedure 93-003, General Summary, above). 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.

Valves RHR-57 and RHR-66 are the isolation valves for the Residual Heat Removal (RHR) system discharge to the Waste Collector Tank, TK-9-1A. Both valves are normally closed are manually opened to transfer Torus water to the Waste Collector Tank. Both valves have the safety function to close, when open, upon receipt of a PCIS Group II signal in order to properly direct RHR System flow for core and primary containment cooling and to assure primary containment integrity.

Safety Evaluation Summary

With the reactor in cold shutdown condition and the drywell and RPV heads removed, primary containment integrity was not required; the only transient or accident parameter variations related to the testing which could have occurred, were coolant inventory decreases. The testing involved the transfer of water from the torus to radwaste; the reactor vessel was not affected and the remaining ECCS subsystems and components remained available. To preclude inadvertent coolant inventory decreases, the RHR valves in lines connected to the primary coolant system were verified to be shut prior to testing. If a decrease in water level had occurred, the leak path could have been manually closed or would automatically close upon receipt of a reactor low water level signal. If the water level continued to decrease such that adequate core cooling was no longer assured, the remaining ECCS subsystems provide 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 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.

kk. Special Test Procedure 93-009 In-Situ Differential Pressure Testing of Valves CS-11A and CS-12A was complete 10/14/93.

General Summary

Special Test Procedure 93-009 was conducted to address NRC Generic Letter 89-10 (see Special Test Procedure 93-003, General Summary, above). 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.

Valve CS-11A and CS-12A are the injection isolation valves for the "A" Core Spray Subsystem. Valve CS-11A is normally open and valve CS-12A is normally closed. Both valves have the safety function to open, when closed, upon receipt of a Core Spray System initiation signal in order to properly direct Core Spray System flow to the reactor pressure vessel. The tests provided data on the effects of multiple strokes on valve performance and correlations between static and dynamic test results.

Safety Evaluation Summary

The Core Spray System is an Emergency Core Cooling system (ECCS) which, in conjunction with the primary and secondary containments, has the safety objective of limiting the release of radioactive materials to the environs following a Loss of Coolant Accident (LOCA). With the reactor in cold shutdown condition and the drywell and rpv heads removed, the only transient or accident parameter variations which could have occurred related to the testing were coolant inventory increases or decreases. Prior to the start of testing and removal of the "A" Core Spray Subsystem from service, alternate testing of the active components of the "B" Core spray Subsystem was performed to ensure operability during the time period when the "A" Core Spray Subsystem 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 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. Service Water Lineup Change went into effect 10/19/94.

General Summary

This valve lineup change revised the normal standby lineup of the service water system. Manually operated gate valves V70-16A and V70-16B are located in a series with only a 3/4" normally closed manual vent valve between the valves. During alternate cooling operation, both of these valves are required to be open. Functionally, only one of the valves is required; however, the second valve was added for isolation capability due to the long length of piping from the cooling tower basin to the reactor building. Previously, valve V70-16A was open and valve V70-16B closed. After this lineup change, both valves V70-16A and V70-16B are closed. The purpose is to prevent or minimize leakage in suction piping 24"SW-17 to minimize growth on the internal piping surfaces.

Safety Evaluation Summary

Initiation of alternate cooling previously required local operator actions to open the V70-16B valve which is located in the torus area. Operator action was also required to open V70-17 which is in the same pit as 16A. Closure of both the 16A and 16B valves may result in slightly longer initiation times; however, this does not have an adverse impact since sufficient time is available to align the valves. This lineup change had no detrimental effect on the functioning of the service water system or alternate cooling system, nor on any equipment important to safety. The service water system and alternate cooling system are not accident initiators.

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

mm. <u>Burning Non-Contaminated Waste Oil in the Containment Access Building</u>

<u>General Summary</u>

Vermont Yankee burns non-contaminated waste oil in the Waste Oil Burner located in the Containment Access Building (CAB). Operation of the waste oil burner is conducted in full compliance with all Federal and State requirements relating to the burning of waste oil. The burning of waste oil does not directly or indirectly increase the radioactive material release from any of the four radioactive material release barriers.

<u>Safety Evaluation Summary</u>

The location of the waste oil burner is approximately 50 feet from the outer railroad airlock door, which comprises part of the secondary containment. This door and its seal are Safety Class 2. All other systems important to safety are isolated from the waste oil burner and are not affected by its operation. The possibility that a fire in the waste oil burner might damage the outer railroad lock door inflatable seal is extremely small and no impact on the capability to maintain secondary containment will likely occur. Any failure of the waste oil burner would not cause or threaten failure of the four radioactive material barriers and nuclear safety/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. The burning of non-contaminated waste oil does 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.

nn. Cycle 17 Core Operating Limits Report

General Summary

This report provided the cycle-specific limits for operation of the Vermont Yankee Power Station in Cycle 17. It included the limits for the maximum average planar linear heat generation rate, maximum linear heat generation rate, and minimum critical power ratio.

The Cycle 17 core contains 240 irradiated GE-9B bundles and 128 fresh GE-9B bundles, manufactured by General Electric. The average initial enrichment for the irradiated bundles is 3.11 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 17 also contains 86 irradiated control rods, 3 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 17 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 LOCA/ECCS (Loss of Cooling Accident/Emergency Core Cooling System) analysis, assuming the operating limits in the COLR (Core Operating Limits Report), shows that Cycle 17 meets the acceptance criteria of 10CFR50.46. Therefore, the operation of cycle 17 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 17 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.

oo. Bundle LYV667 Replacement

General Summary

During fuel moves for Cycle 17, the bail handle on bundle LYV667 was damaged. This bundle was intended to be inserted on the edge of the Cycle 17 core in location 37-38. Bundle LYV686, which was slated for discharge, was used as a replacement for Bundle LYV667 because it was the same fuel type and had a similar exposure.

Safety Evaluation Summary

The replacement bundle and the discharged bundle differ only in the amount of exposure: LYV686 had a higher burnup than LYV667 by 308 MWd/St. This increase in exposure did not increase the probability of an accident or transient because the exposure was within the exposure range of the mechanical and CPAR analyses. The core change did not increase the probability of a thermal-hydraulic instability. The replacement bundle, LYV686, was mechanically the same as the replaced bundle, LYV667. The core change did not affect the operation of any other equipment important to safety.

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

pp. Use of Reflective 3D Monicore began on 8/6/93.

General Summary

The indexer motor on the #1 Traversing In-Core Probe (TIP) failed in 1992, rendering the #1 TIP machine inoperable. Continuing safe plant operation was possible with the 3D Monicore core monitoring code. However, the thermal limits as calculated by 3D

Monicore diverged conservatively from the Simulate-3 code. Simulate-3 is used as the Vermont Yankee licensing code for reload analysis. It was determined that the 3D Monicore did not handle missing TIP data as well as had been previously thought; and the thermal limits core-wide were skewed, or asymmetric.

Reflective 3D Monicore is a set of instructions for the 3D Monicore software, that tells 3D Monicore to use the actual LPRM data where instrumentation exists; but in the case of the LPRM pseudo-string locations, the code is now told to reflect the data from the physical string location into the pseudo-string location. This is a more accurate methodology, because the model is told to fit its results to the actual data instead of relying solely on the diffusion model results.

Safety Evaluation Summary

Reflective 3D Monicore is a steady-state core performance monitoring code. It plays no role in the sequence-of-events leading to a previously analyzed accident, nor does it play any role in the mitigation of a previously analyzed accident. Reflective 3D Monicore interfaces with no plant equipment important to safety. This change to Reflective 3D Monicore did not affect any plant hardware, plant design, safety limit settings, or plant systems operation. Use of Reflective 3D Monicore reduces core-wide asymmetries and reduces the uncertainties in the calculated operating margins relative to the core performance analysis report's licensing analysis.

There was no increase in the probability of occurrence or consequences of an accident or malfunction as previously evaluated in the FSAR. The decision to use reflective 3D Monicore 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.