VERMONT YANKEE CYCLE 19 OPERATING REPORT

Between November 2, 1996 and June 1, 1998 Vermont Yankee implemented a number of changes requiring evaluation in accordance with 10CFR50.59(a)(2). This report includes the safety evaluation summaries for twenty Engineering Design Change Requests (EDCRs), thirteen Temporary Modifications (TMs), four Basis for Maintaining Operation (BMOs), one revised Basis for Maintaining Operation (BMO), eleven Special Test Procedures (STPs), seven procedure changes, four Final Safety Analysis Report (FSAR) changes, three Safety Classification Changes, one Setpoint change, one Technical Specification Bases change and the following additional subjects: Low Level Waste Storage in Sealand Containers, Main Steam Line Valve Leak, Removal of Reactor Building Fresh Air Supply Roll Filter Medium During Winter, Downgrade of Thermal Performance Requirements for RRU's 5 and 6, Dosimetry Office Move – E-Plan and OP 3504, Tornado Missile Comparison for Sealand Containers, Change to Residual Heat Removal (RHR), RHR Service Water Maximum Flow Rates and Clarification of Separation Criteria for Instrument and Control Cables. There were no Safety Relief Valve challenges during this operating cycle.

I. Changes in Facility Design

A. 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).

EDCR 97-401, EDG Modification

General Summary

This design change provided the engineering details required to improve the Emergency Diesel Generator (EDG) mechanical availability and reliability. It involved the Diesel Engine, Diesel Jacket and Lube Oil Cooling System and the Diesel Starting Air Systems. TM's 94-18 and 94-19, which installed 3/8" Whitey ball valves on the No. 14 bearing and Governor Servo boosters were also incorporated into this design change. These ball valves allowed for rolling over of the engine to remove the residual oil in the upper pistons.

Safety Evaluation Summary (SE 98-005)

This design change increased the availability of the EDGs by installing equipment that eliminated having to make temporary modifications for testing purposes. This design change also increased the starting air margin by enlarging the receiver tanks.

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 a reasonable assurance that the health and safety of the public was not endangered.

EDCR 98-402, HPCI/RCIC Vacuum Breaker

General Summary

The intent of EDCR 98-402 was to eliminate the potential problems associated with an elevated Torus pressure by reconfiguring the High Pressure Coolant Injection (HPCI) and Reactor Core Isolation Cooling (RCIC) exhaust line vacuum breakers. This new configuration now draws air/nitrogen from the Torus air space, thus equalizing the pressure differential between the Torus and the HPCI/RCIC exhaust line vacuum breakers.

The scope of work included replacing the HPCI exhaust vacuum breaker line with 2" and 3" diameter pipe from the 18" torus/drywell main vacuum breaker line and replacing the RCIC exhaust vacuum breaker line with 1" and 2" diameter piping from the 18" torus/drywell main vacuum breaker line. The HPCI-exhaust vacuum breaker line and the RCIC-exhaust vacuum line were removed along with the HPCI-exhaust vacuum breaker line valves. The remaining lines were abandoned in place and modified to ensure seismic adequacy.

Safety Evaluation Summary (SE 98-027)

The activity in EDCR 98-402 did not increase 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 a reasonable assurance that the health and safety of the public was not endangered.

EDCR 97-403, Stack Gas III Replacement

General Summary

This EDCR was written to replace the plant High Range Stack Monitor (Stack Gas III) system and its lightning protection. This new system provides diagnostic capabilities, equal range coverage and equivalent or better sensitivity while requiring less operator intervention. The new system meets NRC Post Three Mile Island accident requirements to measure noble gas effluent, NUREG 0578 and NUREG 0737. Additionally it meets the requirements of Regulatory Guide 1.97.

Safety Evaluation Summary (SE 97-035)

The High Range Stack Monitor system does not provide any accident mitigating functions and is not considered an accident initiator. No new malfunctions were introduced as the new equipment functions in the same manner as the equipment that was replaced.

As the High Range Stack Monitor system does not provide a safety function associated with a design basis accident, the margin to safety is not reduced.

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

EDCR 97-404, MOV Electrical and Pressure Locking Modifications

General Summary

This design change addresses Generic Letter 89-10, "Safety-Related Motor Operated Valve (MOV) Testing and Surveillance" to ensure that the MOV's will perform their safety function for design basis conditions. The scope includes standardized wiring, pressure locking prevention, replaces some Environmentally Qualified (EQ) Motor Control Center (MCC) cubicles and implements some outage efficiency improvements.

This change does not modify the pressure boundary. All electrical components are Safety Class Electrical and are Seismic Class I. There are no required changes to Technical Specifications and only minor changes to the FSAR. Installation was completed during LCO periods and the 1998 refueling outage.

Safety Evaluation Summary (SE 97-024)

The accidents described in the FSAR have been reviewed and it was determined that the modified valves are not accident initiators, nor can they initiate any operational transients. The modifications do not change or prevent any automatic operation of the equipment required to mitigate the consequences of an accident nor will they affect system operational requirements. Therefore, the margin of safety, as defined in the basis for any Technical Specification, is not reduced.

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

EDCR 97-405, Chemical Containment Area Modifications

General Summary

This design change made modifications to the sulfuric acid storage tank, the sulfuric acid and sodium hypochlorite secondary containment berms, the level indication for the sodium Hypochlorite, Sulfuric Acid, and Sodium Bromide tanks, and the auxiliary chemical addition system.

The 5000 gallon sulfuric acid storage was replaced with a 1500 gallon tank of a different material due to the original tank condition and smaller capacity requirements. The sodium Hypochlorite tank berm was discovered to be too small for the capacity of the tank. Subsequently this modification limits the level in the tank, indicated by an alarm, to just below the berm capacity. The berm for the sodium hypochlorite tank was replaced and coated with an epoxy paint that is resistant to this chemical. The existing berm for the Sulfuric Acid tank was enhanced with a polypropylene liner. The Sodium Bromide tank berm was not modified.

The pumping system for chemical addition was modified such that one pump was replaced with a pump that more accurately pumps at the new flow rate.

Safety Evaluation Summary (SE 97-015)

This system is not an initiator of any design basis accident or any abnormal operational transients. This change provides no safety function and does not support any safety related equipment. Therefore, the margin to safety is not reduced.

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

EDCR 97-406, FCV 2-39 and 2-40 Position Indication Modification

General Summary

This design change was written to modify the Recirculation Loop sample line isolation valves FCV 2-39 and 2-40 to provide a means of monitoring the status of these valves to ensure compliance with the Vermont Yankee Reg. Guide 1.97 submittal and to facilitate the In-Service Testing Program.

These valves are air-operated fail closed valves which provide an alternate reactor coolant sample path in the event the normal sample path through the Reactor Water Cleanup System is unavailable. They are Primary Containment Isolation Valves and are closed automatically upon a Group I isolation signal.

In addition to the position indication components added to the valves, power to the valves was separated to provide more diverse sources from redundant busses.

Safety Evaluation Summary (SE 97-030)

These valves are not considered accident initiators or initiators of any abnormal operational transients. The mechanical and seismic integrity of the valves and process lines are affected by this installation as added weight to the valves. Additional supports have been added to ensure seismic and mechanical loads are within the analysis.

Operation of this equipment was not changed by this installation. The installation provides control room operators with a more direct and reliable indication of valve position.

Modifications to these valves do not change their automatic PCIS function; therefore, the margin of safety has not changed.

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.

EDCR 97-407, Conversion of TM 95-049, Addition of Protection to K-186 Line

General Summary

This design change will convert TM 95-049 to a permanent change. TM 95-049 added protective relaying to the secondary trip circuit of the K-186 line to actuate on Zone 1 directional overcurrent faults and initiate a transfer trip signal to the Keene and Vernon Road Substations.

The modifications made by TM 95-049 were Non-Nuclear Safety (NNS) and had no affect on any safety systems required for safe shutdown of the reactor. The TM did not affect design basis calculations.

Safety Evaluation Summary (SE 98-003)

Prior to making TM 95-049 a permanent change, the TM package was re-reviewed to ensure that the intent of the design change was met and to ensure that the modification had been correctly implemented. The review verified that each step identified in the TM was signed off as complete and that the appropriate tests and inspections had been performed and documented.

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

EDCR 97-408, Underground Storage Tank Upgrades

General Summary

This design change replaced the underground gasoline and diesel fuel storage tanks with new double-wall fiberglass tanks and upgraded the appropriate subsystems. This was completed to comply with the State of Vermont regulations regarding underground storage tanks.

This equipment is non-nuclear safety. There are no licensing or safety implications. No changes to the FSAR were required and this design change does not require any changes to the Technical Specifications.

Safety Evaluation Summary (SE 97-025)

The underground storage tanks are not described in the VY design basis and as such have no impact on the existing VY design basis. The design change is considered NNS. This design change will not affect any safety systems required for the safe shutdown of the reactor.

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.

EDCR 97-409, Structural Enhancements and Venting Additions to Emergency Diesel Generator Rooms to Provide Better Tornado Protection

General Summary

This design change provides additional structural support for the Emergency Diesel Generator and Daytank room walls and new Safety Class 3 tornado relief dampers.

The modifications implemented by this change are comprised of structural hardware that augments the existing masonry wall capacity. Additionally, it provides dedicated tornado venting capability by supplying relief dampers and openings. Structural modifications and door gaps are passive changes while the relief dampers are self-contained and self-actuated components. The structural modifications strengthen the walls to resist design basis loads and the dampers/door gaps ensure that the tornado pressure differential forces remain below the walls' capacity.

These changes do not adversely affect the structural or functional integrity, operation, safety objective or safety design basis of the Diesel Generator System.

Safety Evaluation Summary (SE 97-005)

These changes are not initiators of any analyzed accidents or operational transients and function to protect the equipment that they surround so that the equipment will be available to perform their required safety functions. This change does not interfere with or change any functions of the Emergency 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 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.

EDCR 97-410, Control Rod Drive Scram Discharge Volume Vent and Drain Valves Replacement

General Summary

This design change will replace the air-operated vent valve and the two air-operated drain valves and their actuators in each Scram Discharge Volume. The new ball-type valves replaced the installed Y-Globe valves. Prior problems with the valves and a lack of vendor support dictated that the new valves would be of a different type and manufacturer. These valves function to vent and drain the Scram Discharge Volume following a reactor scram that isolates the volume to retain reactor vessel water until the scram can be reset. During the time the valves are open they function as PCIS valves.

The operation and function of the system remains the same, such that no Technical Specification changes were required.

Safety Evaluation Summary (SE 97-034)

The Scram Discharge Volume vent and drain valves are not accident initiators and any malfunction would not adversely impact any of the analyzed accidents. This design change did not change any safety function and only replaced existing equipment.

This design change doesn't affect any safety limits and therefore the margin of safety was not reduced.

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

EDCR 97-412, Reactor Building Fire Zone RB-3 and RB-4

General Summary

This design change was written to provide wide-area fire detection in the Reactor Building Fire Zones RB-3 and RB-4 to facilitate use of alternative shutdown methods to achieve safe shutdown. Additionally, it provided physical separation in the form of a steel barrier for divisional electrical circuits to ensure that at least one train of equipment remains available to achieve safe shutdown. Implementation of this design change ensures safe shutdown capability for a fire in the Reactor Building and for an electrical fault in divisional raceways.

Safety Evaluation Summary (SE 97-021)

The automatic operation of plant equipment affected by this EDCR has not changed. The changes to the fire detection systems do not change any description or intent of the FSAR. The cable separation enhancement does not change any operation of the Auxiliary Power System as defined in the FSAR.

The affected systems, fire detection and auxiliary power systems, are not accident initiators. The Station Auxiliary Power could be involved in some of the abnormal operational transients but this design change does not change the operation of any of these systems. No new failure modes are introduced by this design change. As there are no changes to the operation of this equipment, there is no change in any 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 FSAR, and there is reasonable assurance that the health and safety of the public was not endangered.

EDCR 97-414, Advanced Off Gas (AOG) Modifications

General Summary

This design change addressed maintenance and operational reliability concerns with the electrical instrument power distribution and certain electrical components of the AOG system. The scope of work included reconfiguration of CRP 9-50 instrumentation power, elimination of annunciator nuisance alarms, installation of new hydrogen monitors and analyzers and replacement of the SJAE suction isolation valve (PCV-OG-516A & B) automatic controllers with manual indicating controllers. New reset buttons, power and temperature switches were also installed. All components are classified as Non-Nuclear Safety (NNS) with the exception of repairs and modifications made to the outer shell and frame of the Main Control Board.

Safety Evaluation Summary (SE 98-002)

The AOG System is a Radioactive Gaseous Waste Treatment System. This design change implements modifications to the AOG System to improve system reliability and operations and to enhance the performance of maintenance activities.

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 a reasonable assurance that the health and safety of the public was not endangered.

EDCR 97-415, NES Fuel Rack Installation

General Summary

This design change was written to install an NES fuel rack in the Southeast corner of the Spent Fuel Pool. This spent fuel rack will store irradiated fuel assemblies in the Spent Fuel Pool until the fuel assemblies can be shipped to an off-site repository. The design service life for this rack is 40 years. The rack will not be physically installed unless needed for a full core off-load.

Implementation of this design change will require removal of a Fuel Pool Cooling line, associated supports from the Spent Fuel Pool and completion of the fuel pool cleanup program in order to provide sufficient space in the Southeast corner.

Safety Evaluation Summary (SE 98-004)

Materials, general arrangements of the NES 15 x 12 rack and appropriate dimensions provided by this modification are sufficient to assure that the final design will conform to the design basis with adequate 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 FSAR, and there is a reasonable assurance that the health and safety of the public was not endangered.

EDCR 97-416 TIP Purge Modifications

General Summary

This design modified the TIP Purge line and brought it into conformance with the Appendix J Program and RG 1.97. The scope of work included the installation of two 3/8" block valves and a variety of 3/8" test connections to the TIP purge line to facilitate Appendix J testing. Valve

SOV-7-107 was also replaced with a Safety Class valve that has direct position indication. SOV-7-107 is an automatic valve and did not previously have position indication. The added feature does not affect the operation of the valve or the system. This modification required a change to the Appendix J Program.

Safety Evaluation Summary (SE 98-014)

The components in the TIP Purge system were upgraded in order to bring them into compliance with Appendix J testing requirements and Regulatory Guide (RG) 1.97. The primary containment isolation function of the TIP Purge system was enhanced because 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 does not present a significant hazard not described or implicit in the FSAR, and there is a reasonable assurance that the health and safety of the public is not endangered.

EDCR 97-419, Turbine Building Blowout Panels

General Summary

This EDCR was written to install blowout panels in the Turbine Building to act as a relief device for venting the building during High Energy Line Breaks (HELB) and Tornadoes. This will protect the emergency diesel generator and day tank block walls from a high differential pressure so that they will remain intact. Although the analytical model used for Turbine Building HELBs took credit for the blowout panels, recent investigation revealed that the blowout panels did not exist.

The blowout panels used the existing siding to maintain the function of providing a weatherproof structure. The panels were cut to form an approximate 1000 square foot area and secured with calibrated shear pins made to allow the panels to blowout at 0.5 psid. The panels are tethered to the building structural steel to limit their movement once they have actuated.

Safety Evaluation Summary (SE 97-033)

The modifications implemented by this design and equipment with which they interface are not accident initiators nor will they initiate any abnormal operational transients. These panels are credited in the analysis and as such, already take into consideration any radiological releases.

The margin of safety is inherently increased relative to the diesel generator masonry walls structural capability to physically remain in place during and subsequent to a design basis accident. Therefore, the margin of safety is not reduced.

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

EDCR 97-420, Upgrade of CS and RHR Flow Instrumentation Loop

General Summary

This design change upgraded RHR/CS flow instrumentation loops to satisfy the following:

- RHR/CS pump minimum flow monitoring
- Flow control following RHR/CS pump suction strainer installation, and
- ASME OMa-1988 in-service testing of pumps in light water reactor power plants

Implementation of this design change provided one set of instruments to be used for minimum flow, strainer application and in-service testing.

This design change also addressed long-term minimum flow requirements for V14-26A and V14-26B. Circuit changes were made which now allow the operator to override an automatic "close" signal in the control scheme of V14-26A and V14-26B. This ability to override will allow the operator to open these valves in the presence of an autoclosure signal for long-term minimum flow purposes.

Safety Evaluation Summary (SE 98-008)

The upgrade of the RHR and CS System flow rate instrumentation improved the system reliability, accuracy, and enhanced the operational performance. This improved accuracy will result in usage of higher LPCI flows than currently assumed in the LOCA analyses.

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 does not present a significant hazard not described or implicit in the FSAR, and there is a reasonable assurance that the health and safety of the public is not endangered.

EDCR 97-421, Generator No-Load Disconnect Switch

General Summary

Installation of this design change brought Vermont Yankee into full compliance with the Station Blackout (SBO) Rule while maintaining the original design basis for offsite power. The design change installed a generator "no-load" manual disconnect switch into the isolated phase bus in the Turbine Building between the main step up transformer and the main generator. The switch will be operated via a hand crank. Position switches were wired into the existing generator lockout trip circuits to automatically reconfigure the protection circuits when the disconnect is opened.

The design change significantly shortened the time required to establish conditions necessary to backfeed the Main Transformer from the 345 kV switchyard and energize the Unit Auxiliary Transformer and station 4160 Volt buses. This allowed Vermont Yankee to fully credit this source as a second source of offsite power. It also allowed the Vernon Tie Line to be designated as the Station Blackout Alternate AC Source and not credited as an offsite power source.

Safety Evaluation Summary (SE 98-006)

Installation of a Generator No-Load Disconnect Switch provided a more immediate source of offsite power through the main step-up transformer and unit auxiliary transformer. Installation of the Disconnect Switch upgraded the offsite power circuits by enabling the plant to establish a delayed access source within one hour and allows the Vernon Tie Line to satisfy the Station Blackout source requirement.

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 does not present a significant hazard not described or implicit in the FSAR, and there is a reasonable assurance that the health and safety of the public is not endangered.

EDCR 97-422 Cycle 20 Reload

General Summary

This design provides descriptions and analysis results pertaining to the mechanical, thermal-hydraulic and reactor physics analyses required for Cycle 20 operation. The Cycle 19/20 refueling involved the discharge of 112 irradiated fuel assemblies and the insertion of 112 GE-13 type fuel assemblies. This design change also provided details for the replacement of several control rods and LPRM strings.

The impact of this design change on the licensing bases was evaluated and it was determined that there would be no change in the existing power operation or safety design bases.

Safety Evaluation Summary – Core Shroud Tensioning (SE 98-016)

This Safety Evaluation addressed the re-tensioning of the core shroud tie-rods to support the GE-13 reload. An evaluation of the core shroud repair, completed in 1996, was conducted and it was concluded that there was sufficient margin in the repair design for the increased RIPD associated with the use of GE-13 fuel.

Re-tensioning of the shroud ties did not increase the probability of occurrence or consequences of an accident or malfunction as previously evaluated in the FSAR. The re-tensioning also did not present a significant hazard not described or implicit in the Vermont Yankee FSAR, and there is a reasonable assurance that the health and safety of the public was not endangered.

Safety Evaluation Summary – Fuel Shuffling (SE 98-017)

Safety Evaluation 98-017 addressed the refueling operations associated with the fuel shuffling prior to Cycle 20. A review of Abnormal Operating Transients, Station Blackout (SBO), Anticipated Transient Without Scrams (ATWS), Appendix R, Alternate Shutdown, Thermal Hydraulic Stability and the safety functions of Structures, Systems and Components determined that the refueling operations associated with Cycle 20 would not increase the probability of occurrence or consequences of an accident or malfunction as previously evaluated in the FSAR. It was also determined that the fuel shuffling did not present a significant hazard not described or implicit in the FSAR, and there is a reasonable assurance that the health and safety of the public was not endangered.

EDCR 97-422, ECN 1 Cycle 20 Reload

General Summary

This design change provided descriptions and analysis results pertaining to the mechanical, thermal-hydraulic and reactor physics analyses required for Cycle 20 operation. The Cycle 19/20 refueling involved the discharge of 112 irradiated fuel assemblies and the insertion of 112 GE-13 type fuel assemblies. This design change also provided details for the replacement of several control rods and LPRM strings.

The impact of this design change on the licensing bases was evaluated and it was determined that there would be no change in the existing power operation or safety design bases.

Safety Evaluation Summary – Operation of Cycle 20 (SE 98-030)

Safety Evaluation 98-030 addressed the reload core design and operation of Cycle 20 itself through End-of-Cycle. Operation of Cycle 20 does not increase the probability of occurrence or consequences of an accident or malfunction as previously evaluated in the FSAR nor does Operation of Cycle 20 present a significant hazard not described or implicit in the FSAR. There is a reasonable assurance that the health and safety of the public will not be endangered as a result of Cycle 20 operation.

EDCR 97-423, RHR and CS Strainer Replacement

General Summary

This design change installed four replacement large-capacity passive suction strainers on the Core Spray (CS) and Residual Heat Removal (RHR) Torus penetrations. These strainers were designed with sufficient capacity to ensure that debris loading effects following a LOCA would have minimal effect on the Net Positive Suction Head (NPSH) for the low pressure Core Standby Cooling System (CSCS) pumps. The strainers were installed in response to NRC Bulletin 96-03.

The assumptions used in sizing and the requirements for the new strainers are consistent with the provisions of Regulatory Guide (RG) 1.82, Rev. 2, "Water Sources for Long-Term Recirculation Cooling Following a Loss-of-Coolant Accident" and NUREG/CR-6224, "Parametric Study of the Potential for BWR ECCS Strainers Blockage Due to LOCA Generated Debris."

Other work performed under this design change included Torus painting, modification of the downcomers and the installation of narrow range level instrument taps. A temporary access opening was made in the Torus and internal vent header to facilitate this work.

Safety Evaluation Summary (SE 98-012)

The installation of the new strainers, modification of the downcomers and other work detailed in this design change did not increase 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 in the FSAR, and there is a reasonable assurance that the health and safety of the public was not endangered.

Temporary Modification 96-047

General Summary

This Temporary Modification was written to address a steam leak past the pressure seal gasket in Main Steam valve V2-77.

The leak was repaired using a Furmanite Leak repair process whereby sealing material is injected through ports drilled in the valve body. The ports are threaded and adapters are installed to ensure there is no leakage from the threaded holes and to accommodate the sealant injection gun. Once the seal is injected, the gun is removed and a plug is installed on the adapter.

Safety Evaluation Summary (SE 96-071)

The Main Steam lines are not considered initiators of any analyzed accidents and do not initiate any operational transients. Any failure mode of this valve is bounded by the Main Steam Line break analysis and the Main Steam Isolation Valves would terminate any release path. This modification was performed with the upstream valve isolated to ensure Primary Containment Isolation. The implementation of this Temporary Modification did not reduce the margin of safety, failure point or accepted safety limit of the Primary Containment or the Primary Containment Isolation 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 FSAR, and there is a reasonable assurance that the health and safety of the public was not endangered.

Temporary Modification 97-003

General Summary

This Temporary Modification installed a freeze seal downstream of the Reactor Building Closed Cooling Water (RBCCW) outlet isolation valve for the "B" Fuel Pool Cooling (FPC) heat exchanger. The "B" isolation valve had a broken stem, and with the exception of shutting down the entire system, there was no other way to isolate the valve. The freeze seal was applied to the heat exchanger common discharge line, which required that both heat exchangers be taken out of service.

During the repair, there was a potential for flooding and loss of the RBCCW system if the freeze plug was not maintained during disassembly. This potential condition was highly unlikely due to the following: 1) the establishment of the freeze seal was controlled by a "Continuous Use" procedure, 2) Vermont Yankee has successfully used freeze seals in the past, 3) the integrity of the seal was maintained and additional reserve liquid N^2 was immediately at hand, 4) monitoring the temperature of the pipe on the live side of the freeze seal, and 5) the installation of a gasketed plate over the valve body opening when the bonnet was removed.

Safety Evaluation Summary (SE 97-002)

The purpose of this safety evaluation was to ensure that no unreviewed safety question would exist while performing maintenance on the outlet isolation valve for the "B" Fuel Pool Cooling (FPC) heat exchanger.

The FPC and RBCCW systems are not accident initiators for any of the design basis accidents nor can they initiate any abnormal operational transients. Both systems form closed loops and are not connected directly to the reactor coolant pressure boundary or the primary containment atmosphere.

The Standby Fuel Pool Cooling System was available to maintain fuel pool temperatures in accordance with Technical Specification 3.12.

The use of a freeze seal does not increase the radiological consequences of any accident analyzed in the FSAR nor does it produce any new release paths to the environment.

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

Temporary Modification 97-005

General Summary

This Temporary Modification installed a freeze seal in a 6-inch pipe to isolate V70-18A, the waste sludge mixing pump flush connection valve. This valve would not isolate and leaked past its seat which required the valve to be rebuilt. If this valve had to be isolated using isolation valves in the affected lines, extensive portions of the Condensate Transfer System (CST) including the transfer pumps would have to be removed from service. This would not have been a practical alternative as the use of the CST for fuel pool filling or ECCS Systems backup water supply would not have been available.

Failure of the freeze plug would be mitigated by the downstream 6-inch by 4-inch reducers which would limit the flooding potential and allow time to secure the open system. Additionally, any flooding due to freeze seal failure would be confined to the lower level of the Radwaste building and not affect any SSCs.

Safety Evaluation Summary (SE 97-003)

The purpose of this Safety Evaluation was to ensure no unreviewed safety question would exist while performing maintenance on the waste sludge mixing pump flush connection valve. The CST supply to various Radwaste components was isolated during repair of this valve.

The safety objective of the CST system is to provide a backup source of water for HPCI and RCIC operation. Installation of this freeze seal did not affect the availability to perform this function as the CST system suction point is above the "reserved" volume of water for the

HPCI/RCIC function. The CST and Radwaste systems are not initiators of any of the design basis accidents nor do they initiate any abnormal operational transients.

The use of a freeze seal does not increase the radiological consequences of any accident analyzed in the FSAR nor does it produce any new release paths to the environment.

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

Temporary Modification 97-007

General Summary

This TM temporarily blocked off the normal HVAC supply to the East and West Switchgear rooms by installing a blanking plate in the supply duct from TSF-2A and TSF-2B. The plate was installed in the duct, west of the switchgear room west wall at an existing access hatch. The East Switchgear Room exhaust fan was tagged to provide a supply path for fresh air into the room. The existing ductwork distributed air from the East to the West Switchgear Rooms. The existing West Switchgear Room exhaust fan provided the motive force to move the air through the two rooms.

The purpose of this TM was to ensure that a postulated high energy line break (HELB) in the Turbine Building would not adversely affect equipment in the Switchgear Room.

Safety Evaluation Summary (SE 97-006)

Installation of this TM would not initiate any of the accidents described in the FSAR. The TM blocked off the normal flow from TSF-2A and 2B in the turbine hallway, west of the swtichgear room west wall, which is physically separated from components which can affect accidents.

This TM only changed the source of inlet air to the switchgear rooms. The air flow was redistributed to the turbine building where operational flexibility in the turbine building HVAC system and/or system rebalancing assured that there were no adverse effects on the system or the turbine building. The equipment in the switchgear room and the turbine building continued to function in the same fashion as they always did, there was no increase in the probability of occurrence or consequences of an accident or malfunction as previously evaluated in the FSAR. This TM 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.

Temporary Modification 97-009

General Summary

This Temporary Modification installed temporary diesel driven pumps, piping, and instruments in the Alternate Cooling System Cooling Tower Cell CT-2-1 for performance testing utilizing heat effluent from the main condenser cooling water.

This temporary system was an NNS system that discharged into a Safety Class 3, seismic structure. The vertical and horizontal piping runs, inside the cooling tower, were installed such that the alternate cooling structure would not be challenged during a seismic event.

Safety Evaluation Summary (SE 97-009)

These systems are not initiators of any accident or transient, and no margin of safety is associated with the Circulating Water System. It should also be noted that the safety functions of the Alternate Cooling System and Service Water System were not adversely affected.

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

Temporary Modification 97-013

General Summary

This TM was written to establish a freeze seal on a Turbine Building Closed Cooling Water (TBCCW) ¾ inch carbon steel pipe to provide isolation to repair the chemical feed tank isolation valve. In order to isolate the valve without the use of a freeze seal the entire TBCCW system would have to be shutdown which would prevent plant operation.

During the repair, there was a potential for flooding and loss of the TBCCW system if the freeze plug was not maintained during valve disassembly. This potential condition was highly unlikely due to the following: 1) the establishment of the freeze seal was controlled by a "Continuous Use" procedure, 2) Vermont Yankee has successfully used freeze seals in the past, 3) the integrity of the seal was maintained and additional reserve liquid N^2 was immediately at hand, and 4) monitoring the temperature of the pipe on the live side of the freeze seal.

Safety Evaluation Summary (SE 97-020)

The TBCCW system is not an initiator of any of the design basis accidents or any abnormal operational transient. This system is a closed loop system that does not connect with the reactor coolant pressure boundary or the primary containment.

The use of a freeze seal DOES not increase the radiological consequences of any accident analyzed in the FSAR nor does it produce any new release paths to the environment. No safety margins were affected, as this system is a non-nuclear safety 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 FSAR, and there is reasonable assurance that the health and safety of the public was not endangered.

Temporary Modification 97-021.

General Summary

This Temporary Modification was installed to facilitate repair of one of the SAC-1 air handling units. The barrier was necessary because of a common header, with another air handling unit, utilized to distribute air. The barrier allowed operation of the opposite unit during the repair process.

Safety Evaluation Summary (SE 97-023)

The installed barrier did not increase the probability of occurrence of any accident analyzed in the FSAR. SAC-1 is not an initiator of any accidents or operational transients. This installation could increase the potential radiological consequences in the Control Room if the Control Room HVAC was placed in "emergency" recirculation mode. In-leakage by the barrier could occur until the SAC-1 access panels were re-installed and sealed. This effort took approximately 30 minutes during which time the operators in the Control Room could, if necessary, continue to stay in the Control Room utilizing self-contained breathing apparatus.

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

Temporary Modification 97-26

General Summary

This Temporary Modification disabled the turbine-generator current to cooling flow comparator. The current runback circuit was found to be a less than adequate design and this modification will prevent the generation of a generator runback.

The lack of this protection could result in generator damage and disconnection from the electrical grid due to undetected failures in the generator coolant system. However, to reduce the potential for inadvertent generator runback, which could result in a significant nuclear system transient, the protection was removed. Other inputs that initiate a generator runback will remain functional, as well as individual parameter alarms.

Safety Evaluation Summary (SE 97-032)

The turbine-generator runback circuit is not an initiator for any accident. Loss of stator cooling with a generator runback is an event that results in a reactor water temperature decrease which requires operator action to reduce reactor power. The removal of this protective function will therefore not have an adverse safety impact on the reactor.

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

Temporary Modification 97-030

General Summary

This TM was written in conjunction with EDCR 97-423 and provided details for draining the Torus for RHR and CS Suction Strainer Replacement and Torus Internal Coating Rework. The Torus was drained using temporary submersible pumps placed in the Torus and connected to Fuel Pool Cooling line 6"-FPC-16. Electrical issues associated with this work were controlled via TM 98-005. Water inventory was controlled by Operations prior to plant shutdown to ensure a sufficient capacity would be available during Torus drain down.

Safety Evaluation Summary (SE 98-007)

This Safety Evaluation was written to address the effect of the temporary system installed by TM 97-030 on operation of permanent plant systems and the effect of a failure in the temporary system. This work occurred with the plant in a cold shutdown condition.

This TM utilized temporary pumps, hoses, piping and valves to drain the Torus through the Fuel Pool Filter Demineralizers to the Condensate Storage Tank.

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 does not present a significant hazard not described or implicit in the FSAR, and there is a reasonable assurance that the health and safety of the public is not endangered.

Temporary Modification 98-001 and TM 98-001, Rev. 1

General Summary

TM 98-001 installed four air compressors, four Air Dryers and an Air Receiver tank in the courtyard area east of the Reactor Building in order to support planned Torus work during the 1998 Refueling Outage. Two existing Reactor Building penetrations were used to deliver compressed air to various parts of the Reactor Building. The air system was operated in accordance with an approved operating procedure and was continuously manned by a trained operator. This operator had remote indication of Reactor Building pressure and Reactor Building fan status in order for the system to be isolated from secondary containment in the event that normal Reactor Building ventilation was lost.

The original version of TM 98-001 required the mode switch to be in "Shutdown" while using the construction air system. Additional calculations and analysis were performed which determined that usage limited to 250 scfm or less would not impact the Standby Gas Treatment System. Therefore, Revision 1 stated that grit blasting evolutions would be performed in the "Shutdown" mode but that clean-up evolutions and painting could be performed with the mode switch in "Refuel".

Safety Evaluation Summary (SE 98-19 Rev 1)

Installation and operation of this temporary air supply system was not an accident initiator for any accident previously evaluated in the FSAR, including Refueling Accidents. This system did not interface with any other system and there was no effect on any other system. There is reasonable assurance that the health and safety of the public was not endangered due to the use of this temporary air supply system.

Temporary Modification 98-010

General Summary

This TM provided clean, dehumidified air for circulation in the Torus during the Torus Coating Project conducted under EDCR 97-423. This TM was only in effect while the plant was in the Shutdown or Refuel mode.

Ventilation equipment, which included four dehumidifier units, an airhouse, pre-filter unit and the associated ductwork, was provided by an approved vendor and secured in the Reactor Building according to an approved VY procedure. Exhaust waste heat from electrical equipment was vented into the Reactor Building.

An assessment was conducted and the applicable precautions taken for potential fire, flood and radiation hazards.

Safety Evaluation Summary (SE 98-022)

This TM did not increase the probability or occurrence of an accident previously evaluated in the FSAR nor did this TM introduce any new equipment malfunctions which were different from those previously evaluated. The design bases of the Reactor Building HVAC system and SBGT system were not affected by this TM therefore, it was concluded that this TM did not reduce any 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 Temporary Modification 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.

Temporary Modification 98-030

General Summary

This TM superceded TM 98-010 which installed a temporary Torus ventilation system in support of the Torus Coating and Strainer Replacement Project (EDCR 97-423). TM 98-030 modified the existing installation (TM 98-010) to route the exhausted air from the temporary ventilation system predominantly into the Reactor Building normal ventilation system exhaust duct. TM 98-030 also provided the necessary details for removal of the ventilation system installed under TM 98-010 and closure of TM 98-010.

Safety Evaluation Summary (SE 98-024)

The design bases of the Reactor Building HVAC system and the Standby Gas Treatment system were not affected as a result of this modification. There was no reduction in the margin of safety as defined in the Technical Specifications. There was reasonable assurance that the health and safety of the public was not endangered due to the implementation of TM 98-030.

Temporary Modification 98-032

General Summary

TM 98-032 installed new lines of code in the refuel platform PLC "Hoist Loaded" reset logic. The new code resulted in the "seal in" of the hoist load until either the grapple was opened or the platform was no longer over the core. This new logic is different from the preset logic which only required the hoist loaded signal to go below a pre-determined setpoint.

Safety Evaluation Summary (SE 98-026)

This logic change did not increase the probability of occurrence or consequences of an accident or malfunction as previously evaluated in the FSAR. This logic change actually added another level of logic to ensure that no fuel was on the Main Refuel Hoist before allowing the "Rod Block" to clear. The test did not present significant hazards not described or implicit in the FSAR, and there is a reasonable assurance that the health and safety of the public was not endangered.

Basis for Maintaining Operability (BMO 96-05), Revision 4 Increase in Maximum Torus Temperature Post-LOCA

General Summary

This BMO, provides the justification for an increase in maximum Torus temperature from 166°F to 185°F from an initial Torus temperature of 90°F due to a re-evaluation of Torus temperature following a LOCA.

Revision 4 to this BMO documents the satisfactory resolution of issues identified in revisions 0 through 3 and supports continued operation through 1998 with the BMO open while confirmatory analyses and FSAR updates are completed. This plan has been presented to the USNRC in several meetings and has been accepted by the USNRC in a letter dated April 16, 1998.

This BMO revision states that the analysis for containment temperature response following various design basis accidents scenarios have identified that the peak Torus temperature can approach 185°F. Based on further analysis, the conservatively calculated peak Torus temperature is no greater than 183.2°F. This is in excess of the 176°F evaluated in previous revisions. This higher temperature does not affect EQ equipment as the equipment is qualified to 185°F. All Torus structures and attached equipment and piping are acceptable up to 185°F and NPSH margins for the Residual Heat Removal and Core Spray pumps remain adequate with the new elevated temperature.

Thus it can be concluded that there are no adverse safety consequences of the higher peak Torus temperature.

Safety Evaluation Summary, Revision 2, (SE 96-008)

Revision 2 of the safety analysis was prepared for the increase in maximum Torus temperature from 166°F to 185°F, assuming an initial Torus temperature of 90°F.

The changes to Torus temperature and its associated effects do not affect any of the parameters that could initiate an accident or an abnormal operational transient. No increase in the consequences of any analyzed event since the radiation release barriers are either unaffected or maintained within existing limits.

Containment integrity and CSCS performance are not adversely affected by this change. Therefore this change does 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. This 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.

Basis for Maintaining Operability (BMO 97-016), SII Cable Run for a Short Distance in a DI Pull Box

General Summary

During drawing reviews, due to Appendix R questions, it was determined that an SII cable passed through a DI pull box for a short distance. Additionally, seven NNS cables are also run through the same box. The concern was a potential problem whereby if a fault were to occur in the SII cable and cause a loss of the DI cable (or vice versa) then the High Pressure Coolant Injection (HPCI) system could be lost. The separation criteria are a requirement of the FSAR and a Vermont Yankee document, VYS-027, that addresses cable separation. Due to breaker classification and coordination, adequate protection is provided for all the affected circuits except one, which is not normally in service and has been de-energized as a precaution.

Safety Evaluation Summary (SE 97-011)

This safety evaluation was performed to evaluate the improper cable routing that existed prior to prompt repair of the problem.

The equipment supplied by the cables routed through the DI pull box can not initiate an accident or transient but could potentially cause the loss of the HPCI system. However, due to the short length of this routing, the cables not touching each other, the cables being lightly loaded, and the condition of the cables, pull boxes and fire seals are excellent, the probability of a problem occurring is very low. If this had occurred, the Automatic Depressurization System (ADS) which is the backup for HPCI would have been available to function as required.

These changes do not result in a reduction in a margin to safety as described in the FSAR or the basis for any Technical Specification.

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

<u>Basis for Maintaining Operability (BMO 97-015), Torus Vent Valve Position Change</u>

<u>General Summary</u>

This BMO was written to provide a basis for maintaining operation when the positionof the Torus Vent Valve TVS-86 was changed from normally open to normally closed.

During investigation of an Event Report, it was determined that actions taken in accordance with the Vermont Yankee Emergency Operating Procedures (EOPs) could result in actuation of the Torus Vent System (TVS) rupture disc. The EOPs direct the operators to initiate containment flooding if certain plant conditions cannot be met and further directs the operators to vent the containment to maintain containment pressure at $\leq$ 62 psig. However, the additional weight of the water, if the containment were flooded, would increase the pressure by 10.3 psig with a resultant total pressure of 72.3 psig. This exceeds the setting of the TVS rupture disc and the maximum pressure shown in the FSAR.

Design Change, EDCR 90-406 was implemented to install the Torus Vent System which added a rupture disc and an isolation valve as a method of venting the Torus. This design change and safety evaluation was for limiting containment pressure during a beyond design basis loss of containment heat removal event. Following the loss of containment heat removal event, containment pressure would gradually increase to the point where the rupture diaphragm would open in approximately 12 - 24 hours after the start of the event, which allows sufficient time for re-establishing containment cooling.

<u>Safety Evaluation Summary</u> (SE 97-016)

The Torus Vent System is not an initiator of any analyzed accidents or transients. Closing the isolation valve would not increase the radiological consequences due to the integrity of the Torus Vent System. No margin of safety is reduced, as closing the isolation valve would not change the setting of the rupture disc and the operation of the vent is not needed to mitigate any design basis accident or abnormal operational transient.

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

<u>Basis for Maintaining Operation (BMO 97-032)</u>

<u>General Summary</u>

This BMO supports the change in operation of fail open air operated valves PCV–104-69A/B from an air operated valve to a manually operated valve. These valves control the pressure in the Alternate Cooling System (ACS) when the ACS is in operation. It was determined that if a problem occurred in the air supply regulator, controller or valve positioner, it could cause an increase in the diaphragm pressure and close the valve. If the valve fails closed the hand operator cannot open it.

This change isolates the air supply from the valve and vents the diaphragm which fails the valves open. The valves can then be controlled, by the hand operator, to control pressure in the event that ACS is required.

Safety Evaluation Summary (SE 97-018)

The ACS is not an accident initiator for any design basis accidents nor can it initiate any abnormal operational transients. This change decreases the probability of a malfunction by eliminating the reliance on non-safety/non-seismic regulators and controllers. These components are part of an auxiliary support system and their failure would have no impact on any safety limits. This change does not reduce the capability of the ACS to provide cooling to 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 BMO 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.

Basis for Maintaining Operation (BMO 96-08), Effect of Design Basis Tornado Pressure Loading on Diesel/Day Tank Enclosures

General Summary

This BMO was written to ensure that an unreviewed safety question did not exist due to the compensatory actions that were taken regarding vent paths and pressure relieving capability requirements as stated in the FSAR for tornado protection.

Compensatory actions, based on a BMO, include the use of weather alert radios in the Access Control Office (ACO) and Central Alarm Station (CAS) whereby the security force will immediately notify the Operations Shift Supervisor (SS) of a weather alert broadcast for the area. As conditions warrant, the SS will have the diesel and day tank doors blocked open and will post watches as required.

Both Security and Operations have assessed their capability for this response and concluded that they can support the manual actions required to complete these actions.

Personnel staffing issues are acceptable since a timing constraint for implementing the manual actions has been defined as two hours. This requirement is commensurate with Technical Specification requirements, which allows the Fire Brigade composition to be less than the minimum complement for a period not to exceed two hours, while personnel are called in.

Safety Evaluation Summary (SE 97-007)

The compensatory measures and alteration of normal plant configuration (diesel and day tank room doors) are not initiators of accidents or transients. Blocking open the doors degrades the Fire Protection and EQ functions they provide; however, both Security and Fire Watch personnel have been trained to close the doors in the event of changing environmental conditions that would be indicative of a steam line break or fire. While in this configuration, qualified Security and Fire Watch personnel will be posted in accordance with procedure requirements.

The margin of safety is not affected by this evolution, as the two-hour time frame is consistent with Technical Specifications requirements for fire brigade staffing.

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

Special Test Procedure 96-012, Reactor Core Isolation Cooling Pump Vibration Evaluation

General Summary

The purpose of this test was to gather information on the Reactor Core Isolation Cooling (RCIC) pump and turbine vibration levels when operating under specific conditions of varying speed and constant flow. This test will be used in order to justify a relief request from the ASME OM-6 code vibration limits for this pump. As a result of the In-Service Testing (IST) program third ten-year interval update, an alert range limit of 0.325 in/sec peak was required by the code. Due to the small deviation between the reference value and the alert limit, normal fluctuations in readings caused the pump to be placed unnecessarily in the alert range.

Safety Evaluation Summary (SE 97-008)

The performance of this test will not initiate any design basis accidents or cause any abnormal operational transients. The pump was operated within its design limits during the test. The control system for the RCIC system continued to be operable during the test, and would have provided an automatic return from the test to operating mode if system initiation was required.

This test did not reduce the margin to safety as the system was operated within its design limits using only the full flow test line which pumped water to the Condensate Storage Tank (CST) and had no effects on reactor water temperature.

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

Special Test Procedure 97-002, Alternate Cooling System Performance Testing with Reduced Flow

General Summary

This test was written to demonstrate that the Alternate Cooling System could meet its hydraulic and thermal performance requirements at less than design flows.

A temporary pumping system removed heated water from the main condenser discharge, at the discharge structure forebay, and pumped it to the Alternate Cooling Tower cell CT-2-1 distribution trays at the top of the cell. Flow then descended through the cell into the deep basin and out to the discharge structure. Instrumentation was placed to measure temperatures, flows and humidity to evaluate performance of the cell.

This test did not impact the normal operation of the station Circulating Water, Alternate Cooling or Service Water Systems.

Safety Evaluation Summary (SE 97-010)

These systems are not initiators of any accident or transient. It should also be noted that the capability of the Alternate Cooling System, Circulating Water System, and Service Water System were not adversely affected by this test. The Circulating Water System is not specifically addressed in the Technical Specifications; however, the Circulating Water System flows are used in determining off-site doses in the Off-Site Dose Calculation Manual (ODCM). There are no margins of safety associated with the Circulating Water System or in the basis of any Technical Specification, which are affected by this test.

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

Special Test Procedure 97-003, John Deere Diesel Voltage and Frequency Evaluation

General Summary

The purpose of this test was to determine if the John Deere Diesel Generator (JDDG) and all connected loads would operate satisfactorily with 480 volt Bus 11 unavailable. Previous problems occurred in that voltage fluctuations were experienced when the diesel was equipped with a mechanical governor.

This test required that the voltage and frequency of the JDDG be measured and recorded to determine the performance of the newly installed diesel electronic governor.

Safety Evaluation Summary (SE 97-031)

Equipment fed from the JDDG during a loss of normal power are not initiators of any design basis accident or any abnormal operational transient. These loads are not required to mitigate the consequences of any design basis accidents.

This test configures the loads for the generator, as designed under the condition of the loss of Bus 11. Performance of this Special Test Procedure does not reduce any margins to safety as none of the loads are required to perform any safety functions.

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

Special Test Procedure 97-004, RHR Minimum Flow Evaluation

General Summary

The purpose of this special test procedure is to gather information on the RHR System when operating in the minimum flow recirculation mode. A non-intrusive ultrasonic flow measurement device will be temporarily installed to obtain flow rates. The data gathered for this test supports vendor approved proceduralized minimum flow requirements.

The RHR pump vendor revised the recommended minimum flow rates required for pump protection and has decreased the time the pumps can run using the installed minimum flow line.

This Special Test began by establishing Torus Cooling. Once this configuration had been stabilized, the discharge flow path was secured and after approximately 23 seconds, the minimum flow valve opened. Once the flow was stabilized and the data taken, the RHR Pump was secured and the system was returned to its normal configuration

Safety Evaluation Summary (SE 97-029)

This Special Test is not an initiator of any design basis accident or any abnormal operational transient. During the test, the RHR system was operated in the Torus Cooling mode and the minimum flow bypass mode which is within the design and licensing basis of the RHR System.

This test does not reduce the margin of safety as described in the FSAR or the basis for any Technical Specification. An administrative Technical Specification LCO was entered during performance of this Special Test Procedure due to the RHR configuration. The LCO is specifically associated with Torus Cooling and was not a result of the Special Test Procedure.

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

Special Test Procedure 97-005, Pressure Testing of 10" RHR-41A and 10" RHR-41B

General Summary

This test procedure was performed to satisfy the requirements of ASME Section XI, Articles IWC and IWD-5000, System Pressure Tests for piping lines 10" RHR-41A and 10" RHR-41B. This section of piping is a small portion of the RHRSW to RHR emergency fill inter-tie and was hydrostaticly tested during quarterly RHR Pump Operability Testing. Performing this pressure test in parallel with RHR operation requires pressurization of only the test boundary by the hydro pump. Otherwise major portion of the RHR System would need to be pressurized. The test pressure was equivalent to the normal system operating pressure of 210 psig.

Safety Evaluation Summary (SE 97-027)

The pressurization of the piping to the specified test pressure is not an initiator of any accident analyzed in the FSAR nor could this test cause any abnormal operational transient. The inter-tie is only utilized as an "emergency fill" from the station Service Water System in the event of a loss of primary coolant inventory and failure of the ECCS. The hydro pump cannot overpressurize the RHR piping due to its small size.

Due to the hydrostatic pressure being equal to system operating pressure, a heat exchanger leak was not expected. If one had occurred, the system was capable of being immediately isolated. Additionally, the test medium was demineralized water, which precludes the release of radioactive material.

This test did not modify or change either the RHRSW or RHR systems, thus, no safety margins were affected.

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

Special Test Procedure 97-006, Core Spray Minimum Flow Evaluation

General Summary

The purpose of STP 97-006 was to gather information on the Core Spray (CS) System when operated in the minimum flow recirculation mode. Data obtained during the testing included:

- Core Spray Full Flow and Minimum Flow
- Core Spray Pump Suction Pressure in Full Flow and Minimum Flow
- Core Spray Pump Discharge Pressure in Full Flow and Minimum Flow
- Core Spray Pump Vibration Data in Full Flow and Minimum Flow

Safety Evaluation Summary (SE 97-036)

The performance of this test does not initiate any design basis accidents or cause any abnormal operational transients. The test included the temporary installation of an ultrasonic flow measurement device which was used to measure CS minimum flow in the "B" loop of CS. The "B" CS pump was the only operating pump in the system during the performance of the test. The ultrasonic flow measurement device was nonintrusive.

The performance of this STP did not reduce the difference between a system failure point and the accepted safety limits, nor did it reduce the margin of safety as defined in the basis for any Technical Specification. Both CS subsystems were available for injection if required during the performance of the STP.

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

<u>Special Test Procedure 98-001, Dynamic Testing of V70-19A and V70-20</u>

<u>General Summary</u>

The purpose of this special test was to gather information on motor operated valve (MOV) performance when the tested valves are operated under design basis test differential pressure and flow conditions.

Service Water valves V70-19A and V70-20 were previously tested, however, in response to a commitment made by the BWR Owners Group under the JOG PV Program to conduct periodic in-plant differential pressure (DP) testing on selected gate, globe and butterfly valves, VY committed to DP test MOVs V70-19A and V70-20.

The data collected consisted of upstream and downstream time history pressures, differential pressures, ambient temperature, process fluid temperature MOV thrust, torque, motor current switch actuation and spring pack displacement time histories.

<u>Safety Evaluation Summary</u> (SE 98-020)

The performance of STP 98-001 did not increase the probability of occurrence or consequences of an accident or malfunction as previously evaluated in the FSAR. This test did not present a significant hazard not described or implicit in the FSAR, and there is reasonable assurance that the health and safety of the public was not endangered.

<u>Special Test Procedure 98-002, V10-26A Dynamic Pressure Locking Test</u>

<u>General Summary</u>

The purpose of this test was to verify the ability of V10-26A (Lower Drywell Spray Outboard Isolation Valve) to open when the bonnet is pressurized. Pressure locking can occur when operating sequences or temperature changes cause the pressure of the fluid in the bonnet of a gate valve to be higher than the pressure on the upstream and downstream sides of the disc assembly. Under a LOCA scenario, V10-26A could be subject to bonnet pressure. After Torus cooling is established after a LOCA, the valve is exposed to water at a temperature approximately equal to RHR heat exchanger outlet temperature. This temperature was estimated to be 35°F higher than reactor building ambient conditions. Valve V10-26A would be exposed to this increase in temperature for approximately 20 minutes before it is opened.

During performance of STP 98-002 it was desired that a bonnet pressure of 50 psig above upstream system pressure be maintained and then the valve would be opened while diagnostically monitoring the valve.

<u>Safety Evaluation Summary</u> (SE 98-021)

The performance of STP 98-002 did not increase the probability of occurrence or consequences of an accident or malfunction as previously evaluated in the FSAR. This test did not present a significant hazard not described or implicit in the FSAR, and there is reasonable assurance that the health and safety of the public was not endangered.

Special Test Procedure 98-003, RHR Heat Exchangers Thermal Performance Test

General Summary

This STP provided the steps necessary to perform the Residual Heat Removal (RHR) heat exchanger thermal performance test to ensure heat removal capability during worst case accident scenarios. Completion of this test satisfied Generic Letter 89-13 commitments.

The test included both A and B loops of RHR. The test determined the heat removal capability of the heat exchangers at limiting conditions and the associated fouling factors. All tubing, hoses and mounting hardware were safety class. All temporary test equipment involved with the test was removed once the test data was collected.

Safety Evaluation Summary (SE 98-013)

The performance of STP 98-003 did not increase the probability of occurrence or consequences of an accident or malfunction as previously evaluated in the FSAR. This test did not present a significant hazard not described or implicit in the FSAR, and there is reasonable assurance that the health and safety of the public was not endangered.

Special Test Procedure 98-004, Primary Containment Pressurization Test

General Summary

This Special Test involved the pressurizing of the containment to a minimum pressure of 44 psig with a one hour hold point as opposed to the pneumatic leakage test required by ASME Section XI. This provided a retest of the temporary hatch in the Torus which was used for Torus access during the last refueling outage when strainers were replaced in the ECCS equipment. This was also used as a post maintenance test for primary containment penetrations, HPCI/RCIC vacuum breakers and Torus narrow range level instrument penetrations.

Safety Evaluation Summary (SE 98-033)

Performance of this test is not an initiator of any accidents or abnormal operational transients. This test was performed while the reactor was shutdown. The test pressure, 44 psig, was below the design pressure of the containment which is 56 psig. As the test pressure was below the design pressure of the containment, there was no reduction in the margin 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 Special Test Procedure 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.

Special Test Procedure 98-005, Radio Interference Testing

General Summary

The intent of STP 98-005 was to perform radio interference testing in the immediate vicinity of plant equipment to determine whether the use of a wireless system would interfere with or cause anomalies in existing plant equipment. The testing was conducted in the vicinity of plant equipment that included areas where current use of handheld radios was prohibited. The equipment was also tested in the area of the SRM/IRM preamplifier cabinets, Rodney Hunt PLC, Weschler Digital Indicators and the entrance into the Switchgear Room. Benchmarking was conducted and it was noted that similar testing had been conducted at other nuclear facilities with no negative impact on plant equipment.

During the testing period, constant communication with the Control Room was maintained. The results were documented, including the distance the test was performed from the equipment, observed changes in the equipment and other information considered relevant to the test.

Safety Evaluation Summary (SE 98-025)

This test did not increase the probability of occurrence or consequences of an accident or malfunction as previously evaluated in the FSAR. No fuel moves were allowed during this test period and therefore the probability of a control rod drop or refueling accident was not increased.

The test did not present significant hazards not described or implicit in the FSAR, and there is a reasonable assurance that the health and safety of the public was not endangered.

Low Level Waste Storage in Sealand Containers

General Summary

Routine solid radwaste handling practices include the collection and shipment of Dry Active Waste (DAW) material to an offsite contractor. To support this activity, Vermont Yankee collects waste materials from the Radiation Control Areas (RCA) and stores the waste in Sealand containers located outdoors, adjacent to the Radwaste and Reactor Buildings. These containers are stored until there is a sufficient amount of waste accumulated to warrant a shipment. Typically, as many as seven containers may be stored in this area in various stages of fill. The Sealand containers are made of carbon steel and qualify as DOT Strong Tight Packages. By design they are weather tight and protect the waste from environmental conditions that might cause the spread of contamination. Each container has an internal capacity of 1280 cubic feet. Control of these containers is under the plant radiation protection procedures for handling, packaging and shipping radioactive materials.

Safety Evaluation Summary (SE 97-001)

This Safety Evaluation was written to determine if this continuous storage of DAW results in an unreviewed safety question.

The design basis events considered for this temporary type of storage include natural threats of flood, fire, tornado, and earthquake as well as container handling accidents.

The site elevation for the Sealand containers is approximately 0.8 feet below the Maximum Probable Flood (MPF). This places bottom of the containers about 10 inches below the maximum flood depth. However, the MPF would only occur following the probable maximum precipitation. This would allow sufficient time to either sandbag the containers or place them on blocks above the highest MPF level.

Regulatory Guide 1.143 and GL 81-38 do not require seismic or tornado design criteria for structures storing waste in a form suitable for disposal. This is the case with the Sealand containers.

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

Tornado Missile Comparison for Sealand Container Storage

General Summary

Sealand style storage containers are used at Vermont Yankee to store dry radioactive materials. There are seven designated storage areas located within the protected area of the plant. A tornado missile comparison based on the Modified Petry formula and the National Defense Research Committee (NDRC) equations was conducted to ensure that the Sealand and laundry storage containers are bounded by the missile impacts associated with the existing tornado missiles as described in the FSAR. Also included in this comparison were the natural threats of flood, fire and earthquake, as well as container handling accidents in lifting and moving full storage containers onto transport trailers.

Safety Evaluation Summary (SE 97-001, Rev.1)

These containers are physically separated from plant safety class components and do not interface with any plant support service such as water, air, electrical or ventilation

There was no increase in the probability of occurrence or consequences of an accident or malfunction as previously evaluated in the FSAR. Use of these containers does not present a significant hazard not described or implicit in the FSAR, and there is a reasonable assurance that the health and safety of the public is not endangered.

Main Steam Line Valve Leak (MS-77)

General Summary

The steam leak in MS-77 was evaluated for five different concerns as follows: 1) Is the leak significant enough to affect other safety systems or accessibility, 2) could the leakage increase whereby the previous Appendix J leakage results would be invalidated, 3) could the leak degrade whereby the Technical Specification limits on single valve leakage be exceeded, 4) could long term operation of this valve result in further degradation and affect its operability in any other way, and 5) does the inability to open MS-77 and recover from a Group I isolation using the main condenser affect safety.

It was determined that any increases in temperature or humidity would not adversely affect any environmental qualified equipment, or any components or structures in the immediate vicinity of the leak. Component ratings including temperature limitations are above that which would be expected from this leak. Accessibility was not a problem as localized contamination levels were acceptable. The valve was declared inoperable and subsequently isolated by closing upstream and down stream valves. This addressed the potential leakage limits concern. There were no other postulated failure modes that could cause any other negative effects. The operators in lieu of using the main condenser as a heat sink can use other plant systems, the High Pressure Coolant Injection (HPCI), the Reactor Core Isolation Cooling (RCIC) and the Automatic Depressurization System (ADS).

Safety Evaluation Summary (SE 97-004)

This Safety Evaluation was completed to ensure that the steam leak in the Main Steam Line drain valve MS-77 bonnet did not result in an unreviewed safety question.

The Main Steam System is not an initiator of any analyzed accidents or operational transients. Containment isolation requirements were satisfied by the closure and tagging of a series inboard valve that would provide an isolation function regardless of any postulated failure of MS-77. Any large unexpected failure of this valve would not have impacted reactor water level beyond the capabilities of the plant to make-up the leakage. There is no type of transient or malfunction that could have occurred, which is not already bounded by analysis contained 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 condition 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.

Removal of Reactor Building Fresh Air Supply Roll Filter Medium During Winter

General Summary

This evaluation was written to ensure that there were no unresolved safety questions regarding the removal of filter media from the Reactor Building fresh air supply during the winter months.

This media was removed to prevent precipitation from freezing on the filter medium thereby reducing the supply airflow and changing the pressure in the Reactor Building. Although this would be conservative from a radiological standpoint, it would not be desirable to have less ventilation flow.

Safety Evaluation Summary (SE 97-012)

Removal of the filter media from the Reactor Building ventilation supply does not impact the design basis as listed in the FSAR. The Safety Design Basis requires the Reactor Building system to isolate in the event of an accident. The Standby Gas Treatment system would then start and maintain the correct pressure in the Reactor Building.

The lack of filtered air has no impact on the operation of the fifty-four inch ventilation valves.

Reactor Building ventilation flow is not an initiator of any analyzed accident or transient nor does the removal of the filter medium increase the radiological consequences of any analyzed accident. No margin of safety is affected by this change, as ventilation flow is not included in the basis of any Technical Specifications.

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

Downgrade of Thermal Performance Requirements for RRU 5 and 6

General Summary

This safety evaluation was written to document the basis for removing RRU 5 and 6 from the Generic Letter 89-13 Test Program. It has been determined that these units are no longer necessary to meet the cooling requirements for the ECCS corner rooms following an accident. Therefore, the thermal performance for these units has been downgraded, as they provide no safety function for heat removal but are relied upon to retain the pressure integrity of the Service Water system. It has been calculated that RRUs 7 and 8 alone provide sufficient cooling.

Safety Evaluation Summary (SE 97-019)

RRUs 5 and 6 are not initiators of any design basis accident or of any abnormal operational transient. The downgrading of this equipment does not impact the required cooling ability of RRUs 7 and 8 to provide necessary cooling to ECCS components. As RRU 5 and 6 do not provide a safety function, with the exception of maintaining Service Water system integrity, they do not impact or change any 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 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.

Dosimetry Office Move – E-Plan and OP 3504

General Summary

The Dosimetry Office was moved from the Administration Building (OSC) to Warehouse 1. This move affects the Emergency Plan, Section 6 and Vermont Yankee Plant procedure OP 3504, "Emergency Communications". Figure 5, FSAR Section 13.6, also references the Emergency Plan. A dosimetry kit for emergency use has been placed in the OSC so that the referenced figures are still functionally correct.

Safety Evaluation Summary (SE 98-001)

The move of the Dosimetry Office does not increase the probability of occurrence or consequences of an accident or malfunction as previously evaluated in the FSAR. This move will not present significant hazards not described in the FSAR, and there is reasonable assurance that the health and safety of the public was not endangered.

Revision of Safety Class Relief Valves SR-72-3A/B and SR-72-5

General Summary

The safety class designator for the overpressure protection relief valves SR-72-3A/B and SR-72-5 was revised from Non-Nuclear Safety (NNS) to Safety Class 3 (SC3). These valves provide overpressure protection to the Instrument Air System (IAS). The IAS is not a safety related system but it does provide air to safety related solenoid valves (SOVs). These SOVs are required to exhaust air from air operated components which allows the safety function to be accomplished. SOVs have a maximum operating differential pressure (MOPD) above which the valve may not function. Exceeding the MOPD could prevent the SOV from exhausting air from its associated equipment.

Valves SR-72-3A/B and SR-72-5 maintain the IAS pressure less than the MOPD by relieving system pressure. Therefore, the relief valves accomplish a safety related function. The valves serve no safety class pressure boundary function. A failure of the pressure boundary would permit the air operated components to go to the failed/safe position.

Safety Evaluation Summary (SE 98-031)

Safety Evaluation 98-031 addressed the change from NNS to SC3 of the overpressure protection relief valves (SR-72-3A/B and SR-72–5). This change does not increase the probability of occurrence or consequences of an accident or malfunction previously evaluated in the FSAR nor does it present a significant hazard not described or implicit in the FSAR. There is reasonable assurance that the health and safety of the public was not endangered as a result of this upgrade in safety classification.

Revising the Safety Class of a Valve, Pressure Indicator and Transmitter in the HPCI Suction Line

General Summary

An evaluation of the HPCI suction valves concluded that a single failure under certain conditions could cause all HPCI suction MOVs to go open on a loss of logic power. The possibility that MOVs V23-57 and 58 could be open resulted in a requirement for the suction check valve V23-32 to be operable and maintain the water seal in the Torus. A majority of piping in HPCI would also need to be treated as an extension of the primary containment per Appendix J. Using Appendix J criteria, the HPCI system would become a closed seismic loop and an extension of the containment which adds new equipment to the Appendix J Program.

Three components, V23-804B, PI-23-81 and PT-23-83, which would be in the expanded containment, were NNS. To qualify as an Appendix J boundary, the components had be upgraded to SC2. In addition, the requirements of AP 0155, Appendix B "Locking Policy" needed to be applied to the system branch isolation valves V23-152A, 29A, 19A and 59A.

Safety Evaluation Summary (SE 98-032)

The safety impact of this change was reviewed and it was determined that this change does not increase the probability of occurrence or consequences of an accident or malfunction as previously evaluated in the FSAR. This upgrade also does not present a significant hazard not described or implicit in the FSAR. There is reasonable assurance that the health and safety of the public was not endangered as a result of this upgrade.

Revising the Safety Class of Valves, Pressure Indicators and a Transmitter Located in the HPCI Suction Line

General Summary

An evaluation of the HPCI suction valves concluded that a single failure under certain conditions could cause all HPCI suction MOVs to go open on a loss of logic power. The possibility that MOVs V23-57 and 58 could be open resulted in a requirement for the suction check valve V23-32 to be operable and maintain the water seal in the Torus. A majority of piping in HPCI would also need to be treated as an extension of the primary containment per Appendix J. Using Appendix J criteria, the HPCI system would become a closed seismic loop and an extension of the containment which adds new equipment to the Appendix J Program.

The current Appendix J Program lists the containment boundary as MOV V23-58. Until a design change could be implemented, a deficiency existed with the current Appendix J Program boundaries and the extended boundary. To make the existing design agree with the Appendix J Program, the HPCI suction line boundary had to be extended through the HPCI pump suction into the discharge side of the HPCI system.

Components V23-806/808, PI-23-44/45/99 and PT-23-100, in the expanded boundary were listed as Non-Nuclear Safety (NNS). To be an Appendix J boundary, the above components had to be changed to Safety Class 2 (SC2). In addition, the requirements of AP 0155, Appendix B "Locking Policy" had to be applied to system branch isolation valves V23-140A/B/C, 141A/B, 142A/B and 143A/B.

Safety Evaluation Summary (SE 98-034)

The safety impact of this change was reviewed and it was determined that this change does not increase the probability of occurrence or consequences of an accident or malfunction as previously evaluated in the FSAR. This upgrade also does not present a significant hazard not described or implicit in the FSAR. There is reasonable assurance that the health and safety of the public was not endangered as a result of this upgrade.

Change to RHR and RHRSW Maximum Flow Rates

General Summary

Operating procedures were revised to change the flow limits of 2,700 gpm on RHRSW and 7,000 gpm on RHR through the RHR heat exchangers, E-14-1A and B. The changes were made to assure that adequate heat removal capability would be available under postulated design basis accident conditions when maximum potential flow measurement instrument errors are considered. Operators were instructed to not exceed the specified design basis flow rates of 2,700 and 7,000 gpm respectively. Under the worst case combination of flow instrument errors following a design basis Loss of Coolant Accident, actual flow rates through the RHR heat exchanger could have been as low as approximately 6,300 gpm for RHR and approximately 2,450 gpm for RHRSW. This was less than the assumed minimum values of 6,400 gpm and 2,700 gpm in the containment analysis.

Engineering evaluations were done to assess the maximum indicated flow rates that could be used by operators that would assure the minimum actual flow rates exceeded the containment analysis basis values without adversely affecting other components. The evaluation showed that the flow limit on RHR could be removed altogether and the upper bound maximum actual flow rate for one RHR pump operating would be 7,400 gpm. Operation of two RHR pumps in one loop would require the RHR heat exchanger bypass valve to be open to limit flow through the heat exchanger. RHRSW limits could be changed to an upper bound of 3,050 gpm indicated, corresponding to 3,300 gpm actual.

Safety Evaluation Summary (SE 98-036)

The safety impact of changing the RHR and RHRSW flow rates and the associated effects of the increase were reviewed and it was determined that this change did not increase the probability of occurrence or consequences of an accident or malfunction as previously evaluated in the FSAR. The proposed increase in the maximum RHR and RHRSW flow rates through the RHR Heat Exchanger will provide increased margins of safety for heat removal capability. The increased flow rates were evaluated for their effect on the structural integrity of the RHR heat exchanger and associated piping. The impact was determined to be negligible. There is reasonable assurance that the health and safety of the public was not endangered as a result of this increase.

Clarification of Separation Criteria for Instrumentation and Control Cables – FSAR Section 8.4

General Summary

This change addresses the FSAR wording regarding the installation of selected control and instrument cables in the same raceway. This justifies selected installations as not having any adverse consequences from the installation of a control cable in an instrument tray when the instrument cable is used only for indication.

This change also addresses the FSAR wording regarding the installation of power cable in that it must be separate from low-level signal instrumentation cable. This justifies selected installations, specifically 120VAC power supply cables being routed to Reactor Protection System (RPS) instrumentation cabinets in the Reactor Building. Evaluation of this installation showed that it was acceptable because failure of the power cable will not initiate a spurious Reactor Scram or prevent a scram from occurring.

Safety Evaluation Summary (SE 98-035)

The subject cables and any induced signals are not accident initiators and do not impact any system used to mitigate an accident. No malfunctions would occur due to these changes as one change is only used for indication and the other change is limited to the RPS power supply cables and LPRM detector cables which do not control any process systems that can initiate a transient. There is no reduction in the margin of safety, as cable separation does not affect FSAR analyses or the basis for any Technical Specifications.

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

Procedure OP 0044, Volume Reduction Packaging and Shipping of Irradiated Hardware from Spent Fuel Pool - Partial Change

General Summary

This design change will allow the Refuel Floor Process Radiation Monitors (PRM's) Group III initiation to be bypassed during planned evolutions. The evolutions could cause an alarm and initiate a Group III Primary Containment Isolation System isolation during fuel pool clean-up activities. These activities were under the full control of a radiation protection technician and presented no unusual radiation doses to the workers. Although the refuel floor monitors were bypassed, the Reactor Building HVAC monitors remained functional and would initiate a Group III isolation should any sources cause high airborne levels in the Reactor Building.

Activities included loading of liners into the 8-120 cask using the transfer bell and removal of contaminated equipment from the fuel pool.

Safety Evaluation Summary (SE 97-026)

The refuel floor Process Radiation Monitors are not initiators of any accidents nor do they initiate any abnormal operational transients. No higher accident doses would have been possible as the tools, hardware and components being processed had lower dose rates than those analyzed in the FSAR accidents. Any postulated dose rates from the clean-up operation are bounded by the FSAR analyzed accidents. Since none of the activities controlled by this procedure can result in a refueling accident and other installed means of detection are available; there is no reduction in any associated safety margin.

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

Procedure OP 1432, Revision 0, In-Core Fuel Sipping

General Summary

OP 1432 was written to provide instructions for in-core fuel sipping. The process used involved placing a hood over the fuel channels on the fuel bundle. The placement of this hood restricted the flow through the fuel bundles allowing the bundle to heat up. If a leak were present, fission gas would escape through the hole in the fuel rod into the hood area. Water was then drawn up from the hood to a Flow Control and Degas Unit. A gas sample was separated from the water and processed. The material used to seal the hood over the fuel channels was approved for use in the BWR environment.

This process of sipping fuel is an improvement over vacuum sipping because the fuel bundles do not need to be removed from the reactor to the Spent Fuel Pool to be sipped.

Safety Evaluation Summary (SE 98-015)

This type of incore sipping does not increase the probability of occurrence or consequences of an accident or malfunction as previously evaluated in the FSAR because the fuel bundles do not need to be removed form the reactor to be sipped. This process also does not present a significant hazard not described or implicit in the FSAR, and there is a reasonable assurance that the health and safety of the public is not endangered.

OP 2115, Rev. 39, Primary Containment

General Summary

OP 2115, Rev. 39, was revised to allow the use of VNP-1065, the nitrogen purge gas pressure regulator bypass valve. This change was required due to the configuration of the N^2 purge pressure regulator and its attendant downstream temperature sensors. The sensors close N^2 pressure regulator PCV-1001 if a low temperature of 60°F or a high temperature of 120°F is sensed downstream of the pressure control valve. The change required that the correct N^2 purge gas temperature be established prior to placing PCV-1001 in service. Additionally, should PCV-1001 fail, a provision was made to allow the use of VNP-1065, with appropriate controls, to control N^2 purge pressure and temperature.

Safety Evaluation Summary (SE 98-029)

This procedure change did not increase the probability of occurrence or consequences of an accident or malfunction as previously evaluated in the FSAR. The nitrogen purge system is not an initiator of any accident or transient analyzed in the FSAR. This procedure 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.

Procedures OP 2123, Revision 27, Core Spray and OP 2124, Revision 43, Residual Heat Removal System

General Summary

The changes made to these two procedures reflect those changes required to ensure that the RHR and CS pumps have adequate long-term minimum flow protection while meeting their design and licensing requirements. These revisions did not change the short-term automatic protection features of the RHR and CS Systems.

Safety Evaluation Summary (SE 98-009)

There was no increase in the probability of occurrence or consequences of an accident or malfunction as previously evaluated in the FSAR. These procedure changes do not present a significant hazard not described or implicit in the FSAR, and there is a reasonable assurance that the health and safety of the public was not endangered.

Procedure OP 2123, Revision 28, Core Spray

General Summary

OP 2123 was revised to add sections necessary to align the Core Spray system for standby operation during refueling, utilizing suction from the Condensate Storage Tank (CST). During the time the Core Spray System was aligned to the CST, it remained operable.

Safety Evaluation Summary (SE 98-018)

Re-aligning the Core Spray System did not change any setpoints or safety limits nor did it increase the probability of occurrence or consequences of an accident or malfunction as previously evaluated in the FSAR. This re-alignment did not present a significant hazard not described or implicit in the FSAR, and there is a reasonable assurance that the health and safety of the public was not endangered.

Procedure OP 2181, Revision 43, Service Water/Alternate Cooling Operating Procedure, and FSAR Section 10.6.5

General Summary

OP 2181 was revised to change the Administrative Limit and steps associated with the requirements for aligning the SW discharge to either the deep basin or condenser discharge block.

Safety Evaluation Summary (SE 98-023)

The objective of this Safety Evaluation was to verify that the 45°F transition point was only applicable when the plant was operating at power and that discharging the Service Water System to the cooling tower basin with the reactor coolant temperature less than 212°F and reactor power less than 1% would not introduce an unsafe condition or an unreviewed safety issue. An evaluation was also performed to assure that this change did not effect or impact any safety system and would not create an accident or malfunction not previously evaluated in the FSAR. It was concluded that there was reasonable assurance that the health and safety of the public was not endangered due to this change.

Procedure, OP 2143, 480 and Lower Voltage AC System and OT 3122, Loss of Normal Power

General Summary

These changes were made to reflect the change in breaker position for the Alternate Cooling Tower Fan 2-1 from normally closed to normally open on Motor Control Center 8C. This change was necessitated due to a cable separation concern. Upon loss of the normal power or a requirement to utilize the Alternate Cooling System (ACS) due to loss of the Vernon Pond, flooding of the intake structure or a fire in the intake structure, the fan can now be powered from either diesel through manual operator action.

The power supply to the fan is load shed upon loss of normal power. Within the process of establishing the ACS, the operators also close a power supply breaker from either diesel. The operators have two hours to establish the ACS operation, which is more than sufficient time for either breaker to be closed.

Safety Evaluation Summary (SE 97-028)

The ACS is not classified as an Engineered Safeguard System and is not designed to accept the consequences of a design basis LOCA. It is also not single failure proof. The objective of the ACS is to provide an alternate means of heat removal in the unlikely event that the Service Water Pumps become inoperable.

The ACS is not an initiator of any design basis accident nor does it initiate any abnormal operational transients. The Technical Specification requirements for the ACS require that the ACS be demonstrated to be operable within twenty-four hours of discovery and declaration that one Service Water Subsystem is inoperable. The ACS is not associated with any safety limits as described in the Technical Specifications; therefore, the margin of safety is not changed,

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

FSAR Change, Section 8.6-1

General Summary

This revision to the FSAR changed the minimum Main Station Battery voltage from 105 V to 108 V. This change was supported by design basis calculations that demonstrated the adequacy of the battery size and ensured that sufficient voltage would be available to operate the required emergency loads.

Safety Evaluation Summary (SE 98-010)

The Main Station Batteries provide DC power during LNP conditions. This change ensured that sufficient voltage was available to operate control components and MOVs. This change did not result in a physical change to the plant therefore, there were no increased consequences of an accident or malfunction or an increase in the probability of a malfunction nor did it result in any new failure modes. The margin of safety was not affected by this change and there is reasonable assurance that the health and safety of the public was not endangered.

FSAR Change, Sections 6.4-1 and 4.7-4

General Summary

FSAR Sections 6.4-1 and 4.7-4 were revised to describe the design bases for the acceptable containment isolation features for the Condensate Storage Tank suction lines for HPCI and RCIC.

There was no physical change to the plant due to this revision; the revision merely provided a specific description of the primary containment design bases for the HPCI and RCIC CST suction lines. The design bases for these lines were developed in a System Engineering Self-Assessment from design basis statements already existing in the FSAR. By inclusion of the statements in this FSAR change, the FSAR would document the work that was done in the Self-Assessment to determine that a single check valve in the suction line to the CST was acceptable.

Safety Evaluation Summary (SE 97-013)

This FSAR change provided a more accurate description of the existing containment isolation features in the HPCI and RCIC CST suction lines. There was no aspect of this clarification of the design basis that increased the probability of occurrence or consequences of an accident or malfunction as previously evaluated in the FSAR. This revision 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.

FSAR Change, Section 13.9.3

General Summary

This change to the FSAR updated information contained within Section 13.9.3 to reflect the current procedures and work control processes. The current procedures and work control processes ensure that refueling activities are safely and appropriately controlled.

Safety Evaluation Summary (SE 98-011)

The changes made to Section 13.9.3 and the Refueling procedures are administrative in nature and do not result in physical changes to the plant. There was no increase in the probability of occurrence or consequences of an accident or malfunction as previously evaluated in the FSAR. These changes do not present a significant hazard not described or implicit in the Vermont Yankee FSAR, and there is a reasonable assurance that the health and safety of the public was not endangered.

FSAR Change, Section 10.12.3.6

General Summary

This change revises the Vermont Yankee House Heating Boilers (HHB) operating pressure to be less than the 50 psig currently specified in the FSAR. The wording in the FSAR would read 28-50 psig versus 50 psig. This allows more flexibility to operate within known safety margins.

Safety Evaluation Summary (SE 97-022)

The HHBs are not an initiator of any accident or malfunction analyzed in the FSAR. The change in setpoint does not alter the function of the station Heating, Ventilating and Air Conditioning or affect any components that could be considered accident initiators. This change does not result in a reduction in a margin of safety as described in the FSAR or the basis for any Technical Specification.

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

Setpoint Change No. 98C-020, 021 and 022 and OP 4354

General Summary

As a result of changes and revisions to instrument accuracy calculation VYC-714, sections E, F and G, setpoint changes were made to the following equipment and incorporated into OP 4354, "RHR Subsystem A/B Logic Functional/Calibration Test".

Relays 10A-K45A/B	RHR (LPCI) Outboard Injection Valve Open Signal Removal Time Delay
Relays 10A-K50A/B	Time Delay Start of RHR Pumps P-10-1B/1C in the LPCI Mode with Power from Diesel Generators
Relays 10A-K72A/B	Prevent Closure of V10-65A/B on LPCI Initiate Signal

These setpoints were changed to reflect current analysis methods and equipment specific uncertainties.

Safety Evaluation Summary (SE 98-028)

These setpoint changes did not increase the probability of occurrence or consequences of an accident or malfunction as previously evaluated in the FSAR. These setpoint changes also 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.

Technical Specification Change to Section 3.10 Bases – Batteries

General Summary

The change to Technical Specification Section 3.10 bases removed the option of having one cell in a station battery, ECCS Instrumentation battery or an Uninterruptible Power System battery out of service with the battery still considered operable. This change enhances battery capacity and capability and simplifies battery sizing calculations. The subject batteries are now only considered operable with all cells in service.

Safety Evaluation Summary (SE 97-017)

This change is more restrictive than previous requirements and does not allow a battery in a degraded mode (one cell out of service) to be considered operable.

The batteries are not initiators of any analyzed accident or operational transient. As the batteries are supporting features for various safety systems that are used to mitigate accidents, this change enhances overall safety. The margin of safety with the DC system will be increased with this change in Technical Specification bases due to the increased capacity, capability and voltage of the batteries.

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